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Proceedings of the International Symposium on Materials Testing Reactors

July 16-17, 2008, JAEA Oarai R&D Center, Japan

(Eds.) Masahiro ISHIHARA and Hiroshi KAWAMURA

Neutron Irradiation and Testing Reactor Center
Oarai Research and Development Center

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This report is the Proceedings of the International Symposium on Materials Testing Reactors hosted by Japan Atomic Energy Agency (JAEA). The symposium was held on July 16 to 17, 2008, at the Oarai Research and Development Center of JAEA. This symposium was also held for the 40th anniversary ceremony of Japan Materials Testing Reactor (JMTR) from achieving its first criticality. The objective of the symposium is to exchange the information on current status, future plan and so on among each testing reactors for the purpose of mutual understanding. There were 138 participants from Argentina, Belgium, France, Indonesia, Kazakhstan, Korea, the Russian Federation, Sweden, the United State, Vietnam and Japan.

The symposium was divided into four technical sessions and three topical sessions. Technical sessions addressed the general topics of “status and future plan of materials testing reactors”, “material development for research and testing reactors”, “irradiation technology (including PIE technology)”and “utilization with materials testing reactors”, and 21 presentations were made. Also the topical sessions addressed “establishment of strategic partnership”, “management on re-operation work at reactor trouble” and “basic technology for neutron irradiation tests in MTRs”, and panel discussion was made.

Keywords : Material Testing Reactors, International Symposium, Status and Future Plan of MTR, Material Development, Irradiation Technology, PIE, Utilization, World Network, Management on Re-operation

「汎用照射試験炉に関する国際会議」論文集
2008 年 7 月 16-17 日、大洗研究開発センター

日本原子力研究開発機構　大洗研究開発センター
照射試験炉センター
（編）石原　正博、河村　弘

（2008 年 10 月 27 日受理）

本論文集は、（独）日本原子力研究開発機構主催の「汎用照射試験炉に関する国際会議」に提出された論文をまとめたものである。本国際会議は、JMTR 初臨界から 40 周年を記念し、各照射試験炉の相互理解を深めるための情報交換を目的として 2008 年 7 月 16 日から 17 日に JAEA 大洗研究開発センターで開催された。会議には、米国、フランス、スウェーデン、ベルギー、ロシア、カザフスタン、韓国、ベトナム、インドネシア、アルゼンチン及び日本から合計 138 名が出席した。

本国際会議では、「照射試験炉の現状と今後の計画」、「照射試験炉のための材料開発」、「照射技術（照射後試験技術含む）」及び「材料試験炉を用いた利用」のセッションにおいて 21 件の講演が行われ、「相互補完的なパートナーシップの構築」、「照射試験炉の再起動に関する管理」及び「材料試験炉における中性子照射のための基本技術」のトピックスについてパネル討論が行われた。

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1. Introduction

1.1 SYMPOSIUM OBJECTIVE AND PROSPECTS

Opening Remarks by General Chair, Masao Takuma

At the beginning, I express my gratitude of this symposium as a general chairman.

Here, we have so many persons around the world to participate in this symposium from Europe, the United States, South America and Asia as well as from Japan. I say again “thank you very much for coming this symposium”. Under the situation of global warming in this century, the importance of the nuclear energy is re-recognized together with so-called "3S", namely Safety, Security and Safeguard, in Toyako Summit Discussion held at 7- 9th July, 2008. The new nuclear technology era, so-called "Nuclear Renaissance" is coming globally now, in close relation to the future of "Our Earth and Our Human Society".

Here, the first “International Symposium on Material Testing Reactors” is held with objectives of information exchange among each testing reactor for the purpose of mutual understanding of present status, moreover discussion on the world network construction of testing reactors. At first, we make information exchange on “present status and future plan of each materials testing reactor” for a mutual understanding, and also make information exchange on ”the construction of world network from a viewpoint of globalization of user” as well as “a management of re-operation at reactor trouble” in special sessions in the program. Next, we organize three sessions concerning material development, irradiation technology and utilization for materials testing reactors to make information exchange in detail from a technical standpoint. In addition, a special session of a basic technology for neutron irradiation test is also organized to exchange information.

Through above organized sessions during two-symposium-days,

I will expect that deeper understanding of each other for the situation of each reactor in each country, moreover I will also expect that effective proposals concerning the construction of global network will be made to contribute further expansion of nuclear technology. I think that "International cooperation and solidarity towards the peaceful use of nuclear power" is the keyword of the nuclear power renaissance in the 21st century.

For closing my message, I express my gratitude to symposium committee persons, secretaries, as well as to chair persons and to all persons concerned.

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2. Status and Future Plan of Material Testing Reactor

2.1 JHR
A HIGH PERFORMANCE MTR UNDER CONSTRUCTION FOR A SUSTAINABLE NUCLEAR ENERGY

Daniel Iracane[1] and Pierre-Yves Cordier[2]

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The Access to an up-to-date Material Testing Reactor (MTR) is essential to support a sustainable nuclear energy, meeting industry and public needs, and keeping a high level of scientific expertise. This includes services to existing and coming reactor technologies for major stakes such as safety and competitiveness, lifetime management, operation optimization, development of innovative structural material and fuel required for future systems (innovative Gen III, Gen IV, fusion…), etc.

The JHR copes with this context.

Design phase has been completed by the end of 2005 and JHR is now under construction. Start of operation is scheduled in 2014. As a new MTR taking benefit of a large available worldwide experience, JHR offers new major experimental capability that will be presented.

JHR will be operated within an international users' consortium that will guarantee effective and cost-effective operation. This innovative way to operate a MTR, as a user-facility for the benefit of industry and public bodies, will be presented.

Keywords: Material testing reactor, JHR project, fuel and material behavior under irradiation.

1. INTRODUCTION

The development of a sustainable nuclear energy requires R&D on fuel and material behavior under irradiation with a high level of performances in order to meet the following needs and challenges for the benefit of industry and public bodies:

1 - A constant improvement of the performances and safety of present and coming water cooled reactor technologies. Taking into account the lifetime extension and the progressive launch of generation III, NPPs using water coolant will be in operation through the entire century. They will require a continuous R&D support following a long-term trend driven by the plant life management, safety demonstration, flexibility and economics improvement. Lifespan extension of present Generation II reactors and demonstration of the lifespan of coming Generation III reactors is a major economical stake: capital appreciation, paying-off production facility, less investment required for producing electricity. Experimental irradiations of structure materials are necessary to anticipate these material behaviors and will contribute to the operation optimization.

2 - Fuel technology in present and future nuclear power plants is continuously upgraded to achieve better performances and to optimize the fuel cycle, still keeping the best level of safety. Fuel evolution for generation II and III is and will stay a key stake requiring developments, qualification tests and safety experiments to ensure the economical competitiveness and safety: experimental tests exploring the full range of fuel behavior, determine fuel stability limits and safety margins, as a major input for the fuel reliability analysis.

3 - To meet nuclear energy sustainable development objectives in the resources and waste management, generation IV reactors are mandatory and require innovative materials and fuels which resist to high temperatures and/or fast neutron flux in different environments. These environments will be needed for demonstrating the technical, economical and safety performances of these technologies. The selection, optimization and qualification of these innovative materials and fuels raise critical issues concerning their in-service behavior; utilization of high performance Material Testing Reactors and other facilities will be necessary to fix these issues.

4 - In addition, such a research infrastructure will contribute to build up technical skills in the nuclear industry and to train a new generation of research scientists, engineers and, ultimately, executives. Indeed, a high performance Material Testing Reactor, operated within international cooperation and complementary to other domestic research infrastructures, offers the appropriate

framework to attract younger generations and to cross-fertilize the international skill.

Material Testing Reactors (MTR) dedicated to testing material and fuel under irradiation are essential tools to address the aforesaid issues. They have provided key support for nuclear development for over 40 years.

However, most of existing MTRs are facing obsolescence in terms of safety and experimental capacity and a new high performance material testing reactor is needed to meet industrial and public needs, as identified above [1].

The Jules Horowitz Reactor -JHR- copes with this objective.

JHR is designed, built and operated as an international user-facility in order:

- to offer the level of economic and technical efficiency required by a global and mature industry,
- to share costs and results for topics of common interest and requiring large consensus (safety, waste management, etc...).

JHR design is optimized for offering high performance material and fuel irradiation capability for the coming decades.

The JHR is now under construction.

2. JHR STATUS AND PERFORMANCES

2.1 JHR schedule

The detailed design studies have been completed by the end 2005. The Regulatory Body has assessed the preliminary safety options in April 2003 and the preliminary safety analysis report in June 2008. The construction permit has been delivered in September 2007 and the JHR site preparation and excavation (cf. figure 1) have been completed (2007-2008). First concrete is scheduled in spring 2009.

2.2 JHR nuclear infrastructure

The JHR nuclear infrastructure meets modern safety standard [2-4]. Compared to most of (almost 50 years old) existing MTRs, major progress has been performed on the containment building, on the cooling circuit, on seismic hazards management, etc. With these up-dated features, JHR will offer a sustainable and secured irradiation capacity for the coming decades.

Figure 1: Site excavation will be completed The site excavation (200 000 m3 of limestone) has started in spring 2007. The nuclear building excavation (40 000 m3) and related reinforcements have been completed at mid-2008.

2.3 The JHR experimental capacity

The JHR is a research infrastructure to perform screening, qualification and safety experiments on material and fuel behavior under irradiation [5-8].

JHR is a water cooled reactor to provide the necessary flexibility and accessibility for managing several highly instrumented experiments, reproducing different reactor environments (water, gas or liquid metal loops), generating transient regimes (a key point for safety).

The JHR facility gathers (cf figure 2):

- the reactor building including the core, the cooling system and the experimental bunkers directly connected to the core through pool wall penetrations
- and the auxiliary building including the pools and hot cells necessary for the experimental irradiation process.

JHR core (cf figure 3) is optimized to produce high fast neutrons flux to study structural material ageing and high thermal neutrons flux for fuel experiments.

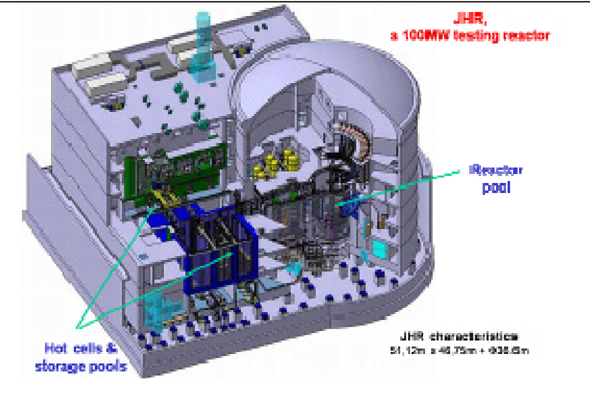

Figure 2: JHR layout showing the Reactor Building and the Auxiliary Building, crossed together by a water block connecting the core, the poles and the hot cells

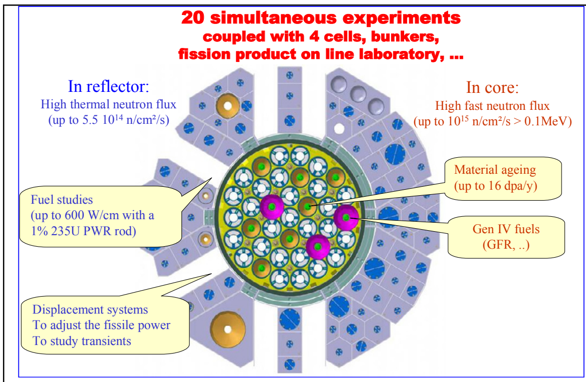

Figure 3: The JHR core, in his core pool, is a high power density fuel rack in a vessel slightly pressurized and surrounded by a Beryllium reflector. Experiments can be implemented in the centre of fuel elements, in place of fuel elements, in beryllium block or in water channels crossing the reflector.

The JHR experimental capability is typically ~ 20 simultaneous experiences in-core plus in reflector providing suited environments relevant for different reactor technologies and high neutron flux (taking into account a full experimental loading):

- Fast neutrons perturbed flux: 10^{15} n/cm²/s >0.1MeV and 5.10^{14} n/cm²/s >1MeV. This corresponds to 16 dpa/year (that is, 8 times larger than the damage on a PWR internal material)
- Thermal neutrons flux: 5,5 10^{14} n/cm²/s, (ie. 600W/cm on 1% enriched PWR fuel pin) to accelerate fuel BU and to simulate power transients

The JHR experimental performance relies also on key out-of-core components:

- Loops for power reactors in normal or non-normal conditions (relevant coolant, T, P features)
- Effective transient devices for safety studies, a major scientific challenge
- Hot cells for the current operation (preparing the experiment, non destructive exams) and alpha cell for Safety fuel experiments (figure 3)
- On line instrumentation and control (more data, better management, extrapolation capability with modelling)
- On line fission product measurement laboratory for gas and liquids

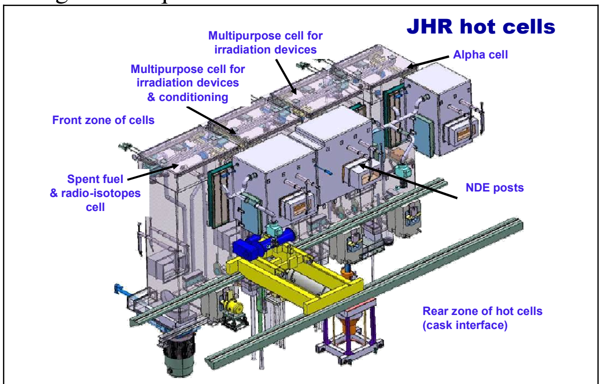

Figure 4: JHR hot cells providing necessary interfaces and non destructive exams. The alpha cell allows effective operations with damaged experimental fuels.

3. JHR MEMBERSHIP

3.1 JHR, a user-facility open to international cooperation

As a major research infrastructure for the international nuclear community, JHR is funded and steered by a Consortium binding Members contributing to the financing of the JHR construction.

The construction cost is 500 Million Euros (500M□2005) from 2006 to 2014 at the 2005 economic situation.

A Member of the JHR Consortium will benefit from a Guaranteed Access Right to the JHR experimental capacity, thus extending its domestic experimental R&D capabilities toward a high performance Material Testing Reactor.

As an illustration, a membership share in the JHR project of 3% (resp. 5%) corresponds to

- A contribution of 15 M□2005 (resp. 25M□2005) to the construction cost,
- Full reserved access rights on 3% (resp. 5%) of JHR experimental capability. Detailed management of these access rights and cumulative possibilities are defined in the Consortium Agreement signed by Members,

- Voting rights in the Consortium Board at the level of 3% (resp. 5%).

CEA is responsible of the JHR construction and final cost. Possible evolution of the JHR construction cost has no impact of non-CEA Members contributions and shares.

3.2 JHR financing structure

Together with CEA, several partners are already committed in the financing of the JHR construction:

- industrial companies such as EDF, AREVA, VATTENFALL
- research labs and public bodies from Belgium, Czech Republic, Finland, India, Japan and Spain.

A JHR Member may gather a domestic association of partners to consolidate its national funding; in this case, the association partners jointly benefit from the Guaranteed Access Rights to the JHR experimental capacity.

CEA is still willing to enlarge the JHR international partnership.

3.3 JHR membership terms

The JHR Consortium Agreement will bind Members contributing to the financing of the JHR construction.

CEA is the owner and nuclear operator of the JHR. CEA shall assume all the liabilities related to the decommissioning phase; for that purpose, CEA will generate a fund as a part of the annual operation cost.

Non-CEA Members are not responsible for any financial and material damage resulting from the construction, the operation or the decommissioning of the reactor by the CEA. The JHR Consortium covers a system without legal personality.

A Member has Guaranteed and secured Access Rights to experimental locations in the JHR:

- To perform proprietary experimental programs for the exclusive benefit of the Member
  And/or
- To participate in a international joint program gathering several Members among which cost and results are shared
  And/or
- To implement bilateral or multilateral programs with other Members to the convenience of involved Members

Operation costs are paid only for utilised rights; access rights can be utilised partly or in totality each year.

A Member participates in the JHR Governing Board with voting rights equal to its membership share. The Governing Board is responsible for the policy and strategic orientation of the Consortium, for the commercial policy toward Non-Members, for adopting the reference JHR operation plan and the annual report, etc…

Non-Members will be able to access the JHR facility under decision of the Governing Board and within conditions defined by the strategic and commercial policy of the JHR Consortium.

4. CONCLUSIONS

Nuclear energy has become a very competitive industry. The renewal of a worldwide interest for nuclear fission technologies demonstrates a general recognition of the merits of this energy source.

Consistently, in the context of a growing demand of energy, an important increase of the nuclear production is expected in the coming decades.

This raises key challenges for the industry and the public bodies regarding the life-time extension, the operation optimization, the evolution of current LWR technologies and the development of fast neutrons reactor technologies, the safety demonstration being a continuous top level objective.

These challenges requires from research institutes to offer renewed and sustainable competences and infrastructures to support and this evolution.

The JHR project, as a new high performance MTR operated as an international users-facility, fit this requirement; indeed, JHR will:

- contribute to the long term securing of the experimental capacity,
- offer high performance experimental capacity,
- attract new generation of scientists and engineers.

On the world scene, a coordination between worldwide MTRs :

- can improve the common performances and skills (personnel exchanges, experimental technology co-development …),
- allows optimizing the services for the benefit of Industry and Public Bodies by accessing specific performances as needed or by decreasing burdens like transportation of sample.

Nevertheless, such coordination will require a careful management taking into account confidentiality issues related to end-users.

REFERENCES

[1] C. Vitanza, D. Iracane, D. Parrat, “Future needs for materials test reactors in Europe (FEUNMARR Findings)”, 7th International Topical Meeting on Research Reactor Fuel Management, ENS, RRFM 2003.

[2] D. Iracane, “The Jules Horowitz Reactor, a new

Material Testing Reactor in Europe" Proc. TRTR-2005 / IGORR-10 Joint Meeting, Gaithersburg, Sept. 2005

[3] JP. Dupuy, G. Ithurralde, G. Perotto, C. Leydier, X. Bravo "Jules Horowitz Reactor: General layout, main design options resulting from safety options, technical performances and operating constraints"; IGORR 10, Gaithersburg, Sept. 2005

[4] M. Boyard, JM. Cherel, C. Pascal, B. Guigon; "The Jules Horowitz Reactor core and cooling system design"; IGORR 10, Gaithersburg, Sept. 2005

[5] D.Iracane, Daniel Parrat, J. Dekeyser, H. Bergmans, K.Bakker, S. Tahtinen, J. Kysela, C. Pascal, A. Jianu, D. Moulin, M. Auclair, L. Fournier, S. Carassou, S. Gaillot, « The JHR Co-ordination Action (JHR-CA), an European collaboration for designing up-to-date irradiation devices for materials and fuels in MTRs", FISA2006, Luxembourg, March 2006

[6] J.F. Villard; Innovative in-pile instrumentation developments for irradiation experiments in MTRs; IGORR 10, Gaithersburg, Sept. 2005

[7] G. Panichi, F. Julien, D. Parrat, D. Moulin, B. Pouchin, L. Buffe, N. Schmidt, L. Roux. "Developing irradiation devices for fuel experiments in the Jules Horowitz Reactor"; IGORR 10, Gaithersburg, Sept. 2005.

[8] S. Carassou, G. Panichi, F. Julien, P. Yvon, M. Auclair, S. Tahtinen, P. Moilanen, S. Maire, L. Roux. "Experimental material irradiation on the Jules Horowitz Reactor"; IGORR 10, Gaithersburg, Sept. 2005.

2.2 Advanced Test Reactor – Testing Capabilities and Plans

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The Advanced Test Reactor (ATR), at the Idaho National Laboratory (INL), is one of the world's premier test reactors for providing the capability for studying the effects of intense neutron and gamma radiation on reactor materials and fuels. The physical configuration of the ATR, a 4-leaf clover shape, allows the reactor to be operated at different power levels in the corner "lobes" to allow for different testing conditions for multiple simultaneous experiments. The combination of high flux (maximum thermal neutron fluxes of 1E15 neutrons per square centimeter per second and maximum fast [E>1.0 MeV] neutron fluxes of 5E14 neutrons per square centimeter per second) and large test volumes (up to 122 cm long and 12.7 cm diameter) provide unique testing opportunities. For future research, some ATR modifications and enhancements are currently planned. In 2007 the US Department of Energy designated the ATR as a National Scientific User Facility (NSUF) to facilitate greater access to the ATR for material testing research by a broader user community. This paper provides more details on some of the ATR capabilities, key design features, experiments, and plans for the NSUF.

Keywords: Advanced Test Reactor, Idaho National Laboratory, test reactor, research reactor, irradiation testing, neutron flux, National Scientific User Facility

1. INTRODUCTION

The Advanced Test Reactor (ATR), located at the Idaho National Laboratory (INL), is one of the most versatile operating research reactors in the United States. The ATR has a long history of supporting reactor fuel and material research for the US government and other test sponsors. The INL is owned by the US Department of Energy (DOE) and currently operated by Battelle Energy Alliance (BEA). The current experiments in the ATR are for a variety of customers – US DOE, foreign governments, and private researchers, and commercial companies that need neutrons. The ATR has several unique features that enable the reactor to perform diverse simultaneous tests for multiple test sponsors. The ATR has been operating since 1967, and is expected to continue operating for several more decades. Also at the INL are several facilities used for experiment preparation and post irradiation examination (PIE). In 2007, DOE designated the ATR as a National Scientific User Facility (NSUF), enabling a broader user community the ability to perform research (irradiation testing and PIE) in the INL facilities. This paper discusses the ATR design features, testing options, previous experiment programs, future plans for the ATR capabilities and experiments, a brief overview of the PIE capabilities at the INL and some discussion of the NSUF plans.

2. ATR DESCRIPTION

The ATR is a pressurized, light-water moderated and cooled, beryllium-reflected highly-enriched uranium fueled, nuclear research reactor with a maximum operating power of 250 MWth. The INL is owned by the US Department of Energy (DOE). The ATR is one of the most versatile operating research reactors in the United States. The ATR core cross section, shown in Figure 1, consists of 40 curved aluminum plate fuel elements configured in a serpentine arrangement around a 3 by 3 array of large irradiation locations in the core termed "flux traps." The flux traps have the highest flux in the reactor due to the close proximity of the fuel. This core configuration creates five main reactor power lobes (regions) that can be operated at different powers during the same operating cycle. In addition to these nine flux traps there are 68 additional irradiation positions in the reactor core reflector tank. There are also 34 low-flux irradiation positions in the irradiation tanks outside the core reflector tank.

General design information and operating characteristics for the ATR are presented in Table 1. The ATR has several unique features that enable the reactor to perform diverse simultaneous tests for multiple test sponsors. The unique design of ATR control devices permits large power variations among its nine flux traps using a combination of control cylinders (drums) and neck shim rods. The beryllium control cylinders contain hafnium plates that can be rotated toward and away from the core, and hafnium shim rods, which withdraw vertically, can be individually inserted or withdrawn for minor power adjustments. Within bounds, the power level in each corner lobe of the reactor can be controlled independently to allow for different power and flux levels in the four corner lobes during the same operating cycle.

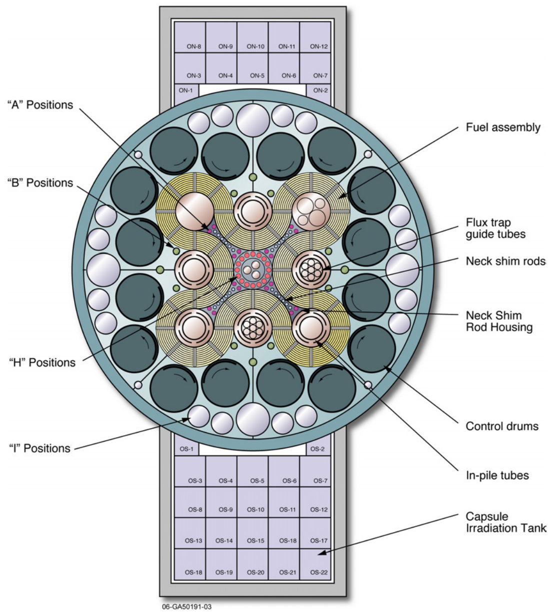

Figure 1. ATR Core Cross Section

Table 1. ATR Design and Operating Information

Reactor	
Thermal Power (Maximum Design Power)	250 MW_{th}
Power Density	1.0 MW/liter
Maximum Thermal Neutron Flux	1.0 $x10^{15}$ n/cm^2-sec
Maximum Fast Flux	5.0 $x10^{14}$ n/cm^2-sec
Number of Flux Traps	9
# Experiment Positions	68
Core	
Number of fuel assemblies	40
Active length of Assemblies	1.2 m (4 ft)
Number of fuel plates per assembly	19
Reactivity Control Drums/Rods	Hafnium
Primary Coolant System	
Design Pressure	2.7 MPa (390 psig)
Design Temperature	115°C (240°F)
Reactor coolant	Light water
Maximum Coolant Flow Rate	3.09 m^3/sec (49,000 gpm)
Coolant Temperature (Operating)	< 52°C (125°F) inlet, < 71°C (160°F) outlet

A typical operating cycle for the ATR consists of 42 to 56 operating days and 14 outages days, during which operators refuel the reactor and insert, remove, or reposition experiments. There are usually 6 operating cycles each year, for an average total of 250 operating days each year. Experiments remain in the ATR for the entire duration of the operating cycle. A hydraulic shuttle irradiation system (HSIS) is being installed into the ATR, and will be operational in October 2008. This will enable small volume, short duration irradiations to be performed in the ATR. This system will enable up to 16 small shuttle capsules will to be inserted into the shuttle system for a single shuttle operation.

Most experiments are handled, using long handled tools, through ports in the reactor vessel head, so that the vessel head does not need to be removed during every outage; some experiments need to be handled with a crane and are lifted directly out of the top of the vessel into a specially designed container.. All other experiments are moved into the adjacent ATR canal area for some cooling time prior to packaging for shipment to other facilities for post irradiation examination.

A unique feature of the ATR is that the core internal components are removed and replaced every 8-10 years, during a core internals changeout (CIC) , an outage of approximately six months duration. Additionally, the ATR reactor vessel is substantially larger than the core size allowing for reduced neutron flux embrittlement of the reactor vessel. Unlike commercial LWRs in the US, the ATR has no established lifetime or shutdown date. Analyses and surveillances are routinely performed to monitor the material condition of the key structural components.

The ATR also contains a separate facility, the Advanced Test Reactor Critical (ATRC) facility (Figure 2), which is a full-size replica of the ATR operated at low power (5 kW maximum) and used to evaluate the potential impact on the ATR core of experiment test trains and assemblies. Mock-ups of experiments can be inserted in the ATRC, and such parameters as control rod worths, reactivities, thermal and fast neutron distributions, gamma heat generation rates, ATR fuel loading requirements, and void/temperature reactivity coefficients can be determined prior to insertion into the ATR.

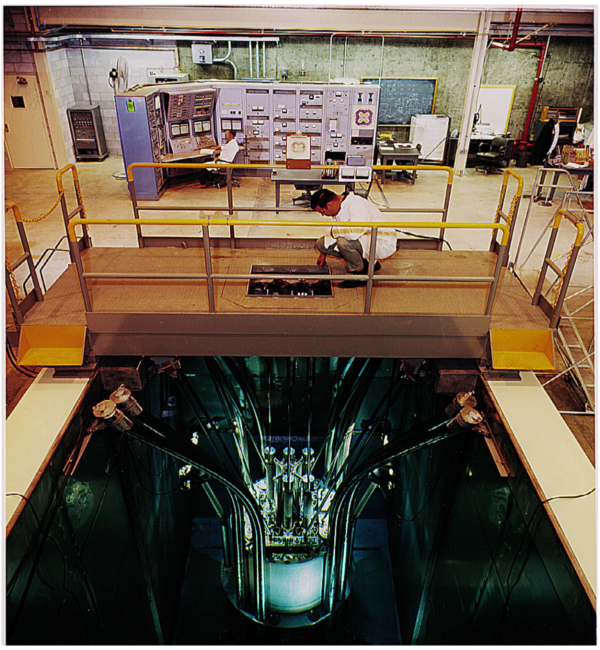

Figure 2. ATR Critical Facility

3. EXPERIMENT CAPABILITIES

There are three basic types of experiment configurations utilized in the ATR – the static capsule, the instrumented lead, and the pressurized water loop experiment. Each is described in more detail below, with some examples of the experiments performed using each type of configuration.

3.1 Static Capsule Experiment

The simplest experiment performed in the ATR is a static capsule experiment. The material to be irradiated is sealed in aluminum, zircaloy, or stainless steel tubing. The sealed tube is placed in a holder that sits in a chosen test position in the ATR. A single capsule can be the full 1.2 m core height, or may be shorter, such that a series of stacked capsules may comprise a single test. Capsules are usually placed in an irradiation basket to facilitate the handling of the experiment in the reactor. Figure 3 shows a simplified drawing of a static test capsule and basket assembly. Some capsule experiments contain material that can be in contact with the ATR primary coolant; these capsules will not be sealed, but in an open configuration, such that the capsule is exposed to and cooled by the ATR primary coolant system. Examples of this are Reduced Enrichment for Research and Test Reactors (RERTR) fuel plate testing, such that the fuel to be tested is in a cladding material similar to (or compatible with) the ATR fuel element cladding.

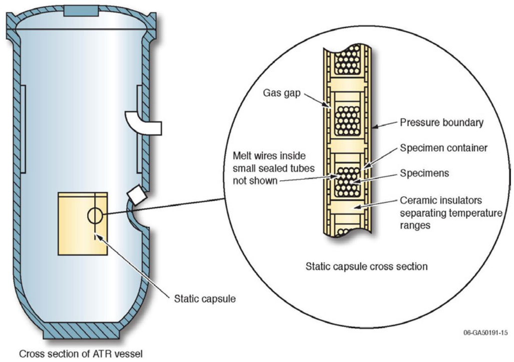

Figure 3. Static Capsule Assembly

Static capsules typically have no instrumentation, but can include flux-monitor wires and temperature melt wires for examination following the irradiation. Limited temperature control can be designed into the capsule through the use of an insulating gas gap between the test specimen and the outside capsule wall. The size of the gap is determined through analysis for the experiment temperature requirements, and an appropriate insulating or conducting gas is sealed into the capsule. An additional adjustment that has been used in static capsule experiments is flux tailoring in a single position – a filtering material can be used to change the fast:thermal ratio and fuel can be added to increase the overall flux.

Static capsule experiments are easier to insert, remove, and reposition than more complex experiment configurations. Relocations to a different irradiation location within the ATR are occasionally desired to compensate for fuel burn-up in a fuel experiment. A static capsule experiment is typically less costly than an instrumented one and requires less time for design and analysis prior to insertion into the ATR.

3.2 Instrumented Lead Experiment

The next level in complexity of ATR experiments is an instrumented lead experiment, which provides active monitoring and control of experiments parameters during the irradiation period. The primary difference between the static capsule and the instrumented lead experiment is an umbilical tube that runs from the experiment in the reactor core region through a penetration in the reactor vessel and houses instrumentation connections that lead to a monitoring/control station elsewhere in the reactor building. In a temperature-controlled experiment, thermocouples continuously monitor the temperature in the experiment and provide feedback to a gas control system to provide the necessary gas cooling mixture to the experiments to achieve the desired experiment conditions. The thermocouple leads and the gas tubing are in the umbilical tube. A conducting (helium) gas and an insulating (typically neon or possibly argon) gas are mixed to control the thermal conductance across a predetermined gas gap. The computer-controlled gas blending system allows for the gas mixture to be up to 98% of one gas and as low as 2% of the other gas to allow for a wide range of experiment temperature ranges. Temperature measurements are typically taken with at least two thermocouples per capsule to provide assurance against an errant thermocouple and to also provide redundancy in the event of a thermocouple failure. The INL has developed (and continues further research on) thermocouples capable of performing at ever higher temperatures – the current tests are operating at 1200°C and the next experiments will be operating closer to 1500°C Figure 4 shows a typical instrumented lead experiment.

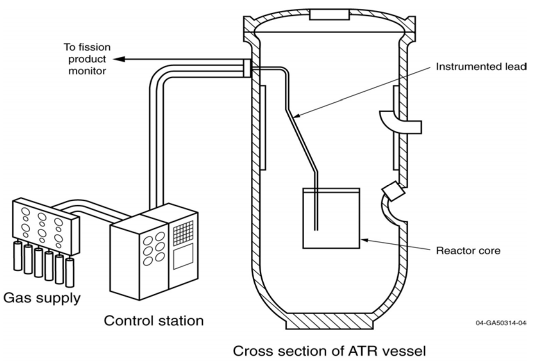

Figure 4. Example of an Instrumented Lead Experiment Configuration

Some of the instrumented lead experiments need specialized environments, such as an oxidized cover gas. The instrumented lead experiment allows for precise environmental conditions to be established and monitored, ensuring that the experiment data objectives can be met satisfactorily. Use of the instrumented lead experiment configuration enables researchers to monitor the gas around the test specimen for changes to the experiment conditions. In a fueled experiment, for example, there is sometimes a desire to test for fission gases, which could indicate a failure of the experiment specimen. Gas chromatography can also be used to monitor oxidation of an experiment specimen. The instrument leads allow for a real time display of the experiment parameters on an operator control panel. The instrumented leads can also be used to provide an alarm to the operators and experimenters if any of the experiment parameters exceed test limits. For any monitored experiment parameter, a data acquisition and archive capability can be provided; typically the data are saved for six months.

The primary advantage of the instrumented lead experiment is the active control of the experiment parameters that is not possible in a static capsule experiment. Additionally, the experiment sponsor does not have to wait until the full irradiation has been completed for all experiment results; the instrumentation provides preliminary results of the experiment and specimen condition.

3.3 Pressurized Water Loop Experiment

The pressurized water loop (PWL) experiment is the most complex and comprehensive type of testing performed in the ATR. Five of the ATR flux traps contain in-pile tubes (IPTs), connected to pressurized water loops, that provide a barrier between the reactor primary coolant system and a secondary pressurized water loop coolant system. The experiments are isolated from the ATR reactor coolant system since the IPT extends through the entire reactor vessel. There are closure plugs at the top and bottom of the vessel to allow the experiments to be independently inserted and removed.

The secondary cooling system includes pumps, coolers, ion exchangers, heaters to control experiment temperature, and chemistry control systems. All of the secondary loop parameters are continuously monitored, and computer controlled to ensure precise testing conditions. Loop tests can precisely represent conditions in a commercial pressurized water reactor. Operator control display stations for each loop continuously display information that is monitored by the ATR staff. Test sponsors receive preliminary irradiation data before the irradiations are completed, so there are opportunities to modify testing conditions if needed. The data from the experiment instruments are collected and archived similar to the data in the instrumented lead experiments. The real-time feedback of experiment conditions and irradiation results can also be an asset to the experiment sponsor..

4. POST IRRADIATION EXAMINATIONS

The Hot Fuels Examination Facility (HFEF) [1] (Fig. 5) located within the Materials and Fuels Complex (MFC) at Idaho National Laboratory is a large alpha-gamma multi-program hot cell facility which is designed to remotely characterize highly irradiated fuel and structural materials. Some of the most commonly performed tests are described below. In addition to the hot cell work, the INL operates several radiochemistry laboratories that are also available for PIE work. The wide range of fuel handling and measurement capabilities at HFEF, coupled with the INL's experience in testing and analyzing fuel behavior make HFEF an ideal facility in which to perform post-irradiation and spent nuclear fuel characterization activities.

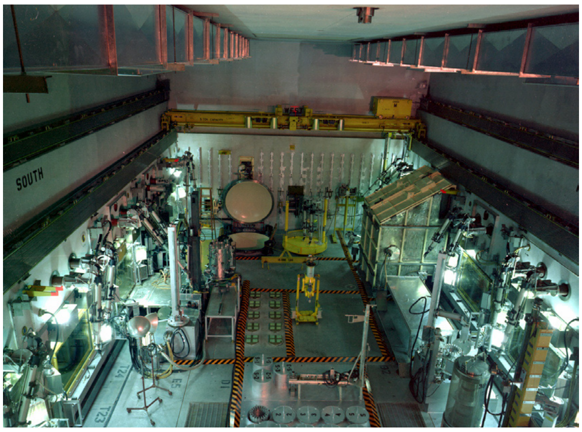

Figure 5. HFEF Main Cell (West View)

4.1 Visual Examination

An experiment capsule can be visually examined to determine its mechanical integrity and surface appearance, and the mass of the capsule can be measured. Deviations from as-built conditions will be examined by close-up photography prior to performing subsequent PIE tasks. Of note are capsule tube and end cap failures, cracks, deformations, blisters, areas of discoloration, corrosion, and loss of material by wear.

4.2 Neutron Radiography

Neutron radiography can be used to examine the internal condition of the capsule assembly prior to disassembly. This non-destructive technique is able to detect the presence of water or bonding material within the capsule cavity indicating a leak or containment failure. Data can be obtained on the general condition of the specimens, including axial position, radial position and gap, distortion or bow, and the fuel column length within the capsule.

Neutron radiography is performed at the MFC Neutron Radiography Facility which interfaces through the floor of the HFEF Main Cell. The neutron beam is generated by a 250 kW Training Research Isotope General Atomics (TRIGA) reactor located below the Main Cell.

4.3 Internal Gas Pressure and Void Volume Analyses

The Gas Assay, Sample, and Recharge system can be used to puncture the capsule assembly, to measure the free volume and internal gas pressure and to collect samples for gas composition and isotopic analyses and total elemental composition.. This system provides internal void volume and gas pressure data to an accuracy within ± 5% in a pressure – volume range of 0.03 to 60 liter-atmosphere.

4.4 Assembly Dimensional Inspections

The diameter profile of a specimen can be measured using the Element Contact Profilometer (ECP), a remotely operated, continuous-contact profilometry gauge for measuring axial and spiral diameter profiles of cylindrical elements and capsules. This is capable of measurements to an accuracy of ±0.0003 in. (7.6 μm) and can measure percentage swelling over a range of 0.13 – 8.7%. The length of the rodlet can be measured with an accuracy of ±0.010 inches and the bow of the fuel rodlet can be measured to an accuracy of ±0.020 inches.

4.5 Gamma Scan

Experimental components can be gamma scanned over their entire accessible length to measure total activity and isotopic activity of select fission products and activation products in structural materials. Gamma spectra can be used to determine fuel pellet or fuel pin separations in the fuel column, fuel redistribution, fission product migration, and the relative axial burnup profile.

4.6 Optical and Electron Microscopy and Radiochemical Analyses

Optical and electron microscopy are used to analyze microscopic features such as fuel restructuring, pore density and size distribution, fuel-cladding chemical interaction, grain size and structure in the fuel and cladding, precipitates, and cladding integrity. The fuel burnup of a representative number of fuel samples can be determined by radiochemical analyses on samples supplied to the Analytical Laboratory.

Sample preparation is performed in an inert, argon atmosphere sub-cell of the HFEF Main Cell equipped with an independent atmosphere control system. The sample preparation sub-cell has facilities for sectioning, mounting, grinding, polishing and etching. The optical microscopy, scanning electron microscopy and radiochemistry samples are transferred by a pneumatic transfer system.

5. ATR NATIONAL SCIENTIFIC USER FACILITY

In 2007, the DOE designated the ATR as a National Scientific User Facility (NSUF). The mission of the ATR NSUF, to provide nuclear energy researchers access to world-class facilities, thereby facilitating the advancement of nuclear science and technology within the U.S, is accomplished by providing experimental irradiation testing and PIE facilities for the user community and technical assistance in designing and analyzing reactor experiments. With this designation, DOE has committed to maintain and enhance the research capabilities necessary to further fuel and material research objectives. Access to the ATR NSUF also includes access to the INL PIE facilities and science and engineering support for experiments. The user facility will:

- Support both basic and applied research and development
- Increase the effectiveness and decrease the uncertainty associated with development of new fuels and materials for existing and advanced reactor concepts
- Facilitate the development and validation of new analytical models to improve nuclear energy systems
- Encourage research collaboration on high-quality scientific experiments using the ATR and assure that unique research capabilities are made available to a broader scientific community
- Provide access to the ATR for production of medical, research, and industrial isotopes
- Provide a platform for educating and training nuclear scientists, radiochemists, engineers, and trades critical to the nuclear industry
- Support new nuclear programs at universities that do not have a dedicated research reactor, and augment those programs that do have research reactors.

The reactor and associated PIE facilities can be accessed for non-proprietary work by university-led teams, at no cost, through award of proposals selected on a competitive basis. The first proposal solicitation was offered in 2008 and four proposals were selected. It is anticipated that in the future, up to 15 experiments may be offered in a single year, depending on the complexity of the experiments proposed and available funding.

There are several projects underway and planned to enhance the research value of the INL facilities. These include addition of PIE equipment to the HFEF and Analytical Labs, reactivation of a PWR loop in the ATR, addition of the shuttle irradiation system, and building a new facility for test train assembly.

6. CONCLUSION

The ATR is a versatile research reactor, with several unique design features and offering several testing configurations that enable the reactor to support multiple diverse experimental programs. In 2007, the DOE designated the ATR as a National Scientific User Facility (NSUF), enabling the INL to support non-proprietary research at no cost to the user. Associated INL facilities, such as experiment assembly, use of experimental instrumentation, and PIE, can also be accessed through the NSUF proposal competition process. The ATR is expected to continue operations for many years into the 21st century.

ACKNOWLEDGEMENTS

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REFERENCES

[1] "FY-2008 Advanced Test Reactor National Scientific User Facility Users' Guide," INL/EXT-07-13577, Idaho National Laboratory (2007).

[2] S. B. Grover, R. V. Furstenau, "The Advanced Test Reactor Irradiation Facilities and Capabilities," First International Conference on Physics and Technology of Reactors and Applications. Marrakech (Morocco), March 14-16, 2007.

[3] C. J. Stanley, F. M. Marshall, M. C. Thelen, "Advanced Test Reactor National Scientific User Facility – Research Proposal Process Lessons Learned," Proceedings of ICAPP '08, Anaheim, CA, USA, June 8-12, 2008.

2.3 Status and future plan of Research Reactors of Scientific Centre of Russian Federation "Research Institute of Atomic Reactors"

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The State Scientific Centre of Russian Federation "Research Institute of Atomic Reactors" (RIAR) is a multi-functional nuclear research centre. At present, six research reactors are available at RIAR united in a single complex. At present, each RIAR reactor facility has a long-term plan of the experimental work within the period of the operation license validity. To analyze the world's experience and coordinate plans under international programs, it is reasonable to set up an effective collaboration with IAEA regarding the strategic plan for each RR. The RR Strategic Plan must be reviewed and approved by the Russian Government Atomic Energy Agency (Rosatom). When drawing up a plan, the terms of confidentiality should be met according to the International and Russian Legislation and requirements of company norms, commitments and commercial contracts.

Keywords: post-irradiation examinations, research reactors, experimental base, strategic plan, the research reactor coalitions.

1. EXECUTIVE SUMMARY

The State Scientific Centre of Russian Federation "Research Institute of Atomic Reactors" (RIAR) is a multi-functional nuclear research centre. At present, six research reactors are available at RIAR united in a single complex. There is also an experimental power facility with the VK-50 boiling reactor; the Russia's largest complex for post-irradiation examinations of full-scale components of nuclear reactors and irradiated materials; radiochemical complex for investigation and production of transuranic elements and various radioisotope products; equipment and facilities to perform research work in the field of fuel cycle and facilities for radwaste handling are available as well.

RIAR carries out research and experimental activities in the following directions:

- reactor testing and post-irradiation examination of materials and components of power and research reactors of different purposes aimed at the improvement of the existing NPP reactors and creation of innovative nuclear facilities;
- development and in-reactor testing of fuel, absorbing and structural materials of nuclear and fusion reactors;
- elaboration of radioisotope production technologies for the industrial and medical purposes and radioisotope production;
- investigations on the closed fuel cycle of nuclear reactors with the use of power- and weapon-grade plutonium and fractioning and transmutation of long-lived fission products.

The main scope of RIAR research and commercial activities is based on the experiments performed at the research reactors (SM-3, MIR-M1, BOR-60, RBT-6 and RBT-10/2). So, the most important task the RIAR management faces is the effective use of the unique experimental base providing its safe operation. This task cannot be solved without rational and realistic planning of the performance of each reactor facility for the near future (2-3 years) and for long-term period as well.

At present, each RIAR reactor facility has a long-term plan of the experimental work within the period of the operation license validity. However these plans need more detailed elaboration and feasibility study of their implementation, a thorough analysis being done regarding the experimental capacities of each reactor and its usage in Russia's and world's nuclear industry. The RIAR Administration considers it necessary to draw up a long-term plan (strategic plan, according to IAEA) for each research reactor (RR) using the IAEA Guidelines "Strategic planning for research reactors", IAEA-TECDOC-1212. The strategic plan for each RIAR RR should contain a long-term program of the reactor usage up to the date of its shutdown, including the decommissioning preparation program as well as the concept and approximate dates of the decommissioning work fulfillment. The strategic plan has to be based on the Russian programs for development of reactor emission-based nuclear, industrial and medical engineering; it should be open for wide international cooperation in the frame of INPRO, GNEP, Generation-IV, RERTR and other programs and involve the national programs of nuclear engineering development in different countries and commercial contracts with domestic and foreign companies. To analyze the world's experience and coordinate plans under international programs, it is reasonable to set up an effective collaboration with IAEA regarding the strategic plan for each RR.

The RIAR Administration supports the IAEA initiative to create the research reactor coalitions so as to optimize the effective usage of the existing RR regarding the development of nuclear engineering, creation and implementation of high-tech radioisotope medical and industrial products accumulated in RR, sharing the experience in RR safe operation, safe radwaste handling and diminishing of their volume, provision of nonproliferation and training of RR personnel. It seems expedient to establish both local and global coalitions to unite the experimental capacities of some specific reactors and to solve such urgent tasks as the creation of new materials for innovative nuclear power systems, RERTR, etc.

The RR Strategic Plan must be reviewed and approved by the Russian Government Atomic Energy Agency (Rosatom). When drawing up a plan, the terms of confidentiality should be met according to the International and Russian Legislation and requirements of company norms, commitments and commercial contracts.

2. INTRODUCTION

The economical changes going on in the Russian Federation related, first of all, to the passing to market-oriented economy, join-stock corporation and privatization of nuclear enterprises, including RIAR, stipulate a high necessity for RIAR RR to be more profitable and to improve the efficiency of their usage. The long-term strategic planning and feasibility study of the RR use are the instruments to solve the above-said task.

Besides, due to a number of objective factors (exhausting of organic energy resources, global heating and environmental pollution), Russia, as well as countries all over the world, undergoes a nuclear power renaissance. The nuclear power systems of the next generation, being implemented worldwide, must be more efficient and safe. All the RIAR research reactors refer to the high-power reactor category. The reactors are multi-purpose, i.e. they are used to carry out research activities, applied investigations, transmutation and examination of properties of different materials under irradiation and to accumulate radioisotopes. So, RIAR research reactors can take a certain niche and play a significant role in the creation of new nuclear power systems. Due to this, it is necessary to perform a thorough analysis of the RR experimental capacities and their further development and to draw up long-term plans for RR to participate in the Russian and International programs for the nuclear power engineering development.

On the whole, to perform a balanced long-term planning of the RIAR activities in other directions (post-irradiation examinations and reactor material science, fuel cycle, radiochemistry and radionuclide production) and develop its infrastructure, strategic planning for the RR experimental base is needed.

The RIAR research reactors, together with a large PIE complex, are the Russia's unique national technological base regarding their characteristics, importance, tasks and scale of application. So, the strategic plan for each RR is the necessary instrument for the Russian Atomic Energy Agency (Rosatom) to effect state regulations and planning of activities and to provide nuclear and radiation safety of research reactors up to their lifetime end.

RIAR is a peculiar research institute since its site locates a number of research reactors with a united infrastructure. To provide the RR operation the following systems have been designed: outer and inner autonomous power supply; distillate and process water supply; collection, reprocessing and disposal of liquid and solid radwaste; ventilation and purification of contaminated air; SNF storage. Also, there are facilities for centralized repair and maintenance of equipment, checking out and metrology of measuring devices, fabrication of non-standard equipment, radiation safety provision and environmental monitoring, security and physical protection, fire safety, fresh NM account and control, logistics and personnel training. RIAR has a unified system for planning and financing. So, the RR strategic plans have to be related to each other and to the strategic planning of the whole experimental base.

The strategic planning is necessary:

- to perform realistic assessment of the RR capabilities accounting the demands in its services and to plan its activities;
- to perform the feasibility study of the long-term RR usage effectiveness accounting the demands in renovations and increasing costs to provide the performance of systems and equipment in accordance with more and more strict safety

requirements and accumulate funds for decommissioning;
- to find ways of enlarging the experimental capabilities and effectiveness of RR usage;
- to advertise the RR capabilities and enlarge the cooperation with different scientific, commercial and educational organizations both inside and outside Russia;
- to create International coalitions of research reactors and nuclear centers on the basis of mutually complementary and beneficial service rendering, exchange of experience in safe RR operation and techniques to carry out experiments and train the personnel;
- to assess the RR final state and determine measures required to prepare it for decommissioning and disposal of nuclear and radwaste.

The RR strategic plan should be revised once a year and, if necessary, corrections should be introduced. Managers on RR operation and performance of experiments are reliable for the development of technical issues of the strategic plan.

3. BRIEF DESCRIPTION OF RIAR RESEARCH REACTORS

3.1 High-flux research reactor SM-3

The high-flux research reactor SM-3 has been under operation since 1961 and is intended to perform experiments on irradiation of reactor material samples under the set conditions, to study the regularities of the changes of different materials under irradiation and to accumulate transplutonium elements and radioactive nuclides of different elements.

The SM reactor has a unique channel design to achieve thermal neutron flux of high density in the moderating trap in the core centre with hard neutron spectrum. Before the reconstruction made in 1992, the reactor was called SM-2. After the reconstruction it became SM-3.

The main specifications of the SM-3 reactor are presented in Table 1.

3.2 Loop-type research reactor MIR.M1

The loop-type research reactor MIR has been operated since 1967. It is used, mainly, to test fuel of different types of nuclear reactors under conditions simulating both normal (stationary and transient) operating conditions and some design-basis accidents. The main technical and operating characteristics of the MIR reactor are presented in Table 2.

Table 1 Technical specifications of the SM-3 reactor

Characteristics	Value
Type of reactor	Pressurized water trap-type reactor on intermediate neutrons
Power (MW)	100
Maximal thermal neutron flux density in the central channel ($s^{-1}\cdot cm^{-2}$)	5×10^{15}
Time of operation at power (day)	230-240
Scheduled operating time	2020
Fuel	Uranium dioxide of 90% enrichment in U-235
Core arrangement	Square with a central trap
Core outer dimensions (mm)	420×420
Core height, mm	350
Number of cells for driver FA	32

Table 2 Technical specifications of the MIR.M1 reactor

Characteristics	Value
Maximal thermal power (MW)	100
Maximal thermal neutron flux density in the loop channel ($s^{-1}\cdot cm^{-2}$)	5×10^{14}
Time of operation at power (day)	230-240
Scheduled operating time	2020
Fuel	Uranium dioxide of 90% enrichment in U-235
Core height (m)	1
Number of cells for driver FA	48 - 58
Number of loop channels	11

3.3 Pool-type reactor RBT-6

The pool-type RBT-6 reactor with a thermal power of 6MW has been under operation since 1975. It is intended to examine material properties under long-term irradiation at a neutron flux density of 10^{13}-10^{14} $s^{-1}\cdot cm^{-2}$. The core is assembled of fuel assemblies spent in the SM reactor. The peculiarity of this reactor is that the number of core cells can be varied and the core arrangement can be easily changed.

Table 3 Technical specifications of the RBT-6 reactor

Characteristics	Value
Nominal thermal power	6

Characteristics	Value
(MW)	
Neutron flux density ($s^{-1}·cm^{-2}$)	$\leq 1\times10^{14}$
Time of operation at power per year (day)	250-320
Scheduled operating time	2025
Fuel	Uranium dioxide of 90% enrichment in U-235
Core height (m)	0.35
Number of cells for driver FA	56

3.4 Pool-type reactors RBT-10/1 and RBT-10/2

Research material science reactor RBT-10 was commissioned in 1982-1984 and consists of two thermal neutron reactors 10MW each. Fuel assemblies spent in the SM reactor are used as fuel. However, unlike RBT-6, these reactors have more cells in the core (10×10) to locate irradiation channels. The RBT-10/1 reactor has not been operated since the beginning of 90-s; fuel is unloaded and the Rosatom decision is approved to decommission this reactor. At present, it is being prepared for the decommissioning.

Table 4 Technical specifications of the RBT-10/2 reactor

Characteristics	Value
Nominal thermal power (MW)	10
Neutron flux density ($s^{-1}·cm^{-2}$)	$\leq 1\times10^{14}$
Scheduled operation time	2034
Fuel	Uranium dioxide of 90% enrichment in U-235
Core height (m)	0.35
Number of cells for driver FA	78

3.5 Fast experimental reactor BOR-60

The fist Russian fast experimental reactor BOR-60 was commissioned in 1969. It is mainly intended for experiments to try out fuel cycle and coolant technologies as well as a wide rage of designs of fast reactors with sodium coolant. Besides, the BOR-60 reactor, being a powerful fast neutron source, is used to examine the effect of neutron irradiation on different types of structural, fuel and absorbing materials. The main characteristics are presented in table 5.

Table 5 Technical specifications of the BOR-60 reactor

Characteristics	Value
Thermal power (MW)	Up to 60
Coolant	Sodium
Electric capacity (MW)	12
Thermal capacity (Gcal/h)	20
Maximal neutron flux density ($cm^{-2}s^{-1}$)	$3.7·10^{15}$
Time of operation at power per year (day)	220-230
Scheduled operation time	2015
Fuel	UO_2 or UO_2-PuO_2
Enrichment in ^{235}U (%)	45-90
Enrichment in ^{239}Pu (%)	Up to 70

4. Capabilities of RIAR research reactors

4.1 High-flux research reactor SM-3

4.1.1 Existing capabilities

The SM-3 reactor is mainly used as a high-flux neutron source to irradiate materials and accumulate radioisotopes. Experimental rigs can be located in the central trap (up to 27 cells), reflector cells (30 cells) and in FA of special purpose (6 FA with 4 cells in each). The experimental capabilities of the SM-3 reactor are presented in Table 6.

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4.1.2 Experimental capabilities of RR SM-3

There are loop facilities VP-1 and VP-3 intended for fuel testing, examination of fission product release from leaky fuel rods and ways of their removal from the primary circuit as well as for irradiation of structural and absorbing materials. The main characteristics of the loop facilities are presented in Table 7.

At present, the SM loop channels are used for high-dose irradiation of promising zirconium alloys, absorbing materials and fusion reactor materials as well as for testing of promising designs of fuel materials for high-flux research reactors.

The unique capabilities of the SM reactor made it a leading Russian reactor in the field of transuranic production and accumulation of radionuclides with high specific activity. All this is due to the following reactor advantages:

- large number of channels in the reflector with the thermal neutron flux density ranging from 1.5×10^{18} to 1.5×10^{19} $m^{-2}s^{-1}$;
- possibility to irradiate targets in the bulk of core fuel rods, where the portion of epithermal and fast neutrons is significant;

Table 6 Experimental capabilities of RR SM-3

Number of cells for irradiation, including:	Up to 37		
• trap	1 (block option 27 cells for targets Ø (12...25)mm; channel option – central channel Ø 50 mm + 18 cells for targets Ø12mm)		
• core	Up to 6 (FA 184.05.000 with 4 cells for targets each)		
• reflector	30 (of which 20 cells can be instrumented or supply by separately coolant).		
Neutron flux densities in irradiation rigs, ($s^{-1} \cdot cm^{-2}$)	Neutron flux		
	≤ 0,67eV	0.67-100eV	≥ 0.1MeV
• trap	1.3×10^{15}	1.7×10^{14}	1.4×10^{15}
• core	1.3×10^{14}	9.3×10^{13}	2.0×10^{15}
• reflector	$1.35\times10^{15}\div9\times10^{13}$	-	3.3×10^{14}-2.5×10^{12}

- availability of a trap in the core centre with unperturbed thermal neutron flux density up to 5×10^{19} $m^{-2}s^{-1}$;
- effective heat removal from targets under irradiation.

A wide range of neutron flux density change and its spectral characteristics make it possible to search and implement optimal schemes of accumulation of these or those radionuclides. A multi-stage accumulation of Cm and Cf-252 heavy isotopes is performed in the neutron trap cells and two reflector cells closest to the core. Besides, the following radionuclides with high specific activity are accumulated: Ni-63; Sn-113; Sn-119m; Fe-55; Fe-59; Cr-51 and other. Activation of Se-75 and Ir-192-based source blanks is performed as well. The core experimental channels are used to irradiated radionuclides, which are accumulated more effectively in the hard neutron spectrum. They are P-32, P-33; Gd-153 and Sn-117m. The cells closest to the core are used for large-scale accumulation of such radionuclides as Co-60 and Ir-192. The annual output achieves 300 kCi of Co-60 and 400 kCi of Ir-192.

Table 7 Technical specifications of the VP-1 and VP-3 loop facilities

Characteristics	VP-1	VP-3
Maximal operating pressure, (MPa)	5.0	18.5
Coolant temperature (°C)	90	300
Flow rate (m^3/h)	30	5-8
Thermal power	500	90
Coolant	water	water

Cell 20 located in the reflector has a special instrumented device providing a continuous control of neutron flux during irradiation and it is intended for precise activation of Co-60-based source blanks of medical purpose. About 1000 sources are produced annually. The diagram below presents the specific activity of some radionuclides achieved in the SM reactor.

Radionuclide	Specific activity (Ci/g)
^{119m}Sn	1
^{133}Ba	5
^{188}W	5
^{63}Ni	14
^{65}Zn	40
^{124}Sb	40
^{55}Fe	70
^{45}Ca	60
^{113}Sn	100
^{153}Gd	100
^{59}Fe	150
^{60}Co	400
^{192}Ir	700
^{169}Yb	1000
^{170}Tm	1000
^{75}Se	1300
^{51}Cr	3500

4.1.3 Potential capabilities

High-dose irradiation of structural and fuel materials for innovative nuclear reactor of Gen IV (gas-cooled high-temperature reactors, fast boiling reactors and supercritical water reactors) can be performed as well as for promising designs of research reactors. Table 8 presents the characteristics of promising irradiation rigs.

Facilities can be designed to improve the effectiveness and increase the volume of radioisotope accumulation as well as to enlarge the nomenclature of radioisotopes (^{131}Ba, ^{177}Lu, ^{99}Mo).

New fuel compositions with low-enriched uranium can be examined for their application in high-flux reactors.

The reactor mock-up (physical model) can be used to train students, post-graduates and reactor personnel.

Table 8 Characteristics of promising irradiation rigs

Irradiation rig	Medium	Testing parameters			
		φ_{fn} (E>0.1MeV) ($cm^{-2}s^{-1}$)	φ_{tn} ($cm^{-2}s^{-1}$)	K (dpa/h)	Kt (dpa/year)
Loop rig in the reflector	Water (300°C, 18.5 MPa)	10^{13}-4×10^{14}	2×10^{13} - 4×10^{14}	3×10^{-5}- -$1.2\cdot10^{-3}$	0.15-6.0
Loop rig in the core	Water (300°C, 18.5 MPa)	1.5×10^{15}	2×10^{14}	$\leq3\times10^{-3}$	15-18
Ampoule rig in the reflector	Boiling water (up to 320°C), supercritical water, helium (400-1150°C)	5×10^{12} - 4×10^{14}	2×10^{13}- - 4×10^{14}	1×10^{-5}- -1.2×10^{-3}	0.1-6.0
Ampoule rig in the core	Boiling water (up to 320°C), supercritical water, helium (400-1150°C)	(1.5-2) $\times10^{15}$	(2-3) $\times10^{15}$	$\leq4\times10^{-3}$	16-22

Table 9 Technical specifications of the loop facilities

Nоп	**Characteristics**	**Loop facilities**						
		PV-1	**PVK-1**	**PV-2**	**PVK-2**	**PVP-1**	**PVP-2**	**PG**
1.	Coolant	water	water, boiling water	water	water, boiling water	water, boiling water, steam	water, boiling water, steam	nitrogen, helium
2.	Number of experimental channels	2	2	2	2	1	1	1
3.	Maximal channel heat power (kW)	1500	1500	1500	1500	100	2000	160
4.	Maximal coolant temperature (°C)	350	350	350	365	500	550	500
5.	Maximal coolant temperature outlet of irradiation device (°C)	350	350	350	365	1000	1200	1300
6.	Maximal pressure (MPa)	17.0	17.0	18.0	18.0	8.5	15.0	20.0
7.	Maximal coolant flow rate through the channel (m^3/h)	16.0	16.0	13.0	13.0	0.6	10.0	

4.2 Loop-type fuel test reactor MIR-M1

4.2.1 Existing capabilities

MIR M1 is a specialized reactor to perform loop testing of fuel for reactors with water, boiling water and overheated steam, gas, metal coolant under stationary and transient conditions and under design-basis accidents. The reactor facility comprises a reactor, two hot cells, cooling pools and mock-up (physical model).

RR fuel is usually tested in the driver fuel channels. Physical characteristics of the reactor and large number (up to 41pcs.) control rods to change reactivity allow simultaneous testing at different thermal neutron flux density differing in 50 times. High thermal neutron flux density (~(5-7)$\times10^{14}$ $cm^{-2}s^{-1}$) provides additional irradiation of standard or experimental WWER and PWR fuel rods with a burnup of ~70MW·d/kgU and higher. Table 9 presents the main characteristics of the loop facilities.

Several types of irradiation rigs were developed to test fuel of water-cooled reactors:

- dismountable rigs to test shortened mock-ups (≤250mm) of "Garland-type" fuel rods; up to 4 rigs can be installed in one channel;
- dismountable irradiation rigs to test fuel rods with an active part of ~1000mm; up to 19 fuel rods can be tested;

- rigs to test non-instrumented re-fabricated (≤1000mm) and full-size (≤3500mm) fuel rods from spent NPP fuel assemblies;
- dismountable rigs to test instrumented and non-instrumented re-fabricated (≤1000 mm) and full-size (≤3500mm) fuel rods;
- dismountable rigs to test fuel rods under the power ramping conditions using rotating or movable screens;
- instrumented multi-component devices to perform testing under simulated LOCA;
- instrumented devices to perform testing under simulated RIA;
- irradiation rigs and experimental devices of loop facilities to test the leaky fuel rod behavior.

To control the main parameters of the fuel rod performance, transducers and devices were developed to provide on-line measurements during the experiment. The main specifications of the in-pile control transducers are presented in Table 10.

Current experimental activities performed in the MIR M-1 reactor:

1. Examinations of advanced VVER-1000 fuel with the increased grain size, new pellet geometry and decreased cladding thickness.
2. Testing of VVER-1000 fuel rods with a burnup of ≥60MW·d/kgU under the power ramping and cycling conditions.
3. Comparative testing of VVER fuel rods with new cladding materials.
4. Examination of fission product release from re-fabricated VVER –1000 fuel rods with artificial defects at a high burnup of ≥ 50MW·d/kgU.
5. Comparative testing of fuel rods of different design for floating NPPs.
6. Testing of VVER fuel with a high burnup of ≥ 60MW·d/kgU under simulated RIA.
7. Testing of VVER fuel with a high burnup of ≥ 60MW·d/kgU under simulated LOCA.
8. Testing and examinations in justification of the fuel serviceability under normal operation and transient and design-basis accident conditions performed for advanced reactors in the frame of the AES (NPP)-2006 Program.
9. Testing of low-enriched fuel under the RERTR Program.
10. Accumulation of industrial-purpose isotopes (^{192}Ir).

4.2.2 Potential capabilities

Upgrading of the gas-cooled PG-1 loop is planned to increase the channel outlet temperature up to 1100°C and perform in-reactor fuel testing for promising high-temperature gas-cooled reactors. The steam-water PVP-2 loop is also planned for upgrading to increase the pressure up to 22.5Mpa and provide testing conditions for fuel and materials of super-critical water-cooled reactors.

Experimental rigs can be designed to simulate both severe accidents with fuel damage and beyond the design-basis accidents.

Loop testing can be performed for fuel with innovative types of fuel elements, new cladding materials (ceramic, metal ceramics), new fuel compositions (UO_2 with absorber and moderator additives), etc.

The scope and nomenclature of accumulated isotope can be widened (^{14}C, ^{60}Co, etc.). The neutron radiography facility can be developed. The reactor mock-up (physical model) can be used to train students, post-graduates and reactor personnel.

Table 10 In-pile transducers specifications

Parameter	Design type	Measurement range	Error	Dimensions, mm	
				Diameter	Length
Temperature of coolant (Tc) and cladding (Tcl)	Chromel-Alumel thermocouple	Up to 1100°C	0.75%	0.5	
Fuel temperature (Tf)	Chromel-Alumel thermoprobe	Up to 1100°C	0.75%	1-1.5	
	W-Re thermoprobe	Up to 2300°C	~ 1.5%	1.2-2	
Cladding elongation (δL)	LVDT	(0-5)mm	± 30μm	16	80
Diameter change (δD)	LVDT	(0-200)μm	± 2μm	16	80
Under-cladding gas pressure (Pf)	Bellows + LVDT	(0-20)MPa	~ 1.5 %	16	80
Neutron-flux density (F)	Rh-, V-, Hf –neutron detectors	(1015-1019) $m^{-2}s^{-1}$	~ 1%	2-4	50-100
Volumetric steam content (β)	Cable-type conductivity transducer	20-100%	10%	1.5	

4.3 Pool-type reactors RBT-6 and RBT-10/2

4.3.1 Existing capabilities

The reactor is licensed to operate till 2009 and work is being done to justify its lifetime prolongation till 2024. The reactor is operated at an annual usage factor of ~ (0.7-0.9). The reactor is mainly used to examine changes of material properties during long-term irradiation at a neutron flux density of $10^{13} \div 10^{14}$ $cm^{-2} \cdot s^{-1}$. RBT-6 is located in the same building as SM and these reactors have common heat and water supply systems, common secondary circuit and are maintained by the SM personnel.

The experiments are performed in eight vertical channels of the core, three channels of the reflector and in the KORPUS facility intended for testing of the VVER, PWR and BWR reactors vessel steel under a wide range of vessel operation conditions simulated by neutron density and power spectrum, irradiation temperature, gradients of these parameters and modes of parameter changes under operation. The reactor is also used to test structural material for the international fusion reactor ITER. There are channels for nuclear doping of silicon ingots up to 120mm in diameter. There are ampoule channels to accumulate radioisotopes at a neutron flux density of (10^{13}-10^{14}) $cm^{-2} \cdot s^{-1}$. The reactor is also equipped with devices for neutron-activation analysis and neutron radiography.

The RBT-10/2 reactor is under operation since 1982 and it is licensed to operate till 2010. Its operating lifetime is scheduled till 2034. The reactor is used to examine changes of fuel and structural material properties under irradiation at a neutron flux density of 10^{13}-$10^{14}$$cm^{-2}$ s^{-1}, accumulation of radionuclides, nuclear silicon doping and radiation coloring of minerals. There are 27 channels to irradiate materials up to 60mm in diameter. Also channels are available for nuclear doping of silicon ingots 120mm in diameter.

At present, the reactor is used to accumulate ^{131}I and to perform nuclear doping of silicon and radiation coloring of minerals.

4.3.2 Potential capabilities

The RBT-type reactors being operated at full power most of time, the experimental channels are not loaded enough (≤50%). The main task is to increase the usage of reactor experimental capabilities. First o all, it is necessary to increase the scope and nomenclature of radionuclide production (^{131}Cs, ^{14}C, ^{99}Mo, ^{60}Co, etc.).

Channels can be designed for nuclear doping of silicon ingots 160mm and 200mm in diameter. Radiation technologies can be developed on transmutation and change of physical and chemical properties of industrial-purpose materials.

The liquid fuel loop facilities can be developed for ^{99}Mo, ^{89}Sr, etc. production.

The RBT-6 devices for neutron-activation analysis and neutron radiography can be modernized to improve significantly their characteristics.

A channel for neutron-capturing therapy can be designed at the RBT-10/2 reactor.

4.4 Fast experimental reactor BOR-60

4.4.1 Existing capabilities

The BOR-60 reactor is licensed to operate till 2010. At present, work is being performed to examine the state of reactor systems and equipment and to justify the prolongation of the reactor operation till 2015.

