Report of the Consultants’ Meeting on

Role of Research Reactors in Material Research for Nuclear Fusion Technology

13 - 15 December 2010
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1. BACKGROUND AND OBJECTIVES

Background
An ambitious programme on fusion materials is one of the main requirements to the successful development of future fusion energy. The fusion of deuterium and tritium produces helium and neutrons of 14 MeV, which is typically one order of magnitude higher than that of neutrons produced in fission reactors. This results in the production of extensive radiation damage in the bulk of the materials, a significant amount of helium and hydrogen, including other transmutation products, which can result in swelling and alteration of the mechanical properties. Therefore, the specificity of the reactions produced by the 14 MeV neutrons together with the operating conditions required for the materials (intense fast neutron fluxes and high operating temperatures in the range from 400 to 600 °C) constitute a challenge which fusion materials R&D has to take up.

Indeed, among a number of important technological issues related with the realization of a future nuclear fusion reactor, the availability of qualified structural materials, functional components and advanced joining technologies still has to be confirmed. Considering unprecedented features of the nuclear fusion radiation environment in terms of high-energy and high-flux neutron-photon fluxes, including high temperature and the presence of a magnetic field, completely new concepts of advanced material irradiation and testing facilities have been discussed and their development progressed since 1960s, converging to such dedicated projects as the Fusion Materials Irradiation Test (FMIT) and the International Fusion Materials Irradiation Facility (IFMIF).

While waiting for the construction and full-power operation of a dedicated material irradiation facility, high-flux material test reactors (MTR) remain a compromise solution to advance in this specific domain of material research, testing and qualification. It must be admitted that the irradiation environments, which research reactors are presently able to supply, are not fully equivalent to the required irradiation environments in nuclear fusion systems. However, both experimental studies and theoretical efforts in the last decades have produced more comprehensive bridges between them.

The purpose of this meeting was to bring together the teams from material test research reactors, spallation neutron sources, multiple beam irradiation facilities and the future IFMIF installation, all working on the theoretical, experimental, and engineering aspects of material research for fusion technology. The state of the art was to be discussed, areas of needed research and development be identified, and possibilities for enhanced collaboration be explored. The meeting was expected to re-examine and better define the role of research reactors in the development of nuclear fusion systems in an appropriate time span and within the international closer collaborations among fission- and fusion-related material research communities.

Objectives
The specific objectives of the meeting were:

- Discuss experimental conditions and parameters required at research reactors for their enhanced contribution to the development of nuclear fusion systems; recommend needed development

- Review correlations and links between nuclear fusion irradiation environments and the ones available or presently under development at research reactors; propose required efforts
• Discuss and propose dedicated validation experiments and modelling efforts to bridge the gap between research reactor-based test irradiations and qualification of materials for nuclear fusion technologies

• Discuss and recommend collaboration initiatives among fission- and fusion-related material research communities

The meeting also provided a forum to exchange ideas and information through scientific presentations and brainstorming discussions, leading to the following overall objectives: 1) enhancement of RR utilization in Member States for practical applications, 2) increased cooperation between different RR centres and user communities, and 3) promotion and development of specific applications of RRs.

2. WORK DONE AND RESULTS ACHIEVED

The consultancy meeting was attended by 14 participants from 10 Member States. The meeting started off with welcome, opening and introductory remarks by Mr Pablo Adelfang, head of Research Reactor Section, NEFW. Later a welcome address was given by Mr Danas Ridikas and Mr Richard Kamendje, the IAEA Scientific Secretaries of the meeting, Physics Section, NAPC, followed by a self-presentation of all meeting participants. Mr Eberhard Diegele (F4E, EU) was nominated as chair person and Mr Bob van der Schaaf (NRG, the Netherlands) was appointed as rapporteur of the meeting. Then Mr D. Ridikas (IAEA) outlined the specific objectives of the meeting within the ongoing IAEA project D2.01 on Enhancement of Utilization and Applications of Research Reactors.

All participants presented their views on the subject of this meeting. The presentations were followed by lively discussions amongst the participants. Further, intermediate summaries and compilations of findings and comments contributed to involving participants into the aims of the meeting and the strengthening of the exchange of knowledge and experience.

The Annexes of this report include: 1) requirements for structural materials in fusion 2) future work plan, 3) book of individual abstracts, 4) meeting agenda, and 5) list of meeting participants. Copies of the presentations, papers and administrative information were distributed at the end of the meeting to all participants and may be obtained from the Scientific Secretaries on request. The full meeting report as a working document is also available on request from the Scientific Secretaries.

2.1. Discussion on necessary experimental conditions for development of nuclear fusion systems

Fusion road map
The main commonly agreed elements in the overall road map toward commercial fusion reactors are ITER, IFMIF and a demonstration reactor, usually referred to as DEMO. The DEMO engineering design will be carried out based on the information supplied from ITER and IFMIF (and underlying science and technology programs).
Definition of DEMO

The meeting participants noted that today a clear and consistent definition of design parameters of a DEMO is missing, in terms of plasma physics, technology, and operational scenarios. In particular, the vision and the requirements in different countries vary quite significantly. Plasma physics mostly is defined as “ITER-like”, however, to what extent the mode of operation is continuous or consists of “long” pulses is under dispute. In addition, continuous electricity delivery to the grid may not be required at the first stage of operation. Similarly, the step in technology taken from ITER to a DEMO will be associated with different levels of risk, from very conservative to more advanced ones. It is anticipated that a DEMO might become operational in the 2030ties.

Finally, there could be more than one DEMO world-wide such as an early conservative DEMO followed by more advanced DEMO types. The “conservative” options are defined as those options that have the highest technical readiness commonly using Reduced Activation Ferritic-Martensitic, RAFM, steel and breeding blankets operated within the RAFM temperature window. The advanced options of DEMO, where advanced might be defined by improved overall thermal efficiency, may include the use of advanced blanket options as well as advanced plasma physics operation. Structural materials used with advanced breeder blanket designs such as V-alloys and SiC/SiC are usually called “advanced materials”. Fe-based oxide dispersion strengthened (ODS) materials hold an intermediate place which could be either applied for enhanced performance of the conventional options by reinforcing the RAFM structure or be applied for advanced high temperature blankets.

- In the above context the meeting participants concluded that even though the exact path from an early conservative to a more advanced DEMO cannot be detailed now, the international fusion community, as in the case of ITER, should develop reference documents on DEMO requirements. These DEMO reference documents will stimulate the selection of candidate materials to facilitate the fission community.
to enlarge cooperation and assist in fusion materials science and engineering related irradiation and research.

**Role of IFMIF**

Presently there are a few 14 MeV neutron sources available in the world, for example FNS in Japan and FNG in the EU. The intensity of these 14 MeV neutron sources is of the order of $10^{11}$ neutrons/s and therefore by many orders of magnitude too small to do technology-oriented experiments capable of underpinning designs for ITER, DEMO or IFMIF.

IFMIF is a neutron irradiation testing device specific with respect to:
- The neutron spectrum (most similar to fusion in-vessel components),
- The high neutron flux (allowing accelerated testing) at both, acceptable flux and temperature gradients across test samples for generating engineering data,
- Flexibility in loading conditions and instrumentation, but limited in available test volumes.

Consequently, the role of IFMIF in the initial period of operation is foreseen to be the supply of key data necessary for engineering design of DEMO. Given the limitations in time, irradiation volume and construction costs, candidate (structural) materials to be tested in IFMIF should have undergone a rigorous pre-selection procedure. The pre-selection will take into account evidence on radiation resistance to be proven in other irradiation campaigns including fission reactors, multi-ion beam facilities and spallation neutron sources. Materials to be tested in IFMIF should have prior established databases on irradiation properties. Those databases for a DEMO and IFMIF itself will have to be generated with fission RRs. That approach is reviewed in the next section 2.2.

**2.1. Review of irradiation environments in nuclear fusion and research reactors**

It is the common understanding that most of the data for DEMO design activities (Conceptual Design Activities, Engineering Design Activities, etc.) will have to be provided from irradiations at dedicated fission RRs. These facilities will continue to provide a large platform for the generation of a major part of the needed data base. The demands put on RRs will go beyond the operation of IFMIF with new materials, including new operational regimes of DEMO.

The contributions might come from a variety of fission RRs:
- High fluence/flux for mechanical properties and breeder functional materials
- Testing of (sub) components/mock-ups and integrated tests with bulky objects
- Medium fluence/flux for specific knowledge (e.g. instrumentation development)
- Low flux RRs in support of modeling, nuclear data measurements, etc.

The value of RRs for fusion materials characterization in addition to the availability of neutron flux is defined by specific needs driven objectives (e.g. irradiation of structural materials or functional materials) and scope of generated data (e.g. generic development, licensing, modelling verification). Depending on these objectives some requirements as precise temperature control, on-line measurements of tritium release, etc. might be considered essential or advantageous. Without being complete, the list of requirements includes:

(A) Main requirements for in-pile test
Working Material

- Materials palette
- Type of static and dynamic testing devices in-pile
  - Flux, instrumentation, control of parameters (control of temperature mandatory; other requirements, e.g. strain/stress, will depend on type of test)

(B) Advantageous requirements for peripherals
  - Access to hot cells and advanced test capabilities
  - Specific knowledge (corrosion, ..., handling of beryllium, ... ceramics)

(C) Characterization of structural materials
  - Full and deep characterisation for whole operational temperature window

(D) Characterization of functional materials
  - Breeder blanket
  - Coatings
  - Insulators / Windows

(E) Other issues
  - Validation of (irradiation) models/tools, and very precise control of all parameters
  - Contribution to fundamental understanding (science) of material behaviour in specific environment/conditions
  - Medium flux
    - Specialized, ... special experiments, technology, ... accessibility
  - Requirements are specific to the class and type mentioned above

As the below table indicates, at present there are about 15 RRs with ~50MW or higher thermal power, consequently with very high neutron fluxes, available for materials irradiation and research.

