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2011-003

Proceedings of the 4th International Symposium on Material Testing Reactors
December 5-9, 2011, Oarai, Japan

(Eds.) Masahiro ISHIHARA and Masahide SUZUKI

Neutron Irradiation and Testing Reactor Center
Oarai Research and Development Center

March 2012
Japan Atomic Energy Agency

日本原子力研究開発機構

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Proceedings of the 4th International Symposium on Material Testing Reactors
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Oarai-machi, Higashiibaraki-gun, Ibaraki-ken

(Received December 21, 2011)

This report is the Proceedings of the fourth International Symposium on Material Testing Reactors hosted by Japan Atomic Energy Agency (JAEA). The first symposium was held on 2008, at the Oarai Research and Development Center of JAEA, the second, 2009, Idaho National Laboratory (INL) of United States and the third 2010, Nuclear Research Institute (NRI) in Czech Republic to exchange information for deep mutual understanding of material testing reactors. The fourth symposium was originally scheduled to be held INVAP in Argentina. However, the aftermath of volcanic explosion at Chili forced the symposium to change place. Total 111 participants attended from Argentina, Belgium, France, Germany, Indonesia, Malasia, Korea, South Africa, Switzerland, the United State and Japan.

This symposium addressed the general topics of “status and future plan of material testing reactors”, “advancement of irradiation technology”, “expansion of industry use(RI)”, “facility, upgrade, aging management”, ”new generation MTR”, “advancement of PIE technology”, “development of advanced driver fuel”, and “nuclear human resource development(HRD) for next generation”, and 39 presentations were made. Furthermore, three topics, “Necessity of cooperation for Mo-99 production by (n,gamma) reaction”, “ Necessity of standardization of irradiation technology” and “Conceptual design of next generation materials testing reactor by collaboration”, were selected and discussed.

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

「第4回汎用照射試験炉に関する国際会議」論文集
2011年12月5-9日、大洗

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、鈴木　雅秀

（2011年12月21日受理）

本論文集は、（独）日本原子力研究開発機構主催の「第4回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、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年に第1回がJAEA大洗研究開発センター、2009年に第2回目が米国INL、2010年に第3回がチェコNRIで開催された。今年で第4回目となる本会議は、当初、アルゼンチンINVAPで行われる予定であったが、火山噴火の影響で急遽、日本で開催されることになった。会議にはアルゼンチン、ベルギー、ブラジル、フランス、ドイツ、インドネシア、マレーシア、フィリピン、韓国、南アフリカ、スイス、ウクライナ、米国及び日本から合計111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」、「照射技術」、「産業利用の拡大(RI)」、「施設、改良、高経年化の管理」、「次世代の材料照射試験炉」、「照射後試験技術」、「改良ドライバー燃料の開発」、及び「次世代の原子力人材教育」のセッションにおいて39件の講演が行われ、「(n, γ)法によるMo-99製造に関する協力の必要性」、「照射技術の標準化の必要性」、及び「次世代材料試験炉の共同概念設計」のトピックスについて討論が行われた。

大洗研究開発センター：〒311-1393　茨城県東茨城郡大洗町成田町4002

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1. Introduction

The 4th International Symposium on Material Testing Reactors was held from December 5 to 9, 2011 at the Oarai Park Hotel, Oarai-town, Japan. This symposium annually congregates worldwide specialist related to Material testing reactor (MTR), to exchange experiences, insights and ideas for fostering a community network. The first symposium was held at Oarai Research & Development center, Japan Atomic Energy Agency (JAEA), Japan, on July 16th - 17th, 2008, the second at Idaho National Laboratory (INL), Unites States, on September 28th – October 1st, 2009, and the third at Nuclear Research Institute (NRI), Czech Republic, June 21st – 23rd, 2010. The 4th symposium was initially planned to be held at Argentina, however Puyehue volcano eruption at Chili unfortunately forced the symposium to change place. Then, JAEA held this symposium on behalf of the INVAP, Argentina.

For the 4th Symposium, papers were called for the following areas:

- National and international policies for strategic planning on MTR,
- MTR utilization, irradiation and PIE technology,
- Recent Operational Experiences,
- New and innovative MTR designs,
- Facility upgrades refurbishment, life extension and decommissioning,
- Nuclear regulation and licensing,
- Human resources, training and qualification,
- MTR fuels, design, manufacturing, and qualification.

Total of 42 abstracts were applied for the presentation, and nine fields of sessions, including an invited session for circumstances surrounding research reactors, were planned. Finally, we have 39 presentations in the symposium as is shown in attached symposium program. Total 111 persons from 14 countries (Argentina, Belgium, Brazil, France, Germany, Indonesia, Malaysia, Philippines, Korea, South Africa, Switzerland, Ukraine, United States and Japan) attended the symposium. After 39 presentations, three topics, "Necessity of cooperation for Mo-99 production by (n,gamma) reaction", " Necessity of standardization of irradiation technology" and "Conceptual design of next generation materials testing reactor by collaboration" were selected and discussed. Participants and presentations are shown in Figs.1and 2.

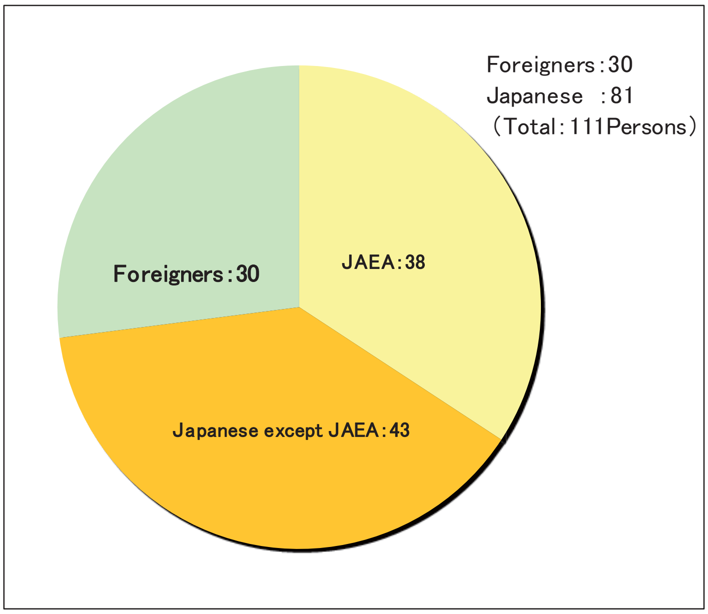

Fig. 1 Participants of the 4th International Symposium on Material Testing Reactors.

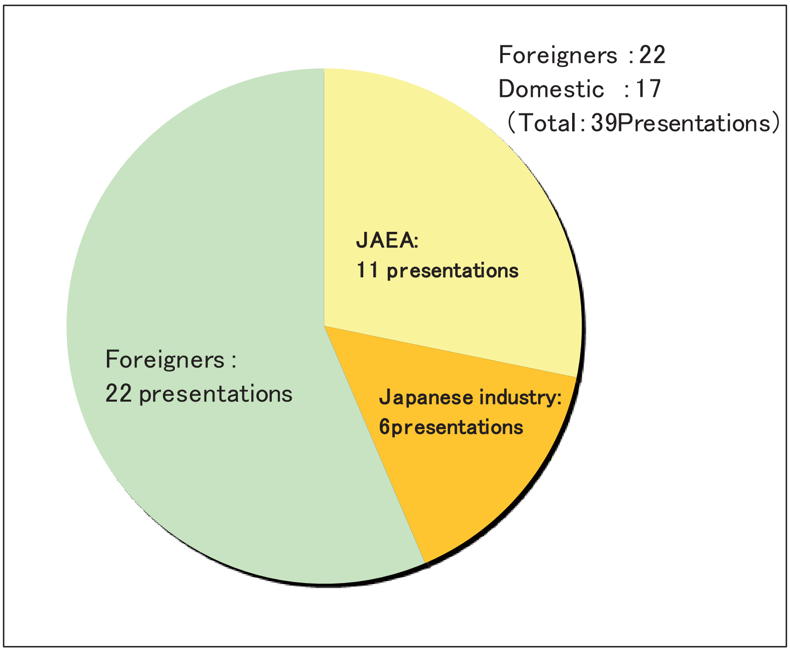

Fig. 2 Presentations of the 4th International Symposium on Material Testing Reactors.

2. Circumstances Surrounding RR's in the World

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2.1 Research with Neutrons in Germany in View of the Phase Out of Nuclear Power

Joachim Knebel[1], Winfried Petry[2]

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E-mail : joachim.knebel@kit.edu

[2] *Technische Universität München*
Forschungsneutronenquelle Heinz Maier-Leibnitz (FRM II),
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E-mail : winfried.petry@frm2.tum.de

The central goals of German energy policy are guided by economic efficiency, security of supply and environmental compatibility. The German path towards an era of renewable energies requires intensified efforts and successes in innovation and new technologies in order to heavily accelerate the modification of the existing energy system. Priority is given to research and development in the fields of energy efficiency and renewable energies together with issues such as energy storages and energy grids.

Thus, Germany has the potential and aim to develop into one of the most energy effective and environmentally friendly national economy world-wide.

The deployment of nuclear energy for electricity production in Germany is limited to the year 2022 in order to allow the renewable energies to take this role, technologically and economically. In the frame of national provident research and to provide know-how in nuclear safety at the internationally highest level of science and technology, nuclear safety research is of high importance in Germany. This includes the operation of research and teaching reactors. National research centres and universities are equally contributing to both nuclear safety research and education and training.

Traditionally Germany's research reactors focus on the delivery of intense neutron beams for basic research with neutrons. Both universities and national research centres within the Helmholtz Association are operating these neutron sources. Today Germany operates three neutron sources, the TRIGA Reactor operated by the University of Mainz, the swimming pool reactor BER2 operated by the Helmholtz Zentrum Berlin and the recently inaugurated multiple purpose Neutron Research Source Heinz Maier Leibnitz (FRM II). This is complemented by four teaching reactors at almost zero thermal power: at the TU Dresden, the University Stuttgart and the Universities of Applied Sciences in Furtwangen and Ulm.

Beside this Germany is also stakeholder of the world's most famous neutron centre, the Institute Laue Langevin at Grenoble, and is heavily engaged at the upcoming European Spallation Neutron Source (ESS) in Lund Sweden.

This base of neutron beams of highest brilliance is used to give answers to the great challenges of modern society like Energy, Health, Information Technology, Mobility … For example in Germany neutrons are used to cure tumors by irradiaton with fast neutrons, to produce homogenously doped Silicon for future high tension direct current transport or to reveal the physics of high temperature (T_c) supraconductivity.

Research with Neutrons in Germany in View of the Phase Out of Nuclear Power

6.12.2011 - ISMTR-4 - Oarai Japan

Dr.-Ing. Joachim U. Knebel, Chief Science Officer Bereich 4

KIT – University of the State of Baden-Wuerttemberg and National Research Center of the Helmholtz Association

www.kit.edu

Research with Neutrons in Germany in View of the Phase Out of Nuclear Power

Reprint and dissemination of this contribution is only permitted with written approval of Karlsruhe Institute of Technology (KIT) and Technical University of Munich (TUM) and citation of the authors: Joachim Knebel and Winfried Petry. Sending of the reprint to the author is requested.

The spoken work is valid only.

2 6.12.2011 Joachim Knebel & Winfried Petry - ISMTR-4 - Oarai Japan Chief Science Officer Bereich 4 (CSO-4)

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- Fukushima
- Energy Policy and Energy Research in Germany (after Fukushima)
- Personal Conclusions
- Research Reactors in Germany

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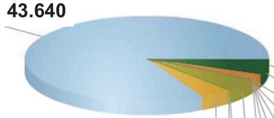
Articles in the Media „after Fukushima“: Status 14.04.2011



Source:
AREVA & Meltwater News

- **Until 12.04.2011: KIT contributions concerning Fukushima**
  - **More than 150 articles in print media,**
  - **More than 50 TV contributions,**
  - **More than 80 radio interviews,**
  - **More than 270 articeles in online media.**

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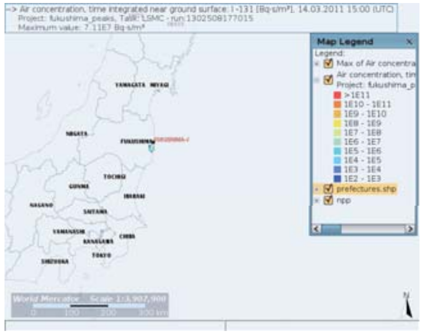
KIT-RODOS Calculations of Potential Releases

- Daily 24 hour forecast on KIT Homepage: http://www.kit.edu/besuchen/6042.php.
- Today, RODOS is operational in more than 10 European countries.



Example: I-131 concentration in the air during 14.03-16.03, calculated with JRODOS

Source:
Wolfgang Raskob, KIT-IKET

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Angela Merkel
www.bundeskanzlerin.de

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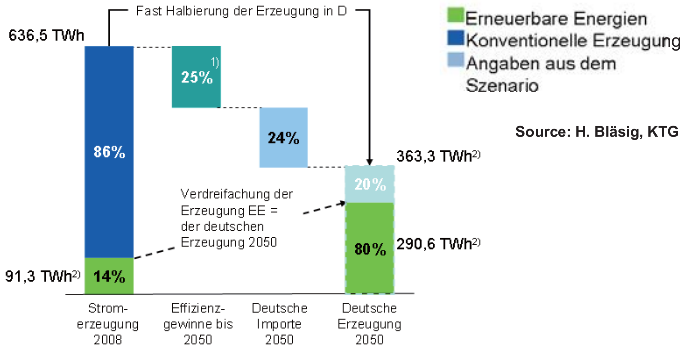
Energy Concept of the German Government 2010
(Base Scenario)



Source: H. Bläsig, KTG

- **BM Röttgen BMU on 30.8.2011: „20% renewables in electricity!“**
- **But: Electricity consumption increases by 3.6% from 2009 to 2010.**

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11.3.2011: Fukushima

Block 4 Block 3 Block 2 Block 1

www.n-tv.de – ap, dpa

- **Immediate shutdown of older NPPs in Germany,**
- **Stress tests for NPP in Germany and Europe,**
- **Ethic Commission in Germany,**
- **Phase-out of nuclear electricity production in Germany until 2022!**

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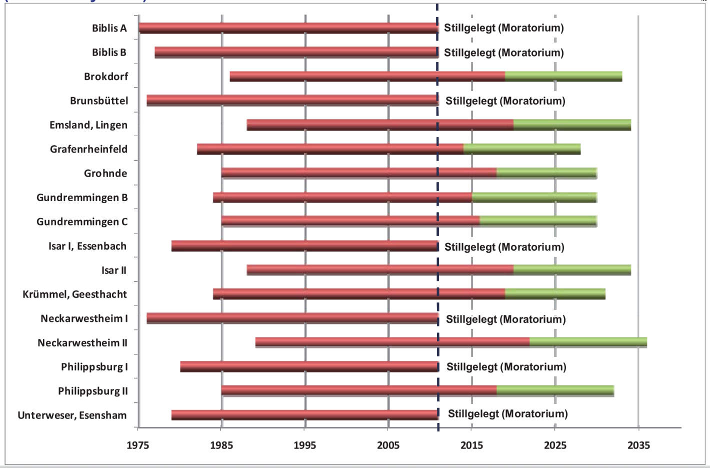
Operation Times of German NPPs
(Status May 2011)



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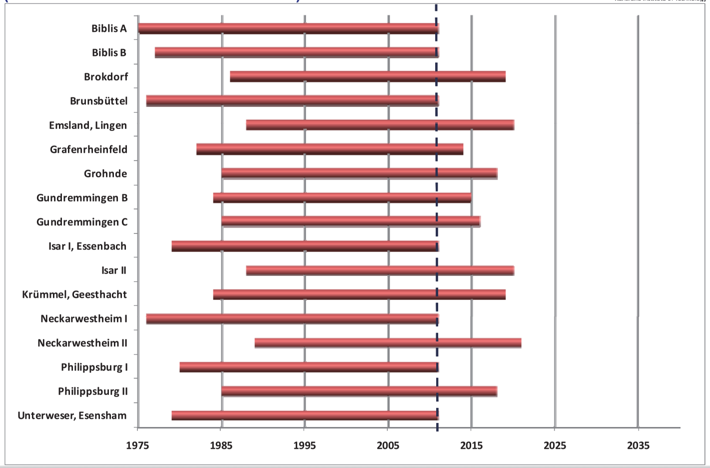
Operation Times of German NPPs
(Decision of the Merkel Government)



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Die Bundesregierung

New Energy for Germany (10/2011) www.bundesregierung.de (1/2)

Germany will be one of the most energy efficient and environmentally friendly economies world-wide.

- From 2009 to 2011: Electricity production from renewable energies increases from 17 to 20 percent. Goal 2020: 35%, Goal 2050: 80%.
- End of 2022, the last German NPP will be shut down. Temporarily, there will be more electricity from coal and gas, in order to guarantee a reliable and financable electricity supply. Modernisation of existing and start of operation of new conventional power plants.
- Erection of large wind parks, strong support of biomass.

Quelle: http://de.wikipedia.org/wiki/Liste_der_Offshore-Windparks

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Die Bundesregierung

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- Energetic redecoration of buildings: Increase of programme funds for CO_2-efficient buildings.

Quelle: http://hallo.news352.lu/edito-77382-rieseninvest-in-die-stromautobahn.html

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Federal Ministry for Economics and Technology: „Research for an environmentally friendly, reliable and financable energy supply“

- 6th Energy Research Programme of the Federal Government defines strategically important topics:
  - Energy efficiency,
  - Renewable energies,
  - Energy storage,
  - Grids.

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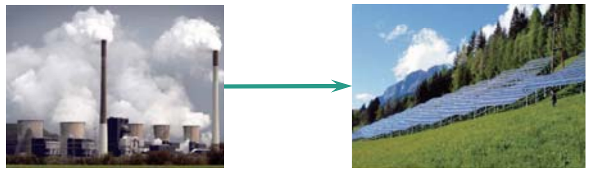
Fundamental, New Challenges to our Energy System:

Energie für Deutschland

- From point to area
- From central to decentral
- From direct consumption to storage
  Consumer attitude follows energy offers
- Consumer actively controls and provides energy services



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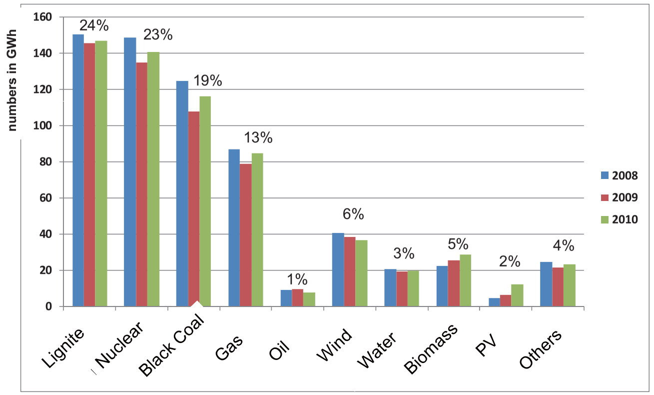
Brutto Electricity Production in Germany according to Energy Sources



Quelle: BDEW

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Die Bundesregierung — New Energy for Germany (10/2011) www.bundesregierung.de (1/2)

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Quelle: http://de.wikipedia.org/wiki/Liste_der_Offshore-Windparks

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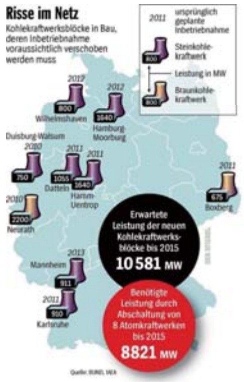
New Coal Fired Power Plants

- New plants in Duisburg-Walsum, Hamburg-Moorburg, Boxberg und Wilhelmshaven, … (total of $10GW_{el}$) show hydrogen-induced stress corrosion cracking behaviour of new steel T24 (7 CrMo VTiB 10 10) used in steam generators.
- 

Quelle: www.panoramio.com/photo/22685844



Quelle: www.spiegel.de/spiegel/print/d-79175803.html

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New Energy for Germany (10/2011) www.bundesregierung.de (1/2)

Die Bundesregierung

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Wind Offshore: Strategie & Portfolio von EnBW

EnBW

- Errichtung von Windparks in Ostsee und Nordsee.
- Schrittweise Steigerung des Schwierigkeitsgrades (Wassertiefe, Küstenentfernung, Turbinengröße, Turbinenanzahl)

	EnBW Windpark Baltic 1	EnBW Windpark Baltic 2	EnBW Windpark Hohe See	EnBW Windpark Hohe Dreiht
Küstenentfernung:	16 km	32 km	85 km	92 km
Meerestiefe:	18 m	30 m	39 m	39 m
Anlagen-/Parkgröße:	2,3 / 48,3 MWel	3,6 / ~288 MWel	5-6 / ~4-500 Mwel	5-6 / ~4-700 MWel
	2010	2013	2014	2016

Quelle: EnBW November 2011

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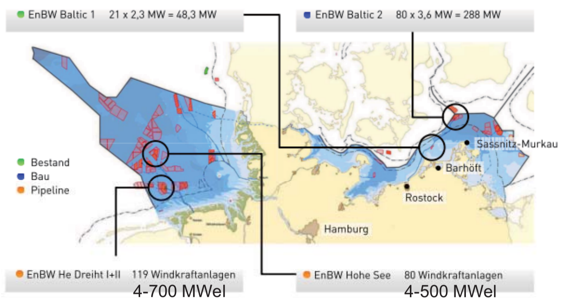
Wind Offshore: Strategy & Portfolio of EnBW

EnBW

- Erection of windparcs in North Sea and Baltic Sea.
- Stepwise increase of difficulties (Water depth – max 39m, distance from shore – max 92 km, turbine size – max 6.16 MW, turbine number - 119)
- Baden-Württemberg: from 0.1% today to 10% wind in 2020.



Quelle: EnBW November 2011

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Wind Offshore: Strategie & Portfolio von EnBW

- Frage: Wie kommt Strom von Produktionsorten zu Nutzern.

Folie von Stromnetzen => TEMME (Tel.)

BEISPIEL Baden-Württemberg:
Faktor 10 von 0.1 auf 10%: Linie entlang Schwarzwald in 2er Reihen
Oder zählen die GW in Nord- und Ostsee zu BW-Strom?

BEISPIEL: Kombination PV und Wind

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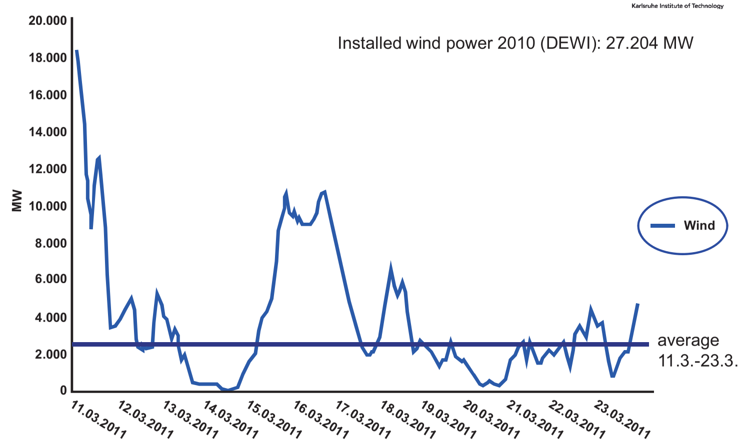
Electricity from Wind and PV in Germany



Quelle: Graphik nach Bundesnetzagentur 2011

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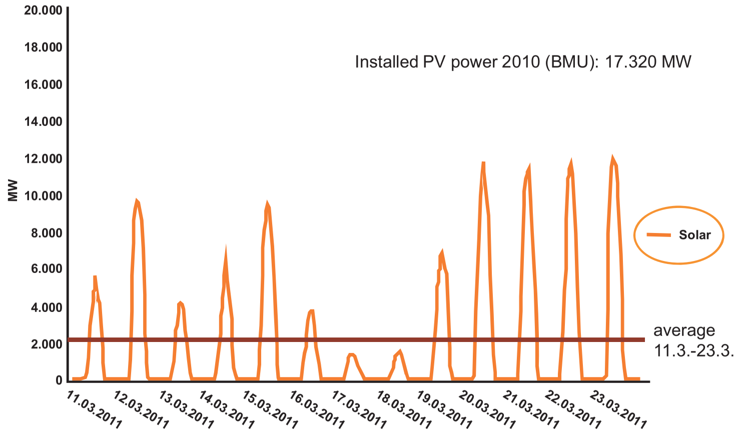
Electricity from Wind and PV in Germany



Quelle: Graphik nach Bundesnetzagentur 2011

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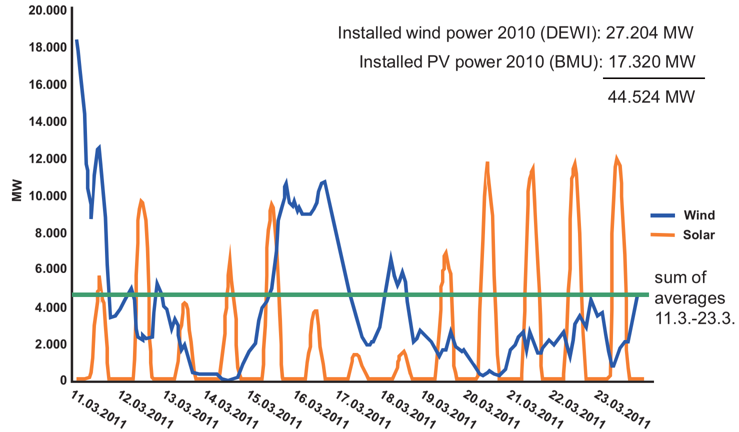
Electricity from Wind and PV in Germany



Quelle: Graphik nach Bundesnetzagentur 2011

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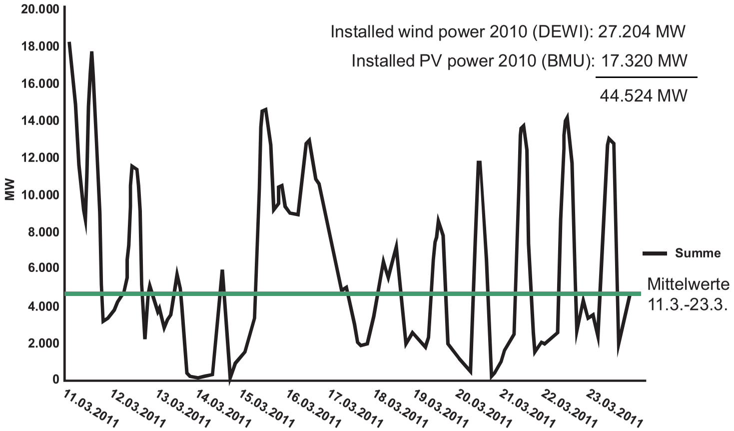
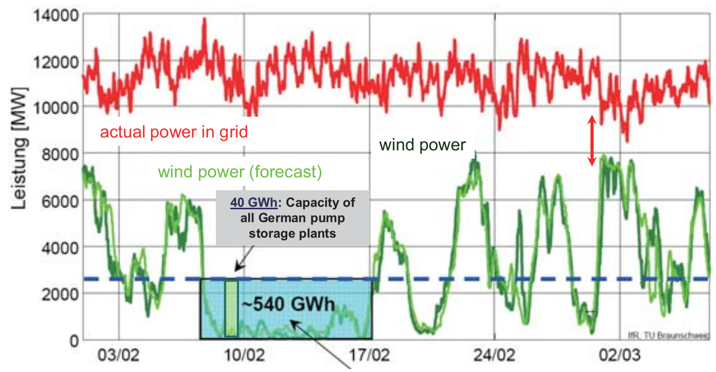
Electricity from Wind and PV in Germany



Quelle: Graphik nach Bundesnetzagentur 2011

Wind Energy and Required Storage Capacity

Example: Grid of Vattenfall Europe February 2008



Quelle: IfR, TU Braunschweig, ABB

Die Bundesregierung

New Energy for Germany (10/2011) www.bundesregierung.de (1/2)

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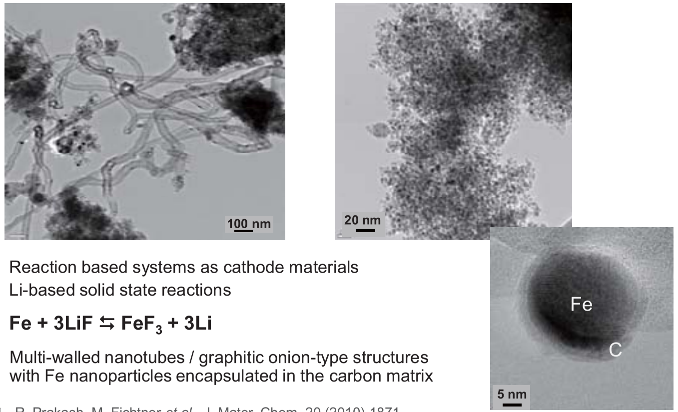
Quelle: http://hallo.news352.lu/edito-77382-rieseninvest-in-die-stromautobahn.html

Chances of an (intelligent) Electromobility?

Structure of C/Fe+LiF Nanocomposite



TEM /HR TEM

- Reaction based systems as cathode materials
- Li-based solid state reactions

Fe + 3LiF ⇆ FeF_3 + 3Li

- Multi-walled nanotubes / graphitic onion-type structures with Fe nanoparticles encapsulated in the carbon matrix

R. Prakash, M. Fichtner *et al.*, J. Mater. Chem. 20 (2010) 1871
R. Prakash, M. Fichtner *et al.*, J. Pow. Sources 196 (2011) 5936

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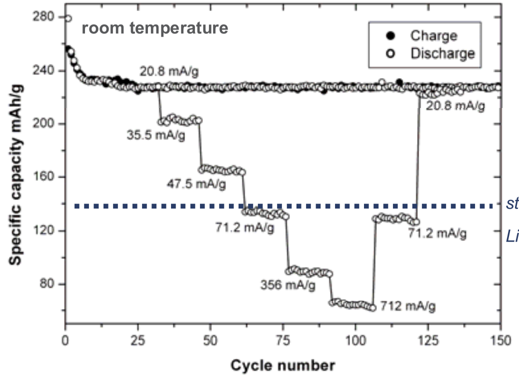
Conversion Material with High Cyclic Stability



R. Prakash, *et al.*, J. Power Sources 196 (2011)

30 6.12.2011 Joachim Knebel & Winfried Petry - ISMTR-4 - Oarai Japan Chief Science Officer Bereich 4 (CSO-4)

MeRegioMobil: System Integration User – Energy Sources – Storage Systems

- Intelligent Integration of Electromobility

Quelle: KIT-Zentrum Mobilitätssysteme

BOSCH DAIMLER EnBW Fraunhofer Institut System- und Innovationsforschung KIT OPEL SAP STADTWERKE KARLSRUHE

31 6.12.2011 Joachim Knebel & Winfried Petry - ISMTR-4 - Oarai Japan Chief Science Officer Bereich 4 (CSO-4)

Can Electric Cars Contribute to Electricity Storage?

- Comparison of storage capacities

Electric Car	Pumped-storage Hydropower Plant
Assumption: • 1 Mio cars (goal 2020 in D) as storage in grid • each 20 kWh battery capacity • 70% availability for storage • 50% average free capacity	Assumption: • All pump storage plants in Germany are considered as one unit • *Atdorf: plus 12.5 GWh*
 Quelle: www.think.no	 Quelle: Schluchseewerk AG
7.000 MWh capacity	**40.000 MWh capacity**

32 6.12.2011 Joachim Knebel & Winfried Petry - ISMTR-4 - Oarai Japan Chief Science Officer Bereich 4 (CSO-4)

Solar Electricity in Germany

- In 2009, about 45% of all PV plants world-wide were operated in Germany.
- In 2009, the share of electricity produced by PV in Germany was 1,9% of all electricity produced in Germany
- The majority of PV cells is produced in the far east regions.

Social Acceptance of Energy Technologies

Fears and Concerns:

- Nuclear: Safety, final repository
- Coal: Climate, limited resources
- Gas: Climate, limited resources
- Wind: Nature conservation, residents
- Geothermal: Risk of earthquakes
- Biomass: Food competition, mono cultures
- PV: Use of land, landscape protection
- Electricity grid: Nature conservation, electro smog

Personal Conclusions

- What is not questioned by public:
  - Increase of renewable energies,
  - Increase of energy efficiency.
- No consensus in public concerning:
  - How fast? How expensive?
  - Do we have to go all the way?
  - Which personal sacrifices?
- *Politics cannot overrule physics.*
- False promises and expectations hinder acceptability and success.
- Probably no ‚magic' energy solution, but a *concert* of small solutions.
- Investment in education and research creates know-how, creativity and innovative solutions.

KIT Chamber Orchestra

Contents

- Fukushima
- Energy policy and energy research in Germany (after Fukushima)
- Personal Conclusions
- Research Reactors in Germany

1st nuclear installation in Germany
Forschungsreaktor München

1[st] divergence: **31 October 1957**; **28 July 2000** taken out of service in order to take into operation the Neutron Research Source Heinz Maier-Leibnitz, 2 March 2004 1[st] divergence

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Forschungsreaktor Rossendorf

- 1[st] criticality **16 Dec. 1957**,
- heavy water moderator, 10 MW,
- taken out of service **27 Juni 1991**

38 Chief Science Officer Bereich 4 (CSO-4)

Today

- HFR, Institut Laue-Langevin, Grenoble, stake holders Germany, France, UK, 1[st] criticality 31 Aug. 1971, the world leading neutron source for fundamental science, Φ_{therm} = 13 x 10^{14} n/cm^{2}s
- BR2, Helmholtz Zentrum Berlin, 1[st] criticality 13 April 1991, Φ_{therm} = 1.2 x 10^{14} n/cm^{2}s
- Forschungsneutronenquelle Heinz Maier-Leibnitz (FRM II), Technische Universität München, 1[st] criticality 2 March 2004, leading multiple purpose research reactor, Φ_{therm} = 8 x 10^{14} n/cm^{2}s

and the European Spallation Source, Lund, Sweden at horizon 2019

⇒ more neutron instrumentation in Germany and Europe than ever before

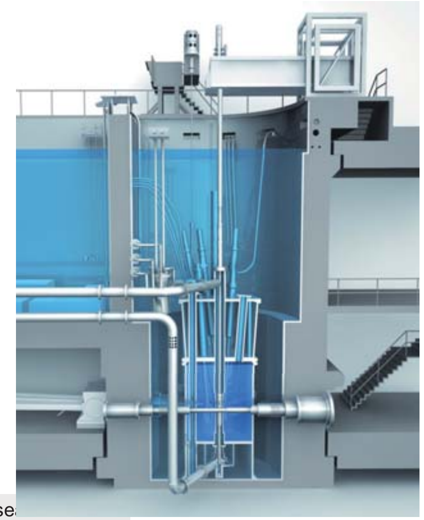
39 Chief Science Officer Bereich 4 (CSO-4)

Forschungsreaktor Heinz Maier-Leibnitz (FRM II), Multiple Purpose Reactor for the 21[st] Centrury

Forschungsneutronenquelle Heinz Maier-Leibnitz (FRM II)

- Thermal power 20 MW
- 1 compact core, 8,1 Kg highly enriched Uranium
- Vessel in swimming pool reactor 700 m³ H_2O, T ~ 35°C
- Heavy water moderator
- External neutron beams
- 30 instruments
- Several irradiation facilities
- Ca. 800 scientific visitors per year
- Access to beam time by scientific competition
- Factor 2 oversubscribed
- 60% of the users from Germany,
- 40% international users.

⇒ Most modern multiple purpose rese



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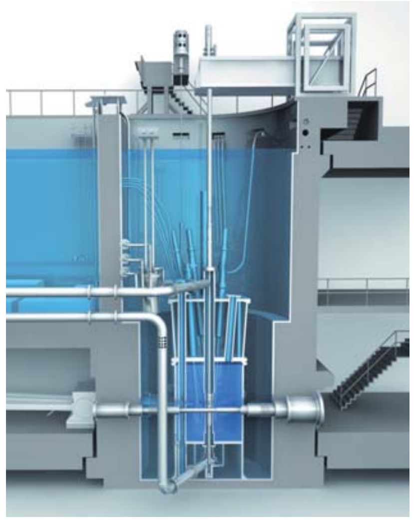
FRM II, a Safe Nuclear Facility

- 4 barriers to retain radioactivity
- Full protection against simultaneously arriving
  - earthquakes
  - air plane crash, also A380
  - flooding
- Due to small figure of thermal power no active cooling after shut down needed
- FRM II is prepared for beyond design events



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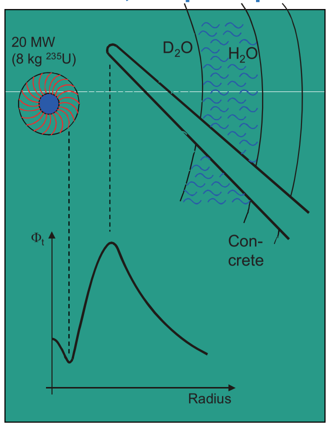
FRM II, the principle



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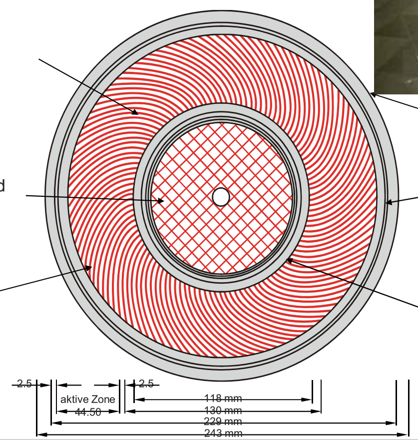
Compact Core



7.6 kg ^{235}Uranium
cycle length: 60 d

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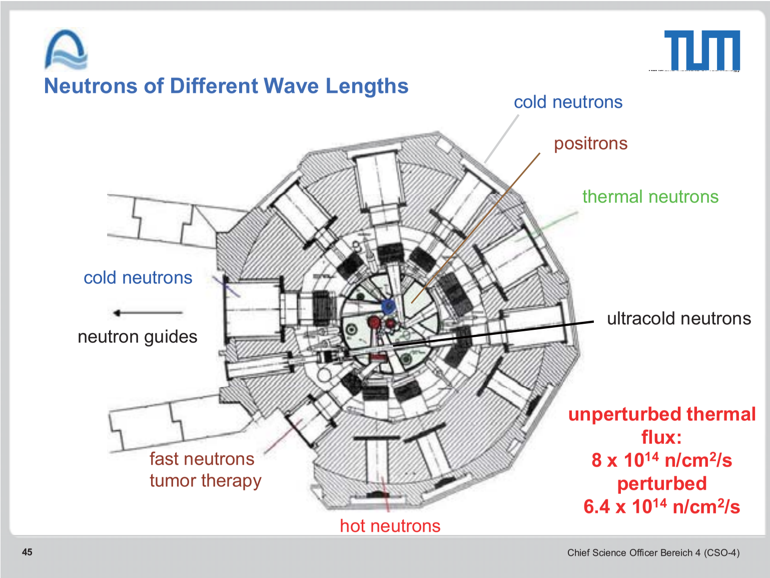
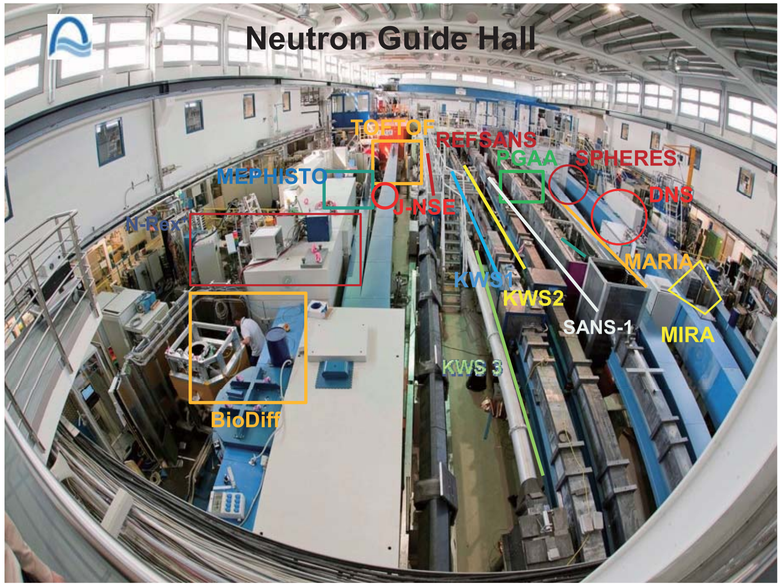


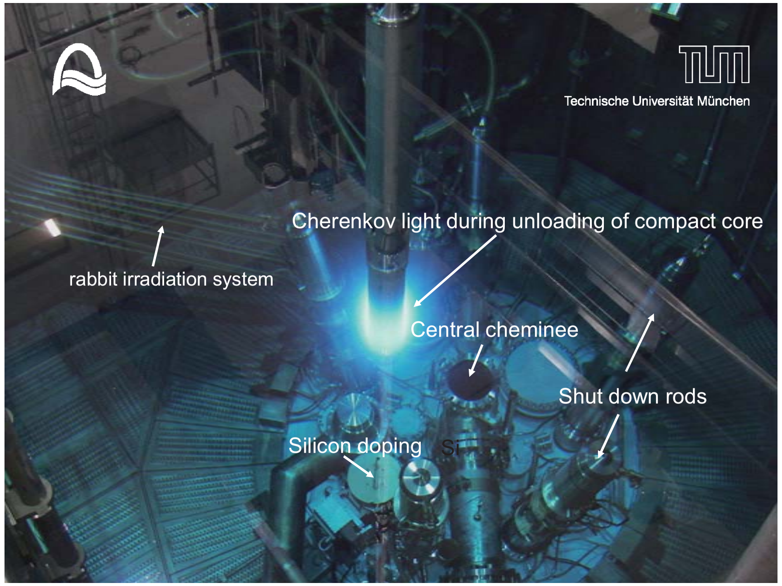


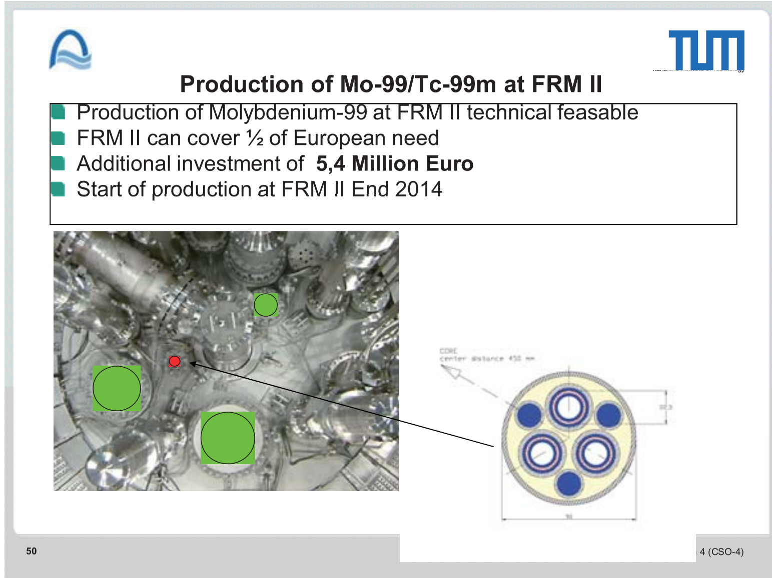
Irradiation devices operational

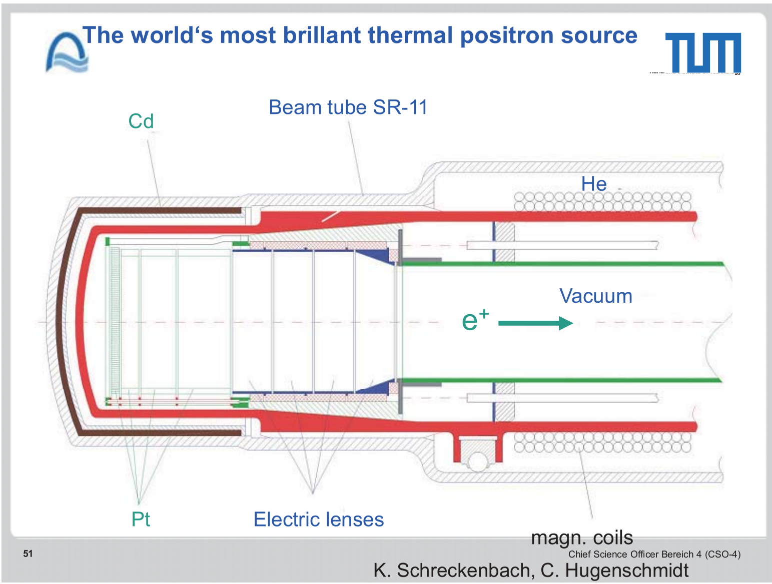
Facility	Sample Conveying	Φ_{th} ($cm^{-2} \cdot s^{-1}$)	Φ_{th}/Φ_f	Irradiation Period	Positions
Fishing line	manually	$1.2 \cdot 10^{13}$	$1.2 \cdot 10^{3}$	10 minutes … days	1
Standard Rabbit Irradiation System RPA	pneumatic (CO_2)	4.8×10^{12} 7.3×10^{13}	6.66×10^{4} 1.3×10^{3}	30 seconds … hours	2 × 3
Capsule Irradiation Facility KBA	hydraulic (pool water)	$7.7 \cdot 10^{13}$ $1.3 \cdot 10^{14}$	$7.7 \cdot 10^{2}$ $3.3 \cdot 10^{2}$	minutes … days	2 x 5
Silicon Doping Installation SDA	mechanically automatized	2×10^{13}	?	10 minutes … 1 day	1
control rod position	isotope production	6×10^{14} fast	-	52 days	1

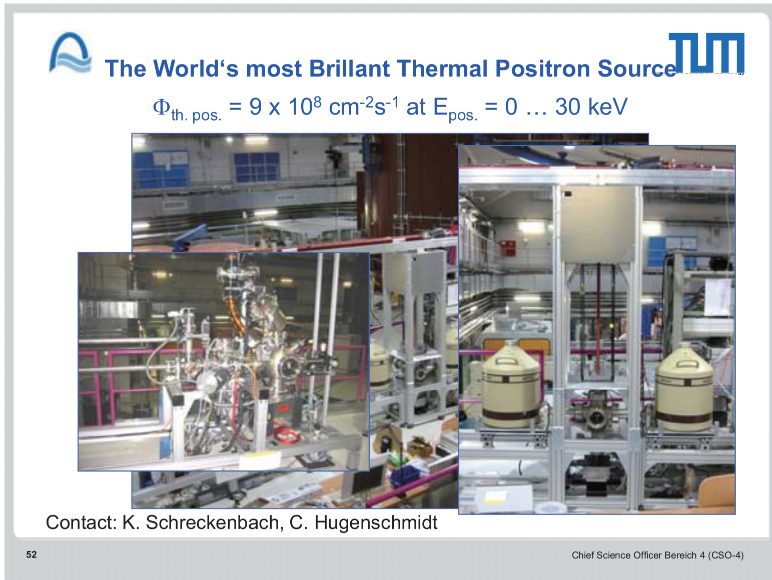
Contact: H. Gerstenberg, X. Li, I. Neuhaus

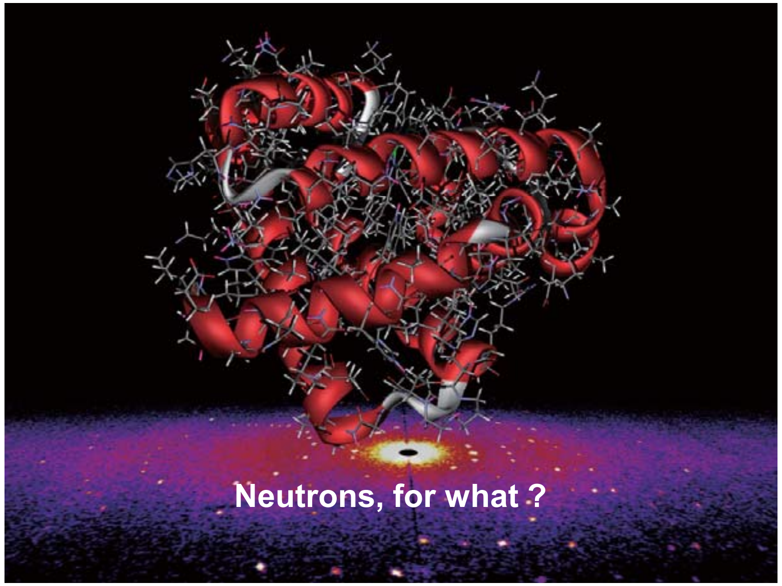
48 Chief Science Officer Bereich 4 (CSO-4)

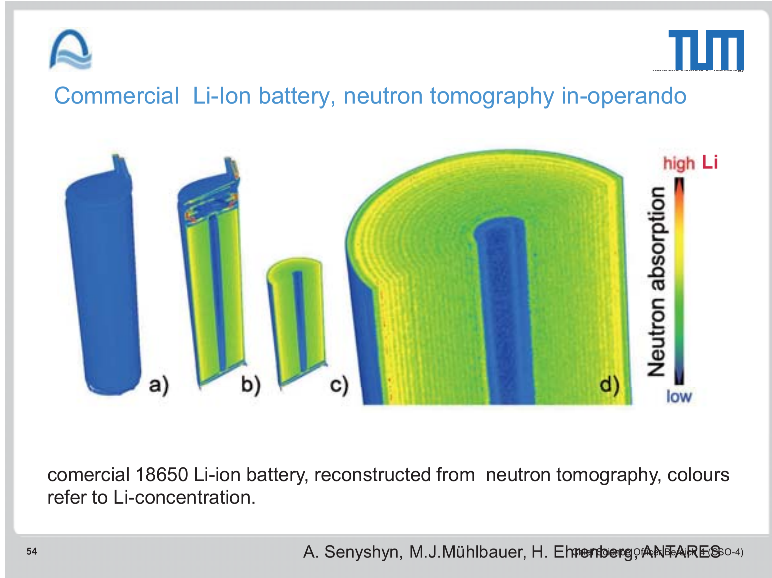

Technische Universität München
Cherenkov light during unloading of compact core
rabbit irradiation system
Central cheminee
Shut down rods
Silicon doping

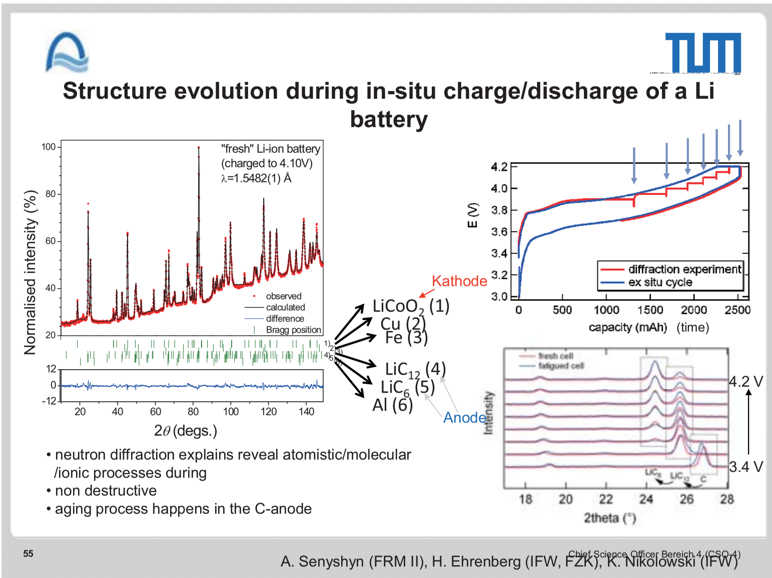

Production of Mo-99/Tc-99m at FRM II
Production of Molybdenium-99 at FRM II technical feasable
FRM II can cover ½ of European need
Additional investment of 5,4 Million Euro
Start of production at FRM II End 2014
50
4 (CSO-4)

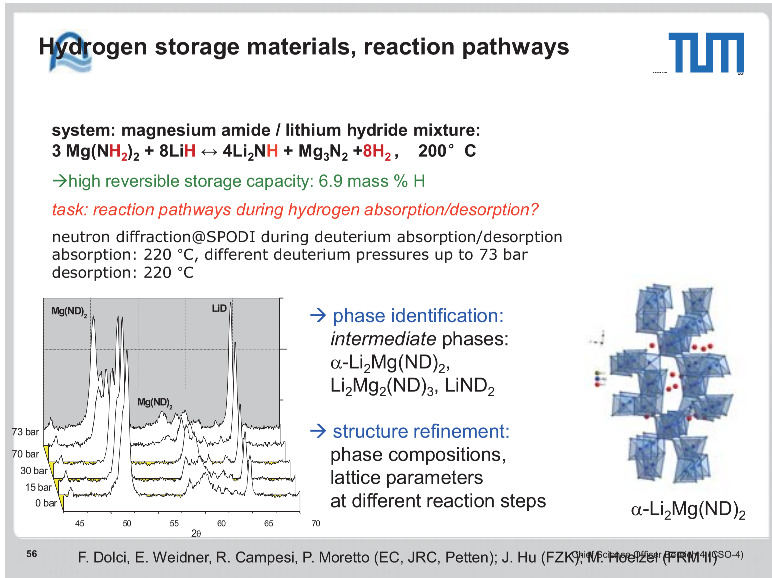

The world's most brillant thermal positron source
Cd
Beam tube SR-11
He
Vacuum
e+
Pt
Electric lenses
magn. coils
51
Chief Science Officer Bereich 4 (CSO-4)
K. Schreckenbach, C. Hugenschmidt


The World's most Brillant Thermal Positron Source
Φth. pos. = 9 x 10^8 cm^-2s^-1 at Epos. = 0 … 30 keV
Contact: K. Schreckenbach, C. Hugenschmidt
52
Chief Science Officer Bereich 4 (CSO-4)


Neutrons, for what ?


Commercial Li-Ion battery, neutron tomography in-operando
a)
b)
c)
d)
Neutron absorption
high Li
low
comercial 18650 Li-ion battery, reconstructed from neutron tomography, colours refer to Li-concentration.
54
A. Senyshyn, M.J.Mühlbauer, H. Ehrenberg, ANTARES

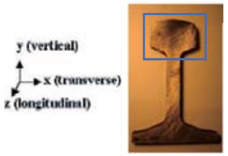

Structure evolution during in-situ charge/discharge of a Li battery
"fresh" Li-ion battery (charged to 4.10V)
λ=1.5482(1) Å
observed
calculated
difference
Bragg position
Normalised intensity (%)
2θ (degs.)
Kathode
LiCoO2 (1)
Cu (2)
Fe (3)
LiC12 (4)
LiC6 (5)
Al (6)
Anode
E (V)
diffraction experiment
ex situ cycle
capacity (mAh) (time)
fresh cell
fatigued cell
Intensity
4.2 V
3.4 V
2theta (°)
• neutron diffraction explains reveal atomistic/molecular /ionic processes during
• non destructive
• aging process happens in the C-anode
55
A. Senyshyn (FRM II), H. Ehrenberg (IFW, FZK), K. Nikolowski (IFW)


Hydrogen storage materials, reaction pathways
system: magnesium amide / lithium hydride mixture:
3 Mg(NH2)2 + 8LiH ↔ 4Li2NH + Mg3N2 +8H2 , 200° C
→high reversible storage capacity: 6.9 mass % H
task: reaction pathways during hydrogen absorption/desorption?
neutron diffraction@SPODI during deuterium absorption/desorption
absorption: 220 °C, different deuterium pressures up to 73 bar
desorption: 220 °C
Mg(ND)2
LiD
73 bar
70 bar
30 bar
15 bar
0 bar
2θ
→ phase identification:
intermediate phases:
α-Li2Mg(ND)2,
Li2Mg2(ND)3, LiND2
→ structure refinement:
phase compositions,
lattice parameters
at different reaction steps
α-Li2Mg(ND)2
56
F. Dolci, E. Weidner, R. Campesi, P. Moretto (EC, JRC, Petten); J. Hu (FZK); M. Hoelzel (FRM II)

Strain mapping

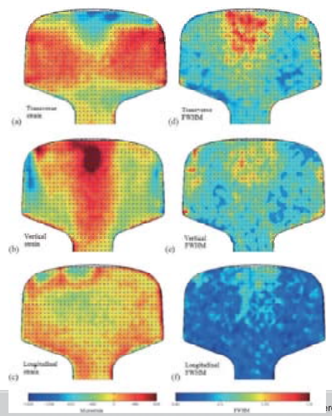
Railway rail

Accumulation of residual stresses in rails during service can contribute to crack initiation and fracture and may result in serious accidents.



Strain in transv., vert. and long. direction in [mm/m], x- in mm

FWHM from (211) interference line show plastic deformation

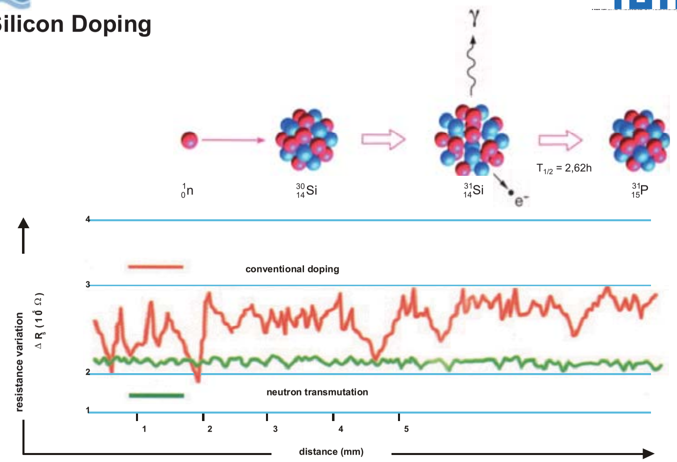


57 T.-S. Jun *et al*, J. Strain Anal. **44** (2009) pp 563

Silicon

58 Chief Science Officer Bereich 4 (CSO-4)

Silicon Doping



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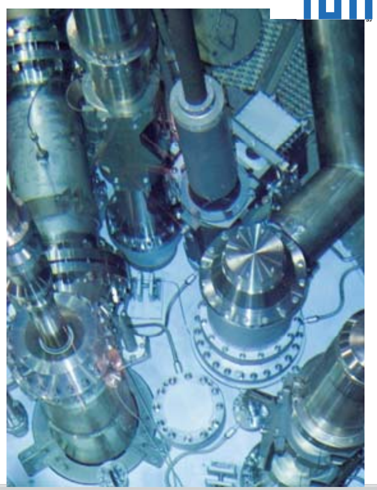
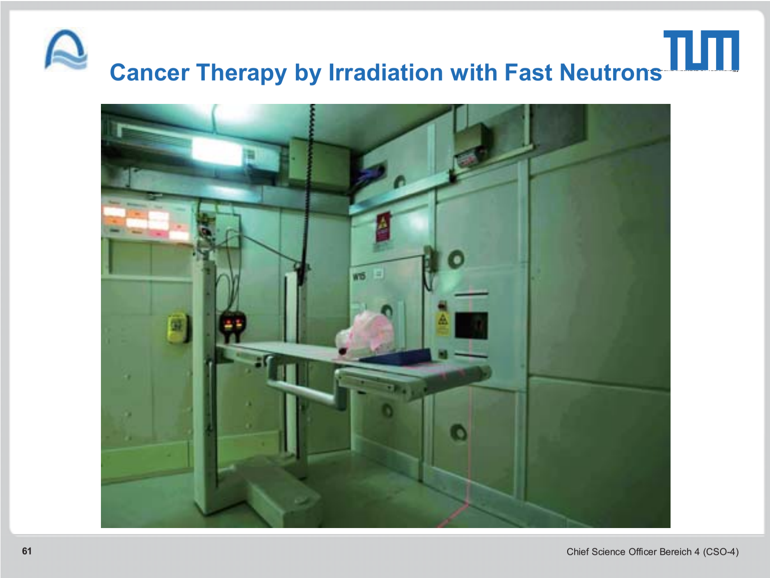
Irradiation of Si-single crystal

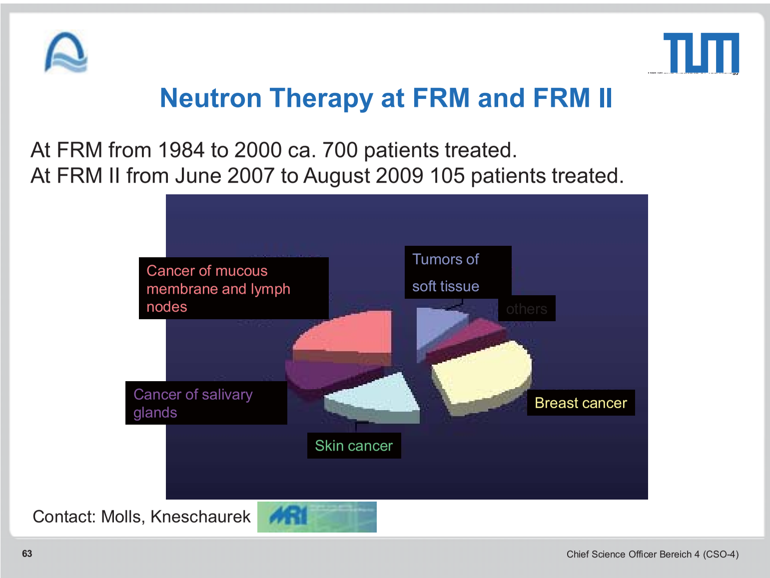
FRM II produces 15 t doped Si / a, corresponds to 10-15% of world market



60 Chief Science Officer Bereich 4 (CSO-4)


Cancer Therapy by Irradiation with Fast Neutrons
61
Chief Science Officer Bereich 4 (CSO-4)


Neutron therapy, metastases at the breast wall
27/06/2008
23/06/2008
12/08/2008
27/06/2008
62
Molls, Kneschaurek, Loeper, Klilnikum rechts der Isar, FRM II


Neutron Therapy at FRM and FRM II
At FRM from 1984 to 2000 ca. 700 patients treated.
At FRM II from June 2007 to August 2009 105 patients treated.
Cancer of mucous membrane and lymph nodes
Tumors of soft tissue
others
Cancer of salivary glands
Breast cancer
Skin cancer
Contact: Molls, Kneschaurek
63
Chief Science Officer Bereich 4 (CSO-4)


You are always welcome to
FRMII, Munich
and Germany
Munich
City of Garching
Campus Garching
FRM II
Subway


Karlsruhe Institute of Technology
Thanks a lot for listening.
www.kit.edu/eps/cso-4
joachim.knebel@kit.edu, winfried.petry@frm2.tum.de & Colleagues
Dr.-Ing. Joachim U. Knebel, Chief Science Officer Bereich 4
KIT – University of the State of Baden-Wuerttemberg and
National Research Center of the Helmholtz Association
www.kit.edu

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2.2 Nuclear Research Policy in the Evolving Political Context of Switzerland

J.-M. Cavedon, J. Dreier, P. Hardegger

Paul Scherrer Institute
Villigen PSI
5232 Villigen
Switzerland
Tel : +41 56 310 27 42
E-Mail: jean-marc.cavedon@psi.ch

The five Swiss nuclear power plants cover 39% of its electricity needs. Together with the 55% hydro share, they bring an almost CO_2-free electricity supply to the country since decades. In the aftermath of the Fukushima nuclear event, the Swiss Federal Council has rapidly frozen the ongoing process of general licenses for replacing or building new nuclear plants. It has proposed this summer to ban any new nuclear license, while letting the existing plants operate until the end of their life cycle, assumed to last 50 years in average. The Swiss nuclear plants were built between 1969 and 1984.
A new energy strategy will be drafted by spring 2012, new measures and instruments will be submitted to the National Council and the Council of States in 2013, with final decision by public vote (referendum) in 2014. Ambitious targets for energy efficiency are being set: reduction of energy and electricity consumption of 13% by 2020, and of 28% (energy) and 35% (electricity) by 2035. No technology shall be forbidden and the Federal Council shall report regularly on the progress of energy technologies, especially on new nuclear technologies and their safety and waste reduction capabilities.
The Paul Scherrer Institute acknowledges this political will. In agreement with the vast majority of the federal councilors and members of the parliament, it considers that nuclear energy research must be pursued. Nuclear power plants will be operated for decades and need safe operation with state-of-the-art safety measures and devices, as well as highly educated personnel.
Nuclear research at PSI is already devoted to safety, waste management and new reactor concepts. This portfolio will be further pursued, with increased attention given to the educational potential of new reactors concepts, where increased safety and waste reduction are central topics.
Highlights of today’s research activities at PSI comprise materials degradation, passive safety integral tests, safety margins for fuel and cladding temperature, hydrogen risk, aerosol retention, dynamic probabilistic safety assessment and reactivity of fast neutron cores. All these topics contribute to the constant progression in reactor safety.
The nuclear waste repository plans for Switzerland have converged on Opalinus clay as a host rock for deep underground repository. PSI’s geochemical expertise is instrumental in assessing the safety of the repository concept and has contributed to the selection of six repository sites that are now formally approved for the second selection phase by all stakeholders.

PAUL SCHERRER INSTITUT

Paul Scherrer Institut

Jean-Marc Cavedon, Department Head Nuclear Energy and Safety

Nuclear Energy Research Policy in the evolving political context of Switzerland

Presentation to ISMTR-4, Oarai, Japan, December 6th 2011

PSI, 14 February 2012

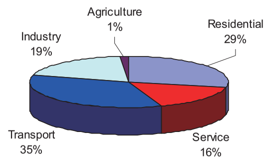
Outline

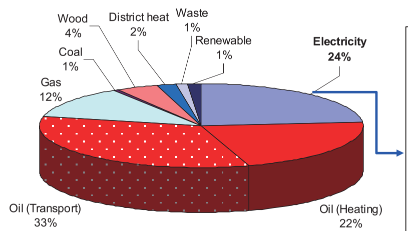
- Energy and electricity supply in Switzerland
- Status of political decisions
- The mission of PSI
- The research portfolio : PSI's highlights in safety research

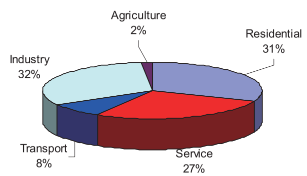
Overview of Swiss energy system

Energy Economy (2009)

- Energy expenditure: CHF 27.1 Billion (5.1% of GDP)
- Energy import: CHF 8.7 Billion (4% of import expenditure)
- **Energy import dependency:** 80%
- 243 TWh energy, 58 TWh electricity



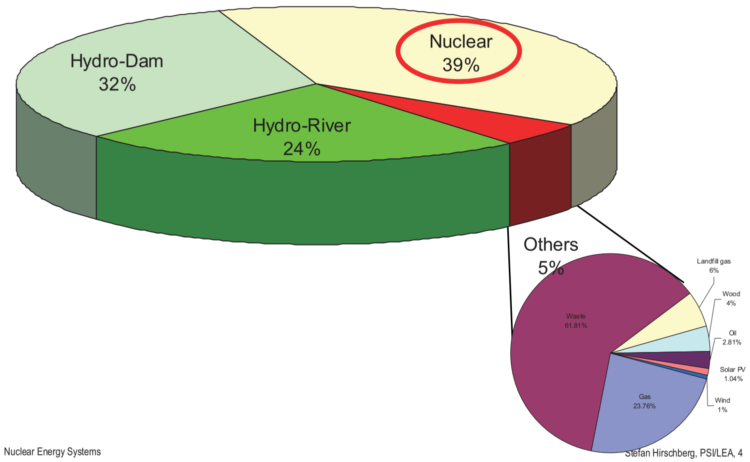




Nuclear Energy Systems

Stefan Hirschberg, PSI/LEA, 3

Swiss electricity system



Nuclear Energy Systems

Stefan Hirschberg, PSI/LEA, 4

PAUL SCHERRER INSTITUT PSI

Switzerland's decision to abandon nuclear energy, a very political debate

"Responding in early summer to the problems at the Fukushima nuclear power plant, Switzerland's executive branch, the Federal Council, decided to review our energy perspective 2035 – our base for energy policy decisions…."

•No replacement of the five current nuclear power plants at the end of their life cycle (the last one in 2034 assuming an average 50-year life span)
•New Energy policy: efficiency, renewable energy, combined heat and power, gas fired power plant.
•2012 Federal Council decides about the measures and instruments which will be necessary
•2013 debate in Parliament
•2014 final decision by a public vote.

"Regardless of the outcome of the debate, the vast majority of federal councillors and members of parliament agree on one point: **nuclear research must continue in Switzerland**. Swiss researchers involved in nuclear projects can be considered both economically and materially reliable.
Public funding for nuclear research is unlikely to be curtailed and we are confident that Swiss researchers are committed to future research in this field. Their **work with international partners is certain to continue for a long time to come".**

Address of Dr. Pascal Previdoli, Deputy Director of the Swiss Federal Office for Energy, to the Generation IV International Forum, Lucerne, October 6th 2011

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Status of nuclear phaseout at the Parliament level

30.09.2011 The Ständerat (Council of States) supports the principle of nuclear phase-out and modifies the message as such

- **Exclude that nuclear technology is totally forbidden**
- **Demand that the Federal Council (=government) follows the path forward in nuclear technology**

Text to be approved by the National Council and the Council of States in December 2011, without modification :

The Government should propose to change the law as such :

- ***No general permit for nuclear power plants shall be issued***
- ***No technology shall be forbidden***
- ***Education, training and research in all energy technologies, in Switzerland and in the framework of international collaborations shall be further supported***
- ***The Federal Council shall report regularly on the progress of techniques and in particular on the ones of nuclear technology***
- ***The Federal Council shall at the same time express its position towards safety, waste disposal, and the effects on environment, economy and climate policy***

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The first steps in a new energy strategy

Swiss Federal Council 1.Dec. 2011 concretizes its energy strategy for 2050

•Energetic efficiency :
– Buildings, Electric appliances, Industry and services, Mobility
– **Targets for energy and electricity savings until 2020 (~-13%) and 2035 (~-28-35%)**
•Renewable energies
•Fossil plants
•Electric grids
•Research : Pilot and demonstration plants
•Report in spring 2012
•Project for consultation by end of 2012

N.B. Nuclear electricity is today 40% of electricity and 10% of total energy

PAUL SCHERRER INSTITUT PSI

Concept for energy research in Switzerland 2012-2016

Draft from the Federal Commission for research in energy CORE, 28.11.2011

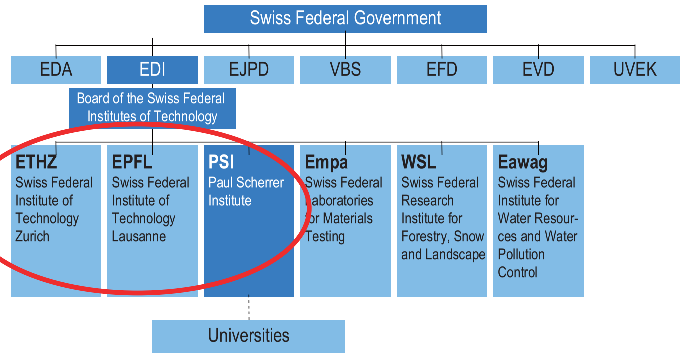
Target for 2050-2100

Primary energy divided by 2 or 3
1 tonne CO_2/p/y
No problematic health or environmental impacts
Widely reduced mass flows and closure of materials flows in the energy sector

Guidelines for 2012-2016

•Housing and jobs of the future
•Mobility of the future
•Energy systems of the future (excerpts re nuclear)
– **Continue regulatory nuclear safety research**
– **Continue research on waste management and waste reduction**
– **Continue development on safer and more efficient technologies**
– **Keep capacity of continuous evaluation of technology progress (e.g. Gen IV) in terms of safety and waste management**
•Processes of the future

PSI, a federal research institute with academic sisters



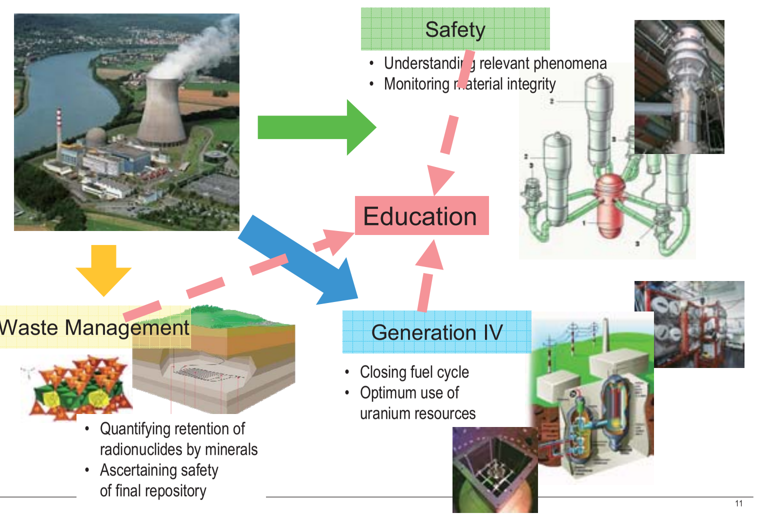
Mission of the Paul Scherrer Institute

- To play a leading role on an international level in
  - physics of condensed matter and materials sciences
  - structural biology
  - radiochemistry, radiopharmacy and proton radiation therapy
  - particle physics and accelerator technology

  by using large-scale facilities (SLS, SINQ, SμS, particle beams)
- To be a UserLab for external science community
- **Energy research towards an efficient, environmentally friendly and reliable energy supply (primarily using complex facilities)**

"The nuclear phase-out does not mean the end of nuclear research" says Joël Mesot, Director of the Paul Scherrer Institute.
"Nuclear power plants will be operated until 2035 in Switzerland. Their safety must be assured until then. Our duty is to educate the personnel that will ensure safe operation"
Aargauer Zeitung, Switzerland, 26.11.11

Department Nuclear Energy and Safety

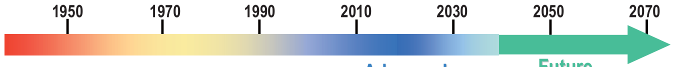


Swiss Nuclear Education

- ❑ Joint EPFL-ETHZ Master in Nuclear Engineering
  - o First – and currently still only – joint degree of the two universities
- ❑ Active PSI role, as national R&D centre for nuclear fission
  - o Unique expertise, facilities of NES/PSI (part of ETH-Domain)
  - o Senior scientists, lecturers in several courses
  - o Direct PSI-NES links to Chairs at EPFL & ETHZ, crucial factor
- ❑ Industry actively involved, incl. lecturers in several courses
- ❑ Evidence of valuable synergies between Master and PhD students
- ❑ EPFL-ETHZ-PSI contributions also to "continuing education"
- ❑ 3rd year ongoing
- ❑ 4th year started on Sept. 16th 2011 with 17 students

R. Chawla, H.-M. Prasser, J.-M. Cavedon — Swiss Nuclear Education, June 2011

Towards a sustainable nuclear energy



First successes Gen I

Thorough exploration of experimental and first commercial systems

Current reactors Gen II

PWR, BWR, Candu, VVER

Ageing and being shut down

Advanced reactors Gen III / III+

ABWR, EPR, ESBWR...

Can be ordered today

Future systems Gen IV

SFR (demo) VHTR,GFR, SCWR, LFR, MSR

Demo in planning

Source: GIF, 2003

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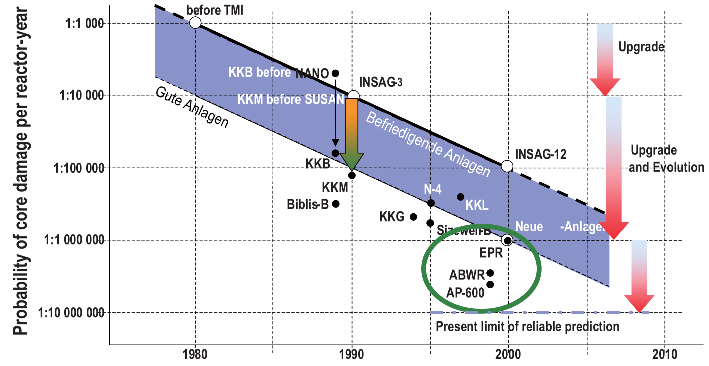
Constant progression in safety from Gen II to Gen III



Quelle : IAEA, 1995

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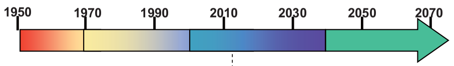
Nuclear science and technology at PSI



Neutrons	thermal	fast
Coolant	water	gas liquid metals supercrit. H_2O
Materials & Fuels	low T	high T
Waste	radio-active	all toxicities
Energy Systems	nuclear and non-nuclear	intra-nuclear and others

Education

Reactor Operators, Masters, PhDs, Post-Docs

Continuous education engineers

Large nuclear facilities

Hotlab, Zero power reactor, irradiation facilities, cooling loops,

Large analytical facilities

X-rays, neutrons in-pile and out-of-pile, charged particles

15

Department Nuclear Energy and Safety

Zero power Research reactor PROTEUS: experiments for efficient use of nuclear fuel

Large-scale facility PANDA: heat removal by natural circulation

Materials degradation

Nuclear Materials degradation questions ...

from individual pellet ...

to

... reactor pressure vessel (RPV) and piping

Research with Focus on Safety and Performance – Link to Hotlab (AHL)

Issue	Safety relevance	PSI contribution	Added value
Fuel pellet • fission gas production • swelling	Pressure increase (r integrity)	Fuel particle the SLS **Partnership with operators and vendors**	Determination of microscopic strains as function of the pellet radius and burn-up
Fuel cladding • Corrosion • Hydrogen uptake	Embrittlement by fo hydrides (risk of loosing integrity)	Hydrides distribution rod cross sections, as function of stresses **Partnership with vendors**	Better assessment of the influence of stress and hydrides on the cladding degradation
RPV & piping • Irradiation induced embrittlement • Fracture mechanics • Combined loading and degradation effects	Loss of ductility and ability to with- s s (risk of loosing integrity of the most important components due to brittle failure)	• Analysis of micro-structural changes FS at SLS) • Probabilistic fracture mechanics analyses (PTS, earthquake, water hammer) **Partnership with the Swiss regulator**	• Step towards non-destructive embrittle-ment determination • Better understanding of early clustering of minor alloying elements leading to embrittlement • Probabilities for crack growth and component failure

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PANDA Integral Containment System Tests for GIII+ Certification Support

Concept demo for certification, with a vendor for USNRC certification

*Simplified Boiling Water reactor (**SBWR**)*

- **Passive Safety Concept demonstration** at large scale for US NRC certification

- Data base for containment code validation, in particular for US NRC certification of GE's TRACG code, code validation contributions

*European and Economic SBWR (**ESBWR**)*

- Integral test series supporting the USNRC certification including light gas effects

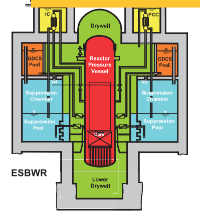
- Further contributions to US NRC certification of the GE code TRACG

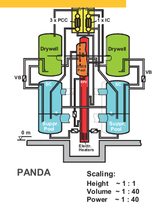
- Large-scale start-up stability test series

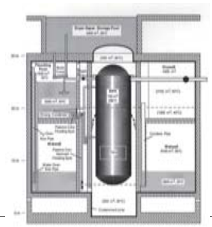
***SWR1000** (KERENA)*

- Large-scale integral containment system concept demonstration including light gases

- Simulation tool validation for SWR1000







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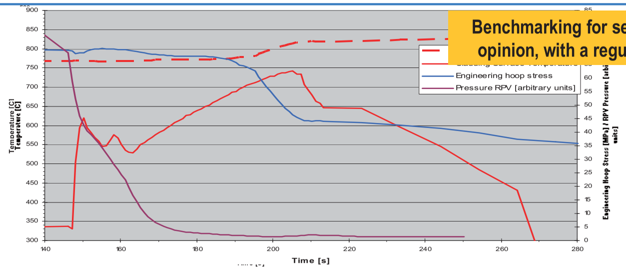
Safety margins of fuel temperature in EPR Finland



- Loss Of Coolant Accident performance of Finnish EPR (PWR) evaluated with the STARS systems codes (in cooperation with the Finnish regulator STUK)
- Regulatory LOCA limits:
  - $T_{cladding}$ < 1204 C (2200 F)
  - Oxidation < 17%
- Analysis showed: $T_{cladding}$ < 750 C (very large margin)
- In addition: $T_{burst} - T_{cladding}$ > 50 C -> Cladding will not burst open -> zero release!
- **EPR design shows a very benign behaviour during LBLOCA**

Added value:
- Integrated analysis (System–Analysis / Thermo-Mechanics shows large margins
- Independent second opinion provided for Finnish regulator (STUK)

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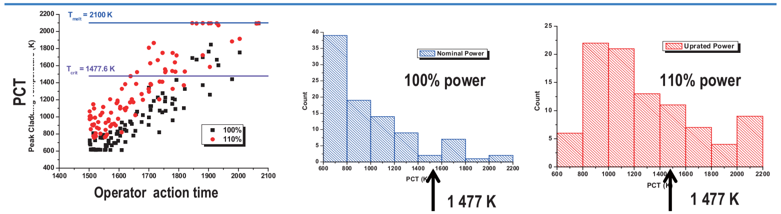
Safety Margins on fuel cladding temperature



Plant safety assessments requires a holistic approach:
- Many scenarios for a given initiating event (e.g. Medium-sized Loss Of Coolant Accident event tree) -> need to be analyzed
- Impact of plant modification can be assessed, e.g. 110% power uprate
- Change of Safety Margin is identified as change of exceedance probability of acceptance limit:
  - Allows for straight-forward aggregation of margins against different acceptance limits (safety barriers)
- For MLOCA initiating event:
  - Conditional probability increases from 0.123 to 0.286 (Δ= 0.163) due to power uprate of 10 %
  - for the time window of relevance for operator action
  - Approach: Best-Estimate Plus Uncertainty (BEPU) -> Dynamic Event Trees

PSI-driven international research in an OECD/NEA framework

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Hydrogen risk in Light Water Reactors

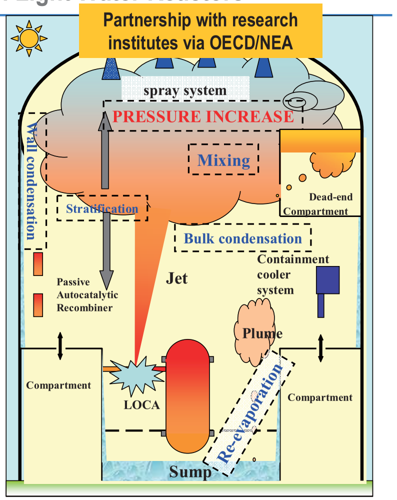
Severe Accident scenarios: hydrogen released by Metal-Water reaction, hydrolysis, Molten Core – Concrete Interaction MCCI

Hydrogen risk is a safety issue - detonation and explosions

- **Hydrogen stratification or mixing**
  - ➢OECD/SETH PANDA project (2003-2006)
- **Hydrogen stratification break-up**
  - negatively buoyant plumes or jets,
  - safety systems (spray, cooler, recombiner)
  - ➢OECD/SETH-2 PANDA project (2007-2010)
- **Facility Scaling effects and realistic scenarios**
  - ➢EU-ROSATOM: ERCOSAM-SAMARA projects (2010-2013)
- Flow structures (floor, vertical wall),
- Combination of two safety systems (e.g. spray-cooler, recombiner-recombiner, etc.)
- Containment compartments (opening hatches)
  - ➢OECD/HYMERES project proposal (2012-2015)

The PANDA data in the various projects are used for Lumped Parameter and Computer Fluid Dynamics code validations

Partnership with research institutes via OECD/NEA



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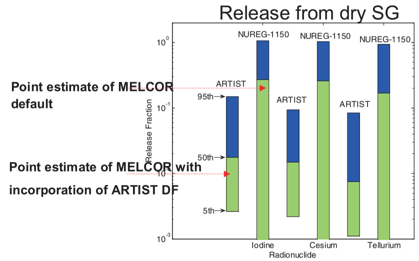
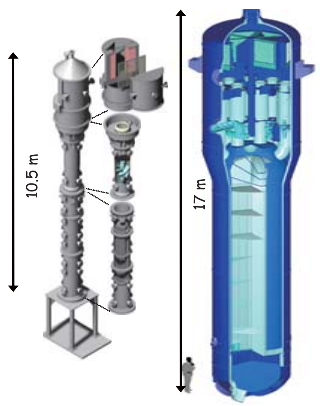
Aerosol retention in steam generators of PWRs (ARTIST)

- ARTIST data used in **system codes**
- Significant **aerosol retention** in the SG secondary side
  - Water in the bundle efficiently suppresses the release
- **Release of radioactivity smaller** than previously estimated
- **More accurate** estimate of the release
- Data for different accident scenarios
- Work for Design Basis Accident on-going

PSI-centered partnership with regulators, vendors, research institutes, operators





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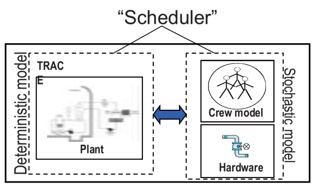
Dynamic Probabilistic Safety Assessment (DynPSA)

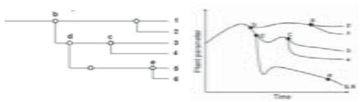
Research initiated on own funds

Technical issue:
- **Current PSAs compartmentalize**
  - **analysis of NPP physics** in accidents (transient analysis)
  - probabilistic models (accident sequence modeling, event tree analysis, estimation of probabilities)

Safety relevance:
- **separated assessment of uncertainties** in the two models, difficult to understand joint effect and interactions
- assumptions needed to model **accident dynamics**, effect of **stochastic uncertainties**, and **context for operator decision-making**

PSI:
- **integrated transient analysis and modeling of operator actions**
- **re-evaluation of success criteria**
- **characterization of decision contexts** for Human Reliability Analysis





Added value:
- **new capabilities for uncertainty analysis, realism of accident analyses,** assessment of **contexts for operator decisions (HRA)**
- **more sensitive estimation of plant risk, safety margin evaluation**

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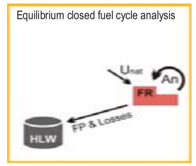
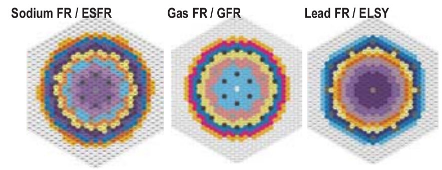
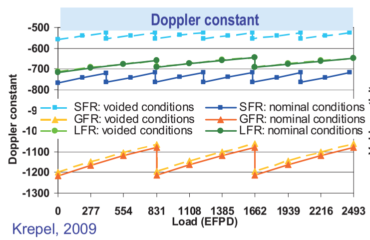
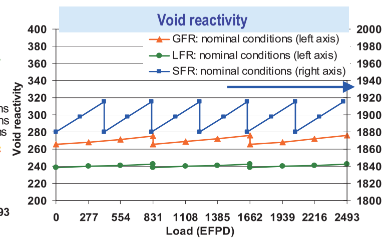
FAST: Safety related parameters of Gen-IV Fast Reactors

Own funding in international contractual exchange (GIF)









Krepel, 2009

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Scope of Generation IV systems

reliable, affordable, sober, clean, harmless and peaceful

	Acronym	Spectrum	Fuel Cycle
Sodium-Cooled Fast	**SFR**	fast	closed
Lead Alloy-Cooled Fast	**LFR**	fast	closed
Gas-Cooled Fast	**GFR**	fast	closed
Very High Temperature	**VHTR**	thermal	open
Supercritical Water-Cooled	**SCWR**	thermal & fast	both
Molten Salt	**MSR**	epithermal	closed

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Nuclear waste calls for underground disposal

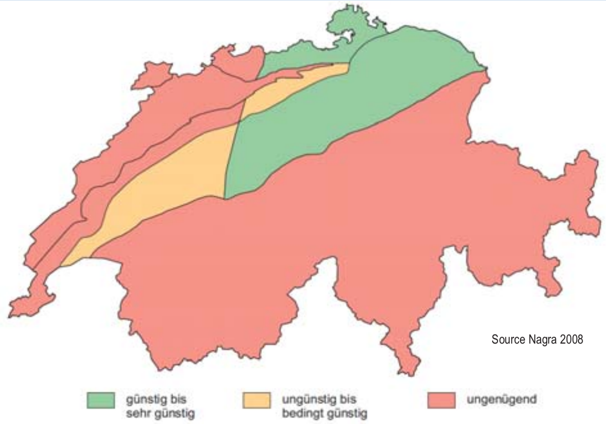


Fig. 3.5-2: Darstellung der gesamtheitlichen 3-stufigen Bewertung der geologisch-tektonischen Grossräume für das HAA-Lager.

ISMTR4_Oarai 2011

Opalinus clay is the best host rock in CH

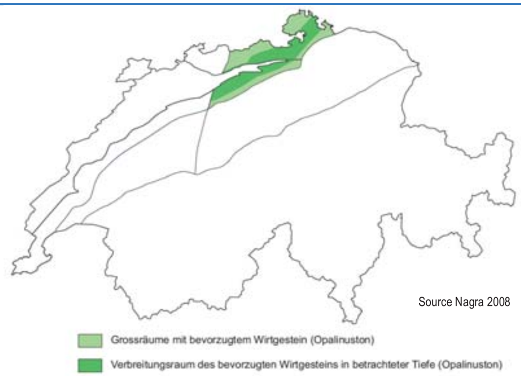
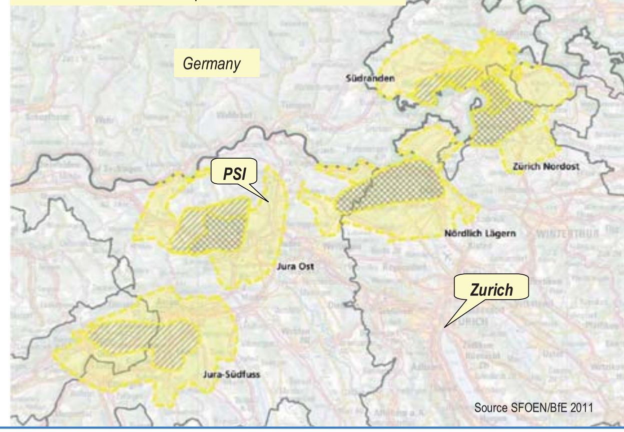


Fig. 4.7-2: Das für das HAA-Lager bevorzugte Wirtgestein Opalinuston und sein Verbreitungsraum in betrachteter Tiefenlage.
Das für das HAA-Lager bevorzugte Wirtgestein – der Opalinuston mit seinen Rahmengesteinen – wird in den flachliegenden und wenig bis mässig zergliederten Grossräumen (östlicher Tafeljura, östliche Subjurassische Zone) in der Tiefenlage 400 bis 900 m u.T. weiter verfolgt.

ISMTR4_Oarai 2011

Phase 2 of site selection starts now

Repository sites accepted by the Swiss Government for the second selection phase on Nov. 30th 2011



ISMTR4_Oarai 2011

Waste isolation : a scientific challenge

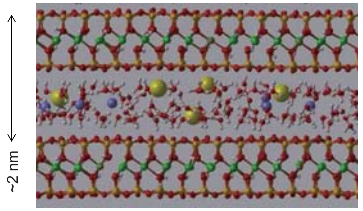
Convince/prove that waste isolation is technically possible over 10^6 years

- Questions are asked by Nagra, the waste cooperative, within performance assessments
- Clay is an excellent retention medium
  - low diffusion
  - high sorption
  - reliable resealing
- Cement and glass are very good matrices to incorporate radionuclides
  - high retention
  - low leaching rates
- Surface and interface chemistry should prove the above points even at atomic level
  - MD codes and XAS experiments, together with all conventional tools
  - Multiscale studies: transfer the conviction/proof from the atomic level to the field scale

ISMTR4_Oarai 2011

SwissFEL: a tool for nuclear waste research

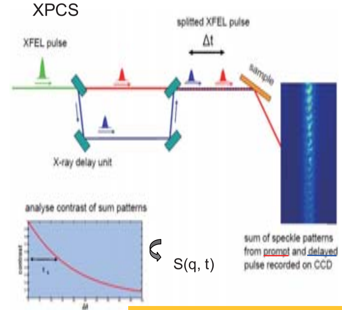
- Dynamics of mass transport in clay minerals



Molecular Dynamics simulation
S. Churakov PSI/NES / LES, 2011

t: 0.1 - 1000 ps, L: 0.1 - 100 nm

- Method: split-pulse XPCS



Partnership within PSI and with a basic research institute

- Partners: LES SwissFEL DESY LCLS

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Concluding remarks

Management
Nuclear Energy and Safety Department

- Nuclear phase-out is a new situation in Switzerland, not yet politically stabilized
- First signs for alternative strategy show a substantial challenge for energy saving and growth in renewable energy production
- Gas will be a growing part of the energy and electricity supply
- Climate policy remains unchanged
- Economic constraints remain unchanged
- How will be filled the missing gap, if any ? Gas, import ? nuclear ??
- Nuclear power will be operational for at least the next 25 years (declining)
- Nuclear research will be pursued for safety of ageing Gen II reactors
- …and for evaluating Gen III and of Gen IV reactors, both in their safety and waste reduction capabilities
- International cooperation is essential in energy research, and now especially in nuclear research !

ISMTR4_Oarai 2011

Thank you for your attention

PSI, 14. Februar 2012

2.3 Advancing Nuclear Technology and Research: The Advanced Test Reactor National Scientific User Facility

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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 INL also has several hot cells and other laboratories in which irradiated material can be examined to study material radiation effects. In 2007 the US Department of Energy (DOE) designated the ATR as a National Scientific User Facility (NSUF) to facilitate greater access to the ATR and the associated INL laboratories for material testing research. The mission of the ATR NSUF is to provide access to world-class facilities, thereby facilitating the advancement of nuclear science and technology. Cost free access to the ATR, INL post irradiation examination facilities, and partner facilities is granted based on technical merit to U.S. university-led experiment teams conducting non-proprietary research. Proposals are selected via independent technical peer review and relevance to United States Department of Energy. To increase overall research capability, ATR NSUF seeks to form strategic partnerships with university facilities that add significant nuclear research capability to the ATR NSUF and are accessible to all ATR NSUF users.

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

1. INTRODUCTION

In 2007, the Advanced Test Reactor (ATR), located at Idaho National Laboratory (INL), was designated by the U. S. Department of Energy (DOE) as a National Scientific User Facility (NSUF). This designation made test space within the ATR and post-irradiation examination (PIE) equipment at INL available for use by approved researchers via a proposal and peer review process. The goal of the ATR NSUF is to provide those researchers with the best ideas access to the most advanced test capability, regardless of the proposer's physical location.

Goals of the ATR NSUF are to define the cutting edge of nuclear technology research in high temperature and radiation environments, contribute to improved industry performance of current and future light water reactors, and stimulate cooperative research between user groups conducting basic and applied research. As part of meeting each of these three goals, the ATR NSUF has developed a broad educational program aimed at increasing the number of researchers knowledgeable about reactor experimentation, post irradiation examination techniques, and material radiation effect fundamentals. The educational program also includes a wide variety of internship opportunities, faculty/student research team projects, partnerships with other DOE laboratory and university experimental facilities, annual Users Week, which includes several seminars on ATR and partner facility research, collaborative experiment projects, graduate research fellowships, and opportunities for postdoctoral researchers and visiting scientists.

Since 2007, the ATR NSUF has expanded its reactor test space, obtained access to additional PIE equipment, taken steps to ensure the most advanced post irradiation analysis possible, and initiated an educational program and digital learning library to help potential users better understand the critical issues in reactor technology and how a test reactor facility could be used to address this critical research. This article describes these expanded capabilities and services so that researchers can take full advantage of this national resource.

To increase overall research capability, ATR NSUF seeks to form strategic partnerships with university facilities that add significant nuclear research capability to the ATR NSUF and are accessible to all ATR NSUF users. Recognizing that INL may not have all the desired PIE equipment, or that some equipment may become oversubscribed, the ATR NSUF established a Partnership Program. This program invited universities to nominate their capability to become part of a broader user facility. Several universities and one national laboratory have been added to the ATR NSUF with capabilities that include reactor-testing space, PIE equipment, and ion beam irradiation facilities.