The following activities are performed at the BOR-60 reactor:

Reactor material testing:

- Reactor testing of structural materials, in particular, accelerated tests of:
- new cladding materials for fast reactor fuel pins
- steels for water-cooled reactor internals;
- zirconium alloys for VVER cores;
- vanadium-based alloys in the lithium medium for fusion reactors;
- graphite for the RBMK reactor;
- insulation, magnetic and refractory materials for fusion reactors.

Fuel cycle:

- Tests of fuel rods and FA up to burnup more than 30%h.a. under steady-state and transient conditions for the statistics verification of serviceability of fuel rods with vibropacked fuel;
- Demonstration of weapon-grade plutonium burning;
- Tests of fuel rod mock-ups with nitride fuel for promising fast reactors;
- Tests of fuel rod mock-ups with different fuel compositions for promising fast reactors;
- Examination of transmutation and burning of long-lived radionuclides from SNF from different reactors;
- Try out of closed fuel cycle with nitride fuel;
- Tests of fuel for the BREST-type reactor with lead coolant in the ampoule loops to solve the same problems as for the BN-800 reactor (enhanced safety, non-proliferation, decrease of the environment burden, etc.) However, as

compared to BN-800, the absence of sodium, large pitch of the spacer grid and lead substrate in the fuel rods improve significantly the facility safety and a two-circuit design increases its efficiency.

Radionuclide production:

- Accumulation of some commercial radionuclides generated by the threshold reactions: ^{32}P, ^{33}P, ^{35}S, ^{89}Sr (reaction (n,p)) and ^{117m}Sn (reaction (n,n'));
- Examination of parameters Th-229 accumulation by reaction $^{230}Th(n,2n)^{229}Th$ for alpha-radioimmuno therapy;
- Irradiation of samples and targets for the commercial accumulation of different radioisotopes ^{63}Ni, ^{89}Sr, ^{153}Gd.

Sodium technology:

- Improvement of equipment decontamination systems that are in contact with sodium and elimination of nondraining sodium from the equipment under repair and decommissioning using new technologies;
- Examination of oxide cold trap regeneration methods to operate them without replacement;
- Experimental justification of deep purification of sodium coolant from cesium radionuclides using compact traps of different designs.

Testing of process equipment:

- Lifetime tests and examination of operating modes of “sodium-water” steam generators;
- Lifetime tests of electromagnetic pumps;
- Lifetime tests of experimental samples of different BN reactor equipment.

4.4.2 Potential capabilities

Reactor material science:

- Testing of structural materials for Gen-IV fast reactors in specialized capsules and ampoule loops and in different media (He, Pb, Pb-Bi) to select cladding materials;

Fuel cycle:

- Justification of serviceability of fuel rods with nitride fuel under stationary and transient conditions;
- Testing of new types of fuel compositions for fast reactors of next generation;
- Examination of fuel rod behavior under accident conditions using specialized ampoule loops (excess of the safe operation limits regarding temperature, sodium boiling, FP release in case of fuel rod leakage, etc.);
- Designing and reactor testing of ultrasonoscopy and reactor diagnostics systems;

Safety issues:

- Examination of accidental processes to justify NPP BN-reactor safety;
- Experiments to simulate accidental and abnormal situations to master the means and methods of diagnostics (ultrasonoscopy under the sodium layer, calculation of the reactivity balance, vibro-acoustic and nose diagnostics).

5. STAKEHOLDERS

5.1 Government

At present, RIAR research reactors are owned by the State and managed by the Russia's Federal Atomic Energy Agency (Rosatom). A decision has been taken to establish a State Rosatom join-stock corporation, in which RIAR will integrate.

Rosatom effects a financial support of the work aimed at the enhancement of the operation safety, supply of fuel and RR upgrading. The Managing Body sets up a long-term program of experimental activities and controls the allocation of funds, observance of Federal Norms and Rules in the field of nuclear power usage, fulfillment of International Non-proliferation Treaties and account, control and storage of nuclear materials as well.

5.2 Top Management

Since the RIAR experimental base is located on one site and has a single infrastructure, resources needed for the RR operation are controlled by the top management. The RIAR Administration solves all the problems to satisfy operating needs, including fuel, materials, power supply and engaging and training the personnel.

5.3 Educational and scientific institutions

RIAR renders free of charge services in the practical training of university and college students at the RR facilities regarding RR operation and performance of experiments. Students from faculties of two Dimitrovgrad Technical Universities have practical training at the RIAR RR facilities. Besides, any Russia’s Technical University preparing specialists in the field of nuclear engineering can take advantage of these services.

Together with other research centers RIAR performs joint investigations in the field of fundamental science. Some of them are in the frame of non-commercial agreements, but the majority of such investigations are performed on the commercial basis.

5.4 Commercial customers

RIAR RR facilities and departments that perform reactor tests and calculations, develop and fabricate gages, devices, measurement systems, etc., render commercial-based services both inside RIAR (to radioisotope production department, material science department and fuel cycle department) and to Russian and foreign entities.

The main commercial customers are enterprises of Rosatom (research and design institutes that develop fuel and nuclear facility components and NPP owners and operators) as well as some industrial and scientific entities from other industrial sectors.

RIAR RRs are open for collaboration and render services to any foreign organization in the field of reactor radiation usage and training of personnel in the frame of international agreements and Russia's commitments in the peaceful usage of nuclear energy.

At present, the income coming from commercial customers is the most important for RIAR RRs, so the technical policy and reactor schedules are mainly determined by the commercial orders.

5.5 Regulatory Body

The Rostehnadzor Department for Control of Nuclear and Radiation Safety (Gosatomnadzor) issues licenses and sets conditions of their validity, including requirements on safe operation of RRs and RIAR as a whole.

The regional Gosatomnadzor Department is located at the RIAR site. It performs regular inspections of the RR technical state and checks the license validity and observation of norms and rules. The Department considers and approves all the changes in the design and operating documents influencing the RR safety.

Gosatomnadzor agrees the list of RR personnel that must be licensed to operate the reactor, controls the personnel training and examines the theoretical and practical knowledge of the operating personnel.

One of the conditions to issue a license for the RR operation is the availability of the RR decommissioning concept.

5.6 Personnel

At present, all RIAR RRs vacancies are filled with high-skilled licensed personnel. RIAR has its own training center to train personnel for research reactors and other nuclear- and radiation-hazardous work. Job descriptions and training programs have been elaborated for each position. The personnel knowledge is checked up regularly.

Each reactor facility has its own personnel training point equipped with manuals, layouts and guidelines. The most urgent task for the personnel training points is to upgrade the aids, computerize the training process and use analytical simulators. The personnel of the RIAR support departments deals with power and process water supply to the research reactors. This personnel is also trained and the key specialists are licensed.

The average salary of the RR personnel is higher than the average salary in RIAR and in the region but it is lower that the NPP operator salary. To increase the motivation, the salary should be raised, employment benefits should be provides and the working conditions should be constantly improved. For this purpose, the efficiency of the RR usage must be enhanced.

5.7 Population

The attitude of the population of the city and region towards RIAR and RR operation is loyal, on the whole. There is a PR Group that informs the population about all important events that take place at RIAR as well as about incidents during the RR operation. This Group organizes tours to RR facilities for pupils and students. During these tours brief information is given about the RR design and purpose and tasks of its usage.

Some ecological public association have appeared in the city recently that perform monitoring of the RIAR system for radioactive substance handling and make a public expertise of the radiation environment data. The Department for Radiation Safety and Protection of RIAR Environment provides the public associations with information. It is reasonable to establish a specialized communication center equipped with the RR mock-ups and computer-based interactive systems that simulate the nuclear reaction processes and main types of the experimental research.

5.8 IAEA

Contacts with IAEA on the issues of purchasing or providing technical data are rather periodical than regular. The IAEA Guidelines on the operation, maintenance, safety, strategic planning, etc. are used as manuals at the RIAR RRs. The RR specialists take part in the meetings, workshops and conferences on the issues of usage, safety, handling of fuel and radioactive substances that are organized and/or sponsored by IAEA.

A regular cooperation with IAEA on the RR usage issues should be organized. For this purpose it is reasonable to conclude an agreement or Technical Project for Collaboration with IAEA and to appoint a person at each RR to be responsible for regular contacts with IAEA.

5.9 Attraction of new stakeholders

To make the advertisement of the RR capabilities more effective we have to:

- make an informative, access-free and useful Internet-site;
- reprint and distribute booklets about RIAR RR facilities;
- perform technical conferences and workshops on the RR usage at the RIAR site;
- take an active part in the conferences on nuclear physics and engineering, nuclear power engineering, radiation medical and industrial technologies;
- attend exhibitions and scientific actions of the state-of-the-art engineering;
- visit the existing customers;
- develop actively the international cooperation and participate in the international projects organized, first of all, by IAEA.

To make the RR facilities more competent, we have to:

- improve the quality of the performed experimental work and rendered service (timeliness, correspondence to specifications, reliability);
- decrease the net cost due to increase of RR personnel labor productivity;
- use fuel effectively, implement up-to-date energy- and material-saving technologies;
- create an up-to-date radwaste handling system to decrease totally the radwaste amount;
- create an effective economically profitable management system.

Search of new directions of the RR usage and enlargement of its experimental capabilities should be supported by:

- effective analysis of modern scientific and technical data so as to find promising ways of the RR usage (development of promising materials for nuclear power engineering, creation of new types of nuclear reactors, usage of reactor emission for material properties transmutations, etc.);
- data acquisition, study of experience in usage and experimental activities at other RRs of the same type;
- enlargement of contacts with the leading technical universities and scientific centers, conclusion of agreements for the scientific and technical collaboration and attraction of students and specialists to carry out research.

6. CONCLUSION

Information presented above shows that the RIAR RRs relate to a high-power test reactor class and, being a single complex, have great capabilities to perform applied investigations in nuclear power engineering, radioisotope productions, transmutation and examination of properties of various materials under irradiation. It should be emphasized that during the last 15-20 years that were marked by the economic changes in Russia and a significant decrease of the investigations in the field of nuclear power engineering both in Russia and all over the world, RIAR RRs kept their serviceability and experimental capabilities. The BOR-60 operating lifetime can be extended at least till 2015, and that of SM-3 and MIR.M1 – up to 2020.

Thank to the nuclear renaissance Russia and all countries are going through, RIAR research reactors face the following ambitious tasks:

- to provide high level of serviceability and experimental capabilities and confirm its status of the main experimental stage of Russia's nuclear engineering;
- to take a leading positioning the current and promising experimental programs of the Russia's nuclear engineering development;
- to establish a wide international cooperation under INPRO, GNEP, Generation-IV, RERTR and other programs as well as in the frame of national programs of nuclear engineering development in other countries;
- to raise the RR usage efficiency due to the increasing volume and nomenclature of accumulated radioisotopes and industrial and medical purpose to satisfy completely the increasing demand of the Russia's market and foreign commercial customers.

2.4 The Best Research Reactor, therefore we call it BR2 at Mol

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A high-pressure water loop (CALLISTO) providing PWR reactor operating conditions, reusable irradiation device for materials (MISTRAL), pressurized water capsule for power transient irradiations of LWR fuels, more than 50 radioisotopes for medical and industrial applications etc. , were introduced as the status and future plan of the BR2.

Keywords: BR2, CALLISTO, MISTRAL, PWC/CCD, POSEIDON

1. INTRODUCTION

Located at the Belgian Nuclear Research Centre, SCK•CEN, BR2 is one of the World principal high-flux Materials Testing Reactors providing a multipurpose capability for many nuclear and non-nuclear scientific research and commercial applications.

Acknowledged to be a key European research tool and by participating in European, national and international R&D programmes, BR2 contributes to the evolution of science and technology, particularly in the field of nuclear safety. Whilst within a national context, it exists primarily as research tool in support of the Belgian commercial PWR's.

2. BR2

The design of the BR2 are particularly well adapted to these R&D options:

- A core with a central vertical 200 mm diameter channel, with all its other channels inclined to form a hyperboloidal arrangement around it. This geometry combines compactness leading to high fission power density, with easy access at the top and bottom covers, allowing complex irradiation devices to be inserted and withdrawn;
- A large number of experimental positions of 84 mm with in addition 4 peripheral 200 mm channels for large irradiation devices. Through loop experiments can be installed through penetrations in the top and bottom covers of the vessel;
- A remarkable flexibility of utilization: the reactor core configuration and operation mode are adapted to experimental requirements;
- Irradiation conditions representative of those of various power reactor types - neutron spectrum tailoring;
- High neutron fluxes, both thermal and fast (up to 10^{15} n/cm².s).

Photograph of reactor pool, 3D image of tank, reactor vessel and example of core arrangement for BR2 are shown in Fig. 1 - 4.

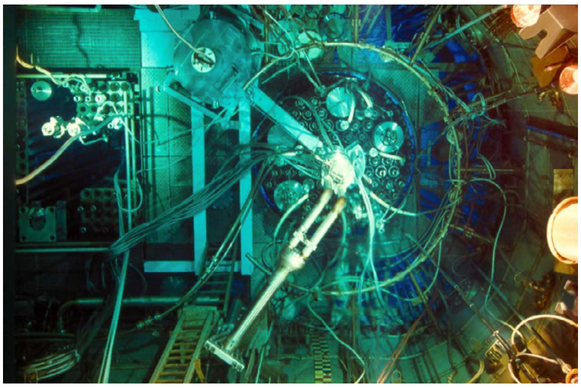


Fig. 1 Photograph of reactor pool.

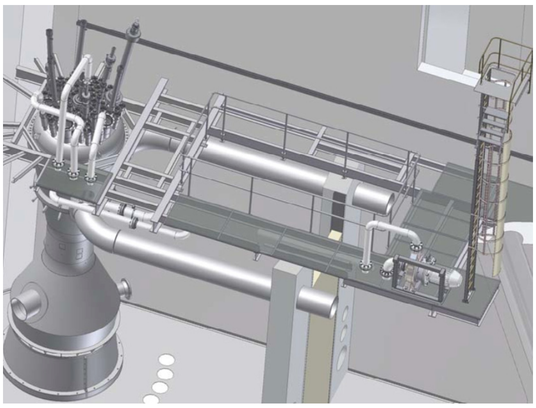


Fig. 2 3D image of BR2 tank.

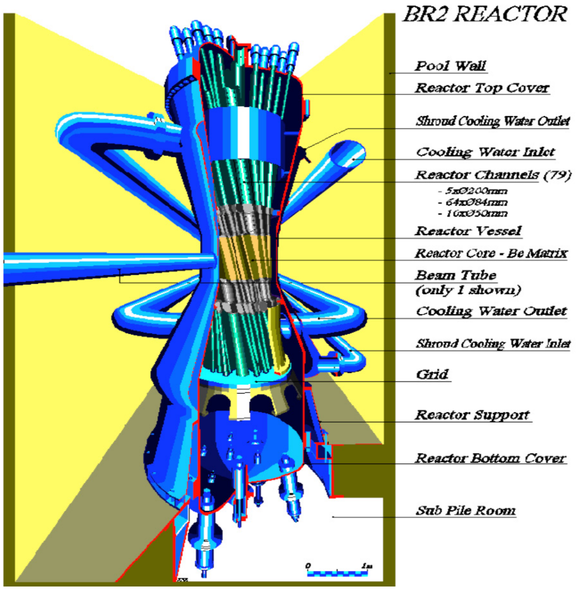


Fig. 3 Reactor vessel of BR2.

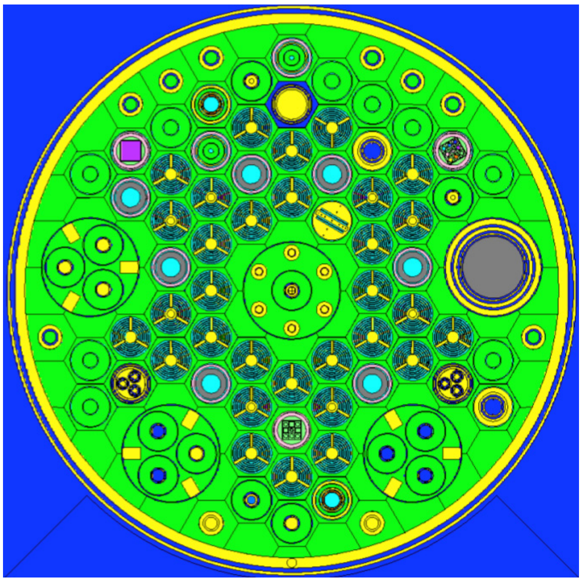


Fig. 4 Example of core arrangement of BR2.

3. IRRADIATION SERVICE IN BR2

BR2 team in collaboration with various SCK•CEN expert groups are able to provide a fully comprehensive irradiation service from planning the experiment to post irradiation inspection and interpretation of the final results.

As a high-flux reactor, BR2 is able to produce life-time irradiation effects in fissile and reactor structural materials within escalated time-scales to accurately predict their mechanical behaviour and the resultant changes in their nuclear chemistry. Among the main irradiation devices currently used for these purposes, we operate:

- CALLISTO: The CALLISTO facility is a high-pressure water loop that provides predictive model validation and qualification testing under realistic PWR reactor operating conditions.
- MISTRAL: MISTRAL is a reusable irradiation device for research on reactor materials exposed to a high fast neutron flux at temperatures below 350˚C.
- PWC/CCD: The Pressurised Water Capsule (PWC) devices are used for power transient irradiations of fresh or pre-irradiated LWR fuels.
- Several dedicated devices are used for in-situ corrosion as well as on-line mechanical testing of material specimens under neutron flux.
- POSEIDON : POol Side Equipment for Irradiation and Doping Of silicon by Neutrons.

CALLISTO IPS-fuel irradiation system, PWC capsule & VNS screen system and POSEIDON system are shown in Fig. 5 – Fig. 7.

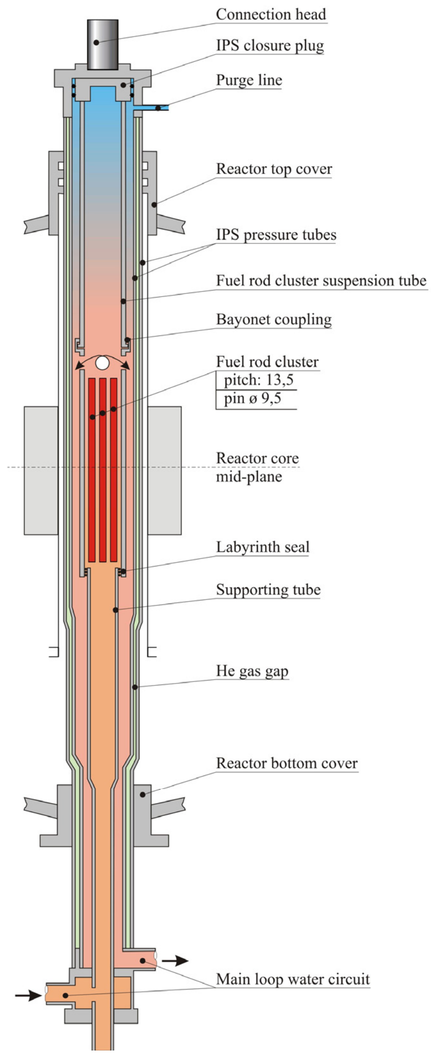


Fig. 5 CALLISTO IPS-fuel irradiation system.

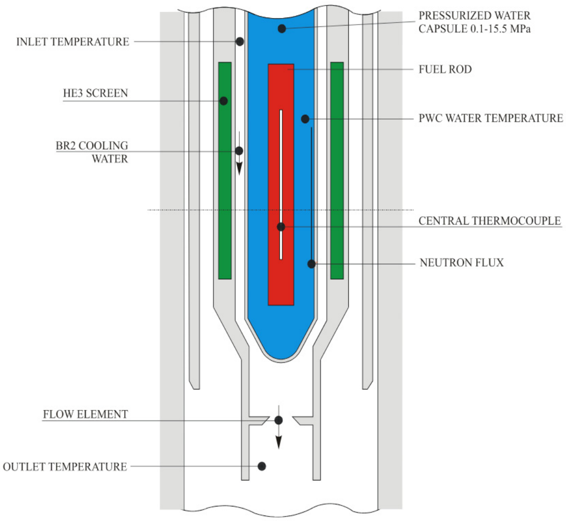
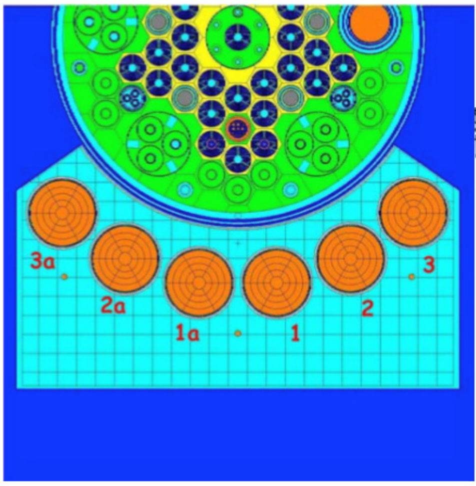


Fig. 6 PWC capsule & VNS screen system.

BR2's high thermal neutron flux (10^{15}n/cm^2.s peak) is ideally suited for the routine production of many important radioisotopes that require a high specific activity. More than 50 radioisotopes are produced in BR2 for medical and industrial applications.

Refurbished to provide a technical life expectancy beyond 2020, it is foreseen that the reactor will continue to play a central role within Europe for producing many of the World essential radiopharmaceuticals as well as carrying out fundamental and industrial nuclear materials research.



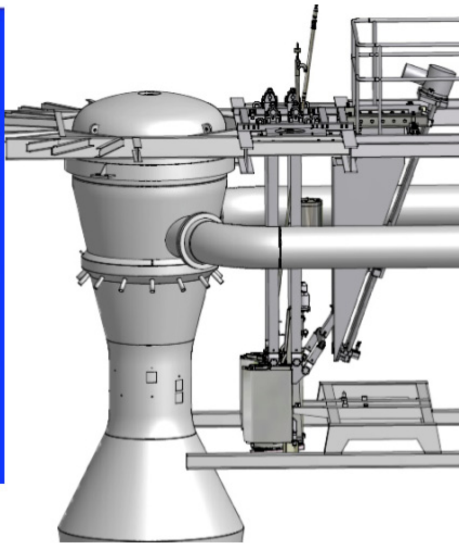


Fig. 7 POSEIDON system.

BR2's high thermal neutron flux (10^{15}n/cm^2.s peak)

2.5 STATUS AND FUTURE PLANS ON RESEARCH REACTORS AND IRRADIATION TECHNOLOGY IN ARGENTINA

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The National Atomic Energy Commission of Argentina (CNEA) was founded on May 31st, 1950. Since January 17th, 1958 when our first research reactor RA-1 achieved for first time criticality, CNEA built and operated several research reactors (RA-0, RA-1, RA-2, RA-3, RA-4, RA-6 and RA-8). Recent features related to them are the conversion of the RA-6 reactor core from HEU to LEU, the conversion from HEU to LEU of RA-3's target irradiation for fission radioisotope production and the qualification program of high and very high fuels through irradiations in RA-3 core. Collaboration between CNEA and NNSA-DoE enhanced the RA-6 core conversion. The RA-3 reactor core was converted from HEU to LEU in 1989 and upgraded from 5 to 10MW in 2001, to enhance the radioisotope production and the fuel assembly testing program for high and very high density LEU fuels. Due to final restrictions on HEU supply in 2001, CNEA turned from HEU into LEU its target fabrication, satisfying most stringent pharmacopeia quality requirements. CNEA is involved with R&D and fabrication of MTR fuels since more than 40 years, providing domestic research reactors and overseas clients. A comprehensive qualification plan for U_3Si_2 compound up to 4.8 g/cc based on US-NRC NUREG-1313 technology specifications and criteria and through CNEA's technical capabilities (basic research laboratories, irradiation in the core of RA-3 reactor and hot cells for PIE analysis) successfully finished in 2003.

Keywords: research reactor, core conversion, power upgrade, target conversion, MTR qualification program

1. INTRODUCTION

This work is related to CNEA's research reactor improvements. Those are the RA-6 reactor core conversion from HEU to LEU; the conversion of target to be irradiated in the RA-3 reactor core for fission radioisotope production and the qualification program for high and very high LEU fuels.

2. RA-6 CORE CONVERSION

The RA-6 reactor is a pool-type one, sited in the Bariloche Atomic Center. It is dedicated to Human Resources training in Nuclear Engineering, BNCT on melanoma and organ treatment and Neutron Activation Analysis for environmental study and characterization.

On October 30th, 2005 two contracts between CNEA and NNSA-DoE were signed to enhance the core conversion from HEU to LEU, including the option to take part of the Acceptance Program FRR SNF to return to USA the SNF fabricated by CNEA using US origin HEU. Almost all tasks were already accomplished [1]:

- July, 2006: Swapping of fresh HEU (Arg)-LEU (USA) materials
- July, 2007: Fabrication of new LEU-silicide core (28 SFA + 5 CFA) for RA-6 reactor and reflectors, construction of interim storage facilities to store LL and MLW to contain the wastes generated during the HEU-SNF conditioning.
- October, 2007: Removal of RA-6 spent HEU core and conditioning of spent fuels
- November, 2007: Transport cask loading and transportation to port and delivery to USA

It remains to finish the primary and secondary loop piping refurbishing, expecting to restart the RA-6 reactor during the last trimester of present year.

3. FISSION RADIOISOTOPE PRODUCTION IN RA-3 REACTOR CORE:

The RA-3 reactor was built in 1967 to irradiate targets for fission radioisotope production, to test new fuel materials and assemblies and to perform several nuclear techniques including training of human resources in Reactor Engineering. It used HEU fuels at 5 MW thermal power achieving 1.0E+14 n/cm^2/s thermal flux. It was converted from HEU to LEU-uranium oxide based core in 1989 and was upgraded from 5 to 10MW thermal power in 2001, to enhance the radioisotope production and the fuel assembly testing program for high and very high density uranium compound.

The fission radioisotope production by target irradiation in the RA-3 core covers about 90% of RI used in Argentina for Nuclear Medicine applications.

During 2001 Argentina had a final restriction on the supply of HEU material for the production of fission Mo99, so CNEA decided to turn its target fabrication from HEU into LEU material.

This change was achieved maintaining the chemical route for the production (alkaline chemical process) in order to make the smallest impact in RI provision. An adequate replacement meat for targets was developed, tested and since 2002 routinely used under irradiation up to date.

Our LEU technology satisfies the most stringent requirements of quality for its use in nuclear medicine applications [2].

4. QUALIFICATION PROGRAM

CNEA has defined a program to qualify the technology and the associated facilities for the production of fuel elements for research reactors, using high density uranium compound.

CNEA has been involved with the development and fabrication of MTR fuels since more than 40 years, crossing from the dispersion fuel based on U_3O_8 to the more recently high density uranium silicide compound. Since that time more than 800 fuel elements of different kind have been supplied to domestic and overseas clients.

A comprehensive qualification plan for U_3Si_2 compound with up to 4.8 g/cc was successful finished. These capabilities included several laboratories for basic research, the upgrading of the RA-3 reactor and hot cells preparation for the post-irradiation analysis.

Here is presented relevant information related to the fabrication, irradiation history and Post-irradiation Examination for three full-scale fuel prototypes (the precursor P-04, P-06 and P-07 which has similar plate geometry to RRR Opal reactor designed fuel assembly)

The qualification process involved the technology expressed in US-NRC NUREG-1313 which provides the necessary technical basis to establish the use of U_3Si_2 dispersion in Al with U densities of up to 4.8 g/cm^3 for research reactor fuel assemblies. The manufacturing of the FE and their component parts was done under strict quality control program.

The irradiation of each FA was done in the RA3 reactor, with forced downward coolant and at a thermal power of 10 MW, finishing in 2003.

During irradiation, visual inspections of each FE were done approximately every 10 % of burn up in the pool bay of the reactor by a periscope. The gap between fuel plates and possible presence of blisters, distortions or other phenomena affecting it was visually evaluated with the aid of back lighting. Also sipping test was performed monitoring the activity of the water passing through the plates to verify the integrity of then.

The progressive irradiation of prototypes reached 24% BU (atU%) for P04, 55% for P06, and finally 67% for P07. Once each irradiation was completed, and after adequate cooling time, the prototypes were transferred to the CELCA hot cells facilities for post-irradiation inspection. Post-irradiation Examination involved a complete sequence of non destructive and destructive measurements: visual inspection and metrology of each FE, removal of plates, plate thickness measurement, density determination by immersion methods, gamma scanning profile for relative BU, blister test, metallographic examination by OM and SEM and radiochemical analysis for absolute burn up determination.

These test allowed the complete evaluation of each prototype, the integrity of fuel plates, oxide layer formation, meat and fuel particle swelling and the interaction reaction between fuel compound and Al matrix. The final evaluation was that CNEA fulfilled the qualification grade for low enriched U3Si2 based fuel assembly manufacturer.

Nowadays a program to test and qualify very high density fuel is running by means of irradiations in the RA-3 and foreign reactors, particularly directed to U-Mo dispersed fuels in Al alloy cladding and U-Mo monolithic fuel in Zry-4 cladding.

5. CONCLUSIONS

CNEA is deploying an intense activity in reactors enhancement including fuel cycle length extension, LEU technology for fission irradiation production to get the best use of them according to best international practices.

REFERENCES

[1] S. Balart, O. Calzetta, P. Cristini, J. Garcés, A.G. Gonzalez, J.D. Hermida, M. Mirandou, O. Novara, H. Taboada, RERTR meeting Progress on RERTR Activities in Argentina (2007)

[2] D. Cestau, P. Cristini, A. Novello, M. Bronca, R. Centurión, R. Bavaro, J. Cestau, E. Carranza, RERTR meeting HEU and LEU Comparison in the Production of Molybdenum-99.(2007)

2.6 WWR-K Research Reactor – Status and Future Plans

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Nuclear reactor WWR-K is a light water tank-type research reactor commissioned in 1967. Reactor site is located on the territory of the Institute of Nuclear Physics of the National Nuclear Center of Kazakhstan, 20 km far from the city of Almaty, the largest city in Kazakhstan. The reactor operated at 10 MW thermal power for 20 years until its temporary shutdown in 1988 for safety upgrades. After re-commissioning in 1998, it operates at 6 MW power by cycles of up to 30 days duration. The reactor has several dozens of irradiation channels with thermal neutron flux 10^{12} to 10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$ and is equipped with hydraulic and pneumatic rabbit systems, neutron radiography facility, gas/vacuum loop facility, and a critical assembly which allows modeling of the reactor core, precise measurement of neutronic characteristics of in-core devices, and verification of neutronic calculations. Five concrete hot cells and four iron hot cells in the reactor building are used for handling of irradiated material. Current activities at the reactor are concentrated in production of radioisotopes for medicine and industry, irradiation tests of materials and components, and neutron activation analysis. Post-irradiation examinations use various methods, depending on specific tasks. A recent example is a high burnup irradiation test of ceramic tritium breeder material for fusion reactor which was conducted jointly with JAEA. Future plans are related to expansion of these activities, in particular for testing of prospective nuclear fuel for advanced reactor systems, and conversion of the reactor to low enriched uranium fuel.

1. HISTORY

Nuclear reactor WWR-K is a light water tank-type research reactor commissioned in 1967. Reactor site is located on the territory of the Institute of Nuclear Physics of the National Nuclear Center of Kazakhstan, nearby the Alatau village, 20 km far from the Almaty City, on the submontane plain in the bottom of Zailiyskiy Alatau Mountains, 780 m above the sea level.

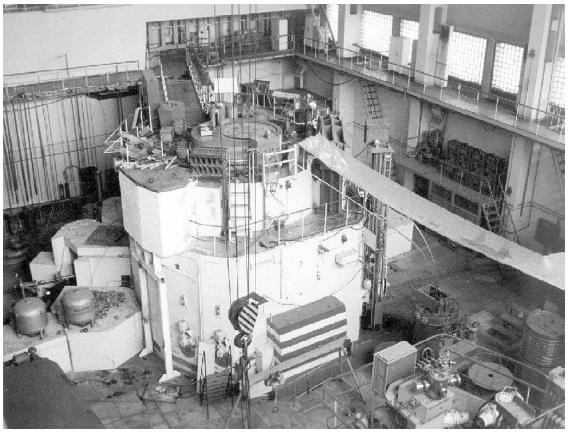


Fig. 1. Research reactor WWR-K.

The reactor operated at 10 MW thermal power for 20 years until its temporary shutdown in 1988, which was a result of regulatory authorities' decision to strengthen the safety measures for reactors operating in high seismic conditions. The shutdown fell at the same time as disintegration of the Soviet Union and fundamental economic reforms in newly independent states, so the most of safety improvement work has been done already in the Republic of Kazakhstan. This work was strongly supported by the International Atomic Energy Agency (IAEA) and the International Science and Technology Center (ISTC). The project #K-012 "Studies of Safety Issues of WWR-K Reactor Facility under Enhanced Seismic Activity and in View of its Future Use", funded by Japan, was the first ISTC project implemented in Kazakhstan. Safety measures undertaken in 1994-1997 years included detailed study of tectonic conditions at the reactor site, evaluation of seismic stability and reinforcement of vulnerable structures and equipment, installation of new emergency cooling system, duplication of power supply and control systems, etc. Power rating was reduced to 6 MW, but a new configuration of the core allowed saving neutron fluxes in central channels. The reactor was restarted in 1998, and its operation time increased gradually during 1998-2003. In 2004-2008, it operates 50-160 days per year by 5-30-day cycles.

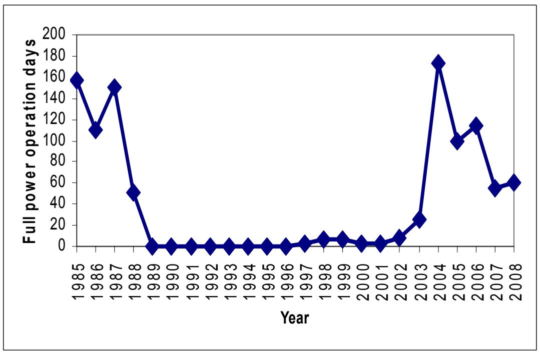


Fig. 2. WWR-K reactor operation history.

2. TECHNICAL FEATURES OF THE REACTOR

Reactor tank located inside concrete biological shielding has 5.7 m height and 2.3 m diameter. The core with fuel assemblies, the channels for control rods and ionizing chambers, vertical and horizontal irradiation channels, transport channels, niche of thermal column are mounted in the tank. Primary circuit, drainage and auxiliary pipes are welded into the bottom of the tank. All internals are made of aluminum alloy. Core height is 60 cm, diameter is 72 cm. Cooling water temperature: inlet 30-45°C, outlet 35-50°C.

WWR-K reactor uses hexagonal tubular fuel of two types: five-tube and three-tube assemblies. A central tube in the three-tube fuel assembly is assigned for location of control rod, but can be used also for placing irradiation ampoule (3.2 cm diameter, 60 cm height) into it.

Transport channels are used for unloading spent fuel from the reactor to storage pool, and for delivery of irradiated ampoules to the hot cell.

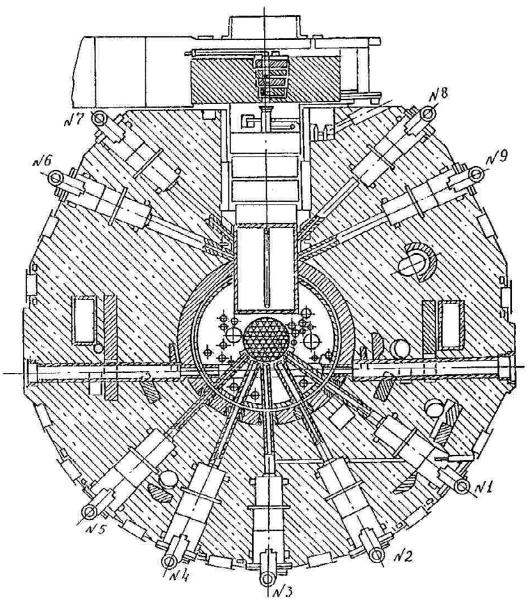


Fig. 3. Horizontal section of the reactor.

A number of irradiation channels in the reactor is several dozens. Irradiation channels with thermal neutron flux density higher than 10^{12} $n \cdot cm^{-2} \cdot s^{-1}$ are listed in the table below:

	Diameter, mm	Thermal neutron flux, $n \cdot cm^{-2} \cdot s^{-1}$
Vertical channels		
Core	60	$(0.3 \div 1.2) \cdot 10^{14}$
Reflector	70 ÷ 200	$10^{12} \div 10^{13}$
Horizontal channels		
Radial	100	$2 \cdot 10^{13}$
Tangential	193	$6 \cdot 10^{12}$
Thermal column cavity	1150	$1 \cdot 10^{13}$

Two pneumatic transfer systems are installed in the reactor: one in radial channel for usual gamma activation analysis, and one in vertical channel connected to delayed neutron measurement facility.

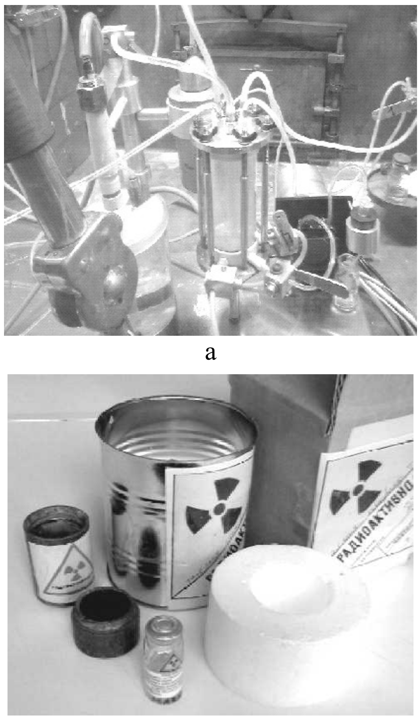
A neutron radiography facility is mounted at one of radial beam tubes. There is a gas/vacuum loop facility, which is used mainly for high-temperature irradiation tests, and a critical assembly which allows modeling of the reactor core, precise measurement of neutronic characteristics of in-core devices, and verification of neutronic calculations.

Five concrete hot cells and four iron hot cells in the reactor building are used for handling of irradiated material.

3. CURRENT ACTIVITIES

3.1 Radioisotope production

Development of isotope production technology at WWR-K was started after reactor re-commissioning in 1998. To date, three isotopes are produced routinely: ^{99m}Tc, ^{131}I and ^{192}Ir. For Technetium-99m, so called “gel technology” is used. The column of central generator is charged with particles of polymolybdate gel produced from irradiated molybdenum oxide (natural enrichment, 3-4 days of irradiation in central channel). The generator is eluted daily, and pertechnetate obtained is supplied to local clinics in form of ready injection solution., The total activity of generator is 60-80 GBq, and radioactive concentration of technetium on the second day of elution is 2-2.5 GBq/ml.



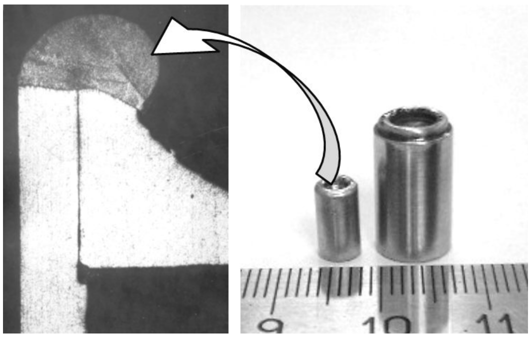
a

b

Fig 4. (a) Central $^{99}Mo/^{99m}Tc$ generator; (b) radiopharmaceutical package.

Iodine-131 is produced by traditional dry distillation from irradiated tellurium oxide, with iodine absorption in alkaline solution. Activity of bulk solution depends on irradiation time and amount of target material, defined by ordered quantity. Usual radioactive concentration of ^{131}I radiopharmaceutical solution supplied to local clinics is 20-100 MBq/ml.

Iridium-192, with specific activity of 200-300 Ci/g, is used for manufacturing of sealed sources for industrial radiography. Other types of sealed sources produced at WWR-K for local industry are cobalt-60 (low and medium activity), antimony-124 and thallium-204.



a

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Fig. 5. (a) Sealed sources produced at WWR-K.
(b) Ampoule for cobalt irradiation.

3.2 Neutron activation analysis

Traditional NAA with small specimens (0.1-1.0 g) is used at WWR-K for a long time. Recent development is analysis of large specimens (10-100 g) which simplifies sample preparation and provides more precise data for wider range of microelements in both geological and environmental samples, but requires special more accurate calibration.

R&D for transmutation doping of Silicon

Several irradiation positions were evaluated in respect to neutron transmutation doping of silicon ingots with ± 5% homogeneity. The results are:

- Thermal column cavity, in spite of very large diameter, was rejected because of sharp non-uniformity of neutron flux in both vertical and horizontal directions, and its rapid drop in radial direction.
- Tangential channel may be used for irradiation of 6-inch ingots, with continuous moving of ingot along the channel and axial rotation. 10 ingots, 0.5 m long each, can be uniformly irradiated up to $5 \cdot 10^{17}$ thermal n/cm^2 in about 15 days. Commercial realization requires designing and installation of fairly complex equipment (loading/unloading, precise movement, continuous flux monitoring) and may be feasible in case of long-term order.
- Vertical channels (diameter 100 mm and 200 mm) in water reflector are suitable for irradiation of 3-inch and 6-inch silicon ingots, respectively. Irradiation equipment is much simpler than for tangential channel. Acceptable radial homogeneity is provided by rotation of ingot. Axial (vertical) flattening of neutron flax is reached by installation of a profiled neutron absorber around the irradiated ingot. Estimated irradiation time is about 2 days for 40 cm high silicon ingot.

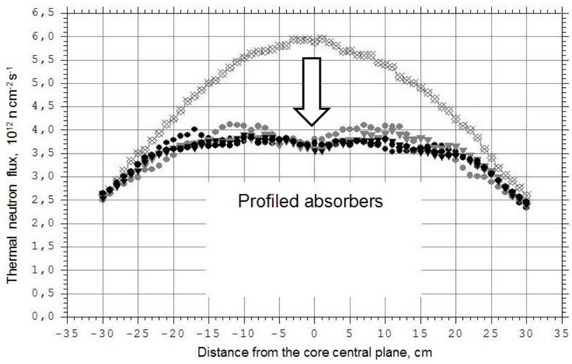


Fig. 6. Flattening of axial neutron flux distribution
by profiled absorbers.

3.3 R&D for BNCT

Experiments on formation of neutron beam suitable for boron neutron capture therapy (BNCT) were carried out at one of radial horizontal channels. In order to reduce a gamma dose rate and a share of fast neutrons in the beam, three filters were installed: lead blocks in core periphery, lead screen in the region of biological shielding, and beryllium oxide filter after lead screen. With these filters, satisfactory fluxes of thermal ($1 \cdot 10^9$ cm^{-2} s^{-1}) and epithermal ($5 \cdot 10^8$ cm^{-2} s^{-1}) neutrons was reached ~1 meter deep in concrete shielding, that makes a consideration of thermal column cavity as possible irradiation position for BNCT more practical than radial channel.

3.4 Material testing

Irradiation testing of various materials and components is a traditional application of WWR-K reactor for many years. A list of materials studied includes mainly pure metals, alloys and stainless steels. At present, the experiment on high burnup irradiation test of lithium ceramics as a candidate material for fusion reactor tritium breeder is carried out jointly with the JAEA.

4. Post-irradiation examinations

The techniques used in post-irradiation examinations are listed below:

- Mechanical tests
- Transmission electron microscopy
- Scanning electron microscopy
- Light microscopy
- X-ray diffraction
- Electrical resistivity measurements
- Gas thermodesorption spectroscopy

For high-burnup samples, mass-spectrometry and gamma-spectrometry measurements are also used.

Specimen preparation techniques in use are cutting, grinding, mechanical polishing, chemical and electrochemical polishing and etching, ion etching.

Hot radioactive samples are treated in the hot cells equipped with cutting and grinding equipment.

A list of mechanical testing techniques include tensile tests, compression tests, microhardness measurements. Relatively new technique is Shear Punch test with miniature specimens (usually 3 mm disks, the same as for TEM), which can be safely applied for very high activity samples, and a specimen can be easily prepared from any kind of irradiated component.

Preliminary calibration experiments with samples in different thermomechanically evolved conditions give a correlation between uniaxial tensile yield stress $\sigma_{0.2}$ and a critical shear yield stress τ for the specific material, which then allows estimating $\sigma_{0.2}$ value from τ measured on hot irradiated samples.

Post-irradiation thermal treatment is normally used to study kinetics of material structure and properties evolution. Transmission electron microscopy (TEM) is the basic technique used at INP for microstructural examinations of irradiated materials.

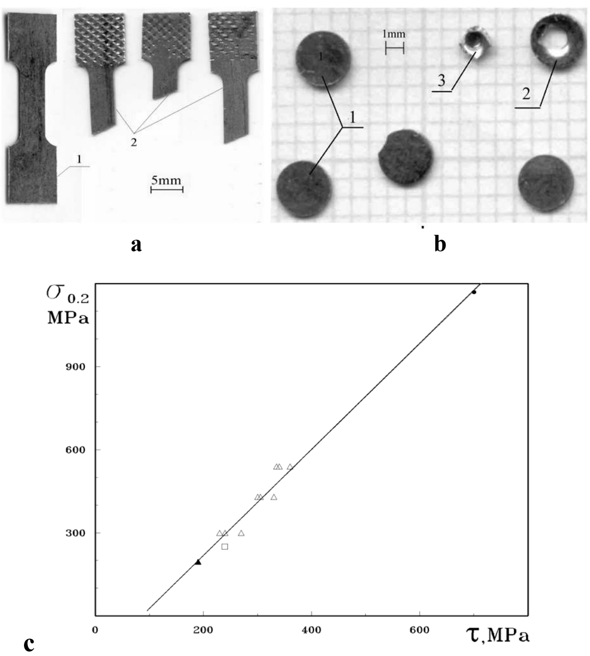


Fig. 7. (a,b) The shape of specimens before (1) and after (2,3) mechanical tests:
(a) uniaxial tensile, (b) Shear Punch;
(c) Tensile/Shear Punch correlation dependence.

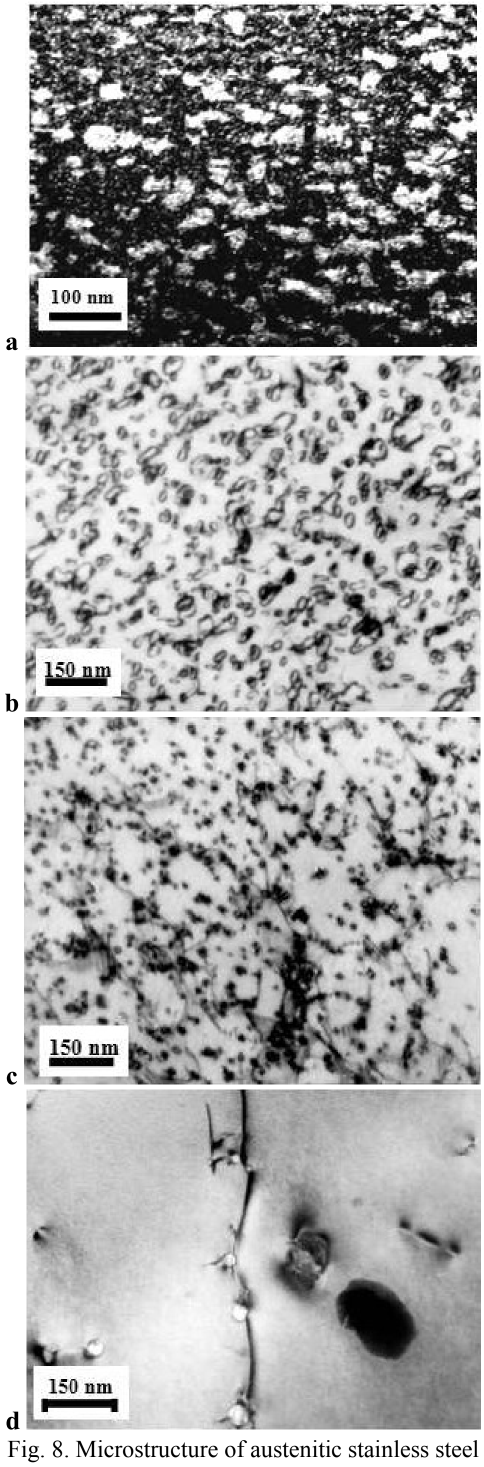


Fig. 8. Microstructure of austenitic stainless steel Cr18Ni9Ti irradiated in WWR-K reactor up to 1.3 x 10^{22} n/cm^2, E>0.1 MeV (transmission electron microscopy): (a) As-irradiated, Frank loops and defect clusters; (b) 1-hour annealing at 650^{o}C, complete dislocation loops; (c) 1-hour annealing at 950^{o}C, secondary phase precipitates; (d) 1-hour annealing at 1050^{o}C, voids and precipitates.

To study peculiarities of martensitic $\gamma \rightarrow \alpha'$ transformations in irradiated steel under deformation, mechanical testing can be accompanied by measurements of content of ferromagnetic α'-phase induced by deformation, with miniature magnetic probe.

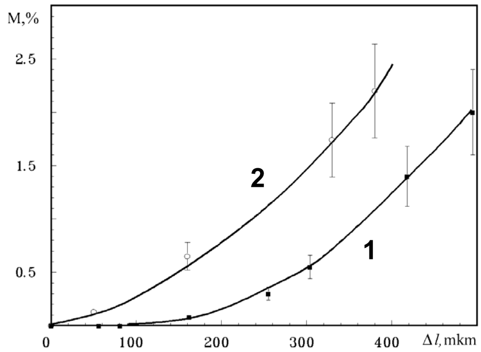
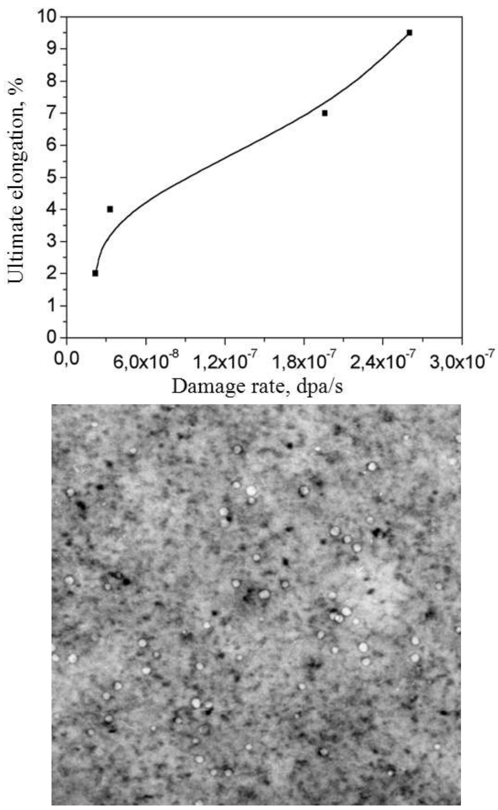


Fig. 9. Content of ferromagnetic α'-phase versus deformation (depth of punch penetration, *Δl*) in non-irradiated (1) and irradiated (2) Cr18Ni9Ti stainless steel.

Study of the samples irradiated under different conditions allowed, in particular, to reveal a dose rate effect on degradation of material structure and mechanical properties.



a

b

Fig. 10. (a) Dose rate effect on embrittlement of Cr16Ni11Mo3 stainless steel;
(b) voids in Cr16Ni11Mo3 stainless steel irradiated to ~1.5 dpa at 2 x 10^{-8} dpa/s (no voids were observed after irradiation at 2 x 10^{-7} dpa/s).

5. HIGH BURNUP IRRADIATION TEST OF CERAMIC TRITIUM BREEDER FOR FUSION REACTOR

The experiment on high burnup irradiation and post-irradiation examinations of lithium ceramics is funded by Japan through ISTS project #K-578. The objectives are:

- To examine tritium release dynamics from the samples under long-term neutron irradiation at different temperatures;
- To obtain the samples of Li_2TiO_3 ceramics (96% ^6Li enrichment) with 20% or more lithium burn-up;
- To perform post-irradiation examinations of high-burnup ceramic samples.

Outline of project activities:

❑ Preparation of needed experimental setup for ceramics irradiation at WWR-K reactor

- Neutronics studies,
- Thermal studies,
- Development of tritium measurement technique,
- Design and manufacturing of irradiation ampoules,
- Preparation of WWR-K reactor and loop facility for test,
- Composing of test program for WWR-K reactor;

❑ Irradiation experiment

- Provision of reactor and ULF operability,
- Control and adjustment of test parameters,
- *In situ* registration of Tritium release,
- Processing of current data;

❑ Post-irradiation examinations after completion of irradiation program

- Unloading of irradiation ampoules from the reactor,
- Dismantling of irradiation ampoules,
- Checking the condition of the samples,
- Measurement of Lithium isotope ratio,
- Measurement of residual Tritium content in the samples,
- Study the radiation-induced changes in composition, structure, density, mechanical and thermal properties of the samples.

Main irradiation parameters:

- Irradiation time - 5350 hrs;
- Reactor power – 6 MW;
- Two irradiation channels in the reactor core were used;
- Total amount of lithium ceramics samples irradiated (pebbles and pellets) – 16.2 gram;
- The samples were irradiated in 3 “active” ampoules (temperature control + *in situ* tritium measurement) and 3 “passive” ampoules (temperature control only);
- Lithium atoms burn-up reached about 22%;
- Total amount of tritium generated ~2000 Ci;
- Irradiation temperature - 400°C…900°C, heat generation in ceramics under irradiation - 100…130 W/g.

Irradiation ampoules were installed in two vertical channels in the reactor core, as shown at fig. 11.

Each irradiation ampoule contained a flat stainless steel capsule with the pebble or pellet samples, attached to cylindrical heat screen. Sample temperature was regulated by changing helium gas pressure (5 Pa……1000 Pa) in the gap between heat screen and ampoule wall. Helium pressure was controlled by helium loop facility. Arrangement of the samples inside irradiation ampoule is shown at fig. 12.

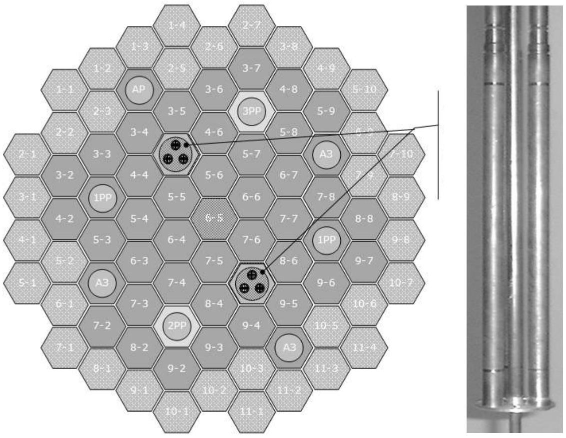


Fig. 11. Location of ampoules with lithium ceramics in WWR-K reactor core.

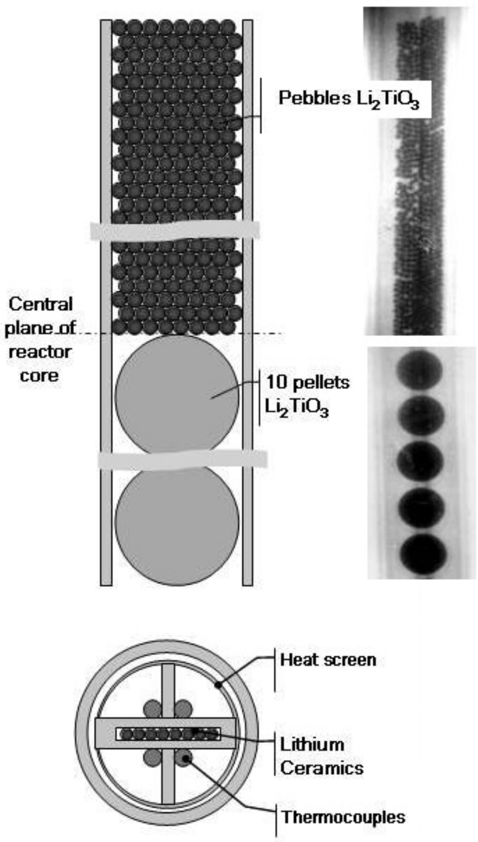


Fig. 12. Arrangement of the samples inside irradiation ampoule (drawing and neutron radiography image).

Inlet helium gas was purified in cryogenic absorber. Tritium released from active ampoules was measured by three mass-spectrometers with Pd-Ag filters, and the residual tritium is removed from outlet gas by catalytic oxidation and cryosorption. Temperature was measured by thermocouples installed in each irradiation ampoule, and neutron flux was monitored by self-powered neutron detectors installed in both irradiation channels and in water reflector. Gas pressure, temperature and neutron flux data are recorded continuously by computed data acquisition system.

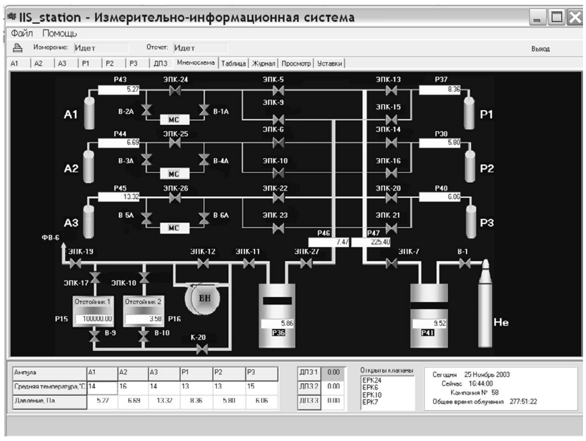


Fig. 13. Screenshot of Data Acquisition System.

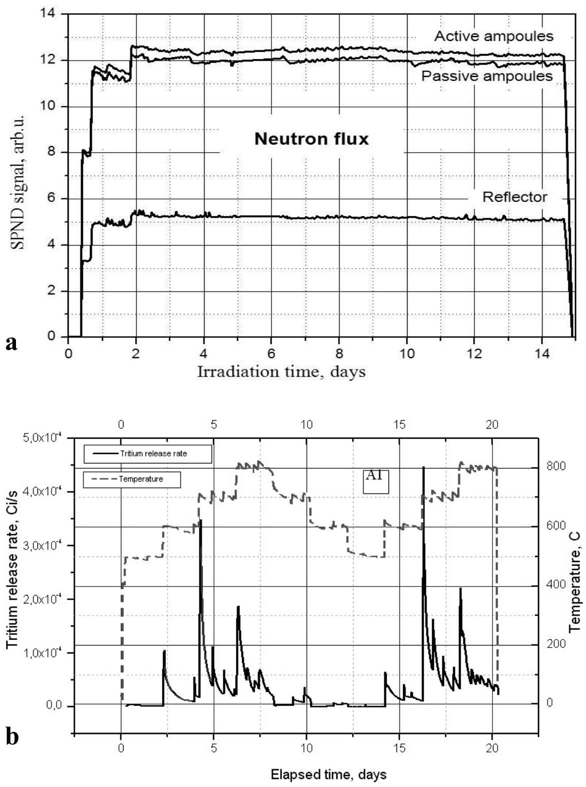


Fig. 14.
(a) Neutron flux variation during 14-day operation cycle;
(b) Variation of temperature in “active” ampoule and tritium release during 20-day operation cycle.

6. POST-IRRADIATION EXAMINATIONS OF LITHIUM CERAMICS SAMPLES

The shape of irradiated Li_2TiO_3 ceramics samples remained unchanged, while their color has changed from white to dark-gray.

No density changes could be revealed due to irregular shape of the pebbles and a presence of cavities in some of them. The measured dimensions were 1.0±0.1 mm and density ~1.75 g/cm^3 ± 20% in both irradiated and unirradiated samples. However, X-ray diffraction showed significant change in phase composition of the samples: the new phases $LiTi_2O_4$, $LiTiO_2$ and $Li_4Ti_5O_{12}$ were observed in irradiated samples, along with basic Li_2TiO_3 and TiO_2 presented in the initial samples.

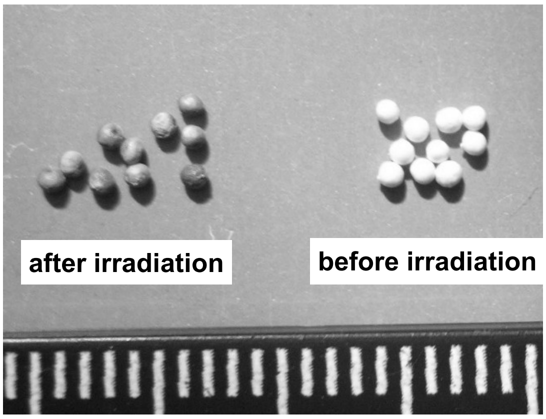


Fig. 15. Photograph of lithium ceramics pebbles before and after irradiation.

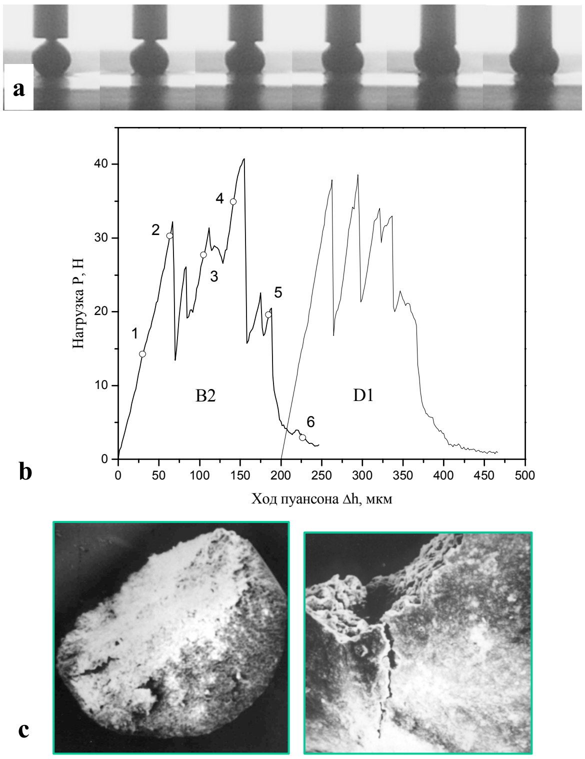


Fig. 16. (a) Stages of 1-mm pebble crush test; (b) pebble deformation diagram; (c) scanning electron microscopy image of fractured sample.

Mechanical tests showed some reduction of crushing strength of the pebbles due to irradiation: brittle fracture started at 40-50 MPa in unirradiated samples and at 20-47 MPa in irradiated samples. Also, a significant difference in microhardness of different crystallographic phases was revealed.

Residual Tritium content was found to be (6.6 ± 0.6)x10^{11} Bq/kg in the samples irradiated at 490^{o}C, and (1.7 ± 0.3)x10^{11} Bq/kg in the samples irradiated at 660^{o}C.

Post-irradiation examinations of high burnup lithium ceramics samples are continuing in 2008.

7. WWR-K REACTOR CONVERSION TO LOW-ENRICHED URANIUM FUEL

Regular fuel of WWR-K reactor contains 36%-enriched uranium, which exceeds the internationally accepted limit of 20% for non-weapon usable low-enriched uranium (LEU). In 2003, the program was started to convert the reactor to LEU fuel.

Neutronic and thermal-hydraulic calculations showed that replacement of 5-tube fuel assemblies with low density fuel meat by 8-tube fuel assemblies with higher density fuel meat, accompanied by installation of beryllium reflector around the core will allow to retain high irradiation characteristics and safety of the reactor. Moreover, neutron flux can be increased up to 2x10^{14} n cm^{-2}s^{-1} in central irradiation positions and about 1x10^{14} n cm^{-2}s^{-1} in core periphery.

Conceptual design of new fuel assembly has been developed. Further plan includes manufacturing of 3 lead test assemblies which should be irradiated in the reactor up to 60% burnup of uranium-235. If the results of irradiation and post-irradiation examinations will confirm reliability of new fuel, the reactor core will be re-arranged to accommodate the beryllium reflector and more control rods. The conversion to LEU may be fully completed in 2013-2014.

2.7 Status and Perspective of Material Irradiation Tests in the HANARO

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HANARO (High flux Advanced Neutron Application ReactOr) is a multipurpose research reactor of an open-tank-in-pool type. It was designed to provide peak thermal and fast flux of 5×10^{14} n/cm^2·sec (E<0.625eV) and 2.1×10^{14} n/cm^2·sec (E>1.0MeV) at a 30MW thermal power, respectively. Since the commencement of HANARO operation in 1995, some parts of the reactor systems have been gradually improved for a stable operation of the reactor, while the operation mode has been adjusted to meet users' demands. During the same period, a significant number of experimental facilities have been developed and installed for the use of the 32 vertical holes and the 7 horizontal beam ports. Owing to a stable operation of the reactor and a rapid proliferation in the utilization fields, more experimental facilities are continuously being added to satisfy various fields of study and new research needs. As a nation-wide neutron research facility, the HANARO is now successfully utilized in various fields including nuclear fuel and material irradiation tests. The equipments for the irradiation tests of the nuclear fuels and materials in the HANARO are classified into a capsule and an FTL (Fuel Test Loop). Capsules for irradiation tests of nuclear fuels and materials in the HANARO have been developed. Also, extensive efforts have been made to establish the design/manufacturing and irradiation technologies for irradiating nuclear fuels and materials by using these capsules and their control systems, which should be compatible with HANARO's characteristics. Other devices consisting of a fixing of the capsule during an irradiation test in the HANARO, a cutting and a transporting of the capsule main body after an irradiation test were also developed. These capsules and others have been actively utilized for the various material irradiation tests requested by users. Based on the accumulated experiences and the user's sophisticated requirements, capsules for a creep test and a fatigue test of materials during an irradiation in the HANARO have been developed. And, the irradiation plans related to developing the Gen-IV reactor systems by using capsules in the HANARO will provide more emphasis on the development of capsules by focusing on the irradiation tests of materials or nuclear fuels for the Gen-IV reactor systems, such as the SFR and the VHTR. The FTL is one of the irradiation devices, which can conduct an irradiation test of a nuclear fuel in the HANARO under the operating conditions of commercial nuclear power plants. The 3-test fuel rods can be irradiated in the HANARO by using the FTL. The installation of the FTL was completed in March 2007. Currently, the commissioning test of the FTL is being performed. And, the FTL at first will be used for the irradiation test of an advanced nuclear fuel for a PWR from the end of this year.

Keywords: HANARO, Research Reactor, Material, nuclear Fuel, Irradiation Test, Capsule, FTL (Fuel Test Loop)

1. INTRODUCTION

The HANARO has the maximum thermal power of 30MW. HANARO (High flux Advanced Neutron Application ReactOr) is a multipurpose research reactor of an open-tank-in-pool type. Its general design feature is given in Table 1. Detailed information is available at the IAEA's RRDB or HANARO home page (http://hanaro.kaeri.re.kr).

HANARO has been operated and the functions of its systems have been improved continuously [1] since its first criticality in February 1995, and it is now successfully utilized in such areas as neutron beam research, fuel and materials irradiation tests, radioisotopes production, neutron activation analysis, and neutron transmutation doping, etc. A significant number of experimental facilities have been developed and installed since the beginning of the reactor's operation, and continued efforts for developing more facilities are in progress. As new experimental facilities are added, we have seen a rapid growth in its utilization in terms of the number of users as well as fields of application. In other words, HANARO has established its status as a national neutron research facility. Internationally competitive experimental facilities of the reactor and the support of the government for HANARO users has promoted new researches in a wide range of areas to include the use of neutron in research activities, which is confirmed by the high growth record of the HANARO utilization every year. Also, an international symposium on research reactor and neutron science in commemoration of the 10th anniversary of HANARO was held in 2005 [2,3]. In this paper, the status and the perspective in the field of material irradiation tests in the HANARO are described.

Table 1. General Design Feature of HANARO

Reactor Type	Open-Tank-In-Pool
Thermal Power	30 MW
Thermal Neutron Flux (Max.)	5.4×10^{14} n/cm^2-s
Fuel Element	19.75% enrichment, U_3Si-Al Meat, Al Clad
Coolant	H_2O
Moderator	H_2O/D_2O
Reflector	D_2O
Core Cooling	Upward Forced Convection Flow
Absorber Material	Hafnium

2. HANARO and TEST HOLES

The HANARO was designed to provide a peak thermal and fast flux of $5\text{x}10^{14}$ n/cm^2·sec (E<0.625eV) and $2.1\text{x}10^{14}$ n/cm^2·sec (E>1.0MeV) at a 30MW thermal power, respectively. Since the commencement of HANARO operation in 1995, some parts of the reactor systems have been gradually improved for a stable operation of the reactor, while the operation mode has been flexibly adjusted to meet users' demands. During the same period, a significant number of experimental facilities has been developed and installed for the use of the 32 vertical holes and the 7 horizontal beam ports. The arrangement of the vertical holes and the beam ports are shown in Fig. 1.

