**Proceedings of the IEA Workshop/Working Group
Meeting on Ferritic/Martensitic Steels***

ECN Nuclear
Petten, Netherlands

October 1-2, 1998

Prepared by: R. L. Klueh

IEA WORKING GROUP - TASK ANNEX II

Implementing Agreement
for a program of research
and development on fusion materials

*Research sponsored by the Office of Fusion Energy Sciences, U. S. Department of Energy, under contract DE-AC05-96OR22464 with Lockheed Martin Energy Research Corp.

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Workshop Participants

European Union

- A. Alamo, CEA Saclay, France
- E. Materna-Morris, FZK, Germany
- R. Schaublin, PSI, Switzerland
- B. van der Schaaf, ECN Petten, Netherlands
- E. V. van Osch, ECN Petten, Netherlands

Japan

- A. Hishinuma, JAERI, Japan
- A. Kimura, Kyoto University, Japan
- A. Kohyama, Kyoto University, Japan
- K. Shiba, JAERI, Japan

United States

- R. L. Klueh, Oak Ridge National Laboratory, USA
- G. R. Odette, University of California at Santa Barbara, USA
- F. W. Wiffen, U. S. Department of Energy, USA

IEA WORKSHOP ON REDUCED-ACTIVATION FERRITIC/MARTENSITIC STEELS

ECN, Petten, The Netherlands

1-2 October 1998

1 October

9:00 Welcome and Introduction

1. Ferromagnetic Effects and Steel Properties (Ch: G. R. Odette)

09:15 M. Ruatto Magnetization of RAFM Steels

09:45 A. Kohyama Fatigue Data on JLF-1 and the Future Plan

10:15 A. Alamo Thermal Ageing Behaviour of RAFM Steels and F82H Weldments

10:45 Break

11:00 K. Shiba Unirradiated Properties of F82H IEA Heat—Status and Recent Results

11:30 A. Hishinuma Development of Spark Plasma Sintering (SPS) Joining Techniques for RAM Steels

11:40 Discussion of Ferromagnetic Effects and Steel Properties

12:00 Lunch

2. Irradiated Properties and Composition Effects (Ch: A. Alamo)

13:00 K. Shiba Present Status of JAERI Irradiation Program for LAM

13:30 E. V. van Osch Post-Irradiation Mechanical Properties of F82H plate, TIG, and EB Welds

14:00 A. Kohyama Present Status of JUPITER Program

14:30 R. L. Klueh Effect of Tantalum on Properties of Irradiated 9Cr-2WVTa Steel

15:00 Break

3. ODS Alloys and Alloy Development (Ch: A. Kohyama)

15:15 B. van der Schaaf ODS RAFM Steels: Review and Prospects

15:45 A. Alamo Development of Ferritic/Martensitic ODS alloys

16:15 A. Hishinuma Preliminary Studies for Developing ODS RAM Steels

16:30	G. R. Odette	Integrating Modeling, Experiment and Data Base Development: A Mechanism-Based Approach to Developing Martensitic Steels for Fusion Applications
17:00		Discussion of Irradiation Effects, Composition Effects, ODS Steels and Alloy Development
17:30		Adjourn

2 October

4. Helium Effects Studies

(Ch: F. W. Wiffen)

09:00	R. Schäublin	Electron Microscopy of RAFM Steels With Helium
09:30	K. Shiba	Tensile and Charpy Properties of B-Doped F82H After Irradiation
10:00	E. Materna-Morris	Microstructural Investigations of He-Generation Effects.
10:30		Break
10:45	A. Kimura	Effects of Helium Implantation on the Ductile-Brittle Transition Behavior of Reduced Activation 9Cr-2W Martensitic Steel
11:15	E.V. van Osch	Post-Irradiation Welding and Helium Generation With ¹⁰ B in RAFM Steel
11:45		Discussion of Helium Effects Studies
12:15		Lunch

5. Strategy For Development of Ferritic/Martensitic Steels

(Ch: A. Hishinuma and B. van der Schaaf)

13:30	A. Hishinuma	Summary of Japanese Strategy
13:50	B. van der Schaaf	Summary of European Union Strategy
14:10	F. W. Wiffen	Summary of U. S. Strategy
14:30		Discussion to Develop a Proposed Strategy for Riso Meeting

6. Progress and Future Cooperation

(Ch: B. van der Schaaf and R. L. Klueh)

15:15		Discussion on Progress and Status of Action Items Recommendations for Future Cooperation
16:00		Adjourn

REPORT ON THE IEA WORKSHOP/WORKING GROUP MEETING ON FERRITIC/MARTENSITIC STEELS FOR FUSION

ECN Nuclear Research, Petten, The Netherlands, 1-2 October 1998

R. L. Klueh

Executive Summary

The International Energy Agency (IEA) Working Group on Ferritic/Martensitic Steels for Fusion held a workshop at ECN Nuclear Research, Petten, The Netherlands, 1-2 October 1998. The Working Group, consisting of researchers from Japan, the European Union, the United States, and Switzerland, met to review research that has been completed since the previous meeting and to continue planning and coordinating an international collaborative test program on reduced-activation ferritic/martensitic steels for fusion applications. At the workshop, data were presented from the continuing research on the IEA heats of steel that are being studied in the collaboration. Data on these and other reduced-activation steels in the irradiated and unirradiated condition were presented. Other subjects that were discussed included effects of a ferromagnetic steel in a fusion machine, the effect of helium on properties, and the development and application of oxide dispersion-strengthened steels for fusion. A Working Group status-review meeting is planned in conjunction with the International Conference on Fusion Reactor Materials (ICFRM-9) in Colorado Springs, Colorado, USA, 10-15 October 1999, at which time plans for a workshop to be held in 2000 will be finalized.

Introduction

The IEA Working Group on Ferritic/Martensitic Steels for Fusion under the auspices of the IEA Executive Committee for the Implementing Agreement on Fusion Materials conducted a workshop at ECN Nuclear Research, Petten, The Netherlands, 1-2 October 1998. Researchers from Japan (4), the European Union (5), the United States (3), and Switzerland (1) participated. Russian Federation participation was invited, but no one from there attended the meeting. The objective of the Working Group is the establishment and coordination of an international collaborative test program to determine the feasibility of using ferritic/martensitic steels for fusion.

This workshop was the ninth meeting of the Working Group, which was formed as a result of a workshop on ferritic/martensitic steels in Tokyo in October 1992. At the first meeting following the Tokyo workshop, the Working Group developed specifications for large heats of reduced-activation steels and outlined a collaborative research program. Two 5-ton heats of the IEA-modified F82H steel and two 1-ton heats of JLF-1 steel were produced, fabricated into plates, and distributed to the participants of the collaboration. Subsequent meetings have been used to plan a test program and to coordinate the acquisition of the data needed to prove feasibility for the steels for fusion.

The Petten meeting was a follow up to the meeting at Tokyo, Japan, 3-4 November, 1997, in which information was presented that indicated helium has an embrittling effect on ferritic/martensitic steels irradiated at 250-400°C. At the Tokyo meeting and at ICFRM-8 at Sendai, Japan, 27-31 October 1997, several investigators presented data on oxide dispersion-strengthened (ODS) steels as possible structural materials that will allow higher operating temperatures. At Petten, information was presented on both of these subjects, along with recently developed information on the properties of the IEA heats of reduced-activation steels and other reduced-activation steels. Information was also presented on work designed to determine the effects produced by ferritic/martensitic steels in the high magnetic fields of a magnetically confined fusion reactor.

Research and Development Activities

The following is a brief description of the information presented at the Petten workshop. Copies of viewgraphs and other information presented at the workshop are appended to this summary.

Ferromagnetic Effects

At one time it was believed that the expected strong interaction of a ferromagnetic material with the magnetic fields of a magnetically confined fusion system would make it impossible to use the ferritic/martensitic steels as structural materials. Calculations in the early 1980s indicated that the effect could be taken into account in the design of the reactor. Because these early studies were but cursory analyses of the problem, questions as to the magnitude of the effect persist, and detailed design studies as well as experimental investigations are required to eliminate this uncertainty.

P. Ruatto has been involved in a program at FZK Karlsruhe to study the transient eddy current problems and magnetic fields and forces that can develop with the use of ferritic/martensitic steels. Ruatto's presentation (given by E. Materna-Morris) discussed some of the information derived with the three-dimensional finite element method program AENEAS that was developed at FZK. Results were presented for calculations that examined the effect of plasma disruptions on the outboard blanket segment of the DEMO helium-cooled pebble bed outboard blanket segment and the European Helium cooled pebble bed test blanket module in ITER. A centered plasma disruption was considered for the DEMO, and the forces on the components were calculated and described for the MANET steel. The force calculations for a plasma disruption in ITER were also summarized. The conclusion was that for a correct mechanical design of a fusion power plant it is necessary to include an electromagnetic analysis, and the AENEAS is an appropriate tool for this task.

K. Shiba discussed the Japan Atomic Energy Research Institute (JAERI) research effort on the effect of a ferromagnetic material (ferritic/martensitic steel) on the operation of a fusion machine. (Note that this presentation was part of Shiba's presentation in the section on Steel Properties—Irradiated, and the viewgraphs on the ferromagnetic effects are included in his presentation in that

section.) Three subject areas are being pursued by JAERI: (1) an experimental study of the use of a ferritic steel for plasma ripple reduction in ITER by the installation of a "ferritic board" on the JFT-2M tokamak, (2) an experimental study of the possibility of producing rippleless plasma operation with a reduced-activation martensitic steel as the vacuum vessel by the installation of a reduced-activation (F82H) ferritic steel liner in a small tokamak, and (3) using this lined tokamak to conduct research on possible undesirable effects due to ferritic/martensitic steels on plasma production and control. Work began on (1) this past year and involved a computer simulation and preliminary experiment using JFT-2M. The results indicated a reduction in ripple magnitude and a modification of the magnetic field over the whole plasma region due to the insertion of the ferritic board.

R. Klueh has attempted to determine what work has been conducted and what work is ongoing throughout the world on ferromagnetic effects, and the results of that have been summarized. A copy of that summary is included in the appended material following Ruatto's presentation.

Steel Properties—Unirradiated

The work of the Monbusho fatigue test program in Japan to develop mini-sized test techniques, study size effects, develop a strain-control test technique without contacting the specimen, and determine the fatigue behavior of JLF-1 steel was described by A. Kohyama. A hydraulic servo-controlled testing machine using laser measurements has been developed that should be applicable to hot-lab testing. Testing of full-sized hour-glass specimens (100 mm long, 9 mm at the center of the specimen) and miniature specimens (25.4 mm long, 1.25 mm at the center) measured comparable properties except under the condition of very low cycle fatigue. The machine was used to test JLF-1 base metal and TIG weldments, and the results indicated that fatigue strength (S-N curve) of the base metal was less than that of the weld metal. A correlation was developed between the fatigue limit and Vicker hardness that was related to the tensile strength, which was shown to provide good predictions for the fatigue limit.

A. Alamo presented data on the effect of thermal aging on the tensile and Charpy behavior of six European Union (EU) reduced-activation ferritic/martensitic steels and F82H and JLF-1. Aging was for up to 13400 hours at 250, 350, 400, 450, and 550°C. The EU steels included steels with high carbon and nitrogen (LA12Ta), low C with (LA12TaLC) and without (LA12LC) tantalum and low nitrogen (LA12TaLN). There was also high (11%) chromium (LA4Ta) and high (3%) tungsten (LA13Ta).

There was little effect of aging on the yield stress of the F82H after aging 13400 h, but the reduction of area was significantly reduced above 400°C. The Charpy results for the EU steels indicated that there were chemical composition effects. For example, aging the high-tungsten steel to 10000 h at 350, 400, 450, and 550°C, resulted in a reduction of upper-shelf energy (USE) and an increase of transition temperature for the higher temperatures, which was associated with Laves phase. There were also indications of chromium effects, especially at 400°C aging, which may be due to chromium-rich α' formation. The F82H and JLF-1 showed a reduction in USE and

an increase in transition temperature after thermal aging 13400 h at 550°C, but little effect after aging at 250, 350, 400, and 450°C. Laves phase formation may play a role in this behavior. Tensile, creep, and Charpy tests were also made on thermally aged F82H weldments produced by the TIG and electron beam (EB) processes. TIG welds, which were post-weld heat treated (PWHT), displayed a similar strength but a slightly lower ductility and USE compared to the base metal. In particular, some degradation of impact properties was found after aging at 550°C. However, the results for EB welds without a post-weld heat treatment still need to be compared with steels that have a PWHT to fully evaluate the properties of the EB weldments.

K. Shiba reported on the continued progress on the JAERI round robin tests that are generating a range of mechanical and physical properties data for the IEA heat of F82H. Mechanical property tests that have been made or that are in progress include hardness, tensile, Charpy impact, fatigue, fracture toughness, and creep. The range of physical properties include density, specific heat, thermal expansion, thermal and electrical conductivity, melting point, Young's modulus, Poisson's ratio, modulus of rigidity, and magnetic hysteresis. Other measurements include the determination of a continuous-cooling-transformation diagram, water corrosion, hydrogen permeability, and hydrogen cracking. Some of the mechanical property tests have also been conducted on aged steel and on weldments. Shiba presented recent tensile, Charpy, fatigue, and fracture toughness results on thermally aged and unaged F82H steel. Analysis of the extracted precipitates from the aged steel indicated a tendency toward the production of Laves phase for steel aged at 550, 600, and 650°C. Mechanical properties of the weldments were generally comparable to the base metal.

Fabrication of the blanket structure of a fusion power plant presents many difficulties, especially welding and joining, and A. Hishinuma presented information on a potential joining technique. Hot Isostatic Pressing (HIP) bonding is a potential technique for certain geometries. However, the optimum conditions for HIP bonding are 150 MPa at 1040°C for 2 h followed by tempering. Such a high temperature and long hold time can have negative effects on the properties due to austenite grain growth. Spark plasma sintering (SPS) bonding, which involves the formation of a plasma between the parts being joined, is being studied by JAERI as an alternative to HIP bonding. SPS conditions are 20-50 MPa at 800-900°C with hold times of 0.08-1 h. Excellent joints have been obtained with this technique; the joints are improvements over HIP-bonded material in metallographic appearance and strength.

Steel Properties—Irradiated

The status of the JAERI irradiation program on F82H was reviewed by K. Shiba. Irradiations are being carried out in the High Flux Isotope Reactor (HFIR) in the U.S./JAERI collaboration and in the Japan Materials Test Reactor (JMTR) and the Japan Research Reactor (JRR-2/JRR-3/JRR-4). Accelerator (dual/triple beam) irradiations are also being conducted. The program involves tensile, Charpy, and fracture toughness measurements and microstructural studies of the irradiated steel.

E. V. van Osch reported on results of work at ECN at Petten on post-irradiation properties of the IEA F82H plate and welds. Irradiation was in the High Flux Reactor (HFR) to 2-3 dpa and the testing (tensile, impact, and static fracture toughness) is in progress. A 65 kg heat of steel (ECN-BS) was obtained and irradiated with F82H. ECN-BS contained somewhat more Cr, C, and Ta and less B than the F82H. The ECN-BS steel showed improved Charpy properties over the F82H after irradiation to 2.5 dpa at 300°C. Comparison was made between EB and TIG weldments of the IEA F82H. Before irradiation, the EB welds had a higher strength and ductility; testing of the irradiated welds is in progress. Irradiations to 10 dpa at 300°C are in progress, with the testing to be performed under the next EU Framework Program (1999-2002). The F82H is included in this experiment, but emphasis of this framework program will be on the new EUROFER steels. This experiment will also include work on B-doped steels to investigate the effect of helium on properties, to investigate the distribution of the boron in the steel, and to measure the helium content.

The Japanese universities (Monbusho) program on the properties of irradiated reduced-activation ferritic steels for fusion reactors was reviewed by A. Kohyama. Most of this work was on the JLF-1. The first irradiations were carried out in FFTF. The results included: tensile studies conducted on steels irradiated to 60 dpa at 365-600°C, swelling data obtained after irradiation to 70 dpa at 420°C, Δ DBTT (change in ductile-brittle transition temperature) data obtained from irradiations to 50°C at $\approx$ 400°C, and pressurized-tube irradiation creep tests for specimens irradiated to 35 dpa at 520°C. Current work involves experiments in HFIR, JOYO, and JMTR. At present a dual-beam ion-irradiation facility (DuET) is being constructed at Kyoto University that will be used for future in-beam studies. The facility is expected to begin operation in FY 1999.

Another Monbusho effort is the Ferritic Isotopic Tailoring (FIST) experiment in which isotopic-tailored F82H disks were irradiated in HFIR to simulate the fusion environment effects of producing hydrogen and helium in the steel. Preliminary results from TEM and shear-punch tests have been obtained and are being evaluated.

The effect of tantalum in the ORNL 9Cr-2WVTa steel on Charpy and tensile properties after irradiation was discussed by R. L. Klueh. The steel has excellent strength and impact toughness before and after irradiation in the Fast Flux Test Facility (FFTF) and the High Flux Reactor (HFR). The ductile-brittle transition temperature (DBTT) increased only 32°C after 28 dpa at 365°C in FFTF, compared to a shift of $\approx$ 60°C for a 9Cr-2WV steel—the same as the 9Cr-2WVTa steel but without tantalum. This difference occurred despite the two steels having similar tensile properties before and after irradiation. The 9Cr-2WVTa steel has a smaller prior-austenite grain size, but otherwise microstructures are similar before irradiation and show similar changes during irradiation. The irradiation behavior of the 9Cr-2WVTa steel differs from the 9Cr-2WV steel and other similar steels in two ways: (1) the shift in DBTT of the 9Cr-2WVTa steel irradiated in FFTF does not saturate with fluence by $\approx$ 28 dpa, whereas for the 9Cr-2WV steel and most similar steels, saturation occurs by $<$ 10 dpa, and (2) the shift in DBTT for 9Cr-2WVTa steel irradiated in FFTF and HFR increased with irradiation temperature, whereas it decreased for the

9Cr-2WV steel, as it does for most similar steels. The improved properties of the 9Cr-2WVTa steel and the differences with other steels were attributed to tantalum in solution and the loss of that tantalum during irradiation by precipitation. The precipitation still needs to be confirmed.

ODS Steels and Alloy Development

B. van der Schaaf reviewed the possibility of oxide dispersion-strengthened (ODS) steels for fusion applications. These steels contain a high number density of (TiO_2 or Y_2O_3) oxide particles that provide enhanced creep strength. One problem with the conventional and reduced-activation ferritic/martensitic steels being investigated for fusion is that the upper operating temperature will be limited to $\approx 550^\circ\text{C}$, and this limits the systems in which they can be used (e.g. water-cooled system). ODS steels with their improved creep properties offer the possibility of extending that temperature to 600°C and higher. Because they are strengthened by a high number density of small oxide particles, the oxide particles could provide sites for defect recombination and helium trapping and thus reduce swelling and suppress helium bubble effects. Most of the prior work on these materials for nuclear applications were for fuel canning for fast breeder reactors. The results for that application indicated significant improvement in creep strength over conventional steels with the helium effects suppressed. The major problems involved the anisotropy due to the powder metallurgy fabrication techniques used to make the tubes. There is limited experience on thick-wall parts, and although reduced-activation ODS steels are being developed, there is as yet no literature information available on them.

Van der Schaaf concluded that the ODS reduced-activation steels being developed show considerable promise that indicates they could, if developed, extend operating temperatures above 600°C (assuming creep controls and not corrosion) and reduce helium effects. However, the fabrication route needs to be developed for the larger sections needed in a fusion reactor blanket. Joining may present some difficulty and should be addressed early in the development stage.

Work on the development of ODS steels was described by A. Alamo. The high-chromium ferritic steels (MA 956 and MA 957) had elongated grains (recrystallized grain size $d > 1$ mm, recrystallization temperature $> 1300^\circ\text{C}$) with a high texture, anisotropic properties, and low ductility. The creep and aging behavior of these steels was studied. The MA 957 with an optimized grain size showed excellent creep resistance at 650°C relative to a 15-15 austenitic stainless steel, especially for longer rupture times ($> 10^4$). Precipitation of intermetallic phases (χ , Laves, and α' phases) was detected in the thermal aging studies and irradiation experiments.

The development of a 9Cr ODS steel which can transform to martensite is being pursued. The objective is to avoid intermetallic phase precipitation and reduce the anisotropy of the properties compared to the fully ferritic materials. 9Cr-Mo and 9Cr-W steels containing Y_2O_3 are being examined. These steels developed an equiaxed grain structure when normalized and tempered, and there was no grain growth in the range 1000 - 1250°C . The yield stress for each of these steels was higher than that for MA 957 with a somewhat reduced, but still high, reduction of area. This development study is continuing.

A. Hishinuma reported on the JAERI efforts to produce an ODS reduced-activation ferritic/martensitic steel. The compositions that have been investigated were variations on the F82H with 8% Cr, 0-1.75% W, 0.1-0.28% Ti, 0.15% O, 0.1-0.23% Y, 0.12% C. The manufacturing process involved mechanical alloying the powders followed by hot extrusion at 1050°C to fabricate the steel, after which it was normalized and tempered. Microstructures have been produced that have a fine-grain structure that appears relatively equiaxed. Excellent Charpy and tensile properties were obtained from several of the experimental steels. The compositional variations indicated that the tensile properties depended on the Y_2O_3 and tungsten content, but much less on the titanium content.

G. R. Odette discussed the recent review of the fusion materials program in the U.S. by The Fusion Energy Systems Advisory Committee of which Odette was a member. They recommended that the fusion materials program seek to integrate modeling, experiment, and data-base development to develop advanced materials for fusion. This means bringing more modeling into the materials program, as the committee viewed the program as being deficient in this area. Odette feels that one area where such an approach can be applied is the study of fracture of fusion reactor components. Since fracture behavior of irradiated materials is of critical importance for fusion, micromechanical-based local fracture models need to be applied with small specimen measurement of fracture resistance on unirradiated and irradiated material to provide the resultant properties necessary to predict limits for fusion structures. These results need to be further combined with microstructure-property models that reflect the effect of alloy composition, processing variables, and irradiation. The implementation of such an integrated approach was discussed in terms of work being conducted at the University of California at Santa Barbara. As one example, work on reactor pressure vessel embrittlement was cited and discussed.

Helium Effects Studies

Electron microscopy studies of the reduced-activation ferritic/martensitic steels IEA F82H and OPTIMAX A were discussed by R. Schaublin. The F82H was irradiated to 0.5 and 1.7 dpa with 590 MeV protons at PSI in Switzerland, and F82H and OPTIMAX A were irradiated to 2.5 dpa at 250°C in HFR in Petten. The dislocation structure, carbide composition and size distribution, and grain/lath boundary chemistry of the F82H irradiated with protons were determined for the unirradiated (before and after tensile deformation) and irradiated steels. The results for the proton irradiation of the IEA F82H generally indicated that there was essentially no difference in the microstructural defects in the as-received (unirradiated), deformed (material taken outside the necked region), and irradiated conditions. The $M_{23}C_6$ particles, which constituted the majority of the precipitate, were found to be coherent with the matrix. Chromium enrichment at prior austenite grain boundaries was detected for the normalized-and-tempered steel, but after irradiation, chromium depletion was observed.