2. FACILITY CAPABILITY

2.1 Advanced Test Reactor

The ATR was designed to optimize fuel and material testing for the Navy's nuclear propulsion program. It began operation in 1967, and has operated continuously since then, averaging about 250 operating days per year. Irradiation of material and fuel in the ATR can simulate many years of prototypical operation in a few months or years of testing. This capability is valuable for testing materials and fuels in

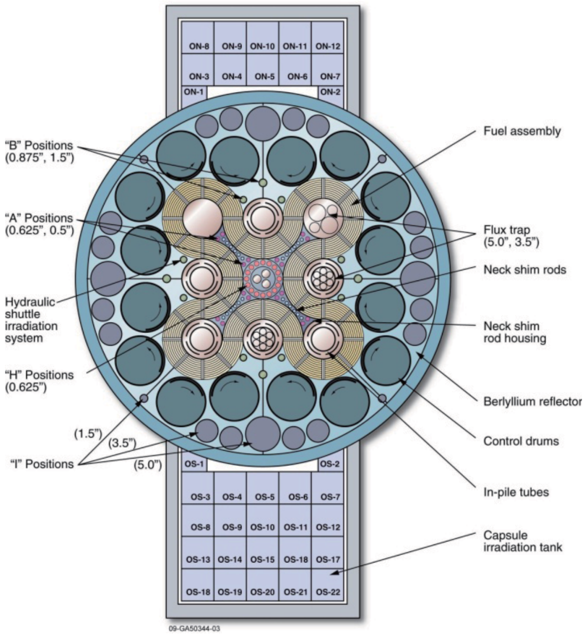

Figure 1 ATR ore Cross Section.

support of light water reactors (LWRs) and more advanced reactor designs. Unlike U.S. commercial LWRs, the ATR has no established lifetime or shutdown date. All core internal components are removed and replaced every eight to ten years during a core internals changeout outage, which typically takes about six months.

The ATR is a pressurized, light-water moderated and cooled, beryllium-reflected, enriched uranium fueled reactor with a maximum operating power of 250 MWth. The ATR core cross section, shown in Figure 1, consists of 40 curved aluminum plate fuel elements configured in a serpentine arrangement around a three-by-three array of large irradiation locations in the core or flux traps, where the peak thermal flux can reach 1.0 × 1015 n/cm2-sec, and peak fast flux (E>1.0 MeV) 5x1014 n/cm2-sec. This core configuration creates five main reactor power lobes (regions) that can be operated at different powers during the same operating cycle. Along with the nine flux traps, there are 68 irradiation test positions ranging in diameter from 1.27 to 12.7 cm and are 122 cm long, and the irradiation tanks outside the core reflector tank have 34 low-flux irradiation positions.

There are three primary experiment configurations in the ATR - static capsule, instrumented lead, and pressurized water loop. Experiments must remain in the ATR for the entire duration of the operating cycle (average length of 49 days), except for experiments performed in the Hydraulic Shuttle Irradiation System (HSIS). The HSIS enables small volume, short duration, irradiations to be performed in the ATR, and can include up to 14 small shuttle capsules in a single shuttle operation.

The ATR building also houses the ATR Critical (ATRC) facility, which is a full-size replica of the ATR, but operates at low power (5 kW maximum). It is used to evaluate an experiment's potential impact on the ATR core, by measuring experiment control rod worths, reactivities, thermal and fast neutron distributions, gamma heat generation rates, and void/temperature reactivity coefficients before inserting an experiment into the ATR.

2.2 Post-Irradiation Examination Capabilities

Post-irradiation examination (PIE) capabilities are available to ATR NSUF users at numerous facilities at the INL, including the Hot Fuel Examination Facility (HFEF), Analytical Laboratory (AL), Electron Microscopy Laboratory (EML), and Fuels and Applied Science Building (FASB). These facilities house equipment and processes used for nondestructive examination, sample preparation, chemical, isotope, and radiological analysis, mechanical and thermal property examination, and microstructure property analysis.

2.2.1 Nondestructive Examinations

Nondestructive examination activities are available at the HFEF. Capabilities include neutron radiography using 250 kW TRIGA reactor, with two beam tubes and two separate radiography stations, precision gamma, dimensional inspections using a continuous contact profilometer, element/capsule bow and length examinations to measure distortion (bow) and length of fuel elements, visual exams, eddy current examinations to measures material defects, and high precision specific gravity measurements using pycnometer and immersion scales.

2.2.2 Sample Preparation

Samples preparation capabilities include solid metallography, which consists of sectioning and cutting, mounting into metallographic bases, and grinding and polishing processes and equipment, and gas sampling using laser puncture and gas collection processes.

2.2.3 Chemical, Isotopic, and Radiological Analysis

Chemical, isotopic, and radiological analysis of irradiated fuel and material meeting National Institute of Standards and Technology traceability standards capabilities include inductively coupled plasma mass spectrometry with dynamic reaction cell, inductively coupled plasma atomic emission spectroscopy, atomic absorption analysis, thermal ionization mass spectrometry, gas mass analysis, isotope mass separator, gross and isotopic radiological analysis, gross alpha/beta analysis, alpha, beta, and gamma spectroscopy analysis.

2.2.4 Mechanical Property Examination

Mechanical property examination activities are available for high radiation samples in the EML, HFEF Main Cell, and the lower-dose, contact-handled FASB. Capabilities include metallography, microhardness testing, tensile testing, and shear punch testing.

2.2.5 Thermal Property Examinations

Thermal property examination instruments and processes are available at the INL Materials and Fuels Complex. Capabilities include: thermal diffusivity (laser flash method and scanning diffusivity analysis), differential scanning calorimitry, and high temperature furnace for accident testing of high temperature gas-cooled reactor fuel.

2.2.6 Microstructure Property Analysis

State-of-the-art microstructure property analysis instruments capable of micro and nanoscale characterization are available at INL. Capabilities include scanning transmission electron microscope (STEM) with energy dispersive x-ray spectrometer, scanning electron microscope (SEM) with energy dispersive and wavelength dispersive x-ray spectrometers and electron back scatter diffraction detector, field emission gun (FEG) SEM, dual beam focused ion beam (FIB) that enables site specific sectioning of materials for 3D analysis or high resolution, TEM characterization, shielded electron microprobe to analyze elements from Be through Cm with full matrix correction, including fission gases on samples, and x-ray diffractometer to perform microscale phase identification, small-sample powder diffraction, and texture determination.

2.2.7 Instruments in the Center for Advanced Energy Studies (CAES)

State-of-the-art microstructure property analysis instruments capable of micro and nanoscale characterization are available at INL. Capabilities include scanning transmission electron microscope (STEM) with energy dispersive x-ray spectrometer (See Figure 2), scanning electron microscope (SEM)

Figure 2 CAES Scanning Transmission Electron Microscope.

with energy dispersive and wavelength dispersive x-ray spectrometers and electron back scatter diffraction detector, field emission gun (FEG) SEM, dual beam focused ion beam (FIB) that enables site specific sectioning of materials for 3D analysis or high resolution, TEM characterization, shielded electron microprobe to analyze elements from Be through Cm with full matrix correction, including fission gases on samples, and x-ray diffractometer to perform microscale phase identification, small-sample powder diffraction, and texture determination.

3. PARTNER FACILITIES

The ATR NSUF and its partners facilities represent a prototype laboratory for the future. This unique model is best described as a distributed partnership with each facility bringing exceptional capabilities to the relationship including reactors, beamlines, state-of-the-art instruments, hot cells, and most importantly expert mentors.

Potential partners interested in having their facility or capabilities added to the NSUF are able to nominate themselves through a partner proposal that is submitted to the ATR NSUF. If selected the partner is then added to the suite of capabilities of the user facility. The following is a current list of partners:

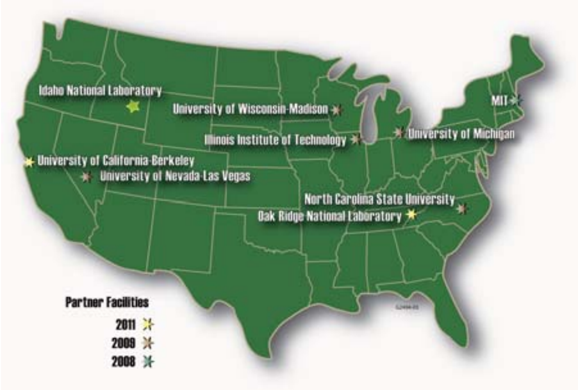

Figure 3 ATR NSUF Partners.

3.1.1 Massachusetts Institute of Technology (MIT) Reactor

The MIT reactor is a 5 MWth tank-type research reactor. It has three positions available for in-core fuel and materials experiments for water loops at pressurized water reactor/boiling water reactor conditions, high-temperature gas reactor environments at temperatures up to 1400°C and fuel tests at LWR temperatures have been operated and custom conditions can also be provided. Fast and thermal neutron fluxes are up to $1x10^{14}$ and $5x10^{14}$ n/cm^2–s, respectively.

3.1.2 North Carolina State University (NCSU) PULSTAR Reactor

The PULSTAR reactor is a 1 MWth research reactor, fueled by uranium dioxide pellets in zircaloy cladding. The fuel provides response characteristics that are similar to commercial LWRs, which allows teaching experiments to measure moderator temperature, power reactivity coefficients, and doppler feedback. In 2007, the PULSTAR reactor produced the most intense low-energy positron beam with the highest positron rate of any comparable facility worldwide.

3.1.3 Nuclear Services Laboratories

Nuclear Services laboratories at North Carolina State University (NC-State) offer neutron activation analysis, radiography, imaging, and positron spectrometry capabilities.

3.1.4 Irradiated Materials Complex (IMC) at University of Michigan

The UM IMC Complex houses laboratories and hot cells for conducting high-temperature mechanical property, corrosion and stress corrosion cracking experiments on neutron irradiated materials in an aqueous environment and for characterizing fracture surfaces after failure.

3.1.5 Harry Reid Center Radiochemistry Laboratories

The Radiochemistry Laboratories at University of Nevada, Las Vegas (UNLV) offer metallographic microscopy, x-ray powder diffraction, Rietveld analysis, SEM and STEM, electron probe microanalysis, and x-ray fluorescence spectrometry.

3.1.6 Characterization Laboratory for Irradiated Materials

The Characterization Laboratory for Irradiated Materials at the University of Wisconsin–Madison (UW-M) SEM and STEM on neutron-irradiated materials.

3.1.7 Tandem Accelerator Ion Beam

A 1.7 MV terminal voltage tandem ion accelerator at UW-M features dual ion sources for producing negative ions with a sputtering source or using a radio frequency plasma source. The analysis beamline is capable of elastic recoil detection and nuclear reaction analysis.

3.1.8 Michigan Ion Beam Laboratory

The 1.7 MV Tandetron accelerator in the Michigan Ion Beam Laboratory at the University of Michigan (U-M) offers controlled temperature proton irradiation capabilities with energies up to 3.4 MeV as well as heavy ion irradiation.

3.1.9 Illinois Institute of Technology (IIT) Beamline

The MRCAT beamline at Argonne National Laboratory's Advanced Photon Source (APS) offers

Figure 4 Illinois Institute of Technology MRCAT Beamline at APS

synchrotron radiation experiment capabilities, including x-ray diffraction, x-ray absorption, x-ray fluorescence and 5 μm spot size fluorescence microscopy.

3.1.10 University of California at Berkeley (UCB)

At the UCB Nuclear Engineering laboratory, nanoindenter capabilities are available for testing on low radioactive samples.

3.1.11 High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboartory (ORNL)

The HFIR provides a high flux (up to 5x1015 n/cm2-s thermal) material irradiation test capabilities are similar to those available at the ATR.

For a more details on ATR NSUF partner facilities and for specific capabilities of each facility go to http://atrnsuf.inl.gov [1].

4. RESEARCH PROPOSAL OPTIONS

Researchers can gain access to the ATR NSUF facilities described above through several proposal options. These have evolved over time to meet researcher requests and provide the maximum possible flexibility. All proposal submittals are completed through the web site at http://atrnsuf.inl.gov/[1]. All proposals received against open calls and Rapid Turnaround Experiments (RTE) are subject to a peer-review process before selection. An accredited U.S. university or college must lead research proposals for irradiation/post-irradiation experiments. Collaborations with other national laboratories, federal agencies, non-U.S. universities, and industry are encouraged. Any U.S.-based entities, including universities, national laboratories, and industry can propose research that would use the MRCAT beamline at the APS or would be conducted as an RTE

4.1.1 Open Calls

The annual open call for reactor irradiation or major PIE proposals is a continuously open rolling call with project selections twice a year, in the fall and in the spring. This gives researchers the flexibility of writing proposals at their leisure and allows ATR NSUF to make two sets of awards each year. Proposals for these calls focus on irradiation/post irradiation examination of materials and fuels and on post irradiation examination of previously irradiated materials or fuels from the ATR NSUF Sample Library. These calls also offer researchers the option to submit proposals for synchrotron radiation experiments through the ATR NSUF partnership with IIT.

4.1.2 Rapid-Turnaround Experiments (RTEs)

An experiment is considered an RTE if it can be completed in two months or less, such as PIE of previously irradiated fuels or materials, ion beam irradiation, and neutron scattering experiments. The call for RTEs is always open, allowing proposals to be submitted at any time. RTE proposals are reviewed within a month of submittal and awarded throughout the year based on ranking and the availability of funds.

4.1.3 New User Experiment

In response to requests from university faculty members, ATR NSUF developed a New-User Experiment to provide an opportunity for university researchers to experience the intricacies of designing and conducting an in-reactor test. The ATR NSUF Director selects the materials to be irradiated and each university researcher involved in the project can work with INL staff to design an experiment that meets the data objectives. To participate, researchers submit a letter of interest through the web site.

5. EDUCATION PROGRAMS

The Objective of the ATR NSUF educational programs help establish a cadre of nuclear energy researchers, facilitating the advancement of nuclear science and technology through reactor-based testing. It optimizes the value of these programs by developing strategic partnerships with universities and helps inform the academic user community of nuclear energy issues and tools available to address research questions. ATR NSUF education programs are used to maximize access and serve as a resource to an informed, engaged, and equipped academic user community.

ATR NSUF uses focused internships, fellowships, and faculty/student exchanges to encourage faculty and student access to cutting-edge and one-of-a-kind tools for conducting reactor-base research in nuclear science and technology, fuels, and materials. This enables researchers to gain access to key mentors, world-class facilities, and equipment. NSUF also created the ATR NSUF User's Week where research forums and specialized courses are presented. This educational tool has been instrumental in developing new text on irradiation test planning and execution. A major emphasis of all education programs is to allow for maximum interaction and access to the critical components of the nation's experimental nuclear research infrastructure.

5.1.1 Internships

Internships are the direct mechanism by which undergraduate and graduate students can be introduced to mentors. Each year, approximately 23 interns are exposed to ATR NSUF research and gain experience with tools in reactor-based nuclear science and technology. Interns typically spend 10 to 12 weeks at the INL in the summer. Graduate students may use their intern experience to conduct thesis or dissertation research, in a more focused experience than the undergraduate internship, that can last for up to one year. Internships are also used to support the increased impact of the ATR NSUF on facility operations.

5.1.2 Fellowships

Post-doctoral fellowships give recent doctoral graduates an opportunity for a short (up to three years) duration appointment in areas that align with current or future ATR NSUF research.

5.1.3 Visiting Scientists

The ATR NSUF education program has two programs for visiting scientists and students. The Faculty and Student Research Team (FSRT) program awards faculty-led team contracts to partner with an INL mentor and work on building capability needed in the user facility. In addition, teams gain an understanding of INL, build technical knowledge, and establish relationships with INL researchers. The ATR NSUF also uses an INL program called the Faculty and Staff Exchange program, in which participants are sent to universities or other research facilities and university faculty can visit INL. Researchers are encouraged to spend time at a university/INL to teach, perform research, collaborate, and be involved in campus/laboratory life.

5.1.4 Users Week

Annually, the ATR NSUF hosts a User's Week to provide a venue to inform the nuclear science and technology community of current issues and the tools and facilities available through the use of the ATR NSUF to address these issues. Users Week is comprised of a research forum that discusses current nuclear technology research being conducted in the NSUF. Sessions are held to familiarize participants with the ATR NSUF research facilities and capabilities. Discussions are held to facilitate potential industry-laboratory-university collaborations. User Week offers extended courses on fuels and materials and how to plan and execute irradiation experiments. Up to 50 travel scholarships are available to faculty and student participants.

5.1.5 Short Courses/Workshops

Portions of courses from Users Week have been

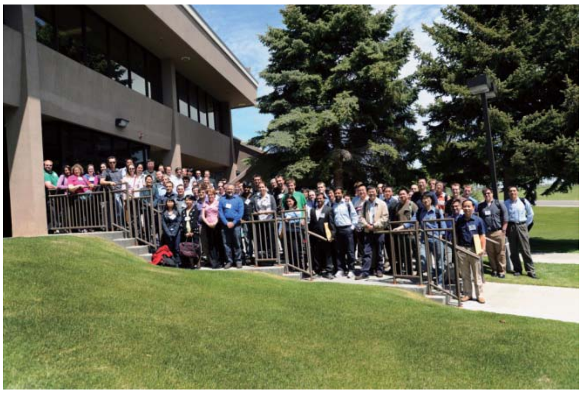

Figure 5 2011 Users Week participants

available as short courses at universities, technical society meetings, or technical meetings. As examples, short courses are adapted from the Experimenters' Course or the Fuels and Materials course.

6. CONCLUSIONS

The ATR NSUF and its partner facilities represent a prototype laboratory for the future. This unique model is best described as a distributed partnership with each facility bringing exceptional capabilities to the relationship including reactors, beamlines, state-of-the-art instruments, hot cells, and most importantly expert mentors. Together these capabilities and people create a nation-wide infrastructure that allows the best ideas to be proven using the most advanced capabilities. Through ATR NSUF , university researchers and their collaborators are building on current knowledge to better understand the complex behavior of materials and fuels in the radiation environment of a nuclear reactor.

Since ATR NSUF established the partnership program, 7 universities have offered their facility's capabilities, greatly expanding the kinds of research that can be offered. The avenues opened through these partnerships facilitate cooperative research across the country, matching people with capabilities and students with mentors.

Detailed information about current experiments, collaborations, and partner facilities can be found in the 2009 and 2010 ATR NSUF Annual Report [2,3].

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3. Status, Future Plans

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3.1 RSG-GAS: Current and Strategic Plan of Future Activities

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ABSTRACT
The Multi Purpose Reactor G.A. Siwabessy, so-called RSG-GAS, is the third research reactor built in Indonesia and operated by National Nuclear Energy of Indonesia (BATAN). This MTR type reactor achieved its first criticality in July 1987. Since then it has been operated about 77.300 hours. It uses LEU fuel in the form of U_3O_8-Al at the beginning, but then converted to silicide fuel, U_3Si_2-Al by a density of 2.96 g/cm^3 preserving its nominal power of 30 MW and average flux of 2.5×10^{14} n/cm^2 sec. In the last two years the reactor is operated more than 3.500 hours per year. Several equipments have been refurbished due to ageing. The reactor is operated mainly for serving radioisotopes production, neutron activation analysis (NAA), neutron beam experiments and general irradiation for research and development activities. The future activities of the RSG-GAS reactor will be prioritized first on increasing the reliability and availability of the reactor operation to assure the fulfillment of the demand of the isotope production, along with the improvement of neutron beam instruments and NAA facilities to increase its utilization. In this context, the collaboration in the utilization of RSG-GAS for many application with the countries in the region is also expected. The human resource development for the new recruited RSG-GAS staff is also a challenge and become a priority.

Keywords: research reactor, MTR, silicide fuel, radioisotope, NAA, neutron beam experiment

1. INTRODUCTION

The activity in the field of nuclear science and technology in Indonesia was started by establishment of Natural Radioactivity Measurement Committee in 1958. Since then, the activities has been developing in various technical area. The most remarkable activities which marked the development of nuclear science and technology in Indonesia was the construction of three research reactors. The first reactor, TRIGA MARK-II type, was built in Bandung and reached its first criticality in 1964. The second reactor was built in Yogyakarta, so-called Kartini Reactor of 100 kW thermal power. The construction of this reactor, whic is also TRIGA type, was undertaken by Indonesian scientist and engineers benefiting the experience obtained from construction of the first reactor. The first criticality was reached in 1979. This reactor is mainly for education and training purposes. And, the third research reactor built is the Multi Purpose Reactor (MPR), namely in Indonesian as RSG-GAS or MPR – GA Siwabessy. The name GA Siwabessy is taken from the name of the pioneer of nuclear science and technology in Indonesia, especially in the use of radioisotopes in the medicine. All those research reactors are operated by National Nuclear Energy Agency of Indonesia, or BATAN.

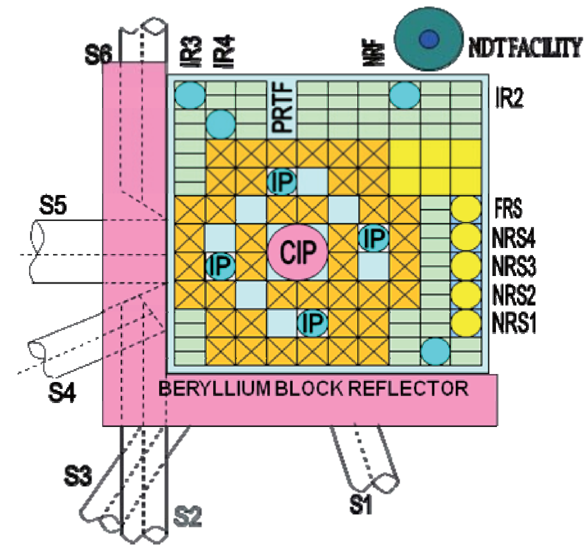
The RSG-GAS is located at Serpong Nuclear Area in the heart of science and technology park so-called Puspiptek. The last is a huge area where various scientific and technological laboratories are erected. The RSG-GAS is constructed with the aims to become a central part of various activities related to nuclear science and technology such as radioisotopes development and production, nuclear fuel element development, nuclear material research and development, reactor safety research etc. The results of these all activity is expected to accelerate the development of nuclear industry in Indonesia. The reactor which has 30 MW thermal power is a MTR type reactor. The reactor is bulit since 1985 and has achieved its first criticality in 1987. Since then, the reactor has been operated safely for more than 77.000 hours for several activities, mainly radioisotopes production, nuclear material research using neutron beam facilities, neutron activation analysis and many sample irradiation.

This paper aims to describe the activities has being done in the RSG-GAS until now and the strategic plan for the next five years development plan.

2. REACTOR DESCRIPTION

The RSG-GAS reactor was designed as a multipurpose research reactor with nominal thermal power of 30 MW, producing thermal neutron flux in the order within 2.5×10^{14} $n.cm^{-2}.s^{-1}$. As a multipurpose reactor, RSG-GAS provides facilities for utilization on material testing, radioisotope production, R&D using neutron irradiation as well as training.

The Fig. 1 shows the configuration of RSG-GAS reactor core and the most important technical specification of RSG-GAS is given in the Table 1.



: Fuel element IP: Irradiation Position

Fig. 1. The Configuration of RSG-GAS Core.

The core is cooled and moderated by light water. The reactor core is arranged on a grid plate with 10×10 positions. To guide the coolant flow through the core components, the grid plate is surrounded by a core shroud. Outside the core shroud, the L-shaped berryllium block reflector is placed adjacent to and covers two sides of the core. A support structure caries the grid plate and the core shroud, both of which are made from aluminum alloys, as well as the berylium block.

Table 1. Main Design Features of RSG-GAS

Parameter	Characteristic
Type	Open pool
Power (max.)	30 MWth
Thermal neutron flux	2.5×10^{14} $n/cm^2.s$
Moderator	Light Water
Reflector	Beryllium
Fuel type	U_3Si_2Al (plate)
No. FE in TWC	40 bundels
No. CFE in TWC	8
^{235}U enrichment	19.75%
^{235}U density	2.96 g/cm^3

The Typical Working Core consists of 40 fuel elements, 8 control elements and irradiation inserts comprising 8-fuel element position all together. The fuel elements are based on the proven MTR-technology. Each fuel element consists of a lower end fitting and upper handling device, two side plates and 21 fuel plates. Each fuel plate is composed of an AlMg-frame and two cover sheets of the same material, which enclose the U_3O_8-Al dispersion meat plate. By core conversion programme, this oxide fuel was converted to the silicide fuel U_3Si_2-Al with the same fuel density of 2.96 gU/cm^3.

The control elements are designed as a fork-type absorbers. The fueled part of the control element is very similar to the fueled part of the fuel element. In each absorber, 15 inner fuel plates are held together by two side plates. 3 fuel plates are removed at each end of the fueled zone to provide space for the insertion of the absorber blades. Aluminum plates replace two of the three fuel plates removed. The absorber assembly is composed of two stainless steel (material 1.4541, equivalent to SS 321) clad AgInCd blades.

Irradiation facilities as shown in Fig. 1 consits of one Central Irradiation Position (CIP), four Irradiation Position (IP) in the core and four Irradiation Position (IR) in berrylium reflector region. Other experimental facilities in RSG-GAS include four Normal Rabbit System (NRS), one Fast Rabbit System (FRS), one Power Ramp Test Facility (PRTF), one Neutron Radiography Facility (NRF), one Neutron Transmuttation Dopping (NTD) Facility and six horizontal neutron beam tubes. Four neutron beam tubes are actually to provide neutron beam for several diffractometer and spectrometer instrumentations in the Neutron Scattering Laboratory (NSL), one beam tube (S1) is to be used as Iodine Loop Facility, and one tube (S3) is still idle.

The NSL is equiped with four scattering instruments installed in the experimental hall of reactor (XHR) and three instruments which are located in the neutron guide hall (NGH). Those two halls are connected by a tunnel accommodating two neutron guides (NG1 and NG2) transferring neutron from the reactor to the instruments in the NGH. The Fig. 2 shows the layout of neutron scattering instruments in the XHR and NGH. The instruments comprise:

1. Residual stress measurement diffractometer, DN1
2. Four-circle diffractometer / texture diffractometer, DN2
3. High-resolution powder diffractometer, DN3
4. Triple axis spectrometer, SN1
5. Small angle neutron scattering spectrometer (SMARTer), SN2
6. High-resolution small angle neutron scattering spectrometer (HRSANS), SN3
7. Neutron radiography facility, RN1

3. CURRENT ACTIVITIES

Over the periode of 1995-1998, the reactor was operated around 5000 h/years at 25-30 MW power level, 5 – 6 cycles/year, 750 MWD per cycle. Since the year 1998, the reactor was operated at the power level of 15MW, up to max. 4 cores cycles annually, 540 – 600 MWD per cycle, based on optimization of the fuel availability, user requirement as well as efficiency. Since 1999 BATAN has implemented the RSG-GAS reactor core conversion programmed from oxide fuel U_3O_8-Al to silicide fuel U_3Si_2-Al with the same uranium density in meat of 2.96gU/cm^3. Conversion of the RSG-GAS reactor core was carried out by operating a core of oxide-silicide mixture. It took 10 operational cycles to get the full silicide core. Full silicide core was reached at core 45th in August 2002.

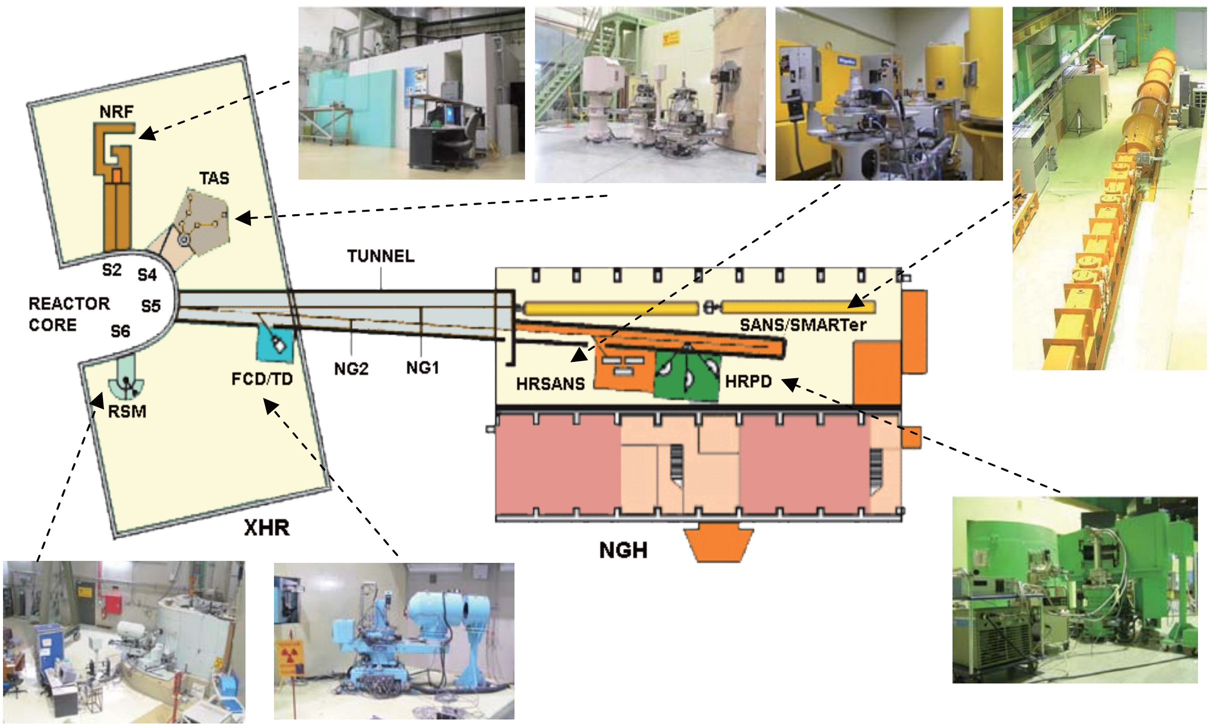


Fig. 2. The lay out of neutron scattering instruments.

Several activities have being done in RSG-GAS as given below.

3.1. Radioisotopes Production

A high neutron flux in the reactor core is potential to produce high quality radioisotopes by faster and more efficient irradiation. All irradiation positions is used for radioisotopes development and production. Various types of radioisotopes from fission and activation process have been produced in RSG-GAS reactor including for medical purpose: ^{131}I, ^{99}Mo, ^{133}I; etc, for industrial purpose, such as ^{192}Ir, ^{82}Br, ^{60}Co, etc, for tracer, such as ^{14}C, ^{32}P, ^{35}S. The radioisotopes which are commercially produced by state owned company, namely PT. Batek, are for meeting the domestic needs and partly for supplying regional demand.

3.2. Neutron Beam Experiments

The neutron beams coming from RSG-GAS are used to carry out several experiments, research and development using various neutron beam instruments. With regard to the neutron scattering related topic of activities, along with hard matter research, research in soft condensed matter and life sciences are emerging recently and significantly on investigating biological macromolecule structures in solution. Based on the arrangement of RSG-GAS operation mode and length, the neutron beams are available for about 14 – 15 days monthly which is about 170 days yearly.

Since at least the last five years, a major work on increasing the neutron beam intensity at the several neutron beam instruments, modernization of control and data acquisition system have been performed. However, a number of research experimental works on material characterization has also been conducted. The results of the experiments are published in various scientific conferences and journals. The activities using each instruments are described very briefly below [1].

1. *Diffractometer for residual stress measurement (RSM), DN1-M*
   Using the instrument namely DN1 (or DN1-M), several measurements on TIG-welded SUS304, W/Cu composite, fiber reinforced composite and titanium casting alloys have been performed. The experiments are aimed to investigate the internal stress and its distribution. Two example of the experimental results could be found in [2,3].

2. *Four circle diffractometer / texture diffractometer (FCD/TD), DN2*
   In 2009, a 1-meter flight tube made of borkron glass and coated by 56Ni was installed in the position between monochromator and sample table of instrument DN2. This flight tube resulted the increase of incident neutron beam intensity by factor of 2.5. It benefits to improve the experimental results which are mostly dealing with the measurement of texture of materials such as aluminum alloys (A 3104, A6060 and A7075 series) [4], brass, steel, zircaloy and copper [5].

3. *High resolution powder diffractometer (HRPD), DN3*
   At the beginning, the utilization of the difractometer DN3 was mostly for collecting the data from superconductor and magnetic sample. Cryostat equipment has been set up at the sample position to allow the measurement at a low temperature for crystal and magnetic structureal studies from magnetic ample. Several experiments to study on crystal structure using cryostat or furnace have been carried out and published recently [6-8].

4. *Triple axis spectrometer (TAS), SN1*
   The instrument is still under repair, especially for assembling the controller interfacing with the peripheral interface. The development of the software for controling and data acquisition in both double and triple axis mode has done as well. New developed GUI software is also being developed.

5. *Small-angle neutron scattering (SANS) spectrometer (SMARTer), SN2*
   After having completed a five years program of revitalizing of the instrument, especially the electronic and mechanics system, the computer software including control system, data acquisition, the SANS instruments has put back in operation and works with better performance. Since then, many experimental activities using this instrument has been done. The umber of external users, local and regional users coming from universities and research institutes is increasing. While, the

subject of micellar solutions, biopolymers, ceramics and porous materials, also protein and virus in solution are appearing as the major issues [9, 10]

6. *High resolution SANS spectrometer (HRSANS), SN3*
   The recent activity is improving the control and data acquisition system by installing the new one. The test has ben performed and showed a good performance of the new system. Further examination and improvement of the instruments performances are still in progress.

7. *Neutron radiography facility (NRF), RN1*
   The present development of the facility is to set up the neutron tomography instrument, including construction of a rotary and translation table, installation of a new CCD camera, along with Li_6-ZnS scintillator screen and high reflective TiO_2 coated mirror [11, 12]. Fig 3. shows an example of test's result.

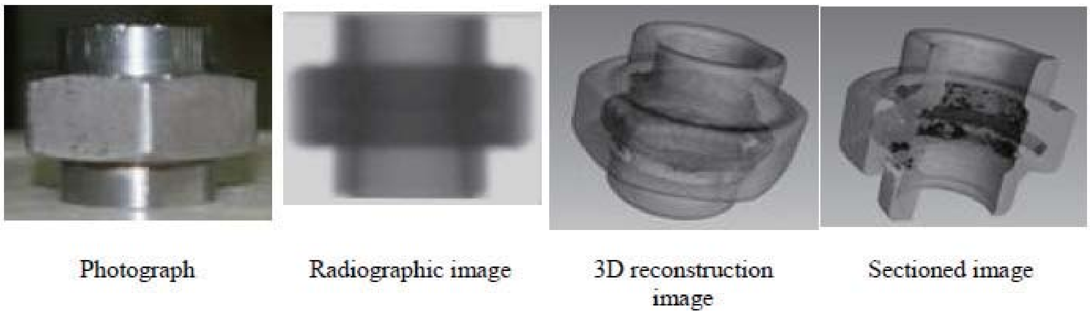


Fig. 3. Radiographic image and view of 3D volume reconstruction of pipe connector [12].

3.3. Neutron Activation Analysis

The RSG-GAS is equipped with the facility to perform neutron activation analysis (NAA), especially utilizing thermal neutron. For instance, the NAA facility in the RSG-GAS has limited utilization due to limited capacity of the related equipments. However, the activities of NAA in the RSG-GAS covering the various areas of application are being done as following:

- Area of health, such as: trace elemental analysis on whole blood, serum, hair samples, trace elemental analysis on food sample (macro-micro nutrient on food)
- Area of environment, such as: to monitor the environment contamination at industrial region, airborne particulate samples, marine river pollutant (fish, water, sediment, oyster etc), soil, river sediment, lichen and geochemical mapping
- Area of industry, such as: elements determination in tea, cigarret, cosmetic product, can food, industrial waste samples

The NAA activities is also done in the laboratories in the Bandung Nuclear Research Center and Yogyakarta Nuckear Research Center where the research reactors are located. In this context, Indonesia has been participating actively in the regional activities such in IAEA RCA and FNCA.

3.4. Component Refurbishment

Along with the effort to assure the reliability and safe operation of the RSG-GAS, and because RSG-GAS has entering its 20 years of operation, since about three years, several component and equipment refurbishment have been under taken as follow:

- Replacement of all secondary pumps; 2 out of 3 pumps have been successfully replaced
- Cooling tower : replacement of blowers
- Isolation for Chilled Water System had been replaced.
- Radiation Monitoring System had been renewed.

- Instrumentation system of symatic 5 has been replaced by Instrumentation system of symatic 7
- Replacement of all Control Rod Absorption

Other component refurbishment and modification are planned.

3.5. Development of Higher Density of Silicide Fuel Element

After succeeding the conversion of the reactor fuel from oxide to silicide fuel, a development activity has recently performed to obtain a higher density of silicide fuel, i.e. 4.8 gU/cm^3 and 5.2 gU/cm^3 [13]. Several plates of both type have been fabricated, but only the one with 4.8 gU/cm^3 of density has been irradiated in the reactor until more than 40% burn up. The post irradiation examination is not yet performed, but a preliminary inspection showed a good condition of that irradiated fuel. Along with that activity, an analysis of the RSG-GAS safety performance using the higher density of silicide fuel has also been performed [14]

4. STRATEGIC PLAN FOR FUTURE ACTIVITIES

The RSG-GAS has been and would remains the most important facility to support the R&D on and utilization of nuclear sciences and technology in Indonesia for many years to come. A strategic plan for future activities using the RSG-GAS should be drawn and set to assure the strategic goals could be achieved.

The vision of RSG-GAS has been set to be recognized as a reliable neutron source facility for application and development of nuclear science and technology contributing to improvement of socio-economic well-being. The Mission RSG-GAS is:

- To assure the sustainability of irradiation services for commercial purposes and R&D
- To develop a network for increasing the utilization of the reactor
- To enhance the safe operation and maintenance best practices and technology
- To promote the benefits of nuclear science and technologies.

In order to achieve the goal and accomplish the mission, several strategic steps have been determined as follow:

- *To perform maintenance, modification and refurbishment, if ncessary, the component and system of the RSG-GAS to maintain and improve the reactor operation reliability and safety*.

  To date, the RSG-GAS has been 22 years in operation. Several system, structure and component has suffered from ageing. A rigourous maintenance programme is becoming more important. Several reactor components had been refurbished and others are in schedule in the next future. In line with this, a target to increase the operation hours per year up to 4200 hours/year is being set up.

- *To improve the irradiation and other related facilities both in terms of type and quantity*.

  With regard to the improvement of the capacity and capability of radioisotopes production for commercial purposes, RSG-GAS would always cooperate with PT. Batek as the responsible party. In principle, considering the technical performance of reactor system, RSG-GAS is committed to support the developent plan of PT. Batek.

  For developing the application of NAA facility, it is planned in the near future to develop the facility towards utilization of epithermal neutron as irradiation source. Along with that plan, the automation of sample irradiation and counting system has been started. The development of the NAA facility is to anticipate the increase of application of NAA.

  Considering that the neutron scattering experiments is a major user of the RSG-GAS, the avalaibility of the neutron beam would be increased accordingly to the increase of the operation hour. On the other hand, the improvement of neutron scattering instruments, which has been performed, will be continued to accomodate the need of development of research areas and providing a better experimental instruments. Special attention is given to the considerable increase of ustilization of neutron radiography and SANS that needs consequently improvement of those instruments.

- *To develop a network with potential reactor users to increase the utilization of the reactor.*
  At a present, beside the local researchers, the researchers from several universities and research institutions, even from foreign countries, are using the experimental facilities in RSG-GAS, especially neutron beam instruments. BATAN as the operator of RSG-GAS participate actively in various international and regional activities, such as IAEA RCP, IAEA RCA, and FNCA, to utilize the RSG-GAS. However, considering the potential capacity of the RSG-GAS, a development of network to enlarge the potetial users of the RSG-GAS.

- *To develop the human resources programme and to enhance the safety culture*
  As a consequence of the government policy several years ago in terms of recruitment of government employee, the numbers of staff in the RSG-GAS is slightly decreasing. Only since about three years ago, several new staff have been recruited. While, a quite number of senior and experienced staff are getting closer to retirement age. The problem of knowledge and experience gap between senior and younger staff has became a major concern. The program namely coaching mentoring which aims to accelerate the transfer of knowledge from senior to junior staff has been set to priority. That program is also aimed to maintain and even enhance the safety culture.

5. CONCLUSIONS

The RSG-GAS has been operated safely and reliably since 22 years after its criticality. Several activities in radioisotopes development and production, fuel development, NAA, neutron beam experiments and other irradiation application has been conducted. However, the utilization level needs to be increased. Along with the refurbishment of several reactor components to maintain the reliability and safety of operation, the improvement and development of research and experimental facilities has been done and planed for the next five years. The collaborating activities with different institutes in the country and in the region is pursuing. The human resource development for the reactor staff is also an important program to fill the gap of kowledge among the reactor staff and maintain the safe operation of the reactor.

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3.2 The University of Missouri Research Reactor HEU to LEU Conversion Project Status

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The University of Missouri Research Reactor (MURR®) is one of five U.S. high performance research and test reactors that are actively collaborating with the U.S. Department of Energy (DOE) to find a suitable low-enriched uranium (LEU) fuel replacement for the currently required highly-enriched uranium (HEU) fuel. A conversion feasibility study based on U-10Mo monolithic LEU fuel was completed in 2009. It was concluded that the proposed LEU fuel assembly design, in conjunction with an increase in power level from 10 to 12 MW_{th}, will (1) maintain safety margins during operation, (2) allow operating fuel cycle lengths to be maintained for efficient and effective use of the facility, and (3) preserve an acceptable level and spectrum of key neutron fluxes to meet the scientific mission of the facility. The MURR and Argonne National Laboratory (ANL) team is continuing to work toward realization of the conversion.

The "Preliminary Safety Analysis Report Methodologies and Scenarios for LEU Conversion of MURR" was completed in June 2011. This report documents design parameter values critical to the Fuel Development (FD), Fuel Fabrication Capability (FFC) and Hydromechanical Fuel Test Facility (HMFTF) projects. The report also provides a preliminary evaluation of safety analysis techniques and data that will be needed to complete the fuel conversion Safety Analysis Report (SAR), especially those related to the U-10Mo monolithic LEU fuel.

Specific studies are underway to validate the proposed path to an LEU fuel conversion. Coupled fluid-structure simulations and experiments are being conducted to understand the hydrodynamic plate deformation risk for 0.965 mm (38 mil) thick fuel plates. Methodologies that were recently developed to answer the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) regarding the MURR 2006 relicensing submittal will be used in the LEU conversion effort. Transition LEU fuel elements that will have a minimal impact on the MURR users have been identified, and the transitional fuel shuffling plan is being developed. Finally, the desired regulatory path for licensing the facility as a 12 MW LEU Research Reactor has been identified.

Keywords: fuel conversion, U-10Mo, boron, RERTR, GTRI

1. INTRODUCTION

Because of its compact core design (33 liters), which requires a very high loading density of ^{235}U, the University of Missouri Research Reactor (MURR®) could not perform its mission with any previously qualified low-enriched uranium (LEU) fuels. However, in 2006 with the prospect of the Global Threat Reduction Initiative (GTRI) Fuel Development Program validating the performance of U-Mo monolithic LEU foil fuels, MURR began actively collaborating with the GTRI Conversion Program, and four other U.S. high-performance research and test reactors that use highly-enriched uranium (HEU) fuel, to find a suitable LEU fuel replacement.

This paper provides the current status of converting the MURR from an aluminide dispersion HEU fuel to a U-10Mo monolithic LEU fuel. Specific topics discussed include the LEU Preliminary Safety Analysis Report (PSAR) [1], coupled fluid-structure simulations and experiments on fuel plates, methodologies developed to answer relicensing questions that will aid in the conversion analyses, the LEU transition fuel elements and fuel cycle, and licensing challenges in uprating power level from 10 to 12 MW.

2. GENERAL DESCRIPTION OF FACILITY, REACTOR AND FUEL

The MURR is a multi-disciplinary research and education facility providing a broad range of analytical and irradiation services to the research community and the commercial sector [see www.murr.missouri.edu]. The MURR has six types of experimental facilities designed to support these services and research programs: the Center Test Hole (Flux Trap); the Pneumatic Tube System; the Graphite Reflector Region; the Bulk Pool Area; the (six) Beamports; and the Thermal Column. The first four experimental facilities provide areas for the placement of sample holders, or carriers, in different regions of the reactor core assembly for the purposes of material irradiation. Some of the material irradiation services include transmutation doping of silicon, isotope production for the development of radiopharmaceuticals and other life-science research, and neutron activation analysis. The six beamports channel neutron radiation from the reactor core to experimental equipment which is used primarily to determine the structure of solids and liquids through neutron scattering and to perform Boron Neutron Capture Therapy (BNCT) experiments.

2.1 Basic Reactor Description

The MURR is a pressurized, reflected (beryllium and graphite), heterogeneous, open pool-type reactor, which is light-water moderated and cooled. The reactor is designed and licensed to operate at a maximum thermal power level of 10 MW with forced cooling, or up to 50 kW in the natural convection mode.

The reactor core assembly is located eccentrically within a cylindrically-shaped, aluminum-lined pool, approximately 10 feet (3.0 m) in diameter and 30 feet (9.1 m) deep. The reactor core consists of four major regions: center test hole (flux trap), fuel, control blade and reflector. A three-dimensional view of the reactor core assembly is shown in Figure 2.1. The fuel region has a fixed geometry consisting of eight fuel elements having identical physical dimensions placed vertically around an annulus between two cylindrical aluminum reactor pressure vessels.

Each fuel assembly is comprised of 24 circumferential plates. The HEU plates contain uranium enriched to approximately 93% in the isotope ^{235}U as the fuel material. The control blade region is an annular gap between the outer pressure vessel and the inner reflector annulus, so that no penetration of the reactor pressure vessels is required. Five control blades operate vertically within this gap: four Boral® and one stainless steel. The blades control the reactor reactivity by varying neutron reflection. The reflector region consists of two concentric right circular annuli surrounding the control blade region. The inner reflector annulus is a 2.71 inch (6.9 cm) thick solid sleeve of beryllium metal. The outer reflector annulus consists of twelve 30° arc length vertical elements of mostly graphite canned in aluminum, having a total radial thickness of 8.89 inches (22.6 cm).

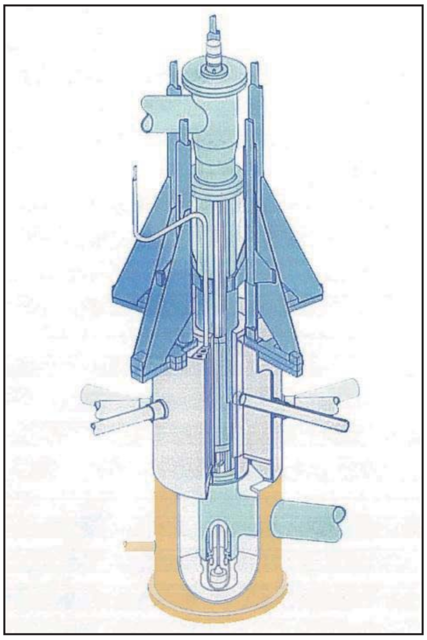

Fig. 2.1. Reactor Core Assembly.

2.2 Current Fuel Design and Operating Cycle

In 1971, the MURR was converted from the original uranium-aluminum alloy fuel to a uranium-aluminide dispersion U-Al_x fuel material with a maximum loading of 775 grams of ^{235}U per element. The U-Al_x dispersion fuel system was developed at the Idaho National Engineering Laboratory (INEL) for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the Materials Test Reactor (MTR) and Engineering Test Reactor (ETR) prior to its use at MURR [2, 3]. A drawing of the MURR fuel element is shown in Figure 2.2.

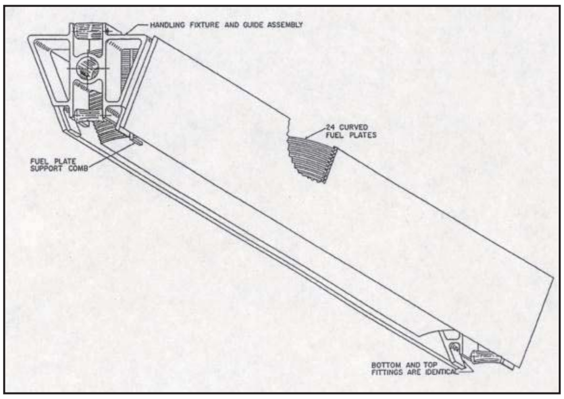

Fig. 2.2. MURR Fuel Element – Pictorial View.

The MURR operates continuously with the exception of a weekly scheduled shutdown. Over the past 35 years of operation, the MURR has averaged approximately 6.3 days/week at full power. The weekly shutdown provides an opportunity to access samples in the center test hole, to perform surveillance tests and maintenance, and to replace all eight fuel elements in the core. Replacing the fuel elements provides a chance to remix or shuffle the elements that will be used and to restart the reactor with a xenon-free core.

3. PRELIMINARY SAFETY ANALYSIS REPORT

Before conducting the detailed analyses needed to prepare the LEU conversion final Safety Analysis Report (SAR), and certainly prior to finalizing the design of the LEU fuel elements, data relevant to the needs of the GTRI Program "pillars" needed to be documented. The "Preliminary Safety Analysis Report Methodologies and Scenarios for LEU Conversion of the University of Missouri Research Reactor (MURR)" was completed in June 2011 and formally issued as a MURR Technical Data Report in August 2011 [4]. This report (also referred to as the PSAR) documents certain design parameter values critical to the Fuel Development (FD), Fuel Fabrication Capability (FFC), and the Hydromechanical Fuel Test Facility (HMFTF) projects. Furthermore, this report provides a preliminary evaluation of safety analysis techniques (e.g., methodologies and codes) and data that will be needed for completing the fuel conversion SAR, especially as they relate to the high density, U-10Mo monolithic LEU foil fuel.

The structure of the LEU PSAR follows the chapters and sections, and associated content of the existing MURR HEU SAR, which was developed in accordance with NUREG-1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Parts 1 and 2 [5,6]. Where design changes due to the conversion affect the safety analysis, discussion of the HEU safety analyses are presented alongside requirements for LEU analyses under the appropriate section for easy reference to the SAR. Chapters and/or sections of the SAR that would not be changed due to a new fuel type were not included in the PSAR. A new appendix (Appendix D) was added to address MURR specific fuel qualification.

The LEU fuel plate and element design described in the PSAR was developed as part of the 2009 Fuel Conversion Feasibility Study [4]. Figure 3.1 provides a cross sectional illustration of an LEU fuel plate with the fuel meat, Zr interlayer, and Al-6061 clad tolerances noted. The fuel design, measurement uncertainties, and fabrication tolerances are crucial parameters for the performance and safety analyses. These parameters are needed for the FD and FFC projects. Additionally, flow testing of the LEU fuel plates, which are thinner and have a different fuel meat/core than the current HEU fuel plates, is needed to ensure mechanical stability in the MURR thermal-hydraulic environment. These flow tests will be performed at the HMFTF.

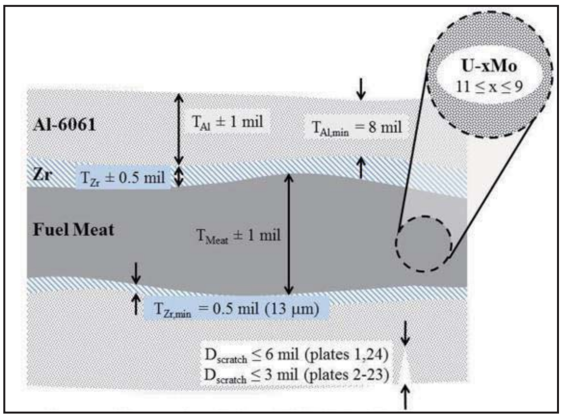

Fig. 3.1. Cross Sectional View of an LEU Fuel.

Another parameter of interest is the fission product release fractions from the U-Mo monolithic LEU fuel. The metallic matrix differs from the U-Al_x matrix of the HEU fuel. Hence, the fission product retention characteristics during a Maximum Hypothetical Accident (MHA) will vary to some degree relative to the current U-Al_x dispersion fuel system. Although, there is no assumption of fission product retention of iodine, krypton, or xenon in the current MURR MHA analysis, a confirmatory experiment is requested to evaluate if the assumption of a complete release of iodine, krypton, and xenon from a melt is overly conservative. Additionally, any release of actinides from the LEU fuel should also be assessed and appropriately included in the consequences of the MHA and/or the mishandling or malfunction of fuel accident scenarios.

A revised/improved RELAP model developed for the Loss of Flow and Loss of Coolant Accident (LOFA and LOCA) analyses will be used to confirm that peak fuel temperatures remain well below the fuel blistering temperature. In these analyses, properly conservative assumptions for an LEU core will be used.

4. TRANSITION LEU FUEL ELEMENTS/FUEL CYCLE

The MURR active fuel cycle typically consists of 32 to 40 fuel elements, which corresponds to 16 to 20 pairs of elements. A weekly core loading is always made up of four pairs, with two elements in each pair being loaded opposite each other and having approximately the same burnup history. The compact core volume limits excess reactivity and results in the control rods being fully withdrawn when an HEU core with equilibrium xenon activity has approximately 670 MWDs on it. This results in the maximum achievable burnup limit for an HEU element being 150 MWDs, which corresponds to slightly less than a 25% ^{235}U burnup. This also results in the hot spot burnup in a fuel element being less than 1.8E21 fissions/cc, so the Technical Specification limit of 2.3E21 fissions/cc is not achievable due to excess reactivity limitations.

Cores are loaded so that the average fuel element has around 75 MWDs on it. Typically a fuel element will be used in 18 to 20 different core loadings before being retired from the fuel cycle. A core with fuel elements approaching the burnup limit will also include a corresponding number of elements with very little power history on them. This maximizes the number of MWDs obtainable per fuel element and results in the control blades being almost fully withdrawn with equilibrium xenon—less than 0.5% of excess reactivity in the core. With the control blades almost fully withdrawn during more than 80% of the weekly operation time, this provides the best flux distribution for research and irradiation utilization of the MURR.

The LEU fuel cycle use will utilize this same approach because it maximizes the MWD obtainable per fuel element and continues to provide the best flux distribution for the reactor users. The U-10Mo LEU fuel tests performed by Idaho National Laboratory (INL) have not indicated any fission/cc burnup limit that would be achievable in MURR. So the limiting factor for MURR LEU fuel burnup will be excess reactivity and safety limits for mixed burnup cores. The LEU conversion fuel element design for MURR requires increasing the water-to-metal ratio to gain the required additional reactivity to avoid a significant increase in the number of fuel elements needed for full time operation of the reactor. This was done by decreasing the plate thicknesses and increasing the width of the coolant channels. The twenty-three internal coolant channels in the LEU element are 92 mils (2.34 mm) wide versus 80 mils (2.03 mm) in the HEU fuel elements. Because of the difference in coolant channel widths, the HEU and LEU fuel elements cannot be used in a combined mixed core.

For conversion to LEU fuel, transition fuel elements have been studied to achieve the required flux distribution while the necessary fuel element burnup distribution is generated in the MURR typical 32 to 40 element fuel cycle inventory. Three possible types of transition fuel elements were considered: thin cadmium wires swaged into the element side plates; thinner U-10Mo foil thicknesses relative to the nominal LEU design; or a small concentration of boron in the 6061-T6 aluminum 0.150 inch (3.81 mm) thick element side plates. The use of aluminum side plates with less than 1% boron is the current choice and has been evaluated in greater detail.

REBUS-PC is the code used for modeling the fuel cycle as recommended and performed by the Argonne National Laboratory (ANL) group. The REBUS-DIF3D code and modeling of MURR was benchmarked with the current HEU fuel cycle and then used to determine the performance of the proposed LEU core in an earlier feasibility study [4].

A discretization study was performed. Figure 4.1 illustrates a significant axial profile variation of the neutron capture rate in the ^{10}B, strongly affected by the control blade position. Radially, the profile is somewhat flatter, although the reaction rates are higher near the flux trap (innermost radial position) and beryllium/graphite reflector regions. It has also been noted that the control blade position has an impact on the radial reaction rate profile.

The k_{eff} for the cases with 11, 13 and 14 axial zones are within the 1 σ of the MCNP calculation, confirming that the discretization of the fuel element

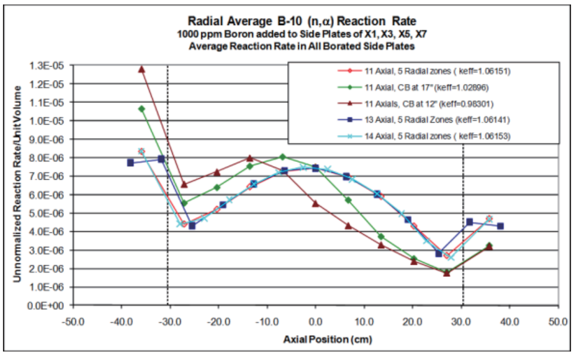

Fig. 4.1. Axial Profile of Neutron Capture Rate in Borated Side Plates.

side plates does not affect the calculated k_{eff}. The 14 axial zone case, with 12 axial zones in the range of the fuel meat, and one zone each in the regions above and below the fuel meat, is the preferable discretization since it captures the axial reaction rate profile and uses an axial discretization similar to that needed for the fuel depletion in REBUS-DIF3D.

For mixed-burnup core calculations with REBUS-DIF3D and MCNP, a set of lumped fission product (LFP) cross sections are needed. The DIF3D calculations use cross sections in a 10-group structure that was developed specifically for the MURR spectrum. For MCNP, a 69-group set of LFP cross sections are utilized. The cross sections are generated with WIMS-ANL using a 1-D core representation. For the high concentrations of boron included in the startup cores, it was necessary to determine the impact of the poison on the neutron spectrum, which could affect the LFP cross section.

The effect of 2000 ppm boron in the side plates of the odd-numbered elements on the neutron spectrum in the fuel plates was studied in the REBUS-DIF3D analysis. Within the fuel meat alone, which is modeled explicitly in the MCNP model, the impact of the borated side plates on the neutron spectrum is less than 3%. The impact is larger when the fuel meat and clad are smeared together, as in the DIF3D model. However, the effect on the spectrum appears only in the two most thermal groups of the energy spectrum, and is less than 10%. Consequently, it does not appear necessary to represent the poisoned side plates in the WIMS-ANL model used to generate the LFP cross sections for DIF3D or MCNP.

Figure 4.2 plots the ratio of the neutron spectrum by neutron energy group in the side plates for the borated/standard cases (standard means non-borated); with 2000 ppm boron poison added to the odd-numbered element side plates. In the side plates themselves, there is a significant depression of the spectrum in the thermal groups <0.2 eV due to the presence of the boron poison. The strongest depression (nearly 50%) is seen in the borated element side plates. The effect in the neighboring side plates is also quite strong. Consequently, a 10-group set of burnup-dependent ^{10}B cross sections was utilized in the REBUS-DIF3D analysis of the fuel cycle. The cross sections were fit as a function of the 10B concentration in the side plate at each statepoint in the fuel cycle.

REBUS-DIF3D is used to deplete the complex fuel cycle of MURR, but MCNP is used to apply the depleted compositions for all other neutronic

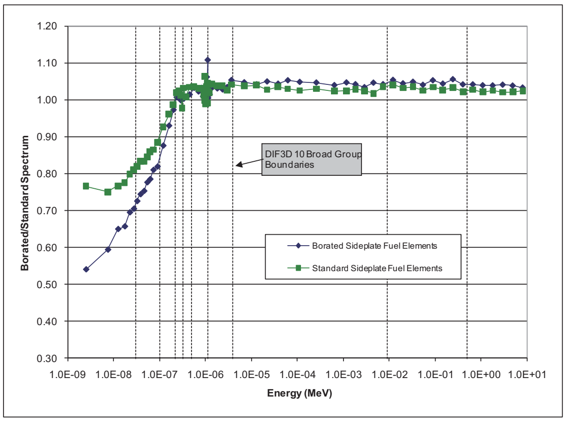

Fig. 4.2. Ratio of Neutron Spectrum in Side Plates for Borated/Standard Cases.

calculations, such as power distribution generation or experimental performance prediction. As described in the Feasibility Report [4], the compositions calculated by REBUS-DIF3D for each axial zone of each fuel plate and side plate were processed to create MCNP material cards.

The azimuthal power peaking factors for transition fuel cycle cores with borated side plates are less than for the intended routine LEU cores without borated side plates. As would be expected, the azimuthal peaks, which occur at the ends of the fuel plates near the side plates, increase as the boron in the side plates burns out.

The DIF3D results for a mixed-burnup LEU transition core were compared with results for the same mixed core calculated with MCNP. Table 4.1 compares the core k_{eff} calculated by DIF3D with the k_{eff} calculated by MCNP. The control blades were banked at 23 inches withdrawn in both the DIF3D and MCNP calculations. The total core burnup at BOC (Beginning of Cycle) was 642.3 MWD. REBUS-DIF3D composition data were collected at the BOC (Xe-free conditions) and on the second day of the week (equilibrium Xe conditions). Borated side plates were present in the even-numbered elements of this core. The boron in the side plates was about 52%

depleted at BOC. The results show that the k_{eff} calculated by DIF3D are with 0.2% $\Delta k/k$ of the MCNP results for the mixed-burnup LEU transition core.

Table 4.1 Core k_{eff} For LEU Transition Fuel Cycle, Week 32, Control Blades Banked at 23 Inches

	MCNP		DIF3D	Difference
	k_{eff}	σ	k_{eff}	(Δk/k)
Day 0 (Xe-free)	1.02172	0.00011	1.02345	0.17%
Day 2 (Eq. Xe)	0.98986	0.00010	0.99029	0.04%

Figure 4.3 presents a comparison of the axial heat flux profiles calculated by DIF3D and MCNP for fuel plates 1 and 24 of element X1, an element with a BOC burnup of 23.4 MWD and no boron in the side plates. The results indicate a very good agreement between the two codes for the heat flux profile.

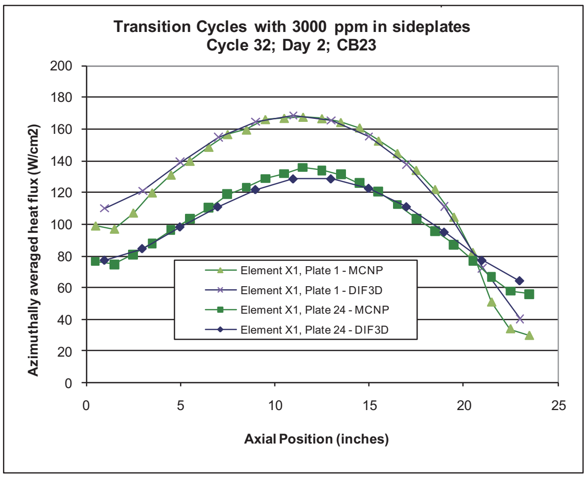

Fig. 4.3. Axial Heat Flux Profile in LEU Transition Core; Week 32 with Equilibrium Xenon and Control Blades at 23 Inches.

The boron loading in the side plates was selected after modeling 1000, 2000, 3500 and 7000 ppm boron in the side plates and plotting the core k_{eff} versus the power history on the core. A boron loading of 3000 ppm was selected as providing the appropriate balance between initial reactivity hold-down and the boron depletion rate in the MURR transition fuel cycle.

Table 4.2 below shows the first proposed transition fuel cycle that has been recently modeled. The cycle starts out with 12 LEU fuel elements with borated side plates containing 3000 ppm boron and eight standard LEU fuel elements. The initial 38 cores consist of four borated LEU elements with four standard LEU elements. The mix of elements is then altered over time to preserve the desired reactivity while building up an equilibrium inventory of partially depleted standard elements.

As noted in the last column, the total MWD of the cores increase during the transitional fuel cycle. By core 79, a sufficient inventory of standard LEU elements has been built up to operate in the normal MURR fuel cycle mode.

Table 4.2 LEU Fuel Elements in Proposed Transition Fuel Cycle

Weekly LEU Cores	Core Elements (Boron/ Standard)	Boron Side Plate LEU Fuel Elements	Standard Side Plate LEU Fuel Elements	BOC Core MWDs
C1 → C16	4/4	12	8	0-342
C17 → C38	4/4	12	16	340-682
C39 → C58	2/6	12	24	586-872
C59 → C62	2/6	12	32	625-627
C63 → C66	4/4	12	32	639-714
C67 → C78	2/6	12	32	758-800
C79	0/8	0	32	916

Figure 4.4 shows the equilibrium xenon k_{eff} with the control blades banked at 23 inches (88.5%) withdrawn for three different fuel cycles. The green triangles indicate the k_{eff} of the intended routine LEU fuel cycle cores [4] with four pairs of fuel elements without borated side plates, but with different burnup between the four pairs. The average burnup of the eight elements in the core will be approximately half of the targeted full burnup. It should be noted that the calculated k_{eff} includes a model bias of about -0.5% $\Delta k/k$, as determined by benchmark studies in the feasibility analyses [4]. The red squares indicate the core k_{eff} of proposed transition fuel cycle cores containing borated fuel elements that are modeled with 3000 ppm boron in the side plates. The blue diamonds indicate the fuel cycle with the same cycling between fuel elements as the red squares, except the borated fuel elements are modeled with 3500 ppm boron in the side plates. This shows the effect of an additional 500 ppm of boron in half of the side plates in the core. The loading of 3000 ppm boron in the transitional element side plates works

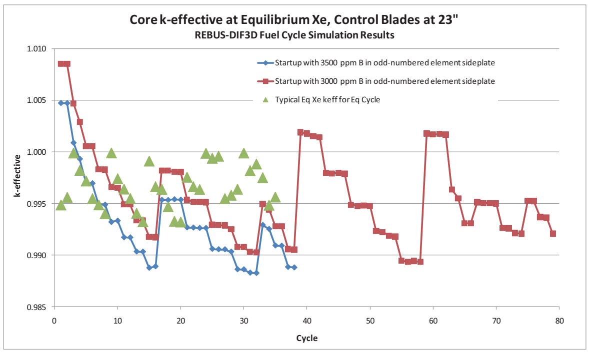

Fig. 4.4. Potential Transition Fuel Cycle Cores Compared to the Intended Routine LEU Fuel Cycle.

well since only 25% of the boron is left after about 186 MWD of fuel element power history, which corresponds to the end of cycle 78. These results show that 3000 ppm boron in transitional element side plates can serve well for the transition fuel cycle. After obtaining these results, the fuel cycle can be improved for cores 39 through 62 by adjusting which fuel elements are in the core loadings, thus improving the variability of k_{eff} for those cores.

5. FLOW-INDUCED DEFORMATION ANALYSIS

With the transition to new aluminum clad, U-10Mo monolithic LEU foil fuel, significant changes are necessary for the fuel plate design. Some of the new fuel plates are 24% thinner than the current HEU design and have a potential slip plane at the interface between the foil and the cladding. These changes have led to concerns about mechanical stability under high velocity coolant flows. Therefore, work is currently underway at the University of Missouri to quantify the potential for deflection of the new fuel plates. This work is focused on numeric modeling through fluid-structure interaction (FSI) simulations, as well as flow experiments to validate the numeric models.

5.1 Fluid-Structure Interaction Simulations

The FSI simulation process involves explicitly coupling a commercial computational fluid dynamics (CFD) code, Star-CCM+ [6], with a commercial finite element analysis (FEA) code, Abaqus [7]. By allowing the CFD and FEA codes to solve for their respective fluid and solid domains independently, the final result should yield high accuracy. Presently, the geometry modeled in the simulations consists of a large channel leading to a flat plate with channels of differing thickness on either side, as shown in Figure 5.1. As the simulation iterates between the two programs, the static pressure field in the CFD model is passed as a load to the FEA model, allowing it to solve for deflection. Once the FEA code has determined the new profile of the plate, this information is passed back to the CFD code, and the CFD mesh is morphed around the updated geometry. The fluid field is then solved again, and the process continues iterating until a converged, steady state solution is reached.

Given the high aspect ratio of the model, as well as the influence of high velocity coolant flows on

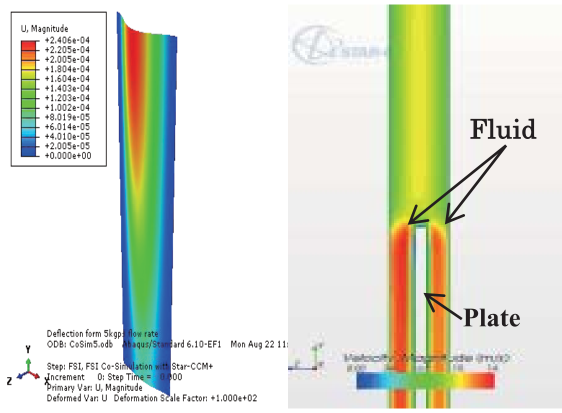

Fig. 5.1. FEA Deflection Solution (left), and CFD Velocity Solution along the Mid-Span of a Plate at the Leading Edge (right).

plate deflection, stability in FSI simulations has been a significant challenge. In the past year, techniques to achieve solution stability have been developed, and include damping the motion of the plate and pressure field in early iterations. As shown in Figure 5.1, by using both the CFD and FEA codes together, it is now possible to discern the deflection and stress profile of a plate, and the velocity and pressure profile of the fluid.

Now that the FSI process is understood well enough to ensure stability and convergence to a final solution, the emphasis is turning to validate the accuracy of the simulations. This involves running flow experiments and comparing the results to a computational analog of the same assembly. Once the computational method is validated, more complex assemblies can be modeled in a short time frame.

5.2 Flow Loop Experiments

Over the past year, the flow loop at the University of Missouri has been redesigned to accommodate testing of a wide range of assemblies. The loop includes a flow meter, pressure transducer, and a system for precise flow rate control. Recent enhancements now allow for the recording of data on the flow induced deflection phenomenon for small scale, single and dual plate models. While the primary goal of the flow loop tests is to help validate the numeric models, they will also be helpful in evaluating the effect of plate curvature and a leading-edge comb.

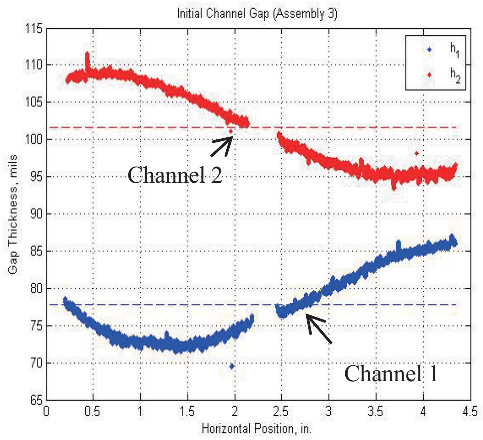

Fig. 5.2. Initial Channel Gap Profile and Mean Values (32 mil plate).

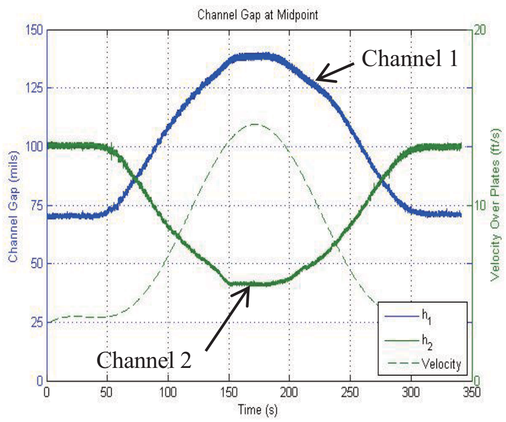

Fig. 5.3. Flow Channel Gap and Average Velocity through the Channels (32 mil plate).

These tests are designed to analyze the impact of plate thickness, flow velocity, channel gap, curvature, and a comb on deflection. Initial tests are designed to achieve as much deflection as possible and provide an upper bound for deflection determination. These tests are focusing on flat plates 110 mm (4.3 inches) wide without a comb. Additionally, two plate thicknesses of 0.81 mm (32 mils) and 1.02 mm (40 mils) are being studied. The plates are placed in test assemblies with flow channels of differing thickness on either side, as shown in the CFD model of Figure 5.1.

After assembly, the profile of the flow channels is mapped using an optical laser system. For the example test outlined here, the expected channel gaps were 80 mils and 100 mils. As can be seen from Figure 5.2, the actual channel gaps vary slightly along the width of the plate, with average values of 77.86 mils and 101.70 mils. Additionally the plate thickness was measured to be 31.3 mils, a slight deviation from the designed value of 32 mils.

Once the channel gaps are mapped, the flow loop is used to move water through the test section at a variable flow rate. Flow is started at a minimum value, increased gradually to its maximum, and then returned to the minimum. During the test, the data shown in Figure 5.3 is collected on the thickness of the channel gap at the midpoint of the leading edge, where the maximum deflection is expected. Additionally, the pressure drop over the length of the plate is recorded for later comparison to numeric and analytic models.

While the elastic deflections shown in current tests are significant, it is important to realize that these tests utilize wide, flat plates with no support comb. Future tests will investigate the effect of plate curvature and comb, and should yield significantly less deflection.

6. RELICENSING REQUEST FOR ADDITIONAL INFORMATION

The MURR was licensed by the U.S. Nuclear Regulatory Commission (NRC) in October 1966 to operate a maximum thermal power level of 5 MW. This license was based on the NRC's review of the Hazards Summary Report (HSR) [July 1965], and Addendum 1 [February 1966] and Addendum 2 [May 1966]. In 1974, after thermal-hydraulic and instrumentation enhancements to the reactor coolant systems were performed, the MURR was relicensed to operate at 10 MW based on the NRC's review of Addendum 3 [August 1972], Addendum 4 [October 1973] and Addendum 5 [January 1974] to the HSR. After 40 years of operation, the MURR operating license was set to expire in October 2006.

On August 31, 2006 the MURR submitted a request to the NRC to renew Amended Facility Operating License No. R-103 for an additional twenty years of operation. On May 6, 2010, and then again on June 1, 2010, the MURR received a total of 186 technical questions [also referred to as Request for Additional Information (RAI)] to assist the NRC staff in determining if the renewal application met the requirements of the regulations, in particular the regulations in Title 10 of the Code of Federal Regulations (10 CFR) Parts 20 and 50. The questions were based on a review of the renewal application using the NRC staff's standard review plan in NUREG-1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Part 2 [6].

Many of the questions focused on the material presented in Chapter 4, "Reactor Description," and Chapter 13, "Accident Analyses," of the MURR Safety Analysis Report (SAR) that was submitted as part of the renewal application. Newer, more detailed neutronics and thermal-hydraulic analyses, which were benchmarked against MURR HEU measured values, were used to support the LEU conversion feasibility analysis. The 2009 Fuel Conversion Feasibility Study [4] produced perhaps the best benchmarked modeling of the MURR core since original startup in 1966. This benchmarked modeling greatly aided the MURR staff in answering many of the relicensing questions, especially those regarding Chapter 4. Similarly, answering the NRC's questions will help in identifying details of analyses needed for the LEU 12 MW conversion SAR.

7. LICENSING CHALLENGES IN A POWER LEVEL UPRATE FROM 10 TO 12 MW

An uprate in power level to 12 MW using LEU fuel is required in order to preserve approximately the same level and spectrum of key neutron fluxes as the current 10 MW HEU fuel provides. It has been previously calculated that the flux and reaction rate losses would vary between 12 to 15% if the current power level of 10 MW is maintained with LEU fuel.

Under 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," the MURR is classified as a non-power reactor. 10 CFR 50.2 defines a non-power reactor as a research reactor or *testing facility* licensed under §§ 50.21(c) or 50.22 of Part 50 for research and development. The MURR is currently licensed as a research reactor.

10 CFR 50.2 defines a *testing facility* as a nuclear reactor which is authorized to operate at:

(1) A thermal power level in excess of 10 megawatts; or

(2) A thermal power level in excess of 1 megawatt, if the reactor is to contain:

(i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments; or

(ii) A liquid fuel loading; or

(iii) An experimental facility in the core in excess of 16 square inches in cross-section.

The proposed MURR power uprate to overcome the conversion flux penalty will result in a thermal power level greater than criterion (1). The following three regulatory paths or options are available for the MURR to uprate in power and be licensed at 12 MW:

(1) Apply for an *exemption* from the *testing facility* classification based on the need to increase power in order to maintain the same performance as the current HEU fuel provides; or
(2) Petition a *rule making change* to revise the definition of a *testing facility*; or
(3) *License the facility as a testing facility*.

Preliminary informal discussions with the NRC have been held and it appears that the exemption from the *testing facility* classification is the most logical uprate licensing path for both the NRC and MURR, since the only reason for the specific proposed uprate is to maintain rather than extend the current capabilities of the MURR after conversion. A public meeting is scheduled for November 2011 to discuss what is required going forward, including regulatory and Licensee actions, review process and schedule.

8. SUMMARY AND FUTURE WORK

Good progress continues towards the conversion of MURR from HEU to LEU fuel. The PSAR [1] methodologies and scenarios for conversion describe the set of experiments and analyses that will be performed. Discussion of HEU safety analyses are presented alongside requirements for LEU analyses, where changes for conversion are indicated. Fuel element design parameters, associated tolerances, and fuel specifications are summarized. Methodologies are discussed for the scenarios and methods planned for the LEU safety analysis. A transitional LEU fuel cycle has been identified that will provide the MURR users the required/desired flux profiles while the traditional fuel cycle inventory of fuel element power histories from new to almost end-of-life is built up. This will require only lightly borated side plates in twelve LEU fuel elements.

It is important to note that the U-10Mo monolithic LEU fuel is not yet qualified or commercially available. The FD and FFC efforts within the GTRI Conversion Program are both working to clarify the fuel specifications that will be supported for the new LEU fuel. A goal of the PSAR is to facilitate the FD and FFC in clarifying the MURR specific fuel specification requirements.

The positive feasibility results reported in Reference 4 were predicated on the best information available in September 2009. Sufficient excess reactivity to maintain operating lifetime with LEU fuel has been obtained by increasing the water-to-metal ratio in the core by thinning the fuel plates and increasing the coolant channel widths. Plates 3 through 23 were decreased from a 50 mil thickness to 38 mils by designing an 18 mil U-10Mo foil with 10 mil cladding (including any Zr interlayer to control fuel swelling behavior). Reduced foil thicknesses have been designed in three of the fuel plates in order to control power peaking and assure safety margins. It is not yet clear whether the combined 10 mil cladding thickness will prove too difficult or expensive to fabricate. Furthermore, experiments and analyses to prove the hydrodynamic stability of the thin 38 mil thick fuel plates must still be completed. Should thicker cladding and/or a stiffer plate be required, then the inherent penalty of displacing moderating water will need to be addressed to prove technical feasibility of an alternate fuel design. Work on a preliminary contingency design for thicker fuel plates began last year and is continuing, in parallel with safety analyses of the current LEU design.

The LEU feasibility analyses included HEU benchmark modeling that has facilitated answering the NRC's request for additional information received in May and June 2010 regarding MURR's relicensing application that was submitted in August 2006. Similarly, answering the NRC's relicensing questions has identified details of analyses that need to be performed for the LEU 12 MW conversion SAR.

Furthermore, acceptable experimental fluxes will only be maintained if the reactor power can be increased from 10 to 12 MW in order to offset the inherent penalty of introducing more 238U into the core. Feasibility studies to date have indicated that safety margins will be maintained with LEU fuel operated at 12 MW. The power uprate will be modeled in the safety analyses. Efforts to address the regulatory issues of the uprate are ongoing to assure successful conversion on the GTRI schedule.

Finally, we must also note that the economic feasibility of conversion cannot be declared until commercial availability of the fuel has been developed, including credible fuel cost projections. The MURR understands that GTRI is committed to addressing fuel cost increases due to a conversion from HEU to LEU fuel. The MURR will continue to work within the GTRI Conversion Program to assist the FFC in development of cost models and/or to pursue redesigns (as possible) once key cost factors are better understood.

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3.3 SAFARI-1 Research Reactor Beryllium Reflector Element Replacement, Management and Relocation

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The beryllium (Be) reflector elements of the SAFARI-1 Research Reactor were replaced in October 2011 as part of the Ageing Management Programme of the reactor. After more than three million MWh of operation over a period of 47 years, core reloading became more difficult due to the geometric deformation of the beryllium reflector elements. During the replacement of the reflector elements, criticality and reactivity worth experiments were performed and found to compare favorably with calculated values.

A Beryllium Management Programme was established at SAFARI-1 to identify and apply effective and appropriate actions and practices for managing the ageing of the new beryllium reflector elements. This will provide timely detection and mitigation of ageing mechanisms relevant to beryllium reflector elements, supporting the life extension of these elements. These actions and practices include monitoring of the tritium levels in the primary water, calculating and measuring the fluxes within the beryllium reflector positions, measuring the straightness of the elements to track geometric deformation and visually inspecting the reflector elements for crack formation. Acceptance criteria indicating the end of life of the elements were established. These criteria take into account the smallest gap that could exist between elements, sudden changes in the tritium levels and formation of cracks. All the data obtained through the Beryllium Management Programme are recorded in a database.

Additional benefits gained through a Beryllium Management Programme are the availability of a complete irradiation history of the beryllium reflector elements at any point in time and the establishment of a knowledge base to assists in the understanding of the behavior of the beryllium reflector elements in an irradiation environment.

Straightness baseline measurements of the new beryllium reflector elements were performed with a beryllium straightness measurement tool, designed at SAFARI-1. The decommissioned beryllium elements were also measured to investigate the occurrence of ageing mechanisms and their consequences, and to determine the location of the element's vulnerable areas. This supports the Beryllium Management Programme in terms of an optimal shuffling scheme to mitigate the effects of ageing.

Currently, the decommissioned beryllium reflector elements are stored in a storage rack within the reactor pool. Due to limited space it is necessary to relocate these elements. A proposed relocation strategy based on existing literature is discussed. This strategy covers the requirements, relocation process and types of storage to be considered.

Keywords: Beryllium reflector elements, ageing management, replacement, relocation, disposal.

1. INTRODUCTION

SAFARI-1 research reactor is characterized as a multi-purpose tank-in-pool reactor with a thermal power of 20 MW. This facility is a high neutron flux reactor that is light water-moderated and cooled as well as beryllium and light water reflected. The reactor has been in operation for 47 years and is rated by the IAEA as a high utilized reactor, with more than three million MWh of operations over this period. SAFARI-1 contributes largely to South Africa's scientific and technical growth and provides important services of industrial interest (e.g. isotope production for medical and industrial use).

The facility was designed to have an operational life of at least 40 years. Due to commercial interest it is important to invest in the life extension of SAFARI-1, to ensure safe and continual operation.

Currently, an Ageing Management Programme is underway at SAFARI-1 and part of this programme is the replacement of the beryllium reflector elements. Ageing implies the anticipated or actual degradation of the functionality of a system, structure and component (SSC). Therefore, activities such as repair, refurbishment, optimization and replacement of SSCs over the lifetime of the reactor are necessary to ensure the functionality of the SSC is maintained.

As per [10], effective ageing management requires the use of an ageing management methodology. This methodology entails the detection and evaluation of ageing degradation of an SSC as a consequence of service conditions. It also involves the application of countermeasures for prevention and mitigation of ageing degradation.

Ageing management principles from both research reactors and nuclear power reactors [9] are used to manage the ageing of the beryllium reflector elements at SAFARI-1. These principles are discussed in this paper.

2. SAFARI-1 REFLECTOR DESCRIPTION

The function of beryllium reflector elements in the SAFARI-1 reactor is to enhance the neutron density by reducing neutron leakage. The current decommissioned beryllium reflector elements were used for 47 years of reactor operations. During this period the reflector elements were never replaced. Figure 1 presents the cross section of the SAFARI-1 reactor core. The shaded area is a representation of the beryllium reflector elements in the different core positions. There are 19 reflector elements in the core situated at the south, east and west faces of the core as presented in Figure 1. The reflector elements have a rectangular shape, with dimensions 80x76x924mm. Aluminum adaptors are located on both ends of the elements for handling purposes and to fit the element into the grid plate. Certain hollow elements (e.g. position A4) are not filled with beryllium plugs to allow for irradiation of samples. Ten beryllium plugs are used to fill a hollow beryllium reflector element. The weight of a single solid element is about 9 kg.

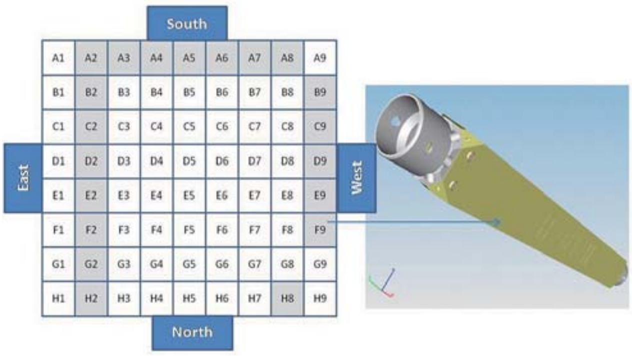

Figure 1: Beryllium Core Map and Element

3. CORE LOADING DIFFICULTIES

In November 2010 the beryllium replacement project became high priority, due to core reloading becoming more difficult as a result of the geometric deformation of the beryllium reflector elements. The beryllium reflector elements that caused core loading difficulties were those in core positions F2, A4 and H8. These elements were located in the high fast fluence positions during that particular time. The beryllium reflector element, A4, is presented in Figure 2.

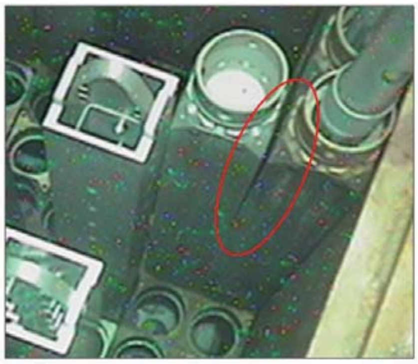

Figure 2: Beryllium Elements in SAFARI-1 Core

To ensure the continual and safe operation of the SAFARI-1 reactor, ten solid elements, ten hollow elements and 100 plugs were purchased from the beryllium manufacturer Materion, previously known as Brush Wellman Company. The new beryllium reflector elements were commissioned during October 2011. The procedure for replacement of the beryllium reflector elements is described in Section 3 of this paper.

4. BERYLLIUM REPLACEMENT

Neutronic calculations were performed, as per [7],with a typical core configuration to ensure that it is safe to replace all the beryllium elements at once. The neutronic assessment was based on an estimation of the neutron poison content of the beryllium reflector elements over the full irradiation history. Results indicated an insignificant change of the neutronic parameters between the two core configurations (old beryllium reflector elements vs. new beryllium reflector elements). However, to confirm this assessment in practice an experiment with the following scenarios was performed during the replacement to determine the change in reactivity worth and flux:

Scenario 1: Obtain a stable critical bank at low power for at least 15 minutes with the previous core cycle configuration. This entails using the fuel from the previous core cycle and the old beryllium reflector elements.

Scenario2: Increase the reactor power to 2 MW for 30 minutes and then shut the reactor down. Cobalt and nickel foils were irradiated within certain beryllium positions to obtain the fluxes.

Scenario3: Obtain a stable critical bank at the same power as scenario 1 for at least 15 minutes with the previous core cycle configuration and two old beryllium reflector elements replaced with two new beryllium reflector elements. After 15 minutes the reactor is shut down.

Scenario4: : Obtain a stable critical bank at the same power as scenario 1 and 2 for at least 15 minutes with the previous core cycle configuration and all except two of the beryllium reflector elements replaced with new beryllium reflector elements. Two positions retained their old hollow beryllium reflector elements due to the RINGAS system installed in it. These remaining two old elements will be replaced as soon as the required maintenance on the RINGAS system can be done.

Scenario5: Increase the reactor power to 2 MW for 30 minutes and then shut the reactor down. Cobalt and nickel foils were irradiated within certain beryllium positions to obtain the fluxes.

The values for the fluxes are not yet available and for this reason are not discussed in this paper. The results for the different scenarios are given in Table1. These scenarios were simulated with the Monte Carlo N-Particle (MCNP) code, as per [6]. The MCNP results for the reactivity worth of the scenarios as well as the actual values are given in Table 1.

Table1: Results of Experiments during Replacement

Scenarios	**Actual**	**Reactivity worth (MCNP)**
Scenario 1		
Critical Bank (cm)	624.0	
Scenario 3		
Critical Bank (cm)	Rods 1-4 & 6: 624.0	
	Rod 5: 604.5	
Reactivity worth	0.15 $	0.12 $
Scenario4		
Critical Bank (cm)	Rods 1-4 & 6: 624.0	
	Rod 5: 516.7	
Reactivity worth	0.55 $	0.60 $

Considering Table 1, there is a small acceptable difference between the calculated and actual values. Performing these scenarios verified that the results presented in [7] are correct. The reactor was then taken to full power.

5. BERYLLIUM MANAGEMENT PROGRAMME

A Beryllium Management Programme, as per [1], is established at SAFARI-1 to ensure:

- Availability of a complete irradiation history of the beryllium elements at any point in time,
- The establishment of a knowledgebase to enhance the understanding of the behavior of beryllium in an irradiation environment,
- The extension of the lifetime of the beryllium elements.

As part of the Beryllium Management Programme a methodology is proposed that involves the application of countermeasures for prevention and mitigation of the ageing degradation of the beryllium reflector elements. The ageing mechanisms to consider when formulating such a methodology are changes to the material properties due to neutron irradiation, motion, fatigue or wear and corrosion. Neutron irradiation induces swelling and bending/bowing of the reflectors. Fatigue or wear may cause changes such as displacement, material damage, deformation, deterioration of surfaces and changes in the dimensions.

The methodology is based on Deming's quality approach that entails to plan, do, check and act. This approach is presented in Figure 3.

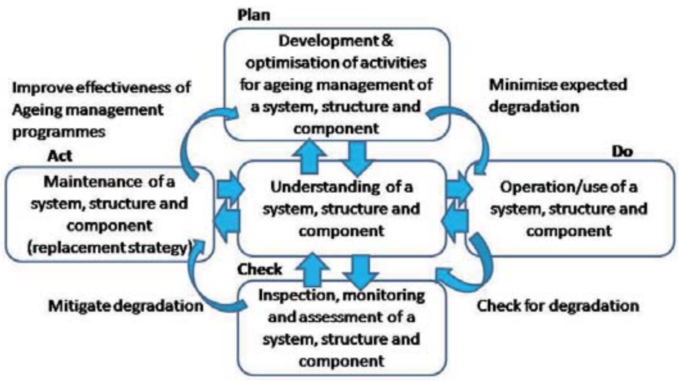

Figure 3: Deming's Approach ([9])

As part of this methodology putting in place measures for the timely detection and mitigation of the degradation of the elements necessitates the establishment of certain practices and actions. This forms part of the check and act of Deming's approach. The practices entail:

- monitoring of the tritium levels in the primary water,
- calculating and measuring the fluxes within the beryllium reflector positions,
- measuring the straightness of the elements to track geometric deformation, and
- visually inspecting the reflector elements for crack formation.

Through recording these results on a database effective evaluation of the ageing degradation of the reflectors can be achieved. The actions require rotation and shuffling to ensure minimization of ageing degradation of the elements, this is described in section 5.5.

5.1 Geometric Deformation Measurements

Geometric deformation such as swelling and bowing of the reflector elements are measured with a beryllium straightness measurement tool as presented in Figure 4.

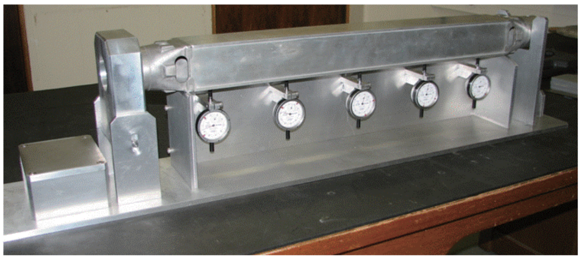

Figure 4: Beryllium Straightness Measurement Tool

The element is measured two meters under water in a horizontal position. The measurement tool has five modified dial gauges. The modification ensures no leakage under water by means of ensuring the pressure within the gauge is equalized with the environmental pressure. Five measurements are taken along the length of the element to establish whether the element has swelled or bowed. All four faces of the beryllium elements are measured. Tool calibration is performed before any measurements are conducted with a calibration element. The reasons for using a mechanical method instead of electronic instrumentation to measure the straightness of the elements are to reduce the cost of the apparatus, to reduce the risk of having obsolete instruments in future and to have no interference with the instruments in a flux environment. Baseline measurements of all the new beryllium reflector elements were performed. These measurements are recorded within the database for future reference. All straightness measurements of the beryllium reflector elements are compared with the baseline measurements to track the geometric deformation over time.

To reduce handling, the element in the highest fast fluence position is to be measured every twelve months. This is the reference element and due to the high fast fluence exposure it is expected to show signs of geometric deformation first. When nearing the fast fluence of $2x10^{22}$ $n.cm^{-2}$ the measurement frequency may need to be increased. Straightness measurement of the beryllium reflector element located in the experimental facility (A4 position) is to coincide with maintenance and replacement of the RINGAS system.

It is necessary to have acceptance criteria for the amount of geometric deformation. A thermal hydraulic calculation needs to be performed to establish the minimum allowable gap between a fuel element and a beryllium element that still ensures effective heat removal. It is also needed to investigate how much an element can bend or swell before preventing core loading. Using these results a replacement criterion can be established.

5.2 Flux Calculations and Measurements

Flux measurements by means of Co or Ni foils are to be conducted every six months in the identified positions. For each cycle the flux map is kept on record to maintain the history of each beryllium reflector element. The measurements and calculations need to be compared to validate and verify the results.

5.3 Tritium measurements

Samples of the primary water are taken each month and are analyzed to obtain the tritium concentration. It is important to take into account that the tritium concentration in the primary water may be due to the release of fission products from fuel elements or from the beryllium reflector elements. According to [5], test performed showed that 99% of tritium was retained in the beryllium reflector samples of the ATR even after long anneals at temperatures less than 511 degrees Celsius. Bursts of tritium are only released at high temperatures (> 600 degrees Celsius). SAFARI-1 is operated at low temperatures of less than 125 degrees Celsius. Large volume tritium releases from SAFARI-1s beryllium reflector elements are therefore not expected considering the operating temperature. As discussed in [5], the mechanisms of high volumes of tritium retention in beryllium reflectors may be due to internal and/or external oxide concentrations and/or He defects/bubbles.

An important factor to investigate is the release of tritium due to damage during handling. This programme may help to give some insight on this factor.

Table 2 gives the tritium concentration over a four month period. The first three samples were conducted before the beryllium reflector elements were replaced. The last sample was taken shortly after the beryllium reflector elements were replaced.

Table 2: Measured Concentrations for Tritium

Sample	**Concentration (kBq/l)**
1	271+/-4
2	189+/-3
3	225+/-4
4	249+/-4

No noteworthy change in the tritium concentration is observed when the new beryllium reflector elements were commissioned. One of the reasons may be that the decommissioned beryllium reflector elements are still in contact with the primary pool water. For this reason the source of the tritium (fuel or beryllium reflector elements) can only be established once the decommissioned reflector elements are removed from the reactor pool and relocated to the selected storage area.

5.4 Visual Inspections

Visual inspections using an underwater camera are conducted twice a year whenever the elements are removed or handled during reactor shutdown periods. Any cracks or deposits on the beryllium reflector elements are recorded and tracked over time. The increase in concentration of the tritium levels and crack formation should be monitored and the correlation noted.

5.5Rotation Criteria

Initially, the core is loaded so that the identifier (number) of each element is directed to the west side of the core. When the elements are unloaded during a shutdown all elements shall be replaced in the same configuration and with the same rotation as before unloaded.

All the elements shall be rotated 180 degrees after every 11 cycles. This will ensure an even exposure to fast fluence on all sides of the reflector elements over the existing lifetime of the reactor.

At this point in time no shuffling is considered in order to reduce handling.

6. MEASUREMENTS

As part of the knowledgebase, to aid in the understanding of the behavior of the reflector elements within a radiation environment, the geometric deformation of five decommissioned elements where measured. Four hollow elements and one solid element were measured. Measurements were performed with the measurement tool presented in section 5.1. All four sides of the beryllium reflector elements were measured. The two most severe cases from the five measurements are presented in this paper. Figure 5 presents the deviation from zero for a hollow beryllium reflector element. This particular bowed beryllium reflector element closes the gap between two adjacent elements.

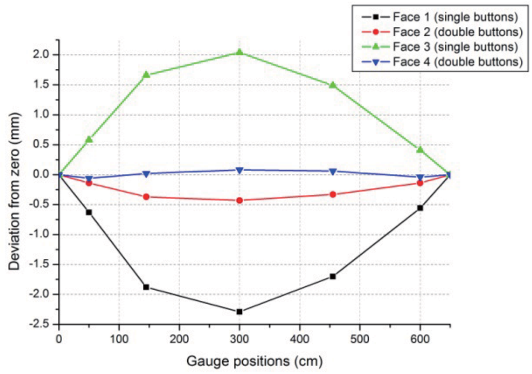

Figure 5: Deviation from Zero for a Hollow Beryllium Element.