<table>
<thead>
<tr>
<th>Country</th>
<th>Name</th>
<th>Reactor Type</th>
<th>Thermal Power, kW</th>
<th>Thermal Flux, n/cm²/s</th>
<th>Fast Flux, n/cm²/s</th>
<th>Criticality Date</th>
</tr>
</thead>
<tbody>
<tr>
<td>Belgium</td>
<td>BR-2</td>
<td>TANK</td>
<td>100000</td>
<td>1.0E15</td>
<td>7.0E14</td>
<td>1981-06-29</td>
</tr>
<tr>
<td>Canada</td>
<td>HSU</td>
<td>HEAVY WATER</td>
<td>132000</td>
<td>4.6E14</td>
<td>4.5E13</td>
<td>1957-11-03</td>
</tr>
<tr>
<td>China</td>
<td>HESTA</td>
<td>TANK</td>
<td>120000</td>
<td>6.2E14</td>
<td>1.7E13</td>
<td>1979-12-27</td>
</tr>
<tr>
<td>France</td>
<td>FFER</td>
<td>HEAVY WATER</td>
<td>58300</td>
<td>1.5E15</td>
<td></td>
<td>1971-07-01</td>
</tr>
<tr>
<td>France</td>
<td>GISUS</td>
<td>POOL</td>
<td>70000</td>
<td>2.7E14</td>
<td>2.6E14</td>
<td>1966-09-08</td>
</tr>
<tr>
<td>India</td>
<td>FETS</td>
<td>FAST BREEDER</td>
<td>40000</td>
<td>3.8E13</td>
<td></td>
<td>1935-10-18</td>
</tr>
<tr>
<td>India</td>
<td>GHSUVA</td>
<td>HEAVY WATER</td>
<td>100000</td>
<td>1.8E14</td>
<td>4.5E13</td>
<td>1985-08-08</td>
</tr>
<tr>
<td>Japan</td>
<td>FFTR</td>
<td>TANK</td>
<td>50000</td>
<td>4.0E14</td>
<td>4.0E14</td>
<td>1969-03-30</td>
</tr>
<tr>
<td>Japan</td>
<td>KOE2</td>
<td>FAST, NA COOLED</td>
<td>140000</td>
<td>4.0E13</td>
<td></td>
<td>1977-04-24</td>
</tr>
<tr>
<td>Netherlands</td>
<td>NEP</td>
<td>TANK IN POOL</td>
<td>45000</td>
<td>2.7E14</td>
<td>5.1E14</td>
<td>1961-11-09</td>
</tr>
<tr>
<td>Russian Federation</td>
<td>BOR-60</td>
<td>FAST BREEDER</td>
<td>60000</td>
<td>2.0E14</td>
<td>2.5E13</td>
<td>1969-12-01</td>
</tr>
<tr>
<td>Russian Federation</td>
<td>MStN1</td>
<td>POOL/CHANNELS</td>
<td>100000</td>
<td>5.0E14</td>
<td>3.0E14</td>
<td>1966-12-01</td>
</tr>
<tr>
<td>Russian Federation</td>
<td>PH</td>
<td>PRESS. VESSEL</td>
<td>100000</td>
<td>3.0E13</td>
<td>2.0E13</td>
<td>1961-01-10</td>
</tr>
<tr>
<td>United States of America</td>
<td>ATR</td>
<td>TANK</td>
<td>250000</td>
<td>8.5E14</td>
<td>1.8E14</td>
<td>1967-07-02</td>
</tr>
<tr>
<td>United States of America</td>
<td>MFE5</td>
<td>TANK</td>
<td>200000</td>
<td>2.1E15</td>
<td>1.0E15</td>
<td>1965-08-01</td>
</tr>
</tbody>
</table>


Two types are distinguished: the moderated spectrum reactors with the potential of testing at very low temperature (below 200°C) and the fast spectrum reactors, where without very special arrangements test temperatures below 350°C are the lower limit. The radiation damage
rate of the fast spectrum RRs is twice or three times the value for moderated spectrum reactors. Some examples of the RRs with existing experience in and considerable potential for the experimental fusion field are:

- BOR-60, IVV-2M, BR-2, HFR, LVR15, and OSIRIS in Europe
- HFIR in the US
- CARR, CEFR, FBTR, JMTR, JOYO, in Asia

Some of these RRs have been recently overhauled. Others, especially in the EU will be closed, and replaced in the medium term by new facilities. For the longer term, in the next decades, the new-built RRs as CARR and CEFR in China, RJH, MYRRHA, and Pallas in the EU, and MBIR in the Russian Federation might become partly available for fusion development support. It is expected that neutron fluxes over $5 \times 10^{14}$ n/(cm$^2$ s) will be effective to contribute valuable data. High operational availability is another prerequisite to reach DEMO targets of 30 – 80 dpa in acceptable time. For fusion power plants (FPP) damage rates up to 150 dpa at the end of life are projected. Those levels will be difficult to simulate even in the most powerful RRs in reasonable irradiation time frames. Furthermore, the fission neutron spectrum does not provide the 14 MeV-based He, Hydrogen and dpa co-generation and the specific conditions related to the presence of magnetic fields. Here accelerated particle beams will support theory and models to qualitatively allow prediction of the phenomena relevant for DEMO.

In the longer term specific plasma volumetric sources with a real fusion environment-based He and dpa co-generation might offer the required irradiation volumes so dearly missed in the future IFMIF. The EVEDA effort for preparing the building track of IFMIF should result in final design (built-to-print) within the second half of this decade. As of today not any political decision and financial commitment has been taken, the predictions for the development and building path for such a volumetric source differ widely, and the starting date for the use of such a device is therefore unclear. Operation somewhere around 2025 is reasonably expected, but still should not be taken for granted.

It is for sure that fission RRs will provide the experimental results that support the design and reliability and consequently licensing of IFMIF components exposed to intense neutron irradiation, including those exposed to 14 MeV neutrons. For a DEMO the RRs will also have to contribute to the reliability of its operation, since large volumetric sources for testing of fusion materials, devices and (sub) components will only materialize late.

The main tasks for the fission RRs in the development of fusion materials are:

- Generate the major part of material properties for engineering design
- Evaluate and confirm performance limits
- Validate materials and component behaviour models on macroscopic scale
- Reduce risks of component malfunctioning during operation in DEMO
- Identify and learn about the critical operational conditions
- Explore unknown failure mode consequences for plant reliability

Important differences between the use of RRs and IFMIF are:

- Ease and space of testing for engineering application standards
- Irradiation condition control
- Availability
- Costs
- Time scale
These differences are also valid for the use of particle beams for fusion materials science and engineering. Most of the beams have intensities much below that of IFMIF. Since the particles are usually ions, the relevance for fusion applications has to be assessed very carefully and complemented by advanced modelling.

- The meeting participants proposed that a specific template should be prepared on irradiation devices, and related information on irradiation capabilities and post-irradiation examination, and that the IAEA will collect such data on all available RR facilities for material irradiation and research.

2.2. Fusion materials research through different stages

In the main requirements for the validation and modelling activities four levels are distinguished for all devices involved, be they particle beams or moderated or fast spectrum neutron sources:

(a.) First level (environment control and instrumentation)
- Monitoring of parameters is required
- Good to excellent control, and monitoring of temperature (in particular, stability)
- Good monitoring of stress/load/strain in instrumented experiments (e.g. stress-relaxation)
- High dose excess, say, up to 50-100dpa

(b.) Second level (active control of load history in mechanical tests)
- Static in-pile experiments
- Dynamic testing: uni- or bi-axial, cyclic loading, hold times, and relaxation

(c.) Third level (multi-effect tests)
- Mechanical + environment (e.g. corrosion) under irradiation
- Interaction of solid, gaseous and liquid materials in realistic contact during operation

Remark: The material interaction and corrosion does not necessarily require very high flux RR, although this will certainly need specific knowledge and experience.

(d.) Fourth level (verification/validation of modelling)
Here irradiation is only one parameter. For example: tritium extraction and permeation are also important. The modelling of fusion materials goes through four stages:
- Screening of candidate materials
- Demonstration of materials performance limits
- Qualified material with demonstration and performance (keep in mind the He issue!)
- Materials performance and component specific loading

More or less consecutively the character of the research and the up-scaling of the testing devices changes in four steps:
- Basic science for improved understanding and accelerated development cycles
- Modelling from micro- to macro-scale
- Modelling interpretation and transferability of measurements into design data/rules
- Macroscopic phenomena related to operation.
Typical examples for the first category are damage in optical fibers, thermal conductivity behaviour under radiation, bolometer and magnetic sensor changes due to radiation damage, as in the JUPITER programme. The outcome of the just mentioned experiments is critical for the diagnostics instrumentation to control the plasma in fusion devices.

Applications from the first to the second category are exemplified in the JANNuS triple beam arrangement intended for the modelling verification of co-generation of radiation damage with ion particles and He-injection, preferably observed in beam with a transmission electron microscope. These are only examples from a working field, where many more devices are used to pursue the four steps from the sub atomic phenomena studied with beams and microscopes to the induced collapse of full size test components.

In this area it is essential to observe that materials properties must not be studied in optimized, but isolated test environments. One should also keep in mind that for the structures envisaged for a DEMO thousands of tons of high quality material must be made within one narrow specification window. In this regard, both design and also manufacturing practices might have an irreversible influence that must not be neglected (e.g. manufacturing and heat input from welding must not deteriorate the carefully conditioned bulk half product material). The close co-operation with materials manufacturing and materials processing industrial practice must be established at the earliest development stage. In particular, this implies that the resources, irradiation campaigns and testing will not only be spent on the base materials but also on material systems such as similar or dissimilar welds, coatings etc.

Another influence that must be taken into account for materials applications is the notion of the societal changes in handling nuclear waste materials. In several countries law makers are preparing rules for the allowable time for radioactive waste material to decay. These requirements might become mandatory before establishing the requirements on recycling fusion materials after service.

It has been a policy in the fusion material community since decades, in particular Japan, the US and the EU, to develop materials that comply with social and environmental requirements and development of “low activation” materials has been one of the main criteria. Steels as developed by today F82H and EUROFER are aiming for this objective and today are at or close to fulfil requirements of recycling within 100 years.

- The meeting participants noted that the RR community should contribute not only through nuclear technology development but also through basic scientific research aspects of materials (e.g. use of neutron beams for advanced analysis and characterization, modelling, etc.). In this regard, the IAEA should encourage Member States to make available mechanisms to involve and support academia in the field of basic fusion material research and related technology development as part of fusion energy development strategy.

2.3. Synergy between fission and fusion materials research

Service conditions of fusion structural materials
The current primary candidate materials for a DEMO in the EU and Japan are reduced activation ferritic/martensitic (RAFM) steels in a water or helium gas cooling environment. ODS steels, Vanadium alloys and SiC/SiC composite materials are candidate structural
materials for advanced blanket concepts. Tungsten alloys are options for gas-cooled divertor concepts. Typical service conditions for these structural materials are summarized in ANNEX I.