Neutron irradiation of OPTIMAX A at 250°C produced no defects, but faceted cavities were observed. For the F82H, on the other hand, no cavities were present, but black dot (loops) damage was observed.

Helium effects studies using boron-doped F82H steel irradiated in HFIR and JMTR were reported by K. Shiba. Standard F82H, which contains a small amount of natural boron, F82H to which natural boron was added, and F82H to which ^{10}B was added were compared. The ^{10}B is transmuted to helium; natural boron contains $\approx 20\%$ ^{10}B . Irradiation in HFIR at 300-500°C up to ≈ 30 dpa produced very little effect on the tensile properties (yield stress and total elongation). Tensile specimens irradiated in JMTR to 0.7 dpa and 120 appm He for the ^{10}B -doped steel had little effect on the yield stress, but there was an indication of a slight reduction in total elongation and reduction of area. Although the standard F82H and the F82H containing the ^{10}B addition had similar Charpy impact properties in the unirradiated condition, irradiation to 0.2-0.6 dpa at 250-350°C in JMTR produced a much larger shift in the Charpy transition temperature for the ^{10}B -doped (≈ 100 appm He) steel. At temperatures above $\approx 400^\circ\text{C}$, there was only a small difference in the Charpy behavior of the steels with and without ^{10}B . Microstructural examination of steels irradiated to 57 dpa in HFIR indicated $2 \times 10^{21} \text{ m}^{-3}$ (3 nm) cavities present in the ^{10}B -doped steel but none in the non-doped steel.

E. Materna-Morris reported on the effect of helium on steels after dual-beam irradiation and neutron irradiation in the HFR. MANET I hardened more than the F82H did during dual-beam irradiation to 0.3 dpa and 500 appm He. For irradiations in HFR at 300°C, a larger shift in Charpy transition temperature was observed for MANET I and OPTIFER II than for the ORNL 9Cr-2WVTa and F82H. Dual-beam irradiation of the F82H to 0.8 dpa and 300 appm He at 250°C produced a larger shift in the transition temperature than for a similar irradiation in HFR. The excess shift was attributed to helium. Likewise, to explain the relative Charpy transition temperature behavior of MANET I, OPTIFER II, F82H, and 9Cr-2WVTa (listed in order of decreasing transition temperature shift) after irradiation in HFR, the results were correlated with ^{10}B content, which transmuted to helium, although it was stated that the helium contribution to the shift in transition temperature cannot be determined quantitatively because it is not possible to separate helium and alloying effects. Scanning electron microscopy observations of relative amounts of cleavage and intergranular fracture on the fracture surfaces were correlated with the Charpy results (change in transition temperature).

A. Kimura discussed a small punch test procedure used to evaluate the effect of helium on the DBTT of 9Cr-2W steels. Disk specimens 3-mm in diameter and 0.22-mm thick were irradiated in a 36 MeV α -particle beam from a cyclotron. An energy degrader was used to uniformly implant 120 and 580 appm He (0.048 and 0.23 dpa) in the disk. Irradiation was at $<150^\circ\text{C}$. Hardness data were used to estimate a yield stress and yield stress increase ($\Delta\sigma_y$) during irradiation. Data from JMTR irradiations where no helium was present indicated that the shift in yield stress fit a $\text{dpa}^{1/4}$ law, which agreed with the results for the cyclotron-irradiated specimens, indicating no helium effect on hardening (just the effect of displacement damage). The data for the cyclotron-irradiated material fit the linear correlation between $\Delta\sigma_y$ and ΔDBTT obtained from the JMTR data, indicating that helium did not affect the shift in DBTT. From hardening changes during annealing, it was found that helium reduced the rate of recovery of the irradiation hardening, suggesting that helium stabilizes the defect clusters. TEM indicated that helium decreased the

size of the clusters but increased the number density. In this experiment, irradiation to 0.23 dpa and 580 appm He at <150°C did not affect irradiation hardening and embrittlement .

E. V. van Osch reported on work being started to study the post-irradiation welding of helium-containing steel at ECN in Petten. Neutron irradiated F82H plates of 1, 3, and 5 mm thickness that were irradiated in HFR to 2 dpa (≈ 5 appm He) and some 1 mm plates that were irradiated to 2.5 dpa were available for the study. The irradiated 1, 3, and 5 mm plates were successfully TIG welded to unirradiated plates with no external defects detected by SEM. The 1 and 3 mm plates were welded in a single pass with no filler metal, and the 5 mm plate contained a Y-groove and was welded with 4-6 passes. Further inspection of the welds is planned. Heats of steel have been ordered with ^{10}B , ^{11}B , and natural boron, so that it will be possible to generate various amounts of helium up to 250 appm He and higher.

Strategy for the Development of Ferritic/Martensitic Steels for Fusion

Presentations were made on the strategy for the development of ferritic/martensitic steels in Japan, EU, and the U.S. by A. Hishinuma, B. van der Schaaf, and F. W. Wiffen, respectively. The stated goal of this session was the development of a united strategy that could be presented by representatives from the Working Group (van der Schaaf and Hishinuma) to an IEA panel that was meeting in Copenhagen the following week, 5-9 October 1998, to consider a coordinated strategy for fusion materials development.

The strategies for Japan and the EU are pointed toward a DEMO using a martensitic steel, and this gives rise to dates for selecting a given material for the construction of the plant. Japan has a potential date of 2015 for selecting a material for DEMO, and the EU has a date for a DEMO-relevant design by 2009 based on conventional-type ferritic/martensitic steels. Should ODS steels be successfully developed, the date for a DEMO-relevant design for this material would be 2015.

In contrast to Japan and the EU, the U.S. has no plans for a DEMO and instead is involved in a science-based approach in which the technical program will emphasize, "enabling technologies for plasma experiments, domestic and internationally." The materials work will be targeted at developing materials that will support economically attractive, environmentally attractive, and safe fusion energy source designs.

A. Kohyama presented some further views on the Japanese strategy. He expressed concern about what should be done beyond the work presently being carried on the large heats of the IEA F82H and JLF-1 that are being studied in the IEA collaboration. He emphasized the need for a clear strategy for ferritic/martensitic steel development to be presented to the fusion community.

The discussion on the strategies of the various programs indicated that at present it appears there are common features in the strategies of the three programs, starting with the need to coordinate the materials development with the design and engineering community. The question of whether ferromagnetic structural materials are acceptable for magnetically confined fusion still needs to be

answered, as do questions on the effect of the simultaneous helium and displacement damage on properties (embrittlement). An expansion of the design window for ferritic/martensitic steels is desirable, and it is agreed that the ODS steels offer the best approach to achieve that goal by raising the operating temperature. This development needs to be pursued.

Other questions vital to the application of ferritic/martensitic steels to fusion include nuclear transmutations that will burn out elements of the steel (e.g., W, Ta, etc.), the effect of tungsten on the breeding ratio, compatibility issues and the need for barriers or other coatings for the steel. The urgent need for a 14 MeV neutron source was again emphasized.

Despite different objectives of the European Union, Japan, and the United States and given the time and financial constraints on the programs, the complexity of the common problems standing in the way of the three programs meeting their respective goals makes a coordinated effort of international collaboration by the three programs essential if their goals are to be achieved.

Action Items

No formal action items were set forth at this meeting. However, the following action item from the Tokyo meeting has not yet been completed:

Considerable work has now been completed on the IEA heats of ferritic/martensitic steels. Compilations of the work on the IEA heat of F82H by the Japanese and European Union are being prepared by K. Shiba and R. Lindau, respectively, who will consult on an exchange of reports and a distribution of the reports to other members involved in the IEA collaboration. In the future, a report summarizing the work being carried out in the European Union, Japan, and the United States will be prepared.

K. Shiba has agreed to continue this cooperative effort with R. Lindau. In addition, E. van Osch has expressed his interest to Shiba in participating in the effort.

On an informal basis, the EU approached Shiba requesting the JAERI irradiation matrix for F82H, so they can avoid duplication in their program. Shiba has this information in his data base, but he also agreed to prepare a hard copy and distribute it.

Other Information

Although it was not discussed formally in the meeting, the EU has ordered a 4000 kg heat of EUROFER 97, the EU reduced-activation reference steel for DEMO. Delivery is expected in the spring of 1999. Most of the ingot will be processed into plate to be used for the EU testing program for wrought and weld products. Tens of meters of tubes will be produced that will be used for welding trials and component mock-ups. Welds for testing will be made by fusion and HIP processes. There will be a limited number of forged bars, some of which will be atomized for

powder products that will be made by the HIP process for qualification of the process. There are plans to offer material to participants in the IEA program for evaluation.

Next Meeting

The next meeting of the Working Group will occur on one evening of the ICFRM-9 Conference in Colorado Springs, Colorado, USA, during the week of 10-15 October 1999. This meeting will serve as the planning meeting for the next workshop, which will be held in the fall of 2000.

Appendix

Viewgraphs and Handouts

Ferromagnetic Effects and Steel Properties

Forschungszentrum Karlsruhe
Technik und Umwelt

Paolo Ruatto
Institute for Neutron Physics and Reactor Technology

Effects of Ferromagnetic Structural Material on the Electromagnetic Behaviour of Fusion Reactor Components

Introduction

The Finite Element Method program AENEAS, developed at FZK to study transient eddy current problems as well as magnetic fields and forces in non-linear magnetic materials, has been applied to investigate the electromagnetic behaviour during a reference plasma disruption for

- ① the DEMO Helium cooled Pebble Bed (HCPB) outboard Blanket segment,
- ② the European Helium Cooled Pebble Bed (HCPB) Test Blanket Module in ITER

Plasma Disruption

When a plasma disruption occurs, eddy currents are induced in the conducting structures surrounding the plasma region.

Such currents interact with the magnetic field and cause an electromagnetic load (Lorentz forces) acting on the structure.

If the material is ferromagnetic - to determine the effective loading of the structure - attention has to be paid to:

- ① changes in the eddy current distribution,
- ② changes in the magnetic field in the structure,
- ③ additional contributions to the load due to direct interaction between the magnetic field and the magnetized material.

Plasma Disruption

$$\vec{F} = \vec{J}_w \times \vec{B}$$

Effect of Ferromagnetic Properties

$$\vec{B}_{\text{ferr}} > \vec{B}_{\text{nichtferr}}$$

$$\vec{B} = \mu_0 (\vec{M} + \vec{H})$$

$$\vec{J}_{\text{ferr}} \neq \vec{J}_{\text{nichtferr}}$$

$$\vec{C} = \vec{M} \times \vec{B} \quad \vec{F} = \vec{M} \cdot \nabla \vec{B}$$

DEMO HCPB Solid Breeder Blanket

Outboard Segment

Mathematical Formulation

$$\vec{H}(\vec{r}) \vec{J}(\vec{r}, t) + \frac{\mu_0}{4\pi} \int_{\Omega} \frac{1}{|\vec{r} - \vec{r}'|} \frac{\partial \vec{J}(\vec{r}', t)}{\partial t} d\tau' = - \frac{\partial \vec{A}_e(\vec{r}, t)}{\partial t} - \nabla \phi(\vec{r}, t) - \left[\frac{\mu_0}{4\pi} \int_{\Omega_M} \frac{\partial \vec{M}(\vec{r}', t)}{\partial t} \times \nabla' \frac{1}{|\vec{r} - \vec{r}'|} d\tau' \right] \quad (1)$$

$$\vec{M}(\vec{r}, t) = \chi(|\vec{H}(\vec{r}, t)|) \vec{H}(\vec{r}, t) \quad (2)$$

$$\vec{H}(\vec{r}, t) = \frac{1}{4\pi} \int_{\Omega} \vec{J}(\vec{r}', t) \times \nabla' \frac{1}{|\vec{r} - \vec{r}'|} d\tau' + \frac{1}{\mu_0} \nabla \times \vec{A}_e(\vec{r}, t) + \frac{1}{4\pi} \nabla \times \int_{\Omega_M} \vec{M}(\vec{r}', t) \times \nabla' \frac{1}{|\vec{r} - \vec{r}'|} d\tau' - \vec{M}(\vec{r}, t) \quad (3)$$

Numerical Formulation

$$[L] \frac{d\{I\}}{dt} + [R] \{I\} = \{V\} + \{U\} \quad (1')$$

$$\{M\} = [X] \{H\} \quad (2')$$

$$\{H\} = [H_r] \{I\} + \{H_e\} + [H_M] \{M\} \quad (3')$$

Time integration

$$[D]\{I\}_{w+1} = [D]\{I\}_w - \Delta t [R]\{I\} + \int_{t_w}^{t_{w+1}} \{V\} dt + \int_{t_w}^{t_{w+1}} \{U\} dt \quad (1'')$$

$$[D] = [L] + \omega \Delta t [R]$$

$\Delta t = t_{w+1} - t_w$, integration time-step

$0 \leq \omega \leq 1$, Crank-Nicolson-Parameter

$$\{I\}_{w+1} = ([1] - [A])\{I\}_w + [B](\{f\}_{w+1} - \{f\}_w) + [C](\{M\}_{w+1} - \{M\}_w) \quad (1''')$$

$$[A] = \Delta t [D]^{-1} [R], \quad [B] = [D]^{-1} [V_{koe}], \quad [C] = [D]^{-1} [U_{koe}]$$

Iterative Procedure

Zeit t_{w+1} , Iterationsschritt (i)

$$1) \{I\}_{w+1}^{(i)} = \{I_c\}_{w+1} + [C]\{M\}_{w+1}^{(i-1)},$$

$$2) \{H\}_{w+1}^{(i)} = \{H_e\}_{w+1} + [H_J]\{I\}_{w+1}^{(i)} + [H_M]\{M\}_{w+1}^{(i-1)} \\ = \{H_c\}_{w+1} + ([H_J][C] + [H_M])\{M\}_{w+1}^{(i-1)},$$

$$3) [X]_{w+1}^{(i)} = f(\{H\}_{w+1}^{(i)}),$$

$$4) \{M\}_{w+1}^* = [X]_{w+1}^{(i)}\{H\}_{w+1}^{(i)},$$

$$5) \{M\}_{w+1}^{(i)} = ([1] - [\beta]_{w+1}^{(i)})\{M\}_{w+1}^{(i-1)} + [\beta]_{w+1}^{(i)}\{M\}_{w+1}^*$$

$$\left\{ \begin{array}{l} [\beta]_{w+1}^{(i)} = ([1] - [X]_{w+1}^{(i)}([H_J][C] + [H_M]))^{-1} \\ \{M\}_{w+1}^{(i)} = [\beta]_{w+1}^{(i)}[X]_{w+1}^{(i)}\{H_c\}_{w+1} \end{array} \right. \quad \rightarrow \quad \beta = \frac{1}{1 + \chi_{\max}}$$

Forces Calculation

Lorentz-Kräfte:

$$\vec{F}_1 = \int_{\Omega} \vec{J}(\vec{r}, t) \times \vec{B}_0(\vec{r}, t) d\tau,$$

$$C_1 = \int_{\Omega} \vec{r} \times (\vec{J}(\vec{r}, t) \times \vec{B}_0(\vec{r}, t)) d\tau,$$

Magnetisierungskräfte:

$$\vec{F}_2 = \int_{\Omega_M} (\nabla \times \vec{M}(\vec{r}, t)) \times \vec{B}_0(\vec{r}, t) d\tau$$

$$+ \int_{\partial\Omega_M} (\vec{M}(\vec{r}, t) \times \vec{n}) \times \vec{B}_0(\vec{r}, t) dS,$$

$$C_2 = \int_{\Omega} \vec{r} \times [(\nabla \times \vec{M}(\vec{r}, t)) \times \vec{B}_0(\vec{r}, t)] d\tau$$

$$+ \int_{\partial\Omega_M} \vec{r} \times [(\vec{M}(\vec{r}, t) \times \vec{n}) \times \vec{B}_0(\vec{r}, t)] dS$$

$$\vec{K}_{M,12} = (\vec{M}_1 - \vec{M}_2) \times \vec{n}_{12}$$

$$\vec{f}_{12} = \vec{K}_{M,12} \times \vec{B}_{0,12}$$

$$\vec{J}_M = \nabla \times \vec{M}, \quad \vec{K}_M = \vec{M} \times \vec{n}$$

AENEAS

- 3D Finite-Element-Method Program
- Integral-Volume-Method
- Symmetry
- Magnetization Curve
- S.O.R. iterative Method
- Analytical Algorithms for magnetic field calculations
- Magnetization forces (EMC-Method)

Electromagnetic Analysis for DEMO

- A centered plasma disruption with a linear decay of the plasma current from 20MA to zero in 20ms has been considered.
- The structure is fully saturated by the strong toroidal magnetic field (6 tesla) and the magnetization $\mathbf{M}$ is directed almost entirely toroidally causing a thickening of the toroidal component of the magnetic flux density $\mathbf{B}$, whereas the influence of $\mathbf{M}$ on the poloidal component of $\mathbf{B}$ is slight.
- It means that when a plasma disruption occurs, the varying poloidal magnetic field of the plasma „sees“ a similar situation as the structure would be non-magnetic, so that behaviour and distribution of the eddy currents induced in the structure are approximately the same with or without ferromagnetic structural material.

Measurement of the Magnetization Curve for MANET Steel

$$H_{\text{sat}} = 24 \text{ kA/m}, \quad B_{\text{sat}} = 1.64 \text{ T}, \quad B_r = 1.15 \text{ T}$$

DEMO $\rightarrow$ 4800 kA/m !

Magnetic Flux Density Distribution

[T]

Poloidal Component B_p

Toroidal Component B_T

Electromagnetic Analysis for DEMO (continued)

- The electromagnetic forces (and couples) caused by interaction of the eddy currents ($\mathbf{J}$) with the magnetic field $\mathbf{B}$ (Lorentz forces $\mathbf{J} \times \mathbf{B}$) increase only their magnitude due to the contribution of the magnetization $\mathbf{M}$ to the toroidal component of $\mathbf{B}$, whereas their direction remains practically unchanged. In the most stressed part of the outboard segment box, like the side walls, the increase is about 11%. Considering the resultant forces and couples, we observe that a large torque acts on X-axis and a torsion in Z-direction.
- Additional forces and couples are present in the structure caused by direct interaction of $\mathbf{M}$ with $\mathbf{B}$. The resultant of such forces produces a stretching of the structure. This contribution, which is present only in a ferromagnetic structure even during normal operation, determines in case of a plasma disruption a completely different electromagnetic load for the DEMO outboard segment as in the case with non-magnetic structural material.

Electromagnetic Forces

Distribution of the Lorentz forces

	Non-magnetic Steel		MANET			
	Lorentz Forces		Lorentz Forces		Magnetization Forces	
O	$F_x = 1.1$	$C_x = 30.4$	$F_x = 0.8$	$C_x = 38.0$	$F_x = -3.9$	$C_x = -10.4$
	$F_y = -2.4$	$C_y = 7.1$	$F_y = -3.6$	$C_y = 10.2$	$F_y = 3.8$	$C_y = -16.1$
	$F_z = -0.2$	$C_z = 22.6$	$F_z = -0.7$	$C_z = 25.8$	$F_z = 13.7$	$C_z = -0.6$
U	$F_x = 1.0$	$C_x = 24.0$	$F_x = 0.7$	$C_x = 27.5$	$F_x = -2.4$	$C_x = 0.5$
	$F_y = 2.5$	$C_y = -4.9$	$F_y = 2.8$	$C_y = -6.1$	$F_y = -1.5$	$C_y = 9.8$
	$F_z = -0.1$	$C_z = -23.0$	$F_z = 0.9$	$C_z = -25.1$	$F_z = -5.7$	$C_z = -0.8$

Resultants of Forces (MN) and Couples (MNm)

Electromagnetic Analysis for TBM in ITER

Following critical issues have to be studied by means of an electromagnetic analysis:

- ① magnetized matter interacts directly with the external magnetic field causing a magnetic loading (Magnetization Force, MF) on the structure even during normal operation which has to be considered for the mechanical design of the TBMs;
- ② when a plasma disruption occurs, an electromagnetic load (Lorentz force, LF) due to interaction between eddy currents and the magnetic field is added to the structure; the value of this load and the effect of the magnetization on it have to be investigated to assess the capability of the TBM structure to withstand the mechanical effects of plasma disruptions;
- ③ the magnetized structure of the TBMs produces a non-axisymmetrical magnetic field in the plasma region which can affect plasma stability and lead to disruption.

ITER Test Blanket Module

Total FEM Model

Detail of TBM

Force Calculation for TBM in ITER

Electromagnetic force distributions have been calculated for the reference centered disruption (CD, 50ms) and for upward and downward VDEs (50ms).

Resultant forces and torques have been derived at the TBM support. If we consider normal components of force and torques for the 50ms CD, we have that:

- A pulling force of 0.2MN in the direction of the plasma center acts on the TBM even during normal operation;
- the resultant F_N of the LF distribution originating during the disruption achieves a maximum value of 0.12MN in positive normal direction, contributing to a reduction of the total F_N acting on the support. However, if we consider the spatial distribution of the forces on the TBM structure, we can observe that whereas the MFs act prevalently in the radial direction, the LFs are differently directed depending on the eddy current patterns in the structure. Therefore, during a disruption the loading of the structure can be locally higher than during normal operation (for example on the first wall).
- Considering the T_N we note that the only significant contribution is given from the LF (max. 0.8 MNm).

Force Calculation for TBM in ITER (continue)

- The effect of the magnetization on the LF (stronger magnetic flux density) has been evaluated carrying out the same electromagnetic analyses for a non-magnetic structural material. For the reference CD (50ms) peak values of the LF contribution have been calculated as $F_N=0.07$ MN and $T_N=0.65$ MNm.
- Upward and downward VDE have been also analysed, but from the table below – showing the maximum values of the total resultant forces and torques acting on the support for the reference CD (50ms) and upward and downward VDEs - it is evident that the most critical event for the TBM is represented by a centered plasma disruption.