Figure 6 presents the amount of swelling of this hollow element.

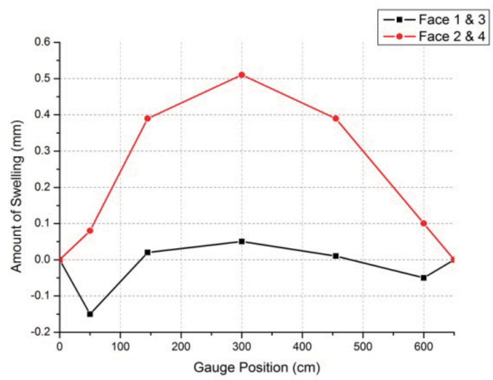

Figure 6: Amount of Swelling for a Hollow Beryllium Element

Figure 7 presents the deviation from zero for a solid beryllium reflector element. As with the previous case this beryllium reflector element closes the gap between two adjacent elements.

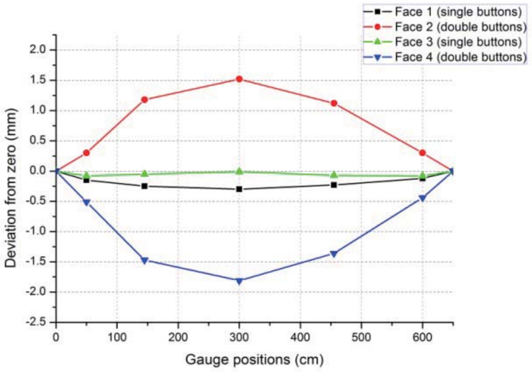

Figure 7: Deviation from zero for a Solid Beryllium Element.

Figure 8 presents the amount of swelling of the solid beryllium reflector element shown in Figure 7.

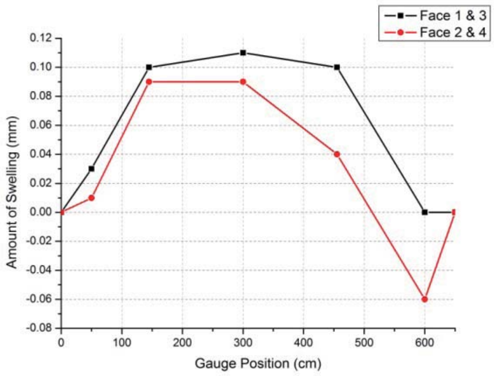

Figure 8:Amount of Swelling of the Solid Reflector element

As it is not known where these elements were located in the core during the last 47 years of operations, it is difficult to make any conclusive remarks other than swelling and bowing did occur during the lifetime of the reactor. It is therefore imperative to have a Beryllium Management Programme in place to manage and record the position, orientation and exposure of the beryllium reflector elements and so ensure the early detection and mitigation of ageing mechanisms of the reflector elements to enhance the continual and safe operations of the reactor.

7. DISPOSAL OF DECOMMISSIONED BE

Currently, SAFARI-1s decommissioned beryllium reflector elements are stored in an aluminum storage rack within the reactor pool as presented in Figure 9.

Figure 9: Current Beryllium Storage

The decommissioned beryllium elements cannot be stored in the reactor pool indefinitely, and for this reason the disposal path needs to be identified. According to the IAEA's position paper on waste disposal [4], it is essential to have a plan in place for the disposal of waste and in this case the decommissioned beryllium. The reason being, that beryllium waste is a liability for the facility due to financial and environmental implications. This section describes this plan.

A proper disposal path can only be determined through characterizing the beryllium waste (decommissioned beryllium reflector elements). Waste characterizing entails obtaining the radionuclide inventories (e.g. transuranic isotopes, ^{14}C, ^{60}Co and ^{3}H) and shielding requirements. It is also essential to investigate the amount of possible heat generated by the reflector elements, change of inventories due to the 'out of reactor effect' described in [3] and the typical hazards presented by beryllium.

The disposal plan entails characterizing the waste through calculations and measurements that is described in Figure 10: Waste Characterizing. Figure 10.

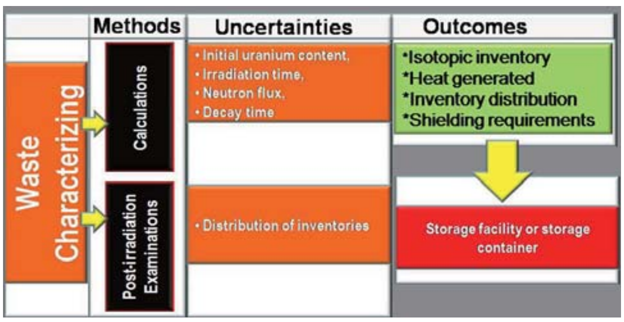

Figure 10: Waste Characterizing.

7.1 Calculation Approach to Characterize Waste

In order to characterize waste by means of calculation methods, uncertainties such as the initial uranium content, irradiation time, neutron flux and decay time of the beryllium reflector elements need to be addressed. These uncertainties are the basis for the following input that is required to make informed decisions regarding the disposal path of the beryllium reflector elements:

- build-up of isotopic inventories in the elements while used in the reactor,
- transmutation of uranium into transuranic isotopes,
- half-life of long lived radio isotopes.

Inventories are to be calculated with the ORIGEN or FISPACT code.

In support of obtaining samples for post-irradiation examinations that are representative of the distribution of the isotopic inventory of the elements, it is required to calculate how homogeneous the inventory distribution is.

The initial uranium content present in the decommissioned reflector elements is a major uncertainty due to not knowing who the supplier of the beryllium reflector elements was. Sources of beryllium ore may differ from manufacturer to manufacturer. For this reason the elemental composition of the beryllium reflector elements could vary depending on who the supplier was. Hence, it is essential to do a literature survey to obtain representative beginning of life elemental compositions of the beryllium reflector elements. According to [3], manufacturers of beryllium reflector elements in the United States (US) during the early sixties were Kawecki Berylco and Brush Wellman Company. Also, for that period most of the beryllium ore material used by these two suppliers originated from Brazil.

SAFARI-1 was commissioned in 1964 by a US based company Allis Chalmers Nuclear Manufacturers. Due to the political turmoil during that era it is assumed that Allis Chalmers would not have used Russia (Kazakhstan) as beryllium reflector suppliers. Therefore, it is assumed that either Kawecki Berylco or Brush Wellman supplied the beryllium reflector elements that were commissioned by Allis Chalmers at SAFARI-1.

Reactors in the US such as the Advanced Test Reactor (ATR), Material Test Reactor (MTR) and Engineering Test Reactor(ETR)were also supplied with beryllium reflector elements by the two mentioned companies. Idaho National Engineering and Environmental Laboratories have done extensive work to characterize irradiated beryllium reflector elements buried in the subsurface disposal areas from these three reactors.

The measured elemental composition data and manufacturer data sheets of the beginning of life impurities of the beryllium waste of these reactors are presented in [3]. It is showed here that the beryllium reflector elements contain uranium as impurity.

Initially, SAFARI-1 made use of [2] as the beginning of life impurity content to calculate the core neutronic parameters for the replacement of the beryllium reflector elements, as described in section 5 of this paper. According to [3], this impurity content is inconclusive due to uranium not being present. Having a comprehensive elemental composition is crucial to do proper waste characterization. It is therefore, necessary to redo all the initial calculations performed using [2]. It is proposed to use the elemental composition given in [3] as input to obtain a more representative inventory.

With respect to the other uncertainties mentioned such as the irradiation time, neutron flux and decay time it is necessary to take into consideration that no formal Beryllium Management Programme were in place during the 47 years of operation. A representative inventory may be calculated by:

- Using [3] and available operating history of the reactor as input,
- Assuming the worst case scenario of no shuffling of the reflector elements over 47 years,
- Taking into account the time that the reflector elements have been removed from the core.

7.2 Measurement Approach to Characterize Waste

Post-irradiation examinations of the decommissioned beryllium reflector elements are to be performed at Necsa. Samples of selected beryllium reflector elements are to be prepared and analyzed. The calculations will provide valuable input on how homogeneous the distribution of the inventory is along the length and width of the element.

With this information it can be established whether a cutting or drilling tool is required or only surface shavings to obtain representative samples. This tool needs to be operated either in a hotcell or under water.

7.3 Storage Possibilities

Underground or surface storage may be

considered for disposal of the decommissioned beryllium reflector elements. The storage areas or containers that were considered for interim or final storage are:

- Vaalputs waste repository. This is an off-site low/medium level underground waste storage facility.
- Pipe storage facility at Necsa, which is a sub-surface dry storage facility.
- Surface storage container manufactured in-house or by a supplier.

The requirement for waste to be stored at Vaalputs repository is an activity of less than 10.8 nCi/g. Using [3], transuranic waste has an activity of more than 100 nCi/g. For this reason it is not possible to store the beryllium reflector elements at Vaalputs.

The pipe storage facility is used to store SAFARI-1s spent fuel. Storing decommissioned beryllium reflector elements at this facility will require extensive safety analysis. It is necessary to prove that the storage of beryllium reflector elements in the pipe store will cause no criticality accidents. A major change to the current nuclear license would have to be done if this storage option is considered. For these reasons other storage options also need to be considered.

Surface storage may be the most plausible method of disposal of the decommissioned beryllium reflector elements at SAFARI-1.The main functions of a surface storage container are to contain the decommissioned beryllium reflector elements and to protect the employees, equipment and environment from chemical toxicity as well as radiation. To design a storage container that perform these functions it is necessary to consider the following influencing factors:

- The material type used for the container should be corrosion resistant, radiation tolerant and shield the environment against radiation
- The container should be gas tight
- The container should be resistant to fire
- Retrieval of waste should be possible
- Required lifespan of the cask (taking into account the half-life of the long lived radio isotopes).

These influencing factors are described in more detail in the following paragraphs.

Unlike underground storage, a surface storage container needs to be actively under surveillance to detect and mitigate deficiencies of the container and the contents. It is therefore a necessity to have a maintenance and In-Service-Inspection (ISI) programme in place.

During irradiation there is a build-up of gas within the reflector elements. High volumes of gases such as ^3H and helium are formed within the reflector elements, as per [7]. As discussed in section 6.3 these gases may diffuse out of the elements under certain conditions. Another hazard posed by the reflector elements is the release of beryllium metal particles into the environment during breakage or when exposed to fire, as stated in reference [8]. In order to prevent leakage of hazardous gasses or beryllium metal particles into the environment it is required to have a gas tight container that is fire resistant.

The material type of the container needs to be non-corrosive. It is also required that the container is designed such that the contents do not come into contact with water or air in order to prevent degradation of the beryllium reflector elements.

The container shall provide radiological shielding for the employees and the environment. To obtain the shielding requirements for the container it is proposed that shielding analysis is performed with either MicroShield or MCNP. To prevent degradation of the container it is required that the material type is radiation tolerant. The material type of the container should ensure effective heat transfer. Internal source of heat is the reflector elements and the external source may be e.g. the sun if the container is located outside. If heat cannot be removed efficiently from the cask, it is recommended to explore active coolant systems.

It is also required to enable sampling; hence it should be possible to retrieve elements from the container for visual inspections and regular gas samples should also be taken. Allowing retrieval of the elements will ensure the possibility of beryllium recycling in the future.

Waste stores are vulnerable to accidental and deliberate intrusion by personnel and non-personnel. It is therefore necessary to keep the storage container under close security surveillance according to [4].

8. CONCLUSIONS

It was proven through straightness measurements, by means of an in-house designed beryllium measurement tool that bowed and swelled elements caused the core loading difficulties during November 2011.

In order to manage the ageing of the beryllium commissioned during October 2011, the ageing management IAEA safety guides of both research reactors and nuclear power plants were consulted.

Nuclear power plant lessons learnt as well as the ageing management matrix in [10] are the foundation for the Beryllium Management Programme. Benefits to be gained from this programme are the knowledge obtained from the behaviour of beryllium reflector elements within an irradiation environment as well as the life extension of the elements.

Possibilities of disposal of the decommissioned beryllium reflector elements were discussed in this paper. It is essential for SAFARI-1 to characterise the beryllium waste in order to establish a proper disposal path. Extensive calculations and post-irradiation examinations need to be performed for this. With current knowledge, the most feasible choice to consider appears to be the use of a surface storage cask on Necsa site.

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3.4 MATERIAL TEST REACTOR FUEL RESEARCH AT THE BR2 REACTOR

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Abstract - The construction of new, high performance material test reactor or the conversion of such reactors' core from high enriched uranium (HEU) to low enriched uranium (LEU) based fuel requires several fuel qualification steps. For the conversion of high performance reactors, high density dispersion or monolithic fuel types are being developed. The Uranium-Molybdenum fuel system has been selected as reference system for the qualification of LEU fuels. For reactors with lower performance characteristics, or as medium enriched fuel for high performance reactors, uranium silicide dispersion fuel is applied. However, on the longer term, the U-Mo based fuel types may offer a more efficient fuel alternative and-or an easier back-end solution with respect to the silicide based fuels.
At the BR2 reactor of the Belgian nuclear research center, SCK•CEN in Mol, several types of fuel testing opportunities are present to contribute to such qualification process. A generic validation test for a selected fuel system is the irradiation of flat plates with representative dimensions for a fuel element. By flexible positioning and core loading, bounding irradiation conditions for fuel elements can be performed in a standard device in the BR2. For fuel element designs with curved plates, the element fabrication method compatibility of the fuel type can be addressed by incorporating a set of prototype fuel plates in a mixed driver fuel element of the BR2 reactor. These generic types of tests are performed directly in the primary coolant flow conditions of the BR2 reactor. The experiment control and interpretation is supported by detailed neutronic and thermal-hydraulic modeling of the experiments.
Finally, the BR2 reactor offers the flexibility for irradiation of full size prototype fuel elements, as 200mm diameter irradiation channels are available. These channels allow the accommodation of various types of prototype fuel elements, eventually using a dedicated cooling loop to provide the required thermal and hydraulic conditions. The availability of a comprehensive set of post irradiation examination facilities on site complements the versatile BR2 reactor to provide a set of high performance tools for MTR fuel qualification.

Keywords: test reactor fuel, qualification experiments.

1. INTRODUCTION

The primary function of research reactors is to produce excess neutrons to experiments, either inside the reactor vessel as for material test reactors or outside the reactor vessel as for neutron beam research reactors. To perform this function in the most efficient way possible, a maximum fission density and a minimum parasitic absorption of neutrons is required, with efficient thermalisation of the neutron flux and cooling of the fuel. As a consequence, the majority of test reactors has been conceived to be fueled with metallic, highly enriched uranium (HEU) based fuels, as this allows combining the requirements, cited above, in an efficient way. Metallic HEU based fuel combine a high density of fissile material with a low parasitic neutron absorption and good thermal properties, allowing high performance in a test reactor. Moreover, metallic matrix fuels offer a significant advantage from the fission product retention point of view.

The fuel designs differ from reactor to reactor (see Figure 1), but can be classified as pin- or plate-type. Most research reactors, especially the high power ones, use metallic plate-type fuel, in which thin plates (flat or curved), containing fissile material, are arranged in assemblies (box- or tube-type or even more complex). Most of the fuels are dispersion fuels in which the fissile material is dispersed in a matrix, mostly Al (a few exceptions exist to this, eg. Zr, ZrH, SS, …), with an Al alloy cladding.

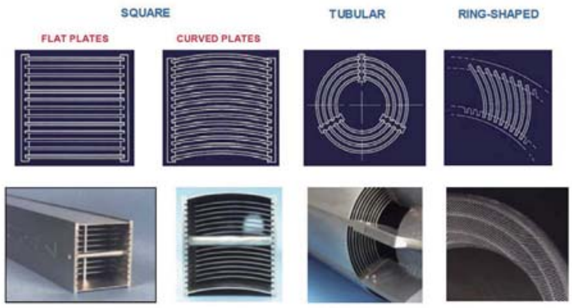

Fig 1. Examples of different research reactor fuel designs (courtesy of AREVA-CERCA)

Because of the proliferation risks associated to the high enrichment of the fissile material used, the US-DOE has started a program for Reduced Enrichment for Research and Test Reactors (RERTR) in 1978 to reduce and eventually eliminate the civil use of high-enriched uranium [1]. A main goal of this program was and is to develop the technology, including fuels, required to convert research reactors from High Enriched Uranium (HEU) to Low Enriched Uranium (LEU) based fuel [2]. This conversion should occur without significant loss of performance of the research reactors (neutron flux in key positions) and maintaining the economy of the fuel cycle of the reactor (reactor cycle length, extent of final fuel burn-up, fuel cycle cost). For the performance of the reactor, the essential goal is to increase the uranium concentration of the fuel to compensate for decreasing its enrichment so that the density of ^{235}U remains equal. This can be achieved by increasing the fuel to matrix ratio (typically 1.7 gU/cm^3 for HEU) or changing the fuel material to another material with a higher uranium density. For qualification of a new MTR fuel, demonstration of a satisfactory irradiation behaviour is an essential point to prove the feasibilityof the fuel system in terms of cladding integrity, coolability and dimnesional stability under irradiation at the desired power levels and burn-up.

As an alternative fuel material to HEU based Ual_x dispersions, U_3Si_2 and U_3Si based fuels were developed, with the latter only successful in pin-type fuel thanks to the more important constraints, which reduce swelling. Plates with U_3Si_2 dispersion in Al up to loadings of 4.8 gU/cm^3 have been tested and found stable under irradiation in plate-type configurations, which has eventually led to the NUREG-1313 Safety Evaluation Report issued by the U.S. Nuclear Regulatory Commission in 1988 [3], which approved the fuel for general use. Since then, a large number of research reactors have converted to LEU U_3Si_2 fuels, but some reactors are as yet incapable of moving away from HEU without significant loss in performance, due to the limitations of the density of ^{235}U in silicide based fuels. Therefor, the quest for alternative fuels for high performance MTRs continues, requiring a number of screening, qualification and demonstration tests.

The BR2 reactor of the Belgian nuclear research centre in Mol is one of the high performance material test reactors in Europe. Due to the high specific power rating of its fuel elements, it is not capable of converting its current HEU based UAl_x driver fuel to a qualified, LEU based silicide fuel. Therefor, SCK•CEN, as owner of the BR2 reactor, is engaged in a qualification programme of innovative LEU fuel of the dispersed U(Mo) type. This fuel type has the potential to achieve the required density for the high performance reactors ($8gU/cm^3$) [2]; addition of silicon to the matrix in which the U(Mo) is dsipersed may allow the suppression of undesired swelling of the fuel during irradiation at high power to high burn up. However, further optimalisation of the dispersed U(Mo) fuel type is required before its qualification for high performance MTRs can be done [4].

In this paper, the different types of tests that have been performed at the BR2 reactor to qualify new MTR fuels are discussed. These tests include the following:

- Plate testing: in these tests, geometrically identical plates are irradiated in a single basket. These tests allow the optimisation of the plate production process and the fuel metallurgy and demonstrate the feasibility of the selected fuel system at the represenative scale for the fuel element.
- Mixed element testing: especially for the reactors with curved or cilindrical fuel plates, the stability of curved plates in more stringent flow conditions than the flat plate irradiation is evaluated.
- Irradiation of prototype fuel elements: the BR2 reactor offers the possibility to load fuel elements or sections thereof due to the availability of 200mm irradiation channels in the reactor core, allowing the installation of elements, larger than its standard fuel element size.

2. General characteristics of the BR2 reactor

The BR2 reactor is a high performance material test reactor. The reactor core is constructed as a number of cylindrical irradiation channels, arranged in a hyperboloid of revolution. This geometry is shown in figure 2 and allows a high core density on one hand, while maintaining good accessibility on the other hand: the diameter of the reactor vessel at the top and bottom covers is about double the diameter at the core level. Each irradiation channel is materialized in a beryllium block with hexagonal cross section. This beryllium serve as moderator, together with the light water in the channels, which acts as coolant. The primary coolant is pressurized at 1.2MPa nominally; the inlet temperature is 40°C. Each irradiation channel is cooled separately by forced top down flow between the upper and lower plenum of the vessel. The nominal capacity of the primary coolant circuit is 100MW; the limit for the heat flux on the driver fuel plates is set at 470W/cm²; for experiments, a maximum heat flux of 600W/cm² can be allowed for the nominal cooling conditions of the BR2 primary circuit.

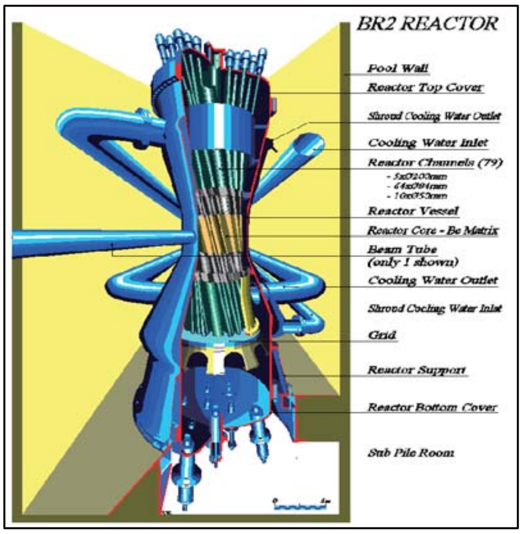

Fig.2. Reactor vessel internal structure of the BR2 reactor.

The core configuration of the BR2 reactor is flexible and tailored according to the experimental needs and safety requirements. Therefor, the irradiation conditions of several simultaneous experiments can be satisfied with large flexibility. A typical core configuration of the BR2 reactor is shown in figure 3.

The reactor core management is supported by 3 dimensional modeling of the neutron and gamma fluxes in the core, using an MCNP model. The evolution of the irradiation condition during the reactor cycle is predicted by linking a burn-up code (SCALE) to the MCNP model. This type of modeling also is applied for the design and evaluation of experiments.

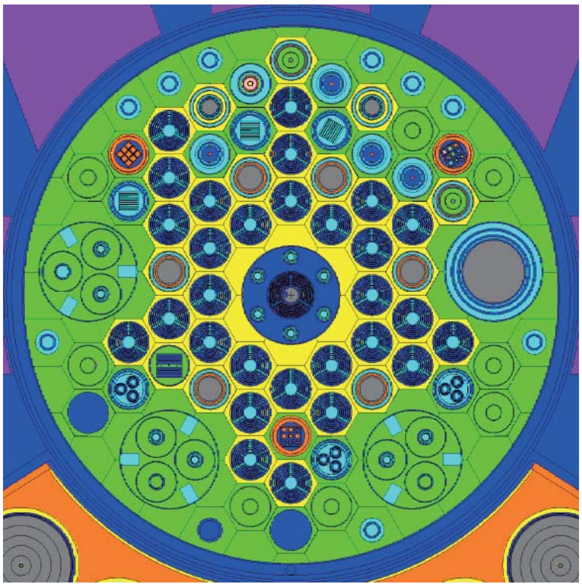

Fig.3. Typical reactor core configuration of the BR2.

3. Flat plate testing: FUTURE type basket irradiation in BR2

The FUTURE (Fuel test utility for research reactors) type basket is conceived to irradiate a number of geometrically identical fuel plates in primary water flow of the BR2 2 reactor. The in core section of the basket consists of a cylindrical body, fitting in a standard 84mm diameter irradiation channel of the BR2 core. Fuel plates are loaded in a rectangular cavity in the basket. The design of the irradiation basket is an iterative process to verify the safety criteria of the experiment:

- With the specified pressure drop over the core, the presence of the basket should not disturb the cooling of the neighboring fuel elements by creating a hydraulic bypass (higher flow than in a standard fuel element).
- With the given flux distribution in the selected irradiation channel, the power generated in the experiment should be limited in order not to exceed the maximum heat flux on the fuel plates.

The hydraulic design of this cavity is an iterative process: an initial design is established using a simplified hydraulic model to determine the flow rate of the coolant and the convective heat transfer coefficient. This calculation is validated by hydraulic testing of the basket or a mock up in a test loop, simulating the conditions of a standard BR2 irradiation channel (see figure 4). These hydraulic data are then combined with the typical flux profile (at first a cosine shaped approximation is used) to optimize the irradiation position for the basket (respecting the safety limits and the requirements of the experiment).

The final design of the rig is validated by detailed modeling of the thermal and hydraulic conditions, using three dimensional models and advanced calculation methods for the hydraulics (CFD type calculation) and neutronics (MCNP) for the exact power distribution. These calculations serve as final check for the selection of the irradiation conditions. The comparison between the simplified and detailed calculation of the plate surface temperature for the fuel plates in the FUTURE basket under irradiation, for the same peak power density value. The comparison between these results shows that the variation in peak plate temperature is small between the two methods.

The mechanical design of the irradiation basket has to provide means for loading and maintaining the basket in the reactor channel, as well as to be able to unload and load the fuel plates in between irradiation cycles for inspection under water. These requirements are met by combining two different coupling mechanisms, a first, standard coupling between the basket carrier rod and the basket coupling fork and a second one between the coupling fork and the basket itself. The fuel plates can be extracted under water when the basket is locked into the standard vertical transfer chariot and both couplings are removed.

The irradiation experiment for the selection of optimized metallurgical processing conditions of the full size U(Mo) fuel plates has been described in [5]. Four fuel plates have been irradiated during 3 reactor cycles in order to obtain a plate average burn up above 50%, with initial peak power density on the plate between 450 and 500W/cm². After the first irradiation cycle, the plates were inspected visually (figure 5) and leak testing was performed in order to detect cladding failure. Neither of these tests indicated failure of the fuel after the first irradiation cycle. After the second irradiation cycle, no cladding failure was detected, nor after the third one. Visual inspection after the third irradiation cycle did reveal blistering on 3 of the four plates. Non-destructive analysis in the hot-cell showed significant swelling or pillowing on all plates, resulting in a local thickness increase over 10% (figure 6). This result is likely to be linked to the insufficient inhibition of the interaction between the U(Mo) particles and the Al(Si) matrix material. Further analysis is ongoing and a follow-up experiment is planned to further optimize the metallurgical processing of the dispersed U(Mo) LEU based MTR fuel.

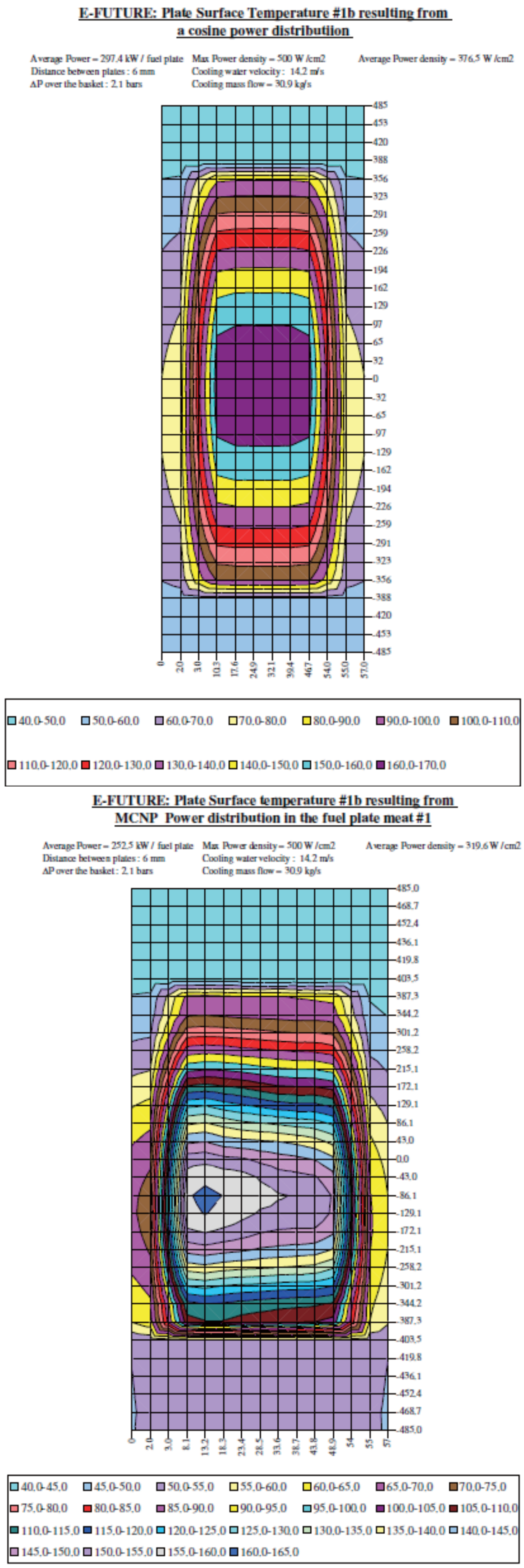

Fig.4: Comparison of the calculated surface temperature of a fuel plate in the FUTURE basket using the simplified (top) and detailed (bottom) method.

Figure 5: plate inspection between irradiation cycles in the FUTURE basket.

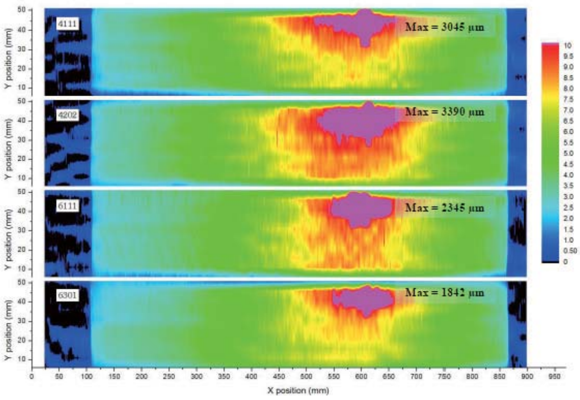

Figure 6: swelling of the U(Mo) fuel plates after completion of the irradiation in the FUTURE basket.

4. Curved plate testing: mixed element irradiation in BR2

The most direct approach to test cylindrically curved fuel plates of full-scale relevant dimensions is to incorporate the test plates in a standard fuel element for the BR2 reactor. The most demanding irradiation conditions are obtained by replacing the outer ring of HEU fuel plates by the experimental plates, as the absorption of neutrons by the fuel is decreasing the fission density when moving from the outer to the inner plates.

Since the geometry of a mixed element is identical to the one of a standard fuel element, no hydraulic design of the experiment is required. The hydraulic performance of the mixed element is experimentally verified in a mock up loop, simulating the hydraulic conditions of the BR2 primary circuit. The irradiation conditions are optimized using the same tools as for the FUTURE basket irradiations.

For illustration, the conditions for a high power LEU U_3Si_2 dispersion fuel test with 4.8gU/cm³ are summarized. The requested irradiation conditions were:

- Cladding temperature between 140 and 150°C for 10% of the irradiation time; always below 150°C.
- Average burn-up accumulation of 55%
- Average power level 70% of peak power level.

Detailed analysis of the thermal conditions of the mixed element allowed to satisfy these conditions at a peak power rating between 410 and 450W/cm², compliant to the BR2 safety limits. The maximum power density was obtained at a total reactor power level of 57MW. The positioning of the fuel element was carefully selected in order to orient the center of the mixed element plates towards the stiffeners of the surrounding element. (figure 7). In this way, the optimum symmetry and uniformity of the flux is obtained, allowing to satisfy the requirement for peak to average power ratio.

Fig.7: Arrangement of the mixed element in the b60 channel of BR2.

5. Prototype fuel element testing: irradiation in the EVITA loop in BR2

Full size prototype fuel elements can generally not be irradiated in standard irradiation channels in the BR2 reactor. For such elements, the 200mm diameter irradiation channels can be fit with dedicated irradiation baskets. If the primary flow conditions are not compatible with the requirements of the prototype fuel element, a dedicated cooling loop has to be foreseen. For illustration, the irradiation of prototype fuel elements for the French Jules Horowitz reactor (RJH) in BR2 is discussed. The objective of the irradiation campaign is to test full size fuel elements for the future Jules Horowitz reactor under representative thermal and hydraulic conditions. These objectives imply the following:

- Power density up to 500W/cm
- Total power of the fuel element >3.8MW during at least 30 irradiation days.
- Average burn up of 60% at the end of the irradiation.
- Mean surface flux dispersion between 1.15 and 1.31
- Coolant velocity 14.6m/s.

The geometry of the RJH element significantly differs from the standard BR2 elements: the number of fuel plates is larger (8 instead of 6), the outer diameter is 96.2mm instead of 80mm, the spacing between the plates is smaller (1.95mm instead of 3mm) and the element length is smaller. In order to provide the required cooling conditions, a semi-open loop is installed in the BR2 reactor vessel. The loop consists of the following parts:

- The in-pile section, containing the fuel element. It is installed in the central 200mm diameter channel of the BR2 reactor for sake of symmetry in the irradiation conditions. The space between the channel wall and the irradiation channel can be filled with 2 types of plugs, one in Be and one in Al with water channels. The availability of 2 plugs is to tailor the total reactivity of the fuel element and irradiation rig: with increasing burn-up, the Al + water plug is replaced by the Be plug to compensate for the uranium consumption in the element.
- The instrumentation cane, inserted in the axis of the experimental fuel element. The instrumentation provides on line recording of the gamma and neutron fluxes, the pressure drop and temperature measurements in order to determine the thermal balance in the loop.
- The booster pump, installed in the reactor pool, which feeds the coolant to the in-pile section.
- The 2 aspiration lines, taking primary water from two peripheral reactor channels to the booster pump.
- The instrumentation and control out-of pile equipment to record the data on the experiment, regulate the flow through the in-pile section and provide the necessary safety functions.

As the experimental fuel element contains significantly more ^{235}U than a standard fuel element of BR2 and that there are no burnable poisons foreseen in this type of fuel element, the reactivity effect of the fuel element and the loop is evaluated both by modeling as well as by zero power nuclear measurements. As the objective of the irradiation program is to obtain an average burn up of 60%, the corresponding consumption of uranium is compensated by replacing the plug around the in pile section. After 2 initial irradiation cycles, the aluminum-water composite plug is replaced by a beryllium one.

The safety provisions of the EVITA loop are to prevent damage to the fuel element due to loss of flow accidents in the loop. So far, it has not been demonstrated that the RJH element will be cooled by natural convection after the stop of the pumps and SCRAM of the reactor, as it is the case for the standard BR2 fuel elements. When the BR2 primary pumps are functioning, sufficient flow occurs in the EVITA loop to evacuate the residual heat of the element. In case of failure of the primary pumps of the BR2, the EVITA booster pump itself can evacuate the residual heat from the element. Its availability is required during one hour after the end of the irradiation in order to prevent fuel damage. Therefor, an uninterruptible power supply (UPS) is connected to the EVITA loop booster pump, to provide the required cooling after unforeseen stop of the primary pumps of the reactor. Simultaneous failure of both the primary pumps of the reactor and the EVITA booster pump is considered an accident beyond design; however, there is a basket foreseen at the lower extremity of the in-pile section in order to capture fuel element debris in case of fuel damage or mechanical failure of the element. In case of fuel failure, the consequences are limited by the safety systems, as installed in the BR2 reactor's primary circuit and auxiliary systems.

The typical power evolution, as measured in the EVITA loop during one BR2 reactor cycle, is shown in figure 5. Up to now, 3 RJH prototype fuel elements have been irradiated in the EVITA loop in the BR2 successfully. Post irradiation examinations have not shown any cladding failure and subsequent fission product releases, neither any abnormal deformation nor swelling of the fuel meat in the plates.

4. CONCLUSIONS

The current paper provides an overview of the different types of irradiation experiments, as performed in the BR2 reactor for the qualification of material test reactor fuel under high performance conditions. These tests take advantage of the high flexibility of the BR2 reactor in order to accommodate the experimental goals, both from the nuclear as well as from the geometrical point of view.

Irradiation of flat fuel plates is a versatile method to optimize fabrication methods for new types of MTR fuel. Comparison of the full size plate irradiation with earlier mini plate irradiations show the more severe character of the former type of test and show that this test is an essential step to qualify the fuel for utilization in high performance conditions.

The mixed element irradiations are a cost effective

step to validate a metallurgical optimized fuel in a realistic geometry. The mixed element test evaluates the compatibility of plate bending with the irradiation and thermal-hydraulic conditions.
Finally, the irradiation of full scale prototype fuel elements represents the final step to qualify the new fuel element. Although more challenging from the technical and nuclear point of view, the successful realization of irradiations in the EVITA loop show that the BR2 reactor and its supporting teams are very suited for this type of challenge.

It should be noted that successful MTR fuel testing is not only depending on the availability of the necessary irradiation technology, but that the supporting capabilities for pre-and post irradiation analysis are equally important for the fuel qualification process.

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3.5 Review on the seismic safety of JRR-3 according to the revised regulatory code on seismic design for nuclear reactors

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JRR-3(Japan Research Reactor No.3) with the thermal power of 20MW is a light water moderated and cooled, swimming pool type research reactor. JRR-3 has been operated without major troubles. This paper presents about review on the seismic safety of JRR-3 according to the revised regulatory code on seismic design for nuclear reactors. In addition, some topics concerning damages in JRR-3 due to the Great East Japan Earthquake are presented.

Keywords: revised regulatory code, seismic safety, seismic design, the Great East Japan Earthquake, JRR-3

1. INTRODUCTION

JRR-3(Japan Research Reactor No.3) was built as the first domestic reactor in 1962.The large-scale modification including the removal and re-installation of the core was carried out from 1985 to 1990. The modified JRR-3 with the thermal power of 20MW is a light water moderated and cooled, swimming pool type research reactor. After the modification, JRR-3 has been operated without major troubles. This paper presents about review on the seismic safety of JRR-3 according to the revised regulatory code on seismic design for nuclear reactors. In addition, some topics concerning damages in JRR-3 due to the Great East Japan Earthquake are presented.

2. Revision of the regulatory code on seismic design

“regulatory code on seismic design for nuclear reactors” was established as a criterion when we examine it safely about the seismic safety of nuclear reactor institution (established in 1978). The nuclear reactor of Japan has been designed according to this guidance until now. In September 2006, For the further safety improvement of the nuclear reactor, the Nuclear Safety Commission let a code reflect the latest seismological knowledge and revised the code.

The main point in the revision of the regulatory code is as follows.

① Employ the advanced technique for the geological survey
② Employ the advanced technique for the determination method of the basic design earthquake ground motion
③ Change of the seismic design classification
④ Application of probabilistic safety evaluation

3. Review on the seismic safety of JRR-3

During 2007 to 2010, we carried out review on the seismic safety for JRR-3 and reported the result to the Ministry of Education Culture, Sports , Science and technology. Flow of the review is as follows.

3.1 Determination of the basic design earthquake ground motion

In this step, according to the revised code, we investigated having active fault or not, a past earthquake by the latest survey method, and determined the basic design earthquake ground motion. Fig.1 shows the basic design earthquake ground motion.

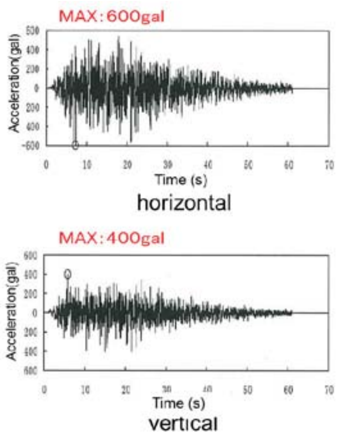

Fig.1. Basic design earthquake ground motion of JRR-3

3.2 Review on seismic safety for reactor building

We evaluated the seismic safety of the building. Items of evaluation of reactor building are as follows. Fig.2 shows the example of the evaluation model.

- ・Stability evaluation of the reactor building basics ground
- ・Seismic safety evaluation of reactor building
- ・Core covering function of reactor pool
- ・Seismic safety evaluation of roof truss

As a result of having evaluated these, we confirmed that the seismic safety of the nuclear reactor building was secured.

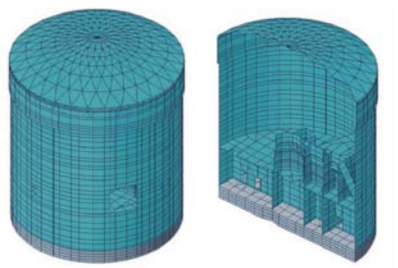

Fig.2. 3D FEM model of reactor building

3.3 Calculation of floor response spectrum

Based on a result of the earthquake response analysis of the nuclear reactor building, we calculated a floor response spectrum of each building floor. Fig.3 shows floor response spectrum of the first floor of reactor building.

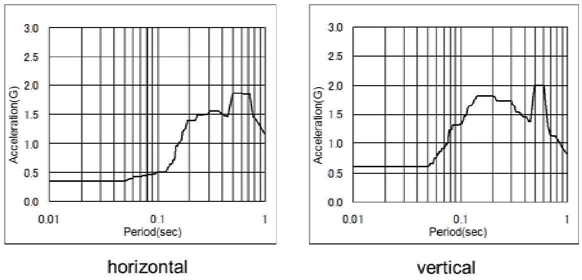

Fig.3. Floor response spectrum of the first floor of nuclear reactor building

3.4 Review on seismic safety for components and their supporting structure

If a function of reactor shut down and core covering are kept, JRR-3 can remove the Decay heat without a pump .（By the natural circulation cooling) Therefore, we chose components and their supporting structure from the viewpoint of reactor shut down and core covering. Fig.4 and Fig.5 shows the fuel element and core components, as the example of components and their supporting structure for evaluation.

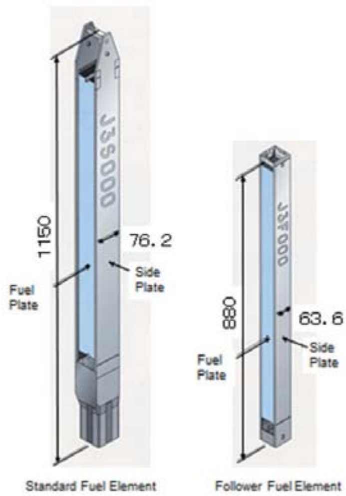

Fig.4. Fuel element

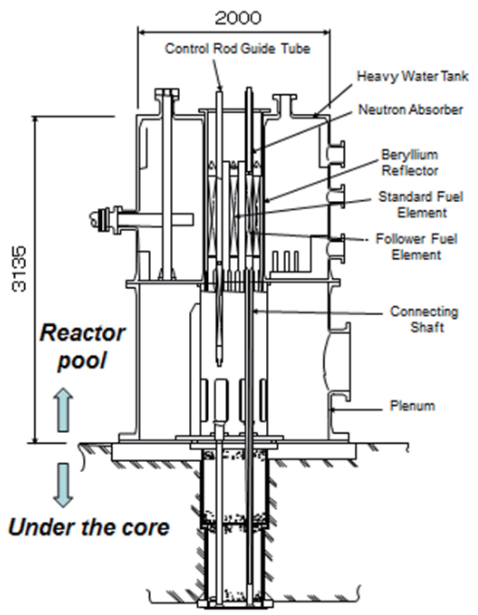

Fig.5. Core components

As an evaluation method, we used the method called "The response Magnification Method".

Firstly, the response ratio is calculated by expression(1) with the acceleration which used by original design and review. Next, calculate the initiation stress(σ_{ss}) by substituting response ratio into the expression(2). Finally, we judge the seismic safety of the equipment by comparison of the initiation stress(σ_{ss}) with evaluation criterion. If σ_{ss} smaller than evaluation criterion, we judge that the seismic safety of the equipment is secured.

$$\beta = \frac{\sqrt{C_H{}^2 + (1+C_V)^2}}{\sqrt{C_{H0}{}^2 + (1+C_{V0})^2}} \quad (1)$$

$$\sigma_{ss} = \sigma \times \beta \quad (2)$$

Table 1 shows result of evaluation. The response ratio shows that 20-30% of stress increased than original design. However, the initiation stress is very smaller than the evaluation criterion, because each component is designed with margin. In this review, we confirmed that each component had enough strength according to the demand of the revised code.

Table 1 Result of evaluation

Kind of components	Response ratio	Initiation stress (N/mm²)	Evaluation Criterion (N/mm²)	Judgment
Standard Fuel Element	1.31	4	66	○
Follower Fuel Element	1.31	20	90	○
Beryllium Reflector	1.28	3	135	○
Heavy Water Tank	1.28	14	100	○
Neutron Absorber	1.28	640	1272	○
Control Rod Guide Tube	1.28	7	114	○

4. Summary

As a result of review, we confirmed that the seismic safety of reactor building, components and their supporting structure was secured. The initiation stress is very smaller than the evaluation criterion. It is thought because each component is designed with margin.

Some topics concerning damage in JRR-3 due to the Great East Japan Earthquake

The Great East Japan Earthquake is a Great earthquake with the seismic energy of magnitude 9.0. This earthquake gave serious damage to the East Japan with tsunami. In Tokai-mura, observed a tsunami of 5m. At the time of the earthquake occurrence, JRR-3 was not operational. There is a lot of sink in ground around the reactor building due to the earthquake. But the serious damage such as the afunction was not find to the equipment important to safety. Fig.6 shows reactor core after the earthquake. The present time, with the repair and performance inspection, we are making efforts for the reopening of the nuclear reactor operation.

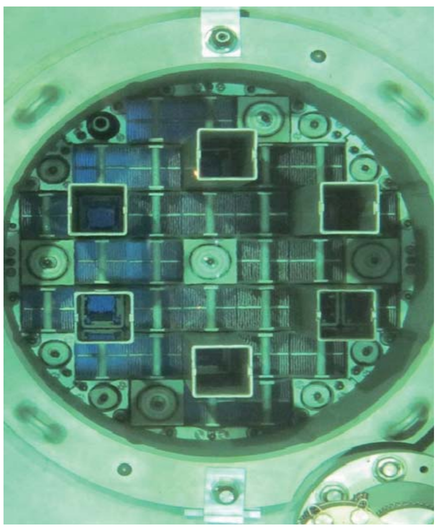

Fig.6. Reactor Core after the earthquake

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3.6 Present Status of Japan Materials Testing Reactor

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The Japan Materials Testing Reactor (JMTR) in Japan Atomic Energy Agency (JAEA) is a light water cooled tank type reactor with first criticality in March 1968. Owing to the connection between the JMTR and hot laboratory by a canal, easy re-irradiation tests can be conducted with safe and quick transportation of irradiated samples.

The JMTR has been applied to fuel/material irradiation examinations for LWRs, HTGR, fusion reactor and RI production. However, the JMTR operation was once stopped in August 2006, and check & review on the reoperation had been conducted by internal as well as external committees. As a result of the discussion, the JMTR reoperation was determined, and refurbishment works started from the beginning of JFY 2007. The refurbishment works have finished in March 2011 taking four years from JFY 2007.

Unfortunately, at the end of the JFY 2010 on March 11, the Great-Eastern-Japan-Earthquake occurred, and functional tests before the JMTR restart, such as cooling system, reactor control system and so on, were delayed by the earthquake. Moreover, a detail inspection found some damages such as slight deformation of the truss structure at the roof of the JMTR reactor building. Consequently, the restart of the JMTR will be delayed from June to next October, 2012. Now, the safety evaluation after the earthquake disaster is being carried out aiming at the restart of the JMTR.

The renewed JMTR will be started from JFY 2012 and operated for a period of about 20 years until around JFY 2030. The usability improvement of the JMTR, e.g. higher reactor availability, shortening turnaround time to get irradiation results, attractive irradiation cost, business confidence, is also discussed with users as the preparations for re-operation.

Keywords: JMTR, Refurbishment, Re-Operation, Future plan, Irradiation Test, World Network, Asian Network, Human Resource Development

1. INTRODUCTION

The Japan Materials Testing Reactor (JMTR) in Japan Atomic Energy Agency (JAEA) is a light water cooled tank type reactor with thermal power of 50MW.

The JMTR is connected with the hot laboratory by a water canal as shown in Fig.1, cross section of the core is shown in Fig.2. Specification of the JMTR is shown in table 1. Outline of the JMTR hot laboratory is shown in Fig.3.

The purpose for construction of JMTR was to perform irradiation tests for LWR fuels, materials to establish domestic technology for developing nuclear power plants, and also constructed to produce radio isotopes and education and training.

The first criticality was achieved in March 1968, and the JMTR has been utilized for basic and applied researchers on fuels/materials of fission reactors and fusion reactor. Power ramping tests for the nuclear fuels were, for example, performed to study the integrity/safety of fuels. Radioisotopes were also produced using the JMTR, and were widely used in the medical treatment and industries [1-3].

The reactor operation was stopped from August, 2006, and then the refurbishment works started from the beginning of JFY 2007 by the user's strong request to the JMTR utilization. The renewed and upgraded JMTR will restart in October, 2012 and will operate for a period of about 20 years (until around JFY 2030).

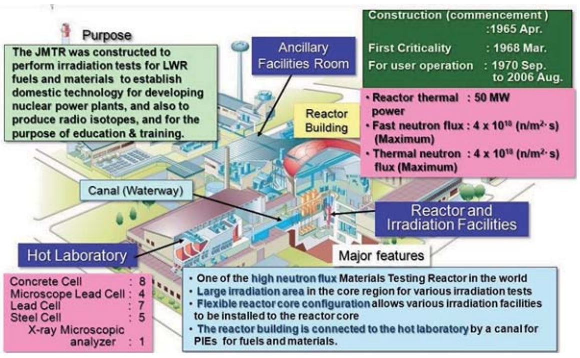


Fig.1. Outline of the JMTR

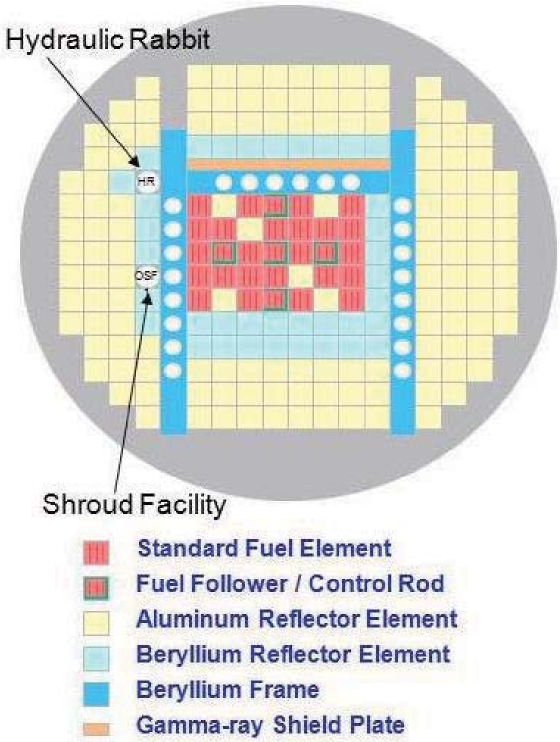


Fig.2. Cross Section of the Core

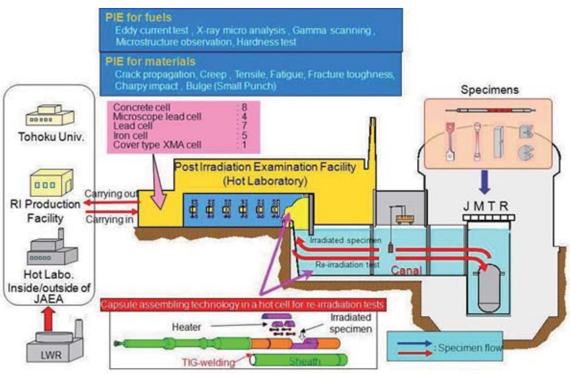


Fig.3. Outline of the JMTR Hot Laboratory

Table.1 Specifications of the JMTR

Reactor Power	50MWt
Fast Neutron Flux (Max.)	4×10^{18} n/m^2・s
Thermal Neutron Flux (Max.)	4×10^{18} n/m^2・s
Flow Primary Coolant	6,000 m^3/h
Coolant Temperature	49 ℃ / 56 ℃
Core Height	750mm
Fuel	Plate type, 19.8% ^{235}U
Irradiation Capability (Max.)	60(20*) capsules
Fluence/y (Max.)	3×10^{25} n/m^2・y
dpa of Stainless Steel (Max.)	4 dpa
Diameter of Capsule	30 - 65 mm
Temp. Control (Max.)	2,000 ℃

* : Capsule with in-situ measurement

2. JMTR REFURBISHMENT PROGRAM

Repairing and replacement works of the JMTR have been carried out according to the following process (1) to (3).

(1) Investigation of Aged Components

Aged components were surveyed and selected by evaluating whether those could be used safety after re-operation of the JMTR.

(2) Replacement of Reactor Components

Replacements were carried out within the range of licensing permission of the JMTR.

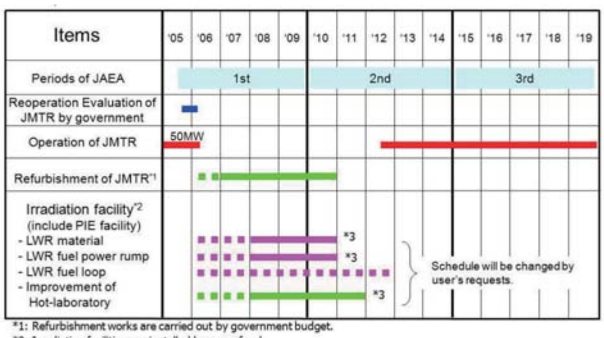
Replacements of the systems and components, such as, boiler system, refrigerator for air conditioning system, power supply system, and air supply/exhaust system, process control and instrumentation system, and so on, were conducted.

(3) Installation of New Irradiation Facilities

Corresponding to the user's requests, new irradiation facilities, such as test facilities for materials/fuels, production facility for medical isotopes etc. are being installed.

The JMTR refurbishment schedule is shown in Table 2.

Table 2 The JMTR Refurbishment Schedule



*1: Refurbishment works are carried out by government budget.
*2: Irradiation facilities are installed by users fund.
*3: IASCC, Irradiation embrittlement, Hafnium irradiation and fuel ramp tests are being prepared in a NISA (Nuclear and Industrial Safety Agency) project.

3. FUTURE PLAN OF JMTR

After finishing the refurbishment works, the JMTR will be operated for a period of about 20 years until around JFY 2030. The expected utilization fields of irradiation are :

1) *Lifetime extension of LWRs,* which includes the aging management of LWRs and the development of the next generation of LWRs.
2) *Progress of science and technologies*, which includes the development of fusion reactor materials, development of HTGR (High Temperature Gas cooled Reactor) fuels and materials, the basic research on nuclear energy, etc.
3) *Expansion of industrial use*, which includes the production of silicon semiconductor for the hybrid car and the production of 99mTc for the medical diagnosis medicine.
4) *Education and training* of nuclear scientists and engineers.

Availability factor of the JMTR would be increased in order to cope with these increasing of irradiation utilization. At first, 4 cycles are planning in JFY 2012 and 7 cycles (about 60 %) are planning in JFY 2013. In near future, availability factor of the JMTR would be increased up to world-class. [4]

Moreover, the JMTR is planned to contribute the research/development utilization as well as the industrial utilization by offering excellent irradiation technology. In irradiation, an attractive irradiation test will be proposed by developments of advanced technologies such as new irradiation technology, new measurement technology and new PIE technology. Furthermore, the cooperation with various nearby PIE facilities surrounding the JMTR will be established to extend the capability of PIEs after ongoing discussion with the nearby facilities.

In Asian area, some excellent testing reactors are operated now. Each of these reactors has individual and original characteristics and takes supplementary role in each other. The JMTR has a plan to contribute greatly to users by construction of the internationally utilized facility as an Asian center of testing reactors.

3.1 Birth of the nuclear techno-park with the JMTR

In June 2010, Japanese Government selected 14 specialized projects of advanced research infrastructure in order to promote basic as well as applied researches. One of these 14 projects is "Birth of the nuclear techno-park with the JMTR". In this project, new irradiation facilities and PIE equipments will be installed up to FY 2013. The purpose is to build international research and development infrastructure. In the project, development of user-friendly environment especially for young and female researchers is highlighted For example, preparation of irradiation area with LWRs environments, development of ^{99m}Tc production technique by (n, γ) method, construction of atomic level analyzer of PIE and achievement of very high temperature (2000 degree) irradiation environment would be installed up to JFY 2013 as shown in Table 3.

Table 3 Schedule of the project 'Birth of the nuclear techno-park with the JMTR'

Installation of advanced equipments	FY	2010	2011	2012	2013
Irradiation facilities	LWRs water-environment demonstration test facility				
	High accuracy time-control irradiation facility				
	High accuracy capsule temperature control unit				
PIE equipments	High grade manipulator with visual function				
	Complex type fine texture analyzer				
For education& training	Testing-reactor simulator for nuclear education				

3.2. World network and Asian network

Construction of world network is proposed by JAEA to achieve efficient facility utilization and providing high quality irradiation data by role sharing of irradiation tests with materials testing reactors. Image of the MTRs network and World and Asian network is shown in Fig.4. As the activity to construct of the world network, the International Symposium on Material Testing Reactors (ISMTR) has been held every year. The objective of this symposium is to exchange the information among each testing reactors for mutual understanding of status of each reactor, the world network construction of testing reactors and so on.

Furthermore, by construction of the world network and Asian network, it would be able to achieve efficient facility utilization and to provide high quality irradiation data by role sharing of irradiation tests among materials testing reactors in the world. Moreover, the JMTR is addressed as a kernel of Asian MTR, and fuel/material study for LWRs, education & training etc. are promoting.

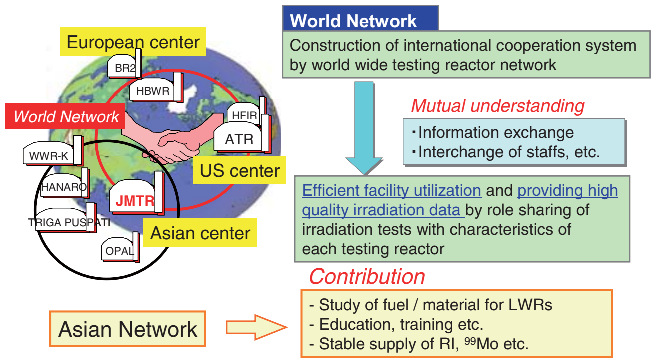


Fig.4 World network and Asian network for material testing reactors

3.3. Next generation research and test reactor

The feasibility study on next generation research and test reactor is started for the growing demand for new research and test reactor in the world. This project will be proceeded with cooperation works by universities, industries and Asian countries as shown in Fig.5.

This study would be contributed for domestic and international development of human resource. Furthermore, as the results, its work study will be contributed for introduction of nuclear power plant to these countries in future

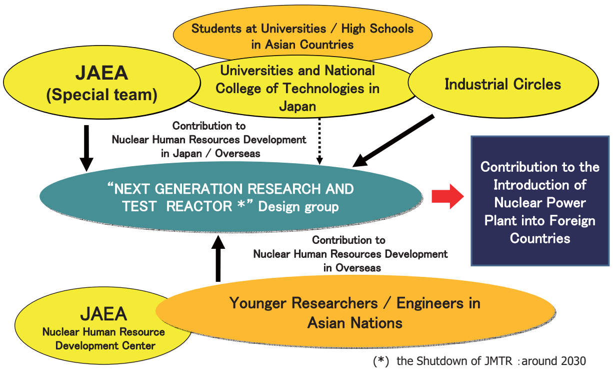


Fig.5 Design and construction of Next Generation Research and Test Reactor

4. CONCLUSIONS

JAEA placed the JMTR as a testing reactor which supports the basic technology of the nuclear energy, and carried out the refurbishment of the reactor facilities taking four years from JFY 2007 for prolonged operation; the new JMTR would be re-operated promptly after the completion of seismic influence evaluation.

New irradiation facilities are under installation in the JMTR; LWR fuels/materials irradiation facilities by the "Birth of the nuclear techno-park with the JMTR" , industrial use for RI production, and so on. Furthermore, the new irradiation facilities are also under installation, which would be completed by JFY 2013.

The new JMTR would be one of the kernel irradiation testing reactor in Asia, and contribute to the research and development utilization as well as the industrial utilization by offering advanced irradiation/post-irradiation technologies. Moreover, new JMTR contribute to the human resource development for domestic as well as foreign researcher/engineers including Asian countries where nuclear power plants will be planned to construct.

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4. Advancement of Irradiation Technology (1)

4.1 Status of Material Development for Lifetime Expansion of Beryllium Reflector

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Beryllium has been used as the reflector element material in the reactor, specifically S-200F structural grade beryllium manufactured by Materion Brush Beryllium & Composites (former, Brush Wellman Inc.). As a part of the reactor upgrade, the Japan Atomic Energy Agency (JAEA) also has carried out the cooperation experiments to extend the operating lifetime of the beryllium reflector elements. It will first be necessary to determine which of the material's physical, mechanical and chemical properties will be the most influential on that choice. The irradiation testing plans to evaluate the various beryllium grades are also briefly considered and prepared. In this paper, material selection, irradiation test plan and PEI development for lifetime expansion of beryllium are described for material testing reactors.

Keywords: Beryllium, Reflector, Lifetime expansion, Material Testing Reactor, JMTR, WWR-K

1. Introduction

As a structural material, beryllium (Be) is a light material which has high tensile strength. Be surface forms a thin oxidation film by interacting with air like aluminum, and Be 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, Be has been utilized as a moderator and/or reflector in a number of material testing reactors. In fact, the nuclear properties of Be 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 Be exist in many places throughout the world, and a lot of Be was used in materials testing reactors (MTR) from the beginning of atomic energy development. Usage of Be 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 Be is achieved. The activation issues for Be 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 Be. At the same time, tritium (^3H) is produced in the Be by a well known reaction sequence. It is difficult to reprocess irradiated Be because of high induced radioactivity.

In this paper, material selection, irradiation test plan and development of post-irradiation examinations (PIEs) for lifetime expansion of Be are described for material testing reactors. These items have been discussed in the "Specialist Meeting on Recycling of Irradiated Beryllium".

2. Beryllium Utilization in JMTR

Figure 1 shows the core arrangement of JMTR. As the engineering data of JMTR, thermal power is 50MW, and maximum fast and thermal neutron flux are 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 made of Be and has a 224 array of 77mm squares arranged in a square lattice [3].

Two kinds of Be reflectors are used in the JMTR (see Figure 2). One is the Beryllium elements. The Be reflector elements have not been changed during JMTR operation. Because these elements are changed the installation place and controlled the irradiation fluence. The other is the Be frame. It is necessary to exchange the beryllium frames every fixed period because of the deformation by neutron irradiation. Thus, there frames were exchanged six times up to the JMTR operation periods of 165 cycles. The deformation of beryllium frame has been measured and managed during every JMTR overhaul for management.

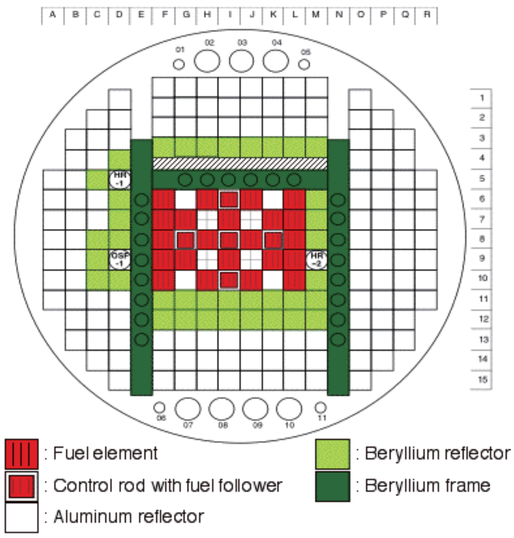
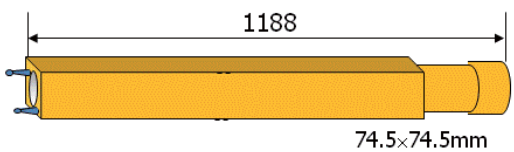
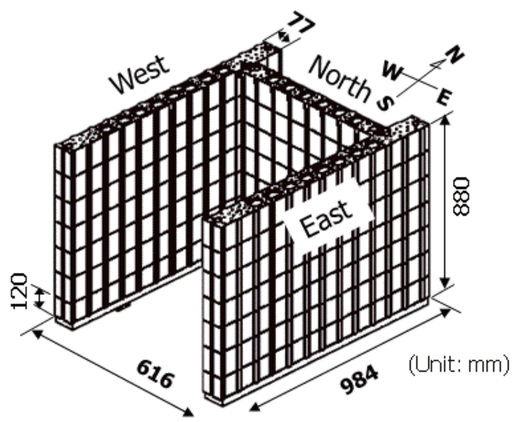


Fig. 1 Core arrangement of JMTR.



(a) Be Reflector Element



(b) Be Reflector Frame

Fig. 2 Configuration of Beryllium reflectors in JMTR.

The JMTR has used beryllium reflector since it began operation in 1968. The reactor has been operated using structural grade beryllium made by Materion Brush Beryllium & Composites (MBBe&C, former, 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.

3. Material Selection for Lifetime Expansion

Table 1 shows general properties of Be. Be 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. Be surface forms a thin oxidation film by interacting with air like aluminum, and Be is highly resistant to corrosion in dry gases.

S-200F fabricated by the vacuum hot press is used as the standard material and the typical purity of nuclear grade Be such as S-200F is 99.1%. For the Be frame, it means an operational service lifetime goal of 15-20 years (180,000MWD), rather than the current about five years. In order for that to happen, it will be necessary to consider fundamental changes to the frame design, starting with the selection of Be material grade. For the selection of Be materials, shape and purity of beryllium power, and uniformity of grain are considered as the cooperation program between JAEA and MBBe&C. Especially, uniformity is changed from VHP to HIP. Thus, three kinds of beryllium materials were selected for lifetime expansion in the specialist group. S-200F is the reference material as the reflector. S-65H will be tested due to its higher purity and better isotropy than S-200F and I-220H will be tested due to its higher mechanical strength and better isotropy than S-200F. Table 2 shows properties of candidate materials [4].

Figure 3 shows the test flow for lifetime expansion of beryllium reflector. For the material selection, it is necessary to construct the database of each Be material grade. From the utilization of Be reflectors, mechanical and chemical properties should be tested under un-irradiation and irradiation condition. Especially, chemical properties is important items for lifetime expansion.

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

Table 2. Candidate materials of beryllium grades for lifetime expansion

(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

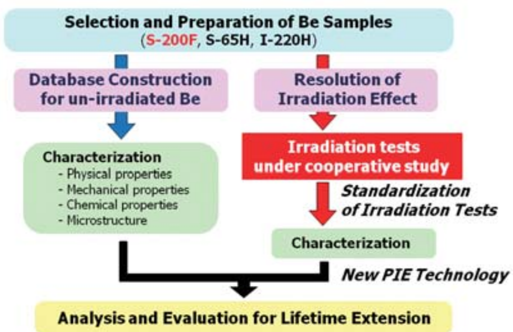


Fig. 3 Test flow for lifetime expansion of beryllium reflector.

4. Characterization of Beryllium Specimen

The corrosion test was carried out under pure water at 50°C [5]. In the test, the surface interactions of these beryllium samples were evaluated by X-Ray Diffraction (XRD) and X-ray Photoelectron Spectroscopy (XPS). Water analysis was carried out during the corrosion test by the pH/conductivity meter.

In the corrosion test, the pH of water in the vessel installed Be samples was almost 6 at a stationary value. The electric conductivity increased gently and the value was about 400μS/m. During the corrosion test, the white product was generated on the surface of each Be sample. From the XRD patterns of each beryllium before/after corrosion test, the (100), (002) and (101) peaks of beryllium were observed in each Be sample. These peaks of each Be sample decreased after the corrosion test and the decrease of peaks in S-200F was larger than that in I-220H.

Figure 4 shows the weight change of each Be sample during corrosion test. From the result, the weight change of S-200F was larger than that of I-220H. It seems that the weight change influenced by the content of BeO in each Be sample. In future, the surface analysis of each Be sample after the corrosion test will be carried out by XPS.

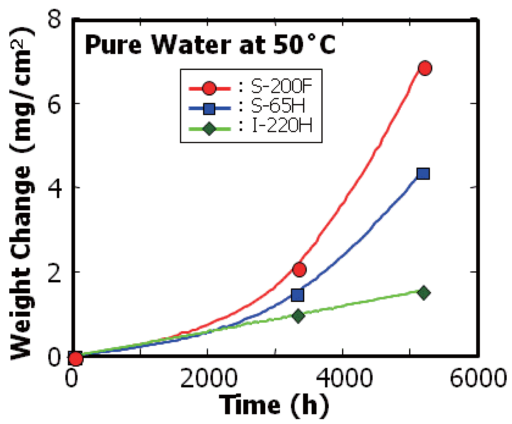


Fig. 4. Weight change of Be samples during corrosion test.

5. Irradiation Tests

Candidate test reactor facilities for performing the irradiation include: the JMTR and JRR-3 (JNP), the BR2 (Belgium), the ATR (U.S.A.), the SM-3 (RF) and WWR-K (KZ).

In fact, the irradiation tests of each Be material grade have been carried out in JRR-3 and WWR-K and the irradiation tests in JMTR will be started at the JMTR re-start (Oct., 2012). The irradiation tests have been negotiated in BR2 (Belgium), the ATR (U.S.A.) and the SM-3. (RF). Five kinds of irradiation samples were prepared for tensile test, bending test, impact test, H/He release test and observation in MBBe&C.

In JRR-3, maximum fast and thermal neutron flux were about 0.85×10^{18} and 1.8×10^{18} /m^2/s, respectively.

Irradiation time was about 4200h. Irradiation temperatures of Be samples were about 50˚C and 150˚C in cooling water and He gas atmosphere, respectively. Now, dismantlement of irradiation capsules and transportation from Tokai to Oarai are planned.

Irradiation test started at the WWR-K Reactor in Kazakhstan under ISTC partner project from March 2010 [6]. The Be samples have been irradiated up to the fast neutron fluence (E>1MeV) ~1×10^{24} /m^2 in the irradiation channels of the WWR-K. Irradiation position in WWR-K is shown in Fig. 5. The density of fast neutron flux was measured by Indium detectors screened by Cadmium. Beryllium samples under irradiation in irradiation device are washed by coolant, which is desalted water. Chemical composition of coolant meets relevant regulatory requirements. Characterization of the irradiated specimens will be carried out by the PIEs such as size change and measurements of helium and tritium.

For the irradiation tests in JMTR, the irradiation capsules were fabricated. In JMTR tests, maximum fast and thermal neutron flux were about 1×10^{18} and 3×10^{18} /m^2/s, respectively. Irradiation temperature of Be samples was about 50˚C in cooling water.

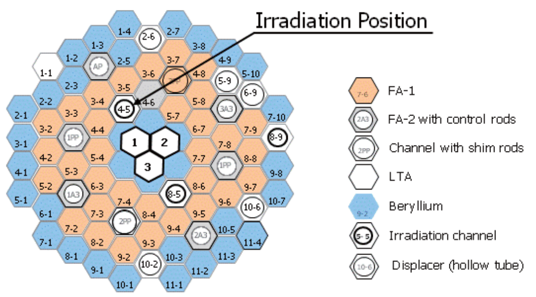


Fig. 5 WWR-K core and irradiation position of beryllium samples.

6. Development of PIE Techniques

As the development of new PIE techniques, non-destructive inspections such as Electromagnetic Transducer (E-MAT) and electrical resistor, and the transmission electron microscope (TEM) observation have been investigated for research of characteristics of Be samples after neutron irradiation tests.

The preliminary tests were carried out with the un-irradiated Be samples by EMAT and electrical resistor and the sound velocity and the resistivity were evaluated [7]. The preparation method of Be samples and TEM observation results was being examined as a evaluation method of the un-irradiated Be samples [8].

In future plans, the irradiated S-200F, S-65H and I-220H will be measured be the developed techniques after irradiation testing in JRR-3.

7. Conclusions

Material Selection of Beryllium Grades as MTR Reflectors was discussed in the Be Specialist Meetings and the following items were decided:

(1) Selection of beryllium grades : Reference (S-200F), Isotropy (S-65H), Strength (I-220H)
(2) High-irradiation tests for lifetime extension : Performance of irradiation tests (JRR-3, WWR-K, JMTR)

Status of Be study are as follows;

(1) Preparation of Be Samples for irradiation tests
(2) Irradiation tests in JRR-3 and WWR-K
(3) Preparation of irradiation test in JMTR
(4) Corrosion evaluation of Be specimens in out-of-pile test
(5) Development of measurement techniques for new PIEs

Acknowledgement

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4.2 Beryllium Reflectors for Research Reactors. Review and Preliminary Finite Element Analysis.

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Beryllium is used in numerous research reactors to moderate neutron energy and to reflect neutrons back into the core, thus intensifying the thermal neutron flux. However, beryllium is degraded by radiation damage, as a result of both displacement and transmutation. Displacement damage leads to point defect clustering, irradiation hardening and embrittlement. Transmutation produces helium, which results in high levels of gas and swelling, even at low temperatures. A brief state-of-the-art review on the use of reflector assemblies reveals that each user has adopted a different method for overcoming problems related to swelling: strengthening, cracking and distortion. In the present work a preliminary study about the geometry influence on the reflector assembly behavior was performed by a Finite Element Analysis (FEA). A simplified study was made varying its geometry in height, thickness and width. The results showed that the most influencing parameter in avoiding distortion due to swelling is firstly the reflector's assembly height, H; secondly its thickness, L, and lastly its angle/width, θ. These results contribute to the understanding of distortion behavior and the stresses generated in a simple geometry Be bar subjected to radiation, which can be a useful tool for mechanical design of more complex components.

Keywords: Beryllium reflector assembly, swelling, Finite Element Analysis, geometry variation, distortion, deflection.

1. INTRODUCTION

1.1 Beryllium in nuclear applications

Beryllium (Be) has a rather high neutron-scattering cross section (σ_S=6b) and the lowest neutron-absorption cross section of all the metals (σ_a=8mb) because of its low atomic weight (9.012182 g/mol) and high atomic density (0.123 atm/b-cm). These factors make Be an excellent material for reflectors in numerous research reactors, reflecting neutrons back into the core and thus intensifying the thermal neutron flux density. Beryllium is also adequate as moderator; an example is the BR2 reactor [1]. Besides, Be alloys are also used for other good properties [2]:

- Light weight, 66% lighter than aluminum (ρ=1.85 gr/cm^3)
- Rigidity, 4X > Aluminum, 1.5X > Steel
- Heat Capacity, highest of all metals,
- Thermal conductivity, equivalent to aluminum

However, Be structure is altered during neutron irradiation by displacement damage and by transmutation. Displacement damage generates a wide range of defects in the Be crystal structure. Transmutation produces helium and tritium in the crystal structure which acts as defect, causing swelling. Figure 1 shows the simplified transmutation diagram for ^{9}Be. The effect of defects and swelling on thermomechanical properties depends on the irradiation temperature. [3-9].

$$^9Be + n \rightarrow 2\,^4He + 2n \qquad (E_n > 2.7\ MeV)$$
$$^9Be + n \rightarrow \,^7Li + \,^3H \qquad (E_n > 10.5\ MeV)$$
$$^7Li + n \rightarrow \,^4He + \,^3H + n$$
$$^9Be + n \rightarrow \,^6He + \,^3H \qquad (E_n > 0.6\ MeV)$$
$$^6He \rightarrow \,^6Li + \beta^- + \nu$$
$$^6Li + n \rightarrow \,^4He + \,^3H$$

Fig. 1. Beryllium fission due to fast neutron interactions [6]

Solubility of ^{4}He and ^{3}H in beryllium is very low and they will easily precipitate into small gas bubbles. If irradiated at low temperature (50°C-150°C), the swelling is limited [10], but the Be exhibits substantial hardening and embrittlement. The He atoms are mostly trapped at vacant Be lattice sites [11] or in the vicinity of dislocations [12]. The dislocations are pinned by the gas atoms, causing hardening and embrittlement. At higher temperatures, the He atoms and vacancies become moving and internal gas-filled bubbles and voids are formed. With increasing temperature, the gas bubbles grow and eventually coalesce, inducing important swelling to the material. These effects undermine the structural integrity of Be reflectors.

1.2 Beryllium components in research reactors

Bibliographic data allow for making an approximate estimation of Beryllium components' lifespan in high flux Research Reactors worldwide. The following section summarizes such information.

HFIR

The HFIR [13-15] core is cylindrical and it is surrounded by a concentric ring of beryllium reflector which is subdivided into three regions:

- Removable
- Semi Permanent
- Permanent

Due to beryllium swelling, these different parts need to be replaced after a certain amount of operation cycles, as displayed in Table 1.

Table 1 Lifetime of Be reflector in HFIR (85 MW)

Type of reflector	Acc. Power [MWD]	Neutron Fluence [$x10^{22}$ n.cm^{-2}]
Removable	83700	3,4
Semi Permanent	167400	3,3
Permanent	279000	2,3

ATR

The ATR (Advanced test reactor) has a serpentine core design and Beryllium reflector surround the serpentine fuel elements [16,17]. The beryllium reflector is fabricated in 8 separate sectors. Since the ATR began to operate in 1967, there have been four planned core internals change-outs, when the beryllium reflector blocks were replaced. Table 2.

Table 2 Be reflector lifetime in the ATR (120 MW)

Period	Acc. Power [MWD]	Neutron Fluence [$x10^{22}$ n.cm^{-2}]
1968/1972	28894	1,83
1973/1977	36285	2,31
1977/1986	72987	4,64

The first change, in the early 1970s, was not planned but was necessary when the reflector blocks developed major cracks. As a consequence, blocks since then have deep stress relieving saw cuts perpendicular to the core axis which extended the Beryllium life. After that, the replacement took place in 1977, 1986, 1994, and 2004, at nominally 10-year intervals.

JMTR

The JMTR (Japan Material Testing Reactor) is utilized for irradiation experiments of fuel and materials due to the high neutron flux generated in its core [18]. The fuel region is surrounded by Be reflector frames. Because its exposure to high fast neutron fluxes, swelling promotes Beryllium reflectors cambering and they must be replaced after a fixed amount of accumulated energy (MWD). Since its startup JMTR the reflector frame has been changed in 6 opportunities. From the fourth generation, the followings improvements were done in order to extend the lifespan of the Be frame reflector. The Table 3 summarizes the information [19].

- Design improvement: enlargement of the gap between Be frame and fuel element, shape change of bottom block, joint design change.
- Monitoring of swelling: camber monitoring apparatus, yearly monitoring, setting the controlling criteria

Table 3 Be reflector lifetime in the JMTR (50 MW)

Period [years]	Acc. Power [MWD]	Neutron Fluence [$x10^{22}$ n.cm^{-2}]	Camber [mm]
1966/1974	24000	0.96	0.71
1975/1983	28000	1.12	0.84
1984/1987	25000	1.00	0.75
1988/1995	36000	1.44	1.24
1996/2002	29000	1.15	1.09
2003/2007	25000	1.00	-

BR2

Beryllium is used in the BR2 as a neutron moderator and reflector under the form of a reactor core matrix [1]. The central region of the reactor pressure vessel contains a matrix of beryllium metal which acts to position all fuel elements, control rods, beryllium plugs and experiments. The matrix is mainly an assembly of a great number of prismatic Irregular hexagonal bars, each with a cylindrical bore which forms the channel. The flux reached in BR2 is up to 10^{15} n.cm^{-2}.s^{-1} thermal and 8 x 10^{14} n.cm^{-2}.s^{-1} (>0.1 MeV). BR2 was first criticality in 1961, full power operation startup in early 1963 was 224.000 MWd. In 1974, a visual inspection of the inner surfaces of the reactor channels showed signs of cracks, not observed three years earlier during the previous examination. A safety evaluation concluded that further operation was possible without undue risk to the operators and the public. The reactor was finally shutdown by the end of 1978. At this stage the matrix was heavily cracked and the "hottest" channel had reached a fast fluence of 7.9 x10^{22} n.cm^{-2} (E>1 MeV). The replacement of the first BR2 beryllium matrix was carried out in 1979-80. Investigation of the dimensional stability and the swelling of the first matrix led to the definition of a maximum allowed fast fluence for the second matrix. Agreement has been reached with the National Safety Authority concerning this value: 6.4x10^{22} n.cm^{-2} (E>1MeV). A 2nd major shutdown with major refurbishment was in 1996. Table 4.

Table 4 Lifespan of Be matrix in the BR2 (100MW)

Matrix	Period	Acc. Power [MWD]	Neutron Fluence [x10^{22}n.cm^{-2}]
1^{st}	1963/1978	224000	7.90
2^{nd}	1980/1996	180000	6,40
3^{rd}	1997/2016	------	------

2. PRELIMINARY Be REFLECTOR FEM ANALYSIS

As can be seen from previous section each reactor has adopted different strategies in order manage the problems associated to swelling: strengthening, cracking and distortion.

The lifespan of the Beryllium components, in fluence terms, typically vary between 1 and 6.4 x 10^{22} n/cm^2. This variation mainly is attributed to the material quality, impurities content, especially BeO content and, manufacturing process [20]. A lot of effort has been put in improving Beryllium properties, and still up to day. However an additional variable which is essential in the lifespan of Beryllium component subjected to radiation generally it is not discussed in open literature: the mechanical design, basically the geometry.

This preliminary study takes a hypothetical Beryllium reflector bar and analyzes, by means finite element method analysis (FEM), the distortions and associated stresses promoted by swelling in varing its geometry (height, thickness, width).

2.1 Material and method

2.1.1 Material

For these preliminary calculations, it was assumed that the alloy is S200-F from Materion Products (ex Brush Wellman), which mechanical properties are shown in the following table. [21]

Table 5 Be alloy S200-F. Mechanical properties.

Yield Strength, min[MPa]	241
Tensile Strength, min [MPa]	324
Modulus of Elasticity[GPa]	303
Poisson Ratio	0.18
Elongation at break, min [%]	2

2.1.2 Study cases

A reflector assembly of height (H) 820 mm, thicknes (L) 90mm and angle/width 16° was taken as reference case, figures 1 and 2.

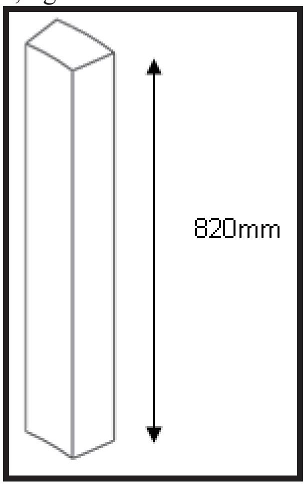


Fig. 1. Reference reflector assembly scheme

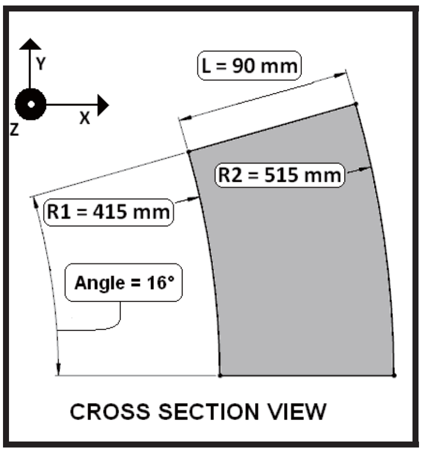


Fig. 2. Cross section dimensions of reflector assembly

The effect of swelling has been studied in varying the geometry assembly as follow:

- Variation on thickness (L): 30 mm, 90 mm, 150 mm and 180 mm.
- Variation on height (H): 205mm, 275 mm, 420 mm and 820 mm
- Variation on angle/width (θ): 16°, 32°, 180° and 360 °

2.1.3 Calculi

All calculi were made using the SolidWorks Simulation application, which is part of the SolidWorks design package. It was assumed that the material have an isotropic behavior for calculus purpose. Stress strengthening was no taking account for this analysis level. On other hand, as boundary condition was imposed that the bottom of the reflector assembly was fixed.

A moderated incident fast neutron flux was assumed, $\varphi = 6.1x10^{13}$ $n.cm^{-2}.s^{-1}$. Figure 3 shows the variation of fast neutron flux (E>0.85 MeV) with the penetration depth in Beryllium thickness.

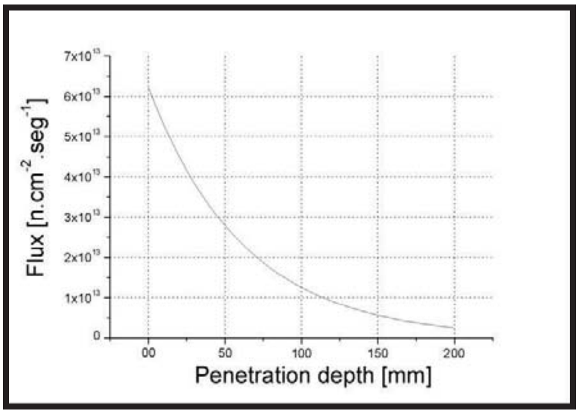


Fig. 3. Attenuation of fast neutron flux through Be.

The swelling was assumed isotropic and it was estimated through the linear approximation fitted from bibliographic data and which given in Equation 1.

$$\frac{\Delta V}{V}[\%] \cong 3\frac{\Delta L}{L}[\%] = 2,0x10^{-23}.(\varphi.t)\,, \qquad (1)$$

where, V: Volume [cm3]; L: length [cm]; φ: flux [$n.cm^{-2}.s^{-1}$]; t: time [s]

2.2 Results

Figure 4 shows the schematic distortion undergone by the reference case Be reflector assembly. In addition to the expected growth of volume due to the swelling, the simulation results predict a deflection of the free bar end (δ). This phenomenon was observed in all case studies, what led to establishment of deflection as a consequence parameter of swelling damage.

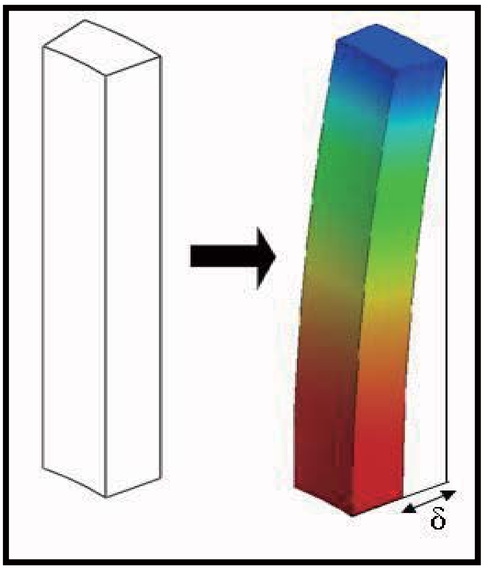


Fig. 4. Deflection of a Be reflector assembly.

The following graphics shows the results for the different study cases: variation in height, thickness and width or angle. The filled data point represent the reference case results and the open point the different variants.

Thickness Variation:

As shown in Figure 5, deflection of reflector assembly diminishes when thickness bar increases.

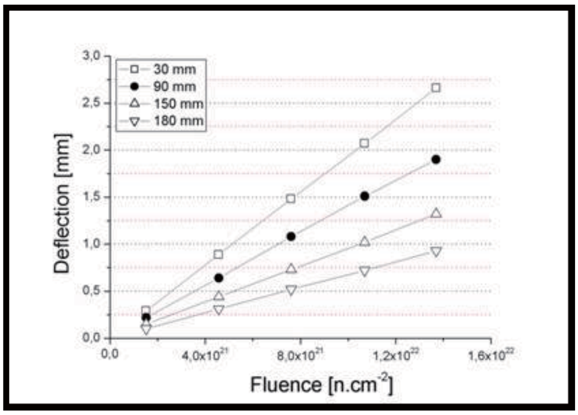


Fig. 5. Deflection variation with reflector thickness

However, stresses increase their values (Figure 6). It can be noted that between 150 and 180 mm thickness, stresses do not present a significant increment.

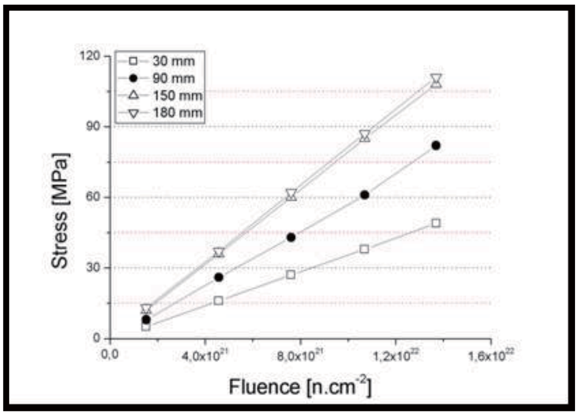


Fig. 6. Stress variation with reflector thickness

Height Variation:

Figure 7 shows the deflection change as per the assembly height, while Figure 8 shows the stresses generated due to swelling. When the bar height decreases, the deflection also diminishes. However, the stresses experiment a significant reduction when the reflector assembly height is the half of that taken as reference case height but, there is not important change in stresses for shorter bars, Figure 8.

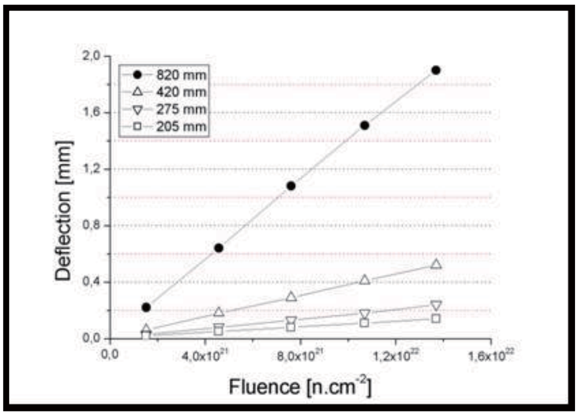


Fig. 7. Deflection variation with reflector height

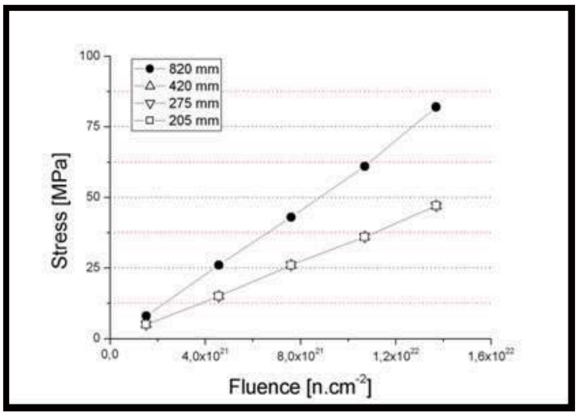


Fig. 8. Stress variation with reflector height

Angle/Width Variation:

Deflection decreases as the angle increases (Figure 9). On other side, for pieces with angles of 16° and 32°, deflections are very similar. For a complete ring the deflection does diminish considerably. On the contrary, and as can be expected, the stresses increase as angles increase (figure 10) due to the constraint effect.

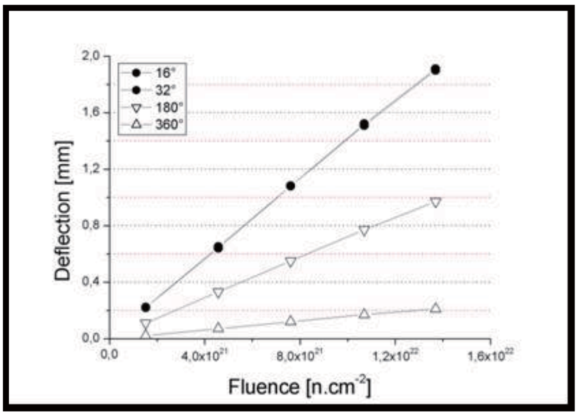


Fig. 9. Deflection variation with reflector angle/width

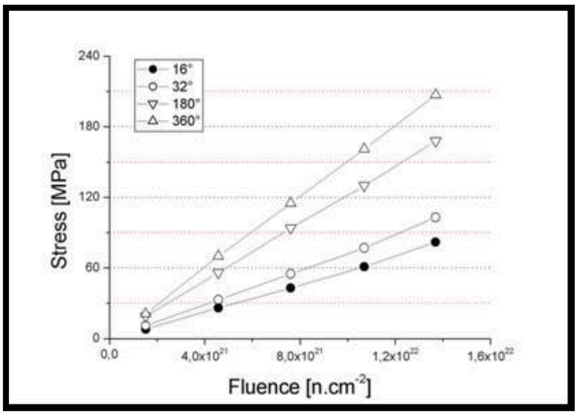
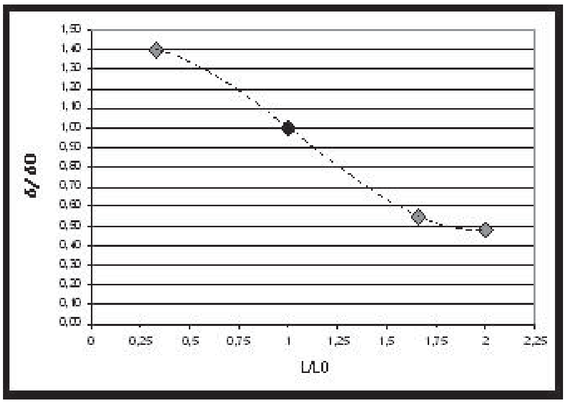
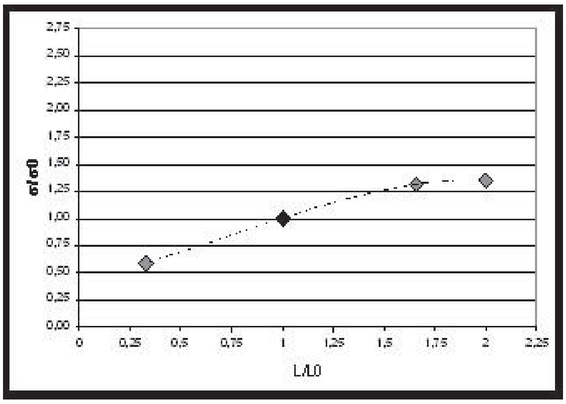


Fig. 10. Stress variation with reflector angle/width

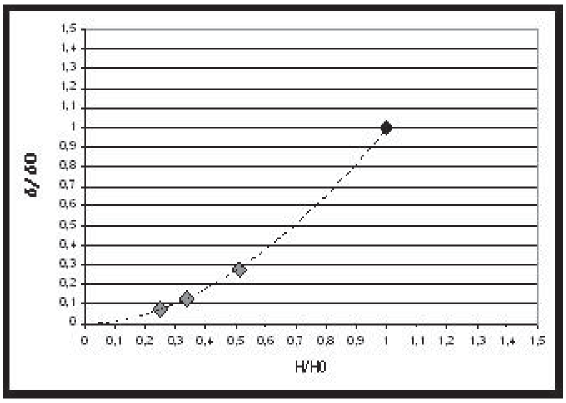
Graphs in Figure 11 show comparatively the relative influence of each geometric parameter on the deflection variation and the maximum stress generated by swelling. These graphs show how geometric parameters should be changed according to the reference condition (L/L0; H/H0; θ/θ0) for inducing a change in deflection and consequently in stresses (δ/δ0; σ/σ0). δ0 and σ0, respectively, refer to deflection and stresses generated for the reference case while L0, H0 and θ0 refer to the reference geometry. The dashed lines are only a tendency line, they are not intended to be a fitting line. The dark points indicate the reference condition (point (1,1)).



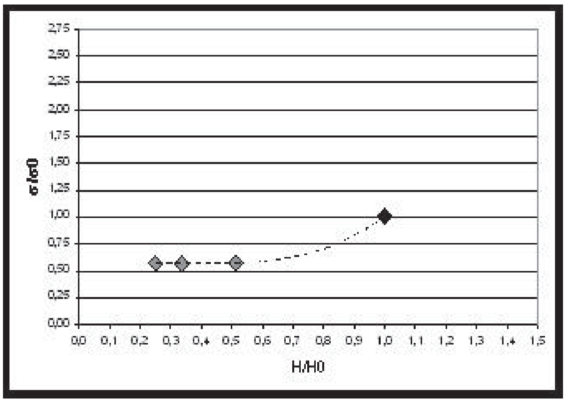
a)



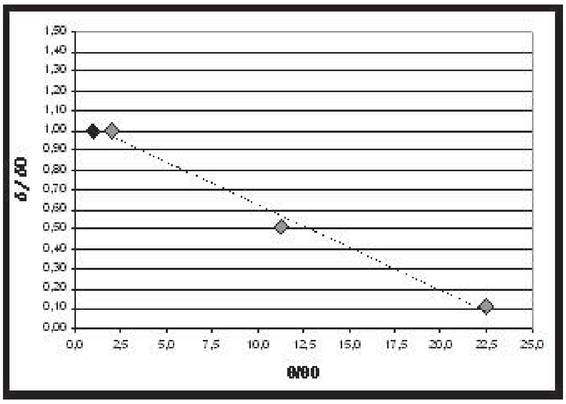
b)



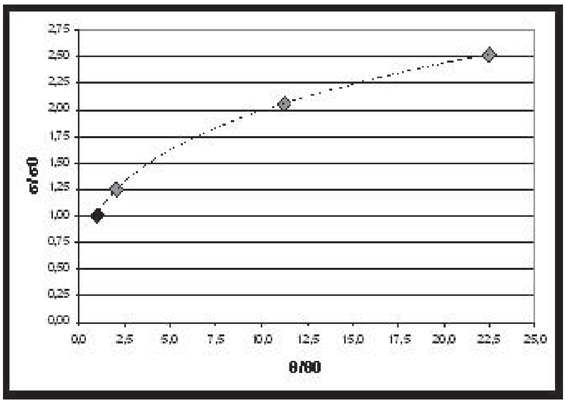
c)



d)



e)



f)

Fig. 11. Relative variation in δ and σ as per reference condition in function of relative variation of geometric parameters: a) δ with thickness, b) σ with thickness, c) δ with height, d) σ with height, e) δ with angle/width, f) σ with angle/width.

