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Proceedings of 2012 JAEA/KAERI Joint Seminar on Advanced Irradiation and PIE Technologies March 28-30, 2012, Mito, Japan

(Eds.) Masahiro ISHIHARA, Etsuo ISHITSUKA and Masahide SUZUKI

Neutron Irradiation and Testing Reactor Center
Oarai Research and Development Center

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Under the "Arrangement for Corporation in the field of peaceful uses of Nuclear Energy between the Japan Atomic Energy Agency (JAEA) and the Korean Atomic Energy Research Institute (KAERI)", the 2012 JAEA/KAERI Joint Seminar on Advanced Irradiation and PIE (post-irradiation examination) Technologies has been held at Mito, Japan from March 28 to 30, 2012.

This triennial seminar is the seventh in series of bilateral exchange of irradiation and PIE technologies and research reactor management. Since the first joint seminar on the PIE Technology between JAERI (Japan Atomic Energy Research Institute, former agency of JAEA) and KAERI was held at JAERI Oarai Research Institute, Japan in 1992, the international cooperation program between JAEA and KAERI has been actively carried out in the field of neutron irradiation. At the fifth seminar in 2005 and sixth in 2008, the irradiation technology and the research reactor management fields were included, respectively, to the joint seminar, and it covers whole areas of irradiation using research reactors.

In this seminar total 37 presentations were made in three technical sessions, which are "research reactor management", "advanced irradiation technology" and "post-irradiation examination technology", and active information exchange was done among participants. Papers or manuscripts presented in the 2012 JAEA/KAERI Joint Seminar on Advanced Irradiation and PIE Technologies are contained in the proceedings.

Keywords ; Material Testing Reactor, Research Reactor, Hot Laboratory, Irradiation Technology, PIE Technology

「2012 照射試験・照射後試験技術に関する日韓セミナー」　論文集
2012 年 3 月 28-30 日、水戸

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、石塚　悦男、鈴木　雅秀

（2012 年 10 月 19 日受理）

韓国原子力研究院（以下、「KAERI」という）と（独）日本原子力研究開発機構（以下、「JAEA」という）が締結した「原子力の平和利用分野における協力のための取決め」に基づいて、水戸において「2012 照射試験・照射後試験技術に関する日韓セミナー」が 2012 年 3 月 28 日～30 日に開催された。

照射技術に関する情報交換として３年毎に開催されているこの日韓セミナーは、今回で７回目である。1992 年に日本原子力研究所（以下、「JAERI」という）大洗研究所（現在の JAEA 大洗研究開発センター）において、JAERI と KAERI との照射後試験技術に関する第 1 回日韓セミナーが開催されて以来、中性子照射の分野における JAEA と KAERI の国際協力が進められてきた。2005 年の第 5 回日韓セミナーにおいて、照射技術の分野、さらに第 6 回目において原子炉の管理分野が加わり、試験研究炉を用いた広範な照射利用の情報交換となった。

本セミナーでは「試験研究炉の管理」、「照射技術」及び「照射後試験技術」の三つのセッションにおいて 37 件の講演が活発に行われた。本報告書は、この「2012 照射試験・照射後試験技術に関する日韓セミナー」で発表された論文を収録したものである。

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1. Opening Address

Soju Suzuki
Director General,
Oarai Research and Development Center,
Japan Atomic Energy Agency

It is our great pleasure to hold “2012 KAERI-JAEA Joint Seminar on Advanced Irradiation and PIE Technologies” at Mito, Ibaraki, Japan. I would like to welcome all of you to the 2012 joint seminar; especially to express my special appreciation to 21 Korean participants taking long time travel to be here today.

This triennial seminar has been held alternately every three years at the KAERI and the JAERI, former organization of the JAEA, since the first seminar was held in 1992 at the Oarai Research Establishment, JAERI. We have done bilateral valuable information exchange in the field of irradiation technology and post irradiation technology through past six seminars. From this activity, good friendship and cooperation between KAERI and JAEA have been constructed in these fields.

By the way, as everyone knows, the earthquake, off the Pacific coast of Tohoku that occurred on March 11 last year, caused the serious damage at the Tohoku district, and also caused slightly damage to the JMTR etc. Here, I would like to express our deepest gratitude for many words to encourage from you. Now, we are carefully checking the integrity of the reactor facility towards the reoperation of the JMTR, and its restart will be planned in 2012.

In this triennial seminar, research reactor management, advanced irradiation technology and post irradiation examination technology are addressed as information exchange items. I would expect that fruitful discussions will be done here, and that more close relationship will be made to promote the neutron irradiation field in both organizations to meet the domestic as well as international requirements on nuclear energy.

Finally, once again, I would welcome all the participants to attend this seminar, and I wish that this seminar will be valuable for participants gathering here. Moreover, I express my gratitude to members of the steering committee, Chairmen as well as persons of the Secretariat.

Thank you for your attention.