**Service conditions of fission structural materials**
The service conditions of fission technology materials have a very wide spectrum. Existing reactors have already a wide spectrum in structural and functional materials. The fission RRs will be instrumental in the development of GEN-IV reactors. Their structural materials spectrum is from low alloy carbon vessel steels to SiCSiC ceramic composites anticipated for control elements in High Temperature Reactors. Functional materials range from beryllium for reflectors to thorium ceramics for the new generation of fuel. More details can be found through GEN-IV ([http://www.gen-4.org/](http://www.gen-4.org/)) or INPRO ([http://www.iaea.org/INPRO/](http://www.iaea.org/INPRO/)).

**Synergy of fission and fusion materials development**
There is a significant overlap between the application spectra in the fission and fusion technology domains. The materials science and engineering and the manufacturing technology have similar roots below them. The environments differ principally in the neutron spectrum from the first wall of a fusion reactor to half a meter inside the blankets. There the co-generation of H, He and radiation damage from neutron interaction with matter differs strongly from the fission reactor core neutron radiation damage. The rest of the fusion device related structures undergo neutron interaction in a way comparable with that of materials in a fission reactor core, except for the presence of magnetic fields.

Indeed, fission and fusion domains can both benefit from a coherent programme for basic material science and modelling from micro- to macro-scale. The nature of the fusion nuclear environment has led to fusion-specific activities addressing the co-generation of H, He, solid transmutation products, and radiation damage. It should be noted that adverse transmutation effects have resulted in the late 70- and early 80ties in the complete abandon of applications based on nickel bearing steels and in the development of low activation steels. For the same reasons high temperature alloys of niobium and molybdenum were abandoned, and the advocacy of the vanadium alloy case was made. In the context of advanced fission reactors the fission community has also developed a much pronounced interest in the use of low activation alloys. Cross fertilization between the fusion and fission communities is certainly there and will become more intense also through the necessary input from manufacturing practices and nuclear materials use and recycling. The fusion domain can largely gain from the fission design and licensing procedures from the nuclear industry. The component and materials data base production for the fusion environment sets its own agenda but its impact will only be at a later stage (see section 2.3).

Similar experience existed and exists in the shift of the moderated neutron energy spectrum based power reactors and the fast spectrum power reactors. The environmental differences between them are not as pronounced as they are between fission and fusion. The development to fast reactor technology, e.g. the development of austenitic steels, can serve as an example of benefiting from the basic insight originally gained in the moderated neutron energy spectrum power generation. Basic science and material behaviour modelling are good examples here as well. This does not need to be restricted to metals or ceramic doted metals, including ODS. The knowledge and applications in ceramic fuels and blanket component technology have attractive similarities to stimulate the development of synergy mechanisms.
Paramount in the synergy are also the services provided by the irradiation experts connected to the RRs. The ties with hot laboratories and universities specialized in radiation damage studies in solids are essential for the successful initiation, and completion of irradiation programs with adequate post-irradiation experiments, and analyses.

Last but not the least, although fusion and fission are addressing similar materials however the development goals are different because of the different use. For example, fusion breeding blanket materials are subjected to high thermal load (at moderate primary stresses) – therefore breeding blanket and divertor will be exchanged every 3-5 years. Creep resistance is hence not the key issue. Contrary, in fission technology the critical components are designed and licensed to last at least for 40-50 years…

- The IAEA should facilitate cooperation, exchange of information and call for joint meetings-workshops between fusion-fission communities. In this regard, the IAEA should assist in organizing around June 2011 a dedicated technical meeting to bring the fusion and fission materials development communities together and share their experience and good practices.

3. SUMMARY AND CONCLUSIONS

There was unanimous agreement that there are significant opportunities for collaboration and useful interaction between fission and fusion material research communities. The following points summarize the current situation as it pertains to the majority of material research laboratories:

- The important role of RRs in fusion material and fusion nuclear technology development is confirmed and strongly supported. Many data have already been obtained to validate the design of ITER components, and in support of the conceptual designs of subsystems towards DEMO and beyond.

- Dedicated particle beam experiments must provide qualitative data to validate theory and modelling in order to complement the lack of cogeneration of gas and displacement damage before IFMIF is able to deliver the required irradiation conditions, in particular for structural materials.

- An essential link in the chain of processes leading to reliable design data for a DEMO in connection to the RRs are the test devices realization, and the use of well-equipped laboratories for post-irradiation examination and testing. The available knowledge, ability and experience of qualified personnel in fission nuclear technology and sciences are another advantage one should use efficiently in the chain of processes for DEMO success.
4. RECOMMENDATIONS

The meeting adopted the following concrete recommendations, satisfying the stated objectives of this Consultancy Meeting:

- The international fusion community, as in the case of ITER, should develop reference documents on DEMO requirements and candidate materials to facilitate the fission community to enlarge cooperation and assist in fusion materials science and engineering related irradiations and research.

- A template shall be prepared on irradiation devices, and related information on irradiation capabilities and post-irradiation examination, and the IAEA will collect data on all available RR facilities for material irradiation and research.

- The RR community should contribute not only through irradiations and technology development, but also through basic scientific research aspects of materials (e.g. use of advanced neutron beams, modelling, etc.).

- The IAEA should encourage Member States to make available mechanisms to involve and support academia in the field of basic fusion material research and related technology development as part of fusion energy development strategy.

- The IAEA should facilitate cooperation, exchange of information and call for joint meetings-workshops between fusion-fission communities. Therefore, a follow up technical meeting to bring the fusion and fission materials development communities together and share their experience and good practices is recommended in 2011.
ANNEX I. REQUIREMENTS FOR STRUCTURAL MATERIALS IN FUSION

End-of-life service conditions for fusion structural materials are defined by the neutron fluence of 10-20MWa/m² at temperatures ranging between 300 and 550°C for RAFM steels.

The following two tables provide a summary of the present fusion structural materials developments (as a rule of thumb neutron loading on divertor target plate is 1/3 FW, i.e. 10-30 dpa (in steel))

| Service condition breeder blanket RA Ferritic/Martensitic Steel (DEMO target) |
|-----------------------------|---------------------------------|----------------|
| Temperature                 | 285-550 °C /Max. temperature for ODS < 650/750 °C | |
| Neutron fluence             | First DEMO BB 3-5 MWa/m²dpa / Final goal 10 MWa/m²dpa | |
| Damage levels               | 30 - 100 dpa (steel)             | |
| Coolant                     | Water / Helium                  | |

| Service condition of the materials for advanced concept blankets and divertors |
|--------------------------------|----------------|----------------|
| Material                      | Vanadium alloys | SiC/SiC composites |
| Maximum Temperature           | < 800 °C       | < 1,100 °C       |
| Coolant                       | Liquid Metal    | He gas / liquid metal |

The application of Small Specimen Test Technique (SSTT), in particular the high flux test module of IFMIF, is favourable for several reasons:

(i) to make best use of available irradiation volume

(ii) to keep temperature gradients in the specimen as low as possible [examples presented, step gradients in change of material properties within 20-50°C, e.g. DBTT, tensile strength 500°C in RAFM]

(iii) SSTT is already used for tensile, creep, fatigue specimens (widely) in irradiation campaigns with fission RRs in the EU, Japan, the USA and the Russian Federation. In these areas the fundamentals of [IFMIF-type] SSTT have been already widely established.

However, SSTT for fracture toughness and fatigue crack growth tests are still under development. Moreover, it is worthwhile to be noted that the gap between SSTT and the design methodology for the high temperature structure irradiated to high damage levels by high energy neutrons still remains. Finally, SSTT might become even stronger requirement from the fusion community to harmonize data bases.

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ANNEX II.  WORK PLAN OF THE FUTURE ACTIONS

The following table sets out a draft work plan of actions that should be taken in order to implement the recommendations in a timely fashion.

<table>
<thead>
<tr>
<th>Activity</th>
<th>Coordination</th>
<th>Start date</th>
<th>Delivery date</th>
</tr>
</thead>
<tbody>
<tr>
<td>Technical Meeting to bring the fusion and fission materials development communities together</td>
<td>IAEA</td>
<td>01 March 2011</td>
<td>01 July 2011</td>
</tr>
<tr>
<td>Template to be prepared on RR related irradiation capabilities and post-irradiation examination</td>
<td>RR community (in cooperation with the IAEA)</td>
<td>01 March 2011</td>
<td>01 May 2011</td>
</tr>
<tr>
<td>Reference document on DEMO requirements and candidate materials to facilitate the fission community</td>
<td>Fusion community (in cooperation with the IAEA)</td>
<td>01 March 2011</td>
<td>15 June 2011</td>
</tr>
<tr>
<td>Technical Meeting report on the intensification of joint activities related to fusion and fission materials science and engineering</td>
<td>IAEA</td>
<td>01 July 2011</td>
<td>01 September 2011</td>
</tr>
</tbody>
</table>
ANNEX III. INDIVIDUAL CONTRIBUTIONS

1 E. Diegele, F4E, the European Union

Structural Materials Development for DEMO and beyond in Europe

In Fusion development the next step ahead is ITER, the facility to demonstrate the scientific and technological feasibility of fusion power. In parallel to ITER construction and operation, still some open technology issues need to be addressed and solved before building a demonstration reactor (DEMO). This includes in particular structural materials R&D for in-vessel components. The loading is determined by high heat flux and damage through high energy 14 MeV neutrons. At the First Wall this typically corresponds to 30-80 dpa for DEMO, approximately two orders of magnitude higher than in ITER.

Materials development for breeder blankets and divertors has a history of more than two decades. It is very specific for in-service and loading conditions and also for required properties in combination with safety standards and that create a unique set of specifications. In particular, social-economic demands of low level waste and low activation reduce the choice significantly to materials based on Cr, Ti, V, Fe, W, Si, C and a few others. Consequently, four classes of structural materials, (i) the reduced activation ferritic/martensitic (RAFM) steels, including nano-dispersion strengthened variants, (ii) the vanadium alloys, (iii) the tungsten alloys, and (iv) the SiC fibre reinforced ceramic composites, are investigated.

The main objective is to have DEMO materials and key fabrication technologies fully developed and qualified (for full DEMO life) within two decades. Nevertheless, a major part of the task has to be completed much earlier: DEMO relevant technologies will be tested in tritium breeding test blanket modules (TBM) in ITER. Materials and materials technologies have, therefore, to be fully qualified for licensing processes within the next decade (up to a few dpa). Therefore,

- Materials for use in DEMO are based on present technologies and knowledge with some reasonable extrapolation. Hence RAFM steels are the primary candidate. Additionally, ODS (Oxide Dispersion Strengthened) steels are developed for high temperature application.