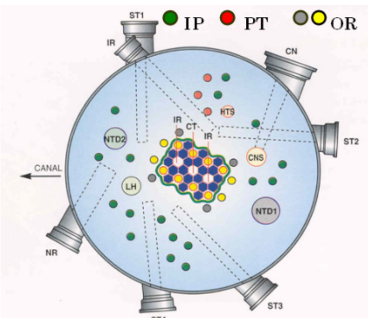

Fig.1. Core arrangement of HANARO

Three flux traps in the core (CT, IR1, IR2), providing a high fast neutron flux, can be used for material and fuel irradiation tests. They are also proper for the production of high specific activity radioisotopes. Four vertical holes in the outer core region, abundant in epithermal neutrons, are used for fuel or material irradiation tests and a radioisotope production. In the heavy water reflector region, 25 vertical holes abundant in high quality thermal neutrons are located for a radioisotope production, neutron activation analysis, neutron transmutation doping (NTD) and cold neutron source (CNS) installation. The two large holes named NTD1 and NTD2 are for NTD, the CNS for the cold neutron source installation, and the LH (large hole) for the irradiation of large targets. HTS is equipped with a hydraulic transfer system for a short half-life radioisotope production, a pneumatic transfer system (PTS #1~3) for a neutron activation analysis is installed in three NAA holes, and the 17 IP holes are for various target irradiations. Thermal and fast neutron fluxes of these test holes are listed in Table 2.

Table 2. Neutron flux of vertical test holes

Location	Hole Name	Hole No.	Inside Dia. (cm)	Neutron Flux (n/cm^2·sec) Fast (>0.82 MeV)	Neutron Flux (n/cm^2·sec) Thermal (<0.625 eV)	Remarks
Core	CT	1	7.44	2.10×10^{14}	4.59×10^{14}	Fuel material irradiation isotope production
	IR	2	7.44	1.95×10^{14}	3.95×10^{14}	
	OR	4	6.00	2.25×10^{13}	3.56×10^{13}	
Reflector	[illegible]	1	15.0	$6.62\times10^{[illegible]}$	[illegible]	Fuel material irradiation Isotope production
	[illegible]	1	10.0	$9.44\times10^{[illegible]}$	[illegible]	
	IP	17	6.0	$1.45\times10^{[illegible]}$	$2.40\times10^{[illegible]}$	
				$2.20\times10^{[illegible]}$	1.95×10^{14}	

There are 7 horizontal beam ports of different types available for researches on a neutron scattering, neutron radiography, prompt gamma neutron activation analysis (PGAA) and medical applications such as a BNCT (boron neutron capture therapy). All the beam ports are tangentially arranged and their beam noses except for the NR are located at the peak thermal neutron flux location in the reflector. Five beam ports (ST1~ST4 & CN) are for neutron scattering experiments, the IR is for ex-core neutron irradiation experiments such as a BNCT or dynamic neutron radiography, and the NR is for a neutron radiography. Owing to a stable operation of the reactor and a rapid proliferation in the utilization fields, more experimental facilities are continuously being added to satisfy various fields of study increasing and new research needs arising. As a nation-wide neutron research facility, the HANARO is now successfully utilized in various fields including nuclear fuel and material irradiation tests.

3. IRRADIATION DEVICES

The equipments for irradiation tests of nuclear fuels and materials in the HANARO are classified into two categories, such as, capsule and FTL (Fuel Test Loop). Capsules for irradiation tests of nuclear fuels and materials in the HANARO have been developed. And then, extensive efforts have been made to establish design/manufacturing and irradiation technologies for irradiating nuclear fuels and materials by using capsules and their control systems, which should be compatible with HANARO's characteristics. Other devices consisting of a fixing of the capsule during an irradiation test in the HANARO, a cutting and a transporting of the capsule main body after an irradiation test were also developed. These capsules and others have been actively utilized for the various material irradiation tests requested by users. Based on the accumulated experiences and the user's sophisticated requirements, capsules for creep test and fatigue test of materials during irradiation in the HANARO have been developed. And, the irradiation plans related to develop the Gen-IV reactor systems by using capsules in the HANARO will give more emphasis on development of capsules focusing on the irradiation tests of materials or nuclear fuels for the Gen-IV reactor systems, such as the SFR and the VHTR. The FTL is one of the irradiation devices, which can conduct the irradiation test of nuclear fuel in the HANARO under the operating conditions of commercial nuclear power plants. The 3-test fuel rods can be irradiated in the HANARO by using the FTL. The installation of the FTL was completed in March 2007. Currently, the commissioning test of the FTL is being performed. And, the FTL will be firstly used for the irradiation test of an advanced nuclear fuel for PWR from the end of this year.

3.1 Capsules

The main activities of the capsule development and utilization programs in the HANARO are focused on in-reactor material tests, new and advanced nuclear fuel research and development, safety-related research and development for nuclear reactor materials and components, and basic research. Now, capsules were developed and are utilizing for the irradiation test of materials and nuclear fuel in the HANARO, and creep and fatigue capsules have been developed to study for creep and fatigue behavior of materials under the irradiation.

A capsule for material irradiation test is one of the irradiation devices which can evaluate the irradiation performance of nuclear and high-technology materials in the HANARO. The development of the instrumented irradiation capsules and related technology started from 1995, and the capsule was first installed in HANARO in 1998. Now the capsule has an important role in the integrity evaluation of reactor core materials and the development of new materials through the precise irradiation tests of specimens such as RPV, reactor core, pressure tube, fuel cladding and high-technology materials.

A non-instrumented capsule is typically about 1 m in length and 60 mm in diameter. The temperatures of specimens is controlled by varying the widths of gas-filled gaps between the specimens and specimen holders, and monitored with passive fluence and temperature monitors(a eutectic alloys). A variety of specimen can be included in 5 stages of the capsule.

A typical instrumented capsule for material irradiation test in the HANARO is shown in Fig. 2. An instrumented capsule is of cylindrical shape and its main-body is 60mm in diameter and 870mm in length. The basic instruments of the capsule are thermocouples, fluence monitors, and heaters to fulfill user requirements. The temperature of specimens in 5 stages is independently controlled by a capsule temperature control system.

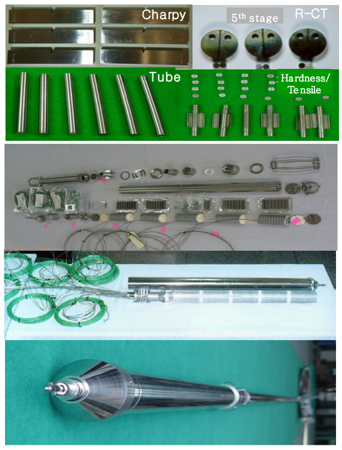

Fig. 2. Instrumented capsule for material irradiation

The utilization fields of the these capsules are as follows;

- Irradiation tests and damage evaluation of pressure vessel, reactor core, and high-technology materials, etc.
- Safety/integrity evaluation and life-extension researches of commercial reactor and production of design data of new nuclear materials for next generation nuclear power reactors
- Basic researches of irradiation effects in materials

A capsule for nuclear fuel irradiation test is applicable to study on an irradiation characteristics of the nuclear fuel pellet and to obtain the in-core performance and the design data of the nuclear fuel in the HANARO. The non-instrumented capsule was developed in 1999, and has been utilizing the irradiation characteristics test of DUPIC fuel and advanced PWR fuel pellets. The design verification test of an instrumented capsule was completed in the HANARO's test hole in 2003. Now, the instrumented capsule can be used to measure the fuel temperature, internal pressure of fuel rod, fuel deformation and neutron flux during fuel irradiation test.

A non-instrumented capsule is utilized in studies on irradiation characteristics of nuclear fuel in OR test hole of the HANARO. Its main-body is 960mm in length and 56mm in diameter, and includes 3~6 test fuel rods, length of about 200mm in the capsule.

A typical instrumented capsule for nuclear fuel irradiation test in the HANARO is shown in Fig. 3. An instrumented capsule is installed in OR4 or OR5 test hole of the HANARO, its main-body is 56mm in diameter and about 1 m in length and total length including the protection tube is about 5m. The capsule includes several test fuel rods instrumented thermocouple, pressure transducer and elongation detector to measure fuel temperature, internal pressure of fuel rod, and fuel deformation, respectively, and SPNDs to detect neutron flux.

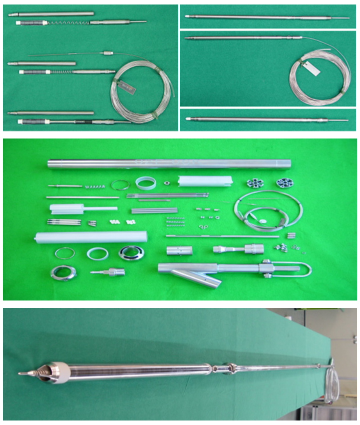

Fig. 3. Instrumented capsule for nuclear fuel

The utilization fields of the these capsules are as follows;

- Irradiation of nuclear fuel for DUPIC, advanced PWR and CANDU
- Study on the in-core characteristics of UO2 pellet and UO2 pellet including additives
- Basic study of the fission gas release, etc.

A creep capsule was developed to obtain the creep characteristics of nuclear material during irradiation in 2002, and was improved to increase the number of specimen. A typical capsule for creep test in the HANARO is shown in Fig. 4. A fatigue capsule was also developed. In order to get the reliable data of creep and fatigue test, the design of these capsules is under improving. Also, other advanced capsules will be continuously developed to support the development of materials and nuclear fuel for the Gen-IV reactor systems in the near future. An advanced capsule is a device to measure the changes of nuclear material or fuel properties and to control the irradiation conditions during irradiation test in the HANARO.

Fig. 4. Capsule for a creep test

3.2 Fuel Test Loop (FTL)

The FTL (Fuel Test Loop) simulates commercial NPPs' steady state operating conditions such as their pressure, temperature, flow, water chemistry conditions and neutron flux levels to conduct the irradiation and thermo-hydraulic tests [4]. The conceptual design of the FTL was started at the end of 2001 and both the basic and detailed design had been finished by March 2004. The installation of the FTL was successfully completed in March 2007. At present, the commissioning of the FTL is being conducted. The FTL will be used for the irradiation test of high burn-up PWR fuels after its commissioning is completed.

The fuel test loop provides the test conditions of a high pressure and temperature similar to those of commercial PWR and CANDU reactors. The FTL is composed of an OPS (Out Pile system) and an IPS (In-Pile test Section). The OPS is composed of a process system and an I & C (Instrumentation and Control) system. The IPS is to be loaded into the IR1 position in the HANARO core. The FTL coolant is supplied to the IPS at the required temperature, pressure and flow conditions that are consistent with a test fuel. The nuclear heat added within the IPS is removed by the main cooling water. Fig. 5 shows a schematic diagram of the FTL.

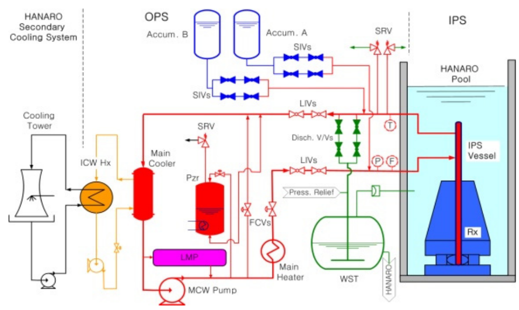

Fig. 5. Schematic diagram of the FTL

The process system contains several equipments such as a pressurizer, a cooler, a heater, pumps, and a purification system which are necessary to maintain the proper fluid conditions. The FTL coolant is supplied to the IPS at the required temperature, pressure and flow conditions that are consistent with a test fuel. The nuclear heat generated within the IPS is removed by the main circulating water cooler. The main circulating pump provides the motive power to circulate the FTL coolant within the loop. After a pump discharge, an in-line heater provides the capability to increase the temperature for a start-up and for a positive temperature control. A pressurizer is provided to establish and maintain the coolant pressure for the test fuel type. The process system includes the following systems [5].

- Main cooling water system
- Emergency cooling water system
- Penetration cooling water system
- Letdown, makeup, and purification system
- Waste storage and transfer system
- Intermediate cooling water system

- Test loop sampling system
- IPS inter-space gas filling and monitoring system
- Miscellaneous systems

The main test conditions for the FTL are given in Table 3. The I & C system has the following functions [6].

- Maintaining the irradiation test conditions by an automatic control,
- HANARO trip and a FTL safe shutdown during transient or accident conditions,
- Simultaneous operation of the FTL with the HANARO
- Data acquisition from the IPS.

Table 3. Main test conditions in the FTL

Test conditions	Value
Operation cycles a year	9
Operation cycle length (EFPD/cycle)	28
Number of test rods	3
LHGR (W/cm)	≤ 320
Peak to average heat rate	≤ 1.16
Fast neutron flux in cladding surface ($n/cm^2 \cdot sec$)	1.2×10^{14}
Coolant temperature (℃)	300 ~ 308
Coolant pressure (kg/cm^2)	150 ~ 159
Coolant velocity (kg/s)	1.52 ~ 1.8
B concentration (ppm)	≤ 1500
Dissolved oxygen concentration (ppm)	≤ 0.1
pH at 300℃	5.5 ~ 8.0
Electric conductivity (μS/cm)	≤ 50
Cl and F concentration (ppm)	≤ 0.2

The commissioning of the FTL was started from April 2007. The commissioning of the FTL is performed in three stages. An individual system performance test under room temperature is performed in the first stage, and the integral system performance test with mock-up fuels under a high temperature is performed in the second stage, and finally the integral system performance test with test fuels under a high temperature is performed in the third stage. The individual system performance test had been successfully completed. The integral system performance test with mock-up fuels under a high temperature is being performed. The integral system performance test with test fuels under a high temperature will be performed from November 2008.

The irradiation test for PWR fuels will be started after the commissioning is completed. The test rig includes 3 pins. Three SPNDs are installed in the upper, middle and lower parts of the irradiated section. Three thermocouples are installed in the inlet, middle and outlet points of the test rig. A LVDT is installed to measure the fission product pressure, and thermocouples are installed to measure the centreline temperature of a test fuel.

4. PIE FACILITIES

There are two facilities to perform PIE (Post -Irradiation Examination) in KAERI, that are IMEF (Irradiated Material Examination Facility) and PIEF (Post-Irradiation Examination Facility). The main mission of IMEF is to provide PIE services for the irradiated fuels and materials in the HANARO. It has 8 concrete hot cell, 1 lead hot cells and 1 service pool. And, PIEF is essentially employed for testing and then evaluating the performance and the integrity of nuclear fuels discharged from reactors. It has 4 concrete hot cell, 2 lead hot cells and 1 service pool.

5. CONCLUSIONS

Since the commencement of the HANARO operation in 1995, some reactor systems have been gradually improved for a stable operation of the reactor, while the operation mode has been flexibly adjusted to meet the reactor's demands. Meanwhile we started a full power operation at 30 MW from a conditional operation at 24 MW, and changed the operation mode to a 23-day operation and a 12-day shutdown. During the past years, a significant number of experimental facilities have been developed and installed to make efficient use of the 3 vertical holes in the inner core region, the 4 vertical holes in the outer core region, the 25 vertical holes in the heavy water reflector region, and the 7 horizontal beam ports. Among them, the development of the Fuel Test Loop (FTL) and the cold neutron research facility (CNRF) project are the most important on-going efforts. As a nation-wide neutron research facility, the HANARO is now successfully utilized in various fields such as neutron beam research, fuel and material irradiation tests, radioisotope production, neutron activation analysis, and neutron transmutation doping, etc.

Some capsule for irradiation test of materials and nuclear fuel were developed and perform the irradiation tests for a last decay, and continuously the development of capsules will be carried on to support the development of the Gen-IV reactor systems in the HANARO. The commissioning of the FTL will be conducted in 2008. The FTL will be used for an irradiation test of a high burn-up PWR fuels after its commissioning is completed. The capsule and FTL will also be used for materials and nuclear fuel irradiation test in the HANARO in succession and could act a role of maximizing the utilization of the HANARO.

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REFERENCES

[1] Chang-oong Choi, et. al., "Present Situations and Perspective on the Advanced Utilization of HANARO", *Physica B*, 311 (2002) 34-39.

[2] Proceeding of International Symposium on Research Reactor and Neutron Science in commemoration of the 10th anniversary of HANARO, Apr. 11-13, 2005, Daejeon, Korea, KAERI/GP-234/2005.

[3] Special Issue on International Symposium on Research Reactor and Neutron Science in commemoration of the 10th anniversary of HANARO, *Nucl. Eng. and Tech.*, Vol. 38, No. 5, 2006.

[4] S. K. Park., et. al., "Prediction on the Nuclear Fuel Cladding Temperatures for the LBLOCAs of the HANARO Fuel Test Loop", Journal of Korea Society of Mechanical Technology, Vol.9, No. 3, pp. 53-58, 2007.

[5] S. K. Park., et. al., "Consideration on the Conservatism of LOCA Analysis for the HANARO Fuel Test Loop", Journal of Korea Society of Mechanical Technology, Vol.9, No. 2, pp. 1-7, 2007.

[6] S. H. Ahn, et. al., "Development of safety related control panels for HANARO fuel test loop", (Proc. of the Korea Nuclear Society Spring Meeting), Korea, (2007) (CD-ROM).

2.8 Status and Future Plan of Japan Materials Testing Reactor

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The Japan Materials Testing Reactor (JMTR) of Japan Atomic Energy Agency (JAEA) is a light water cooling tank typed reactor. JMTR has been used for fuel and material irradiation studies for LWRs, HTGR, fusion reactor and RI production. Since the JMTR is connected with hot laboratory through the canal, re-irradiation tests can conduct easily by safety and quick transportation of irradiation samples. First criticality was achieved in March 1968, and operation was stopped from August, 2006 for the refurbishment. The reactor facilities are refurbished during four years from the beginning of FY 2007, and necessary examination and work are carrying out on schedule. The renewed and upgraded JMTR will start from FY 2011 and operate for a period of about 20 years (until around FY 2030). The usability improvement of the JMTR, such as higher reactor available factor, shortening turnaround time to get irradiation results, attractive irradiation cost, business confidence, is also discussing as the preparations for re-operation.

Keywords: JMTR, Material Testing, Refurbishment, Irradiation, LWR, RI Production

1. INTRODUCTION

JMTR of JAEA is a light water cooling tank typed reactor. JMTR has been used for fuel and material irradiation studies for LWRs, HTGR, fusion reactor and RI production. Since the JMTR is connected with hot laboratory through the canal, re-irradiation tests can conduct easily by safety and quick transportation of irradiation samples. First criticality was achieved in March 1968, and operation was stopped from August, 2006 for the refurbishment.

The reactor facilities are refurbished during four years from the beginning of FY 2007, and necessary examination and work are carrying out on schedule as follows.

- Aged-investigation: It was confirmed that the condition of primary and secondly cooling tube and so on was good by the investigation.
- Component replacement: Control rod drive mechanism, reactor control system, primary cooling pumps, secondary cooling pumps, electric power supply system and so on, were decided to replace.
- Specific designs for component replacement; The designs were finished, and replacement components were decided from a viewpoint of future maintenance, reliability and so on.

The renewed and upgraded JMTR will start from FY 2011 and operate for a period of about 20 years (until around FY 2030). The usability improvement of the JMTR, such as higher reactor available factor, shortening turnaround time to get irradiation results, attractive irradiation cost, business confidence, is also discussing as the preparations for re-operation.

2. OUTLINE OF JMTR

JMTR is a testing reactor dedicated to the irradiation tests of materials and fuels. It had achieved first criticality in March 1968. High neutron flux generated in the core of JMTR is utilized for the irradiation experiments of fuels and materials, as well as for radioisotope productions. JMTR provides various irradiation facilities, such as many types of irradiation capsules, shroud irradiation facility and hydraulic rabbit irradiation facility.

Irradiated capsules or specimens are transferred to the hot laboratory, which is connected to the reactor building through a water canal, for post irradiation examinations (PIE). Owing to the shielding capability of the water, irradiated radioactive capsules or specimens are safely transferred underwater through the canal. Cross section of JMTR and hot laboratory are shown in Fig.1.

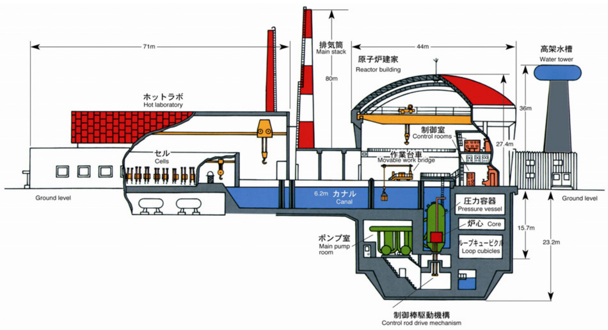

Fig.1 Cross section of JMTR and hot laboratory.

The reactor pressure vessel, 9.5m high with 3m in inner diameter, is made of low carbon stainless steel (SUS304L) and is located in the reactor pool, which is 13m deep. The control rod drive mechanisms are located under the pressure vessel for easy handling of the irradiation facilities and fuels in the core.

The core of the JMTR is in a cylindrical shape with 1.56m in diameter and 0.75m high, it consists of 24 standard fuel elements, five control rods with fuel followers, reflectors and H-shaped beryllium frame.

Cooling water in primary cooling system is pressurized at about 1.5MPa to avoid local boiling in the core during power operation.

The heat generated in the core is removed by the cooling water in the primary cooling system. The cooling water flows downwards in the core and transfers the heat from the core to secondary cooling system through heat exchangers. The heat transferred to the secondary cooling system is removed away into the atmosphere in cooling towers. Cutaway view of reactor is shown in Fig.2.

JMTR is utilized for the basic and the applied researchers on the fuels and materials of fission reactors and fusion reactor, and radioisotope productions. Power ramping tests for the nuclear fuels are also performed to study the integrity and safety of the fuels.

Test specimens irradiated in the JMTR are transferred to the hot laboratory for PIE. The data obtained are used for the development of nuclear fuels and materials and safety assessment of the reactor.

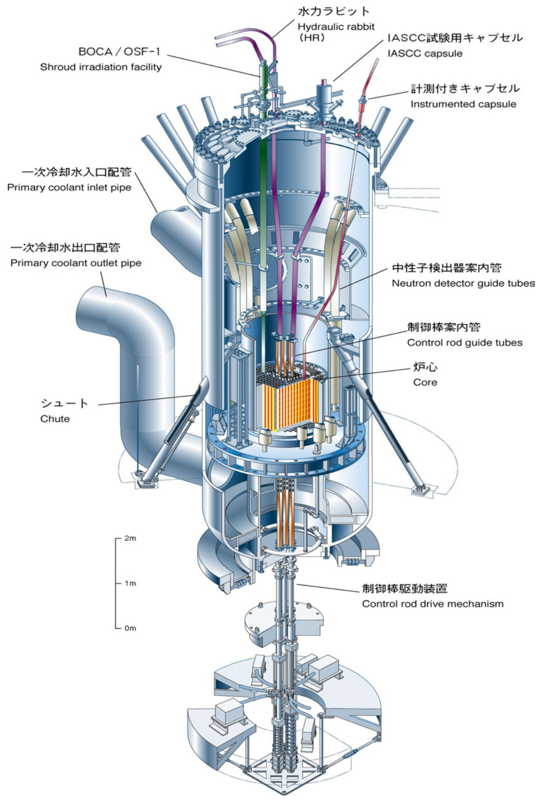

Fig.2 Cutaway view of reactor.

Radioisotopes produced in the JMTR are widely utilized in the medical treatment, industries and agriculture [1-3].

3. START OF NEW JMTR

The reactor facilities are refurbished during four years from beginning of FY 2007, and the operation of new JMTR will start in FY 2011.

3.1 The usability improvement of the JMTR

The usability of JMTR will also be improved to be attractive to users, shown as follows.

(1) Achievement of the reactor available factor from 50% to 70%.

(2) Shortening of turnaround time to get irradiation results earlier.

(3) Realization of more attractive irradiation cost in comparison with other testing reactors in the world.

(4) Establishment of more simple irradiation procedure and more satisfied technical support system.

(5) Guard of the business confidence by perfect information control, etc.

As for the item (1), the possibility of reactor scram by the accident will be decreased by the replacement of reactor components as described above. In addition, even if the failure of components occurs, the repairing the failed components will also become easier. These will shorten the time of out-of-operation. Actually, JMTR has already had a experience of high capability of operation, that is more than 180days in a year in two times. Then, the replacement of old and unreliably components leads the higher capability of operation. Furthermore, optimization of the overhaul time of the reactor defined once per year by the Japanese regulation will take longer operation period during one year. Items from (2) to (5) are now under discussion taking users requests into consideration.

3.2 Target of new JMTR

(1) Proposal of Attractive Irradiation Tests

Proposal of attractive irradiation test will be carried out by advanced technologies such as new irradiation technology, new measuring technology and new PIE technology. Cooperation with various nearby PIE facilities, surrounding the JMTR is also under discussion in order to extend the capability of PIE.

(2) Establishment of International Center

Construction of the research base utilized internationally as the Asian center of testing reactors is now under consideration. In Asian area, some excellent testing reactors are operated, such as HANARO in Korea, OPAL in Australia. Each of these reactors has individual and original characteristics and can take supplementary role in each other.

(3) User-Friendly Management

User-friendly management must be established by above-mentioned improvement of usability of JMTR. The technical support system for the users will be established by place the specialists of irradiation technology and irradiation research, such as specialists of reactor fuel and reactor materials. The users will be able to discuss sufficiently on the detail irradiation method with these specialists at the planning stage of irradiation. This is an example of the improvement of the usability which is easy to use for many users due to the fulfillment of the technical support system.

3.3 Expected roles of new JMTR

As described above, JMTR will be refurbished by the replacement of old-designed components and development of new irradiation facilities. Also, the usability is planned to be improved. As for these improvements, the following roles are expected on the new JMTR.

(1) Lifetime Extension of LWRs
- Aging management of LWRs
- Development of next generation LWRs

(2) Progress of Science and Technologies
- Development of fusion reactor materials and developments
- Development of HTGR (High Temperature Gas cooled Reactor) fuels and materials
- Basic research on nuclear energy, etc.

(3) Expansion of Industrial Use
- Production of silicon semiconductor for hybrid car
- Production of ^{99m}Tc for medical diagnosis medicine

(4) Education and training of nuclear scientists and engineers

The new JMTR is planned to contribute the worldwide research fields and industrial fields by playing these important roles.

4. REFURBISHMENT OF JMTR

Refurbishment of JMTR can be divided into two categories, "replacement of reactor components" and "construction of new irradiation facilities".

(i) Replacement of reactor components

The replaced components are decided from the accumulated experience. Aged or old-designed components of control rod drive mechanism, reactor control system, primary cooling pumps, secondary cooling pumps, electric power supply system and so on, will be replaced by present-designed ones. For example, the circuits of reactor control system and process control system which consist of a huge amount of relays and soldered wirings will be replaced by present-designed integrated circuits.

As for the facilities which are not replaced, for example heat exchangers, pressure vessel, secondary cooling towers etc, safety has being investigated from a view point of view of the aging. The capability of long-term operation in the future has been certified by this investigation.

By these replacements for safe and steady reactor operation, decrease of the possibility of failure of each component and capability of prompt repairing of the failed component will be established. This leads improvement of the rate of reactor operation in future.

Replacement of reactor component is shown in Fig.3 and component replacement schedule is shown in Fig.4.

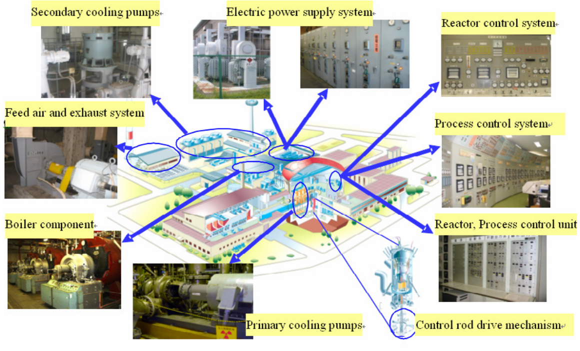

Fig.3 Replacement of reactor components.

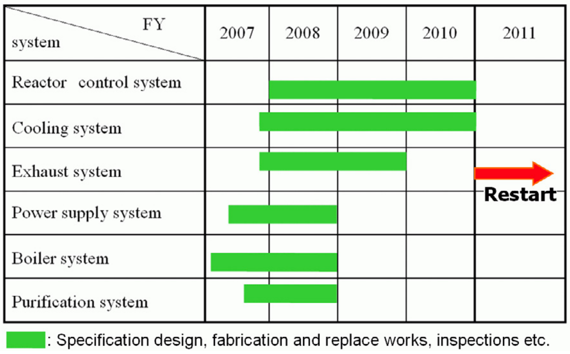

Fig.4 Component replacement schedule.

(ii) Construction of new irradiation facilities

New irradiation facilities, i.e. irradiation test facilities of materials and fuels, production facilities for silicon semiconductor and medical radioisotopes, will be established in JMTR.

- New Material and Fuel Irradiation Tests

Irradiation test facilities of materials and fuels are being developed and will be prepared in the JMTR during a shutdown period of about 4 years starting from April 2007 by requirements from the regulatory and developing uses of LWRs for the purpose of the long term and up-graded operations.

Requirements are addressed on higher performance uses of LWRs, e.g. power up rating, longer operation cycles and modified water chemistries for lifetime extension of the power plants to obtain evaluation data of fuel and materials under irradiation conditions. To meet one of these requirements, an irradiation capsule with a larger test section for tests with large sized specimen of reactor materials in order to investigate the scale effect on the IASCC behavior is now being developed. A new type of a power ramp test facility is also under development to provide the constant surface temperature of test fuel rod during a boiling transient. It is planned to realize the linear power of a test fuel by controlling the pressure of surrounding ^{3}He gas screen, absorber of neutrons.

- New Irradiation Facility for Industrial Purpose

Present development plan of irradiation facility for industrial purpose includes the development of irradiation facility for production of silicon semiconductor. Target of development is to establish the irradiation facility of large sized silicon ingot with 8 inches in diameter which meets the trend requirement of the field of hybrid cars and so on.

Another plan is to provide the ^{99m}Tc for medical use. However, the hydraulic rabbit irradiation facility, which is well developed and already used for irradiation in JMTR, can be applied. Now, investigation of production performance and costs are carried out [4].

5. CONCLUSIONS

The JAEA placed that the JMTR is a testing reactor which supports the basic technology of the nuclear energy, and decided the refurbishment of the reactor facilities during four years from FY 2007; operation of the new JMTR will be started from FY 2011.

In the same time, irradiation facilities corresponding to the user needs, such as Nuclear and Industrial Safety Agency, will be installed to contribute the lifetime extension of LWRs by the user's fund. Additionally, the contribution to the development of the ITER and the industrial use etc., are discussed in the JAEA.

In the practical use of the JMTR, the JAEA will promote the expansion of use and improvement of usability (e.g. improvement of the reactor available factor, shortening of the turnaround time, achievement of the attractive irradiation cost, establishment of the satisfied technical support system, retention of the business confidence) taking account of the opinion obtained from external experts such as “the JMTR user's committee”, “the Council for Science and Technology Policy” and so on, and also taking account of the management considering the outside-user including the industry.

REFERENCES

[1] Department of JMTR, "The Course of JMTR", (2006)

[2] Neutron Irradiation and Testing Reactor Center, "The evolution of the JMTR", leaflet, (2007)

[3] Department of JMTR, "JMTR Japan Material Testing

Reactor", booklet, (2005)
[4] Koichi IIMURA, other, "Conceptual Study of 99Mo production Facility in JMTR", JAEA-Technology 2008-035

3. Material Development for Research and Testing Reactor

3.1 Problems and Future Plan on Material Development of Beryllium in Materials Testing Reactors

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Beryllium has been utilized as a moderator and/or reflector in a number of material testing reactors. The attractive nuclear properties of beryllium are its low atomic number, low atomic weight, low parasitic capture cross section for thermal neutrons, readiness to part with one of its own neutrons, and good neutron elastic scattering characteristics. However, it is difficult to reprocess irradiated beryllium because of high induced radioactivity. Disposal has also been difficult because of toxicity issues and special nuclear material controls. In this paper, problems and future plans of beryllium technology are introduced for nuclear reactors.

Keywords: Beryllium, Reflector, Beryllium recycling, Lifetime extension, Testing reactor

1. INTRODUCTION

As a structural material, beryllium is a light material which has high tensile strength. Beryllium surfaces form a thin oxidation film by interacting with air, like aluminum, and beryllium is highly resistant to corrosion in dry gases. Its useful properties such as thermal conductivity and good elevated-temperature mechanical properties for light element and high melting point make the metal attractive for nuclear reactors. Especially, beryllium has been utilized as a moderator and/or reflector in a number of material testing reactors. Among the attractive nuclear properties of beryllium are its low atomic number, low atomic weight, low parasitic capture cross section for thermal neutrons, readiness to part with one of its own neutrons (n, 2n), and good neutron elastic scattering characteristics [1-2].

Reactors with beryllium exist in many places throughout the world, and a lot of beryllium was used in materials testing reactors (MTR) from the beginning of atomic energy development. Usage of beryllium in neutron fields causes its mechanical properties to become worse. Possible durability in this case is determined by that neutron fluence at which minimum allowed quality of beryllium is achieved. The activation issues for beryllium in nuclear reactors under neutron irradiation arise mainly via (n, γ) and (n, p) reactions with impurities such as iron, nickel and nitrogen in the beryllium. At the same time, tritium (^3H) is produced in the beryllium by a well known reaction sequence. It is difficult to reprocess irradiated beryllium because of high induced radioactivity. Disposal has also been difficult because of toxicity issues and special nuclear material controls.

In this paper, problems and future plans of beryllium technology are introduced for nuclear reactors. Material modification and waste issues of beryllium reflectors are discussed for lifetime extension and recycle of used beryllium irradiated in MTR. These items were discussed in the specialist meetings on beryllium study at INL (Jul. 2007) and in Lisbon (Dec., 2007).

2. STATUS OF IRRADIATED BERYLLIUM IN MTR

Table 1 shows general properties of beryllium. Beryllium is a light material which has high tensile strength. Density is about 1.85g/cm^3, melting point is 1285˚C and thermal conductivity is 188 W/m/K at 25˚C. Beryllium surfaces form a thin oxidation film by interacting with air, like aluminum, and beryllium

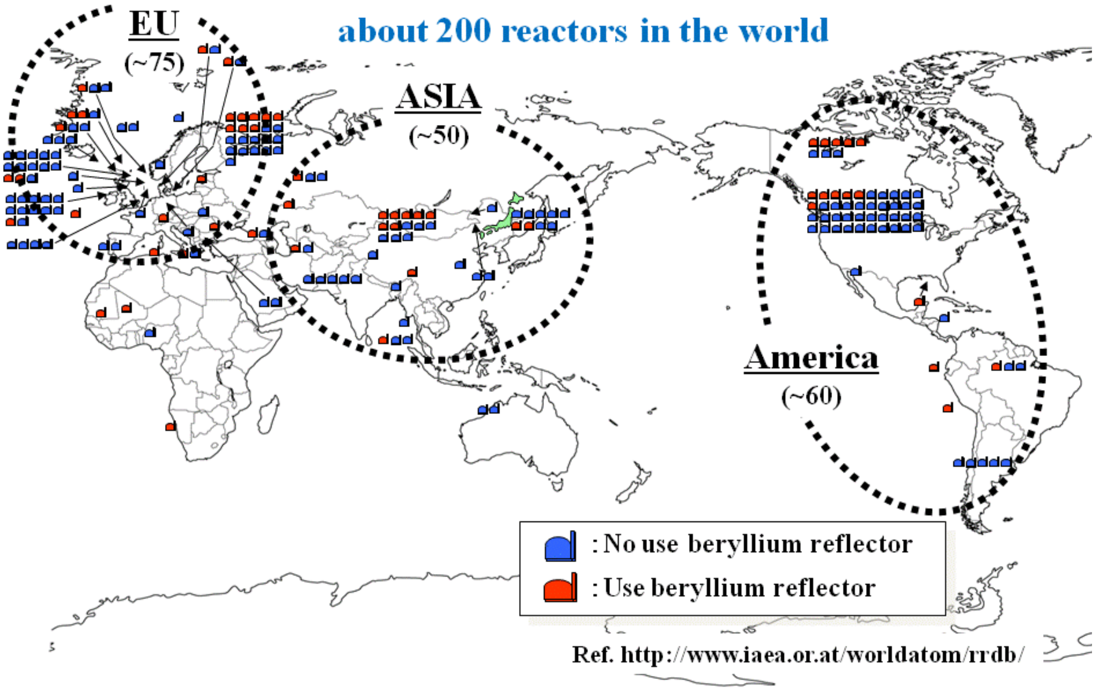

Fig. 1 Research and testing reactors with beryllium as the reflector or moderator in the world.

Table 1 Properties of beryllium

Properties	Values
Density (g/cm^3, at 25°C)	1.8477±0.0007
Atomic weight	9.013±0.0004
Melting point (°C)	1285±10
Boiling point (°C)	2507-2970
Sublimation point (°C)	482 ($BeCl_2$)
Specific heat (J/kg/K, at 25°C)	1923
Thermal conductivity (W/m/K, at 25°C)	188
Thermal expansion (x10^{-6} °C)	13

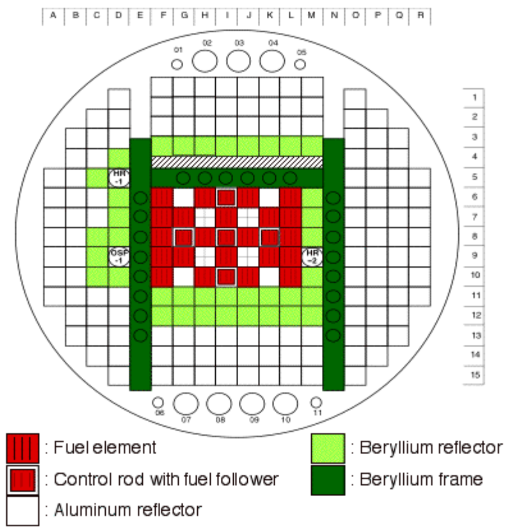

Fig.2. Core arrangement of JMTR.

is highly resistant to corrosion in dry gases.

Figure 1 shows research and test reactors in the world. There are about 200 reactors in the world [3], for example, 50 reactors in Asia region, 75 reactors in EU region and 60 reactors in America region. In this figure, beryllium is used as the reflector in the research and testing reactors of 30% in the world.

For example, figure 2 shows the core arrangement of JMTR. As the engineering data of JMTR, thermal power is 50MW, and maximum total neutron flux is 4×10^{18} n/m^2/s. The core, 1560mm in diameter and 750mm in effective height, is divided into four regions by the H shape partition wall (beryllium frame) made of beryllium and has an array of 224 squares, 77mm on each side, arranged in a square lattice [4].

The status of irradiated beryllium was investigated. Characteristics of irradiated beryllium are (1) internationally regulated, (2) specific chemical hazards such as BeO, (3) contains tritium and (4) activated material. Therefore, wastes of irradiated beryllium are difficult to handle and increase on and on.

Now, the amount of irradiated beryllium is about 3 tones in Japan, and what's more, is about 40 tones in the world. In USA, the beryllium used at ATR, etc was buried in the desert until 1990. Since then, it has been kept in the ATR canal. However, as the water pollution occurred by ^{14}C, buried beryllium was encapsulated in wax, and surface storage has been considered. In EU, Russia and Japan, irradiated beryllium has been kept in the pool generally.

3. PROPOSAL FOR SOLUTION OF BERYLLIUM ISSUES

There are two points for consideration in the solution of irradiated beryllium wastes. One is to manage the inventory of irradiated beryllium waste accumulated up to now. The irradiated beryllium has been kept at the reactor sites. Recycling of irradiated beryllium is proposed. The other is to reduce amount of irradiated beryllium waste generated from now. Irradiation tests are planned for the design modification of beryllium materials.

3.1 Lifetime Extension of Beryllium Reflector

Beryllium is fabricated by vacuum hot pressing. S-200F is used as the standard material, and the typical purity of nuclear grade beryllium such as S-200F is 99.1%. Beryllium reflectors are irradiated at temperatures from 50 to 150°C in cooling water of MTR.

Now, the refurbishment of the JMTR is underway, and it will run through 2011. The plan is for the new JMTR to operate through the year 2030. The upgrade will include new irradiation facilities and core configuration. JAEA is also targeting the reduction of reactor down-time as one of the goals. For the beryllium frame, it means an operational service lifetime goal of 15-20 years (180,000MWD), rather than the current five years. In order for that to happen, it will be necessary to consider fundamental changes to the frame design, starting with the choice of beryllium material grade.

For the choice of beryllium materials, shape and purity of beryllium powder, and uniformity of grain size are considered in a cooperation program between JAEA and Brush Wellman. Especially, uniformity is different in vacuum hot pressing than when hot isostatic pressing is used. Therefore, three kinds of beryllium materials were selected for lifetime extension in the specialist group. S-200F is the reference material as the reflector. S-65-H will be tested due to its higher purity and better isotropy than S-200F and I-220-H will be tested due to its higher mechanical strength and better isotropy than S-200F. Table 2 shows properties of candidate materials [5].

Candidate test reactor facilities for performing the irradiation include: the JRR-3M (JA), the BR2 (Belgium), the ATR (U.S.A.), and the SM-3 (RF).

3.2 Beryllium Waste Disposal

The reprocessing of the irradiated beryllium from nuclear reactors has been proposed in Japan, and beryllium will be purified by this technology. Figure 3 shows the concept of recycling of irradiated beryllium reflectors [6-7]. This recycling process consists of (1) beryllium separation from activated nuclides as the impurities in the beryllium, (2) beryllium purification from recovered beryllium compounds, and (3) refabrication of metallic beryllium. The beryllium pebbles for fusion reactors will be fabricated with the purified beryllium. It is possible to establish the recycling technology of the irradiated beryllium using traditional atomic techniques.

Table 2. Candidate materials of beryllium grades for lifetime extension

(a) Purity & Grain Size Comparison

Technical Factor	Be Grade	Be Assay		Grain Size
		min	typ	max
		(%)		(μm)
Reference	S-200F	98.5	99.1	20
Isotropy	S-65-H	99.0	99.4	15
Strength	I-220-H	98.0	98.6	15

(b) Mechanical Property Comparison

Be Grade	YS		UTS		Elongation	
	min	typ	min	typ	min	typ
	(MPa)		(MPa)		(%)	
S-200F	241	260	324	380	2.0	3.0
S-65-H	206	280	345	450	2.0	5.1
I-220-H	345	498	448	577	2.0	3.2

The industrial technology for the recycling of irradiated beryllium was published in the previous paper. In ref. 6, the preliminary tests on beryllium separation step and purification step of the irradiated beryllium were carried out in the initial stage. The beryllium separation utilizes the reaction between beryllium and chlorine, and beryllium chloride ($BeCl_2$) with sufficiently low melting temperature was generated by this reaction. Purification of the irradiated beryllium is based on the difference of phases of beryllium chloride and chlorine interaction products with ^{3}H and ^{60}Co.

A kg-scale demonstration test with used beryllium is proposed under the ISTC project. The project leader is IAE NNC-RK in Kazakhstan, and Japan and EU are collaborated in this project. In Japan, about 10g scale recycle test with irradiated beryllium was carried out, and the experience will be utilized in this project. Additionally, a small scale recycle test with beryllium was also carried out in EU, and the experience will be utilized in this project.

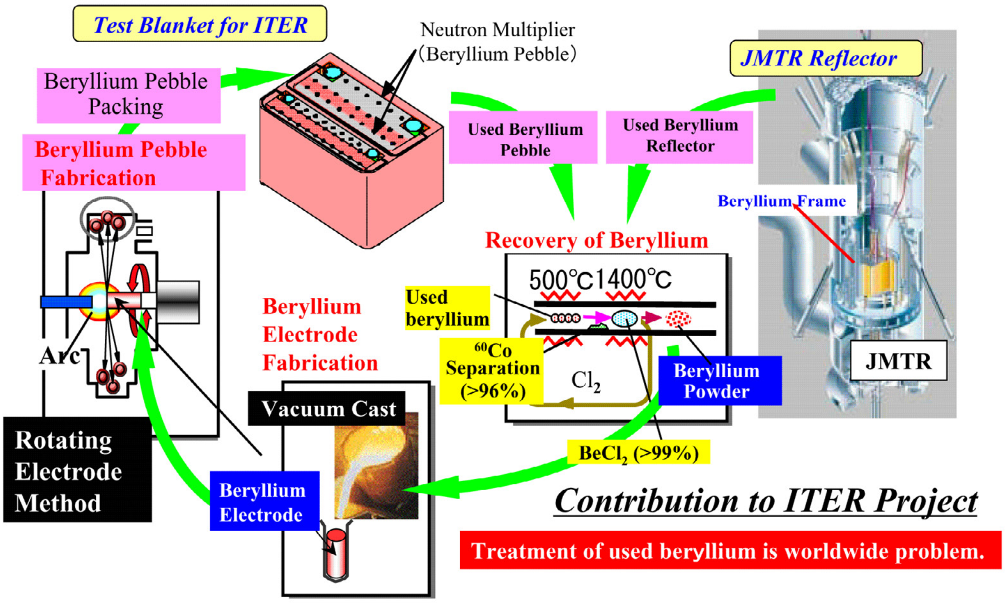

Fig. 3 Concept of recycling of irradiated beryllium reflectors.

Project purposes are research of possibility of irradiated beryllium purification from radioactive tritium and cobalt by the technology with conversion of beryllium to beryllium chloride and R&D on the purification of the beryllium irradiated in the nuclear reactor. Additionally, the irradiated beryllium will be transported from Japan to Kazakhstan in this project.

This study of beryllium recycle with irradiated beryllium will be started in 2008, and assimilation of each technology will be expected under this project.

4. CONCLUSIONS

Summary in the beryllium specialist meeting is shown as follows;

1) Investigation of Amount of Irradiated Beryllium
- In ATR : 6.9 ton (storage)
- In JMTR : ~3 ton (storage)
- In EU including BR-2 : ~3.5 ton (storage)
- In SM-3 and MIR : ~3 ton (storage)

2) Lifetime Extension Program
- Material selection for beryllium reflector (specification decision).
- Irradiation test for lifetime extension under cooperation study in Material Testing Reactor Communities.

3) Be Waste Disposal Program
- Development of the kg-scale recycle process with JAEA and EU under ISTC project.
- Consideration of ton-scale recycle program based on kg-scale recycle test results.
- Construction of cooperation program among Beryllium Meeting attendees.

Future plan of beryllium study is shown in Fig.4, and material tests for lifetime extension and reprocessing test of irradiated beryllium will be started in 2008.

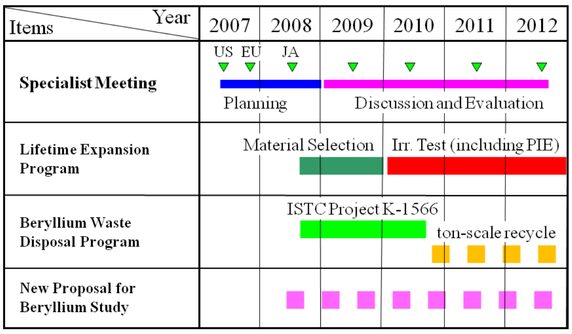

Fig.4 Future plan of beryllium study for MTR development.

REFERENCES

[1] Edited by C.R.Tipton, Reactor handbook – Second edition in four volumes - , Interscience Publishers, Inc., New York, (1960) p.897.
[2] J.M. Beeston, Nuclear Engineering and Design, 14 (1970) 445-474.
[3] http//www.iaea.or.at/worldatom/rrdb/
[4] JMTR project, "Conceptual Design of the Japan Materials Testing Reactor", JAERI 1056 (1964).
[5] C. Dorn, et al. in this presentation.
[6] H. Kawamura, et al., Proc. 2nd IEA International Workshop on Beryllium Technology for Fusion, Sept. 6-8 (1995), Jackson Lake Lodge, Wyoming USA, P.261.
[7] E. Ishitsuka, H. Kawamura, et al., Proc. 6th IEA International Workshop on Beryllium Technology for Fusion, JAERI-Conf 2004-006, (2004) P.262.

3.2 Material Selection for Extended Life of The Beryllium Reflectors in the JMTR

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The Japan Materials Test Reactor (JMTR) has been one of the most significant high-energy test reactors in the world since achieving its first criticality in 1968. Beryllium has been used as the reflector element material in the reactor, specifically S-200F structural grade beryllium manufactured by Brush Wellman Inc. The JMTR is currently in the process of being refurbished, and the upgraded reactor will return to service in 2011. As a part of the reactor upgrade, the Japan Atomic Energy Agency (JAEA) also has plans to extend the operating lifetime of the beryllium reflector elements. In order to do that, it will first be necessary to determine which of the material's physical and mechanical properties will be the most influential on that choice. Selecting a different grade of beryllium material for the reflector elements to extend operational lifetime under neutron irradiation is discussed in detail. A new plan for irradiation testing to evaluate the various beryllium grades under consideration is also briefly described.

Keywords: JMTR, reflector elements, nuclear grade beryllium, lifetime, vacuum hot-press (VHP), hot isostatic press (HIP), mechanical properties, purity

1. INTRODUCTION

The Japan Materials Testing Reactor (JMTR) is a world-class high-energy nuclear test reactor which has been in use for 40 years at this writing. The JMTR's primary mission is to perform irradiation tests on fuels and materials, and its high-neutron flux can also be used for the production of radioisotopes. Power ramp testing of nuclear fuels is also carried out to evaluate the integrity and safety of the fuels under study [1].

Beryllium has always been used as the reflector element material in the JMTR. The core of the JMTR, which is 1560mm in diameter and 750mm in effective height, is divided into four regions by an H-shaped partition wall (beryllium frame), which is made of beryllium and has a 224-unit array of 77.2mm squares arranged in a square lattice. The effective part of this element is also made of beryllium [2].

Current issues with and future plans for beryllium technology in the field of material development for nuclear test reactors has been and will continue to be discussed on an ongoing basis in specialist meetings [3]. Two main issues have been identified in these meetings. The first is the potential for changing the material specification (grade) for the beryllium reflectors. The second is the possible reduction of waste by recycling the irradiated beryllium.

In this paper, alternative beryllium material grades to extend the lifetime of the reflectors under neutron irradiation are considered.

2. BERYLLIUM IN THE JMTR

The JMTR has used beryllium reflector elements since it began operation in 1968. The reactor has been operated using structural grade beryllium made by Brush Wellman Inc. in Elmore, Ohio, U.S.A. Since Brush's introduction of S-200F Structural Grade Beryllium in 1985, it has been specified as the reflector element material for the JMTR.

2.1 S-200F Structural Grade Beryllium

Over the last 20 years, S-200F beryllium has become the Brush Beryllium Products largest volume production material. Like most beryllium grades, S-200F is a powder-metal product. What makes this material different from its predecessors (S-200E, etc.) is that the powder is made by impact grinding. This process gives a relatively equi-axed particle, which gives S-200F more isotropic mechanical properties and lower oxide content than the previous materials which were made with attrition-milled powders, which consisted of more flake-like particles. Figure 1 shows an illustration of the difference between the powder types.

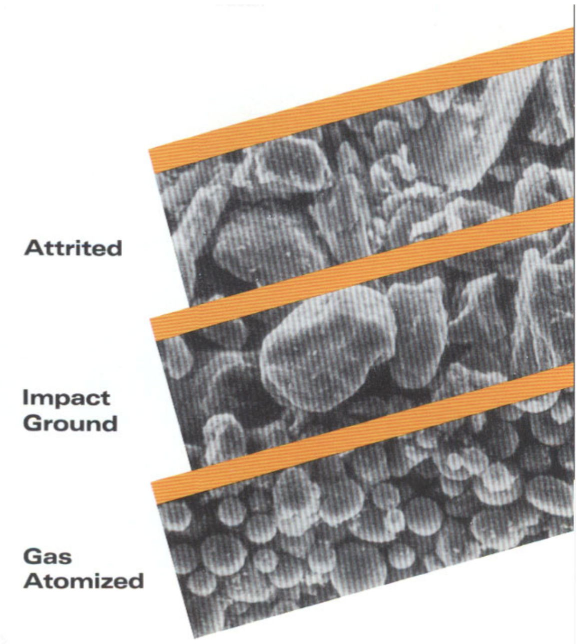

Fig.1. Photomicrographs of beryllium powder types.

2.2 Reflector Element Failure Mode

At the current JMTR operating parameters of 50MW for six cycles per year (consisting of about 30 operational days per cycle or 180 days per year), the beryllium frames generally last for about 5 years (about 36,000MWD) before replacement. "MWD" signifies "Mega-Watt Days", which is the amount of power in MW multiplied by the number of days of operation. It is well established that the failure of the beryllium reflector elements is due to the swelling of the material which is caused by its irradiation during the operation of the reactor. The swelling leads to dimensional change in the frame, which results in bending of the parts.

The real question is: which technical aspect of the beryllium's properties has potentially the greatest influence in reducing the radiation-induced swelling? This is not yet known, but working together, JAEA and Brush Wellman have a good idea of where to look for answers.

3. THE JMTR UPGRADE

The planned refurbishment of the JMTR is now underway, and it will run through 2011. The plan is for the new JMTR to operate through the year 2030. The upgrade will include new irradiation facilities and core configuration. JAEA is also targeting the reduction of reactor down-time as one of the goals. For the beryllium frame, it means an operational service lifetime goal of 15-20 years (180,000MWD), rather than the current five years. In order for that to happen, it will be necessary to consider fundamental changes to the frame design, starting with the choice of beryllium material grade.

4. BERYLLIUM TECHNICAL FACTORS

Based on a cooperative effort between JAEA and Brush, the technical factors for the beryllium frame material have been considered as follows.

4.1 Purity of the Beryllium

Pure beryllium has a very low mass absorption coefficient, especially in comparison to other metals. As noted earlier in this paper, beryllium irradiated in a nuclear test reactor like the JMTR will eventually swell. Metals which have much higher mass absorption coefficients will absorb the radiation at a higher rate and will consequently incur greater overall swelling, and they will swell at faster rates as well. For this reason, the other metallic trace elements found in commercially "pure" beryllium are a potentially important technical factor in understanding what causes the swelling.

For purposes of this discussion, "purity" will be defined as the chemical composition of the beryllium material. This is indicated primarily by the beryllium assay (i.e. the percentage of beryllium by weight in the material). The remainder of the chemical composition consists of beryllium oxide (BeO) and other trace elements found in the material.

4.2 Isotropy

There are two main aspects in the manufacturing of beryllium which will affect its isotropy, and those are the powder morphology and the consolidation process. When speaking of isotropy in the context of this paper, the reference is to the uniformity of properties of the material, irrespective of the direction of testing.

4.2.1 Powder-Derived vs. Cast

The reason that structural grade material is a powder metal product in the first place is due to beryllium's hexagonal close-pack (HCP) crystal structure, which is inherently anisotropic.

In cast form, the material grows very large grains which do not mechanically deform uniformly, and it has virtually no ductility in one direction (i.e. it shows preferential cleavage in the basal plane of the crystal). This means that beryllium with a large grain structure will have lower mechanical strength. Powder metallurgy permits the material to be produced in a fine-grained form, which overcomes the crystal structure problem by distributing loads in low ductility-oriented grains to grains oriented with high ductility.

The use of impact-ground powder instead of attrition-milled powder is also an added improvement to the isotropy of the resulting material because the grains are more randomly distributed.

4.2.2 Consolidating the Powder

The process used for consolidating the beryllium powder is also a factor in the isotropy of the final product. Traditionally, most powder-derived grades of beryllium have been consolidated by a process called "vacuum hot-pressing (VHP)". Please see Figure 2 for a schematic illustration.

This is a directional process, in which "a column of loose beryllium powder is compacted under vacuum by the pressure of the opposed upper and lower punches. The temperature is increased, and the powder is simultaneously compacted and sintered in the final stages of pressing, bringing the billet to final density" [4].

Since the VHP process is directional in nature, it contributes to the anisotropy of the material properties. As a result, Brush has worked with alternative methods of consolidating beryllium powder which will enhance its isotropy. The process which has given the best results is hot isostatic pressing (HIPing).

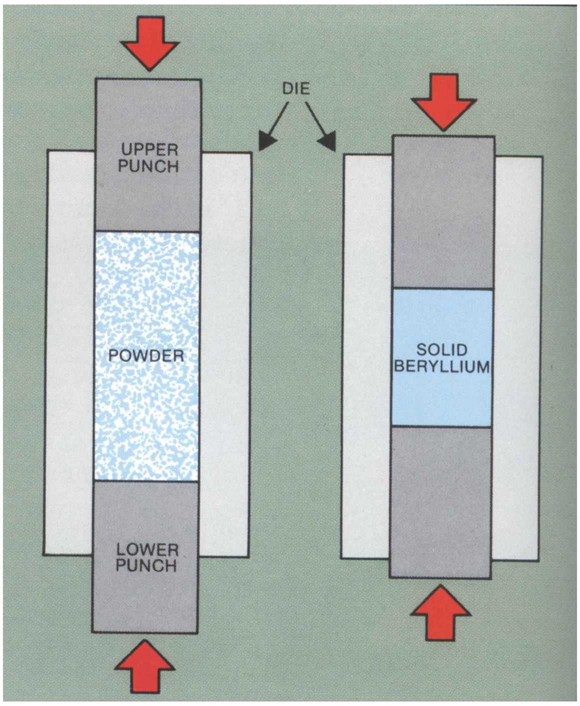

Fig.2. Cutaway schematic view of a vacuum hot press.

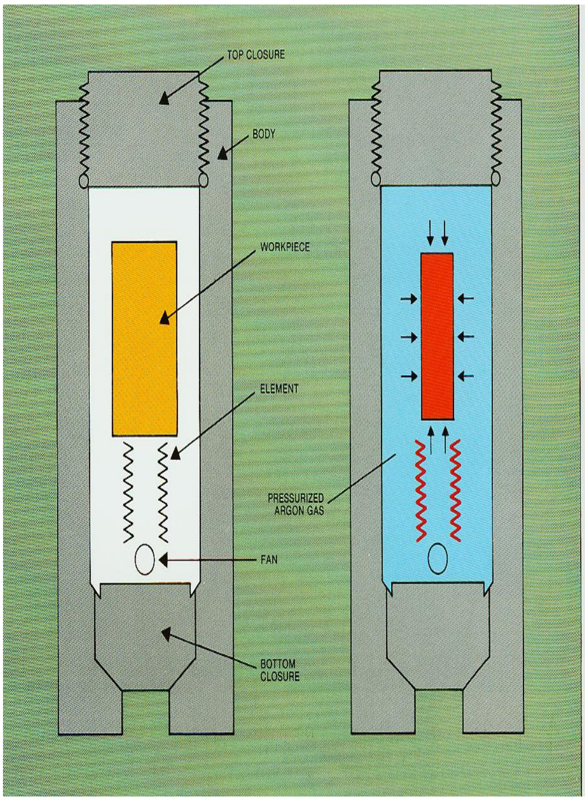

Fig.3. Cutaway schematic view of a hot isostatic press.

In order to prepare beryllium powder for HIPing, it must be sealed in a sheet metal container (can) and have any residual gases evacuated. This loaded can is then placed in the HIP furnace. The HIPing process simultaneously consolidates and sinters the powder by using elevated temperature and pressurized argon gas. The gas exerts pressure uniformly in all directions on the can containing the beryllium powder, as shown in the illustration in Figure3.

To summarize, both powder morphology and the consolidation method will play roles in the isotropy of the resulting material. Both these aspects must be considered when assessing the impact of isotropy as a technical factor in the operational lifetime of beryllium frames in the JMTR.

4.3 Tensile Strength

As previously noted, once exposed to radiation in the test reactor, the beryllium reflector elements will eventually swell, bend, and ultimately even crack. The material's ability to resist this progression will be dependent on its tensile properties, particularly yield strength (YS) and ultimate tensile strength (UTS).

Tensile strength in powder-derived beryllium is affected by the material's chemical composition, consolidation method, and its thermal history. Higher BeO content will result in higher YS and UTS, but lower elongation. Just as the HIP process results in more isotropic properties, it also gives higher tensile properties in general, due to the greater pressure and lower temperature during consolidation compared with the VHP process.

It should also be noted that every Brush structural grade beryllium has a minimum value for each tensile property, which is guaranteed for material which is purchased to the specification. The actual values of these properties for any specific lot of beryllium are reported on certificates which accompany the material. The average of these actual material properties will be referred to as "typical" values in this paper.

4.4 Stress Condition

This technical factor refers to residual stress in the material as a result of the fabrication processes needed to make the finished reflector elements. It is important to machine the parts using speeds and feeds which are appropriate for the material. The use of "progressively diminishing depths of cut" is a technique which will also help to minimize the creation of internal stresses in the machined parts. The technique of diminishing cuts means that the final passes of the cutting tool will each in succession remove less material. For example, the third-to-last cut might remove 0.075mm, the next-to- last cut might remove 0.025mm, and the final cut might only remove a few microns of material. The use of this technique will help to internal stresses in the part as a direct result of the machining process.

Once the machining is complete, thermal cycling the parts in a vacuum furnace at 788°C (1450°F) will relieve any remaining residual stresses.

5. BERYLLIUM GRADE OPTIONS

Brush produces several different grades of beryllium, each of which has been optimized for certain properties or characteristics. These optimizations will typically have particular end-uses in mind, such as structural, instrument, or optical applications. The various grades may also be considered as micro-alloys of beryllium, due to effects which relatively small changes in composition and processing may have on the materials' properties and performance [5-11].

The families of these various grades of beryllium are distinguished by their individual specifications. There are three main families of Brush beryllium grades: 1) structural, 2) instrument, and 3) optical. Wrought types of beryllium, such as rolled sheet and extruded products are excluded from this discussion.

The Brush beryllium grade naming system uses an "S" to designate structural grades, an "I" to designate instrument grades, and an "O" for optical grades. The numbers which follow the letter in the specification name were originally based on the approximate amount beryllium oxide contained in the material. "S-200" would have indicated a structural grade of beryllium with around 2% BeO content. As it happens, S-200 material has evolved and improved over the many decades since its inception, and today's version of this material contains a maximum of 1.5% BeO. All of the other names currently in use reflect fairly accurately the BeO content on a percentage basis (e.g. O-30 indicates an optical grade of beryllium with about 0.3% BeO).

The reason that BeO content is reflected in the name of the grade is because it is one of the key areas which will impact the material's mechanical properties and its desirability for specific end-use applications.

The letters which may occur after the number in the specification name indicate either the current version of the material for VHP grades, such as S-200F versus S-200E, or S-65C versus S-65B. The letter "H" after the number in the specification name indicates that the material is consolidated by the HIP process rather than by vacuum hot-pressing (VHP). Please see Table 1 for a summary of the various Brush beryllium grades grouped by their optimized technical factors and indicating their respective methods of consolidation.

Table 1. Grades & Consolidation Methods

Technical Factor	Be Grade	Consolidation Method
Reference	S-200F	VHP
Purity	S-65	VHP
Isotropy	S-65-H	HIP
	I-70-H	HIP
	O-30-H	HIP
Strength	S-200F-H	HIP
	I-220-H	HIP

5.1 Structural Grades

S-200F beryllium is the current JMTR reflector element material. It has good mechanical strength, but some slight anisotropy due to the fact that it is VHP. Upgrades within the structural grade family would include S-65, which is higher in purity and tensile elongation, but lower in strength. S-65 is also made by VHP, so no difference there. S-200F-H and S-65-H are HIPed versions of S-200F and S-65, respectively. These grades will have higher overall mechanical strengths than their VHP counterparts and improved isotropy between longitudinal and transverse values.

5.2 Instrument Grades

The instrument grades of beryllium are optimized for mechanical strength, generally at the expense of tensile elongation. For most instrument applications, dimensional stability under high g-loading is critically important. These applications include gyroscopes and other inertial guidance components and assemblies.

I-220-H and I-70-H are the two instrument grades which have been considered for JMTR. I-220-H has the highest tensile strength of all Brush grades, but the lowest purity (due to high BeO content). I-70-H is similar in purity to S-65. Both of these materials are HIPed, making them more isotropic than VHP grades.

5.3 Optical Grades

Brush optimizes optical grade material for purity and isotropy. The purity is needed to facilitate the polishing of bare beryllium mirrors. The isotropy is the best of all Brush grades, because this material is not only HIPed, it is also made with inert-gas atomized spherical beryllium powder.

Table 2. Purity & Grain Size Comparison

Technical Factor	Be Grade	Be Assay		Grain Size
		min	typ	max
		(%)		(μm)
Reference	S-200F	98.5	99.1	20
Purity	S-65	99.0	99.4	20
Isotropy	S-65-H	99.0	99.4	15
	I-70-H	99.0	99.4	12
	O-30-H	99.0	99.5	15
Strength	S-200F-H	98.5	99.1	12
	I-220-H	98.0	98.6	15

The primary optical grade considered for the new JMTR reflectors has been O-30-H. This material has the highest typical purity of all grades being considered, and the best isotropy. Its mechanical strength is good, although slightly lower than that of S-200F-H.

6. MATERIALS TESTING PROGRAM

From the information presented in this paper up to this point, one can see that each of the beryllium grades under consideration to replace S-200F has some properties which would be an improvement.

Table 3. Mechanical Property Comparison

Be Grade	YS		UTS		Elongation	
	min	typ	min	typ	min	typ
	(MPa)		(MPa)		(%)	
S-200F	241	260	324	380	2.0	3.0
S-65	206	230	289	386	3.0	5.2
S-65-H	206	280	345	450	2.0	5.1
I-70-H	207	290	345	460	2.0	5.4
O-30-H	297	302	400	425	3.0	3.1

S-200F-H	296	336	414	450	3.0	4.6
I-220-H	345	498	448	577	2.0	3.2

At the same time, it can also be seen this is not true for every property being considered. There is no single grade which is higher in purity, more isotropic, and has better mechanical properties all at the same time. That is why it is necessary to establish which technical factor(s) is (are) the most important to be optimized to extend the lifetime of the JMTR reflector elements.

The aforementioned specialist meeting group, which includes JAEA, Brush and other participants from the nuclear test reactor community is working collectively to put together a thorough materials testing program. The critical issues identified by JAEA are the irradiation temperature (50-150°C), the beryllium consolidation method (VHP versus HIP), and the purity of the material (Be assay). Preliminarily, the specialist group is leaning toward performing the irradiation testing on the following beryllium grades for the reason indicated: S-200F, because it should be included as the reference, since this is the material currently in use; S-65-H should be tested due to its higher purity and better isotropy than S-200F; and I-220-H should be tested due to its higher mechanical strength and better isotropy than S-200F.

Candidate test reactor facilities for performing the irradiation include: the JRR-3M (Japan), the BR2 (Belgium), the ATR (U.S.A.), and the SM-3 (Russia). JAEA has not yet finalized the right choice from among these four candidates, but expects to do so in the near future.

7. CONCLUSION

The proposed testing plan outlined in this paper is an ambitious one, to be sure. It may well represent the most comprehensive irradiated beryllium testing program ever attempted. If operational performance of the JMTR is to be improved, it is clear that finding a way to extend the lifetime of the beryllium reflector elements is a key part in that strategy.

Not only will an improved beryllium reflector mean better overall performance of the reactor, it will signify greater reliability of the JMTR itself by reducing downtime for maintenance and the need to replace the reflectors.

The results of the JMTR Beryllium Reflector Materials Testing Program will not only positively impact the performance of the refurbished JMTR, but will also serve as a new benchmark of performance quality which will have the potential to benefit all high-energy test reactors around the world.

REFERENCES

[1] Japan Atomic Energy Agency, *JMTR Japan Materials Testing Reactor* brochure (year unknown) pp. 3-7.

[2] Japan Atomic Energy Research Institute, *Conceptual Design of the Japan Materials Testing Reactor*, (1964) p. 1056.

[3] Tsuchiya, et al. in this presentation.

[4] Brush Wellman Inc., *Producing Defect-Free Beryllium and Beryllium Oxide* brochure (1985) pp. 12-15.

[5] Brush Wellman Inc., *S-200F Standard Grade Beryllium Block* specification, Revision A, (1987) pp. 1-4.