2. Plenary Session

2.1 Current Status of JMTR

Naohiko Hori[1], Masanori Kaminaga, Tsuyoshi Kusunoki, Masahiro Ishihara, Yoshihiro Komori and Masahide Suzuki

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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 the 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, 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, and cross section of the core is shown in Fig.2. Specification of the JMTR is summarized 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 and materials to establish domestic technology for developing nuclear power plants to produce radio isotopes and to conduct 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 these 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 reoperation. The renewed and upgraded JMTR will restart in October, 2012 and will operate for a period of about 20 years (until around JFY 2030).

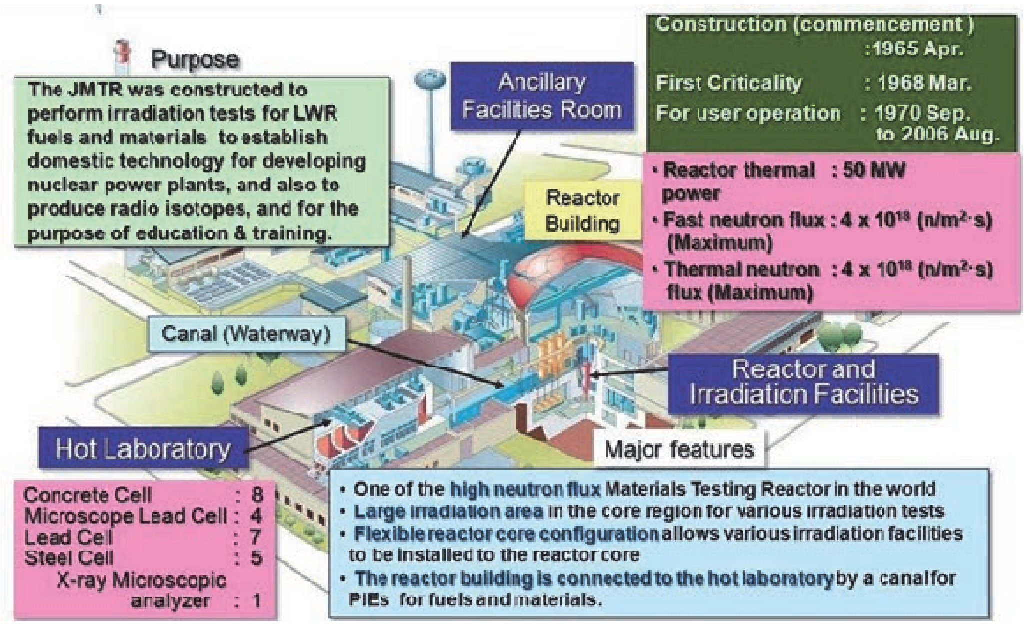

Fig.1 Outline of the JMTR.

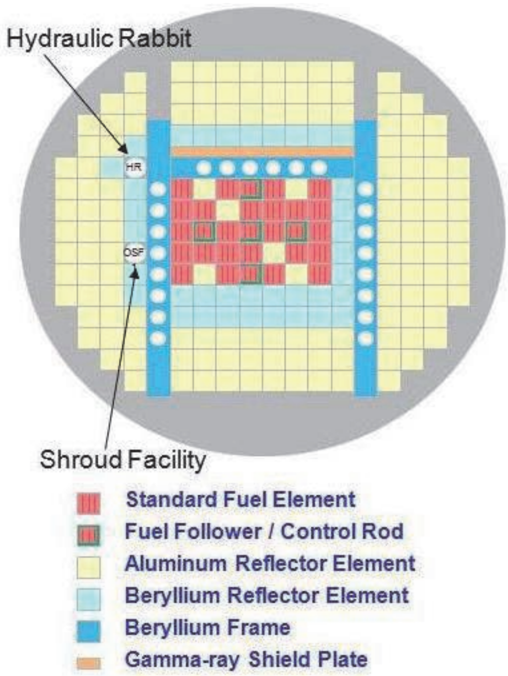

Fig.2 Cross Section of the Core.