Max Values	F_N (MN)	$ F_T $ (MN)	T_N (MNm)	$ T_T $ (MNm)
CD (50ms)	-0.20	1.14	0.72	0.32
Up. VDE (50ms)	-0.20	0.08	0.47	0.25
Down. VDE (50ms)	-0.20	0.12	0.49	0.22

Force Calculation for TBM in ITER

CENTERED DISRUPTION (50ms)

CENTERED DISRUPTION (50ms)

Error Field Calculation for TBM in ITER

The magnetic field produced by the magnetized matter of the TBM (error field) has been calculated to evaluate the effect of the TBMs on the toroidal magnetic field in the plasma region. Results are reported on the next slide:

- On the left, the magnitude of the error field is shown as a function of the toroidal angle in the plane $Z=1.7\text{m}$ and for different radius values.
- Toroidal magnetic field ripple values δ_{Fe} for the same plane are reported on the right and have been calculated following the expression

$$\delta_{Fe}(R, Z=1.7) = \frac{B_{\varphi}^{(max)}(R, \varphi^{(max)}) - B_{\varphi}^{(min)}(R, \varphi^{(min)})}{2B_{\varphi}(R)} \cdot 100\%$$

- At $R=10.947\text{m}$ - outer edge of the plasma - $\delta_{Fe} \cong 0.85\%$, but it can be observed that near the module, δ_{Fe} changes very rapidly with R . Changes in δ_{Fe} have also been obtained by varying the Z coordinate.

Error Field Calculation for TBM in ITER

Magnitude of Error Field ($Z=1.7\text{m}$)

Toroidal Magnetic Field Ripple ($Z=1.7\text{m}$)

Conclusions

- An electromagnetic analysis taking into account the ferromagnetic properties of the structural material is necessary for a correct mechanical design of the DEMO outboard blanket segment as well as of the TBMs in ITER.
- Furthermore, the real structure of the TBMs has to be included in the calculations to evaluate accurately the error field effect on the plasma.
- The Finite Element program AENEAS is an appropriate tool to fulfil these requirements.

PRESENT AND PAST RESEARCH ON THE
EFFECT OF FERROMAGNETIC MATERIALS
IN FUSION REACTORS:
A BRIEF REVIEW

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A STATUS REPORT ON THE REVIEW OF RESEARCH ON THE EFFECT OF FERROMAGNETIC MATERIALS IN FUSION REACTORS

R. L. Klueh

People who are familiar with this research area were contacted in the United States, Europe, and Japan. In our query to those experts, we asked for their assessment of the problem, what had been done in their organizations in the past, and what was being done at present. We have received four replies from Japan, two from the U.S., and one from Europe. A brief summary of what was obtained is given below.

Work in the United States

The first work in the world in this area appears to have been done in the United States in the early 1980s. Attaya et al. [1] considered the effect of a ferritic steel on the first wall, blanket, and coolant circuit of a tandem mirror machine. The field perturbations were found to be small and confined to the end region and on the same order of magnitude as the field ripples produced by the central cell magnets. Based on the calculations of the magnetostatic forces on a ferritic steel pipe in the magnetic field of the machine, the stresses were found to be small, but not negligible, and they must be incorporated in the stress analysis of the design.

Lechtenberg et al. [2], considered the ferromagnetic stresses on a coolant pipe in a tokamak (Starfire) from the toroidal field magnets. They concluded that the stresses were at such a level that they had to be accounted for, but they were not of the level that they were unmanageable.

Earlier work at GA [3], which was summarized by Lechtenberg et al. [2], considered the effect of a ferromagnetic material (Sandvik HT9) on the toroidal field ripple for three conceptual designs (GA TNS, ETF, and Starfire); the concern was that the ferromagnetic steel could introduce an additional ripple resulting in a loss of plasma energy or a plasma disruption. These studies concluded that due to the high level of saturation, the ferromagnetic blankets have an effective magnetic permeability close to unity and so do not cause any significant changes to the toroidal field ripple. Perturbations of up to 3% were found possible for the poloidal fields (calculated for Starfire), but these could be handled by normal plasma control techniques. From a simple force calculation based on simple cylinders in the magnetic fields, it was concluded that a "simple 'keystoned' blanket structure could be designed to accommodate this minor parasitic force."

The references for that early work are:

1. H. Attaya, K. Y. Yuan, W. G. Wolfer, and G. L. Kulcinski, in: Proceedings of Topical Conference on Ferritic Steels for use in Nuclear Energy Technologies, Eds. J. W. Davis and D. J. Michel (The Metallurgical Society of AIME, Warrendale, PA, 1984) p. 169.

2. T. Lechtenberg, C. Dahms, and H. Attaya, *ibid*, p. 179.
3. J. Rawis, W. Chen, E. Chung, J. Dillassandro, P. Miller, S. Rosenwasser and L. Thompson, *Assessment of Martensitic Steels as Structural Materials in Magnetic Fusion Devices*, General Atomic Report GA-A15749, January 1980.

In one of the replies to our queries to U.S. researchers, J. Blanchard indicated that not much has been done in the U.S. beyond the work of Attaya, Lechtenberg, and the workers at GA in the 1980s. Blanchard bases that on "a quick review of ferromagnetic effects for a system design study (ARIES)" that he carried out. His review resulted in a recommendation that a further study be carried out.

The only other information I received from the U.S. was a copy of an ITER Memo, "The Study of Ferromagnetic Inserts in the Vacuum Vessel (No. 1)," sent by J. Schmidt of PPL. The inserts were meant to reduce toroidal field ripple, and the report showed how this could be accomplished with the ferritic 430 stainless steel.

Both Blanchard and Schmidt expressed the importance of the problem and indicated that their organizations have the tools and personnel to carry out a detailed analysis of the problem should funds for such a project become available.

Work in Japan

Japan has an ongoing program, which is being carried out at JAERI and Hitachi (in an early e-mail message from Kohyama, he mentioned that work might be going on at the universities, but I have not received anything yet on that work). The Japanese work includes the installation of an F82H liner in a small tokamak. Some of this work was reported in the following papers:

1. S. Takeji et al., "Error Field Analysis of Steady State Tokamak Reactor with Ferromagnetic (F82H) Blanket," in: *Proc. 16th IEEE/NPSS SOFE*, Vol. 2, 1995, p. 1214.
2. M. Sato et al., "Analysis of Error Field due to Ferritic Steel in the Advanced Material Testing Program of JFT-2M," *J. Nucl. Mater.*, to be published.
3. T. Nakayama, et al., "Evaluation of Magnetic Fields due to Ferromagnetic Vacuum Vessel And their Influence on Plasma Discharge in Tokamak Devices," *J. Nucl. Mater.*, to be published.

The first of these papers analyzes the error fields generated by a ferromagnetic steel blanket for the conceptual design of a steady state tokamak reactor. The second paper is a computational comparison of the case of a vacuum vessel of non-magnetic material with a "ferritic board" [which I think means ferritic steel liner] and one made entirely of ferritic steel. The analysis involves the ripple amplitude reduction, and they show that the results are the same for both types of vacuum vessels.

For the third paper, the magnetic fields due to the ferromagnetic vacuum vessel in the Hitachi Tokamak HT-2 were calculated and determined experimentally to verify that plasma discharge is possible with a ferritic steel vacuum vessel. The results were extrapolated to JFT-2M and ITER. Plasma discharge was possible for the HT-2, and the effect was concluded to be smaller for the larger tokamak devices, indicating plasma discharge should be possible for the larger vessels.

Work in Europe

A list of sixteen papers/reports was received from P. Ruatto of Karlsruhe, who is working with L. Boccaccini. Ruatto later sent me five papers—the most important ones—that contained the main substance of their work. Their goal appears to be an assessment of the mechanical loading on the DEMO reactor components during plasma disruption. The following are the titles of the publications Ruatto sent:

1. L. V. Boccaccini et al., "Analysis of Mechanical Effects Caused by Plasma Disruptions in the European BOT Solid Breeder Blanket Design with MANET as Structural Material," Third International Symposium on Fusion Nuclear Technology, Los Angeles, CA, June 27-July 1, 1994, Fusion Engineering and Design 27 (1995) Part A, 407.
2. P. Ruatto, "Computation of the Generation and Impact of Electromagnetic Fields on Ferromagnetic Structures Surrounding the Plasma in a DEMO Fusion Reactor," ISEM 95 International Symposium on Non-Linear Electromagnetic Systems, Cardiff, UK, September 17-20, 1995, in: *Nonlinear Electromagnetic Systems*, A. J. Moses and A. Basak, eds. (IOS Press, Amsterdam, 1996) 835.
3. L. V. Boccaccini and P. Ruatto, "Effect of the Presence of Ferromagnetic Structural Material on the DEMO Helium Cooled Pebble Bed Blanket During Plasma Disruptions," Fusion Technology (1997) 1519.
4. T. Takagi, J. Tani, P. Ruatto, B. Krevet, and L. V. Boccaccini, "Numerical Evaluation of Natural Frequency Change of a Ferromagnetic Plate Due to Magnetic Stiffness Effect," IEEE Transactions on Magnetics, 32 (3) (1996) 1054.
5. P. Ruatto, "Development of a Numerical of a Numerical Method to Compute Electromagnetic Forces Generated by Magnetic Field Changes in Magnetic Structures and Application of this Method to a Plasma Disruption in a Tokamak Fusion Reactor," Forschungszentrum Karlsruhe, Wissenschaftliche Berichte, FZKA 5683, March 1996 (in German)

According to Ruatto, the early European work involved a code that could not consider nonlinear magnetic materials. A new code was developed by Ruatto that has been used and is being used for the analyses discussed in the publications. Until now they have not considered the influence of the ferromagnetic material on the error field of the plasma, but they are presently getting ready to study that problem.

HARAI Workshop
October 1-2 1998
ECN Petten, Netherlands

Fatigue Properties of Low Activation Ferritic Steel; *JLF-1 and Its Welded Joints*

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Background and Objective

Institute of Advanced Energy

Kyoto University

JLF-1 steel (Fe-9Cr-2W low activation ferritic steel):

- one of the steels for the IEA test program
- provided from the Japanese university activity, JUPITER program

First large heat:

- 1 metric ton, produced in 1986, plate thickness of 7 and 15 mm
- the mission is to obtain data sets of basic mechanical properties with and without radiation damage; neutron and charged particle irradiation.

Second large heat:

- 1.5 metric ton, produced in 1995, plate thickness of 15 and 25 mm
- the mission is to investigate joining technology and to evaluate structural integrity on TIG and EB welded joints (fatigue, creep, fracture toughness, etc.,)

Objective:

- to develop the mini-sized fatigue test techniques and to clarify the specimen size effects on the fatigue properties.
- to develop the strain controlled fatigue test techniques without contact on the specimen.

Fabrication of Specimen for Fatigue Test

Institute of Advanced Energy

Kyoto University

plate thickness: 25mm

Dimension:mm

Series **PMHT**
(Post Machining Heat Treatment)

- PMHT-0 No heat treatment
- PMHT-1 $600^{\circ}\text{C} \times 1\text{h/F. C.}$
- PMHT-2 $740^{\circ}\text{C} \times 3\text{h/F. C.}$

Strategy to Evaluate Fatigue Properties Under irradiation

Institute of Advanced Energy

Kyoto University

Fatigue Specimen Used

Institute of Advanced Energy

Kyoto University

ASTM recommended

$$\frac{R}{d} \approx 8$$

Full-sized fatigue specimen

Power Reactor and Nuclear Fuel Development Corporation

Miniature-sized fatigue specimen

II-MIF-CDA

Experimental Procedure (2)

Institute of Advanced Energy

Kyoto University

Push-Pull

Scheme of radius strain measurement system

Wave form

- Hydraulic servo controlled testing machine
Maximum load: 10tf
- Wave form : sine
- Stress ratio : $R=-1$
- Frequency : 1Hz
- Total radius strain controlled

Experimental Procedure

Institute of Advanced Energy

Kyoto University

Facilities

Pre-irradiation

Institute of Advanced Energy,
Kyoto University

Post-irradiation

Oarai Branch of Institute for Material Research,
Tohoku University

In-situ

TBD

Test Machine

Full-sized fatigue specimen

- Hydraulic servo controlled testing machine
Maximum load: 10tf

Mini-sized fatigue specimen

- Electrical testing machine
Maximum load: 200kgf

Welding Conditions

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Kyoto University

Tungsten insert gas (TIG) welding

Conditions

CURRENT	230~250A
Voltage	10.5V
Travel speed	10cm/min
Heat input	14.5~15.8kJ/cm
Preheat temp	$\leq 200^{\circ}\text{C}$
Interlayer temp	$\leq 200^{\circ}\text{C}$
Number of passes	≈ 20
Wire diameter	1.2mm

PWHT $740^{\circ}\text{C} \times 3\text{h/F. C.}$
(Post Welding Heat Treatment)

Groove

Macrostructure

Results of Radius Strain Controlled Fatigue Test

Institute of Advanced Energy

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Cyclic Stress Response Curves / Base Metal

Institute of Advanced Energy

Kyoto University

Cyclic Stress Response Curves / Weld Metal

Institute of Advanced Energy

Kyoto University

Fatigue Life Curve (S-N Curve)

Institute of Advanced Energy

Kyoto University

Correlation between Fatigue Limit and Tensile properties

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Kyoto University

Fatigue limit (σ_{wo})

Correlation

Tensile strength (σ_B)

Vickers hardness (H_V)

$$\sigma_{wo} \approx 0.5\sigma_B$$

$$\sigma_{wo} \approx 1.6H_V \pm 0.1H_V \quad (H_V \leq 400)$$

Result of tensile strength and Vickers hardness distribution

Fatigue Life Curve (S-N Curve)

Institute of Advanced Energy

Kyoto University

Prediction of Fatigue Limit

Institute of Advanced Energy
Kyoto University

Prediction of fatigue limit
using following formula

$$\sigma_{WO} \approx 0.5\sigma_B$$

Number of cycles to failure, Nf

*The predicted value from
Vickers hardness and tensile
strength was very accurate*

Effect of Specimen Size on Fatigue Properties

Institute of Advanced Energy

Kyoto University

Speci. Size effect
small

Slightly decline of
Fatigue life curve
Slight shift to
higher stress

Effect of Post Machining Heat Treatment on

Fatigue Life

Institute of Advanced Energy

Kyoto University

- *Fatigue life is not influenced by the three types of heat treatment*
- *Effect of residual stress and strain due to machining on fatigue life can be neglected*

Conclusion

Institute of Advanced Energy

Kyoto University

- *A miniature fatigue specimen testing machine for a hot-laboratory has been successfully developed and the specimen size effect was studied. So far, there is no significant specimen size effect on fatigue properties except at very low cycle fatigue region.*
- *Fatigue life of JLF-1, TIG weld metal was longer and the fatigue strength was larger than those of base metal.*

-

IEA WORKSHOP ON REDUCED-ACTIVATION F/M STEELS
ECN, Petten, The Netherlands
1-2 October 1998

MECHANICAL BEHAVIOUR AFTER THERMAL AGEING
OF RAFM STEELS AND F82H WELDMENTS

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Commissariat à l'Energie Atomique
Service de Recherches Métallurgiques Appliquées
CEA - Saclay, France

MATERIALS

Chemical composition of LA martensitic steels (in wt%).

Steel	C	Si	Mn	Cr	V	W	N	Ta
LA12Ta	0.155	0.03	0.88	9.86	0.28	0.84	0.0430	0.10
LA12TaLN	0.165	0.02	0.84	9.04	0.24	0.75	0.0048	0.10
LA12TaLC	0.090	0.03	1.13	8.80	0.30	0.73	0.0190	0.10
LA12LC	0.089	0.03	1.13	8.92	0.30	0.73	0.0350	0.010
LA4Ta	0.142	0.03	0.78	11.08	0.23	0.72	0.0410	0.07
LA13Ta	0.179	0.04	0.79	8.39	0.24	2.79	0.0480	0.09
F82H	0.087	0.10	0.21	7.46	0.15	1.96	0.0059	0.023
JLF-1	0.106	0.05	0.52	8.70	0.18	1.91	0.0280	0.08

LA.... alloys, F82H and JLF-1 steels have been supplied respectively by AEA-Culham (U.K.), JAERI and Tokyo University (Japan).

Types of materials :

- 9Cr - 0.8W - V - Ta : LA12Ta (high C and N), LA12TaLC (low C, Ta), LA12LC (low C, no Ta), LA12TaLN (low N).
- 11Cr - 0.8W - V - Ta : LA4Ta (high Cr)
- 9Cr - 3W - V - Ta : LA13Ta (high W)
- 7.5/9Cr - 2W - V - Ta : F82H (low Cr, low Ta), JLF-1

METALLURGICAL CONDITION

Experimental LA-steels have been produced as plates of 3.5 mm thick. The last steps of the fabrication route consisted of normalisation and tempering treatments followed by a final cold-working pass. This condition is denoted « **N&T - CW** ».

Normalisation : 1030°C - 40 minutes

Tempering : 750/790°C - 1h

Final CW : 10%

Large-scale heats, F82H and JLF-1, have been supplied as plates of 7.5 mm and 15 mm thick in the normalised and tempered condition, denoted « **N&T** ».

Normalisation : F82H : 1040°C - 40 minutes

JLF-1 : 1050°C - 1h

Tempering : F82H : 750°C - 1h

JLF-1 : 780°C - 1h

F82H - TENSILE PROPERTIES AFTER AGEING

IMPACT PROPERTIES AFTER AGEING - EFFECTS OF CHEMICAL COMPOSITION

IMPACT PROPERTIES AFTER AGEING - EFFECTS OF CHEMICAL COMPOSITION

IMPACT PROPERTIES AFTER AGEING

F82H WELDMENTS

Profiles of Cr and Mn concentrations measured on the weld cross-section (mid-thickness). TIG specimen KG820-1-31W-37 (heat n° 9753) of 25 mm thick.

F82H WELDMENTS

Hardness values

(HV₁₀, average of 10 measurements)

Welds	Welded Metal	Heat Affected Zone	Base Metal
TIG	221	223	210
EB	390	400	214

Soudage FE monopasse de l'acier F82H JAERI (4-5)
Duretés dans l'axe transverse à la zone fondue

F82H WELDMENTS

Creep and tensile specimens

creep : 4 mm dia. , $l_0 = 20 / 25$ mm

tensile : 2 mm dia., $l_0 = 12$ mm

F82H WELDMENTS

Creep properties - Time to rupture

F82H WELDMENTS

(15 mm thick)

1 mm

longitudinal metallography showing the rupture location for TIG weld

TENSILE PROPERTIES OF F82H WELDMENTS

TIG

F82H - TIG 15 ET 25 mm
sens transversal

F82H WELDMENTS

Charpy V specimens - TL orientation

(55 mm long, 10 mm wide, 2.5 mm thick)

F82H weldments - 15mm thick TIG joints thermal aged for 10000h

F82H weldments - 15mm thick EB joints thermal aged for 10000h

UNIRRADIATED PROPERTIES OF F82H IEA HEAT

- Status and Recent Results -

K. Shiba/JAERI

IEA Working Group Meeting
on
Reduced Activation Ferritic/Martensitic Steels

1-2, October, 1998

ECN Petten, The Netherlands

JAPANESE IEA ROUND ROBIN TEST PROGRAM ON F82H

1. Metallurgical tests (Base Metal) (1995 - 1996)

Test	Schedule	Remarks
Macro/Micro structure	1995	Optical, SEM, TEM microscopy
Hardness	1995	
CCT curve	1995 - 1996	

2. Mechanical properties (Base Metal) (1995 - 1997)

Property	Test condition	Schedule	Remarks
Tensile	RT-600°C	1995 - 1996	
Charpy Impact	Full curve	1995 - 1996	
Fracture toughness(K_{IC} , J_{IC})	RT, 100, -30°C	1995 - 1996	
Creep			
Creep Rapture	500 - 650°C	1995 - 1997	max: 50000h
Creep Rate	500 - 650°C	1995 - 1997	max: 3000h
Creep Curve	500 - 650°C	1995 - 1997	max: 3000h
Creep Fatigue	500 - 650°C	1995 - 1997	holding time: 0,1,3,10,30min
Fatigue	RT - 600°C	1995 - 1997	ϵ : 0.5 - 1.5%(push/pull)

3. Aging tests

Aging condition: (400), (500), 550, 600, 650°C; 1000, 3000, 10000, (30000) h

	Property	Test condition	Schedule
Metallurgical tests			1995 -
Mechanical properties	Tensile	RT, 550°C	1995 -
	Charpy Impact	Full curve	1995 -
	Fracture toughness(K_{IC} , J_{IC})	RT, 100, -30°C	1995 -

4. Weldments (TIG/EB)

Metallurgical tests

Test	Schedule	Remarks
Macro/Micro structure	1995	Optical, SEM, TEM microscopy
Hardness	1995	
CCT curve	1995 - 1996	

Mechanical properties

Property	Test condition	Schedule	Remarks
Tensile	RT-600°C	1995 - 1996	max: 10000h ±0.5 - 1.0%(push/pull)
Charpy Impact	Full curve	1995 - 1996	
Fracture toughness(K_{IC} , J_{IC})	RT, 100, -30°C	1995 - 1996	
Creep Rapture	500 - 600°C	1996 - 1998	
Fatigue	550°C	1995 - 1997	

Aging tests

Aging condition: 550, 600, 650°C; 1000, 3000, 10000 h

	Property	Test condition	Schedule
Metallurgical tests			1995 -
Mechanical properties	Tensile	RT, 550°C	1995 -
	Charpy Impact	Full curve	1995 -
	Fracture toughness(K_{IC} , J_{IC})	RT, 100, -30°C	1995 -

5. Physical properties (1995 - 1996)

Property	Temperature	Schedule
Density	RT - 700°C	1995
Specific Heat	RT - 700°C	1995
Thermal expansion	RT - 700°C	1995
Thermal conductivity	RT - 700°C	1995
Electrical conductivity	RT - 700°C	1995
Melting point		1995
Young's modules	RT - 700°C	1995
Poisson ratio	RT - 700°C	1995
Modules of rigidity	RT - 700°C	1995
Magnetic hysteresis	RT - 800°C	1995 - 1996

6. Other properties

Property	Measurment	Schedule	Remarks
Vacuum Properties	Released gas measurement	1995 - 1996	Corrosion loss SSRT test Hydrogen solubility after aging SSRT test
Corrosion	Corrosion test	1995 - 1996	
Resistance	In high temperature water	1995 - 1996	
Hydrogen Test	Tritium permeability	1995 - 1996	
	Hydrogen cracking	1995 - 1996	SSRT test

PROGRESS IN ROUND-ROBIN TEST

- TOPICS -

1. Base Metal Properties

- as N&T
 - Fatigue, Fracture Toughness, Tensile (low temp)
- Aging
 - Tensile, Charpy, Fracture Toughness

2. Welded Joints (TIG/EB)

- as Weld (PWHT: 720°Cx1h)
 - Hardness, Tensile, Charpy, Fracture Toughness
- Aging in progress (500-650°C/up to 100000h)
 - Hardness, Tensile, Charpy, Fracture Toughness

(Optional) Magnetic Properties (as N&T)

- presented previously

Fracture Toughness of F82H IEA heat

Tensile Properties of F82H IEA heat

BASE METAL PROPERTIES (as N&T)

- FATIGUE

Test Conditions

Strain rate: 0.1%/s (saw wave)