[6] Brush Wellman Inc., *S-65 Structural Grade Beryllium Block* specification, Revision C, (1987) pp. 1-4.

[7] Brush Wellman Inc., *Material Specification for S-65-H Structural Grade Beryllium Block*, Revision A, (1998) pp. 1-4.

[8] Brush Wellman Inc., *S-200F-H Grade Beryllium Block* specification, Revision B, (1990) pp. 1-4.

[9] Brush Wellman Inc., *Preliminary Material Specification for I-70-H Optical Grade Beryllium Block*, Revision -, (1990) pp. 1-5.

[10] Brush Wellman Inc., *Material Specification for I-220-H Instrument Grade Beryllium Block*, Revision A, (1997) pp. 1-5.

[11] Brush Wellman Inc., *O-30-H Optical Grade Beryllium*, (1998) pp. 1-5.

3.3 Development of U-Mo Research Reactor Fuel in KAERI

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In connection with the back-end option as well as the need for upgrading the performance in the HANARO reactor core, KAERI has developed a rod type U-Mo fuel with a high U density, up to 6 g-U/cc, since the middle of 1990s. Because U-Mo alloy has a ductile nature, the alloy cannot be easily converted to powder by a comminution process. The centrifugal atomization process has been applied to fabricate U-Mo fuel powders. The qualification experiments, the KOMO-1,2, and 3 irradiation tests, for both atomized U-Mo dispersion fuels with a U-loading of up to 6 g-U/cc and monolithic fuels with U-loadings of 6~16 g-U/cc, in the HANARO research reactor were carried out. The first irradiation test for U-Mo dispersion rod fuels revealed the fuels with U-loading of 6.0 g-U/cc not to be acceptable due to a complete interaction between the U-Mo fuel particles and the Al matrix at low burn-ups(~10 at% BU). In the 2nd irradiation test (KOMO-2), rod-type U-Mo dispersion fuels with 4 g-U/cc and 4.5 g-U/cc were irradiated up to 60 at% BU in HANARO. PIE results showed that a whole range of the fuel particles were interacted with the Al matrix at the central region of the fuel meat. The latest KOMO-3 irradiation test has been prepared by reflecting the KOMO-2 test results. In order to alleviate the severe interaction problem, several approaches have been adapted. Remedies for a dispersion fuel include 1) a use of large-sized U-Mo fuel particles to reduce the total interfacial area between the U-Mo and Al matrix, 2) a modification of U-Mo by adding a small amount of Zr into the U-Mo alloy, and 3) an addition of Si into the matrix Al. From the current PIE analysis on the KOMO-3 test fuels, the U-7Mo/Al dispersion fuel (4.5 g-U/cc) contained large-sized U-Mo particles(210-300 μm) which appeared to exhibit a sound irradiation performance. The fuels modified with an alloying (Zr and Si) also showed a considerably lower IL thickness than U-7Mo/Al. The centerline temperature (~204^{o}C) of the large-sized U-7Mo/Al particle fuel was much lower than that of a similar fuel with smaller fuel particles irradiated in the KOMO-2 test(~480^{o}C), demonstrating the advantage of using large-sized U-Mo particles. This supports that a use of a large-sized particle is one of the promising methods worth pursuing further in the future tests combining a alloy modification to deal with the interaction growth problem in a U-Mo/Al dispersion fuel.

Keywords: research reactor, U-Mo, atomization, dispersion fuel, irradiation test

1. INTRODUCTION

The development of a low enriched uranium nuclear fuel has been the center of attraction for research reactors according to the non-proliferation policy under the reduced enrichment for research and test reactors (RERTR) program [1]. Uranium silicide dispersion fuels such as U_3Si_2/Al and U_3Si/Al are being commonly used in about 90% of the world's research reactors due to their stable irradiation behavior [1-2]. However, high uranium density dispersion fuels (8-9 g/cm^3) are required for some high performance research reactors [3,4]. U-Mo alloys have been considered as one of the most promising uranium alloys for a dispersion fuel due to the good irradiation performance of its cubic uranium phase. In connection with the end of the US return policy, an accelerated qualification program to replace a uranium silicide dispersion fuel with a U-Mo dispersion fuel was undertaken by the RERTR program [5].

In KAERI, a necessity for its fuel production at KAERI was proposed in conjunction with the construction of HANARO in the middle of the 1980s. A R&D project for the HANARO fuel fabrication was established in 1987 and it started with a basic study about the technology related to a fuel fabrication. In order to overcome the difficulties of a comminuting process to produce a fuel powder, the development of a atomization process was launched in 1989. Atomized U_3Si powder having a spherical shape could be produced successfully from U-Si alloy melt. It was found that the spherical fuel powder had beneficial effects in the fabrication process as well as its fuel performance [6-7]. Therefore, a U_3Si-Al dispersion fuel by applying atomization technology is now being used as a driver fuel at the HANARO reactor.

U-Mo alloy itself has a ductile nature, which means that it is very difficult to make U-Mo powders by a mechanical comminution of the as-cast U-Mo alloys. In order to simplify the preparation process and improve the properties, a rotating-disk centrifugal atomization method has been developed [8]. The centrifugally atomized powders have some advantages in that the powder has a rapidly solidified γ uranium structure, a relatively narrow particle size distribution, and a spherical shape. Therefore, KAERI has carried out irradiation tests, named the KOMO irradiation tests, to qualify a rod type U-Mo dispersion fuel with U-loadings of ~6 g-U/cc by applying atomization technology for the back-end option of a spent fuel as well as upgrading the HANARO reactor [8].

In this paper, the PIE results of the KOMO irradiation tests on rod type U-Mo dispersion fuels are presented based on the microscopic observations of the irradiated fuels. The KOMO-4 irradiation test plan, by reflecting the previous test results, is also included.

2. IRRADIATION TEST RESULTS OF A ROD TYPE U-Mo/Al DISPERSION FUEL

2.1. THE KOMO-1 TEST

The representative irradiation test conditions of the KOMO fuels are summarized in Table 1. The KOMO-1 irradiation test, in which rod type U-Mo/Al dispersion fuels with U-loadings of 3.4 and 6.0 g-U/cc were included, had been carried out in HANARO since June 26, 2001 and the test assembly was discharged on August 27, 2001 due to a failure occurrence of a fuel rod [9]. The maximum linear power was evaluated to be about 112 kW/m, in which the maximum BOL temperature of a fuel rod was estimated to be 276 ^{o}C during an irradiation.

Table 1. The KOMO irradiation test conditions.

	Fuel Composition	Particle Size (μm)	Matrix	U-loading (g-U/cc)	Max. LP (kW/m)	Max. BOL T. (°C)	Max. BU (at%U^{235})	Status
KOMO-1	U-7Mo U-9Mo	<150	Pure Al	3.4 and 6.0	112	276	~13	Irr. Stopped (Fuel Failure) 2001.7~8
KOMO-2	U-7Mo U-9Mo	mostly <150	Pure Al	4.0 and 4.5	108	196	68	Irr. Test + PIE Completed 2002.1-2003.1
KOMO-3	U-7Mo U-7Mo-1Zr U-7Mo-0.2Si	100-210 210-300 300-425	Pure Al Al-0.4Si Al-2Si	4.5	95	181	66	Irr. Test Completed PIE in Progress 2006.2-2007.7
KOMO-4	U-7Mo U-7Mo-1Zr U-7Mo-1Ti	Mostly 210-300	mostly Al-2Si Al-5Si	4.5 ~ 5.5	105	200~ 210	65~70	Planned in 2008

The first irradiation test (KOMO-1) revealed that the U-Mo/Al dispersion fuel with a U-loading of 6.0 g-U/cc was not acceptable due to a complete interaction between the U-Mo particles and the Al matrix. Fig. 1 shows a typical example of a heavily interacted microstructure of the atomized U-7Mo/Al dispersion fuel (428A-H5) sample after ~13 at% U-235 BU. The parameter with the strongest affect was concluded as the fuel temperature. Intermetallic compounds in the form of $(U,Mo)Al_x$ are formed as a result of the irradiation-induced diffusion reaction. Because the intermetallic compounds are less dense than the combined reactants, the volume of the fuel meat increases after the interaction. In addition to the effect on the swelling performance, the reaction layers between the U-Mo and the Al matrix induce a degradation of the thermal properties of the U-Mo/Al dispersion fuels [10]. A further PIE revealed a completely interacted region in the highly U-loaded fuels with 6.0 g-U/cc, where some voids and cracks were observed. The swelling was measured to be 17.1 % at a maximum. However, the fuels with a U-density of 3.4 g-U/cc showed a smaller swelling of 7.8 % and much thinner interaction layer thicknesses for various regions.

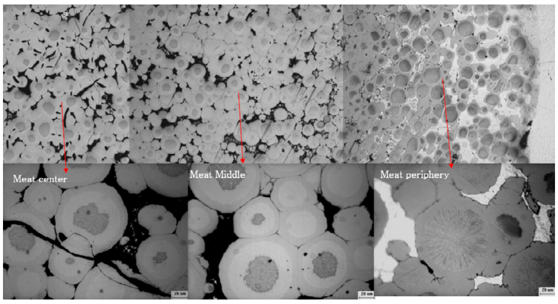

Fig. 1. Microstructure of the cross section of the irradiated U-7Mo/Al dispersion fuel meat (428A-H5).

2.2. THE KOMO-2 TEST

In the 2nd irradiation test, fuels with a lower U-density than the KOMO-1 fuels were designed in order to reduce the fuel centerline temperature. Uranium loading in the dispersion fuel was changed from 6.0 g-U/cc to 4.0 and 4.5 g-U/cc, in which the max. BOL temperature was less than 200^{o}C by slightly reducing the linear power. The irradiation test bundle was loaded at the OR5 hole of HANARO on Jan. 9, 2003 and discharged on Jan. 27, 2004. Average burnup was calculated as 60.8 at.%U-235 and the maximum local burnup was 71.2 at.% of U-235 [11,12].

Post-irradiation examination of the KOMO-2 fuels revealled various microstructural features with the burn-up, linear power and location in the fuel meat. In the case of the U-7Mo/Al dispersion fuel with 4.5 g-U/cc(494-H2) after a 68 at% U-235 burnup, as shown in Fig. 2, all the fuel particles were extensively interacted with the Al matrix except for the periphery region in the fuel meat.

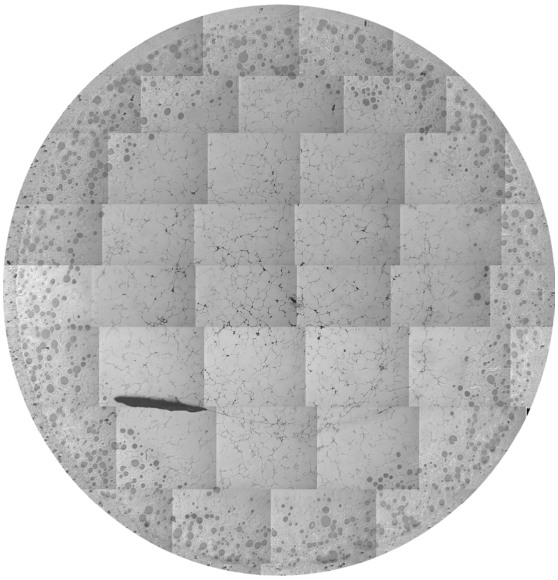

Fig. 2. Microstructure of the cross section of the irradiated U-7Mo/Al dispersion fuel meat (494A-H2).

Fuel temperature history of the rod-type U-Mo/Al dispersion fuel was calculated by using a modified correlation of the interaction layer growth, as shown in equation (1). Fuel temperature of the rod-type dispersion fuel revealled a strong feedback behavior due to the lower thermal conductivity of the interaction layer. From the result of the temperature histories for the top section of the 494H2 rod, it appeared that the centerline temperature increased sharply after a 20% U-235 burnup and reached a peak temperature at around 480^{o}C after a 40% U-235 burnup [13]. It was also calculated that a dispersion of the large fuel particles (>150 μm) was effective in mitigating the thermal degradation associated with the interaction layer.

$$y^2 = A \cdot \dot{f}^{1/2} \cdot \Delta t \cdot \exp\left(\frac{-Q}{RT}\right) \qquad (1)$$

For an average fuel particle size smaller than 75 μm, an undesirable temperature jump that can lead to a melting of the Al matrix was predicted, whereas for a fuel particle size larger than 150 μm, the fuel temperature gradually decreased toward the end of its life without a break-away temperature increase.

2.3. THE KOMO-3 TEST

The latest KOMO-3 irradiation test has been prepared by reflecting the KOMO-2 test results [14]. In order to alleviate the severe interaction problem, several approaches have been adapted. Remedies for the dispersion fuel included 1) a use of large-sized U-Mo fuel particles to reduce the total interfacial area between the U-Mo and the Al matrix, 2) a modification of U-Mo by adding a small amount of Zr into the U-Mo alloy, and 3) an addition of a small amount of Si into the matrix Al.

The irradiation test bundle consisting of 12 different fuel rods, as given in Table 2, was loaded at the OR5 test hole of the HANARO reactor on February 8, 2006 and discharged on July 2, 2007 after 206 EFPD (effective full power days) of an irradiation. The maximum linear power of the dispersion fuels with the U-density of 4.5 g-U/cc was slightly lower than that of the KOMO-2 fuels with the same U-density. The maximum temperature in the dispersion fuel meat was estimated in the range of 170-180^{o}C. The average and peak burn-ups were calculated as 54 at.%U-235 and ~68 at.% of U-235, respectively.

Table 2. . Irradiation conditions of the KOMO-3 test fuels.

No	Fuel I.D.	Fuel Material	Fuel Particle Size	Tmax(BOL) (°C)	Max. BU (at%U-235)	Max. Lin. Power (kW/m)
1	567-MC1	U-7Mo(Multi-C)	ϕ 2.0 mm×4 ea	322	39.63	73.91
2	569-MC1	U-7Mo-0.2Si(Multi-C)	ϕ 2.0 mm×4 ea	168	40.16	77.42
3	557-SD1	U-7Mo/Al	105-210 μm	175	65.22	94.0
4	557-MD1	U-7Mo/Al	210-300 μm	170	63.09	90.32
5	557-MD2	U-7Mo/Al(No oxid.)	210-300 μm	181	65.65	98.40
6	557-LS1	U-7Mo/Al-0.4Si	210-300 μm	181	62.39	88.78
7	557-HS1	U-7Mo/Al-2Si	210-300 μm	174	64.87	93.57
8	557-LD1	U-7Mo/Al	300-425 μm	181	66.19	98.49
9	558-MD1	U-7Mo-1.0Zr/Al	210-300 μm	175	64.20	94.07
10	559-MD2	U-7Mo-0.2Si/Al	210-300 μm	177	64.56	95.62
11	560-SD2	U3Si2/Al	Under 150 μm	175	68.82	94.21
12	568-T1	U-7Mo(tube)	ϕ 6.35 ×T0.7XL5	262		88.94

Compared to the previous KOMO-2 result (494-H2) where U-Mo powders with normal size ranges(<150 μm) were used[2-3], as shown in Fig. 3, the U-7Mo/Al dispersion fuel(4.5 g-U/cc, 557-MD1) with large-sized U-Mo particles(210-300 μm) appeared to exhibit a sound irradiation performance from the evolution of the microstructures. Interaction layers(IL) have not been developed extensively, even at the fuel meat center.

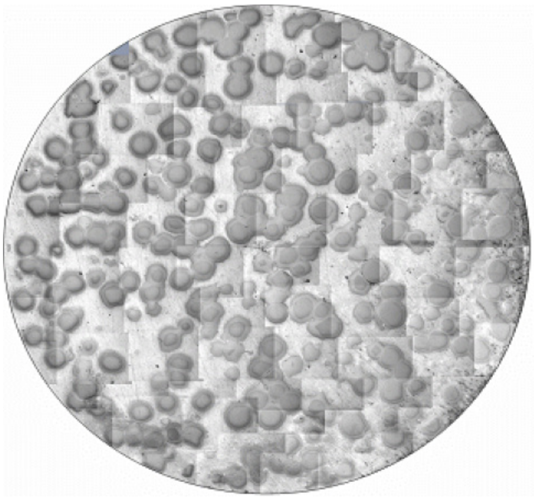

Fig. 3. Microstructure of the cross section of the irradiated U-7Mo/Al dispersion fuel meat (557-MD1).

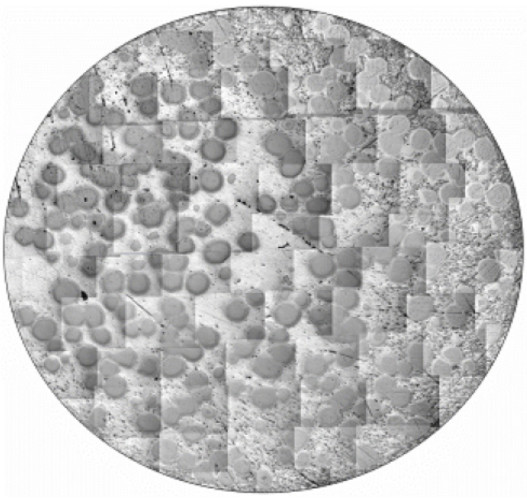

Fig. 4. Microstructure of the cross section of the irradiated U-7Mo/Al-2Si dispersion fuel meat (557-HS1).

The use of large-sized U-Mo particles reduces the total interfacial area between the U-Mo and the Al matrix which results in a relatively small fraction of the IL. This in turn leads to a smaller increase of the fuel temperature during an irradiation than the KOMO-2 case. Similar microstructures as the U-7Mo/Al fuel were observed, in general, on the U-7Mo-1Zr/Al and U-7Mo/Al-2Si fuels (see Fig.4). It appeared that U-7Mo/Al has a IL thickness of ~70 μm at the center and ~25 μm at the periphery, whereas U-7Mo-1Zr and U-7Mo/Al-2Si have decreased IL thicknesses of 40-50 μm.

Fig. 5 shows the swelling as a function of the BOL temperature of the fuel rods. Similar to the KOMO-2 test, swelling of the KOMO-3 test fuels can be correlated as a linear function of the BOL temperature and appeared to be slightly smaller than the KOMO-2 result because of the smaller IL fraction.

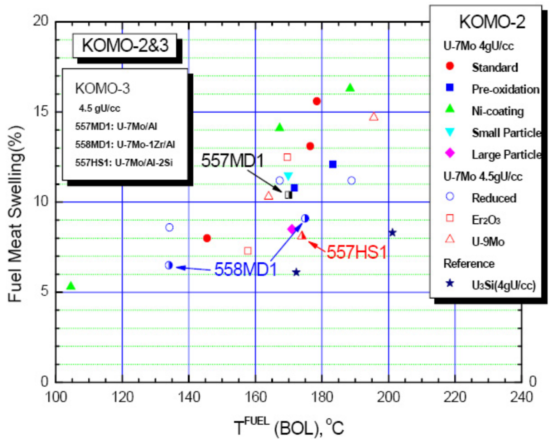

Fig. 5. Swelling vs. BOL T. of the irradiated KOMO-2&3 fuels.

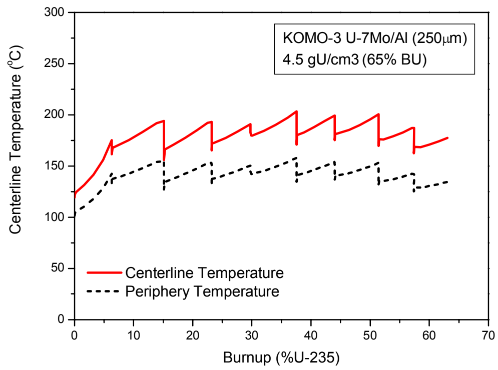

Fig. 6. Centerline and periphery temperature histories with the BU in the 63.2% BU 557-MD1 rod.

Figure 6 shows the predicted temperature histories of the centerline and periphery temperatures in the U-7Mo/Al dispersion fuel rod(557-MD1) irradiated to 63.2% BU. The centerline temperature reached a peak temperature at around 204°C after a 37% BU. The temperature at the EOL was 177°C. The prediction for a similar fuel (494-H2, 4.5 g-U/cc) from the KOMO-2 test revealled that the peak temperature was ~480°C. Since the use of large-sized U-Mo particles was the only difference for the current case from the KOMO-2 case, the much lower fuel temperature is attributed to the fuel particle size. Therefore, the use of large-sized fuel particles can be one of the promising methods to solve the interaction growth problem in U-Mo/Al. The effect of a large-sized U-Mo fuel can be enhanced even further by simultaneous modifications of the fuel with Zr or Ti and the matrix with Si.

3. THE KOMO-4 IRRADIATION TEST PLAN

U-Mo dispersion fuels for the KOMO-4 irradiation test are being designed by reflecting the previous KOMO test results. In order to find the ultimate limit of the linear power in a rod type U-Mo dispersion fuel with U-loadings of 5~6 g-U/cc, the KOMO-4 fuels are going to include fuels containing 1) Large-sized U-Mo powder, mostly 210~300 μm in size, 2) Al-Si alloy matrix where the Si content will be 5 wt% at a maximum, 3) Alloy modification of U-7Mo with Zr and Ti. The peak linear power will be 105~110 kW/m, which can produce a peak BOL temperature of 200~210°C at the fuel meat center. The test bundle is going to be fabricated in the near future and it is expected that the irradiation test will start by the end of 2008.

Table 3. Irradiation conditions of the KOMO-4 test fuels.

No.	Fuel Powder	F. Particle Size (mm)	Matrix	Fuel Meat Length (mm)	U-density (g-U/cc)	BOL T (°C) (max)	Linear Power (KW/m) (max)	Remarks
1	U-7Mo	105-210	Al	100	4.5	190	100	Reference
2		210-300						
3		300-425						
4	U-7Mo	210-300	Al-2Si	300	5	190	100	Si Effect (2wt%)
5	U-7Mo	210-300	Al-5Si	300	5~5.5	200~210	105~110	Si Effect (5wt%) + Ternary Alloy(Zr, Ti)
6	U-7Mo-1Zr							
7	U-7Mo-1Ti							
8	U-7Mo	<150	Al-5Si	200	5	190	100	Si Effect under normal fuel particle size distribution
9	U-7Mo	210-300	Al-5Si	150	5	200	105	Pre–Interaction layer at Particle Surface by Annealing Treatment
10	U-7Mo-1Zr							
11	U-7Mo-1Ti							
12	U-7Mo	Multi-core or Tube	Al	150	6		80~90	Monolithic (2 mm Dia. X 4 EA)

4. CONCLUSION

From the previous qualification experiments, the KOMO-1,2, and 3 irradiation tests, of rod type U-Mo dispersion fuels with a U-loading of up to 6 g-U/cc based on atomization technology in the HANARO research reactor, the use of large-sized fuel particles can be one of the promising methods to solve the extensive interaction growth problem between the U-Mo fuel and the Al matrix. Further improvement of the relatively reduced IL growth was observed, in general, on the U-7Mo-1Zr/Al and U-7Mo/Al-2Si fuels. The effect of a large-sized U-Mo fuel can be enhanced even further by simultaneous modifications of the fuel with Zr or Ti and the matrix with Si. The preparation of the KOMO-4 test fuels are now undergoing to find the ultimate limit of the linear power in a rod type U-Mo dispersion fuel with U-loadings of 5~6 g-U/cc.

5. ACKNOWLEDGMENTS

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6. REFERENCES

[1] G.L. Hofman, J. Nucl. Mater., 140(1986)256.

[2]. R.C. Birther, C.W. Allen, L.E. Rehn, G.L. Hofman, J. Nucl. Mater., 152(1988)73.

[3] J.L. Snelgrove, G.L. Hofman, C.L. Trybus, T.C. Wiencek, in: Proceedings of the 18th International Meeting on RERTR, Seoul, Korea, 1996.

[4] J.L. Snelgrove, G.L. Hofman, M.K. Meyer, C.L. Trybus, T.C. Wiencek, Nucl. Eng. Des. 178 (1997) 119.

[5] A.Travelli, in: Proceedings of the 14th International Meeting on RERTR, Sao Paulo, Brazil, 1998.

[6] C.K.Kim, J.M. Park, H.J. Ryu, Nuclear Engineering and technology, Vol.39 Vo.5 (2007) 617.

[7] C.K. Kim et al., "Effect of Particle Shape and Distribution on Thermal and Electrical Conductivities in U_3Si -Al Dispersion Fuels", Journal of Nuclear Materials, Vol. 209, 1994.

[8] K.H. Kim et al., J. Kor. Nuc. Soc. 33(2001)365.

[9] C.K. Kim, et al., "Status of the Qualification Program for Atomized U-Mo Dispersion Rod Type Fuel in Korea," The proceedings of the 6th RRFM meeting, Ghent, Belgium, 2002.

[10] S. H. Lee, J. M. Park, and C. K. Kim, International Journal of Thermophysics, Vol. 28, No. 5 (2007) 1578.

[11] J.M. Park, H.J. Ryu, Y.S. Lee, D.B. Lee, S.J. Oh, B.O. Yoo, Y.H. Jung, D.S. Sohn, and C.K. Kim, in: Proceedings 26th International Meeting on RERTR, Vienna, Austria, 2004.

[12] C.K. Kim, H.J. Ryu, D.B. Lee, S.J Oh, K.H Kim, J.M Park, Y.S. Choo, D.G. Park, H.T. Chae, C.S. Lee, D.S Sohn, in: Proceedings 26th International Meeting on RERTR, Vienna, Austria, 2004.

[13] Ho Jin Ryu, Jong Man Park, Hee Taek Chae, Chang Kyu Kim, in: Proceedings 27th International Meeting on RERTR, Boston, USA, 2005.

[14] J.M. Park, H.J. Ryu, Y.S. Lee, B.O. Yoo, Y.H. Jung, C.K. Kim, and Y.S. Kim, The proceedings of the 12th RRFM meeting, Hamburg, Germany, 2008.

4. Irradiation Technology (including PIE Technology)

4.1 JMTR Strategy of Restart and Dosimetry for Standardization of Irradiation Technology

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Calculated neutron flux/fluence was verified against measurements of irradiated fluence monitors. With regard to gamma dose, calculated gamma heating rate were verified against measurements of the nuclear heating evaluation capsule which was developed in order to measure nuclear heating rate. It was confirmed that the calculated fast and thermal neutron flux/fluence agreed with measurements within ±10%, ±30%, respectively, and the calculated gamma dose agreed within -3～+21%. The attempt to improve accuracy of calculated irradiation parameter, especially thermal neutron flux, is conducted for restart of JMTR on FY2011.

Keywords: JMTR, MCNP, Fluence monitor, Neutron flux, Gamma heating rate.

1. INTRODUCTION

The JMTR (50MWth) will restart on FY2011 after the refurbishment for new irradiation researches and utilizations. Current irradiation research on aging of LWR core internal materials, specifically IASCC, tritium release of fusion blankets under neutron irradiation, etc. generally needs more accurate prediction, control, and evaluation of irradiation parameter such as neutron flux/fluence, gamma dose, etc.

In the neutron irradiation tests of nuclear materials, it is difficult to evaluate neutron flux and energy spectrum at each position of the specimen installed in the complicated capsule (i.e. irradiation rig) by conventional diffusion or transport calculation codes due to their limitation of modeling capability. An analysis procedure using Monte Carlo method has been therefore introduced in the JMTR to evaluate irradiation field at each specimen by using MCNP[1] code, which has a capability to model complicated structure of the capsule directly.

The effects of neutron energy spectrum and the influence of irradiation damage due to gamma-rays dose have attracted special interest in current irradiation researches. Furthermore, accurate prediction and control of temperature at specimens in irradiation capsules under high neutron flux and high gamma heating rate conditions are required to aging research on LWRs and development of fusion reactor.

Current situation was therefore examined as improvement of the accuracy of calculated irradiation parameters of neutron flux/fluence[2-4] and gamma dose[5-6]. Furthermore, the attempt to improve accuracy of calculated irradiation parameter, especially thermal neutron flux, is conducted for restart of JMTR on FY2011.

2. JMTR

The JMTR is a tank-in-pool type reactor with thermal power of 50MW and both coolant and moderator are light water. The typical core configuration is shown in Fig. 1. The reactor core, which is 1560mm in diameter and 750mm in effective height, consists of fuel elements, control rod, reflectors and H-shaped beryllium frame. Each reflector element has irradiation hole, which is loaded with a capsule for irradiation tests or a solid plug of the same material as the reflector element. The H-shaped beryllium frame has also irradiation holes. An irradiation channel can be chosen among 195 possible positions in the core.

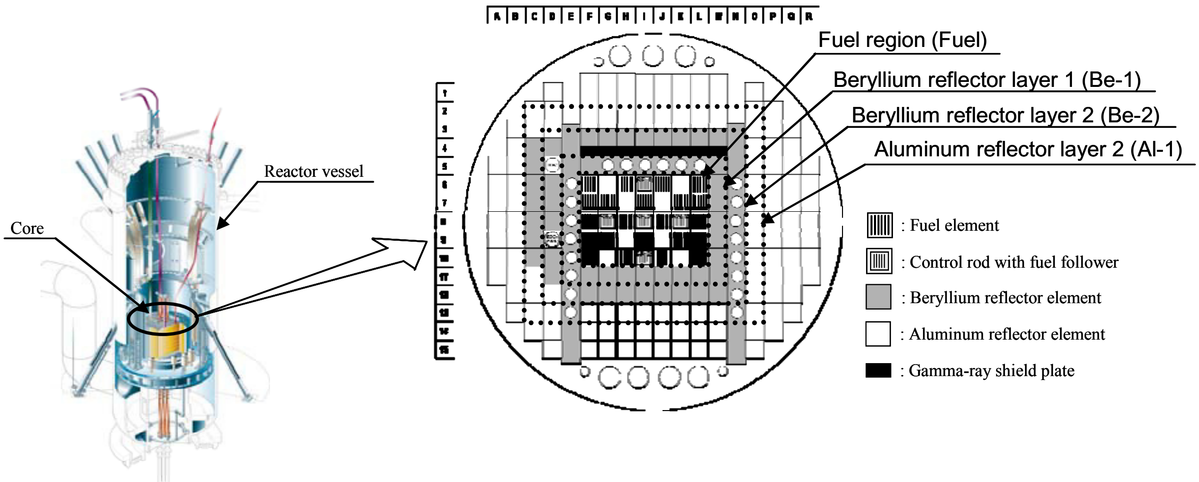


Fig. 1. Typical core configuration of JMTR

3. Current Evaluation of Neutron Flux/Fluence

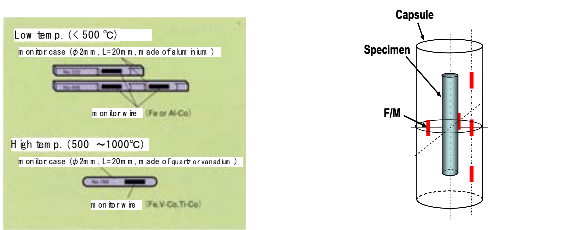
3.1 Measurement

Neutron flux/fluence at local positions have been measured by using the fluence monitors. The typical fluence monitors used in the JMTR are illustrated in Fig. 2(a). ^{54}Fe(n,p)^{54}Mn reaction of iron and ^{59}Co(n,γ)^{60}Co reaction of aluminum-cobalt (0.11wt% of Cobalt) wires are used as fast and thermal neutron flux/fluence respectively. As usual practice, five fluence monitors (include coupled iron and aluminum-cobalt wires) are prepared for one irradiation capsule (φ 40～ φ 60mm x 880mm, see Fig. 2(b)).

After irradiation tests, radiation activities are measured with the germanium detector. The reaction rates are calculated by using radiation activities, and neutron flux/fluence are obtained from the reaction rates and the weighted neutron cross section with calculated neutron spectrum at fluence monitor position.

3.2 Calculation

Neutronic calculations are conducted using the Monte Carlo code MCNP(ver. 4B) with continuous energy neutron cross section library FSXLIBJ3R2[7] (derived from JENDL3.2) and thermal S(α,β) libraries of ENDF-B/III [1]. The JMTR core for each operation cycle is modeled and whole core of JMTR include irradiation capsules are modeled in detail. Fast and thermal neutron fluxes at each sample position are calculated by KCODE option in MCNP.



(a) Fluence monitor. (b) Typical arrangement of flence

Fig. 2. Fluence monitor for irradiation tests of JMTR

3.3 Results

The measured and calculated neutron flux data [8-10] are shown in Fig. 3, using the measured and calculated data of fluence monitors (fast neutron: 576 items (132 capsules), thermal neutron: 457 items (103 capsules)) from FY1998 to FY2007. The calculated fast neutron fluxes agreed with measured ones within ±10% error. The other hand, the calculated thermal neutron fluxes agreed with measured ones within ±30% error.

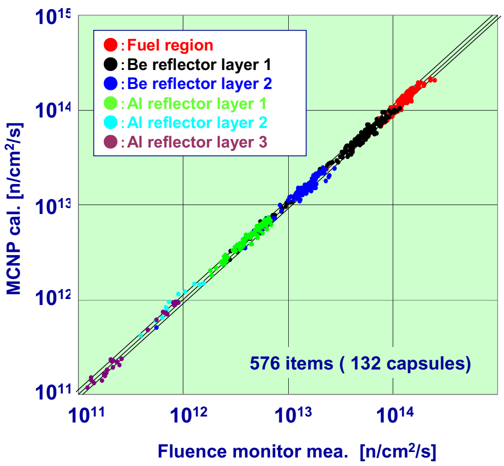
The calculated thermal neutron fluxes tend to overestimate the measured ones as contrasted with the fast neutron fluxes.

3.4 Sensitive analysis

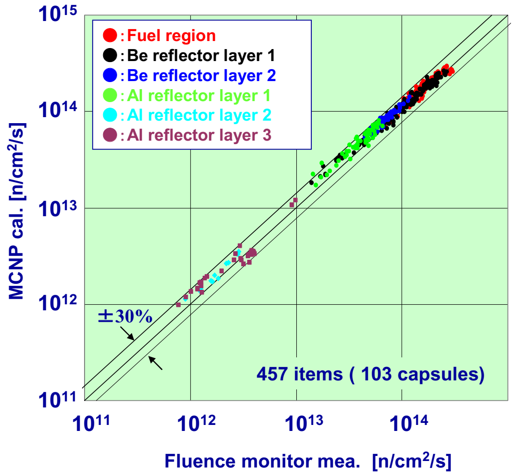
The accuracy of calculated thermal neutron flux was examined from the viewpoint of irradiation region in the JMTR core. As the result, calculated thermal neutron flux tends to overestimate in beryllium reflector layer 2 and aluminum reflector layer 1 as comparison with the other irradiation regions.

Relating to cross section data of thermal energy range, thermal S(α,β) library of ENDF-B/III in materials of light water and beryllium was used to the calculation of JMTR core.

Sensitive analysis[14] was therefore conducted concerning the thermal S(α,β) libraries, at first. The thermal S(α,β) libraries of light water and beryllium form ENDF-B/VII.0 and JEFF-3.1 were compiled by NJOY99 for JMTR (temperature of approx. 300 K). The calculation model of JMTR exclusive of



(a) Fast neutron flux (E>1MeV)



(b) Thermal neutron flux (E<0.683eV)

Fig. 3. Comparison of measured and calculated neutron flux at fluence monitor positions from FY1998 to FY2007

Table 1 Sensitive analysis of thermal S(α, β) libraries of ENDF/B-III, ENDF/B-VII and JEF3.1

Region	Irradiation hole	Fast neutron flux (E > 1MeV) [n/cm²/s]			ratio		Thermal neutron flux (E < 0.683eV) [n/cm²/s]			ratio	
		ENDF/B-III(B-III)	ENDF/B-VII(B-VII)	JEFF-3.1(JEF)	B-VII/B-III	JEF/B-III	ENDF/B-III(B-III)	ENDF/B-VII(B-VII)	JEFF-3.1(JEF)	B-VII/B-III	JEF/B-III
Fuel	J-9	1.50E+14	1.44E+14	1.43E+14	0.96	0.95	1.66E+14	1.59E+14	1.60E+14	0.96	0.96
Fuel	H-7	1.42E+14	1.42E+14	1.43E+14	1.00	1.00	1.55E+14	1.52E+14	1.58E+14	0.98	1.02
Fuel	K-10	1.10E+14	1.07E+14	1.07E+14	0.98	0.97	1.95E+14	1.89E+14	1.92E+14	0.97	0.98
Fuel	G-6	1.01E+14	1.03E+14	1.05E+14	1.02	1.04	1.67E+14	1.65E+14	1.73E+14	0.99	1.04
Be-1	M-8	5.67E+13	5.42E+13	5.54E+13	0.96	0.98	3.59E+14	3.48E+14	3.48E+14	0.97	0.97
Be-1	E-8	5.51E+13	5.72E+13	5.74E+13	1.04	1.04	3.89E+14	3.97E+14	4.08E+14	1.02	1.05
Be-1	I-11	4.51E+13	4.41E+13	4.42E+13	0.98	0.98	3.44E+14	3.35E+14	3.47E+14	0.97	1.01
Be-1	I-5	4.29E+13	4.24E+13	4.13E+13	0.99	0.96	2.81E+14	2.81E+14	2.80E+14	1.00	0.99
Be-2	N-8	1.22E+13	1.19E+13	1.23E+13	0.98	1.01	2.45E+14	2.41E+14	2.40E+14	0.99	0.98
Be-2	D-8	1.21E+13	1.23E+13	1.28E+13	1.02	1.06	3.40E+14	3.42E+14	3.48E+14	1.01	1.03
Be-2	I-12	1.04E+13	1.07E+13	1.01E+13	1.03	0.97	2.32E+14	2.35E+14	2.36E+14	1.01	1.01
Be-2	I-4	6.36E+12	6.60E+12	6.31E+12	1.04	0.99	1.58E+14	1.59E+14	1.55E+14	1.00	0.98
Al-1	O-8	4.36E+12	4.02E+12	4.28E+12	0.92	0.98	7.55E+13	7.41E+13	7.32E+13	0.98	0.97
Al-1	I-13	3.90E+12	3.98E+12	3.48E+12	1.02	0.89	7.93E+13	7.70E+13	7.69E+13	0.97	0.97
Al-1	C-8	2.81E+12	3.01E+12	2.98E+12	1.07	1.06	1.53E+14	1.56E+14	1.58E+14	1.02	1.03
Al-1	I-3	2.63E+12	2.32E+12	2.21E+12	0.88	0.84	5.09E+13	4.95E+13	5.00E+13	0.97	0.98

irradiation capsules was selected for the calculations.

Calculation results were tabulated in Table 1. The values of calculated thermal neutron fluxes by thermal S(α,β) libraries of ENDF-B/VII.0 and JEFF-3.1 were about the same as ones of the ENDF-B/III. As the results, it was confirmed that difference of thermal S(α,β) libraries was not effected by evaluation of thermal neutron fluxes.

4. Verification of Calculated Gamma Heating Rate

4.1 Measurement

An irradiation capsule was developed in order to evaluate the gamma heating rate (generated from interaction between materials and neutron or gamma ray) in the JMTR core by measuring temperature. The schematic of the irradiation capsule was shown in Fig. 4. The irradiation capsule has a simple structure to reduce uncertainties in the thermal calculation.

As the test specimens for measurement of gamma heating rate, four materials (iron, SS316, zircaloy-2, and titanium) were used to examine dependence on the materials. The irradiation tests were conducted at the 141-147 operation cycle in JMTR from 2001 to 2002. The loading positions of the irradiation capsule were shown in Fig.5.

4.2 Calculation

The neutron-gamma coupled calculations by MCNP were applied to evaluate the gamma heating in the JMTR core. Contributions of prompt fission, capture, fission products decay gamma-rays and prompt fission neutrons to heat deposition were considered. For the fission products decay gamma-ray spectrum, following formula [11] was applied.

The neutron-gamma coupled calculations by MCNP were applied to evaluate the gamma heating in the JMTR core. Contributions of prompt fission, capture, fission products decay gamma-rays and prompt fission neutrons to heat deposition were considered. For the fission products decay gamma-ray spectrum, following formula [11] was applied.

$$N(E) = 6.0\exp(-1.1E)\ [MeV^{-1}] \qquad (1)$$

The continuous energy cross section libraries MCPLIB[1] (derived from DLC-7E) for the gamma rays and FSXLIBJ3R2 for the neutron was used. The whole core of the JMTR with a configuration in each operation cycles including irradiation capsules was modeled in detail. Thermal calculations were carried out to obtain the temperature at each thermocouple by the 1-D geometry capsule design code GENGTC[12] using gamma heating rate calculated by MCNP.

4.3 Results and Discussion

Calculated gamma heating rates were shown in Table 2. The measured and calculated temperatures are shown in Fig. 6. As the results, calculated temperature data agreed with measured data within -3～+21% error. Contributions of prompt fission neutron to heat deposition were less than 1.5% of the total heat depositions in these calculations [13].

Therefore, it was confirm that the specimen temperature in irradiation capsules could be evaluated by practical accuracy in this procedure.

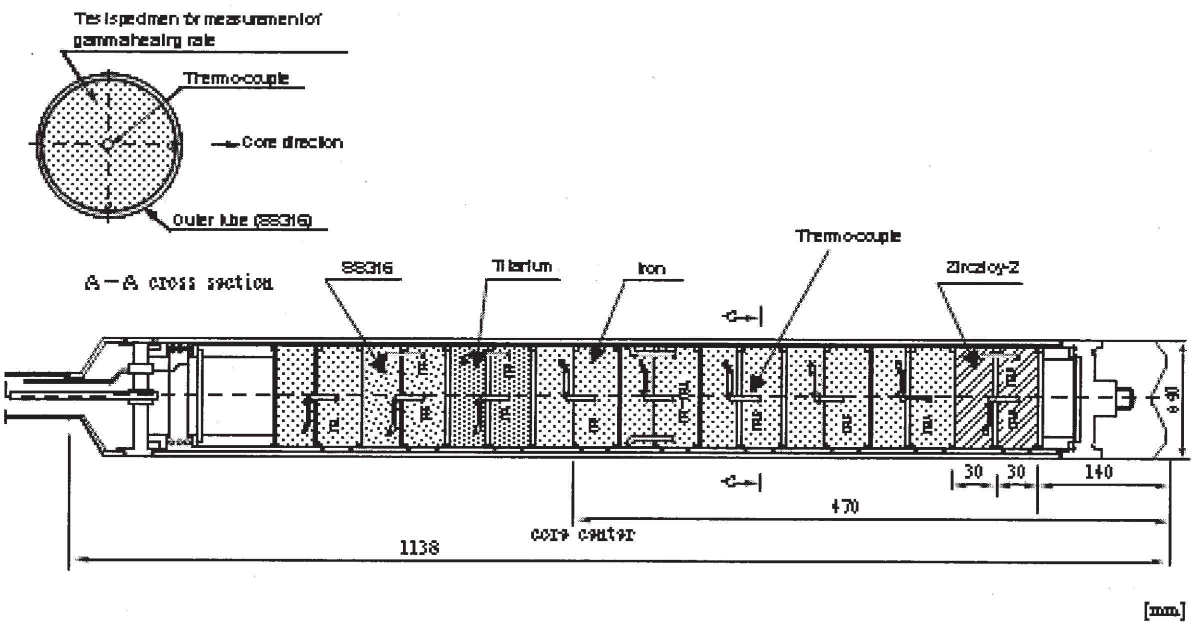


Fig. 4. Irradiation capsule for evaluation of gamma heating rate.

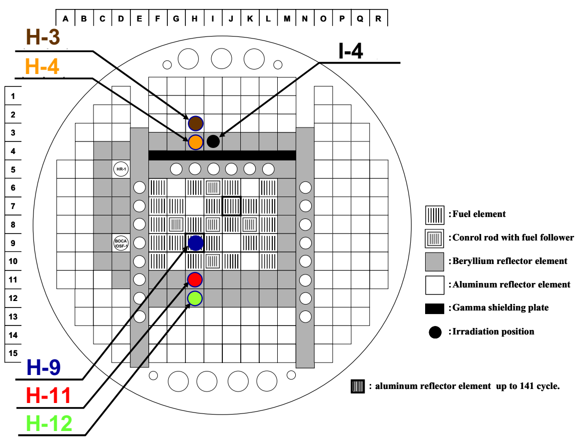


Fig. 5 Loading position of Irradiation capsule for evaluation of gamma heating rate.

Table 2 Calculated gamma heating rate in specimens

Position	Gamma heating rate [W/g]					
	H-9	H-11	H-12	I-4	H-4	H-3
No.1 Fe	4.62	2.72	1.65	0.87	0.71	0.43
No.2 SS316	5.73	3.38	1.97	1.14	0.89	0.51
No.3 Fe	6.98	4.41	2.54	1.45	1.13	0.63
No.4 Fe	6.72	3.96	2.25	1.27	1.00	0.55
No.5 Fe	7.06	4.23	2.29	1.29	1.01	0.55
No.6 Fe	7.29	3.94	2.23	1.33	1.00	0.53
No.7 Fe	7.08	3.78	2.06	1.21	0.96	0.49
No.8 Fe	6.39	3.34	1.79	1.13	0.83	0.42
Np.9 Zry-2	5.51	2.47	1.28	0.64	0.51	0.30

5. SUMMARY

This paper presents verification results of neutron flux and gamma heating for the irradiation tests of the JMTR. Based on the MCNP code and the whole core model, the neutron-gamma coupled calculations were applied on the JMTR to evaluate neutron flux and gamma heating at sample positions in the irradiation capsules.

As the verification results of neutron flux, it was confirmed that the calculated fast and thermal neutron flux were agreed with measured ones within ±10%, ±30%, respectively. It was found that the accuracy of thermal neutron flux was relevant to irradiation regions in the JMTR. As the sensitive analyses, difference of thermal S(α,β) libraries was not effected by evaluation of thermal neutron fluxes.

Concerning gamma heating, it was confirmed that the calculated temperature was agreed with measured ones within -3～+21%.

The evaluations of neutron flux/fluence and specimens temperature with practical accuracy are therefore possible in the irradiation test of the JMTR.

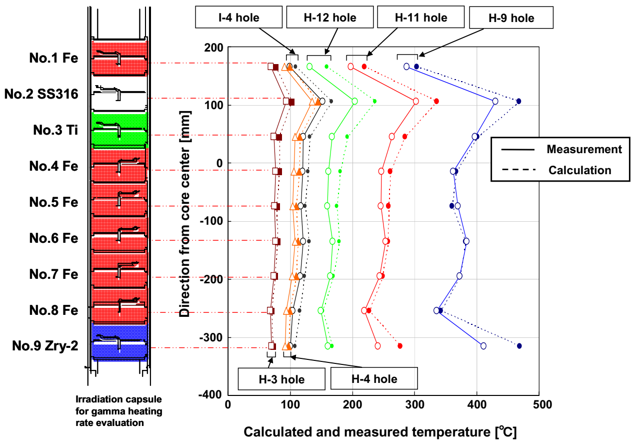


Fig. 6. Comparison of measured and calculated temperature at thermocouple positions for evaluation of gamma heating rate

REFERENCES

[1] J.F.BRIESMEISTER,(Ed.), "MCNP-A General Monte Carlo N-Particle Transport Code, Version 4B", LA-12625-M, (1997).

[2] Y.Nagao, "Core calculation of JMTR", JAERI-Review 97-003, pp.74-95, (1997) [in Japanese].

[3] Y.Nagao, "Core calculation of JMTR", JAERI-Review 98-010, pp.128-165, (1998) [in Japanese].

[4] Y.Nagao, "Core calculation of the JMTR using MCNP", JAERI-Conf 2000-018, pp.156-167, (2000) [in Japanese].

[5] Nihon Gensiryoku Gakkai, "Monte Carlo hou niyoru ryuusi simyuresyon no genjyou to kadai", (2002) [in Japanese].

[6] Y.Nagao, Y.Itabashi, Y.Komori, et al., "Evaluation of neutron and gamma heating for irradiation tests of JMTR", 2002 Annual meeting of the atomic energy society of Japan, K7, (2002) [in Japanese]

[7] K.Kosako, F.Maekawa, Y.Oyama, et al., "FSXLIB-J3R2: A Continuous Energy Cross Section Library for MCNP Based on JENDL-3.2", JAERI-Data/Code 94-020, (1994).

[8] Y.Nagao, M.Nakamich, M.Tsuchiya, et al., "Verification of tritium production evaluation procedure using Monte Carlo code MCNP for in-pile test of fusion blanket with JMTR", Fusion Eng. Des. 51-52, pp.829-835, (2000)

[9] Department of JMTR JAERI, "JMTR jihou No.127", pp.10-15, (2000) [in Japanese]

[10] Y. Nagao, private communication, (2001)

[11] Everitt P. Blozard(Ed), "Reactor handbook Volume III Part B Shielding", ORNL, (1962)

[12] H.Someya, et al., "GENGTC-JB: A computer program to calculate temperature distribution for cylindrical geometry capsule", JAERI-M 87-148, (1987) [in Japanese]

[13] Y.Nagao, M.Sato, M.Niimi, "Evaluation of nuclear heating rate in JMTR", JAEA-Technology 2007-051, (2007) [in Japanese].Y. Lastname and K. Lastname, J. Plasma Fusion Res **1**, (2001) 40-45.

[14] Y.Nagao, M.Niimi, H.Kawamura, T.Iguchi, "Improvement of neutron/gamma field evaluation for restart of JMTR", Proc. of 13th Int. Symp. on Reactor Dosimetry May 25-30 2008 PETTEN, the Netherlands.

4.2 Current Status and Future Plan of JMTR Hot Laboratory

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The hot laboratory (JMTR-HL) was founded to examine the objects mainly irradiated in the Japan materials testing reactor (JMTR) in 1971. The JMTR-HL has an advantage that the hot cell is connected with the reactor vessel of the JMTR by a canal. Hence it is easy to transport irradiated radioactive capsules and specimens through the canal. Since 1971, about 2,400 irradiated capsules have been treated in the JMTR-HL and various post irradiation examinations (PIEs) have been widely performed there. In recent years, several new techniques, e.g., an in-cell irradiation assisted stress corrosion cracking (IASCC) test, a scanning-electron microscope (SEM) / electron-back scattering-diffraction pattern (EBSD) observation, were added to the conventional PIEs. In addition, the JMTR-HL had contributed to realize an in-pile IASCC test program at the JMTR through the development of a TIG welding technique by remote-handling with manipulators in the hot cell for re-assembling of capsules. A modification of the facility to treat high burn-up fuels, up to about 100 GWD/t, is planned at the JMTR-HL now.

Keywords: Japan Material Testing Reactor, Hot Laboratory, Post Irradiation Examination, BOCA, IASCC, SEM-EBSD.

1. INTRODUCTION

The hot laboratory (JMTR-HL) associated with the JMTR was founded to examine the material specimens and fuel specimens irradiated mainly in the JMTR, and has been operated since 1971. The JMTR-HL is directly connected with reactor core by a water canal which is 6m deep and 3m wide, as shown in Fig.1. Hence irradiated radioactive capsules are efficiently transported under water through the canal in a short time. In this report, we describe the current status and future of the JMTR-HL.

2. OUTLINE OF FACILITIES OF THE JMTR-HL AND MAIN TEST APPARATUSES

The JMTR-HL has three hot cell lines for post irradiation examinations (PIEs); a concrete cell line, a lead cell line, and a steel cell line, respectively. Figure 2 shows the year each facility was put into service.

The concrete cell line consists of eight cells and it is shielded by heavy concrete of 1.1m in thickness, as shown in Fig.3. It was put into service in 1971. In the concrete cells, dismantling irradiated capsules, re-capsuling, visual inspections, X-ray radiography, dimensional measurements, gamma scans, and eddy current tests are carried out on fuel specimens as PIEs. Metallography by optical microscope and hardening tests are carried out in four microscope cells that are shielded by lead of 0.18m in thickness and are connected to the concrete cells.

The lead cell line consists of seven cells shielded by lead of 0.15m in thickness, as shown in Fig.4. It was put into service in 1971 for PIEs on irradiated materials.

The steel cell line consists of five cells shielded by steel of 0.35m in thickness. It was put into service in 1982 for PIEs on irradiated materials. The tensile test, instrumented impact tests on reactor material, and PIEs related to mechanical properties on fusion reactor material are performed in these cells.

3. EXPERIENCES OF PIES IN THE JMTR-HL

The JMTR-HL has been operated for about 40 years for PIEs of many objects irradiated JMTR and other facilities. Through the PIEs, the JMTR-HL contribute not only to research of materials for light water reactors, fast reactors, high temperature gas reactor, and fusion reactor fuels, but also to the production of domestic industrial radio isotopes like ^{192}Ir.

The JMTR-HL treated about 2,400 capsules and about 90,000 irradiated specimens, and performed over 9,500 PIEs from 1971 to 2006 in the JMTR-HL[1]. Figure 5 shows statistics of the irradiated capsules according to their purpose. The contribution of material tests is 55%, of fuel tests is 21% and of isotope productions is 24%. The fuel tests consist of 65% tests for light water reactors, 13% tests for high temperature gas reactors, and 8% tests for basic research. The material tests consist of 29% tests for basic research, 28% tests for light water reactors, 9% tests for fusion reactors, and 9% tests for high temperature reactors.

4. THE TREND OF PIES IN THE JMTR-HL

The followings are topics of recent PIEs.

4.1 Assembling and dismantling of BOCA capsules

Power ramping tests in JMTR using BOCA (Boiling Water Capsule) were performed for the purpose of safety research of load following operations on LWR fuels. The JMTR-HL developed an installing apparatus for irradiated fuel pins into the BOCA, an assembling apparatus for capsules, and a dismantling apparatus of capsules after irradiation[2]. These apparatuses have been installed in the hot cells, as shown in Fig.6. After these tests, BOCA was used for the power ramping tests in JMTR for the purpose of research of high burn up tests of LWR. These tests were performed over the course of about 20 years from 1981 to 1999, and it contributed to safety research and achievement of high burn-up of LWR fuels.

4.2 Irradiation assisted stress corrosion cracking experiments in a hot cell

Irradiation assisted stress corrosion cracking (IASCC) occurred in stainless steel is considered to be one of the key issues from a viewpoint of the life management of core components in the aged LWRs. To simulate IASCC behavior in LWRs for PIEs, tensile tests are performed under high temperature and high pressure water conditions on specimens irradiated up to a neutron fluence that is higher than the so-called IASCC threshold fluence in a test reactor. Figure 7 shows the developed IASCC experiment apparatus installed in the hot cell[3]. Over 200 irradiated specimens were tested by this apparatus. It contributed to basic research of materials of LWRs.

4.3 The in-pile IASCC experiments -Assembling techniques for test capsules

In-pile crack growth and crack initiation studies in the JMTR were performed to evaluate factors affecting IASCC behavior. These studies were performed by in-pile IASCC test capsules that simulate LWR water conditions under irradiation. The results were compared with these of PIEs.

There were, however, some technical hurdles to overcome for the experiments. To perform in-pile IASCC tests, pre-irradiated specimens were relocated from pre-irradiation capsules to an in-pile test capsule in a hot cell by remote handling. Hence, a remote TIG welding technique was developed for assembling the in-pile test capsules [4,5]. Figure 8 shows the outline of the remote-assembly work. Eight in-pile IASCC test capsules were assembled and the testing time about 20,000 hours in total was achieved.

4.4 SEM/EBSD

In order to obtain a fractography of the test specimens after mechanical tests, a remote-handling type scanning-electron microscope (SEM) was developed and introduced into the JMTR-HL in 1995. It contributes greatly to the study of generation mechanisms of IASCC and IGSCC (inter granular stress corrosion cracking) in structural materials of LWRs.

Furthermore, in order to investigate generation mechanisms of IASCC and IGSCC in more detail, the Orientation Imaging Microscopy (OIM) was introduced in the JMTR-HL in 2001. A new hardware device, an electron-backscattering diffraction-pattern (EBSD) detector was added onto the conventional SEM, as shown in Fig.9. Additionally a new software system, called OIM, was added. Thus a remote-handling type crystal orientation analyzer was realized as an in-cell system for the first time[6]. It is the only one of its kind in the world. Through the OIM observations using heavily irradiated specimens such as structural materials in light water reactor, the JMTR-HL is contributing to the further development of the research in the fields of IASCC and IGSCC.

5. RENEWAL PLAN OF THE JMTR-HL

In order to accept a higher burn-up fuel, up to about 100 GWD/t, a refabrication of the JMTR-HL from 2008 to 2011 is planned as follows;

5.1 Correspondence to high burn-up fuel

A power ramping test for high burn-up LWR fuels is planned using BOCA in JMTR. LWR fuel pins irradiated at commercial power plants will be loaded into the BOCA capsules at the hot cell of the JMTR-HL, then they will be re-irradiated in the reactor core of the JMTR.

(a) Reinforcement of the neutron shielding capacity of hot cell :
Because the burn-up rate of the accepted fuel will be higher than that of the present license, a reinforcement of the neutron shielding capability is required. Therefore, a reinforced design is planned for the concrete cell. (See Fig.10)

(b) Center-hole drilling technique for irradiated fuel pellet :
The BOCA irradiation program requires installation of instrumentations for fuel-center temperature and fuel-gas pressure. As shown in Fig.11, the center-hole drilling technique for installation of a thermocouple and the welding technique for the end-plug of the fuel pins had been developed by the JMTR-HL in the previous BOCA program. It is planned to apply the same technique this time.

(c) Assembling and dismantling apparatus for BOCA capsule :
Design of in-cell apparatus for assembling and dismantling of BOCA capsules is planned to increase the efficiency of the work, as shown in Fig.12. The shielding container becomes unnecessary by changing the loading method from a horizontal gamma gate loading to the underwater loading, thus man power and working hours can be saved.

5.2 Renewal of aged facilities

The facilities of the JMTR-HL have become aged by continuous operation for about 40 years. Therefore, replacement of main equipment, e.g., power supplies, power manipulators, normal manipulators, etc., is planned before restart of JMTR in 2011.

6. CONCLUSIONS

Through the experiences of many PIEs and the development of the new techniques for PIEs, the JMTR-HL has contributed to the research and development for fuels and materials of various nuclear reactors for many years. By the renewal plan in progress, the JMTR-HL will become the most important facility for PIEs in Japan.

REFERENCES

[1] “JMTR Hot Laboratory”, pp.126-128, History of Japan Atomic Energy Research Institute, 2005, (In Japanese).

[2] Toshimitsu ISHII, et al., “Assembling techniques for re-irradiation capsules in the JMTR-HL,” p.9-1, UTNL-R 0446, 2005, (In Japanese).

[3] Toshimitsu ISHII, et al., “Remote-welding technique for assembling in-pile IASCC capsule in hot cell,” pp.46-54, Proc., of 2005 JAEA-KAERI Joint seminar on advanced irradiation and PIE technologies, JAEA-Conf 2006-003, 2006.

[4] Kazuo KAWAMATA et al., “Remote-welding technique for assembling in-pile IASCC capsule in hot cell,” pp.115-125, Proc., of 2005 JAEA-KAERI Joint seminar on advanced irradiation and PIE technologies, JAEA-Conf 2006-003, 2006.

[5] Akira SHIBATA, et al., “Technical development for IASCC irradiation experiments at the JMTR,”ICONE16-48588, Proc. 16th Int. Conf., on Nucl., Eng., May 11-15, 2008, Oriando, Florida, U.S.A.

[6] Yoshiaki KATO, et al., “Installation of remote-handling typed EBSD-OIM analyzer for heavy irradiated reactor materials,” JAEA-Testing 2008-005, 2008.

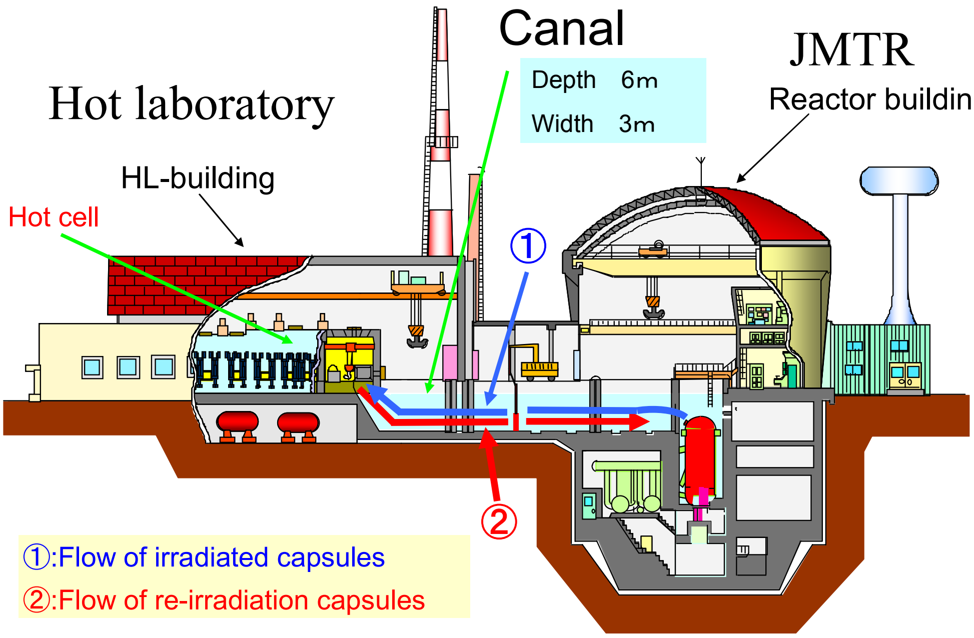


Fig.1 Canal between the JMTR and the Hot laboratory

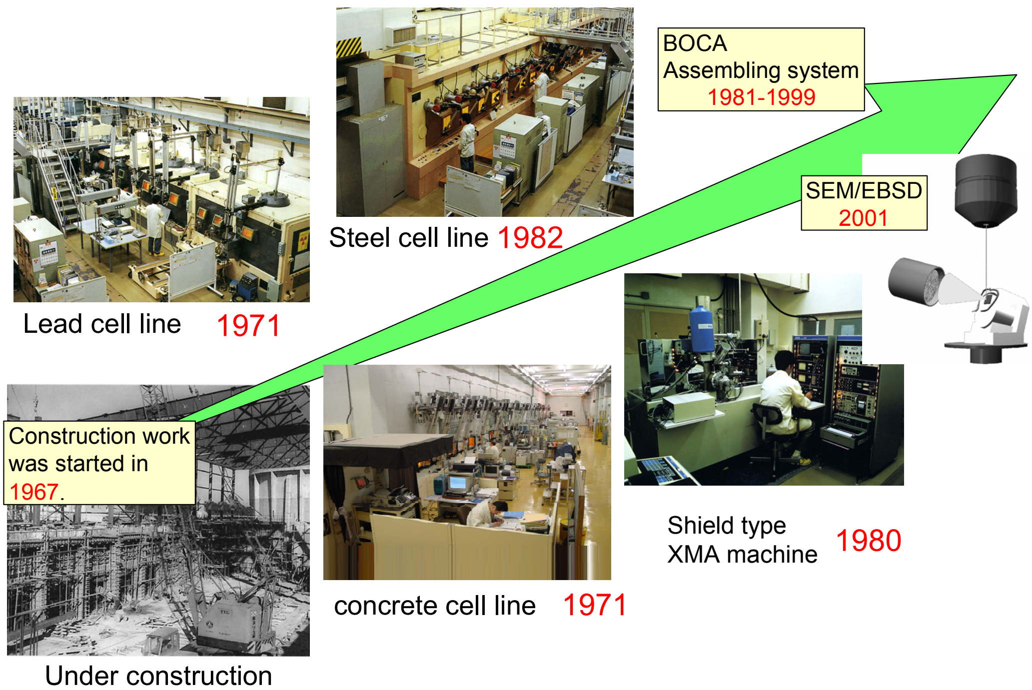


Fig.2 History of the JMTR Hot Laboratory

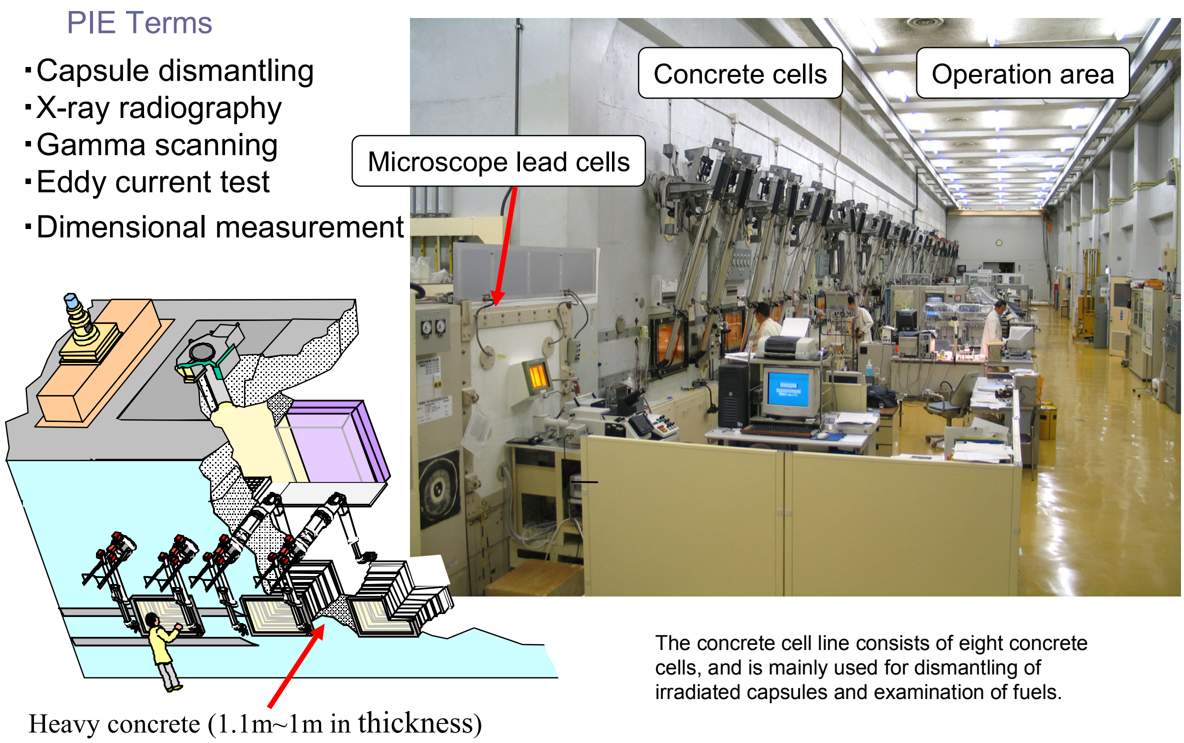


Fig.3 Concrete Cell Line

Fig. 4 Lead and Steel Cell Lines

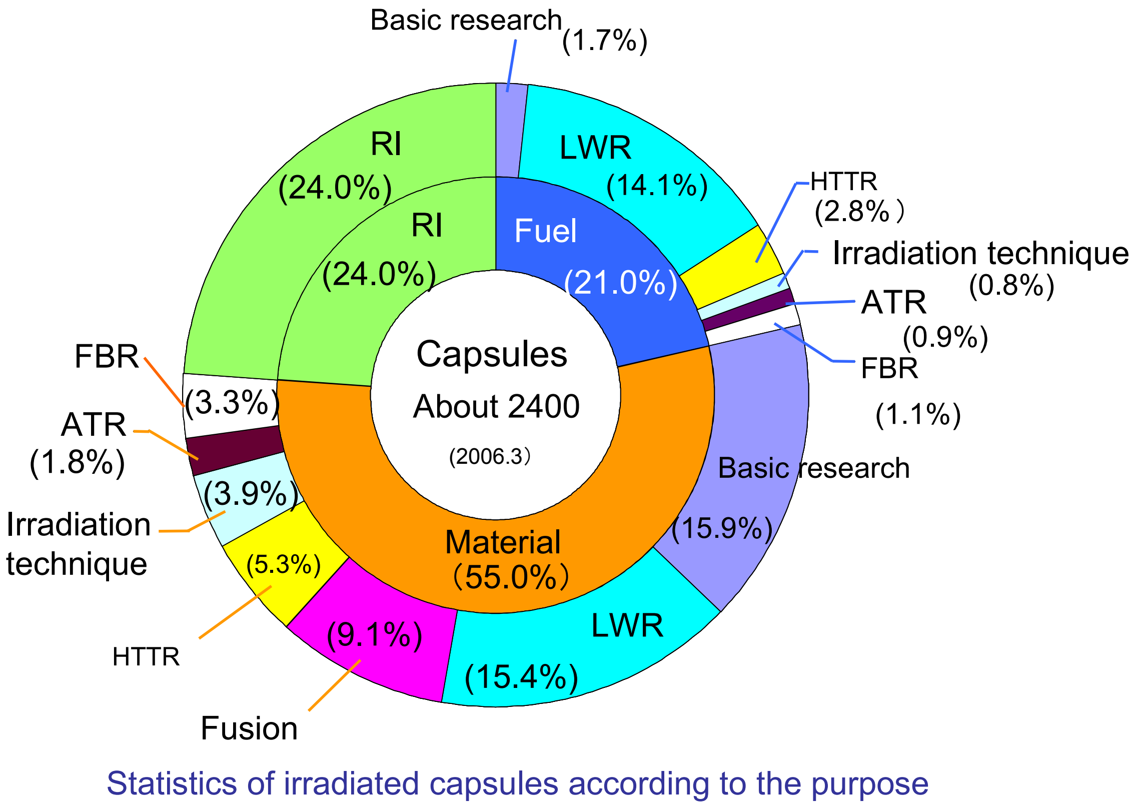


Fig.5 Experiences of PIEs in the JMTR-HL from 1971 to 2006.