- The large fraction of high energy neutrons in the fusion neutron spectrum results in large amounts of gaseous transmutations (He and H) that are more than one order of magnitude higher than in fission based neutron spectrum.

- Fission based material test reactors (MTRs) are essential pillars of the irradiation qualification programmes and are needed for at least two decades. Their main contribution will be the wide and intensive characterization of physical properties and mechanical properties under irradiation. In particular, studies on multiple combination of effects (corrosion/irradiation) or irradiations on small mock-up of components or more complex loading conditions (multi-axial, time-dependent loading) are of interest to close the gap in present knowledge.

- MTRs, however, cannot provide sufficient data for a successful licensing process towards DEMO. Therefore, IFMIF (International Fusion Materials Irradiation Facility) a facility designed for simulating as closely as possible the fusion neutron spectrum is mandatory for the fusion materials R&D path-way.
Complimentary to IFMIF, an enhanced material science programme should increase knowledge and understanding of radiation effects. In a longer term perspective, this should result in the implementation of an integrated approach involving modeling and model-oriented experimental validation into a strategy of accelerated development and testing of candidate fusion materials, material systems and material technologies.
2 T. Shikama, IMR, Japan

Development of fusion reactor materials utilizing fission RRs in Japan

Several irradiation tools are presently available such as accelerators including HVEM (High Voltage Electron Microscope), moderate intensity 14MeV neutron sources, but the IFMIF (International Fusion Materials Irradiation Facility) is still expected to be incarnated. In the meantime, based on accumulated experimental data and, thanks to remarkable advance of computer technologies, the development of so-called multi-scale modelling is rapidly progressing. In this context, fission reactors will be still expected to play an important role in studying and testing materials performance under fusion adopted irradiation conditions. Here, it must be admitted that the fission-reactor irradiation has some definite setbacks, such as difficulty of simulating nuclear transmutation effects, which will be one among major irradiation effects of the high energy fusion neutrons. However, many of major setbacks can be overcome by alternative sophisticated irradiation techniques such as isotope-tailoring as well as neutron spectrum tailoring. Also, some setbacks could be covered by other irradiation techniques, such that dependence of irradiation effects upon some detailed irradiation parameters could be followed by ion-irradiations under well-controlled irradiation conditions. Finally, the future IFMIF will bridge the deep gap between the setbacks which fission reactor irradiation has and the comprehensive understandings of irradiation effects.

For irradiation studies with nuclear fusion materials beyond ITER, a high dpa as well as big irradiation volumes will be mandatory, and therefore high flux fission reactors will be needed to play their role extensively there. The candidate reactors could be JOYO (Japan), HFIR (USA), and JHR (EU). Extensive utilization of these high flux reactors for development of fusion reactor materials is mandatory, especially for the preparation for final evaluation in the IFMIF. Here, promotion of utilization of these reactors under international collaboration is indispensable. In the meantime, well-controlled irradiation and in-situ studies with well equipped installations will not be easily done in such high flux fission reactors, mainly due to associated high-nuclear heating rates and some difficult accessibility. In general, reliable installations for controlling irradiation conditions and for measuring material properties in-situ would not be realized easily in the irradiation field, where the nuclear heating rate exceeds 10W/g for iron. Here, medium sized reactors with more flexible accessibility will have their indispensable value especially for in-situ type studies which will be very important for functional nuclear fusion materials. Some of candidate reactors here will be JMTR (Japan), HFR (Netherlands), BR-2 (Belgium) and Halden (Norway). In some cases, dynamic and transient irradiation effects could be important to study and pulse-type fission reactors such as in Japan and the RF will be expected to play a role. One important but forgotten irradiation will be a cryogenic irradiation for nuclear fusion materials. There, demanded neutron fluence will not be high but in the range lower than $10^{23}n/m^2$. There, some small reactors may play an important role.
Radiation damage

In fusion power reactors, the plasma facing (first wall, divertor) and breeding blanket components will suffer irradiation by an intense flux of 14.1 MeV neutrons coming from the plasma. These fusion neutrons will produce nuclear transmutation reactions and atomic displacement cascades inside the various encountered, and therefore irradiated, materials. From the point of view of material science, the nuclear transmutation reactions will produce impurities such as helium and hydrogen gas atoms as well as (eventually radioactive) metallic impurities, while atomic displacement cascades will produce mainly point structure defects (vacancies and interstitial atoms) and clusters of point structure defects as well as segregation of alloying elements. The final microstructure of the irradiated materials will result from a balance between the primary radiation damage and thermal annealing. It will be composed of complex secondary defects including small defect clusters, interstitial dislocation loops, vacancy dislocation loops, stacking fault tetrahedra, precipitates, voids and/or helium bubbles. This degraded microstructure will have a strong impact on the physical and mechanical properties of the irradiated materials. It may engender local changes in the chemical composition as well as a decrease of electrical conductivity (especially at low temperatures) and/or a decrease of thermal conductivity (especially in the case of ceramic materials). It may also lead to degradation of the mechanical properties including hardening and embrittlement effects and/or a decrease of creep strength. It may also lead to loss of dimensional stability resulting from the phenomena of swelling (as the helium bubbles may engender a macroscopic increase in the volume of the materials), irradiation creep and/or irradiation growth. Irradiation-assisted stress corrosion cracking may also occur. The materials may also become radioactive due to the formation of radioactive metallic impurities by nuclear transmutation reactions.

The key irradiation parameters include the accumulated damage, expressed in ‘dpa’ (number of displacements per atom), the damage rate (in ‘dpa s\(^{-1}\)’), the rates of production of impurities (e.g. appm \(\text{He dpa}^{-1}\) and appm \(\text{H dpa}^{-1}\) ratios) and the temperature of the materials under irradiation. At low temperatures (e.g. < 673 K in the case of steels), one observes mainly hardening and embrittlement effects, including a loss of ductility (as measured in tensile tests), a loss of fracture toughness and an increase in the ductile-to-brittle transition temperature (DBTT) in the case of body-centred cubic (bcc) materials. At intermediate temperatures (e.g. 573–873 K in the case of steels), one observes a peak in swelling, located at about 723 K for reduced activation ferritic/martensitic (RAFM) steels. At high temperatures (e.g. above 873 K in the case of steels), one observes mainly irradiation-enhanced precipitation and creep effects as well as helium embrittlement effects. How to investigate in detail the effects of irradiation by fusion neutrons on the properties of candidate materials for fusion power reactors? Unfortunately, the existing sources of 14 MeV neutrons have very small intensity and do not allow us to get significant damage accumulation in a reasonable time. Therefore, it is necessary to simulate irradiation by fusion neutrons by using fission neutrons, high-energy protons or heavy ions. However, the irradiation conditions provided by such particles are quite different from those expected to occur in a fusion power reactor, especially in terms of damage rate and rates of production of impurities. For instance, fission neutrons produce not enough helium and hydrogen with respect to fusion...
neutrons, while high-energy protons produce too much helium and hydrogen as well too many metallic impurities.

However, such an approach is far from being adequate, because it is difficult to separate effects of particle type, particle energy, temperature, accumulated damage, damage rate and rates of production of impurities, and in addition, candidate materials will have to be submitted to actual fusion irradiation conditions in order to be fully qualified for the designers and engineers who will construct the fusion power reactors.

**JANNuS Facility - Joint Accelerators for Nano-science and Nuclear Simulation**

In France JANNuS was designed to supply a large range of ion irradiation and implantation conditions, allowing in-situ Transmission Electron Microscopy (TEM) and ion beam analysis with single, dual or triple beam combinations. Such a facility has no equivalent in Europe and will play an essential role for multi-scale modelling of irradiation effects in materials.

The advantages of this simulation process are the versatility of the available experimental irradiation conditions (temperature, dose rate, fluence, damaged thickness) and the possibility to carry on in situ or ex situ physico-chemical and structural characterization.

The main uses of this new facility are the study of the evolution of the microstructure of the material during irradiation and its physical and mechanical consequences, plus cumulative effects of simultaneous multi-irradiation.

**IFMIF**

IFMIF is an intense source of 14 MeV neutrons (up to $10^{15}$ n s$^{-1}$ cm$^{-2}$, produced from 250 mA incident 40MeV deuteron beams on liquid Li target), where the neutron spectrum should meet the first wall neutron spectrum as near as possible.

The missions of this facility are:

- Qualification of candidate materials up to about full lifetime of anticipated use in a fusion DEMO reactor
- Calibration and validation of data generated from fission reactors and particle accelerators
  - Identify possible new phenomena which might occur due to the high energy neutron exposure

**Modelling of radiation damage and radiation damage effects must contribute to the design of experiments and interpretation of data from IFMIF.**

**Work under EFDA**

Materials at EFDA are running under the Fusion Materials Topical Group, which is organized in four main lines:

- **MAT-REMEV: Radiation Effects Modelling and Experimental Validation**
Objective: development of a conceptual and quantitative framework for the interpretation of experimental tests on steels and iron-based alloys, and predicting the performance of these materials under DEMO-relevant operating conditions.

- MAT-ODSFS: Nano-structured ODS Ferritic Steel Development
  Objective: development of an ODS ferritic steel with high tensile and creep strength and sufficient ductility and fracture toughness up to about 750°C as well as good radiation resistance.

- MAT-SiC/SiC: SiCf/SiC Composite for Structural Application in Fusion Reactor
  Objective: development of a reference SiC-based composite suitable for fusion applications

- MAT-W&WALLOYS: Tungsten and Tungsten Alloys Development
  Objective: development structural as well as armour materials in combination with the necessary production and fabrication technologies for DEMO divertor components.

Neutron irradiations needed for Fusion

In the case of steels, the best is to irradiate them in the range of 300-350°C, where hardening and embrittlement effects are maximum, and at the expected upper temperature for use. For W-base materials, the situation is a little more complex, as these temperatures are still uncertain, but the strategy should remain the same. Assuming for instance a DBTT close to 600°C, they should be irradiated at 300-500°C and 1000-1100°C.

The irradiation doses should be up to about 100-150 dpa, assuming a damage rate of 30 dpa/year in steels and a maximum lifetime of about 5 years.