Test Temp (°C)	Total Strain Range (%)	Cycles to Rupture	Remarks
RT	0.6	10504	
	1.0	3028	
	1.5	923	
300	0.5	14726	Buckled
	0.5	9875	Retest
	0.75	2814	
	0.75	4831	Retest
	1.0	1597	
	1.5	746	Buckled
550	0.5	3850	
	0.75	1860	
	1.0	880	
	1.0	987	Retest
	1.5	247	

FATIGUE TEST RESULTS OF F82H IEA HEAT (RT Test)

FATIGUE TEST RESULTS OF F82H IEA HEAT (300°C Test)

FATIGUE TEST RESULTS OF F82H IEA HEAT (550°C Test)

FATIGUE TEST RESULTS OF F82H IEA HEAT

- Detailed data on fatigue results are available from WWW homepage (Database)
<http://realab01.tokai.jaeri.go.jp/lafdb01/fmain.htm>

Magnetic Property of F82H IEA heat

BASE METAL PROPERTIES (Aging)

Current Status of Aging Tests

Aging Condition \ Test Items		Microstructure		Precipitation Analysis	X-ray Analysis	SEM (EDX, etc.)	TEM	Hardness	Tensile (RT, 550°C)	Charpy (full curve)	Fracture Toughness (J-R)
		X100	X400								
Base Metal		○	○	○	○	○	○	○	○	○	○
1000h	500°C	○	○	—	—	—	—	○	○	○	—
	550°C	○	○	—	—	—	—	○	○	○	—
	600°C	○	○	○	○	○	—	○	○	○	—
	650°C	○	○	○	○	○	—	○	○	○	—
3000h	500°C	○	○	○	○	○	—	○	○	○	—
	550°C	○	○	○	○	○	—	○	○	○	—
	600°C	○	○	○	○	○	—	○	○	○	—
	650°C	○	○	○	○	○	○	○	○	○	—
10000h	500°C	○	○	○	○	○	—	○	○	○	—
	550°C	○	○	○	○	○	—	○	○	○	—
	600°C	○	○	○	○	○	—	○	○	○	—
	650°C	○	○	○	○	○	○	○	○	○	—
30000h	500°C	Aging in progress									
	550°C										
	600°C										
	650°C										
100000h	500°C	Aging in progress									
	550°C										
	600°C										
	650°C										

○: finished, △: in progress, —: not scheduled

Tensile Properties of Aged F82H IEA heat

F82H IEA Heat after 10000H Aging

Charpy Impact Test Results of Thermal Aged F82H IEA Heat

Chemical Analysis of Precipitation in Thermal Aged F82H IEA Heat

Chemical Analysis of Precipitation in Thermal Aged F82H IEA Heat

Thermal Aging Condition of F82H IEA Heat (TTP Diagram)

PROPERTIES OF WELDMENTS (as weld)

TIG weldments

Method: Narrow Gap Oscillating Electrode TIG

(a) Groove (15 mm)

(b) Layers

EB weldments

Method: I-groove welding (without filler metal)

○ Post-Welding Heat Treatment (PWHT):

720°Cx1h for both weldments

○ Thermal aging is in progress

500-650°C/up to 10000h

TIG WELDED JOINT

TIG Welded Joint (1W-4: 15 mm)

HAZ

EB WELDED JOINT

EB Welded Joint (4-21: 15 mmt)

Tensile Properties of F82H IEA heat TIG welded Joint

Tensile Properties of F82H IEA heat TIG and EB welded Joint

Fracture Toughness of F82H IEA heat (1"CT, RT test)

PROPERTIES OF WELDMENTS (as weld)

Summary

		TIG		EB	
		15 mm	25 mm	15 mm	25 mm
WM		10-15 mm	10-20 mm	2-8 mm	3-10 mm
HAZ		2-5 mm	3-10 mm	1-3 mm	2-4 mm
Total		20 mm	23 mm	12 mm	14 mm
Hv max		275	285	280	300
Hv min		187	200	210	210
RT/WJ	YS	510 MPa	470 MPa	474 MPa	520 MPa
	UTS	629 MPa	605 MPa	605 MPa	640 MPa
	FS	1947 MPa	1475 MPa	1455 MPa	1464 MPa
	Eu	4 %	4 %	7 %	5 %
	Et	16 %	14 %	20 %	19 %
	RA	82 %	77 %	78 %	77 %
RT/J _Q	HAZ	-	800 kJ/m ²	-	550 kJ/m ²
	WM	-	4020 kJ/m ²	-	-
DBTT	HAZ	-87°C	-82°C	-44°C	-38°C
	WM	12°C	-6°C	-	-

- EB weld has almost the same properties as BM
- TIG has hard/brittle WM and annealed HAZ
- Both weld has the same or better fracture toughness than BM at room temperature

PROGRESS IN ROUND-ROBIN TEST

- SUMMARY -

Base Metal

- Fracture toughness in brittle region was $\sim 90-100 \text{ MPam}^{1/2}$.
- Reduction of area and fracture stress changes in brittle region.
- Cyclic softening occurs at any temperature in fatigue test.

Aging

- 500°C aging has almost no effect till up to 10000h.
- Tensile strength decreases by high-temp aging.
- Both decrease in USE and shift in DBTT occurs by high-temp aging.

Weldments

- EB weld has almost the same properties as BM.
- TIG has hard/brittle WM and annealed HAZ.
- Both welds have the same or better fracture toughness than BM at room temperature.

Development of SPS Bonding Procedure for XXXXXXXXXX RAF Steel

JAERI A. Hishinuma

Background:

- The RAF steel is the first candidate material for DEMO and beyond.
- Fabrication techniques are important for fusion application as well as material properties themselves.
- HIP (Hot Isostatic Pressing) Bonding is an important techniques for the blanket construction
- And then, development studies for the HIP bonding techniques have been executed.
- Optimum conditions of the HIP bonding for RAM steels are:
150 MPa x 1040 °C x 2 hr , (too high, see fig.1)

and tempering condition:

740 °C x 2 hr

- Problems: HIP temperature is too high from a viewpoint of phase instability (fig.1)

Fig.1 Relation between grain size number and holding time of HIP-equivalent heat treatment

SPS (Spark Plasma Sintering) Bonding

Feature of the SPS process

To make possible sintering and sinter-bonding at low temperature and short periods by charging the intervals between powder particles with electrical energy and effectively applying a high temperature spark plasma generated momentarily.

PAS粉体粒子間現象
PAS Phenomenon, caused between
Powder particles

SPS system configuration

Experiments procedures

1. Specimens used:

F82H

size: $20\text{mm } \phi \times 7\text{mm } h$ or $15\text{mm } h$

roughness: 0.2, 16, 40 μm

2. SPS conditions;

Pressure: 20 - 50 MPa

Temperature: 800, 850, 900°C

Holding time: 300 - 3600 sec.

Vacuum: 1×10^{-2} torr

4. Tensile test

As SPS

PSHT(Post SPS-bonding Heat Treatment): 740°C x 40 min

Tensile specimen: SS-3 type

Temperature: Room temperature – 600°C

Strain rate: 5×10^{-3} sec

Vacuum: 1×10^{-5} torr

W1 = 1.52 mm

W2 = $1.52^{+0.025}_{-0.038}$ mm

Fig.1 Optical microstructures of the SPS bonding area of F82H bonded under conditions of (a),(b) 800°C (c),(d) 850°C and (e),(f) 900°C for 1 hr under pressure of 20 MPa.

Fig. 2 Stress-strain curves tested at room temperature for joints of F82H SPS-bonded at 800 and 900°C for 1 hr under pressure of 20 MPa with that of the base metal.

Summary

- Preliminary experiments for PSP bonding techniques have been carried out in order to examine a possibility to fusion devices
- PSP bonding techniques can be applicable to fabrication of blanket structure.
- An advantage of the SPS bonding compared with the HIP bonding techniques, is that the bonding can be done at lower temperature and short periods, without changing any microstructures.

Irradiated Properties and Composition Effects

PRESENT STATUS OF JAERI IRRADIATION PROGRAM FOR LAM

K. Shiba/JAERI

IEA Working Group Meeting
on
Reduced Activation Ferritic/Martensitic Steels

1-2, October, 1998

ECN Petten, The Netherlands

IRRADIATION PROGRAM IN JAERI

©Japan/US Collaboration

- HFIR target Irradiation
- HFIR RB Irradiation
- LAF research started from Phase 2 (1987-1995)
- Currently Phase 3 (1995-1999)
- Continue to Phase 4 (1999-2005)

©JAERI Domestic Program

- JMTR/JRR-2/JRR-3/JRR-4 Irradiation
- Accelerator Irradiation
(TIARA dual/triple-beam irradiation)

JAPAN/US COLLABORATION

HFIR Phase 2 Irradiation

- Basic Irradiation Behavior on Mechanical Properties/Microstructure
- F82H/F82H+¹⁰B(Helium effects)/⁵⁴Fe-F82H(Hydrogen effects)
- Tensile/TEM
- 200/250/300/400/500/600°C: 2.5-57 dpa

HFIR Phase 3 Irradiation

- Irradiation Behavior on Fracture Mechanism
- F82H IEA heat/F82H+2%Ni
- Tensile/Charpy/Fracture Toughness/Creep
- 300/500°C: 8-20 dpa
- Irradiation of 1 capsule has finished. Waiting for PIE.

HFIR Phase 4 Irradiation

- Details are in discussion

JAERI REACTOR IRRADIATION

Preliminary Irradiation Data on F82H

- F82/F82H/F82H+¹⁰B
- Tensile/Charpy
- 250-600°C: 0.05-0.9 dpa

Advanced Research on Irr. Behavior of LAF

- F82H IEA/F82H+¹⁰B
- Tensile/Charpy/Fracture Toughness/Fatigue
- 250-350°C: 0.5-2 dpa

JAERI REACTOR IRRADIATION

Scheduled JAERI Reactor Irradiation

JRR-3: - 250°C/2dpa (Apr.1998 – Mar.2000)
- F82H IEA
- Fatigue/Tensile (irr. effects on fatigue)

JMTR: - 250°C/1.8 dpa (Apr.1999 – Mar.2000)
- F82H IEA/F82H+¹⁰B
- Tensile/Charpy/Fracture Toughness
(correlation between properties)

JRR-3: - 300-400°C/1 dpa (Apr.1999 – Mar.2000)
- F82H/F82H/IEA/F82H+¹⁰B/F82H+2%Ni
- TEM/Tensile
(reliability of helium simulation)

JMTR: - ?? °C/1.8 dpa (Apr.2000 – Mar.2001)
- In planning

Tensile Properties of F82H after HFIR Irradiation

F82H Tensile Test Results (Yield Stress)

Microstructure of F82H (HFIR Irradiation: 250°C 2.5dpa)

Bright Field Dislocation Image

Weak-Beam Dark Field Image (g/3g)

OTHER RESEARCH ON RAF/MS IN JAERI

AMTEX Program (JFT-2M)

- Research on the Availability of Ferritic/Martensitic Material in Plasma Device
- Computer Simulation and Preliminary Experiment using JFT-2M were conducted in this year.

Subjects:

- Ripple reduction in the ITER
- Availability of ripple-less plasma operation with RAF/MS vacuum vessel
- Research on the undesirable effects for plasma production, control, etc.

Advanced Material Tokamak Experiment (AMTEX) Experimental Plan

1998	1999	2000	2001	2002	2003
Design for AMTEX and preliminary experiment (Including V-alloy Research)	Reduction of toroidal field ripple				
	Reactor Physics R&D issue: reduce the ripple loss of high energy particles in a reactor scenario				
	Install the ferritic boards on JFT-2M				
	Ferritic steel-plasma matching test Reactor Physics R&D issue: verify the validity of ferritic steel as the reactor structural material from the view-points of plasma confinement install the ferritic boards inside the vacuum vessel(VV), and then replace the VV with ferromagnetic VV				

Computational Results[3,5]

Insertion of Ferrite Board near VV

Ripple magnitude (δ) is reduced over the whole plasma region.

The structure of the magnetic field (B_t) is extremely modified. Toroidal mode number (n) of the TF is about twice the number of TFC in the outer region. [3]

$$D_{rp_{cl}}^{rp} \sim n^5 \delta^{4.5} \text{ or } n^4 \delta^4 [4]$$

⇒ Ripple loss depends on only δ and n .

$D_{rp_{cl}}^{rp}$: Diffusion coefficient of ripple trapped particle in collisionless region

⇒ In order to obtain the guide line for reduction of higher mode ripple, δ is analyzed by Fourier expansion.

[3] M. Sato, et al., at ICFRM-8, Sendai, Japan, 1997, P2-C060.

[4] P. N. Yushmanov, Reviews of Plasma Physics 16, 1990, p117.

In order to reduce both $n=16$ mode and higher mode, wider and thicker FB is better be located at further position from plasma[5]

Dependence of δ_{16} on R

(a) FB $H=1\text{m}$, $W=0.15\text{m}$, $Bt=2.2\text{T}$, $R=1.65\text{m}$

Dependence of δ_{32} on R

(b) $H=1\text{m}$, $W=0.15\text{m}$, $Bt=2.2\text{T}$, $R=1.65\text{m}$

[5] M. Sato, et al., at 20th SOFT, Marseille, France, 1998, P2-137.

Summary

Computational results

- In order to reduce fast ion losses
→the ripple magnitude (δ) have to be reduced and its higher toroidal mode number have to be suppressed.

Guideline of the design for reduction of δ and higher mode
(Number of toroidal field coil =16)

- Wider and thicker FB is better to be located at further position from plasma.

Prospect of Ferritic Steel

- Even if a good new material will be used for a demo fusion reactor, the ferritic steel have a big possibility to use in the reactor.
- ←Because the ferritic steel will be used for the vacuum vessel and ripple reduction.

SUMMARY

- Japan/US collaboration will be extended to 2005 (HFIR Phase 4).**
- Some HFIR Phase 3 capsules has been finished the irradiation and waiting for PIE.**
- Several JAERI reactor capsules are planned for fracture mechanics research.**
- AMTEX program is conducted to research the availability of RAF/M steels in plasma device.**

Post Irradiation Properties of F82H plate, TIG and EB welds

IEA-RAFM Workshop
Petten, 1-2 October 1998

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30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Outline

- Program overview
- Recent results
- Future work
- Conclusions

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Status F82H Irradiation Programme

capsule	300°C	Type	Irradiation	testing	contents
ILAS-4	2.5 dpa	Tensile	Completed	Completed	Plate
ILAS-6	2.5 dpa	Tensile, LCF	Completed	In progress	EB15, TIG15, EB15-CEA, T91-EB15-CEA.
ILAS-7	10 dpa	Tensile	In progress	-	EB15, TIG15.
ILAS-8	2.5 dpa	Tensile	Completed, cooling	-	EB25, TIG25, EB25- Schelde
CHARIOT-2	2.5 dpa	CT, KLST	Completed	Completed	Plate
CHARIOT-4	2.5 dpa	CT, KLST	Completed, cooling	-	T91-EB15-CEA, plate.
CHARIOT-5	2.5 dpa	CT, KLST	Completed, cooling	-	TIG15, EB25, EB15, EB15- CEA
CHARIOT-6	10 dpa	CT, KLST	In progress	-	Plate.
CHARIOT-7	10 dpa	CT, KLST	In progress	-	TIG15, EB15, EB25.

Programme 1998: FTT and Impact tests of reference and 2.5 dpa irradiated,
LCF and remainder tensile tests 2.5 dpa irradiated.

30-Sep-98

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F82H vs. a 65kg Lab. heat "ECN-BS" (SOFT-20 paper)

Table 1. Chemical composition of materials (weight %).

	C	Si	Mn	P	S	Cr	Ni	Mo	N	Cu
F82H	0.09	0.11	0.16	0.002	0.002	7.71	0.02	0.03	0.006	0.01
ECN-BS	0.12	0.09	0.50	<0.005	0.004	9.00	<0.02	<0.02	0.005	<0.02

(Table 1 - continued.)

	Co	Ta	B	Ti	Nb	V	Al	W	Fe
F82H	0.005	0.02	0.0002	0.01	0.0001	0.16	0.003	1.95	bal.
ECN-BS	-	0.07	<0.0005	<0.02	<0.02	0.28	0.01	2.17	bal.

Table 2. Details on heat treatment.

	thickness (mm)	normalizing	tempering	hardness (HV5)	grain size (µm)
F82H	7.5	1040 °C/38min	750 °C/60min	204	75
ECN-BS	25	1050 °C/1h/AC	750 °C/2h/AC	212	22

Experiment	Material	Specimens	T _{irr} range (°C)	Dose (dpa)	Helium (appm)
ILAS-4	F82H	Tensile	325 - 335	2.4 - 3.0	~3
ILAS-5	ECN-BS	Tensile	320 - 335	2.7 - 3.1	~6
CHARIOT-2	F82H	KLST	280 - 320	2.3 - 3.1	~3
CHARIOT-3	ECN-BS	KLST	345 - 370	2.8 - 3.8	~6

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Tensile properties, 325°C irradiated

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KLST impact tests F82H and ECN-BS

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μ -Hardness check on KLST-specimens

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Reference and PI-test Results on welded IEA-F82H

- Reference tensile tests
 - EB15, EB25, TIG15, TIG25, EB15-CEA, T91-EB15-CEA, EB15-Schelde, EB25-Schelde.
 - EB15: no significant difference between root and cap
 - TIG15: difference: root 10-20 MPa lower YS and UTS than cap, ductility slightly higher (RA 2-4%, UE and TE ~1 %)
 - EB better than TIG: higher strength, higher ductility.
- 2.5 dpa Post irradiation (ILAS-6) tensile tests
 - EB15, TIG15, EB15-CEA, T91-EB15-CEA: in progress
 - preliminary results: weldments (ILAS-6) show significant higher irradiation hardening than 7.5 mm plate (ILAS-4)

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Comparison EB and TIG weldments (300°C), unirradiated

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Future Work

- This year:
 - Completion of PI testing of 2.5 dpa 300°C plate and welded specimens: tensile, LCF, impact and fracture toughness (CT-10mm and -5mm)
 - FTT 300°C 10 dpa CT-2.5mm
- Future:
 - Tensile and CT-10mm FTT 300°C 10 dpa (2000)
 - Next FP programme: EUROFER 97 and EUROFER 2000
 - B-distribution measurement (1999-)
 - He-content measurement (1999-)

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Conclusions-1 (programmatic)

- Testing of 2.5 dpa 300°C irradiated F82H specimens (plate and EB and TIG-weldments) will be completed by the end of 1998.
- FTT of 10 dpa 300°C irradiated 2.5mm CT's will be available by the end of this year.
- Bulk of 10 dpa 300°C F82H data will be generated in 2000.
- The next EU Programme (1999-2002) will focus on EUROFER

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Conclusions-2 (irradiation behaviour)

- Comparison of F82H and ECN-BS seem to show non-hardening embrittlement: F82H larger DBTT-shift, however same level of hardness.
- Preliminary results of ILAS-6 irradiation seem to show strong sensitivity of F82H hardening to irradiation temperature:
40° lower irr. temperature yields up to 100MPa more hardening!

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Japanese University Program of RAFS R & D for Fusion Reactor

A. Kohyama and A. Kimura

Institute of Advanced Energy

Kyoto University

IEA RAF Workshop

Oct.1-2, 1998 ECN, Petten, Netherland

Blanket Structure of FFHR

I.A.E, Kyoto University

Protection wall

$P_n = 1.5 \text{ MW/m}^2$

$N_d = 450 \text{ dpa/30y}$

Self-cooled

T breeder

$TBR > 1.2$

Radiation

shield

5 order

Vacuum vessel

& T boundary

Material Issues for FFHR

I.A.E, Kyoto University

● Reduced Activation Ferritic Steels; JLF-1

1. High Temperature Strength ($T > 500^{\circ}\text{C}$)
2. Compatibility with FLiBe
3. High resistance to Neutron Irradiation
4. Ferro-magnetism

#1: Operation temperature of Flibe cooling system is recommended to be higher than 500°C .

#2: Corrosion tests of JLF-1 in FLiBe is going on.

#3: Microstructural examination of a RAF steel revealed that void swelling was still less than 1% after neutron irradiation to 200dpa.

#4: Investigations on the effects of magnetism are strongly demanded.

for CREST (CRIPEI, U. Tokyo, Kyoto U. KHI)
water cooling:

Accomplishments in Monbusho/DOE Collaboration (1)

I.A.E, Kyoto University

- 1. Tensile data up to 60dpa at temperatures between 365 and 600°C**
 - 1) hardening and softening**
 - 2) saturation of hardening at around 10dpa**
 - 3) no saturation of softening at 30dpa**
 - a) mechanism of irradiation hardening and softening (I-cluster)**
 - b) superiority of martensitic single phase steels (9Cr-2W)**
 - c) an addition of Al to improve high temperature strength**
- 2. Swelling data and TEM examinations up to 70dpa at 420°C**
 - 1) swelling peak temperature; 420°C**
 - 2) incubation period (35-50dpa)**
 - 3) swelling rate (35-70dpa); 0.013%/dpa**
 - 4) irradiation up to 200dpa; 0.6%**
 - a) no large effects of 30wtppm of B addition**
 - b) incubation period appears to be correlated with the changes in the precipitation morphology**
 - c) relation between irradiation hardening and swelling (Tirr dependence)**

Accomplishments in Monbusho/DOE Collaboration (3)

I.A.E, Kyoto University

5. He related experiments

- 1) B addition; no enhancement of DBTT shift (FFTF/MOTA)**
- 2) 480atppmHe implantation; no enhancement of irradiation hardening and embrittlement**
- 3) Ni addition; large irradiation hardening below 170°C**
 - a) He suppress the growth of V and I-cluster**
 - b) He stabilized thermally V and I-cluster**
 - c) He-V or He-I cluster does not influence the irradiation hardening**
 - d) Ni-I clusters have strong potential of irradiation hardening**
 - e) controlling irradiation temperature is strongly demanded for this type of experiments**

Current Irradiation Research in JUPITER and Domestic University Program (1)

I.A.E, Kyoto University

1. HFIR-MFE-RB-11J,12J;

· Thermal Neutron Shielded, 300 and 500°C

Materials; JLF-1, mod.-JLF-1, F82H

Specimen type; PT, CVN1.5, SS-Tensile, DCT, TEM

- 1) low temperature irradiation effects; toughness, creep
(also in ATR; 200 and 300°C)
- 2) fracture toughness measurements
- 3) effects of Ni addition

2. HFIR-MFE-RB-14J;

· Unshielded, 300, 500, 800°C

Materials; JLF-1, mod.JLF-1, F82H

Specimen type; PT, CVN1.5, SS-Tensile, DCT, TEM

- 1) transmutation effects
- 2) ODS steel (800°C)

Current Irradiation Research in JUPITER and Domestic University Program(2)

I.A.E, Kyoto University

3 HFIR-MFE-RB-13J;

- Thermal Neutron Shielded,
- Varying Temperature; 200/350°C (350°C), 300/500°C (500°C)

Materials; JLF-1, mod.-JLF-1, F82H

Specimen type; CVN1.5, SS-Tensile, TEM

1) factors controlling irradiation hardening (200/350°C)

V-cluster, I-cluster, precipitation behavior

2) stability of defect clusters (300/500°C)

4 JOYO (Japan)-Marico;

- 370-600°C, 5-20dpa

Materials; JLF-1, mod.JLF-1, F82H

Specimen type; CVN1.5, SS-Tensile, SS-Fatigue, TEM

1) post-irradiation fatigue tests, specimen size dependence

2) DBTT/ Fatigue/ Hv/ Swelling correlation

Current Irradiation Research in JUPITER and Domestic University Program(3)

I.A.E., Kyoto University

5 JOYO (Japan)-SVIR;

·370-600°C, 5-20dpa

Materials; JLF-1, mod.JLF-1, F82H

Specimen type; CVN1.5, SS-Tensile, SS-Fatigue, TEM

- 1) post-irradiation fatigue tests, specimen size dependence
- 2) DBTT/ Fatigue/ Hv/ Swelling correlation

6 JMTR;

·Thermal Neutron Trapped or Shielded,

·Varying Temperature; 200/350°C (350°C), 300/500°C (500°C)

·In-pile creep

·High Temperature/Pressure Water loop

Materials; JLF-1, mod.-JLF-1, F82H

- 1) factors controlling irradiation hardening (200/350°C)
V-cluster, I-cluster, precipitation behavior
- 2) stability of defect clusters (300/500°C)

The Monbusho/US-DOE Collaboration on fusion Materials

— JUPITER —

Ver.15:9/1/98

HFIR

In-situ electrical resistivity

High dpa Irrad.
(with thermal shield)

Vary. Temp. Exp.