2.3 Discussion

As expressed in Equation 1, the swelling is proportional to fast neutron fluence. Thus, the face exposed to irradiation source, i.e. the highest fluence, has the largest volumetric strain. On the other hand, the opposite face has the smallest volumetric change due to swelling. This inhomogeneous change in volume causes deflection to the reflector's bar in the direction of the flux/fluence gradient, Figures 3 and 4. The deflection can produce contact between assemblies, difficulties during the exchange of components, and eventually the closure of cooling channels. This phenomenon was observed in at least two reactors before; the JMRT [19], where the chamber is a surveillance parameter, and the Plum Brook Reactor control rod [9].

This deflection may vary according to geometry and fluence. Deflection is inversely proportional to thickness increment (Figure 6). Such results seem consistent because the assembly has more geometrical constraint as assemblies' thickness increases. From Figures 11-a) and b), can be see that deflection and stress seem to reach a plateau for a double thickness bar.

A decrease in the assembly height implies a heavy reduction in deflection for all fluence (Figure 7); when height decreases to half the reference deflection value, it lowers to 0.3 of the original, Figure 11-c).

Concerning the generated stresses showed in Figure 8, an unexpected behavior is predicted by the model. When the bar height is taken down to half the reference case, the maximum stresses diminish up to approximately half of the initial value. However, a subsequent reduction in height does not result in lower stresses, Figure 11-d).

Angle variation refers to the change in the assembly width. In Figure 9 and 11-e) it can be seen that deflection does not vary for an assembly with a double width (32°). For a bar with 180° angle (i.e. a semi circle) deflection diminishes by half the original value approximately. For a 360° angle, geometric restriction is maximum and "deflection" reduces around 10 times in comparison to the reference case, Figure 11-e). In the last case, deflection is not the correct term because the assembly's geometry could be considered like a barreling distortion, as can be seen in Figure 12.

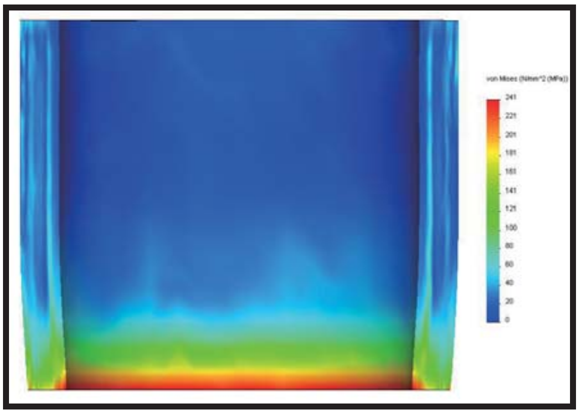


Fig. 12. View of stresses in the 360° case, deformation is magnified 100X

Concerning the stresses (Figure 10), it can be seen that as the angle contributes to the stresses increase due to the geometric constraint generated. For assemblies with 360°, complete ring, the stress is markedly high, reaching values close to the yield stress of the alloy used.

A comparative analysis can be made from Figure 11. The variation in deflection is shown in Figures 11-a), c) and e). When the height is half the reference case, the deflection drops to $0.3\delta_0$, whereas for reaching a reduction of $0.5\delta_0$, an increment of $L>1.5L_0$ and $\theta>11\theta_0$ is needed. On other hand, stresses increase when δ drops, as in the latter two cases, Figures 11-b) and f). The worst case occurs with a complete ring, where the stress is 2.5 times higher than the reference case, and it is close to the yield stress of Be alloy when fluence is about $1.4\text{x}10^{22}$ n.cm^{-2}, Figures 10 and 11-f). On the contrary, the stresses decrease when δ drops, as in the height variation case. Therefore, it can be concluded that for the simple geometry analyzed here, the most influencing parameter is the bar height.

3. SUMMARY

Beryllium structure is altered during neutron irradiation by displacement damage and by transmutation. This leads to embrittlement and swelling.

A brief review of the main high flux Research Reactors that use Beryllium components was made and it can be concluded that each reactor has adopted different strategies in order manage the problems associated to swelling: strengthening, cracking and distortion. The Be component lifespan ranges from 1 to $6.4\text{x}10^{22}$ n.cm^{-2} for fast neutron E>1MeV. This variation is mainly attributed to the material quality, impurities contents, especially BeO content and, manufacturing process. Nevertheless, geometry is not discussed by published literature.

A simplified analysis of swelling effect on a hypothetical Beryllium reflector was made varying its geometry in height, thickness and width.

The results showed that:

- The face exposed to the source of irradiation has larger volumetric strain than the opposite face leading to inhomogeneous strain and the reflector bar deflection in the direction of the fluence gradient.
- The deflection decreases when: thickness and angle/width growth and height diminishes.
- The stresses increase with deflection drop for thickness and angle/width variation cases.
- The worst case, from stresses point of view, is given for a complete Beryllium ring where the stresses are close to the yield strength when the fluence is about of $1.4x10^{22}$ $n.cm^{-2}$.
- For variation of height the stresses drop as the deflection decreases. Then, the height variation is the most sensible parameter.

Results summarized above help to understand the distortion behavior and the stresses generated in a simple geometry Be bar subjected to neutron iradiation. However, these can be turned into a useful tool for mechanical design of more complex components.

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4.3 IRRADIATION TESTS OF NUCLEAR GRADE GRAPHITE AT HIGH NEUTRON DOSE AND LOW TEMPERATURE AT HANARO

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The irradiation characteristics of nuclear grade graphite have been widely studied, particularly the dimensional changes and thermal conductivity, when used as a moderator in high temperature gas-cooled reactors. However, despite the extensive use of graphite in research reactors, there has historically been little irradiation data on isotropic nuclear grade graphite at low temperatures (below 150^{o}C) for high fast neutron fluence (~ 10^{25} n/m^2) relevant to research reactors. Some irradiation results of nuclear grade graphite have shown that thermal conductivity is remarkably reduced and significant irradiation growth occurs at low irradiation temperatures and high fast neutron fluence. These results tend to restrict the use of graphite as a reflector material in research reactors. Therefore, it is necessary to generate irradiation data for predicting the behavior and operating performance of nuclear grade graphite in new research reactors.

Pre- and post-irradiation characterization such as irradiation growth, thermal conductivity, and hardness properties will be conducted on two isotropic nuclear grades of graphite, IG-110 (TOYO TANSO Co., Ltd.) and NBG-17 (SGL Group). Each of two experimental capsules consists of over 80 graphite specimens irradiated in the HANARO reactor located at Korea Atomic Energy Research Institute (KAERI). The graphite specimens are canned in Type 316L stainless steel and filled with helium gas. Fast neutron fluence is expected to be in the range of 1.41 x 10^{25} n/m^2 and 5.28 x 10^{25} n/m^2 (E > 0.18 MeV). The irradiation temperature was calculated to be in the range of 70 to 120^{o}C depending on the location of the specimens and gap between the graphite specimen and canning material. These results will provide design data for a new research reactor being developed in Korea.

Irradiation Tests of Nuclear Grade Graphite at High Neutron Dose and Low Temperature at HANARO

2011. 12. 06

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Nuclear Materials Division
Korea Atomic Energy Research Institute

4th International Symposium of Materials Test Reactors, Oarai, Japan

Presentation Outline

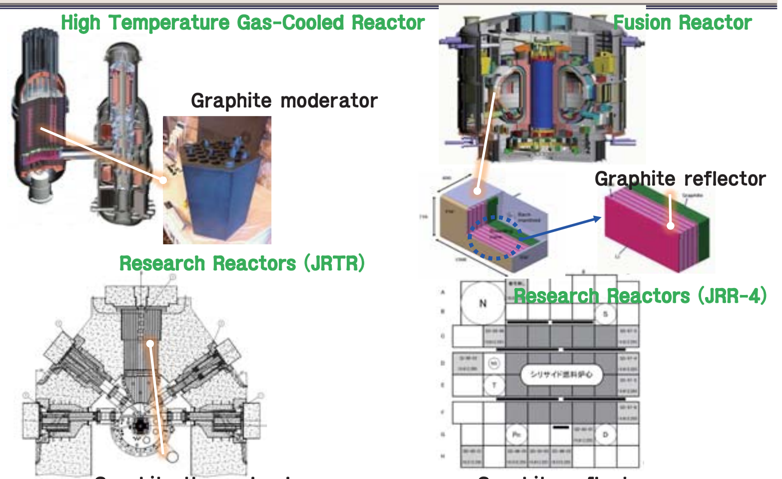
I. **Introduction**
 - Nuclear Grade Graphite
 - Irradiation Behaviors of Nuclear Grade Graphite

II. **Irradiation Test Plan at HANARO**
 - Irradiation Test
 - Irradiation Schedule
 - Pre-& Post-Irradiation Examinations

III. **Summary**

4th International Symposium of Materials Test Reactors, Oarai, Japan

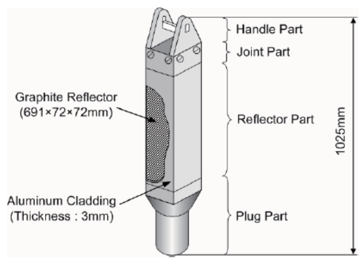
Nuclear Grade Graphite



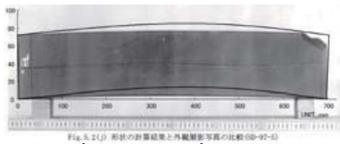
4th International Symposium of Materials Test Reactors, Oarai, Japan

Why the Interest in Graphite Irradiation?

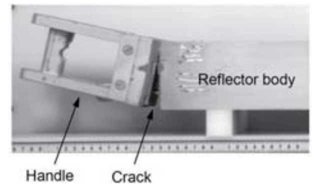
- **Large irradiation growth of graphite reflectors under JRR-4 operation condition (< 200ºC)**
 → failure of aluminum cladding due to irradiation growth of graphite



Reflector element of JRR-4



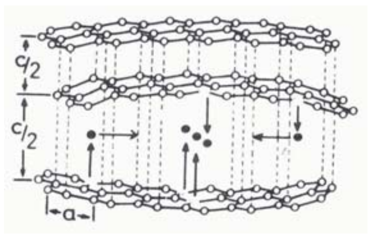
Buckling of graphite reflector

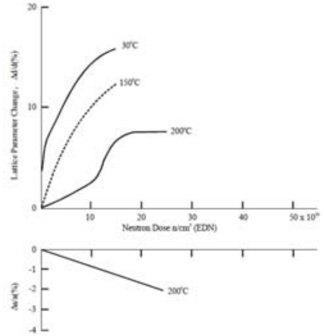


Failure of aluminum cladding

H. Horiguchi, et. al, *RRFM*, 2009

Irradiation Damage of Graphite

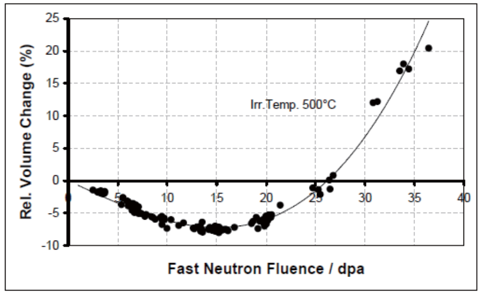




1. Displacement of atoms into interlayer positions due to neutron irradiation
2. Crystal undergoes <c>-axis expansion and <a>-axis shrinkage
3. Crystal structure changes lead to permanent dimensional changes in graphite.

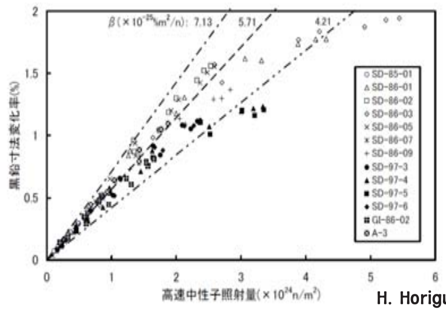
Dimensional Changes at High Temperatures

- **At High Temperatures above ~ 300ºC**
  - Initial shrinkage due to densification
  - High temperature (T_{irr}>300ºC) shrinkage with turnaround to swelling at higher doses
  - Higher dose turnaround into volume swelling due to incompatibility of crystal stains causing new pore generation



Dimensional Changes at Low Temperatures

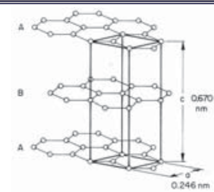
- **At Low Temperatures below ~ 200ºC**
  - No shrinkage below ~ 200ºC
  - Linear growth has been attributed to rapid accumulation of irradiation damage during irradiation at low temperatures (lack of vacancy mobility)
  - Maximum growth rate : 7.13 x 10^{-21} %cm²/n



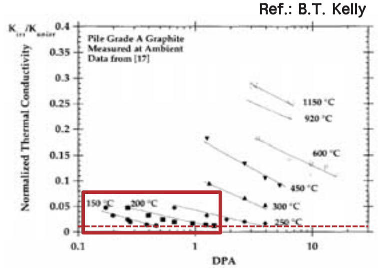
H. Horiguchi, et. al, *RRFM*, 2009

Irradiation Induced Changes in Thermal Conductivity

- **Generation of vacancies in carbon layers by neutron irradiation**
  - ➔ Reduction of harmonic oscillation (decrease in phonon mobility)
  - ➔ Bad thermal conductivity due to amorphorization



Ref.: B.T. Kelly



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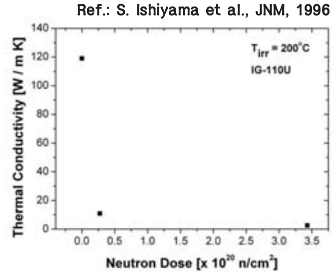


Material Selection

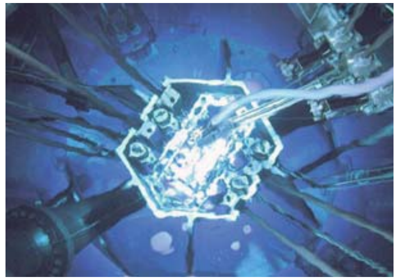
- **Isotropic Nuclear Grade Graphite**
  - IG-110, NBG-17 (ASTM D7210)

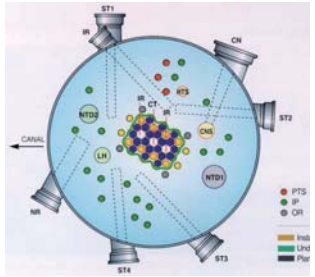
SGL (Germany)	Pitch Coke	Pet. Coke
Extruded	NBG-10	NBG-20
Iso-moulded	NBG-17	NBG-25

GRAFTECH (USA)	Pitch Coke	Pet. Coke
Extruded	PPEA	PCEA
Iso-moulded		PCIB-SFG

TOYO TANSO (JAPAN)	Pitch Coke	Pet. Coke
Extruded		
Iso-moulded	IG-430	IG-110

Test Holes for Material Irradiation at HANARO

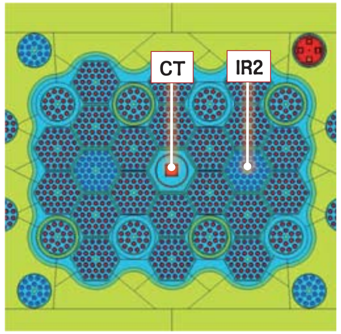




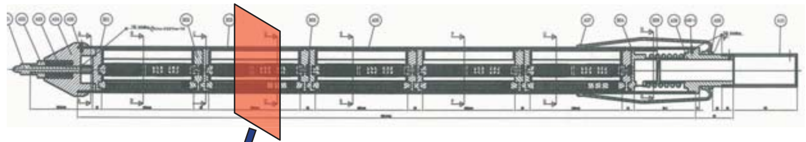
Location	Hole		Inside Dia. (cm)	Neutron Flux (n/cm^2 . sec)		Remarks
	Name	No.		Fast (>0.82 Mev)	Thermal (<0.625 ev)	
Core	CT	1	7.44	2.10 x 10^{14}	4.39 x 10^{14}	Fuel/material irradiation, isotope production
	IR	2	7.44	1.95 x 10^{14}	3.93 x 10^{14}	
	OR	4	6.00	2.23 x 10^{13}	3.36 x 10^{14}	

Irradiation

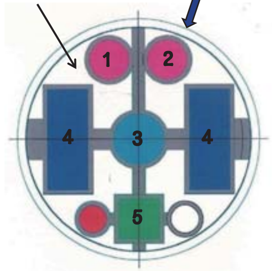
- **HANARO in-core position CT, IR2**
  - Expected fast neutron flux : 1.41 x 10^{21} to 5.28 x 10^{21} n/cm^2 (>0.18MeV)
- **Irradiation temperature**
  - 59.4°C - 133.1°C
  - Constant temperature by means of controlled gamma heat loss through an inert gas (helium) gap (1.1 mm)



Capsule Design Features



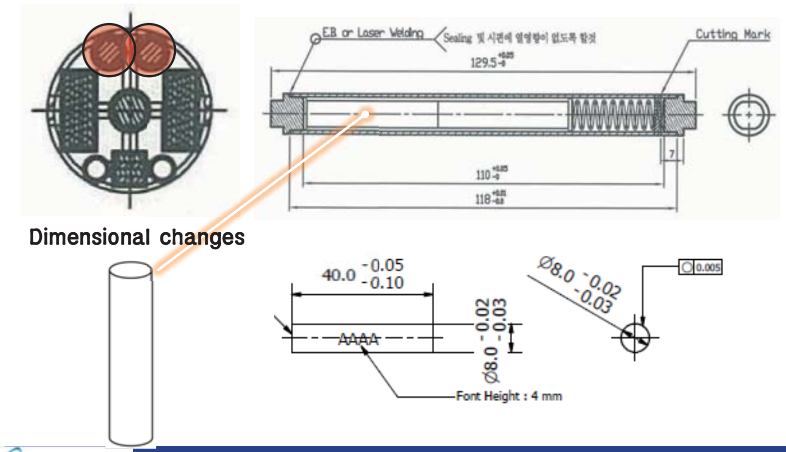
Coolant flow



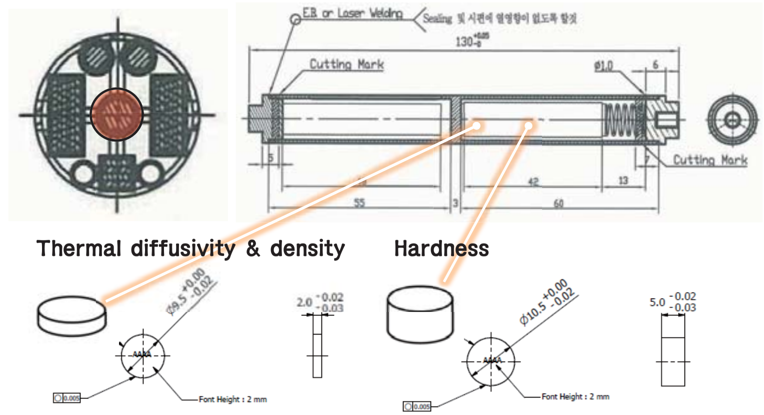
Capsule cross-section

1. Dimensional change of graphite (IG-110)
2. Dimensional change of graphite (NBG-17)
3. Thermal diffusivity, density, and hardness of graphite (IG-110 & NBG-17), Zr alloys & Be
4. Tensile properties of Zr alloys & Be
5. Swelling of Zr alloys & Be

Graphite Specimen Features



Graphite Specimen Features



Irradiation Temperature and Fast Neutron Dose

#1 capsule

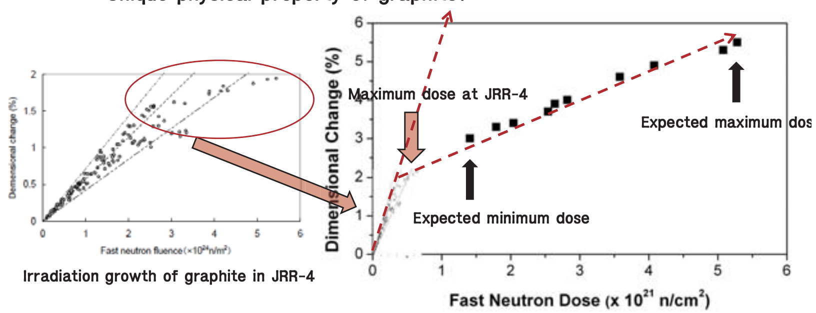
Fluence	**1.41×10^{21} n/cm^2**
Temp.	59 - 78°C
Fluence	2.04×10^{21} n/cm^2
Temp.	77 - 110°C
Fluence	2.64×10^{21} n/cm^2
Temp.	90 - 132°C
Fluence	2.54×10^{21} n/cm^2
Temp.	91 - 149°C
Fluence	1.79×10^{21} n/cm^2
Temp.	74 - 105°C

#2 capsule

Fluence	2.82×10^{21} n/cm^2
Temp.	59 - 78°C
Fluence	4.08×10^{21} n/cm^2
Temp.	77 - 110°C
Fluence	**5.28×10^{21} n/cm^2**
Temp.	90 - 132°C
Fluence	5.08×10^{21} n/cm^2
Temp.	91 - 149°C
Fluence	3.58×10^{21} n/cm^2
Temp.	74 - 105°C

Expected Dimensional Changes

- **Expected irradiation growth of IG-110 graphite at high neutron dose**
  - Maximum neutron dose (JRR-4) : ~ 5.5×10^{20} n/cm^2
  - Decrease in the growth rate after neutron dose of 2.5×10^{20} n/cm^2
    - Effect of compressive stress introduced by the aluminum cladding?
    - Unique physical property of graphite?



Irradiation growth of graphite in JRR-4

Post-Irradiation Examination

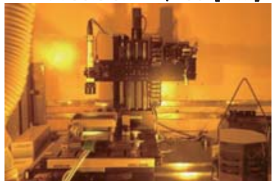
- Dimensions, R.T. (ASTM C781)
- Density, R.T. (ASTM E1461)
- Hardness, R.T (ASTM C886)
- Coefficient of diffusivity, R.T-T_{irr} (ASTM C781/E1461/C714, laser flash)

Optional:

- Coefficient of thermal expansion, R.T-T_{irr} (ASTM C781/E228)
- Crystallite sizes Lc, La (X-ray diffraction)
- Crystal growth and shrinkage (c-axis, a-axis, X-ray diffraction)
- Microstructural evolution (Mrozowski cracks, pores)
- Axial compression test

Hot Cell Facility for Post-irradiation Examination

Dimension measurement
- Precision : 1/100 [mm]



Thermal diffusivity test
- Laser flash method

Density measurement
- Immersion method

Irradiation Schedule

2011	2012	2013	2014

- Manufacturing and instrumentation of capsules (2 capsules)
- Preparation of graphite specimens

- Irradiation at HANARO
- Pre-irradiation examination

- Post-irradiation examination
  √ Density, thermal diffusivity,
  √ Dimensional change, hardness

Summary

- In order to use graphite for a reflector or moderator in research reactors, irradiation tests of nuclear grade graphite will be performed from 2012 to 2013 at low temperatures below 150°C and high fast neutron doses up to 5 x 10^{21} n/cm^2.
- Post-irradiation examination such as dimensional changes, thermal diffusivity, density, and hardness of nuclear graphites, IG-110 and NBG-17 will be performed.

5. Advancement of Irradiation Technology (2)

5.1 In-pile IASCC Growth Tests of Irradiated Stainless Steels in JMTR

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The Japan Atomic Energy Agency (JAEA) has an in-pile irradiation-assisted stress corrosion cracking (IASCC) test plan to evaluate in-situ effects of neutron/γ-ray irradiation on crack growth of irradiated stainless steels under high-temperature water conditions for commercial boiling water reactors (BWRs) using the Japan Materials Testing Reactor (JMTR). Crack growth rate and its electrochemical corrosion potential (ECP) dependence are different between in-pile test and post irradiation examination (PIE), but these differences are not fully understood. The objectives of the present study are to understand the difference between in-pile and out-of-pile IASCC growth and to confirm the effectiveness of mitigation due to lowering ECP on in-pile crack growth rates. For in-pile crack growth tests, we have selected a large compact tension specimen such as 0.5T-CT because of validity of SCC growth test at a high stress intensity factor (K-value). For loading a 0.5T-CT specimen up to K~30 MPa√m, we have adopted a lever type loading unit for in-pile crack growth tests in the JMTR. In this report, an in-pile test plan for crack growth of irradiated SUS316L stainless steels under simulated BWR conditions in the JMTR and current status of development of in-pile crack growth test techniques are presented.

Keywords: IASCC, In-pile test, stainless steel, JMTR, BWR, ECP, crack growth rate, 0.5T-CT specimen, lever type loading unit, SUS316L

1. INTRODUCTION

In order to ensure the integrity of reactor core internals for BWRs, effects of neutron irradiation on SCC have been evaluated by using mainly PIE results. For nuclear regulation in Japan, an evaluation guide for IASCC is now under construction by using PIE results [1, 2]. In the reactor core, however, in-situ effects of neutron and/or γ-ray irradiation on materials and water chemistry are present. Therefore, it is necessary to confirm whether the PIE results are appropriate for the evaluation of materials degradation in the reactor core.

2. IN-PILE IASCC GROWTH TESTS

The JAEA has a plan of in-pile IASCC growth tests for measuring crack growth rates (CGRs) of irradiated stainless steels under simulated BWR conditions in the JMTR. The objectives of this study are to understand the difference between in-pile and out-of pile IASCC growth and to confirm the effectiveness of mitigation due to lowering ECP on in-pile IASCC growth rates. These in-pile irradiation tests will lead to a more reasonable and reliable method for evaluating materials susceptibility to IASCC.

In this study, considering the previous in-pile and PIE results on SCC growth of irradiated stainless steels and the situation of Japanese commercial BWR plants, in-pile IASCC test parameters and their ranges have been selected as follows.

2.1 MATERIALS

A candidate area for evaluation of integrity to IASCC is around H4 weld line in Japanese BWR core shroud, because it is at the highest neutron fluence position in the core shroud. We have selected SUS316L stainless steel, which is used for most of the Japanese commercial BWR core shroud. There is no CGR data for in-pile tests of irradiated 316L stainless steels in the previous studies. In commercial BWR plants, SCC has been seen at the base metal and heat affected zone (HAZ) near the weld line of core shroud. For this reason we have selected the base metal and HAZ as test materials for irradiation.

2.2 FAST NEUTRON FLUENCE

In order to simulate a long-term operation, specimens will be pre-irradiated at a high-flux region (~1×10^{18} n/m^2/s) in the JMTR before the in-pile tests. We have selected a range of fast neutron fluence from 3×10^{24} to 3×10^{25} n/m^2 (E > 1 MeV). A large number of previous PIE data are available in this fluence range.

2.3 K-VALUE

A range of K-value from 10 to 30 MPa√m has been selected for the in-pile test, which covers the distribution of K-value estimated from the residual stress around the core shroud H4 weld line [3]. In order to compare the in-pile tests and PIE results precisely, the K dependences of CGR at a low K region (10 – 15 MPa√m) and a high K region (25 – 30 MPa√m) are important. The actual value at high Ks has to be determined in consideration of the validity of K-value for the irradiated specimen.

2.4 ECP

In order to clarify the ECP dependence of CGR, in-pile crack growth tests will be performed at two ECP levels (> 100 mV$_{SHE}$ and < -100 mV$_{SHE}$) in the JMTR. If a much lower ECP in the irradiation region is achievable, a CGR test will also be conducted at that level. In order to confirm the possibility of lower ECP, we will perform other in-pile test for measuring ECP directly in the irradiation region. The CGR response to ECP will be evaluated on the same specimen. It is noted that the value of ECP in the irradiation region depends not only on the irradiation dose rate and the inlet water condition such as dissolved oxygen (DO) and/or dissolved hydrogen (DH), but also on the water flow condition inside irradiation capsule. Thus, ECP inside the irradiation capsule will be monitored during the in-pile crack growth tests.

2.5 SPECIMEN

Due to validity of SCC growth test at a high K region, technical limitations such as the inner diameter of the irradiation capsule, and temperature increase by γ-heating, a 0.5T-CT specimen with thickness of 12.7 mm has been selected. To compare with the previous results, some 6.35-mm-thick specimens are also included. The properties of oxide films on the fracture surface of the specimen after the in-pile tests will be analyzed to understand the effects of the specimen thickness on the water condition inside the crack.

3. DEVELOPMENT OF IN-PILE CRACK GROWTH TEST TECHNIQUES

In this study, it is necessary to develop techniques to load a 0.5T-CT specimen up to K~30 MPa√m and to monitor the crack length by potential drop method (PDM) during irradiation in the JMTR. An in-pile crack growth test unit consists of a loading unit and a CT specimen with lead wires for PDM measurement. We have developed new lever type loading unit (Fig. 1) to apply a large load on a specimen.

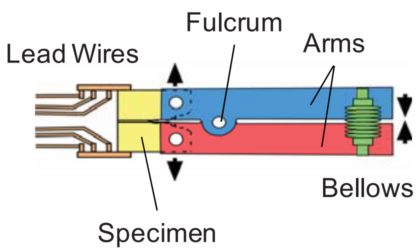

Fig.1. Lever type loading unit.

In this unit, a bellows shrinks to generate force by controlling inner helium gas pressure and an applied load on a specimen is determined by pressure difference, an effective diameter of a bellows and a lever ratio. Fig. 2 shows a schematic and a photo of a prototype in-pile crack growth test unit.

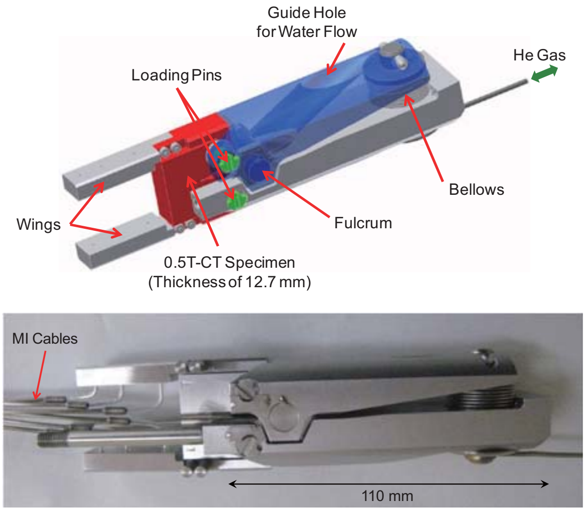

Fig.2. Schematic and photo of a prototype in-pile crack growth test unit.

Features of this unit are as follows;

- Lever type loading unit made of type 630 stainless steel using a bellows (lever ratio 1:6)
- Bellows is fixed to only one of the unit arms to avoid a compressive load on the specimen when the bellows expands.
- PDM measurement using mineral insulator (MI) cables
- Specimen with "the Wings", which facilitate a connection between a pre-irradiated CT specimen and MI cables for PDM measurement in hot-cell by remote controlling
- Guide hole can lead water flow to the specimen crack region.
- Two test units can be arranged in an irradiation capsule (Fig. 3).
- ECP measurement during in-pile test

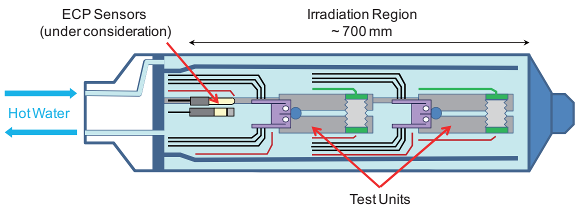

Fig.3. Arrangement image of in-pile test units in an irradiation capsule of JMTR.

3.1 LOADING UNIT FOR A 0.5T-CT SPECIMEN

In a lever type loading unit, a large applied load causes elastic strain of a specimen and unit arms, and consequently causes displacement of a bellows. Since the bellows has a property as a spring, repulsive force arises from displacement of itself and reduces a generated force by a pressure difference. Therefore, an estimation procedure of an applied load has to be considered with the repulsive force of the bellows spring.

Firstly, an applied load on a specimen position was measured by using a load cell, when mechanical force was applied on a bellows by a testing machine in the air at room temperature. Fig. 4 shows calculated values of applied loads on a specimen with and without the repulsive force of the bellows spring. As shown in the figure, an applied load on a specimen is reduced by the repulsive force (the red solid line).

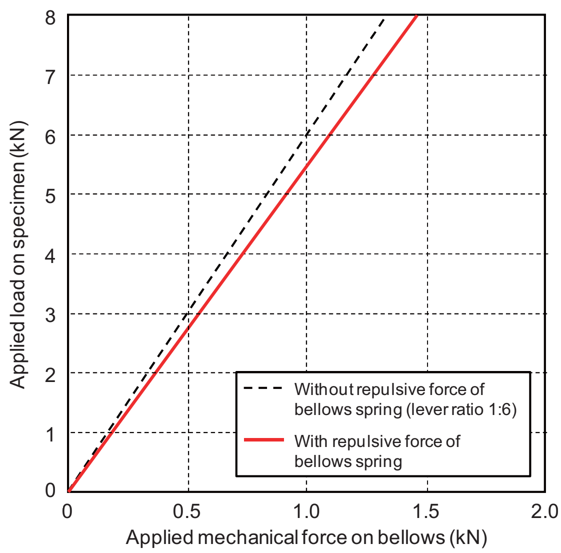

Fig.4. Calculated values of applied loads on a specimen with and without repulsive force of a bellows spring.

Fig. 5 shows a result of the load measurement test. A solid line denotes the calculated value with the repulsive force. The measured value indicates that loading up to ~7 kN corresponding to K~30 MPa$\sqrt{\text{m}}$ has been achieved, but there is hysteresis of applied load between loading and unloading processes. It is considered that the hysteresis arises mainly from friction around the fulcrum and bending/torsional rigidity of the fulcrum. The result also shows that an applied load on a specimen is different even at the same applied mechanical force and it depends on the history of loading-unloading processes. This result implies that strict control of an applied load with the operation procedure of the gas control facility is necessary.

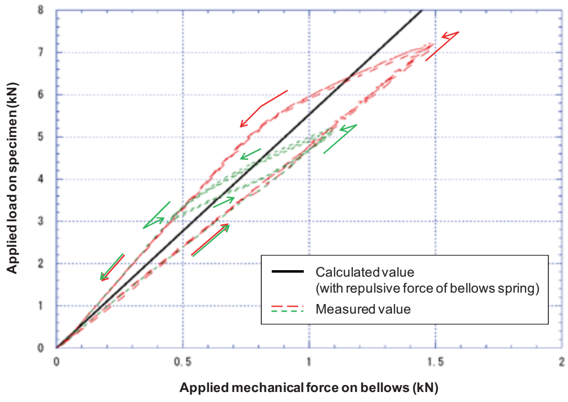

Fig.5. Load measurement test in the air at room temperature by using a testing machine.

Load measurement test in high pressure water at room temperature was performed by controlling inner gas pressure of the bellows as shown in Fig. 6. For the results of load measurement tests in the air, an applied mechanical force is converted to the corresponding pressure difference by using an effective diameter of the bellows. Almost the same tendency as load measurement tests in the air is indicated except for around a small pressure difference. It implies that an initial bending of the bellows may cause additional friction.

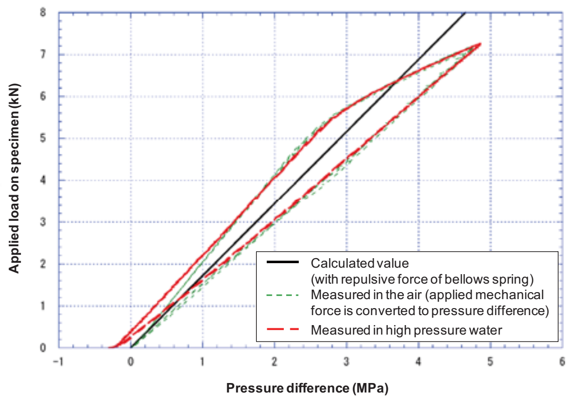

Fig.6. Load measurement test in high pressure water at room temperature by controlling inner gas pressure of a bellows.

Thus, in order to estimate an applied load on a CT specimen installed into an irradiation capsule during irradiation, we have to confirm a fundamental relationship between an applied load on a specimen and an applied mechanical force on the bellows in the air (or a pressure difference in high pressure water) for each loading unit in several ranges of loading/unloading processes, which simulate the actual crack growth test procedure.

3.2 CRACK LENGTH MONITORING BY PDM

We have determined a shape of a CT specimen with non-separate "Wings" (a non-separate-type CT specimen) and positions of lead wires for PDM measurement for an in-pile crack growth test unit as shown in Fig. 7. In fatigue crack growth tests at room temperature by using a non-separate-type CT specimen, a linear correlation between PDM signals and crack length was obtained (Fig. 8). For two measurement points of voltage, voltages of V_1 and V_2 were different due to length of current path (i.e. $V_2 > V_1$), but changes in voltage, ΔV_1 and ΔV_2, were almost the same. This indicates that ΔV_1 and ΔV_2 reflect only the increase in electrical resistance around a crack.

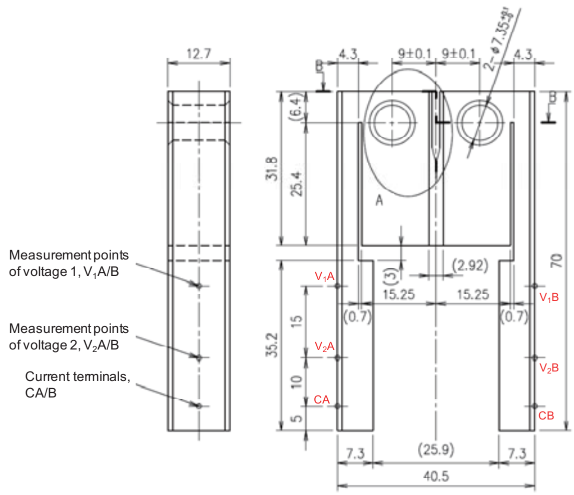

Fig.7. Shape of a CT specimen with non-separate "Wings" and positions of lead wires for PDM measurement.

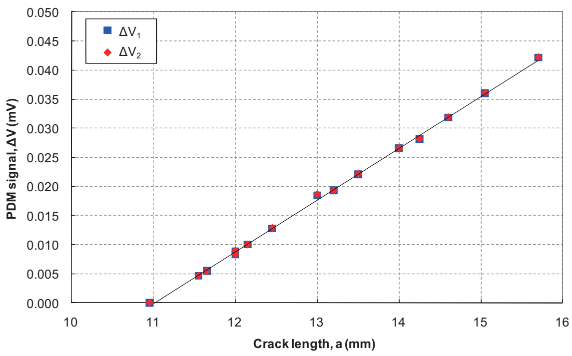

Fig.8. PDM measurement of a non-separate-type CT specimen during fatigue crack growth tests at room temperature.

In an irradiation capsule, MI cables for PDM measurement have to be terminated with stainless steel plugs (Fig. 9) because of no robust insulating materials in high temperature water under irradiation, therefore inner signal wires and outer sheaths are conducted electrically near the Wings.

Since MI cable sheaths are in touch with each other in an irradiation capsule and an applied current can escape through MI cable sheaths, a specimen current may be changed with a connected point of MI cable sheaths. For precise PDM measurement, it is necessary that a connected point of MI cable sheaths is fixed and as far as possible from the plugs of MI cables, and MI cable sheaths are insulated each other between a connected point and the plugs (Fig. 10).

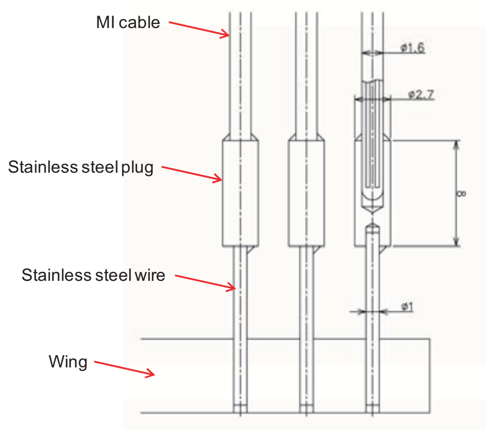

Fig.9. Connection between “the Wing” and MI cables for PDM measurement. An MI cable is terminated with a stainless steel plug and connected to a stainless steel wire.

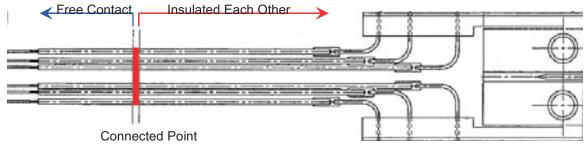

Fig.10. Connected point of MI cable sheaths.

In order to confirm where an appropriate connected point is, we have measured PDM signals with changing connected points of MI cable sheaths. Fig. 11 shows a result of time evolution of PDM signals when MI cable sheaths are not connected. As shown in the figure, we can compensate a temperature difference during PDM measurement by calculating V_1 / (V_2-V_1), which means a cancelation of a temperature-dependent electrical resistivity of the metal.

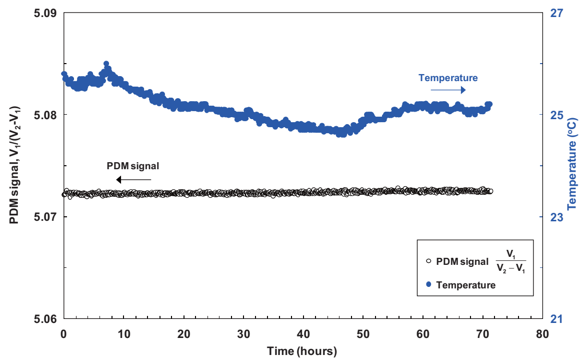

Fig.11. Time evolution of PDM signals when MI cable sheaths are not connected.

PDM signals and their standard deviations as a function of a connected point of MI cable sheaths from the plugs are shown in Figs. 12 and 13, respectively. The results indicate that an intensity and scattering of PDM signals are almost constant when a connected point of MI cable sheaths is above 0.5 m. Therefore, we have determined that a connected point of MI cable sheaths is 0.5 m from the plugs and MI cables in this region are insulated each other in an irradiation capsule.

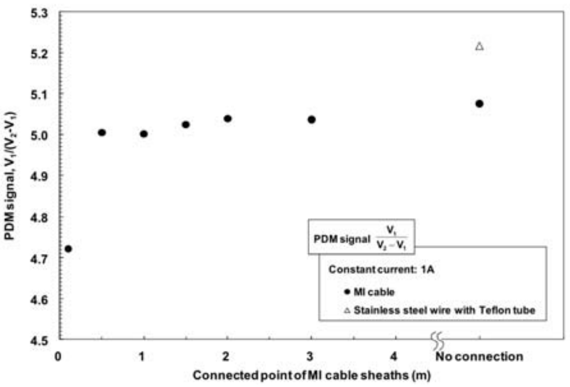

Fig.12. PDM signals as a function of connected point of MI cable sheaths.

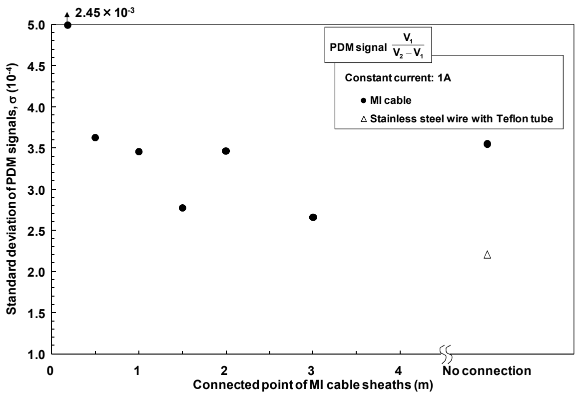

Fig.13. Standard deviations of PDM signals as a function of connected point of MI cable sheaths.

4. IN-PILE TEST SCHEDULE

The construction of the irradiation test facilities for BWR water conditions and a loading control system was finished in 2011. In-pile mock-up tests will be performed in 2012 with confirmation of a procedure of setting a CT specimen into a loading unit by remote controlling. Pre-irradiation of CT specimens will be performed in 2012 – 2014, and in-pile crack growth tests of irradiated CT specimens will be started in 2014.

5. SUMMARY

In-pile IASCC growth tests of irradiated SUS316L stainless steels under simulated BWR conditions in the JMTR are being planned to understand the difference between in-pile and out-of-pile crack growth behavior and the effect of ECP on in-pile crack growth. Based on the previous studies and conditions in commercial BWR plants, test parameters of the in-pile IASCC tests have been determined. For the in-pile crack growth tests, we have developed new lever type loading unit using a 0.5T-CT specimen with a thickness of B=12.7 mm up to K~30 MPa√m. A procedure of estimating an applied load on a CT specimen has been confirmed from the results of load measurement tests in the air and high pressure water at room temperature. From PDM measurements by using non-separate-type CT specimen with MI cables, we have confirmed a precision of the PDM signals. Out-of-pile mock-up tests for more precise estimation of an applied load and crack length at high temperature (~288°C) are now in progress.

ACKNOWLEDGEMENTS

The authors wish to thank Drs. T. Nakamura, H. Nakajima, Y. Kaji, J. Nakamura and K. Tsuchiya for helpful discussion on the irradiation test plan and the development of in-pile test techniques. A part of this study is being conducted under a contract with the Nuclear and Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI).

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5.2 Design Modification of the In-Pile Test Section for Increase of Sealing Capability

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Since KAERI established the fuel test loop (FTL) at HANARO in 2009, KAERI has carried out several experiments to verify the performances of the equipment. Based on the experiments, the design modification of the In-Pile test Section (IPS) has been processed to improve some difficulties such as difficulty in ejecting the inner assembly of the IPS from the pressure vessel, difficulty of the sealing process of the cooling water, etc. At first, because the cooling water of HANARO in KAERI consists of an open-pool type, if a certain shock is generated during the disassembly process, the cooling water can be spattered out of the pool. Therefore, two jacking bolts will be added on the top flange part of the inner assembly to decrease the shock. Second, at the pressure boundary of the IPS where MI-cables go through, the brazing process has been used to seal out the cooling water. However, because the length of the IPS is up to 5.5 meters, it is too difficult and time consuming to carry out the brazing process at the end part of the IPS. Therefore, the brazing process will be replaced with the mechanical sealing structure to simplify the assembly process.

Keywords: Fuel Test Loop (FTL), In-Pile test Section (IPS), jacking bolt, assembly structure, mechanical sealing, HANARO

1. INTRODUCTION

A 3-pin Fuel Test Loop (FTL) established at HANARO (High-flux Advanced Neutron Application Reactor) consists of an in-pile test section (IPS) and an out-pile system (OPS). IPS is located in the IR-1 irradiation hole of HANARO core to test the neutron irradiation characteristics and the thermal hydraulic characteristics of the nuclear fuel under the same condition with the commercial nuclear power plant [1-3]. That is, cooling water with high temperature (300˚C~310˚C) and high pressure (15.5MPa) flows through the IPS. Therefore, high attention needs to be paid on the sealing of cooling water in the IPS. As shown in Fig. 1, IPS can be distinguished as a pressure vessel and an inner assembly. The pressure vessel is used to provide a reliable pressure boundary and an effective heat insulation of the cooling water. The inner assembly should be installed in the pressure vessels during the irradiation test and it should be safely disassembled from pressure vessels when the test is finished. The inner assembly, the main part of IPS, consists of fuel rods, sensors and instrumentation lines, a flow guide of the coolant and supporting structures. In particular, instrumentation lines pass through the end part of the inner assembly and it needs to seal out the cooling water at the pressure boundary. In this paper, a safe disassemble structure of the inner assembly and pressure vessels and a mechanical sealing structure at a narrow pressure boundary where instrumentation lines pass through have been studied.

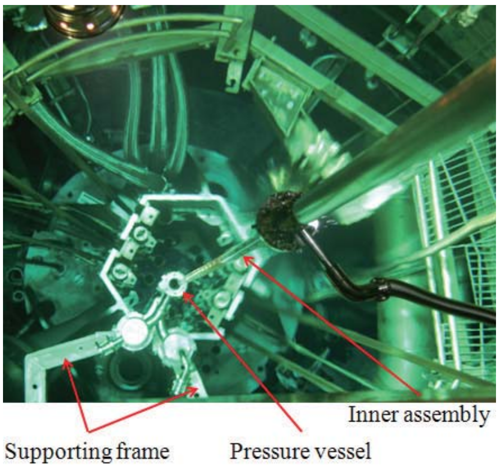
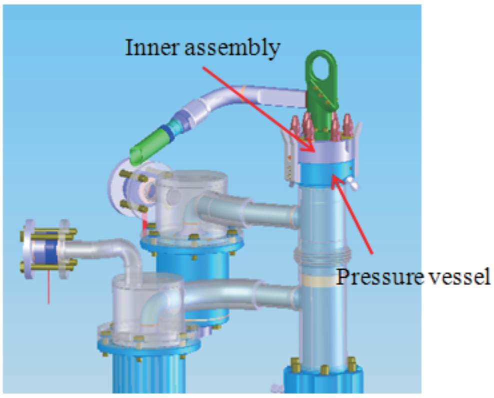


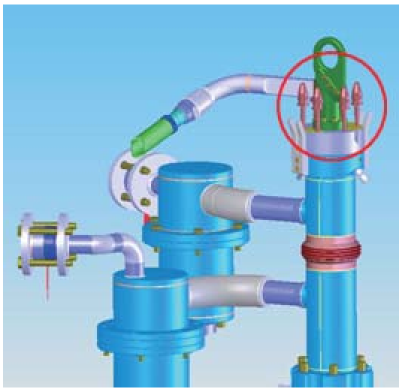
Fig.1. FTL at HANARO core

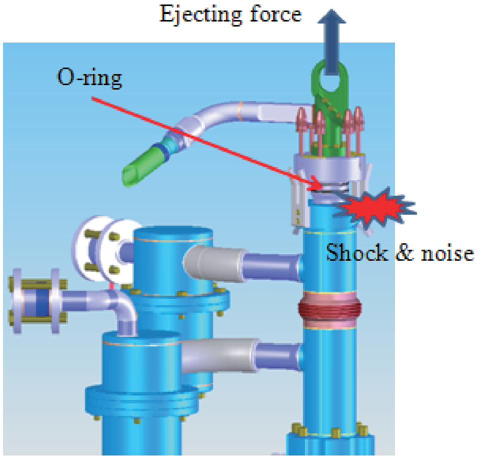
2. IMPROVEMENT OF SAFETY IN EJECTING INNER ASSEMBLY FROM PRESSURE VESSELS

All parts of IPS are sealed with high strength sealing components. Therefore, some parts are not easy to disassemble due to the resistant pressure of sealing components. In particular, when the inner assembly of the IPS is ejected from the pressure vessel, a big sound and shock are generated due to the sealing components (Fig.2).



(a)IPS assembly

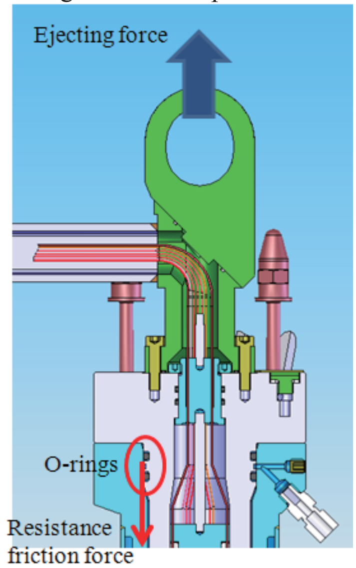

(b)Loosen fasten bolts



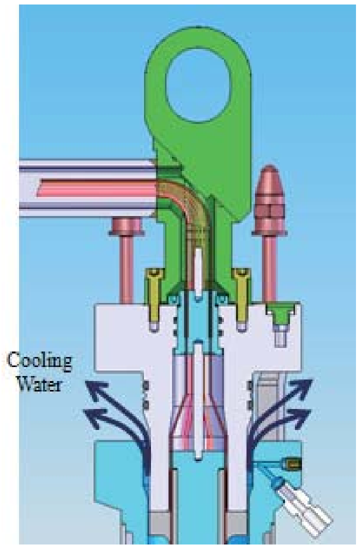
(c)Eject the inner assembly from the pressure vessel

Fig.2. Ejecting process of the inner assembly from the pressure vessel

As shown in Fig.3, O-rings are installed on the contact face between the inner assembly and pressure vessel to prevent the leakage of cooling water while running the rig. When assembling and disassembling the inner assembly, resistance friction force is generated due to the elastic property of the O-rings. At the moment when the inner assembly is detached from the pressure vessel, the resistance friction force disappears and a big sound including a certain shock is generated due to the pressure difference between the internal region of the loop and its outer region. Because cooling water of HANARO in KAERI consists of an open-pool type, if a certain shock is generated during the ejecting process, the cooling water can be spattered out of the pool. In addition, the cooling water remaining in the pressure vessel is poured out with an instant pressure difference of 15.4MPa. If there is no equipment to compensate the shock generated by the pressure difference, the cooling water in the rig can also be spattered out of the pool.



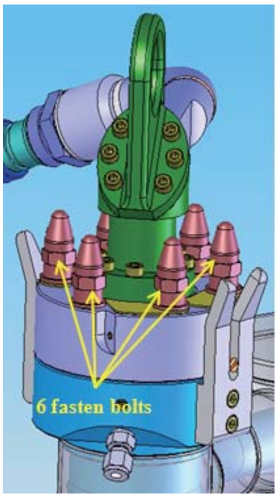
(a)Friction force induced on the sealing component



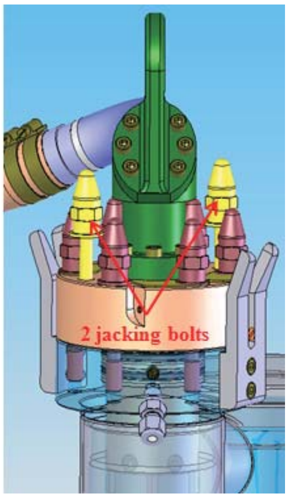
(b)Spouting of cooling water from pressure vessel

Fig.3. Section view of the sealing part of IPS

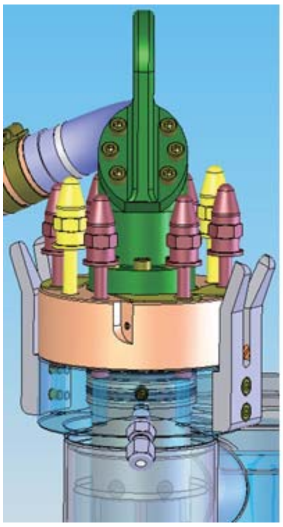
Therefore, in order to decrease the shock induced on the inner assembly, two jacking bolts will be installed on the top flange part of the inner assembly of the IPS where six fastening bolts are installed (Fig.4.(b)). The ejecting process of the inner assembly will be carried out by loosening the fasten bolts and jacking bolts sequentially (Fig.4.(c) and (d)). Then, fasten bolts will guide the inner assembly to avoid moving abruptly and it will also compensate for the shock generated by the sealing component.



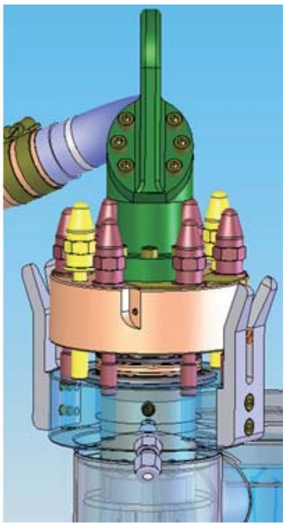
(a)Original design



(b)Add 2 jacking bolts



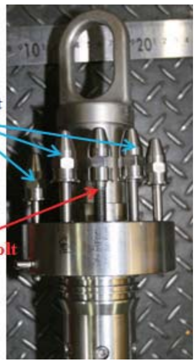
(c)Loosen fasten bolts



(d)Eject the inner assembly by fastening jacking bolts

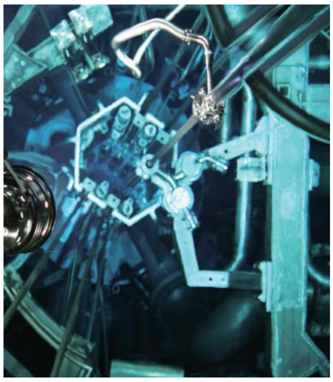
Fig.4. Operating mechanism of jacking bolts

An improved design was applied to the mock up test and showed a good performance. During the ejecting process of the inner assembly, noise was not generated and cooling water was not spattered out of the pool. Fig.5 shows the ejecting experiment of the inner assembly at HANARO with the mock up equipment.



(a)A test mock up of flange part with jacking bolt

(b)Loosen fasten/jacking bolts with a special tool



(c)Eject the inner assembly with a lifting tool

Fig.5. Ejecting experiment of the inner assembly at HANARO

3. MECHANICAL SEALING STRUCTURE AT THE PRESSURE BOUNDARY OF INTRUMENTATION FEED TRHOUGH

The IPS has 15 sensors such as a thermocouple, LVDT and SPND to instantly measure the fuel performance during the irradiation test. The sensors installed on the internal wall of the IPS are connected with MI-cables (Diameter 1.0mm, Length 1600mm) to deliver signals to the measuring device located in the outside of the IPS and go through the pressure boundary. The specification of the MI-cable is given in Table 1.

Table 1 The specification of MI-cable

	Material	Diameter
Sheath	AISI 316L	OD:1.0±0.01mm ID: ≒ 0.82mm
Insulator	Al_2O_3	
Wires (C-type)	Alloy 405[+] / Alloy426[-]	0.18mm

At the pressure boundary of the IPS where MI-cables pass through (Fig.6), the brazing process has been commonly used to seal out the cooling water [4]. However, because the length of the inner assembly is up to 5290mm, it is too difficult and time consuming to carry out the brazing process at the end part of the IPS as shown in Fig.7. In addition, if it fails the sealing test after brazing, the component including MI-cables should be scrapped out and it needs to re-assemble the inner assembly with a new spare component and MI-cables.

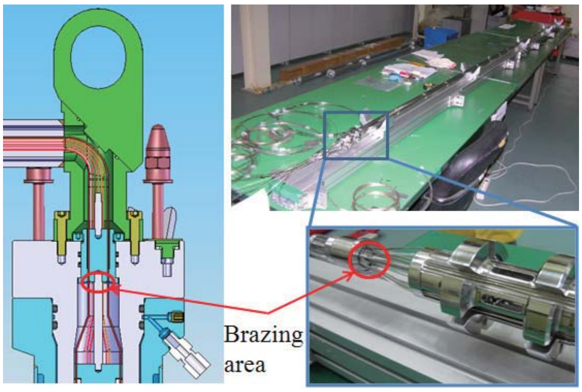


Fig.6. Brazing area of the inner assembly

Fig.7. Brazing process of MI-cables and sealing plug

Therefore, the sealing structure using the brazing process needs to be replaced with a mechanical sealing structure to simplify the assembly process. Generally, graphite is easy to be deformed with small stress and highly compressed graphite has a high sealing performance. As shown in Fig.8, a mock up sample was made by welding the top flange with a sealing plug which uses graphite sealant. When fastening torque induced on a sealing plug is delivered to the graphite through the sealing plug, the graphite is deformed and makes a pressure boundary with the MI-cables and sealing plugs.

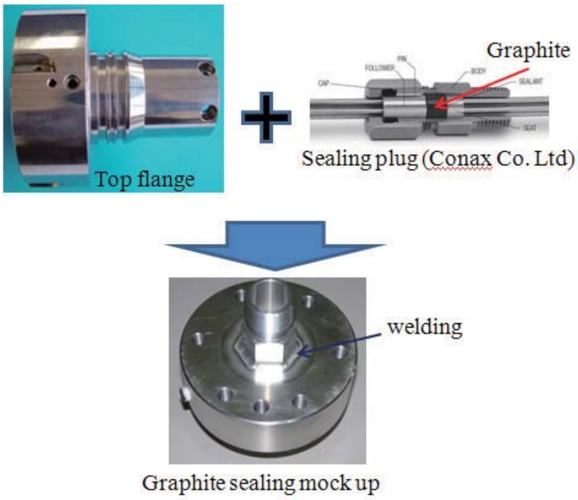
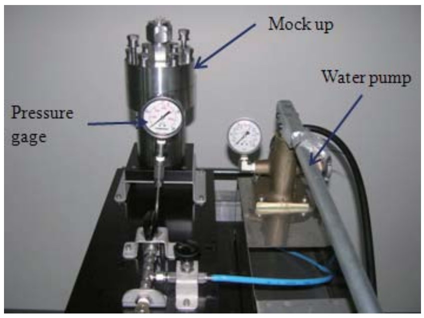


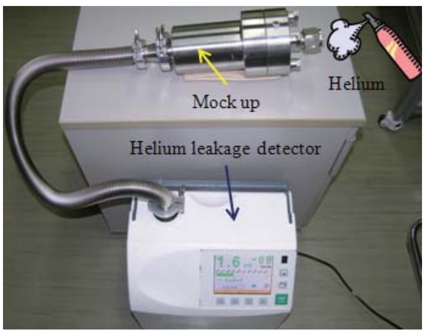
Fig.8. A test mock up of mechanical sealing

The design condition of the IPS is 17.5MPa, 350˚C and graphite sealing should satisfy the criterion designated on the ASME section III (125% of design pressure: ≥ 22.5MPa, Helium leakage test: 5×10^{-9} torr · liter / sec) [5].

In order to verify the performance of the graphite sealing, a mock up, a hydraulic pressure test device and a helium leakage test device were developed as shown in Fig.9.



(a)Hydraulic pressure test device



(b)Helium leakage test device

Fig.9. Equipments for sealing performance test

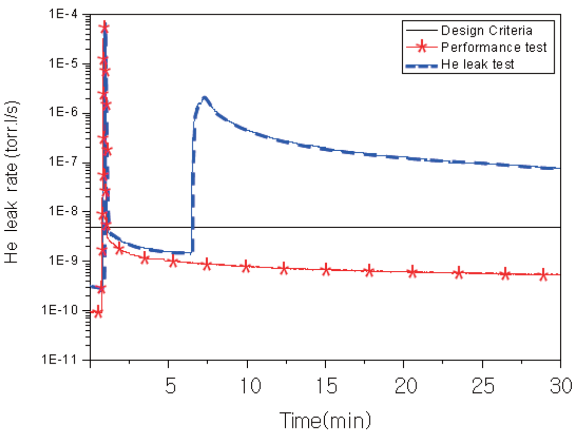
Hydraulic sealing criterion is used to check whether the mock up keeps the sealing status for 10minutes under 22.5MPa (125% of design pressure, 17.5MPa) of hydraulic pressure. An experiment was carried out by inducing an assembly torque on the pressure plug. Leakage was checked at 102 N·m, 129 N·m, 156 N·m, 183 N·m, 210 N·m and 237 N·m. Hydraulic pressure was induced with a manual pump. As listed in Table 2, it satisfies the sealing criterion (no pressure drop, no leakage) above 183 N·m of the assembly torque.

Table 2 Hydraulic pressure test

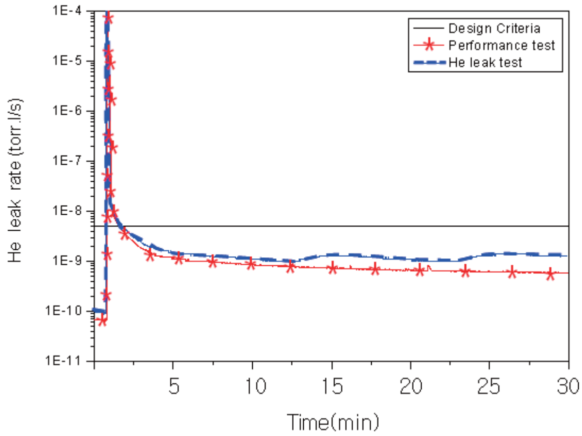
Assembly torque (N·m)	Pressure drop (22MPa, 10min.)	Leakage
102	22 MPa	O
129	0.4 MPa	O
156	0.2 MPa	O
183	0.0 MPa	X
210	0.0 MPa	X
237	0.0 MPa	X

A helium leakage test is used to check whether helium gas passes through the graphite sealing part at $1x10^{-3}$ torr of the vacuum state. After spraying helium gas at the outside of the vessel, if a helium detector (ASM310) connected with the opposite side of the vessel indicates a helium gas of more than $5x10^{-9}$ torr · liter / sec, it fails the helium leakage test. Fig.10 shows a graph of the helium leak rate at the designated assembly torque induced on the pressure plug. At 183 N·m of the assembly torque, the helium leak rate is $2.1x10^{-6}$ torr · liter / sec, which does not satisfy the criterion. However, at 237 N·m of the assembly torque, the helium leak rate reaches up to $1.4x10^{-9}$ torr · liter / sec, which satisfies the criterion.

From the results of sealing test, when graphite is compressed more than 237 N·m of the assembly torque, the sealing criteria is satisfied.



(a)Helium leak rate (at 183 N·m)



(b)Helium leak rate (at 237 N·m)

Fig.10. Helium leak rate at each assembly torque

4. CONCLUSIONS

In the study, design modification on the inner assembly part of the IPS has been carried out to improve the safety and working efficiency. At first, two jacking bolts are added on the flange part of the inner assembly to decrease the shock generated while ejecting the inner assembly. It helps cooling water not to spatter out of the pool and improves safety. Secondly, brazing at the feedthrough part of the instrumentation cables is replaced with a mechanical sealing design. Graphite is placed on the pressure boundary and compressed with a bolt-typed pressure plug to seal out the cooling water. A sealing performance test was carried out with a hydraulic pressure pump and helium leak detector. It satisfied all the design criteria at 237 N·m of the assembly torque. The modified design will be applied to make a new IPS in order to carry out an irradiation test at the loop.

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[5] ASME, "Testing," Boiler and pressure vessel code, sec III, subsection NB-6000

5.3 Status for Development of a Capsule and Instruments for High-Temperature Irradiation in HANARO

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As the reactors planned in the Gen-IV program will be operated at high temperature and under high neutron flux, the requirements for irradiation of materials at high temperature are recently being gradually increased. The irradiation tests of materials in HANARO up to the present have been performed usually at temperatures below 300℃ at which the RPV materials of the commercial reactors are being operated. To overcome the restriction for high-temperature use of Al thermal media of the existing standard capsule, a new capsule with double thermal media composed of two kinds of materials such as Al-Ti and Al-graphite was designed and fabricated as a more advanced capsule than the single thermal media capsule.

Keywords: HANARO, capsule, high-temperature irradiation, single thermal media, double thermal media, VHTR, SFR, Gen-IV nuclear system, instrumentation, thermocouple, micro-heater

1. Introduction

The nuclear systems have evolved in accordance with concerns over energy resource availability, climate change, energy security. Among them, Gen-IV nuclear systems are in the spotlight as future energy sources. It pursues sustainability, safety and cost-effectiveness, and proliferation risk reduction for future commercial development, which includes SCWR (Super critical water reactor), SFR (Sodium-cooled fast reactor), GFR (Gas-cooled fast reactor), LFR (Lead-cooled fast reactor), MSR (Mol-ten salt reactor), VHTR (Very high temperature reactor). KAERI (Korea Atomic Energy Research Institute) takes part in the development of VHTR and SFR among them[1]. The characteristics of these reactors are that the operating temperature and the neutron fluence are higher than those of the existing ones. At HANARO, the irradiation devices which require high-temperature and relatively low fluence were recently developed to meet the irradiation needs described above. The irradiation test which was required at I-NERI project (International Nuclear Research Initiative) of USA, was performed in the range of 380~450°C, 10^{18}~10^{19} n/cm^2 (E> 1.0 MeV) [2].

2. Irradiation test holes in HAHARO

In HANARO, there are 32 vertical and 7 horizontal test holes. The vertical test holes are used for irradiation tests. The horizontal holes are all embedded in the reflector area, which are used for neutron scattering. Among 32 vertical holes, 3 ones are located in the inner core, 4 in the outer core and 25 in the reflector tank.

The CT, IR1 and IR2 holes are used for the material irradiation test using high flux neutron and the fuel irradiation test requiring the high output density. The 4 OR holes are used for RI production and fuel burn-up test as they have the high thermal/epi-thermal neutron flux. There are various test holes in the reflector area, which are used in the applied fields using thermal neutrons. Among them, 17 IP holes, LH and HTS holes are used for RI production and 3 NAA holes for neutron activation analysis by using a pneumatic transfer system. 2 NTD holes, which are the biggest in HANARO, are used for Si doping for semiconductor production. Cold neutron source is installed in the CN hole.

To obtain the proposed neutron fluence by I-NERI, the holes in the out-core region were selected. The holes in the in-core area are used when higher neutron fluence is required. Fig. 1 shows the layout of the HANARO core.

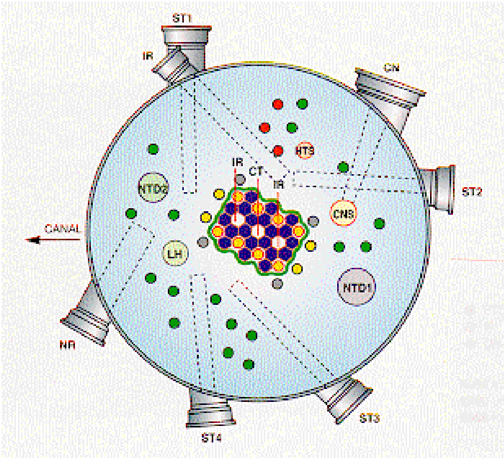

Fig. 1. Layout of HANARO core

3. High-temperature Irradiation Capsule

The reactors planned in the Gen-IV program will be operated at high temperature and flux as shown in Table 1. The outlet temperatures of VHTR and SFR are 1,000°C and 550°C respectively [3], which are much higher than the irradiation temperatures of material capsules tested in HANARO up to recently.

Table 1. Operation conditions of Gen-IV reactors

Type	Temp (℃)		Max. dose (dpa)	Pressure (MPa)	Coolant
	Inlet	Outlet			
PWR	290	320	100	16	Water
VHTR	600	1,000	1~10	7	He
SFR	370	550	200	0.1	Sodium

The capsule for high-temperature materials was designed as two types. One using single thermal media (specimen holder) made of materials such as Al, Ti, graphite, Mo, Ni. The other has double thermal media composed of two kinds of materials, in which the outer one is aluminum and the inner is made of material such as Al, Ti, graphite, Mo, Ni. The capsule with single thermal media was designed to be irradiated at temperatures up to 550℃. The capsule with double thermal media was designed for irradiation at temperatures up to 1,000℃. The first capsule with double thermal media, of which materials consist of Al, Ti and graphite, was irradiated at temperatures up to 700℃.

1) A capsule with single thermal media

This capsule was designed for irradiation of high-temperature materials with a relatively lower neutron fluence in the HANARO out-core region. As an alternative of Al thermal media, Zr, Fe, Ti and Mo in this capsule were used. Fig. 2 shows a single thermal media.

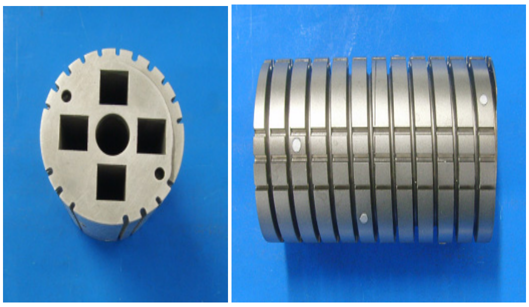
Fig. 2. A single thermal media

The hydraulic, vibration and thermal requirements for the inserts such as capsules at the OR test holes of HANARO were reviewed. The capsule with 56 mm in diameter was proved to satisfy all these requirements. In the thermal performance test for the φ56 mm capsule, the temperature of the specimens at the 2nd stage reached 400 °C at 101 kPa, and 527 °C at 13.3 kPa. It reached high temperature in the order of Fe, Ti, Mo, Zr and Al. The temperatures are inversely proportional to the values of the thermal conductance.

This capsule was utilized for irradiation of materials demanded in the I-NERI project included in the Gen-IV program of the USA. This work required an irradiation test in the range of 380~450℃ and 10^{18}~10^{19} n/cm^2 (E>1.0MeV). The irradiation test was undertaken in 2008 and 2009 at the OR5 hole. The specimens of 9Cr-1Mo material were installed at the first capsule and the second one 9Cr-1Mo-1W. They had 91 numbers of standard and 1/2-size Charpy, plate tensile specimens consisted of matrix, welded and HAZ (Heat Affected Zone) parts. The irradiation was conducted for 24 days (1cycle), and the temperatures of the specimens were maintained in the range of 400±10℃ and the neutron fluence (E>1.0 MeV) 1.1~4.4×10^{19} (n/cm^2), which are in the range to meet the values required in the project [4]. Fig 3 shows the irradiation temperatures of the I-NERI specimens measured by thermocouples at each stage during irradiation.

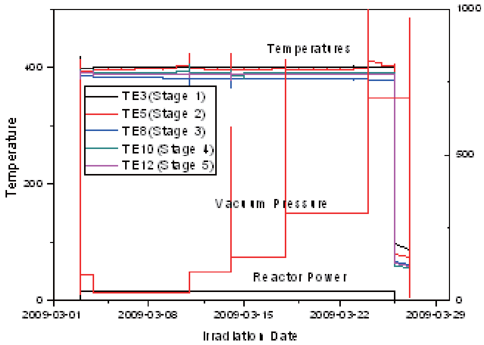

Fig. 3. Irradiation temperatures
(I-NERI specimens)

2) Capsule with double thermal media

A new capsule was fabricated for irradiation at temperatures higher than 700°C. Ti and Gr were selected as materials for the thermal media in this capsule. Ti is lower in density and price when compared with other materials such as Zr, Nb, or Mo, but the cross-section absorption of thermal neutrons is relatively high. Graphite was also selected as a candidate material because its mechanical properties are excellent at high temperature, and it is therefore used as the in-core material in VHTR. Fig. 4 shows double-layered thermal media composed of Al/Gr or Al/Ti. The gap between the holder and specimen is 0.1mm, while that between the inner and outer thermal media is 0.15mm, and between the outer thermal media and outer tube the gap is 0.16-0.36mm, which was designed to effectively control the temperature of each stage. All gaps were filled with He gas at 101kPa.

Fig 4. Double-layered thermal media (Al/Gr or Ti)

The appearance of the capsule is shown in Fig. 5. The rod tip of the bottom guide tube was assembled with a receptacle in the reactor core, and the protection and guide tubes play the role of a guide for various signal lines such as thermocouples, micro-heaters, and helium supply tubes. The main body is a major part of the capsule in which specimens, measuring devices, and various components were installed, and it includes an external tube with a cylindrical shell 60 mm in external diameter, 2.0 mm in thickness, and 870 mm in length.

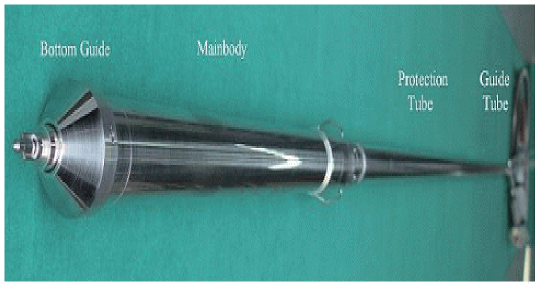

Fig. 5. Appearance of material capsule

As the double thermal media were new kinds of structure and Gr, Ti were at first irradiated material in HANARO, the evaluation of nuclear characteristics was important in the design of the capsule and the safety of irradiation. The reactivity was calculated to be 9.6 mk even if the thermal media were all Ti, which will make it highest. Therefore, the irradiation test was proved to be safe as it is less than +12.5 mk of the limit value required in HANARO [5].

ANSYS program was used for the thermal analysis. The two-dimensional model for the specimen section was generated. The temperature of the cooling water in the reactor in-core is about 33 ℃ and the heat transfer coefficient at the outer surface of the external tube is 30.3 x 10^3 W/m^2℃, which was experimentally determined [6].

Fig 6 and 7 show the irradiation temperatures of specimens, which were measured when temperatures were rising with the increase of reactor power and reached at the equilibrium at the HANARO full power of 30MW.

The temperature of specimen was rising up to 635 ℃ at the stage 2 at the condition of 760 torr without no heater power. The gamma flux is most high at the position on the way of reactor power-up initially at HANARO. The temperatures of specimens were controlled by the internal He pressure and micro-heater according to user's requirement. The temperatures of each stage were adjusted to 600~900 ℃ as shown in the Fig. 7.

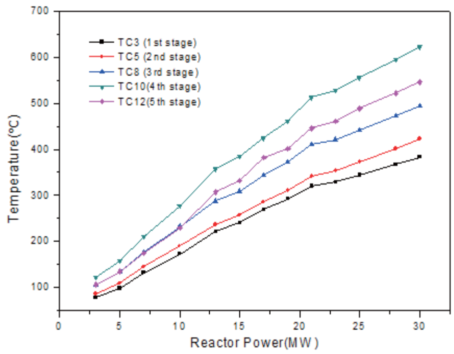

Fig. 6. Irradiation temperatures
(Initial at 760 torr)

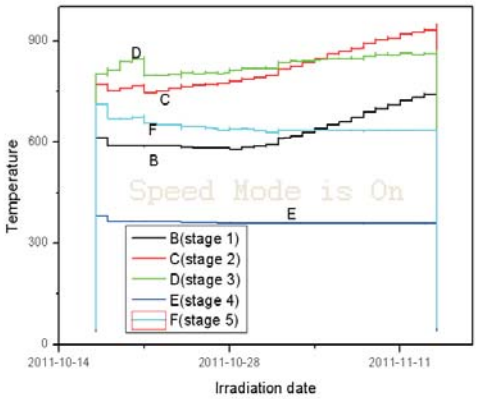

Fig. 7. Irradiation temperatures
(He pressure of 30~50 torr)

Fig. 8 shows the temperature distribution calculated at the stage 3 as an example of the results of temperature analysis. It shows the temperature of the specimen can reach up to 970 °C and aluminum thermal media 525 °C with a control of internal He pressure during the normal operation of 30MW. The neutron flux at the position of stage 3 is the highest at HANARO. The outer gap between the external tube and the outer thermal media also plays an important role on the control of the temperatures of specimens by a change of the helium pressure in the analysis.

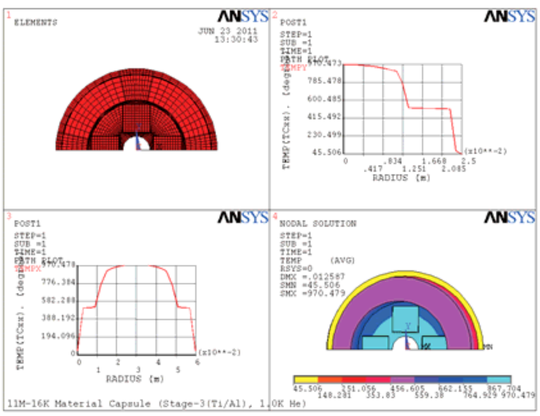

Fig. 8. Temperature distribution at stage 3

Measured and calculated temperatures of the capsule specimens are listed in Table 2. The ANSYS and GENGTC code were used for the calculation of temperatures. In the experiences, the temperatures by ANSYS code show a little higher than by GENGTC code. In comparison with the calculated and measured temperatures of the specimens, all differences are less than 10%. The maximum difference is 8.6% at the stage 3. It showed that the forecast by a code coincides with the measured ones.

Table 2. Comparison of specimen temperatures between measurement and analysis

Stage	Gap size (mm)		Temperature at 101kPa		Error (%)
	g2	g3	Calculat -ed	Measur- ed	
1	0.4	0.65	554	548	1.1
2	0.5	0.45	678	635	6.8
3	0.2	0.30	661	614	7.6
4	0.2	0.32	625	580	7.7

Error=(Calculated-Measured)/Measured x 100(%)

3. Instrumentation

Thermocouples and heaters of which the sheath materials are STS 310 and Inconel 600 were selected for use at temperature up to 1150 ℃ [7] instead of 304L which has been used to the standard capsule for medium temperature irradiation. The characteristics of the instruments used for the standard and high-temperature capsules are listed in Table 3.

The instruments penetrate at the top end plug of the capsule and are sealed by brazing for water resistance. Brazing is a manual task in which a welder applies heat to melt a filler material. Therefore, the experience and technique of the welder are important for proper capsule brazing. The instruments in an Inconel 600 sheath were broken during brazing and the filler material did not stick well to the base material. Among the many material capsules fabricated until now, the sheath material of the instruments was the first to break. Fig. 9 shows the normal and broken surface after brazing.

Table 3. Characteristics of Insturments

	Standard capsule	High temperature capsule
Irradiation temperature	~400℃	700~1,000℃
Thermocouple		
• manufacturer	Thermocoax	Thermocoax
•sheath	STS 304L	STS 310/ Inconel
• working T	700℃	1,150℃
Heater		
• manufacturer	Thermocoax	Thermocoax
• sheath	STS 304L	STS 310/ Inconel
• working T	600℃	1,150℃

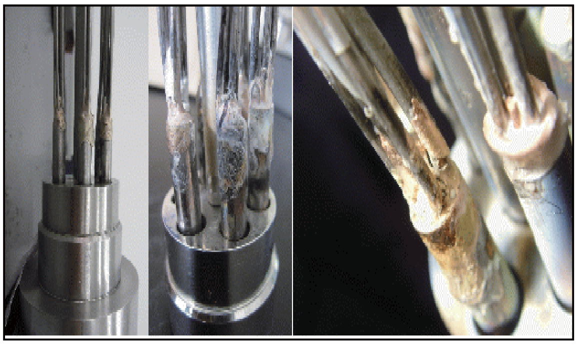

Fig. 9. Intact and damaged surfaces after brazing

The temperature of brazing flame used in the standard capsule was higher than 1,427℃. This is contrary to the principle of brazing, as the temperature is higher than the melting point of base material (STS 304L) of a standard capsule. The flame can be maintained at lower than 1,300℃ in the torch using butane gas. The fracture at brazing is dependent on the temperature of flame and the twisting angle of instruments.

STS 310 material was finally selected as an alternative of Inconel 600 material, which is easy to break due to an increase in brittleness at high temperatures. The results of brazing test for the Inconel and STS 310 materials are shown in Fig. 10 and 11. The integrities of the instruments sheathed in an STS 310 material are maintained even if when twisted up to 720°. However, the instruments housed in an Inconel sheath were damaged during brazing with a 480° twist. Therefore, Inconel material is inappropriate for use in high temperature instruments. In conclusion, STS 310 material was chosen for use with high temperature instruments.

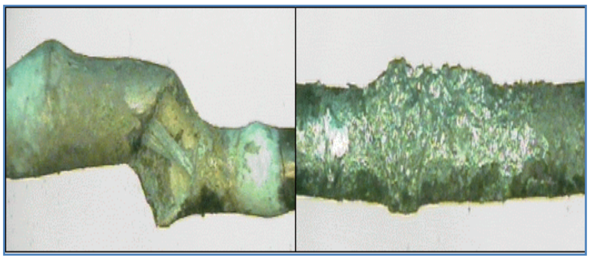

Fig. 10. Brazing after a twist(Inconel/STS310)

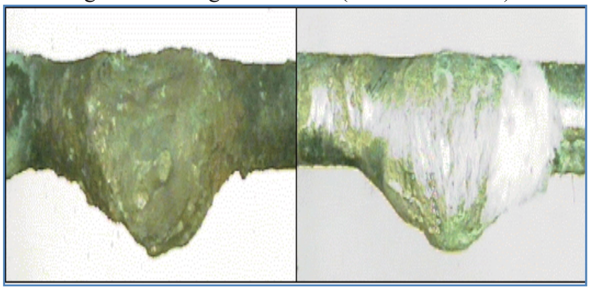

Fig. 11. Status after flaming heating without twist (Inconel/STS310)

4. Fluence Monitor for High-temperature

To measure the neutron flux at high temperature, the quartz or Zr-4 tube instead of Al was used as the outer capsule containing the specimens of Fe, Ni, Ti material. Then, the container will not melt as it will be used at temperature below 1,000℃. The quartz material was first used as a F/M container for the high-temperature capsule. A quartz material must be heated to high temperature to make a container with the help of skilled glass craftsmen. In addition, the quartz container should be broken to take out the specimen after irradiation. In the future, Zircaloy material will be used to replace the quartz container. Fig. 12 shows the conceptual shape of High-temperature F/M.

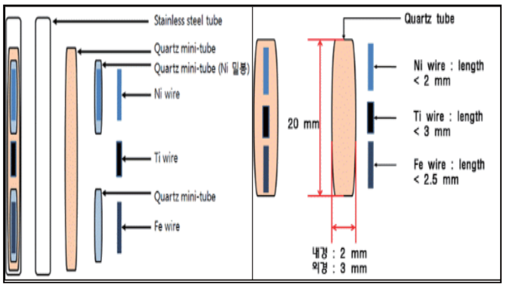

Fig. 12. High-temperature F/M

4. Conclusions

In accordance with the development plan of future nuclear systems in Korea, which are to be operated at high temperatures, a capsule suitable for irradiation testing at high temperatures was developed to overcome restrictions in use of aluminum at high temperature. The new capsule with a double-layered thermal media structure, of which the outer section is aluminum and the inner section is Ti or graphite, was fabricated. The high-temperature instruments of STS 310 material were used in the new capsule. Also, the technology for the brazing of instruments at very high temperature was developed.

Acknowledgements

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6. Expansion of Industry Use (RI)

6.1 Status of ^{99}Mo-^{99m}Tc Production Development by (n, γ) Reaction

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Technetium-99m (^{99m}Tc) is one of the most commonly used radioisotopes in the field of nuclear medicine. As one of effective uses of the JMTR, JAEA has a plan to produce ^{99}Mo by (n, γ) method, a parent nuclide of ^{99m}Tc. JAEA has performed R&D on production method of ^{99}Mo-^{99m}Tc in JMTR cooperating with foreign organizations and relevant Japanese enterprises under the cooperation programs. In this paper, present status of R&D for production of ^{99}Mo-^{99m}Tc in JMTR under international cooperation is introduced and constructions of the irradiation and PIE facilities at the JMTR site are also described.

Keywords: Radiopharmaceutical, ^{99}Mo-^{99m}Tc Production, (n, γ) method, Irradiation target, Mo-Tc generator, Solvent extraction, ^{99m}Tc concentration, Mo recycle

1. Introduction

Medical imaging techniques using technetium -99m (^{99m}Tc; $T_{1/2}$=6h) account for roughly 80% of all nuclear medicine procedures, representing over 30 million examinations worldwide every year. Thus, ^{99m}Tc is most commonly used radiopharmaceuticals in the field of nuclear medicine.

^{99m}Tc is generated by decay of molybdenum-99 (^{99}Mo; $T_{1/2}$=66h). Production of ^{99}Mo is carried out by fission reaction ((n, f) method) with high enriched uranium targets and five research reactors commissioned between 45 and 55 years ago are currently producing to meet 90 to 95% of global supply of ^{99}Mo. Recently, this supply chain is indicated that it is difficult to carry out a stable supply of ^{99}Mo/^{99m}Tc for some problems such as aging of these reactors and obstructions in transportation. Responding these circumstances, the OECD-NEA Steering Committee established the High-level Group on the Security of Supply of Medical Radioisotopes (HLG-MR) in 2009 and the report on “The Supply of Medical Radioisotopes” was published in June, 2011 [1-3].

On the other hand, the renewed JMTR will be started from the later half of JFY2012, and it is expected to contribute to various fields. As one of effective uses, JAEA has a plan to produce ^{99}Mo in JMTR [4-5]. In case of Japan, supply of ^{99}Mo depends only on imports from foreign countries, therefore JAEA has been performed R&D on ^{99}Mo production by (n, γ) method in JMTR cooperating with foreign countries and Japanese manufacturers under the cooperation programs.

In this paper, the status of R&D for the production of ^{99}Mo-^{99m}Tc under international cooperation is introduced. Construction of the irradiation and PIE facilities are also described.

2. R&D Items for Establishment of ^{99}Mo Production by (n, γ) Method

Virtually all ^{99}Mo used over the world is fission Mo (^{99}Mo produced by (n, f) method) and four companies have shared more than 95% of the world ^{99}Mo supply using five reactors and four post processing facilities. On the other hand, there are many other reactors over the world capable of producing the ^{99}Mo. Thus, the risk of the shortage can be significantly reduced if the ^{99}Mo production is shared by many reactors. However, the production of fission Mo at many reactors has difficulties in terms of safety, nuclear proliferation resistance and waste management.

The (n, γ)^{99}Mo (^{99}Mo produced by (n, γ) method) had been used at the initial era of ^{99}Mo until the fission Mo became available. It is remained for local supplies of very limited demands. The (n, γ)^{99}Mo has several advantages compared to the fission Mo, but the extremely low specific activity makes its uses less convenient than the fission Mo [6].

There are some subjects for the establishment of ^{99}Mo production by (n, γ) method and the main R&D items are as follows;

(1) Fabrication of irradiation target such as sintered MoO_3 pellets,
(2) Extraction and concentration of ^{99m}Tc from Mo solution,
(3) Examination of ^{99m}Tc solution for medical use,
(4) Mo recycling from the used Mo generators and solutions.

3. Status for ^{99}Mo Production by (n, γ) Method

The method of production for (n, γ)^{99}Mo has been developed by some reactor sites and the ^{99}Mo is produced and supplied from these reactors though the amount of (n, γ)^{99}Mo is smaller than that of fission ^{99}Mo. The developmental status of (n, γ)^{99}Mo is described in this section.

3.1 Irradiation Targets

Molybdenum oxide (MoO_3) is the most commonly used chemical form as irradiation target for the (n, γ)^{99}Mo production. Table 1 shows the characteristics of MoO_3. This material is chemically stable and its processing is easily performed by dissolving in the NaOH. On the other hand, MoO_3 can be sublimated at relatively low temperature. Figure 1 shows the phase diagram of Mo-O system [7]. Thus, the MoO_3 powder is usually irradiated in the reactor and (n, γ)^{99}Mo is generated. In this case, it is necessary to have large irradiation volume in the irradiation capsule. In JAEA, the MoO_3 pellet form with high density has been developed for large quantity (n, γ)^{99}Mo production. The preliminary fabrication tests of MoO_3 pellets were carried out in JAEA [8]. MoO_3 pellets were fabricated by the cold pressing and sintering method and the Spark Plasma Sintering (SPS) method. As a result, high density MoO_3 pellets were fabricated by the SPS method and the characteristics such as chemical, thermal and dissolving properties were confirmed.

Table 1 Material Properties of MoO_3

Properties	Values
Molecular weight	143.94
Density	4.696 g/cm^3(4°C/26°C)
Melting point	795°C
Boiling point	1155°C
Solubility in water	0.14g/100g (20°C)

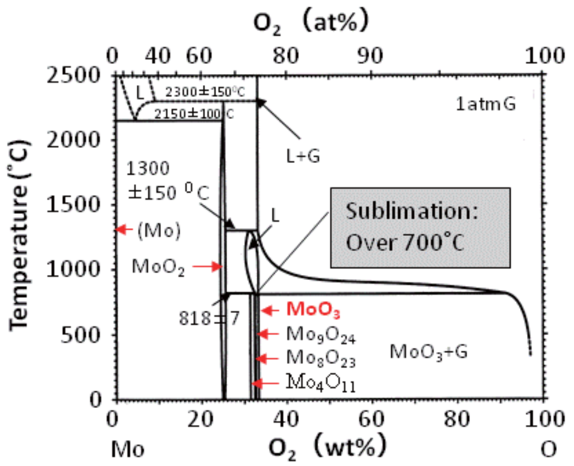

Fig.1 Phase diagram of Mo-O system.

3.2 Extraction and Concentration of ^{99m}Tc

Actually, the production capacity is low because of much lower ^{98}Mo(n, γ) cross section compared to ^{235}U fission and the fraction of ^{99}Mo in Mo is a few ppm at the end of irradiation even at high flux reactors. Thus, it is necessary for utilization by the (n, γ)^{99}Mo to develop the extraction and concentration methods of ^{99m}Tc through Mo-Tc generators or Mo-Tc extraction devices.

It is important to establish the separation and extraction methods as described in the item (2) in section 2 therefore the experiments and information exchanges about some methods have been carried out under the international cooperation. A joint research between BATAN and JAEA started in 2009, and performance tests and evaluation by the PZC column and solvent extraction with MEK have been carried out in BATAN. On the other hand, a joint research between INP-NNC-RK and JAEA also started in 2009, and joint experiments have been carried out for the comparison with separation and extraction properties of ^{99m}Tc by PZC and Zr-gel.

(1) Solvent Extraction

In general, solvent extraction method employs methyl ethyl ketone (MEK) to extract ^{99m}Tc from ^{99}Mo. After dissolution of MoO_3 powder in a NaOH solution, the solution is mixed thoroughly with MEK, and then the MEK containing ^{99m}Tc is separated. Substantial steps are required to re-extract the ^{99m}Tc from the MEK and purify it as a radio-pharmaceutical. This method is still used for small local supply of ^{99m}Tc at several research reactor centers [9-11].

On the other hand, this method may be used by radiopharmaceutical manufacturers, especially by the master milker operators for the commercial scale (n, γ)^{99}Mo supply. Up to now, there has been no experience of a large scale ^{99m}Tc extraction by this method. Presently, the preliminary tests are jointly carried out between JAEA and Chiyoda Technol Corporation and the hot tests with irradiated MoO_3 have been carried out between JAEA and BATAN under the international cooperation [12].

(2) Mo Absorbent

The Mo-Tc generators loaded with (n, γ)^{99}Mo adsorbed in zirconium molybdate gel have been used at hospitals in some countries.

The molybdenum zirconium gel (Zr gel) was also developed as Mo absorbent in Kazakhstan. The Zr gel is synthesized by the following processes; (1) resolution of irradiated $^{99}MoO_3$ into ammonia solution, (2) mixture of Mo solution and HNO_3 for synthesizing sodium poly-molybdate (PMNa), (3) Mixture of PMNa and $ZrOCl_2 \cdot 8H_2O$ for synthesizing Zirconium molybdate complex [13]. The synthesized Zr gel is dried, crushed and packed in column (Zr gel generator). ^{99m}Tc is eluted by milking with saline from Zr gel generator.

The poly zirconium compound (PZC) was developed as Mo adsorbent in Japan and characterization had been carried out under the international cooperation in the framework of the Forum for Nuclear Cooperation in Asia (FNCA). PZC is synthesized from zirconium tetra-chloride ($ZrCl_4$) and isopropyl alcohol ($(CH_3)_2CH_2OH$) as raw materials by the reactions of hydrolysis and polymerization. PZC is an inorganic polymer that consists of zirconium, oxygen and chlorine [14]. ^{99}Mo is adsorbed to PZC, and it is packed in a column (PZC generator). ^{99m}Tc is eluted by milking with saline from the PZC based generator.

An information exchange on the fabrication methods and characteristics of PZC and Zr gel were evaluated between INP-NNC-RK and JAEA under international cooperation [15].

3.3 ^{99m}Tc solution for a medicine

For the medical applications, the ^{99m}Tc produced by β-decay of ^{99}Mo is chemically separated and a solution of sodium pertechnetate ($^{99m}TcO_4^-$ solution) is obtained. Depending on the purpose of diagnosis, the $^{99m}TcO_4^-$ solution itself or a ^{99m}Tc labeled compound made by the $^{99m}TcO_4^-$ solution with a compound is given to a patient by intravenous or oral administration. The specifications of ^{99m}Tc solution are determined in pharmacopoeia in each country. For example, Table 2 shows pharmacopoeia in USA. In the United States Pharmacopeia (USP), specifications of both fission Mo and (n, γ)^{99}Mo are determined. Especially, when ^{99m}Tc solution is separated by the liquid-liquid extraction using MEK from the (n, γ)^{99}Mo, the content of MEK is limited to less than 0.1%.