<table>
<thead>
<tr>
<th>Material</th>
<th>Temperature (°C)</th>
<th>dpa</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steels</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RAFM</td>
<td>300-350, 550</td>
<td></td>
</tr>
<tr>
<td>ODS RAFM</td>
<td>300-350, 650</td>
<td></td>
</tr>
<tr>
<td>ODS RAF</td>
<td>300-350, 750</td>
<td>100-150</td>
</tr>
<tr>
<td>W</td>
<td>W-base materials</td>
<td>300-500, 1000 – 1100</td>
</tr>
</tbody>
</table>

Conclusions

- IFMIF is essential for qualification of candidate materials for future fusion reactors (DEMO and beyond)
  …but, taking into account the fact that it will be available only in, say, beyond 2020, Fission Research Reactors must continue to play an essential role in materials testing, and not only these facilities, but also, particle accelerators, other neutron sources, etc.

- In a fusion machine, not only FW materials are affected by radiation, **structural and functional materials are also subject to radiation damage.**
Fusion materials research in China

Neutron irradiation experiment by fission reactors is of great importance before and even after IFMIF starting its operation. There is a very limited space/volume in IFMIF and most of neutron irradiation should still be carried out in fission reactors. The candidate materials and research subjects for IFMIF experiment will be decided with experiments of the existing fission reactors.

Together with neutron beams, ion beams and electron beams will also be used for the scientific and technological research in irradiation damage in materials. These type of investigations can help us to understand the details in irradiation process.

China has recently built two research reactors, CARR and CEFR, and will provide its capability of neutron irradiation to the rest of the world. However, it needs several years for China to equip its hot cells for PIE work. More international cooperation would be helpful to Chinese research projects of neutron irradiation and by this kind of cooperation China may have more chance to use other countries’ reactors. In any case, the cooperation on neutron irradiation beyond countries is necessary for the development of fusion materials.

The IAEA will play an important role to enhance the neutron irradiation experiment for developing fusion energy. The advice from IAEA would promote the research inside China greatly. Therefore, in my opinion the IAEA can encourage Chinese government to wider its international cooperation on neutron irradiation in future.

Such kind of consultancy meeting should be held regularly and then a long term program of neutron irradiation by fission reactors may be launched at the second stage.

The universities in China are very active to the research on irradiation damage because China is booming nuclear technology now. China started a huge program to develop its nuclear industry, covering the area from fission power station to fusion tokamak devices.
5  S. Saroja, IGCAR, India

Materials Research at IGCAR for Indian Nuclear Energy and Fusion Programmes

The Indian Fast Breeder and the Fusion Program has evolved a comprehensive strategy for materials research in the last few decades. The major thrust has been to forge a seamless synergy between various essential building blocks: atomistic understanding of materials behaviour, materials technology, component fabrication and the evaluation of its real-life performance. The nuclear industry has been witnessing a metamorphosis be it in developing newer materials, processing methods or fabrication technologies, from the empirical methods to a more scientific and knowledge based design. A multidisciplinary approach involving strong R&D expertise in DAE, academia and industries of the country, over the later half of the last century, has provided the required confidence to launch commercial fast breeder reactors. This rich experience has laid the necessary foundation for participating in the fusion program of India in collaboration with the Institute of Plasma Research, Gandhinagar.

The second stage of the three stage program, namely the fast breeder reactors was successfully started in the 80’s with the commissioning of the Fast Breeder Teat Reactor (FBTR) at Kalpakkam in 1985. This reactor has the unique distinction of using a mixed carbide of uranium and plutonium and has been operated for twenty five years without a single pin failure. FBTR, the flagship of IGCAR, Kalpakkam, has been the test bed for demonstrating the fast reactor technology, sodium technology, the carbide fuel and the materials technology. The post irradiation data generated using the in house facilities at various burn-up levels from 25 to 155MWd/t provided valuable data on materials behaviour in a fast reactor. The clad material, 20% cold worked Type 316 austenitic stainless steel, retained an elongation of about 5% at 80dpa. The void swelling and defect structures were also evaluated in detail. The volumetric swelling data of ~4% at 80 dpa and the formation of Ni-Si intermetallic phases paved the way for an intense material development activity for high burn-up and high temperature materials.

The major thrust in our materials program was identified to be the following four important features: enhanced burn-up, high temperature, high breeding ratio and closing the fuel cycle. The development of materials and technologies was directed towards meeting the above targets.

The 90s witnessed the development, characterisation and performance evaluation of austenitic steels with modification of composition, to enhance the burn-up. The availability of accelerator facilities for studying the ion irradiation behavior enabled the screening of large number of various compositions of developmental alloys, from time to time. Alloy D9 and improved D9 varieties with 15Cr, 15Ni, Ti and minor element variations, a complete indigenous effort, has shown that a two fold enhancement in radiation resistance (upto150 dpa) and also higher temperature of operation (~550°C) and ~ 700°C during transients can be achieved. An indigenous austenitic alloy of INDFAC has been developed for the clad component applications.

The next phase towards meeting the criteria of high burn up and high breeding involved the development of ferritic steels namely the 9Cr-1Mo class, which had marginal advantage over the high Cr counterpart, 12Cr steels, w.r.t irradiation embrittlement. The similarity in environment with respect to fast neutrons, flux and radiation damage has led to our
participation in material development for test blanket module in ITER. The 9Cr-W-0.1C alloys have been indigenously produced and an elaborate program for assessing the creep, tensile, impact and thermal stability behavior is in progress.

The major disadvantage in shifting the emphasis from austenitic to ferritic steels is the reduction in the high temperature limit of operation. Hence, oxide dispersion strengthened 9Cr ferritic martensitic steel clad tubes were developed to enhance the high temperature creep properties of ferritic steels. The complex manufacturing technology was standardized in collaboration with Indian industries and R&D organizations. We have successfully produced clad tubes whose properties meet the design criteria. Intense R&D in high chromium ODS is initiated to improve chemical compatibility and for fusion applications.

The advanced technologies for fabrication of complex components have been developed with the involvement from Indian industries. This has led to the active participation of our country in the international effort to establish fusion technology, through ITER. India is involved in the fabrication of Test Blanket Module and the diverter assemblies for ITER. The use of advanced welding methods such as electron beam, laser and laser hybrid welding, narrow gap Tungsten Inert Gas welding, testing and qualification procedures are being standardized in collaboration with several R&D organizations in the country.

Although the knowledge, experience and data obtained from the fast breeder reactor have given deep insights into materials behavior and established complex materials technologies, yet there are issues of concern in extrapolating the understanding to the fusion scenario. It is possible that the damage mechanisms in a fusion reactor could be entirely different from the lessons learnt from fission reactors. The availability of a dedicated irradiation facility with 14MeV neutrons and higher dose rate is an essential requirement for developing materials technology for fusion. However, materials surveillance, for which methodologies and expertise have been established over several decades across the globe, should be a collaborative venture, to reduce the cost and time required for the development of more challenging materials technology for fusion.
6 O. Yeliseyeva, NASU, Ukraine

Compatibility of Structural Materials with Liquid Metal Breeders/Coolants

(as applied for fusion and fission reactor concepts)

An excellent thermo-physical and nuclear properties of liquid metals (Li, Pb, Pb-Li and others) allow to increase the temperature range of both fusion and fission reactors. Under investigation the main attention needs to be given to corrosion aggressiveness of liquid metals with regard to the structural materials (RAFM steels, V-alloys). While the basic phenomena - dissolution and mass-transfer of solid metal by pure liquid metal - are studied well enough (in static and dynamic condition) the influence of non-metallic impurities on the corrosion behavior of structure materials in the liquid-metal environment is unclear up to now.

The problem of our investigations was aimed at the understanding of interaction mechanisms of structure materials with liquid metal (Pb, Pb-Bi, Li etc.) with taking into account the role of non-metallic impurities (O, N, C). The main attention was focused on the cases when the diffusion counter flows of the components promote in formation of protective coating at the interface.

The systems Fe[Cr]-Pb[O], V[Ti]-Li[N], V[O]-Li[Er] with in-situ self healing (or self-recovering) surface layers were investigated.

At the optimal concentration range of oxygen in Pb[O] there is a positive result of passivation of traditional steels (with 9-12% Cr) at low temperatures (400-550°C). In order to keep protective oxide layer at 550-650°C the chromium content in the steel must be higher (up to ~18%). Most probably the protective Cr-oxide film can be formed on the high-chromium steel in the eutectic Pb-17Li without special doping of melt by oxygen. This hypothesis deserves to be verified.

Investigation of V[O]-Li[Er] system was dedicated to diminution of magneto-hydrodynamic pressure drop when Li flows in the duct of self-cooled V/Li blanket. This negative phenomenon can be prevented by the insulator layer formed on the inner wall of vanadium ducts. The feasibility of Er2O3 oxide layer formation was demonstrated up to 650-700°C owing to counter-flows of oxygen (from vanadium alloy) and erbium (from lithium). The mechanism and adequate model of creation of such kind of coating was proposed.

The purpose of research of V[Ti]-Li[N] system: to optimize a V-Ti-Cr composition for operation in lithium with variable concentration of nitrogen. It was determined that the V-(8-10)Ti-(4-5)Cr possessed the best corrosion resistance and stable surface nitride layer.

Adsorption property of heavy liquid metals (Pb, Bi) becomes apparent in contact with solid metal without surface oxide film. The main influence of liquid metal consists in liquid-metal embrittlement of iron and FM steel (at 300-440 °C) and acceleration of creep (at 500-650°). This experience should be taken into account in fusion reactor with Pb-Li coolant because of dissociation of surface oxide film.

- The compatibility data “solid metal-liquid metal” obtained for model materials and conventional steels must be verified experimentally for Low Activation Materials (V-alloys and RAF/M steels).
- The main mechanisms of “liquid metal-solid metal” interaction should be elucidated and taken into account for development of model of corrosion behavior of new structure materials in the liquid-metal heat-transfer with non-metallic impurities.
- The combined influence of irradiation and corrosion should be investigated.
7 V. Chernov, Bochvar Institute, Russian Federation

Energy Related Material Research in Russian

The current RF nuclear energy R&D program in the first half of the 21-st century has a major emphasis on the Closed Nuclear Fuel Cycle on the base of fast reactors and the R&D for fusion power plant (FPP). The RF fusion and fast power reactor material programs are closely linked. Now in the RF there are the research (fast BOR-60, intermediate IVV-2M) and power (fast BN-600) reactors. The new fast reactors: power BN-800 (2014), multi-functional research MBIR (2018) and power BN-1200 (2020) will be constructed. Also the plasma fusion neutron sources type Tokamak (TIN-0, TIN-1, TIN-2) are under the R&D to construct during 2015-2030 years. The requirements for structural materials (SMs) include the neutron loads 110–150–200 dpa and the operation temperature up to 700 – 800°C with high energy efficiency during operation and after operation include good storage (corrosion resistance), reprocessing and fast decay of radioactive inventory. Improved and new core SMs are required for further widening of temperature, stress and dose application windows for innovative fast and fusion power reactors.