High dpa Irrad.
(no-thermal shield)

High dpa Irrad.
(with thermal shield)

PTP Mater. Irrad.

In-situ thermal conductivity

HFBR

In-situ electrical resistivity

ATR

Mater. Irrad.

Precisely Temperature Controlling Experiment

Future Planning of Irradiation Research (1)

I.A.E., Kyoto University

1. Continuous effort to built irradiation data base
 - 1) temperature dependence
 - 2) dose dependence
 - 3) irradiation effects on material system component
 - structure gradient, temperature and stress gradient
 - compatibility (including H and He)
 2. Modeling material performance under irradiation
 - 1) understanding physics of radiation effects
 - 2) bridging over the gaps between basic research and engineering
- ***International collaboration is strongly demanded***●
JUPITER-II (2001-2006) -planning stage-

Future Planning of Irradiation Research (2)

I.A.E, Kyoto University

Period	1st Period (1999~2005)	2nd Period (2006~2010)	3rd Period (2011~2015)
Stage	Material Design	Material Selection	Pre-Demonstration
C&R	<ul style="list-style-type: none"> ○Irradiation Effects (100dpa) <ol style="list-style-type: none"> 1)Physical Properties¹⁾ 2)Swelling Rate 3)Creep Rate 4)DBTT 5)Strength 6)He effects ○Ferro-Magnetism (JFT-2M) ○Material System Compatibility) <ol style="list-style-type: none"> 1)H₂O, He, FLiBe(HF) 2)IASCC 	<ul style="list-style-type: none"> ○Irradiation Effects(>100dpa) <ol style="list-style-type: none"> 1)Physical Properties¹⁾ 2)Mechanical Properties²⁾ 3)Dynamic Effects(In-situ creep, transmutation, IASCC) ○Irradiation Effects on Material System Blanket Components <ol style="list-style-type: none"> 1)Joining, Welding 2)Dynamic Effects ○Modeling Material Behavior ○Ferro-Magnetism(JT-60) 	<ul style="list-style-type: none"> ○Irradiation Effects on Blanket Material System (>100dpa) <ol style="list-style-type: none"> 1)Joining, Welding 2)Dynamic Effects³⁾ ○Modeling Material Behavior of Blanket System ○Ferro-Magnetism (JT-60)
Facility	HFIR, ATR, JOYO, JMTR, etc JFT-2M	HFIR, ATR, JOYO, etc, JT-60	HFIR, ATR, JOYO, etc, JT-60
IFMIF	<div>1st stage</div> <div>2nd stage</div>		

1)Thermal conductivity , Thermal expansion, Heat capacity, Magnetic/electric properties

2)elasticity, tensile, impact, fracture toughness , fatigue, dimensional stability

3)IASCC, creep, fatigue, creep/fatigue

Application of Multiple Beams to Energy Materials Research

Institute of Advanced Energy

Kyoto University

PARTICLE EMISSION

RBS / ERDA / Etc.
Chemical Analysis
Defect Analysis
Structural Analysis

X-RAY EMISSION

PIXE
Ultra-Sensitive
Chemical Analysis

LIGHT EMISSION

Optical / IR / UV
Defect Analysis
Structural Analysis
Impurity Analysis

EX-SITU EXPERIMENTS

Microstructural Analysis
Surface Analysis
Microchemical Analysis
Mechanical Property Evaluation, Etc.

Heating / Cooling

HT Materials
Environmental Effects
Reactivity Control
Defects / Lattice Study

Specification of DuET Dual-Beam Facility at IAE Kyoto Univ.

Institute of Advanced Energy

Kyoto University

	DuET Kyoto University	HIT University of Tokyo
Location	Uji, Kyoto, Japan	Tokai, Ibaragi, Japan
Year built	1999	1983
Primary accelerator	1.7MV Tandetron	1MV Tandetron
Secondary accelerator	1MV Singletron	3.75MV Van de Graaff
Primary beam	6.8MeV Ni 5 μ A 6.8MeV Si 40 μ A 5.1MeV He 2 μ A <i>Etc.</i>	4MeV Ni 1.5 μ A
Secondary beam	1MeV H 75 μ A 1MeV He 15 μ A	1MeV H 50 μ A 1MeV He 10 μ A
Target temperature	4 ~ 2073K	300 ~ 973K
Target environment	Vacuum, He, O ₂ , etc.	Vacuum
In-beam experiments (*Planned for FY1999)	RBS, ERDA, PIXE*, Optical/IR /UV spectrometry*, Photoluminescence*, Laser ablation*	RBS, Optical/IR/UV spectrometry

IFARF Workshop
October 1-2 1998
ECN Petten, Netherlands

Ferritic Isotopic Tailoring (FIST) Experiment

- present status and future plan -

*Institute of Advanced Energy
Kyoto University*

**S. Ohnuki, Y. Kohno+, A. Kohyama++
K. Shiba* and A. Hishinuma***

Hokkaido University, +: Univ. of Tokyo

++: Kyoto University, *: JAERI

collaborators:

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Background and Objective

Institute of Advanced Energy

Kyoto University

The Past:

- **isotope tailored F82H disks were irradiated in HFIR**
- **these discs were provided to JUPITER**
- **FIST experiment was proposed at JUPITER WS (8/21-23/97)**

Present Status: (FIST-1)

- **F82H discs were shipped to PNNL 5/98**
- **SP tests finished at PNNL (by S. Ohnuki, M.L. Hamilton) 7/98**
- **TEM inspection method was established and TEM inspection done (by S. Ohnuki, D.S. Gelles) 7/98**
- **Gas Analysis finished 8/98**

Objective:

- **to simulate the fusion environment effects of hydrogen and helium on ferritic alloys**

FIST-1 is the first step for go/no go decision of FIST-2 and beyond

Outline of the FIST Experiment

Institute of Advanced Energy

Kyoto University

Use Fe isotope tailoring to produce H and He in HFIR

Fe(54) at \$20K/gm and Fe(56) at \$2K/gm for comparison with natural Fe

Alloy classes of interest:

JLF-1, (ODS ferritic, Fe-9Cr), F82H (supplied from JAERI)

Issues:

Cost (dependent on smallest melt size possible),

Cost of materials: Japan/(US?) Alloy production: Japan

Irradiation in HFIR

PIE/ gas analysis: PNNL or IMR-Tohoku

Schedule (probably for FY2000)

A Scope

(1) Isotope-tailored specimen:

JLF1 + Fe54 or Fe56(B10), ODS ferrite + Fe54 or Fe56(B10)

Model 9Cr + Fe54 or Fe56(B10)

(2) Specimen fabrication

Isotopes: ORNL Fabrication: Japan

(3) Irradiation

HFIR (RB or PTP)

300, 500 C (+ 400, 600C) 5 dpa, 10 dpa

(4) Specimen type & mass

TEM specimen (TEM & shear punch/Disk Bend test) : 0.01gm

TN: tensile specimen(SS-J): 0.10gm

Charpy(1.5mm): 0.35gm

Cost Estimate:

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Kyoto University

Plan 2 (near-minimum cost)

3 alloys (+ 1 alloy): JLF-1, Fe-9Cr model alloy, ODS Ferritic ,
(F82H from JAERI)

Specimen matrix

SP (lrr temp)x(4 verification) = 3x4 = 12/unit

DB (lrr temp)x(4 verification) = 3x4 = 12/unit

TEM (lrr temp)x(5 verification) = 3x5 = 15/unit

mass needed

JLF-1: 1 gm Fe-9Cr model alloy: 1 gm

ODS Ferritic (minimum for SPEX mill): 4 gm

Materials costs: \$150K

Cost Estimate -1-

*Institute of Advanced Energy
Kyoto University*

3 alloys: JLF-1, Fe-9Cr model alloy, ODS Ferritic (F82H from JAERI)

Specimen matrix

Charpy 6 specimens/unit

Tensile (@RT, @lrr temp)x(2 strain rates)x(2 verification) + (2 extra) = $2 \times 2 \times 2 + 2 = 10$ /unit

SP (@RT, @lrr temp)x(4 verification) + (2 extra) = $2 \times 4 + 2 = 10$ /unit

DB (@RT, @lrr temp)x(4 verification) + (2 extra) = $2 \times 4 + 2 = 6$ /unit

TEM (@RT, @lrr temp)x(2 verification) + (2 extra) = $2 \times 2 + 2 = 6$ /unit

mass needed

JLF-1 including toughness 20 gm

Fe-9Cr model alloy 8 gm

ODS Ferritic 8 gm

Total mass: $20 + 8 + 8 = 36$ gm (Assumed yield rate of 70%: 50 gm needed)

Materials costs:

Fe54 \$20K/gm x 50 gm = \$1M (120M¥)

Fe56 \$1K/gm x 50 gm = \$50K (6M¥) Natural Fe: negligible

Total \$1.05M

Cost Estimate -2-

Institute of Advanced Energy

Kyoto University

Alloy fabrication cost:

Nippon Steel Corporation estimate \$50K-\$100K

PIE cost: negligibly small compared to isotope cost

+ Gas analysis	H, He	~\$2K/sample	\$20K for 10 samples
+ Charpy		Tests at Oarai	
+ DB		Tests at Oarai	
+ Tensile (total 30 specimens)	->	MATRON 1 or 2 weeks	\$5K-\$10K
+ SP		Tests at Oarai or PNNL(unknown)	
+ TEM		total 32 specimens	less than \$1K (?)
		Total PIE cost	~ \$200K

Total Experimental Cost: ~ \$1.25M

F82H IAE

Fe54-F82H

F82H Alloys

Fe54-F82H

Fe54-F82H

Preliminary Results on FIST-1

Microstructural Inspection

*institute of Materials Science
University of California, Berkeley*

F82H-Fe54 (300C, 34dpa) 650appmH
Voids: small size (~5nm), low number density
Dislocation: high density

F82H-B (300C, 34dpa) 320appmHe + Li
Cavities: on sub-boundaries
Dislocation: medium density

F82H STD (300C, 34dpa)
Cavities: no
Dislocation: medium density

F82H-Fe54 (250C, 2.3dpa) 46appmH
Cavities: no
Dislocation: medium density

FLST-1; Preliminary Summary

Report of Laboratory 1, 2, 3, 4

Page 1 of 10

Radiation hardening (increase in shear strength)

- by high dislocation density.
- by cavities (contribution is not so clear).

Radiation induced / enhanced precipitation

- not detectable

Hydrogen effects

- not clear (may affects dislocation evolution)
- synergistic effect ?

EFFECT OF TANTALUM ON PROPERTIES OF 9Cr-2WVTa STEEL

R. L. Klueh

Oak Ridge National Laboratory

**IEA Working Group Meeting on
Ferritic/Martensitic Steels**

Petten, The Netherlands, October 1-2, 1998

9Cr-2WVTa STEEL HAS EXCELLENT STRENGTH AND IMPACT PROPERTIES

- 9Cr-2WVTa steel: Fe-9Cr-2W-0.25V-0.07Ta-0.1C (wt. %)
- Steel is used in normalized and tempered condition
- Steel has extra low Charpy transition temperature prior to irradiation: -88°C (1/3-size Charpy specimen)
- DBTT after irradiation to 28 dpa at 365°C in FFTF was extremely low at -56°C (a shift in DBTT of only 32°C)
- The shift in DBTT is less than for other steels for similar conditions in FFTF and HFR

BEHAVIOR OF 9Cr-2WVTa STEEL IS IMPROVEMENT OVER SANDVIK HT9

- HT9 was first martensitic steel in U.S. fusion program
- Properties are improved before and after irradiation

9Cr-2WVTa HAS SUPERIOR PROPERTIES TO OTHER STEELS IRRADIATED IN HFR

- Irradiation in HFR to 0.8 dpa at 250-450°C
- Shifts in transition temperature of 20-30°C for 9Cr-2WVTa
- DBTT and shift in DBTT for 9Cr-2WVTa steel increases with irradiation temperature from 350 to 450°C

9Cr-2WVTa STEEL SHOWS DIFFERENT BEHAVIOR FROM MOST OTHER STEELS

- Inverse temperature effect
 - For most steels, DBTT and shift in DBTT decreases with irradiation temperature because irradiation hardening decreases with irradiation temperature
 - DBTT for 9Cr-2WVTa increased with temperature
 - Increase between 350 and 450°C in HFR irradiation
 - Increase between 365 and 393°C in FFTF irradiation
- Increase in DBTT with fluence
 - DBTT for 9Cr-2WVTa increases with fluence in FFTF at 365°C; for most other steels it saturates with fluence

9Cr-2WVTa STEEL HAS SIMILARITIES AND DIFFERENCES WITH 9Cr-2WV

- Steels are the same except for the 0.07% Ta
- Two steels have much in common
 - Similarities in microstructure before and after irradiation
 - Similar strengths before and after irradiation
- Steels have different Charpy properties
 - 9Cr-2WVTa has better Charpy properties than 9Cr-2WV before and after irradiation
- 9Cr-2WV steel behaves like most steels during irradiation
 - No inverse temperature effect
 - DBTT saturates with fluence

MICROSTRUCTURES OF 9Cr-2WVTa AND 9Cr-2WV HAVE FEW DIFFERENCES

- Similarities before and after irradiation in FFTF:
 - similar dislocation structures and lath size
 - similar precipitate structures ($M_{23}C_6$ and MC)
 - similar dislocation loop structure formed during irradiation (40-100 nm at $3 \times 10^{19} \text{ m}^{-3}$)
- Differences before and after irradiation:
 - 9Cr-2WVTa has smaller prior-austenite grain size
 - small portion of tantalum is in MC precipitate
 - Atom probe indicates >90% Ta is in solution before irradiation

9Cr2WVTa and 9Cr-2WV STEELS HAVE SIMILAR PRECIPITATE STRUCTURES

Steel	Ppt	Before Irradiation		After Irradiation	
		Density (m ⁻³)	Av Diam (nm)	Density (m ⁻³)	Av Diam (nm)
9Cr-2WV	M ₂₃ C ₆	5.9x10 ¹⁹	125	3.2x10 ¹⁹	160
	MC	1.2x10 ¹⁸	54	1.1x10 ¹⁸	60
9Cr-2WVTa	M ₂₃ C ₆	4.5x10 ¹⁹	136	4.1x10 ¹⁹	143
	MC	7.5x10 ¹⁸	29	5.6x10 ¹⁸	36

9Cr-2WVTa and 9Cr-2WV STEELS HAVE SIMILAR TENSILE PROPERTIES

- Properties of steels are similar from 0-28 dpa at 356°C
- Hardening saturates with fluence

CHARPY PROPERTIES OF 9Cr-2WVTa AND 9Cr-2WV STEELS ARE DIFFERENT

- Transition temperature saturates with fluence for 9Cr-2WV but not for 9Cr-2WVTa irradiated in FFTF to 28 dpa

TANTALUM IN SOLUTION AFFECTS CHARPY PROPERTIES OF 9Cr-2WVTa

- Difference in prior-austenite grain size could explain difference in the normalized-and-tempered Charpy impact properties of 9Cr-2WV and 9Cr-2WVTa
- The prior-austenite grain size cannot explain the inverse temperature effect and fluence effect for 9Cr-2WVTa
- Similar lath sizes and precipitate characteristics in 9Cr-2WV and 9Cr-2WVTa before and after irradiation means these cannot explain difference in irradiation effects
- Tantalum in solution must favorably affect Charpy impact properties of 9Cr-2WVTa steel

HYPOTHESIS: TANTALUM AFFECTS FLOW STRESS OR FRACTURE STRESS

- Strength of 9Cr-2WVTa and 9Cr-2WV steels is not substantially different before and after irradiation
- Tantalum effect on fracture stress or flow stress can explain differences in 9Cr-2WVTa and 9Cr-2WV

TANTALUM PRECIPITATION CAN EXPLAIN IRRADIATION BEHAVIOR OF 9Cr-2WVTa

- If tantalum precipitates during irradiation, tantalum removal from solution would lower the fracture stress or change the flow stress
- Precipitation depends on diffusion, and irradiation at higher temperatures (for a given fluence) will accelerate loss of tantalum and produce the inverse temperature effect observed in HFR and FFTF
- At lower temperatures, higher fluences (longer times) are required for precipitation, which causes the observed continuous increase in DBTT with increasing fluence
- Loss of tantalum from solution still needs to be verified

9Cr-2WVTa MAINTAINS SUPERIOR PROPERTIES DESPITE LOSS OF TANTALUM

- Once tantalum is lost from solution, matrix composition and microstructure (except for prior austenite grain size) will be similar for the 9Cr-2WV and 9Cr-2WVTa steels
- This should result in similar DBTT shifts for the steels
- After 14 dpa in FFTF at 393°C, similar shifts were observed for two irradiations of each steel: 46 and 32°C vs. 35 and 43°C for the 9Cr-2WV and 9Cr-2WVTa, respectively.
- 9Cr-2WVTa still had lower DBTT because of lower value before irradiation due to smaller prior-austenite grain size

OTHER TANTALUM-CONTAINING STEELS SHOULD SHOW SIMILAR BEHAVIOR

- F82H and OPTIFER-Ia contain tantalum
- Effect of tantalum loss on other steels not evident after 0.8 dpa in HFR as for the 9Cr-2WVTa

F82H AND OPTIFER-1a SHOW INVERSE TEMPERATURE EFFECT AFTER 2.4 dpa

MANET STEELS AND OPTIFER II—NO Ta AND NO INVERSE TEMPERATURE EFFECT

MANET-I

MANET-II

OPTIFER-II

SUMMARY AND CONCLUSIONS

- Excellent irradiation resistance of 9Cr-2WVTa steel is due to smaller prior-austenite grain size and tantalum in solution, which can affect fracture stress or flow stress
- Irradiation resistance due to tantalum in solution is lost during irradiation; change is attributed to precipitation of tantalum, which still needs to be verified
- Similar effect is observed in other tantalum-containing steels (F82H and OPTIFER-Ia)
- Irradiation resistance of steels without tantalum should be improved by proper heat treatment to change prior-austenite grain size

ODS Alloys
and
Alloy Development

OXIDE DISPERSION STRENGTHENED RAFM STEELS REVIEW AND PROSPECTS

Presentation at IEA RAFM Workshop
Petten, 1 October 1998

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OVERVIEW

- Objective ODS steels for blankets
- Investigated materials
- Testing parameters
- Results
- Conclusion
- Prospects

OBJECTIVE ODS STEELS FOR BLANKETS

- High creep strength at temperatures of 875 K or higher
 - Suppress helium bubble effects
 - Low swelling
 - Low irradiation creep
 - Component manufacturing easy
 - Cost in line with improvement
-

ORIGINAL OBJECTIVE OF ODS STEEL

- Develop LMFBFR canning material for up to 200 dpa with maximum temperatures in the range 875 - 975 K
 - Resist creep deformation caused by internal fuel pin pressure build-up
 - Strike the balance between:
 - Fabricability of cans typically 0.3 mm wall thickness and 6 mm diameter
 - Acceptable mechanical properties under high temperature high stress condition
-

Typical ODS steel chemical compositions

ID	C	Cr	Ti	Mo	W	Ti ₂ O ₃	Y ₂ O ₃	Origin
DT02	< 0.08	13	3.5	1.5	-	2	-	SCK
DY05	< 0.08	13	3.5	1.5	-	-	0.5	SCK
DT2203Y5	na	13	2.2	1.5	-	0.9	0.5	SCK
CEA-A	na	14	1	0.3	-	-	0.25	CEA
EM10+Y	na	9	-	1.0	-	-	na	CEA
SUKO01	0.013	11	0.4	-	2.6	-	0.52	PRNFD
SUKO02	0.031	13.7	-	-	3.0	0.17	0.22	PRNFD
KO01	0.19	106	0.5	-	2.3	0.93	-	PRNFD
MA956	0.019	19.1	0.34	na	-	-	0.48	PNL
MA957	na	13.1	0.95	0.27	-	0.95	0.25	PNL

TYPICAL POWDER SIZES

Elemental	100 - 150 μm
Pre-alloyed	< 150 μm
Yttrium oxide*	< 44 μm
Yttrium oxide	< 20 nm
Inert atmosphere essential	
- vacuum	
- inert gas	

* Depending on process

SCK FABRICATION ROUTE

ECN

POST-IRRADIATION PARAMETERS

TESTING

● Irradiations

Electrons ≤ 20 dpa

Fast Fission Neutrons ≤ 200 dpa

● Mechanical testing

– Thermal creep : ≤ 10000 hrs

– Irradiation creep : ≤ 925 K

– Tensile

– Charpy energy

● Microstructure

Maximum dose 200 dpa

● Swelling

Maximum dose 200 dpa

ODS Ferritic Steel (No.4)

ODS Martensitic Steel (No.16)

DT2906

DT 2203Y05

UTS versus RA at 325°C and 3,4 dpa

ECN

CONCLUSION

- Oxide dispersion strengthening improves conventional Cr steels properties
 - Helium effects seem to be suppressed by the ODS microstructure
 - Most information is strongly oriented to LMFBR fuel canning
 - Severe anisotropy observed
 - Thick-walled parts or large diameter tube experience is not available in open literature
 - In open literature no information is available yet on ODS RAFM steel
-

ECN

PROSPECT

- Conventional Cr steels with ODS have properties promising for ODS RAFM high temperature application under high neutron loads
 - ODS potential in RAFM is:
 - Operating temperature increase to 900 K
 - Helium effects reduction
 - Fabrication route must be developed for ODS RAFM blanket “heavy sections”
 - Joining of ODS RAFM steel must be addressed in an early stage
 - Too early to decide on which $\text{Ti}_2\text{O}_3/\text{Y}_2\text{O}_3$ mixture to apply
-

DEVELOPMENT PRIORITIES

- Assure that the conventional fabrication routes produce acceptable ODS RAFM properties improvement
 - Find a manufacturing route for thick walled (10-20 mm) ODS RAFM parts with homogeneous, isotropic properties
 - Develop joining methods resulting in acceptable properties
 - Keep corrosion properties in the required range
-

IEA WORKSHOP ON REDUCED-ACTIVATION F/M STEELS
ECN, Petten, The Netherlands, 1-2 October 1998

DEVELOPMENT OF 9Cr FERRITIC- MARTENSITIC STEELS STRENGTHENED BY OXIDE DISPERSION

V. LAMBARD, A. ALAMO

Commissariat à l'Energie Atomique
CEA- Saclay, SRMA
Metallurgical Research Lab.
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Oxide Dispersion Strengthened «ODS» ferritic alloys

MECHANICAL ALLOYING

Attrition

Metallic powders + oxide powder(s)

POWDER COMPACTION

Extrusion, HIP

**Metallic matrix with a homogeneous
distribution of oxide particles
(average size 15 nm)**

*High temperature strength
Resistant to environmental degradation*

*High dimensional stability under
irradiation (swelling, in-pile creep)*

ODS FERRITIC STEELS

Distribution of Y_2O_3 Oxide Particles

100 nm

ODS FERRITIC STEELS

AS-EXTRUDED MICROSTRUCTURE

Longitudinal Section

Transverse Cross-section

500 nm

ODS FERRITIC STEELS.