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

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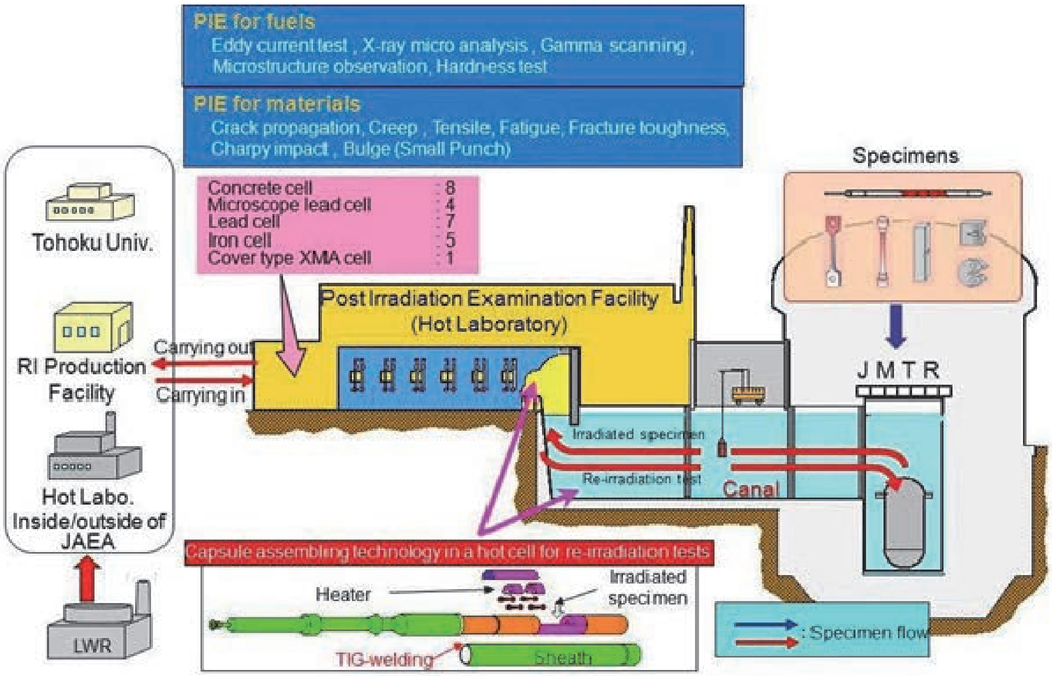

Fig.3 Outline of the JMTR Hot Laboratory.

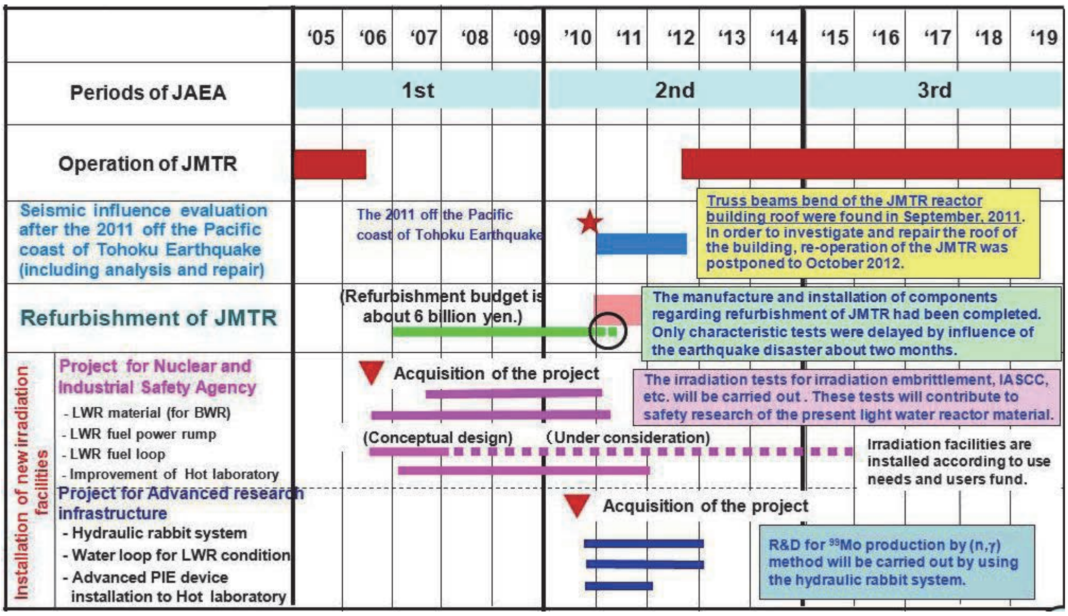

Fig.4 The JMTR Refurbishment Schedule

2. JMTR REFURBISHMENT PROGRAM

Repair and replacement works of the JMTR have been carried out in accordance with the following process (1) to (3).

(1) Investigation of Aged Components

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

(2) Replacement of Reactor Components

Replacement was carried out within the range of licensing permission of the JMTR.

Replacement of the systems and components, such as boiler system, refrigerator for air conditioning system, power supply system, air supply/exhaust system, process control and instrumentation system was 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 Fig.4.

3. FUTURE PLAN OF JMTR

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

1) *Safety research of LWRs,* which includes the aging management of LWRs and so on,

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 ^{99m}Tc for the medical diagnosis medicine,

4) *Education and training* of nuclear scientists and engineers.

Availability factor of the JMTR would increase with increasing those irradiation utilizations. 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 increase up to about 70 %. [4]

Moreover, the JMTR is planned to expand the 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, 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 specialty and should have 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 for 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 of this project 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, installation facilities are water-environment demonstration test facility for LWRs environments, high accuracy time-control irradiation facility, complex type fine texture analyzer, and those facilities would be installed up to JFY 2013 as shown in Fig.5.