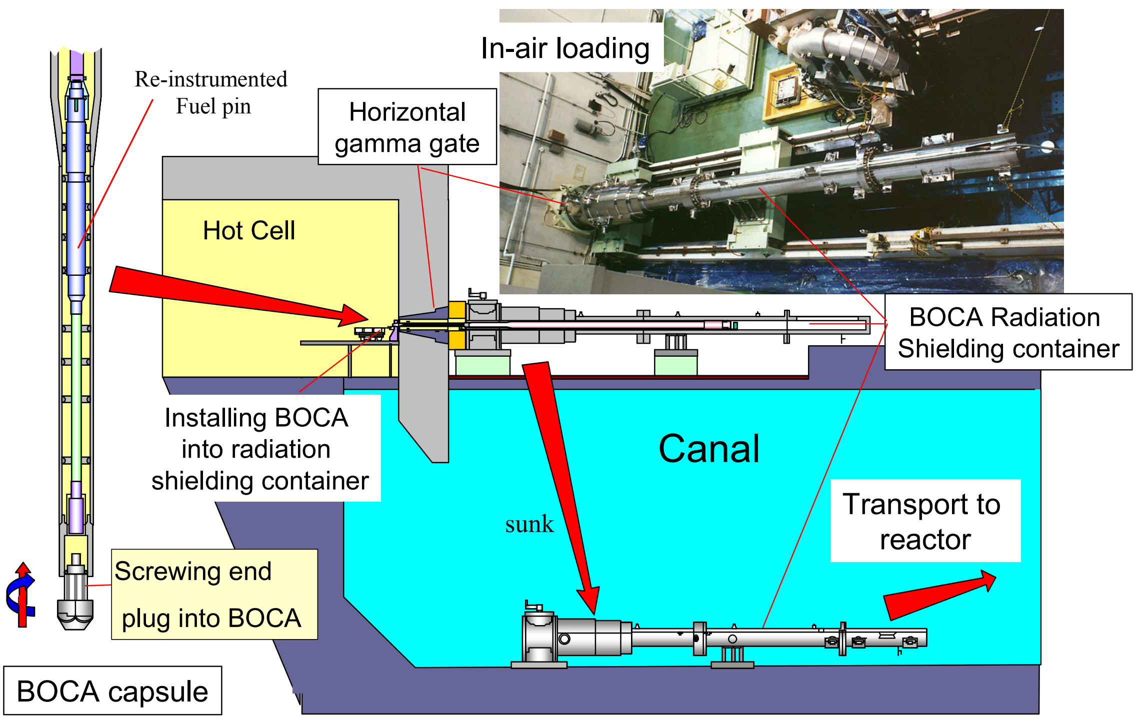
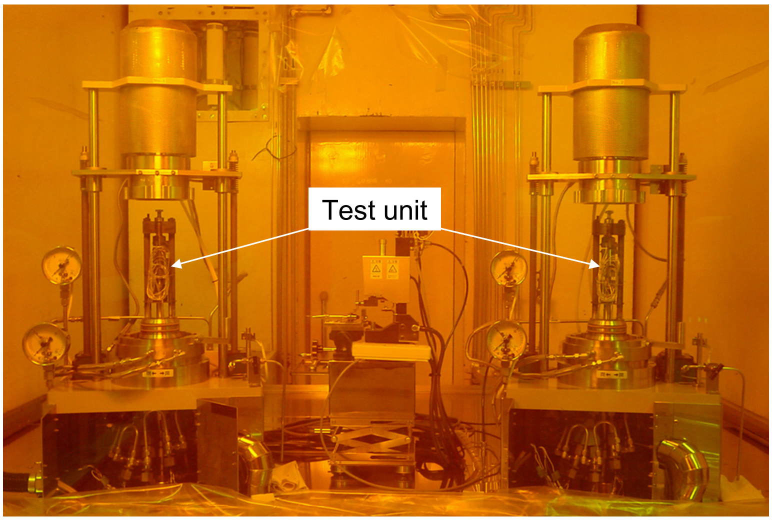


Fig.6 Assembling and dismantling of BOCA capsule



Specifications	
Maximum load	10 kN
Maximum stroke	30 mm
Maximum temperature	573 K
Maximum pressure	10 MPa
Maximum flow rate	30 ℓ/h [total] (15 ℓ/h [1 autoclave])

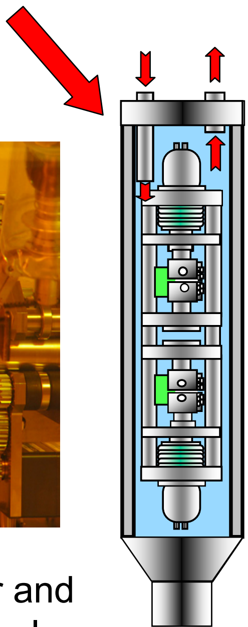
Test time 1500~2000Hr

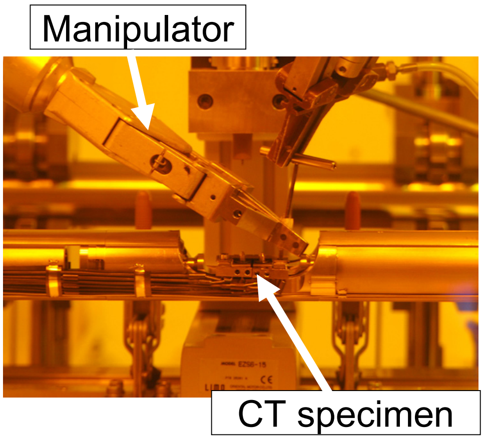
This apparatus simulates environmental conditions in the reactor core of BWRs, excepting with neutron irradiation.

Test specimens; 0.5T-CT type specimens irradiated in the JMTR.

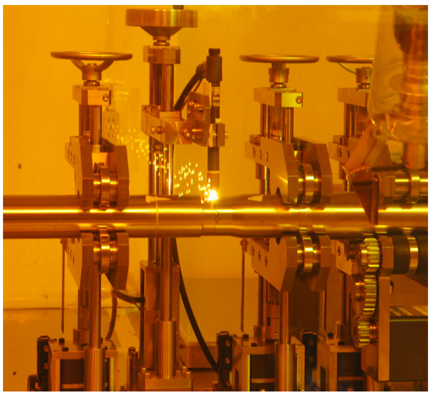
Fig.7 Outline of IASCC growth test apparatus

In-pile IASCC test capsule





Installation of pre-irradiated specimen into the capsule



In-cell welding of inner and outer tubes of the capsule

Fig.8 Assembling of In-pile IASCC test capsule in the hot cell

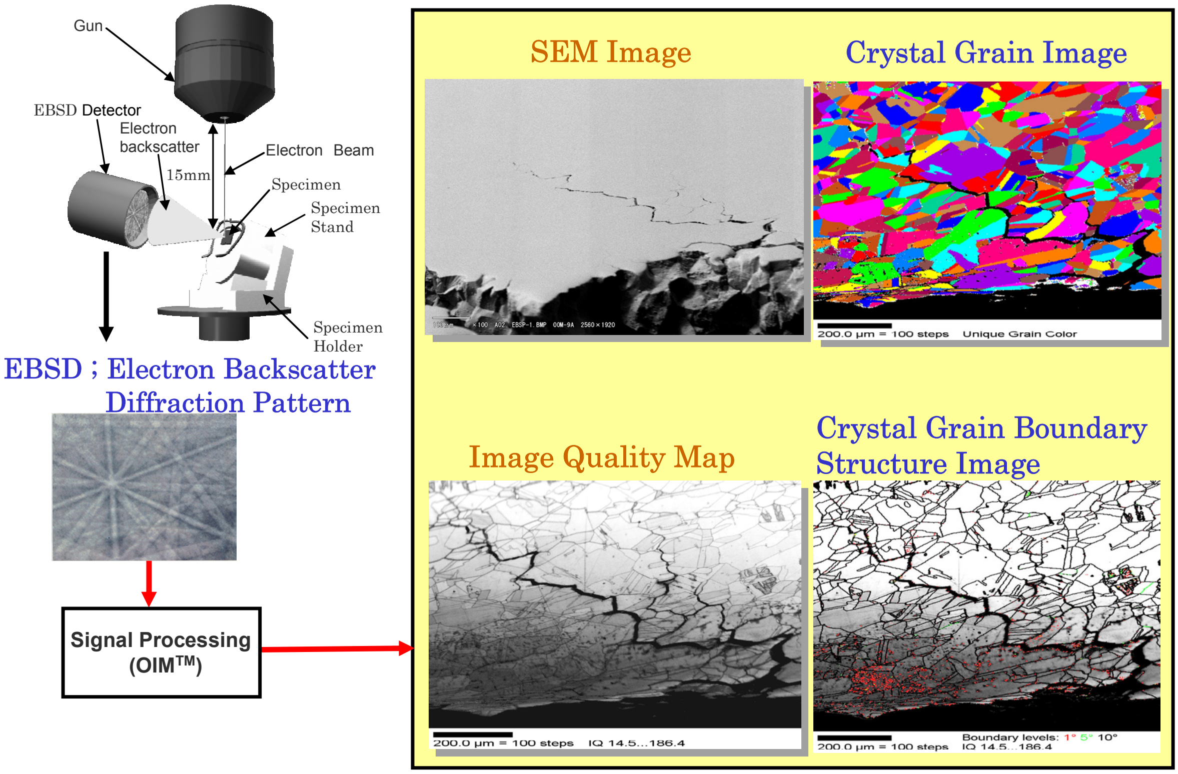


Fig.9 Orientation Imaging Microscopy using EBSD

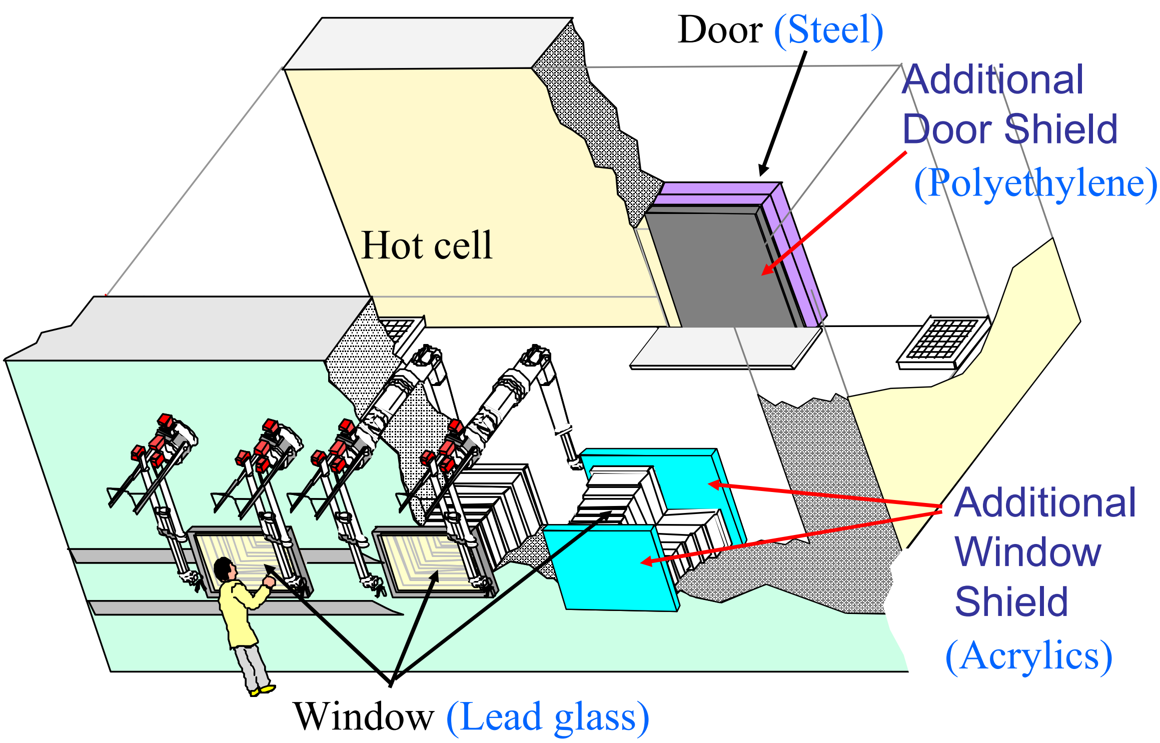


Fig.10 Reinforcement of neutron shielding capacity of the hot cell

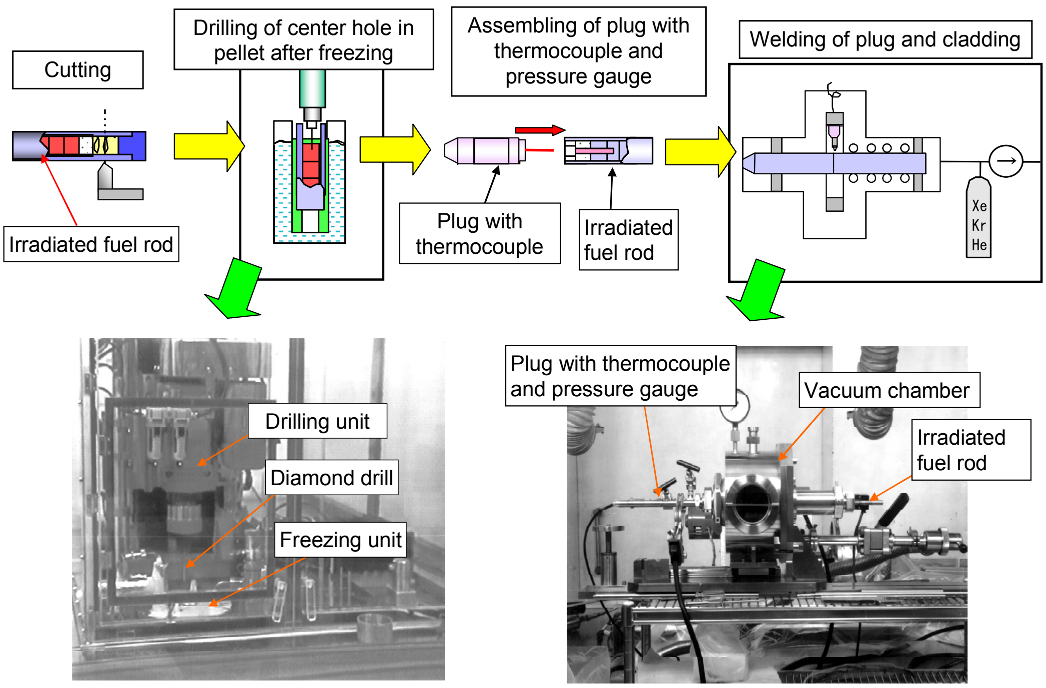


Fig.11 Outline of installing of a thermocouple and a pressure gauge in a fuel pin

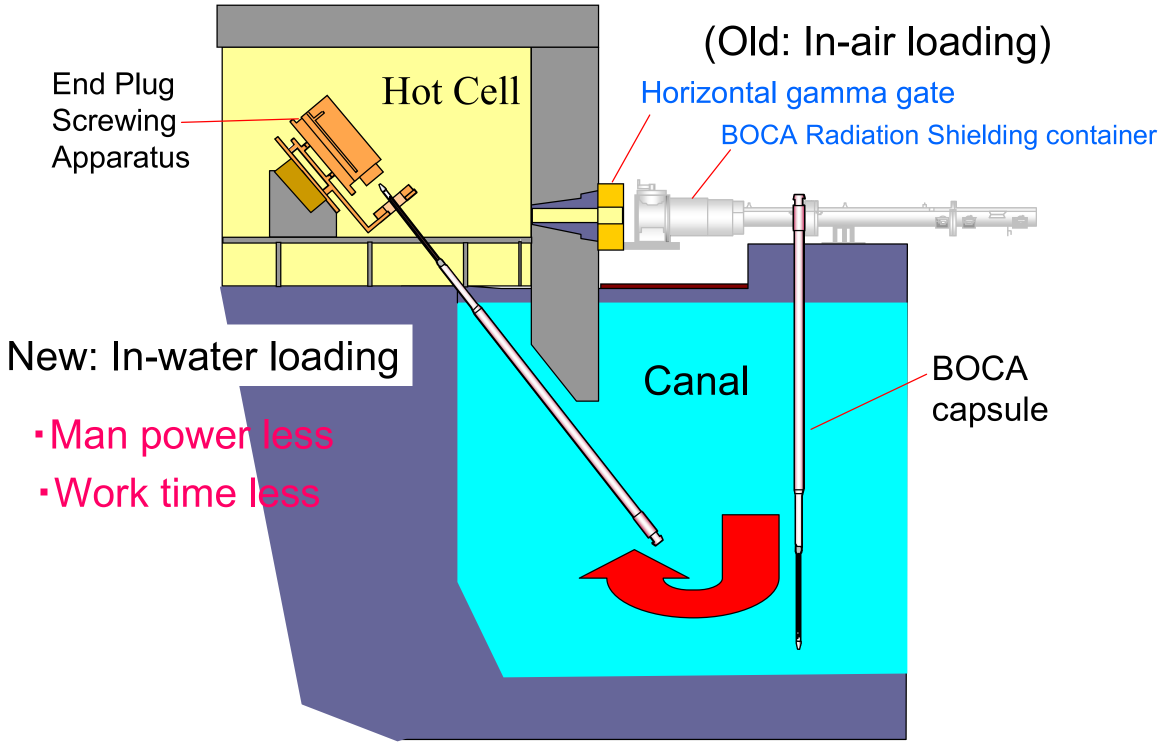


Fig.12 New loading method for BOCA capsules

4.3 Post Irradiation Examinations Cooperation and Worldwide Utilization of Facilities

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Status of post irradiation examinations in Studsvik's facilities, cooperation and worldwide utilization of facilities, was described. Studsvik cooperate with irradiation facilities, as Halden, CEA and JAEA, as well as other hot cell facilities (examples, PSI, ITU and NFD) universities (example, the Royal Institute of Technology in Sweden) in order to be able to provide everything asked for by the nuclear community. Worldwide cooperation for effective use of expensive and highly specialized facilities is important, and the necessity of cooperation will be more and more recognized in the future.

Keywords: Studsvik, Post irradiation examinations, Worldwide cooperation

1. INTRODUCTION

The situation for MTRs (Multipurpose Test Reactors) and PIE (Post Irradiation Facilities) have changed dramatically comparing with the situation some 50 years ago when it all started.

Nuclear energy and the glorious opportunities for the nuclear based technology at that time had no limit, with nuclear driven cars, aerospace projects and boats, as well as cities heated by the cooling water as promising opportunities. Governments in many countries put a lot of effort as well as resources into this new promising technology, starting national hot cell laboratories as well as test reactors and production facilities. There was no lack of either resources or projects.

We have today a mature industry with nuclear produced electricity and some special nuclear operated larger vessels. The governments look upon the nuclear technology as a mature technology and their focus is today on Gen IV and fusion, implying less resource for the facilities and the number of facilities has decreased during the last years. Studsvik as an example shut down the test reactor, the R2, which was very similar to the JAEAs JMTR, a few years ago. The new situation just recently with nuclear again being a politically acceptable and even favorable energy source has not and will probably not imply a dramatic increase in number of facilities. Established nuclear energy is regarded as a mature technology and the R&D level is accordingly.

Clients require better and better quality on the data produced at the MTRs (focusing on online data) and at the hot cells. Most clients also demand delivery exactly as planned even though some work is more like research, which by nature is more difficult to predict and plan. Being a successful MTR implies that you deliver as planned, what was agreed and with high quality.

It is not only a technical and timely performance issue to operate an MTR but also a management, owner and regulatory issue.

The future view is a smaller number of facilities which will have to share the limited amount of resources, both from governments and industry. It is today very difficult to manage everything by yourself as special technologies exist only in a few places. Regulatory issues as well as operational issues are mostly common and only some are special to certain countries. Shared lessons learned in these common areas are a way to make the MTR operation easier.

Studsvik cooperate with Halden, CEA and JAEA as well as a university (the Royal Institute of Technology in Sweden) in order to be able to provide everything asked for by the nuclear community. Cooperation will be even more necessary in the future when looking at the future of nuclear with concepts requiring expensive and highly specialized facilities. The Studsvik activities and outline of cooperation image is shown in Fig.1 and Fig.2.

Fig. 1 Activities of Studsvik.

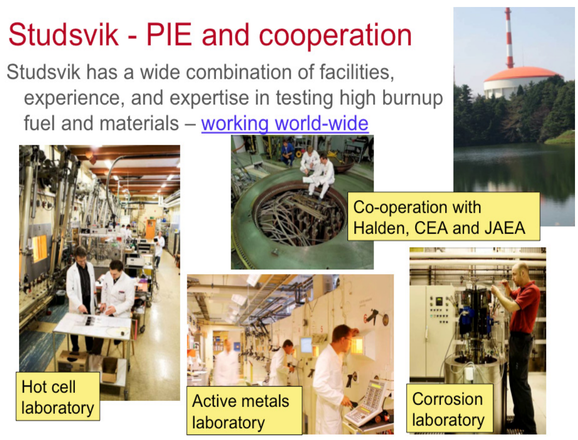

Fig. 2 PIE and cooperation on Studsvik.

2. PIE ON STUDSVIK

Studsvik performs PIE for clients world-wide and all main vendors are clients at Studsvik, as well as utilities world-wide. Examples of PIE on Studsvik are shown in Fig.3 and Fig.4. Studsvik's mission is to be a complete supplier irrespective of the client need, delivering everything from initial transport to final disposal of the examined or tested material, and Studsvik thus has to cooperate with different specialist organizations. Studsvik has a good relation with and is supported by the Swedish authorities, not with funds but with clear regulations and expected behavior, which is necessary to be able to work world-wide with different clients, issues and materials.

Worldwide cooperation, which is asked for by many (from authorities to industry) requires, in addition to expertise, experience and good facilities, some amount of Authority benevolence to be feasible.

Cooperation agreements is just one way forward, it is also possible to work together with special topics, tests, areas of expertise, etc. without a formal cooperation agreement. Studsvik work with NFD (Japan), PSI, ITU (Europe) and others without formal agreements to be able to provide all services asked for. Facilities have typically different areas of expertise and Studsvik work with any facility needed to get the work done.

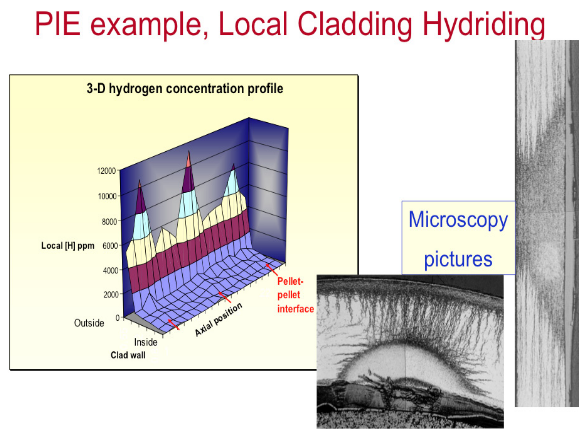

Fig. 3 Example of PIE on Studsvik (local cladding hydriding).

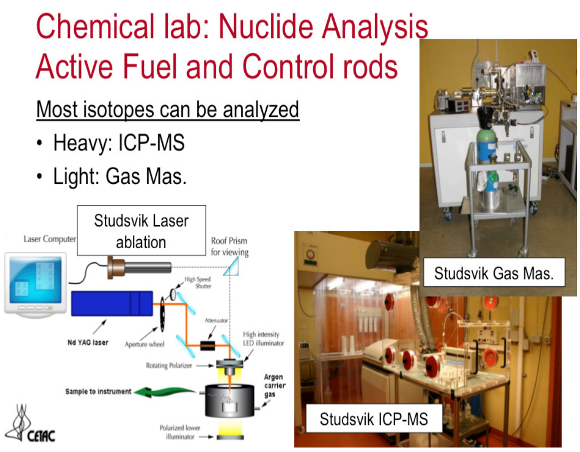

Fig. 4 Example of PIE on Studsvik (nuclide analysis)

4.4 PIE Activities in NFD Hot Laboratory

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The BWR fuel design, which has been revised for step-by-step burnup extension, has been verified at each step through comprehensive PIEs. A large number of fuels and materials have been examined in various research and development programs. Recently, the axial cracks on the outer surface have been observed for some high burnup fuel claddings during the power ramp test. Detail PIEs showed that the cracks initiated at the outer surface of the fuel cladding and propagated in a radial direction possibly by the delayed hydride cracking mechanism. Therefore, mechanical properties of fuel cladding under increased fuel burnup were studied from a viewpoint of the hydrogen concentration. Some new PIE techniques were developed in order to clarify the failure mechanism of high burnup fuel cladding under power ramp conditions.

Reactor core structural materials have also been studied for plant life management and development of remedies. Irradiation assisted stress corrosion cracking (IASCC) is one of the critical concerns in materials for nuclear in-reactor components of Light Water Reactor. IASCC studies require various kinds of PIE techniques, because IASCC is caused by a complicated synergistic effect of stress and chemical environment on material that suffered degradations by irradiation. In order to investigate the mechanism of IASCC, we developed and carried out SCC test and microchemical analysis of irradiated stainless steels.

Keywords: BWR Fuel, High Burnup, Cladding, Mechanical Properties, Hydrogen Concentration, Structural Material, IASCC, Crack Growth Rate

1. INTRODUCTION

Nippon Nuclear Fuel Development Co., Ltd. (NFD), a joint venture of Hitachi LTD. and Toshiba Corp., has been operating hot laboratory facility since 1977 for extensive post-irradiation examinations (PIE) of boiling water reactor (BWR) fuels and structural materials. NFD hot laboratory has capability to accommodate full size commercial BWR fuel bundles. Comprehensive PIE programs have been carried out on many BWR fuel bundles [1-5] including failed fuel bundles [6] and MOX fuel bundles [7], as well as structural materials irradiated in BWRs [8-10]

To meet the demands for detailed mechanistic understanding of fuel performance and material degradation, more precise microscopic and specific PIE techniques are required. Various examination techniques have been developed during the course of the PIE programs to meet the research requirements. This paper presents the overview of recent PIE activities in NFD hot laboratory.

2. OUTLINE OF THE NFD HOT LABORATRY

NFD hot laboratory consists of a storage pool, an inspection pool, six concrete cells, six steel shielded cells, two waste storage cells, and an isolation area. It has a capability to accommodate full size commercial BWR fuel bundles.

Standard PIE procedures include non-destructive examinations followed by destructive ones. A window is installed in the side wall of the inspection pool so that the fuel bundles can be observed directly. Non-destructive examinations on fuel bundle and fuel rods can be performed in the inspection pool and the monitoring cell. Concrete cells, shielded by heavy concrete, are used for destructive examinations such as metallography using optical microscope and scanning electron microscope (SEM). Steel shielded cells annexed to the concrete cells are used for the tests on relatively small specimens, such as mechanical and stress corrosion cracking testing.

Various kinds of microscopic equipment, including transmission electron microscope (TEM), field emission electron gun equipped TEM (FEG-TEM), FEG-SEM, electron probe micro-analyzer (EPMA), X-ray diffractometer (XRD), X-ray fluorescence spectrometer (XRF), and focused ion beam (FIB) are installed in precise measurement labs.

3. OVERVIEW OF PIE ACTIVITIES

3.1 Development of New Pie Techniques for High Burnup Fuel Claddings

Recently, the axial cracks on the outer surface have been observed for some high burnup fuel claddings during the power ramp test. Detail PIEs showed that the cracks initiated at radial hydrides of outer surface. Therefore, some new PIE techniques were developed in order to clarify the failure mechanism of high burnup fuel cladding.

3.1.1 Power Ramp Simulation Test

As schematically shown in figure 1, the power ramp simulator developed in this work mainly consists of an internal heater, high-pressure pumps, a pressure vessel, and a cooling/heating system [11]. A cladding tube specimen equipped with the internal heater is set in the pressure vessel filled with water, which corresponds to media for cooling specimen. The internal heater, which has a heating portion of 20 mm length, simulates heating the fuel at a linear heating rate of 45 kW/m, inducing the radial temperature gradient in the specimen.

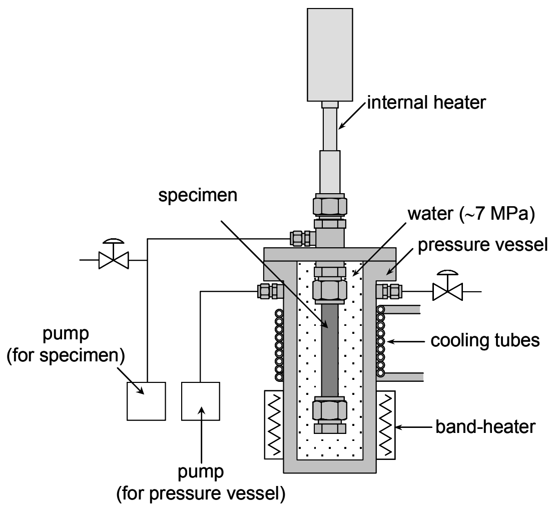


Fig.1. Schematic figure of the power ramp simulator [11].

Figures 2(a) and (b) show the cross sectional metallography at the internally heated region after the 30-min and 2-hour tests, respectively [11]. For both tests, radial hydrides were observed at the periphery of the internally heated region. The morphology of the hydrides was very similar to those observed in the cladding examined after the power ramp tests, indicating good capability of the simulator for power ramp tests. Meanwhile, circumferential hydrides that precipitated before the test were also observed. These results imply that the radial hydrides are mainly formed by the hydrogen that thermally diffuses from the inside of the specimen. The cross sectional metallography observed in the non-heated region after the 2 hour test is shown in figure 2(c) for reference. In contrast with the internally heated region, the non-heated region showed no precipitation of radial hydride.

The power ramp test is very important for evaluating the integrity of fuel. However, it is very hard test and it is difficult to control a test parameter independently. Since this power ramp simulator is easy to control a test parameter independently, we expect that it complements a power ramp test.

3.1.2 In-situ Observation of the Crack Growth Process under the SEM

A schematic of the test apparatus for the in-situ observation under the SEM is shown in Figure 3 [12]. The ring tensile method was used, since it is a simple technique with a compact and simply-shaped specimen. The specimens were put on a tensile device which was placed in the chamber of the SEM and driven by an alternative current servo-driver. Load and displacement were measured by a 5 kN small-sized load cell and an inductive displacement transducer, respectively. A specimen was heated by a micro-heater and temperature was measured by an infrared thermometer. The maximum temperature and load were 623 K and 3 kN, respectively. Crack propagation in the radial direction was continuously observed and recorded. Hydride phase could also be observed by use of a back scattered electron imaging technique.

SEM images of cracks in the circumferentially-hydrided material are represented in Figure 4 separately for each cycle. In cycle 1, as shown in Figure 4(a), the crack propagation started at A in the figure after 88 minutes passed from the start of the tensile test [12].

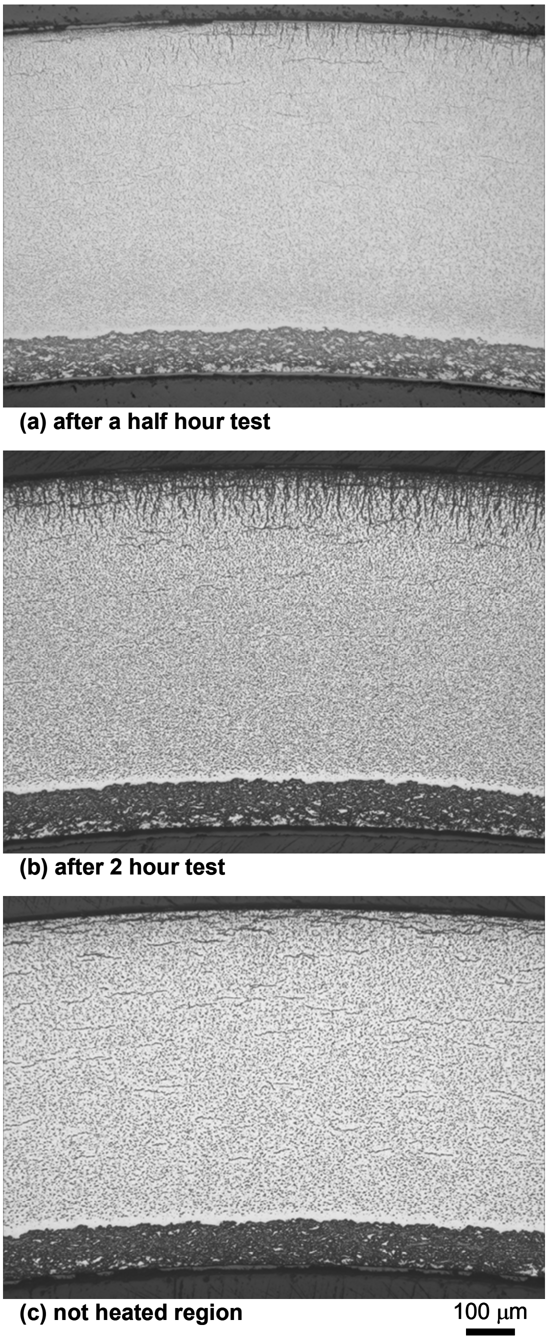


Fig.2. Cross sectional metallography observed at internally heated region after a half hour test and (b) 2 hour test, and (c) at not heated region after 2 hour test [11].

The crack extended rapidly from A to B in the figure, while the velocity immediately decreased to a very low level after B. The stagnation of the crack propagation was accompanied by the crack tip blunting as shown in micrograph 4. In cycle 2, a new sharp crack appeared at the blunted crack tip and propagated rapidly from 4 to 5 as shown in Figure 4(b). As was the case in cycle 1, the crack propagation stagnated again from 5 to 7. The blunting of the crack tip was also observed. The crack velocity in cycle 3 was one step faster than cycle 1 and cycle 2 (see Figure 4(c)). The appearance of new sharp cracks followed by the blunting were observed qualitatively in the similar manner as cycle 1 and cycle 2. In cycle 4 the crack propagated another step faster than cycle 3. Sharp and almost straight cracks propagated in a zigzag manner as shown in Figure 4(d).

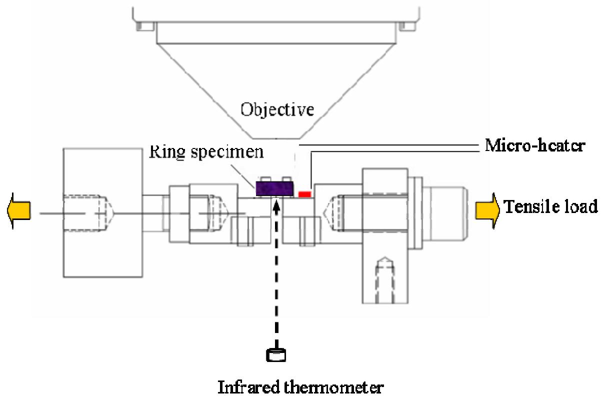
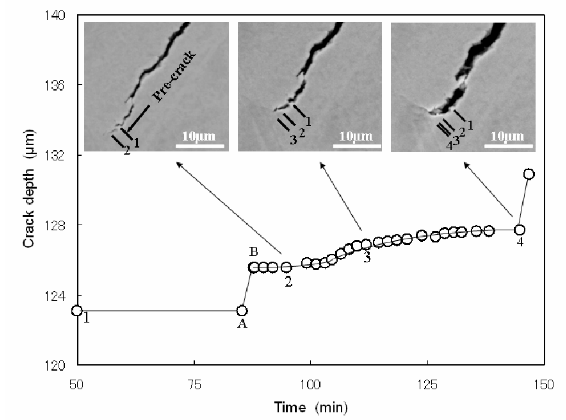
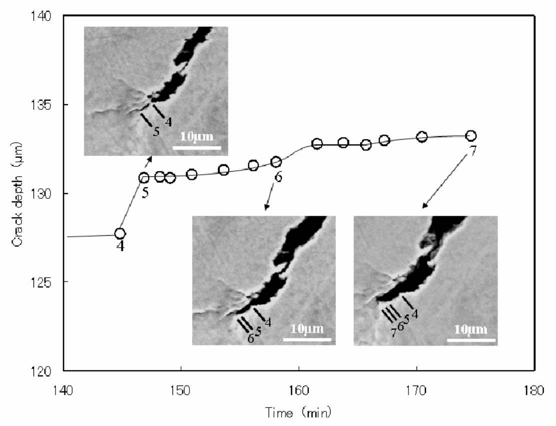


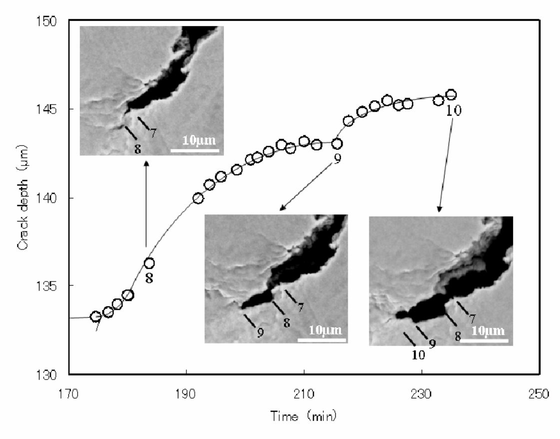
Fig.3. Schematic of the test apparatus for the in-situ observation of the DHC process under the SEM [12].



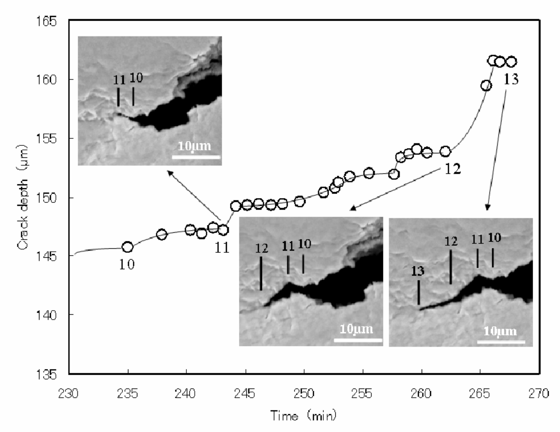
(a) Cycle 1



(b) Cycle 2



(c) Cycle 3



(d) Cycle 4

Fig.4 SEM micrographs of incipient cracks in the circumferentially-hydrided material : (a) Cycle 1, (b) Cycle 2, (c) Cycle 3, (d) Cycle 4 [12].

A new technique was developed for the ring tensile tests on fuel cladding tubes in a chamber of the SEM to directly observe the crack propagation process. The crack propagation process was clearly seen from the relatively early phase of the process to the failure.

3.2 PIEs on Structural Material for IASCC

Irradiation assisted stress corrosion cracking (IASCC) is a recent concern in materials for nuclear in-reactor components of light water reactors (LWRs) [13-15]. It takes the form of intergranular stress corrosion cracking (IGSCC) and the critical fluence level has been reported to be about $5x10^{24}n/m^2$ (E>1MeV) in Type 304 stainless steels (SS)

In order to evaluate the integrity of the in-reactor components, the crack growth rate (CGR) test is carried out in many research institutes. In PIE, a small-sized specimen is used for a CGR test from limitation, such as the irradiation space of Material Testing Reactors. However, CGR tests using a large-size specimen also are required from a viewpoint of the stress conditions at the crack tip. We developed the specimen machining techniques from the LWR in-core structure materials. CT specimen is machined using a numeric control milling machine and electric discharge machine which are installed in a hot cell. We have machined a maximum of 1.5 T-CT specimens (Figure 5) from the block of approximately 650w x 140h x38.1t mm until now. Although remote handling was difficult for this block, it could be done and machining was carried out using various devices.

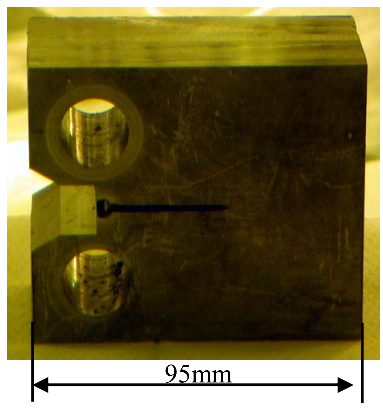


Fig.5 1.5T-CT specimen (maximum size)

Figure 6 shows the stress intensity dependence of CGR data under normal water chemistry (DO: 32ppm) using large-sized specimens [16]. All of the data are below the upper limit and low fluence data are below the disposition curves of The Japan Society of Mechanical Engineers (JSME) code.

In order to investigate the mechanism of IASCC, recently, it is necessary to observe the microstructure of the position which the crack has occurred. We applied FIB systems to sampling the TEM specimen from crack tip. TEM specimen can be prepared with rapidity and accuracy using focused ion beam. In-situ microsample extraction from specific site is possible. For example, they are worked layer, crack tip, heat affected zone and fusion line of weld joint. Micro-sampling technique is enable us to observe and identify the oxides in cracks and the dislocation structure near cracks and also to estimate the deformation behaviour near cracks. Figure 7 shows the example of FIB sampling procedure. The around of the observation area at crack tip are cut off by an ion beam. A micro specimen is picked up by the mico-manipulator. After sampling, this sample is made thin to 50nm thickness using FIB.

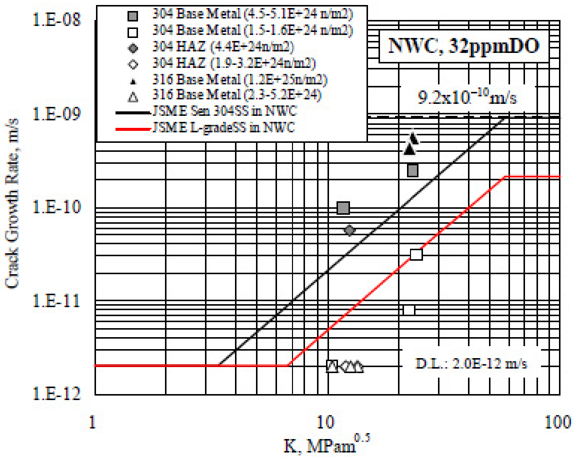


Fig.6 Dependence of CGR on neutron fluence [16]

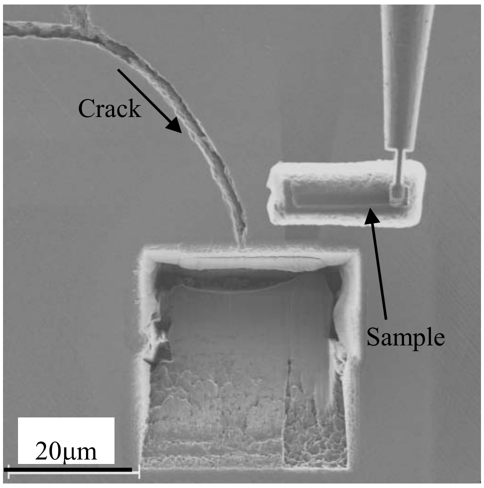


Fig.7 Example of FIB sampling procedure

4. CONCLUSIONS

Some new PIE techniques were developed in order to clarify the failure mechanism of high burnup fuel claddings under power ramp conditions.

Various material evaluation techniques were developed and the data required for an IASCC mechanism elucidation and IASCC evaluation guide proposal is obtained.

A large number of fuels and materials irradiated in commercial reactors as well as those in test reactors have been examined in various research and development programs. Detailed PIEs on fuel pellets, cladding materials, and core structural materials have been successfully carried out and significant amounts of data have been accumulated.

REFERENCES

[1] Y. Mishima and T. Aoki, Paper presented at IAEA International Symposium on Improvements in Water Reactor Fuel Technology and Utilization, IAEA-SM-288/58, Stockholm, Sept. 15-19 (1986).

[2] M. Oishi, Paper presented at IAEA Technical Committee Meeting on Fuel Performance at High Burnup for Water Reactors, Nykoping (1990).

[3] H. Ohara, et al., Proc. International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, April 17-21 (1994).

[4] H. Hayashi, et al., Proc. International Topical Meeting on Light Water Reactor Fuel Performance, Portland, May 2-6 (1997).

[5] H. Hayashi et al., Proc. International Topical Meeting on Light Water Reactor Fuel Performance, Park City, April 10-13 (2000).

[6] K. Ogata, et al., Proc. International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, April 17-21 (1994).

[7] M. Ichkawa, et al., J. At. Energy Soc. Japan, **39**, 2 (1997) pp.93-111 (in Japanese).

[8] M. Kodama, et al., Proc. 5th International Symposium on Environmental Degradation of Materials in Nuclear Power System-Water Reactors, Monterey, August (1991) 948-954.

[9] M. Kodama et al., Proc. 6th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, August, (1993) 583-588.

[10] Y. Tanaka et al., Proc.8th International Symposium on Environmental Degradation of Materials in Nuclear Power System-Water Reactors, August (1991) 803-811.

[11] T. Higuchi et al., Proc. the 2008 Water Reactor Fuel Performance Meeting, Seoul, Korea, October (2008) to be published.

[12] T. Kubo et al., Proc. the 2008 Water Reactor Fuel Performance Meeting, Seoul, Korea, October (2008) to be published.

[13] W.L. Clarke and A.J. Jacobs, Proc. 1st International Symposium on Environmental Degradation of Materials in Nuclear Power System-Water Reactors, Myrtle Beach, August (1983) 451-461.

[14] P.L. Andresen, et al., Proc. 4th International Symposium on Environmental Degradation of Materials in Nuclear Power System-Water Reactors, Jekyll Island, August (1989) 83-121.

[15] E. P. Simonen and S. M. Bruemmer, J. Nucl. Mater., **239** (1996) 185.

[16] S. Ooki et al., Proceedings of Twelfth International Conference on Environmental Degradation of Materials in Nuclear Power System-Water Reactors, Salt Lake City, August 2005, 365-376.

4.5 Status of PIEs in the Department of Hot Laboratories and Facilities

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Department of Hot Laboratories and Facilities (DHL) have managed three Post-Irradiation Examination (PIE) facilities. The present paper mainly describes several facilities and technical topics of following these apparatuses.

•Pellet Thermal Diffusivity Measurement Apparatus (PTD)
•Specific Heat Capacity Measurement Apparatus (SHC)
•Micro Density Measurement Apparatus (MDM)

Keywords: PIE, Specific Heat Capacity, Thermal Diffusivity, Density, Thermal Conductivity, UO_2 fuel, High Burn-up

1. INTRODUCTION

The Department of Hot Laboratories and Facilities (DHL) have managed three PIEs facilities including the Reactor Fuel Examination Facility (RFEF), the Waste Safety Testing Facility (WASTEF), and the Research Hot Laboratory (RHL) in Nuclear Science Research Institute, Tokai Research and Development Center, Japan Atomic Energy Agency. In RFEF, three kinds of apparatuses have been developed to determine the thermal conductivity of high burn-up fuel pellet.

2. OUTLINE OF EACH FACILITIES

2.1. Reactor Fuel Examination Facility (RFEF)

The RFEF has been established in 1979 to perform the PIEs of spent fuels from Light Water Reactor (LWR) and Advanced Thermal Reactor (ATR) in order to investigate their safety and reliability. RFEF is one of the biggest PIE facility in Japan. The benefit of RFEF is to perform the PIEs for full size fuel assemblies (4 meters in length, 700 kg in weight) from commercial reactor to small samples (a few millimeters). And the contribution of research programs of JAEA, the experimental study of nuclear fuels safety such as Reactivity Initiated Accident (RIA) and Loss of Coolant Accident (LOCA) simulating experiment. PIEs consist of Non-destructive test (NDT) and Destructive test (DT). The main items on NDT are Visual inspection, Gamma scanning, Diameter measurement, Eddy current measurement, Oxide layer measurement, and X-ray radiography. The main items on DT are Tensile test, Burst test, FP gas analysis, Metallography, SEM-EPMA, X-ray diffraction, Density measurement, Melting temperature measurement, and Thermal diffusivity measurement .

2.2. Waste Safety Testing Facility (WASTEF)

The WASTEF has been established in 1981 to investigate the safety storage and disposal of high-level waste from reprocessing of spent fuel. The R&D program of high-level waste was finished in 1998. Current status in WASTEF is that the examination program for the Irradiation Assisted Stress Corrosion Cracking (IASCC) of the nuclear reactor materials has been performed to use the Slow Strain Rate Tensile apparatus (SSRT), Field Emission Transmission Electron Microscope (FE-TEM) is equipped the structural analysis of irradiated materials, and spent fuel dissolution and ion-exchange separation have been carried out for burn-up analysis. As for the TRU nitrides research program, thermal property measurement such as the thermal diffusivity or the specific heat capacity has been continued.

2.3. Research Hot Laboratory (RHL)

The RHL has been established in 1961 as the first PIE facility in Japan for conducting PIEs of fuels and materials irradiated in research reactors. Many kinds of PIEs have been performed during 42 years for the contribution of research programs in JAEA. However, RHL was selected the one of target as the decommissioning facility in the rationalization program for decrepit facilities. Therefore, all PIEs had been finished in March 2003 and we have been started the destruction of hot cell. And the partial area of RHL facility will be used for the temporary storage of un- irradiated fuel samples.

3. DEVELOPMENT OF NEW PIES APPARATUSES AND TECHNIQUES IN RFEF

As the high burn-up fuel, the temperature at the center region of fuel pellet is gradually rise up during irradiation caused by the change of thermal conductivity. The thermal conductivity of irradiated fuel pellets is tend to decreased by increasing the fission products (FP) and irradiation defects in the pellet. The thermal conductivity (λ) is calculated by multiplying the thermal diffusivity (α), the Specific heat capacity (Cp) and the density (ρ), as follows:

$$\lambda = \alpha \times Cp \times \rho \qquad (1)$$

Three kinds of apparatuses have been developed to determine the thermal conductivity of high burn-up fuel pellets.

•Pellet Thermal Diffusivity Measurement Apparatus (PTD)

•Specific Heat Capacity Measurement Apparatus (SHC)

•Micro Density Measurement Apparatus (MDM)

3.1. Outline of the PTD

Figure 1 shows the schematic diagram of the PTD. PTD is composed of the sample holding device, the laser oscillator, vacuum pumps, the tungsten mesh heater, the In-Sb infrared detector, the radiation shielding box, the hood, control units and the computer system. Several parts except for the computer system and control units were covered with the hood made of acrylic for preventing the dispersion of radioactive materials. The maximum radioactivity of loading sample is 43 GBq. The apparatus can be operated by remote handling.

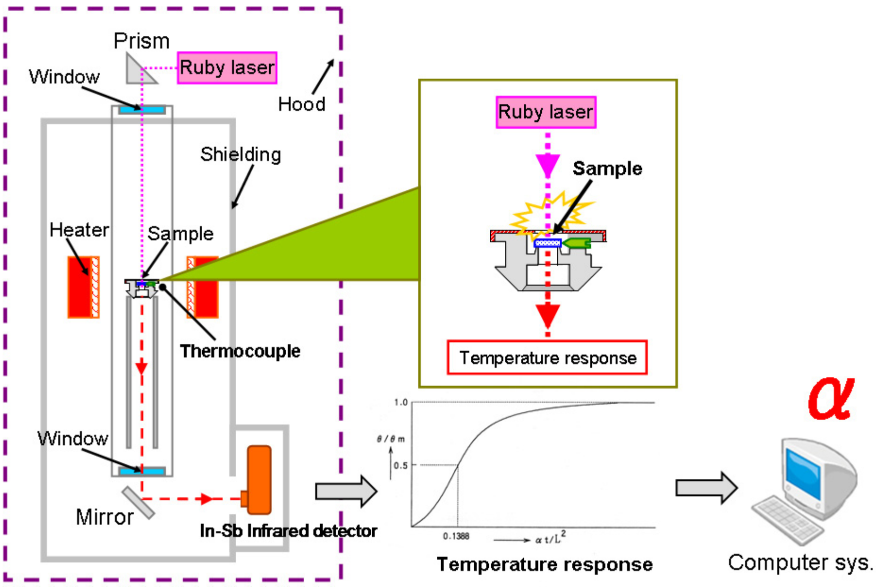

Fig.1. Schematic diagram of PTD

The thermal diffusivity of irradiated sample is measured with the laser flash method. The laser oscillation uses the ruby laser with the maximum energy about 6 J/pulse, the wave length of 694.3 nm and pulse width of 0.5 msec. The sample room is kept in a vacuum condition of less than 2×10^{-4} Pa during experiments by using turbo-molecular and oil-rotary pumps. A tungsten mesh heater is used for heating the sample to keep the examination temperature. The examination temperature range is from R.T to 1800 degrees. The sample temperature is monitored by a W-Re thermocouple located near the sample holder. The thermal energy is induced on the sample by flashing a ruby laser beam. And the temperature response of the sample measured with an In-Sb infrared detector. The temperature response data is transferred to a computer system and the thermal diffusivity is calculated. The thermal diffusivity (**α**) can be calculated by the following equation:

$$\alpha = L^2 / 4t_L \qquad (2)$$

Where: α, the thermal diffusivity
L, the thickness of the sample
t_L, the slope of the temperature response

As an example, the result of thermal diffusivities of the irradiated UO_2 pellet comparison with the un-irradiated UO_2 pellet is shown in Figure.2. The thermal diffusivities were measured from R.T to 1500 degrees. The thermal diffusivity of the irradiated UO_2 pellet could be lower than that of the un-irradiated UO_2 pellet at temperature above 1,000 ℃ because of accumulating fission products (FP) and irradiation induced defects in the UO_2 pellet. The result of this examination shows the thermal diffusivity of the irradiated UO_2 pellet decrease and the deference between these values changed remarkable at the lower temperature conditions.

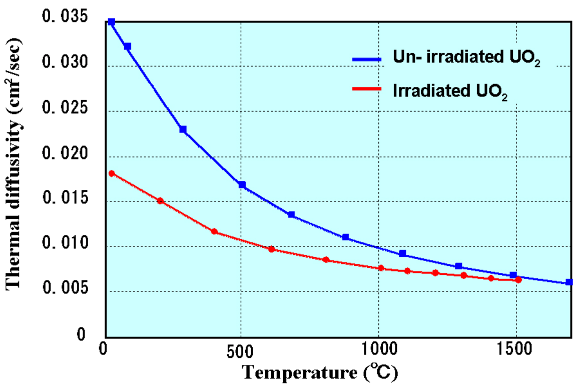

Fig.2. Thermal diffusivities of the irradiated UO_2 pellet and the un-irradiated UO_2 pellet

3.2. Outline of the SHC

Figure 3 shows the schematic diagram of the SHC. SHC is composed of the sensor unit, the base unit and the computer system. The sensor unit is composed of the heating furnace, the two sample pans, the sample chamber and the sample transfer system. The range of temperature is from R.T to 1500 degrees. The temperature is measured with type-R thermocouple. The sample pan and sample holder are made of platinum. The base unit is composed of the temperature control circuit and data transfer system. The base unit is set at the outside of shielding box to avoid radiation damage of the electronic circuit. The apparatus can be operated by remote handling.

The SHC is to measure the thermal capacity by heat-flux type Differential Scanning Calorimeter (DSC). The heat-flux type DSC is generally accepted to collect specific heat capacity data in the high temperature region. The basic composition of heat-flux type DSC and the measuring principle of specific heat capacity are shown in Figure 4. The specific heat capacity is measured as follows:

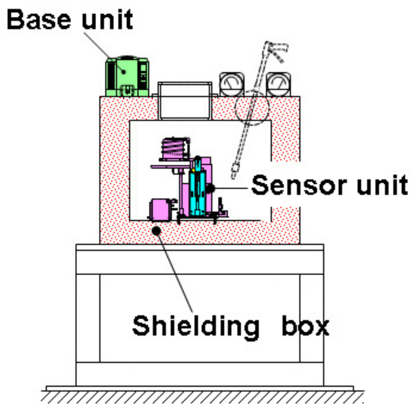

Fig.3.Schematic diagram of SHC

The SHC is to measure the thermal capacity by heat-flux type Differential Scanning Calorimeter (DSC). The heat-flux type DSC is generally accepted to collect specific heat capacity data in the high temperature region. The basic composition of heat-flux type DSC and the measuring principle of specific heat capacity are shown in Figure 4. The specific heat capacity is measured as follows:

At first, the blank pan and sample pan are put on the each holder of heating furnace by using vacuum tweezers. Next, the temperature is controlled with constant heating velocity. The measuring data is transferred to the computer system and the density value is calculated. The specific heat capacity (**Cp**) can be calculated by the following equation:

$$Cp = \frac{Wst \times Hsa}{Wsa \times Hst} \times Cpst \qquad (3)$$

Where:
Wst, the standard sample weight
Wsa, the measured sample weight
Hsa, the base line shift quantity of the measured sample (Hsa=C-A)
Hst, the base line shift quantity of the standard sample (Hst=B-A)
Cpst, the standard sample specific heat capacity at the T_1 temperature

SHC is in the cold mock-up test process.

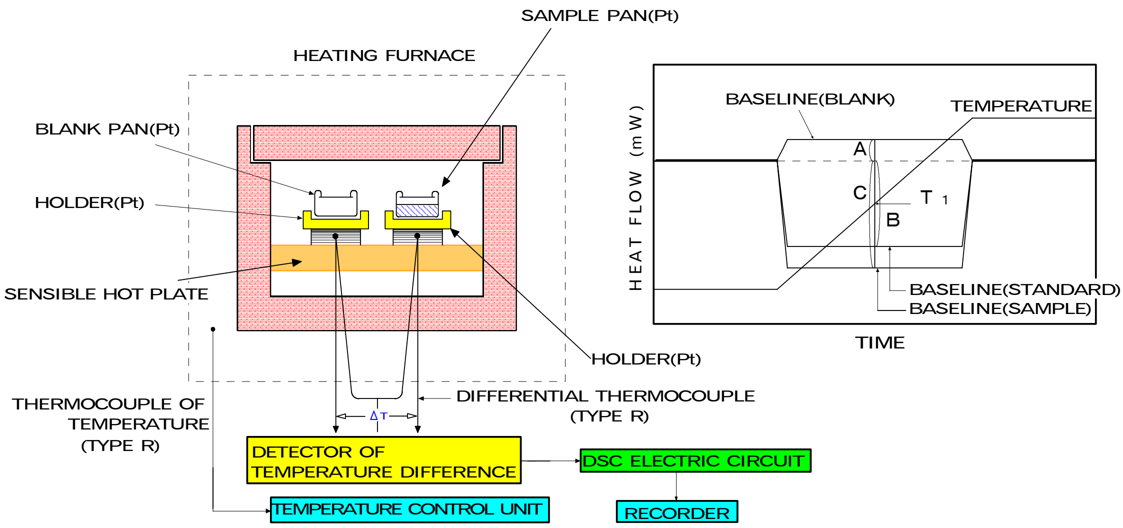

Fig.4.Basic composition of Heat-Flux Type DSC and measuring principle of Specific Heat Capacity

3.3. Outline of the MDM

The density of the small fragments in the pellet is important information for the determination of the swelling rate in the small area and for the calculation of the thermal conductivity. However, the conventional densitometer for irradiated sample in RFEF is designed for measuring the whole fuel pellet and is not available for precise density measurement of the small size sample. The MDM has been developed to measure the density of the small size sample precisely.

The density of the fuel pellet is calculated using the value of sample weight and wet weight (buoyancy) in the immersion liquid by Immersion density method. The immersion liquid was the water contained with a surfactant. Figure 5 shows the schematic diagram of MDM. The apparatus is composed of the balance device, the sample transfer unit and the data processing units. This apparatus can be operated by remote handling.

The sample transfer unit is equipped with twin baskets. Twin baskets set on a string vertically for both the dry and wet weight (buoyancy) measurements. Before measurement, wet weight (buoyancy) and weight of the baskets are offset.

MDM has the once-through system. Figure 6 shows once-through sample loading system.

This system is possible to perform the automatic measurement. At the first process, sample is set into dry basket using small conveyor belt to measure the sample weight. Then, dry basket is lifted up and the sample is dropped down to the wet basket to measure the weight (buoyancy) in the immersion liquid.

When the measurement is finished, the wet basket is lifted up and the sample is automatically transferred out the measuring unit.

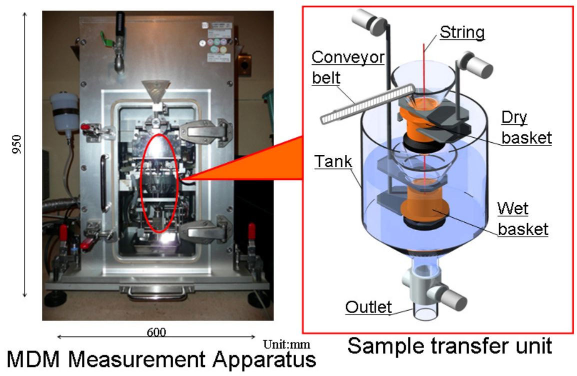

Fig.5. Schematic diagram of MDM

The measuring data is transferred to the computer system and the density value is calculated. The density (**ρ**) can be calculated by the following equation:

$$\rho = \frac{(\rho_a \times W_w - \rho_l \times W_d)}{(W_w - W_d)}$$

Where:

ρ_a, the density of the air
ρ_l, the density of the immersion liquid
W_d, the sample weight in the air
W_w, the sample weight in the immersion liquid

Now MDM is in the cold mock-up test process.

4. SUMMARY

The Department of Hot Laboratories and Facilities (DHL) have managed three PIEs facilities. The DHL progress in the decommissioning work of the facilities with re-arrangement and concentration of PIE items. And development of new PIE techniques for extended burn-up fuels, such as improvement of de-fueling machine and development of miniature sample preparation apparatus for the mechanical property test of cladding tube and development of chemical analysis technique for the determination of burn-up and isotopic abundance.

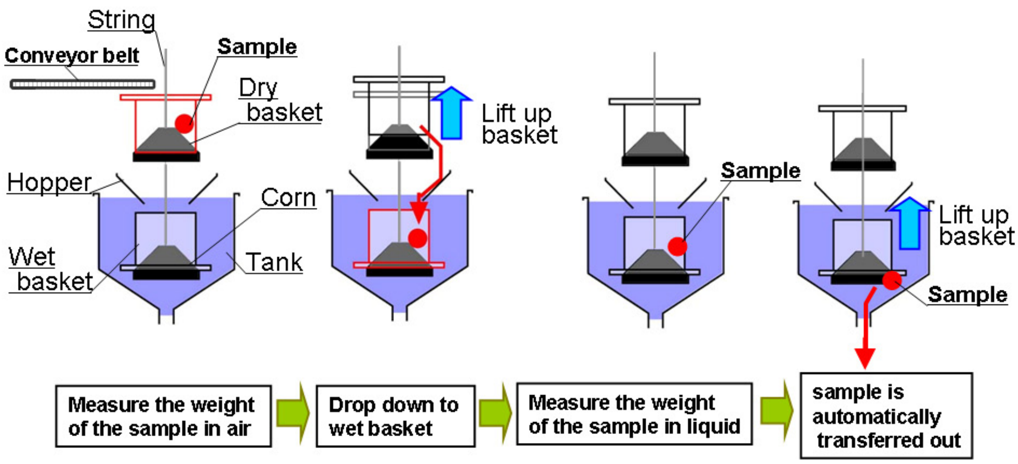

Fig.6. Once-through system

In RFEF three kinds of apparatuses have been developed to determine the thermal conductivity of high burn-up fuel pellets. PTD provide the thermal diffusivity data of the irradiated pellet. SHC and MDM are in the cold mock-up test process. In near future, thermal diffusivity (α), specific teat capacity (Cp) and density (ρ) of the irradiated pellet can be measured by developed apparatus. Therefore, we will determine the thermal conductivity of irradiated pellet.

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REFERENCES

[1] Annual Report on Operation and Management of Hot Laboratories and Facilities (From April 1, 2006 to March 31, 2007); JAEA-Review 2007-059

[2] Progress of Nuclear Safety Research-2003; JAERI-Review 2004-010, PP.132-135

[3] Isao OWADA; DEVELOPMENT OF REMOTE HANDLING APPARATUS FOR MEASURING THERMAL DIFFUSIVITY

[4] Isao OWADA; Development of a Shielded Pellet Thermal Conductivity Measurement Apparatus; JAERI-M 93-244

[5] Masaki AMAYA; Thermal conductivities of irradiated UO_2 and $(U,Gd)O_2$ pellets; Journal of Nuclear Materials 300 (2002) 57-64

[6] Atsushi ONOZAWA; Development of Once-through Type Densitometer; JAERI-Tech 2004-061

[7] Katsuya HARADA; Development and Application of PIE Apparatuses for High-burnup LWR Fuels; JAERI-Conf 99-009; PP.103-111

5. Utilization with Material Testing Reactors

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5.1 From MYRRHA to XT-ADS

-Development of Pb-Bi cooled ADS as a fast spectrum irradiation facility in Europe-

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The MYRRHA project started in 1998 by SCK•CEN in collaboration with Ion Beam Applications (IBA, Louvain-la-Neuve), as an upgrade of the ADONIS project. MYRRHA is designed as a multi-purpose irradiation facility in order to support research programmes on fission and fusion reactor structural materials and nuclear fuel development. Applications of these are found in ADS systems and in present generation as well as in next generation critical reactors. The first objective of MYRRHA however, will be to demonstrate on one hand the ADS concept at a reasonable power level and on the other hand the technological feasibility of transmutation of Minor Actinides (MA) and Long-Lived Fission Products (LLFP) arising from the reprocessing of radioactive waste. MYRRHA will also help the development of the Pb-alloys technology needed for the LFR (Lead Fast Reactor) Gen.IV concept.
Transmutation of MA can be completed in an efficient way in fast neutron spectrum facilities. Both critical reactors and sub-critical Accelerator Driven Systems (ADS) are potential candidates as dedicated transmutation systems. However, critical reactors, heavily loaded with fuel containing large amounts of MA, pose safety problems caused by unfavourable reactivity coefficients and small delayed neutron fraction. A sub-critical ADS operates in a flexible and safe manner even with a core loading containing a high amount of MA leading to a high transmutation rate. Besides the reduction of the HLW burden, the MYRRHA project will serve the purpose of developing the lead alloys technology as a reactor coolant that can be used in one of the Generation IV reactor concepts namely the Lead Fast Reactor (LFR).
This project will trigger the development of various innovative technologies and techniques that are of interest for various nuclear fission and fusion applications. Some of them are described in the paper.
Since April 2005, MYRRHA is serving as a basis for the development of the European experimental ADS (XT-ADS) in the frame of the European Commission FP6 project EUROTRANS. The status of MYRRHA project and its capabilities as a fast neutron irradiation facility are described in this paper.

Keywords: MYRRHA, Accelerator Driven Systems, Fast neutron irradiation facility, transmutation of MA and LLFP

1. INTRODUCTION

A survey of the future needs for research reactors in Europe has been discussed in depth and shared in Europe since 2002 [1]. This survey took into account the needs of the nuclear industry, the strategic importance of future GEN IV reactors developments, the advanced fuel cycle and the public health stakes. It appears that a European policy must include a mid-term roadmap for establishing a European Research Area on Experimental Reactors (ERAER) encompassing at least:

- a high performance material testing reactor
- a reactor optimized for medical applications
- an experimental reactor for innovative fast neutron reactor technology development with capabilities related to test advanced fuel cycles.

In view of the important and long term needs of irradiation facilities making use of fast energy spectrum neutrons, the SCK•CEN teams have designed an original, multi-purpose, largely innovating, research infrastructure called MYRRHA (for **M**ulti-purpose h**Y**brid **R**esearch **R**eactor for **H**igh-tech **A**pplications). MYRRHA consists of a high power proton accelerator, coupled to a lead-bismuth cooled subcritical reactor by means of a spallation target also made of liquid lead-bismuth alloy. MYRRHA answers the needs for a research infrastructure in the fields of:

- transmutation of high-level radioactive wastes,
- development of innovative materials for future energy systems,
- medical radioisotopes production.