Key materials issues include the development of heat, radiation and corrosion resistance SMs, including low (reduced) activation materials (LAMAs) such as ferritic-martensitic (12-14) % chromium steels and vanadium alloys (V-Ti-Cr system). The extension of materials and nuclear databases to fill gaps in existing knowledge base and new engineering data are also needed for high dose neutron irradiated materials. There is also cross-cutting interest in improved fundamental understanding of material science and technology for fast and fusion power reactors.

Our knowledge data bases seem to be appropriate for the DEMO reactor but further progress is anticipated for the innovative FPP conceptions and designs. There is a compelling need:
- to develop science-based engineering design rules for structural materials exposed to the high temperatures and neutron irradiation environment (improved strength-creep-embrittlement-corrosion rules) to replace current empirical correlations,
- to accommodate potential of our materials science knowledge on radiation properties of SMs via research and power fast reactors for fusion reactor requirements that may appear at high temperatures and more higher flux and fluence due to fusion neutron irradiation,
- to improve thermal- and irradiation-induced aging and degradation mechanisms during prolonged operation of research reactors in close coordination with advanced modeling and material characterization including the scale-size effect for specimens.

On the way to FPP via Research/Power (as research for fusion) fast reactors and powerful fusion neutron sources we need (no other way):
- High neutron fluences (100-150-200 dpa-Fe) and wide temperature windows (300-1000 °C) of neutron irradiation of different SMs in the power BN-600 fast reactor. Now the special material science assembly to test the RAFMS RUSFER-EK-181 (Fe-12Cr-2W-V-Ta) is under irradiation (2010-2012 years) and the assembly to test the V-Ti-Cr alloys is under preparation (2010 – 2014 years).
- The use of research reactors (BOR-60, MBIR, IVV-2M) and fusion neutron sources (IFMIF, TIN) to inspection of our knowledge and understanding of radiation models of defect formation and evolution, mechanisms of radiation...
phenomena and properties, the effects of alloy composition and micro-structure formation and evolution on irradiation properties of SMs with different crystal symmetry in different time and size scales over the wide temperature window (up to 1000\(^{0}\)C),
- Adequacy of SMs databases for the designs and construction of DEMO-FPP in time.

On this way to power fusion reactors and nuclear fusion technology the IAEA support, organization, coordination, combination and collaboration of strengths between the IAEA-fusion communities via research nuclear reactors are very important and topical.
Strategy and Current Status of Fusion material R&D in Japan and the role of RRs
Focused on JAEA activity, especially on structural material development

1. General statement

It is important to define the milestone of DEMO development in order to discuss the role of research reactor in fusion material development. It can be identified that the specification of fusion material, especially for structural material, should have been prepared until around 2020 to be used in detail engineering design activity of DEMO to start construction DEMO around 2030, and this would be the requirement for material to be irradiated in IFMIF in first campaign.

Reduced Activation Ferritic/Martensitic (RAFM) steel is the primary candidate structural material, and it would be the only material which has enough potential to be used as structural material in DEMO design activity. The specification of fusion structural material would specify mechanical property and fabrication and joining processes, but it should regard the impact of fusion neutron irradiation.

Fusion neutron irradiation data would be accumulated by IFMIF irradiation, but its irradiation volume is limited, and the irradiation will not start earlier than 2020 at least. This indicates that the initial DEMO design target should be within the range where fusion neutron irradiation data is no too far off from the data trend obtained from fission irradiation experiments (Fig.1). For the specification of fusion material, accumulation of “rich” fission irradiation database within above range would be essential. Thus, it can be concluded that IFMIF is essential for fusion material program, but irradiation in research reactor is also essential for accumulation of irradiation data for design for upcoming 10 to 15 years.

At the same time, it is critical to define the above indicated range. Thus, it is important to characterize and estimate materials performance under high doses fusion neutron irradiation using simulation experiments and computational modeling. This research activity would be defined as the essential scientific activity to support fusion technology development. It should be noted that this strategy should be applied not only for structural material but also for functional material.

Figure 1: Strategy of the fusion neutron irradiation effect prediction technique development
2. Experimental conditions and parameters requirement to the development of nuclear fusion systems

The specification of fusion material will be the specification which provided with irradiation database, and it is expected to be very conservative. Thus, high-flux, high-fluence neutron irradiation with large irradiation volume and well-designed (reliable) temperature control system will be required to fulfill the expected requirement. It should be noted that the requirement for temperature control accuracy would be differ depending on its object, i.e., high accuracy will be asked for investigation of irradiation effect mechanism, but it would be more practical for accumulation of database, since it already foreseen, for example, that we should accept the fact that there is potential temperature gradient in a irradiated sample.

3. Issues on correlations between nuclear fusion irradiation environments and the available research reactors, and modelling efforts to bridge the gap between research reactor-based test irradiations and qualification of materials for nuclear fusion technologies

One potential issue is “DPA”. DPA is calculated based on a certain set of assumption on displacement damage, but it is unclear whether it will be the same for fission neutron and for fusion neutron. Nuclear transmutation effects is a well-known big issue, and defining the critical condition that make fission / fusion data to different by He/H effects would be the most urgent issue. These are the scientific topics where the contributions of modelling / simulation works are highly expected. It should be noted that high energy neutron (over 3~4 MeV) irradiation is needed to see the impact of element changes, as this is not predictable by modelling.

4. Possible collaboration initiatives among (fission- and) fusion-related material research communities

JAEA will continue (and expand) HFIR collaboration as much as possible to solve above listed issues, and JOYO will become the other choice once it back working.

Modelling activity is one of BA-RAFM-R&D program, and all ITER participant countries have right to join BA activity.

Spallation neutron source irradiation (and post-IFMIF-EVEDA) could be the item to be discussed, as it has potential to fill the gap between fission neutron irradiation and fusion neutron irradiation until IFMIF will start.

Utilization of post-IFMIF-EVEDA could be one of the other choices.
9  N. Loginov, IPPE, Russian Federation

Some issues of coolant technology and corrosion of structure materials for fission and fusion reactors

Since September 2002 till March 2006 the ISTC Project #2036 “The thermal-hydraulic and technological investigations for validation of the project of lithium circulating loop and neutron lithium target for IFMIF” was carried in IPPE. As a result Lithium Test Facility (LTF-M) was constructed and features of the IFMIF Lithium Target Mock-up hydrodynamics were studied.

Features of LTF-M: Operation under vacuum $10^{-2}$ - $10^{-3}$ Pa. MHD pump capacity 50m$^3$/h. Height of the facility is 15.6m. Lithium inventory is 270 liters. Maximal lithium temperature is 450$^\circ$C. Maximal electric power is 200kWt.

Besides, small LTF was under operation to provide purification of Li and impurities monitoring. This LTF has pump capacity of 8m$^3$, maximal temperature 600$^\circ$C, Li inventory 60 liters. Both facilities can be used for further researches of Li technology and compatibility of structure materials for fusion.

In addition, a Rotating Disk Test Facility (RDTF) was constructed for investigation of corrosion interaction of lithium and lead-lithium alloy with vanadium alloy and ferritic-martensitic steels. This RDTF include three test sections, each of them contain up to 150 cm$^3$ of liquid metal and immersed rotating disk made from studied material. So it is possible to test simultaneously three samples at three different temperatures. More over, disk samples can be made as composition of several rings or sectors. It allows testing several materials at the same conditions.

Ampoules testing of samples in unmovable liquid metal are possible also. This testing is needed to clarify affect of velocity on corrosion process. Some preliminary testing of V-4Ti-4Cr alloy in lithium at 450$^\circ$C and 600$^\circ$C during 750 and 1450 hours are performed and results were presented.

Monitoring of impurities in lithium and lead-lithium alloy is very important issue for corrosion research. The more important it is at the testing materials under irradiation in fission reactors, because some synergetic effects of corrosion and radiation damage were pointed out in literature. So reactor loops for testing of materials must be equipped with devices for monitoring of oxygen, nitrogen, hydrogen, carbon and chemical elements of testing steels and alloys. IPPE has test facilities, experience and specialists in this area.
Fusion Nuclear Technology activities at NRG Petten

The NRG, the Nuclear Research & consultancy Group, has a first-rate nuclear R&D infrastructure with the High Flux Reactor, the Hot Cell Laboratories and associated laboratories in Petten, The Netherlands. The HFR has already been for decades an important tool in the European Union’s programmes for the development of fusion energy related technologies. The HFR returned to service in September 2010, after an outage period of half a year for inspection and repair. Its availability is now resumed to the typical 285 days per calendar year. This availability results in damage rates of more than 7 dpa in steel per year (see contribution by B. van der Schaaf).

Many data on irradiation behaviour of ITER candidate materials have been obtained, and used in the ITER Material Properties Handbook, including 316L(N) plate, EB & TIG weldments, HIP-bonded, powder-HIPped products and explosive forming and cladding. Temperatures range from 330 to 570 K, and doses are up to 10 dpa. Repair welding work concerned TIG and laser welded material at ITER relevant dpa and helium levels, both for thin and thick sections. Irradiation testing of plasma facing materials concerns beryllium grades, carbon-base materials and tungsten alloys, as well as samples with copper alloy or steel substrates, and mock-ups for high heat flux testing in the JUDITH facility at FZJ. A major step concerns the transition to testing of components. Examples are irradiation stress relaxation of pre-stressed bolts (Ni-alloy and Martensitic steel), and thermal fatigue of primary wall mock-ups under simultaneous thermal and neutron flux cycling.

Qualification of the reduced activation 9Cr-steel Eurofer is ongoing and various manufacturing technologies (welding, HIP) are investigated for the European ITER Test Blanket Modules and DEMO/FPR (HCPB and HCLL concepts). These include tensile, fatigue, creep-fatigue and creep behaviour, fracture mechanics and development of small size specimen techniques required for IFMIF. Temperatures range from 330 to 770 K, and doses are up to 15 dpa.