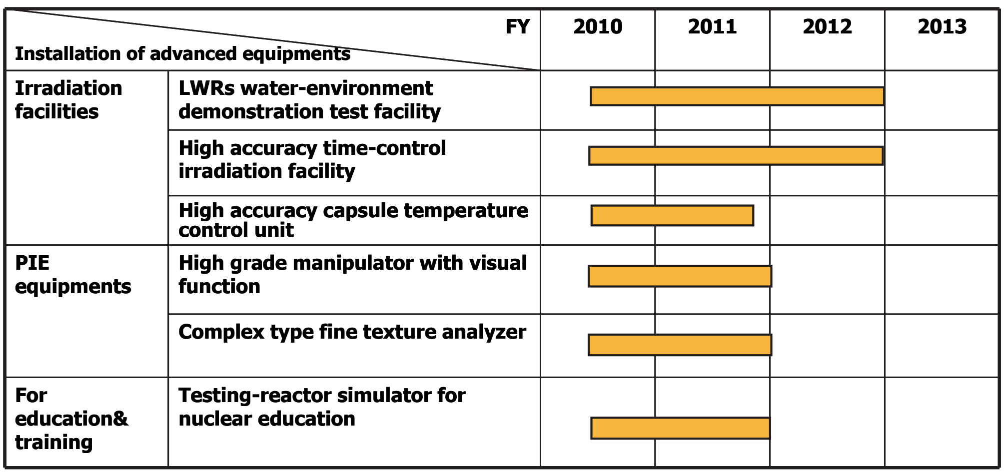

Fig.5 Schedule of the project ‘Birth of the nuclear techno-park with the JMTR’

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 (MTRs). Concept of the MTRs network (World and Asian networks) is shown in Fig.6. Here, as a kernel of Asian MTRs, the JMTR should promote the fuel/material study for LWRs, education & training etc.

For an activity to construct of the world network, the International Symposium on Material Testing Reactors (ISMTR) has been held in every year. In the symposium, information is exchanged in the field of reactor technology such as maintenance as well as aging management, irradiation technologies, post irradiation technologies etc.

3.3. Multipurpose compact research and test reactor

The feasibility study on multipurpose compact 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.7.

This activity is expected to contribute domestic and international human resource developments. Furthermore, as a result of this activity, it is also expected to contribute the construction of nuclear power plants to Asian countries in future

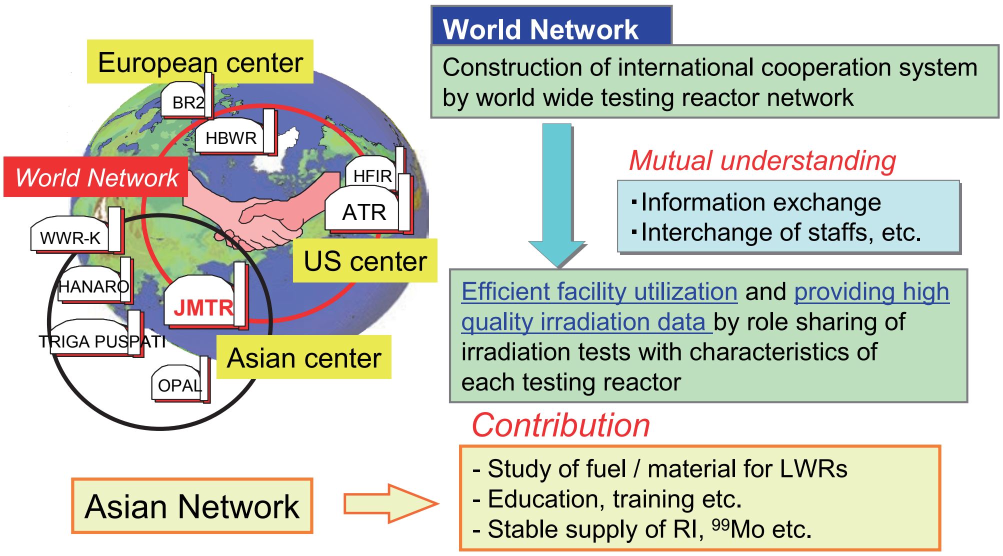

Fig.6 World network and Asian network for material testing reactors

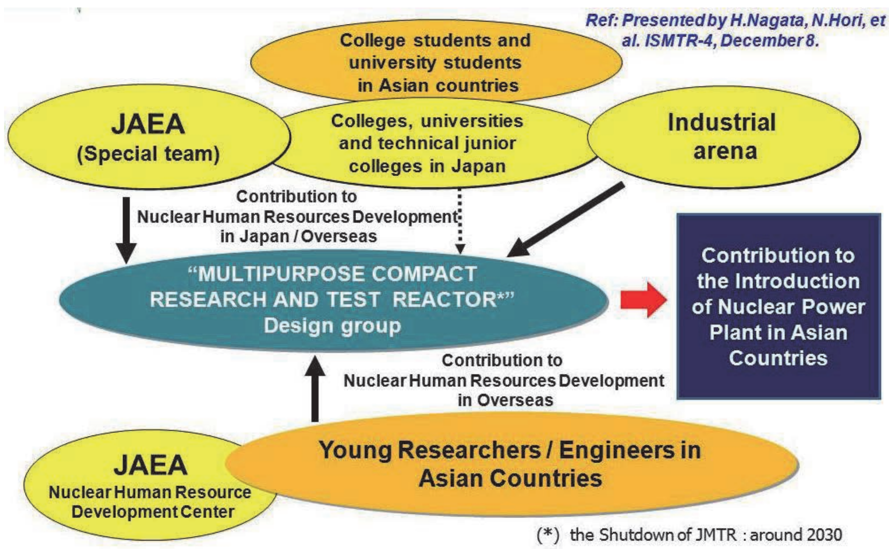

Fig.7 Design and construction of Multipurpose Compact Research and Test Reactor

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 test 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.