MYRRHA is seen as a key element of the European Research Area on Experimental Reactors (ERAER) that allows for:

- continuing studies in the framework of nuclear reactor safety,
- fusion development,
- fundamental research and an adequate tool for the training of specialists in nuclear sciences and technology.

Presently in the 6th Framework Programme (FP6) of the European Commission, MYRRHA is serving as the basis within the integrated project EUROTRANS [2]. At the end of this project in March 2009, the conceptual design of the machine and of its different components will be available. During the period 2009 – 2013, the following tasks will be accomplished in parallel:

- detailed engineering design (2009 – 2011);
- drafting of the technical specifications of the different lots, publication of the call for tenders, comparison of the technical and financial proposals from the contractors and finally awarding of the manufacturing contracts (2012 – 2013);
- development and testing of key innovative components for the accelerator and for the reactor;
- licensing and permitting activities, namely preparation of a Preliminary Safety Assessment Report (PSAR), of the Environment Impact Assessment (EIA) and of the Preliminary Dismantling Plan (PDP). The objective being to obtain the authorization of construction at the end of 2013.

The construction period of the components and the civil engineering work is to be accomplished in a three-year period (2014 – 2016) followed by a one year assembling together of the different components in 2017. The commissioning at progressive levels of power will be accomplished in two year period (2018 – 2019) with the final objective to be in full power operation in 2020.

The main characteristics and innovative features of MYRRHA are briefly reported in this paper. A more detailed description can be found in reference [3].

2. DESIGN OBJECTIVES AND CHARACTERISTICS

Having sketched the scope of MYRRHA and the SCK•CEN objective to meet the international needs in terms of demonstration and irradiation, having listed the targeted catalogue of applications for this facility, in this section we are giving the rationale for the design characteristics of MYRRHA.

As a direct consequence of core performance argument, the central hole in the core (which houses the spallation target) should be of limited dimensions (~10 cm) in order to keep many locations available in the core (to install experiments) where the neutron fluxes are sufficiently high. Because of the desired high fast flux and the high power density gaseous coolants are not eligible for short or medium term deployment of MYRRHA; therefore we opted for liquid metal as coolant. Sodium has not been retained due to the chemical reactivity with water and air and hence fire safety risks. The alternative liquid metal as a coolant were Pb or Pb-Bi and we opted for the latest due to its low melting temperature (123°C), allowing the primary systems to function at rather low temperature levels. The low working temperatures are evident mitigating approach to limit the corrosion problems due to HLM.

Furthermore, we chose from the very beginning of the project a windowless target design because of the high beam load on a hypothetic window that would sustain a proton current density of c.a. 140 μA/cm². No existing material can withstand these conditions. Even if such a design is quite challenging, the experimental and theoretical design work of the windowless target and associated R&D programme conducted since 1998, is giving more and more evidence that the windowless spallation target based on HLM would be feasible.

The sub-criticality level (ks) of 0.95 has been considered as an appropriate level for a first of kind medium-scale ADS. Indeed, this is the criticality level accepted by the safety authorities for fuel storage and would allow for the MYRRHA concept to remain sub-critical in all circumstances even when account for the possible positive reactivity injections.

Taking into account the time schedule envisioned for the deployment of MYRRHA, start up before 2020, the MYRRHA project team decided to use the most mature technologies whenever possible and in particular for the choice of the MYRRHA MOX fast reactor fuel technology has been chosen due to the large experience in Europe and in particular in Belgium. Based on the catalogue of applications listed above and the needed flux levels, a maximum linear power in the MOX fuel is set to 400 W/cm and considering the sub-criticality set up (0.95), the accelerator current intensity for a fixed proton energy of 600 MeV, the total core power as well as the core power density are resulting values from the core design optimization.

To take profit of the thermal inertia provided by a large coolant volume, we opted for a pool-type system in which the components of the primary loop (pumps, heat exchangers, fuel handling tools, experimental rigs, etc) are inserted from the top in penetrations in the cover. The fuel assemblies loaded is foreseen to be from underneath, which is not the classical approach of the sodium fast reactors, but the used technology is the same and the reasons behind the approach is to hold a large flexibility for the experimental devices loading and from the safety point of view as all structures including the spallation module are in place before starting the core loading.

The pool vessel, which contains the core of the MYRRHA machine and the whole series of internals (see Fig. 1), is located in an air-controlled containment environment. Furthermore, several factors lead to the decision to design both operation and maintenance (O&M) and In-Service Inspection & Repair (ISI&R) of MYRRHA with fully-remote handling systems, among them:

- the high availability rate desired for the machine (70 to 75%),
- the high activation on the top of the reactor (due to the neutron leakage through the beam line),
- the Po contamination when extracting components,
- the non-visibility under Pb-Bi and
- the oxygen-free atmosphere in the MYRRHA hall.

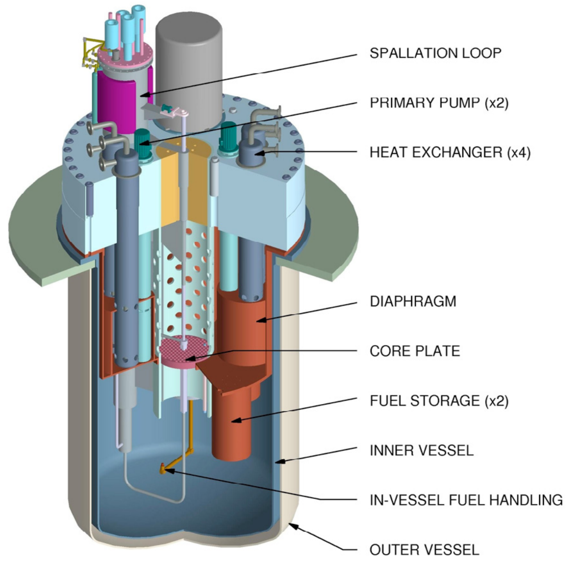

Fig. 1 Overview of current MYRRHA concept, with its main internal components.

The safety aspects of the present design include three main parts:

a) The radiation safety of the installation is investigated and compared with basic radiation protection requirements. The analysis evaluates the necessary shielding for the accelerator, beam line, beam dump and core. It evaluates also the production of activation and spallation products in the accelerator, beam line and spallation target.
b) The general safety analysis first identifies the safety functions that the safety systems must fulfil and the majority of the initiating events for undesired situations. The most limiting accident conditions taking into account bounding events are then investigated and the corresponding accident scenarios are explained. The analysis of the accidents was performed with two calculation codes, RELAP5 mod 3.2 and SITHER.
c) In addition to the general safety analysis, some Computational Fluid Dynamics (CFD) studies where a 3-D modelling is required, were undertaken to deal with specific aspects related to safety considerations.

3. SOME INNOVATIVE ASPECTS OF MYRRHA

3.1 Linear Accelerator System

The main characteristics for the MYRRHA proton beam are presented in Table 1. Each property is the result of technical evaluations based on the requirements mentioned above. The inner diameter of the doughnut footprint depends on the size of the recirculation zone, and thus on the details of the target design.

Table 1 Proton beam specifications.

Proton energy	350 MeV (possible upgrade to 600 MeV)
Max. beam intensity	5 mA Continuous Wave on target (2.5 mA for 600 MeV)
Beam entry	Vertically from above
Beam trip number	Less than 5-10 per year (exceeding 1 second)
Beam stability	Energy: ± 1 %, Intensity: ± 2 %, Size: ± 10 %
Beam footprint on target	Circular, "doughnut" ϕout=72mm, ϕout≈30mm

At present megawatt beam power proton accelerators have been produced in only two basic concepts: sector-focused cyclotrons and linear accelerators. The fundamental difference between the two types is that cyclotrons are essentially monolithic circular devices whereas Linacs that are built from many repetitive accelerating sections are highly modular. Clearly, the cyclotron option is attractive with respect to construction costs and the required beam parameters are within the feasibility of the cyclotron concept. However, because of the compact nature of cyclotrons, they are not modular so a given cyclotron is not upgradeable in energy nor is it well suited for the reliability strategy mentioned above. An equivalent superconducting linear accelerator on the other hand has higher construction costs. Yet, they are highly modular and upgradeable without fundamental limitations in energy and intensity. These properties give an excellent potential for optimised operation costs because of beam efficiency and for high reliability based on the fault tolerance concept. Because of these arguments the Linac has been chosen in the MYRRHA project as reference solution for the accelerator whereas the cyclotron option is retained as back-up solution.

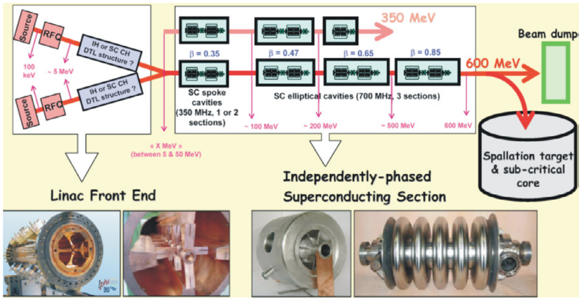

Fig. 2 Schematic layout of the reference design of the Linac for MYRRHA.

The Linac system is based on the concept of over-design, redundancy and fault tolerance. A basic layout of the Linac design is found in Fig. 2. The low energy section uses existing technology for high intensity proton injectors. For MYRRHA a combination of an ECR (electron-cyclotron resonance) ion source and a RFQ (radio frequency quadrupole) accelerating cavity is proposed. This device will create the proton beam and provide the first stage acceleration to 5 MeV. Since the ion source and the initial RFQ cannot be made modular like the high energy part of the Linac, the redundancy principle is applied. Thus, two ECR-RFQ units will be installed to allow rapid takeover by the back-up unit in case of failure.

In the high energy part above energies of about 100 MeV, the well established technology of superconducting multicell elliptical cavities will be used. These have the advantage of high performance regarding accelerating gradients, efficiency, security, reliability and modularity. Finally, they allow comfortable margins on critical values to ascertain a design that is as robust as possible. Combining all cavities, the total length of the high energy part of the MYRRHA Linac would run up to about 111 m.

In the design of the intermediate energy section of the MYRRHA Linac two concepts have been retained : the extension of the injector philosophy towards higher energies using room temperature DTL (drift tube Linac) type structures or alternatively extend the high-energy superconducting (SC) Linac philosophy towards lower energies, low β superconducting resonators. The parameter β here is the ratio between the velocity of the particles in the cavities and the speed of light. It describes to what extent relativistic effects need to be taken into account.

3.2 Liquid metal spallation target

In order to reach the required neutronic performance of the ADS, the target should produce about 10^{17} neutrons/s to feed the sub-critical core at its keff value of 0.95. To do this, it must accept the 350 MeV, 5 mA proton beam delivered by the accelerator. Consequently the target has to be able to evacuate the 1.43 MW heat (81.7 % of 1.75 MW) deposited by the beam. This fact calls for a liquid metal target to allow forced convective heat removal. Liquid lead-bismuth eutectic (LBE) has been chosen as target material. Higher proton energy of 600 MeV would generate the same source intensity with a reduced current of 2.5 mA. On the other hand this reduces the energy deposition in the target and the power density as well due to a larger proton penetration (30 instead of 12 cm). LBE is also the sub-critical core main coolant although both two liquids are separated. The target evidently must fit into the central hole in the sub-critical core that is created by removing three fuel assemblies. Finally the spallation target must reach an appropriate lifetime and it must comply with the role of MYRRHA as a flexible experimental irradiation device.

Inside the compact core geometry only a ϕ72 mm effective target surface is possible. Together with the beam properties, this leads to a beam current density of 125-175 $\mu A/cm^2$, which effectively prohibits the use of a target window close to the spallation zone since it would not realistically survive the harsh conditions for a sufficient period of time. A windowless design means that vacuum conditions of 10^{-5}-10^{-6} Pa above the target surface must be reached to avoid plasma formation and to guarantee compatibility with the vacuum of the accelerator beam line.

The most important part of the spallation target development is the design of the target nozzle. In the current three-feeder design, the LBE flows downwards by gravity through three drag-limited feeders. It is distributed over the full circumference by an annular transition piece and then directed towards the central tube in a funnel-like configuration (Fig. 3). The nozzle is shaped to minimize the recirculation zone that occurs in the centre of the funnel while also avoiding droplet spitting, in order not to jeopardize the beam vacuum. The axis-symmetric target configuration has been under investigation at SCK•CEN and UCL for a number of years. It was demonstrated to exist in a suitable shape in to-scale water, mercury and LBE experiments. In the experiments it was found that a speed match between the annular feeder and the central tube has positive effects on the surface stability.

3.3 Sub-critical core configuration and loading

One of the main objectives of the MYRRHA ADS is to yield very high neutron fluxes within a small size sub-critical core (Fig. 4). The core consists of a lattice of 99 hexagonal channels of which 45 are loaded with fuel assemblies housing 30wt% Pu-enriched (Pu/HM; HM=Heavy Metal) MOX fuel pins arranged in a triangular pitch of 8.55 mm. The fuel rods have an active length of 60 cm. The $(U\text{-}Pu)O_2$ fuel pellets have a density of 10.55 g/cm^3 (95%TD) and each assembly contains 91 fuel pins yielding a 514 kg heavy metal load for the fresh core.

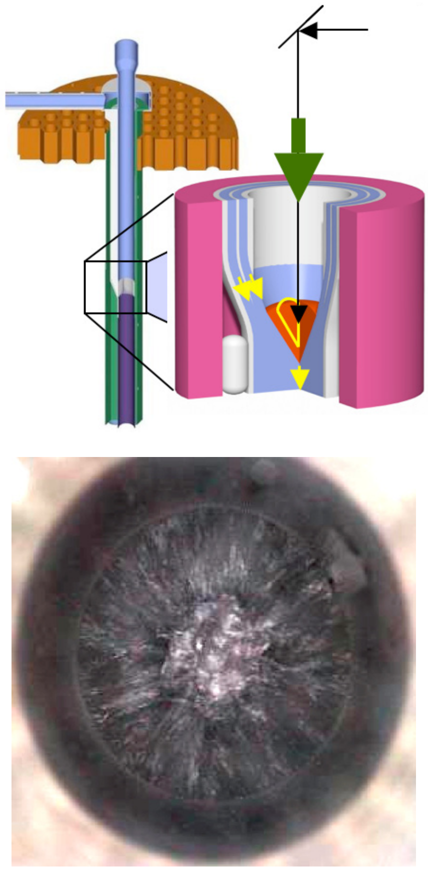

Fig. 3 Top: schematics of the axis-symmetric windowless target configuration; Bottom: top view of the LBE flow in the spallation target unit with the recirculation zone in the centre.

A space of 3 hexagons is cleared at the centre of the sub-critical core to house the spallation target module. The spallation target consists of a windowless column of eutectic Pb-Bi inside a 4-mm-thick 9%-Cr martensitic steel pipe having an inner diameter of 72 mm. The liquid metal target free surface lies 75 mm above the core mid-plane to have an optimum neutron yield and to achieve an axial symmetrical power and flux distribution in the core.

A typical core loading pattern adopted to carry out a preliminary assessment of the potential MYRRHA design concept to the MA transmutations and LLFP incineration in a fast spectrum is displayed in Fig. 3. This shows the MYRRHA core flexibility. The minor actinide load consists of six assemblies similar in geometry to the driver ones, but housing the fuel rods with pellets containing IMF (inert matrix fuel): 40 vol.% $(Pu_{0.4}Am_{0.5}Cm_{0.1})O_{1.88}$ fuel and 60 vol.% MgO matrix. Besides a shorter active length (400 mm), the dimensions of these rods are the same as those of the driver ones. The density of the IMF pellet is 6.077g/cm^3 (~90 % TD) yielding a total amount of 7.24 kg low quality Pu (from PWR spent fuel), 9.04 kg of americium and 1.81 kg of curium irradiated in fast spectrum channels during a 3-years campaign. The calculations yield a net decrease of 2.48 kg in the actinide mass, mainly due to the removal of americium (-2.46 kg). There is a net increase of 0.46 kg in the curium mass due to the build-up of ^{242}Cu and ^{244}Cu from the decay or neutron capture of americium. The burned-out mass of plutonium is 0.51 kg.

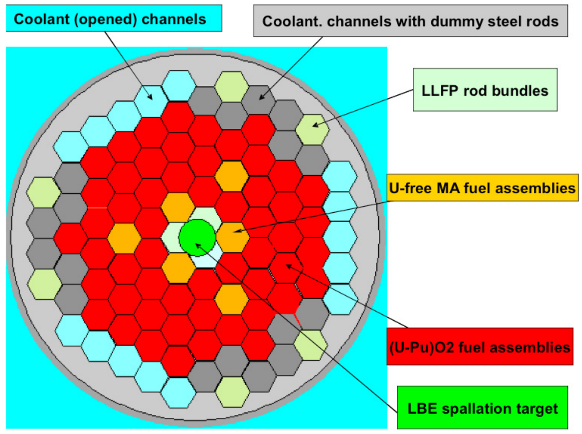

Fig. 4 A dedicated core configuration for MA and LLFP transmutation studies.

3.4 Cladding material

The choice of cladding material is of critical importance both from economic and safety viewpoints. Ferritic-martensitic steel (FMS) T91 has been chosen as the main candidate for the fuel cladding in MYRRHA. This choice was based of the fact that T91 shows a lower swelling rate and embrittlement under irradiation at T > 350°C [5], and higher resistance to dissolution in the oxygen-free LBE, compared to austenitic steels [6]. However, taking into account that all FMS suffer strongly from irradiation embrittlement below 350°C and show higher corrosion rate in present of oxygen, the well-known austenitic steels 15-15 Ti, AISI 316 L and few Russian steels were kept as a backup solution. These steels have already demonstrated their good performances in LMFBR where they were used as the fuel cladding material.

The available experimental data on LBE technology and corrosion resistance of different steels in contact with LBE indicate that their long-term operation (> 10.000 h) is possible only at temperatures lower than 560°C [7]. Therefore, the allowable maximum local temperature of 450°C has been chosen for the LBE coolant normal operation. The minimum coolant temperature has a natural limit of 123°C which is the LBE melting temperature. In order to have a technological margin, the minimum LBE operation temperature of 200°C has been chosen at this stage of the pre-design. However, one should keep in mind that embrittlement problems related to joint effects of LBE and neutron damage on the cladding materials suggests increasing this temperature up to 250-300°C. The LBE velocity limit of 2 m/s has been fixed because of possible erosion of structures during a long-term operation.

3.5 Other innovative features

Besides the above innovative aspects, the project includes several other new important features. Let us mention:

- The development of the heavy liquid metal technology in terms of pumping, conditioning, filtering, monitoring;
- The development of ultrasonic visualisation systems (sensors, full-camera) able to operate under liquid metals at high temperature (300-500°C) and high dose rates (many MGy of gamma-rays and neutrons);
- The development of advanced remote handling (robotics) systems able to operate in radioactive environment and also under heavy liquid metals.

4. CONCLUSIONS

As already indicated above, the generation of research reactors commissioned in the 60's is coming to an end. The European Commission, anxious to build the European research area, is questioning itself on the needs for experimental reactors in the future. As mentioned in this paper, the problem deals with the irradiation needs for medical applications, fundamental research, industrial applications, and the studies on materials and fuels of future nuclear power plants. Moreover, the research reactors provide a unique contribution to the education of a new generation in nuclear science and technology.

In this framework, France, has proposed with success to the European partners to take part in the construction and operation of the future Jules Horowitz reactor in Cadarache, and the Netherlands have introduced the PALLAS project, a reactor dedicated among others to the production of radioisotopes at the Centre of Petten. However, these two projects for new installations do not cover the whole spectrum of needs. The MYRRHA project that SCK•CEN proposes to build on its site (see Fig. 5) is an experimental set-up generating a neutron spectrum that would be able to perform waste transmutation experiments, and the study of future nuclear systems whose objectives include no production of the most radiotoxic waste. It has been shown in this paper that MYRRHA involves fascinating challenges for present and next generations of researchers and industrialists. International support is required to include this project in the future European research platform.

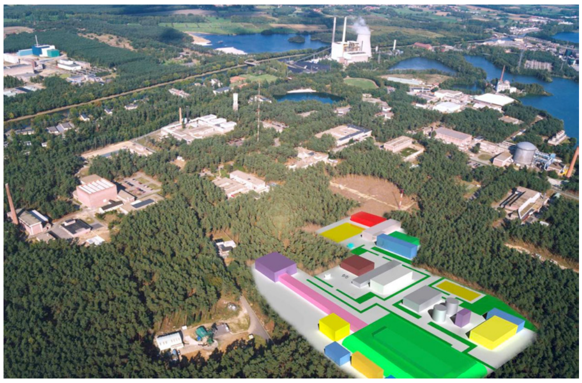

Fig. 5 MYRRHA complex on the technical site of SCK•CEN in Mol.

ACKNOWLEDGMENTS

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REFERENCES

[1] FEUNMARR: "Future EU Needs in Material Research Reactors", FP5 Thematic Network FIR1-CT2001-20122.

[2] FI6W-CT-2004-516520: Integrated Project on European Transmutation (EUROTRANS)

[3] MYRRHA Project – Technical description, SCK•CEN Report ANS/HAA/DDB/3900.B043000 /85/07-17bis, April 2007, 57 p.

[4] A.C. Rolfe, S.F. Mills & S. Sanders (2002) "MYRRHA Remote Handling Study". OTL Final report for SCK•CEN. Contract CO-9002 1581.00, 60 pg. + appendices.

[5] R. L. Klueh, D. S. Gelles, S. Jitsukawa, A. Kimura, G R. Odette, B. van der Schaaf, and M. Victoria, "Ferritic/martensitic steels – overview of recent results." J. Nucl. Mater. 307-311, 2002, pp. 455-465.

[6] V. Sobolev, "Compatibility of structure materials of a typical ADS with heavy liquid metal coolants: Brief review of experimental results". Report R-3532, SCK·CEN, Mol, May 2001, 55 p.

[7] G S. Yachmenev, et al., "The Problems of Structural Material Corrosion in Lead-Bismuth Coolant". In: Proc. "Heavy Liquid Metal Coolants in Nuclear Technology", v. 1, Obninsk, RF, 1999, pp.133-140.

5.2 Beryllium Application for Fission and Fusion

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At present beryllium is widely used as neutron reflector and moderator in research nuclear reactors. On the other hand, beryllium is also considered as a plasma-facing material in the vacuum vessel of the ITER and as a neutron multiplier in the breeding blanket of the DEMO. Basic mechanisms of radiation damage in beryllium are common for both fission and fusion environments. Strong radiation embrittlement determines the state of beryllium under irradiation. The main problem both in fission and fusion is the storage of radioactive beryllium waste. As very attractive alternative the recycling of irradiated beryllium was recently proposed.

Keywords: reactor, beryllium, irradiation, embrittlement, swelling, microstructure, modeling, waste.

1. INTRODUCTION

There are about 30 research nuclear reactors in the world in which beryllium is used as neutron reflector and moderator, in particular the SM and MIR reactors, Dimitrovgrad, Russia. Irradiation leads to strong radiation embrittlement of beryllium blocks above a critical neutron dose, crack formation and propagation into blocks. In this situation it is necessary periodically to replace the irradiated blocks after achieve to new one.

Beryllium is also considered as a plasma-facing material in the vacuum vessel of the ITER and as a neutron multiplier in the breeding blanket of the DEMO. The European Power Plant Conceptual Study is based on a helium-cooled pebble bed (HCPB) blanket design. The Test Blanket Module (TBM) in the ITER will use beryllium as neutron multiplier in form of pebbles with diameter of 1 mm.

The main problem both in fission and fusion is the storage of radioactive beryllium waste. The amount of beryllium radwaste can achieve several hundreds tons. In this case the recycling of irradiated beryllium can be as very attractive process.

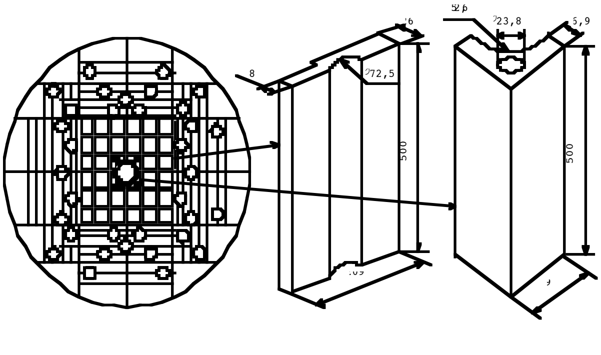
2. BERYLLIUM APPLICATION IN FISSION

2.1. Beryllium Reflector and Moderator in Research Reactors

The SM research reactor has one of the highest fast neutron flux in the world therefore it causes large interest as a place where it is possible to accumulate the strong radiation damage in irradiating materials for an acceptable time period [1]. The maximal fast neutron flux is in the core position channels where the neutron flux reaches of 2×10^{15} cm^{-2} (E>0.1MeV) in the central plane of core. Fig. 1(a) presents the cross-section section of the SM reactor core. The beryllium reflector which consists of separate blocks and is placed on perimeter of the core forms of a high neutron flux in the core. As a result the first row reflector blocks (Fig.1(b)) are exposed to the most radiation damage. The neutron trap is located in a center of the SM reactor core. It consists of four beryllium blocks (Fig.1(c)) which are as neutron moderator. As a result the higher thermal neutron flux is formed in the central cylindrical region limited by four beryllium blocks.

The MIR reactor core is placed in the pool with water and arranged of hexagonal beryllium blocks moderator. The cross-section of the MIR reactor core is showed on Fig.2(a). Beryllium blocks have hexagonal form (Fig.2(b)). The fuel assemblies are placed in the channels located inside beryllium blocks and the regulating rods are placed along joint planes of the next beryllium blocks.

During operation of the SM and MIR reactors as a material of blocks of a reflector and moderator beryllium oxide and metal beryllium were used. Last time there are mainly used the beryllium grades with the grain size of 60mm made by hot extrusion technology. Some characteristics of the TE-56 reactor beryllium grade are presented in the Table 1.



a b c

Fig.1. The SM reactor core:
(a: cross section, b: block of reflector; c: block of moderator)

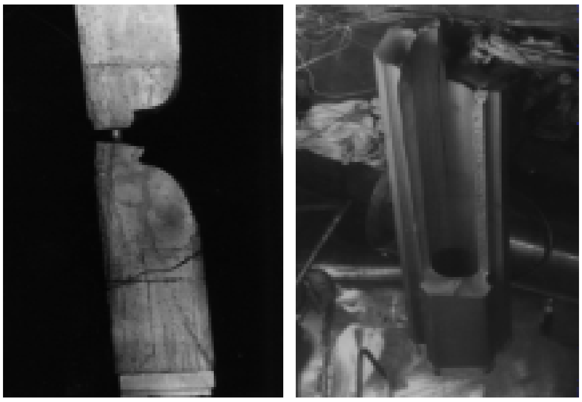


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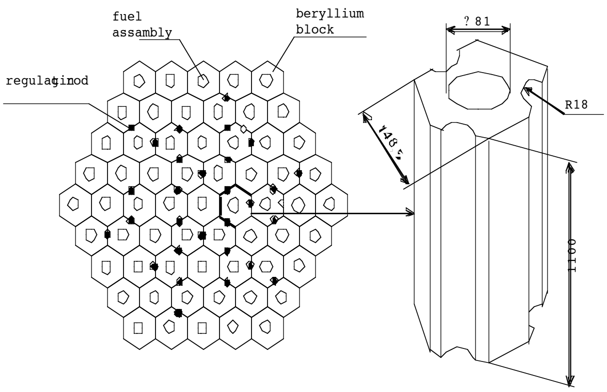
Fig.3. The views of the SM and MIR reactors irradiated beryllium blocks:
(a: block of the SM reactor, b: block of the MIR reactor)



a b

Fig.2. The MIR reactor core.
(a: cross section, b: block of moderator)

Table 1. The characteristics of the TE-56 beryllium grade.

Technology	Hot extrusion
Maximal grain size, μm	56
Middle grain size, μm	25
Content O,	0.98

2.2. Radiation Damage of Beryllium under Low Temperature High Dose Neutron Irradiation

The dependence of beryllium swelling on neutron fluence is showed in the Fig.4 [2]. The irradiation at temperature of 70℃ (Fig.4) results in monotonous increase of swelling with increase of neutron fluence F. At fluence of (9-10)×10^{22} cm^{-2} growth of swelling is slowed down and practically leaved to saturation at value of 3 %. After irradiation at temperature of 200°C the dose dependence of beryllium swelling has similar behaviour.

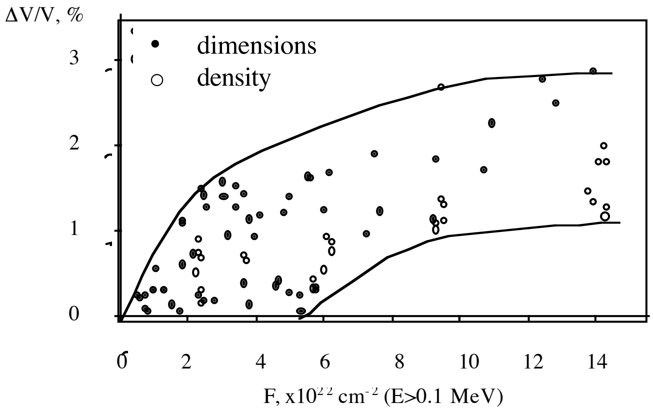


Fig. 4. Dependence of beryllium swelling ΔV/V on neutron fluence F for T_{irr}= 70°C

The dependence of beryllium microhardness on neutron fluence is showed in the Fig.5. The beryllium microhardness increases with growth of neutron fluence F as for the swelling dose dependence. It is gradually slowed down and leaved to saturation in the fluence range of (10-12)×$10^{22}$$cm^{-2}$. The maximal values of microhardness both at irradiation temperature

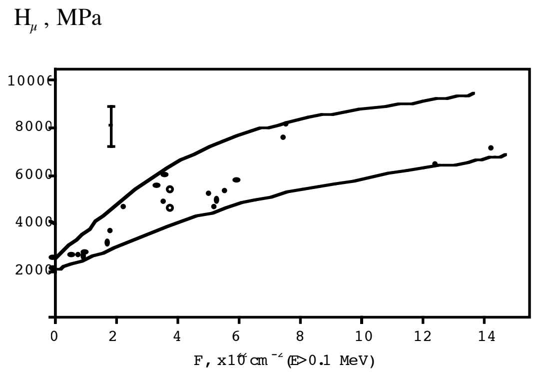


Fig.5. Dependence of beryllium microhardness H_μ on neutron fluence F for T_{irr}=70°C

of T_{irr}=70°C and at T_{irr}=200°C does not exceed of 8000 MPa.

All irradiated beryllium specimens under tensile and compression tests were absolutely brittle. Fig. 6 presents the dependence of beryllium ultimate tensile strength after irradiation at 70°C and testing at room temperature. The comparison of strength of the beryllium specimens, which have been cut out along the extrusion axis (Fig.6(a)) and across the axis (Fig.6(b)), shows that the strength of longitudinal specimens is essentially more then of cross-sectional ones both in initial state and after irradiation. Thus for both kinds of irradiated specimens there is a significant decrease in strength to growth of neutron fluence. The character of decrease is not monotonous, the maximal decrease in the strength occurs in the range before of neutron fluence F=3×10^{22} cm^{-2}. The recession and stabilization of strength at the level of 20-100MPa with further growth of neutron fluence takes place.

The results of compression tests after irradiation at 70°C show that the strength dependence on neutron fluence is similarly to tensile test data. However some differences take place. In particular the specimens cut along the axis are rather less strength by what the specimens cut across the axis.

The dependence of thermal conductivity specific factor for beryllium on neutron fluence at irradiation temperature of T_{irr} =70°C is showed at Fig.7. The drop in thermal conductivity occurs in the fluence range of F=(0-3)×10^{22} cm^{-2}. The growth of neutron fluence F =12·10^{22} cm^{-2} does not result in the further decrease of thermal conductivity, its value is kept at a level of 30-50W/m·K. The irradiation of beryllium at T_{irr} =200°C also leads to decrease in thermal conductivity, but it takes place with a smaller rate.

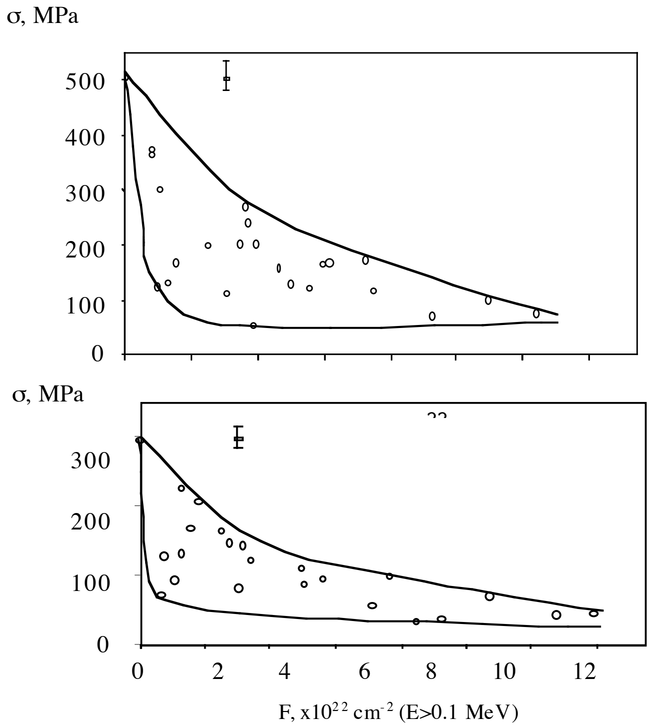


Fig.6. Dependence of ultimate tensile strength σ of beryllium on neutron fluence F for T_{irr}=70°C (a: along the axis, b: across the axis)

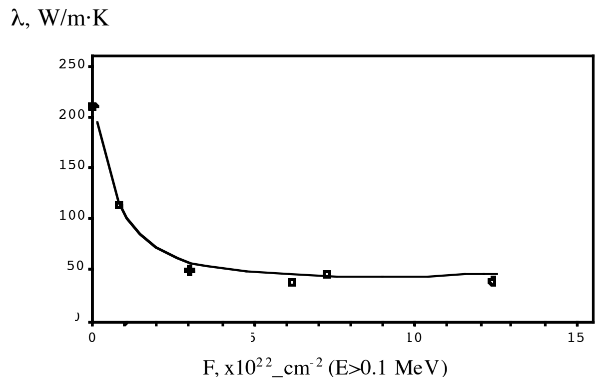


Fig. 7. Dependence of beryllium thermal conductivity λ on neutron fluence F for T_{irr}=70°C (T_{test}=T_{irr}).

2.3. Some Ways to Improve The Radiation Resistance of Beryllium

Fig.8 presents the influence of annealing at 500°C during 3h on mechanical properties of irradiated beryllium specimens. Annealing after irradiation results in partial recovering of mechanical properties for both as after tensile tests and after compression tests. Though after annealing the all specimens is still absolutely brittle raptured however the level of brittle strength after annealing grew. The degree of recovering rather differs from a specimen to a specimen. The significant dispersedness of values is caused by instability of strength level that is always observed at absolutely brittle rapture. However despite of essential dispersedness of experimental data, the tendency of recovering of mechanical properties at annealing after irradiation does not cause of doubts.

The irradiation of beryllium leads to increase in microhardness and annealing after irradiation results in its decrease, that is the partial return to initial level before irradiation (Fig.8). The size of the recovering effect is of 30-40 %.

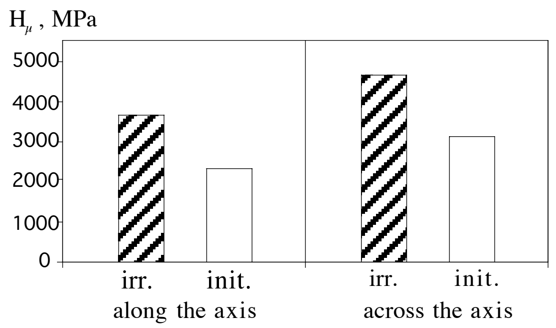


Fig.8. Influence of annealing at 500℃, 3h to microhardness of beryllium irradiated at T_{irr} =70°C up to F= 2×10^{22} cm^{-2}

Annealing after irradiation results also to recovery of thermal conductivity of irradiated beryllium, the effect has also as partial character (Fig.9). The increase in annealing temperature or in annealing duration is possible that will lead to increase of recovery effect.

It is necessary to understand that there is the serious restriction on annealing temperature value after low temperature irradiation of beryllium, which is expressed in swelling sharp increase with growth of annealing temperature (Fig.10). Especially brightly this effect is shown at the annealing temperatures of 800℃ and higher, when the swelling value can reach of 20-25%. Such substantial growth of beryllium volume can lead to problems with operation of beryllium blocks in a nuclear reactor.

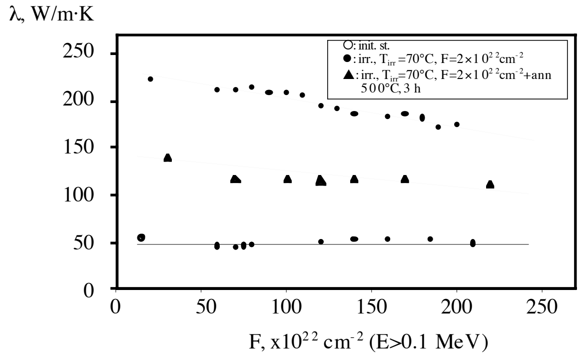


Fig. 9. Influence of annealing at 500°C, 3 h to thermal conductivity λ of beryllium.

The most detailed information about character of microstructure changes in irradiated beryllium under annealing after irradiation can be received by TEM examination. The basic radiation defects in beryllium irradiated at T_{irr}=70℃ are the dislocation loops. Presence of helium bubbles has only individual character. Amount of bubbles practically does not increase up to annealing temperature of 500°C. Only at this temperature the bubbles are formed in large amount. The bubble size is of 1-2 nm, and they are uniformly distributed in microstructure with high density (2×10^{23} m^{-3}). With increase in the annealing temperature the bubble size considerably grows, the volumetric density decreases. At the annealing temperature of 1200℃ the bubble size becomes comparable with the grain size.

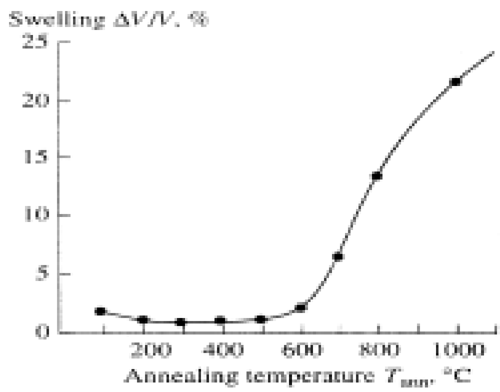


Fig. 10. Swelling of beryllium after irradiation at T_{irr}= 70°C, F=5.7×10^{22} cm^{-2} and next high temperature annealing during 1 h

3. BERYLLIUM APPLICATION AS NEUTRON MULTIPLIER IN DEMO

A key issue of Helium Cooled Pebble Bed (HCPB) blanket is its behaviour under fusion neutron irradiation. In the present European HCPB blanket concept and within a 40,000 hours lifetime, an integral neutron dose of typically $3x10^{26}$ n $m^{-2}s^{-1}$ (E_n> 1 MeV) results in the production of about 80 dpa, 25,700appm helium, and typically 640appm tritium in beryllium [3]. Depending on the local neutron spectrum, the helium/tritium total yield ratio can vary between 10 and 100. According to present HCPB blanket designs an amount of about 300 tons metallic beryllium is foreseen as a neutron multiplier in form of layers of small pebbles. The knowledge of tritium and helium accumulation as well as the gas release kinetics from the pebbles is crucial for the reliable and safe operation of fusion DEMO reactors.

Table 2. Operation conditions for beryllium at fusion reactor HCPB blanket.

Lifetime, hours	40,000
Operation temperature, K	700-1,050
Fast (E>1 MeV) neutron fluence, n/m^2	3×10^{26}
Total damage, displacement per atom, dpa	80
Total He production, atoms per million or appm	27,000
Total T production, appm	640

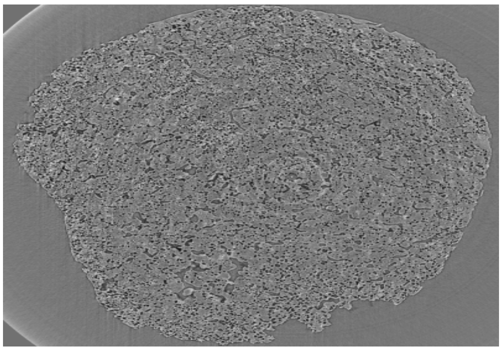
While even large amounts of about 20,000appm helium accumulated in beryllium used as a reflector material in the mixed spectrum reactor BR2 (Mol, Belgium) do not produce bubbles visible via TEM analysis after low temperature neutron irradiation, experiments encompassing post-irradiation temperature ramps revealed pronounced ^{4}He and ^{3}H release peaks [4,5], accompanied by a substantial increase of swelling. Significant swelling and creep were also found by other authors after high dose neutron irradiation at high temperatures or after low temperature irradiation followed by higher temperature annealing [6-9]. A combination of microstructural analysis and gas release measurements supports the evidence that at high temperatures the tritium inventory is concentrated either in helium bubbles or trapped in strain fields in the bubbles' vicinity and can be substantially released only together with helium, i.e. by the formation of open porosity networks often along grain interfaces followed by bubble venting. A holistic 3D view on bubble formation, growth, coalescence and network formation is critically important for the experimental validation of recently started multiscale modelling activities aiming to understand helium and tritium kinetics and predict tritium release characteristics at different irradiation temperatures and neutron doses.

While classical 2D microstructural analysis techniques do not actually suffice to carry out morphologic analyses of extended bubble networks and complete gas percolation paths, X-ray based tomography techniques have substantially enhanced their resolution limit to the micrometer and even sub-micrometer range during the past years. 3D computer aided microtomography (CMT) has already shown its capability in the non destructive analysis of the packing factor of beryllium beds [10,11] with a spatial resolution of 10μm, and its potential in analysing 3D porosity networks in irradiated beryllium [12]. In the present study, the initial results reported in ref. [12] are extended. The analysis includes now (i) the whole reconstructed volume of irradiated and unirradiated beryllium pebbles, (ii) 3D rendering of surface and porosity channels, and (iii) analysis of the open-to-total-porosity ratio.

Beryllium pebbles with a diameter of 2mm and grain sizes between 40-150μm were produced by Brush-Wellmann, USA, and were irradiated in the BERYLLIUM experiment at the High Flux Reactor (Petten, The Netherlands) at 770K to a fast neutron fluence of 1.24×10^{25} n m^{-2} (E_n>0.1 MeV), producing 480appm ^{4}He and 12appm ^{3}H. After irradiation, the pebbles were tempered at ~1500K, just below the beryllium melting point (1556K), since a pronounced gas release peak was found to start at about this temperature in thermal ramping experiments [4]. For safety reasons both the non-irradiated and the irradiated samples were embedded in a double Plexiglas cylindrical capsule of about 6 mm external diameter and glued on the very bottom of it to avoid sample jiggling during the CMT rotational scans.

The high resolution microtomography setup at the ID19 beamline of the European Synchrotron Radiation Facility, Grenoble, France, was used with a monochromatic X-ray beam of 7keV. The entire beryllium pebbles were scanned with two different spatial resolutions: 1.4μm to reveal also single bubbles and 4.9 μm to focus on bubble network formation and percolation paths. For the post-processing of the as acquired microtomography data, a filtered back projection reconstruction program written at the ESRF was deployed.

The CMT analyses were performed (i) on six unirradiated Be pebbles at 4.9μm spatial resolution, and (ii) on two irradiated specimens at 4.9 and 1.4μm resolution, respectively. Fig.12 shows a horizontal (a) and a vertical (b) cross section of an irradiated specimen, recorded at 1.4μm "voxel" resolution. While still a quite high fraction of smaller, partly isolated bubbles can be observed, the post-processing reveals a high density of large bubbles which are in a vast majority interconnected through percolation paths to the pebble surface. Due to insufficient contrast between grey levels, the smaller bubbles are sometimes hardly visible. The observation that even after high temperature annealing at 1500 K still closed porosities are visible supports the main outcome of the temperature ramping experiments [4], that a small fraction of helium is not released before the melting temperature is reached. Actually, a moderate gas concentration of only 480appm helium and 12appm tritium results after 1500K annealing in an advanced stage of bubble coalescence and network formation with extended percolation paths. It can be expected that after high neutron dose irradiation, i.e. end-of-lifetime conditions, similar bubble coalescence and network formation kinetics might be observed not only after high temperature annealing but already at blanket relevant irradiation temperatures, assuming comparable grain sizes, impurity contents and dislocation densities.



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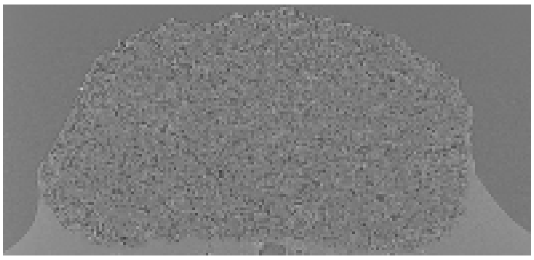
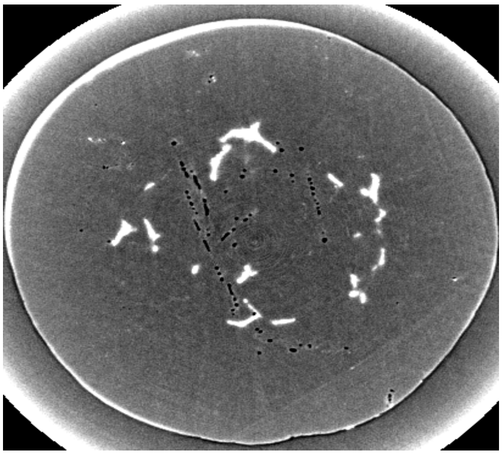


b

Fig.12. Horizontal (a) and vertical (b) CMT cross sections of a Be pebble after neutron irradiation at 770K and annealing at 1500K. The spatial resolution is 1.4μm

Fig.13(a) shows a 4.9μm resolution horizontal cross section of a non-irradiated beryllium pebble before irradiation, revealing typical pores arising from the rapid condensation of the pebble during its fabrication process. The condensation induced voids having diameters ranging between 9-18μm and a total volume fraction of 0.17%. The CMT image of an irradiated and annealed beryllium pebble is shown in Fig.13 (b), highlighting pronounced bubble networks with chain-like percolation paths. The typical distances between the percolation paths as well as their pattern suggest that they have been formed primarily along the grain interfaces. According to the CMT analysis the average diameter of the bubbles in the networks corresponds to the average width of the percolation channels, which is in the vast majority of the cases comprised in a narrow range around 39μm. The mean volume fraction of the irradiation and annealing induced voids is 14.0%, and the void fraction surface within an entire pebble amounts to typically $2.1x10^{-4}$ m^{-2}. Thermal annealing after a relatively short neutron irradiation (480appm helium and 12appm tritium) causes swelling which is almost two orders of magnitude above the fabrication induced porosity. The white and bright grey areas in Fig.13(a) and 13(b) are the signature of impurities present in the beryllium matrix and are featuring different material densities.



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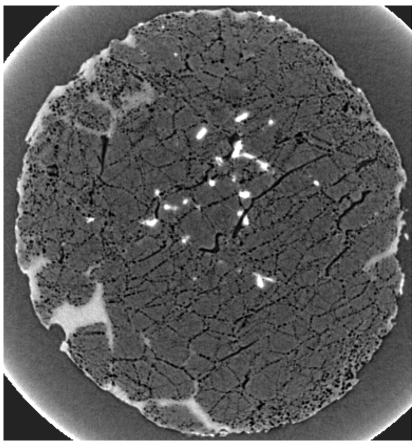

b

Fig.13. Horizontal CMT cross sections of a Be pebble before (a) and after (b) neutron irradiation at 770K and annealing at 1500K. The spatial resolution is 4.9μm

The 3D rendering of an entire irradiated and annealed pebble is as a superposition of the surface contour and the internal complex bubble network. The surface transparency was chosen to be 30%.

4. MODELLING OF RADIATION DAMAGE IN BERYLLIUM

Neutron multiplication on beryllium is accompanied by helium production, which promotes formation of helium bubbles and results in dimensional growth or swelling of beryllium at elevated temperatures. Other neutron reactions induce tritium production in beryllium.

The absence of the experimental results under fusion irradiation conditions and the lack of modeling give large uncertainties in the design calculation of the end-of-life tritium inventory in beryllium under HCPB conditions.

The number of theoretical papers dealing with the defect properties as well as with the properties of helium and tritium solutes in beryllium is very limited.

Relaxations of defect configurations were performed using two widely used DFT codes: VASP and Wien2k [13, 14].

The study of self-interstitial configurations have shown that several interstitials can be formed during the process of radiation damage in beryllium. Occupation preferences of these positions under irradiation will be dependent on the barriers between the stable interstitial configurations. This will also determine mode of interstitial migration: one-dimensional, two-dimensional or bulk diffusion. However, it is possible even now to make rough estimates of diffusion barriers for several special cases using the results obtained. Octahedral configuration (O) is between two crowdion configurations (C) laying in the same basal plane suggesting that the former is a saddle point for C-C jump. If so the diffusion barrier for C-O-C jump is about 0.85eV according to VASP calculations. On the other hand the barrier for interstitial diffusion along c axis (BO-O-BO) is about 0.2eV higher. Another interstitial diffusion path along c axis (T-BT-T) requires lower diffusion barrier of 0.56eV, but is a blind alley. Further movement requires a jump to octahedral position, which diffusion barrier needs to be determined.

Instability of di-vacancies with different orientations confirmed by three independent calculations is very remarkable. Usually, di-vacancies in metals are stable. Energy gain comes from the reduction of the number of bonds due to the absence of the bond between two vacancies. Instability of di-vacancies means that beryllium itself is resistant to formation of vacancy clusters, which serve as precursor for void formation.

Our calculations of He in di-vacancy have shown that helium stabilizes di-vacancies of all orientations. Binding energies of 3.4-4.0eV are sufficient to prevent evaporation of He from di-vacancies up to 1000℃. Therefore below this temperature helium-di-vacancy clusters are stable. Binding energy of substitutional helium with respect to evaporation of He into BO interstitial position is about 3.3eV.

Preliminary calculations have shown that interstitial He is rather mobile although its diffusion is strongly anisotropic. This anisotropy might have interesting implications for helium assisted swelling and should be properly taken into account in rate theory models considering swelling of irradiated beryllium.

5. RECYCLING OF IRRADIATED BERYLLIUM WASTE

At present beryllium finds wide application as a material for moderator and reflector blocks for research nuclear reactors. Research Institute of Atomic Reactors has been operating two research reactors SM and MIR. The SM reactor has beryllium reflector blocks arranged along the core perimeter and beryllium neutron traps located in its center. In the MIR reactor beryllium hexagonal blocks make up the core. While operated in the reactor, the block material is exposed to the life-limiting neutron fluence of $6\times10^{22}cm^{-2}$ (E>0.1MeV). This results in high induced activity of beryllium. When discharged from the reactor, the beryllium blocks are subject to disposal in a high-level storage facility of limited capacity. At present irradiated beryllium as massive blocks is in abundance and increasing further. These facts necessitate investigations on optimization of high-level beryllium waste decontamination and disposal, e.g. categorization into low-level waste. Moreover, beryllium is in sight as a neutron multiplier material for the DEMO fusion reactor blanket, which will apparently produce more irradiated beryllium and have similar disposal problems.

The present paper aims to justify and discuss a prospective investigation plan in optimization of processing and disposal of irradiated beryllium waste as research reactor moderator and reflector blocks.

Application of beryllium in research reactor cores as a neutron reflector and moderator material is dictated by the need to meet special neutron-physical requirements to the core design to increase neutron flux by reflecting and returning the neutrons escaping the core back. For example, beryllium reflector blocks were designed into the SM reactor core perimeter providing for high intensity of high-energy neutron flux, while beryllium moderator blocks located in the core center create high density of thermal neutron flux [15].

Neutron irradiation causes significant worsening of physico-mechanical properties and damages the structure of beryllium items. In particular, severe radiation embrittlement is observed in the blocks and becomes apparent as cracking or even destruction of the items into pieces [16]. Helium and hydrogen (tritium) isotopes abandoning in beryllium under irradiation [17] are recognized to make the most contribution to the degradation of its properties resulting in swelling of the beryllium block [18].

The main factor producing high radioactivity in beryllium items is radiation-induced formation of radioactive isotopes making up several hundreds of appm (e.g. tritium). When formed, helium and tritium are in beryllium crystalline lattice and remain stable that is conditioned by the low irradiation temperature. Helium is considered to be almost insoluble in beryllium, so it tends to form vacancies and bubbles or voids [18]. For instance, radiogenic helium-3 in stored metal tritides can be located both in the lattice octa-voids and helium-vacancy complexes, and in helium bubbles [19, 20]. The latest data [21-23] show radiation-induced tritium either to be captured by beryllium oxide particles or to drain to the grain boundaries or helium voids. Irradiation results in formation of lithium, which is a hydride-forming material [24]. So, irradiated beryllium may also have lithium tritide. At the irradiation temperature in the reflector and moderator blocks (up to 150-200°C) diffusion mobility of the gas atoms is not high, and almost all of them do not escape the material. Therefore, the higher the neutron dose is, the higher radioactivity of beryllium items is.

At present Russian research reactors have beryllium grades (e.g. the TE-56 grade) fabricated by hot extrusion [23] as a reflector and moderator material. The grain of the TE-56 beryllium grade is 15μm in size and contains no more than 2.2% wt. impurities. The impurities include both gases (O, N, H), and other elements (Fe, Al, Ti, Cu, Ni, Mn, Mg, Cr, C, Si, U). When the beryllium blocks are irradiated, foreign atoms react with neutrons to form new elements. Sometimes radioactive isotopes are generated increasing the total radioactivity of the beryllium block.

The prospective work will be to analyze experimental data on the state of the beryllium reflector and moderator blocks of the SM and MIR reactors irradiated up to the life-limiting neutron doses. Analytical and experimental investigations of the induced-radioactivity character and degree for the beryllium blocks after operation in the reactors will be carried out along with determination of dependencies of radioactive block characteristics on operation conditions and terms after irradiation. Contribution of each impurity to the total radioactivity of the beryllium block will be determined.

The research reactors core using beryllium is cooled by distilled water. Circulating in the core the coolant is in direct contact not only with the beryllium reflector and moderator blocks, but also with fuel assemblies, fuel rods, absorber rods, and other structural elements of core. In the process of the reactor operation high-level fission products and other substances or ions are frequently washed into the primary circuit from these structural elements. As a result, the water coolant becomes saturated with dissolved radioactive elements, and hence, gets radioactive itself. Noteworthy is the fact that the SM primary circuit water is pressurized up to several dozens of atmospheres and has a temperature of 60-80°C. Under irradiation water undergoes partial radiolysis and a certain change in acid-alkaline balance (near to the pH is about 7) takes place.

Beryllium corrosion rate in water essentially depends on the chemical composition of the material, production and reprocessing technology, and corrosive environment, i.e. the primary circuit water. In particular, presence of beryllium impurities as secondary phase inclusions (carbide, intermetallide, and others) causes pitting corrosion. The process is intensified by chloride, sulphate, iron, and copper ions found in the water [24]. As for the water characteristics, its aggressivity seems to be increased by the radiolysis. There are certain pH values (5.8-6.0), at which beryllium corrosion rate is minimum. Thus, operation of beryllium items in reactor conditions (i.e. in water under irradiation) leads to the corrosion of the block outside as a spread pitting net contaminating the block surface with radioactive products brought by the coolant and set on the pitting corrosion. Therefore, solution of the beryllium block decontamination task should primarily consider its outside decontamination by etching the outside layer at a depth of corrosion nodules.

Since diffusion rate increases with the temperature growth, high-temperature annealing allows changing thermal stability of radioactive gases in beryllium. Some experimental data demonstrate the high-temperature annealing to de-gas the irradiated beryllium [21, 22]. Intensive radioactive gas release should be monitored for radiation safety. In the process the temperature-time conditions of the annealing should be optimized. Moreover, special procedures for disposal of the released radioactive gases should be developed. Tritium released from beryllium during the high-temperature degassing can be compacted by sorbents (hydride-forming materials) and disposed. It should be taken into account that irradiated beryllium swells (up to several dozens of percent) in the process of the degassing [16]. Significant deformation of the annealed blocks results in high stresses that may destruct the items. This fact should be taken into account and its consequences should be assessed for practical implementation of the high-temperature annealing.

There are two approaches to control over impurity-caused radioactivity of the beryllium block. The first one is assessment of the contribution of each impurity to the total radioactivity, determination of the nuclear reaction chain producing the most radioactive elements, issuing recommendations to decrease the content of the most active impurities in their initial state, i.e. at the beryllium block production stage. The second one is purification of the irradiated beryllium by chemical treatment of radioactive transmutant elements and their separate disposal in a compact form. In particular, decontamination of the irradiated beryllium from radioactive impurities is possible by solid-phase chemical reactions and some other ways.

The work is aimed to develop research and practical recommendations to justify reprocessing of beryllium radioactive waste to categorize them as low-level waste to optimize their subsequent disposal. Different methods for surface and in-bulk decontamination of the irradiated beryllium, i.e. outside etching, high-temperature annealing, and chemical treatment of the irradiated materials will be analyzed. Tritium should be removed to a getter of the composition to be optimized in the process of the work. The last stage is summarizing the research results and issuing recommendations to justify the general procedure for decontamination, treatment and disposal of the radioactive beryllium waste.

Analysis of the literature sources suggests that publications on the radioactive beryllium waste treatment and disposal procedures are not enormous. At the same time the world community recognizes the importance of this problem, and the onset of several research works under the previous working group on beryllium technology is the evidence of it [25-29].

Justification of the radioactive beryllium waste categorization as low-level waste with prospective re-use is of practical significance. Disposal of low-level waste is more preferable from the environmental protection viewpoint. Moreover, the cost of low-level waste disposal is much less (by the factor of 20) than that of high-level one. A single charge of beryllium massive blocks in the SM and MIR reactors operated at FSUE "SSC RF RIAR" makes up several tons; therefore, implementation of the proposed project will contribute to the more optimized use of costly radioactive waste storage facilities and environmental improvements. The experience gained from the research will undoubtedly be useful for American, European and Japanese specialists in this field.

REFERENCES

[1] G.I. Gadgiyev, A.F. Grachyov, V.A. Gremyachkin, V.N. Yefimiv, V.V. Kalygin, A.V. Kliniv, R.I. Korotkov, N.V. Markina, V.Sh. Sulaberidze, Yu.G. Toporov, V.A. Tsykanov, The RIAR's research reactors and experimental capabilities. RIAR's review, Dimitrovgrad, 1992.

[2] V.P. Chakin, Radiation damage of SM reactor beryllium blocks. The Physics of Metals and Metallography, Vol.88, No.2, 1999, 200-204.

[3] Y. Chen, U. Fischer, P. Pereslavtsev, F. Wasastjerna; The EU power plant conceptual study - Neutronic design analyses for near term and advanced reactor models; FZKA report 6763 (April 2003).

[4] *E. Rabaglino,* Helium and tritium in neutron irradiated beryllium; FZKA report 6939 (December 2004).

[5] F. Scaffidi-Argentina, G. Piazza, R. Rolli; Fusion Engineering and Design 69 (2003) 505

[6] V.P. Chakin, A.O. Posevin, I.B. Kupriyanov, J. Nucl. Mater. 367–370 (2007) 1377.

[7] I.B. Kupriyanov, G.N. Nikolaev, V.V. Vlasov, A.M. Kovalev, V.P. Chakin, J. Nucl. Mater 367–370 (2007) 511.

[8] M. Scibetta, A. Pellettieri, L. Sannen, J. Nucl. Mater 367–370 (2007) 1063.

[9] A. Leenaers, G. Verpoucke, A. Pellettieri, L. Sannen, S. Van den Berghe; J. Nucl. Mat. (2007) in press.

[10] J. Reimann, R.A. Pieritz, M. di Michiel, C. Ferrero, Fusion Engineering and Design 75-79 (2005) 1049.

[11] J. Reimann, R.A. Pieritz, R. Rolli, Fusion Engineering and Design 81 (2006) 653.

[12] E. Rabaglino, J. Baruchel, E. Boller, A. Elmoutaouakkil, C. Ferrero, C. Ronchi, T. Wiss, Nucl. Instruments and Methods in Phys. Research B 200 (20003) 352.

[13] G. Kresse, J. Hafner, Phys. Rev. B 47 (1993) 558; G. Kresse, J. Furthmüller, Phys. Rev. B 54 (1996) 1169.

[14] K. Schwarz, P. Blaha, G.K.H. Madsen, Comp. Phys. Comm. 147 (2002) 71-76.

[15] V.P. Goltsev, G.A. Sernyayev, Z.I. Chechyotkina, Radiation material testing of beryllium, Minsk, Science and Engineering, 1977.

[16] G.A. Sernyayev, Radiation damageability of beryllium, The "Ekaterinburg" Publishing House, 2001.

[17] A.V. Vedeneyev, A.N. Golubkov, L.V. Artyomov, et al, Changes in properties of tritide-deuteride titanium caused by tritium β-decay, Preprint 68-98, RNC-VNIIEF, Sarov, 1998.

[18] A.V. Stengach, P.I. Gavrilov, V.P. Sorokin, A.P. Kalachyova N., N.D. Yuryeva, Studies of tritide metal solid phase state by small angle X–ray scattering, Proceedings of the VII International Conference "Hydrogen material science and chemistry of metal hydrides", Kiev, 2001.

[19] I.B. Kupriyanov, V.V. Vlasov, Helium and tritium behavior in neutron irradiated beryllium, Proceedings of the 4-th IEA International Workshop on Beryllium Technology for Fusion, September 15-17, 1999, Karlsruhe, Germany, 264-271.

[20] F. Scaffidi-Argentina, Tritium and helium release from neutron irradiated beryllium pebbles from the EXOTIC-8 irradiation, Proceedings of the 21-st Symposium on Fusion Technology, September 11-15, 2000, Madrid, Spain.
[21] E. Ishitsuka, H. Kawamura, T. Terai, S. Tenaka, Effects of helium bubble and irradiation damage on tritium on release behavior of neutron irradiated pebbles, Proceedings of the Ninth International Conference on Fusion Reactor Materials, October 10-15, 1999, Colorado Springs, Colorado, USA.
[22] K. Makkey, Hydrogen compounds of metals, M., “Mir”, 1968.
[23] Technical specifications TU 95.469-77 “Sintered beryllium”.
[24] Beryllium. Science and technology, Edited by D. Webster, H. London. D. Floid, G. Lowve, Translation from English, Moscow, Metallurgy, 1984.
[25] G.R. Longhurst, M.L. Carboneau, C.K. Mullen, J.W. Sterbentz, Challenges for disposal of irradiated beryllium, Proceedings of the 6-th IEA International Workshop on Beryllium Technology for Fusion, December 2-5, 2003, Miyazaki City, Japan, 225-239.
[26] F. Druyts, D. Mallants, X. Sillen, P. Van Iseghem, Belgian research on fusion beryllium waste, Proceedings of the 6-th IEA International Workshop on Beryllium Technology for Fusion, December 2-5, 2003, Miyazaki City, Japan, 240-248.
[27] V. Shestakov, Y. Chikhray, Yr. Shakhorostov, Some aspects of beryllium disposal in Kazakhstan, Proceedings of the 6-th IEA International Workshop on Beryllium Technology for Fusion, December 2-5, 2003, Miyazaki City, Japan, 249-252.
[28] V.P. Chakin, R.R. Melder, S.V. Belozerov, Experience of beryllium blocks operation in the SM and MIR nuclear reactors useful for fusion, Proceedings of the 6-th IEA International Workshop on Beryllium Technology for Fusion, December 2-5, 2003, Miyazaki City, Japan, 253-261.
[29] E. Ishitsuka, M. Nakamichi, M. Uchida, H. Kawamura, K. Kaminaga, K. Tsuboi, H. Kusunoki, Preliminary survey for recycle of beryllium reflector frames used in the JMTR, Proceedings of the 6-th IEA International Workshop on Beryllium Technology for Fusion, December 2-5, 2003, Miyazaki City, Japan, 262-264.

5.3 New JMTR Irradiation Test Plan on Fuels and Materials

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In order to maintain and enhance safety of light water reactors (LWRs) in long-term and up-graded operations, proper understanding of irradiation behavior of fuels and materials is essentially important. Japanese government and the Japan Atomic Energy Agency (JAEA) have decided to refurbish the Japan Materials Testing Reactor (JMTR) and to install new tests rigs, in order to play an active role for solving irradiation related issues on plant aging and high-duty uses of the current LWRs and on development of next-generation reactors. New tests on fuel integrity under simulated abnormal transients and high-duty irradiation conditions are planned in the JMTR. Power ramp tests of newdesign fuel rods will also be performed in the first stage of the program, which is expected to start in year 2011 after refurbishment of the JMTR. Combination of the JMTR tests with simulated reactivity initiated accident tests in the Nuclear Safety Research Reactor (NSRR) and loss of coolant accident tests in hot laboratories would serve as the integrated fuel safety research on the high performance fuels at extended burnups, covering from the normal to the accident conditions, including abnormal transients. For the materials irradiation, fracture toughness of reactor vessel steels and stress corrosion cracking behavior of stainless steels are being studied in addition to basic irradiation behavior of nuclear materials such as hafnium. The irradiation studies would contribute not only to solve the current problems but also to identify possible seeds of troubles and to make proactive responses.

Keywords: Japan Materials Testing Reactor(JMTR), Irradiation tests, Power ramp test, Fuel Integrity,Abnormal transients, Fracture toughness, Stress corrosion cracking, Light water reactors

1. INTRODUCTION

Extended use of Light Water Reactors (LWRs) is crucially important for the coming decades in order to maintain worldwide economy and environment avoiding global warming with controlled generation of carbon dioxide. Proper life-time management against the aging and high-duty uses of the current LWRs are indispensable, in addition to the steady construction of next-generation LWRs before we go further to Fast Breeder Reactors(FBRs) and other next generation reactors. In order to promote development of the next-generation LWRs, the Ministry of Economy, Trade and Industry (METI), the Federation of Electric Power Companies of Japan (FEPC) and the Japan Electrical Manufacturers' Association (JEMA) have decided to develop new core structure materials highly durable for longterm operations and high-performance fuel rods with new cladding alloys containing UO_2 pellets enriched beyond 5% for highly extended burnups. Japanese government and the Japan Atomic Energy Agency (JAEA) have decided to refurbish the Japan Materials Testing Reactor (JMTR) and to install new tests rigs, in order to play an active role for solving irradiation related issues on the plant aging and the extended uses of the LWRs.

2. OUTLINE OF JMTR

The JMTR is a light water moderated and cooled tank type reactor, which has relatively high thermal neutron flux of 4 x 1018 m2/s at a thermal power of 50MW and is capable for irradiating various capsules as many as about 60 at a time. The reactor has been used for fuel and material irradiation studies, e.g. power ramp tests of BWR fuels [1], irradiation embrittlement tests of reactor pressure vessel (RPV) steels [2] and Irradiation Assisted Stress Corrosion Cracking (IASCC) tests [3] of stainless steels, since 1970. Kinds of irradiation test facilities, e.g. OWL (Oarai Water Loop)-1, 2 [4], OSF (Oarai Shroud irradiation Facility)-1/BOCA (BOiling CApsule), OGL (Oarai Gas Loop)-1 [5] and IASCC capsules, were developed and utilized for the studies. The JMTR, except for its canal connected hot-laboratories, stopped its operation in Aug. 2006. Aged components of the JMTR after 40 years of the operation, such as reactor power control system and cooling system, are being replaced to re-start the reactor in year 2011 and to achieve reliable operation at higher duties above 50%.

3. FUELS AND MATERIALS TEST PROGRAM

New fuels and materials irradiation facilities are being constructed in parallel to the refurbishment at the JMTR. Four types of irradiation rigs, i.e. for fuel transient, fracture toughness, SCC, and hafnium tests, are planned to be installed in the JMTR core and surrounding reflector locations, as illustrated in Figs. 1 and 2. Outlines of the test program are described in the following sections.