In Japan, the supply of ^{99}Mo depends only on imports from foreign countries and it is important for domestic production to determine detail specifications of ^{99m}Tc solution. Meanwhile in Indonesia, BATAN has performed a biodistribution study, test of radiochemical, radionuclidic, chemical purities, and sterility~~,~~ and apyrogenicity tests and also pre-clinical and clinical trials for medical use [16-17].

Table 2 Summary of USP specifications for sodium pertechnetate ^{99m}Tc injection

		Fission ^{99}Mo	(n,γ)^{99}Mo
Sterility		Sterile solution	
Bacterial endotoxins		≤175/V USP Endotoxin Unit per mL	
pH		4.5 ~ 7.5	
Radiochemical purity		≥ 95%	
Radionuclidic impurity (μCi/mCi)	^{99}Mo	≤0.15	≤0.15
	^{131}I	≤0.05	-
	^{103}Ru	≤0.05	-
	^{89}Sr	≤0.0006	-
	^{90}Sr	≤0.00006	-
	Others	β & γ emitters ≤0.1 α emitters ≤1x10^{-6}	γ emitters ≤0.5
Chemical impurity	Al	≤10 μg/mL	≤10 μg/mL
	MEK	-	≤0.1%

3.4 Mo Recycling

The (n, γ)^{99}Mo has some advantages from the points of safety, waste management and nuclear proliferation resistance, but has disadvantage of low specific activity. Thus, ^{98}Mo enriched target is needed to use for the production of ^{99}Mo. Technology development for Mo recycling is required in order to establish cost-effective procedure because ^{98}Mo enriched MoO_3 is very expensive. This opinion is given in the OECD-NEA reports.

In JAEA, the preliminary tests with PZC absorbed un-irradiated Mo were carried out [18]. Figure 2 shows concept of Mo recycling methods with PZC. Two kind of Mo recycling methods are proposed in the preliminary tests. One is the elution method with alkali solution. The other is the sublimation method.

From the results, Mo recovery rate with these methods was more than 95% using PZC. Additionally, the waste volume of PZC was less than 60% with the sublimation method.

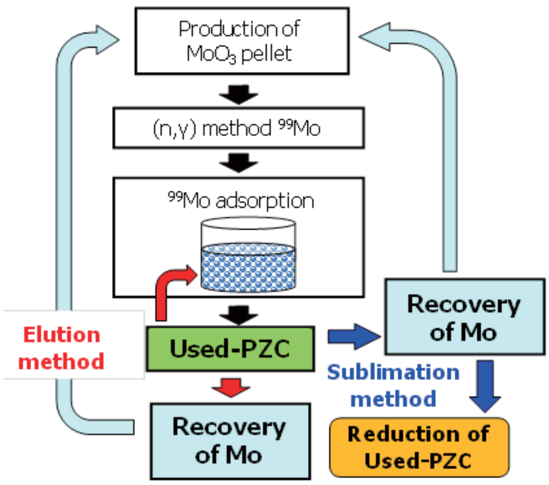

Fig.2 Concept of Mo recycling methods.

4. Status of Constructions of the irradiation and PIE facilities in JMTR

The (n, γ)^{99}Mo production will be carried out with the hydraulic rabbit irradiation facilities in JMTR. Figure 3 shows the flow chart of (n, γ)^{99}Mo production in JMTR [5].

In addition to the existing hydraulic rabbit irradiation facility (irradiation hole : D-5), new hydraulic rabbit irradiation facility (irradiation hole : M-9) will be prepared in 2011. The structure of new hydraulic rabbit is different. Three rabbits and five rabbits will be irradiated in the existing and new facilities, respectively. The amount of (n, γ)^{99}Mo production will reach about 37 TBq/week (1,000 Ci/week) in the new facility.

After irradiation, the irradiated rabbits are transferred with underwater basket to the JMTR hot laboratory through the Canal. After the rabbits of outer tube are opened, the MoO_3 pellets are taken out and ^{99}Mo is commercialized with chemical treatment process in the hot cell. The devices for dismantlement of rabbits and chemical treatments will be installed in the concrete cells and lead cells in JMTR hot laboratory.

The irradiation and PIE facilities are constructed for the preliminary and demonstration tests which are starting from 2012.

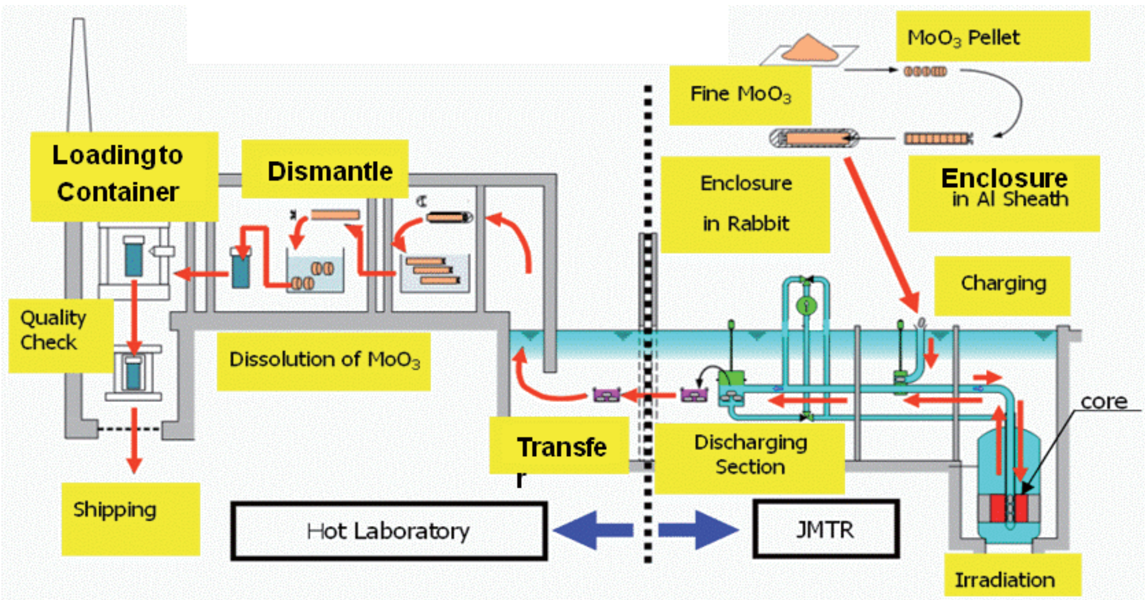

Fig.3 Flow chart of (n, γ)^{99}Mo production in JMTR.

4. Conclusions

The ^{99}Mo production by (n, γ) method was adopted from viewpoints of safety, nuclear proliferation resistance and waste management in JMTR. The R&D has been carried out with foreign organizations and relevant Japanese manufacturers under the cooperation programs. The status of R&D and construction of facilities for demonstration tests are as follows;

(1) Development of the sintered MoO_3 pellet with high density as irradiated target material,
(2) Extraction and concentration of ^{99m}Tc solution by solvent extraction from Mo solution under cooperation between BATAN and JAEA,
(3) Extraction of ^{99m}Tc solution with Mo absorbent from Mo solution under cooperation between INP-NNC-RK and JAEA,
(4) Construction of new hydraulic rabbit irradiation facility in JMTR.

In future plan, information exchange on the (n, γ)^{99}Mo production will be carried out, and road map for R&D will be discussed and established in the forthcoming expert meeting on (n, γ)^{99}Mo production to be held in March 2012.

Acknowledgement

The authors wish to thank Dr. T. Genka, former FNCA ^{99m}Tc generator project leader for his helpful comments in the preparation of this paper.

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6.2 Development of Mo Recycle Technique from Generator Materials

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The domestic ^{99}Mo production by the (n, γ) method is proposed in JMTR because of low amount of radioactive wastes and easy ^{99}Mo/^{99m}Tc production process. For the development of domestic production, it is necessary to use the enriched $^{98}MoO_3$, which is very expensive, for high specific activity of ^{99}Mo. A large amount of used PZC/PTC embraced ^{98}Mo is also generated after the decay of ^{99}Mo. JAEA and Taiyo Koko is proposed to recover molybdenum from the used PZC/PTC for an effective use of resources and reduction of radioactive wastes. Preliminary experiments of Mo recycling with un-irradiated MoO_3 were carried out by the elution and sublimation methods. From the results, Mo recovery rate from the PZC/PTC was more than 95% by two kinds of methods. The prospects are bright for Mo recycle and reduction of radioactive wastes using these methods.

Keywords: ^{99}Mo, ^{99m}Tc, JMTR, (n,f) Method, Radiopharmaceutical, Nuclear Medicine, Mo recycle, Elution method, Sublimation method

1. INTRODUCTION

The Japan Materials Testing Reactor (JMTR) of the Japan Atomic Energy Agency (JAEA) is a light water moderated and cooling tank-type reactor with a thermal power of 50 MW. The first criticality was achieved in March 1968 [1]. The operation of JMTR was halted for the refurbishment in August 2006 after its 165th cycle operation. The renewed JMTR will be started from FY 2012 and operated for a period of about 20 years until around FY 2030. Then, the new JMTR is expected to contribute to many fields: the lifetime extension of LWRs (aging management of LWRs, development of next generation LWRs, etc.), the expansion of industry use (production of the medical radioisotope ^{99m}Tc, etc.) and the progress of science and technology (namely, basic research on nuclear energy). To meet a wide range of users' needs, new irradiation technologies with advanced techniques have been developed [2].

For the production of the medical radioisotope, the ^{99m}Tc is in high demand in radiopharmaceutical. In recent years, the supplying of ^{99}Mo is only dependent on imported from foreign countries, so JAEA has lunched at domestic production of a part of ^{99}Mo in collaboration between industry and academia. This R&D has been extensively conducted by aiming at plan to urge domestic production of ^{99}Mo/^{99m}Tc.

The (n,γ) ^{99}Mo has several advantages compared to the fission Mo, but the extremely low specific activity makes its uses less convenient than the fission Mo[3]. Thus, it is necessary to use ^{98}Mo enriched MoO_3 powder, which is very expensive. Namely, in order to realize domestic ^{99}Mo production by (n, γ) method, so recycling technology development is proposed to recover molybdenum (Mo) as an effective use of resources and reduction of radioactive wastes. In this study, preliminary experiments of Mo recycling were carried out with un-irradiated MoO_3.

2. Mo RECYCLLING SYSTEM

The demand of ^{99m}Tc radiopharmaceutical in Japan is estimated at 4,000-5,000Ci/week [4]. On the other hand, the supply capacity of JMTR is 1,000Ci/week with natural MoO_3 [5], and scale which provides 20% of domestic necessary quantity. For example, by using an enriched $^{98}MoO_3$ increased to 100% concentration, which demands can be theoretically provided all domestic demand. It is necessary to use the high ^{98}Mo enriched MoO_3, which is very expensive. Thus, it is important to recover ^{98}Mo after the process. In fact, the irradiated target is not transmuted into most of ^{98}Mo (less than

1%) from rare metal.

The (n, γ) ^{99}Mo in JMTR is suggested to an adsorbent such as PZC/PTC which is ^{99}Mo-^{99m}Tc generator. Here, PZC (Poly Zilconium Compound) [6, 7] and PTC (Poly Titanium Compound) [8] are the high molecular compounds developed as Mo adsorbent for ^{99}Mo-^{99m}Tc generators, and have 100-200 times as many ability of Mo adsorption compared with alumina column which is conventional Mo adsorbent. PZC/PTC can be used for a generator, as a key material of the realization for the domestic ^{99}Mo production. Two kinds of methods are proposed for Mo recycling from the PZC/PTC.

One is the elution method by eluting chemical ^{98}Mo contained in the sorbent (PZC/PTC) after extracting ^{99m}Tc. The other is the sublimation method from the PZC/PTC at high temperature. Especially, the sublimation method has the additional features can be processed to reduce the adsorbent of used-PZC/PTC. Concept of Mo recycling systems is shown in Fig. 1.

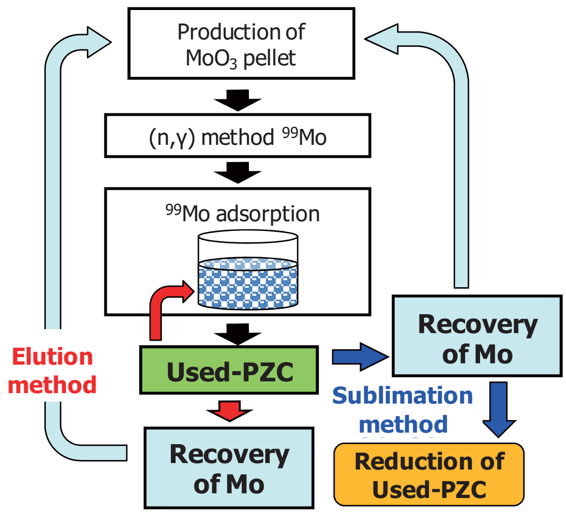

Fig.1. Concept of Mo recycling systems.

3. EXPERIMENTAL

3.1 Sample of molybdenum adsorbent

PZC is an inorganic polymer by using zirconium tetrachloride as a starting material. This is a molybdenum adsorbent PZC has the ability compared to alumina adsorbent that is currently used as a (n,f) method. Thus, when adsorbing ^{99}Mo of low specific radioactivity which produced by (n, γ) method, such highly efficient adsorbent is required.

3.2 Elution method

The elution method was examined. In this method, Mo absorbed in PZC was eluted by the alkali solution such as NaOH. Test apparatus of elution methods is shown in Fig.2. These device consists of a polypropylene column (Diameter: 10-11mm, Length: 118mm, Capacity: 10ml), 50ml syringe and three-way stopcock. The experimental conditions of sodium hydroxide dissolved in water, a predetermined concentration (0.1N, 0.5N and 1.0N) sodium hydroxide solution was prepared.

The column was filled up with Mo adsorbed PZC (5.0g), and water was fully added. Then, 100ml of sodium hydroxide solution was dropped from the top of the column, and effluent was separated from the lower column by 10ml each. The rate of dipping velocity is controlled by three-way stopcock that rate of 1.2ml/min. The sodium hydroxide solution was passed, the water (50ml) was dropped from the top of the column due to inside the extrusion of alkaline solution (rinsing operation) was carried out.

After elution, Mo is recovered from the eluent as molybdic acid. To obtain basic data for the effect on the Mo recovered as a function of predetermined concentration of NaOH.

3.3 Sublimation method

The sublimation method was examined. In this method, Mo absorbed in PZC was recovered by the sublimation of MoO_3 at high temperature, and the effect on sublimation of Na content in PZC was investigated. Schematic drawing of the test apparatus is shown in Fig.3, which is composed of an electric furnace, a quartz-made tube, a dry-air generator and a boat made from Al_2O_3. The quartz tube is set in the center of the electric furnace. The Used-PZC was heated up at a rate of 500˚C/h in a dry air flow, and then heat up 1000˚C higher than 800˚C which is the sublimation temperature of MoO_3. The dry air flow rate was 200ml/min. After the heated up, the MoO_3 gas was quenched with room air the right, then the MoO_3 was deposition in the quartz tube.

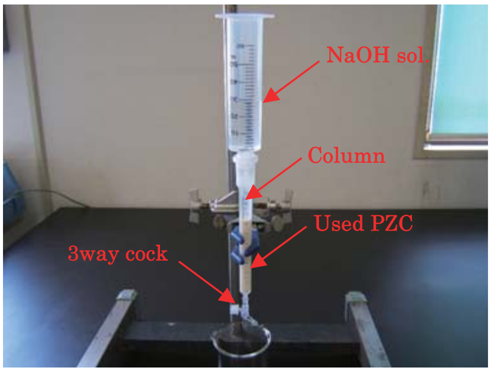

Fig.2. Test apparatus of Elution method.

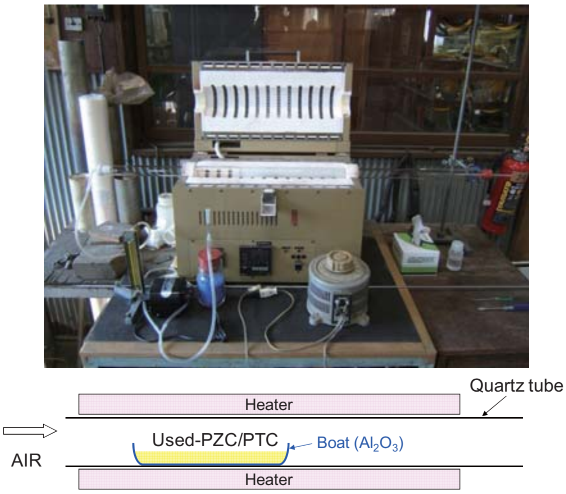

Fig.3. Test apparatus of Sublimation method.

4. RESULTS AND DISCUSSION

4.1 Elution method

After elution, Mo was recovered as molybdic acid from the solution. The effect on the elution of Mo was investigated in different concentration of NaOH is shown in Fig.4. As a result, the eluted rate of Mo increases with the increase the concentration of NaOH. Mo recovery amount with 1N-NaOH was larger than that with 0.1N-NaOH, and the recovery rate of Mo was 98% with 1N-NaOH. Also, after the Mo recovery, characteristics of used-PZC treated with 1N-HCl were about 30% of the initial adsorption of Mo. It has been difficult to complete regeneration of the PZC. In addition, the PTC is under examination.

4.2 Sublimation method

Mo recovery rate from used-PZC by the sublimation method is shown in Fig.5. As a result, the Mo recovery rate from PZC increased with decreasing the Na content in PZC. The recovery rate of Mo was 98% by heating at about 1000˚C after the PZC was washed for Na removal by 0.05N-HCl. Moreover, the volume reduction rate of the PZC was about 65% after the heat-treatment.

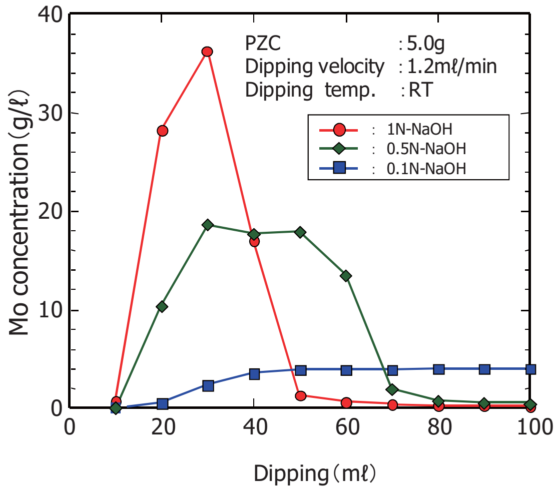

Fig.4. Mo elution profiles as a function of dipping.

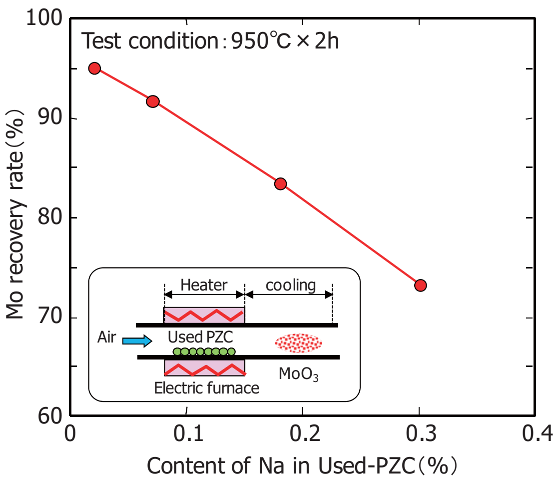

Fig.5. Dependence of the Mo recovery rate by Na content.

5. CONCLUSIONS

Preliminary experiments with two kinds of methods have been carried out with un-irradiated MoO_3 for Mo recycling. As a consequence, Mo recovery rate of two methods was more than 95%, and the used-PZC is reduced was 65% by sublimation method. In this new technique, contributes to an efficient supply and low cost can be realized by (n,γ) method for ^{99}Mo production. Moreover, ^{98}Mo can be effectively used and reduction of radioactive waste. It is future concluded that the hot cell tests of this developed recycling method will be carried out in the renewed JMTR and JMTR hot laboratory.

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6.3 DEVELOPMENT OF ^{99m}Tc EXTRACTION-RECOVERY BY SOLVENT EXTRACTION METHOD

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^{99m}Tc is used as a radiopharmaceutical in the medical field for the diagnosis, and manufactured from ^{99}Mo, the parent nuclide. In this study, the solvent extraction with MEK was selected, and preliminary experiments were carried out using Re instead of ^{99m}Tc. Two tests were carried out in the experiments; the one is the Re extraction test with MEK from Re-Mo solution, the other is the Re recovery test from the Re-MEK. As to the Re extraction test, and it was clear that the Re extraction yield was more than 90%. Two kinds of Re recovery tests, which are an evaporation method using the evaporator and an adsorption/elution method using the alumina column, were carried out. As to the evaporation method, the Re concentration in the collected solution increased more than 150 times. As to the adsorption/elution method, the Re concentration increased in the eluted solution more than 20 times.

Keywords: (n, γ) method, Solvent extraction, MEK, Evaporation method, Adsorption/elution method

1. INTRODUCTION

^{99m}Tc is a crucial radioisotope in the medical field for the diagnosis radiopharmaceuticals in Japan. ^{99m}Tc is obtained from the β^- decay of the ^{99}Mo. The supply of ^{99}Mo in Japan depends entirely on the import from foreign countries. Thus, it is needed to supply ^{99}Mo stably by the domestic manufacturing, because of the unplanned shutdown or aging of the foreign countries' reactors for ^{99}Mo manufacturing and difficult flight by volcano eruption [1-4].

^{99}Mo is generally manufactured by either ^{235}U(n, f)^{99}Mo or ^{98}Mo(n, γ)^{99}Mo reaction using reactors. Japan Atomic Energy Agency (JAEA) and Chiyoda Technol Corporation (CTC) have chosen the manufacturing by the latter reaction and have developed a method of obtaining much amount of ^{99m}Tc[5]. In order to develop the method, it is essential to increase the final concentration of ^{99m}Tc, because the specific activity of ^{99}Mo is very low, comparing with that obtained by ^{235}U(n, f)^{99}Mo reaction.

In this study, preliminary out-pile tests were carried out using the solvent extraction with Methyl Ethyl Ketone (MEK) of Re instead of ^{99m}Tc from Mo solution, and characteristics of extraction and recovery of Re were evaluated.

2. EXPERIMENTAL

In the preliminary experiments, Re was used instead of ^{99m}Tc because Re and Tc were homologous elements. The whole experimental flow is show in Fig.1. As extraction test, Re-Mo solution was prepared and extracted with MEK, followed by Recovery test. In the Re recovery test, the evaporation test and the adsorption/elution test were carried out to evaluate Re recovery yields in each step and also the total recovery yields [6-7].

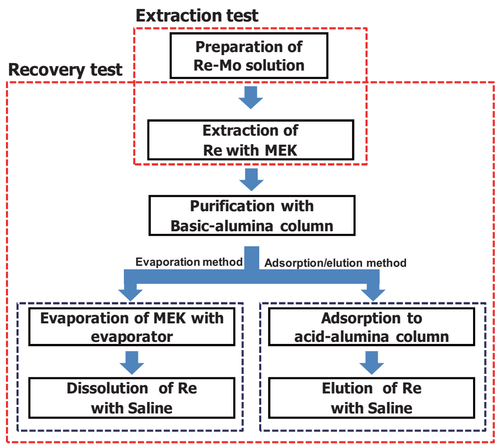

Fig. 1 The whole experimental flow

2.1 Preparation of Re-Mo solution

MoO_3 target containing 100Ci of ^{99}Mo is dissolved. A quantity of ^{99m}Tc and ^{99}Tc to be produced in 1 day was calculated and the amount of Re corresponding to the quantity was evaluated. The tests were carried out based on the amount of Re evaluated.

2.2 Re Extraction test

To determine extraction conditions, three kinds of tests were carried out. In the first test, concentration of NaOH for dissolving MoO_3 target was determined. Test conditions were shown in Table 1. Re extraction test was carried out as parameter of NaOH concentration. In the second test, volume of NaOH solution for dissolving MoO_3 target was determined. Test conditions were shown in Table 2. Re extraction test was carried out as a parameter of ratio of MoO_3 weight to 6M-NaOH volume. In the third test, Re extraction test was carried out under the optimal conditions obtained in above two tests. Test conditions were shown in Table 3.

As the first test, dissolution of MoO_3 powder with NaOH solutions and extraction of Re with MEK were carried out. 200 g of MoO_3 powder was dissolved with 245 ml of 5, 6 and 7M-NaOH solutions. 90 μg of Re was added to the solutions. The solution obtained is called Re-Mo solution in this paper. Re-Mo solution and 60 ml of MEK were mixed for 5 min in a separatory funnel. The mixture was allowed to stand for 30 min to separate Re-MEK phase and aqueous phase. Chronological change of aqueous phase was observed for 6 days.

Table 1 Test conditions for determining NaOH sol. concentration

NaOH sol. concentration	5M	6M	7M
NaOH sol. volume	245 ml		
MoO_3 weight	200 g		
MEK volume	60 ml		
Re weight	90 μg		

Secondly, Re extraction efficiencies were determined under some experimental conditions using 6M-NaOH. The conditions were as follows; (i) 200 g of MoO_3, dissolved with 300 ml of NaOH solution, 90μg of Re to be added, and 70 ml of MEK, 5 min of mixing time, 20 min of standing time, (ii) 150 g of MoO_3, dissolved with 300 ml of NaOH solution, 90μg of Re, 70 ml of MEK, 5 min of mixing time, and 20 min of standing time.

Table 2 Test conditions for determining solution volume

NaOH sol. concentration	6M	
NaOH sol. volume	300 ml	
MoO_3 weight	200 g	150 g
MoO_3 weight /NaOH sol. volume	2g/3ml	1g/2ml
MEK volume	70 ml	
Re weight	90 μg	

Thirdly, Re extraction test was carried out. The conditions were as follows; 200 g of MoO_3, dissolved with 400 ml of 6M-NaOH solution, 271 μg of Re, 92 ml of MEK, 10 min of mixing time, and 30 min of standing time. The amount of Re recovered was determined by Inductively Coupled Plasma Atomic Emission Spectroscopy (ICP-AES) after MEK was removed and Re precipitate was dissolved with water.

Table 3 Re extraction test conditions

NaOH sol. concentration	6M
NaOH sol. volume	400 ml
MoO_3 weight	200 g
MEK volume	92 ml
Re weight	270.6 μg

2.3 Re recovery test

Re recovery tests were carried out by two kinds of experiments; evaporation method using the evaporator and adsorption/elution method using the alumina column. To obtain a high concentration of ^{99m}Tc solution, evaporation method and adsorption/elution method were developed. Re concentration factor and recovery yields of final products obtained by two methods were compared.

(1) Evaporation method

Re-Mo solution was prepared by dissolving 200 g of MoO_3 with 400 ml of 6M-NaOH and adding 271 μg of Re to the solution. The Re-Mo solution and 92 ml of MEK were mixed for 10min in separatory funnel in order to extract Re into MEK. MEK phase and aqueous phase were separated by standing them for 30min. A portion of MEK was taken and the amount of Re determined. MEK containing Re was passed through the basic alumina column (basic alumina: 7g, φ10) to remove impurities such as Mo and water. In order to push out residual Re from the column, 20 ml of MEK was passed through the column. A portion of MEK was taken and the amount of Re determined. MEK containing Re was evaporated by evaporator; MEK was evaporated at 40°C for 15 min and then at 40-80°C for 15 min with new MEK receiver. Finally, the inside of Re container was evacuated for 30 sec to remove MEK completely. Re of precipitate was dissolved with 2 ml of saline. The amount of Re in the solution was also determined by ICP-AES.

(2) Adsorption/elution method

Re-Mo solution was prepared; 200 g of MoO_3 was dissolved with 400 ml of 6M-NaOH solution and 87 μg of Re was added to the solution. Re-Mo solution and 90 ml of MEK were put into separatory funnel, and mixed for 5min. Re was extracted with MEK from Re-Mo solution. Re-MEK phase and aqueous phase were separated after 30min. MEK containing Re was passed through the tandem columns system. The upper part is basic alumina column (basic alumina: 7g, φ14), and the bottom one is acidic alumina column (acidic alumina: 7g, φ14). In order to push out residual Re from the columns, 30ml of MEK was passed through the columns. In order to rinse the acidic alumina column, 30ml of pure water was passed through the acidic alumina column. Finally, 20 ml of saline was passed through the acidic alumina column by 2 ml fraction in order to recover Re as a product.

3. RESULTS and DISCUSSION

3.1 Preparation of Re-Mo solution

Evaluated amount of Re was shown in Table 4. Atomic number of ^{99}Mo of 100Ci is 1.27E18. After one day passes, atomic number of ^{99}Mo is lessoned to 9.86E17. As all of ^{99}Mo decayed become to sum of ^{99}Tc and ^{99m}Tc, if decay of ^{99}Tc is ignored. As a result, the total atomic number of ^{99}Tc+^{99m}Tc becomes 2.83E17. Therefore, the number is equivalent to 87μg of Re. MoO_3 powder was dissolved with NaOH solution and Re-Mo solutions were prepared by adding prescribed amount of Re.

Table 4 Amount of Re

	100Ci	77.7Ci (1day later)
Number of ^{99}Mo	1.27E18	9.86E17
Number of ^{99}Tc+^{99m}Tc	0	2.83E17
Weight of Re	0μg	87μg

3-2. Re extraction test

In the first test of changing the concentration of NaOH solution, it was confirmed whether the precipitate appeared in the aqua phase. Test results were shown in Table 5. Precipitate was observed immediately after the extraction, when 5M-NaOH solution was used, and the precipitate was also observed 24 hrs later when 7M-NaOH solution was used. The precipitate has grown with the time elapse in these aqueous phases. The precipitate was not observed at room temperature for 6 days, when 6M-NaOH solution was used. From the results, the concentration of NaOH solution was decided to be 6M in this study.

Table 5 Results on effect of NaOH sol. concentration

	Alkali concentration		
	5M	6M	7M
Aqua phase	Crystallization	No crystallization	Crystallization (24h later)

In the second test of using 6M-NaOH solution, it was also confirmed whether the precipitate appeared in the aqueous phase. Test results were shown in Table 6. The precipitate was not observed at ambient temperature. But, the precipitate appeared when 200 g of MoO_3 was dissolved with 300 ml of NaOH solution and after extraction the Re-Mo solution was held at temperatures of lower than 10°C. It is thought that the precipitate occurred because it became less solubility to the decrease at lower temperatures. From the results, it was decided that 1 g of MoO_3 is dissolved with 2 ml of 6M-NaOH solution in ratio.

Table 6 Results on effect of 6M-NaOH sol. volume

	MoO_3 weight /NaOH volume	
	2/3	1/2
Ambient temperature	No crystallization	No crystallization
10°C	crystallization	No crystallization

From these results, we determined the dissolution and extraction conditions as follows; 200 g of MoO_3 is dissolved with 400 ml of 6M-NaOH solution and Re is extracted with 92 ml of MEK. Under these conditions, the Re extraction test was carried out and Re was recovered with the extraction efficiency of 90.5%. Test results were shown in Table 7.

Table **7** Re extraction test results

Re weight (μg)		Efficiency (%)
Re-Mo solution	270.6	90.5
MEK	244.8	

3-3. Re recovery test

(1) Evaporation method

Recovery test using evaporator was carried out in order to determine the recovery yield at each step and the concentration factor comparing with Re-Mo solution. Test results were shown in Table 8. The recovery yield in a process of extraction was 96.8%, efficiency on a process of removal of impurities 94.7 %, that in a process of concentration 90.8 %, and the total yield 83.3 %. And the final product of saline was clear in color. As Re concentration in Re-Mo solution was 0.59 μg/ml and Re concentration in the final product was 112.7 μg/ml, the concentration factor was 191.

Table 8 Test results by evaporation method

Process	Re volume (μg)	Efficiency (%)
Extraction	270.6	96.8
	262.0	
Removal of impurities	247.3	94.7
	234.3	
Evaporation	222.7	90.8
	202.3	
Total	-	83.3

(2) Adsorption/elution method

Recovery test using acidic alumina was carried out in order to determine the total recovery yield and the concentration factor. Elution profile is shown in Fig. 2. Re of 95.9% was finally recovered by 20ml of saline. Re concentration in Re-Mo solution was 0.18 μg/ml, and Re concentration in 20 ml of saline was 4.17μg/ml. Therefore, the concentration factor was 23 times. However, when Re eluted in 8 to 20 ml eluate is recovered, the concentration factor increases to 33 times.

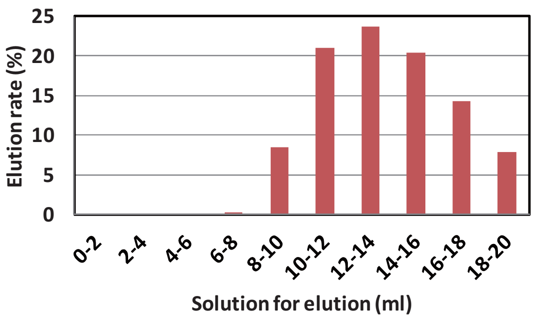

Fig. 2 Elution profile

4. CONCLUSION

Extraction-recovery test results were shown in Table 9. It has been understood that the recovery yields obtained by the methods of using the evaporator and the acidic alumina column were 80% or more. Concentration factor of Re was 191 times in evaporation method using evaporator. The concentration factor of Re will become higher when final Re precipitate is dissolved with smaller volume of saline. But, this method has a fear that impurity concentration of Re product increases when all impurities are not completely removed. On the other hand, the concentration factor of Re was 23 times in adsorption/elution method using the alumina column. It is possible to obtain higher concentration factor of Re by adjusting collecting fractions and also the purity of Re product obtained by adsorption/elution method would keep high because acidic alumina also purifies the Re product.

Following these results, we have been developing the production method using ^{99m}Tc/ ^{99}Mo with BATAN in Indonesia.

Table 9 Re extraction-recovery test results

	Efficiency (%)	Concentration factor (Times)
Evaporation method	83.3	191
Adsorption/elution method	95.9	23

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7. Facility, Upgrade, Aging Management

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4th International Symposium on Material Testing Reactors, Oarai, Japan, December 5-9, 2011

7.1 MIT Research Reactor – Power Uprate and Utilization

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The MIT Research Reactor (MITR) is a university research reactor located on MIT campus. and has a long history in supporting research and education. Recent accomplishments include a 20% power rate to 6 MW and expanding advanced materials fuel testing program. Another important ongoing initiative is the conversion to high density low enrichment uranium (LEU) monolithic U-Mo fuel, which will consist of a new fuel element design and power increase to 7 MW.

Keywords: MITR, power uprate, utilization, advanced material testing, in-core loops.

1. INTRODUCTION

The MIT Research Reactor (MITR) is owned and operated by MIT, and has a track record in carrying out frontline research and educational training in the areas of nuclear fission engineering, advanced materials testing, neutron physics, geochemistry, and environmental studies since it was built in 1958. The MITR currently uses high enrichment uranium (HEU) MTR-type fuel element. Each rhomboid-shaped fuel element consists of fifteen fuel plates in the form of UAlx cermet and cladded with 6061 aluminum alloy. A unique design feature of the MITR fuel is the longitudinal fins which increase the heat transfer area by roughly 100%.

The MITR core is contained in two concentric tanks. The inner one is for the light water coolant/moderator while the outer one contains the heavy water reflector, which is surrounded by a graphite reflector. Reactor control is provided by six boron-impregnated stainless-steel shim blades and one cadmium regulating rod. The close-packed hexagonal reactor core of the MITR is designed with twenty-seven rhomboidal fuel element positions. In general, twenty-four fuel elements are loaded during normal operations. The remaining three positions are filled with either a solid aluminum "dummy" element or an in-core experimental facility. Fig. 1 is a top view of the reactor core with one fuel capsule experiment and two solid dummy elements. The hexagonal core structure is about 38 cm (15”) across and the length of an active fuel length is about 56 cm (22”).

The MITR operates at atmospheric pressure with nominal primary coolant flow rate of 2000 gpm. Primary coolant enters the bottom of the core tank through the core shroud, flows upward through the fuel elements and then exits at the outlet piping at about 2 m above the top of the core. Passive safety is ensured by natural convection after loss of flow transients. The MITR operates 24/7 with fuel cycle lasts from 6-8 weeks.

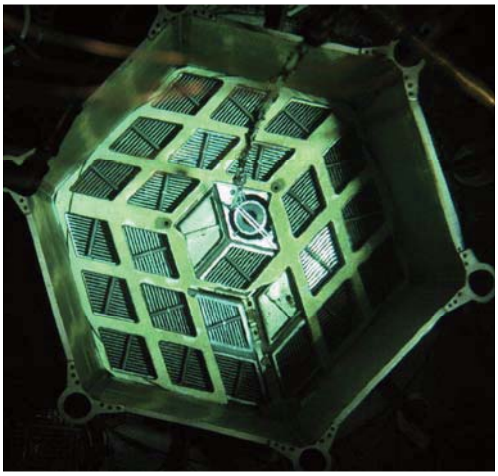

Fig.1. Top view of the MITR core.

2. POWER UPRATE

The MITR received its new license for 6 MW power uprate from US Nuclear Regulatory Commission in November 2010. The 20% power uprate from 5 MW to 6 MW is mainly based on recapturing of safety margin using modern computational tools for core power distribution and transient analysis, and there is no change in core design or operating conditions (coolant outlet temperature, primary coolant flow rate etc.) of the reactor [1]. Gradual power ascension to 6 MW was completed in spring 2011. Recent primary coolant system upgrade with a compact plate-and-frame type heat exchanger improves the efficiency of the cooling system and enables year-around operation at 6 MW. The new license will last until November 2030.

3. ADVANCED MATERIALS AND FUEL TESTING PROGRAM

The MITR's compact core design offers high power density ~ 80 kW/l, and high fast neutron flux ~ 1.3 x10^{14} n/cm2 s, thermal flux ~ 5x10^{13} n/cm2 s.. One of the unique capabilities of the MITR is the design and use of in-core loops that replicate nuclear power reactor conditions to study the behavior of advanced materials and fuel designs. Up to three in-core positions are available to accommodate in-core experiments in the reactor's compact core. The MITR is licensed for fuel irradiations as long as the fissile material mass is limited to 100 grams or less. Thus, small fuel irradiation scoping studies can be performed. The MITR is well-suited for carrying out such basic studies because of its relatively high power density similar to a light water reactor (LWR), its easy-access geometric configuration, and its capability for fast turnaround. The MITR conducted in-core loop experiments with prototypic PWR/BWR operating and chemistry conditions since late 1980's for corrosion, coolant chemistry, and advanced cladding development. A high temperature irradiation facility (HTIF) was developed and demonstrated for high temperature materials irradiation in inert gas environment up to 1400 °C.

The MITR became the first university reactor partner facility of the DOE's Advanced Test Reactor National Scientific User Facility (ATR-NSUF) in 2008. Leveraging the support from ATR-NSUF and other funding sources, the MITR reached a milestone recently in successful operation three in-core experiments simultaneously. These are: 1) a pressurized water loop to test silicon carbide composite cladding under LWR chemistry conditions; 2) high temperature capsules in an in-core sample assembly (ICSA) to irradiate high temperature compound materials up to 900 °C; and 3) uranium hydride metallic fuel rods with liquid metal gap instrumented with fuel centerline thermocouples

3. LEU CORE CONVERSION

The Reduced Enrichment for Research and Test Reactors program (RERTR) is part of the Global Threat Reduction Initiative (GTRI) of the Department of Energy's National Nuclear Security Administration (NNSA) program. The program aims to convert reactors and targets using HEU with LEU fuels in civilian facilities worldwide to reduce the potential global nuclear threat. LEU is defined as U-235 enrichment less than 19.99%. The high density LEU fuel that MITR is planning to use is a monolithic uranium and molybdenum (U-Mo) fuel. The development and testing of U-Mo monolithic fuel is in progress at the Idaho National Laboratory [2]. A study at MIT has demonstrated that LEU conversion is feasible using this fuel, which has a uranium density of 15.5 g/cm^3 with 10 wt% Mo[3].

The new MITR LEU fuel element is designed with 18 fuel plates per element with the same outer dimensions as the HEU element. The increased number of plates per element expands the total heat transfer area. The coolant gap between fuel plates reduces slightly from 78 mils to 72 mils (from fin tip to fin tip). This LEU fuel element design was proposed by Ko et al. based on the criterion that the core tank pressure loading of the LEU core should be equal or less than that of the current pressure loading of the HEU core [4]. A transitional core conversion strategy is proposed for the MITR, which consists of replacing a few depleted HEU elements with fresh LEU elements at each cycle until the core is filled with all LEU elements. Preliminary analysis concluded that in the mixed core is feasible [5] and will reduce the impact of conversion on MITR's utilization program.

4. CONCLUSIONS

MITR in the past decade has achieved major milestones in power upgrade and expanded utilization program. It has a unique capability in the design and operations of in-core experiments for high temperature materials, instrumentation, and fuel irradiations. MITR's accomplishments as a university research reactor underscores the importance of US DOE's continuing support of university research reactors and the need to maintain a national materials testing infrastructure such as the ATR-NSUF

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7.2 OVERVIEW OF REFURBISHMENT PROJECT OF JMTR

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The refurbishment project of the JMTR from the beginning of JFY 2007 was promoted with two subjects; the one is the replacement of reactor components, and the other is the construction of new irradiation facilities.
On the replacement of reactor components, an investigation on aged components (aged-investigation) was carried out, for concrete structures, tanks, tubes in order to identify their integrity. After the investigation, some components were decided to replace from viewpoints of future maintenance and improvement of reliability. On the construction of new irradiation facilities, corresponding to the user's irradiation request, new irradiation facilities, such as irradiation test facilities and equipments for LWRs materials/fuels, were installed in the JMTR. Furthermore, in June 2010, "birth of the nuclear techno-park with the JMTR" was selected by Japanese Government. The new project is to install new irradiation facilities, such as irradiation facility and equipments for Mo-99 production and PIE equipments to JMTR until JFY2012.
The New JMTR will be utilized fully by wide fields of users. Moreover, the JMTR will also contribute the promotion on research and development of the nuclear energy from basic to applied fields as an internationally utilized facility under the international/Asian network collaborations.

Keywords: JMTR, refurbishment, aging, component, irradiation facility, reactor availability-factor

1. INTRODUCTION

The refurbishment project works of the JMTR was started from the beginning of JFY2007. The refurbishment project was promoted with two subjects; the one is the replacement of reactor components, and the other is the construction of new irradiation facilities.
The replacement work was finished at the end of Feb 2011, according to schedule. The construction of new irradiation facilities is in progress as scheduled as shown in Fig1.
Before the replacement of reactor components, an investigation of aged components(aged-investigation) was performed in order to identify integrity of facilities and components to be used for re-operation of JMTR. The equipment which needs replacing before the restart of the JMTR was selected after evaluated on its damage and wear due to aging significance in safety functions, past safety-related maintenance date, and the enhancement of facility operation. The replacement work of power supply system, boiler, radioactive waste facility, reactor control system, nuclear instrumentation system etc. was already carried out as scheduled. [1]
On the other hand, corresponding to the user's irradiation request, new irradiation facilities, such as irradiation test facilities for LWRs materials/fuels with a purpose of long-term and up-graded operations, production facilities for medical radioisotopes for Industrial Purpose, were planned to install in the JMTR.
Furthermore, in June 2010, the project named "birth of the nuclear techno-park with the JMTR" was selected by Japanese Government. The new project is to install new irradiation facilities and PIE equipments to JMTR in order to promote basic as well as applied researches. In the project, development of user-friendly environment especially for young and female researchers is highlighted.

2. AGED-INVESTIGATION

JMTR utilization advisory committee suggested that "JMTR should be operated for more 20 years after refurbishment". Based on this suggestion, following items were considered for selection of "Components to be continuously used" and "Components to be replaced". The fundamental view is shown in Fig2.

Criteria for selecting components to be replaced as follows,

1. Safety point of view
   (1) Aging of components
   (2) Importance of safety feature
   (3) Maintenance experience
2. Improvement of availability
   (4) Affordability of spare parts

In order to evaluate the ageing of components, the integrity investigation for the JMTR facilities and components was carried out in early phase of the refurbishment from the beginning of FY2007. Investigation items for aged components investigation are as follows,

1) Maintenance situation for aging such as corrosion, thinning, fatigue, irradiation effect
2) Possibility of appropriate status supervision after re-operation
3) Validity of current maintenance method

In case of the primary cooling system, verification on reliability of the heat exchangers for primary cooling system was carried out to investigate an integrity of continuously use component. From a result of the significant corrosion, decrease of tube thickness, crack were not observed on the heat exchangers, and integrity of heat exchangers were confirmed.
In the long terms usage of the heat exchangers, the maintenance based on periodical inspection and a long-term maintenance plan was scheduled. [2]

In case of piping of the Piping in Secondary Cooling System, Visual inspection by staff and fiberscope were carried out. As the results, there were cracking on whole inner lining surface. However, decrease of piping thickness could hardly be observed. Therefore, repair of lining were carried out for the long term usage in future. After repair, the lining was strengthened by five layer painting of the tar epoxy paint on coal tar enamel paint. And the future maintenance was reconsidered to investigate once per 5 years from the life of paint.

In case of the Concrete structure, an integrity investigation was carried out for the JMTR reactor building, which was the concrete structure and would be used for the long-term after JMTR restart,
In the integrity investigation, the concrete surface deterioration, the rebound number (nondestructive strength test), compressive strength using drilled concrete core test piece, the static modules of elasticity, the carbonation depth, the reinforced bar corrosion and the chloride ion content were investigated respectively and the integrity of concrete was confirmed. [3]
In order to use the JMTR reactor building continuously for long-term, it is important for maintaining the integrity of a concrete structure by the periodical maintenance and the repairing work including the building outer-wall surface painting that has been conducted up to now.

3. REPLACEMENT OF REACTOR COMPONENTS

Based on criteria for selecting components, following items were reviewed and studied,

- Aging during 20 years during reoperation,
- Importance grade of reactor facilities/equipments,
- Conditions of facilities / equipments,
- Stable supply of spare parts in the maintenance activities during 20 years.

Replacement and renewal of the components were selected from evaluation on their damage and wear in terms of aging as shown in Table 1. Facilities whose replacement parts are no longer manufactured or not likely to be manufactured continuously in near future, were selected as renewal ones. Furthermore, replacement priority was decided with special attention to safety concerns. A monitoring of aging condition by the regular maintenance activity is an important factor in selection of continuous using after the restart. Taking also account of a continuous operation with safety, reactor facilities/equipments to be renewed were decided. [4]
As a result, aged or old-designed components of the control rod drive mechanism, primary cooling system, secondary cooling system, electric power supply system etc., were to be replaced by present-designed ones. Furthermore, the replacements and renewal were possible to carry out within the range of licensing permission of the JMTR.
For facilities which are not replaced, e.g. heat exchangers, pressure vessel, secondary cooling towers and so on, their safety was evaluated from a view point of aging. The long-term operation in future will be possible by maintaining the present condition in accordance with the periodic safety review of the JMTR. [5]

Renewal of the feed and exhaust air system is carried out at first, and also renewal of utility facilities of electric power supply system, boiler component, etc. is carried out at the same time. Then, facilities in the reactor building are to be finally renewed. Renewal of JMTR had been on schedule, and completed.
After restart of the JMTR, the maintenance activity will be carried out by the maintenance program based on the periodic safety review of the JMTR. By the replacement of reactor facilities, the failure possibility

of each component will decrease, and this leads the improvement of the higher reactor availability-factor in future as shown in Fig.3. [6]

4. CONSTRUCTION OF NEW IRRADIATION FACILITIES

Corresponding to the user's irradiation request, new irradiation facilities, such as irradiation test facilities for materials/fuels, production facilities for medical radioisotopes, were planned to install in the JMTR as shown in Fig 4.

(1) New Material and Fuel Irradiation Tests
[Facility for fuel development]
An irradiation facility of fuel behavior test at transient condition has been developed to evaluate the safety for the high burn-up light-water reactor fuels, uranium and MOX fuels in JMTR. The facility is capable of carrying out power ramping and boiling transition tests on light-water reactor fuels. The fuel irradiation test facility. The irradiation facility consists of shroud irradiation equipment, capsule control equipment and He-3 power control equipment. [7]
[Facility for material development]
The material irradiation test facility is developed to study the Stress Corrosion Cracking (SCC) under neutron irradiation for the light-water reactor in-core materials. This facility consists of a water environmental control system in the BWR material irradiation facility simulating the BWR environment and the water chemical test facility simulating the broad water environment such as the BWR, PWR. The BWR material irradiation facility consists of a water environment control system, weight-loading control unit and capsules.

(2) New Irradiation Facility for Industrial Purpose
One of irradiation facilities is intended to provide the ^{99m}Tc for medical use. A hydraulic rabbit irradiation facility, which is well developed and already used for irradiation in the JMTR, can be applied to the production.

5. BIRTH OF THE NUCLEAR TECHNO-PARK WITH THE JMTR

In June 2010, "Birth of the nuclear techno-park with the JMTR" was selected as one of projects of the Leading-edge Research Infrastructure Program by Japanese Government. The new project is installation of advanced irradiation facilities and PIEs equipments in JMTR in order to promote basic as well as applied researches. In the program, development of user-friendly environment especially for young and female researchers is highlighted.
For example, preparation of irradiation area with LWRs environments, development of ^{99m}Tc production technique by (n, γ) method, construction of atomic level analyzer of PIE and achievement of very high temperature (2000 degree) irradiation environment shown in Fig.5 would be installed at least until JFY2012.

6. CONCLUSIONS

The refurbishment works of the JMTR had been completed as scheduled.
By the replacement of the reactor facilities, the failure possibility of each component will decrease, and this leads the improvement of the higher reactor availability-factor in future.
By installation of irradiation facilities, various irradiation tests which corresponding to the role of JMTR will be carried out after reoperation.
Furthermore, the new irradiation facilities and equipments would be installed by the project of "birth of the nuclear techno-park with the JMTR" at least until JFY2012.

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Table 1 Selection of components to be replaced

Criteria for selecting components to be replaced

1.Safety point of view

(1) Aging of components
(O:There is possibility)

(2) Importance of safety feature
(O:Importance is high level)

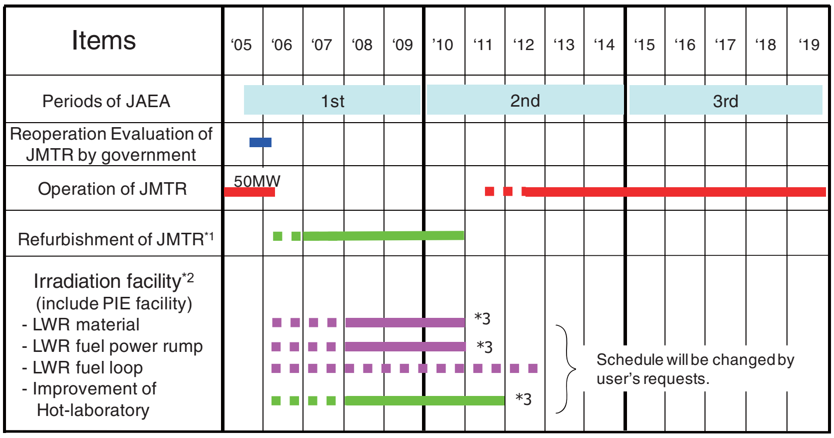
(3) Maintenance experience
(O:High trouble frequency or Short service life time etc.,)

2. Improvement of availability

(4) Affordability of spare parts
(O:Difficulty)

Facility, system	Components	Criteria (1)	(2)	(3)	(4)	
Reactor and process control system	Reactor control panel	O		O	O	Rep.
	Process control panel	O	O	O	O	Rep.
	Neutron instruments	O	O	O	O	Rep.
	CRDM	O	O	O	O	Rep.
Reactor cooling system	Main pump motors	O	O	O	O	Rep.
	Main heat exchangers		O			Cont.
	UCL circulation pump	O	O		O	Rep.
	Secondary cooling system main pipes					Cont.
Radiological waste disposal system	Emergency blowers	O	O		O	Rep.
	Regular blowers					Cont.
Power supply system	Power supply units	O			O	Rep.
	High voltage transformer	O		O	O	Rep.
Other system	Water demineralizer	O			O	Rep.
	Boiler units	O			O	Rep.

Rep. : to be replaced
Cont. : to be continuously used



*1: Refurbishment works are carried out by government budget.
*2: Irradiation facilities are installed by users fund.
*3: IASCC, Irradiation embrittlement, Hafnium irradiation and fuel ramp tests are being prepared in a NISA (Nuclear and Industrial Safety Agency) project.

Fig. 1 JMTR refurbishment work schedule

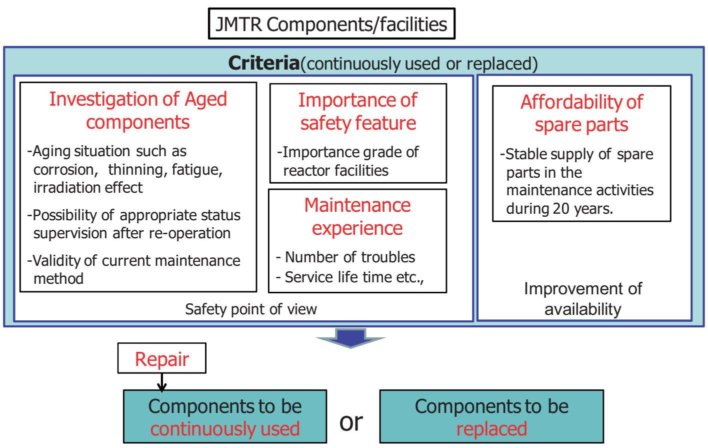


Fig. 2 Fundamental view for the investigation and replacement

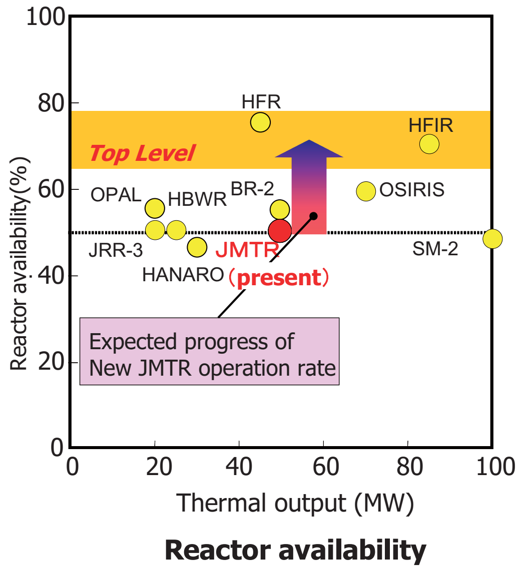


Fig. 3 Expected reactor availability of JMTR

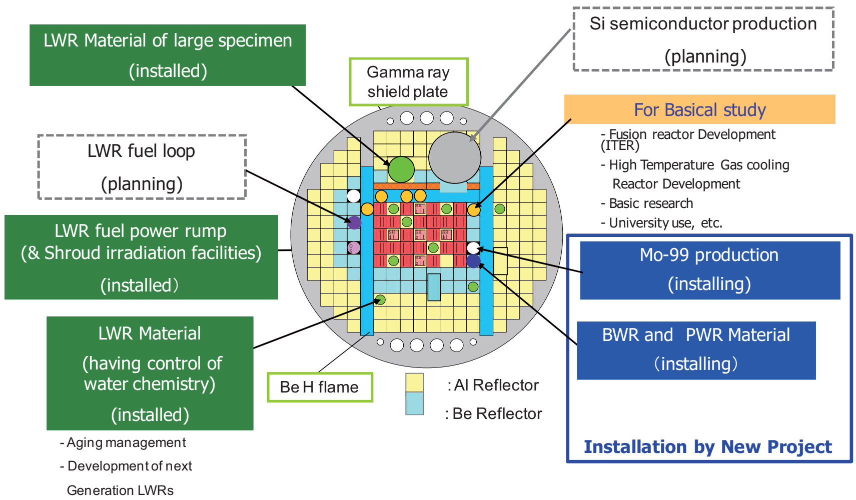


Fig. 4 Outline of JMTR core irradiation facilities

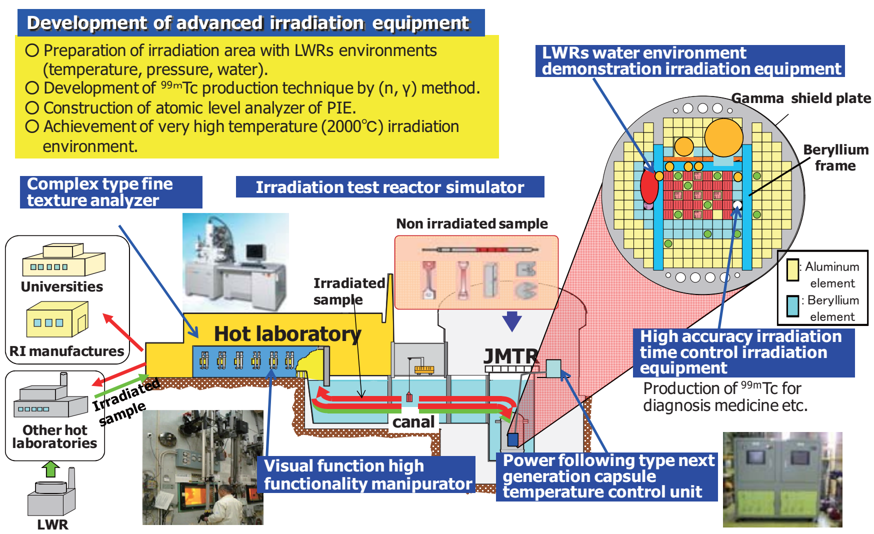


Fig. 5 Outline of JMTR core irradiation facilities

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7.3 Real-time Personal Exposure and Health Condition Monitoring System

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JAEA (Japan Atomic Energy Agency) and HAM (Hitachi Aloka Medical, Ltd) have proposed novel monitoring system for workers of nuclear facility. In these facilities, exposure management for workers is mainly used access control and personal exposure recordings. This system is currently only for reports management but is not confirmative for surveillance when work in progress.
Therefore, JAEA and HAM integrate access control and personal exposure recordings and two real-time monitoring systems which are position sensing and vital sign monitor. Furthermore change personal exposure management to real-time management, this system integration prevents workers from risk of accidents, and makes possible take appropriate action quickly.
This novel system is going to start for tentative operation, using position sensing and real-time personal dosimeter with database in Apr. 2012.

Keywords: real-time monitoring, exposure management, vital sign, position sensing, access control, workers dosimetry, wireless dosimeter,

1. INTRODUCTION

In order to radiation safety during radiation emergency and to justify radiation exposure to workers of nuclear facility, development of total monitoring system of radiation exposure dose is essentially needed. In nuclear facilities, exposure management for workers is mainly used access control and personal exposure recordings. This system is currently only for reports management but is not confirmative for surveillance when work in progress.

JAEA (Japan Atomic Energy Agency) and HAM (Hitachi Aloka Medical, Ltd) have proposed novel monitoring system for workers of nuclear facility.

2. CURRENT SYSTEMS

2.1. Access control and individual exposure management

This is usual systems for exposure management for workers in nuclear facility. Supervise section of facility check the gate and each personal dosimeters which workers wearing. It makes possible for restriction for entering facility, record the time into and left the facility, and record exposed dose of workers day by day, but it makes difficult for clarify current status of work.

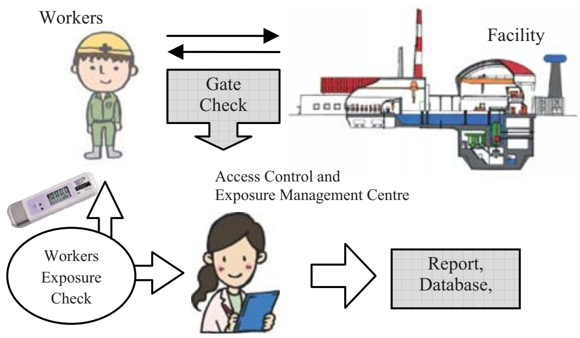

Fig.1. Image of Access Control and Individual Exposure Management System

2.2. Position sensing

When it occur some serious incidents in nuclear facility, positions information where workers are makes supervise section possible to accurate initial action quickly. So real-time position sensing system makes avoidance of risk for when it occur serial accidents. Installing the exciters and receivers facilities, position sensor which workers attached indicates gate check and where they are real-time.

2.3. Vital signs monitor

Workers healthcare are important factor to operate facility, and vital signs of them are important to keep them healthcare. Workers attached vital signs monitor which sends data real-time makes available to check the workers healthcare at any time

2.4. Wireless dosimeter

Dosimeter using individual exposure management measures integrated dose of workers, and supervision section confirms it day by day. Wireless dosimeter provides changes of exposure in working real-time to supervision section.

3. SYSTEM INTEGRATION

Integrating position, vital signs and exposure monitoring of real-time on current system makes real-time personal exposure and health condition monitoring system.

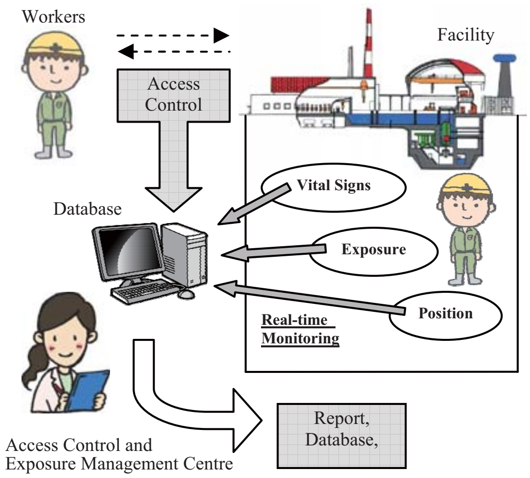

Fig.2. Image of Real-time Personal Exposure and Health Condition Monitoring System

Furthermore, change personal exposure management to real-time management, this system integration prevents workers from risk of accidents, and makes possible to take appropriate action quickly.

4. FUTURE PROSPECTS

This novel system is going to start for tentative operation using position sensing and real-time personal dosimeter with database in Apr. 2012.

In the future, this system is going to integrate internal exposure management for improvement of workers healthcare.

In addition, this management system which is especially for supervisor is going to improve from management to communication tools between workers and supervise section.

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7.4 OPAL Shield Design Performance Assessment: Comparison of Measured Dose Rates Against the Corresponding Design Calculated Values – A Designer Perspective

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A comparison of OPAL shielding calculations against measurements carried out during Commissioning, is presented for relevant structures such as the reactor block, primary shutters, neutron guide bunker, etc. All the results obtained agree very well with the measured values and contribute to establish the confidence on the calculation tools (MCNP4, DORT, etc.) and methodology used for shielding design.

Keywords: shielding, block, shutter, bunker, concrete, MCNP4, DORT

1. INTRODUCTION

The Australian Nuclear Science and Technology Organisation (ANSTO) signed in July 2000 a turn-key contract with the Argentinian engineering company INVAP for the construction of the replacement research reactor of the old HIFAR. The contract was awarded to INVAP based upon INVAP's conceptual engineering design. Detailed design was carried out during the following years, and was eventually followed by civil work, manufacturing and installation. This new reactor, placed nearby the old HIFAR in Sydney, was called Open Pool Australian Lightwater (OPAL) reactor.

The 20 MWt OPAL reactor has 16 fuel assemblies of MTR-type, in a compact 4x4 array. Fuel meat is U_3Si_2 in an Al matrix. Moderator and coolant is light water, while the paralelepiped core is surrounded by a heavy water tank reflector, where irradiation facilities are placed. Other facilities include neutron beams and a cold neutron source.

As design at INVAP progressed, a large number of shielding calculations were performed during the Preliminary and Detailed engineering phases. These official project reports included analysis on nearly all of the shielding features of the planned Facility. In 2006, exhaustive measurement of OPAL's shieldings was carried out during all Commissioning Stages, particularly at Stage C step-by-step power increases, up to full power operation at 20 MW. This later stage took the largest effort to both ANSTO [1] and INVAP teams. Among many other measurement points, more than 50 radiation base points (RBP) were selected to evaluate the effectiveness of the different shielding features. Neutron and gamma dose rates, both at contact and at 1 m away from the shielding were measured at these RBPs [1].

As a result of these measurements, comparison against design values could be done. In the present paper we show, were applicable, a comparison between the calculated and the measured dose rate data for some of the most relevant shieldings in the Facility.

2. REACTOR BLOCK

The Reactor Block is a massive structure of Heavy Weight Concrete (HWC), and serves as Primary Shielding of the Reactor. It surrounds the Reactor Pool and Service Pool, and also houses the Decay Tank room and control rod room, which are beneath the pools, see Figure 1.

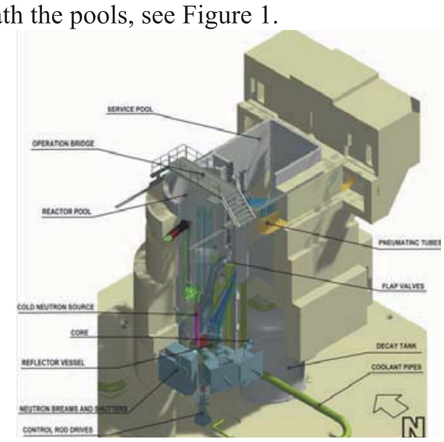

Fig.1. Reactor Block and some main components

Note that the Hot Cells are in the upper level, and are also part of the HWC Block.

The main source enclosed is of course the reactor core, but also relevant sources are the spent fuel assemblies (stored in the service pool), the irradiated molybdenum targets (“rigs”), and the N-16 from the activated coolant, inside decay tank room.

The heavy concrete was developed and produced from local suppliers, meeting stringent quality requirements.

In the HWC, fine and gross aggregates of an ordinary concrete have been replaced by iron ore, giving it a classical red-orange color. The final density was on average, 3.52 gr/cm^3, with an iron (Fe) proportion of 51 % in weight.

Considering full power operation, a Sn 2-D DORT model of the reactor pool section of the reactor block and main internals was done. The Sn method is a numerical technique for solving the finite-difference form of the Boltzmann transport equation. In this finite-difference scheme, not only the space, but also the angle and energy phases are discrete, and the resulting matrix is solved by means of an iterative process.

Nuclide data was obtained from Bugle-96 multigroup cross-section library, which is based on ENDF/B-VI library. Bugle-96 is a broad group library of 47 neutron and 20 gamma groups. Eight materials were used, most of them composed of various nuclides in homogenised composition. A P_3 Legendre polynomial expansion for differential cross sections was used, while the angular quadrature set is a S_8 cylindrical geometry. These are usual approximations for neutron/gamma transport shielding calculations. A uniform volumetric source is assumed in the core, with a typical neutron fission Watt spectrum used. Gamma ray source spectrum corresponds to prompt and decay fission products.

As a result, 2D R-Z flux contour plots were produced, for the neutron and the gamma groups. From these plots, radial and axial results were generated.

As an example, the Figure 2 shows the calculated radial dose rate, starting from the center core axis, at left. Radial shielding thickness is 230-cm HWC at core center level.

Note that HWC is a very good gamma absorber as compared with water. From Figure 2 the gamma dose at reactor face (right outside of the biological shield) is 0.008 μSv/h and neutron doses are negligible.

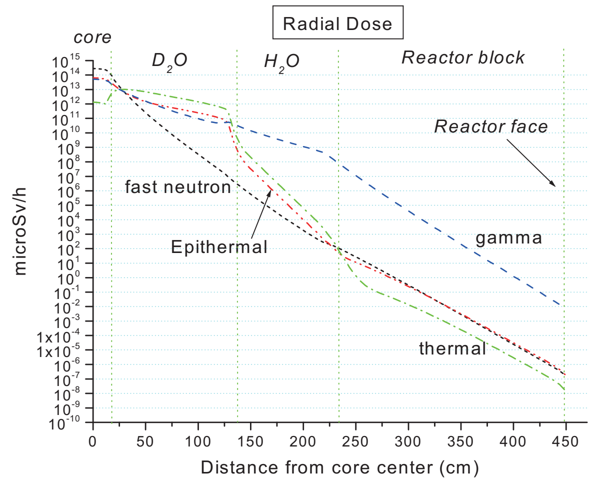

Fig.2. DORT calculated core radial dose rates

During Commissioning measurements, the radial contact dose rate was nearly background (lower than 0.1 μSv/h), i.e. the core contribution was lower than the radiameter minimum detection.

Regarding dose in the lower core axial direction, i.e. at the control rod drives room roof, the calculation showed a maximum dose rate of 8 uSv/h (gamma) for the bulk of the shielding, ouside the mechanism penetration. (which was homogenous in this DORT model). Commissioning measurement showed a dose of around 9.5 μSv/h in an equivalent position, thus a good agreement occurs. This is for the bulk part of the shielding, outside the mechanism penetration which is described on next section.

3. MECHANISM PENETRATION

The mechanism penetration is a heavy grouting-filled cylinder embedded in the Reactor Block. It houses the bars for moving the five control rods, the fuel element fastener rods and other fastener, which enters the core from below, see Figure 3. In this figure, the core is at the upper part, and the control rod room is in the bottom, where the motors, valves, etc. for the mechanism bars are placed.

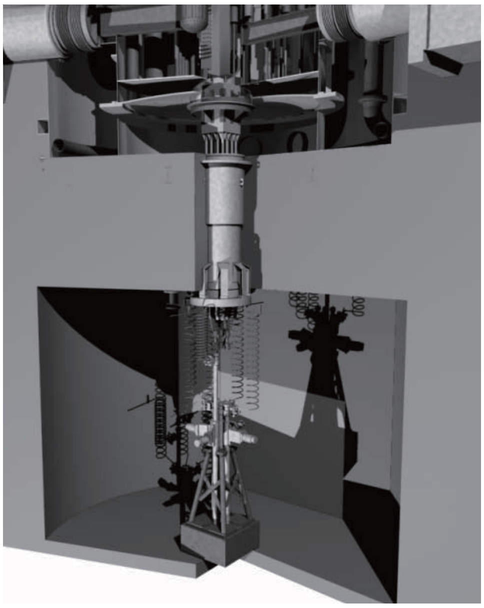

Fig.3. Cutaway of Reactor Block, showing the position of the mechanism penetration (center)

All bars in the penetration are surrounded by a water blanket. Possible neutron or gamma streaming through them could not be neglected, thus a detailed calculation was necessary.

A montecarlo MCNP4c model was done, from the core up to CRD roof. A 3 group, cylindrical weight window-superimposed mesh scheme was implemented for neutrons and photons. In this model, the build-up of radiation from penetration sides was neglected.

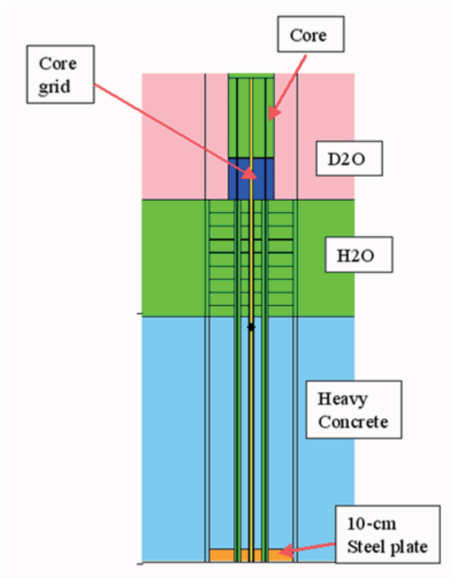

Fig.4. Model of mechanism penetration

Although the model predicted some gamma streaming through the control rods bar (at the steel plate height, see Figure 4), these were impossible to measure during Commissioning, because the mentioned bars continue all along up to the motors, thus the streaming actually does not occur. Where measurable, in points nearby the steel plate, results were in the order of magnitude of the MCNP calculations. Neutrons were not detected, as predicted by model.

However, during the mentioned Commissioning measurements, one little 5x5 cm hot spot was identified, corresponding to the water flushing pipe which was not included –mistakenly - in the MCNP model. Highly collimated gamma radiation emerging from this hot spot was about 570 μSv/h at one meter of the shield surface.

This streaming was conveniently identified in the room, and additional shielding was installed (5-cm lead shielding) to correct this issue.

4. PRIMARY BEAM SHUTTERS

The primary beam shutters must provide adequate shielding - while the corresponding beam is not being operated- against neutrons and gamma rays coming directly from the core through the neutron guides, as well as gamma rays generated due the (n,g) reactions between neutrons and the materials they encounter on their way. See Figure 5

The shutter has impressive atenuation capacity. As an example, the neutron (total) flux at the beam tip, a few cm from the core, is higher than 1E+14 n/cm^{2}s and must be reduced to only a few neutrons/cm^{2}s such as to comply with the dose rate limit set at the reactor face, in the reactor beam hall.

Shutters are of the “dry” type, i.e., they do not have water as a shielding material. Instead, main shielding comprises polyethylene and steel, optimized for the existent neutron/gamma spectrum. To close, the shutter rotates along its axis, in such a way as to stop particles streaming through beam guides. Helium is circulated through the in-pile section of the beam and shutter for cooling purposes.

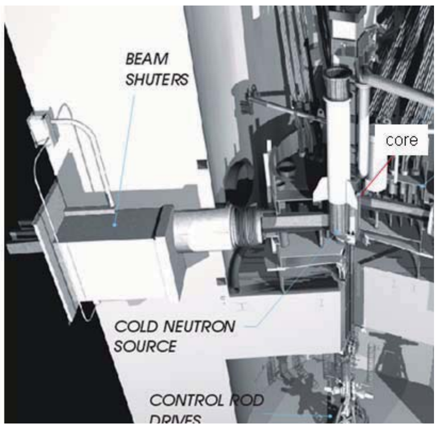

Fig.5. Cutaway showing one of the shutters (at left)

An MCNP4c, 3-D model of the thermal beam and the in-pile components was developed. The purpose of this model was to calculate the neutron and photon source at the inlet of the thermal beam shutter.

This saved source (2E+06 particles) was later used in a detailed MCNP model of the primary shutter (see Figure 6) for the Thermal Guide (TG). An importance scheme for both neutrons and gamma was necessary, in order to transport enough particles up to reactor face, which is located on the upper part of Figure 6.

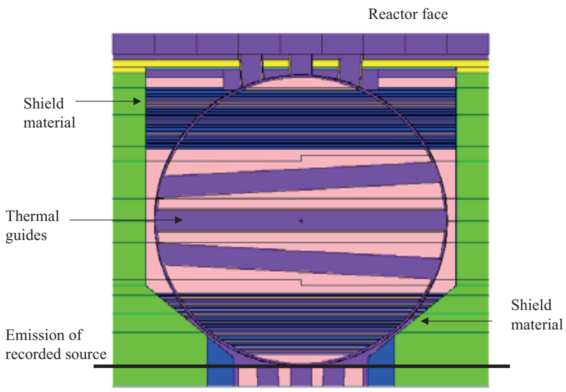

Fig.6. Detailed MCNP model of primary shutter

In a similar way, another MCNP model was prepared for the Hot Beam shutter (HB).

The following Table 2 illustrates the selected calculation results, as compared with the Commissioning measurements carried out, for the relevant Radiation Base Points (RBPs), in those locations capable of comparison. In general, there is a reasonable agreement between calculations and measurements.

Table 2 Comparison of contact dose rates

RBP	Calculated (µSv/h)	Measured (µSv/h)	C/M
RBP#14 (HB1/HB2)	20(***)	42(***)	0.48
RBP#15A (TG1/TG3)	24	10(*)	2.4
RBP#13B:	N/A(**)	9.8	

Notes: (*) Relevant activation
(**) Not directly comparable
(***) Neutron+Gamma

5. NEUTRON GUIDE BUNKER

The neutron guide bunker corresponds to the shielded enclosure around the guides, in the curved section of them, from reactor face up to the experimental facilities in the Neutron Guide Hall building. See Figures 7 and 8.

These guides serve to transport thermal and cold neutrons, from the reactor to the experimental facilities.

Bunker is around 40 meters long, and most of its walls and roof are constructed in ordinary reinforced concrete (2.2 gr/cm^3 design density), with a thickness from 0.7 to 1.2 meters.

Fig.7. Neutron guides inside bunker. Reactor Face is at bottom.

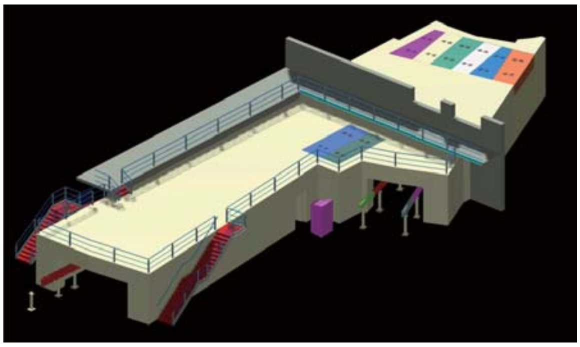

Fig.8. AutoCAD model of the bunker

An MCNP4c detailed model of neutron guides and bunker was developed.

Using the previous model of the shutter, transport of particles up to Reactor Face was carried out. This allowed to save a particle source in Reactor Face, which was used in the consequent transport along neutron guides.

An importance scheme was applied for both neutron and gamma through the model walls and roof.

Main shielding problem arises from fast neutrons, that scatter along the guide, disperse in all directions, and reach the concrete shielding. As these neutrons have a high energy, they penetrate a lot inside concrete.

The hydrogen content of shielding, critical to moderate these fast neutrons, was assumed as a fully dried out concrete (0.25 weight %). That means, all water is chemically bound to the concrete. This was a conservative assumption, since real amount would be fairly higher.

The Table 3 shows a comparison between the calculations and the Commissioning measurements, for the points that are equivalent.

Table 3 Comparison of dose rates at Bunker

RBP	Calculated (μSv/h)	Measured (μSv/h)	C/M
RBP#25	45.5	23.0	2.0
RBP#23A	31.4	10.1	3.1
RBP#23B	30.4	4.1	7.4
RBP#23E	7.2	1.3	5.5
RBP#30A	730	40.0	18.3
RBP#30B	105	8.3	12.7

In general, results are within the order of magnitude, and calculations probed conservative. In support of this conclusion, it should be taken into account, that the actual average density of the ordinary concrete, measured during test of probes, was 2.45 gr/cm3, and that the best-estimate of hydrogen content at the time of Commissioning, based upon mix report compositions, was 0.6 wt %.

To investigate this mater, the calculated effect of hydrogen content can be seen in next Table 4, assuming a fixed concrete density.

Table 4 Calculated effect of hydrogen content (Initial dose rates normalized to 10)

RBP	Hydrogen weight %		Factor
	0.1 %	0.5 %	
RBP#25	10.0	2.92	3.4
RBP#210	10.0	2.02	4.9
RBP#30A	10.0	1.96	5.1
RBP#30B	10.0	3.13	3.2

6. COOLANT ACTIVATION

Here with the term “coolant activation” we group different radionuclides that apear in both the cooling ordinary water and the heavy water reflector. For simplicity purposes, we will focus only in the ordinary water (H_2O), although some statements are also valid for the D_2O reflector.

The radiation sources in the circulating water are due to the activation of natural elements contained in pure water, activation of water impurities, and activity induced by the activation of corrosion products.

First of all, the most relevant activity concerning is the generation of N-16, which is produced near the core through the reaction with fast neutrons: $^{16}O(n,p)^{16}N$. To diminish doses in the process rooms to acceptable levels, decay tanks are incorporated for this isotope, at the outlet of the core, with a residence time of 120 seconds, which is enough, considering the little half life of N-16 (7.1 seconds). These tanks are enclosed inside the heavy concrete of the reactor block.

Regarding activation of corrosion products, in MTR reactors is important the Na-24, which comes mainly through the reaction $^{27}Al(n,alfa)^{24}Na$ from structural aluminium (cladding of fuel assemblies, etc.).

Concerning the activation of water impurities, the most relevant is the production of Ar-41, which comes from activation of dissolved argon in water.

To diminish dose rate at pool top due to these and other isotopes, a Hot Water Layer system was provided. This system assures a non-activated water acting as shielding between the lower part of the pool and the upper zone, where operations are carried out.

A 2D, three-neutron groups calculation was used, for estimation of the reaction rates of the relevant isotopes that occur in the core, reflector and nearby zone of reactor pool.

With all the process data from the involved systems (primary cooling system, service pool cooling, purification system, hot water layer, etc.) it was prepared a compartment model, in which radioactivity flow is modeled, taking into account mass flows, compartment volumes and residence times in each cell.

This model allowed to estimate the water activity of the each isotope, in reference points of the systems (core outlet, hot water layer, service pool inlet, etc.).

In particular, using the calculated concentration of the pool, it was estimated the dose rate in the surface of the pool by means of a very simple MicroShield model. Calculation resulted in 4.1 μSv/h, which is in a good agreement with the measured values of around 0.5 to 2.0 μSv/h, see Figure 9.

Fig.9. Measurements at top of reactor pool

During Commissioning, a number of water samples were taken from plant systems, and some of them analysed by means of high resolution spectrometry. This is the case of the samples of the primary coolant, taken near the operating main pumps. The following Table 5 shows the comparison of calculated activity concentrations against measured ones.

Table 5 Comparison of primary coolant activity values

Isotope	Calculated (Bq/cm^3)	Measured (Bq/cm^3)
Ar-41	617.0	N/V(*)
Na-24	811.0	310.0
O-19	127.0	N/V(**)
Mg-27	80.0	Not detected
Cr-51	9.0	<0.77
Mn-56	0.9	4.7
Cd-115	N/C(***)	59.0
In-117	N/C(***)	74.0
W-187	N/C(***)	28.0

(*) Not valid due to sample degassing/isotopic exchange before measurement (**) Not valid due to decay before measurement. (***) Not Calculated

For those nuclides that can be directly compared, measurement results are within the order of magnitude of calculations. In addition, some non-considered nuclides appeared, probably from activation of burnable poison and structural weldings.

7. CONCLUSIONS

It was shown that a very good agreement between calculations and measurements was obtained for most of the OPAL shieldings.

These good results are explained because of:

a) INVAP's teams previous experience in other reactors (Egypt MPR, etc.), methodology, coordination efforts, etc

b) Widely accepted codes for calculations, such as MCNP and DORT.

c) Large effort of highly-trained people, including ANSTO staff, during Commissioning measurements.

In the few cases with non-expected dose rates (i.e. streaming through shielding), they are easily explained.

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4th International Symposium on Material Testing Reactors, Oarai, Japan, December 5-9, 2011

7.5 OPAL Reactor calculations using the Monte Carlo Code Serpent

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In the present work the Monte Carlo cell code developed by VTT Serpent v1.1.14 [1] is used to model the MTR fuel assemblies (FA) and control rods (CR) from OPAL (Open Pool Australian Light-water) reactor in order to obtain few-group constants with burnup dependence to be used in the already developed reactor core models [2]. These core calculations are performed using CITVAP 3-D [3] diffusion code, which is well-known reactor code based on CITATION [4].

Subsequently the results are compared with those obtained by the deterministic calculation line used by INVAP [3], which uses the Collision Probability Condor cell-code [5] to obtain few-group constants. Finally the results are compared with the experimental data obtained from the reactor information for several operation cycles.

As a result several evaluations are performed, including a code to code cell comparison at cell and core level and calculation-experiment comparison at core level in order to evaluate the Serpent code actual capabilities.

Keywords: OPAL reactor – Monte Carlo cell code Serpent v.1.1.14 – MTR core calculations – Condor CP cell code

1. INTRODUCTION

Monte Carlo neutron transport codes are widely used to perform criticality calculations and to solve shielding problems due to their capability to model complex systems without major approximations. However, as far as these codes usually demand high computational resources, other natural applications such as burnup and cell calculations have been prohibitive in the past decades. Nevertheless, due to the advances in computer performance in last years, the utilization of this solving method to perform cell level calculations arises as a very promising near-future application.

In this work the Monte Carlo Serpent cell-oriented code [1] developed by VTT Technical Research Centre of Finland is used. This code is a three-dimensional continuous-energy code designed to perform cell-level fuel assembly calculations in order to obtain few group constants to be used in core calculations. In addition the code is oriented to pin-type cell calculations and has several interesting features such as burnup capabilities and improved methods to optimize calculation time.

Several tests have been done by INVAP in order to study the code capabilities to model pin-type cells using theoretical benchmarks [6]. Thus, the objective of the present work is to study the capabilities of Serpent code to model MTR-type cells and obtain few group constants to be used in core calculations. Therefore, the OPAL reactor is chosen to evaluate the Serpent code capabilities.

The neutronic model for OPAL reactor has been carried out using a deterministic calculation line [3], which uses the Collision Probability-2D Condor cell code to obtain condensed few-group cross section to perform core calculations with CITVAP core code.

Therefore, in this work the Serpent code is introduced into the neutronic model for OPAL reactor [2] developed by INVAP. In order to simplify the comparison only the FA and the CR are be modeled with Serpent, where the same characteristics that those from Condor models are used. Afterwards the obtained few-group constants were used to perform core level calculations in CITVAP for reactor critical positions for operation cycles 7 to 15.

As a result several comparisons at cell and core levels can be done, including a calculation experiment comparisons that allows evaluating the Serpent code actual capabilities.

2. OPAL NEUTRONIC MODEL

OPAL is a state-of-the-art 20 MW research and production reactor fueled by LEU MTR-type FA, cooled by light-water and reflected by heavy water. The core consists of a 4x4 grid, where each position is filled with a FA with 21 fuel plates. Each fuel plate consists of an Aluminum cladded plate of low-enriched uranium Silicide dispersed into an Aluminum matrix. The core grid is surrounded by a Zicalloy chimney and includes an internal central cross box to allocate five Hafnium CRs. Besides, the reactor counts with four control plates and a central cross-shaped regulating control rod. Beyond the core chimney a reflector tank contains the heavy-water that behaves as neutron reflector.

In addition several types of FA are used in cycles 7 to 15, where the main difference between each type is the Uranium load and the presence of Cadmium wires (used as burnable poison). Furthermore, several irradiation positions and neutron beams are located into the heavy water reflector tank.

The OPAL neutronic model [2] consists of a Cell level calculation for condensation and homogeneization (performed with Condor using multi-group nuclear data) that allows obtaining a 3-group library to be used at core level calculations performed with CITVAP code, where the group condensation scheme is shown in the following table.

Group	Upper limit (eV)	Lower limit (eV)
1	From library	0.821 e6
2	0.821 e6	0.625
3	0.625	From library

Table 1 Group limits for few-group libraries.

In the present work only the FA and CR (in the inserted position) cell level models are developed with Serpent, where the compositions, dimensions, burnup steps and average power are the same as those from Condor models in the OPAL neutronic model [2].

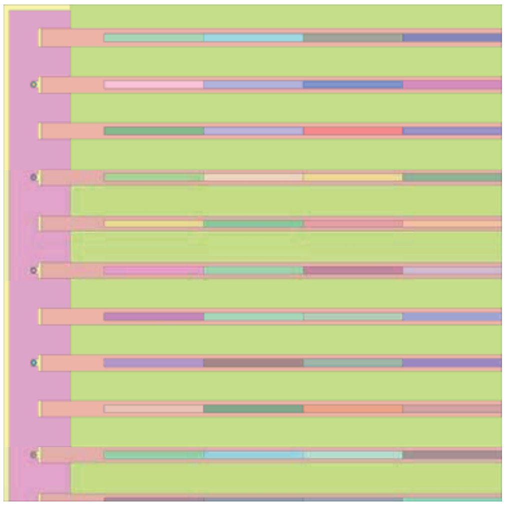
The following sections describe the cell models developed in Serpent and the model in CITVAP used for core-level calculations.

2.1. CELL MODELS FOR FA IN SERPENT

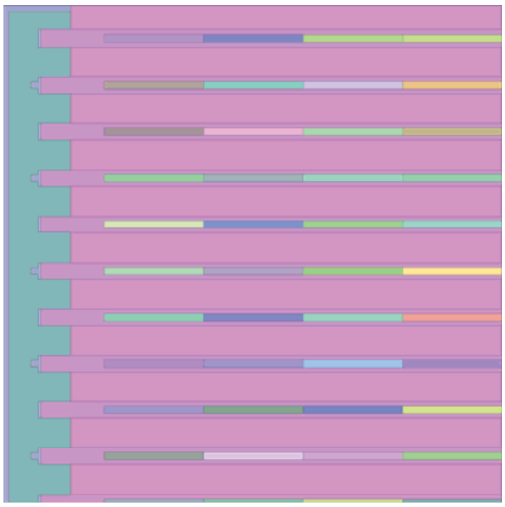
Cell models in Serpent were performed considering a quarter of FA. The FAs modeled correspond to the cycles 7 to 15 of operation (namely si_210, si_380 and si_480), where FAs si_380 and si_480 are represented with and without BP. In addition ENDF-VI.8 (ACE format) nuclear data was chosen for these models.

Afterwards all the few-group cross sections were obtained from Serpent outputs and converted to CITVAP format through a pos-processing code developed ad-hoc.

In order to eliminate modeling differences between Condor and Serpent, the models developed in Serpent for FA considered the same divisions and number of materials than those from Condor. As an example, several captures are presented in the following Figure.



a)



b)

Fig. 1 X-Y views for Serpent FA models. a) SI_480 w/BP b) SI_480 w/o BP

As far as a quarter of FA was modeled, the boundary condition in Serpent was set to specular reflection. In addition the homogenization was performed running two cases for each model, considering the plate zone in one case and the frame zone in the other one. Besides, in order to obtain the accumulated fluence in the frames for each burnup step, the thermal flux reported by Serpent was multiplied by the irradiation time and included in the few-group cross section library.

2.2. CELL MODELS FOR CR IN SERPENT

To model CRs in Serpent a 2-D geometry was used to reproduce 1-D Condor models for Control Plate and the Regulating plate inserted. This models considers a two zones scheme to represent the CRs at core level (namely zones 1 and 2). Again the boundary condition was set to specular and ENDF-VI.8 (ACE format) nuclear data was chosen. Afterwards, all the cross sections were obtained from Serpent outputs and converted to CITVAP format through the same pos-processing code used for FAs models.

2.3. EXTRA CELL MODELS FOR FA IN CONDOR

The FA cell models in Condor were obtained from [2]. It must be noted that these models use the esin2001 69-group nuclear data library, which is primarily based on WIMS-69 (namely Condor+esin).

In addition these models for all FA types were re-run using the Helios 190-group library [7] in order to compare results using a different nuclear data library (namely Condor+Helios). It must be said that as far as esin2001 library has not Hafnium nuclear data, the CRs models in Condor from [2] are originally run using Helios 190-group library.

2.4. CORE MODEL

The core model in CITVAP was obtained from [2]. An X-Y cut in the centre of the active zone is shown in the following Figure.

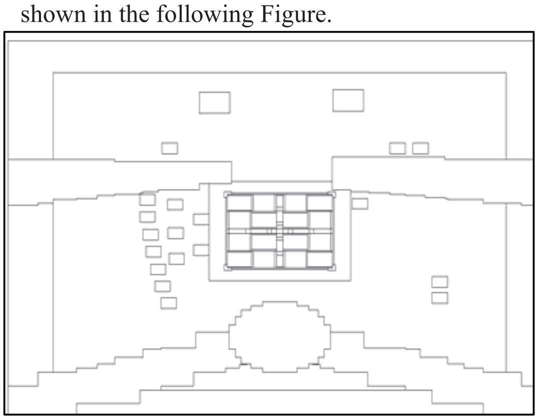


Fig. 2 X-Y cut for CITVAP model (centre of core)

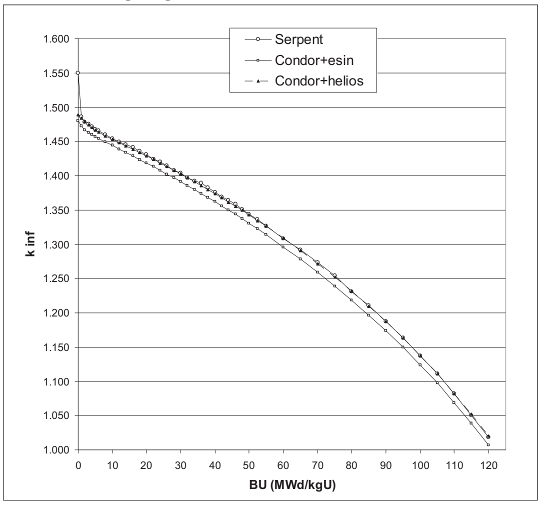
Afterwards, using the CR critical positions reported by the reactor operation data from cycles 7 to 15, a comparison experiment-calculation can be performed using alternatively the few-group cross sections obtained from the cell models from the last sections. In addition it must be noted that cycles 7 to 12 use a combination of all FAs, while further cycles only use si_480 w/BP fuels.

3. RESULTS AND ANALYSIS

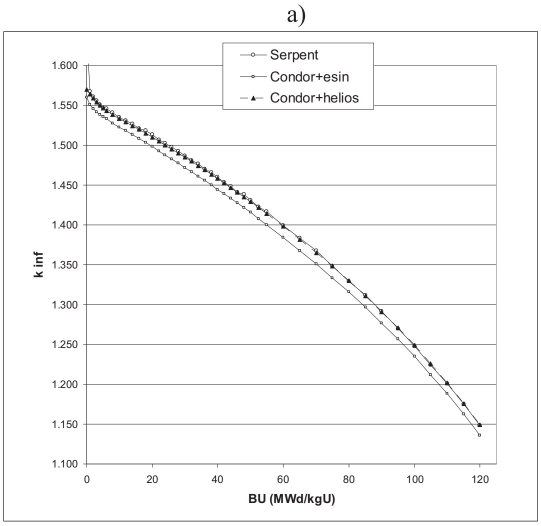
In this section a code to code comparison is performed at cell level, while a comparison with experimental results is performed at core level.

3.1. CELL LEVEL COMPARISONS
3.1.1. INFINITE MULTIPLICATION FACTOR

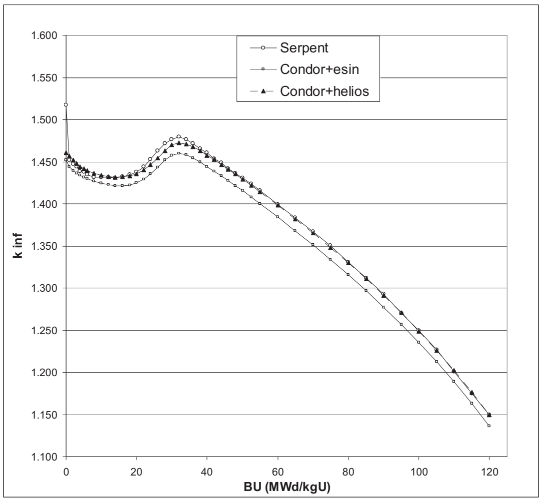
The infinite multiplication factor for the models developed in Condor and Serpent are compared in the following Figure.



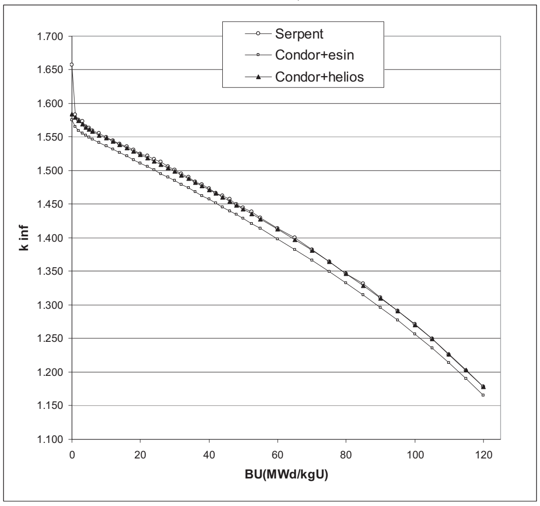
a)



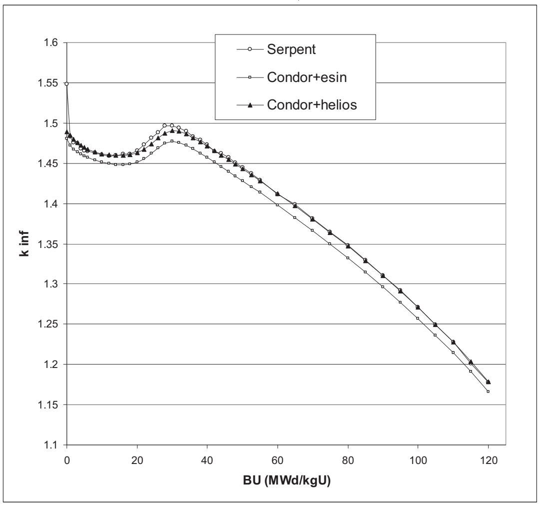
b)



c)



d)



e)

Fig. 3. Excess reactivity comparison between Condor and Serpent models a) SI_210 b)SI_380 w/o BP c)SI_380 w/BP d) SI_480 w/o BP e) SI_480 w/ BP

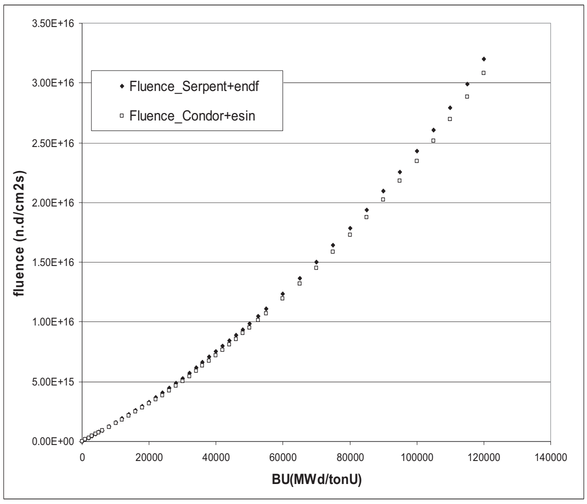
As it can be seen in the Fig. 3 the results from Condor models and Serpent are consistent. It must be noted that the differences between Condor+Helios and Serpent model (<200 pcm) are lower than the difference between Condor+esin and Serpent (<1100pcm). This reduction can be justified from the differences in the nuclear data.