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

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2.2 Current Status of HANAO

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HANARO is a 30 MW_{th} research reactor. Its first criticality was achieved in Feb. 1995 and the normal operation started from Dec. 1995. HANARO has operated 2,607 days and generated power of 63,369 MWD until 2011. The reactor trip occurred 79 times from 1997. In 2011, HANARO operated 8 cycles and total 211 days. The reactor trip occurred a time in 2011. The reactor trip by pool top radiation high took place 20 February 2011 because an activated floater separated from the NTD (Neutron Transmutation Doping) facility and rose to the surface of the reactor pool. As a result, a Radiation Alert was issued. The fuel irradiation tests for the SFR (Sodium-cooled Fast Reactor) fuel development, UMoly fuel development and the irradiations for the radioisotope production, material test and NTD-Si production were performed in 2011. For the repair and system improvement, two secondary cooling pumps and HVAC system in the control room were changed. The replacement of the reactor control computer started from this year.

Keywords: HANARO, research reactor, reactor operation history, reactor trip, material and fuel test.

1. INTRODUCTION

HANARO is an upward flowing light water cooled, heavy water reflected open-tank-in-pool type research reactor. Its first criticality was achieved on Feb. 8, 1995 and normal power operation was started from Dec. 1995 after performing the various reactor physics tests for design verification.

The major utilization of HANARO is RI production, fuel and material irradiation tests, beam experiment, neutron activation analysis and NTD-Si production. The reactor core is composed of an inner core inside an inner shell of the reflector tank and an outer core outside the inner shell as shown in Fig. 1. Three hexagonal holes (CT, IR1 and IR2) in the inner core are provided for tests needing high thermal fluxes, while four holes (OR3 ~ OR6) at the outer core are reserved for experiments using thermal and epi-thermal neutrons. Besides, a total of 25 vertical irradiation holes with different sizes are distributed in the reflector region. Seven tangential beam tubes are also deployed horizontally.

HANARO used to operate at 24 MW. All the licensing issues for a 30 MW operation were resolved in 2003 and the tests to confirm the safety of fuel at a 30 MW operation were completed in Nov. 2004. From that time, HANARO has been operating at 30 MW

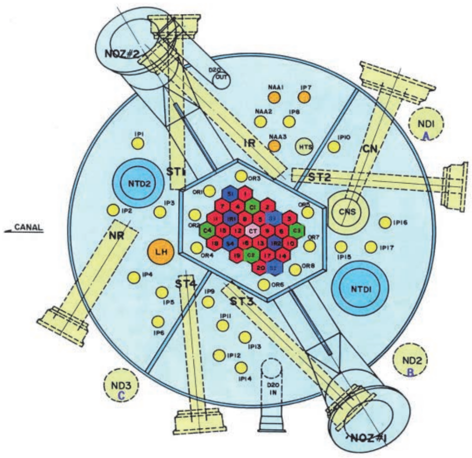

Fig.1. Plan view of HANARO

In this paper, the operation status of HANARO, recent events are described.

2. THE OPERATION STATUS OF HANARO

HANARO operates 8 cycles a year. A cycle is 28 FPDs (Full Power Days). The operation mode is a four-week operation and two-week shutdown. There are six shifts of three staffs each. Each shift is composed of a SRO, RO and system operator.

The operation days and power generation from 1996 to 2011 are shown in Fig. 2. HANARO has operated 2,607 days and generated power of 63,369 MWD until 2011.

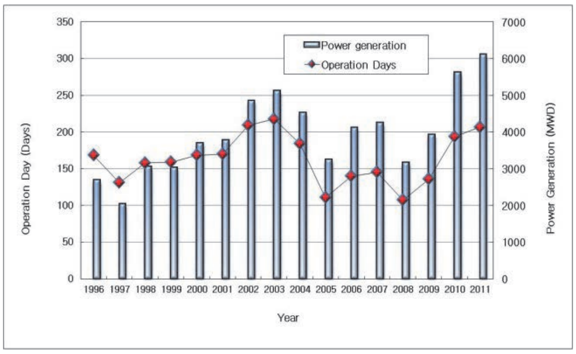

Fig.2. Reactor operation History

The reactor trip history shows in Fig. 3. The reactor trip occurred 79 times from 1997 [1]. Reactor system improvement, accumulation of operation experience and extensive training program for operators have brought a rapid drop in the number of unscheduled trips. In case of 2009 and 2010, the reactor trips were due to FTL(Fuel Test Loop) and CNS(Cold Neutron Source) system commissioning tests.