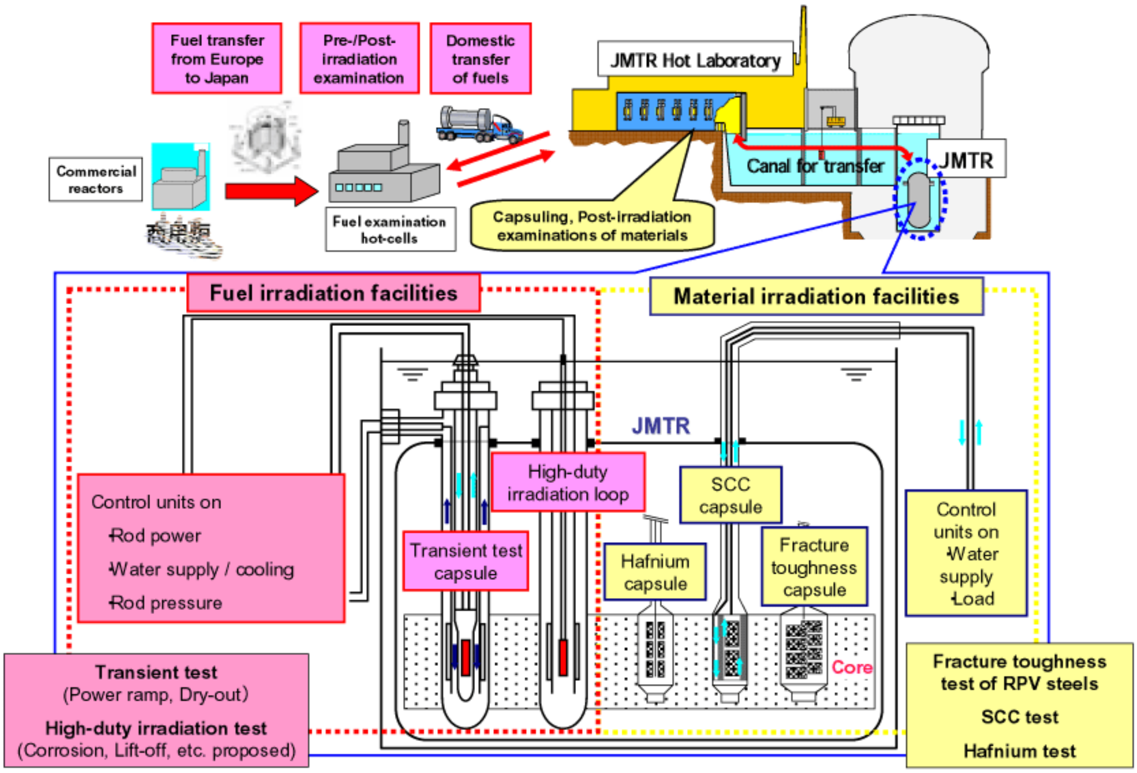


Fig. 1 Fuels and materials irradiation tests and facilities in the JMTR.

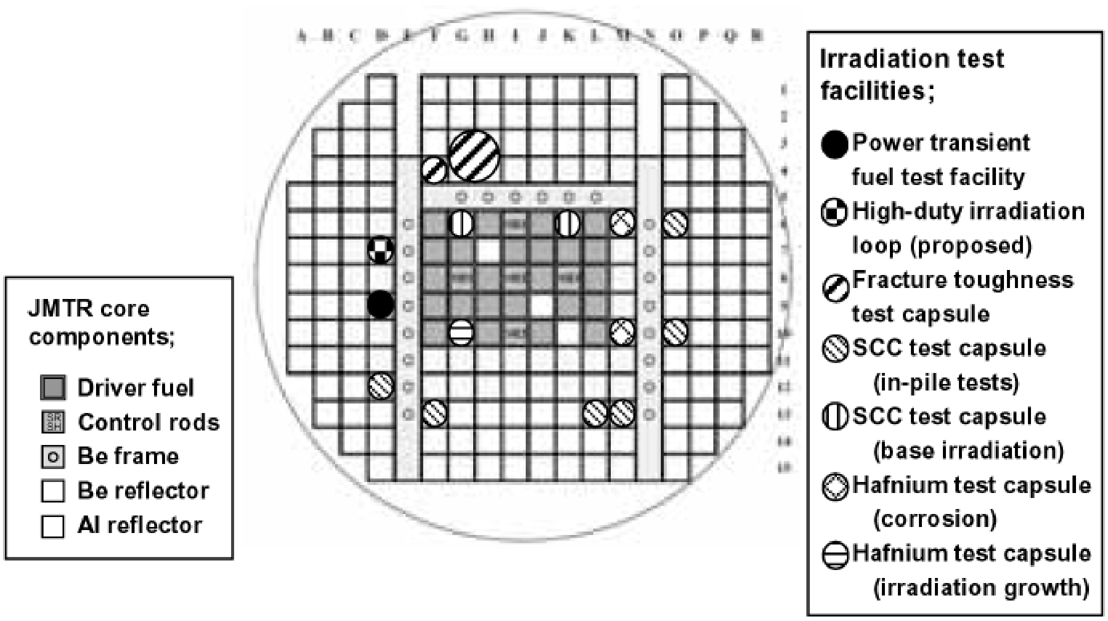


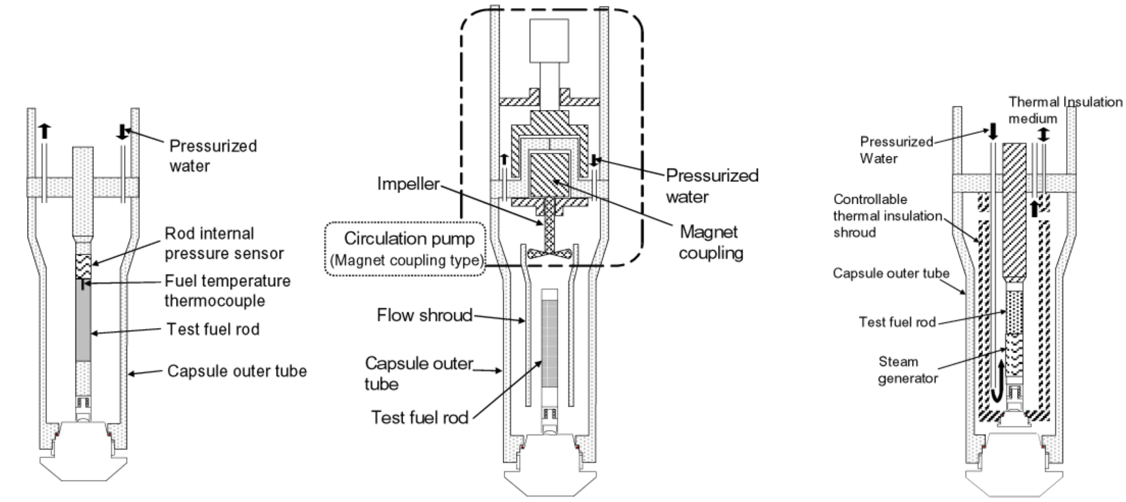
Fig.2 An example of planned JMTR core configuration with test facilities on fuels and materials.

3.1 Fuel tests

Earlier power ramp tests in the JMTR of 8x8 type BWR fuel rods at burnups up to about 60GWd/t revealed that failure mode of the cladding under the transient conditions could shift to hydrogen related cracking from pellet clad interaction driven stress corrosion cracking (PCI/SCC). The rod power level at which the hydrogen related cracking occurred was considerably lower than those for PCI/SCC at lower burnups [1]. The mechanism causing the failure has been studied mainly out-ofpile.

Applicability of the understanding through the mechanism study to higher burnups and to new design fuels needs to be confirmed through integrated transient tests. New criteria on fuel integrity and the evaluation methods of the thermal transients were proposed in a standard by Atomic Energy Society of Japan (AESJ) [6]. The proposal, allowing limited boiling transition in replace of minimum critical power ratio (MCPR), would extend operational limits of LWRs. Safety margin of the fuel integrity criteria under the dry-out conditions are requested to be confirmed for the application of the standard by the Nuclear Safety Commission (NSC) [7].

In order to contribute solving the issues, fuel integrity under abnormal power transient condition is being investigated using a special capsule-type test rig, which has its own power control system under simulated LWR cooling conditions. Natural convection boiling capsules, known as BOCA, have been successfully used in the JMTR for power ramp tests of BWR fuel rods. In addition to the natural convection capsule, new test rigs are being developed and proposed for the future tests. The new rigs are a forced convection capsule to conduct power ramp tests under more representative cooling conditions avoiding boiling transition or departure from nucleate boiling (DNB) and a dry-out capsule to examine fuel behavior under boiling transient conditions. These two kinds of new capsules would be used in the transient test rig, in replace of the natural convection capsule. Schematic configuration of the capsules is shown in Fig. 3. The various capsules could be loaded, unloaded and exchanged during an operational cycle of the JMTR which last for about 30 days, enabling maximum of four transient tests conducted during the cycle. In addition to the abnormal transient test rigs, water loops for simulation of long-term high-duty operation and cladding lift-off tests at controlled rod-internal pressure conditions are also designed and proposed for the future tests. The loop could be used for the development of new cladding alloys and pellets for the next-generation LWRs. Test fuel segment containers for domestic/international transportation, a fuel drilling device for pellet temperature measurement and capsule handling devices are being prepared in hot laboratories for the tests. The containers and hot laboratories will be equipped with extra radiation shielding for handling of high burnup UO_2 and MOX fuels emitting high-intensity neutrons and gamma-rays.



(a) Natural convection capsule (b) Forced convection capsule (proposed) (c) Dry-out capsule (proposed)

Fig.3 Schematic configuration of capsules for fuel transient tests. Types of capsules could be used for simulation of various transient conditions.

Power ramp tests of high burnup BWR fuel rods are planned to start in 2011. Expected test procedure is shown in Fig. 1. The high burnup BWR fuels transported from commercial LWRs will be re-fabricated into about 40cm-long segments, some instrumented with thermocouples for pellet centerline temperature and internal pressure measurements. After pre-test examinations, the segments will be subjected to the power ramps in basically two modes, multi step power ramp for scoping and single step ramp for determination of failure threshold. Failure threshold and its mechanism of various LWR fuels under the transient conditions will be investigated in the irradiation tests with detailed post test examinations. Combination of the JMTR tests with simulated reactivity initiated accident (RIA) tests [8, 9] in the NSRR [10] and loss of coolant accident (LOCA) tests [11] in hot laboratories would provide a comprehensive data for safety evaluation and design progress of the high performance fuels at extended burnups, covering from the normal to the accident conditions.

3.2 Material tests

3.2.1 Neutron irradiation embrittlement of reactor pressure vessel steels

Fracture toughness of reactor pressure vessel (RPV) steels of commercial LWRs has been evaluated indirectly from initial fracture toughness and relevant Charpy basis measurements on ductile-to-brittle transition temperature (DBTT) shift. Extensive efforts have been made on direct determination of the fracture toughness in the ductile-to-brittle transition region, called “Master Curve” method, to exclude uncertainties related to the indirect evaluation. The method that has been standardized in ASTM as E 1921, determines the universal fracture toughness vs. temperature curve by fracture toughness testes at one test temperature using at least 6 specimens. The universal curve was established by statistical analyses using a huge database of unirradiated and irradiated RPV steels, adjusted to the specimen thickness of 1 inch. Since only 10×10×55 mm Charpy size specimens with a rather smaller thickness than standard thickness are available from surveillance capsules loaded in many LWRs, one concern is that irradiation embrittlement effect on fracture toughness is reasonably determined by these relatively small specimens. Preliminary results show clear evidence of possible over-estimation of fracture toughness using the pre-cracked Charpy specimens, compared with results from 1-inch thickness compact tension (1T-CT) specimens [12]. This bias in test results probably due to loss in constraint of the smaller specimen types. For application of the Mater Curve method for direct evaluation of RPV integrity using the currently available pre-cracked Charpy specimens, it is quite important clarify the specimen size effects and establish reliable standards for the proper evaluation. To examine the effects of specimen size and configuration on the fracture toughness in highly irradiated RPV steels up to ~1×10^{24}n/m^2 (E>1MeV), an irradiation hole with a diameter of ~ 120 mm and neutron flux ~5×10^{17}n/m^2/s (E>1MeV) is being prepared in an outside region of beryllium frame of the JMTR. This will enable us to irradiate 1T-CT specimens in inert gas atmosphere. A schematic drawing of the capsule with 1T-CT-specimens is shown in Fig. 4. The estimated temperature around crack tip of the 1T-CT specimens installed in the capsule at the above irradiation hole is below a RPV operational temperature of 290°C by a thermal analysis concerning an increase in temperature due to gamma ray. This enables us to control the irradiation temperature around 290°C by a combination of heating and an adequate gap between the specimens and heating medium made of aluminum.

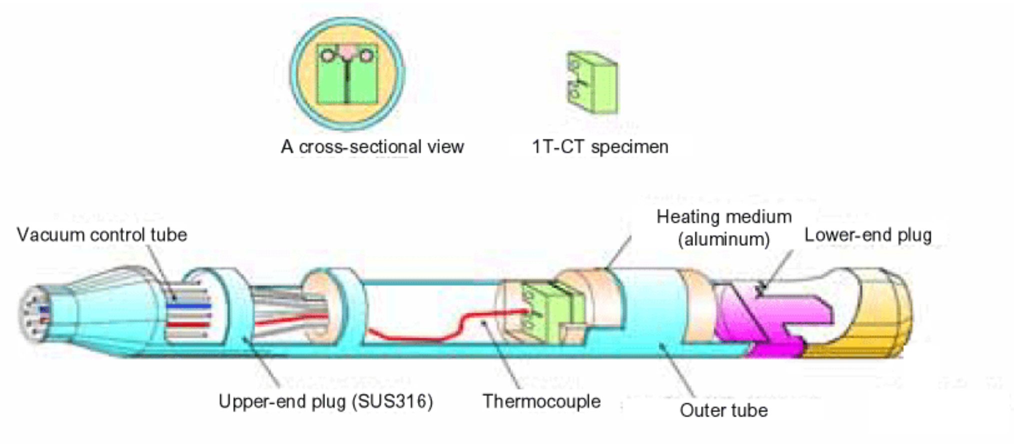


Fig. 4 A schematic drawing the irradiation capsule of 1 inch-thickness compact tension (1T-CT) specimens of RPV steels.

Table 1 Preliminary conditions of in-pile SCC tests under simulated BWR conditions.

Materials[*1]	Fluence Level [*2] (E>1MeV)	Size of Specimen	Range of Stress Intensity Factor, K (MPa•m$^{1/2}$)	Irradiation Temperature	Water Condition[*3]
316L SS	Low (<1×10^{25}n/m^2)	0.5T-CT (B=12.7mm) ／ 0.4T-CT (B=5.6mm)	10 - 30	288℃	NWC／HWC
	High (1×10^{25} -3×10^{25}n/m^2)				

*1 Other materials such as 316(NG) SS and 304L SS are under consideration.
*2 Flux during in-pile tests is ~1 x 10^{16} - 3 x 10^{16} n/m^2/s, which corresponds to that for core shroud in BWRs.
*3 NWC: Normal Water Chemistry, HWC: Hydrogen Water Chemistry

3.2.2 Stress corrosion cracking of stainless steels

Stress corrosion cracking (SCC) of the reactor core components is caused by the synergistic effects of material, stress, corrosion by high temperature water and neutron irradiation. Accelerated crack growth has been observed in out-of-pile SCC tests of irradiated materials faster than those in unirradiated material tests. The test results indicates impacts of radiation induced changes in the material on the crack growth rates. However, research on the other irradiation effects on the SCC, e.g. radiolysis effects in the water [13], have been quite limited and have not been well quantified. Then, in this program the cracking behavior will be investigated systematically in both in-pile tests with on-time radiation and post-irradiation tests out-of-pile. Thickness of the test specimens in the in-pile crack growth tests has been limited to 5.6 mm in the earlier IASCC tests at JMTR, which lead to test conditions under limited loads and constraint. In order to obtain more reliable SCC data under better conditions, new in-pile crack growth test rig, capable of testing thicker 0.5T-CT specimens (12.7mm thick), is being prepared with modified loading devices. The corrosive atmosphere in the test section is planned to be monitored by Electrochemical Corrosion Potential (ECP) sensors, in order to have better understanding on the radiolysis effects. Crack growth rates below and above the neutron fluence of the IASCC sensitivity threshold are planned to be examined by in-pile/out-of-pile tests under BWR conditions, as listed in Table 1. Effects of water chemistry effects, e.g. crack growth rates under normal water chemistry (NWC) and hydrogen water chemistry (HWC) will be investigated in a separate rig. The test rig for the water chemistry study is designed to simulate PWR conditions, as well.

3.2.3 Hafnium tests

Numbers of cracking were found on stainless steel sheath and tie-rods of hafnium plate-type control rods of Japanese BWRs in 2006. The cause of the cracking was considered as IASCC, whose stress was generated by growth of hafnium plates at cumulative thermal neutron irradiation doses level greater or equal to 4.4 × 10^{25} n/m^2. In order to prevent the trouble, a regulatory limit was set by the Nuclear and Industrial Safety Agency (NISA) to cumulative thermal neutron irradiation doses level greater or equal to 4.0×10^{25} n/m^2 as a short-term actions [14]. The NISA investigation report on the trouble pointed out that the material research should be extended in mid/long terms to establish technical standards for structural integrity of the components. Troubles of the hafnium control rods, including earlier events of other types, has been related to irradiation growth, corrosion and hydrogen absorption. In order to reliable data and evaluation standards on the phenomena, irradiation growth tests under gas atmosphere and in-pile corrosion tests under simulated LWR water conditions are planned. Reference tests on un-irradiated material are being conducted in parallel to the irradiation rig preparation. The irradiation rigs could also be used for examination of integrity of other components.

4. SUMMARY

Japan Materials Testing Reactor (JMTR) is being refurbished and new test facilities for irradiation of fuels and materials are being installed, in order to support maintaining and enhancing safety of LWRs in long-term and up-graded uses. Power ramp tests of new design fuel rods for high-duty uses will start in year 2011 after refurbishment of the JMTR. Combination of the JMTR tests with simulated reactivity initiated accident tests in the Nuclear Safety Research Reactor (NSRR) and loss of coolant accident tests in hot laboratories would serve as the integrated fuel safety research on the highperformance fuels at extended burnups. For the materials irradiation, fracture toughness of reactor pressure vessel steels and stress corrosion cracking behavior of stainless steels are being studied in addition to basic irradiation behavior of nuclear materials such as hafnium. The irradiation studies would contribute not only to solve the current problems but also to identify possible seeds of troubles and to make proactive responses.

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REFERENCES

[1] S. Shimada, E. Etoh, H. Hayashi, Y. Tukuta, *J. Nucl. Mater.*, 327, 2-3, pp. 97-113, (2004).
[2] S. Shimada, E. Etoh, H. Hayashi, Y. Tukuta, *J. Nucl. Mater.*, 327, 2-3, pp. 97-113, (2004).
[3] Y. Kaji, M. Ohmi, Y. Matsui, S. Kita, T. Tsukada, H. Tsuji, N. Nagata, K. Dozaki, H.Takiguchi, “Program of in-pile IASCC testing under the simulated actual plant condition :Development of technique for in-pile IASCC growth test in JMTR,” *Proc. ICONE-11*, April 20-23, 2003, Tokyo, Japan, ICONE11-36119, (2003).
[4] M. Hoshi, E. Tachikawa, T. Suwa, C. Sagawa, C. Yonezawa, I. Aoyama, K. Yamamoto, *J. Nucl. Sci. Technol.* 23, 6, pp. 511-521 (1986).
[5] N. Tsuyuzaki, H. Sasajima, Y. Endo, H. Itami, “Fission product plate-out measurement of Oarai helium gas loop OGL-1,” *Proc. IAEA-IWGGCR-13*, Berkeley, UK, Oct. 22-25, 1985, pp.363-370, (1985).
[6] Nuclear Energy Society of Japan, AESJ-SC-P002:2003, (2003).
[7] Nuclear Safety Commission “Special sub-committee report on fuel integrity evaluation after boiling transition,” May 19, 2006, (2006).
[8] T. Nakamura, T. Fuketa, T. Sugiyama, H. Sasajima, *J. Nucl. Sci. Technol.* 41, 1, pp. 37-43, (2004).
[9] T. Fuketa, T. Sugiyama, F. Nagase, *J. Nucl. Sci. Technol.* , 43, 9, 1080 (2006).
[10] T. Nakamura, S. Katanishi, Y. Kashima, S. Yachi, M. Yoshinaga, *J. Nucl. Sci. Technol.* 39, 3, pp. 264-272 (2002).
[11] F. Nagase, T. Fuketa, *J. Nucl. Sci. Technol.* , 43, 9, 1114 (2006).
[12] IAEA, “Guidelines for Application of the Master Curve Approach to Reactor Pressure Vessel Integrity in Nuclear Power Plants,” IAEA Technical Reports Series No.429, (2005).
[13] S.Hanawa, T.Satoh, Y.Mori, J.Ogiyanagi, Y.Kaji, S.Uchida, “Development of Water Radiolysis Code for the JMTR IASCC Test Loop,” *Proc. ICONE-14*, July 17-20, 2006, Miami, USA, ICONE14-89294, (2006).
[14] Nuclear and Industrial Safety Agency, Ministry of Economy, Trade and Industry, “Actions taken for the cracks on hafnium control rods of Fukushima-Daiichi Nuclear Power Station Unit- 6, Tokyo Electric Power Company,” Press release information Feb. 1, 3 and May 31, 2006, http://www2.jnes.go.jp/atom-db/en/trouble/individ/power/f/f20060201/news.html.

5.4 Utilization of Material Testing Reactor for Radioisotope Production

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In April 2000, JAEA (former JAERI) and CTC reached an agreement that we took over the radioisotope production from JAEA. We set up our facility in the Tokai Research and Development Center Nuclear Science Research Institute and started services. In this paper, we state present status of the production of radioisotopes in Japan and development activities in the future.

1. INTRODUCTION

The first commercial production of the radioisotopes and the supply in Japan started in 1962 with a research Reactor JRR-1 at Japan Atomic Energy Agency (JAEA former JAERI). Na-24 was the first supplied nuclide and then nuclides were extended into 6 nuclides including P-32. In later stage, the supply of sealed radiation sources such as Co-60, Ir-192 and Au-198 was started.

2. MAIN NUCLIDES PRODUCED

Regular supplies

[1] Industrial use

Ir-192 sources are mainly used for the welding inspection of steel. Yb-169 is used for much thinner steel particularly for the heat exchanger. Industrial use sources are shown in Table 1.

Table 1

Nuclide	Shape	Dimension (mm)	Activity (MBq)	Delivery /year
Co-60	needle	ϕ 0.46×10 ϕ 0.91×15 ϕ 0.91×15 ϕ 0.91×15	37 185 370 740	on request
Ir-192	pellet	ϕ 2.0×2.0	370,000	Every two months
Yb-169	pellet	ϕ 1.0×1.0 ϕ 1.0×2.0	185,000 370,000	on request

[2] Medical use

Brachytherapy source Au-198 grains have been produced every week for permanent implantation for widely used cancer treatment. Ir-192 hairpin, single pin and thin wire are also produced for low dose brachytherapy purpose. Medical use sources are shown in Table 2.

Table 2

Nuclide	Shape	Dimension (mm)	Activity (MBq)	Delivery /year
Ir-192	pellet line hairpin single pin thinwire	ϕ 1.1×1.2 ϕ 0.6×3.5 ϕ 0.65×93 ϕ 0.65×47 ϕ 0.3× 20,30,50	296,000 370,000 740 370 148 ~370	on request on request every two months
Au-198	seed	ϕ 0.8×2.5	185	weekly

On request production

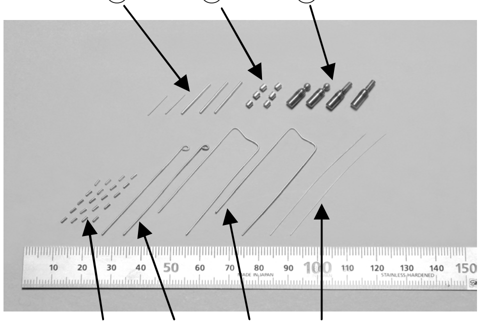
Main products of on request production are the processed radioisotopes such as Na-24, K-42, Cu-64 and Mo-99. Recently, the production of high energy beta emitting nuclides such as Re-186, Re-188 and Lu-177 are going to be produced for cancer therapy application.

Industrial use sources and medical sources are shown in the below photo.

Industrial use

① : Co-60 ② : Ir-192 ③ : Yb-169

① ② ③



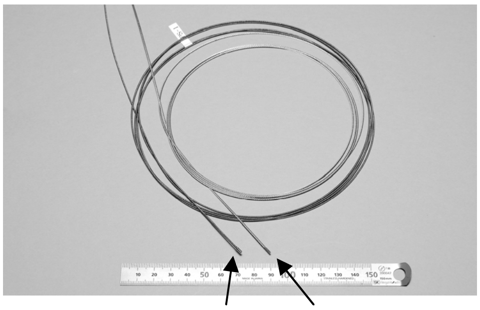
④ ⑤ ⑥ ⑦

Medical use

④ : Au-198 ⑤ : Ir-192 single pin

⑥ : Ir-192 hairpin ⑦ : Ir-192 thinwire

① : Ir-192 φ 1.1 RALS ② : Ir-192 φ 0.6 RALS



① ②

3. SHIPMENT OF RADIATION SOURCES

Ir-192 gamma radiography sources are produced approximately 1,400 sources per year. It is shipped with nominal 10Ci per source in unsealed condition for to the domestic source manufacturer. Yb-169 for non-destructive inspection was a large demand when the nuclear re-cycling facility was built in Rokkasho mura. (2000-2002) Shipment of RI sources for industrial use are shown in Table 3.

Table 3

	2003	2004	2005	2006	2007
Co-60	50	100	110	50	0
Ir-192	1340	1770	1420	1035	940
Yb-169	10	6	4	4	5

Brachytherapy source Au-198 grains supplied about 2,000 sources a year. High specific activity Ir-192 RALS (high dose rate remote after loading system) sources had been manufactured using JMTR but this was discontinued at the moment since JMTR is out of operation. High dose RALS have been getting popular and the import of the sources have been increasing up to 300 sources per year. Ir-192 hairpin, single pin and thinwire are being replaced by high dose Ir-192 after loading systems. Shipment and number of users of RI sources for medical use are shown in Table 4.

Table 4

	2003	2004	2005	2006	2007
Ir-192 RALS[1]	290 15	289 24	312 12	317 13	333 0
Ir-192 Hairpin[2]	144 6	134 6	106 5	89 4	61 4
Ir-192 singlepin[2]	28 4	22 4	14 3	8 2	8 3
Ir-192 Thinwire[2]	284 8	226 8	176 6	180 8	58 6
Au-198[2]	2355 19	2536 19	2630 17	2608 17	2086 12

[1] The upper line : imported
The below line: domestic manufactured

[2] The upper lines : distribution amounts
The below lines: number of users

4. DEVELOPMENT ACTIVIVITIES

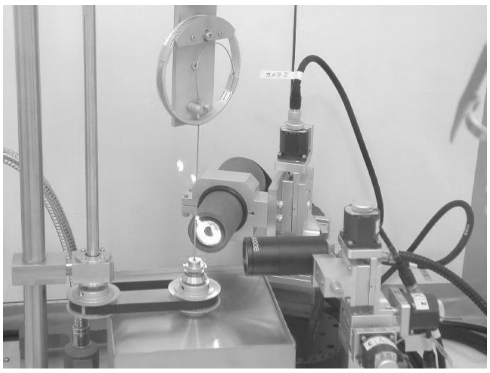
We are working to develop higher performance Ir-192 sources which will be required by the industrial users for non-destructive testing.

With the progress in the medical treatment, more variety of nuclides or special brachytherapy sources will be required. We will work for the development in cooperation with the potential users in the medical sector.

It is very important for us to form a service net work among the research reactor service providers in the world which compensate the shortage of irradiation services during the home reactors are not in use.

Facilities

Hot cell for finishing medical product



Laser welding machine for RALS

5.5 Education and Training for Nuclear Scientists and Engineers at NuTEC/JAEA

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Because of the increasing demand of nuclear engineers in recent years, which is sometimes called as the age of nuclear Renaissance, while nuclear engineers have been decreasing and technical knowledge and expertise have not necessarily been transferred to the younger generations, human resources development (HRD) has been regarded as one of the most important issues in the nuclear field in Japan as well as in the world. Nuclear Technology and Education Center (NuTEC) at Japan Atomic Energy Agency (JAEA) have conducted comprehensive nuclear education and training activities in the past half century, which cover; 1) education and training for domestic nuclear engineers, 2) cooperation with universities, and 3) international cooperation. The main feature of NuTEC's training programs is that emphasis is placed on the laboratory exercise with well-equipped training facilities and expertise of lecturers mostly from JAEA. The wide spectrum of cooperative activities have been pursued with universities, which includes newly developed remote-education system, and also with international organizations, such as with FNCA countries and IAEA. For the nuclear education and trainings, utilization of nuclear reactors is of special importance. Examples of training programs using nuclear reactors are reported. Future plan to use nuclear reactors such as JMTR for the nuclear educations is also introduced.

Keywords: nuclear education and training, JAEA, HRD, cooperation with universities, international cooperation, remote-education, nuclear reactor.

1. INTRODUCTION

Nuclear human resources development (HRD) in Japan has been identified as one of the most important issues these years in nuclear society, mostly due to the decrease of nuclear engineers in industries and students in universities with the coming peak of replacement of nuclear power reactors around 2030, and to the difficulties of technical transfers between old and young generations. The council on Nuclear HRD among industries, government and universities has been established in September 2007 to investigate mid and long term HRD strategy in Japan. Nuclear Technology and Education Center (NuTEC) was established as HRD division in 1958 soon after Japan Atomic Energy Research Institute (JAERI) was founded in 1956. Japan Atomic Energy Agency (JAEA) established by the integration of JAERI and JNC in 2005 clearly identifies nuclear HRD as one of the missions. NuTEC's HRD activities are conducted in line with governmental policy [1] and programs [2], and aims at comprehensive nuclear education and training program. The main feature of the NuTEC's training program is that the curriculum places emphasis on the laboratory exercise using well-equipped training facilities at JAEA and expertise of lecturers mostly of JAEA [3]. NuTEC is aware of the social needs in nuclear HRD and updates its training programs in response to these needs, which include cooperative activities with universities, international training with Asian countries and international cooperation under the scheme of FNCA and IAEA.

2. EDUCATION AND TRAINING FOR DOMESTIC NUCLEAR ENGINEERS

There are 3 categories of training for domestic nuclear engineers: (1) courses for radioisotope and radiation engineers, (2) nuclear reactor engineers and (3) national test examinees. Thoroughly-studied lectures and specially prepared texts are used in each of the courses. Technical visits to related facilities including large-scale and advanced facilities, such as J-PARC, NUCEF, NSRR and HTTR, are arranged in most of the courses to enlarge trainee's experiences.

2.1 Training courses for radioisotope and radiation engineers

Training courses for radioisotope and radiation engineers first began at the Radioisotope School situated in JAERI-Tokyo in 1958. At present, NuTEC provides 6 courses for radioisotope and radiation engineers. All of these courses aim on systematic acquisition of wide variety of knowledge and handling techniques of radioisotopes and radiation through lectures and laboratory exercises. In "Basic Radiation Course", "Radiation Safety Management Course" and "Radiation Protection Basic Course", participants learn subjects, such as radiation safety related law, biological effects of radiation, radiation measurement and safe handling of radioisotopes and radiation. The other 3 courses; working environment expert course and 2 radiation protection supervisor courses are legal qualification courses, in which the participants are entitled to obtain a license after the completion of the courses. The current status and total number of participants from each course are shown in Table 1. The accumulated number of participants is more than 14,300 (as of JFY 2007).

Table 1 Training Courses for Radioisotope and Radiation Engineers

Name of Course	Period (days)	Acceptable number	Frequency (/year)	Total number of participants until JFY. 2007
Basic Radiation Course	15	12	1	8,225
Radiation Safety Management Course	14	12	1	308
Radiation Protection Basic Course	4	12	1	200
1st Class Working Environment Expert Course	2	16	2	587
1st Class Radiation Protection Supervisor Course	5	32	8	4,851
3rd Class Radiation Protection Supervisor Course (since JFY. 2006)	2	32	3	143

2.2 Training courses for nuclear reactor engineers

The Nuclear Engineering School was launched at JAERI-Tokai in 1959. At present, 3 courses are provided for nuclear reactor engineers as shown in Table 2. The most significance of these courses is "Reactor Engineering Course". Since 1959, it has contributed in training nuclear reactor engineers for nuclear power plants, nuclear facilities and research institutes. This course provides comprehensive knowledge of nuclear engineering, nuclear fuel engineering, radiation management and related regulations and laws through various lectures, laboratory exercises and facility visits. Other 2 courses are available as introductory courses: "Nuclear beginners Course" broadly guides through the field of atomic energy, "Introductory Neutron Experiment Course" provides fundamental knowledge required for the use of neutrons, and to familiarize the trainees to its application technology towards the use of J-PARC. The accumulated number of participants is more than 3,000 (as of JFY 2007).

Table 2 Training Courses for Nuclear Reactor Engineers

Name of Course	Period	Acceptable number	Frequency (/year)	Total number of participants until JFY.2007
Nuclear Beginners Course	4 weeks	24	1	1,108
Reactor Engineering Course	3 months	12	12	1,888
Introductory Neutron Experiment Course	3 days	16	1	95

2.3 Training courses for national test examinees

There are 4 courses in preparation of national examinations; "Licensed Reactor Techniques supervisor", "Professional Engineer on Nuclear and Radiation", "1st Class Radiation Protection Supervisor" and "Nuclear fuel Protection Supervisor" as shown in Table 3. The training aims at systematic acquisition of knowledge and consists mostly of lectures. Every course contains subjects on its related law/ordinance/regulations. Past examinations are analyzed and mock examinations are conducted in some training courses. Participants are from electric utilities, nuclear fuel handling plants, RI/radiation handling facilities including staffs from JAEA. The accumulated number of participants is more than 2,300 (as of JFY 2007).

Table 3 Training Courses for National Test Examinees

Name of Course	Period (days)	Acceptable number	Frequency (/year)	Total number of participants until JFY.2007
Licensed Reactor Techniques Supervisor	10	20	2	1,864
Professional Engineer on Nuclear and Radiation	10	32	1	16
1st Class Radiation Protection Supervisor	6.5	30	1	232
Nuclear fuel Protection Supervisor	7.5	25	1	196

3. EDUCATION AND TRAINING FOR JAEA PERSONNEL

NuTEC conducts 39 courses for JAEA Personnel. These courses are off the job-site training (OFF-JT) and are provided to compensate on the job-site training (OJT). There are 2 categories of courses; safety training courses (13 courses) and nuclear engineering training courses (26 courses), and these courses can be participated stepwise from fundamental courses to application courses. The fundamental courses are designed as basic and necessary training for recruits and primary-grade engineers of nuclear facilities, such as, "Radiological measuring training course" and "Nuclear fuel cycle engineering course". On the other hand, the application courses are designed as skill-up training for expert engineers, such as "Safeguard course" and "Reprocessing engineering course". All these courses have special features, i.e., 1) very short period, 2) practicable, 3) open for anyone from 15 JAEA sites (about 4,000 personnel). About 50,000 s have attended these training courses since 1980.

4. COOPERATION WITH UNIVERSITIES

4.1 Cooperation with Graduate School of University of Tokyo

In response to the recent expanding needs in the nuclear field, the University of Tokyo has established a new system in 2005 for the nuclear education by setting up two graduate schools; Nuclear Professional School (NPS), and Department of Nuclear Engineering and Management (DNEM). The former is a one-year education system to produce specially qualified engineers in the nuclear field. As NPS is located next to JAEA Tokai, JAEA has a close and wide-range cooperation in the education of NPS students through NuTEC. JAEA dispatched 5 visiting professors and about 60 lecturers for the lectures in 2007, which covered about 60% of all lectures in this school. Around 90% of the experiments in this school are conducted with JAEA facilities instructed by JAEA researchers and engineers. The number of experimental instructors dispatched from JAEA was about 60 in 2007. The education program of DNEM is performed in Tokyo for the graduate students for 2 or up to 5 years. JAEA dispatched 4 visiting professors to DNEM in 2007.

4.2 Cooperative Graduate School Program

Under an education system provided by MEXT (Ministry of Education Culture, Sports and Science), NuTEC has been cooperating with many graduate schools based on the agreements between JAEA and each university. Currently JAEA has cooperation agreements with 14 graduate and one undergraduate schools. Totally 53 JAEA researchers were dispatched as visiting professors or associate professors to each university in 2007. JAEA also accepted about 20 students from these universities for nuclear studies.

A new remote-education system, called Japan Nuclear Education Network (JNEN), has initiated in April 2007 under the cooperation framework among JAEA and 3 universities. JNEN is a multi-directional education system connecting the remote sites of the participating universities and JAEA through Internet. Many kinds of lectures are available through the system at real time. In the first year, 2005, the special agreement for JNEN was signed among JAEA and three universities; Kanazawa University, Tokyo Institute of Technology, and Fukui University. Through JNEN students of each participating university can take lectures from different universities or JAEA, such as reactor engineering, fuel reprocessing and geological disposal of nuclear wastes. A lecture on the basic nuclear-chemistry from Tokyo Institute of Technology, for example, was distributed to other two universities, and about 50 students in 3 universities took the lecture at each university simultaneously, as seen in Fig. 1.

Fig.1. A scene of a lecture by JNEN system.

Under this program, some experimental courses were also performed using JAEA nuclear facilities to strengthen the effect of the nuclear education by JNEN. The experiments included handling of actual nuclear materials. Such experiments are considered to be highly important and valuable to the participating students. Two universities are scheduled to join JNEN in 2008, and some more are expected to join in the coming years.

4.3 Cooperation with Nuclear HRD Program

A new nuclear HRD Program consisting of 6 subjects has been initiated in May 2007 by MEXT and METI（Ministry of Economy, Trade and Industry）of Japan to support universities and colleges in the education of nuclear engineering and science. For the first year, the programs from 35 universities and 8 technology colleges were adopted, and about half of those universities/colleges expected some kind of cooperation with JAEA, such as dispatch of lecturers, use of nuclear facilities and facility visits. NuTEC has supported in the arrangements to meet these needs.

5. INTERNATIONAL COOPERATION

Soon after its foundation, NuTEC organized an international training course, the UNESCO Isotope Training Course, in 1958. NuTEC continued to conduct International Basic Courses for Radioisotope and Radiation for Asian countries, which were completed successfully in 1971 for the utilization of radioisotopes. From 1977, under the sponsorship of MEXT, NuTEC has been conducting International Atomic Energy Safety Technology Training Project to strengthen the training system of nuclear engineers in Asian countries. For the safe utilization of nuclear energy, the project includes three training programs: Instructor Training Program（ITP）, Joint Training Course（JTC）, and Safeguards Training Course.

5.1 Instructor and Joint Training Courses

NuTEC has been conducting two kinds of training course; Instructor Training Program（ITP）and Joint Training Course（JTC）as more effective and efficient method for developing instructors in a self-sustainable manner for certain Asian countries. ITP is a training program held in Japan to train the instructors who are to be enrolled in JTC, which is held in the partner's country（Fig.2.）. To develop teaching ability and techniques as an instructor, several participants are first invited to NuTEC to join the ITP for 4 to 6 weeks. They will learn teaching techniques that match their countries' needs and then join the JTC as co-instructors with NuTEC's instructors. Through this system, participants accumulate training experiences in JTC in their own country to become main instructors. After four-year JTC, the same course named Follow-up Training Course（FTC）is repeated for four more years to ensure its self-sustainability.

Up to now, ITP and JTC had been conducted bilaterally with Indonesia from 1996, Thailand from 1996 and Vietnam from 2001. The theme of the courses is based on the needs in the steering committee meeting held between each country, but all the courses place emphasis on the laboratory exercise with well-equipped training facilities at JAEA and with key-equipments implemented in each country. We believe that the laboratory exercise is essential for the supply of high quality training course. Indeed, the combination of ITP and JTC has proved very effective in technology transfer and stable enrolment of lecturers. The percentage of enrolment of ITP trained instructors is 70 to 92% in those countries. Also, due to the development of self-sustainable instructors, there has been various extended effects, such as development of local young lecturers, development of a new training course and contribution for educational and research activities using supplied training equipments.

Fig. 2. JTC in Thailand.

5.2 FNCA related activities

Since 1999, NuTEC has organized a workshop to promote HRD activities in Asian countries under the framework of FNCA（Forum for Nuclear Cooperation in Asia）[4]. Currently the project focuses on ANTEP（Asia Nuclear Training and Education Program）activity, a network system by utilizing existing nuclear training and education resources in 10 member states, i.e., training and education programs, nuclear research facilities and experts to meet each country's HRD needs. It was agreed at the FNCA Panel Meeting "Study Panel for Cooperation in the Field of Nuclear Energy in Asia" in Tokyo, 2007 that sharing relevant information among FNCA member states on HRD toward nuclear power is important and recommended that information exchange and cooperation on HRD be enhanced by effectively utilizing the FNCA web-site as the first step. This was approved at the Coordinators Meeting in March 2008［5］. At present, information on the ANTEP needs with its priority are being updated/uploaded and the results for its possible matching between needs and programs are shown on the FNCA HRD web-site（Fig.3.）.

Fig.3. FNCA-HRD website for ANTEP.

5.3 Cooperation with IAEA

NuTEC has been organizing safeguards training courses once every two years and contributing to ANSN (Asian Nuclear Safety Network) in close cooperation with IAEA. The Safeguards Training Course invites about 10 trainees from the countries concluding a safeguards agreement based on the non-proliferation treaty for 2 weeks to join the intensive on-the-job training consist- ing of safeguards technology in Japan, IAEA safeguards technology, supplementary protocol, IAEA system of accounting and physical protection. The course place emphasis on practice, discussion and laboratory exercises to enhance understanding. The selection of trainees and lecturers are conducted in close cooperation with IAEA (Fig.4.).

The Asian Nuclear Safety Network (ANSN) activity has started in 1977 as an Extra Budgetary Program of IAEA supported by Japanese government. ANSN aims to strengthen nuclear safety of nuclear power plants and research reactors in this region by pooling and sharing existing and new technical knowledge and practical experiences for the Asian nuclear facilities of today and future. Within this framework, an Internet-accessible database has been set up and operated by the Education and Training Topical Group, with a hub center organized by Japan Nuclear Energy Safety. NuTEC, in cooperation with Radiation Application Development Association in Japan, has contributed to the ANSN activities by providing the database with a variety of information in the field of nuclear safety.

Fig.4. Classroom Lecture.

6. NUCLEAR EDUCATION AND TRAINING USING A NUCLEAR REACTOR

As well as nuclear reactor simulation devices, utilization of actual reactors is quite useful and effective in the education and training of nuclear scientists and engineers. The JRR-4 of JAEA at Tokai site, for example, has been used for such purposes. Using such reactors, we provide such experiments as; reactor operation exercise, neutron activation analysis, prompt-gamma ray analysis, Xe buildup experiment, criticality approach experiment, etc. Thus the use of research reactors will provide us effective opportunities to learn practical knowledge and techniques related to nuclear reactors.

Japan Material Testing Reactor (JMTR) of JAEA at Oarai site is a research reactor with high neutron flux ($4x10^{18}/m^2s$). The upside view of JMTR core is shown in Fig.5. This reactor is now out of service because it is under refurbishment, as of 2008. The reactor is scheduled to finish the refurbishment and re-start in 2011. NuTEC has a plan to cooperate with JMTR to utilize it for the high-level education and training of the nuclear scientists and engineers through (1) JMTR Refurbishment Program, and (2) Development Program. Through the item (1), we plan to provide the trainees of experiences on the refurbishment of reactor facilities and the establishment of new irradiation facilities. Through the item (2), trainees will obtain advanced knowledge and techniques on such subjects as:

- in-situ observation technology of reactor core under reactor operation,
- remote sensing technology of irradiation behavior without cable,
- basic technology such as high temperature multi-paired thermocouple, uniform irradiation, sensor of oxygen and hydrogen,
- 99Mo production by a new Mo solution irradiation method,
- development of recycling technology of used beryllium for ITER, etc.

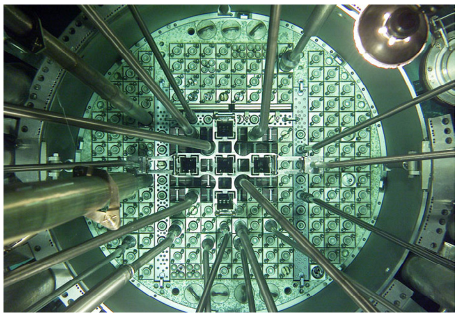

Fig.5. Upside view of JMTR core.

7. SUMMARY

In a situation that we are facing the "Nuclear Renaissance" ahead worldwide, NuTEC at JAEA aims at comprehensive nuclear education and training activities in response to the domestic and international needs. The main feature of NuTEC's training program is that the curricula places emphasis on the laboratory exercise with well-equipped training facilities and expertise of lecturers mostly from JAEA. The wide spectrum of cooperative activities have also been pursued with universities, which includes newly developed remote-education system, JNEN, and with international organizations, such as with FNCA countries and IAEA. The accumulated number of trainees to date amounts to almost 110,000 (Japan:

54,500, JAEA personnel: 52,700, international: 2,550). New programs to utilize research reactors such as JMTR are planned for the effective and high-level nuclear education and trainings. With more extended and close cooperation with domestic and international organizations, NuTEC's HRD activities will hopefully and further be conducted in more effective and efficient manner in the future.

REFERENCES

[1] Atomic Energy Commission, "Framework for Nuclear Energy Policy", October 2005. (in Japanese)

[2] Ministry of Economy, Trade and Industry, "National Nuclear Self-reliant Program", August 2006. (in Japanese)

[3] NuTEC/JAEA, "Annual Report of Nuclear Technology and Education Center (April 1, 2006 - March 31, 2007)", JAEA-Review 2008-005, March 2008. (in Japanese)

[4] NuTEC/JAEA, "The 2006 Activities and the Workshop of the Human Resources Development Project in FNCA", JAEA-Review 2007-029, September 2007.

[5] J. Sugimoto, "Summary Report and Follow-up of 1st Panel Meeting", The 9th FNCA Coordinators Meeting, March 2008, Tokyo, Japan.

6. Summary

JMTR site tour was held in the afternoon of July 15th, and 30 persons joined the JMTR site tour. On the symposium, 138 persons from Argentina, Belgium, France, Indonesia, Kazakhstan, Korea, the Russian Federation, Sweden, the United State, Vietnam and Japan, attended the symposium. Participants are summarized in Fig. 6.1.

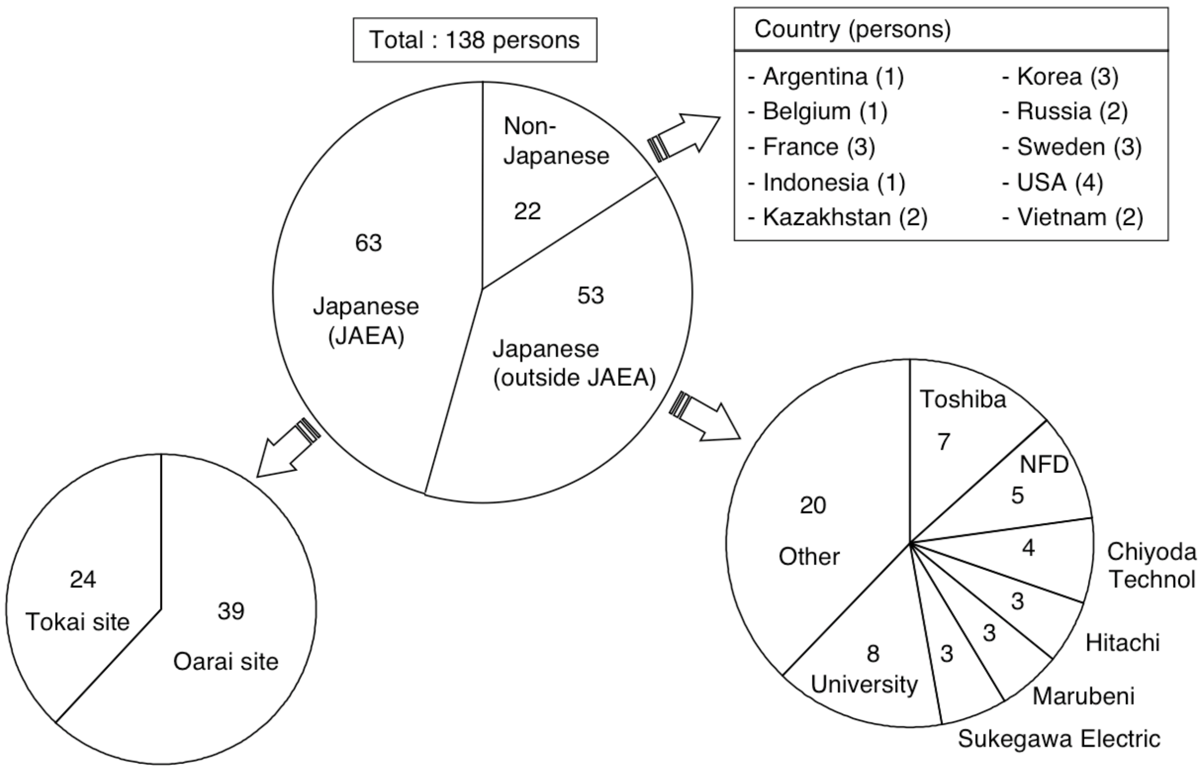

Fig. 6.1 Summary of participants.

The symposium was divided into four technical sessions and three topical sessions. Technical sessions addressed the general topics of [A] Status and future plan of materials testing reactor; [B] Material development for research and testing reactor; [C] Irradiation technology (including PIE technology) and [D] Utilization with materials testing reactors. The topical sessions addressed [T-1] Establishment of the Strategic Partnership, [T-2] Management on Re-operation Work at Reactor Trouble and [T-3] Basic Technology for Neutron Irradiation tests in MTR.

Through the discussion, following rough common sense was recognized by all attendance.

1) Necessity of the international network of MTRs can share when each reactor does not lose profit.

2) Different approach and procedure in each MTRs for the management on re-operation.

3) Importance and potential of information exchange on basic irradiation technology.

4) Next symposium will be planned by ATR to be held in U.S.A.

Highlights of these sessions are as follows.

[A] Status and Future Plan of Materials Testing Reactors

P.Y. Cordier presented the status and future plan of the JHR. Start of JHR operation is scheduled in 2014, and the experimental capability is typically ~ 20 simultaneous experiences in-core and in reflector providing suited environments relevant for different reactor technologies and high neutron flux. It was reported that JHR will be operated within an international users' consortium that will guarantee effective and cost-effective operation for there benefit.

F. M. Marshall reported the status and future plan of the Advanced Test Reactor (ATR). The ATR has been in continuous operation for over 40 years and in April 2007. There are three testing configurations – static capsule, instrumented, and pressurized water loop – with an additional testing capability to be added in 2008, a hydraulic shuttle irradiation system. Tests planned include production of industrial and medical isotopes, new fuel forms for proliferation resistant power reactors, high burn-up structural materials, and materials to be used in high temperature instrumentation for several more decades.

A.L.Petelin presented status and future plan of research reactors of the State Scientific Centre of Russian Federation “Research Institute of Atomic Reactors” (RIAR). RIAR is a multi-functional nuclear research centre. Six research reactors are available at RIAR united in a single complex. At present, each RIAR reactor facility has a long-term plan of the experimental work within the period of the operation license validity, and strategic plan was introduced.

H.Ait Abderrahim presented the status and future plan of the BR2. A high-pressure water loop (CALLISTO) provides PWR reactor operating conditions, reusable irradiation device for materials (MISTRAL), pressurized water capsule for power transient irradiations of LWR fuels, etc, were introduced. More than 50 radioisotopes for medical and industrial applications are also reported.

A. Stankevicius presented the status and future plans of research reactors and irradiation technology in Argentina (CNEA). RA-6 and RA-3 reactor were focused. RA-6 reactor has been upgraded to 3MW thermal power from 0.5MW for gaining a factor 10 in neutron flux intensity, and converted from HEU to LEU fuel. This reactor is mainly used for training. RA-3 reactor is 10 MW thermal power, which is used mainly used for radioisotope production. A dedicated program for production of Mo-99 from LEU target has been successfully conducted in RA-3.

P. Chakrov presented the status and future plans of the WWR-K. The WWW-K reactor started in 1967, and has been stopped for a period of 10 years(1988-1998) for seismic safely upgrades. It operated with a 36% U-235 enriched fuel. The focus was to convert the reactor LEU. In order to regain in flux intensity, the Be reflector was adopted instead of light water. The refurbishment of the reactor was an opportunity to create 2 extra large (φ=60mm) irradiation channel beside the initial 1 center channel. BNCT application is presently under study.

B. G. Kim presented status and perspective of material irradiation tests in the HANARO. HANARO (High flux Advanced Neutron Application ReactOr) is a multipurpose research reactor of an open-tank-in-pool type. The first decade of life of the HANARO reactor served to equipment all the HNARO beam line with various instruments in 2007-2008. The focus is to install the Fuel Test Loop(FTL) in the HANARO core. There is also the ambition to develop now the neutron physics experiment. And also, the reactor operating cycle period increase from 13 days to 23 days presently, due to the high request for the nuclear program.

H.Kawamura presented the status and future plan of the JMTR. Reactor facilities are refurbishing during four years from the beginning of FY 2007, and renewed and upgraded JMTR will start on FY 2011, and operate for a period of about 20 years (by around FY 2030). Replacement of the control rod drive mechanism, reactor control system, primary cooling pumps, secondary cooling pumps, electric power supply system and so on, were decided. The ideas of usability improvement such as higher reactor operating rate, attractive irradiation cost, etc, are also introduced. JMTR view on the future possible would be network activities in the field of MTRs.

[B] Material Development for Research and Testing Reactor

K.Tsuchiya presented problems and future plan on material development of beryllium in Materials Testing Reactors. Material modification and waste issue of beryllium reflectors are explained for lifetime extension and recycle of used beryllium irradiated in MTR. The irradiation study for lifetime extension of beryllium is planned in the reactors such as ATR, SM-3, JRR-3, etc. Kazakhstan under ISTC project of recycle study for used beryllium irradiated in JMTR was introduced. Finally, the future plan of Be study was also explained.

C. Dorn presented selection of beryllium grade as reflector material for the JMTR upgrade. The beryllium frames (S-200F) of JMTR generally last for about 36,000MWD because of swelling under neutron irradiation. After refurbishment of JMTR, the reduction of reactor down-time is considered for high-rate of operation and beryllium frames will be used for 15-20 years (about 180,000MWD). Technical factors and grade option for beryllium elements; fabrication procedure, starting materials (beryllium powder) and it's properties and material testing program were considered.

J.M. Park presented the development of U-Mo research reactor fuel in KAERI. In connection with the back-end option as well as the need of upgrading performance in the HANARO reactor core, KAERI has developed rod type U-Mo fuel with high U density, up to 6 g-U/cc by using atomization process, since the middle of 1990s. It was shown that the alloy cannot be easily converted to powder by the comminution process because U-Mo alloy has a ductile nature. The irradiation result of U-Mo plate type fuel was also explained.

[C] Irradiation Technology (including PIE Technology)

Y. Nagao presented the JMTR strategy of restart and dosimetry for standardization of irradiation technology. In the evaluation procedure using continuous energy Monte Carlo code MCNP and nuclear data library of the JENDL3.2 with a calculation model of the whole 3-D JMTR core, it was reported that the calculated fast and thermal neutron flux/fluence agreed with measurements within ±10%, ±30%, and the calculated gamma dose agreed within -3~+14%, The attempt to improve accuracy of calculated irradiation parameter, especially thermal neutron flux, was introduced. Systematic comparison of calculated and experimental date was carried out.

K. Kawamata presented the current status and future plan of JMTR hot laboratory. New PIE facilities; in-cell IASCC test, SEM/ electron-back scattering-deflection pattern (EBSD) observation, etc. were introduced. Reinforcement of concrete-cells shielding-capacity for neutron for high burn-up (up to 100 GWD/t) fuels test, center-hole drilling technique for fuel pin, and remote re-assembling technique for boiling water capsule (BOCA) were also introduced.

M. Karlsson presented the status of Post Irradiation Examinations in Studsvik's facility which cooperated such as worldwide utilization facilities. Studsvik cooperate with Halden, CEA and JAEA as well as university (the Royal Institute of Technology in Sweden) in order to still be able to provide everything asked for by the nuclear community. The importance of worldwide cooperation for effective use of expensive and highly specialized facilities was emphasized. No one can be a specialists in every area. Studsvik is interested in cooperation with operating RR-2.

M. Kodama presented PIE activities in NFD hot laboratory. Development of new PIE techniques is presented for clarifying the failure mechanism of high burnup fuel claddings under power ramp conditions. PIEs on structural material for Irradiation Assisted Stress Corrosion Cracking are carried out in order to investigate the mechanism of IASCC.

H. Matsui presented the status of department of Hot Laboratories and Facilities (DHL) of Nuclear Science Research Institute of Tokai Research and development center of JAEA for the nuclear safety research. Reactor Fuel Examination Facility (RFEF), the Waste Safety Testing Facility (WASTEF), and the Research Hot Laboratory (RHL) were introduced. The micro density measurement apparatus and specific heat capacity measurement apparatus which have been developed to determine the thermal properties of irradiated fuels were also introduced as a the developments of new PIEs technique in the RFEF.

[D] Utilization with Materials Testing Reactors

H. Aït Abderrahim presented status of MYRRHA project and its capabilities as a fast neutron irradiation facility. The first objective of MYRRHA will be to demonstrate on one hand the Accelerator Driven Systems (ADS) concept at a reasonable power level and on the other hand the technological feasibility of transmutation of Minor Actinides (MA) and Long-Lived Fission Products (LLFP) arising from the reprocessing of radioactive waste. The importance of the project as a trigger of various innovative technologies for the nuclear fission and fusion applications were pointed out.

V.P.Chakin presented the beryllium application for fission and fusion reactors. In fission reactors, the information about low temperature high dose neutron irradiation is necessary to study microstructure of irradiated beryllium, which might provide us with clue on how to extend the lifetime of beryllium blocks in a reactor. The results of xxx experiment beryllium pebbles irradiated with low fluence for fusion reactors were presented.

T. Nakamura presented the utilization of JMTR for safety and development researches of LWRs. In order to contribute for solving the irradiation related issues of the fuels and materials in the LWRs, new tests on fuel integrity under simulated abnormal transients and high-duty irradiation conditions are planned in JMTR. Power ramp tests of new-design fuel rods was introduced as the first stage of the program which is expected to start in year 2011 after refurbishment of the JMTR.

M. Hayashi presented the utilization of radioisotope production. The first commercial radioisotope production and supply started in 1962 in Japan with the research reactor JRR-1 at JAEA. The research reactors, JRR-3, JRR-4 and JMTR have been used for the production of radioisotopes although JMTR is out of services at the moment. To compensate the shortage of RI during the home reactors are not usable, importance of service network among the research reactor service in the world was pointed out.

K. N. Kushita presented the activities of Nuclear Technology and Education Center (NuTEC) at JAEA. The importance of nuclear education and trainings for younger generations by use of nuclear reactors were emphasized, and the examples of training were reported. Future plans to use nuclear reactors such as JMTR was also introduced.

[T-1] How to Establish the Strategic Partnership ?

There are currently some research reactor partnerships, the most mature of which is the European Sustainable Nuclear Energy Technology Platform (SNETP), which extends beyond research reactors to encompass the broader nuclear energy research complex in Europe. There is networking within some countries for their research reactors, Russia and Argentina. There is a newly-formed partnership between the Idaho National Laboratory and JAEA that extends beyond ATR/JMTR collaborations. At this time however, there is no definitive plan to form an international research reactor partnership. There was general agreement that closer collaborations and partnerships could be beneficial, however, there was also discussion of what would be the objectives, goals, and outcomes of such a partnership. In this context, the partnerships are presumed to extend to other technical capabilities and facilities that are used in conjunction with research reactors to achieve research objectives (e.g., hot cells and material testing equipment). Some partnerships could involve sharing of facilities between research programs to optimize the utilization of specialized equipment, others could involve sharing of research staff and educational opportunities. Some of the steps that need to be taken to establish the partnership are: 1) assess the research reactor user needs, compared to the capabilities, 2) develop list of capabilities that need to be obtained (new building or through haring with other facilities) and prioritization of the list, 3) determine suitability of partnerships to meet broader user needs, whether a 2-reactor partnership or a multi-reactor partnership, and 4) establish funding strategies to support the planned partnerships. It is in completion of Step 3 above that the real partnership definition and objectives will occur. There seems to be general concurrence that continuing toward an international partnership is a good idea.

[T-2] Management on Re-operation Work at Reactor Trouble

After presentations on JMTR, HANARO, and ATR, situations of other reactors were introduced briefly for the discussion. It became clear that the system of safety regulation in Japan is very different from other countries. Therefore, it will be necessary and useful for promoting JMTR utilization to know detailed situations in other countries through the information exchange among the test reactor community for the optimization of Japanese system.

[T-3] Basic Technology for Neutron Irradiation tests in MTRs

Basic measurement and analysis methods and procedures were presented and discussed. Comparison of basic technology for neutron irradiation tests in MTRs is summarized in Table 6.1. It was clear by the comparison by each presentations that most reactors have similar technology and use similar codes for neutron / temperature analysis. However, there are also some differences depending on the power and operation situations in the reactor. On the other hands, this differences also mean the possibilities of technology transfer to each MTR. To verify the measurement procedures, “Round Robin” procedure as in Europe was introduced.

Table 6.1 Comparison of basic technology for neutron irradiation tests in MTRs (1/3)

Items	ATR	BR2	HANARO	WWR-K	JMTR
Neutron fluence monitor	-Thermal Co-Al< 500ºC < Co-V - Epithermal Nb - Fast Fe, Ni, SST -Helium Accumulation Fluence Monitors	-Thermal+ Epithermal Al-Co and Al-Ag (bare or under Cd or Gd) -Fast (Equivalent Fission Flux or Flux>1Mev) 57Fe(n,p),58Ni(n,p), 47Ti(n,p), 93Nb(n,n')	-Thermal (<0.625 eV) Ag-Nb (under developing) -Fast (>1 MeV) Fe, Ni, Ti	< Spectrum monitor > -Thermal Au, Cu, Dy - Fast Rh (>0.8 MeV), In (>1.15 MeV), Ni, S (>3 MeV)	-Thermal (<0.683 eV) Al-Co (<500ºC), V-Co,Ti-Co (>500ºC) -Fast (>1MeV) Fe
Evaluation method	Neutron, gamma calculation by full core 3D modeling -Code : MCNP5 -Nuclear data libraries Neutron: JENDL3.2 Libraries ENDF-B/V & B/VI Gamma: MCPLIB Fuel Burn-up ORIGEN2	Neutron calculation by full core 3D modeling -Code:MCNP4C+ENDF-B.VI.8 -Inter-laboratory comparison on experimental basis to check our ability to produce correct experimental values -international (OECD/NEA + IAEA)calculational benchmarks	Neutron, gamma calculation -Code : MCNP-4C2 -Core model: 3D full core, 142 nuclides, 13,104 fuel segments -Nuclear data library Neutron: ENDF-B/VI, TMCCS, ENDL92 - Gamma: MCPLIB	Neutron calculation by full core 3D modeling - Codes: MCU-REA, MCNP5	Neutron, gamma calculation by full core 3D modeling -Code : MCNP4B -Nuclear data library Neutron: FSXLIBJ3R2 (based on JENDL3.2) TMCCS(S(alpha,beta), based on ENDF-B/III) Gamma: MCPLIB (based on DLC-7E)
Error of evaluation	MCNP uncertainty = 2.5%, Total uncertainty dependent on cross section library	Experimental uncertainties on neutron fluxes ~2%, on Gamma ~15% C/E : Neutrons ~+/-10%, Gamma ~+/-20%	- Fast neutron flux (>1 Mev) : ±10%		-Thermal neutron flux (<0.683 eV) : ±30% -Fast neutron flux (>1MeV) : ±10%
Provided data to user	-Thermal and fast neutron flux and fluence (include distribution) -Neutron / gamma spectrum -Fuel burn-up total, distribution, isotopic distribution	-Thermal, Epithermal, Fast (>0.1 MeV), Fast (>1MeV), Eq.Fission -Gamma dose or heating	-Thermal and fast neutron flux and fluence (include distribution) -Neutron / gamma spectrum	-Thermal and fast neutron flux and fluence, -Neutron spectrum	-Thermal and fast neutron flux and fluence (include distribution) -Neutron / gamma spectrum / dpa (option)

Table 6.1 Comparison of basic technology for neutron irradiation tests in MTRs (2/3)

Items	ATR	BR2	HANARO	WWR-K	JMTR
Thermocouple	-K ø3.2, 2.4, 1.6, 1.0, 0.5 (individual conductor) mm - N ø 3.2, 2.4, 1.6, 1.0, 0.5 (individual conductor) mm - C ø 3.2, 2.4, 1.6, 1.0 mm - Mo-Nb ø 3.2, 1.6 mm < 1600º C	- Same as JMTR	- K(~1000ºC) ø1.0, ø1.6mm - W/Re (1000~1900ºC) ø1.4mm	- Allimel-chromel, - W-Re, - Cu-Const - Allimel-Capel	-K(~1000ºC) ø0.5, ø1.0, ø1.6mm -N (800~1100ºC) ø1.0, ø1.5mm -W/Re (1000~1900ºC) ø1.6mm -Multi-paired T/C ø1.8mm (7 points)
Offline measurement	-Melt Wire (In, Sn, Pb, Bi, Ag, Zn, Be, Cu) -SiC, Na Capsules	- From IRMM in Geel (see their Web site)	Melt wire (Eutectic alloys) 200~400ºC	-Melt metals	Melt wire (In,Sn,Pb,Bi,Ag,Zn) 95~420ºC
Calculation code	ABAQUS (3D) RELAP 5 Fluent (3D) FIDAP	-FLICA -SYSTUS -Home Code	-Capsule temperature design GENTIC(1D) ANSYS(2D, 3D) -Evaluation of specimen's temperature ANSYS(2D, 3D), others	-Other institute of NNC	-Capsule temperature design GENTIC(1D) NISA(3D) -Evaluation of specimen's temperature NISA(3D)
Temperature control	Active control - He/Ne or He/Ar mixture with gamma and reaction heating for heat source Passive – Same w/pure gases (He, Ar, Ne, etc.) Heaters on PWR loops Li & Na Bonding to equalize temperatures	-Electrical heater -gas insulator -Temp. Control: +/- 5°C -Range of Temp.: 50 to 650°C	Heater+gas pressure Ex. 290±10ºC (He, Ne) Mixed gas	Heater, gas pressure	Heater+gas pressure (constant temperature control by reactor power feed-forward control) Ex. 290±3ºC

Table 6.1 Comparison of basic technology for neutron irradiation tests in MTRs (3/3)

Items	ATR	BR2	HANARO	WWR-K	JMTR
Thermal neutron flux (Distribution and transient)	SPND Fission Chamber U-Al flux wire	-On-line neutron flux monitoring: miniat. FC (U8, NP7, TH2), SPND -Gamma-ray monitor. SPGD, Calorimeter, Red-Perspex (short irradiat)	- SPND	- SPND - Fission chamber - Neutron counters	-SPND -Fission chamber
Displacement	Linear Position Transducer	- LVDT	- Differential transformer type		-Differential transformer type
Pressure	Impulse line to differential transmitter	- LVDT	- Differential transformer + bellows type	-Differential transformer type, -resistor type, -mechanical manometer	-Differential transformer + bellows type
Gas concentration	Oxygen, carbon monoxide & dioxide, moisture, tritium, fission gases (Xe, Kr & I)			-Radiometers for radioactive gases	-Oxygen, hydrogen sensor (under development)
Other	Force – load cell		-Creep and fatigue test (in-situ measurement)		-Crack growth by current potential drop -Optical measurement by light fiber

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Appendix I Program of Symposium

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Schedule of International Symposium on Material Testing Reactors

	15,July	16,July	17,July
8:00			
9:00		Registration [Session A] Status and Future Plan of MTR	[Session B] Material Development
10:00		[Coffee break] Memorial Photo	[Coffee break] [Session C] Irradiation Technology
11:00		[Session A] Status and Future Plan of MTR	
12:00		[Lunch]	[Lunch]
13:00		[Session A] Status and Future Plan of MTR	[Session D] Utilization with MTR
14:00		[Coffee break]	[Topics 3]
15:00	[Technical tour] JMTR and JMTR/HL Registration	[Topics 1]	[Coffee break]
16:00		[Break] [Topics 2]	[Summary Session]
17:00	Chairman's Meeting		
18:00			
19:00			
20:00			

International Symposium on Material Testing Reactors

Wednesday, July 16, 2008

OPENING	I. Nakajima (Executive Director, JAEA)	9:30 - 9:35
SYMPOSIUM OBJECTIVE AND PROSPECTS	M. Takuma (General Chair, JAIF)	9:35 - 9:40

[A] Status and Future Plan of Material Testing Reactor (1)
Chairman : Mr. V. Chakin, Mr. H. Kawamura

[A-1]	**JHR** A High Performance MTR under Construction to Meet Coming Need for a Sustainable Nuclear Energy	Pierre-Yves Cordier (Embassy of France in Japan)	9:45 - 10:15
[A-2]	**ATR** Advanced Test Reactor – Testing Capabilities and Plans	F. Marshall (INL)	10:15 - 10:45
	＜Coffee break＞ Memorial Photo		10:45 - 11:00
[A-3]	**SM-3** Status and Future Plan of Research Reactors of Scientific Center of Russian Federation RIAR	A.L.Petelin (FSUE "SSC RIAR")	11:00 - 11:30
[A-4]	**BR-2** The Best Research Reactor, therefore we call it BR2 at Mol	H.A. Abderrahim (SCK•CEN)	11:30 - 12:00
	＜Lunch time＞		12:00 - 12:45

[A] Status and Future Plan of Material Testing Reactor (2)
Chairman: Mr. H.A. Abderrahim, Mr. M. Niimi

[A-5]	**RA** Status and Future Plan**s** on Research Reactors and Irradiation Technology in Argentina	H. Taboada (CNEA)	12:45 - 13:15
[A-6]	**WWR-K** WWR-K Research Reactor – Status and Future Plans	P. Chakrov (INP-NNC)	13:15 - 13:45
[A-7]	**HANARO** Status and Perspective of Material Irradiation Tests in the HANARO	B. G. Kim (KAERI)	13:45 - 14:15
[A-8]	**JMTR** Status and Future Plan of Japan Materials Testing Reactor	H. Kawamura (JAEA)	14:15 - 14:45
	＜Coffee break＞		14:45 - 15:00

[T-1] Topics 1 : How to Establish the Strategic Partnership ?
Chairman: Ms. F. Marshall, Mr. M. Ishihara

Discussion on World Network necessary for user friendly management in getting and comparing the irradiation data with more than two reactors	15:00 - 16:00

[T-2] Topics 2 : Management on Re-operation Work at Reactor Trouble
Chairman: Mr. S.Shiroya

Introduction concerning the examples coped with the reactor trouble and the judgments in each reactor	16:00 - 17:15

THURSDAY, July 17, 2008

[B] Material Development for Research and Testing Reactor
Chairman : Mr. B. G. Kim, Mr. H. Sagawa

[B-1]	Problems and Future Plan on Material Development of Beryllium in MTR	K.Tsuchiya (JAEA)	9:00 - 9:20
[B-2]	Material Selection for Extended Life of the Beryllium Reflectors in the JMTR	C. Dorn (Brush Wellman)	9:20 - 9:40
[B-3]	Development of U-Mo Research Reactor Fuel in KAERI	Jong-Man Park (KAERI)	9:40 - 10:00
	<Coffee break>		10:00 - 10:20

[C] Irradiation Technology (including PIE Technology)
Chairman : Mr. P. Chakrov, Mr. M. Suzuki

[C-1]	JMTR Strategy of Re-operation and Dosimetry for Standardization of Irradiation Techniques	Y. Nagao (JAEA)	10:20 - 10:40
[C-2]	Current Status and Future Plan of Hot Laboratory in JMTR	K. Kawamata (JAEA)	10:40 - 11:00
[C-3]	Post Irradiation Examinations Cooperation and Worldwide Utilization of Facilities	M. Karlsson (Studsvik AB)	11:00 - 11:20
[C-4]	PIE Activities in NFD Hot Laboratory	M. Kodama (NFD)	11:20 - 11:40
[C-5]	Status of PIEs in the Development of Hot Laboratories and Facilities	H. Matsui (JAEA)	11:40 - 12:00
	<Lunch time>		12:00 - 13:00

[D] Utilization with Material Testing Reactors
Chairman : Mr. A.L.Petelin, Mr. N. Hori

[D-1]	From MYRRHA to XT-ADS - Development of Pb-Bi cooled ADS as a Fast Spectrum Irradiation Facility in Europe -	H.A. Abderrahim (SCK•CEN)	13:00 - 13:20
[D-2]	Beryllium Application for Fission and Fusion	V. Chakin (FSUE "SSC RIAR")	13:20 - 13:40
[D-3]	Utilization of JMTR for Safety and Development Researches of LWRs	T. Nakamura (JAEA)	13:40 - 14:00
[D-4]	Utilization of Material Testing Reactor for Radioisotope Production	M. Hayashi (Chiyoda Technol)	14:00 - 14:20
[D-5]	Education and Training for Nuclear Scientists and Engineers at NuTEC/JAEA	K. Kushita (JAEA)	14:20 - 14:40

[T-3] Topics 3 : Basic Technology for Neutron Irradiation tests in MTR
Chairman: Mr. M. Karlsson, Mr. E. Ishitsuka

[T-3]	Introduction concerning basic technology like neutron dosimetry, temperature measurement, etc with each MTR.		14:40 - 15:40
	<Coffee break>		15:40 - 16:10
Summary of Each Session by Chairmen		Each chairman	16:10 - 16:50
	CLOSING	M.Ogawa (Deputy Director General, JAEA)	16:50 - 17:00

Appendix II　Presentation Materials in Topics

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Topics 1

How to Establish the Strategic Partnership ?