Furthermore it must be regarded that the Cd burnup is consistent between all models.

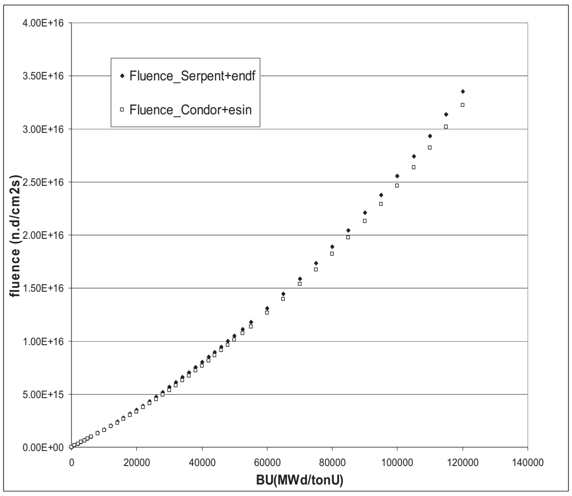
3.1.2. FRAMES FLUENCE

As far as the core-level evolution for frames with burnable poisons is performed using thermal fluence instead of assembly burnup, the frames fluence has been compared between Condor+esin and Serpent models.

The results for FA with BP are presented in the following Figure.



a)



b)

Fig. 4 Frame thermal fluence comparison between Condor+esin and Serpent models a) SI_380 w/BP b)SI_480 w/BP

As it can be seen, the results from both models are consistent.

3.1.3. CONTROL RODS CROSS SECTIONS

The CR cross section calculations have been performed using the models described in the previous section.

The results showed several minor differences in all few group cross sections (<20%), but as far as the most relevant condensed parameter is the absorption cross section, the percent differences are presented in the following table.

	Difference (Condor + Helios) - Serpent [%]		
Cell model	Σa_1	Σa_2	Σa_3
Central CR zone 1	-18	-17	0
Central CR zone 2	-21	-15	0
Plate CR zone 1	-17	-13	2
Plate CR zone 2	-18	-15	1

Table 2 Differences for the condensed absorption cross section.

It can be seen that Serpent condensed absorption cross section are almost equal for the thermal group but ~10/20 % higher for the other groups.

3.1.4. RUNNING TIME

All runs were performed in sequential mode in a Dual-Core AMD Opteron (tm) Processor 2212 (2MHz). The involved running times for FA cases are presented in the following table.

Cell model	**Running time [min]**
Condor+esin	20-25
Condor+Helios	100-120
Serpent + endf	1300-1600

Table 3 Running time for FA cell models.

As it can be seen the running times in Condor code is appreciably lower than Serpent ones, even when the 190-group Helios library is used. In spite of this, the running times for Serpent are fairly acceptable, especially considering that it is a Monte Carlo code. In addition Serpent code has the capability of parallel run, which can significantly diminish the real time of each run.

3.2. CORE LEVEL COMPARISONS

The few group cross sections obtained from cell calculation were introduced in the core model for OPAL reactor for cycles 7 to 15 considering a history record for CR positions in those cycles. It must be noted that the value of reactivity obtained represents a calculation error with experimental results for those cycles.

The following Figure presents the results for the cycles 10 to 15 for the following cases:

1. INVAP calculation line (i.e. Condor + esin cell calculations + CITVAP)
2. INVAP calculation line using Helios library for FA (i.e. Condor + Helios cell calculations + CITVAP)
3. INVAP calculation line plus Serpent cell code for FA and CRs (i.e. Serpent cell calculations + CITVAP)
4. INVAP calculation line plus Serpent cell code only for FA (i.e. Serpent cell calculations + CITVAP)

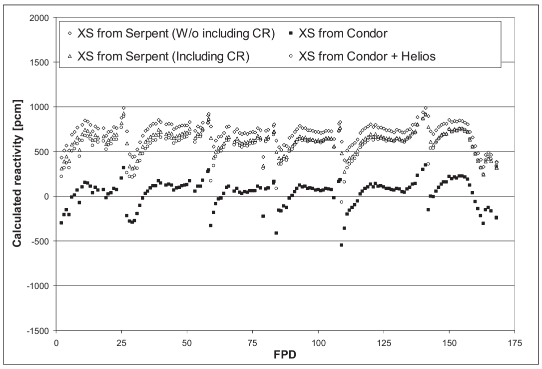


Fig. 5 Core calculations comparison for cycles 10 to 15

It can be seen from Figure 5 that the behaviour of the core results is similar in all cases. Moreover the calculations performed using Serpent or Condor+Helios show the same tendency already observed in cell calculations, i.e. a more reactive result. In addition, the difference with the cases with Condor+esin are lower in core calculations (<700 pcm).

Furthermore, the inclusion of CR cross sections calculated by Serpent show a slight decrease in core reactivity of almost 100 pcm, having the same tendency that the cell-level results.

Finally it must be noted that no change in the burnup behaviour has been observed when including the few group cross sections calculated for the different models.

4. CONCLUSIONS

Calculations for OPAL reactor for cycles 7 to 15 using Serpent v.1.1.14 as cell code have been performed using new cell models for FA and including the models for CRs.

The obtained results have been compared with those obtained from the INVAP´s calculation line and showed to be consistent. The differences encountered (of almost 700 pcm) between the Serpent and the Condor+esin calculations are possibly due to the use of newer nuclear data. Subsequently, when Helios library is used in Condor calculations these differences are much lower (<100pcm).

Besides the running times involved in Serpent models looks fairly acceptable.

Finally, regarding the obtained results, the Serpent code arises as a plausible alternative for MTR-type cell calculations, although it has been designed for pin-type FA calculations.

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8. Advancement of Irradiation Technology (3)

4th International Symposium on Material Testing Reactors, Oarai, Japan, December 5-9, 2011

8.1 Irradiation Creep and Growth Behavior of Zircaloy-4 Inner Shell of HANARO

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The inner shell of the reflector vessel of HANARO was made of Zircaloy-4 rolled plate. Zircaloy-4 rolled plate shows highly anisotropic behavior by fast neutron irradiation. This paper describes the analysis method for the irradiation induced creep and growth of the inner shell of HANARO. The anisotropic irradiation creep behavior was modeled as a uniaxial strain-hardening power law modified by Hill's stress potential and the anisotropic irradiation growth was modeled by using volumetric swelling with anisotropic strain rate. In this study, the irradiation induced creep and growth behavior of the inner shell of the HANARO reflector vessel was re-evaluated. The rolling direction, the fast neutron flux, and the boundary conditions were applied with the same conditions as the actual inner shell. Analysis results show that deformation of the inner shell due to irradiation does not raise any problems for the the lifetime of HANARO.

Keywords: Zircaloy-4, irradiation creep, irradiation growth, HANARO

1. INTRODUCTION

The reflector vessel (Fig.1) of HANARO was made of Zircaloy-4, and it contains heavy water used for reflector. The inner shell of the reflector vessel provides a boundary between the light water cooled and moderated inner core region and the heavy water cooled and moderated outer core region.

The inner shell of the reflector vessel surrounding the core is the most critical part from the viewpoint of a neutron irradiation. When zirconium alloys are exposed to a fast neutron flux, they exhibit permanent anisotropic dimensional changes called 'irradiation growth'. Irradiation growth is independent of the material state of stress. Concurrent with irradiation growth, another phenomenon called 'irradiation creep' occurs. This is irradiation deformation caused by mechanisms activated by an applied stress.

According to the design documents of HANARO, the operating pressure across the thickness of the inner shell wall is about 20kPa, the pressure being higher inside the heavy water region of the reflector vessel than in the light water region on the inner core side. This situation enhances the possibility that the combination of irradiation induced creep and growth may cause sufficient inward deformation of the inner shell wall to interfere with the proper safe operation on the control and/or shutoff absorber units.

To check this situation, the analysis on the irradiation creep and growth of the inner shell was performed at the design phase of HANARO. The thickness and the shape of the inner shell was determined based on this analysis. However the analysis performed at the design phase could not implement the actual material orientation of the inner shell and the plastic behavior of the zirconium alloy due to computing hardware and software constraints.

In this study, the analysis scheme for irradiation creep and growth of a zirconium alloy was re-established and the analysis of the inner shell was performed to reflect the actual operating condition of HANARO. Based on the analysis, the safety of the inner shell on the view point of the irradiation induced deformation was re-evaluated.

2. IRRADIATION CREEP AND GROWTH PROPERTIES OF ZIRCALOY-4

By default, the values in reference [1] are used for the irradiation creep and growth properties of Zircaloy-4. According to reference [1], the creep anisotropic factors were calculated by using the f-texture factor of a cold-worked and annealed Zircaloy-4 slab and the creep resistance factor of a zirconium alloy pressure tube material. Also anisotropic irradiation growth rates were calculated based on the creep anisotropic factors and f, g-texture factors.

The normalized values of irradiation growth rates are as follows:

Circumferential : -0.2978
Longitudinal (rolling direction): 1.0000
Radial (thickness): -0.7023

For irradiation creep at low stresses (<140MPa) the effective stress (σ) and effective creep rate ($\dot{\varepsilon}$) are related by the following equation [2],

$$\dot{\varepsilon} = K_C \cdot \Phi \cdot \sigma \qquad (1)$$

where Φ is the fast neutron flux and K_C is the creep rate constant of the material and depends on the metallurgical structure and temperature. The ratio of the actual to isotropic yield was calculated by using the creep anisotropy factors. Direct tension yielding YRDIR and yield in shear YRSHR are as follows:

YRDIR(11) = 1.0196
YRDIR(22) = 0.8868
YRDIR(33) = 1.1422
YRSHR(12) = 1.0178
YRSHR(13) = 1.2054
YRSHR(23) = 1.1409

3. MODELING

ABAQUS version 6.11 was used for the analysis.

3.1 Geometry and Mesh

HANARO inner shell is a corrugated plate with 7mm thickness and 1160mm height. At the design phase analysis, the inner shell was modeled without the upper/lower plates and the outer shell. Also a quarter of the inner shell was modeled assuming the symmetric loads and neutron flux distribution.

However the whole reflector vessel including the inner shell, the upper/lower plates, and the outer shell was modeled in this study to ensure the boundary effects and to consider the non-symmetric neutron flux distribution (Fig.1).

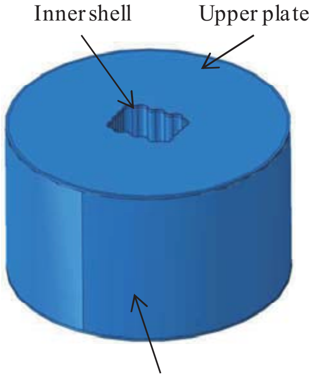

Fig.1 HANARO reflector vessel

Nodes for the inner shell were generated at intervals of 20mm along the axial direction and 10mm along the circumferential direction. A 5 layer mesh was generated along the thickness. The 8-node linear brick with reduced integration (C3D8R) element was used and the number of elements was 98670. Because the neutron flux on the upper/lower plate and the outer shell is much lower than that of the inner shell, there is a little effect on the upper/lower plate and the outer shell due to irradiation. Therefore the upper/lower plate and the outer shell were modeled as an isotropic material and no irradiation induced property was applied to them. To implement the anisotropic behavior to the inner shell, the material orientations were applied (Fig.2). In Fig.2, 1-direction is thickness and 2-direction is plate rolling direction.

3.2 Material Properties

3.2.1 Elastic material properties

Elastic material properties are as follows:

- Young's modulus : 91.36GPa
- Poisson's ratio : 0.33

3.2.2 Irradiation growth

Irradiation growth was simulated by using the swelling option of ABAQUS. According to reference [3] and [4],irradiation growth of Zircaloy-4 plates irradiated at low temperature is saturated at a relatively low fluence level and there is no additional growth due to increased neutron fluence. In this study, irradiation growth was modeled such that 0.05% irradiation induced growth strain occurs at the region of higher than 1E+24/m^2 fast fluence (>0.821MeV) during 1 year and there is no additional irradiation growth during the remaining lifetime. Anisotropic swelling was implemented by setting the swelling rate ratio of the ABAQUS swelling suboption. Anisotropic swelling rate of ABAQUS is defined as the following equation:

$$\dot{\varepsilon}_{11}^{sw} + \dot{\varepsilon}_{22}^{sw} + \dot{\varepsilon}_{33}^{sw} = \frac{1}{3}\left(r_{11} + r_{22} + r_{33}\right) \cdot \dot{\varepsilon}^{sw} \qquad (2)$$

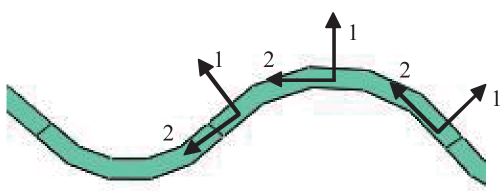

Fig.2 Example of material orientation

where $\dot{\varepsilon}^{sw}$ is the volumetric swelling strain rate and $\dot{\varepsilon}^{sw}_{ii}$ is the i-directional swelling strain rate. Refer to the results of Section 2, the following values were used as anisotropic swelling rates.

r11 = -0.2978 (thickness)
r22 = 1.0 (rolling direction)
r33 = -0.7023 (axial direction)

3.2.3 Irradiation creep

Creep behavior was specified by the equivalent uniaxial behavior. The strain hardening version of the power law creep model was selected to model the creep behavior. The creep strain rate was assumed to be proportional to the uniaxial equivalent deviatoric stress and the fluence. According to reference [5], creep constant was set to 2.8E-31 $m^2 \cdot neutron^{-1} \cdot MPa^{-1}$ for α-annealed zirconium plate material with a dislocation density of 1.4E13m^{-2}.

Anisotropic creep was defined to specify the stress ratios that appear in Hill's function. Creep ratios are used to scale the stress value when the creep strain rate is calculated. Creep stress ratios are defined as constants and they have the same value with ratios of the actual to isotropic yield discussed in Section 2.

3.2.4 Loads

At the design phase of HANARO, the cover gas of the expansion tank of the heavy water cooling system was designed to be pressurized at 20kPa. In that case the operating pressures of the reflector vessel are as follows:

a. Heavy water region
   - Reflector vessel top : 217.2 kPa
   - Reflector vessel bottom : 226.9 kPa
b. Light water region (inner shell)
   - Reflector vessel top : 196.1 kPa
   - Reflector vessel bottom : 207.6 kPa
c. Light water region (except inner shell)
   - Reflector vessel top : 178 kPa
   - Reflector vessel bottom : 191 kPa

During shutdown, only the pressure due to the difference in the static head of light water and heavy water are considered.

a. Heavy water region
   - Reflector vessel top : 158 kPa
   - Reflector vessel bottom : 172 kPa
b. Light water region
   - Reflector vessel top : 197 kPa
   - Reflector vessel bottom : 210 kPa

However HANARO has been operating without pressurizing the cover gas in the expansion tank and this makes the pressure acting on the inner shell during operation at almost zero. Therefore it is assumed that there is no pressure acting on the inner shell during operation. Also the dead weights of the structures installed on the reflector vessel, such as the chimney, control rod driving mechanisms, and etc., are not considered because these loads are much smaller than the loads induced by the hydrostatic pressure.

3.2.5 Boundary conditions

The reflector vessel is firmly fixed to the grid plate by using 30 bolts. Therefore the bottom of the reflector vessel was assumed to be a fixed boundary condition and there was no other boundary condition.

3.2.6 Neutron fluence

Neutron fluence was assumed to have cosine distribution along the axial direction. The fluence of the inner shell varies over circumferential location. In this study, the inner shell was divided by four regions (Fig. 3) and peak fluences of the each region are set

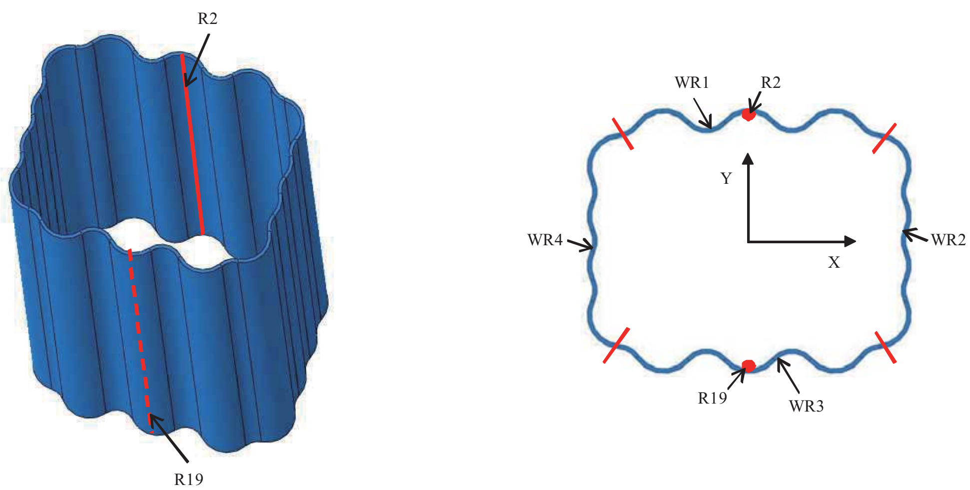

Fig.3 Inner shell

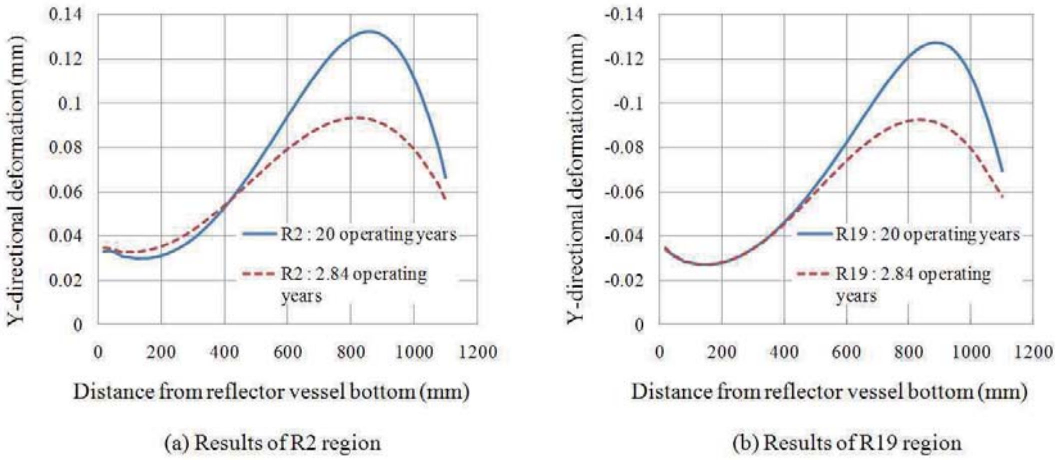

Fig.4 Analysis results

to as follows:

WR1 : 2.4E+25 /m^2/year
WR2 : 3.3E+25 /m^2/year
WR3 : 4.2E+25 /m^2/year
WR4 : 3.6E+25 /m^2/year

4. ANALYSIS RESULTS

The y-directional deformations caused by irradiation creep and growth of R2 and R19 region are depicted in Fig.4. Irradiation creep has little effect on the deformation because there is no pressure acting on the inner shell during operation. In this case, the deformation caused by irradiation growth becomes a dominant factor. However its amount is also small because irradiation growth strain is assumed to be saturated to 0.05% at the first operating year. Therefore it can be expected that irradiation creep and growth of the inner shell does not cause any problems on the safety of HANARO.

5. MEASUREMENT

The periodic measurement of the dimensional changes in the vertical straightness of the inner shell is considered as one of the in-service inspections. To confirm the safety of the reactor, the measurement of the straightness of the centerlines of the two worst sides of the inner shell was performed in Aug. 2004 [6]. At that time, the cumulative full power operating year of HANARO was 2.84 years. The results of the measurement are shown in Fig.5.

As-built dimensions were measured after fabrication of the inner shell. This measurement was performed under atmospheric pressure. However the measurement in 2004 was performed under the submerged condition. Therefore the measurement results include the deformation caused by the pressure due to the difference between the static head of light water and heavy water. To compensate this, the deformation due to the pressure difference is added to the original measured data and re-calculated results are depicted in Fig.5 as ‘As-built’.

However it is difficult to directly compare the as-built and the measured data of Fig.5 because

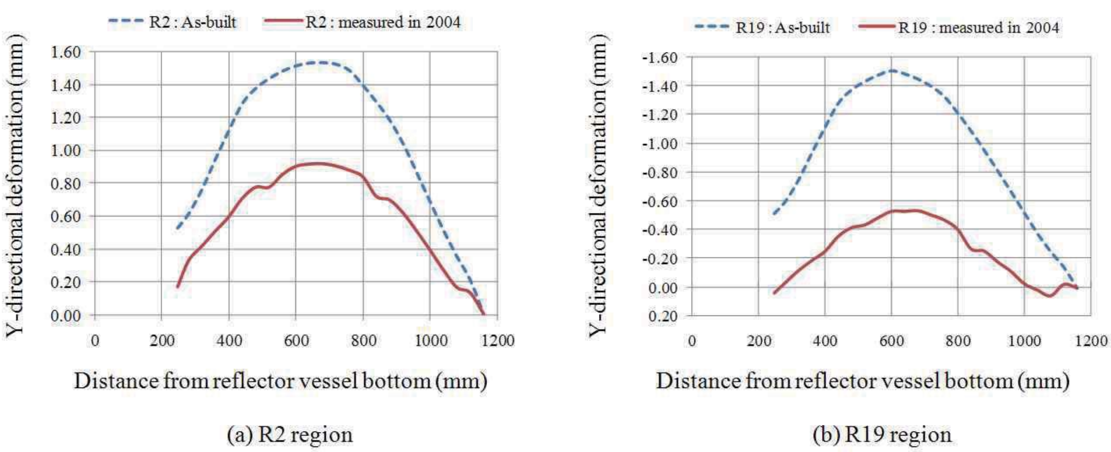

Fig.5 Measured dimensions

as-built dimensions of the inner shell were measured not after completion of the manufacturing but before welding to the upper/lower plates. The measurement of the deformation of the inner shell is planned in 2014. The actual deformation of the inner shell may be able to be estimated after comparing the data measured in 2004 and 2014.

6. CONCLUSIONS

The analysis to predict the deformation of the HANARO inner shell caused by irradiation creep and growth was performed. This study was conducted to generate the results which are more similar to actual situation by applying the enhanced analysis scheme and to predict the deformation for the remaining lifetime. However irradiation creep and growth causes very little deformation because there is no pressure acting on the inner shell during operation. According to the analysis results, the deformation due to irradiation does not raise any problems even at the end of HANARO's design lifetime. The results of this study will be consistently verified by periodic measurement of the inner shell.

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8.2 DEVELOPMENT OF IN-PILE INSTRUMENTS FOR FUEL AND MATERIAL IRRADIATION TESTS

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To get measurement data with high accuracy for fuel and material behavior studies in irradiation tests, two kinds of measuring equipments have been developed; these are the Electrochemical Corrosion Potential (ECP) sensor and the Linear Voltage Differential Transformer (LVDT) type gas pressure gauge. The ECP sensor has been developed to determine the corrosive potential under high temperature and high pressure water conditions. The structure of the joining parts was optimized to avoid stress concentration. The ECP sensor showed enough performance at 288°C and at 9MPa conditions. The LVDT type rod inner gas pressure gauge has been developed to measure gas pressure in a fuel element during neutron irradiation. To perform stable measurements with high accuracy under high temperature, high pressure and high dosed environment, the coil material of LVDT was changed to MI cable. As a result of this development, the LVDT type gas pressure gauge showed high accuracy within 1.8% of a full scale, and good stability.

Keywords: ECP sensor, LVDT gas pressure gauge, in-situ experiment, IASCC, JMTR.

1. INTRODUCTION

In order to maintain and enhance safety of light water reactors (LWRs) in long term and upgraded operations, proper understanding of irradiation behavior of fuels and materials is essentially important. Japan Materials Testing Reactor (JMTR) of Japan Atomic Energy Agency (JAEA) has been used for fuel and materials irradiation studies.

The renewed JMTR will be started from the later half of JFY2011, and it is expected to contribute to various fields.

High quality in-pile instrumentation is very important to study the irradiation behavior of high burn up fuels and aged materials. To get measurement data with high accuracy for fuel and material behavior studies in irradiation tests, two kinds of measuring equipments have been developed; these are the Electrochemical Corrosion Potential (ECP) sensor and the Linear Voltage Differential Transformer (LVDT) type gas pressure gauge.

2. Development of ECP sensor

It is planned to perform in-pile IASCC growth tests of irradiated stainless steels by using the JMTR. The objectives of this study are to understand the difference between in-pile and out-of-pile IASCC growth behavior.

The existence of chemical species formed by radiolysis of neutron and gamma ray irradiation causes the difference of the water chemical environment between primary reactor water and supplied water. This difference affects IASCC of core component materials.

For the in-pile IASCC behavior, the crack growth tests of the 0.5T-CT specimen loaded up to ~7 kN (corresponding to K~30 MPa√m) which is monitored by the Potential Drop Method (PDM) in the irradiation capsule of the JMTR will be performed to confirm the effectiveness of mitigation by lowering electrochemical corrosion potential (ECP) . Fig.1 shows a schematic drawing of the in-pile test.

For these tests, the development of ECP sensor is required.

But the techniques to measure ECP in a reactor were not established at the JMTR. The ECP sensor must be durable enough to stand against high temperature and high pressure reactor water condition. On the other hand, the conventional ECP sensor which is used in JAEA has a problem on durability under neutron irradiation in reactor conditions.

In the conventional sensor, a crack was caused in the stabilized zirconia in the juncture part of the stabilized zirconia and the metal when the neutron irradiation test was performed. And then, the function as the sensor was lost. Fig.2 shows the crack in the juncture neighborhood.

One of the factors which induce crack is the residual stress generated at the joint of stabilized zirconia and metal which have different thermal expansion coefficient.

Thus, structure was optimized to avoid stress concentration and joining condition such as brazing was optimized for connection part. To optimize the dimensions, figure, and brazing method between zirconia and metal sleeve, structure analysis is performed.

Fig 3 shows thermal expansion coefficient of structure materials.

The residual stress distribution at brazing juncture neighborhood was evaluated by the parameters of dimension using the finite elements method for brazing juncture neighborhood (ANSYS Workbench 11.OSP1). Fig. 4 shows example of stress distribution using the finite element analysis.

Various figure were evaluated and Type A with taper (x=5.0,y=0.50) was adopted.

Finally, ZrO_2 membrane type ECP sensors (reference electrode) with Fe/Fe_3O_4 electrode and brazing sealing were produced. Performance tests were performed in high temperature (230-288℃) and high pressure (9MPa) out-pile water loop.

Fig 5 shows SHE value of ECP sensors as a function of temperature. The measured potential value agreed with standard values.

The performance tests of more than 3000hours were performed for the ECP sensors and integrity was confirmed.

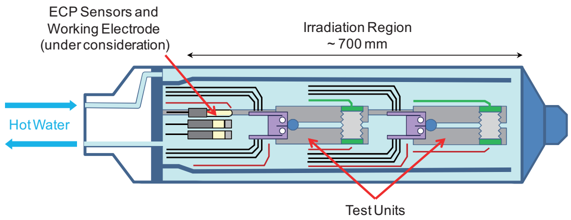


Fig.1 Schematic drawing of the in-pile test

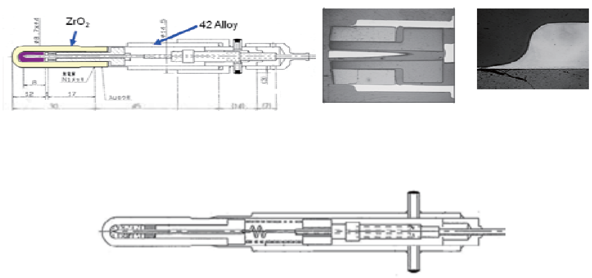


Fig 2 Observation of the crack in conventional ECP sensor

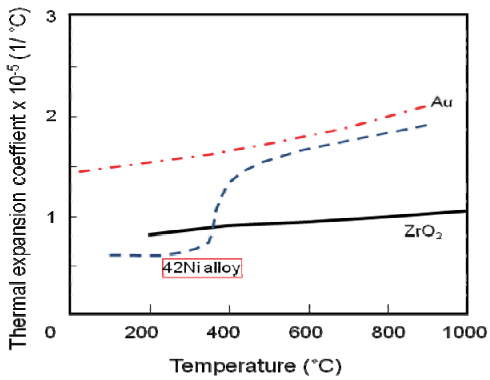
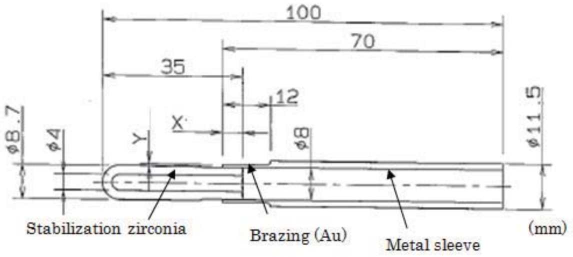
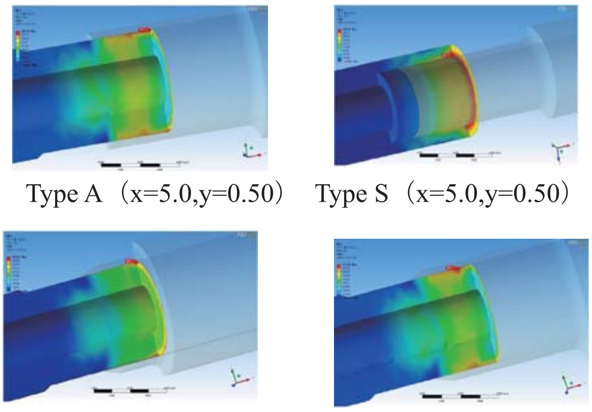


Fig. 3 thermal expansion coefficient of structure materials as a function of temperature





Type A (x=5.0,y=0.50) Type S (x=5.0,y=0.50)

Taper type Type A with taper (x=5.0,y=0.50)

Fig. 4 Stress distribution of brazing juncture neighborhood

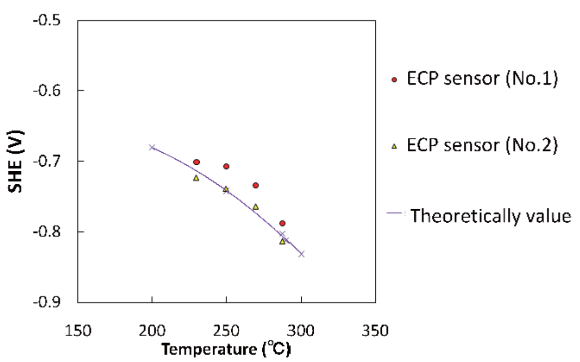


Fig.5 SHE value of ECP sensors as a function of temperature.

3. Development of LVDT type gas pressure gauge

The integrity and evaluation data of irradiation behavior of fuels and materials is important for the higher performance operation of LWRs. To obtain the performance data of current and coming fuels with new cladding alloys and modified pellets, the power ramp tests of the fuels have been planned at the new JMTR.

The fuel integrity under abnormal power transient conditions will be investigated by using three types of capsules. The three types of capsules are as follows: fuel-elongation measurement capsules and fuel center temperature-rod inner pressure measurement capsules. In addition to these capsules, heater capsules for power calibration were designed [6].

The rod inner pressure gauges for 5 and 10 MPa have been developed to deal with the increase of fission gas release in high burn-up LWR fuels, and the trial fabrication tests were carried out. The structure of the fabricated rod inner pressure gauges is shown in Fig. 6. The rod inner pressure gauges are composed of bellows, springs and linear variable differential transformers (LVDT). In trial fabrication tests, the coil of the LVDT was changed from a ceramic covered wire to a reliable MI cable. Especially, two springs were used in the rod inner pressure gauge for 10 MPa because the spring constant is required twice as much as the spring for 5 MPa.

Performance tests of the rod inner pressure gauges were carried out [5]. The structure of the performance test system with the rod inner pressure gauges is shown in Fig. 7. The output characteristics of the Rod inner pressure gauge were measured during increase and decrease of pressure by a step of 1 MPa at room temperature and 300˚C. The results are shown in Fig. 8. The relationship between the pressure and the axial displacement of bellows was linear and there was no zero shifts. The measurement errors were ±1.8% of a full scale.

Irradiation tests of differential transformers for rod inner pressure will be conducted at an Institute of Nuclear Physics (INP), this project started from 21 May, 2010 on the representatives of the National Nuclear Center of Republic of Kazakhstan (NNC-RK) and the JAEA as the ISTC K-1806 Partner Project. Expected duration of reactor irradiation is not less than 100 days. The temperature of differential transformers under study is controlled by in-built electrical heater between 200 and 400˚C. The irradiation devices such as an inner capsule and an outer channel for studies of differential transformers were prepared by JAEA. Two transformers were located inside the inner capsule. Each transformer was equipped with an electrical heater, thermocouple and fluence monitor. Differential transformers delivered from JAEA to INP, and will be subjected to the systematic irradiation tests of durability. This irradiation tests will be started from March, 2011 at WWR-K reactor of INP NNC-RK (Fig.9).

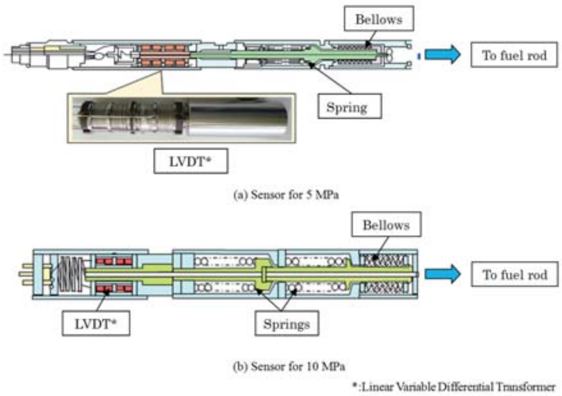


Fig.6 Structure of rod inner pressure gauges..

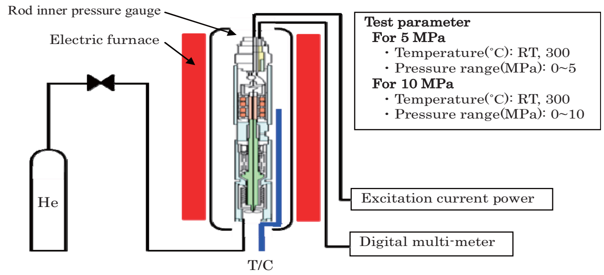


Fig.7 Structure of performance test system with rod inner pressure gauge.

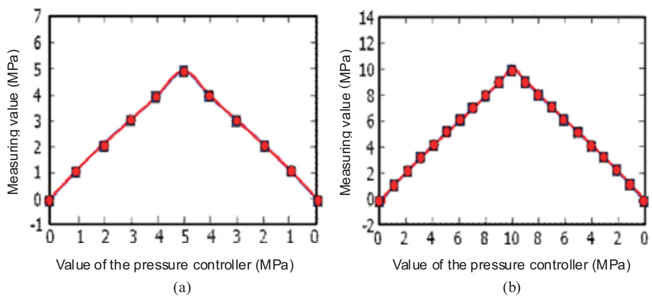


Fig.8 Results of performance test with rod inner pressure gauge:(a) 5 MPa, (b) 10 MPa [5].

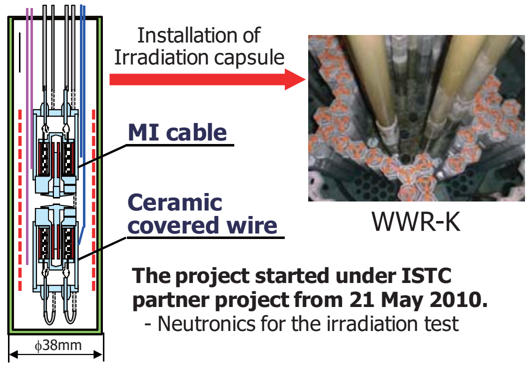


Fig.9 Resolution on Radiation effect of LVDTs with ceramic covered wire or MI cable in WWR-K

4. CONCLUSIONS

ZrO_2 membrane type ECP sensors (reference electrode) with Fe/Fe_3O_4 electrode and brazing sealing were produced. The structure of the joining parts was optimized to avoid stress concentration. The ECP sensor showed enough performance at 288°C and at 9MPa conditions.

The performance tests of more than 3000hours were performed for the ECP sensors and integrity was confirmed.

The irradiation capsule for the Linear Variable Differential Transformer (LVDT), which is irradiated in the WWR-K reactor at Republic of Kazakhstan, was designed and fabricated for the development of the rod inner pressure gauge.

In future, the irradiation tests of these developed instruments will be carried out for the international standard of the Material Testing Reactors using JMTR after the JMTR reoperation and using other reactors.

5. Acknowledgements

The authors are grateful to Drs Y. Chimi, J. Nakano and T/ Tsukada for helpful discution.

A part of this study is being conducted under a contract with the Nuclear Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI).

8.3 Thermal Analysis on the Specimens for Low Irradiation Temperature below 100^{o}C in the HANARO

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A capsule has been used for an irradiation test of various nuclear materials in the research reactor, HANARO. As a part of the research reactor development project with a plate type fuel, the irradiation tests of beryllium, zircaloy-4 and graphite materials using the capsule will be carried out to obtain the mechanical characteristics at low temperatures below 100 ^{o}C with 30 MW reactor power. In this study, in order to obtain the preliminary design data of the capsule with various specimens and the temperature of specimens, a thermal analysis is performed by using an ANSYS program. The finite element models for the cross section of the capsule containing the specimen are generated, and the temperatures are evaluated. The analysis results show that most specimens meet the irradiation target temperature. However, some canned graphite specimens have a slightly high temperature, and the gap size has a significant effect on the specimen temperature. Based on those results a detailed design and analysis of the capsule will be completed this year.

Keywords: HANARO, capsule, thermal analysis, irradiation, canning tube, specimen, graphite, beryllium.

1. INTRODUCTION

The High-flux Advanced Neutron Application Reactor (HANARO) is a multi-purposed research reactor of an open-tank-in pool type located at KAERI in Korea. There are seven vertical test holes which include three flux traps; CT, IR1, IR2 (hexagonal type) in the core and four vertical holes; OR (cylindrical type) in the outer core for nuclear fuels/materials irradiation testing. Among the irradiation facilities at HANARO, the instrumented and non-instrumented capsules have been in development since 1995 and are being utilized for new alloy and fuel developments and for estimation of the lifetime of nuclear power plants. The instrumented capsule for material irradiation tests plays an especially important role in the integrity evaluation of reactor core materials and the development of new materials through precise irradiation tests of specimens such as the reactor pressure vessel, reactor core, pressure tube and fuel cladding materials [1].

As a part of the research reactor development project with a plate type fuel, the capsule for irradiation tests of major reactor materials (graphite, beryllium, zircaloy-4) is under development at KAERI. Those materials in the new reactor will be used in the coolant water of about 40 ^{o}C. Therefore, the irradiation test needs to be carried out at a similar environment to the normal operation of the reactor, and will be performed to obtain the mechanical characteristics such as an irradiation growth, hardness, swelling and tensile strength at the low temperature below 100 ^{o}C up to a fast neutron fluence of 2.16×10^{21}(n/cm^2) (E>1.0MeV). For the irradiation test at a low temperature, it is necessary to design an applicable capsule and estimate the temperatures of specimens.

To obtain the preliminary design data of the capsule with various specimens a thermal analysis is performed by using a finite element (FE) analysis program, ANSYS [2]. The 2-dimensional and axi-symmetrical models for the cross section of the capsule containing specimens are generated, and the maximum temperature and its distribution are evaluated. The effects of the thermal conductivity and the gap size for a canned graphite specimen on the temperature are also discussed.

2. CAPSULE MODEL

The capsule under development has the same outward shape as a typical capsule which consists of the main body, the protection and guide tube etc. [3,4]. The main body is a major part of the capsule in which specimens, supporting structures and various components are installed, and include the external tube of a cylindrical shell with 58 mm in external diameter, 2.0 mm in thickness and 870 mm in length. The capsule's main body has five stages of which contain canned specimens in the axial direction. Fig. 1 shows the geometrical shape of the main body and a typical cross section for an arbitrary stage including the specimens with three different shapes (Type1; Circular, Type2; Rectangular, Type3; Square).

The specimens are basically canned by a tube of 1mm in thickness made of stainless steel. Also, there are two kinds of canning tubes; a closed tube for the graphite circular specimens (Type1 #1&2) and an opened tube with side slots to contact between the specimen and the cooling water (Type1 #5, Type2&3). All canning tubes are fixed by the supporting structure in both ends between the stages. They are also supported by the capsule external tube. The surfaces of canning tubes and the external tube come in contact with the cooling water in the test hole during the irradiation tests. Therefore three models according to the shape of specimens are considered in the analysis.

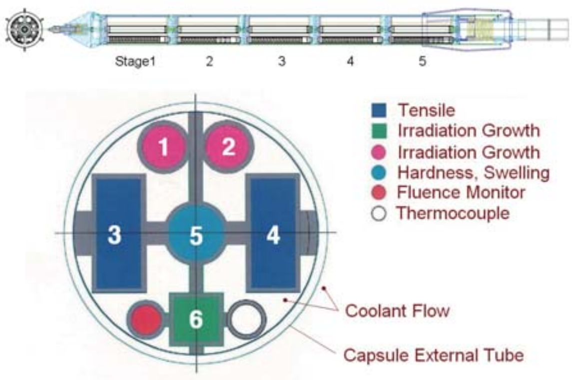


Fig. 1. Cross section view of the capsule.

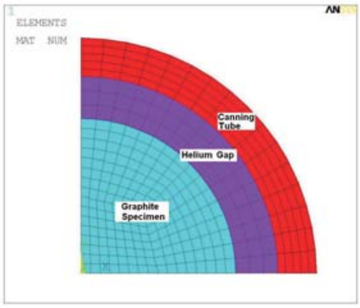
3. THERMAL ANALYSIS

Three FE models in this study are generated using a FE analysis program, ANSYS. One is the 2-dimensional model of a quarter section for all specimens. Another is the axi-symmetric model for circular specimens (Type1 #1,2,5). The other is the 3-dimensional model for rectangular and square specimens (Type2~3 #3,4,6). Fig. 2 shows a typical 2D and axi-symmetric models generated for the canned graphite specimen (Type1 #1&2). In this case the gap size between the specimen and the canning tube is 1.1 mm considering the irradiation growth.

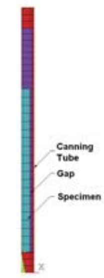
In the thermal analysis the temperature (40 ^oC) of the cooling water and a heat transfer coefficient (h=30.3×10^3 W/m^2 ^oC) at the outer surface of the canning tube are applied as boundary conditions. The heat generation densities of materials in the CT hole of the reactor in-core are used as an input force. Table 1 presents the mechanical properties of materials used in the thermal analysis.

Table 1. Mechanical properties used in the analysis.

Material	Density (g/cm^3)	Thermal Expansion Coefficient (x10^{-6}/^oC)	Thermal Conductivity (W/m ^oC)
Graphite	1.77	3.8	120
Beryllium	1.85	11.3	200
Zircaloy-4	6.47	6.0	11.4



(a) 2D model



(b) Axi-symm. model

Fig. 2. Typical models of canned graphite specimen.

4. RESULTS AND DISCUSSION

Table 2 presents the calculated temperatures of the specimens at a 30 MW HANARO power. They are distributed in a range from 40 ^oC to 153.6 ^oC with different values according to the stages and materials. Depending on the heat generation rates of materials in the reactor core, the temperatures in the middle region (stage3) are high, and those at stage5 are relatively low. The results using three different FE models show nearly the same temperatures, except for the canned graphite specimens having the difference of 12 ^oC between two FE models.

For all zircaloy-4 and beryllium specimens the temperatures are below 61 ^oC regardless of the shape of the specimen and the location in the axial direction.

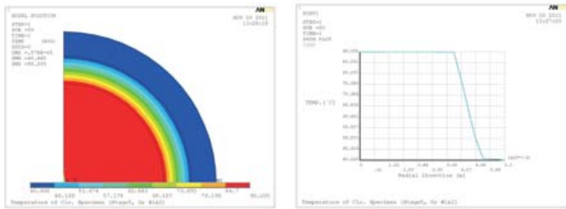
The reason is that the specimens are continuously cooled down by coolant flow through side slots of the canning tube. The zircaloy-4 specimens have higher temperatures than those of the beryllium due to the high density.

On the other hand, in the case of canned graphite specimens (Type1), the temperatures are higher than those of the other materials due to the closed canning of the specimen. The 2D analysis results have slightly higher temperatures than the axi-symmetric model. Also the temperature at stage5 is satisfied with the irradiation target temperature, but stage1&3 have a high temperature.

Table 2. Temperatures of the specimens.

Stage	Specimen No.	Shape	Mat.	2D	Axi-sym. or 3D	
				Max.	Max.	Min.
5 (Top)	1&2	Cir	Gr	90.2	85.0	71.5
	3&4	Rect	Zr	58.4	58.4	40
	5-1	Cir	Be	40.5	41.1	40
	5-2	Cir	Zr	47.9	48.0	40
	6	Squ	Zr	49.8	49.8	40
3 (Mid)	1&2	Cir	Gr	153.6	142	112
	3&4	Rect	Be	42.2	42.2	40
	5-1	Cir	Be	41.1	42.7	40
	5-2	Cir	Zr	60.8	60.9	40
	6	Squ	Be	41.2	41.2	40
1 (Bot)	1&2	Cir	Gr	122.1	114.4	92.6
	3&4	Rect	Be	41.6	41.6	40
	5-1	Cir	Be	40.6	41.7	40
	5-2	Cir	Zr	53.6	53.6	40
	6	Squ	Be	40.9	40.9	40

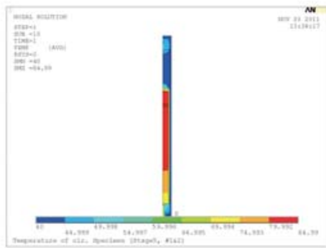
For the five stages of the capsule, the temperature distributions have a similar trend because of the same arrangement of the components at each stage. Fig. 3(a) shows the temperature distribution for the canned graphite specimen at stage5 using the 2D model, and Fig. 3(b) presents the profile in the radial direction at $\theta = 0$ position. The maximum temperature at the specimen is 90.2 ^{o}C, and rapidly decreased at the gap between the specimen and the canning tube. The heat transfer occurs from the inside of the capsule to the outside due to the low temperature of cooling water.



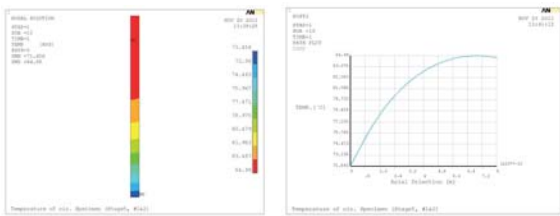
(a) Distribution (b) Profile

Fig. 3. Temperature of the canned graphite specimen at stage5 using the 2D model.

Fig. 4 shows the results of the canned graphite specimen using an axi-symmetric model. The closed canning tube has a similar temperature with that of the cooling water. Fig. 4(b) shows the detailed temperature of the specimen. The temperature in the radial direction is uniformly distributed without the gradient similar to the 2D model. However, it shows the temperature gradient in the axial direction as shown in the figure. In this case the temperature difference between the top and the bottom along the center of the specimen in the axial direction is about 13 ^{o}C. The maximum value is 85 ^{o}C at the top of the specimen center, and the minimum is 71.5 ^{o}C at the bottom, which can increase the heat transfer due to the direct contact between the specimen and the canning tube.



(a) Temperature distribution



(b) Specimen temperature and axial profile

Fig. 4. Temperature of the canned graphite specimen at stage5 using the axi-symmetric model.

In the case of the canned graphite specimens as shown in Figs. 3 and 4 the gap between the specimen and the canning tube has a larger effect than other parameters on the temperature because the helium with low thermal conductivity plays a role of an adiabatic material. Table 3 presents the effect of gap sizes for the canned graphite specimens using a 2D model. When the gap is reduced by half the size, the maximum temperatures of the specimen are decreased by 30~35 percents.

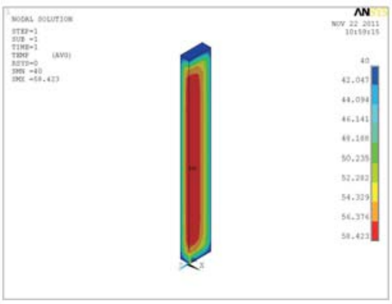
For the present capsule with canned graphite specimens, the temperatures at stage1 and 3 are the higher than the irradiation target temperature. In this case if the gap size is reasonably reduced, the target temperatures will be satisfied. However, considering the irradiation growth of graphite specimens the gap size of 1.1 mm was determined. Therefore the irradiation test for the graphite specimens will be carried out at a little higher temperature.

Table 3. Effect of gap sizes for canned graphite specimen.

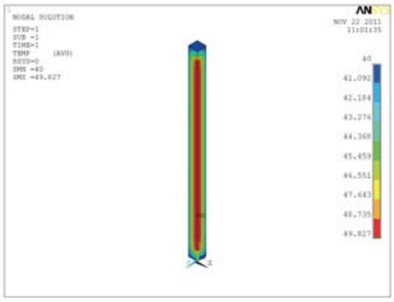
	Stage	Graphite (Specimen #1&2)		
Gap size (mm)		1.1	0.55	0.27
Temp. (^{o}C)	5	90.2	68.0	55.2
	3	153.6	103.7	75.4
	1	122.1	86.3	65.5

The Type2&3 specimens are canned by an opened canning tube. Therefore, zircaloy-4 and beryllium specimens at each stage come in contact with the cooling water, and the analysis results show the similar or a slightly high temperature with that of the water. Fig. 5 shows the temperature distribution of the zircaloy-4 specimens with rectangular and square shapes at stage5 using the 3D model. The result shows exactly the same as the 2D model. The maximum temperature is 58.4 ^{o}C and varies in the range of 40~58.4 ^{o}C.

The temperature of the beryllium specimens is not shown here because they have the same trends with those of the zircaloy-4 specimen, except for the slightly low temperatures.



(a) Type2 (rectangular)



(b) Type3 (square)

Fig. 5. Temperature of zircaloy specimens at stage5 using the 3D model.

5. CONCLUSIONS

1) The analysis results with FE models show in good agreement with one another, except for the canned graphite specimens having a difference of 12 ^{o}C between 2D and axi-symmetric models.
2) The temperatures of beryllium and zircaloy-4 specimens at all stages are 40~60.9 ^{o}C. The zircaloy-4's temperatures are higher than those of the beryllium specimens.
3) In the case of the canned graphite specimens, the analysis results at stage1 and 3 show a slightly high temperature above 100 ^{o}C. But the capsule will be fabricated on the basis of the present design considering the irradiation growth of the graphite.
4) The helium gap for the canned specimens has significant effect on the temperature. When the gap size is half, the temperatures are decreased about 30 percents.
5) The detailed design and analysis of the capsule based on those results will be completed by this year.

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8.4 DEVELOPMENT OF REACTOR WATER LEVEL SENSOR FOR EXTREME CONDITIONS

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In the Fukushima accident, measurement failure of water level was one of the most important factors which caused serious situation. The differential pressure type water level indicators are widely used in various place of nuclear power plant but after the accident of TMI-2, the need of other reliable method has been required.

The BICOTH type and the TRICOTH type water level indicator for light water power reactors had been developed for in-pile water level indicator but currently those are not adopted to nuclear power plant. In this study, the development of new type water level indicator composed of thermocouple and heater is described. Demonstration test and characteristic evaluation of the water level indicator were performed and we had obtained satisfactory results.

Keywords: Temperature measurement, water level, Fukushima, spent fuel stack pool.

1. INTRODUCTION

The differential pressure type water level indicators are widely used in various place of nuclear plant but after the accident of TMI-2, other reliable method has been demanded.

JAEA and Sukegawa Electric Co., Ltd. developed BICOTH type [1],[2] and TRICOTH type [3] water level indicator which are operated by heaters and differential thermocouple train. Those were adopted at Dodewaard BWR in Netherlands and it shows those are possible to use as water level indicator in reactor. But those have not been used in other reactors by some reasons such as port modification and additional electrical works.

The reactor accident in Fukushima was caused by the gigantic tsunami. In this disaster, measurement failure of water level of spent fuel stack pool was one of the most important factors which caused this serious situation.

Therefore, we developed a new type water level indicator which composed of thermocouple and heater. This new water level indicator does not use conventional differential thermocouple train and it has simple structure and it works by low voltage.

The demonstration test and characteristic evaluation of the water level indicator were performed and evaluated.

2. R＆D POLICY

The measurement failure of water level of spent fuel stack pool was one of the most important factors which caused this serious situation, and the development of the water level sensor with high radiation resistance, reliability of measurement, and simple structure for rapid production was required. To solve those, following factors were considered for this development.

Radiation resistance: To achieve radiation resistance, inorganic materials such as metal and ceramics were chosen for structure materials.

Principle: According to the experience of BICOTH type, the ability to detect water level needs temperature difference of about 200 ℃ in atmospheric condition. To measure water level accurately, power density of 0.5W/cm^2 is required.

Power source: Assuming the situation without stable power supply, we designed heater to work under 6V (AC, DC).

Installability: Reactor buildings were exploded in the Fukushima accident. The indicator may be required to install into the gaps of debris. Therefore, the sensor lead was designed as 4mm in diameter.

It was possible to detect the water level by the temperature deference between in water and in air if there is 20℃ temperature difference.

Based on those, the structure of new water level sensor was designed as shown in Fig. 1 for new water level sensor. MI-cable type water level sensor adapts partial heating and thermocouple.

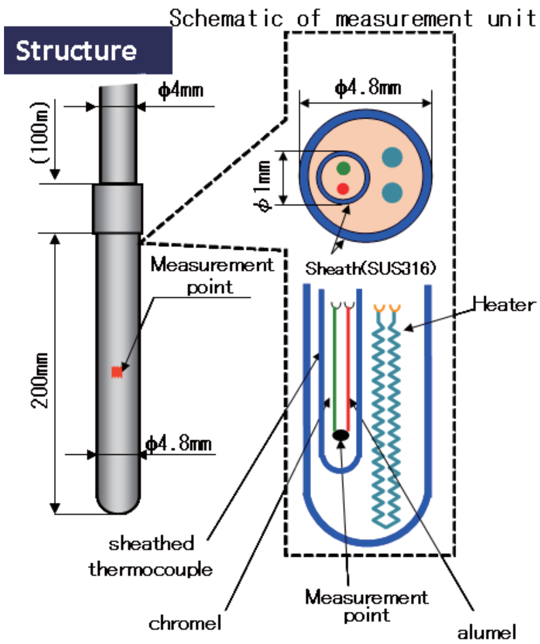

Fig.1 Structure of water level sensor

3. EXPERIMENTAL

Performance test was made with length of 200 mm sensor and 100m MI lead cable. Test apparatus is depicted in Fig. 2.

The sensor unit was installed into water. Temperature of water was controlled to 27-90℃ to simulate spent fuel stack pool. Some additional thermocouples were attached to the surface of water level sensor to measure the temperature distribution along the heated thermocouples position. At various water levels, the output of the thermocouples was measured.

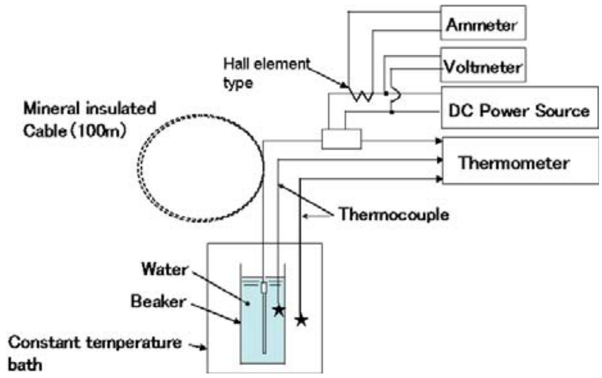

Fig. 2 Test apparatus

4. RESULT AND DISCUSSION

At the water surface level, temperature difference between in water and in air was observed. Fig. 3 shows the results of the experiments. When the measurement point is above 20mm from water level, the temperature difference of 25℃ was observed at 27℃water and the temperature difference of 50℃ was observed at 90℃-water.

According to our experimental data, it is possible to recognize that the heated thermocouples are in air or in water if there is more than 20℃ temperature difference between the two cases. Thus, this water level sensor detects water level with the accuracy of about ±20mm.

Cable length is required minimum 100m distance to the measuring point at the spent fuel pool at Fukushima power plant.

Fig.4 shows actual product with 100m sheathed cable. The manufacturing ability of such a water level sensor with long MI cables was demonstrated.

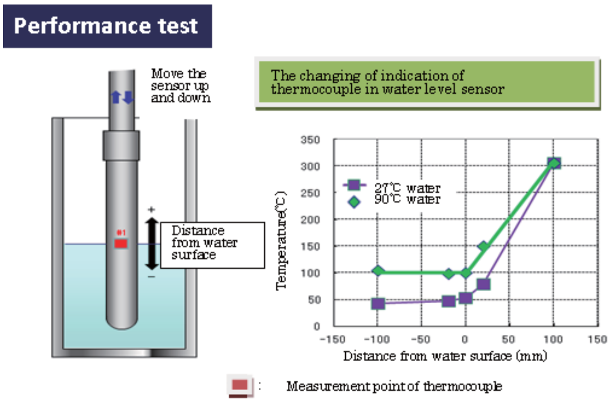

Fig.3 Experimental data

Fig. 4 Water level sensor with 100m length MI cable

5. CONCLUSION

Experience of BICOTH type and TRICOTH type water level sensor helped us to develop new type water level indicator composed of thermocouple and heater.

This sensor has more simple structure and the accuracy of measurement is about ±20mm.

Inorganic material such as metal and ceramics were chosen for structure material and it has high radiation resistance.

Thus, the developed water level gauge has the potential to measure spent fuel pool water level from remote place in extreme conditions.

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9. New Generation MTR

9.1 INVAP NUCLEAR ENERGY DIVISION MAIN ACTIVITIES 2011

Ruben Mazzi

INVAP Australia

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During recent years INVAP HQ has developed own facilities at San Carlos de Bariloche, approaching today a complex of 28.000 sqm intended to support the fast growing demand of the youngest areas: Aerospace, Government and Defence (Radars, Digital TV systems). Jointly with Nuclear, Industrial, Medical and Administrative, INVAP accounts for a total of 900 people 85% being high qualified technical staff distributed in Argentina and four branches worldwide.

INVAP Nuclear Energy Division (NED) is the pioneer activity, at the company founded 35 years ago, and accounts for 200 people with vast experience in project management and nuclear engineering design (i.e. neutronics, process, mechanical, I&C, materials, QA, etc.).

INVAP NED extends the long tradition initiated 60 years ago by the National Atomic Energy Commission (CNEA) with great success as demonstrated by exporting 13 nuclear facilities worldwide in the last 30 years, being the design and construction of the OPAL reactor in Australia, the most relevant to date.

Among the main activities carried out at the INVAP NED during 2011, are:

- Atucha II (750MWe) and CAREM (27MWe) NPP, construction and licensing projects.
- Commissioning of the Radioisiotope Production Facility in Egypt for AEA,
- Successful test recently achieved for producing Mo-99 with the Aqueous Homogenous Reactor
- INVAP Australia including supporting activities to OPAL with the awesome record of >75% availability in the last two years.
- New research reactor developments: RA-10, RMB, DIPR.

INVAP

The Company

- Born: September 1st, 1976
- State owned - Province of Rio Negro
- Annual Sales: 200 Million USD
- 900 employees (85% Professionals and Technicians)
- Headquarters: San Carlos de Bariloche (RN) - Argentina
- Subsidiary in USA.
- Branches in Australia, Brazil, Egypt and Venezuela
- In close cooperation with Atomic Energy Commission (CNEA) and Argentine Space Agencies (CONAE & ARSAT)

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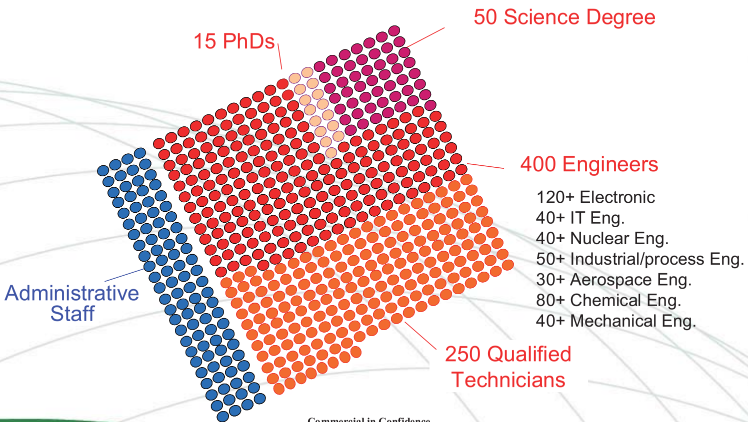
Human Resources

❖ Staff of 900 people / 80% R&D-trained



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New Headquarters

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INVAP Main Technological Areas

- **Nuclear Division:**
  - Research reactors and associated facilities
  - Services to Nuclear Power Plants
  - Engineering services to nuclear industry
  - Waste management plants
  - Independent Advisor to Regulatory Bodies
- **Aerospace Division**
  - Satellites, Ground station & Space Payloads
- **Government and Defence Division**
  - Primary and secondary radars
  - Digital TV Systems (ARG)
  - Nuclear medicine equipment
- **Industrial Applications Division**

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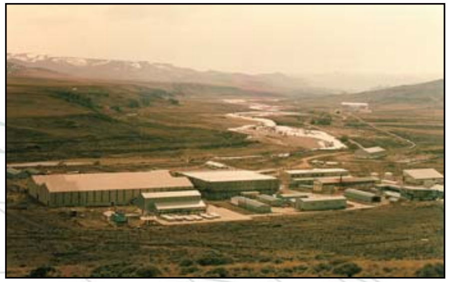
Nuclear Division: the beginning

The first reactor: RA6
1978-1982



The Pilcaniyeu Complex:
Mastering the process of U enrichment
1978-1983

GASSEOUS DIFUSION

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Milestones for Research Reactors

• 1958	RA-1: First Research Reactor
• 1965	RA-0: Critical facility
• 1966	RA-2: Critical facility
• 1968	RA-3: Radioisotope production reactor
• 1972	RA-2: Homogenuos critical facility
• 1978	RP-0: Critical facility in Peru
• 1982	RA-6: Research and training reactor
• 1988	RP-10: Multipurpose reactor in Peru
• 1989	NUR: Research and training in Argelia
• 1997	RA-8: Carem critical facility
• 1998	ETRR-2: Multipurpose reactor in Egypt
• 2006	OPAL: Beam and RP reactor in Australia

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Milestones for RR related projects (only exports)

• 1993	Refurbishment and conversion from HEU to LEU of Teheran reactor in Iran
• 1994	Radioisotope production plant in Cuba
• 1998	Fuel element manufacturing plant in Egypt
• 2000	Fuel element manufacturing plant in Algeria
• 2009	Mo-99 production plant in Australia
• 2009	Refurbishment of I&C for Pitesti reactor in Rumania
• 2011	Radioisotope production plant in Egypt
• 2011	Refurbishment of I&C for Tajoura reactor in Libya

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Nuclear Division

Main Activities 2011

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Our (INVAP/CNEA) export record in the RR field:

All LEU based

5 Reactors
3 Radioisotope Production Plants
2 Fuel Manufacturing Plants
3 Research Reactor Refurbishments

13 facilities in the past 30 years

INVAP

Nuclear Power Plants

Engineering Services
•**CNA II (PHWR)**
Supporting Technical activities for NASA (Construction and Commissioning Phase)

•**CAREM Reactor**
Process & Mechanical Design (Basic Engineering for CNEA)

•**ARN**:
Technical support for licensing of During Commissioning of CNAII

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INVAP Australia

Main activities

- Technical Support: Opal Operations & Maintenance
- Construction & commissioning of the HWUS
- Provision of Mo-99 targets and key spares to OPAL
- New Projects with ANSTO
- Representing INVAP in Australia & Asia-Pacific Region

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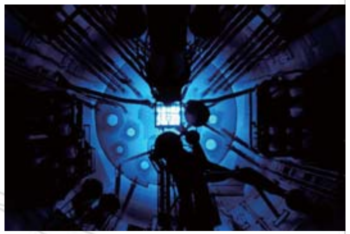
OPAL

5 years after First Full Power !!



Low Radiation Levels at working places + Improved design

Occupational doses (HIFAR) **reduced by factor of 10**

Accumulated Use: ≈1000 FPD **(2.6 FPY)**

Yearly Use : +280 FPD/Y (2010/2011) **Availability over 75%**

Measured neutron fluxes exceed contractual values

Radiation Facilities (Mo-99, NTD Si, RI) increasing productivity

Neutron Guides widely used by Local and International Scientist

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Radioisotope Production Facility (RPF)

A turn-key facility delivered for AEA (Egypt) for production of radioisotopes for Medical, Industrial and Research activities

Main Features:

- Operate with LEU targets irradiated at the ETRR-2 Research Reactor
- Radionuclides:
  - a) Chromium -51
  - b) Iodine -131
  - c) Iodine -125
  - d) Iridium -192
  - e) Molybdenum -99 from fission
  - f) Loading of Molybdenum-99/Technetium-99m generators
- Multipurpose cell for labeling of compounds and/or
- Implementation of new processes.

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Radioisotope Production Facility AEA (Egypt)

Commissioning Test Ended: Nov 2011:

- 4 full Batches
- 3000 Ci Mo-99 produced at RPF Facilities with ETRR-2

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Aqueous Homogeneous Reactor

Medical Isotope Production System*

Basic Principle: Production of Mo-99 by neutron irradiation of an aqueous solution of Uranyl Nitrate (LEU)

Client: Babock & Wilckox Co. (USA)

MIPS Goal: 50% of the USA's market by 2016

Phase: P-o-P (Proof of Principle)

Facilities: Mini-loop (1:400) and Hot Cell at RA-6 (Bariloche)

Results*

✓Several batches of Curies of Mo-99 produced fully complilant with radiopharmaceutical grade specifications

✓Tc-99 Generator also loaded with excellent results !!!

* Presented by B&W, Santa Fe, New Mexico 4-7 Dec 2011)

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Mini-Loop Test & Hot Cell Facilities

(Bariloche at Argentina)

Shielded Cell

Figure 1: Capsule

Figure 2: Installation

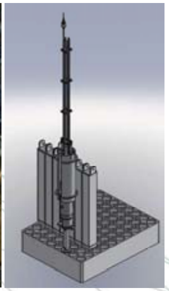


Figure 3: Capsule in grid

New Reactors Designs

RA-10: Argentina CNEA

RMB: Multipurpose Brazilian Reactor

(Preliminary Engineering)

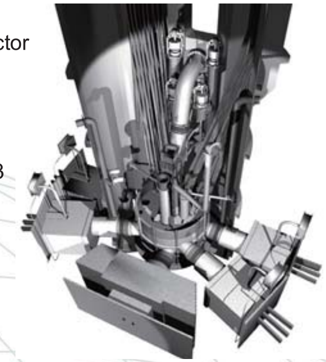
DIPR: Dedicated Isotope Production reactor South Africa

(Reactor Proposal)

RA-10

Main Features

- **Type**:
  Open pool multipurpose research reactor – Reference Reactor OPAL
- **Core**:
  Compact, Variable configuration
  Box 5x5. FE with 6 in-core irrad. Pos.
- **Fuel**: MTR fuel LEU type, 4,8 gU/cm3
- **Control Rods**: #6 Blade type (Hf)
- **Coolant/Moderator:** H_2O
- **Reflector** D_2O
- **Core thermal Power:** 30 MW
- **Neutron Flux** > $3.0 \ 10^{14}$ $n.cm^{-2} \ s^{-1}$



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Countries in which INVAP delivered work in nuclear

Algeria	Lybia
Australia	Mexico
Brazil	Netherlands
Canada	Peru
China	Romania
Egypt	Saudi Arabia
Greece	USA
India	Venezuela
Iran	

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International partnerships, agreements, cooperation (nuclear)

Agreements with organizations from: Australia, Canada, Egypt, Spain, France, Hungary, Russia, USA.

Suppliers from: Australia, China, Francia, Germany, Hungary, Japan, Russia, Singapur, United Kingdom, USA.

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Final remarks

- In contrast with the adverse Impact in the market of nuclear power from Fukushima aftermath (mainly in Europe and some countries in Asia) it is observed a high demand of NEW Research Reactors
- They will be aimed at Basic and Applied Research, Neutron Beams w/CNS, Silicon and Radioisotopes for both Industrial and Medical applications, Nuclear and Non Nuclear Materials Testing and Development and Training
- NEW design will be more likely based on Open Pool, Compact Core High Flux, Fuel Plate Type FA.
- Due to the high demand on radiopharmaceuticals some reactors will be oriented to produce to Mo-99 & other RI

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Thank You

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9.2 MYRRHA
An innovative and unique research facility

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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 Accelerator Driven Systems (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 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 due to the little 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 achieve a high efficient transmutation. Thus, the sub-criticality is not a virtue but rather a necessity for an efficient and economical burning of the MA. 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).

Although carrying out the MYRRHA project will lead to the demonstration of the efficient and safe transmutation of MA in ADS systems as the ultimate goal the implementation of such a project will in addition trigger the development of various innovative technologies and techniques that are of interest for various nuclear fission and fusion applications.

Since March 2010, MYRRHA received the financial support from the Belgian government for 40% share of the 960 M€ investment. A 60 M€ funding has been received for the first stage (2010-2014) additional to the SCK•CEN regular investment in the MYRRHA project.

In this paper the present status of the project and its capabilities as an irradiation facility as well as the ISOL@MYRRHA (Radioactive Ion Beam (RIB) facility) characteristics will be presented.

1. INTRODUCTION

One of the flagships of the nuclear infrastructure of SCK•CEN is the BR2 reactor[1], a flexible irradiation facility known as a multipurpose materials testing reactor (MTR). This reactor is in operation since 1962 and has proven to be an excellent research tool, which has produced remarkable results for the international nuclear energy community in various fields such as material research for fission and fusion reactors, fuel research, reactor safety, reactor technology and for the production of radioisotopes for medical and industrial applications as well as silicon doping for the electronic industry.

The BR2 reactor is now licensed for operating until 2016 with a potential extension for another ten-year period until 2026. The Belgian Nuclear Research Centre at Mol is working since 1998 at the design of a multi-purpose flexible irradiation facility, called MYRRHA, that can replace BR2 MTR and that is innovative to support future oriented research projects needed to sustain the future of the research centre and responding the international needs for a sustainable development of nuclear energy through fission or fusion systems. MYRRHA has been designed as a multipurpose Accelerator Driven System for R&D applications, and consists of a proton accelerator delivering its beam to a liquid Lead-Bismuth spallation target that in turn couples to a sub-critical fast core, also cooled with Lead-Bismuth.

MYRRHA has started from the ADONIS[2] project (1995-1997). ADONIS was the first SCK•CEN project where the coupling between an accelerator, a spallation target and a subcritical core was studied. The ADONIS project has been conceived as a dedicated system for medical radioisotopes production and in particular for ^{99}Mo. The project has been stopped at the conceptual design level. In 1998, a larger ADS project intended to be a multipurpose research facility named MYRRHA[3] has been started. In 2005, MYRRHA has been proposed as a starting base for the XT-ADS design in the frame of the FP6 European Commission project EUROTRANS[4][5]. In the EUROTRANS project, XT-ADS is a short-term (operational around 2020) small-scale (50 to 100 MW_{th} experimental facility that should demonstrate the technical feasibility of transmutation in an accelerator driven system. The history of MYRRHA from ADONIS to FASTEF is discussed in section 2. The present FASTEF design of the project is described in section 3. Of this present design, the following components are discussed in detail: the cooling system (3.1), the core (3.2) and the in-vessel fuel handling machine (3.3). Section 4 describes the research applications at MYRRHA and at last, section 5 describes ISOL@MYRRHA.

2. HISTORICAL EVOLUTION OF MYRRHA: FROM ADONIS TO MYRRHA-FASTEF

The coupling between an accelerator, a spallation target and a subcritical core has been studied for the first time at SCK•CEN in collaboration with Ion Beam Applications (IBA, Louvain-la-Neuve) in the frame of the ADONIS project (1995-1997). ADONIS was a small irradiation facility, based on the ADS concept, having the single objective to produce radioisotopes for medical purposes and more particularly ^{99}Mo as a fission product from highly enriched ^{235}U fissile targets. The proposed design was of limited size with an accelerator of 150 MeV and a core with a power of around 1.5 MW_{th}. The system was a thermal spectrum machine and therefore water was used as coolant and moderator.

The ad-hoc scientific advisory committee recommended extending the purpose of the ADONIS machine to become a material testing reactor for material and fuel research, to study the feasibility of transmutation of the minor actinides and to demonstrate at a reasonable power scale the principle of the ADS. As such in 1998, the MYRRHA project has been started to design a facility based on the ADS concept responding the requested catalogue of applications listed above. In its 2005 version, MYRRHA consisted of a proton accelerator of the LINAC type delivering a 350 MeV * 5 mA beam to a windowless liquid Pb-Bi spallation target that in turn couples to a Pb-Bi cooled, subcritical fast core of 50 MW_{th}. This 2005 design (Figure 1) is also called 'Draft – 2' design[6].

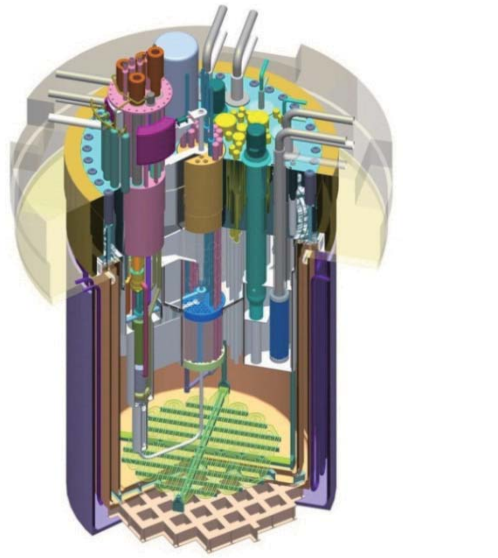


Figure 1. MYRRHA 'Draft – 2'

SCK•CEN proposed to use the MYRRHA 2005 design as a starting base for the XT-ADS design in the FP6 EUROTRANS integrated project[4]. This allowed optimizing an existing design towards the needs of MYRRHA/XT-ADS and within the limits of the safety requirements instead of starting from a blank page.

The MYRRHA/XT-ADS is a pool-type ADS cooled by lead-bismuth eutectic (LBE). The nominal LBE inlet temperature is 300°C and the core outlet temperature is around 400°C. The sub-critical reactor power is 57 MW_{th}. The primary heat exchangers (PHX) have been dimensioned to 70 MW_{th} in order to incorporate the beam power output as well as the decay heat and some extra margin. The reactor main vessel (Figure 2) with an elliptical bottom is around 6 m diameter. The hot and cold pools are separated by a diaphragm and the LBE circulates by means of two mechanical axial pumps. The heat is removed by four (2x2) LBE / water-steam primary heat exchangers. The MYRRHA/XT-ADS heat transfer system dissipates the heat to the atmosphere by means of the primary heat exchangers (evaporators) and the secondary circuit.

For the MYRRHA/XT-ADS subcritical core, the chosen fuel is made of a uranium-plutonium mixed oxide with a plutonium vector coming from the reprocessing of the current generation PWR spent fuel with an initial enrichment of 4.5% in ^{235}U. The sub-criticality level is around 0.95 which is considered as an appropriate level for a first of a kind medium-scale ADS.

The MYRRHA/XT-ADS accelerator is a linear accelerator delivering a 600 MeV * 3,2 mA beam into the spallation target. In order to improve the availability of the accelerator, some strategic principles are adopted:

- The highest reliability of the individual components is ensured;

- Redundancy is included in the design to ensure fault tolerance. This is achieved in 2 ways: a parallel redundancy of the injector and a serial redundancy over the high energy accelerating cavities.

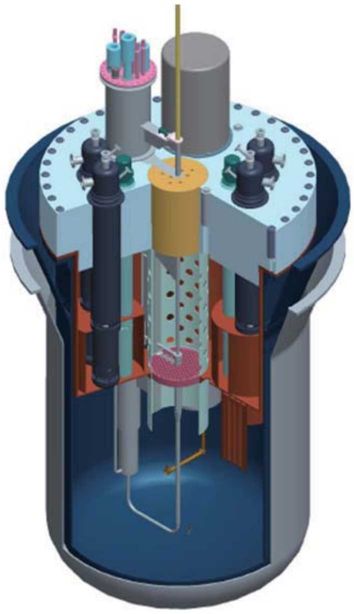


Figure 2. Overall view of MYRRHA/XT-ADS, showing its main internal components.

At the end of the EUROTRANS project, the XT-ADS design has complied with the project main requirements[7]. However, the objectives of the MYRRHA/XT-ADS did not fully correspond with the objectives of MYRRHA. Therefore, it's clear that future design work is needed to bring the design of MYRRHA/XT-ADS in line with the objectives, application catalogue and rationale for implementation. Since 2009, this work is conducted under the FP7 EC Project CDT[8] and called MYRRHA-FASTEF (Fast Spectrum Transmutation Experimental Facility).

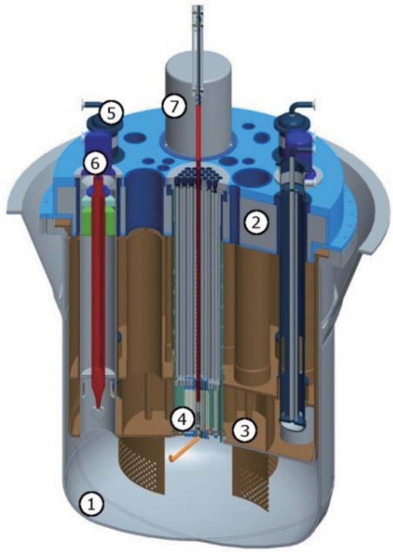
First of all, since the objective of MYRRHA-FASTEF is to operate both in a sub-critical mode and a critical mode, an analysis was performed to establish to which extent the design of MYRRHA/XT-ADS (which only considered sub-critical mode operation) needs to be modified to respond to the objective to operate also in critical mode. In this respect, it is clear that reactivity control and scram systems have to be included in the design. Also, an analysis had to be performed to determine the advantages and disadvantages linked to windowless target concept. Working in critical mode modifies significantly the safety characteristics of the facility and the impact of the safety feedback on the design needs to be implemented.

A second major topic for the update of the MYRRHA design is to obtain the objectives set in the applications catalogue. The MYRRHA/XT-ADS design was not able to reach all the different requirements in terms of irradiation conditions listed in the applications catalogue. To increase the flux level, it became obvious that the total power and power density will have to be increased.

3. CURRENT MYRRHA-FASTEF DESIGN

The main components/systems of MYRRHA-FASTEF are of the same MYRRHA/XT-ADS type with only increased size. The primary and secondary systems have been designed to evacuate a maximum core power of 100 MW_{th}. All the MYRRHA-FASTEF components are optimized for the extensive use of the remote handling system during components replacement, inspection and handling.

While MYRRHA-FASTEF is a pool-type ADS, the reactor vessel houses all the primary systems. In previous designs of MYRRHA, an outer vessel served as secondary containment in case the reactor vessel leaks or breaks. In the current design, the reactor pit implements this function, improving the capabilities of the reactor vault air cooling system. The vessel is closed by the reactor cover which supports all the in-vessel components. Figure 3 represents a vertical cut inside the reactor vessel, showing its main internal components. The main parameters are listed in Table 1.



1. Reactor Vessel
2. Cover
3. Diaphragm
4. Core
5. Primary Heat Exchanger
6. Primary Pump
7. In-vessel fuel handling machine

Figure 3. Vertical cut in MYRRHA-FASTEF, showing its main internal components

Table 1 - Main MYRRHA-FASTEF parameters

FA Length	2000 mm
Nominal Power	100 MW
Core inlet temperature	270 °C
Core outlet temperature	410 °C
Coolant velocity in core	2 m/s
Coolant pressure drop	2.5 bar
Secondary coolant	Saturated water/steam
Tertiary coolant	Air

3.1. The MYRRHA Accelerator

The accelerator is the driver of MYRRHA while it provides the high energy protons that are used in the spallation target to create neutrons which in turn feed the subcritical core. In the current design of MYRRHA, the machine must be able to provide a proton beam with an energy of 600 MeV and an average beam current of 3.2 mA. The beam is delivered to the core in continuous wave (CW) mode. Once a second, the beam is shut off for 200 µs so that accurate on-line measurements and monitoring of the sub-criticality of the reactor can take place. The beam is delivered to the core from above through a beam window.

Accelerator availability is a crucial issue for the operation of the ADS. A high availability is expressed by a long Mean Time Between Failure (MTBF), which is commonly obtained by a combination of over-design and redundancy. On top of these two strategies, fault tolerance must be implemented to obtain the required MTBF. Fault tolerance will allow the accelerator to recover the beam within a beam trip duration tolerance after failure of a single component. In the MYRRHA case, the beam trip duration tolerance is 3 seconds. Within an operational period of MYRRHA the number of allowed beam trips exceeding 3 seconds must remain under 10, shorter beam trips are allowed without limitations. The combination of redundancy and fault tolerance should allow obtaining a MTBF value in excess of 250 hours.

At present proton accelerators with megawatt level beam power in CW mode only exist in two basic concepts: sector-focused cyclotrons and linear accelerators (linacs). Cyclotrons are an attractive option with respect to construction costs, but they don't have any modularity which means that a fault tolerance scheme cannot be implemented. Also, an upgrade of its beam energy is not a realistic option. A linear accelerator, especially if made superconducting, has the potential for implementing a fault tolerance scheme and offers a high modularity, resulting in the possibility to recover the beam within a short time and increasing the beam energy.

A basic layout of the MYRRHA accelerator is provided in Figure 4, aiming at maximizing its efficiency, its reliability (or MTBF) and its modularity.

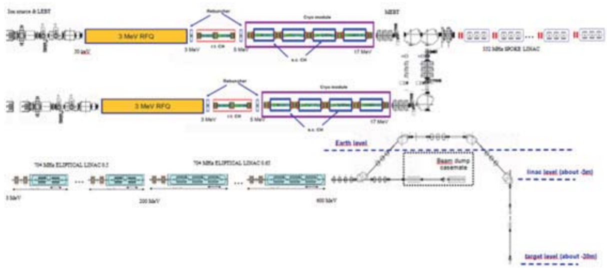


Figure 4. A schematic layout of the reference design of the MYRRHA accelerator

3.2. The core

At the present state of the design, the reactor core consists of mixed oxide (MOX) fuel pins, typical for fast reactors. A major change with respect to the previous version of the core is the switch from a windowless loop-type spallation target to a windowed beam tube-type spallation target. The previous version needed three central hexagons to house the spallation target while the present day design only needs one central hexagon. To better accommodate this central target, the fuel assemblies (FA) size is a little bit increased as compared to the MYRRHA/XT-ADS design. Consequently the In-Pile test Sections (IPS), which will be located in dedicated FA positions, are larger in diameter giving more flexibility for experiments. Thirty seven positions can be occupied by IPS's or by the spallation target (the central one of the core in sub-critical configuration) or by control and shutdown rods (in the core critical configuration). This gives a large flexibility in the choice of the more suitable position (neutron flux) for each experiment.

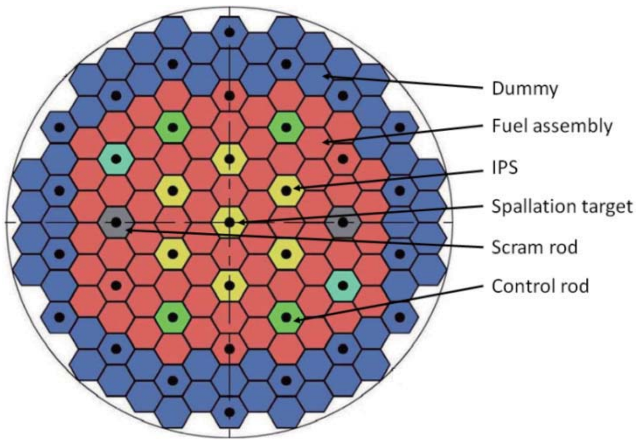


Figure 5. Cut in the MYRRHA-FASTEF core, showing the central target, the different types of fuel assemblies
and dummy components

The requested high fast flux intensity has been obtained optimizing the core configuration geometry (fuel rod diameter and pitch) and maximizing the power density. We will be using, for the first core loadings, 15-15Ti as cladding material instead of T91 that will be qualified progressively further on during MYRRHA operation. Thanks to the use of LBE as coolant, it permits to lower the core inlet operating temperature (down to 270 °C) decreasing the risk of corrosion and allowing to increase the core ΔT. This together with the adoption of reliable and passive shut down systems will permit to meet the high fast flux intensity target.

In subcritical mode the spallation target assembly, located in the central position of the core, brings the proton beam via the beam tube into the central core region. The assembly evacuates the spallation heat deposit, guarantees the barrier between the LBE and the reactor hall and leads the beam to the centre of the core, assuring the optimal conditions for the spallation reaction. The assembly is conceived as an IPS and is easily removable or replaceable.

3.3. The cooling system

The primary and secondary cooling systems have been designed to evacuate a maximum thermal core power of 100 MW. The average coolant temperature increase in the core in nominal conditions is 140 °C with a coolant velocity of 2 m/s. The primary cooling system exists of two pumps and four primary heat exchangers (PHX).

The primary pumps shall deliver the LBE to the core with a mass flow of 4750 kg/s (453 l/s per pump). The working pressure of the pump is 3.10^5 Pa. The pump will be fixed at the top of the reactor cover, which is supposed to be the only supporting and guiding element of the pump assembly.

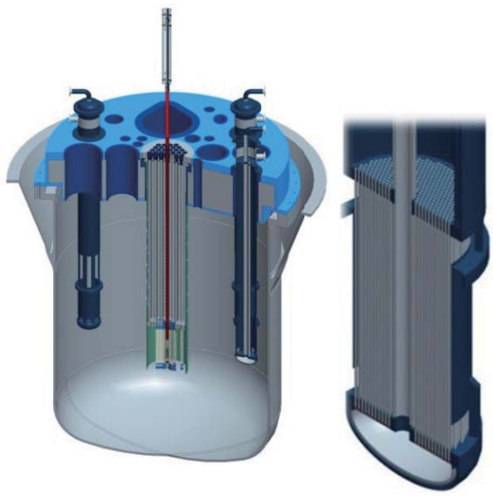


Figure 6. Heat Exchangers

The main thermal connection between the primary and secondary cooling system is provided by the primary heat exchangers (Figure 6). These heat exchangers are from the shell and tube, single-pass and counter-current type. Pressurized water at 200 °C is used as secondary coolant, flowing through the feed-water pipe in the centre of the PHX to the lower dome. All the walls separating the LBE and water plena (feed-water tube, lower dome and upper annular space) are double walled to avoid pre-heating of the secondary coolant and to prevent water leaking in the LBE in case of tube failure.

In case of loss of the primary flow (primary pumps failure), the primary heat exchangers aren't able to extract the full heat power. In such cases, the beam must be shut off in the subcritical case and the shutdown rods inserted in the critical case. The decay heat removal (DHR) is achieved by natural convection. Ultimate DHR is done through the reactor vessel coolant system (RVACS, reactor vessel air cooling system).

3.4. The in-vessel fuel handling machine

The interference of the core with the proton beam, the fact that the room situated directly above the core will be occupied by lots of instrumentations and IPS penetrations, and core compactness result in insufficient space for fuel handling to (un)load the core from above. Since the very first design of MYRRHA, fuel handling is performed from underneath the core. Fuel assemblies are kept by buoyancy under the core support plate.

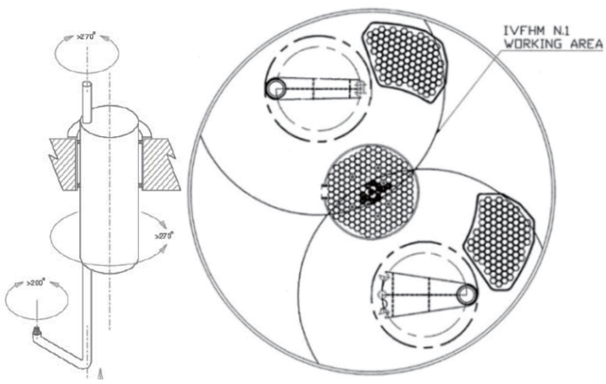


Figure 7. The in-vessel fuel handling machine

Two fuel handling machines are used, located at opposite sides of the core (Figure 7). Each machine covers one side of the core. The use of two machines provides sufficient range to cover the necessary fuel storage positions without the need of an increase for the reactor vessel when only one fuel handling machine is used. Each machine is based on the well-known fast reactor technology of the 'rotating plug' concept using SCARA (Selective Compliant Assembly Robot Arm) robots. To extract or insert the fuel assemblies, the robot arm can move up or down for about 2 meter. A gripper and guide arm is used to handle the FA's: the gripper locks the FA while the guide has to functions, namely to hold the FA in the vertical orientation and to ensure neighbouring FA's are not disturbed when a FA is extracted from the core. An US sensor is used to uniquely identify the FA's.

The in-vessel fuel handling machine will also perform in-vessel inspection and recovery of an unconstrained FA. Incremental single-point scanning of the diaphragm can be performed by an US sensor mounted at the gripper of the IVFHM. The baffle under the diaphragm is crucial of the strategy as it limits the work area where inspection and recovery is needed. It eliminates also the need of additional recovery and inspection manipulators, prevents items from migrating into the space between the diaphragm and the reactor cover, and permits side scanning.

4. RESEARCH APPLICATIONS AT MYRRHA (TBC)

MYRRHA possesses several irradiation stations in and around the reactor core. In this way, a very broad neutron spectrum, ranging from thermal energies to fast neutrons, is available. Both thermal and fast neutron fluxes are very high compared to classic material testing reactors, making it feasible to simulate – within a reasonable time frame – the long-term exposure of materials in commercial reactors. Moreover, the combination of high radiation damage doses (dpa) and the high production of light gasses (H, He) per unit of dpa close to the spallation target is very interesting for the study of materials for fusion research. Therefore, one can summarize the MYRRHA applications catalogue as follows:

- To demonstrate the ADS full concept by coupling the three components (accelerator, spallation target and sub-critical reactor) at reasonable power level to allow operation feedback, scalable to an industrial demonstrator.
- To allow study of the efficient transmutation of high-level nuclear waste, in particular minor actinides that would request high fast flux intensity ($\Phi_{>0.75MeV}$ = 10^{15} n.cm^{-2}.s^{-1}).
- To be operated as a flexible fast spectrum irradiation facility allowing for:
    - Fuel developments for innovative reactor systems, which need irradiation rigs with a representative flux spectrum, a representative irradiation temperature and high total flux levels (Φ_{tot} = 5 . 10^{14} to 10^{15} n.cm^{-2}.s^{-1}).
    - Material developments for GEN IV systems, which need large irradiation volumes (3000 cm³) with high uniform fast flux level ($\Phi_{>1 MeV}$ = 1 to 5.10^{14} n.cm^{-2} .s^{-1}) in various irradiation positions, representative irradiation temperature and representative neutron spectrum conditions.
    - Material development for fusion reactors that need also large irradiation volumes (3000 cm³) with high constant flux level ($\Phi_{>1 MeV}$ = 1 to 5.10^{14} n.cm^{-2}.s^{-1}), a representative irradiation temperature and a representative ratio appm He/dpa(Fe) = 10.
    - Radioisotope production for medical and industrial applications, holding a backup role for the classical medical radioisotopes.
    - Focussing on R&D and production of radioisotopes requesting very high thermal flux levels (Φ_{th} = 2 to 3 . 10^{15} n.cm^{-2}.s^{-1}) due to double capture reactions.
    - Industrial applications, such as Si-doping, which need a thermal flux level depending on the desired irradiation time; for a flux level Φ_{th} = 10^{13} n.cm^{-2}. s^{-1}, an irradiation time in the order of days is needed.

As described in the design of the core, thirty seven positions can be occupied by In-Pile test Sections to perform experiments. Figure 8 shows a MCNPX plot-view displaying a patch of channels in the MYRRHA-FASTEF critical core. Besides the driving fuel assemblies, seven IPS are loaded, each one having seven channels wherein stacks of various steel specimen for material testing as regards to irradiation damage are placed.

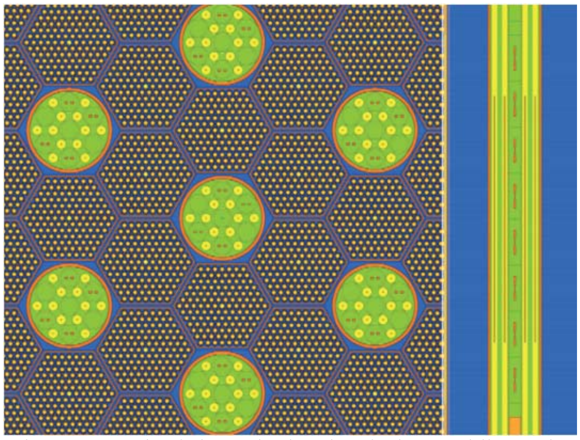


Figure 8. Patch of channels showing fuel assemblies and 7 IPS for material testing samples (left) and a stack of steel specimen (right)

In Table 2, the achievable neutron flux levels and the accumulated dpa over one effective full power year at 100 MW$_{th}$ are given for the typical stack of 8 specimens, as shown on the right side of Figure 8. Both values of the IPS loaded in the central core channel [0 0 0] as an IPS loaded in ring #2 are listed. In terms of fuel irradiation performances, the flux levels are given at different fuel assembly positions. The values for the flux levels given in the second table are for the flux in the central fuel pin of the considered fuel assembly at the mid-plane of the core.

Table 2 - Fluxes and accumulated dpa over one effective full power year at 100 MW$_{th}$

	IPS in Chan [0 0 0]		IPS in Chan [2 0 0]	
Sample n°	dpa/EFPY	Φ_{tot}	dpa/EFPY	Φ_{tot}
8	19.0	2.49E+15	15.9	2.13E+15
7	24.5	3.05E+15	21.3	2.63E+15
6	28.9	3.48E+15	24.9	3.05E+15
5	30.8	3.70E+15	27.3	3.29E+15
4	31.3	**3.77E+15**	27.8	**3.35E+15**
3	29.7	3.62E+15	26.7	3.21E+15
2	26.0	3.26E+15	23.0	2.89E+15
1	20.5	2.75E+15	18.2	2.46E+15

FA centre	[1 0 0]	[1 2 0]	[3 0 0]	[4 0 0]
Φ_{tot}	3.83E+15	3.61E+15	3.21E+15	2.97E+15
$\Phi_{>0.75 MeV}$	7.5E+15	7.06E+14	6.37E+14	5.72E+14

5. ISOL@MYRRHA

Radioactive Ion Beam (RIB) research has been recognized as one of the top priorities in nuclear physics. RIBs create a wide area of research opportunities in several fields of science, such as atomic, nuclear and solid state physics, fundamental interactions, astrophysics, biology and medical applications.