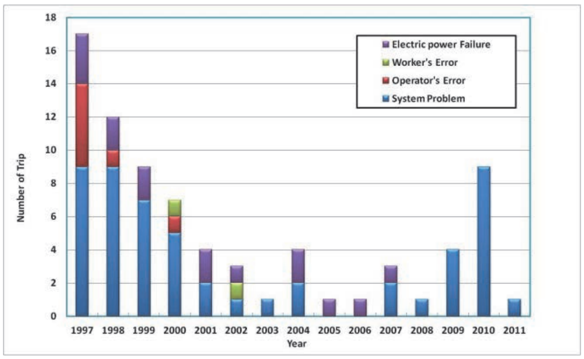

Fig.3. Reactor Trip History

As for the utilization of HANARO, it has been utilized mainly for neutron beam applications, fuel and material tests, RI production, and neutron activation analysis, etc. A significant number of experimental facilities have been developed and installed since the reactor operation started in 1995. For the neuron beam research, NRF(Neutron Radiography Facility), HRPD(High Resolution Powder Diffractometer), FCD(Four Circle Diffractometer) and SANS(Small Angle neutron Scattering) facilities are available for users. CNS system was constructed and commissioning tests were finished in 2010.

For the activation analysis and RI production, NAA and PGNAA(Prompt Gamma Neutron Activation Analysis) are available and ^{131}I, ^{192}Ir, ^{99m}Tc and ^{166}Ho are the major isotopes produced in HANARO. The capsules for material tests and fuel tests were developed and are being used. The fuel irradiation tests for the SFR (Sodium-cooled Fast Reactor) fuel development, UMoly fuel development were performed in 2011.

The commercial NTD (Neutron Transmutation Doping) service is a major work in HANARO since 2003.

3. RECENT EVENT IN HANARO

The reactor trip by pool top radiation high took place 20 February 2011 because an activated floater separated from the NTD facility and rose to the surface of the reactor pool. The trip set point by pool top radiation is 25mGy/h. The maximum value of 6.8×10^5mGy/h was recorded. As a result, a Radiation Alert was issued according to the radiation emergency procedure. An Alert, the lowest level on a three-grade alert system, is usually issued if a radiation leak is found within the reactor hall. The floater was restored to the bottom of the reactor pool, and the surface radiation level was recovered to its normal state. The Radiation Alert was cleared after 7 hours. After this event, the automatic alarm system was installed in the control room to prevent operators from delaying the declaration of Radiation Alert. NTD facilities were redesigned and the related procedures were revised [2].

For the repair and system improvement, two secondary cooling pumps and HVAC system in the control room were changed. The replacement of the reactor control computer started from this year.

After Fukushima accident, the regulatory body reviewed and inspected facilities, documents and procedures including safety assessment with external experts. The regulatory body and external experts ordered to perform reevaluation of flooding effects for the recent extreme weather condition and stress test of reactor building and stack by earthquake.

3. CONCLUSIONS

The Installation of facilities considered in the design stage has been finished. Further work is to maintain and repair the equipments according to aging management schedule for safe operation and effective utilization for RI production, fuel and material irradiation, NTD service, thermal and cold neutron beam facilities.

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2.3 Current Status of JRR-3
- after the 3.11 Earthquake -

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JRR-3 at Tokai site of JAEA was in its regular maintenance period, when the Great East Japan Earthquake was taken place on 11th March 2011. The reactor building with their solid foundations and the equipment important to safety survived the earthquake without serious damage and no radioactive leakage has been occurred. Recovery work is planned to be completed by the end of this March. At the same time, check and test of the integrity of all components and seismic assessment to show resistance with the 3.11 earthquake have been carried out. JRR-3 will restart its operation after completing above mentioned procedures.

Keywords: JRR-3, Earthquake, Recovery Work, Facilities Soundness

1. INTRODUCTION

JRR-3 (Japan Research Reactor No.3) is a light water cooled and moderated swimming pool type research reactor with nominal thermal power of 20MW. Reactor building contains reactor facilities such as a reactor pool, cooling system, instrumentation and control (I&C) system etc. Neutrons coming from the core are transported to a neutron guide hall and several neutron beam experiments are carried out in the hall. Secondary cooling tower receives heat generated in the core and emitted it to the atmosphere. Air with minor radioactivity in the reactor building is filtered and exhausted to the air through an exhaust stack.

JRR-3 has suffered the great earth tremor not previously experienced when the Great East Japan Earthquake with the seismic energy of magnitude 9.0 has occurred on March 11, 2011. At that time, JRR-3 was undergoing regular periodical inspection and the reactor was not operated, but many maintenance people have worked in the reactor building. All workers and JAEA's staff have evacuated successfully just after feeling the strong earthquake. Although commercial electric supply was stopped, necessary minimum facilities were continuously operated with emergency electric generators. There has been checked that no injured person and no increase of radiation dose.