Tritium release characteristics of lithium ceramic pebbles and LiPb eutectic are obtained by in-pile experiments, followed by post-irradiation analyses. Lithium burn-ups relevant for DEMO were already obtained for early candidate breeder. Four sets of HCPB pebble-bed geometries with DEMO relevant thermo-mechanical loads were tested in-pile for about 300 days. HCLL representative LiPb-eutectic assemblies have been tested for tritium release and permeation behaviour of the Eurofer. Controlled gas purge with on-line tritium monitoring and triple containment are key features for irradiation of tritium generating specimens, and already applied for more than two decades.

A spectrally tailored high dose irradiation projects concern ceramic breeder pebbles under DEMO & FPR relevant ratios of fast neutron damage (20 dpa) and lithium burn-up (10-15%). Two high dose irradiation experiments concern beryllium neutron multipliers key performance issues to 3000 and 6000 appm helium. Products include pebbles, as well as porous and dense forms. Temperatures range from 670 to 1020 K.

It is mentioned that close interaction with fission applications is beneficial, and should take place in many disciplines, ranging from modelling to technology validation, and more practicable in sharing facilities and irradiation space. The fusion work is supported by the
Netherlands Ministry of economic Affairs, with financial support of the European Commission, mostly through EFDA and F4E contracts.
PALLAS for Fusion

The Petten research facilities for materials research in Nuclear Fusion Technology include the High Flux reactor, expertise and ability to design and build in pile testing facilities, hot cell and special laboratories with equipment and skills to carry out post irradiation experiments and analyses. NRG has developed a strategy based on the societal needs for the 21st century R&D resulting in dedicated projects related to fusion materials and component testing in parallel with a similar approach for Generation-4 reactors. For fusion technology research the emphasis will be on materials and component testing for blanket materials and divertors. High temperature operation and complex loading effects, in addition to neutron radiation damage, are on the high priority list of NR, because they are the trend list for fusion power plants and preceding devices. Speeding up of radiation times and the use of loops for conditioning components will increase the necessary high frequency of development cycles for design and manufacturing. The sub component tests in Petten should contribute to the reliable operation in fusion devices.

The replacement of the HFR by PALLAS will satisfy the requirements of fission research reactor support for the fusion technology development for the next decades. Other EU research reactors to be operated this century such as RJH, MYRRHA, and ASTRID also will contribute with their specialities in terms of neutron flux spectrum and coolant, as the SNETP analyses have indicated.

The PALLAS properties allow a strong experimental contribution next to its dedication to isotope production for health applications. The new core and fuel design enhance a more economic use of UMo fuel the same time accelerate experiments and isotope production. A wide variety of core and reflector positions enable the selection of the nearest relevant test condition. Validation and modelling of materials must be instrumental to specify the precise purpose and vehicle for the in pile testing, and PIE path. Collaboration with fusion materials engineers, scientists, designers, and manufacturers are essential for the utilization of the experimental data provided. The research reactor PALLAS, and related laboratory activities also form a fertile soil for the training, education and real life nuclear experience of the new generations of fusion power experts.
Use of research reactor for materials research

The LVR-15 is a tank type reactor and currently undergoes the conversion from the IRT-2M fuel of 36 wt.% $^{235}$U enrichment to IRT-4M 19.7% via mixed cores. The LEU fuel will enable to increase the output reactor power from present 10 MW to 11-12 MW and that way to compensate lower fission density of HEU. The thermal and fast neutron flux reach up to $1.5 \times 10^{18}$ n/(m$^2$s) and $2.5 \times 10^{18}$ n/(m$^2$s), respectively. The reactor exploitation is ~ 58% resulting from the average ten 21 days operational cycles.

Due to its power output and achievable neutron fluxes the LVR-15 reactor for the study of combined effects of radiation and ambient media on materials. The reactor is equipped with experimental facilities such as loops and rigs, which permit an exposure under simulated conditions corresponding to those in power reactors.

Irradiation rigs permit the exposure starting from small samples (ring, tensile) up to very large samples (1CT, 2CT). Five loops simulating either PWR or BWR conditions in various irradiation channels, and other specialized facilities are in the operation at the reactor:

Reactor rigs –
Chouca (for Charpy V, tensile, 0.5 CT specimens),
flat rig (for batches of small specimens and/or 1-2 CT specimens).

Reactor loops –
BWR –1 (for structural material testing),
BWR-2 (for reactor pressure vessel (RPV) and internals steels testing),
Zinc loop (for radioactive material transport and water chemistry testing),
RVS-3 (for PWR/VVER water chemistry),
RVS-4 (for testing of fuel cladding corrosion),

Irradiation channels - in-pile channel for RPV steel, in-pile channel for austenitic steel, and in-pile channel for slow strain rate tests (SSRT),
HTHL (high temperature He loop),
SCWL (supercritical water loop),
Pb-Li loop and primary first wall (PFW) materials of the fusion program.

The flexible diameter of irradiation channels and good access to the upper parts of the channels are considered to be the advantage of the reactor with respect to the applications. Moreover, the core can be refueled without outage of the irradiation facilities from the reactor. Other important features in the field of material research are that the material can be pre-irradiated in a rig and then consequently exposed in active channel of a loop enabling also the simulation of the thermal flux or physical stresses. Water chemistry and dosimetry control ensuring the conditions in testing facilities to be as close as to the conditions in power plants is an inevitable constituent part of every loop. The reactor is equipped with hot cells for a post-irradiation sample manipulation, disassembling and assembling core channels.

Reactor LVR-15 has been engaged in the research of the fusion reactor materials and technology under EFDA and F4E/ITER.
Under the EFDA projects there were several tasks solved during the last years.
- Static and dynamic fracture toughness testing at the transition temperature of EUROFER 97 base metal and weld metal. The specimens were irradiated up to 2.5 dpa at the temperature of 200 - 250°C and the static and dynamic fracture toughness was measured at the transition temperature. The results were compared with non-irradiated reference specimens.
- The second task of the EFDA project was in-pile testing of EUROFER weld metal in Pb-17Li eutectic melt. A special rig simulated TBM environment with Pb-17Li liquid metal was designed and operated with the specimens. The target dose was 1.45 dpa at 500°C. The goals were to perform compatibility and corrosion test of EUROFER 97 weld metal with Pb-17Li liquid metal and to investigate chemical and structure stability of Pb-Li eutectic alloy.
- The third task was to develop and test the key components for PbLi ancillary system including mechanical pump, feasibility study of a cold trap and high temperature flanges. The resulted MeLiLo loop confirmed the required parameters of the pump and the effectiveness of the cold trap was evaluated on reductions of Fe, Mn, Ni, Cr, and corrosion products concentrations.

Study of Steady State Magnetic Diagnostic components designed for the measurement of ITER ex-vessel magnetic field was performed at LVR-15. Twenty types of ITER candidate Hall sensors were tested on LVR-15 during previous 6 years. Results from LVR-15 and also from complementary irradiation tests done at IBR-2 and WWR-M (Russia) demonstrate that InSb based Hall sensors manufactured by MSL, Lviv, Ukraine are able to satisfy ITER radiation stability requirements.

The research in the fusion technology has been continuing at LVR-15 reactor under the TBM Consortium of Associates in developing PbLi Ancillary System for HCLL TBM. The main goals are to design and manufacture the system with closed circuit for HCLL TBM blanket concept ensuring circulation, storage and processing of liquid metal breeder Pb-17Li. The system has to enable TBM tritium breeding performance measurement and gravity assisted draining in case of emergency or accident.
Other projects related to TBM are oriented on the qualification campaign for the ITER First Wall (FW) consisted of the fabrication and testing of small-scale mock-ups to demonstrate the ability of the selected fabrication technology to resist to the expected thermal loads. In the first stage an experimental device was designed according to the specifications of out-of-pile thermal fatigue testing that required to provide cyclic heat flux up to 0.625 MW/m² and measurement of temperatures on the joint between Beryllium tiles and CuCrZr heat sink. The BESTH (Beryllium Sample THERmal testing) facility was successfully tested. Afterwards, 5 mock-up have been tested in the BESTH. The first two EU-US FWQM were tested 12 000 cycles; with 300 seconds per cycle, total testing time was estimated to 3 600 000 seconds. The test have continued using Russian and Korean mock-ups (RF-KO FWQM) and were finished with Chinese – EU (CN FWQM - EU PROXY) campaign.

Thermal Fatigue Tests of Be Coated Primary First Wall will continue with in-pile test. A flat rig has been designing to enable to accommodate two small mock-ups according to the F4E proposition: Heat flux: 0.625 MW/m², Inlet water coolant temperature: 100°C, Inlet water coolant pressure: 0.6 MPa, Number of cycles: 20 000, Cycle duration: 390 s (30-180-30-150).

Some fusion relevant materials were characterized (structural changes, thermal, electrical, mechanical properties, heat flux performance) after ~1 dpa irradiation at LVR-15 under Czech Ministry of Industry Project. Especially, plasma facing materials (tungsten - plasma facing armour, copper - heat sink for PFC, stainless steel - construction under PFC, SiC + glass-ceramics - high temperature PFC, joining), electrical insulation (alumina), and construction materials (steels) were irradiated and tested.

Figure 2: Holder with In-pile thermal testing device for Primary First Wall Mock-ups with Be cladding
Utilization of High Neutron Flux Experimental Fast Reactor “Joyo” for Fusion Material Research

The experimental fast reactor Joyo at the Oarai Research and Development Center of the Japan Atomic Energy Agency (JAEA) is the first sodium-cooled fast reactor in Japan. The major objectives of constructing Joyo are to obtain technical information about the liquid metal fast breeder reactor (LMFBR) through experience with its design, construction and operation, and to use the reactor as a fast neutron irradiation facility for the development of fuels, materials and other components required for the LMFBR program. Joyo has recently been upgraded to the high performance MK-III core to provide a more robust and capable irradiation test facility not only for FBR development but also for other fields such as fusion reactor, LWR and non nuclear industry.

Joyo attained initial criticality as a breeder core (MK-I core) in April 1977 and initially operated two 50 MWth and six 75 MWth duty cycles with the MK-I core. From 1982 to 2000, Joyo operated with the MK-II core as an irradiation test bed to develop the fuels and materials for the prototype fast breeder reactor Monju and future fast reactors.