GRAIN STRUCTURE AFTER
RECRYSTALLISATION

1 mm

ODS FERRITIC STEELS

Main characteristics :

- Elongated grains:
 d about 500 nm,
 aspect ratio: 20-50
- High texture
- Anisotropic properties
- Grain size of recrystallised structure :
 $d > 1 \text{ mm}$
- Recrystallisation temperature :
 $T > 1300^{\circ}\text{C}$
- Low ductility

ODS FERRITIC STEELS

RECRYSTALLISATION TEMPERATURES

ODS FERRITIC STEELS

MA957

(Fe - 14 Cr - 1Ti - 0.3Mo - 0.25 Y₂O₃)

Intermediate grain size structure
(cold-drawing and recrystallisation)

Cross-section

Longitudinal
section

100 μm

ODS FERRITIC STEELS

Creep behaviour

Ageing behaviour

Precipitation of intermetallic phases

χ phase (Cr, Ti)

Laves phase (Mo, Ti, Cr)

α' phase (Cr-rich, under irradiation)

ODS - 9Cr MARTENSITIC STEELS

Fast cooling:
austenite $\Rightarrow$ martensite

Several variants/ grain

ODS F/M STEELS

Why a 9Cr-ODS steel ?

To avoid embrittlement by intermetallic phase precipitation .

Control of chemical composition

To decrease anisotropic properties of the fully ferritic materials.

ODS - 9Cr MARTENSITIC ALLOY

9Cr-Mo + Y₂O₃

9Cr-W + Y₂O₃

Martensitic transformation could decrease anisotropy.

Y₂O₃ dispersion assures good creep properties.

9Cr martensitic matrix have a good stability under ageing and irradiation.

ODS - 9Cr MARTENSITIC STEELS

OXIDE PARTICLE DISTRIBUTION

(determined on carbon extraction replica by TEM from about 1000 particles for each material)

Frequency- size diagrams :

Chemical composition (at%):

9Cr-1Mo + Y₂O₃:

55Y - 34Si - 11Mn

9Cr-1Mo + Y₂O₃ +Ti

50Y - 50Ti

ODS - 9Cr STEELS

Equiaxed grain structure

Martensitic matrix

Ferritic matrix

Longitudinal
sections

10 μ m

ODS - 9Cr MARTENSITIC STEELS

Effects of Heat Treatments

Austenitisation :

No changes in hardness and grain growth in the range 1000-1250°C.

Tempering :

ODS - 9Cr MARTENSITIC STEELS

TENSILE PROPERTIES

0.2% Proof Stress

Reduction in area

Preliminary Studies for ODS/RAF Development

JAERI A. Hishinuma

Background:

- RAM Fe-(7-9)Cr ferritic steels are the first candidate alloy for DEMO and beyond.
- However, the design window of the RAM for fusion reactors is not enough, (For example, see Fig.1)
- High temperature strength is required from viewpoints of
 - 1) Safety under off-normal condition (i.e. >600 MPa at 600 °C)
 - 2) Application for other coolant system such as liquid metals or He-gas.
- ODS/RAF is one of the candidates for
 - 1) Overlay materials for RAM (in near future)
 - 2) Alternative materials of current candidate RAM (in future)
- Another advantage – good irradiation resistance (see Fig.2)

Manufacturing process of ODS/RAF

Raw powder of Fe-8Cr alloy and other elements: <150 μ m

Mechanical alloying: 10D type Attrition Mill,

Weight ratio : powder :ball = 1:15

Rotating speed: 220 rpm

Rotating time: 48 hr

Hot extrusion: 1050°C, extrusion ratio = ~7

Heat treatments: 1050°C /1hr/AC $\rightarrow$ 750°C /1hr/AC

Fig. Service condition for F82H (estimated)

Table 1 Chemical compositions of ODS/RAH steels.

(mass%)

符号 \ 元素	C	Si	Mn	P	S	Ni	Cr	W	Ti	Y	O	N
G 9 1	0.12	0.006	<0.005	0.001	0.003	0.052	7.99	0.096	0.092	0.088 (0.11)	0.12	0.0094
G 9 2	0.12	<0.005	<0.005	0.001	0.003	0.012	7.94	<0.005	0.093	0.15 (0.19)	0.13	0.0082
G 9 3	0.12	<0.005	<0.005	0.001	0.002	<0.005	7.92	<0.005	0.093	0.23 (0.29)	0.16	0.0088
G 9 4	0.12	<0.005	<0.005	0.001	0.002	<0.005	7.92	<0.005	0.18	0.24 (0.30)	0.15	0.0090
G 9 5	0.12	<0.005	<0.005	0.001	0.002	<0.005	7.88	<0.005	0.27	0.23 (0.29)	0.15	0.0091
G 9 6	0.12	<0.005	<0.005	0.002	0.003	<0.005	7.89	1.74	0.28	0.23 (0.29)	0.15	0.0096

※Y分析値の () 内は、Y₂O₃の計算値 (Y×1.27)

Fig.1 Optical microstructure of G96 (Fe-0.12C-8Cr-2W-0.3Ti-0.3Y₂O₃) after annealing at 1050°C for 1 hr and tempering at 750°C for 1hr.

Fig. 2 TEM microstructure of G96 (Fe-0.12C-8Cr-2W-0.3Ti-0.3Y₂O₃) after annealing at 1050°C for 1 hr and tempering at 750°C for 1hr.

Fig. 3 TEM microstructure of dispersed particles of G96 (Fe-0.12C-8Cr-2W-0.3Ti-0.3Y₂O) after annealing at 1050°C for 1 hr and tempering at 750°C for 1 hr by extraction carbo replica methods.

Fig. 4 Stress-strain curves of ODS/RAM G96 (Fe-0.12C-8Cr-2W-0.3Ti-0.3Y₂O₃) and RAM F82H at room temperature to 600°C.

Fig.5 Temperature dependence of YS and UTS for ODS/RAM G96 (Fe-0.12C-8Cr-2W-0.3Ti-0.3Y₂O₃) and RAM F82H.

Fig. 6 Y₂O₃, Ti and W contents dependence on fractured area transition temperature(FATT) of ODS/RAM

**Integrating Modeling, Experiment and Data
Base Development:
A Mechanism Based Approach to
Developing Advanced Ferritic Steels for
Fusion Applications**

G. R. Odette

University of California Santa Barbara

**IEA Workshop on Reduced Activation Ferritic Martensitic Steels
October 1-2, 1998
Petten Netherlands**

**Research Sponsored by the US Department of Energy and the US Nuclear
Regulatory Commission**

Outline

Recommendations of the Fusion Energy Systems Advisory Committee (FESAC) Review Panel on the Structural Materials Program (C. Baker and S. Harkness, Co-Chairs)

Recommendations on creating a knowledge base by the integration of modeling, experiment, engineering data compilations and structural integrity assessment methods

Roadmapping and an example from UCSB studies of RPV embrittlement

A brief update on some recent progress UCSB studies of FM steels

FESAC Panel Recommendations

More fully integrate materials R&D with component, system level and advanced power plant design issues

Provide a fundamental knowledge base for assessing and expanding material performance limits for economic, safe and environmentally benign fusion power

New ideas, materials systems, people and collaborations, communities, institutions and international partners

Lever broad advances in materials science, scientific computing and synergisms of cross-cutting issues though an integration of theory, modeling and experiment with fusion specific focus

- Pursue fusion energy materials science and technology as a partner in the international effort.
- It is very important to coordinate the U.S. effort with others based on the complexity of the materials challenge and the level of resources available in the U.S.

Balance of the Program

Based on the information on the U.S. and International Fusion Materials Programs that is included as Appendix 1, the Committee considered the appropriate balance between current programmatic elements in the context of the suggested goal and objectives and reached the following conclusions.

- The fusion materials program would benefit from an increase in the fraction of research related to the modeling of materials behavior.
- The fusion materials program should maintain a focus on key issues related to in-vessel structures in a D-T fueled reactor, with significant emphasis on irradiation experiments. However, the fraction of research related to basic understanding of materials performance in the fusion environment should be increased. This increase is motivated by increased capability and sophistication of computer modeling, the need to make the most effective use of expensive and difficult-to-obtain materials data, and the desire to form stronger connections with the greater scientific community. This basic research should develop mechanistic, micro-structurally based models of irradiation effects on material properties. Such semi-empirical models can be used to evaluate, correlate and extrapolate engineering data; and to provide insight on pathways to improved materials. The modeling approaches should include direct simulation methods, like those based on molecular dynamics and Monte Carlo techniques, which are rapidly developing to link macro and meso size scales with rigorous treatments of key phenomena that occur at the atomic scale.
- In recognition of the new direction in the fusion energy sciences program, additional emphasis should be placed on developing the knowledge base for fusion materials.
- The fusion materials research program should emphasize: a) innovative experiments to address key common and long-term issues; b) assessment of information to provide the best estimates of stress-temperature-displacements per atoms (dpa) -corrosion limits that can be systematically refined and improved with continuing research; c) increasingly reliable property predictions; and d) mechanism based approaches to improved materials. In the near and intermediate term, development of this knowledge base will primarily rely on intermediate dose fission reactor irradiation experiments coupled with an expanded basic research and

What is Modeling

Not simply *theoretical* equations, but includes:

basic mechanism and controlled variable *experiments*
on microstructures/properties focussed on key issues

fundamental *laws* of thermodynamics, kinetics,
structure-property relations, micromechanics and
macromechanics -- quantified in submodels

integrated models of ^{PROPERTY CHANGES} embrittlement

statistical fits to the engineering data base

modeling effort. This effort should be coupled with the recommended increased efforts to develop micro-mechanical predictive models.

- An increase in involvement with integrated component modeling would be beneficial to the fusion materials program.
 - There is a great opportunity to develop and apply modern computational tools to engineering design, analysis and simulation of in-vessel components. Such studies would benefit the materials research program by sharpening the understanding of performance requirements and promoting the development of advanced design methods needed to assure structural integrity without undue conservatism. These studies will also lead to major improvements in the engineering science underlying advanced in-vessel designs. While the fusion materials program should not take the lead in these studies, it should increase its participation in these activities.
- An increased emphasis should be placed on resolving the key feasibility issues raised by each materials systems in conjunction with other parts of the Fusion Program. Examples for each of the three materials systems which are currently under some level of development include:
 - Ferritic Steels
 - * The suitability of the application of ferromagnetic materials for in-vessel components in magnetic confinement devices (in collaboration with researchers outside the materials program and with some emphasis on Tokamaks). This should include an analysis of the perturbations of the device's magnetic fields and issues of dynamic control of the plasma.
 - * The effect of irradiation, including the influence of helium, on fracture toughness, ductility and constitutive properties to better determine performance constraints over the temperature range of interest.
 - * Alloy development and design concepts resulting in higher, maximum operating temperatures in a fusion irradiation environment.
 - Vanadium Alloys
 - * The viability of electric insulators for candidate liquid metal/vanadium systems and the suitability of coolants other than liquid lithium and possible designs to accommodate the chemical reactivity of lithium (in collaboration with researchers outside the materials program).

- * The effect of irradiation, including the influence of helium, on fracture toughness, ductility and constitutive properties to better determine performance constraints over the temperature range of interest.
 - * High temperature limitations with particular emphasis on the role of helium in creep and creep rupture and alloy development strategies to expand these limits.
- SiC/SiC Composites
- * Structural joining methods and identification of properties and methods for designing thermal-mechanically loaded, inherently brittle structures.
 - * Irradiation effects on thermal conductivity, the stability of fibers and fiber coatings, including the effects of transmutations.
 - * Coatings and claddings to provide adequate hermiticity for helium coolants.
- Opportunities for new and innovative approaches to fusion materials research should be expanded.
- The motivation for new approaches is illustrated by Table 1, which shows the results of initial calculations done for the APEX study. The allowable neutron loads are sensitive functions of the assumed fraction of the charged particle energy that reaches the first wall, the thickness of the wall, the coolant temperature and the results from detailed analysis of the specific design. While, therefore, the absolute estimates of allowable neutron wall loadings would be expected to change in other studies, the general trends would be expected to be maintained. The preliminary APEX calculations shown in Table 1 indicate the possible limitations for any of the current candidate materials in allowable neutron wall loading relative to the level of 7 MW/m^2 , which is viewed as an attractive design target. Future inclusion of materials which had previously been eliminated based on long term activation level considerations alone may open up the system design window. These exploratory efforts should consider the needs of innovative magnetic confinement concepts as well as the needs of inertial fusion energy.
- Therefore the introduction of new ideas and people to the fusion materials program should be encouraged. Modest increases in funding and a yearly opportunity for competitive, peer reviewed proposals are possible mechanisms to support such renewal. Innovative approaches to promoting sustained and mutually beneficial collaborations between laboratories, universities and industry should be developed.

Table 1*	
Peak Wall Load Limits for "Dry" First Wall	
Material (Interface Temperature)	Allowable Peak Neutron Wall Loading (MW/m ²)
Ferritic Steel (500°C)	1.5
V-Cr-Ti (600°C)	3.2
SiC-SiC (700°C)	2.5
Oxide-Dispersion Strengthened Ferritic (ODS) (600°C)	2.6
Nb1Zr (700°C)	6.6
Tungsten (800°C)	8.8
TZM (800°C)	13
T111 (800°C)	11.6

- Peer review should be expanded along with the use of well-defined measures of quality and progress towards program goals.
 - Standards of quality, performance and progress towards program goals should be more fully developed and utilized at both universities and national laboratories. These metrics should be used to facilitate an ordering of programmatic priorities toward the more promising new developments in material systems. Periodic expert peer reviews in the context of these metrics should be supported. The review panels should include leading materials scientists from outside the program.

Specialized Facilities for Materials Research and Development

As part of the overall charge, the panel was asked to review the program efforts aimed at a fusion neutron source test facility including US involvement in the international fusion material irradiation facility (IFMIF). The Panel also considered the general topic of specialized neutron irradiation facilities for materials research and development.

The fusion environment in which the materials of the in-vessel system have to function is complex and arguably more challenging than faced by any other potential power generation concept.

*Information taken from "Exploring Novel High Power Density Concepts for Attractive Fusion Reactors," APEX Studies, M. Abdou, et al.

Issues

Clearly need to know basic constitutive and failure properties for design and operation

Changes in key properties in fusion environments
function of *many* variables and *variable combinations*

$$\Delta P = f(\phi, \phi(E), \phi t, T, \text{comp./imp., TMT}, \dots)$$

No fusion experimental facility, reactor experiments are expensive and time consuming, and higher temperature studies will require higher ϕt irradiations - must extrapolate based on models and mechanism experiments

Not only radiation effects

‘Roadmapping’

Realistically identify what the future product and buyer

Identify the existing and new needed constituents to meet objectives (people, tools, materials, support resources, organizational structure, ...)

Provide plans about how various elements fit together (a building and not just bricks)

Methods for assessment and modification of roadmap (knowledge based flexibility of paramount importance)

Enhanced ways for people to interact, cooperate and contribute

RPV Embrittlement 'Roadmap'

Products:

$\Delta T_{\text{cvn},k} = f(\text{Cu, Ni, P, HT, } \phi t, \phi, T, \dots)$ physically based,
calibrated to engineering data base for regulations

aid in technical judgments

early warnings of potential technical surprises

improved methods

Tools (all needed in combination):

workable strategy: damage; micro-nano evolution; basic properties; fracture

relation between ΔT and other basic properties ($\Delta\sigma_y$)

small specimen mechanical tests of measuring basic properties and test automation (large experimental matrices)

alloy sets and controlled irradiations to address key questions and single variable experiments (Cu, Ni, P, HT, ϕt , ϕ , T)

Tools (cont.):

microstructural characterization methods (SANS, AP, FEGSTEM, PIA)

verification and ties to 'real' steels and environments (surveillance samples and PREDB)

modeling (defect production, transport and fate, thermodynamic, kinetic, MD, MC, structure property, macromechanical, micromechanical) - *key role of cascades & thermodynamics in multiconstituent alloys*

integration and multiscale-physics model of $\Delta T = f(\dots)$

statistical fits to the data base

Assessments and Modifications:

verification experiments and tests and data fits (theory led observation in many cases)

learn from mistakes, exploit new opportunities (IVAR delay, piggyback experiment)

develop new capabilities while meeting nearer term objectives (CM-FR)

new initiatives (RPV Master Curve)

Interactions and cooperation

IGRDM

Collaboration with AEA, ORNL,

Synergisms with fusion

Pressure Vessel Embrittlement

- Exposure to neutrons embrittles pressure vessel steels (in both charpy and fracture toughness tests) as manifested by transition temperature increases (ΔT) and upper shelf decreases (ΔUSE)

Alloys

- Q&T Mn-Mo-(...) plates, forgings and welds

Variables

- Cu ($<0.5\%$), Ni ($<1.5\%$), P ($<0.03\%$)
- Microstructure and heat treatment
- Irrad. temp. ($T \approx 255-310^{\circ}\text{C}$)
- Fluence ($\phi t < 6 \times 10^{23} \text{ n/m}^2$)
- Flux ($\phi \approx 10^{12} - 10^{15} \text{ n/m}^2\text{-s}$; tests at $\phi \approx 10^{16} - 10^{18} \text{ n/m}^2\text{-s}$)

Synergistic Interactions

Multiscale Modeling of Neutron Irradiation Embrittlement

Models

Experiments

Atomistic-
Bonding
MC/MD Dynamics
ps, Å, $1-10^5$ atoms)

Thermodynamic-
Rate Theory
Kinetics

Dislocation
Deformation
Mechanics

Macroconstitutive
Local
Fracture Mechanics

Independent Physical
Validation of Engineering
Database
(Gs, m, 10^{31} atoms)

Microstructural Characterization Methods

APFIM

PIA: $T_i + 0$ to 60°C

Figure 19 Trends in embrittlement with key irradiation and PIA variables - see text for discussion.

Fracture and Crack Blunting

Brittle

higher ← Stress → lower

lower ← Strain → higher

Ductile

Progress on MC

New method for direct constraint corrections

Additional data and new analysis of unirradiated F-82H

Applications of the equivalent yield stress model (EYSM)

Initial testing of unirradiated T91

Collaboration with ECN Petten in combining unirradiated
-irradiated data in first evaluation of the MC-shift
procedure

Collaboration with ECN Petten and CRRP/PSI in
ongoing and future irradiations

Constraint Correction F82H

a) Static $K_c(T)$ data for F-82H from pre-cracked Charpy-type specimens with $W = 3.3$ and 10 mm and $a/W \approx 0.1$ and 0.5 and from a bend bar with $W = 30$ mm and $a/W \approx 0.5$. b) $K_c(T)$ data constraint corrected (diamonds) to a common reference geometry $a/W = 0.5$, $W = 10$ mm shown by the open circles (no correction).