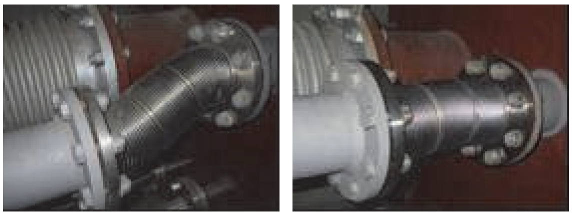
2. DAMAGE AND RECOVERY

It is very important to confirm immediately whether nuclear fuel materials and reactor confinement system are damaged or not. During the aftershocks for a few hours, the reactor pool, nuclear fuels and their storage facilities were checked visually and confirmed to keep their soundness. Although several small cracks were shown on the internal wall of reactor building, they did not result in adverse effect on the integrity of confinement and there were no release of radioactive material to environment.

Some damages visually observed are shown at left side of Fig. 1. Ground around the buildings was sunk about 40cm. The buildings themselves did not sink since they are built on the bedrock. According to the ground sinking, an exhaust duct leading from the reactor building to a stack was slightly damaged at a connection. A liquid nitrogen storage tank, used for feeding nitrogen to experimental facilities, and electric transformers for secondary cooling system were also damaged and leaned. Some of the ceiling panels in the reactor building were dropped.

JRR-3 is in the process of recovery based on the repairing schedule for the utilization with a sufficient extra budget to repair our facilities. Recovery works have been progressing smoothly as shown at right side of Fig.1 and will be completed by the end of this March.

(a) Damage (b) After recovery
Ground sinking around neutron guide hall



(a) Damage (b) After recovery
Flexible pipe in the exhaust duct

(a) Damage (b) After recovery
Ceiling panel in reactor building

Fig.1. (a) Some damages of JRR-3
(b) After recovery work

3. REGULATORY PROCEDURE TO RE-START THE REACTOR

As the 3.11 earthquake measured larger seismic acceleration than that of seismic design of JRR-3, regulatory body has demanded us to evaluate soundness of reactor facilities and report it by reactor re-operation. Several evaluation have been required such as a) the impact on the facilities caused by tsunami, b) the impact in the event of station blackout, c) check and test of all of the structure, systems and components (SSCs), and d) seismic analysis in the light of the knowledge obtained from the 3.11 earthquake.

3.1 Impact of Tsunami

The 3.11 earthquake precipitated a large tsunami of about 15 meter height at the Fukushima Daiichi Nuclear Power Station. The Fukushima power plants withstood the earthquake but the tsunami interrupted the power supply which is necessary to cool down the core. As the JRR-3 is located at the attitude of 19 m, there is no need to take particular countermeasures. For reference, about 5 meter high waves have been observed in Tokai site.

3.2 Impact of Station Blackout

The JRR-3 is operated with the thermal power of 20 MW. When station blackout occurs, the reactor shuts down automatically. The decay heat of JRR-3 is the thermal power of 1.4 MW that is 1/100 of power rector. Fig.2 shows the maximum fuel surface temperature after the auto shutdown by station blackout. The result of analysis says that the core is cooled by free circulation of pumps for about 10 seconds and then cooled by natural circulation. The maximum surface fuel temperature reaches to 123 deg C in about 13 seconds, and then decreases gradually. This shows that integrity of core is kept if station blackout occurs. Pool water evaporates very slightly and it takes about 40 days before the fuels exposed in air.

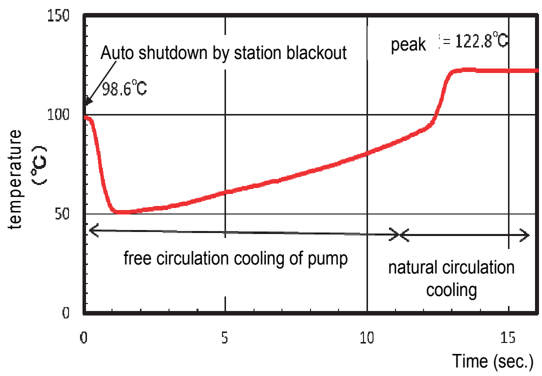


Fig.2. The maximum fuel surface temperature after the auto shutdown by station blackout

3.3 Check and Test of SCCs

Check and test have been carried out for the reactor core facilities, cooling system and I&C system etc as shown in Fig. 3. So far, soundness of the SSCs needed for reactor re-operation has been confirmed. All of the check and test will complete by the end of this March.