From 2003, Joyo is operated with the MK-III core as a high performance irradiation test bed. The maximum fast neutron flux (E≥0.1 MeV) is approximately 3.8x10^{15} n/cm\(^2\)-s and the dpa rate is approximately 45 dpa/year in Fe at the core center. In a reflector region, approximately 1.6x10^{15} n/cm\(^2\)-s and 15 dpa/year can be achieved.

A variety of irradiation experience has been accumulated in Joyo, including monitoring of the driver fuels and control rods performance and irradiation of test fuels and materials using irradiation test devices. Fusion reactor materials such as vanadium alloy, tungsten, low activation ferritic steel and SiC/SiC have been irradiated in cooperation with Japanese universities.

A neutron spectrum is monitored by the activation method using a dosimeter set which consists of Fe, Ni, Cu, Ti, Co, Ta, Sc, Np and \(^{235}\)U. These dosimeters have reactions in different neutron energy. Thermal expansion difference monitor (TED) is used for irradiation temperature measurement. The TED is composed of a metallic sphere lid and either an inconel alloy container. The container is filled with sodium. The TED is loaded in the irradiation capsule with specimens. The sodium inside the container increases as a result of thermal expansion during the irradiation test. The maximum irradiation temperature is calculated by the calibration curve with temperature and volume increase. As a result of these measurements, the accuracy of dpa is 3-5 %, and irradiation temperature is 10 % with offline irradiation test subassemblies. On-line irradiation equipment, MARICO (MAterial testing RIg with temperature COntrol) can control the specimen temperature within ±4 deg-C by changing the gas gap thermal conductivity of the double walled capsule containing the specimen by varying the ratio of argon and helium fill gases.

In Oarai research and development center, there are three post irradiation examination facilities, fuel monitoring facility (FMF), material monitoring facility (MMF) and alpha-gamma facility (AGF). The characteristic non-destructive post-irradiation examination (PIE) technique is X-ray computer tomography (X-ray CT) which has been developed to observe
the structural change in the interior of irradiated fuel subassemblies. The diameters of central voids formed in the fuel pellets could be measured within an error of ±0.1 mm.

Joyo has been suspended its operation since 2007 for the bent irradiation test subassembly (MARICO-2) at an in-vessel storage rack. In-vessel visual inspections were conducted with radiation resistant fiberscopes and cameras. The condition of MARICO-2 and upper core structure which contacted with MARICO-2 were confirmed. The fabrication of retrieval devices will start from fiscal year 2011, and JAEE plans to restart Joyo from fiscal year 2014.
Current Status and Perspectives of Materials Irradiation Tests in HANARO

Korea operates 20 nuclear power plants, accounting for 38% of electricity consumption, with eight new nuclear plants under construction. KAERI was established in 1959 to lay the foundation for achieving national nuclear energy self-reliance, and plays a central role in research and development of national nuclear energy. There are more R&D projects for Gen-IV materials than for fusion materials in Korea. But the development of cross-cutting material technologies is highly anticipated between Gen-IV and fusion nuclear systems. Korea started to build HANARO, a 30MW open-pool type multipurpose research reactor, in 1985. HANARO has been used to conduct research using neutron beams to produce medical and industrial radioisotopes and test materials using irradiation. It is decided to build another research reactor in order to produce radioisotopes and realize new technologies. HANARO was designed to provide a peak thermal of $5.4 \times 10^{14}$ n/cm$^2$/s and a fast flux $2.1 \times 10^{14}$ n/cm$^2$/s. HANARO is equipped with 32 vertical holes for irradiation tests, neutron transmutation doping and radioisotope production and 7 horizontal beam ports of different types available for researches on neutron scattering, neutron radiography, prompt gamma neutron activation analysis and medical applications such as a boron neutron capture therapy.

HANARO follows an established operation mode of 24 days of operation followed by an 11 day shutdown. Various neutron irradiation facilities such as the hydraulic rabbit (small non-instrumented capsule), the non-instrumented and instrumented capsules and the fuel test loop (FTL) facilities for irradiation tests of nuclear materials, fuels, and radioisotope products have been developed at HANARO. The rabbit was originally designed for an isotope production, but it can be used for the irradiation test of a fuel and a material. It is very useful for numerous irradiation tests of small specimens at a low temperature (below 200°C) and neutron flux condition. The dimension of the non-instrumented capsule is typically 1000mm in length and 60 mm in diameter. A specimen temperature is controlled by varying the widths of gas-filled gaps or vacuum gaps between the specimen and a specimen holder. An instrumented capsule has a cylindrical shape and its main body is 60 mm in diameter and 880mm in length. The basic instruments of the capsule are thermocouples, fluence monitors and heaters. The specimen temperature is controlled by a capsule temperature control system. Capsules for performing creep or fatigue test of materials have also been developed. A fluence control capsule system was designed by lifting up a specimen after desired neutron fluence has achieved, hence controlling fluence irrespective of a reactor operation period. The fluence control capsule will make it possible to irradiate specimens at different temperatures and with different fluences.

Fusion reactions generate 14 MeV neutrons from (d, t) reactions. This 14 MeV component produces a higher energy component of the PKA recoil spectra and higher rate of transmutations. Fission research reactors can provide appropriate neutron flux and large irradiation volume, but the produced radiation damage is not comparable. Although fission research reactors do not provide a complete simulation of the fusion environment, they do allow progress to be made in understanding high temperature radiation damage phenomena. To achieve this, advanced capsules including a high temperature irradiation technology up to 1000°C, re-instrumentation, and re-irradiation technology are under development.
ANNEX IV. AGENDA
Consultancy Meeting on
Role of Research Reactors in Material Research for Nuclear Fusion Technology
13-15 December 2010
VIC, Room A2313, IAEA, Vienna, Austria

**Monday, 13 December 2010**

<table>
<thead>
<tr>
<th>Time</th>
<th>Session</th>
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<tbody>
<tr>
<td>08:00-09:00</td>
<td><strong>Registration</strong></td>
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<tr>
<td>09:00-09:30</td>
<td><strong>Welcome &amp; Opening Remarks</strong></td>
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<tr>
<td></td>
<td>Mr Pablo Adelfang (Section Head, Research Reactor Section, IAEA)</td>
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<td>Mr Danas Ridikas and Mr Richard Kamendje (Scientific Secretaries of the Meeting, IAEA)</td>
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<td></td>
<td>Self introduction of the participants; Selection of the Chairperson &amp; Rapporteur</td>
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<td>Approval of the Agenda, Discussion &amp; Administrative Arrangements</td>
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<tr>
<td>09:30-09:45</td>
<td><strong>Mr D. Ridikas &amp; Mr R. Kamendje, IAEA: Introduction &amp; Objectives of the Meeting</strong></td>
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<tr>
<td>09:45-10:30</td>
<td><strong>Mr Eberhard Diegle, F4E, EU</strong></td>
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<tr>
<td>10:30-11:00</td>
<td><strong>Coffee break</strong></td>
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<tr>
<td>11:00-12:30</td>
<td><strong>Mr Tatsuo Shikama, IMR, Japan</strong></td>
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<td>Ms Sehila M. Gonzalez de Vicente, EFDA, Germany</td>
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<tr>
<td>12:30-14:00</td>
<td><strong>Lunch break</strong></td>
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<tr>
<td>14:00-15:30</td>
<td><strong>Mr Farong Wan, USTB, China</strong></td>
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<td>Ms Saibaba Saroja, IGCAR, India</td>
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<tr>
<td>15:30-16:00</td>
<td><strong>Coffee break</strong></td>
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<tr>
<td>16:00-17:30</td>
<td><strong>Ms Olga Yeliseyeva, NASU, Ukraine</strong></td>
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<td>Mr V. M. Chernov, Bochvar Institute, Russian Federation</td>
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<td></td>
<td><strong>Summary Discussion</strong></td>
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<td>18:00</td>
<td><strong>Hospitality Event</strong></td>
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## Tuesday, 14 December 2010

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<th>Time</th>
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<tbody>
<tr>
<td>09:00-10:30</td>
<td>Mr Hiroyasy Tanigawa, JAEA, Japan</td>
<td>Mr Nikolai Loginov, IPPE, Russian Federation</td>
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<tr>
<td>10:30-11:00</td>
<td><em>Coffee break</em></td>
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<tr>
<td>11:00-12:30</td>
<td>Mr Jaap G. van der Laan, NRG, The Netherlands</td>
<td>Mr Bob van der Schaaf, NRG, The Netherlands</td>
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<tr>
<td>12:30-14:00</td>
<td><em>Lunch break</em></td>
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<tr>
<td>14:00-15:30</td>
<td>Mr Marek Milan, NRI Rez, Czech Republic</td>
<td>Mr Takashi Sekine, JAEA, Japan</td>
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<tr>
<td>15:30-16:00</td>
<td><em>Coffee break</em></td>
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<tr>
<td>16:00-17:30</td>
<td>Mr Chansun Shin, KAERI, Republic of Korea</td>
<td>Discussion: Objectives 1+2 of the meeting</td>
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<td>Drafting of conclusions &amp; recommendations</td>
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## Wednesday, 15 December 2010

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<tr>
<td>09:00-10:30</td>
<td><em>Discussion: Objectives 3+4 of the meeting</em></td>
<td><em>Discussion: Drafting of conclusions &amp; recommendations</em></td>
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<td>10:30-11:00</td>
<td><em>Coffee break</em></td>
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<tr>
<td>11:00-12:30</td>
<td><em>Discussion: Finalizing of conclusions &amp; recommendations</em></td>
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<tr>
<td>12:30-14:00</td>
<td><em>Lunch break, end of the Meeting</em></td>
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## ANNEX V. LIST OF PARTICIPANTS

Consultancy Meeting on  
Role of Research Reactors in Material Research for Nuclear Fusion Technology  
13-15 December 2010  
VIC, Room A2313, IAEA, Vienna, Austria

<table>
<thead>
<tr>
<th>Country</th>
<th>Expert’s Contact Information</th>
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<tbody>
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<td>1 China</td>
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<td>Ms Sehila M. Gonzalez de Vicente</td>
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<td>Materials Responsible Officer</td>
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<td>EFDA Close Support Unit - Garching</td>
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<td>Ms Saibaba Saroja</td>
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<td>Head, Nuclear Materials Microscopy Section</td>
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<td>Physical Metallurgy Division</td>
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<td>Mr Tatsuo SHIKAMA</td>
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<td>Institute of Materials Research</td>
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<td></td>
<td>Tohoku University</td>
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<td>Mr Sekine Takashi</td>
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