MC- ΔT -- FMS

Two $K_e(T)$ MC using shifts (- 55 to +140°C) for various sizes, crack geometry, loading rate and irradiation (ECN data 2.5 dpa at 300°C)

MC- ΔT (T91)

Static K_e ($\approx K_{Jc} < 200 \text{ MPa}\sqrt{\text{m}}$) for 0.2T and 0.5T CT specimens -- the unirradiated cleavage initiation 0.2T data (UCSB) is constraint corrected and combined with unirradiated (J-R) and irradiated (K_q) data to estimate shift of $\approx 145^\circ\text{C}$

Collaboration ECN Petten

UCSB and PNL F-82H samples in Petten High Flux Reactor 'CHARIOT' irradiation matrix

	CHARIOT-2	CHARIOT-4	CHARIOT-6
Irradiation Dose and Temperature	2.5 dpa, 300°C (out)	2.5 dpa, 300°C (out)	10 dpa, 300°C (out '99-2000)
Contents			
ECN	10mm CT KLST TEM discs	10mm CT KLST	10mm CT KLST TEM discs
PNL	TEM discs mini-tensiles		TEM discs mini-tensiles
UCSB	SANS coupons	PMC	PMC SANS coupons

Collaboration CRRP/PSI

Data in support of MC- ΔT for F82H -- MAster Curve
Experiments (MACE) in the Hungarian Research Reactor
(~35 mm diameter by 170 mm), with first dose ~0.5 - 5
dpa, ~150°C, 250°C, 350°C starting 1/99

<u>Qty</u>	<u>Specimen</u>	<u>Post-irradiation test</u>
6	0.4T CT's, $a/W \sim 0.5$	K(T) -- static
30	0.2T CT's, 20% SG, $a/W \sim 0.5$	K(T) -- static
13	0.1T 3PB's, $a/W \sim 0.5$	K(T) -- static
13	0.1T 3PB's, $a/W \sim 0.1$	K(T) -- static
19	MCVN, $a/W \sim 0.5$	K(T) -- dynamic
40	Minitensiles	$\sigma = \sigma(\epsilon, \dot{\epsilon}, T)$
4	SANS coupons	
10	TEM discs	

Collaborators at UCSB

Takuya Yamamoto (Tohoku U.) -- Deformation and fracture modeling and experiments (1 year)

Yoshiharu Murase (NRIM) -- creep-fatigue interaction; fracture and failure processes (1 year)

Philippe Spatig (CRPP/PSI) -- Deformation and fracture modeling and experiments MACE experiments

Other

Properties and mobility of SIA and IC in BCC alloys

Cascade vacancy clustering and complex formation

Superposition of net strength contributions from multiple hardening sources

Indentation methods to assess yield/post yield constitutive properties

Constitutive equations for F 82H and T91

Self-Interstitial Cluster Mobility

- SIA clusters are highly mobile (low E^m), migration occurs by 1-dimensional translation (highly anisotropic)
- Highly kinked cluster morphology may be responsible for high cluster mobility

Cascade Aging Simulations

- Fe-0.3% Cu, formation of cascade vacancy-solute clusters

- Resultant formation of copper precipitate embryos

Cascade Vacancy-Solute Clusters

- Continuum of continuously forming and dissolving features

Superposition Rules

Superposition parameter S interpolates from LS to RSS

$$\sigma_t = S (\sigma_1 + \sigma_2) + (1-S) \sqrt{\sigma_1^2 + \sigma_2^2}$$

S determined from (Foreman-Makin type) computer models of dislocation-strong (α_s)/weak (α_w) obstacle interaction

$$S \sim \alpha_s - \alpha_w (4.3 - 2.4 \alpha_s)$$

Predicted strengthening in RPV steels agrees with empirical values; decrease in S during PIA enhances recovery

Indentation Testing

CM measurements of peak pile-up height (h_p)/width (L_{00}) correlate the strain hardening exponent, n , in good agreement with FEM simulations

CONSTITUTIVE EQUATION: $\sigma = \sigma(\epsilon, \dot{\epsilon}, T, \text{irradiation})$

Strong temperature and strain-rate dependence of bcc materials at low temperatures

Thermally activated plastic flow: $\dot{\epsilon}_p = \dot{\epsilon}_{p0} \exp\left(-\frac{\Delta G(\sigma^*, T)}{kT}\right)$

ΔG is the activation energy provided to overcome the barriers for dislocation motion

Investigation by strain rate jumps at the yield stress:

Activation volume: $V = \frac{\partial \Delta G}{\partial \sigma^*} = kT \left(\frac{\partial \ln \dot{\epsilon}_p}{\partial \sigma^*} \right) \cong kT \left(\frac{\Delta \ln \dot{\epsilon}_p}{\Delta \sigma} \right)$

Activation energy: $\Delta G = \delta G - \int_{\sigma_{\max}}^{\sigma} V d\sigma$

ΔG about 0.5 eV for the low temperature mechanism (Peierls)

σ_{μ} is reached at room temperature.

It will be determined how ΔG , σ_{μ} and $\dot{\epsilon}_0$ change after irradiation. This yields useful information about the nature of the irradiation induced defects.

$$\sigma = \sigma^*(\dot{\epsilon}_p, T) + \sigma_{\mu}(\epsilon_p)$$

Helium Effects Studies

ELECTRON MICROSCOPY OF RAFM STEELS WITH HELIUM

R. Schaublin
CRPP - Fusion technology
Materials group
EPFL
5232 Villigen - PSI

• MATERIALS :

F82H $E_2 - 7,65 \text{ Cr} - 2 \text{ W}$
OPTIMAX A $E - 9,0 \text{ Cr} - 1 \text{ W}$

• IRRADIATIONS:

▷ 590 MeV protons (PSI)

F82H $\left[\begin{array}{l} 0,5 \text{ dpa @ } 250^\circ \text{C} \\ 1,7 \text{ dpa @ } 40^\circ \text{C} \end{array} \right.$

▷ neutrons (Petten)

F82H - 2,5 dpa @ 250°C

OPTIMAX A - 2,5 dpa @ 250°C

F82H

Material characterization:

- TEM
- Dislocation type and density
 - Carbide composition and size distribution
 - Grain / lath boundary chemistry

Comparison between:

- Unirradiated
- Unirradiated deformed
- Irradiated to 95 dpa

as received

irradiated
to 0.5 dpa

Fig. 1

Fe 3d

F82H

(P. Spätig)

Figure 3

8211. Carbide size distribution

-82H as received

$g = [\bar{1}10]$

$[01\bar{1}]$

$[10\bar{1}]$

$[110]$

$[020]$

$[200]$

0.25 μm

- Carbide composition: $M_{23}C_6$
 M: 60,6 % Cr
 28,8 % Fe
 6,3 % W
 Ta, V and Ti

- Dislocations:

- Burgers vector: $\frac{1}{2} a_0 \langle 111 \rangle$
- Dominant screw character
- Dislocation density:

as received	$0,85 \times 10^{10} \text{ cm}^{-2}$
deformed	$1,0 \times 10^{10} \text{ cm}^{-2}$
irradiated, 95dpa	$0,9 \times 10^{10} \text{ cm}^{-2}$

- No visible defects in irradiated material

F82.4

as received

Chemical maps using
EFTEM

BF
image

Fe
map

Cr
map

→ Cr enrichment

Fig. 2

F82H
irradiated
to 0.5dpa

3F
image

Fe
map

Cr
map

→ Cr depletion

Fig. 3

#82A
irradiated
to 1,7 dpa

WB-TEM

- Dislocations $\frac{1}{2} a_0 \langle 111 \rangle$
- Black dots in BF

F80H
as received

In-situ tearing

Fig. 4

SUMMARY

- No differences in microstructural defects between as received, deformed and irradiated to 0,5 dpa
 - Difference in boundary chemistry after irradiation
- Unvisible defects?

OPTIMAX A 2.5 dpa, neutrons, 250°C

EX0030 .xxx .xxx .98

General view in BF-TEM

OPTIMAX A 2.5 dpa, neutrons, 250°C

General view in BF-TEM

General view in BF-TEM

OPTIMAX A 2.5 dpa, neutrons, 250°C

Defocussed BF-TEM

Defocussed BF-TEM
Facetted bubbles
Bubble facets on {011} planes

SUMMARY

OPTIMAX A : • No visible defects

• Cavities, faceted

Density : $2 \times 10^{20} \text{ m}^{-3}$ (highest)

Size : 9 nm

F82H : • Black dots (loops)

Density : $3 \times 10^{21} \text{ m}^{-3}$

Size : 3 nm

• No cavities

F82H 2.5 dpa, neutrons, 250°C

50 nm
└─┘

RS0007.XXX.XXX.93

VB image

→ Defects (loops, "black dots")

→ no He bubbles nor cavities

IDENTIFICATION of "SMALL" DEFECTS in a TEM TYPE and SIZE ?

The simulation approach:

MD simulation $\rightarrow$ sample
multislice image simulation $\rightarrow$ image in TEM

Figure 1 : View of a (111) plane cutted in the MD sample containing the 2 nm SFT

LockV

(Y. Dai, PSI)

Figure 2 : a) Experimental weak beam $g(6g)$, $g = (200)$ of an SFT. b) Same as a) but with the wire frame of a 2 nm SFT

Figure 3 : a) Simulated weak beam $g(6g)$, $g = (200)$ of the MD simulated 2 nm SFT. b) Same as a) but with the wire frame of the MD simulated SFT.

$\Rightarrow$ 20% difference between image size and object size

Dislocation loops in Al

Diameters of the loop:

~0.5 nm $\rightarrow$ 8 interstitials (0.57 nm)

~1.0 nm $\rightarrow$ 19 interstitials (1.14 nm)

~2.0 nm $\rightarrow$ 37 interstitials (1.71 nm)

200 kV

$\Rightarrow$ Difference image to object $\nearrow$ when object size $\downarrow$

**TENSILE AND CHARPY
PROPERTIES
OF
B-DOPED F82H
AFTER IRRADIATION**

K. Shiba/JAERI

**IEA Working Group Meeting
on
Reduced Activation Ferritic/Martensitic Steels**

1-2, October, 1998

ECN Petten, The Netherlands

MATERIAL

O Original F82H: F82H std
 O ^{10}B -doped F82H: F82H+10B(320 appm- ^{10}B)
 Onatural B-doped F82H: F82H+nB(320 appm- $^{\text{nat}}\text{B}$)

Chemical Compositions (wt%)

	C	Si	Mn	P	S	Cr	W	V	Ta
std	0.100	0.14	0.49	0.001	0.001	7.44	2.0	0.20	0.04
nB	0.099	0.15	0.50	0.001	0.001	7.49	2.1	0.20	0.04
10B	0.098	0.17	0.50	0.001	0.001	7.23	2.1	0.22	0.04

	Total B	^{11}B	^{10}B	^{10}B (at%)
std	tr.	tr.	tr.	tr.
nB	0.0060	0.0050	0.0010	0.0060
10B	0.0058	tr.	0.0058	0.0324

O Normalizing: 1040°Cx30min
 O Tempering: 740°Cx90min

IRRADIATION MATRIX OF F82H AND B-DOPED F82H

SPECIMENS

OHFIR Irradiation

• SS-3 Sheet Tensile

$$W1 = 1.52 \text{ mm}$$

$$W2 = 1.52^{+0.025}_{-0.038} \text{ mm}$$

○JMTR/JRR-2 Irradiation

- Full-size Charpy (10x10x55 mm; 45°x2mm)
- Round-bar Tensile ($\phi 4 \times 20$ mm)

Tensile Test Results of F82H with and without ^{10}B -doping

Tensile Test Results of F82H with and without ^{10}B -doping

Tensile Test Results of F82H with and without ^{10}B -doping

TENSILE PROPERTIES OF F82H AND B-DOPED F82H

TENSILE PROPERTIES OF F82H AND B-DOPED F82H

TENSILE PROPERTIES OF F82H AND B-DOPED F82H

TENSILE PROPERTIES OF F82H AND B-DOPED F82H

Charpy Impact Test Results of F82H with and without ^{10}B -doping

Charpy Properties of F82H and F82H+10B (400°C/0.4 dpa Irradiation)

Charpy Properties of F82H and F82H+10B (400-530°C Irradiation)

SUMMARY OF MICROSTRUCTURE

		F82H org		F82H+10B
		250°C	300°C	300°C
		2.5dpa	57 dpa	57 dpa 320appmHe
Dislocation Loops*	Number Density	$2 \times 10^{22} \text{ m}^{-3}$	$4 \times 10^{22} \text{ m}^{-3}$	$6 \times 10^{22} \text{ m}^{-3}$
	Average Size	8 nm	8 nm	11 nm
Cavities	Number Density	-	-	$2 \times 10^{21} \text{ m}^{-3}$
	Average Size	-	-	3 nm

* $b = (a/2)\langle 111 \rangle$ on $\{111\}$

OBSERVED PHENOMENA

Tensile Properties

- Small increase in strength and slightly lower elongation. (400°C)
- Less elongation at high temperature above 550°C.
- Less ductility (elongation and RA) at 250°C, but no difference in strength.

Charpy Properties

- DBTT shift to above RT occurred by 250-350°C irradiation to 0.5 dpa (120 appmHe).
- Wider transient temperature range appeared by irradiation above 400°C (>50 appmHe), however, USE stayed the same level of boron-free alloy.

HELIUM EFFECT IN TENSILE PROPERTIES (?)

Fracture Mechanism (?)

SUMMARY

- Helium effects on tensile properties appear as low elongation and reduction of area, however, it does not seem critical.
- Helium caused some shift in transition behavior and DBTT in Charpy properties, but did not decrease USE above 400°C.
- Boron doped specimens had almost the same microstructure as boron-free alloy, except for small cavities after 300°C irradiation to 57 dpa.

IEA Workshop on Reduced-Activation Ferritic/Martensitic Steels

He - Effects on Mechanical Properties and Microstructure of RAFM Steels after Irradiation

**E. Materna-Morris, A. Möslang, R. Lindau, M. Rieth, K. Ehrlich,
Forschungszentrum Karlsruhe
Institut für Materialforschung I + II**

Petten, 1-2 Oct. 1998

■ **Dual Beam He implantation**

■ **HFR irradiation**

Mechanical tests:

Tensile tests

Impact tests

Fracture and microstructural analysis

Experimental

Chemical composition (wt%):

	Cr	W	Mn	N	Ta	C	V	B	Si	Ni	Mo	Nb	Ti	Fe
F82H	7.73	2.06	0.083	0.003	0.018	0.092	0.189	<0.002	0.09	0.032	0.0053	0.006	0.010	bal
F82Hmod	7.7	1.95	0.16	0.008	0.009	0.088	0.162	0.0002	0.11	0.021	0.004	0.0002	0.01	bal
MANET-1	10.8		0.76	0.023		0.13	0.22	0.0085	0.14	0.66	0.59	0.14		bal

Alpha-particle cyclotron irradiations:

	Charpy-V	Tensile	Fatigue
Irrad. temperature (°C)	250	60-550*	250*
α-particle energy (MeV)	0-104	0-60	0-104
Damage rate (dpa/s)	$(4.5-7.5) \times 10^{-7}$	$(1.4-1.8) \times 10^{-6}$	$(4.5-7.5) \times 10^{-7}$
He Impl. rate (appm/s)	$(6-10) \times 10^{-4}$	$(2.5-3.0) \times 10^{-3}$	$(6-10) \times 10^{-4}$
Damage dose (dpa)	0.22	0.30	0.30
He concentration (appm)	300	500	400

*Irradiation temperature equal test temperature

Tensile Properties F82H-mod not irradiated

Experimental conditions:

Sheet tensile specimens (gauge volume: 7.0 mm x 2.0 mm x 0.20 mm)

Small strain rate ($1.2 \times 10^{-4} \text{ s}^{-1}$)

Standard heat treatment ($1040^\circ\text{C}/0.5\text{h} + 750^\circ\text{C}/2\text{h}$)

Vacuum furnace

Main results:

- Compared to MANET steel much lower yield strength and ultimate tensile strength
- In contrast to MANET steel no dynamic strain aging
- Temperature dependent microstructural recovery similar to conventional 9-12 CrMoVNb steels

Helium and damage effects

Tensile properties Helium and damage effects

Tensile properties Helium and damage effects

SEM - Micrographs of F82H Tensile Specimen

$T_{\text{test}} = 60\text{ }^{\circ}\text{C}$

side view
tilt angle: 0°

$T_{\text{test}} = 60\text{ }^{\circ}\text{C}$

side view
tilt angle: 30°

TEM -Micrographs of F82H Charpy Specimen
after Dual-Beam Irradiation

300 appm He
0.2 dpa
 $T_{irr} = 250\text{ }^{\circ}\text{C}$

high
disl. density
 $d_b = 10.5\text{ nm}$
 $\rho_b = 1.5 \times 10^{22}\text{ m}^{-3}$

low
disl. density
 $d_b = 2-5\text{ nm}$
 $\rho_b = 3.7 \times 10^{22}\text{ m}^{-3}$

Irradiation Hardening in f/m steels

Tensile Properties

Irradiation hardening $\Delta\sigma$ of Fe and F82H-mod for different irradiation sources and He/dpa ratios

- [1] Y. Chen, P. Spätig and M. Victoria (ICFRM-8)
 590 MeV proton irradiation from PIREX facility at 320 K;
 average damage rate = 1.3×10^{-6} dpa/s; He/damage rate ≈ 80 appm He/dpa.
- [2] A. Möslang
 75 MeV α -particle irradiation from Dual Beam Facility at 333 K;
 damage rate = 1.1×10^{-7} dpa/s; He/damage rate ≈ 20 appm He/dpa;
 The Fe and F82H-mod specimens were irradiated together.
- [3] R. Lindau*, E. Materna-Morris*, A. Möslang*, D. Preininger*, M. Rieth* and
 H.D. Röhrig (ICFRM-8)
 (0-60) MeV α -particle irradiation from Dual Beam Facility at 333 K;
 damage rate = 1.6×10^{-6} dpa/s; He/damage rate ≈ 1670 appm He/dpa

DPA dependency of the irradiation induced Δ DBTT shift

The Effect of Helium and DPA on DBTT

Impact properties Helium and damage effects

Depending on the He/dpa ratio, either He or dpa damage has the main contribution to $\Delta DBTT$:

Low dpa level (0.2 dpa, 250 °C):

F82H: 8 appm He HFR irradi.: $\Delta DBTT = 18\text{ °C}$
F82H: 300 appm He Implantation: $\Delta DBTT = 44\text{ °C}$
→ He dominates embrittlement

Medium dpa level (0.8 dpa, 250 °C, HFR-irrad.)

MANET-1:	70 appm He	$\Delta DBTT = 145\text{ °C}$
OPTIFER II:	60 appm He	$\Delta DBTT = 95\text{ °C}$
F82H	<20 appm He	$\Delta DBTT = 60\text{ °C}$
ORNL 3791:	<10 appm He	$\Delta DBTT = 25\text{ °C}$

→ The He contribution to $\Delta DBTT$ cannot be determined quantitatively, because He and alloying effects (B, Ni doping) can hardly be separated.

Microstructure at 250 °C:

- He-bubbles are observed both after cyclotron implantation and reactor irradiation
- Non-homogeneous bubble size distribution
($\approx 10\text{ nm}$ in disloc. networks, $\leq 2\text{-}3\text{ nm}$ in matrix)

Phenomenological embrittlement model

- Correlation between dynamic quasi cleavage fracture and irradiation induced changes of strength and fracture stress. Micro-mechanical and microstructural details are not yet completely included.
- It predicts that the relative shift $\Delta DBTT/DBTT_0$ should be basically proportional to $\sqrt{dpa}$ and to $\sqrt{c_{He}}$.

Forschungszentrum Karlsruhe

Technik und Umwelt

HFR-irradiation: 0,8dpa, 250....450°C

Chemical compositions of the alloys

	MANET II	OPTIFER Ia	OPTIFER II	F82H	ORNL 3791
• Cr	9.94	9.3	9.5	7.73	8.9
• C	0.1	0.10	0.125	0.092	0.11
N	0.023	0.0155	0.0159	0.0027	0.0215
Mn	0.79	0.50	0.49	0.083	0.44
P	<0.006	0.0046	0.0043	0.003	0.015
S	<0.007	0.005	0.002	0.003	0.008
V	0.22	0.26	0.28	0.189	0.23
• B	0.007	0.0061	0.0059	0.003	<0.001
Si	0.14	0.06	0.038	0.09	0.21
• Ni	0.66	0.005	0.005	0.032	<0.01
• Mo	0.59	0.005	0.005	0.0053	0.01
Al	<0.2	0.008	0.008	0.01	0.017
Co	<0.2			0.0024	0.012
Cu	<0.01	0.035	0.007	0.0059	0.03
• Nb	0.14	0.009	0.009	0.0057	
• Zr	0.034				<0.001
• Ti				0.0104	<0.01
• W		0.96	0.006	2.06	2.01
• Ta		0.066	0.018	0.018	0.06
Ge			1.2		
• Ce		<0.001	<0.001		
Fe	bal	bal	bal	bal	bal

Tensile specimen

Charpy specimen

Table 2 — Heat treatments, grain size and hardness of the alloys

Heat	Heat treatment	ASTM- grain size	Hardness HV 0.4
OPTIFER Ia	1075 °C 30 min + 780 °C 2 h	6.5	199
OPTIFER II	950 °C 2 h + 780 °C 2 h	7.5	206
F82H	1040 °C 30 min + 750 °C 2 h	6.5	227
ORNL 3791	1050 °C 30 min + 750 °C 2 h	8.5	245
MANET II	965 °C 2h + 1075 °C 30 min + 750 °C 2 h	7.5	274

Table 5 — Δ Hardness after irradiation at 250 and 450 °C.

Heat	Δ Hardness	
	250 °C	450 °C
OPTIFER Ia	31	-1
OPTIFER II	26	-8
F82H	18	-9
ORNL 3791	15	-11
MANET II	53	-2

Table 8 — Calculated helium content after complete ^{10}B burn-up irradiation temperature and the Δ -values of the mechanical properties after $T_{\text{irrad.}} = 250\text{ }^{\circ}\text{C}$.

	OPTIFER Ia	OPTIFER II	F82H	ORNL 3791	MANET II
max. He	60 appm	60 appm	<20 appm	<10 appm	70 appm
ΔDBTT	105	95	65	25	145
$\Delta R_{p0.2}$	83	22	39	-	-
ΔR_m	-7	-62	-35	-	-
ΔA	-7.5	-9.2	-6.2	-	-

Figure 12 — The DBTT shift depending on the He content.

OPTIFER II: 0.8 dpa, $T_{irr.} = 450\text{ }^{\circ}\text{C}$

OPTIFER II: 0.8 dp $T_{irr.} = 250\text{ }^{\circ}\text{C}$

Forschungszentrum Karlsruhe

Technik und Umwelt

Quantitative analysis of the irradiation effects at 250 °C.

Heat	He-bubbles diameter [nm]	He-bubbles density [cm ⁻³]	He-bubbles location	Density of loops [cm ⁻³]	density of α' [cm ⁻³]
OPTIFER Ia	2	1.5×10^{16}	D, H	5.9×10^{15}	1.0×10^{16}
OPTIFER II	3	3.8×10^{18}	D	2.6×10^{17}	2.1×10^{15}
F82H	1.5	$1.4 \times 10^{17*}$	H, S, P	1.9×10^{16}	7.5×10^{15}
ORNL 3791	2	$1.1 \times 10^{15*}$	D, P	1.0×10^{15}	3.5×10^{10}
MANET II	2	3.6×10^{16}	D, S, P	1.4×10^{15}	3.8×10^{15}

Quantitative analysis of the irradiation effects at 450 °C.