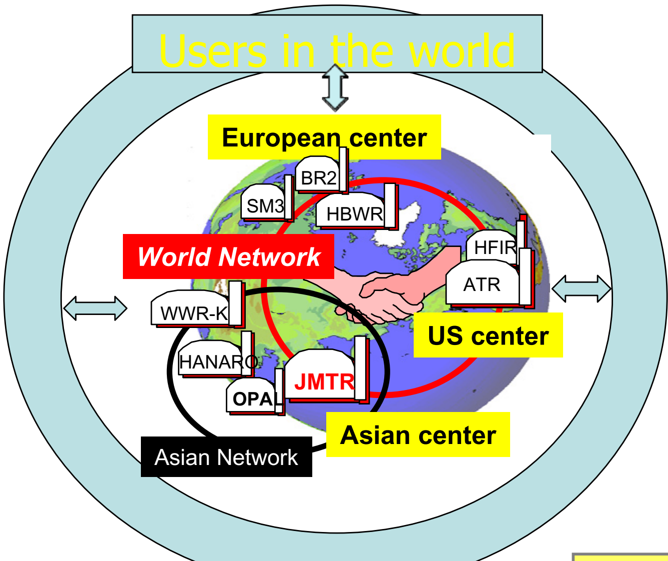
Japan Atomic Energy Agency

JAEA's Proposal on World Network

Example

OBJECTIVES

○Effective facility management and advanced irradiation function by role sharing in irradiation tests with characteristics of each testing reactor.
○Achievement of user friendly system in the world to contribute wide areas of irradiations.



Proposal of World Network

JMTR contributes as an Asian kernel testing reactor.

1stStep:Mutual understanding

- **Information exchange**
- **Interchange of staffs, etc.**

2ndStep:Mutual action

Offer of standardized irradiation tests by the standardization of Basic irradiation technology, etc.

Construction of international cooperation system by world wide testing reactor network

Situation Change *unconsciously*

- Decrease of the Number of Materials Testing Reactor (MTR)
  → **Not only for homeland but also for all over the world**
- Statelessness of User's Needs by Globalization of Nuclear Business
  → **Similar test by each MTR**

Turning Point

Final Goal : Strategic Partnership with keeping User 's Mind

However, Step by Step
First, Problems Solution by information exchange
as to basic technology for irradiation test

Necessity of MTR Network

Advanced Test Reactor National Scientific User Facility – Partnerships and Networks

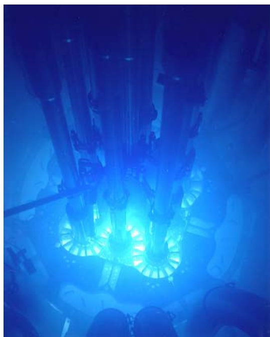
Frances M. Marshall
Irradiation Testing Department

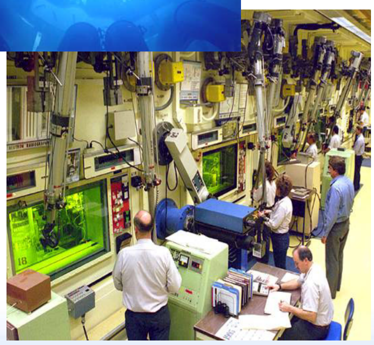
July 16, 2008

Idaho National Laboratory

The ATR NSUF is the Bedrock for INL and NE

- Primary U.S. nuclear technology R&D facility, with primary emphasis on fuels and materials
  - New reactor materials
  - LWR life extension
- Focal point for nuclear energy related expertise for the nation
  - Summer school and internships
  - ATR NSUF Director's colloquium series
  - Engage professional societies (TMS, ANS) for promotional pieces
  - ATR NSUF "Alumni Association"
- INL facilities plus select partnerships optimize utilization of national assets
  - Facility partnerships (MITR, Halden, JHR)
  - National User Facility partnerships (e.g., APS, SHaRE, HFIR)
  - Organizational collaborations (JAEA)
  - Power industry





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2

Building Capability to Meet User Needs

- New Reactor Capabilities
  - Hydraulic shuttle irradiation system, October 2008
  - Pressurized water loop for industry research, July 2010
  - Test Train Assembly Facility, August 2009
- New PIE Capabilities
  - Electron probe micro-analyzer (EPMA)
  - Focused Ion Beam (FIB)
  - Mechanical testing in cell
  - Crack Growth Testing
- New Instrumentation Development
  - Dimension, Conductivity, Temperature, Material flaws
  - Potential Technologies
    - Fiber optics (cracks, temperature, etc.)
    - Ultrasonic techniques (cracks, temperatures, length)
    - SiC (temperature)
    - Wireless technologies (temperature, pressure)

Vision:

•Develop an Irradiated Materials Characterization facility

•Continuously improve the quality of these characterization tools with new techniques or procuring advanced equipment

•Apply to new and existing materials

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3

2008 Summer Session Participation

Universities Represented: 29
Participants 41 (25 scholarships)
Students: 33
Faculty: 6
Industry: 2
14 Guest lecturers (universities and DOE labs)
Select lectures taped to commence development of a "remote learning library."

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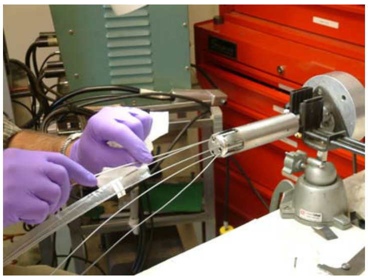
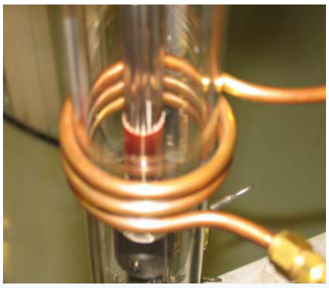
Understanding Customer Needs, Priorities

- Proposals for Reactor Use – 4 in 2008, 5 in 2009
- Meetings with Users
  - Annual User Workshops
  - Other Workshops- ANS or other meetings
  - Summer Sessions – students and researchers
- Data Collection
  - Surveys (equipment and sample library needs)
  - Website questions and comments
- Technical Exchange
  - Colloquium series at INL
  - Technical meeting sponsorship
  - University partnerships - complementary capabilities
  - Researcher/faculty exchange
- External Review
  - Scientific Review Board
  - Independent Assessment Team
  - ATR NSUF Industry Advisory Committee





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High-flux Advanced Neutron Application Reactor

Topics 1

How to Establish the Strategic Partnership ?

Korea Atomic Energy Research Institute

1st International Symposium on Material Testing Reactors,
July 16th - 17th , 2008, Oarai, Japan

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- **KAERI is basically agreed with JAEA's proposal to achieve user friendly system in the field of irradiations area test by using research reactor.**
- **HANARO is always prepared and opened to effectively support users to perform an irradiation test of materials in the HANARO and PIE.**
- **Any time, we can talk about using the HANARO.**

2

Sustainable Nuclear Energy Technology Platform (SNE-TP)

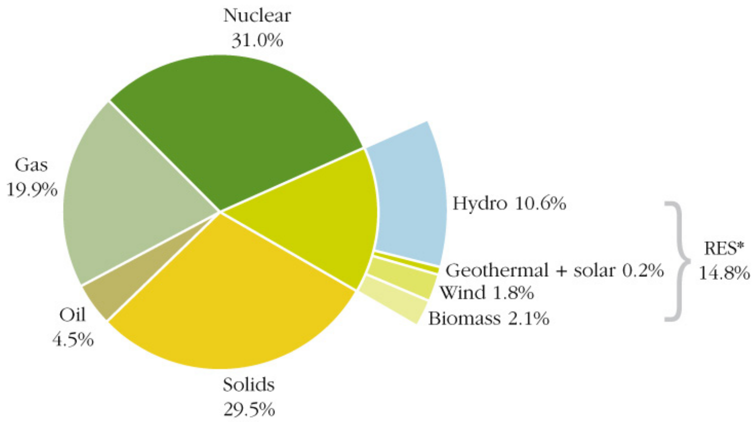
www.snetp.eu

1

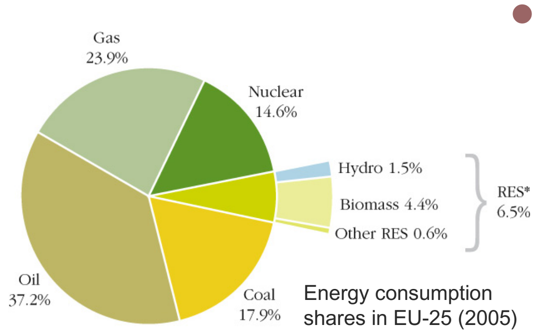
A collective vision, endorsers and contributors

2

Nuclear energy in Europe



Electricity generation shares in EU-25 (2005)

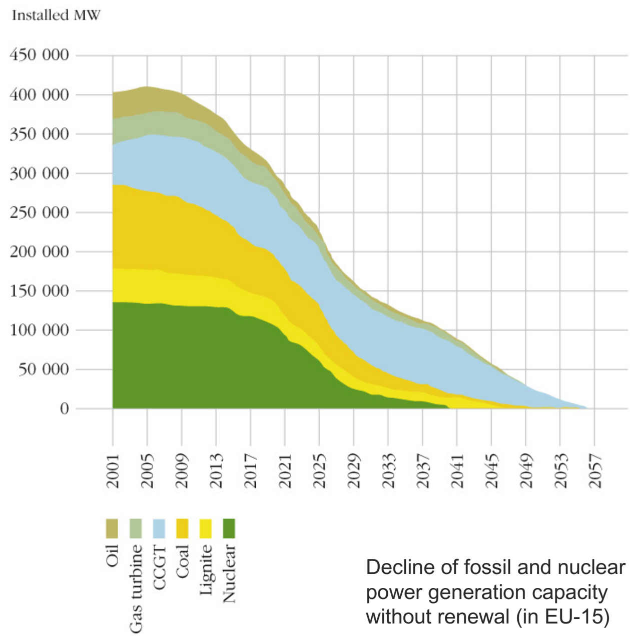


Energy consumption shares in EU-25 (2005)

- 152 reactors in 15 countries in EU-27, producing 31% of EU's electricity
- The largest source of low carbon energy
- Excellent safety record
- Europe, a world leader – but competition is building (Russia, Japan, USA, China, India)

3

Power generation infrastructures



Decline of fossil and nuclear power generation capacity without renewal (in EU-15)

- Fossil and nuclear power generation plants are ageing.
- Need to invest in plant lifetime management and
- Large investments are necessary to build new plants to satisfy demand
  - For nuclear, Gen. III reactors (Finland, France)
- Action is needed now!

4

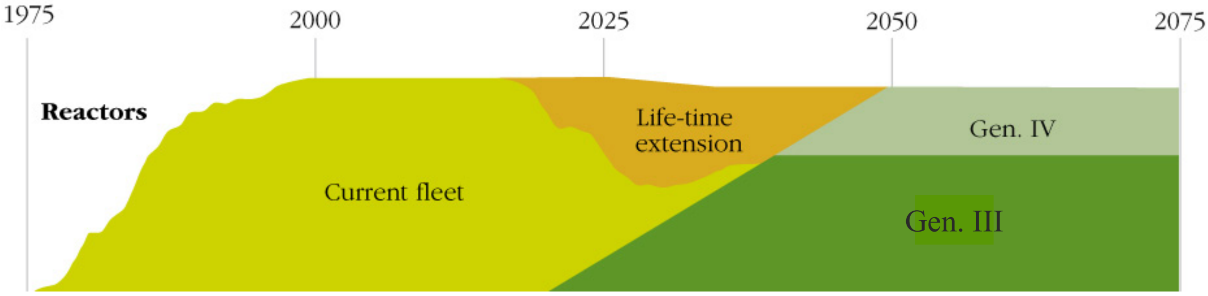
A contribution to Europe's energy challenges

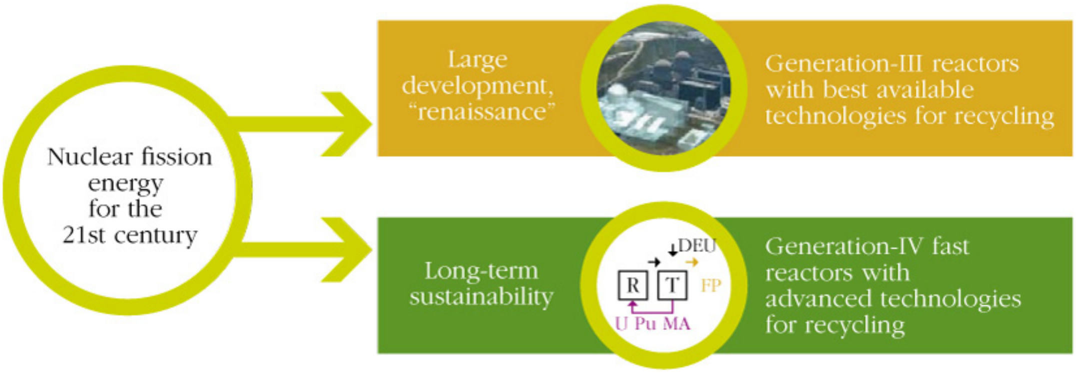
- Security of supply
- Reduction of greenhouse gas emissions
- Competitiveness

- Sustainability of nuclear energy by
  - Continuing to maintain a high level of safety
  - Further developing technical solutions to waste management

5

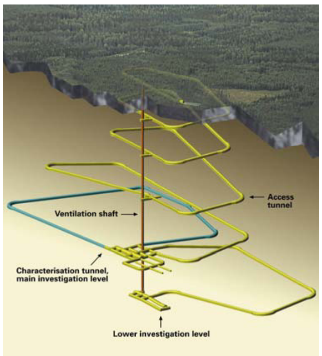
The R&D and industrial challenges



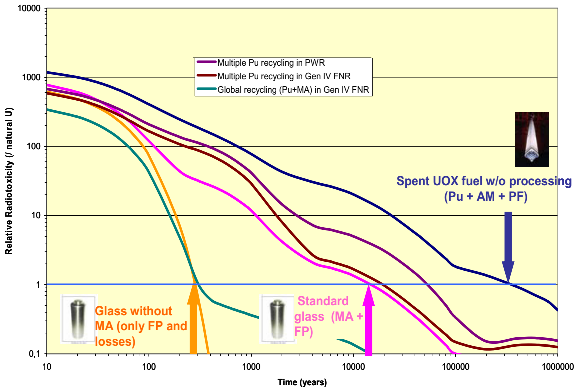


6

Waste management, multi-recycling strategy



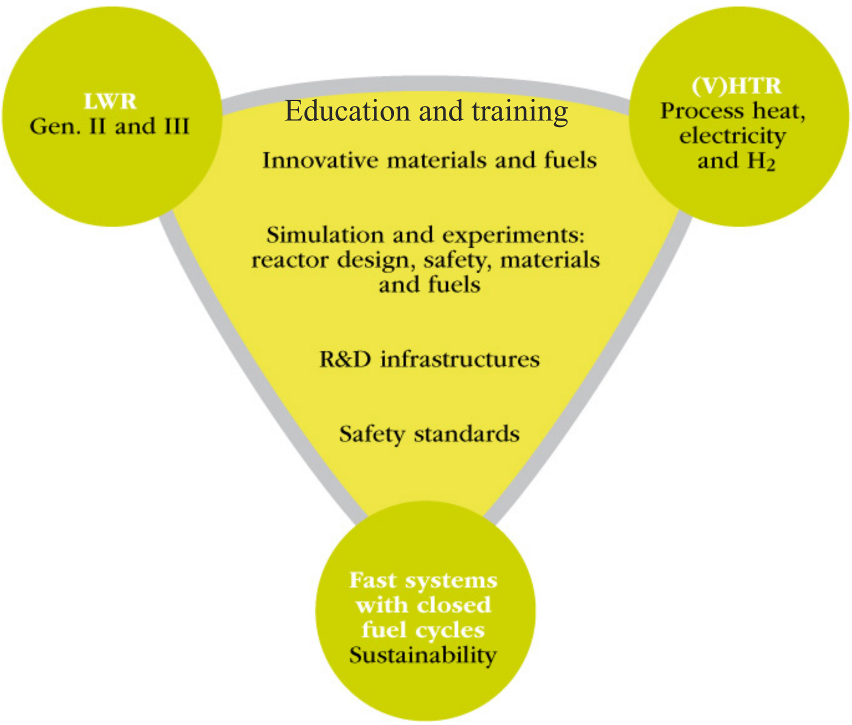
- Continuous progress has been made in the processing of spent fuel, recycling of nuclear material and conditioning of waste
- Reversible geological disposal is the object of an international technical consensus



- Recycling of minor actinides to reduce thermal load and radio-toxicity of waste is the object of on-going research

7

SNE-TP, Strategic Research Agenda



8

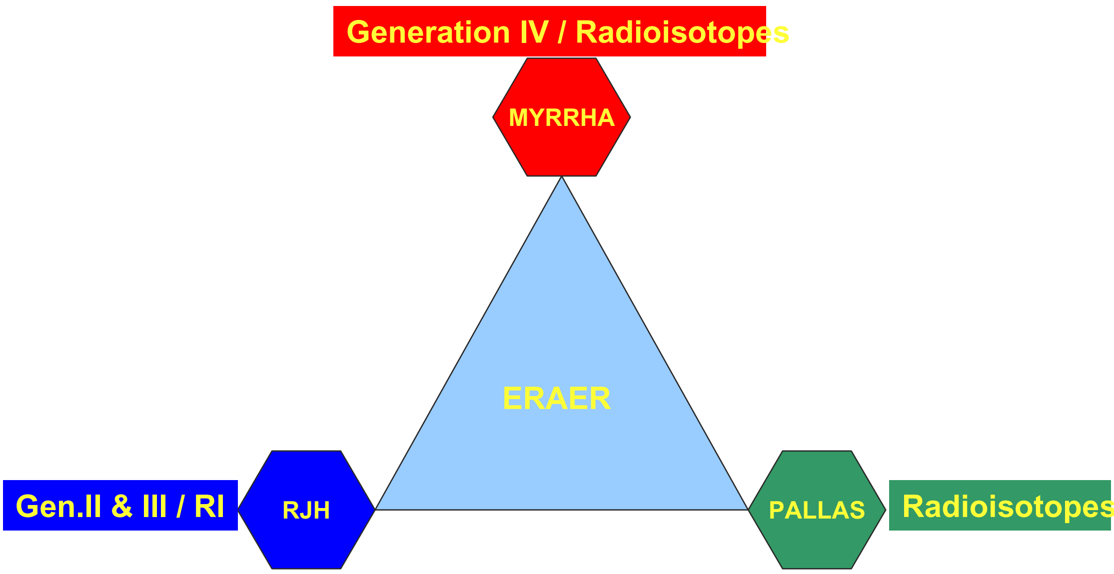
Funding for future research, development, demonstration and deployment programmes

- National programmes (coordinated via SNE-TP)
- Framework programmes (FP7, FP8, …) (but will represent only a small fraction)
- Private/public partnerships
- Euratom loans?
- Other, eg. Regional funds to support new infrastructures?

www.snetp.eu

9

ERAER = European Research Area for Experimental Reactors



ERAER = European Research Area for Experimental Reactors

10

Topics 2

Management on Re-operation Work at Reactor Trouble in JMTR

Neutron Irradiation and Testing Reactor Center, JAEA

Summary on Reactor Scram/Shutdown Events in 2005 (from Jan. to Dec.)

- **Reactor name ………….. JMTR**
- **Power ……………………. 50 MW**
- **Operation ……………….. 132 days at power 50 MW**
- **Availability factor ………. 36 % (operation day / 365[day] x 100)**
- **Total work per year…….. 6251.7 MWd**
- **Operation cycle………… About 30 days/cycle**

- **Unplanned scram/shutdowns…… 1**
- **Interruption of operation…..……… 53 days**
- **Reasons**
  - **Commercial power failure …….. 1 emergency shutdown**
  - **Human error…………..…………. 0**
  - **Experimental devices…………… 0**
  - **Earthquake …………..…………. 0**
  - **Mistake of a test condition in periodical inspection …………….………….. 1 delay in reactor starting date**

Summary on Reactor Scram/Shutdown Events in 2006 (from Jan. to Dec.)

- Reactor name …………... JMTR
- Power ……………………. 50 MW
- Operation ………………. 132 days at power 50 MW
- Availability factor ………. 36 % (operation day / 365[day] x 100)
- Total work per year…….. 6256.8 MWd
- Operation cycle………… About 30 days/cycle

- Unplanned scram/shutdowns…… 1
- Interruption of operation…..…….. 2 days
- Reasons
  - Commercial power failure …….. 1 emergency shutdown
  - Human error…………...………….. 0
  - Experimental devices…………... 0
  - Earthquake …………..…………... 0

Countermeasure Process at Scram/Shutdown Events

(2005, JMTR)

Event	Countermeasure Process	Notes
Reactor Scram (earthquake or commercial power failure.)	1. Notify the regulatory agency of the event, immediately. Safety check of reactor facilities after reactor scram. 2. Notify the regulatory agency of the result of safety check and restart time. 3. Restart the reactor. (Interruption of operation : about 40 hours/event)	Earthquake : 0 commercial power failure : 1 etc.
Unplanned Manual Reactor Shutdown (Cause of event is clear.)	1. Notify the regulatory agency of the event, immediately. Safety check of reactor facilities after reactor scram.. 2. Cause identification and countermeasure. (include repair or replace of reactor equipments, etc.) 3. Report to the regulatory agency for the cause and measures of the event. 4. Restart the reactor. (Interruption of operation : None)	irradiation facility : 0 emergency power unit : 0 Control rod drive : 0 etc.
Reactor Scram (except of earthquake or commercial power failure)	1. Notify the regulatory agency of the event, immediately. Safety check of reactor facilities after reactor scram. 2. Course identification and countermeasures. (include repair or replace of reactor equipments, etc.) 3. Report to the regulatory agency for the cause and measure of the event within 30days (if necessary, final report submit after 30days.) 4. Repair or replace of reactor equipments. 5. Restart the reactor. (Interruption of operation : None)	primary cooling pump : 0 Human error : 0 control rod drive : 0 fuel failure : 0 etc.

Countermeasure Process at Scram/Shutdown Events
(2006, JMTR)

Event	Countermeasure Process	Notes
Reactor Scram (earthquake or commercial power failure.)	1. Notify the regulatory agency of the event, immediately. Safety check of reactor facilities after reactor scram. 2. Notify the regulatory agency of the result of safety check and restart time. 3. Restart the reactor. (Interruption of operation : about 36 hours/event)	Earthquake : 0 commercial power failure : 1 etc.
Unplanned Manual Reactor Shutdown (Cause of event is clear.)	1. Notify the regulatory agency of the event, immediately. Safety check of reactor facilities after reactor scram.. 2. Cause identification and countermeasure. (include repair or replace of reactor equipments, etc.) 3. Report to the regulatory agency for the cause and measures of the event. 4. Restart the reactor. (Interruption of operation : None)	irradiation facility : 0 emergency power unit : 0 Control rod drive : 0 etc.
Reactor Scram (except of earthquake or commercial power failure)	1. Notify the regulatory agency of the event, immediately. Safety check of reactor facilities after reactor scram. 2. Course identification and countermeasures. (include repair or replace of reactor equipments, etc.) 3. Report to the regulatory agency for the cause and measure of the event within 30days (if necessary, final report submit after 30days.) 4. Repair or replace of reactor equipments. 5. Restart the reactor. (Interruption of operation : None)	primary cooling pump : 0 Human error : 0 control rod drive : 0 fuel failure : 0 etc.

Typical Reactor Scram/Shutdown Events in JMTR

■2002.5.14 on 145th cycle

Reactor scram by failure in electric circuit of control rod position detector

・Interruption of operation ………….. 50 days

■2002.12.10 on 147th cycle

Manual shutdown by water leakage from instrumentation pipe of primary cooling system

・Interruption of operation ………….. 188 days

■2003.6.20 on 148th cycle

Manual shutdown by failure of irradiation facility

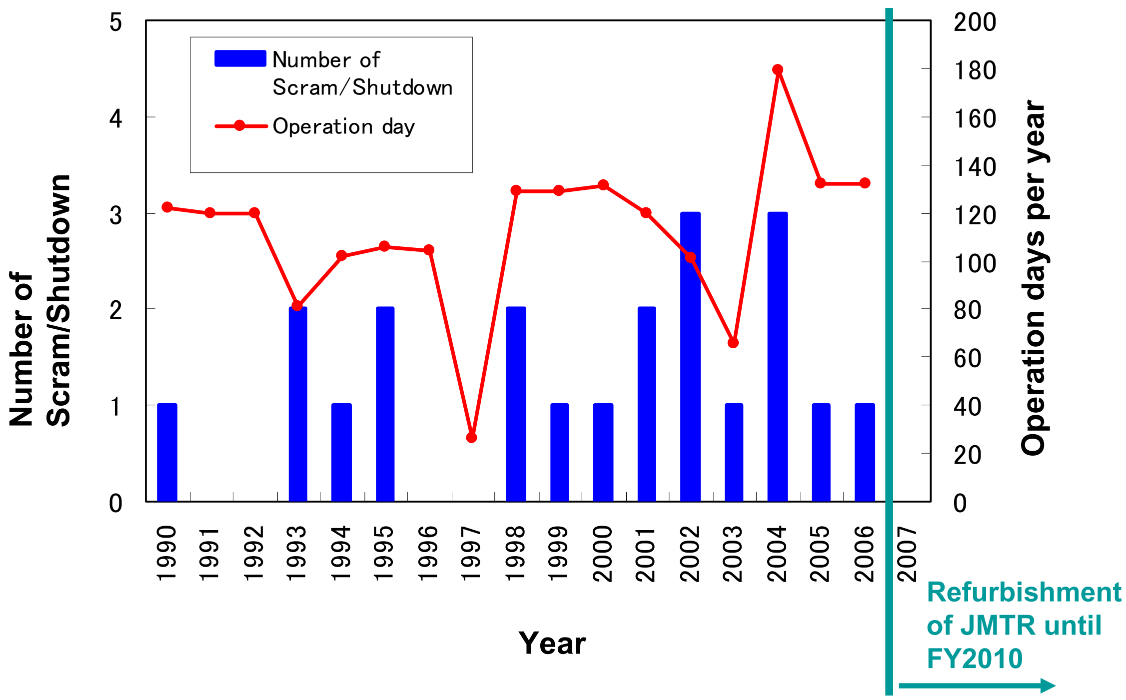
・Interruption of operation ………….. 105 days

■2004.4.4 on 152th cycle

Reactor scram by earthquake

・Discontinuation of operation
（because remained operation days were 2 days.）

History of Operation Days and Number of Reactor Shutdown Events - JMTR -



Topics 2

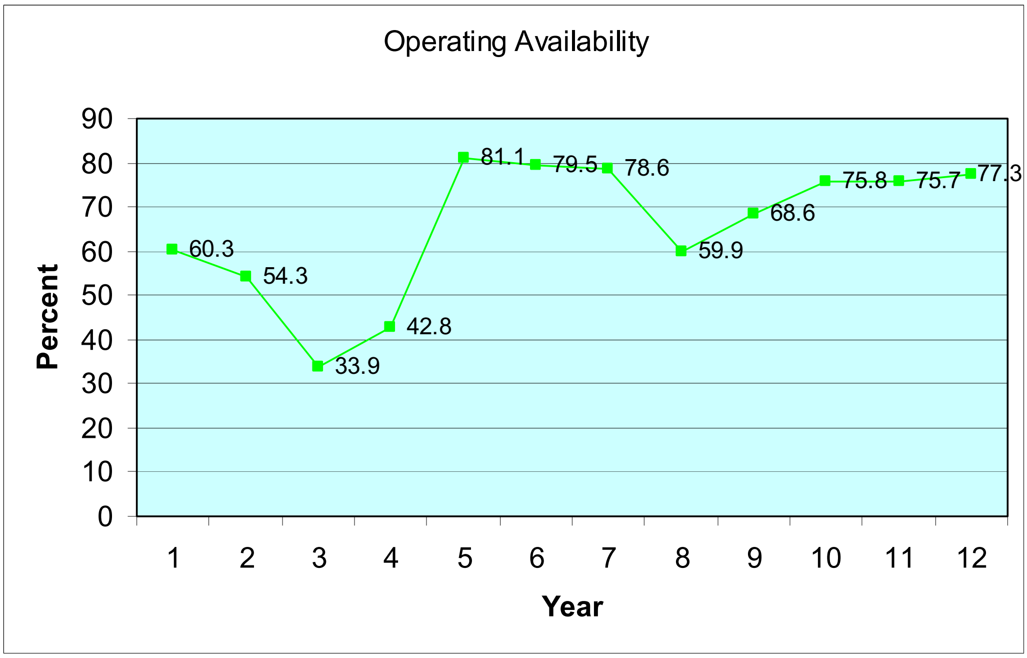
Management on Re-operation Work at Reactor Trouble

Presentation Format as to Each Reactor

Summary on Reactor Operating Routine, ATR (from Jan. to Dec.)

- **Reactor name ………….. ATR**
- **Power ……………………. 250 MWt**
- **Operation ……………….. 250 days at power 110 MWt**
- **Availability factor ………. 68 % (operation day / 365[day] x 100)**
- **Total work per year…….. 27,500 MWd**
- **Operation cycle……………63 days/cycle**

History of Operation Days
- ATR -



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Topics 2

Management on Re-operation Work at Reactor Trouble

Presentation Format as to Each Reactor

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Summary on Reactor Scram/Shutdown Events in 2007 (from Jan. to Dec.)

- **Reactor name ………….. HANARO**
- **Power ……………………. 30 MW**
- **Operation ……………….. 145 days at power 30 MW**
- **Availability factor ………. 39.7 % (operation day / 365[day] x 100)**
- **Total work per year…….. 4,248 MWD**
- **Operation cycle……………23 days/cycle**

- **Unplanned scram/shutdowns…... 3**
- **Interruption of operation….…….. 3.77 days**
- **Reasons**

  Electric supply break ………….. 1 RPS shutdown
  System problem……...…………. 2 RRS shutdown

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Countermeasure Process at Scram/Shutdown Events (2007, HANARO)

Event	Countermeasure Process	notes
Reactor Scram (earthquake or commercial power failure.)	1. For the case of earthquake over OBE (Operating Base Earthquake) - Notify the regulatory body (MEST) of the event, immediately - Safety check of reactor facilities after reactor scram. - Notify the MEST of the result of safety check and restart time. - Restart the reactor. 2. For the case of commercial power failure - Safety check of reactor facilities after reactor scram. - Report to the regulatory institute (KINS) of the event. - Restart the reactor. (Interruption of operation in 2007: about 54 hours/event)	earthquake : 0 commercial power failure : 1 etc.
Unplanned Manual Reactor Shutdown (Cause of event is clear.)	1. Safety check of reactor facilities after reactor shutdown. 2. Cause identification and countermeasure. (include repair or replace of reactor equipments, etc.) 3. Report to the regulatory institute (KINS) of the event and measures of the event. 4. Restart the reactor. (Interruption of operation : about 0 hours/event)	irradiation facility : 0 emergency power unit : 0 control rod drive : 0 etc.
Reactor Scram (except of earthquake or commercial power failure)	1. In the case of RPS(Reactor Protection System) trip - Notify the MEST of the event, immediately. - Safety check of reactor facilities after reactor scram. - Course identification and countermeasures. (include repair or replace of reactor equipments, etc.) - Repair or replace of reactor equipments. - Restart the reactor. - Report to the MEST for the cause and measure of the event within 30days 2. In the case of RRS(Reactor Regulating System) trip - Safety check of reactor facilities after reactor scram. - Cause identification and countermeasures. - Repair or replace of reactor equipments. - Report to the regulatory institute (KINS) of the event and measures of the event. - Restart the reactor. (Interruption of operation in 2007 : about 50 hours/event)	Reactor control system : 2 (RRS trips)

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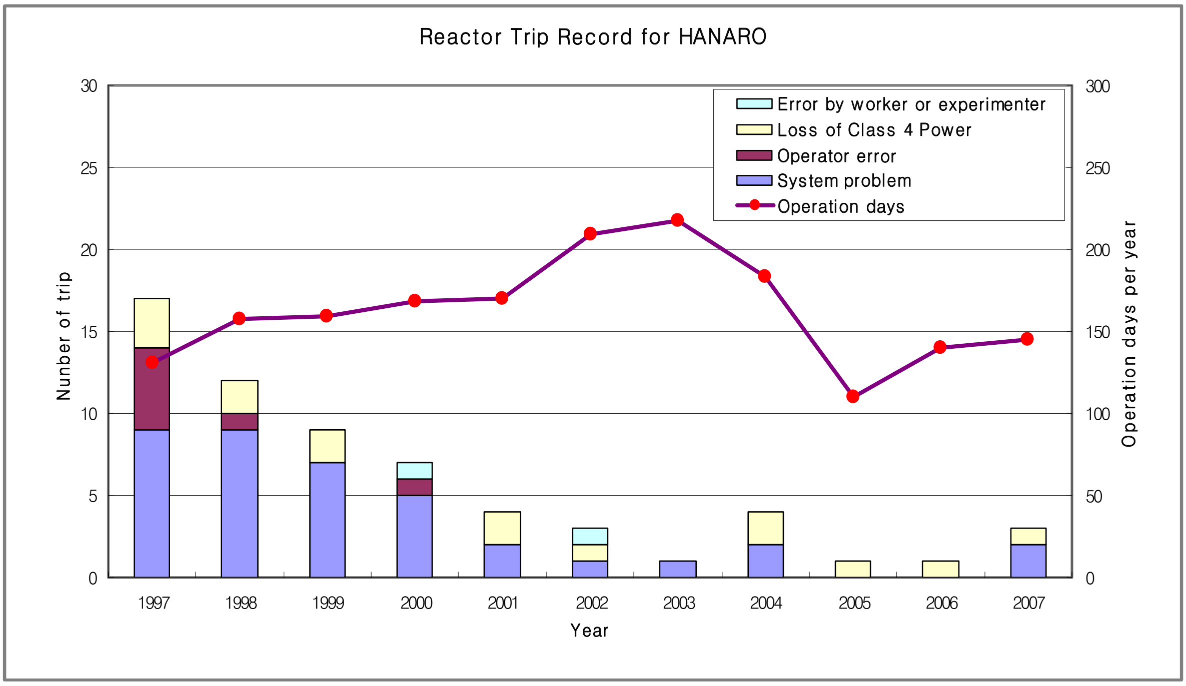
History of Operation Days and Number of Reactor Shutdown Events - HANARO -



International Symposium on Material Testing Reactors
July 16 - 17, 2008, Oarai JAEA, Japan

Topics 3

Basic Technology for Neutron Irradiation Tests in JMTR

K. Tsuchiya
(JAEA)

Items of Basic Technology

(A) Neutron Dosimetry

Fluence Monitor, Evaluation, etc.

(B) Temperature Measurement and Control

Thermocouple, Offline Measurement, etc.

(C) Instrumentation Technology

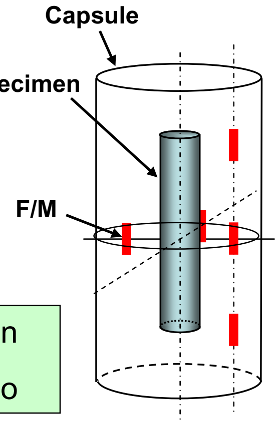
Pressure, Displacement,

Gas Concentration, etc.

Outline and detail of the technology in JMTR is described in the New-JMTR pamphlet.

(A) Neutron Dosimetry

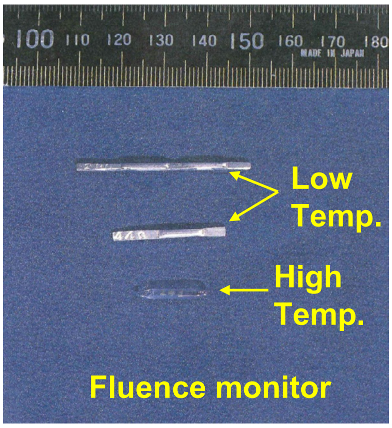
(1) Fluence Monitor



Measurement Method

Neutron fluxes at local positions have been measured by using the fluence monitors (F/Ms).

After irradiation tests, radiation activities of ^{54}Mn and ^{60}Co are measured with the germanium detector.



Fast Neutron : ^{54}Fe(n, p)^{54}Mn

Thermal Neutron : ^{59}Co(n, g)^{60}Co

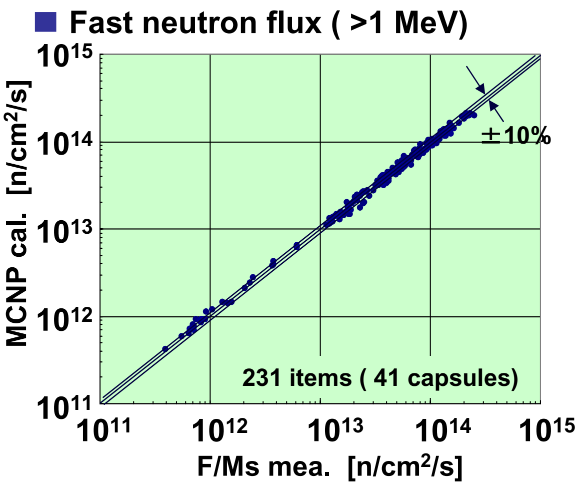
		For low temp. (~500˚C)	For high temp. (500˚C~1000˚C)
Container materials		Al	quartz
Container		ϕ2.0mm L25mm or L40mm	ϕ3mm L15mm
Monitor Materials	Fast neutron	Fe	Fe
	Thermal neutron	Al-Co	V-Co, Ti-Co

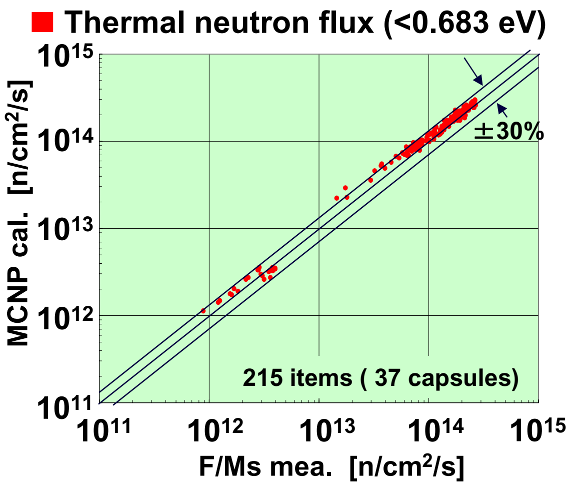
(A) Neutron Dosimetry

(2) Evaluation of Neutron Flux/Fluence

Code : Continuous Energy Monte Carlo Code MCNP
Nucl. data lib. : FSXLIBJ3R2 (based on JENDL3.2)
Cal. model : Full 3D

The calculated fast and thermal neutron flux agreed with measurements within ±10% and ±30%, respectively.





(B) Temperature Measurement and Control

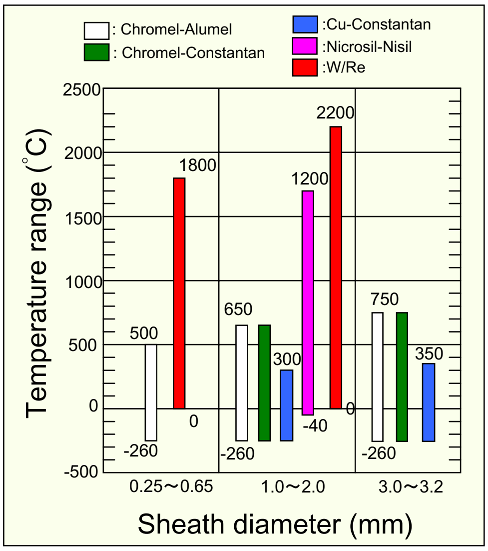
(1) On-line temperature measurement of irradiation specimens

Experiences, etc.

1) The irradiation temperature of nuclear materials and fuels is measured using the sheath type thermocouple with small diameter.
2) Thermocouples are used as monitoring and control for irradiation test equipment.
3) A maximum temperature of 1900 ˚C was measured successfully by W/Re thermocouple.

Specifications of T/C in

Type	K(C/A)	N(Nicrosil-Nisil)	W/Re
Temp. range (Experience)	～1000˚C	800～1100˚C	1000～1900˚C
Sheath diameter	φ0.5, φ1.0, φ1.6	φ1.0, φ1.5	φ1.6
Sheath materials	SUS304, SUS316, Inconel-600	SUS316, Inconel-600	Nb-1%Zr, Mo
Insulator	MgO	MgO	BeO

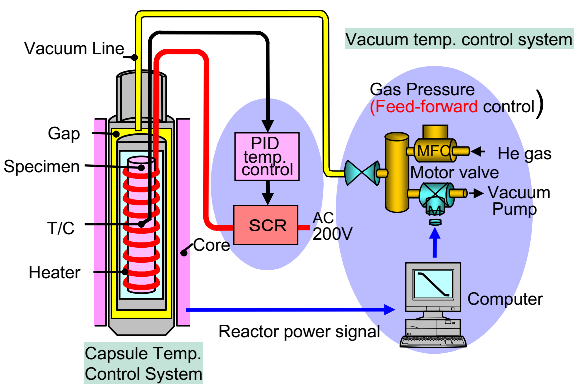


(B) Temperature Measurement and Control

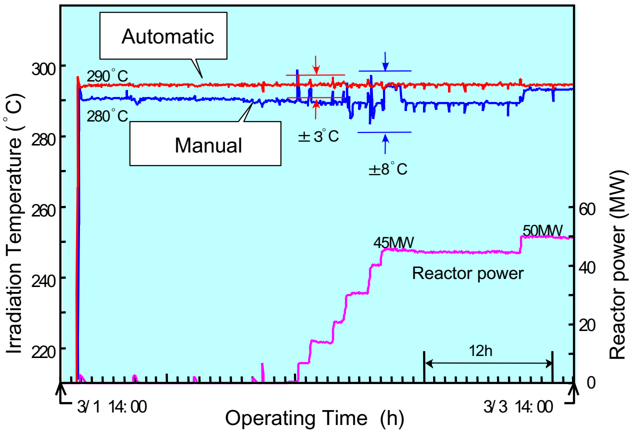
(2) Temperature control of irradiation specimens

Experiences, etc.

1) Temperature control because of losing irradiation defect at low temperature in irradiation-test of material as much as possible by rising temperature up to irradiation temperature when ratings are driven with electric heater before nuclear reactor starts.
2) A steadier temperature can control by antecedent-control-method (The temperature change by the turbulence such as the output changes of the nuclear reactor is forecast, it precedes the temperature change, and the amends operation is done) compared with a past manual operation.



Automatic temperature control device



Example for temperature control

(C) Instrumentation Technology

Items	JMTR(JA)
Thermal neutron flux (Distribution and transient)	-SPND -Fission chamber
Displacement	-Differential transformer type
Pressure	-Differential transformer + bellows type
Gas concentration	-Oxygen, hydrogen sensor (under development)
Other	-Crack growth by current potential drop -Optical measurement by light fiber

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Topics 3

Basic Technology for Neutron Irradiation tests in MTR

Presentation Format as to Each Reactor

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Basic technology for neutron irradiation tests

Items

1. Neutron Dosimetry

Fluence Monitor, Evaluation, etc.

2. Temperature Measurement and Control

Thermocouple, Offline Measurement, etc.

3. Instrumentation Technology

Pressure, Displacement, Gas Concentration, etc.

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Neutron Dosimetry

Items	HANARO(KR)	JMTR(JA)
Fluence monitor	-Thermal Neutron flux (<0.625 eV) Ag-Nb (under developing) - Fast neutron flux (>1 MeV) Fe, Ni, Ti	-Thermal neutron flux (<0.683 eV) Al-Co(<500ºC), V-Co,Ti-Co(>500ºC) -Fast neutron flux (>1MeV) Fe
Evaluation method	Neutron and gamma calculation - Code : MCNP-4C2 -Core model: 3D full core, 142 nuclides, 13,104 fuel segments - Nuclear data library - Neutron: ENDF-B/VI, TMCCS, ENDL92 - Gamma: MCPLIB	Neutron and gamma calculation by full core 3D -Code : MCNP4B -Nuclear data library Neutron: FSXLIBJ3R2 (based on JENDL3.2) TMCCS(S(alpha,beta),based on ENDF-B/III) Gamma: MCPLIB (based on DLC-7E)
Error of evaluation	- Fast neutron flux (>1 Mev) : ±10%	-Thermal neutron flux (<0.683 eV) : ±30% -Fast neutron flux (>1MeV) : ±10%
Provided data to user	- Thermal and fast neutron flux and fluence (include distribution) - Neutron / gamma spectrum	-Thermal and fast neutron flux and fluence (include distribution) -Neutron / gamma spectrum (option)

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Temperature Measurement and Control

Items	HANARO(KR)	JMTR(JA)
Thermocouple	- K(~1000ºC) ø1.0, ø1.6mm - W/Re (1000~1900ºC) ø1.4mm	-K(~1000ºC) ø0.5, ø1.0, ø1.6mm -N (800~1100ºC) ø1.0, ø1.5mm -W/Re (1000~1900ºC) ø1.6mm -Multi-paired T/C ø1.8mm (7 points)
Offline measurement	Melt wire (Eutectic alloys) 200~400ºC	Melt wire (In,Sn,Pb,Bi,Ag,Zn) 95~420ºC
Calculation code	- Capsule temperature design GENTIC(1D) ANSYS(2D, 3D) - Evaluation of specimen's temperatur e ANSYS(2D, 3D), others	-Capsule temperature design GENTIC(1D) NISA(3D) -Evaluation of specimen's temperature NISA(3D)
Temp. control	Heater+gas pressure Ex. 290±10ºC (He, Ne) Mixed gas	Heater+gas pressure (constant temperature control by reacto r power feed-forward control) Ex. 290±3ºC

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Other Instrumentation

Items	HAANRO(KR)	JMTR(JA)
Thermal neutron flux (Distribution and transient)	- SPND	-SPND - Fission chamber
Displacement	- Differential transformer type	- Differential transformer type
Pressure	- Differential transformer + bellows type	- Differential transformer + bellows type
Gas concentration		- Oxygen, hydrogen sensor (under development)
Other	- Creep and fatigue test (in-situ measurement)	- Crack growth by current potential drop - Optical measurement by light fiber

Topics 3

Basic Technology for Neutron Irradiation tests in MTR

Presentation Format as to Each Reactor

Basic technology for neutron irradiation tests

Items

1. Neutron Dosimetry
   Fluence Monitor, Evaluation, etc.
2. Temperature Measurement and Control
   Thermocouple, Offline Measurement, etc.
3. Instrumentation Technology
   Pressure, Displacement, Gas Concentration, etc.

Neutron Dosimetry

Items	Advanced Test Reactor	JMTR(JA)
Fluence monitor	Thermal =>Co-Al < 500ºC < Co-V Epithermal => Nb Fast => Fe, Ni, SST Helium Accumulation Fluence Monitors	-Thermal neutron flux (<0.683 eV) Al-Co(<500ºC), V-Co,Ti-Co(>500ºC) -Fast neutron flux (>1MeV) Fe
Evaluation method	Neutron and gamma calculation by full core 3D Modeling Code : MCNP5 Nuclear data libraries Neutron: JENDL3.2 Libraries ENDF-B/V & B/VI Gamma: MCPLIB Fuel Burn-up ORIGEN2	Neutron and gamma calculation by full core 3D -Code : MCNP4B -Nuclear data library Neutron: FSXLIBJ3R2 (based on JENDL3.2) TMCCS(S(alpha,beta),based on ENDF-B/III) Gamma: MCPLIB (based on DLC-7E)
Error of evaluation	MCNP uncertainty = 2.5%, Total uncertainty dependent on cross section library	-Thermal neutron flux (<0.683 eV) : ±30% -Fast neutron flux (>1MeV) : ±10%
Provided data to user	Thermal and fast neutron flux and fluence (include distribution) Neutron / gamma spectrum Fuel burn-up => total, distribution, isotopic distribution	-Thermal and fast neutron flux and fluence (include distribution) -Neutron / gamma spectrum (option)

Temperature Measurement and Control

Items	Advanced Test Reactor	JMTR(JA)
Thermocouple	Type K => Ø 3.2, 2.4, 1.6, 1.0, 0.5 (individual conductor) mm Type N => Ø 3.2, 2.4, 1.6, 1.0, 0.5 (individual conductor) mm Type C => Ø 3.2, 2.4, 1.6, 1.0 mm Mo-Nb => Ø 3.2, 1.6 mm ≤ 1600 C	-K(~1000ºC) ø0.5, ø1.0, ø1.6mm -N (800~1100ºC) ø1.0, ø1.5mm -W/Re (1000~1900ºC) ø1.6mm -Multi-paired T/C ø1.8mm (7 points)
Offline measurement	Melt Wire (In, Sn, Pb, Bi, Ag, Zn, Be, Cu) SiC, Na Capsules	Melt wire (In,Sn,Pb,Bi,Ag,Zn) 95~420ºC
Calculation code	ABAQUS (3D) RELAP 5 Fluent (3D) FIDAP	-Capsule temperature design GENTIC(1D) NISA(3D) -Evaluation of specimen's temperature NISA(3D)
Temp. control	Active control - He/Ne or He/Ar mixture with gamma and reaction heating for heat source Passive – Same w/pure gases (He, Ar, Ne, etc.) Heaters on PWR loops Li & Na Bonding to equalize temperatures	Heater+gas pressure (constant temperature control by reactor power feed-forward control) Ex. 290±3ºC

Other Instrumentation

Items	Advanced Test Reactor	JMTR(JA)
Thermal neutron flux (Distribution and transient)	SPND Fission Chamber U-Al flux wire	-SPND -Fission chamber
Displacement	Linear Position Transducer	-Differential transformer type
Pressure	Impulse line to differential transmitter	-Differential transformer + bellows type
Gas concentration	Oxygen, carbon monoxide & dioxide, moisture, tritium, fission gases (Xe, Kr & I)	-Oxygen, hydrogen sensor (under development)
Other	Force – load cell	-Crack growth by current potential drop -Optical measurement by light fiber

Topics 3

Basic Technology for Neutron Irradiation tests in MTR

Presentation Format as to Each Reactor

Basic technology for neutron irradiation tests

Items

1. Neutron Dosimetry
   Fluence Monitor, Evaluation, etc.
2. Temperature Measurement and Control
   Thermocouple, Offline Measurement, etc.
3. Instrumentation Technology
   Pressure, Displacement, Gas Concentration, etc.

Neutron Dosimetry

Items	BR-2	JMTR(JA)
Fluence monitor	-Thermal flux + Epithermal : Al-Co and Al-Ag (bare or under Cd or Gd) - Fast flux : Equivalent Fission Flux or Flux> 1Mev : $^{57}Fe(n, p)$, $^{58}Ni(n,p)$, $^{47}Ti(n,p)$, $^{93}Nb(n,n')$	-Thermal neutron flux (<0.683 eV) Al-Co(<500ºC), V-Co,Ti-Co(>500ºC) -Fast neutron flux (>1MeV) Fe
Evaluation method	- Full 3-D MCNP4C + ENDF-B.VI.8 -Inter-laboratory comparison on experimental basis to check our ability to produce correct experimental values - international (OECD/NEA + IAEA) calculational benchmarks	Neutron and gamma calculation by full core 3D -Code : MCNP4B -Nuclear data library Neutron: FSXLIBJ3R2 (based on JENDL3.2) TMCCS(S(alpha,beta),based on ENDF-B/III) Gamma: MCPLIB (based on DLC-7E)
Error of evaluation	Experimental uncertainties on neutron fluxes ~2%, on Gamma ~15% C/E : neutrons ~+/-10%, Gamma ~+/-20%	-Thermal neutron flux (<0.683 eV) : ±30% -Fast neutron flux (>1MeV) : ±10%
Provided data to user	-Thermal, Epithermal, Fast (>0.1 MeV), Fast (> 1 MeV), Eq.Fission - Gamma dose or heating	-Thermal and fast neutron flux and fluence (include distribution) -Neutron / gamma spectrum (option)

Temperature Measurement and Control

Items	BR-2	JMTR(JA)
Thermocouple	- Same as JMTR	-K(~1000ºC) ø0.5, ø1.0, ø1.6mm -N (800~1100ºC) ø1.0, ø1.5mm -W/Re (1000~1900ºC) ø1.6mm -Multi-paired T/C ø1.8mm (7 points)
Offline measurement	- From IRMM in Geel (see their Web site)	Melt wire (In,Sn,Pb,Bi,Ag,Zn) 95~420ºC
Calculation code	- FLICA - SYSTUS - Home Code	-Capsule temperature design GENTIC(1D) NISA(3D) -Evaluation of specimen's temperature NISA(3D)
Temp. control	-Electrical heater - gas insulator - Temp. Control: +/- 5° C -- Range of Temp.: 50 to 650° C	Heater+gas pressure (constant temperature control by reactor power feed-forward control) Ex. 290±3ºC

Other Instrumentation

Items	BR-2	JMTR(JA)
Thermal neutron flux (Distribution and transient)	- On-line neutron flux monitoring: miniat. FC (U8, NP7, TH2), SPND - Gamma-ray monitor. SPGD, Calorimeter, Red-Perspex (short irradiat)	-SPND -Fission chamber
Displacement	- LVDT	-Differential transformer type
Pressure	-LVDT	-Differential transformer + bellows type
Gas concentration		-Oxygen, hydrogen sensor (under development)
Other		-Crack growth by current potential drop -Optical measurement by light fiber

Neutron Dosimetry

Items	WWR-K reactor
Thermal neutron flux (Distribution and transient)	- SPND - Fission chamber - Neutron counters
Spectrum monitor	Au, Cu, Dy – thermal neutrons Rh (>0.8 MeV), In (>1.15 MeV), Ni, S (>3 MeV) – fast neutrons
Evaluation method	Neutron calculation by full core 3D - Monte-Carlo Codes: MCU-REA, MCNP5
Provided data to user	Thermal and fast neutron flux and fluence, Neutron spectrum

Temperature Measurement and Control, other instrumentation

Items	WWR-K Reactor
Thermocouple	- Allimel-chromel, - W-Re, - Cu-Const - Allimel-Capel
Offline measurement	- melt metals
Calculation code	- other institute of NNC
Temp. control	Heater, gas pressure
Pressure	- Differential transformer type, - resistor type, - mechanical manometer
Gas concentration	- Radiometers for radioactive gases

国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位 名称	SI 基本単位 記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 基本単位 名称	SI 基本単位 記号
面積	平方メートル	m^2
体積	立法メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m-1
密度（質量密度）	キログラム毎立法メートル	kg/m^3
質量体積（比体積）	立法メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
（物質量の）濃度	モル毎立方メートル	mol/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率	（数の） 1	1

表５．SI 接頭語

乗数	接頭語	記号	乗数	接頭語	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	µ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表３．固有の名称とその独自の記号で表されるSI組立単位

組立量	SI 組立単位 名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン[a]	rad		$m \cdot m^{-1}=1$[b]
立体角	ステラジアン[a]	sr[c]		$m^2 \cdot m^{-2}=1$[b]
周波数	ヘルツ	Hz		s^{-1}
力	ニュートン	N		$m \cdot kg \cdot s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1} \cdot kg \cdot s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N・m	$m^2 \cdot kg \cdot s^{-2}$
工率，放射束	ワット	W	J/s	$m^2 \cdot kg \cdot s^{-3}$
電荷，電気量	クーロン	C		s・A
電位差（電圧），起電力	ボルト	V	W/A	$m^2 \cdot kg \cdot s^{-3} \cdot A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2} \cdot kg^{-1} \cdot s^4 \cdot A^2$
電気抵抗	オーム	Ω	V/A	$m^2 \cdot kg \cdot s^{-3} \cdot A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2} \cdot kg^{-1} \cdot s^3 \cdot A^2$
磁束	ウエーバ	Wb	V・s	$m^2 \cdot kg \cdot s^{-2} \cdot A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg \cdot s^{-2} \cdot A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2 \cdot kg \cdot s^{-2} \cdot A^{-2}$
セルシウス温度	セルシウス度[d]	℃		K
光束	ルーメン	lm	cd・sr[c]	$m^2 \cdot m^{-2} \cdot cd=cd$
照度	ルクス	lx	lm/m^2	$m^2 \cdot m^{-4} \cdot cd=m^{-2} \cdot cd$
（放射性核種の）放射能	ベクレル	Bq		s^{-1}
吸収線量，質量エネルギー分与，カーマ	グレイ	Gy	J/kg	$m^2 \cdot s^{-2}$
線量当量，周辺線量当量，方向性線量当量，個人線量当量,組織線量当	シーベルト	Sv	J/kg	$m^2 \cdot s^{-2}$

(a) ラジアン及びステラジアンの使用は、同じ次元であっても異なった性質をもった量を区別するときの組立単位の表し方として利点がある。組立単位を形作るときのいくつかの用例は表４に示されている。
(b) 実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号“1”は明示されない。
(c) 測光学では、ステラジアンの名称と記号srを単位の表し方の中にそのまま維持している。
(d) この単位は、例としてミリセルシウス度m℃のようにSI接頭語を伴って用いても良い。

表４．単位の中に固有の名称とその独自の記号を含むSI組立単位の例

組立量	SI 組立単位 名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa・s	$m^{-1} \cdot kg \cdot s^{-1}$
力のモーメント	ニュートンメートル	N・m	$m^2 \cdot kg \cdot s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg \cdot s^{-2}$
角速度	ラジアン毎秒	rad/s	$m \cdot m^{-1} \cdot s^{-1}=s^{-1}$
角加速度	ラジアン毎平方秒	rad/s^2	$m \cdot m^{-1} \cdot s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg \cdot s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2 \cdot kg \cdot s^{-2} \cdot K^{-1}$
質量熱容量（比熱容量），質量エントロピー	ジュール毎キログラム毎ケルビン	J/(kg・K)	$m^2 \cdot s^{-2} \cdot K^{-1}$
質量エネルギー（比エネルギー）	ジュール毎キログラム	J/kg	$m^2 \cdot s^{-2} \cdot K^{-1}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m・K)	$m \cdot kg \cdot s^{-3} \cdot K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1} \cdot kg \cdot s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m \cdot kg \cdot s^{-3} \cdot A^{-1}$
体積電荷	クーロン毎立方メートル	C/m^3	$m^{-3} \cdot s \cdot A$
電気変位	クーロン毎平方メートル	C/m^2	$m^{-2} \cdot s \cdot A$
誘電率	ファラド毎メートル	F/m	$m^{-3} \cdot kg^{-1} \cdot s^4 \cdot A^2$
透磁率	ヘンリー毎メートル	H/m	$m \cdot kg \cdot s^{-2} \cdot A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2 \cdot kg \cdot s^{-2} \cdot mol^{-1}$
モルエントロピー，モル熱容量	ジュール毎モル毎ケルビン	J/(mol・K)	$m^2 \cdot kg \cdot s^{-2} \cdot K^{-1} \cdot mol^{-1}$
照射線量（Ｘ線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1} \cdot s \cdot A$
吸収線量率	グレイ毎秒	Gy/s	$m^2 \cdot s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4 \cdot m^{-2} \cdot kg \cdot s^{-3}=m^2 \cdot kg \cdot s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2 \cdot sr)$	$m^2 \cdot m^{-2} \cdot kg \cdot s^{-3}=kg \cdot s^{-3}$

表６．国際単位系と併用されるが国際単位系に属さない単位

名称	記号	SI 単位による値
分	min	1 min=60s
時	h	1h =60 min=3600 s
日	d	1 d=24 h=86400 s
度	°	1° =(π/180) rad
分	’	1’ =(1/60)° =(π/10800) rad
秒	”	1” =(1/60)’ =(π/648000) rad
リットル	l、L	$1l=1\ dm^3=10^{-3}m^3$
トン	t	$1t=10^3$ kg
ネーパ	Np	1Np=1
ベル	B	1B=(1/2)ln10(Np)

表７．国際単位系と併用されこれに属さない単位で SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位であらわされる数値
電子ボルト	eV	$1eV=1.60217733(49)\times10^{-19}J$
統一原子質量単位	u	$1u=1.6605402(10)\times10^{-27}kg$
天文単位	ua	$1ua=1.49597870691(30)\times10^{11}m$

表８．国際単位系に属さないが国際単位系と併用されるその他の単位

名称	記号	SI 単位であらわされる数値
海里		１海里=1852m
ノット		１ノット=１海里毎時=(1852/3600)m/s
アール	a	$1\ a=1\ dam^2=10^2m^2$
ヘクタール	ha	$1\ ha=1\ hm^2=10^4m^2$
バール	bar	$1\ bar=0.1MPa=100kPa=1000hPa=10^5Pa$
オングストローム	Å	$1\ Å=0.1nm=10^{-10}m$
バーン	b	$1\ b=100fm^2=10^{-28}m^2$

表９．固有の名称を含むCGS組立単位

名称	記号	SI 単位であらわされる数値
エルグ	erg	$1\ erg=10^{-7}\ J$
ダイン	dyn	$1\ dyn=10^{-5}N$
ポアズ	P	$1\ P=1\ dyn \cdot s/cm^2=0.1Pa \cdot s$
ストークス	St	$1\ St=1cm^2/s=10^{-4}m^2/s$
ガウス	G	$1\ G \triangleq 10^{-4}T$
エルステッド	Oe	$1\ Oe \triangleq (1000/4\pi)A/m$
マクスウェル	Mx	$1\ Mx \triangleq 10^{-8}Wb$
スチルブ	sb	$1\ sb=1cd/cm^2=10^4cd/m^2$
ホト	ph	$1\ ph=10^4lx$
ガル	Gal	$1\ Gal=1cm/s^2=10^{-2}m/s^2$

表10．国際単位に属さないその他の単位の例

名称	記号	SI 単位であらわされる数値
キュリー	Ci	$1\ Ci=3.7\times10^{10}Bq$
レントゲン	R	$1\ R = 2.58\times10^{-4}C/kg$
ラド	rad	$1\ rad=1cGy=10^{-2}Gy$
レム	rem	$1\ rem=1\ cSv=10^{-2}Sv$
X線単位		$1X\ unit=1.002\times10^{-4}nm$
ガンマ	γ	$1\gamma=1\ nT=10^{-9}T$
ジャンスキー	Jy	$1\ Jy=10^{-26}W \cdot m^{-2} \cdot Hz^{-1}$
フェルミ		$1\ fermi=1\ fm=10^{-15}m$
メートル系カラット		$1\ metric\ carat = 200\ mg = 2\times10^{-4}kg$
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	
ミクロン	µ	$1\ \mu=1\mu m=10^{-6}m$

（第７版，1998年改訂）