The so-called ISOL@MYRRHA[9][10] facility will use a part of the proton beam delivered by the MYRRHA accelerator to produce pure and intense RIBs by using the Isotope Separator On-Line (ISOL) method. The scenario is to operate both MYRRHA and ISOL@MYRRHA in parallel, which requires a beam-splitting system. Up to 5% of the main beam will be delivered to ISOL@MYRRHA, in a pulsed mode with sufficiently high repetition rate (Figure 9).

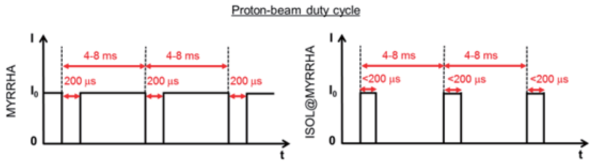


Figure 9. Proton beam duty cycle of MYRRHA and ISOL@MYRRHA

ISOL@MYRRHA will follow closely the RIB production schemes that are developed and successfully used at the ISOLDE-CERN and TRIUMF facilities. It will be equipped with ruggedized target-ion source systems that allow the use of a selection of target materials, including actinide targets, which can withstand the proton-beam power (~60 kW) without compromising the reliability, the longevity, the diffusion and effusion properties, and the yield of particular radioactive isotopes.

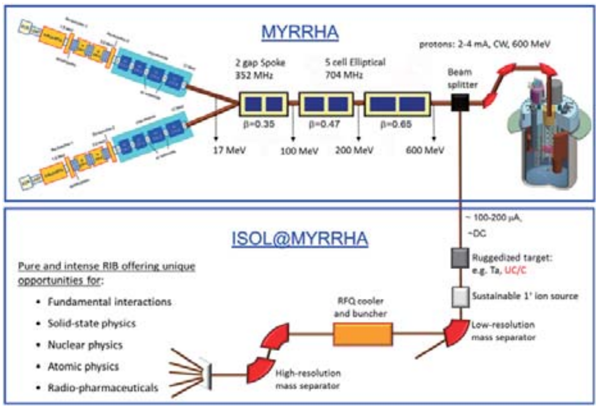


Figure 10. ISOL@MYRRHA: pre-conceptual design

Three types of ion sources are foreseen for selective ionization of the products: the hot-surface ion source, the resonant ionization laser ion source, and the electron cyclotron resonance ion source. Additional purification occurs by mass separation after extracting the ions over a potential difference of up to 60 kV forming the radioactive ion beam.

In order to make effective use of the beam time, the parallel multi-users aspect of ISOL@MYRRHA is an important issue in the design study. Since a high-resolution mass separator prevents the use of different beams at the same time, a pre-separator with low mass resolution is considered. In this way, one could envisage a scenario in which the low mass isotopes (^{8}Li) are used for solid-state physics in parallel with an experiment using heavier nuclei.

Moreover, the pre-separation avoids a too intense RIB loading of the RFQ cooler and buncher, which allows a high-quality low-emittance beam. As a result, the high-resolution magnet can be maximally exploited with a mass-resolving power M/ΔM in the order of 10^4.

The ISOL@MYRRHA facility will deliver RIBs for experimental programs in need of extended beam times, up to a few months, with high intensity and reliability. These are experiments needing high statistics and accurate calibration, experiments that hunt for very rare phenomena, or suffer from intrinsic low detection efficiency. ISOL@MYRRHA is thus complementary to other RIB facilities. Figure 11 shows an overview of typical RIB research, where opportunities specific for ISOL@MYRRHA are indicated in yellow.

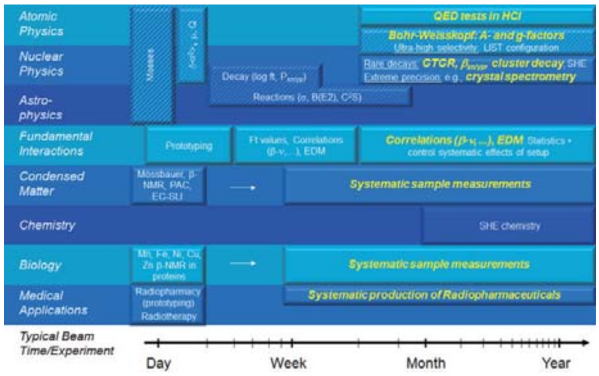


Figure 11. Overview of RIB research, with ISOL@MYRRHA opportunities indicated in yellow. The width of each box corresponds to the typical beam-time required for the indicated experiment. Shaded areas indicate an overlap between two research fields.

The high intensity beams and long/regular beam times are important for all fields in science using RIBs, ranging from fundamental-interaction measurements with extremely high precision over systematic measurements for condensed-matter physics and production of radiopharmaceuticals. Long beam times could be also of interest for astro-physics, when nuclear reactions with small cross sections are involved, but the absence of a post-accelerator in the present design of ISOL@MYRRHA will prevent such kind of studies. Although higher-energy secondary beams are not discarded for a later phase, only research with low-energy beams is addressed for the moment.

CONCLUSION

SCK•CEN is proposing to replace its ageing flagship facility, the Material Testing Reactor BR2, by a new flexible irradiation facility, MYRRHA. Considering the international (and European) needs, MYRRHA is conceived as a flexible fast spectrum irradiation facility able to work in both sub-critical and critical mode.

MYRRHA is now foreseen to be in full operation by 2023 and it will be operated as an Accelerator Driven Systems to demonstrate the ADS technology and the efficient demonstration of transmutation of MA in sub-critical mode and will be able to be run in critical mode. As a fast spectrum irradiation facility, it will address fuel research for innovative reactor systems, material research for GEN IV systems and for fusion reactors, industrial applications, such as silicon doping, and radioisotope production for medical and industrial applications. Being based on heavy liquid metal coolant technology, MYRRHA will also act as the European Technology Pilot Plant for the development of the Lead Fast Reactor.

In parallel with the operation of the sub-critical reactor, a small fraction of the proton-beam delivered by the MYRRHA accelerator will be used for fundamental research, by the production of high-intensity RIBs via the ISOL method.

ACKNOWLEDGEMENTS

The MYRRHA/XT-ADS design work has been performed under the Integrated Project EUROTRANS (Ref. FI6W-CT-2004-516520) cofounded by the European Union in its 6th Framework Programme. The MYRRHA-FASTEF design work is being performed under the Collaborative Project CDT (Ref. FP7-232527) cofounded by the European Union in its 7th Framework Programme. Acknowledgment is also due to all the colleagues of all partner institutes for their contributions in many different topics associated with the MYRRHA project.

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LIST OF ACRONYMS

ADONIS	Accelerator Driven Optimized Nuclear Irradiation System
ADS	Accelerator Driven System
APPM	Atomic Part Per Million
BR2	Belgian Reactor 2
CDT	Central Design Team
DHR	Decay Heat Removal
DPA	Displacement Per Atom
EUROTRANS	European Research Programme for the Transmutation of High Level Nuclear Waste in an Accelerator Driven System
FA	Fuel Assembly

FASTEF	Fast Spectrum Transmutation Experimental Facility
FP7	Seventh Framework Programme (of the European Commission)
HLW	High-level Long-lived radioactive Waste
IPS	In-Pile test Section
ISOL	Isotope Separator On-Line
LBE	Lead-Bismuth Eutectic
LFR	Lead-cooled Fast Reactor
LINAC	Linear Accelerator
LLFP	Long-Lived Fission Products
MA	Minor Actinide
MOX	Mixed Oxide
MTR	Material Testing Reactor
MYRRHA	Multi-purpose Hybrid Research Reactor for High-tech Applications
P&T	Partitioning and Transmutation
PHX	Primary Heat Exchanger
PWR	Pressurized Water Reactor
RFQ	Radio Frequency Quadrupole
RIB	Radioactive Ion Beams
SCARA	Selective Compliant Assembly Robot Arm or Selective Compliant Articulated Robot Arm
XT-ADS	eXperimental demonstration of the technical feasibility of Transmutation in an Accelerator Driven System

9.3 CONCEPTUAL DESIGN OF NEXT GENERATION MTR

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Conceptual design of the high-performance and low-cost next generation materials testing reactor (MTR) which will be expected to construct in the nuclear power plant introduction countries, started from 2010 in JAEA and nuclear-related companies in Japan. The aims of this conceptual design are to achieve highly safe reactor, economical design, high availability factor and advanced irradiation utilization. One of the basic reactor concept was determined as swimming pool type, thermal power of 10MW and water cooled and moderated reactor with plate type fuel element same as the JMTR. It is expected that the research reactors are used for human resource development, progress of the science and technology, expansion of industry use, lifetime extension of LWRs and so on.

Keywords: Compact material testing reactor, Conceptual design, High safety, Low cost, High reactor operation rate, High irradiation technology

1. Introduction

The number of research reactors in the world is decreasing because of their aging. On the other hand, the plan to construct the nuclear power plants is increasing in Asian countries. In these countries, the key issue is the human resource development for operation and management of the constructed nuclear power plants. It is expected that the research reactors are used for ①human resources development, ② progress of the science and technology, ③expansion of industry use, ④lifetime extension of LWRs and so on.

From above backgrounds, the Neutron Irradiation and Testing Reactor Center of JAEA and nuclear-related companies in Japan began to discuss a basic concept of high-performance and low-cost next generation compact materials testing reactor (MTR) for education and training, etc. This activity also has a role of human resource development as the new research reactor design activity in future.

2. Aims of conceptual design

Aims of conceptual design are set as follows. It is necessary to consider the following items to achieve each target.

(1) Economical design
- ① Reduction of construction cost
  - Decrease in expensive parts and simplification of construction
  - Standardization of facilities by increase of the same type reactor
- ② Decrease in operation cost
  - Decrease in fuel cost by high burn-up design

(2) High availability factor
- ① Advanced maintenance
  - Decrease in the number of equipments
  - Maintenance during operation term
- ② High reliability
  - Multiplexing of facilities
  - Apply of outstanding commodity
- ③ Long term operation per cycle
  - Decrease in the number of control rods

・ High burn-up

(3) Advanced irradiation utilization

① Exchange of irradiation capsule in the short term

・ Adoption of pool type reactor of easy handling

② Speedy post-irradiation examination

・ Hot laboratory inside reactor building or neighboring building

③ High neutron flux

・ High power density

④ Flexible irradiation ability

・ Examination of reflector and absorber to control neutron and gamma spectrum (including consideration of new materials)

(4) Highly safe reactor

・ Maintain the reactor core under water

・ Passive decay heat removal

3. Consideration of reactor type

Some types of MTRs such as tank type, swimming pool type, etc. are operated in the world (Fig.1). Swimming pool type reactors are generally economical on running cost. From the viewpoint of neutron economy, high power reactors more than about 30MW are uneconomical on fuel cost because of neutron poisoning.

One of the basic reactor concept was therefore determined as swimming pool type, thermal power of 10MW and water cooled and moderated reactor with plate type fuel element same as the JMTR.

Moreover, it is necessary that neutron flux of the reactor exceeds that of existing LWRs in order to investigate for lifetime extension of introduced LWRs. Therefore, target of maximum fast neutron flux was set 1×10^{18} n/(m^2 ・ s).

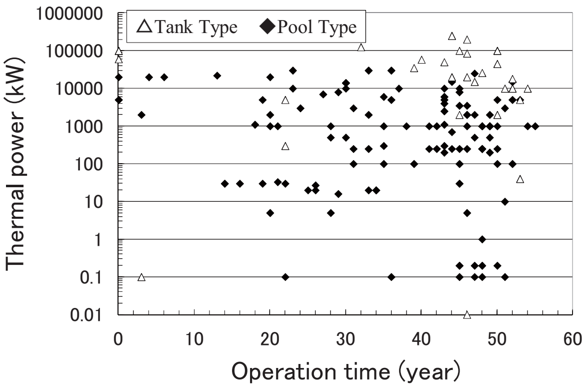

Fig.1. Relation between reactor type (tank type and swimming pool type) and thermal power in world test research reactor.

4. Basic concept of next generation MTR

Present technical specification is shown in the following table.

Reactor Type	Swimming Pool Type
Thermal Power	10 MW
Coolant	Light Water
Moderator	Light Water
Fast Neutron Flux	About 8×10^{17} n/(m^2 ・ s) (E > 1 MeV)
Fuel Type	Plate Type
Control Rod	Hafnium
Reflector	High-Performance Reflector (Be, Al)
Coolant Flow Coolant Direction	About 1,200 m^3/h Down Flow
Coolant Temperature	Inlet Temp. 42℃ Outlet Temp. 50℃
Availability Factor	About 70%
Main Use	Medical RI Industrial RI Education and Trainning

4.1 Preliminary neutronic analysis of the reactor

The target of reactor thermal power is 10MW, and fast neutron flux is 1×10^{18} n/(m^2・s) (E>1MeV).

MCNP (A General-purpose Monte Carlo N-Particle Transport Code) was used for the calculation of the reactor core design study. Maximum fast neutron flux and effective multiplication factor (k_{eff}) was evaluated by this calculation code. As a result, maximum fast neutron flux was about 7.6×10^{17} n/(m^2 ・ s) (k_{eff} = 1.09192) in case of reactor core design where the core consists of 16 fuel elements and 4 control rods with fuel followers (the reference core). The calculation model for MCNP is shown in figure 2.

Furthermore, maximum operation time was calculated by SRAC code system. The calculated result shows that maximum operation time is about 250 day in the case of the reference core. The standard core consists of 16 fuel elements and 4 control rods with fuel followers, surrounded by beryllium reflector element and aluminum reflector element. The calculation was carried out assuming that all fuels are fresh fuels, all control rods are pulled out and there is no irradiation capsule (Fig.3 and Fig.4).

For the next step, study of core re-arrangement and burn-up calculation will be carried out to increase the performance.

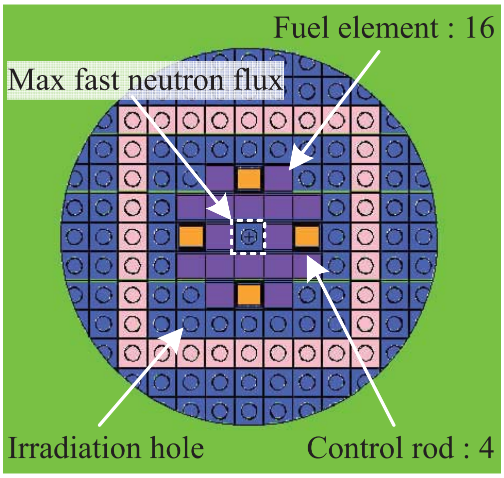

Fig.2. Calculation model by MCNP (standard core)

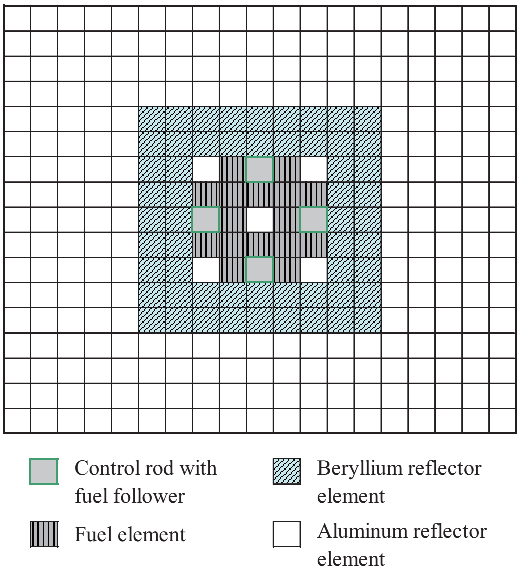

Fig.3. Calculation model by COREBN

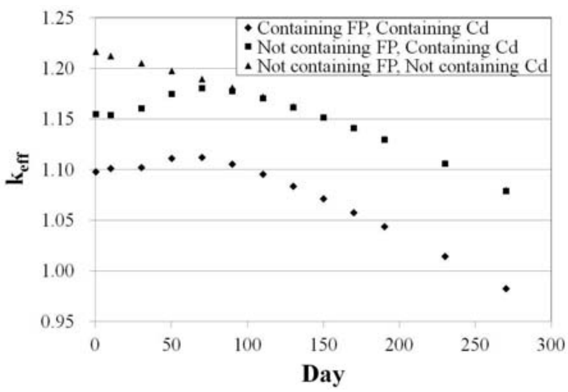

Fig.4. Result by COREBN

4.2 Preliminary thermal hydraulic analysis of reference core

The thermal hydraulic calculation was carried out by using COOLOD (steady-state thermal hydraulic analysis code). It is assumed that the fuel element is the same as that of JMTR, and the core is the reference core which is described in 4.1.

The assumed thermal hydraulic parameters are as follows;

① Inlet temperature : 40 ℃
② Inlet flow rate : 1200 m^3/h
③ Thermal power : 10 MW
④ Radial peaking factor : 1.120
⑤ Engineering hot channel factor for bulk coolant temperature rise : 1.33
⑥ Engineering hot channel factor for film temperature rise : 1.57
⑦ Axial power distribution(Axial peaking factor) : 1.47

The steady-state thermal hydraulic calculation was carried out to examine effect of core flow rate, inlet temperature and thermal power on minimum DNBR.

When inlet temperature changed from 35 ℃ to 90 ℃, minimum DNBR changed from 4.5 to 1.2 (Fig.5). DNBR was calculated to be 4.2 in case of the core inlet temperature 40℃. This result shows that the core has enough safety margins against DNB during normal operation.

When core flow rate changed from 200 to 1800 m^3/h, minimum DNBR changed from 0.7 to 6.3 (Fig.6). Then, outlet temperature changed from 83 ℃ to 45 ℃. DNBR was calculated to be 4.2 and outlet temperature is about 47 ℃ in case of the core flow rate 1200 m^3/h.

When thermal power changed from 10 MW to 20 MW, minimum DNBR changed from 4.2 to 2.1 (Fig.8). DNBR is 4.2 in case of 10 MW thermal power. This result shows that thermal power is able to be increased for performance upgrade.

The effect of thermal hydraulic parameters on minimum DNBR is investigated. In consequence, it was confirmed that design parameters of thermal hydraulic analysis are reasonable.

For the next step, analysis of abnormal transient during operation will be carried out to decide major thermal hydraulic design parameters.

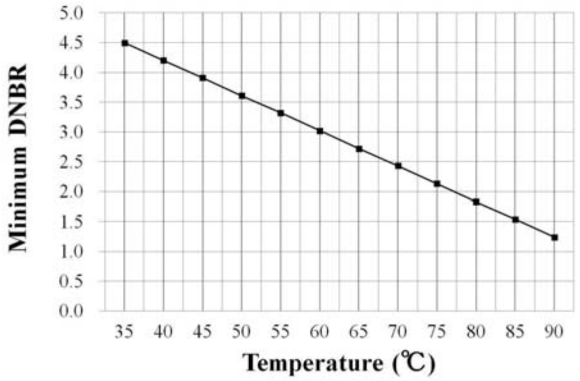

Fig.5. Change of minimum DNBR as a function of core inlet temperature.

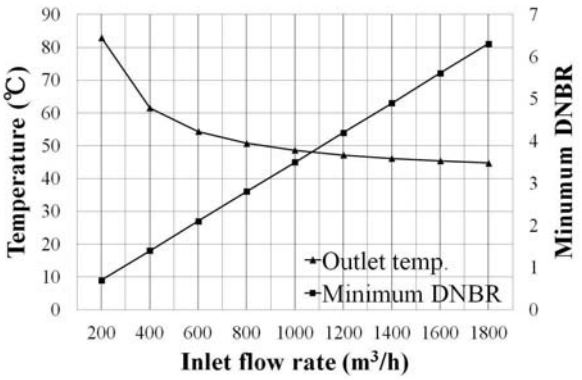

Fig.6. Change of minimum DNBR and reactor outlet temperature as a function of inlet flow rate.

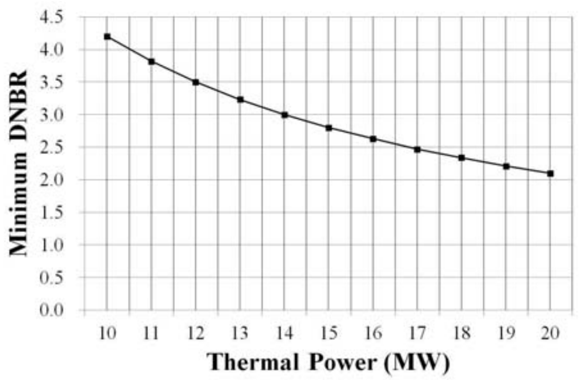

Fig.7. Change of minimum DNBR as a function of thermal power.

4.3 Study of facilities

(1) Reactor core components

Materials to coordinate gamma spectrum and neutron spectrum were investigated from its feasibility and durability point of view, and shielding capabilities of these materials were calculated.

(2) Irradiation facilities

Irradiation facilities experienced in JMTR and JRR-3, such as rabbit irradiation facility and loop irradiation facility, were selected as basic facilities. These facilities are necessary to change corresponding to user’s needs. For the next step, irradiation capabilities, such as neutron flux, irradiation temperature, etc. will be evaluated.

(3) Cooling system

Cooling system of existing test and research reactors such as JMTR and JRR-3 was investigated, and cooling system required for the reactor was determined.

(4) Measurement control facility

Type and characteristic of neutron detector used in each test and research reactor was investigated. Neutron detectors were selected from reliability, economy, durability and maintainability point of view.

(5) Radiation control facility

The radiation control facilities in existing power plants and research reactor were investigated, and subjects to study were selected for the next step.

(6) Hot laboratory

It is desirable that the hot laboratory is arranged in the reactor building or connected by a water canal same as the JMTR from operation point of view. Re-irradiation of irradiated samples is prospective in future by irradiation needs. Taking into consideration of this re-irradiation needs, necessary facilities to be arranged in the hot laboratory was determined.

5. Conclusion

Conceptual design of the high-performance and low-cost next generation materials testing reactor (MTR) which will be expected to construct in the nuclear power plant introduction countries, started from 2010 in JAEA and nuclear-related companies in Japan. The basic concept is a multipurpose low-power research reactor for education and training, etc.

Aims of the conceptual design study aer to achieve ① Economical design, ② High availability factor, ③Advanced irradiation utilization, ④Highly safe reactor. Applying design aims, basic concept of the reactor was discussed and investigated.

The reference core consists of 16 plate type fuel elements and 4 control rods with fuel followers. Followings were made clear from neutronic design.

- The maximum fast neutron flux is 7.6×10^{17} n/m^2/s.
- 250 days continuous operation is possible under the condition of all fresh fuels and no irradiation materials.

On the other hand, the following were made clear from thermal hydraulic design.

- DNBR is 4.2 in case of the core flow rate 1200 m^3/s and core inlet temperature 40˚C,
- The core has enough safety margins against DNB during normal operation.

6. Future Plan

In future, neutronic and thermal hydraulic analyses will be investigated in detail, and conceptual design of the cooling system, irradiation facilities, and hot laboratory etc. will be carried out. Furthermore, the evaluation of reactor kinetics and safety analysis will be conducted.

Requirement of safety has been tightened in the world after the accident of the Tokyo electric fukushima daiichi nuclear power plant. The Review Guide for Safety Design of nuclear power plants will be reconsidered in Japan. It is necessary that conceptual design study of the next generation compact materials testing reactor should be examined with attention to these tendencies.

References

[1] J.F.BRIESMEISTER,(Ed), LA-12625-M (1997).
[2] K. Tsuchihashi, Y. Ishiguro, K. Kaneko et al, JAERI 1302 (1986).

Note:

This design study was performed by JAEA as a joint research with Hitachi-GE Nuclear Energy, Ltd., KAWASAKI HEAVY INDUSTRIES, LTD., NGK INSULATORS, LTD., SUKEGAWA ELECTRIC CO.,LTD., Fuji Electric Co.,Ltd., CHIYODA TECHNOL CORPORATION.

A main part that these companies took charge is;

(1) Hitachi-GE Nuclear Energy, Ltd.
: Investigation of technology that apply to next generation MTR
(2) KAWASAKI HEAVY INDUSTRIES, LTD.
: Study of reactor cooling facility
: Study of irradiation test facility
(3) NGK INSULATORS, LTD.
: Study of part of structure in reactor core (neutron reflector, etc.)
(4) SUKEGAWA ELECTRIC CO.,LTD.
: Study of neutron detector facility
(5) CHIYODA TECHNOL CORPORATION
: Study of radiation monitoring facility
(6) Fuji Electric Co.,Ltd.
: Study of radiation control system

10. Advancement of PIE Technology

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10.1 DEVELOPMENT OF FRACTURE TOUGHNESS ESTIMATION METHOD USING THIN TUBE

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The fracture mechanics method in which existence of a crack was assumed is used for evaluation in the maintainability of the device and structure in a nuclear reactor. A large test piece is required for a fracture toughness test in order to carry out on condition of the plane-strain state at the tip of a crack. Therefore, a small component like the thin tube which cannot obtain CT test piece of required size has been considered that it is difficult to evaluate fracture toughness. The purpose of this research is to perform fracture toughness evaluation with a thin tube (10 mm or less in diameter) like the thimble tube used at the nuclear reactor of PWR.

Fracture toughness evaluation was performed in proportion to the bending test shown in ASTME1820-08. SUS304 cold-worked 20% was used as simulated material of an irradiated material. A specimen configuration is an arc which divided a thimble tube (5.2 mm in inside diameter, outside diameter of 7.6 mm) 20 mm in length into four. Thereby, four examinations are possible from the same material. The fixture of the arc test piece was designed and manufactured so that a 55 mm [in overall length] bending test could be performed. This fixture prepares a slot with the curvature united with the specimen configuration, inserts a test piece in that slot, and fixes to it. The fixture for bending tests was manufactured in consideration of the manipulator operatively in a hot cell. The machined slot and the fatigue precrack were introduced, and then bending fracture toughness testing by the unloading compliance method was carried out at the room temperature in the atmosphere. The fracture toughness test by the 1.0TCT specimen obtained from the same material was carried out for comparison. After the examination, the length of the machined slot, the fatigue precrack, and the ductile crack was measured from the fracture surface. JQ value was determined from the J-R curve obtained by examination. Both examinations showed the almost of the same degree value. The prospect that the fracture toughness of the tubing material which cannot obtain CT specimen could be evaluated was acquired.

1) presented address: Toshiba Corp.

4th International Symposium on Material Testing Reactors, Oarai, Japan, December 5-9, 2011

Development of fracture toughness estimation method using thin tube

Masanari SUGIYAMA (NFD)
Tadahiko TORIMARU (NFD)
Kazuhiro CHATANI (NFD)

Background(1)

- Fracture toughness decreases with increasing fast neutron fluence.
- It is very important to evaluate neutron fluence dependency on fracture toughness of core internals in a nuclear reactor.

Background(2)

- Fracture toughness is usually obtained with relatively large specimen like CT, 3PB.
- The test method for irradiated thin tube like flux thimble tube have not been developed mostly.

Objectives

- To design the technique of fracture toughness test for irradiated thin tube in hot cell.
- To review the technique from the results of fracture toughness tests using unirradiated cold work materials

Bending tests(ASTM E1820-08)

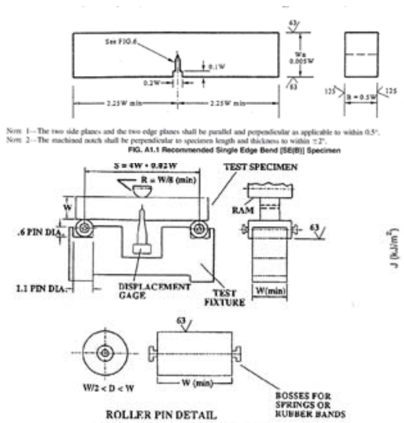

FIG. 4 Bend Test Fixture Design

The test method was designed based on this standard.

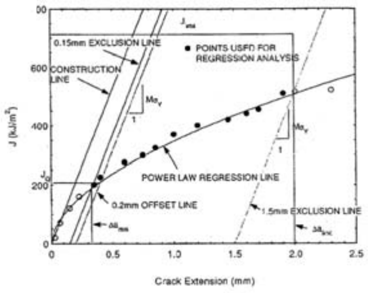

FIG. A9.1 Definition of Construction Lines for Data Qualification

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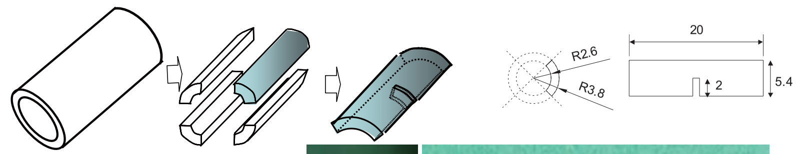
Specimen

A specimen configuration is an arc which divided a thimble tube into four.



Thin tube
- ID 5.2 mm, OD 7.6 mm
- 20 mm in length

Feature
Processing within a hot cell is easy.
Four examinations are possible from the same material.
→ Effective use of irradiation material
The increase in the number of examinations

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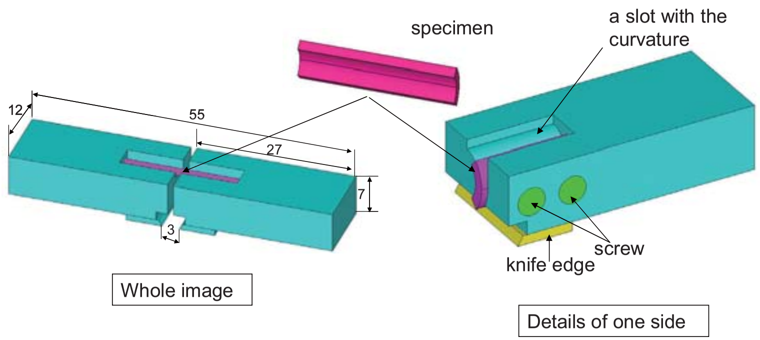
Fixture for the thin tube



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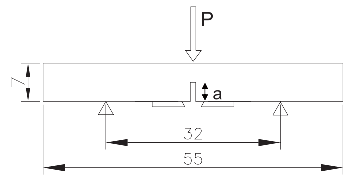
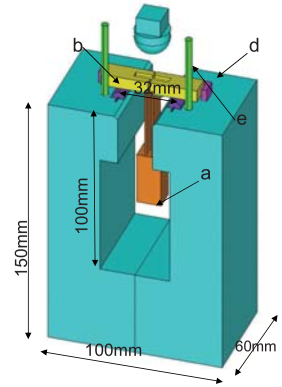
The fixture for the bending test

- The fixture designed is based on ASTM E1820-08.
- It is easy to treat within a cell using manipulator.





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The fracture toughness test

NFD

Cold-worked Materials
→Machining of specimen.
↓
the machined slot and the fatigue precrack were introduced.
↓
Fracture toughness testing by the unloading compliance method were carried out .
・thin tube specimen
(Bend testing)
・1.0CT specimen
(Compact testing)

In room-temperature atmospheric air,

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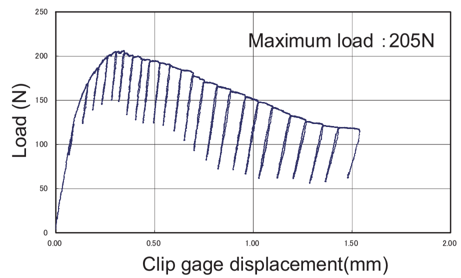
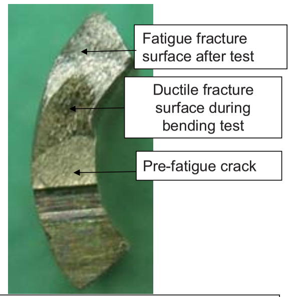
The result of fracture toughness test

NFD



Bending fracture toughness testing by the unloading compliance method.



After the examination, the length of crack was measured from the fracture surface.
・The machined slot
・The fatigue precrack
・The ductile crack

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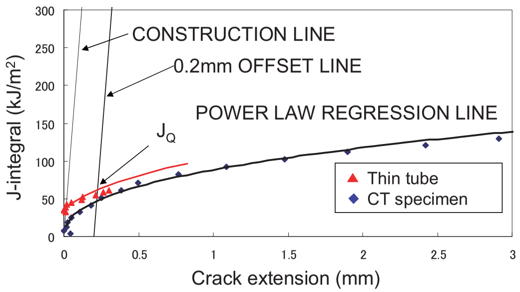
Fracture toughness evaluation

NFD



J_Q value obtained by the bending test was 57kJ/m^2.
J_Q value obtained by CT examination was 52kJ/m^2.

Both examinations showed the almost of the same degree value

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Summary

NFD

- Fracture toughness technique for thin tube in hot cell was considered.
- The fixture based on bending test technique was designed in consideration of the manipulator operatively in a hot cell.
- Fracture toughness tests using thin tube specimen and CT specimen were carried out.
- The fracture toughness value JQ of the same degree was obtained from the bending test and CT examination.
- The prospect of fracture toughness evaluation of thin tube material was acquired with this method.
- This method is applicable in other tubing materials.
- Moreover, application into the material in a reactor which cannot obtain a big test piece is also expectable.

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10.2 Improvement of the center boring device for the irradiated fuel pellets

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The power ramp tests performed at JMTR in Oarai R&D Center are objected to study the safety margin of the high burnup fuels. One of the important parameters measured during this test is the center temperature of the fuel pellet. For this measurement, a thermocouple is installed into the hole bored at the pellet center by the center boring device, which can fix the fuel pellet with the frozen CO_2 gas during its boring process. At the Reactor Fuel Examination Facility (RFEF) in Tokai R&D Center, several improvements were applied for the previous boring device to gain its performance and reliability. The major improvements are the change of the drill bit, modification of the boring process and the optimization of the remote operability. The mock-up test will be performed with the irradiated fuel pellet to confirm the benefit of improvement. This study was conducted under a contract with the Nuclear and Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI).

Keywords: Power ramp test, Frozen CO_2 gas, Fuel pellet, BWR, PIE, RFEF, JMTR

1 INTRODUCTION

The power ramp tests performed at JMTR in Oarai Research and Development Center are objected to study the safety margin of the high burnup nuclear fuels and their behavior under the transient condition. One of the important parameters measured during the tests is the center temperature of the fuel pellet. For this measurement, a thermocouple is installed into the hole bored at the pellet center (Fig. 1) by the center boring device.

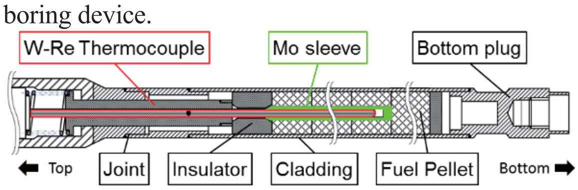

Fig. 1 Fuel rod for Power ramp test

Fig. 2 illustrates the processing steps of the center boring device previously developed at the JMTR Hot Laboratory in Oarai R&D Center [1]. This device consists of several units, such as the freezing unit, boring unit, cleaning unit and vaporizing unit. For the preparation of the pellet boring, the fuel rod is set into the rod chamber and CO_2 gas is injected into the chamber. The rod chamber is surrounded by the Dewar vessel filled with LN_2 to cool down the CO_2 gas (Step1). Therefore, the CO_2, which is spread into the pellet-clad gap and also the cracks inside of the irradiated pellet, is frozen as the dry ice and keeps the pellet position and shape away from their collapses during the boring. And next, the center of the fixed pellet is bored by the drill bit with the frequent cleaning of the turnings. The steps of the boring (Step2) and the cleaning (Step3) are repeated alternately until the boring depth reaches at 40mm. After these steps, the molybdenum sleeve is inserted into the hole to keep its shape, and the dry ice is vaporized by heating up with monitoring the humidity and CO_2 concentration in the vapored gas (Step4). The processing is completed after the humidity and CO_2 concentration are decreased enough. This is one of the best processing ways to bore a hole to the irradiated fuel pellets which have many cracks inside.

At the RFEF in Tokai R&D Center, this type of the boring device has been improved and installed to supply the test fuels for the power ramp tests at the JMTR.

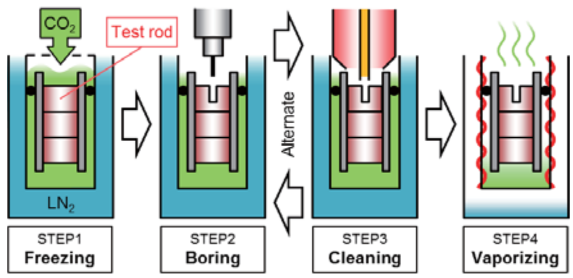

Fig. 2 Processing steps of center boring device

2 IMPROVEMENTS

The preliminary tests were performed with the prototype device and the dummy pellets. The test results indicated that the boring performance and the reliability of the device could be upgraded.

2.1 Reduction of boring time

For this boring device, one of the most important points to be considered is how to prevent the breakage of the drill bit. Once the breakage occurred, it is difficult to pick out the broken bit (or tip) from the bored hole and that pellet cannot be used anymore for the test rod.

The diameter of the drill bit is decided as φ2.5mm for the insertion of the φ2.3mm molybdenum sleeve. To bear against the rotation torque with this thin drill bit, the cylindrical drill bit with brazed chip was selected for the prototype device as shown in Fig. 3 (upper image). The slit was processed at the top of the straight shank and the diamond tip was brazed into that slit. However, the bored hole is the blind and that drill bit didn't have the function to eject the turnings due to its shape. The turnings remained at the bottom of the bored hole and they blocked the boring itself. Therefore the frequent cleanings of the turnings were necessary with the frequent interruptions of the boring. As a result, it took over 20 hours to bore a single hole. Additionally, the diamond tip at the bit top was sometimes broken off from the shank and remained inside of the hole. It could be estimated that the turnings remaining increased the frictional resistance during the boring and the slit part at the bit top, which is weaker than the shank part, cannot bear the rotation torque.

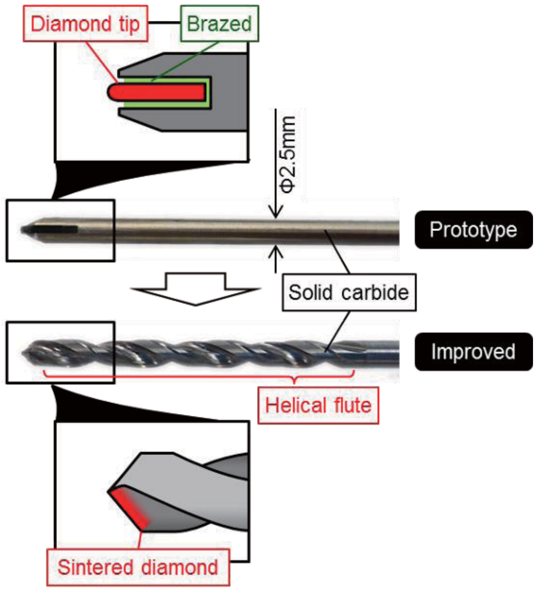

Fig. 3 Comparison of drill bits

To improve these drawbacks, the twist drill bit with diamond sintered head was selected shown as Fig. 3 (lower image). The helical flutes of the drill bit functions as the turnings outlet due to the drill rotation so that the turnings are automatically ejected through these flutes without the interruption of the boring. And the cutting edge at the bit top is sintered to the shank as solid so that it has the enough strength to bear against the rotation torque. Therefore, it is expected that the new drill bit can reduce the boring time dramatically without the breakage of the drill bit.

2.2 Turnings collection

As described above, most of the turnings are ejected automatically from the bored hole. Meanwhile the ejected turnings were piled up beside to the bored hole placed at the top side of the pellet. These turnings should be collected to keep the device and the hot cell away from the contamination as possible. Therefore, the dedicated turnings cleaner was designed as shown in Fig. 4.

The duplex pipe is employed on this cleaner. The dry air from the inner pipe blows the turnings up and the blown turnings are suctioned into the outer pipe at the same moment. The cleaner is inserted into the rod chamber instead of the drill head and sealed hermetically to prevent the leak of the blown turnings. The suctioned turnings are caught by the subsequent filter of 5μm mesh and collected after the boring. Additionally, the air used for the blowing is dehydrated to dry up the turnings and to prevent the turnings attachments inside of the surface of blowing pipe.

Its collection rate indicated in the preliminary test is over 95%.

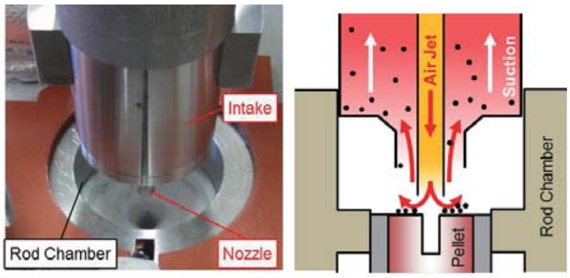

Fig. 4 Turnings cleaner

2.3 Reduction of frictional heat

Another improvement was applied for the loading system of the drill head to upgrade the processing accuracy.

As is the same as other boring or cutting devices, the frictional heat is generated by the contact of the drill bit with the pellet during the boring. Especially for this boring device, the frictional heat is one of the serious problems, because it may cause the dissolve of the dry ice and the unfixing of the pellet (and the fuel rod). The previous device applied the constant load on the drill bit during the boring, and this load was continuously applied until the next cleaning. It means that the longer boring time makes the unfixing risk higher.

For the improvement of the unfixing risk, the inching motion was added to the head loading system, which moves the drill bit slightly up and down. The inching motion can save the contact time in the same manner as the generating of the frictional heat.

2.4 Defrosting device

As described in chapter 1, the sample rods were held with the dry ice frozen by LN_2. During the preliminary tests, a mass of the frost was observed on the Dewar lid as shown in Fig. 5 (upper-left image), due to the temperature difference between R.T. and the surface of the Dewar vessel. That frost blocked the maneuvering of the drill loading systems and made it difficult to bore the hole correctly. Similarly, the LN_2 level detector was also frosted so that it led its malfunction due to the ice clogged.

To avoid the frost, the dedicated heater was equipped on the Dewar lid and the LN_2 level detector as shown in Fig. 5 (lower-right image). The shape of the Dewar lid is quite complicated and the space remaining under the maneuvering area is quite tight. And it is also needed to prevent the conflict with the drill bit. Therefore, the rubber heater was hired according to its flexibility, thickness and cost. With this improvement, no condensation was observed during the processing and it made possible to ensure the correct maneuvering of the driving system for the accurate boring.

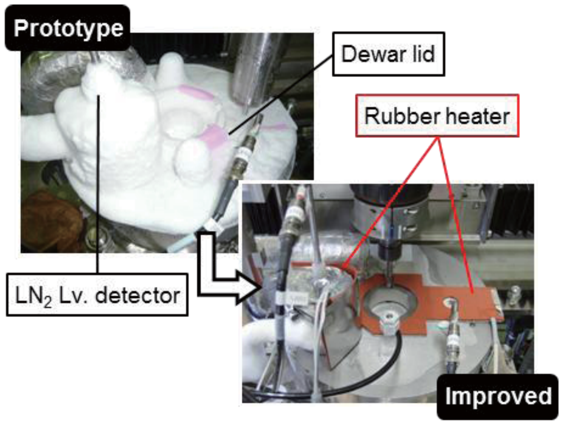

Fig. 5 Rubber heater on the Dewar lid

2.5 Other improvements

Fig. 6 shows the outside view of the improved boring device.

The most of the boring processes, such as the cooling, boring, turning removal, sleeve insertion and un-freezing, are automated for the easy operations by the master-slave manipulators.

The upper unit, which includes the processing unit and the turning cleaner, can be disconnected from the bench of the hot cell by the remote operation, and the vacant space can be furnished as the flat surface with the cover plate. It is beneficial for the easy maintenance and decontamination of itself, and also for the efficient utilization of the hot cell.

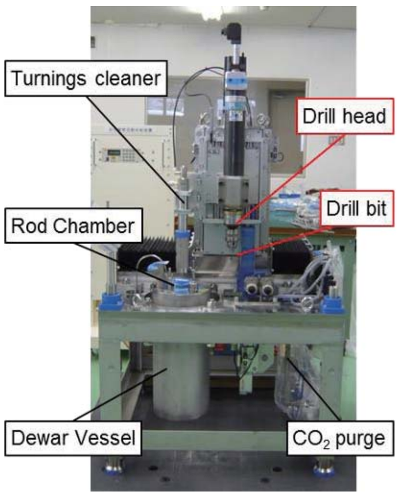

Fig. 6 Improved center boring device

3 CONCLUSIONS

The center boring device for the irradiated fuel pellet was improved to fabricate the fuel rod for the power ramp test. Based on the preliminary test with the prototype device, several improvements were applied to upgrade its boring performance and reliability.

It is expected that the new drill bit can reduce the boring time dramatically without the breakage, and the inching motion and the rubber heater will ensure its processing reliability with the enough boring accuracy. Additionally, the other improvements will be beneficial for the remote operation, and the utilization of the hot cell.

The mock-up test will be performed with the dummy pellets again to confirm the efficiency of the improvements. After this confirmation, the improved device will be installed into the hot cell to confirm its applicability for the irradiated fuels.

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[1] M. Shimizu et al, Development of Thermocouple Re-instrumentation Technique for Irradiated Fuel Rod, **JAERI-Tech 95-037**, (1995)

10.3 DEVELOPMENT OF HIGH RESOLUTION X-RAY CT TECHNIQUE FOR IRRADIATED FUEL ASSEMBLY

AKIHIRO ISHIMI, KOZO KATSUYAMA, KOJI MAEDA AND TAKEO ASAGA Author's

Japan Atomic Energy Agency

High X-ray CT technique was developed to observe the irradiation performance of FBR fuel assembly and MOX fuel. In this technique, the high energy X-ray pulse (12MeV) was used synchronizing detection system with the X-ray pulse to reduce the effect of the gamma ray emissions from the irradiated fuel assembly. In this study, this technique was upgraded to obtain high resolution X-ray CT image. In this upgrading, the collimator which had slit width of 0.1 mm and X-ray detector of a highly sensitive silicon semiconductor detector (100 channels) was introduced in the X-ray CT system. As a result of these developments, high resolution X-ray CT images could be obtained on the transverse cross section of irradiated fuel assembly.

Keywords: FBR MOX Fuel, irradiated fuel assembly, X-ray CT, Post irradiation examination, high resolution X-ray CT image

1. INTRODUCTION

The high energy X-ray computer tomography (X-ray CT) technique, widely used in the medical field, is one of the most powerful non-destructive test tools for characterizing an inner structure. In order to measure the fuel pin position in the irradiated Fast Breeder Reactor (FBR) fuel assembly, the X-ray CT apparatus has been developed at June, 1999 in O-arai research and development center of JAEA. In this X-ray CT system, the 12 MeV X-ray pulse was used in synchronization with the switch on of the detector to minimize the effects of the gamma ray emissions from the irradiated fuel assembly. Conventional post irradiation examinations (PIEs) give results on fuel rod performance after dismantling the fuel assembly, but X-ray CT technology has a great merit of being able to examine the fuel rods performance in the intact fuel assembly [1-5].

The X-ray CT technique has been upgraded in order to increase the resolution of the X-ray CT images obtained. This paper describes the development of high resolution X-ray CT technique and PIE results of the irradiated fuel assembly.

2. Development of X-ray CT system

2.1 X-ray CT apparatus

Figure 1 shows a cut-away view of the X-ray CT apparatus [1]. The fuel assembly is examined by the traverse and rotating X-ray scans. In this X-ray CT system, the 12 MeV X-ray pulse was used in synchronization with the switch on of the X-ray detector to minimize the effects of the gamma ray emissions from the irradiated fuel assembly.

A collimator made of tungsten was installed in front of the X-ray detector to shield the gamma ray from the irradiated fuel assembly.

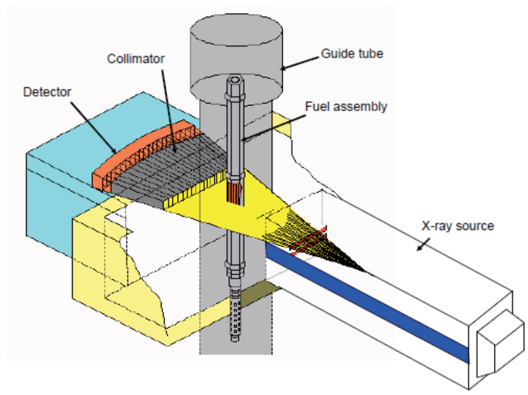

Fig.1 Cut-away view of X-ray CT apparatus

2.2 Details of upgraded X-ray CT apparatus

Figure 2 summarizes the upgrading made for the X-ray transmission process from the source to the X-ray detector, passing through the specimen and collimator.

This X-ray CT apparatus was upgraded to obtain high resolution X-ray CT image of the irradiated fuel assembly. In this upgrading, the width of the collimator slit was decreased from 0.3 to 0.1 mm. In addition, a highly sensitive silicon semiconductor detector (100 channels) was used in place of a $CdWO_4$ scintillator (30 channels).

In order to increase X-ray intensity on the detectors and improve the resolution of X-ray CT images, the X-ray beam shape was changed from a circular shape to an elliptical one which was the similar as the collimator slit shape.

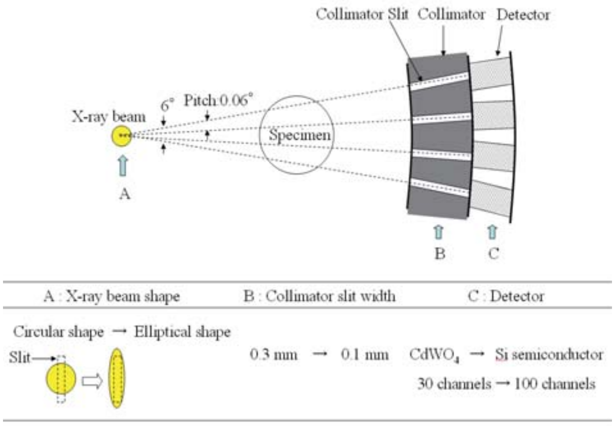

Fig.2 Improvements made for the X-ray CT apparatus

3. Results and discussion

3.1 Specimens of the irradiated FBR MOX fuel assembly subjected to X-ray CT

The X-ray CT images were taken on cross-sections of the irradiated fuel assembly using the upgraded X-ray CT apparatus. Specifications of this fuel assembly are listed in Table 1. After taking the X-ray images, the fuel assembly was dismantled and three fuel rods were subjected to metallographic observations. The burnups of these fuel rods became higher when approaching the reactor core.

Table 1 Fuel assembly specification

Items	Assembly
Total length	2772 mm
Outer face to face length of duct tube	78.5 mm
Number of fuel rods	127
Outer diameter of cladding	5.5 mm
Outer diameter of MOX fuel pellet	4.63 mm
Pellet density	94 %T.D.
Burnup (average)	53.5 GWd/t
Linear heat rating (maximum)	401 W/cm

3.3 Application for irradiated fuel assembly

Figure 3 shows X-ray CT images of the fuel assembly irradiated to 53.5 GWd/t. Figure 3 was taken on the transverse cross-section at the highest neutron flux positions by the upgraded apparatuses. The high resolution X-ray technique made it possible to obtain high resolution X-ray CT image of the irradiated fuel assembly.

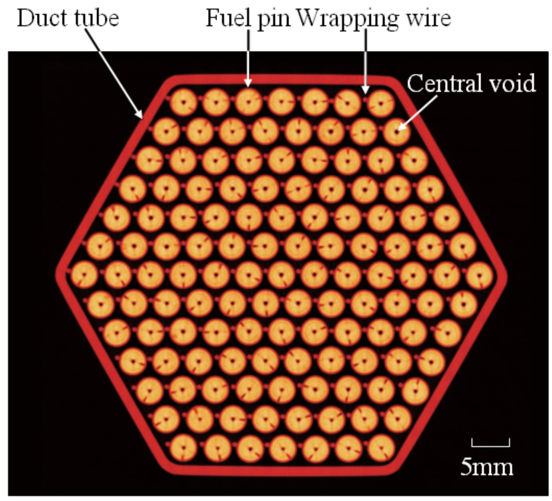

Fig.3 X-ray CT image the irradiated fuel assembly

The central voids can be clearly observed in the X-ray CT image obtained after upgrading. Figure 4 reproduces enlarged X-ray CT images taken on the transverse cross-sections of a fuel rod located at the center of the fuel assembly by the upgraded apparatuses. In the enlarged X-ray CT image of fuel pin, the wrapping wire, cladding, central void, and cracks are distinctly observed in X-ray CT image taken by the upgraded apparatus.

The X-ray CT image of the rod was analyzed to elucidate the density change in the fuel by the CT

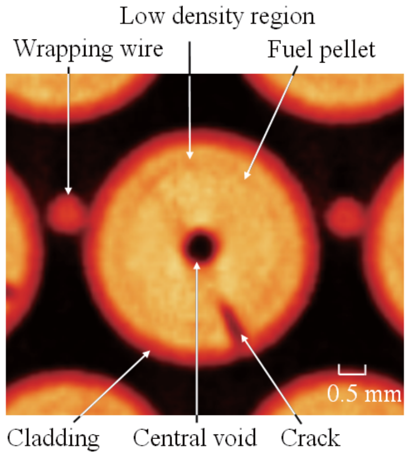

Fig.4 Enlarged X-ray CT image of the fuel pin

image analysis code. The image was taken on the transverse cross-section at an axial position, 250 mm from the bottom of the fuel column. Figure 5 (a) shows a metallographic image. Figure 5 (b) shows an X-ray CT image after the density change was analyzed on the X-ray CT image. The low density region (undisturbed region) and high density region (equi-axial + columnar regions) are discriminated from each other.

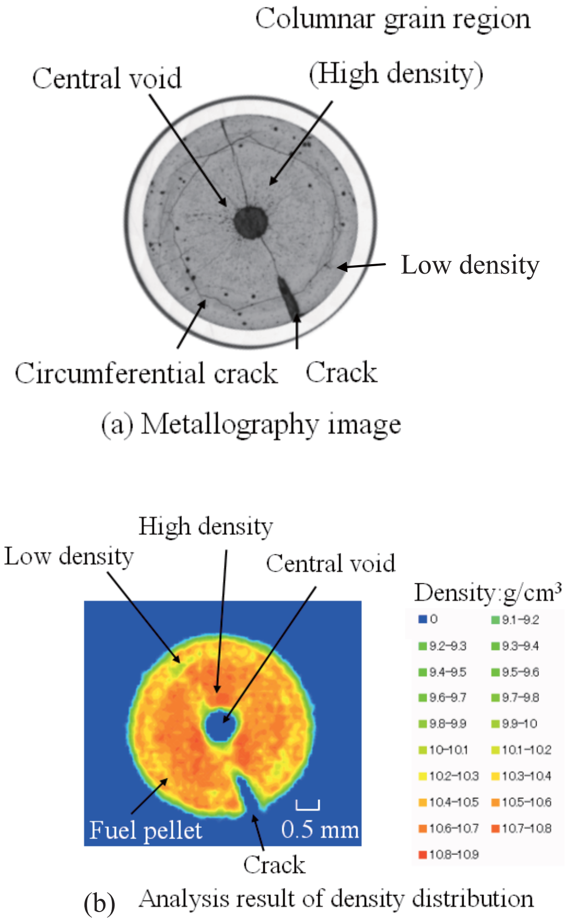

Fig.5 Metallography image of fuel pellet and analysis result of density distribution

3.3 Three-dimensional X-ray CT image

Figure 6 shows three-dimensional X-ray CT image of the irradiated fuel assembly in the core region. Three-dimensional X-ray CT image is composed of the transverse cross-section image. The surface condition of wrapper tube and deformation of all fuel pins can be observed from this three-dimensional X-ray CT image. Furthermore it is possible to observe from any angle of fuel assembly.

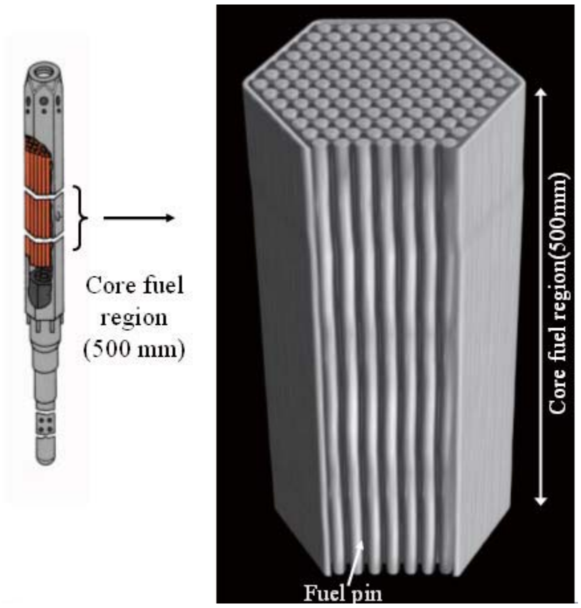

Fig.6 Three-dimensional X-ray CT image of fuel assembly

4. Conclusion

The high resolution X-ray CT technology developed by JAEA was upgraded to examine irradiation behavior of FBR MOX fuel assembly. To obtain the high resolution X-ray CT image, the number of X-ray detectors has been increased from 30 to 100 channels, and the collimator equipped with 100 slits (0.1 mm wide, 2 mm long and 230 mm deep) was utilized. Additionally, the detector material has been changed from $CdWO_4$ to a semiconducting material made of silicon alloy. These improvements made it possible to obtain the high resolution X-ray CT image of the irradiated fuel assembly.

This technique was applied to a fuel assembly irradiated in the experimental fast reactor "JOYO," and the high resolution X-ray CT images of the fuel pellet were successfully obtained. The fuel pellets, central voids in them, claddings and wrapping wires can be clearly distinguished from each other. In the case of the fast reactor fuel, the central void is formed because of the influence of a steep temperature gradient in the radial direction of a pellet. Until now, the central void size has been measured by metallurgical observation on a cross section of the fuel after sectioning it. This image enables us to observe the irradiation behavior of fuel pellets such as a central void size while still in the fuel pins. And a large number of data can be obtained in a short time and applied to the analyses of fuel performance, which leads to improvement in the safety analyses of fuel assemblies and fuel pins.

This technique using the X-ray CT can eject destructive methods in many measurements. In addition, this technique also can be applied further to observation of high density and high radioactivity specimens, such as high-level radioactive wastes, activated equipment used in nuclear reactor and so on.

ACKNOWLEDGMENTS

This paper contains some results obtained within the task “Study of the irradiation behavior of the fuel pellets using the using the high resolution X-ray CT technique” entrusted from the Ministry of Education, Culture, Sports, Science and Technology of Japan.

Gratitude is due to S. Misawa and H. Kubo for their assistance in conducting the study.

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11. Development of Advanced Driver Fuel

4th International Symposium on Material Testing Reactors, Oarai, Japan, December 5-9, 2011

11.1 CERCA industrial approach to customers' needs

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AREVA-CERCA has developed more than 70 fuel and targets designs over the last 50 years. Strong of this experience and the support of a company such as AREVA, AREVA-CERCA has implemented industrial approaches that benefit to the customers.

Some of these industrial approaches will be reviewed to emphasis the benefits in term product development, risk management for projects and regular productions, and quality management to reach customer satisfaction.

This presentation will also cover some latest developments about UMo tests and current policy about spent fuel reprocessing.

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CERCA industrial approach to customer's needs

UMo developments in CERCA

D. Geslin, B. Stepnik, C. Jarousse
CERCA-AREVA,
ISMTR conference, Oarai, Japan, 05-09/12/2011
Rev 0

AREVA

Table of content

1. 12 years of engagement (1/2)

- CERCA start UMo developments in 1999
- A dedicated R&D team
- Strong partnership and common projects with European research centers and research reactors (CEA, ILL, SCK-CEN, TUM)
- World wide partnerships with experts, researchers and reactors

1. 12 years of engagement (2/2)

- More than 100 prototypes realized in all the UMo area:
  - Dispersed / Monolithic UMo
  - Atomized, ground, chemical UMo powder
  - Plane and curve plates
  - Assembly (BR2 UMo element)
  - UMo plates irradiated in 8 irradiation experiments (UMUS, IRIS 1, 2, 3, 4, IRIS-TUM, FUTURE, E-FUTURE)
- About 50 paper contributions published in international conferences

2. Customer oriented approach (1/2)

- From close collaboration between CERCA and reactors experts joint development programs have been built in order to answer to the reactor objectives and constraints:
  - Fuel performances (time cycle, burn-up, power, design …),
  - Minimize the reactor impact (design, cost …),
  - Schedule (HEU driven),
  - Quality Assurance,
  - Commercial

2. Customer oriented approach (2/2)

Two main programs:

- LEONIDAS program:
  - Common program between CEA, SCK-CEN, ILL, USA DoE and CERCA
  - Dispersion UMo fuel program
  - Generic fuel qualification for high performance reactors (RJH, RHF, BR2)

- ALPS program:
  - Common program between TUM and CERCA
  - Dispersion and monolithic UMo program
  - Fuel validation for FRMII reactor

Figure 2: UMo foil manufactured by hot and cold roll processing - 350 µm-

3. Industrial approach (1/4)

- Each year CERCA produces thousands of fuel plates and hundreds of fuel elements.

3. Industrial approach (2/4)

- Each development step is realized with that industrial production in mind:
  - **Design:**
    - The product design is the customer responsibility
    - CERCA interact and advises its customer from the very beginning
    - Industrial feasibility, manufacturing performances, fuel QA inspection criteria
  - **Prototype:**
    - Using production equipment
    - Demonstrate the technical feasibility in an industrial environment
    - Address the technical, safety, QA improvements needed

3. Industrial approach (3/4)

- Each development step is realized with that industrial production in mind:
  - **Development and validation:**
    - Using tools and production equipment in industrial configuration
    - Using improvement techniques (lean 6σ) to fix all the parameters
    - Realizing complete QA inspections (NDT, DT) and measurement campaign
  - **Qualification:**
    - One complete production batch
    - Statistic of the manufacturing performance reached
    - Yield characterizations

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3. Industrial approach (4/4)

- Some examples:
  - Dispersed UMo fuel more adapted to an industrial approach than monolithic one
    - UMo dispersed fuel preferably
    - UMo monolithic fuel also investigated as a back up
  - Si content in UMo fuel plate tested by industrial feasibility:
    - First, Si blended in the Al matrix
    - Second, Si alloyed in the Al matrix
    - Third, Si coated on the UMo powder
  - Powder production:
    - Prototype powder realized in CERCA (atomized, ground, chemical) to learn
    - Assessment of the existing available powder technology planned
    - Technology transfer of an UMo powder process planned

Binocular picture

of atomized UMo powder

Binocular picture

of ground UMo powder

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4. Global approach (1/2)

- AREVA can address the global UMo fuel cycle
- U material transports world wide
- UMo fuel manufacturing:
  - UMo powder (in a few years)
  - UMo plate (flat, curve, …): dispersed and monolithic fuel
  - UMo assembly (all designs)

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4. Global approach (2/2)

- Reprocessing of UMo fresh fuel
  - Successful tests realized on UMo fuel without Si content
  - Test on UMo fuel with Si content to be planned
- Reprocessing of UMo irradiated fuel
  - Can be addressed by AREVA NC in La Hague

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5. Main results obtained (1/2)

- UMo dispersed fuel : LEONIDAS program
  - Manufacturing of high density UMo dispersed fuel demonstrated:
    - U7Mo, 8 gU/cm3, respect of all the QA inspection criteria
    - UMo without Si: industrialization flat and curve plates
    - UMo with Si: prototype realized up to 12% Si
  - Performance of the UMo dispersed fuel:
    - E-FUTURE plate irradiation experiment
    - U7Mo + 4-6% Si, 8 gU/cm^3, 450 W/cm^2
    - No fuel failure observed
    - No blister observed up to 65% burn up
    - Van den Berghe et al. 2011, Charolais et al 2011

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5. Main results obtained (2/2)

- UMo monolithic fuel: ALPS program
  - Development of a CERCA process to bond monolithic UMo foils
  - Manufacturing of monolithic UMo mini-plate demonstrated

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6. UMo program in progress

- UMo powder
  - UMo powder atomization prototype equipment under construction
  - Assessment of the UMo powder technology available in order to build an industrial capacity in CERCA
- UMo dispersed fuel: LEONIDAS program
  - E-FUTURE II plate irradiation experiment in progress (irradiated in 2012)
  - BR2 MIXED ELEMENT irradiation experiment planned (irradiated in 2014)
- UMo monolithic fuel: ALPS program
  - Mini-plates to be irradiated in RERTR-11A in progress (irradiation in 2012)
  - Full-size plates manufacturing in development

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7. Conclusion

- Key factors for success:
  - Set the Goal to achieve by and with customers
  - Customers and partners network
  - Industrial experience, tools and methods
- Results will be an industrial solution for reliable supply
- Development is an adventure but can also be short and efficient (LEU targets…)

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ANY QUESTION ?

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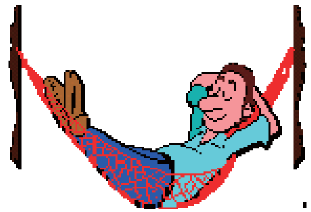
THANK YOU



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11.2 Development of U-Mo/Al Dispersion Fuel for Research Reactors

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Currently, the KOMO-5 irradiation test for full size U-Mo/Al dispersion fuel rods has been underway since May 23, 2011. The purpose of the KOMO-5 test includes an investigation of the irradiation behaviors of silicide or nitride coated U-7Mo/Al(-Si) dispersion fuels and the effects of pre-formed interaction layers on U-Mo particles. It is expected that the irradiation test will be finished after attaining 60 at% U-235 burnup in May 2012, and the first PIE results of the KOMO-5 will be obtained in September 2012. In addition, an international cooperation program on the qualification of U-Mo dispersion fuels for small and medium size research reactors is going to be proposed in cooperation with the IAEA. Conversion from silicide fuel to U-Mo fuel will increase the cycle length with a smaller number of fuel assemblies and allow more flexible back-end options for spent fuel due to of the reprocessibility of U-Mo.

Keywords: Research Reactor, Dispersion Fuel, Atomization, U-Mo Alloys, Irradiation Tests, Post-irradiation Examination.

1. INTRODUCTION

The Reduced Enrichment for Research and Test Reactors (RERTR) program was launched by the USA in 1978 in order to convert HEU research reactor fuels to LEU fuels by developing high uranium density fuel alloys. While many research reactors have been converted by adopting LEU U_3Si_2 fuel, some high power research reactors still requires advanced fuel with higher uranium densities than uranium silicides[1]. As the global R&D activities has concluded that gamma phase U-Mo alloys are promising candidates to be used as advanced high uranium density fuel for the high power reactors, efficient production of U-Mo powder became a major challenge for the fabrication of U-Mo dispersion fuel. A centrifugal atomization process of uranium alloys and compounds developed in KAERI has been considered an economical way to produce gamma phase U-Mo powder with a low loss and low contamination when compared to other possible methods such as grinding or hydriding-dehydriding methods[2].

However, severe pore formation attributed to an extensive interaction between the U-Mo and Al matrix, although an irradiation performance of U-Mo itself showed most stable, tackled a use of U-Mo fuel for high performance research reactors[3]. Because the reaction products, i.e. UAl_x, is less dense than the mixed reactants, the volume of the fuel meat increases after formation of interaction layer(IL). In addition to the impact on the swelling performance, the reaction layers between the U-Mo and Al matrix induces a degradation of the thermal properties of the U-Mo/Al dispersion fuels[4].

Since 2000, KAERI has devoted much attention to qualify rod type U-Mo fuel with U-densities of 5-6 gU/cc by applying atomized U-Mo powder process, named as the KOMO test, for multi-purpose reason, 1) upgrading HANARO research reactor to get more compact core, 2) solving the back-end option of spent fuel, and 3) scientific contribution to understand U-Mo fuel performance. Post-irradiation test in the Irradiated Materials Examination Facility (IMEF) showed the variation of fuel swelling, interaction layer thicknesses and oxide thicknesses with fuel design parameters and irradiation conditions. The KOMO irradiation tests at HANARO have contributed to understand the effects of U-Mo particle size, U-Mo volume fraction, irradiation temperatures, and Si content in the Al matrix.

2. Results of the KOMO-4 irradiation test

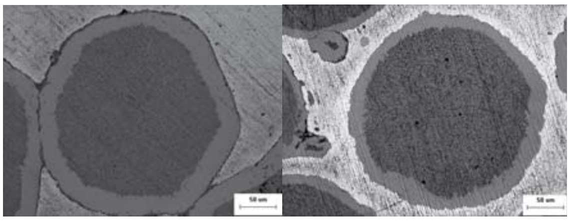
According to the previous KOMO-1,2,3 tests, the formation of IL in rod type U-Mo/Al dispersion fuel with U-loadings up to 5~6 gU/cc is very severe and inevitable due to the high temperature irradiation condition. Although a use of large-sized U-Mo particles in dispersion fuel showed better performance by reducing the interfacial area between U-Mo and Al, this could not be a permanent remedy to solve the interaction problem. Instead, small addition of Si up to 2wt% into Al matrix and a use of ternary U-7Mo-1Zr gave us positive sign to solve intrinsic nature of U-Mo fuel[5].

KOMO-4 irradiation test of rod type U-Mo-X(X=Zr,Ti)/Al-xSi(x=2,5,8wt%) dispersion fuel with 5.0 g-U/cc U loadings was designed, by reflecting the previous KOMO-1,2,3 irradiation results, in order to investigate the power level that can be allowed, the optimum Si content in matrix and the effects of alloying element addition(Ti, Zr) on U-Mo with a combination of Al-Si matrix at a high temperature condition of ~200°C. Several U-7Mo/Al dispersion fuels with different fuel particle sizes (105~210 μm, 210~300 μm, and 300~425 μm) were also included to compare the IL growth behavior.

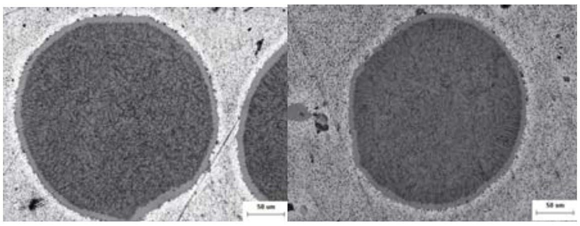
The KOMO-4 irradiation test bundle was loaded at OR-3 hole of the HANARO reactor on 22 Dec. 2008 and was discharged on 03 Jan. 2010 after 132.1 EFPD (Effective Full Power Day) irradiation[6]. The KOMO-4 test assembly kept its directionality toward the HANARO core position during irradiation time in order to evaluate the power history of each fuel precisely.

When the IL thicknesses with the Si content in the Al matrix were compared, higher Si content in the matrix reduces IL growth further up to 8% Si due to higher temperatures as shown in Fig. 1 and 2. However, the higher temperatures of the KOMO-4 test than the RERTR tests produced thicker ILs as compared in Fig. 3[7]. Therefore higher Si contents were needed to suppress the IL growth in high temperature irradiation conditions. Fig. 4 shows that calculated temperature histories of U-Mo/Al-Si fuels with fuel burnup. The average fuel temperature of the KOMO-4 test is around 200°C, which is much higher than 120°C of the RERTR-6 irradiation test conditions.

Si was accumulated preferentially at the IL/matrix interface as shown in Fig. 5 and more Si was observed in the higher Si content added dispersion fuel rod. The Al/U ratios were less than 3 for all Si added dispersion fuel rods, while the ratios were more than 4 for the dispersion fuel rods without Si. Reduction of the population of Si precipitates in the recoil zone around the U-Mo particle is observed. Ti and Zr additions to U-Mo also reduce the IL growth additionally, consistent with the lower temperature RERTR-8 test results.



(a) U-7Mo/Al (b) U-7Mo/Al-2Si



(c) U-7Mo/Al-5Si (d) U-7Mo/Al-8Si

Fig. 1. Microstructures of KOMO-4 irradiated U-Mo/Al-Si fuel. (~50% BU)

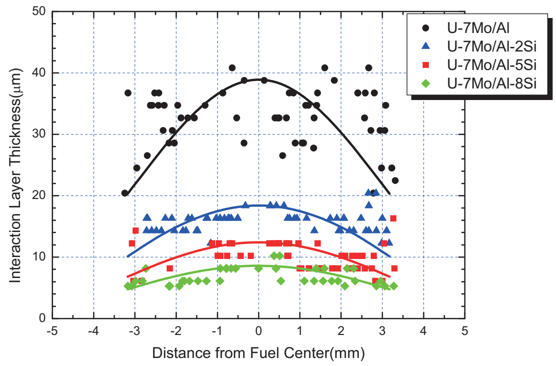


Fig. 2. IL thickness distribution of irradiated U-Mo/Al-Si samples.(~50% BU)

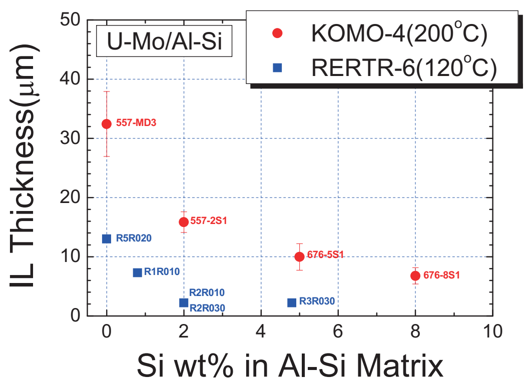


Fig. 3. Comparison of IL thickness between the KOMO-4 and RERTR-6 tests.

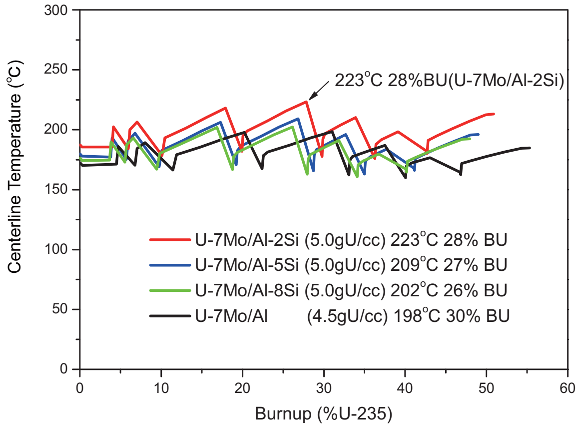


Fig. 4. Calculated temperature history of U-Mo/Al-Si fuels vs. BU.

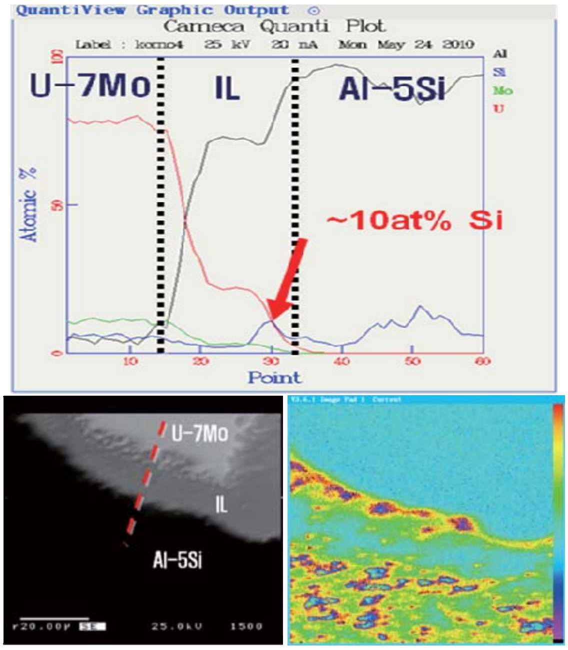


Fig. 5. EPMA scanning and Si X-ray mapping on IL of irradiated U-7Mo/Al-5Si fuel. (49%BU)

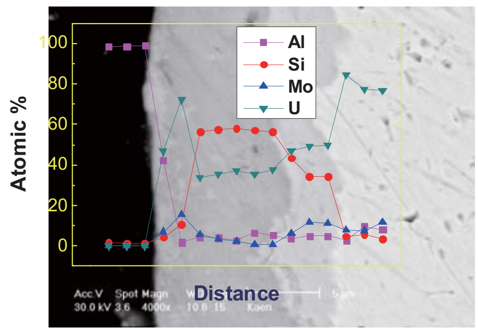
Highly accumulated Si in the IL (average ~40% Si) during pre-irradiation heating was diluted to ~15% Si uniformly during irradiation as the IL grew. Contrary to the observations from out-of-pile tests, the pre-irradiation heated sample showed no improved retardation of the IL growth when compared to Si added dispersion fuel rods. One difference is the uniform distribution of Si throughout the IL in the pre-irradiation heated sample.

From the results of the KOMO-4 test, it is found that 5% of Si in the matrix, which has been derived from the out-of-pile diffusion tests for the minimum Si content for stabilization of the IL, might be inapplicable for in-pile tests[8].

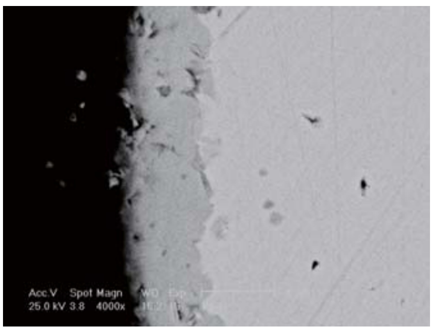
3. KOMO-5 Irradiation Test

The KOMO-5 irradiation test was designed by reflecting the PIE results of the KOMO-4 irradiation test. Since it appeared that Al-5Si is sufficient to suppress the IL growth during irradiation at high temperature and U-Mo alloy modification by adding small amount of Zr or Ti is effective, main consideration of the KOMO-5 fuel design is :

1) U-Mo/Al-Si and U-Mo-X/Al-Si(X=Ti or Zr) dispersion fuel
   - 5 gU/cc with full size fuel (700 mm in length)
   - 5 wt%Si content in matrix
   - Burnable absorbers are included
   - U-Mo particle size less than 150 μm
   - Irradiation temperature (>200°C)
2) Silicide or nitride coated U-Mo/Al and U-Mo/Al-Si dispersion fuel
   - 5 gU/cc with mini-sized fuel (200 mm in length)
   - Less than 5 wt%Si content in matrix
   - Use of silicide or nitride coated U-Mo particles
   - Variable irradiation power condition
3) Target BU : 60% in average



(a)



(b)

Fig. 6. SEM micrographs of (a) a silicide coated U-Mo particle and (b) a nitride coated U-Mo particle.

Silicide or nitride coating technologies have been developed in KAERI in order to reduce the interaction between the U-Mo and Al matrix during irradiation[9,10]. The silicide or nitride coating will act as diffusion barrier layer on the U-Mo particle and out-of-pile diffusion tests showed improved diffusion barrier performances of the silicide and nitride layers. Fig. 6 shows the microstructure of a silicide coated U-Mo particle and a nitride coated U-Mo particle. .

Metallographic cross sections of fresh dispersion fuel rods for the KOMO-5 irradiation test were collected in Fig. 7. Each dispersion fuel rod shows uniform distribution of U-Mo particles in the Al(-Si) matrix after hot extrusion of dispersion fuel meats. Fig. 8 confirms the uniform distribution of each element in the dispersion fuel by EDS elemental mapping. Even when small amount of burnable absorber such as B_4C was added to the Al(-Si) matrix, the distribution of the burnable absorber powder, which can be identified by boron mapping, is very uniform. In the KOMO-5 test, CdO, Gd_2O_3 and B_4C were added to the full size fuel rods as burnable absorbers. Fig. 9 shows the irradiation test fuel assembly containing 12 dispersion sample rods for the KOMO-5 test. The assembly with a special end plug will also maintain its directionality toward the HANARO core position during irradiation.

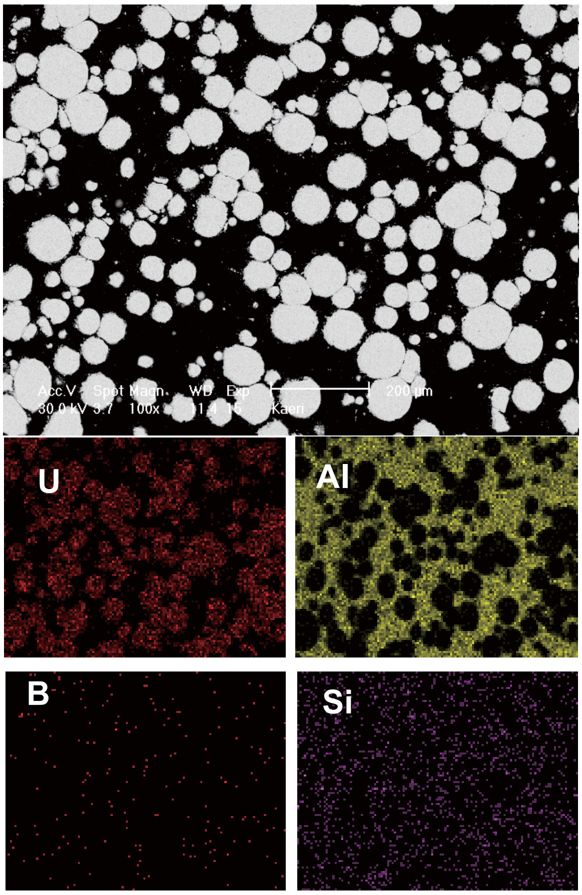


Fig. 8. EDS elemental mapping of burnable absorber added U-7Mo/Al-5Si.

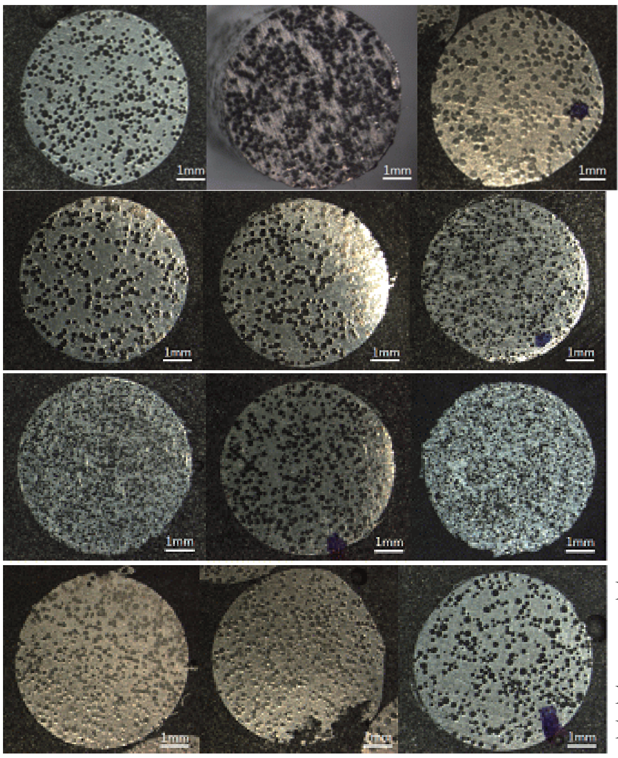


Fig. 7. Metallographic cross sections of fresh dispersion fuels for the KOMO-5 irradiation test.

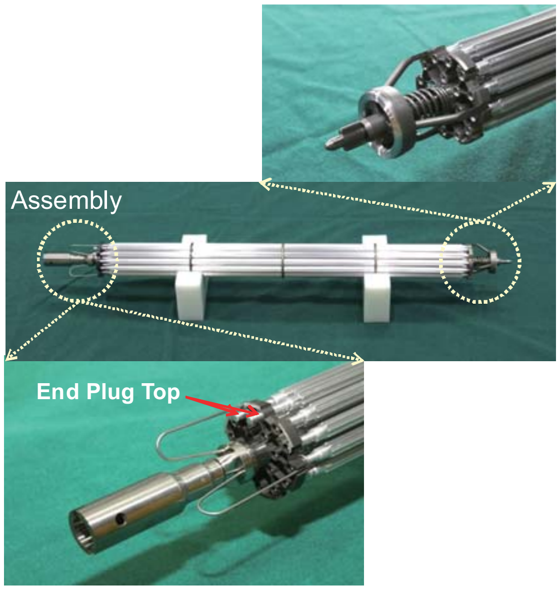


Fig. 9. KOMO-5 irradiation test fuel assembly

The KOMO-5 irradiation test is underway since May 23, 2011. Intermediate visual inspections on the KOMO-5 irradiated fuels revealed a sound fuel surface without any break-away swelling, as shown in Fig. 10[11]. Average BU was calculated to be 16.5

at.% U-235 as of Oct. 4. 2011. It is expected that the irradiation test will be finished after attaining 60 at% U-235 burnup in May 2012 and the first PIE results of the KOMO-5 will be obtained in September 2012.

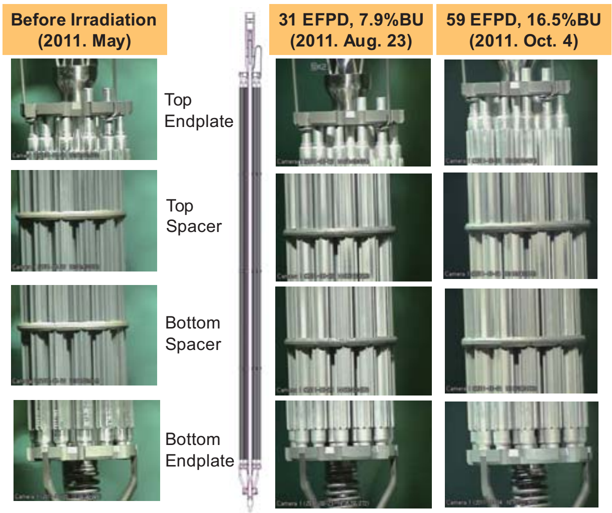


Fig. 10. Visual inspection of KOMO-5 test.

5. JOINT QUALIFICATION OF U-MO FUEL

While many small and medium flux research reactors are waiting for the qualification of U-Mo dispersion fuel for various reasons, major international R&D efforts are given to the qualification of very high uranium density U-Mo fuel including monolithic U-Mo fuel because the conversion of high flux research reactors using HEU has been the first priority under the Reduced Enrichment of Reseearch and Test Reactors (RERTR) program. Expected benefits of the conversion of silicide fuel to U-Mo dispersion fuel would be the flexible back-end options, extended fuel life cycle, reduced spent fuel volume, etc. However, delayed qualification of monolithic U-Mo fuel and very high uranium density U-Mo fuel hinders the use of U-Mo dispersion fuel in small and medium flux research reactors. If an international irradiation and qualification program for U-Mo dispersion fuel for small and medium flux research reactor launches, the commercialization of U-Mo dispersion fuel would be expedited and then, future conversion of many HEU research reactors in Russia and Eastern Europe would be accelerated.

Many countries have expressed interest in participating in an international irradiation and qualification programme, including the ROK and other fabricators, for dispersion U-Mo fuel.

It has been generally accepted that the programme would focus on U-Mo dispersion fuel fabricated for small and medium flux research reactor, would use a qualification approach similar to the one used successfully for silicide fuel as reported in NUREG 1313 and would be coordinated by the IAEA.

Given the potential commercial interests of various fuel fabricators, the IAEA is well suited to design and implement an open international programme in a neutral manner, intended to ensure that no particular commercial/industrial firm receives any special information/benefits. Information/results produced by the international programme should be available to all potential suppliers.

Possible approaches to actual implementation of the joint program should be discussed with potential stakeholders on the following topics.

- The feasibility and timelines of initiating an international irradiation and qualification program for dispersion U-Mo fuel and dispersion U-Mo fuel fabricators, coordinated by the IAEA;
- The approach to be used for the program, especially if it will be similar to NUREG 1313 or will use a different methodology;
- Possible irradiation reactors and irradiation conditions including the representativeness of the irradiation conditions for the reactors that are likely to be converted with the dispersion U-Mo to be qualified;
- Potential participants in the international program, especially fuel fabricators and developers;
- The type of fuel to be irradiated under the international program, including possible geometries and fuel densities;
- The scope of post-irradiation examination;
- The back-end of the irradiation experiments including taking back the U-Mo fuels to the US under the FRRSNF
- The organization of the international program, especially on the composition of the program's scientific supervisory group; and
- Financial aspects of the international program, especially funding sources.

It is expected that this international program will result in an action plan on the international irradiation/qualification program of dispersion U-Mo fuel.

ACKNOWLEDGMENT

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[11] H.J.Ryu et al. RERTR-2011, Santiago, Chile, Oct. 23-27, 2011.

12. Nuclear HRD for Next Generation, Other

4th International Symposium on Material Testing Reactors, Oarai, Japan, December 5-9, 2011

12.1 ACTIVITY OF THE NUCLEAR HUMAN RESOURCE DEVELOPMENT CENTER IN JAPAN ATOMIC ENERGY AGENCY

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The Nuclear Human Resource Development Center of the Japan Atomic Energy Agency (NuHRDeC/JAEA) has continuously and comprehensively conducted a variety of education and training courses in the nuclear field over a half century since its establishment(1958). The main recent activities are 1)Education and training for domestic nuclear engineers , 2)Education and training for foreign nuclear engineers, 3) Cooperation with domestic universities for nuclear education, 4) Cooperation with international organization for nuclear human resource development, and 5)Central role as secretariat for Japan Nuclear Human Resource Development Network (JN-HRD Net).

A variety of training courses has been conducted for education and training for domestic nuclear engineers, such as Nuclear Reactor General Course, Radiation Basic Course, Radiation Safety Management Course etc. The characteristic feature is that the curricula places emphasis on the practical exercise with well-equipped training facilities, experimental laboratories and the expertise of lecturers mostly from JAEA.

Instructor Training Program (ITP) has been launched to contribute to education and training for foreign nuclear engineers since 1996. The purpose of ITP is to develop a self-sustainable instructors in Asian countries. The main specialized fields of the ITP are reactor engineering, environmental monitoring, nuclear emergency preparedness etc. As a part of the ITP, Plant Safety Training Course, Administration Training Courses etc. are also conducted .

We have cooperated with many universities in Japan, such as University of Tokyo, Tokyo Institute of Technology and Ibaraki University. Currently, the JAEA has cooperation agreements with 21 universities including a college. We have especially-close cooperation with the Nuclear Professional School of the University of Tokyo, dispatching totally more than 150 staff for lectures and experiments. Japan Nuclear Education Network is a special framework for the real-time distance learning system connecting six universities and the JAEA.

As an international cooperation in Asia, we collaborate with HRD Project of Forum for Nuclear Cooperation (FNCA). Currently the project focuses on Asian Nuclear Training and Education Program (ANTEP). We also contribute to IAEA Asian Nuclear Safety Network (ANSN) by providing information in nuclear education and training.

JN-HRD Network was established on November 19, 2010 among Japanese industries, government, institutes and universities in order to conduct and promote various domestic and international HRD activities in strategic and integrated manner, effectively and efficiently. We play an important role in hub as Secretariat of Japan-HRD Network, in cooperation with Japan Atomic Industry Forum.

ACTIVITY OF THE NUCLEAR HUMAN RESOURCE DEVELOPMENT CENTER IN JAEA

Dec. 8th., 2011

K.Yamashita, H.Murakami, N.Nakagawa, N.Arai, K.Matsuura, Y.Ohzeki

Nuclear Human Resource Development Center (**NuHRDeC**)
Japan Atomic Energy Agency (JAEA)

1

Features of Nuclear HRD

NuHRDeC is a major HRD center in Japan that conducts technological lectures and exercises for engineers, students, etc. in the fields of nuclear energy and radiation application.

Nuclear Human Resource Development Center (NuHRDeC)

Lecture at NuHRDeC

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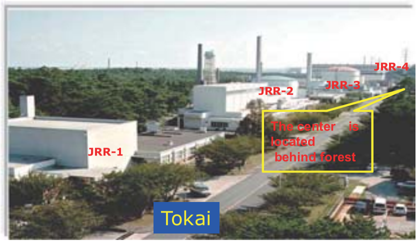
Nuclear Facilities for Training and Visiting

- ◆Various types of nuclear facilities and experimental arrangements
  - Research Reactors (JRR-3, JRR-4, JMTR, HTTR, JOYO etc.)
  - Quantum Beam Facilities (Japan Proton Accelerator Research Complex [J-PARC], etc.)
  - Handling facilities for Radioactive materials (Hot Cell)
- ◆Lecturers in well-experienced in Nuclear HRD
  - In about 20
  - In JAEA about 500
- ◆53 years educational experiences and accumulated know-how



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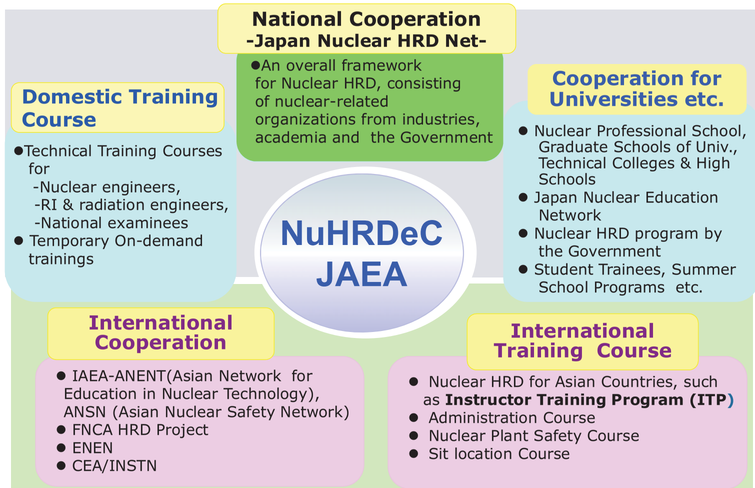
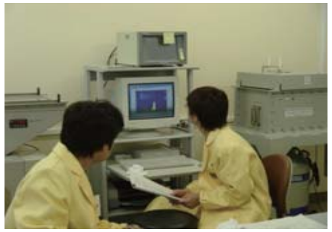
Outline of JAEA-NuHRDeC Activities



Domestic Training Courses

NuHRDeC currently provides three regular training courses:

1. **Reactor Engineering**
2. **Radio Isotope & Radiation**
3. **National Examinees**



Technical Exercises at NuHRDeC Laboratories 3

Cooperation with Universities

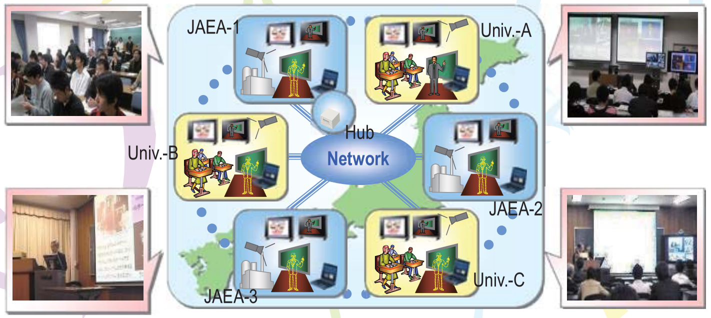
- Cooperation with Universities under Comprehensive Agreements with JAEA (7 Universities: such as Univ. of Tokyo and Ibaraki Univ.)
- Cooperation for Graduate School Program (20 Univ. and college)
- Japan Nuclear Education Network (JNEN : 6 Univ.)
- Contribution to Nuclear HRD Program by Ministry of Education, Culture, Science and Technology(MEXT) & Ministry of International Trade and Industry (MITI)

Handling of nuclear fuels

Remote education system in JNEN

6

Japan Nuclear Education Network (JNEN)



JNEN is a real-time, interactive, remote-education system, initiated in 2007, connecting JAEA with 6 universities. Some laboratory exercises at JAEA are also provided in JNEN.