Heat	He-bubbles diameter [nm]	He-bubbles density [cm ⁻³]	He-bubbles location	Density of loops [cm ⁻³]	Density of α' (+) [cm ⁻³]
OPTIFER Ia	2	5.7×10^{15}	D, S, P	-	-
OPTIFER II	6	1.2×10^{15}	D, S	-	-
F82H	3	$3.7 \times 10^{16*}$	H, D	-	1.6×10^{16}
ORNL 3791	3	$4.6 \times 10^{15*}$	D	-	-
MANET II	4	1.1×10^{16}	D, P	-	-

D = Dislocations, S = Sub-grain boundaries, P = Precipitates,
H = Homogenous distributed, * = Maximum concentration

OPTIFER II: 0 dpa
Impact test at -60 °C (DBTT)

Impact test at 50 °C (DBLL)
OBLIFER II: 0.8 qbs, $T_{1/2} = 520$ °C

OPTIFER II: 0.8 dpa, $T_{irr.} = 450\text{ }^{\circ}\text{C}$
Impact test at $-20\text{ }^{\circ}\text{C}$ (DBTT)

OPTIFER Ia: 0 dpa
Impact test at -80 °C (DBTT)

**OPTIFER Ia: 0.8 dpa, $T_{irr.} = 450\text{ }^{\circ}\text{C}$
Impact test at $-60\text{ }^{\circ}\text{C}$ (DBTT)**

ORNL 3791: 0 dpa
Impact test at -60 °C (DBTT)

ORNL 3791: 0.8 dpa, $T_{irr} = 250\text{ }^{\circ}\text{C}$
Impact test at $-60\text{ }^{\circ}\text{C}$ (DBTT)

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Further Assessment of Helium Induced Embrittlement in RAMS

**A. Kimura
Institute of Advanced Energy
Kyoto University**

**IEA Workshop on Reduced Activation Ferritic/Martensitic Steels
Oct.1-2, 1998 ECN, Petten, Netherlands**

Back Ground (1)

I.A.E, Kyoto University

- 1) 9Cr-2W martensitic steels are much more highly resistant to irradiation-induced degradation than the other candidates, which results from the large capacity of the martensitic structure trapping point defects.
- 2) It is considered that the trapping effect works on transmutation helium, resulting in the high resistance to the helium embrittlement in martensitic steels, since the martensitic structure prevents helium atoms from segregating at a crack nucleation site, such as grain boundary or in the matrix.
- 3) However, the results of isotope tailoring experiments utilizing addition of nickel or boron-10 suggested severe helium embrittlement of the martensitic steels. Helium implantation experiment of F82H also showed a larger shift of DBTT than neutron irradiated one.

Back Ground (2)

I.A.E, Kyoto University

4) In contrast, the DBTT shift of a 9Cr-2W steel implanted with 120at.ppm of helium by cyclotron was explained in terms of displacement damage, suggesting no helium embrittlement of the steel. Furthermore, irradiation hardening of martensitic steel was three times larger in the 1%Ni added steel than without the addition when irradiated at temperatures less than 170°C up to only 0.15dpa.

Methodology

I.A.E., Kyoto University

●Concerns;

1) Isotope Tailoring Experiment

There is a concern about the influence of irradiation induced precipitates, such as nickel silicides and borides, on the irradiation embrittlement.

2) Reactor Irradiation Experiment

A low temperature irradiation during shut down high flux reactor easily give influences on the microstructure and/or mechanical properties of martensitic steel.

●Approach to get real effects of helium;

- * Helium implantation
- * Temperature monitoring
- * Small specimen technology

Experimental

I.A.E, Kyoto University

Material

RAMS; 9Cr-2W-V,Ta,Ti, B

◎ SP test

Test Temp.: RT~77K

Cross Head Speed: 0.2mm/min

◎ SEM, TEM Observation

◎ Annealing

Temp.: RT~873K

Period: 1hour

Vac.: 5.0×10^{-7} torr

◎ Vickers Hardness Measurement

Loading weight: 200g

◎ Positron Life Time Measurement

Hardness Measurement

I.A.E, Kyoto University

He Implantation

I.A.E, Kyoto University

● Cyclotron Irradiation

- 1) Beam; 36MeV- α
- 2) Degradar
- 3) Temperature; $< 150^{\circ}\text{C}$
- 4) He conc.;
120, 580at.ppm
- 5) Displ. Damage;
0.048, 0.232dpa
- 6) Range; 0.23mm

SP Energy - Temperature

I.A.E, Kyoto University

SEM fractograph

I.A.E, Kyoto University

10µm

9Cr-2W steels

Base: 0.01dpa
Base: 0.05dpa
Base: 0.15dpa

1%Ni Unirr.
1%Ni 0.01dpa
1%Ni 0.05dpa
1%Ni 0.15dpa
JLM-1: Unirr.

Yield Stress / MPa

580appm
120appm
Unimplanted

$\sigma_y = 1.54 \times H_V + 213$

Tensile tested at R.T.
Irradiated at 493K in the JMTR

HV

580appm(0.226dpa): $\Delta \sigma_y = 148 \text{ MPa}$

Dose dependence of $\Delta\sigma_y$

I.A.E. Kyoto University

- 9Cr-2W(JLM-1) at 353K(80°C)
- ◆ 9Cr-2W-1Ni at 493K(220°C)
- 9Cr-2W-LowSi/Mn at 493K(220°C)
- 9Cr-2W-LowSi/Mn at 613K(340°C)
- ▼ 9Cr-3W-VTa at 538K(265°C); Abe et al.(NRIM)
- He-implanted JLM-1 below 423K(150°C)

● The estimation of $\Delta\sigma_y$ in the $1/4$ power law, which was derived from the data obtained for the neutron irradiated material, where the transmutation He was negligible, overestimates the hardening by He implantation, due to the displacement of atoms and therefore no calculation of relation on the hardening.

Estimation of ΔDBTT - CVN10

I.A.E, Kyoto University

SP- DBTT(K)

=0.4 × CVN10- DBTT(K)

120appm : $\Delta\text{DBTT-CVN10} = 50\text{K}$

580appm : $\Delta\text{DBTT-CVN10} = 83\text{K}$

I.A.E., Kyoto University

Relationship between $\Delta\sigma_y$ and $\Delta DBTT$

● Obeying the linear relationship; # $\Delta\sigma_y$ (Hardness meas.), # $\Delta DBTT$ (SP-tests)

- 1) SP-test technique is predictable the $\Delta DBTT$ -CVN10.
- 2) A linear relationship indicates the DBTT shift is caused by hardening of the matrix (not by reduction of fracture stress such as grain boundary embrittlement).

Annealing Behavior (Hv)

I.A.E, Kyoto University

- He implantation reduced the recovery of irradiation hardening, suggesting that the thermal stability of I-clusters was increased by helium.
- There is a possibility that helium retard the recovery of dislocation structure and reduces the thermal creep rate.

TEM micrograph

I.A.E, Kyoto University

He-implanted up to 580appm

100nm

TEM micrograph

I.A.E, Kyoto University

Post-implantation annealing up to 873K

200nm

PAS Measurement

I.A.E, Kyoto University

●He implantation reduced the size of microvoids and increased their density. (suppress the vacancy migration, resultantly the growth of microvoids)

Annealing Behavior (PAS)

I.A.E, Kyoto University

- No decrease in the intensity of the longer life time component, τ_2 , was observed in the 9Cr steel implanted with He following the annealing to 600°C.
- He stabilized the microvoids.

TEM micrograph

L. I. E. Kigoto University

Post-implantation annealing up to 873K
(Cavity image)

100nm

Conclusion

I.A.E, Kyoto University

● Effects of He implantation (580atppm) on DBTT ●

- Fracture mode; **CLEAVAGE**
- Δ DBTT (CVN-10); 83K (explained in terms of hardening mechanism)
- $\Delta\sigma_y$ ($\Delta H_v=90$); 150MPa (explained in terms of **DISPLACEMENT DAMAGE**)
- Linear relationship; Δ DBTT and $\Delta\sigma_y$ (validity of SP technique to estimate Δ DBTT (CVN-10))

● Effects of He implantation (580atppm) on microstructure ●

- V, I-cluster;
 - 1) He reduced the size of V-clusters but increased the density
 - 2) He increased the thermal stability of V-clusters
- Hardening mechanism; good relation to the I-cluster behavior

● Thus, 580atppmHe does NOT enhance the irradiation hardening and/or embrittlement of RAJS which consists of martensitic structure containing a number of trapping site for He atoms.

● Next step is to find out the critical He concentration to cause He induced embrittlement.

Post Irradiation Welding and Helium generation with ^{10}B in RAFM steel

IEA-RAFM Workshop
Petten, 1-2 October 1998

E.V. van Osch, M.G.Horsten, D.S. d'Hulst, E.W. Schuring
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30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Outline

- Post Irradiation Weldability
- Status PI Welding
- Helium generation with ^{10}B
- Conclusions

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Post Irradiation Weldability: Why?

- In view of Repair/Replacement of Irradiated FM components
- Austenitic steels (ITER/BWR/PWR): Weld Cracking due to He-bubble coalescence and growth on Grain Boundaries during Cool-down; cracking threshold ~1 appm He.
- Mitigation strategies:
 - reduce amount helium
 - low heat input welding methods: e.g. Laser welding
 - reduce/compensate thermal stresses
- FM-steels expected to have much higher He-tolerance
- DEMO He-levels up to 100's appm

10-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Status Welding of Irradiated F82H

- Irradiated F82H available: 2 dpa (~5 appm He) 1, 3 and 5 mm plates and limited amount 2.5 dpa 1 mm plates.
- First 1, 3 and 5mm TIG welded heterogeneous joints (irr. to unirr) succesful, no external defects (SEM)
- 1 and 3 mm plates: single pass, no filler wire
- 5 mm plate: Y-groove, 4-6 passes

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

Current and future activities:

- Cross section inspection of TIG welds
- Nd:YAG Laser welding 1 mm plates
- Tensile test
- Welding of 10 dpa irradiated F82H: 1 and 2.5 mm plates

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

In-cell TIG welding facility

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

In-cell TIG welding facility

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

F82H 1mm TIG weld 2 dpa irradiated (left) unirradiated (right)

30-Sep-98

IEA-RAFM, 1-2 Oct 1998, Petten

NRG
AN NRC BUREAU COMPANY

Helium generation with ^{10}B

- Ordered 3 65kg lab. heats:
 - one identical to “ECN-BS” -> up to 10 appm He due to ^{10}B
 - one with 50 wppm ^{10}B addition -> 250 appm He
 - one with 50 wppm ^{11}B addition -> 10 appm He

30-Sep-98 IEA-RAFM, 1-2 Oct 1998, Petten

Conclusion

- First TIG welding experiments started:
 - Preliminary results do not show external defects
 - Cross sections being made
- Ordered several lab heats:
 - the reproduce the promising ECN-BS results
 - included ^{10}B and ^{11}B additions to study He (and B) -effects

Strategy for Development of Ferritic/Martensitic steels

EU STRATEGY FOR RAFM DEVELOPMENT

Presentation at IEA RAFM Workshop
Petten, 2 October 1998

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OVERVIEW

- Main goals
 - Milestones period 1998 - 2009
 - Major subjects
 - Time schedule
-

MAIN GOALS

- Engineering RAFM data for ITER test module, ITM, in 2004
 - Manufacturing procedures for ITM, RAFM parts in 2004
 - DEMO relevant design data in 2009
 - High temperature ODS RAFM basis established in 2009
 - In case of ODS RAFM success demo relevant design data in 2015
-

ECN

MILESTONES STRUCTURAL MATERIALS SUBPROJECT

Version: 14 July 1998

Author : B. van der Schaaf

Mile-stone	Time Scale	Items available at milestone date	EBP project phase
4	98/11	<ul style="list-style-type: none">- Phase IA neutron irradiation- Character irradiated welds- F82 characterization- Screening test alloys complete	Selection ITM Primary candidate Alloy.
5	99/06	<ul style="list-style-type: none">- Preliminary ITM design rules- Data Base Evaluation- Corrosion data for design	First version ITM design rules and data base.
6	00/06	<ul style="list-style-type: none">- Evaluation data EUROFER97- Specification EUROFER2000- Phase II neutron irradiation- Compatibility	Selection ITM structural steel specification and primary radiation effect limits.
7	01/05	<ul style="list-style-type: none">- ITM design rules- Data Base evaluation- Compatibility	Second version ITM design rules and data base.
8	02/03	<ul style="list-style-type: none">- Qualification and Fabrication processes- Metallurgical and Mechanical Character	Confirmation of materials and fabrication feasibility.
9	02/12	<ul style="list-style-type: none">- Irradiation Performance- Compatibility- ITM rules and data base	Rules and data base available for detailed ITM design

MILESTONES STRUCTURAL MATERIALS SUBPROJECT

Project period: 2003 - 2009

Mile-stone	Time Scale	Items available at milestone date	Project phase
11	03/09	<ul style="list-style-type: none">– Screening ODS RAFM– Manufacturing ODS RAFM	Advanced HT RAFM
12	04/09	<ul style="list-style-type: none">– RAFM design data verification– Definition additions	Consolidation ITM design
10	03/06	<ul style="list-style-type: none">– Materials for 150 dpa irradiations available– Rigs and reactor choice	High dose RAFM properties for DEMO
13	05/06	<ul style="list-style-type: none">– Choice reference ODS RAFM alloy– Joining, manufacturing routes– Assessment data	HT RAFM engineering data
14	06/06	<ul style="list-style-type: none">– High dose RAFM PIE– Design limits	High dose RAFM data for DEMO
15	07/09	<ul style="list-style-type: none">– Assessment ODS RAFM potential– Decision on continuation	Advanced HT RAFM development
16	09/09	<ul style="list-style-type: none">– Engineering data 150 dpa validated	High dose RAFM data for DEMO

EBP, STRUCTURAL MATERIALS PROGRAM

WP	TASK
SM 1	Irradiation Performance
SM 2	Metallurgical and Mechanical Characterization
SM 3	Compatibility with Liquids and Hydrogen
SM 4	Qualification of Fabrication Processes
SM 5	Rules for Design, Fabrication and Inspection
SM 6	Qualification for DEMO

Version: 980925

Author : B. van der Schaaf

Fusion Materials Programs in Japan

Specialist Group Organized under Fusion Council of JAEC

A brief Summary of the Draft Report as of 24th July, 1998

- A specialist group organized under The Japanese Atomic Energy Commission (JAEC) has been discussed on how could R & D of the first-wall structural materials be carried out timely and efficiently. A draft report of this activity was provided in June, 1998.
- The draft report will be subjected to discussion at the Fusion Council audited by the public and then, will be opened to the public reviewing for a period of time.
- The report is expected to be finalized by Atomic Energy Commission in September, 1998.

Summary of Report (draft as of July, 1998)

Strategy of Fusion Materials Development

- First Wall Structural Materials -

- ☐ Definition of evolutionary stages for the class of materials
 - Exploratory/Development Stage
 - Optimization/Verification Stage
 - Engineering Application Stage
- ☐ Candidate materials and their strategic priority - Three candidate material categories are selected and prioritization is made from view points of their technical maturity and potential properties
 - The first priority category : prime candidate material
 - Reduced activation ferritic steels**
 - Current stage; Optimization / Verification Stage
 - Current tasks; Establishing database for base line properties and selecting the final candidate specification
 - The secondary priority materials category: advanced materials

Vanadium alloys, and SiC/SiC composites

Current stage; Exploratory / Development Stage

☐ Strategy for development procedure

- Development studies for the three candidate materials have been carried out in parallel and will be continued up to the point where a definitive conclusion for candidacy is shown, while the current priority is now given to the reduced activation ferritic steels.
- The potential suitability of the three categories of materials are checked and reviewed every five years, in terms of technical maturity and their established properties regarding the requirements of the systems to be applied.
- The strategic goal for DEMO reactors is set around the year 2015 after the third evaluating stage.

☐ General comments

- A high energy- and high flux neutron source with adequate test capacity (e.g. IFMIF) is indispensable for the development and evaluation of candidate materials, with conventional supplementary test means being utilized.
- Technical review / evaluation of the three candidate materials is continually necessary in order to invest funding and manpower efficiently.
- Importance of close collaborative domestic activities among national institutes, universities, and the industrial firms in parallel with international cooperation, is emphasized for achieving the mission. JAERI is expected to serve as a core organization for the domestic activities.

STRUCTURAL MATERIALS DEVELOPMENT FOR FUSION REACTOR

***U.S. Department of Energy
Fusion Energy Science Program
and the
Fusion Materials Program***

F. W. Wiffen

Office of Fusion Energy Sciences
U.S. Department of Energy

IEA Ferritic Steels Workshop

Petten, The Netherlands

October 1-2, 1998

U.S. Fusion Energy Sciences Program Mission

“Acquire the knowledge base needed
for an economically and environmentally
attractive fusion energy source.”

Technology Program

Transition Highlights

- o ITER tasks have been the focus through FY 1998
- o In FY 1999, will begin transition to broad portfolio of activities serving the domestic program and our needs for international collaborations
- o Transition will be completed in FY 2000; Technology Program will emphasize **enabling technologies** for plasma experiments, domestically and internationally

Program Goals -- Fusion Materials

o Ultimate Goal

- To provide validated materials and the engineering data base for structures in the high neutron flux regions of Fusion Power Systems

o Interim Goals

- Develop candidate materials with the potential of meeting the needs established by conceptual power system design
- Build the understanding needed to predict the effects of the fusion environment on candidate materials
- Understand the effects of compositional and microstructural variables in optimizing materials for service

General Approach -- Fusion Materials

- o Include only materials systems that show promise for application in high temperature service and meet low activation criteria
- o Focus on questions of feasibility for fusion use and critical issues for acceptable service lifetime
- o Concentrate on aspects unique to fusion applications
 - The fusion chemical environment
 - Neutron irradiation effects in the D-T fusion spectrum
- o Balance efforts among candidates for promising design concepts, especially related to choice of coolants
- o Develop understanding at the controlling mechanisms level, so that results can be used to interpolate/extrapolate from available data base

Fusion Energy Sciences FY 1999 Budget

Program Elements

\$228.2 Million

Materials Research Categories

\$6.2 Million

Presidential Budget

Materials Program

(\$ in Millions)

Budget Summary

FY 1998*

FY 1999

8.3

7.0

Goal:

- o Explore innovations in materials technologies needed in the long term to advance fusion science and to achieve fusion's potential as an attractive energy source
 - Focus on low-activation structural materials (vanadium alloys, ferritic steels, and silicon carbide composites) for high power density fusion devices
 - Small complementary efforts on non-structural materials research is conducted under Fusion Technologies (coolants, insulators, coatings and tritium breeders) and Plasma Technologies (plasma facing materials)

*ITER activities are included

Fusion Materials Program

The Modeling Component of the Fusion Materials Program will be expanded

Goal is to examine what is being done now, and to define what is needed in a comprehensive modeling program

Changes will be made to assure that the most important elements of the modeling activity are in place

Performance Targets for Fusion Reactor Structural Materials

- o Develop materials that will support economically attractive fusion power reactor designs
 - Fabricability, weldability, joining technology for field assembly
 - Compatibility with operating environment (coolant, tritium breeder, ambient atmosphere, special purpose coatings, hydrogen plasma)
 - Adequate performance design window (operating temperature limits, stress limits, irradiation damage)
 - Reliability and maintainability
- o Develop materials that will achieve the potential of fusion as an environmentally attractive energy source
 - Materials not requiring long term geological disposal
 - Potential for recycle of materials to minimize environmental impact
- o Develop materials that will achieve the potential of fusion as a safe energy source from the viewpoint of the worker and the public
 - Acceptable levels of heat from radioactive decay and chemical reactions
 - Limited dispersability of radioactivity
 - Acceptable BHP (needs to be defined)

Technological Challenges to Establish Feasibility

Advanced Ferritic Steels

- o Establish that ferromagnetic structural materials are acceptable for MFE concepts
- o Establish the potential design window for advanced ferritic steels -- focus on the role of displacement damage concurrent with helium production on low temperature fracture behavior
- o Determine the effects of nuclear transmutations (e.g., production of He, burnout of W and Ta)
- o Resolve system specific compatibility issues (e.g., T barrier for He or PbLi coolants)

Fusion Materials Program

Issues with the use of ferromagnetic steels in magnetically confined fusion systems require a more complete evaluation

1. The materials program (R. L. Klueh) has surveyed the information available worldwide
2. Experiments in Japan need to be monitored -- HT-2 and JFT-2M structures using ferritic steel components
3. Complete analysis can only be done by the fusion physics/design analysis teams, not by the materials community

Technological Challenges for the Materials Development Step

Advanced Ferritic Steels

Explore oxide dispersion strengthening as an approach to improving the high temperature strength of martensitic steels while maintaining acceptable low temperature toughness characteristics

Focus on Research in the Advanced Materials Program

Advanced Ferritic Steels

- o Carry out research to determine and understand the effects of displacement damage concurrent with helium on the low temperature fracture behavior of F82H and closely related compositions
- o Explore oxide dispersion strengthening as an approach to improving the high temperature strength of martensitic steels while maintaining acceptable low temperature toughness characteristics

Technology Program

Transition Highlights

- o **Virtual Laboratory for Technology** (VLT) is a mechanism for organizing and integrating work of many institutions
 - Coordinating program elements
 - Advocacy and representation
 - Governance with an external Program Advisory Committee and an internal Coordinating Committee

A Comment for RISO Meeting
- Materials R & D Strategy Proposal -

Institute of Advanced Energy

Kyoto University

A. Kohyama

Institute of Advanced Energy

Kyoto University

IEA RAF Workshop
Oct.1-2, 1998 ECN, Petten, Netherland

What is the definition of Strategy ?

- what are we (they) trying to do at RISO? -

Institute of Advanced Energy

Kyoto University

Strategy meeting is not a place for a simple information exchange

In military usage, a distinction is made between STRATEGY and TACTICS. STRATEGY is the utilization, during both peace and war, of all of a nation's forces, through large-scale, long-range planning and development, to ensure security or victory. TACTICS deals with the use and deployment of troops in actual combat.

(from Random House dictionary)

Q1: for security or victory ?

Q1-1: are we in peace or in war ?

Q2: do you discuss TACTICS ?

Q2-1: what is the goal ?

Q2-2: where do you fight ?

Q2-3: how do you fight ?

Where does the Japanese fight and what for ?

Institute of Advanced Energy

Kyoto University

1: Fusion is one of the energy options for the future (in Japan)

- *a back-up or insurance to the other options* Make it an important option
- (ITER special committee report: Japan Domestic) or *to keep the position*

2: Fusion Research has three major areas (under MOE, STA)

- *fusion engineering is still a supporting activity* Make it the major one

3: Fusion engineering is under restructuring (toward MOE+STA)

- *materials comm. has a strong influence* Restru. Under materials' leadership

4: Materials research has to be more emphasized

- *goal of materials R & D is not clear* Get a better financial situation
- a clear strategy will make the way* within F.E. or even within materials

5: Materials community is under strong criticisms

- *should be involved other F.E. activities* Try not to be killed
- *materials R & D activity should be down sized*

Japanese National Strategy on Materials R & D for DEMO

(Version on 6/25/98 by JAERI)

Institute of Advanced Energy

Kyoto University

What is the real situation of RAFs ?

- what do you want to do ? -

Institute of Advanced Energy

Kyoto University

1: Can RAFs become materials for power reactor ?

- *need ODS options with strong will*

Strategy with a clear goal
of materials R&D

2: Can RAFs become candidate materials for ITER ?

- *RAF activity should lead fusion engineering*

Strategy with a clear goal
of materials R&D

3: Can RAFs keep their major position ?

- *RAF have been the most matured materials*

- *what is the next step ?*

Have to create a new phase
after the current IEA large
heats activities (bottom line for Japan)

4: Can RAFs maintain their activity ?

- *not likely to be survived as a part of
fusion energy research, as it is*

- *to move to energy science research or materials
science research is an option*

Strategy with a clear goal

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