7

International Contribution

International Training for Asian countries (1/2)

- **Instructor Training Program (ITP)** (since 1998)
  - **Training courses:**
    Radiation Protection, Reactor Engineering, Emergency Preparedness, etc
  - **Indonesia, Thailand, Vietnam, Bangladesh, Malaysia, Philippines, Kazakhstan**
  - **Totally about 100 lecturer-candidates trained**
  - **Totally about 1700 participants in each country's training course (FTC)**

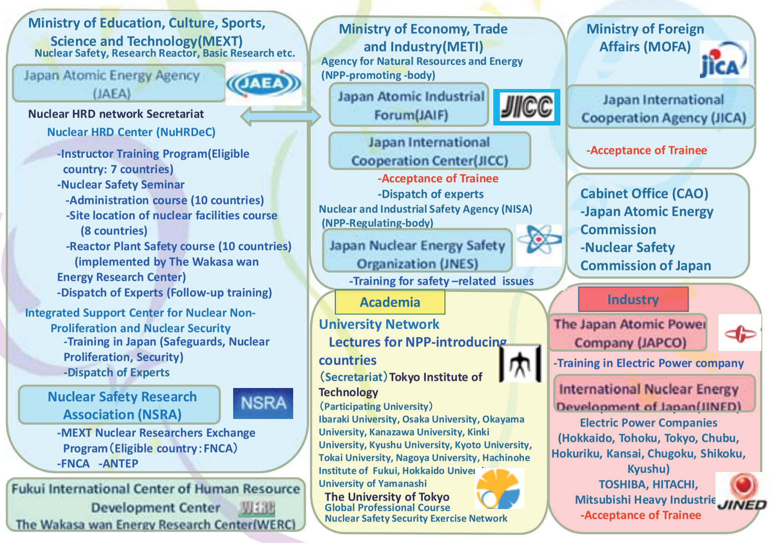
Reactor Engineering Course in Vietnam

Emergency Preparedness Course in Thailand

Reactor Engineering Course in JRR-1

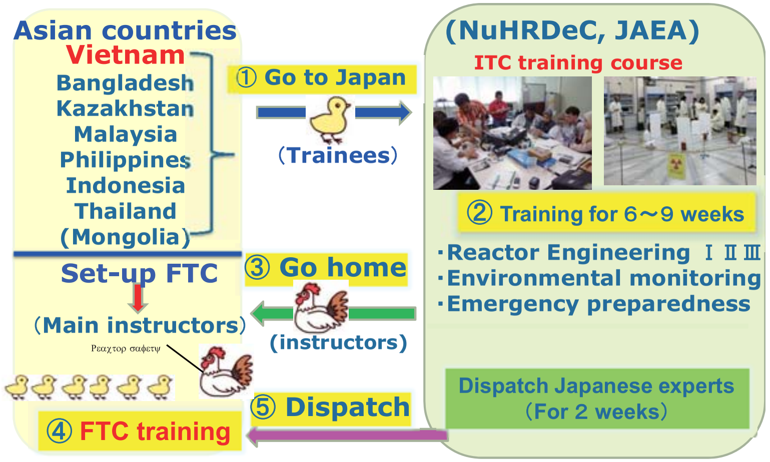
7

Japan' Nuclear Human Resource Development network supporting Asian countries



Instructor Training Program(ITC)

Purpose:
ITP is a training course to level up teaching ability of instructors from Asian countries , who will be instructors of Follow-up Training Course (FTC) in own countries.



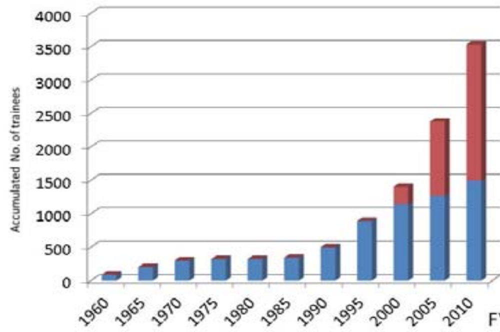
International Training for Asian countries (2/2)

- **Nuclear Plant Safety Course**
  - Training for nuclear power plant safety using PWR, BWR and FBR
  - Totally about 40 participants since 2007
- **Safeguards Training**(Jointly with IAEA)
  Conducted by "Integrated Support Center for Nuclear Nonproliferation and Nuclear Security" since 2011 FY
  - Training for safeguards and accountancy
  - Totally about 190 participants since 1996
- **Administration Course**
  - Training for administrative officers such as energy planning, regulator
  - New course from 2010
- **Siting Course**
  - Training for site planning-related personnel
  - Planning new course from 2011

Administration Course at JAEA

7

Accumulated Number of Trainees of NuHRDeC's International Training Courses



UNESCO Course at NuHRDeC in 1958

Our center has contributed to human resource development in Asian countries since 1958, using Japan's knowledge and experiences in peaceful uses of nuclear energy,

International cooperation

1. Cooperation with IAEA
- ●ANENT (Asian Network for Education in Nuclear Technology)
  Japan participated in ANENT in Dec. 2010.
- ●ANSN (Asian Nuclear Safety Network)
  JAEA provides teaching materials and related information.

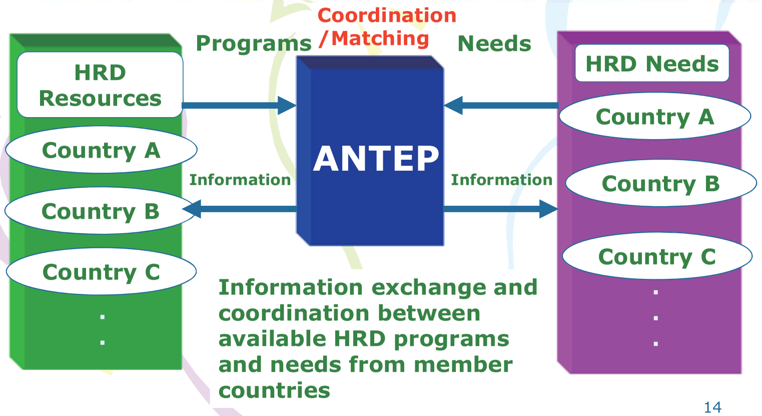
2. FNCA Related Activities
- ●HRD Project
  Focusing on ANTEP (Asia Nuclear Training and Education Program) aiming at network system for information exchange on needs and educational programs by utilizing existing nuclear training and education resources.

FNCA 2011 Workshop on HRD Vietnam

FNCA 2010 Workshop on HRD Seoul, Korea

8

Concept and Function of ANTEP (Asian Nuclear Training and Education Program)



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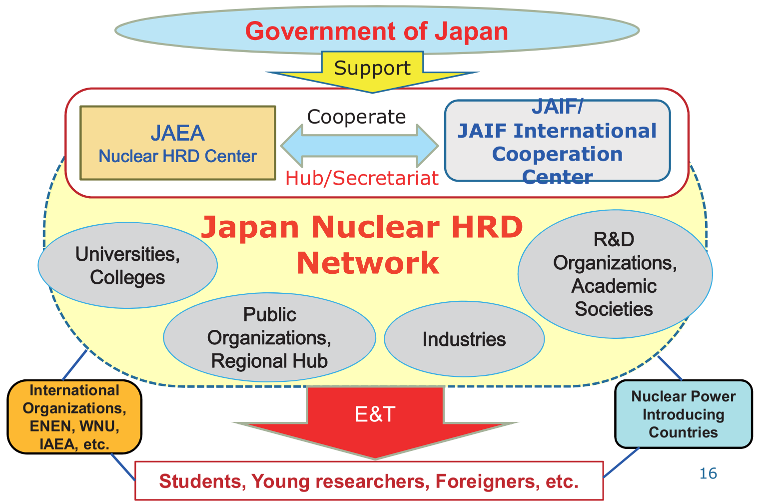
Main activities of JN-HRD Net

- ➢ Establishment of close collaboration with national and international HRD related organizations.
- ➢ Support to participant organization to J-HRD Network
- ➢ National and international public information on HRD
- ➢ Establishment of international network
- ➢ Planning and operation of inter-organ HRD activities
- ➢ Promotion of support and cooperation mostly for nuclear power introducing countries

JN-HRD Net Establishment Event on November 19, 2010

Japan Nuclear HRD Network



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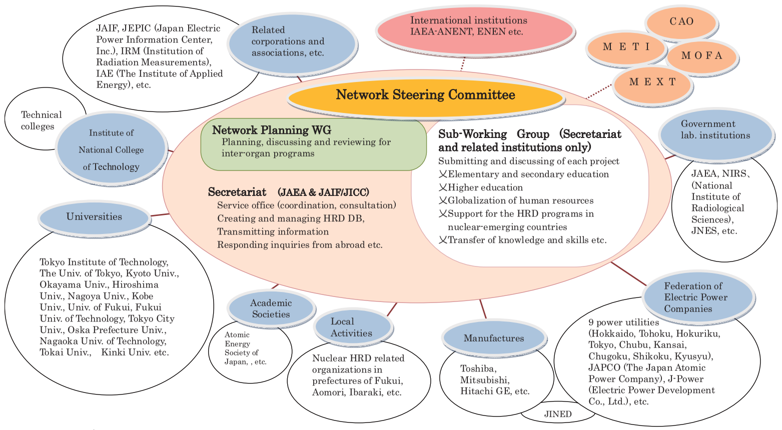
Organization of JN-HRD Net



Obligation 1) Providing information regarding nuclear HRD to this network
Obligation 2) Appointing contact person

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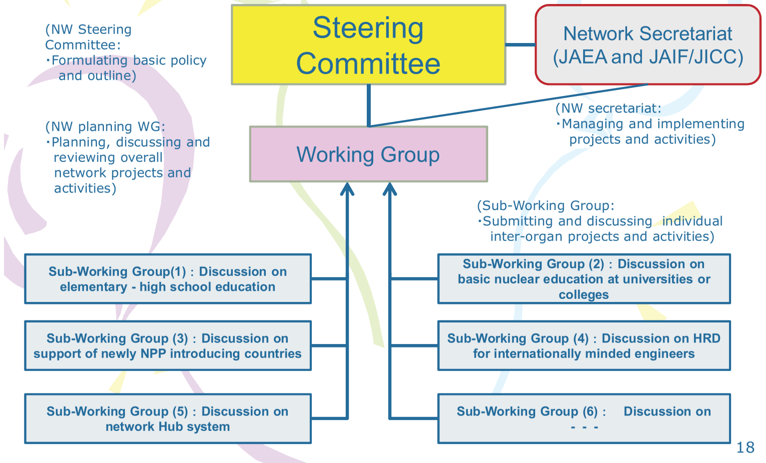
Framework of JN-HRD Net



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Secretariat of JN-HRD net

- NuHRDeC/JAEA
- JICC/JAIF

NuHRDeC: Nuclear Human Resource Development Center
JAEA: Japan Atomic Energy Agency
JAIF: Japan Atomic Industrial Forum, Inc.
JICC: JAIF International Cooperation Center

1. Collect information on nuclear HRD activities of Japan and abroad.
2. Assemble a data base system on nuclear HRD.
3. Support activities discussed and planned by sub-working group.
4. Spread information on the JN-HRD-NET and related activities (as Public Relations and Public Information).
5. Play a role of contact office for inquiries about nuclear HRD from organizations of Japan and abroad.

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Contact

If you have any questions or requests of support in the field of nuclear HRD, please contact us below.

Secretariat

For nuclear HRD in general:
Japan Nuclear HRD Network Office
E-mail: jn-hrd.net@jaea.go.jp

For NPP related HRD from outside Japan:
JAIF International Cooperation Center (JICC)
E-mail: info@jaif-icc.com

TEL: +81-3-3592-2185 FAX: +81-3-3592-2185
URL: //nutec.jaea.go.jp/network/index.htm

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Summary

National Cooperation
-Japan Nuclear HRD Net-

- An overall framework for Nuclear HRD, consisting of nuclear-related organizations from industries, academia and the Government

Domestic Training Course

- Technical Training Courses for
  - -Nuclear engineers,
  - -RI & radiation engineers,
  - -National examinees
- Temporary On-demand trainings

NuHRDeC JAEA

Cooperation for Universities etc.

- Nuclear Professional School, Graduate Schools of Univ., Technical Colleges & High Schools
- Japan Nuclear Education Network
- Nuclear HRD program by the Government
- Student Trainees, Summer School Programs etc.

International Cooperation

- IAEA-ANENT(Asian Network for Education in Nuclear Technology), ANSN (Asian Nuclear Safety Network)
- FNCA HRD Project
- ENEN
- CEA/INSTN

International Training Course

- Nuclear HRD for Asian Countries, such as **Instructor Training Program (ITP)**
- Administration Course
- Nuclear Plant Safety Course
- Siting Course

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Additional Information

12.2 TRAINING PROGRAM FOR STUDENTS AND YOUNG ENGINEERS IN JMTR

N. Takemoto, H. Izumo, N. Hori, E. Ishitsuka and M. Suzuki

Neutron Irradiation and Testing Reactor Center,
Oarai Research and Development Center, Japan Atomic Energy Agency (JAEA)

The JMTR is expected to be a key infrastructure to contribute the nuclear Human Resource Development (HRD) by a research and On-Job-Training (OJT) in order to support global expansion of nuclear power industry. The training program for Asian young researchers and engineers were started from JFY 2011 in JAEA, and ten trainees from Kazakhstan and Thailand had attended in this program in JFY 2011. In addition, in the nuclear HRD initiative program sponsored by the MEXT, the training course was newly established for domestic students and young engineers from JFY 2010 to JFY 2012. In this course, basic understanding on irradiation test and post irradiation examination is aimed to achieve by overall and practical training such as the neutronic/thermal designs of irradiation capsule, post irradiation examination, measurement and evaluation of neutron fluence, etc. using the JMTR and the related facilities. The 1st training course was held with 10 trainees in JFY 2010. The 2nd and 3rd training courses were also held with 19 trainees and 16 trainees in JFY 2011. From JFY 2012, two courses will be held in every year, and 20 trainees will be accepted in each course.

Keywords: JMTR, HRD, OJT, Irradiation test, Post irradiation examination

1. INTRODUCTION

In order to support global expansion of nuclear power industry, the nuclear Human Resource Development (HRD) is addressed one of urgent issues because of the lack of nuclear engineers. In this situation, the JMTR [1] is expected to be an important infrastructure to contribute the HRD through activities on researches and OJTs for students, as well as members of industrial and government [2].

It is possible to conduct wide fields of researches and OJTs using the JMTR, one of the largest testing reactors in Asia, and the JMTR-hot-laboratory connected to the JMTR. The HRD programs for researches and OJTs in the JMTR are customized for each person, and are conducted by many HRD fragments such as the visiting researcher program of JAEA, JAEA research fellow, JAEA reactor engineering course, the Ministry of Education, Culture, Sports, Science & Technology in Japan (MEXT) nuclear researchers exchange program, the nuclear HRD initiative program sponsored by the MEXT, etc.

2. POSSIBLE HRD PROGRAM AND RECENT ACTIVITIES USING JMTR

2.1 HRD AREA

HRD programs will be more effective by collaboration of the Nuclear Human Resource Development Center (NuHRDeC), the Neutron Irradiation and Testing Reactor Center (NITRC), etc., in JAEA. The dispatch countries can apply with either window (NuHRDeC and NITRC).

Example of possible training using the JMTR is summarized as follows;

1) Project management
   - Planning and management of refurbishment program
2) Aging management
   - Integrity inspection for reactor systems (non destructive testing method, etc.)
   - Reactor maintenance and service planning management,
3) Reactor management
   - Reactor core, instrument and control of reactor
   - Reactor cooling system, reactor auxiliary system
4) Irradiation facilities management
   - Capsule irradiation facility, shroud irradiation facility, hydraulic rabbit irradiation facility
   - Irradiation test of nuclear fuels and materials, silicone doping facility

5) Post irradiation facility / hot cell management
   - Hot cell maintenance management, irradiated nuclear fuels / materials testing
6) Radioisotope production
   - Radioisotope production for medical and industry
7) Waste management
   - Low level waste, medium level waste, high level waste
8) Legal and regulation procedures
   - Licensing of reactor, refurbishment approval by regulatory, safety analysis
9) Development of advanced irradiation technology
   - Development of advanced instrumentation technology
   - Development of advanced PIE technology
10) Design study of next generation MTR
   - Core Design, thermohydraulic design, etc.

2.2 HRD PROGRAM USING JMTR

Following two programs were carried out with some engineers and researchers visited to the JMTR.

1) OJT program by the refurbishment of reactor facilities and the installation of new irradiation facilities
2) Education and training program by the development of advanced irradiation technology
   - Development of advanced in-situ instrumentation technology
   - ^{99}Mo production technology development by new Mo solution irradiation method
   - Development of recycling technology on used beryllium reflector irradiated by neutron

Recent HRD programs using the JMTR are summarized as follows;

1) Irradiation technology development
   - MEXT fellow, Vietnamese (1 trainee), 2008.10.8-2009.6.26
   - JAEA research fellow, Korean (1 trainee), 2009.10.1-2010.9.30
   - JAEA research fellow, Korean (1 trainee), 2011.12.1-2012.11.30 (on going)
2) Training through JMTR refurbishment
   - JAEA visiting researcher, Malaysian (1 trainee), 2008.11.25-2008.12.19
   - JAEA Reactor engineering course, Malaysian (1 trainee) & Thailand (1 trainee), 2010.10.04-2010.10.15
   - MEXT fellow, Malaysian (1 trainee), 2010.10.20-2011.3.11
3) Training of university students and young engineers
   - Postgraduate student, Kazakhstan (4 trainees), 2010.8.12
   - Training for Asian young researcher and engineer, Kazakhstan (5 trainees) & Thailand (5 trainees), 2011.8.1-2011.8.12
   - Nuclear HRD initiative program sponsored by the MEXT, Japanese
     (10 trainees), 2010.2.14-2010.2.25
     (19 trainees), 2011.8.17-2011.8.30
     (16 trainees), 2011.8.31-2011.9.13

3. TRAINING FOR ASIAN YOUNG RESEARCHERS AND ENGINEERS

From the viewpoint of encouragement of the study on nuclear science/technology for Asian countries, the training program for Asian young researchers and engineers was started from JFY 2011 in the JAEA. The training program contains basic lecture on the nuclear and thermal calculations of irradiation capsule in the JMTR, training by JMTR simplified simulator, training on HTGR technology etc. As an example, the training schedule of 2011 is shown in Table 1. This program was supported by NuHRDeC, NITRC, etc. in the JAEA. All fee was supported by the JAEA.

Ten trainees from Kazakhstan and Thailand had studied in this program in JFY 2011, and all trainees were reported that the program satisfied them. Photograph of lecture in the training is shown in Fig.1.

4. NUCLEAR HRD INITIATIVE PROGRAM SPONSORED BY THE MEXT

In the nuclear HRD initiative program sponsored by the MEXT, a training course using the JMTR and the related facilities was newly organized for domestic students and young engineers from JFY 2010 to JFY 2012. In this course, basic understanding on irradiation test and post irradiation examination is aimed to achieve by overall and practical trainings such as the neutronic/thermal designs of irradiation capsule, post irradiation examination, measurement and evaluation of neutron fluence [3]. concerning the irradiation test using hydraulic rabbit irradiation facility. The hydraulic rabbit irradiation facility is a water loop system to transfer the small sized (150mm length) capsule, so called rabbit, into and take out from the core under reactor operation by the water flow in the loop. This facility is widely utilized mainly for the basic researches and for the production of short-lived radioisotopes. Outline of the training course and the training subjects are shown in Fig.2 and Fig.3.

4.1 FIRST TRINING COURSE

The 1st training course was held on February 14 to 25, 2011, and 10 trainees were studied. In this course, mainly carried out was the training of neutronic design for the irradiation test. Training with compact training simulator [4] was also carried out by the Japan Atomic Power company (JAPC).

In the training of the neutronic design, criticality calculations for simple geometries and practical neutronic calculations for the irradiation test were carried out using Monte Carlo method which has been introduced in the evaluation of irradiation field of the JMTR core by using of MCNP code [5]. Especially in the neutronic calculation for the irradiation test, the aluminum rabbit including a stainless steel specimen was assumed as a calculation target, and fast and thermal neutron flux, neutron spectrum, etc. in the rabbit in the JMTR core were calculated by trainees with changing irradiation test conditions such as structure material of the rabbit, dimension of specimen, irradiation position, irradiation time, etc.

4.2 SECOND AND THIRD TRINING COURSE

In JFY 2011, two training courses, the 2nd and the 3rd, were held. In the 2nd training course on August 17 to 30, 14 students and 5 engineers were studied. In the 3rd training course on August 31 to September 13, 9 students and 7 engineers were studied. In these courses, practical training of thermal design for irradiation test, activation calculation, radiological control, post irradiation examination, evaluation of neutron fluence were added to the 1st training course. As an example, the training schedule and the photograph of trainees in the 2nd course are shown in Table 2 and in Fig.4 as an example.

In the training of thermal design for irradiation test, irradiation temperature of specimens, structure materials, atmosphere, etc. in the rabbit were calculated as a practice using GENGTC code [6] and NISA code [7] used for thermal analysis in the JMTR. In the training of the post irradiation examination, dismantlement of the dummy rabbit, tensile test and Scanning Electron Microscope (SEM) test of the unirradiated specimen, etc. were experienced in the JMTR-hot-laboratory by remote handling with manipulators installed in the hot cells.

In these courses, trainees could understand well the behavior of neutron, irradiation test, post irradiation examination, radiological control etc. in an irradiation test reactor.

4.3 NEXT TRINING COURSES

From JFY 2012, two courses will be held, and 20 trainees will be accepted in each course in the same way as JFY 2011. From JFY 2012, training of simulated operation of nuclear reactor will start using a simulator of irradiation test reactors, which is under fabrication now. After the restart of the JMTR, rabbits including specimens, fluence monitors and melt wires will be irradiated in the JMTR for the training course. After irradiation, post irradiation examinations such as the dismantlement of rabbits, evaluation of irradiated melt wires and fluence monitors, tensile test of irradiated specimens, etc. will be carried out in the JMTR-hot-laboratory in order to utilize irradiation results such as irradiation temperature, neutron flux and tensile property of specimens in training courses.

5. CONCLUSIONS

The training program for Asian young researchers and engineers was started from JFY 2011 in JAEA, and ten trainees from Kazakhstan and Thailand had studied in JFY 2011. This program will be held in every year and 10 to 20 trainees will be accepted in each course.

In addition, in the nuclear HRD initiative program sponsored by the MEXT, training course using the JMTR and the related facilities was newly organized, and 45 domestic students and engineers were studied in the course from JFY 2010 to JFY 2011. From JFY 2012, two courses will be held in every year and 20 trainees will be accepted in each course.

Trainees will be accepted in these training programs for students and young engineers from domestic and abroad, and the JMTR and the related facilities will continue to contribute to the nuclear HRD with applying characteristics of them in order to support global expansion of nuclear power industry.

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[7] M. Tobita and Y. Matsui, Development of capsule design support subprograms for 3-dimensional temperature calculation using FEM Code NISA, JAERI-Tech 2003-042 (2003) (in Japanese).

Table 1 Training schedule for Asian young researchers and engineers (2011)

Date		AM	PM
31 Jul.	SUN	Arrival	
1 Aug.	MON	JAEA-Oarai Enter Procedure	Commemorative Photo, Orientation, Self-introduction, Introduction of JAEA-Oarai, Status of Research Reactor in World, Outline of Conceptual Design of Next Generation Research Reactor
2 Aug.	TUE	Outline of JMTR and Refurbishment	Facility Tour (JMTR and JMTR-HL)
3 Aug.	WED	Basics of Neutronic Calculation	Training of Neutronic Calculation
4 Aug.	THU	Training of Neutronic Calculation	Outline and Lesson Learned from Fukushima Dai-ichi NPP Severe Accident (2011.3.11)
5 Aug.	FRI	Training of Neutronic Calculation	
8 Aug.	MON	Basics of Thermal Calculation	Training of Thermal Calculation
9 Aug.	TUE	Training of Thermal Calculation	
10 Aug.	WED	JMTR Simplified Simulator	Irradiation and PIE Technology
11 Aug.	THU	Safety Evaluation of JMTR	HTGR Technology
12 Aug.	FRI	Facility Tour (HTTR)	Discussion, Questionnaire Survey, Diploma, Commemorative Photo, Other related Procedure
13 Aug.	SAT	Departure	

Table 2 Training schedule for nuclear HRD initiative program (2nd course)

Date		AM	PM
17 Aug.	WED	Orientation, Commemorative Photo, Introduction of JAEA, NITRC	Training with Compact Training Simulator (JAPC)
18 Aug.	THU	Basics of Nuclear Power	Facility Tour of JMTR, Way of Nuclear Power - Lesson Learned from Fukushima Dai-ichi NPP Severe Accident -, Outline of Hydraulic Rabbit Irradiation Facility
19 Aug.	FRI	Theory of Neutronic Calculation	Theory of Neutronic Calculation, Training of Neutronic Calculation
22 Aug.	MON	Training of Neutronic Calculation	
23 Aug.	TUE	Training of Neutronic Calculation	
24 Aug.	WED	Theory of Thermal Calculation	Training of Thermal Calculation
25 Aug.	THU	Activation Calculation	Facility Tour of Irradiation Engineering Building, Radiological Control
26 Aug.	FRI	Outline of Post Irradiation Examination	Training of Post Irradiation Examination
29 Aug.	MON	Training of Post Irradiation Examination	Safety Analysis of JMTR, Outline and Training of Evaluation of Neutron Fluence
30 Aug.	THU	Discussion, Questionnaire Survey, Diploma, Commemorative Photo, Other Related Procedure	-

Fig.1. Photograph of trainees of the training program for Asian young researchers and engineers (2011).

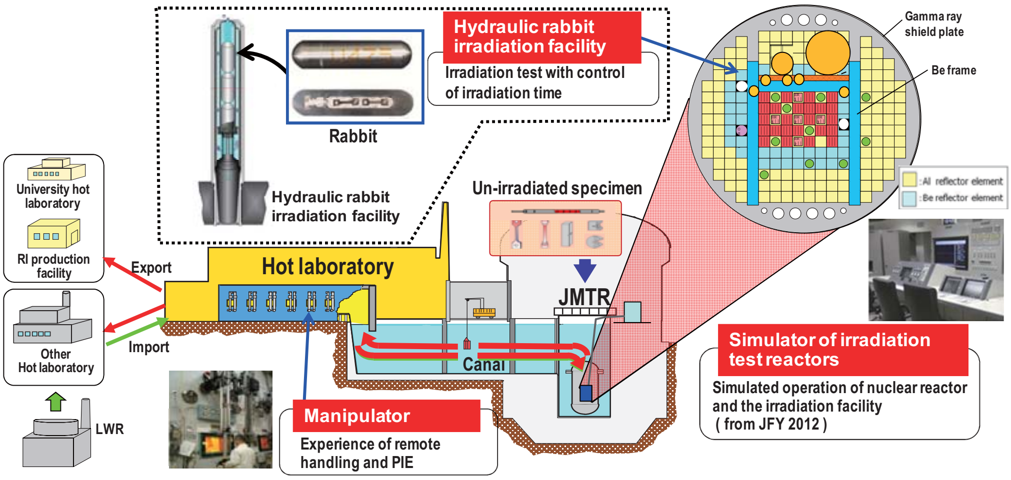

Fig.2. Outline of training course in nuclear HRD initiative program.

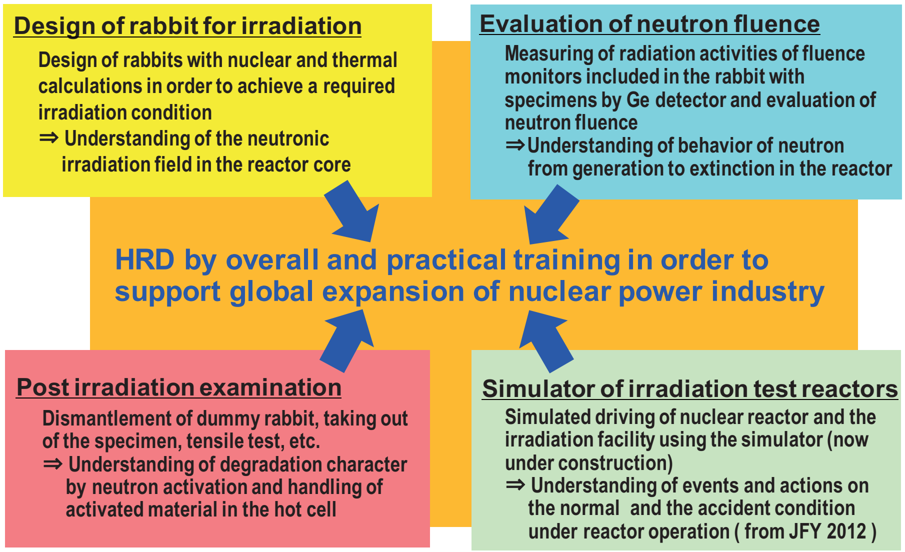

Fig.3. Training subjects in nuclear HRD initiative program.

Fig.4. Photograph of trainees of nuclear HRD initiative program (2nd course).

12.3 Real Time Simulator for Material Testing Reactor

Noriyuki TAKEMOTO[1], Tomomi IMAIZUMI[1], Hironobu IZUMO[1], Naohiko HORI[1], Masahide SUZUKI[1], Tatsuo ISHITSUKA[2] and Kazuo TAMURA[2]

[1]*Neutron Irradiation and Testing Reactor Center,*
Oarai Research and Development Center, Japan Atomic Energy Agency (JAEA)
[2]*Itochu Techno-solution Corporation*

Japan Atomic Energy Agency (JAEA) is now developing a real time simulator for a material testing reactor based on Japan Materials Testing Reactor (JMTR). The simulator treats reactor core system, primary and secondary cooling system, electricity system and irradiation facility systems. Possible simulations are normal reactor operation, unusual transient operation and accidental operation. The developed simulator also contains tool to revise/add facility in it for the future development.

Keywords: simulator, real time, irradiation test, malfunction, reactor operation, station black-out, transient operation

1. INTRODUCTION

Japan Atomic Energy Agency (JAEA) is now developing a real time simulator for a material testing reactor in order to utilize a nuclear human resource development and to promote partnership with developing countries which have a plan to introduce nuclear power plant and/or experimental research reactor.

The simulator is based on Japan Materials Testing Reactor (JMTR) [1], and treats systems/facilities of the JMTR, e.g. reactor, primary cooling system, secondary cooling system, emergency cooling system, instrumentation and control systems, safety and protection systems and electricity system. It can simulate behavior under normal operation condition, transient operation condition such as excess reactivity addition due to irradiation sample insertion and accidental condition such as LOCA. Moreover, it can simulate behavior of irradiation test under irradiation test condition such as material testing under BWR condition.

2. OUTLINE OF JMTR

The JMTR is a tank-in-pool type reactor with thermal power 50MW, and both coolant and moderator are light water. Overview of the JMTR is shown in Fig.1, and the typical core configuration is shown in Fig.2. Typical parameter of the JMTR is summarized in Table 1. The reactor core, which is 1560 mm in diameter and 750 mm in effective height, consists of fuel elements control rods, 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 from within 195 possible positions in the core.

Fig.3 shows outline of primary, secondary and emergency cooling systems. The primary cooling system consists of four main pumps, two emergency pumps, an emergency cooling system and three heat exchangers. The heat removed from the core is transferred to the secondary coolant in the heat exchangers and dissipated into the atmosphere in the cooling tower.

The distinctive characters of the JMTR are summarized as follows.

- One of the high neutron flux materials testing reactor in the world
- Large irradiation area in the core for various irradiation tests
- Flexible reactor core configuration which allows various irradiation facilities installation to the core, etc.

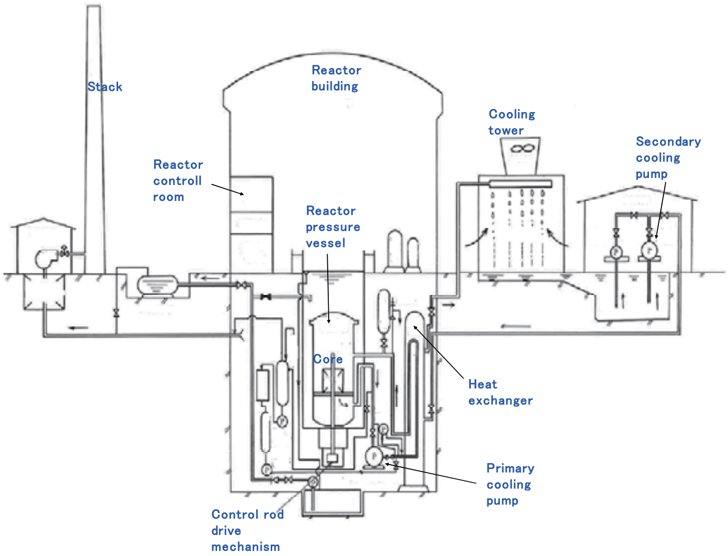

Fig.1 Overview of JMTR

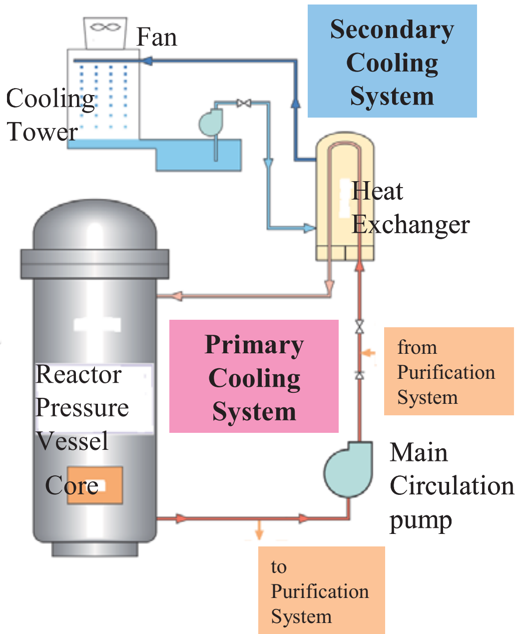

Fig.3 Outline of primary, secondary and emergency cooling systems

Table 1 Typical parameter of JMTR

Reactor	
Thermal output (MW)	50
Neutron flux (max) Thermal ($m^{-2}\cdot s^{-1}$)	$4*10^{18}$
Fast ($m^{-2}\cdot s^{-1}$)	$4*10^{18}$
Effective height (mm)	750
Volume (l)	102
Effective core radius (mm)	416
Type of fuel	plate
Number of standard fuel elements	24
Enrichment of U235 (%)	20

Primary coolant	
Temperature of inlet coolnat (max) (℃)	49
Temperature of outlet coolnat (℃)	56
Flow rate (m^3/h)	6,000
Operation pressure (MPa)	1.5

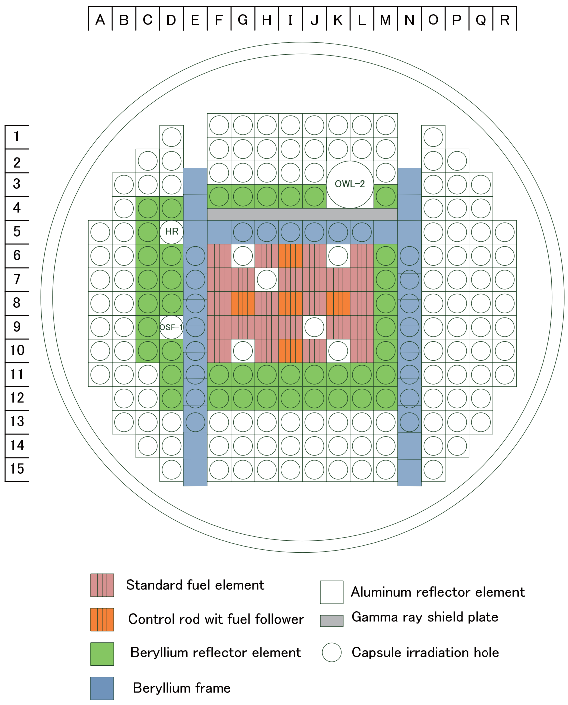

Fig.2 Plane view of core configuration

3. SCOPE OF SIMULATOR

Systems/facilities to be modeled in the simulator are reactor core, primary cooling system including purification system, secondary cooling system, emergency cooling system and irradiation facilities. Electric system is also modeled.

Irradiation facilities to be modeled are following six systems.

1) Capsule irradiation facility with

temperature control by heater and helium gas pressure
2) Capsule irradiation facility with neutron fluence control
3) Advanced water-chemistry controlled irradiation facility
4) Shroud irradiation facility
5) Hydraulic rabbit irradiation facility
6) OWL-2 irradiation facility

Fig.4 shows advanced water-chemistry controlled irradiation facility. Thermo-hydraulic dynamics in cooling system is simulated for each facility. Irradiation material and geometrical dimensions of capsule are determined using capsule design supporting system.

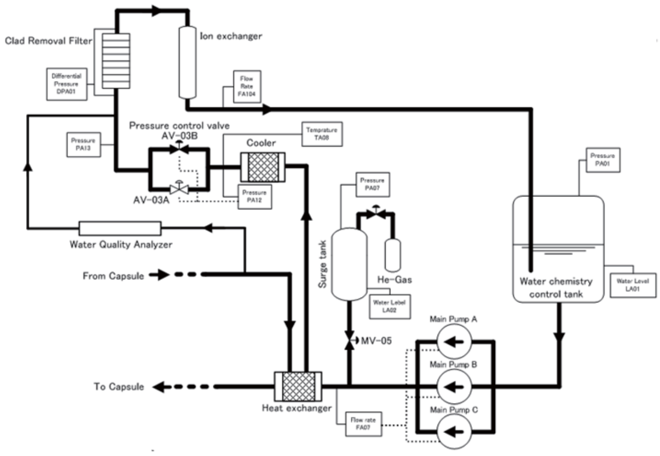

Fig.4 Advanced water-chemistry controlled irradiation facility

Reactor instrumentation and control system are simulated. Fission counters (FC) for start-up and Compensated Ionization chamber (CIC) for log-power and linear power are simulated as reactor instrumentation. Reactor instrumentation for each power has 3 channels respectively and is operated under 2-out-of 3 method. Fig.5 shows outline of instrumentation and control system. Safety actions such as scram caused by unusual condition of the plant are simulated. Malfunctions such as LOCA are also simulated.

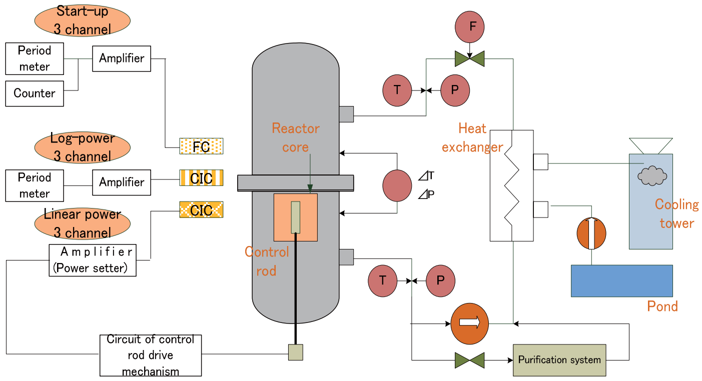

Fig.5 Outline of reactor instrumentation and control system

4. SIMULATOR CONFIGURATION

4.1 Hardware

The simulator is composed of a computation server, a process control computer, an irradiation control computer, an instructor computer and each console. There are three operator stations for the simulator: reactor console, process console and irradiation facility console. The reactor console consists of both hardware panels and of pictures emulating reactor-console-display-graphics. The process console and irradiation console consist of pictures emulating Digital Control System (DCS) graphics and hardware panels. Switches and buttons for reactor operation are arranged on the console. Simulation results are displayed not only at each console computer but also at large LCD. All simulator computers are connected through a LAN network with 100 Mbps. Fig.6 shows network overview.

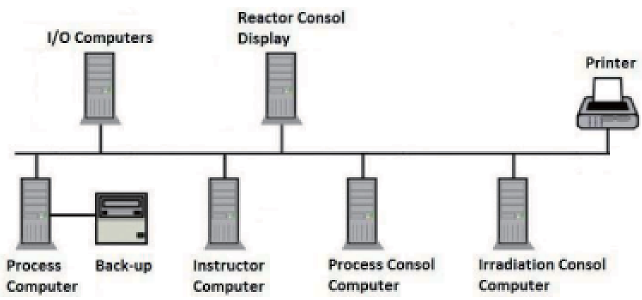

Fig.6 Simulator network overview

4.2 Software/Simulation Model

The simulator is developing now mainly using a simulation developing tool, JADE developed by GSE Inc. Neutronics in the core is calculated by 3 dimensional computer code, REMARK. 4 groups neutron cross section for

REMARK is generated by SRAC code system [2] and JENDL3.3 library [3]. REMARK also calculates decay heat power after shut-down of the reactor.

Thermo-hydraulic dynamics inside of pressure vessel is calculated by 3 dimensional computer code RELAP5-HD, which is a modified version of INL RELAP5-3D [4]. Thermo-hydraulic dynamics outside the reactor pressure vessel is calculated by JTopmeret in the JADE. Fig.7 and Fig.8 show nodalization of the reactor pressure vessel and heat structure design for the fuel plate.

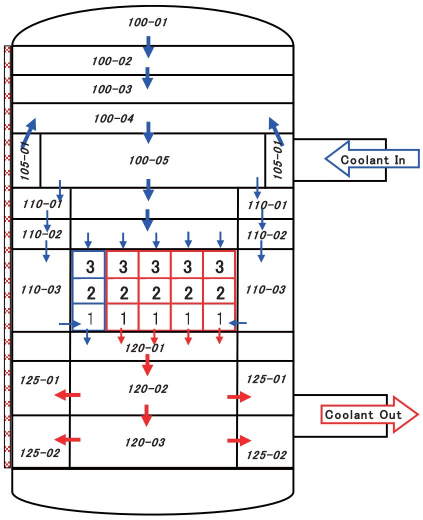

Fig.7 Nodalization in reactor pressure vessel

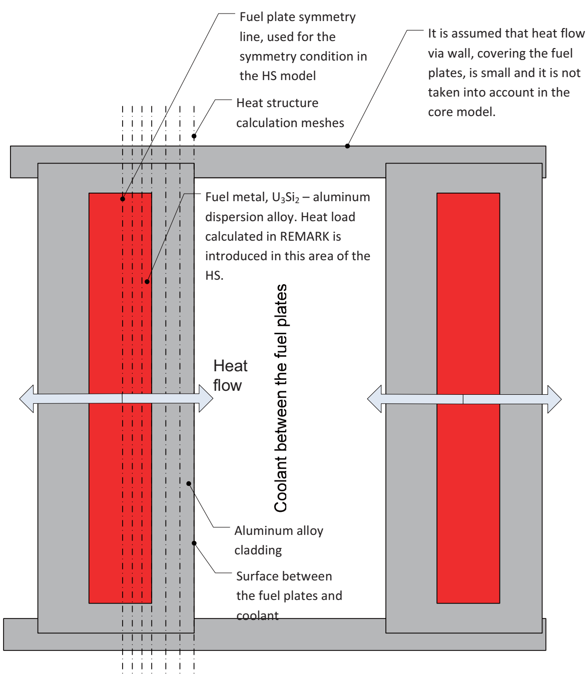

Fig.8 Heat structure design for fuel plate

DCS for instruments such as circulation pumps and various valves, and control of control rods are developed by JControl and/or Jlogic in the JADE. Electric system is developed by JElectric in the JADE.

SimExec, an integrated software system in the JADE is used to support a real time simulation. All modules under the SimExec are synchronized and all variables under the SimExec are used as common variables in all modules. Fig.9 shows a data flow during simulation. Two SimExecs are used for complete modeling of the reactor system. The SimExec 1 is acting as a master and running all modules in the simulator (BOP). SimExec 2, containing RELAP5-HD and REMARK, is used as a server to perform the actual calculations requested by the SimExec 1. Time step in the simulator is aimed at 40Hz.

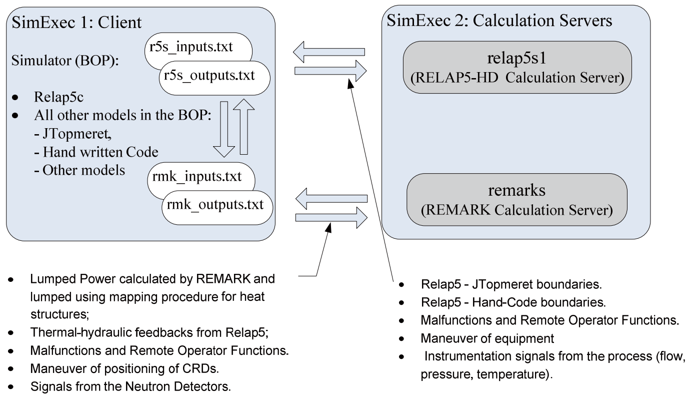

Fig.9 Data flow during simulation

5. OPERATION OF SIMULATOR

5.1 Operation of Reactor

Simulation of reactor operation covers whole operation, start-up, 50MW steady, and shut-down operation according to the JMTR operation procedure. Reactor physics experiments such as measurement of excess reactivity for EOC (End of Cycle) and measurement of effective multiplication factor (k_{eff}) at all control rods insertion and so on, are simulated before start-up of reactor. Reactor power goes up by operation of five control rods using reactor period meter. Automatic operation is executed over 500kW of reactor power and performed by one fine rod corresponding to fluctuation from target reactor power. Compensation operation by control rod is executed at 50MW power corresponding to reactivity fluctuation due to fuel burn-up, insertion of irradiation sample, change of coolant temperature and so on.

5.2 Operation of Irradiation Facility

Operations of insertion/extraction of sample, adjustment of temperature of facility, coolant flow and He-gas pressure are executed. REMARK core calculation takes account of reactivity change due to these operations.

5.3 Instructor Station

The instructor Station provides following functions:

- simulator control functions
  Load and store initial condition
  Control execution (run/freeze/fast/slow)
  Save backtrack snapshots at regular intervals
  Load a stored backtrack snapshot
- Monitoring function
- Create and activating training scenarios
- Training performance review
- Expert mode
- Instructor/Operator actions log

5.4 Malfunction

Typical malfunctions for the simulator are activated by Instructor. Table 2 shows typical malfunctions of the simulator.

Table 2 Typical malfunction

System name	Item
Reactor core/Control rods	Unusual extraction of control rod at start-up condition
Reactor core/Control rods	Unusual extraction of control rod at normal condition
Reactor core/Irradiation equipment	Reactivity addition by irradiation sample
Primary cooling system	Pump trip
Primary cooling system	Outflow of primary coolant
Electric system	Loss of commercial electric system
Electric system	All electricity black-out

6. EXTENSION OF SIMULATOR

The simulator can respond to following future modifications with comparative ease using JADE tools.

- Revision/addition of malfunction
- Revision/addition of irradiation facility
- Revision/addition of equipment in cooling system

7. CONCLUSION

Japan Atomic Energy Agency (JAEA) is now developing a real time simulator for a material testing reactor based on Japan Materials Testing Reactor (JMTR) in order to utilize a nuclear human resource development and to promote partnership with a developing country which has a plan to introduce nuclear power plant and/or experimental research reactor. The simulator is scheduled to be operated in JFY2012.

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[1] JMTR project, “Conceptual Design of the Japan Materials Testing Reactor”, JAERI 1056 (1964).
[2] K.Okumura, et al., “SRAC2006: A Comprehensive Neutronics Calculation Code System”, JAEI-DATA/Code 2007-004 (2007).
[3] K.Shibata, et al., “Japanese Evaluated Nuclear Data Library Version 3 revisin-3: JENDL3.3”, J. Nucl. Sci. Technol., 39[11], 1125 (2002).
[4] The RELAP5-3D Code Development Team. RELAP5-3D Code Manual, INEEL-EXT-98-00834 Revision 2.3. (20005).

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12.4 Report of The 1st Asian Symposium on Materials Testing Reactor

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One of the most important platforms for the establishment of World Networking for Material Testing Reactors is communication. Strong participation among countries in Asian region as well as other region is essential for making the network successful. This paper will discuss on the 1st Asian Symposium on Materials Testing Reactor which was held in Malaysia in early 2011.

Keywords: 1st Asian symposium on materials testing reactor, Asian network for materials testing reactor, World network for materials testing reactor, Status and future plan of testing reactors, Research and development of irradiation technology.

1. INTRODUCTION

1st Asian Symposium on Materials Testing Reactor (ASMTR2011) was held from February 16 to 18, 2011 in Kuala Lumpur, Malaysia. The objectives of this symposium were for the exchange of information among each testing reactors as well as to mutually understand the status of each reactor, and to discuss on the formation of Asian network for testing reactors. The symposium was jointly organized by Malaysian Nuclear Agency (Nuclear Malaysia) and Japan Atomic Energy Agency (JAEA). Papers presented were categorized into following areas; Status and future plan of testing reactors, Research and development of Irradiation tests which also include RI and Si semiconductor production technology, Research and development of PIEs, Utilization of testing reactors, Nuclear Human Resource Development and Best practices for Nuclear Non-Proliferation. After the discussion session, facility tour to TRIGA PUSPATI (RTP) was carried out in the afternoon of 18th February, 2011.

2. OUTLINE OF SYMPOSIUM

In this symposium 47 participants from the Asian countries have participated, and 20 papers were presented. Breakdown of the participants and number of presentations by country is shown in Fig.1. Commemorative photograph is shown in Fig.2.

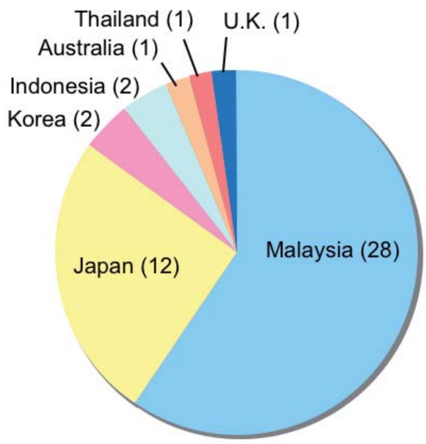
Status and future plans, irradiation technology, industry, discussion on Materials Testing Reactor (MTR) in Asian network, nuclear human resource development and best practice for nuclear non-proliferation were presented and discussed according to the program schedule shown in the appendix. Outline of discussion is shown as follows.

(1) Status and future plans

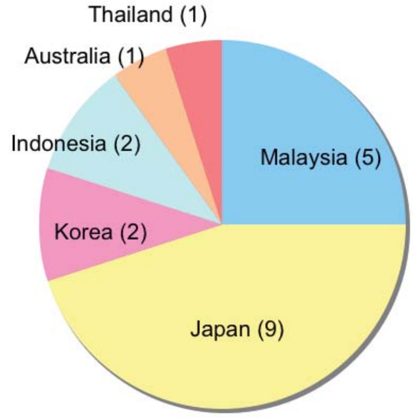
Status and future plans of the reactor facilities and experimental facilities which include hot laboratories were presented by Japan, Thailand, Indonesia, Australia, Malaysia and the Republic of Korea. Problems were discussed and shared with the participants as lesson learned.

(2) Irradiation technology

Neutron Irradiation Facilities of HANARO reactor, SCC tests facilities and instrumentation in the JMTR, Material Studies in Malaysia and PIE technology development in the JAEA-Oarai were presented. Also presented were on new research reactor for the RI production at KAERI and PIE of MOX fuels at JAEA-Oarai.



(a) Participants



(b) Presentations

( ) : Number of persons or presentations

Fig. 1 Breakdown of the participants and number of presentations by country.

Fig.2 Commemorative photograph.

(3) Industry use

Plan for production of ^{99}Mo by (n,γ) reaction at JMTR, status and future plan of the ^{99}Mo production in Malaysia and the characteristics of PZC column for ^{99}Mo/^{99m}Tc generator were presented. As for ^{99}Mo supply, Asian countries except Japan had no serious problems caused by the reactors trouble of NRU in Canada and HFR in Netherlands. This was because the SAFARI in South Africa and Multi Purpose Reactor G.A. Siwabessy in Indonesia was able to complement the shortage of ^{99}Mo supply. However, the importance of steady supply was emphasized, and the necessity for information exchange of production status which also includes experiences was recognized. It was also recognized that information exchange be considered as one of the roles for the Asian networks.

(4) Discussion on MTR in Asian network

During the panel discussion on MTR network in Asia, following topics; “Necessity of Materials Testing Reactor in countries introducing Nuclear Power Plant”, “Necessity of cooperation for RI production” and “What do you think of a design collaboration for Asian MTR?” were discussed.

As for the Necessity of Materials Testing Reactor, participants from Malaysia, Thailand, Indonesia and Korea agreed with the importance of fuels and materials irradiation tests for the LWRs and the RI production for medical use. Additionally, an important role for basic studies was also suggested by a participant form Australia.

As to the cooperation for RI production, although a serious lack of supply of ^{99}Mo for Asian countries occurred recently, an increased production by the SAFARI and Multi Purpose Reactor G.A. Siwabessy ease the issue, except for Japan. However, a necessary action for stable supply was recognized by all participants.

As for the design collaboration of Asian MTR, interest was shown by participants from Malaysia, Thailand, and Indonesia and the necessity of it was well recognized.

(5) Nuclear Human Resource Development

Necessity of personnel training for the nuclear power plant operators was emphasized by participants, as each Asian country has moved forward in a program development for the atomic energy personnel training. To carry out effective nuclear HRD, the importance of information sharing of each program was recognized.

(6) Best practice for Nuclear Non-Proliferation

Japan's efforts and the view of Malaysia on nuclear Non-Proliferation were introduced, and cooperation by Asian countries in the peaceful use of nuclear was proposed by Japanese participant. During the panel discussion, the problems for treatment/ disposal of LEU driver fuels and waste generated by ^{99}Mo production from LEU fuels were recognized. Such problems should be solved in near future in the NPT and were pointed out by participants. Again there was a necessity for problems sharing and continuous deliberation on the subject.

3. SUMMARY

The discussion on ASMTR2011 are summarize as follows;

- Necessity of the MTR was basically recognized by all attendees. Additionally, the importance of MTR was also confirmed not only for NPP development, but also basic studies like neutron diffraction etc.
- Importance of stable RI production, especially ^{99}Mo was agreed upon, and recognizes information exchange as one of the roles for the Asian networks.
- Needs for design collaboration of Asian MTR was recognized by some attendees.
- To carry out effective nuclear HRD, the importance of information sharing for each program was recognized.
- Treatment / disposal of spend driver LEU fuels and waste from ^{99}Mo process by LEU fuel were pointed out as a problem which should be solved in the near future under NPT.
- The necessity of continuous information exchange which also include above topics through ASMTR was recognized by all participants.
- It was decided that the next meeting will be held in Thailand in 2012.

The Proceedings of ASMTR2011 will be published by the Malaysian Nuclear Agency in the near future.

APPENDIX

1st Asian Symposium on Material Testing Reactor 2011 (ASMTR2011)

Program Schedule

16th February 2011, Wednesday	
14:00 – 17:30	Pre-Registration At The Legend Hotel (Only For Participant Staying In The Legend Hotel)

17th February 2011, Thursday		
08:00	Registration	
09:00	Opening Ceremony of 1st Asian Symposium on Material Testing Reactor 2011 by Director General, Malaysian Nuclear agency	
Session I : Status And Future Plans Chairperson : Abdul Aziz Mohamed		
09:25	Status of Refurbishment Project and Future Program in JMTR *H,Kawamura, M.Niimi, T.Kusunoki, Y.Nagao, H.Hiroi*	H. Kawamura
09:45	Study of Applications for the New Research Reactor in Vietnam *Nguyen Minh Tuan and Nguyen Nhi Dien*	Nguyen Minh Tuan
10:05	Current Status and Future Plan of the Research Reactors in Thailand *S. Chue-Inta, N. Klaysuban, C. Tippayakul, S. Loaharojpand and S. Chongkum*	S.Chue-Inta
10:25	Current Status of Multi Purpose Reactor G.A. Siwabessy, Serpong - Indonesia *Alim Tarigan*	A. Tarigan
10:45	Coffee Break	
Session II : Status And Future Plans Chairperson : Jun Sugimoto		
11:00	Neutron and Ion Beam Irradiation and Materials Characterisation and Testing Of Geniv and Fusion Materials at the OPAL Research Reactor and Other Facilities At the Australian Nuclear Science and Technology Organisation *R. P. Harrison, L. Edwards, G. R. Lumpkin, D. G. Carr and M. Law*	L. Edwards
11:20	RTP Upgrading: Study of Bundle Type Core *C. Iorgulis, M. D. Usang, M. H. Rabir, N. S. Hamzah, M. H. Abdul Khalil, J. A. Karim, M. F. Abdul Farid, Z. Hashim, T. Lanyau, M. S. Kassim, M. A. S. Salleh and M. P. Abu*	Mark Dennis Usang
11:40	Status of the Post-Irradiation Facilities in KAERI *Woo-Seog Ryu, Sangbok Ahn, Yong-Bum Chun*	Woo-Seog Ryu
12:00	Commemorative Photo & Lunch	
Session III : Irradiation Technology Chairperson : H. Izumo		
13:30	Neutron Irradiation Facilities and Utilization in HANARO *Kee Nam Choo, Man Soon Cho , Young Hwan Kang , Bong Goo Kim, Cheol Yong Lee, Sung Ho Ahn, In Cheol Lim, Jae Joo Ha*	Kee Nam Choo
13:50	Design Study of Material Irradiation Test Facility for SCC Tests In JMTR *Y. Okada, M. Aoyama, Y. Nomoto, Y. Matsui, M. Kanno, Y. Mori, K. Matsunami, S. Yamazaki and M. Niimi*	Y. Nomoto
14:10	Development of Instruments for International Standard *M. Tanimoto, S. Aoyama, S. Shibata, T. Saito, M. Ohmi And K. Tsuchiya*	M. Tanimoto
14:30	Material Studies In Malaysia *Abdul Aziz Mohamed, Megat Harun Al Rashid Megat Ahmad, Azraf Azman and Faridah Mohd Idris*	Abdul Aziz Mohamed

14:50	Status of PIE Teachnology Development in JAEA-Oarai *K.Tanaka*	K. Tanaka
15:10	Coffee Break	
Session IV : Industry Chairperson : E. Ishitsuka		
15:20	99Mo Production Plan from 98Mo By (N, γ) Reaction in JMTR *H.Izumo, A. Kimura, N. Hori, K. Tsuchiya, M. Ishihara*	H.Izumo
15:40	Mo-99 Production In Malaysia (Existing & Future Planning) *Rehir Dahalan*	Rehir Dahalan
16:00	Characteristics of PZC Column Loaded with High Radioactivity Mo-99 obtained from Irradiated Natural Molybdenum for Mo-99/Tc99m Generator *Abdul Mutalib, Rohadi Awaludin, Adang Hardi Gunawan, Hotman Lubis, Sriyono, Muhammad Subur, Masahiro Ishihara, Tsuguo Genka, Akihiro Kimura, Masakazu Tanase*	R. Awaludin
16:20	Coffee Break	
16:30	Discussion On MTR In Asian Network Panelist : Muhd Noor Muhd Yunus And H.Kawamura	
17:30	Adjourn	

18th February 2011, Friday		
Session V : Nuclear Human Resource Development Chairperson : H. Kawamura		
09:00	JAEA/NuHRDeC Activities and Japan Human Resource Development Network *SUGIMOTO, Jun and MURAKAMI, Hiroyuki*	J.Sugimoto
Panel Discussion On Human Resource Development Panelist : Muhd Noor Muhd Yunus, T.Shikama And E. Ishitsuka		
09:20	Nuclear Malaysia Human Resource Development Planning *Muhd Noor Muhd Yunus*	Muhd Noor Muhd Yunus
09:35	Role of the Oarai Branch of IMR, Tohoku University *Tatsuo Shikama*	T.Shikama
09:50	Human Resource Development Program Using JMTR *E. Ishitsuka, S. Kitagishi, M.Aoyama, K. Kawamata, Y. Nagao, M. Ishihara And H. Kawamura*	E. Ishitsuka
10:05	Discussion	
10:35	Coffee Break	
Session VI : Best Practice For Nuclear Non-Proliferation Chairperson : Jamal Khaer Ibrahim		
10:45	Japan's Efforts on Nuclear Non-Proliferation *Hiroshi TAMAI*	H.Tamai
11:05	NPT, Malaysia Overview *YM Raja Dato' Abd. Aziz Raja Adnan and Marina Mishar*	Marina Mishar
11:25	Panel Discussion on Non-Proliferation Treaty Panelist : Jamal Khaer Ibrahim, Hiroshi TAMAI, Alim Tarigan	
12:15	Lunch & End Of ASMTR2011	

13. Summary

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Session 1 : Invited presentation for circumstances surrounding RR's in the world

J. Knebel presented the situation of nuclear deployment in Germany. Electricity production of nuclear energy in Germany is limited to the year 2022 in order to allow the renewable energies to take this role, technologically and economically. In the frame of national provident research and to provide know-how in nuclear safety at the internationally highest level of science and technology, nuclear safety research is of high importance in Germany. Research reactors are still expecting as a neutron source for basic research and training.

J-M. Cavedon presented the nuclear energy research policy in Switzerland. Nuclear is 39% of total electricity of Switzerland. However, the Federal Council decided to stop nuclear power plants by 2034, because of Fukushima accidents. Final decision will be made by national vote in 2014. However, they admitted the necessity of nuclear research and decided to continue nuclear research in Switzerland. He explained about the nuclear research of Gen. III and Gen. IV, and emphasized the importance of the international cooperation in nuclear research.

J. Benson presented on how the Idaho National Laboratory is providing access to world-class capability to the Advanced Test Reactor, Idaho National Laboratory facilities to study irradiated materials, and strategic university facilities to study intense neutron and gamma effects on materials and fuels.

Session 2 : Status, Future plans

A.R. Antariksawan presented the current and strategic plan of future activities for the Multi Purpose Reactor G.A. Siwabessy (RSG-GAS) in Indonesia. In the last two years the reactor is operated more than 3,500 hours per year. The reactor is operated mainly for serving radioisotopes production, neutron activation analysis (NAA), neutron beam experiments and general irradiation for research and development activities. The future activities of the RSG-GAS reactor will be prioritized first on increasing the reliability and availability of the reactor operation to assure the demand of the isotope production, along with the improvement of neutron beam facilities to increase its utilization.

L. Foyto presented HEU to LEU conversion project status at the University of Missouri Research Reactor. A conversion feasibility study based on U-10Mo monolithic LEU fuel was completed in 2009. The “Preliminary Safety Analysis Report Methodologies and Scenarios for LEU Conversion of MURR” was completed in June 2011, and the desired regulatory path for licensing the facility as a 12 MW LEU Research Reactor has been identified.

M. De Kock presented the beryllium reflector element replacement, management and relocation of SAFARI-1 research reactor. These actions and practices include monitoring of the tritium levels in the primary water, calculating and measuring the fluxes within the Be reflector positions, measuring the straightness of the elements to track geometric deformation and visually inspecting the reflector elements for crack formation. Acceptance criteria indicating the end of life of the elements were established. All data obtained through the Be Management Programe are recorded in a database.

S. Van Dyck presented the material test reactor fuel research at the BR2 reactor. Uranium silicide dispersion fuels and U-Mo based fuels were irradiated by the baskets for the one of qualification tests which could irradiate flexible positioning. For the qualification irradiation of prototype fuel elements of the new

Jules Horowitz reactor by the 200mm diameter irradiation channels, an irradiation test is performed in a dedicated semi-open loop to fully simulate the thermal-hydraulic conditions.

T. Kobayashi presented a review on the seismic safety of JRR-3 according to the revised regulatory code on seismic design for nuclear reactors, and presented some topics concerning damages in JRR-3 due to the Great East Japan Earthquake.

M. Suzuki presented the present status of the Japan Materials Testing Reactor. Unfortunately, at the end of the JFY2010 on March 11, the Great East Japan Earthquake occurred, and functional tests before the JMTR restart were delayed by the earthquake. Moreover, a detail inspection found some damages such as slight deformation of the truss structure at the roof of the JMTR reactor building. Consequently, the restart of the JMTR will be delayed from June to next October, 2012. Now, the safety evaluation after the earthquake disaster is carrying out aiming at the restart of the JMTR.

Session 3 : Advancement of irradiation technology (1)

C. Dorn presented the status of material development for lifetime expansion of beryllium reflector. Three kinds of beryllium metals such as S-200F, S-65H and I-220H were selected from viewpoints of production methods, impurities and grain size of beryllium starting powders, mechanical properties. Irradiation tests in the JRR-3 and WWR-K, corrosion tests, surface analysis for chemical combination, etc. are on going as a material characterization test.

P.S. Bejarano presented the preliminary study about the geometry influence on the reflector assembly behavior which it was performed by a Finite Element Analysis. A simplified analysis was made varying the geometry of a prismatic Be assembly in height, thickness and width. The results showed that the most influent parameter in avoiding distortion due to swelling is the reflector assembly height, H; in second place is the thickness, L, and finally the angle/width, θ.

D. Kim presented the irradiation tests of nuclear grade graphite to provide design data for a new research reactor being developed in Korea. Two isotropic nuclear grades of graphite, IG-110 and NBG-17 are irradiated in HANARO up to around 5×10^{25} n/m^2 (E > 0.18 MeV) at temperature range of 70 to 120 °C. Pre- and post-irradiation characterization such as irradiation growth, thermal conductivity, and hardness properties will be conducted.

Session 4 : Advancement of irradiation technology (2)

Y. Chimi presented the in-pile IASCC growth tests of irradiated stainless steels in JMTR. In order to evaluate in-situ irradiation effects on IASCC growth behavior, the JAEA has planned in-pile crack growth tests of irradiated stainless steels under simulated BWR conditions in the JMTR. He has summarized previous literature data for in-pile and PIE SCC growth of irradiated stainless steels. After that, he showed the in-pile test plan by using the JMTR on the basis of the summary of the previous data. He also reported current status of development of in-pile crack growth test techniques such as a loading unit for a 0.5T-CT specimen and PDM measurement system.

J.T. Hong presented the design modification of the In-Pile Test Section for the fuel test loop to increase sealing capability. At first, two jacking bolts will be added on the top flange part of the inner assembly to decrease the shock induced on the inner assembly. The second, brazing process at the pressure boundary will be replaced with mechanical sealing structure. Above items are under verifying test in advance to the design change of IPS. After several simulated experiments, mock-up tests will be planed.

M.S. Cho presented the status for development of a capsule and instruments for high-temperature irradiation in HANARO. A new capsule with double thermal media composed of two kinds of materials such as Al-Ti and Al-graphite was designed and fabricated. The irradiation test was undertaken at OR5 for 28 days in the mid of 2010. At the irradiation test of the capsule, the temperature of the specimens successfully reached 700˚C and the integrity of the inner and outer thermal media was maintained. Furthermore, instruments such thermocouples and heaters were operated without any problem. A more advanced capsule which can be used up to 1,000˚C will be developed until 2012.

Session 5: Expansion of industry use (RI)

K. Tsuchiya presented the status of ^{99}Mo-^{99m}Tc production development by (n,γ) reaction with the research and testing reactors in JAEA. Especially, it is important to establish the separation, extraction and concentration procedures of ^{99m}Tc from Mo solution. Thus, the joint experiments and information exchanges have been carried out under the international cooperation from 2009. As the joint research between BATAN and JAEA, performance tests and evaluation by the PZC column and solvent extraction with MEK have been carried out in BATAN. As the joint research between INP-KNNC and JAEA, the cooperation experiments have been carried out for the comparison with separation and extraction properties of ^{99m}Tc by PZC and Zr-gel. In future, new R&Ds will be considered for the establishment of ^{99}Mo-^{99m}Tc production development by (n,γ) reaction under the international cooperation.

H. Yoshinaga presented the development of Mo recycle technique from generator materials. JAEA and Taiyo Koko Corporation, who have developed recycling technology of the used PZC, are proposed to recover molybdenum as an effective use of resources and reduction of radioactive wastes. In this study, preliminary experiments by two methods have been carried out with un-irradiated MoO_3 for Mo recycling that Mo recovery rate of these methods was more than 95%. As a result, the establishment of this new concept, contributes to an efficient supply and low cost can be realized by (n, γ) method for ^{99}Mo production.

M. Tanase presented the development of ^{99m}Tc extraction-recovery by solvent extraction method. In this study, the solvent extraction with MEK was selected to increase the concentration of ^{99m}Tc, and preliminary experiments were carried out using Re instead of ^{99m}Tc. Two tests were carried out as preliminary experiments. The one is the Re extraction test with MEK and Mo-Re mixing solution, and the other is the Re recovery test from the MEK which contains the Re. For the Re extraction test, it was clear that the Re extraction ratio is more than 90%. Two kinds of Re recovery tests, which are an evaporation test using the evaporator and an adsorption/elution test using the alumina column, were carried out. For the evaporation test, the Re concentration in the collected solution increased by more than 150 times. For the adsorption/elution test, the Re concentration in the eluted solution increased by more than 30 times.

Session 6 : Facility, upgrades, aging management

L. Hu provided an overview of MITR's recent power up from 5 MW to 6MW, utilization activities in in-core materials and fuel testing, and plan for core conversion from HEU to high density U-Mo LEU fuel. The planned LEU conversion will require a power increase to 7 MW in order to maintain the same neutron flux performance. Work performed at MITR demonstrates the wide range of research and utilization activities that are carried out at a medium-sized university research reactor.

H. Izumo presented the overview of refurbishment project of JMTR from viewpoints of aged-investigation for reactor components and new irradiation facilities installation. JMTR refurbishment project was completed on schedule. JMTR would be operated safely about twenty years operation after re-start, and would be used fully the irradiation facilities for the various field of irradiation tests with international/Asian network collaborations.

I. Saito presented the real-time personal exposure and health condition monitoring system. This system consists of a personal exposure management system and an area entry/exit management system. The system will provide information that who is currently in what part of which room and whether he has some health problems or has been exposed to radiation, and this information will be shared at a central location. After the accumulation of data from the JMTR and refining the system, it will be adapted to larger scale.

M. Brizuela presented the shield design performance of OPAL, Australian research reactor, by means of comparison of measured dose rates against the corresponding design calculated values. The reactor block, the control rod mechanism penetration, primary shutters, and neutron guide bunker were evaluated by radiation transport codes such as DORT and MCNP4. For coolant activation calculations, the ORIGEN2 code and specific models were used. All results obtained agree very well with the measured values, and contribute to establish the confidence on the calculation tools and methodology used for shielding design.

M. Brizuela presented the OPAL reactor calculations using the Monte Carlo cell code developed by VTT Serpent v1.1.14. These core calculations are performed using CITVAP 3-D diffusion code, which is well-known reactor code based on CITATION. Subsequently, results are compared with those obtained by the deterministic calculation used by INVAP, which uses the Collision Probability Condor cell-code to obtain few-group constants. Finally, results are compared with the experimental data obtained from the reactor information for several operation cycles. As a result, several evaluations are performed, including a code to code cell comparison at cell and core level and calculation-experiment comparison at core level, in order to evaluate the Serpent code actual capabilities.

<u>Session 7 : Advancement of irradiation technology (3)</u>

Y. G Cho presented the analysis method for the irradiation induced creep and growth of Zircaloy-4 inner shell of HANARO. The anisotropic irradiation creep behavior is modeled as a uniaxial strain-hardening power law modified by Hill's stress potential, and the anisotropic irradiation growth is modeled by using volumetric swelling with anisotropic strain rate. The analysis results are compared with the deformation of HANARO measured after 9 years in operation.

A. Shibata presented the development of in-pile instruments for fuel and material irradiation tests. Two kinds of measuring equipments have been developed; these are the Electrochemical Corrosion Potential (ECP) sensor, the Linear Voltage Differential Transformer (LVDT) type gas pressure gauge. The ECP sensor has been developed to determine the corrosive potential under high temperature and high pressure water condition. The structure of the joining parts was optimized to avoid stress concentration. The ECP sensor showed enough performance at 288°C and at 9MPa conditions. The LVDT type gas pressure gauge has been developed to measure gas pressure in a fuel element during neutron irradiation. To perform stable measurements with high accuracy under high temperature, high pressure and high dosed environment, the coil material of LVDT was changed to MI cable. As a result of this development, the LVDT type gas pressure gauge showed high accuracy at 1.8% of a full scale, and good stability.

M. H. Choi presented the thermal analysis of specimens for low irradiation temperature below 100˚C in the HANARO, as a part of the research reactor development project with a plate type fuel. Capsule for irradiation tests of major reactor materials is under development in KAERI. Thermal analysis is performed by using an ANSYS program, and the temperatures are evaluated. Analysis results show that almost specimens meet the irradiation target temperature. However, some canned specimens need the design change, and the gap size has significantly effect on the specimen temperature.

K. Miura presented the development of reactor water level sensor for extreme conditions. The new type water level indicator, which composed of thermocouple and heater, was developed. Demonstration test and characteristic evaluation of the water level indicator were performed by 200 mm measurement unit and 1000 mm MI cable. It was demonstrated that the water level can be measured within 20 mm accuracy.

Session 8 : New generation MTR

R. Mazzi presented main activities 2011 of INVAP Nuclear Energy Division(NED). These are asfollows:

Atucha II (750MWe) and CAREM (27MWe) NPP, construction and licensing projects.

Commissioning of the Radioisiotope Production Facility in Egypt for AEA,

Successful test recently achieved for producing Mo-99 with the Aqueous Homogenous Reactor

INVAP Australia including supporting activities to OPAL with the awesome record of >75% availability in the last two years.

New research reactor developments: RA-10, RMB, DIPR.

S.Van Dyck presented the status of MYRRHA project started from 1998 by SCK•CEN. MYRRHA is designed as a multi-purpose irradiation facility in order to support research programs on fission and fusion reactor structural materials and nuclear fuel development. Present status of the design of MYRRHA and the work program for the first stage 2010-2014 in terms of design, licensing and associated support R&D were reported.

N. Hori presented the conceptual design of the high-performance and low-cost next generation materials testing reactor. Targets of this conceptual design is 10MW thermal power by plate type fuel elements, swimming pool type, four hot-cells in reactor facility, and connecting of large-scale hot-cells in the hot laboratory for a re-irradiation examination.

Session 9 : Advancement of PIE technology

M. Sugiyama presented the development of fracture toughness estimation method using thin tube. To perform fracture toughness evaluation with a thin tube like the thimble tube used at the nuclear reactor of PWR, the fracture toughness evaluation was performed in proportion to the bending test. The prospect to evaluate the fracture toughness of the tubing material, which cannot obtain CT specimen, was acquired.

A. Onozawa presented the improvement of the center boring device for the irradiated fuel pellets for the power ramp tests. One of the important parameters measured during fuel power ramp test is the center temperature of the fuel pellet. For this measurement, a thermocouple is installed into the hole bored at the pellet center by the center boring device, which can fix the fuel pellet with the frozen CO_2 gas during its boring process. At the Reactor Fuel Examination Facility in Tokai R&D Center, several improvements were applied for the previous boring device to gain its performance and reliability. The major improvements are the change of the drill bit, modification of the boring process and the optimization of the remote operability. The mock-up test will be performed with the irradiated fuel pellet to confirm its performance and reliability.

A. Ishimi presented the development of high resolution x-ray ct technique for irradiated fuel assembly. To obtain the high resolution X-ray CT image, the number of X-ray detectors has been increased from 30 to 100, and the collimator equipped with 100 slits was utilized. Additionally, the detector material has been changed from $CdWO_4$ to a semiconducting material made of silicon alloy. This technique was applied to a fuel assembly irradiated in the experimental fast reactor "JOYO," and the high resolution X-ray CT images were successfully obtained of the fuel pellet.

Session 10 : Development of advanced driver fuel

D. Geslin presented the industrial approach of MTR fuels development in the CERCA. AREVA-CERCA has developed more than 70 fuel and targets designs over the last 50 years. Store of this experience and the support of a company such as AREVA, AREVA-CERCA have implemented industrial approaches that benefit to the customers. Some of these industrial approaches will be reviewed to emphasis the benefits in terms of product development, risk management for projects and regular productions, and quality management to reach customer satisfaction. The latest developments about U-Mo tests and current policy about spent fuel reprocessing were also presented.

J. M. Park presented the development of U-Mo/Al dispersion fuel for research reactors. KOMO-5 irradiation test for full size U-Mo fuel rods is underway since May 23, 2011. The purpose of the KOMO-5 test includes the investigation of the irradiation behaviors of silicide or nitride coated U-7Mo/Al(-Si) fuels and the effects of pre-formed interaction layers. It is expected that the irradiation test will be terminated after reaching 60 at% burnup in May 2012, and the first PIE results of the KOMO-5 will be released in September 2012.

Session 11 : Nuclear HRD for next generation, other

K. Yamashita presented the activity of Nuclear Human Resource Development Center in JAEA. The main recent activities are 1)Education and training for domestic nuclear engineers, 2)Education and training for foreign nuclear engineers, 3) Cooperation with domestic universities for nuclear education, 4) Cooperation with international organization for nuclear human resource development, and 5)Central role as secretariat for Japan Nuclear Human Resource Development Network (JN-HRD Net).

N. Takemoto presented training program for students and young engineers held using JMTR and the related facilities as part of nuclear Human Resource Development (HRD) in order to support global expansion of nuclear power industry. In the nuclear HRD initiative program sponsored by the MEXT, the training course was newly established from JFY 2010 to JFY 2012. In this course, basic understanding on irradiation test and post irradiation examination is aimed to be learnt by overall and practical training such as the nuclear/thermal designs of irradiation capsule, post irradiation examination, measurement and evaluation of neutron fluence, etc. using the JMTR and the related facilities. In the 1st training course on February 14 to 25, 2011, ten students were joined as trainees. From JFY 2011, two courses will be held in every year, and 20 trainees will be accepted in every course.

T. Ishitsuka presented the development of the real time simulator for a materials testing reactor based on Japan Materials Testing Reactor (JMTR) in order to utilize a nuclear human resource development and promotion of partnership with developing countries which have a plan to introduce nuclear power plant and/or experimental research reactor. The simulator operation is scheduled in JFY2012.

Mohd Ashhar reported the 1st Asian Symposium on Materials Testing Reactor which was held from February 16 to 18, 2011 at the Kuala Lumpur, Malaysia. In this symposium, 47 persons from Asian countries have attended. Construction of the Asian network for testing reactors was discussed, and importance of continuous information exchange was recognized by all attendances.

Discussion of MTR

In the final discussion, following three items are discussed from an active collaboration point with seven panelists chaired by Dr. Masahide Suzuki, JAEA, and Dr. Leslie P. Foyto, University of Missouri.

(1) Necessity of cooperation for Mo-99 production by (n, γ) method,

(2) Necessity of standardization of irradiation technology, and proposal of cooperative subjects by reactor network,

(3) Conceptual design of next generation materials testing reactor by collaboration.

For the first item, there are many remarks/ideas such as,

- integrated approach among target manufacture, irradiation facilities, transportations, processing facilities and radio-pharmaceutical partners is necessary,
- cooperation activities may involve in choice of irradiation target, method, waste management, transportation etc.
- target with Mo-98 enrichment should be interest,
- favorable to corroboration considering increasing demand,
- not to involve due to the competitive technology of (n, fission) method.

From the common interest view point, the Mo-99 production by (n, γ) method will continue to be discuss in more detail for common technical aspects such as target fabrication, irradiation facility.

For the second item, there are also many remarks/ideas such as,

- for general development of irradiation facilities, standardization of irradiation technology can be valuable, however not easy due to the difference country by country as well as facility by facility,
- standardization is better to share good practice,
- standardization between high and low power reactors may not be achievable, however some basic area such as neutronic calculation of irradiated target using common code and benchmarking is possible,
- beneficial to perform the round robin test with same test condition at verified reactors, and discuss the test results,

- standardization is useful to offer of best irradiation test to worldwide users for limited period and to develop instruments efficiently. As cooperation items, basic irradiation techniques, such as neutron dosimetry, temperature measurement, are proposed.

From the discussion, it is concluded that the standardization of irradiation technology is common interest to reduce the construction cost for irradiation facilities as well as to enhance the human resource development, and cooperation subjects such as proposed round robin test, neutron/temperature measurement including the evaluation will be discussed in more detail.

For the third item, remarks/ideas here are such as,

- one example of collaboration is MYRRHA as international project for next generation of fast spectrum irradiation facility,
- now, new MTRs are designed in the world, and similar parts design, for example reflector etc., is possible to collaborate,
- thorium fuel development technology is welcome to collaborate,
- cooperation with many country makes better reactor with good technology and save money,
- conceptual design with proven technology aiming at high performance as well as lower cost reactor is useful to human resource development in domestic as well as joining country,

As a discussion result, it is not easy to find common cooperation area, because the design is carried out by each country, and the construction purpose is different from country by country. However, new research reactors construction is interesting for participants, then the information exchange will be continue.

Appendix Symposium Program

PROGRAM OF ISMTR-4

Monday 5th Dec., 2011

13:30 - 17:00	**Registration (Oarai Park Hotel Lobby)**		
13:30 - 17:40	6th Specialist Meeting on Recycling of Irradiated Beryllium		

Tuesday 6th Dec., 2011

8:30 - 15:00	**Registration (Oarai Park Hotel Lobby)**		
10:10 - 10:20	**Opening Session**	A.Suzuki	JAEA (President)
10:20 - 10:40	**Nuclear R&D in Japan**	H.Ohashi	Univ. of Tokyo (General chair)
Session 1 : Invited presentation for circumstances surrounding RR's in the world		**Chairman:**	**M.Suzuki**
10:40 - 11:10	Research with Neutrons in Germany in View of the Phase Out of Nuclear Power	J.Knebel, W.Petry	Karlsruhe Institute of Technology
11:10 - 11:40	Nuclear Research Policy in the Evolving Political Context of Switzerland	J.Cavedon, J.Dreier, P.Hardegger	Paul Scherrer Institute
11:40 - 12:10	Advancing Nuclear Technology and Research: The Advanced Test Reactor National Scientific User Facility	J.Benson, T.Allen, F.Marshall	Idaho National Laboratory
12:10 - 13:30	COMMEMORATIVE PHOTOGRAPH and LUNCH		
Session 2 : Status, Future plans		**Chairman:**	**J.Knebel**
13:30 - 13:55	RSG-GAS: Current and Strategic Plan for the Future Activities	A.R. Antariksawan, A.Tarigan	National Nuclear Energy Agency, Indonesia (BATAN)
13:55 - 14:20	The University of Missouri Research Reactor HEU to LEU Conversion Project Status	L.Foyto, J.C.McKibben, K.Kutikkad, N.Peters, G.Solbrekken, J.Kennedy, J.Stillman, E.Feldman, C.Tzanos, J.Stevens	University of Missouri
14:20 - 14:45	SAFARI-1 Research Reactor Beryllium Reflector Element Replacement, Management and Relocation	M.De Kock, JWH Vlok, M.Belal	South African Nuclear Energy
14:45 - 15:10	Material Test Reactor Fuel Research at the BR2 reactor	S.Van Dyck, E.Koonen, S.Van den Berghe	SCK•CEN
15:10 - 15:35	Review on Seismic Safety of JRR-3 according to Revised Regulatory Code on Seismic Design for Nuclear Reactors	T.Kobayashi, M.Araki, M.Takeuchi, T.Ohba, Y.Torii	JAEA-Tokai
15:35 - 16:00	Present Status of Japan Materials Testing Reactor	N.Hori, M.Kaminaga, T.Kusunoki, M.Ishihara, M.Niimi, Y.Komori, M.Suzuki, H.Kawamura	JAEA-Oarai
16:00 - 16:15	BREAK		
Session 3 : Advancement of irradiation technology (1)		**Chairman:**	**T.Nakamura**
16:15 - 16:40	Status of Material Development for Lifetime Expansion of Beryllium Reflector	C.Dorn, K.Tsuchiya, Y.Hatano, P.Chakrov, M.Kodama, H.Kawamura	Materion Brush Beryllium & Composites
16:40 - 17:05	Beryllium Reflectors for Research Reactors. Review and Preliminary Finite Element Analysis	P.S.Bejarano, R.G.Cocco	INVAP, Argentine
17:05 - 17:30	Irradiation Tests of Nuclear Grade Graphite at High Neutron Dose and Low Temperature at HANARO	D.Kim, S.H.Kang, H.H.Jin, B.G.Kim, M.H.Choi, T.K.Kim, Y.H.Jeong	KAERI

Wednesday 7th Dec., 2011

8:30 - 15:00 **Registration (Oarai Park Hotel Lobby)**

Session 4 : Advancement of irradiation technology (2) **Chairman: J.Benson**

9:00 - 9:25	In-pile IASCC Growth Tests of Irradiated Stainless Steels in JMTR	Y.Chimi, S.Kasahara, A.Shibata, H.Ise, Y.Kawaguchi, J.Nakano, M.Ohmi, Y.Nishiyama	JAEA-Tokai
9:25 - 9:50	Design Modification of In-Pile Test Section for Increase of Sealing Capability	J.T.Hong, S.H.Ahn, C.Y.Joung, H.Y.Jeong	KAERI
9:50 - 10:15	Status for Development of Capsule and Instruments for High-Temperature Irradiation in HANARO	M.S.Cho, K.N.Choo, S.W.Yang, K.T.Shim, C.Y.Lee	KAERI

10:15 - 10:55 BREAK

Session 5: Expansion of industry use (RI) **Chairman: A.Antariksawan**

10:55 - 11:20	Status of ^{99}Mo-^{99m}Tc Production Development by (n,γ) Reaction	K.Tsuchiya, A.Mutalib, P.Chakrov, M.Kaminaga, M.Ishihara, H.Kawamura	JAEA-Oarai
11:20 - 11:45	Development of Mo Recycle Technique from Generator Materials	M.Tanimoto, S.Kakei, M.Kurosawa, A.Kimura, K.Nishikata, H.Yoshinaga, K.Tsuchiya	Taiyo Koko
11:45 - 12:10	Development of ^{99m}Tc Extraction-recovery by Solvent Extraction Method	A.Kimura, K.Nishikata, M.Tanase, S.Fujisaki, H.Izumo, K.Tsuchiya, T.Shiina, A.Ohta, N.Takeuchi, M.Ishihara	Chiyoda Technol

12:10 - 13:20 LUNCH

Session 6: Facility, upgrades, aging management **Chairman: L.Foyto**

13:20 - 13:45	MIT Research Reactor –power Uprate and Utilization	L.W.Hu	MIT
13:45 - 14:10	Overview of Refurbishment Project of JMTR	H.Izumo, N.Hori, M.Kaminaga, T.Kusunoki, M.Ishihara,Y.Komori, M.Suzuki, H.Kawamura	JAEA-Oarai
14:10 - 14:35	Real-time Personal Exposure and Health Condition Monitoring System	N.Tateishi, H.Kanda, A.Asai, I.Saito, Y.Oota, N.Hanawa, H.Ueda, T.Kusunoki, H.Kawamura	Hitachi Aloka Medical
14:35 - 15:00	OPAL Shield Design Performance Assessment: Comparison of Measured Dose Rates Against the Corresponding Design Calculated Values – A Designer Perspective	M.Brizuela, F. Albornoz	INVAP, Argentine
15:00 - 15:20	OPAL Reactor Calculations using the Monte Carlo Code Serpent	D.Ferraro, E.Villarino (M.Brizuela)	INVAP, Argentine

15:20 - 15:35 BREAK

Session 7: Advancement of irradiation technology (3) **Chairman: L.W. Hu**

15:35 - 16:00	Irradiation Creep and Growth Behaviour of Zircaloy-4 Inner Shell of HANARO	J.H.Chung, Y.G.Cho, J.I. Kim	KAERI
16:00 - 16:25	Development of In-pile Instruments for fuel and material irradiation tests	A.Shibata, S.Kitagishi, N.Kimura,T.Saito, M.Tanimoto J.Nakamura, M.Ohmi, H.Izumo, K.Tsuchiya	JAEA-Oarai
16:25 - 16:50	Thermal Analysis on the Specimens for Low Irradiation Temperature below 100°C in the HANARO	M.H.Choi, B.G.Kim, B.C.Lee, T.K.Kim	KAERI
16:50 - 17:15	Development of Reactor Water Level Sensor for Extreme Conditions	K.Miura, A.Shibata, J.Nakamura, T.Ogasawara, T.Saito, K.Tsuchiya	Sukegawa Electric

Thursday 8th Dec., 2011

8:30 - 15:00 **Registration (Oarai Park Hotel Lobby)**

Session 8 :New generation MTR		**Chairman:**	**Mohd Ashhar Hj Khalid**
9:00 - 9:25	Nuclear Energy Division Main Activities 2011	R. Mazzi	INVAP, Argentine
9:25 - 9:50	MYRRHA An Innovative and Unique Research Facility	H.A.Abderrahim (S.Van Dyck)	SCK•CEN
9:50 - 10:15	Conceptual Design of Next Generation MTR	H.Nagata, T.Yamaura, M.Naka, K.Kawamata, H.Izumo, N.Hori, Y. Nagao, T.Kusunoki, M.Kaminaga, Y.Komori, M.Suzuki, M.Mine, S.Yamazaki, S.Ishikawa, K.Miura, S.Nakashima, K.Yamaguchi, H. Kawamura	JAEA-Oarai

10:15 - 10:25 BREAK

Session 9 :Advancement of PIE technology		**Chairman:**	**M.Brizuela**
10:25 - 10:50	Development of Fracture Toughness Estimation Method using Thin Tube	M.Sugiyama, T.Torimaru, K.Chatani	Nippon Nuclear Fuel Development
10:50 - 11:15	Improvement of Center Boring Device for Irradiated Fuel Pellets	K.Usami, A.Onozawa, Y.Kimura, N.Sakuraba, H.Shiina, A.Harada, M.Nakata	JAEA-Tokai
11:15 - 11:40	Development of High Resolution X-Ray CT Technique for Irradiated Fuel Assembly	A.Ishimi, K.Kastuyama, K.Maeda, T.Asaga	JAEA-Oarai

11:40 - 12:40 LUNCH

Session 10 : Development of advanced driver fuel		**Chairman:**	**S.Van Dyck**
12:40 - 13:05	CERCA Industrial Approach to Customer's Needs	D.Geslin	CERCA
13:05 - 13:30	Development of U-Mo/Al Dispersion Fuel for Research Reactors	J.M.Park, H.J.Ryu, J.H.Yang, Y.J.Jeong, Y.S.Lee	KAERI

13:30 - 14:05 BREAK

Session 11: Nuclear HRD for next generation, other		**Chairman:**	**M.Kaminaga**
14:05 - 14:30	Activity of Nuclear Human Resource Development Center in JAEA	K.Yamashita, H.Murakami, N.Nakagawa, N.Arai, K.Matsuura, Y.Ohzeki	JAEA-Tokai
14:30 - 14:55	Training Program for Students and Young Engineers in JMTR	N.Takemoto, H.Izumo, N.Hori, E.Ishitsuka, M.Suzuki	JAEA-Oarai
14:55 - 15:20	Real Time Simulator for Material Testing Reactor	N.Takemoto, T.Imaizumi, H.Izumo, N.Hori , M.Suzuki, T.Ishitsuka, K.Tamura	Itochu Techno-solutions
15:20 - 15:35	1st Asian Symposium on Materials Testing Reactor Report	H.Kawamura, E.Ishitsuka, M. Fairus Abdul Farid, Julia Abdul Karim, H.Izumo, Mohamad Puad Haji Abu, Muhd Noor Muhd Yunus, Mohd Ashhar Hj Khalid	Malaysian Nuclear Agency

15:35 - 15:45 BREAK

Discussion of MTR		**Chairman:**	**M.Suzuki** **L.Foyto**
15:45 - 17:45	(by panel presentation) (1) Necessity of cooperation for Mo-99 production by (n,gamma) reaction ? (2) Necessity of standardization of irradiation technology, and proposal of active cooperation subjects by reactor network ? (3) How do you think about conceptual design of next generation materials testing reactor by collaboration ?		All

Friday 9th Dec., 2011

9:00 - 12:00 **Technical tour (JMTR, AGF etc.)**

国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位 名称	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	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	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方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
量濃度 (a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立法メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率 (b)	（数字の） 1	1
比透磁率 (b)	（数字の） 1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表される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(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ (d)	Hz		s^{-1}
力	ニュートン	N		$m\ kg\ s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1}\ kg\ s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N m	$m^2\ kg\ s^{-2}$
仕事率，工率，放射束	ワット	W	J/s	$m^2\ kg\ s^{-3}$
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	$m^2\ kg\ s^{-3}\ A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2}\ kg^{-1}\ s^4\ A^2$
電気抵抗	オーム	Ω	V/A	$m^2\ kg\ s^{-3}\ A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2}\ kg^{-1}\ s^3\ A^2$
磁束	ウエーバ	Wb	Vs	$m^2\ kg\ s^{-2}\ A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg\ s^{-2}\ A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2\ kg\ s^{-2}\ A^{-2}$
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	$m^{-2}\ cd$
放射性核種の放射能 (f)	ベクレル (d)	Bq		s^{-1}
吸収線量, 比エネルギー分与, カーマ	グレイ	Gy	J/kg	$m^2\ s^{-2}$
線量当量, 周辺線量当量, 方向性線量当量, 個人線量当量	シーベルト (g)	Sv	J/kg	$m^2\ s^{-2}$
酸素活性	カタール	kat		$s^{-1}\ mol$

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。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である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で"radioactivity"と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含む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	$m^{-1}\ kg\ s^{-1}$
力のモーメント	ニュートンメートル	N m	$m^2\ kg\ s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg\ s^{-2}$
角速度	ラジアン毎秒	rad/s	$m\ m^{-1}\ s^{-1}=s^{-1}$
角加速度	ラジアン毎秒毎秒	rad/s^2	$m\ m^{-1}\ s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg\ s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2\ kg\ s^{-2}\ K^{-1}$
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	$m^2\ s^{-2}\ K^{-1}$
比エネルギー	ジュール毎キログラム	J/kg	$m^2\ s^{-2}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	$m\ kg\ s^{-3}\ K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1}\ kg\ s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m\ kg\ s^{-3}\ A^{-1}$
電荷密度	クーロン毎立方メートル	C/m^3	$m^{-3}\ sA$
表面電荷	クーロン毎平方メートル	C/m^2	$m^{-2}\ sA$
電束密度，電気変位	クーロン毎平方メートル	C/m^2	$m^{-2}\ sA$
誘電率	ファラド毎メートル	F/m	$m^{-3}\ kg^{-1}\ s^4\ A^2$
透磁率	ヘンリー毎メートル	H/m	$m\ kg\ s^{-2}\ A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2\ kg\ s^{-2}\ mol^{-1}$
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)	$m^2\ kg\ s^{-2}\ K^{-1}\ mol^{-1}$
照射線量（Ｘ線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1}\ sA$
吸収線量率	グレイ毎秒	Gy/s	$m^2\ s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4\ m^{-2}\ kg\ s^{-3}=m^2\ kg\ s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2\ sr)$	$m^2\ m^{-2}\ kg\ s^{-3}=kg\ s^{-3}$
酵素活性濃度	カタール毎立方メートル	kat/m^3	$m^{-3}\ s^{-1}\ mol$

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６．SIに属さないが、SIと併用される単位

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=86 400 s
度	°	1°=(π/180) rad
分	’	1’=(1/60)°=(π/10800) rad
秒	”	1”=(1/60)’=(π/648000) rad
ヘクタール	ha	$1ha=1hm^2=10^4m^2$
リットル	L，l	$1L=1l=1dm^3=10^3cm^3=10^{-3}m^3$
トン	t	$1t=10^3\ kg$

表７．SIに属さないが、SIと併用される単位で、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 単位で表される数値
電子ボルト	eV	$1eV=1.602\ 176\ 53(14)\times10^{-19}J$
ダルトン	Da	$1Da=1.660\ 538\ 86(28)\times10^{-27}kg$
統一原子質量単位	u	1u=1 Da
天文単位	ua	$1ua=1.495\ 978\ 706\ 91(6)\times10^{11}m$

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	$1\ bar=0.1MPa=100kPa=10^5Pa$
水銀柱ミリメートル	mmHg	1mmHg=133.322Pa
オングストローム	Å	$1\ Å=0.1nm=100pm=10^{-10}m$
海里	M	1 M=1852m
バーン	b	$1\ b=100fm^2=(10^{-12}cm)2=10^{-28}m^2$
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デジベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	$1\ erg=10^{-7}\ J$
ダイン	dyn	$1\ dyn=10^{-5}N$
ポアズ	P	$1\ P=1\ dyn\ s\ cm^{-2}=0.1Pa\ s$
ストークス	St	$1\ St\ =1cm^2\ s^{-1}=10^{-4}m^2\ s^{-1}$
スチルブ	sb	$1\ sb\ =1cd\ cm^{-2}=10^4cd\ m^{-2}$
フォト	ph	$1\ ph=1cd\ sr\ cm^{-2}\ 10^4lx$
ガル	Gal	$1\ Gal\ =1cm\ s^{-2}=10^{-2}ms^{-2}$
マクスウェル	Mx	$1\ Mx = 1G\ cm^2=10^{-8}Wb$
ガウス	G	$1\ G\ =1Mx\ cm^{-2}\ =10^{-4}T$
エルステッド (c)	Oe	$1\ Oe \triangleq (10^3/4\pi)A\ m^{-1}$

(c) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
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	$1\ rem=1\ cSv=10^{-2}Sv$
ガンマ	γ	1 γ =1 nT=10-9T
フェルミ		1フェルミ=1 fm=10-15m
メートル系カラット		1メートル系カラット = 200 mg = 2×10-4kg
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	1cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー）4.184J（「熱化学」カロリー）
ミクロン	µ	$1\ \mu = 1\mu m=10^{-6}m$

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