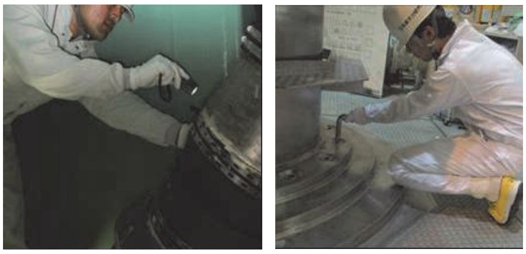


Fig.3. Check and test of all of the SSCs

3.4 Seismic Analysis

The 3.11 earthquake registered 9.0 on the Richter scale, and the intensity was a lower 6 at Tokai. The maximum seismic acceleration of 1183 cm/s^2 in horizontal and of 512 cm/s^2 in vertical were observed at JRR-3. Those are larger seismic acceleration than that of seismic design of JRR-3.

Seismic analysis has been carried out in order to confirm the JRR-3 would have been resistant with the 3.11 earthquake adequately. Simple structures such as reactor building, roof etc. have been assessed as sound enough. Complex structure such as reactor pool is in evaluation. Analysis of other components such as fuel elements, control rod driving mechanism shows there would be no damage.

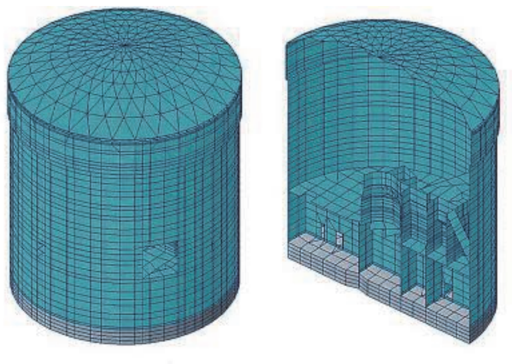


Fig.4. 3D FEM model of reactor building

4. CONCLUSION

Damages by the 311 earthquake would not diminish the safety of the JRR-3. Recovery work mainly for ground sinking has been carried out smoothly. SSCs needed for reactor operation have been checked to be reusable without major repair. Seismic assessment shows almost SCCs are soundness, and only reactor pool is under analysis. We are planning to start a service after the regulatory body confirms the soundness of the facility.

2.4 Recent and Future PIE Activities in KAERI

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KAERI has two PIE hot cell facilities. One is Irradiated Material Examination Facility (IMEF) to examine irradiated nuclear fuels and core structural materials at HANARO, and the other is a Post Irradiation Examination Facility (PIEF) to do the operated PWR nuclear fuels and skeleton. Their activities are categorized by hot cell examinations, new technique developments, maintenance of facility utilities and a public acceptance work. In the past three years hot cell tests were executed over 2,000 times related to R&D projects of the future and next generation reactors such as SMART, U-Mo fuels, I-NERI etc., including 30 rods to be operated in domestic PWR's. To supply PIE data on time to the user schedules new equipment and techniques are being developed for 10 items which include an annealing fission gas release rate, thermal diffusivity measurement, a laser puncturing fission gas collection system and mechanical property measurements on PWR FA skeleton etc. For maintenance of facilities the aged utility equipment are refurbished and exchanged for new ones. The partly cracked and leaking building walls are completely repaired, and the aged electric UPS's, HVAC system parts, heating and cooling system are exchanged for new ones. Additionally efforts to upgrade the public acceptance for facility visiting events, open training courses to the university and graduated students have been made.

Regarding the future of domestic R&D and a power reactor operating plan the demand for hot cell examinations will be steadily increased. Along with them new PIE techniques on TRISO fuels, minor actinide SFR fuels and CANDU fuels etc are required to be developed. New space to install small scale hot cells, and develop for the extreme conditioned test to meet future demands will be taken.

Keywords: KAERI, PIEF, IMEF, HANARO, hot cell, long-term PIE plan, technique, facility maintenance.

1. INTRODUCTION

KAERI has two PIE hot cell facilities. One is Post Irradiation Examination Facility (PIEF) which has been operated from 1982 year the other is Irradiated Material Examination Facilities (IMEF) from 1993 year. IMEF has one pool, seven concrete and one lead hot cells, and performs examinations for the irradiated materials and fuels in HANARO from the national R&D plans. PIEF has three large pools, four concrete and two lead hot cells, and performs for the operated nuclear fuels in domestic nuclear power plants.

PIE facilities have an operation goal to supply PIE data to the user groups on time. To acquire its goal efficiently, they categorize four targets regarding on the working characteristics shown in Fig. 1.

The targets are a production of PIE data, a new technique development, a facility arrangements operation and an upgrade of public acceptance. In this paper the recent activities of them and a future PIE plan are described.

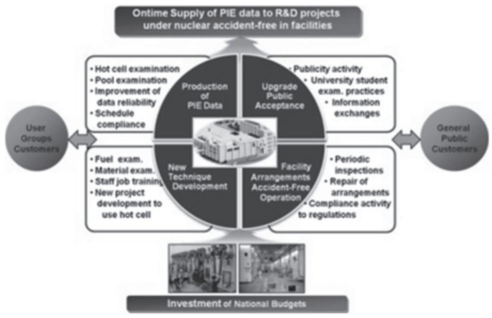

Fig.1 Schematics of a goal and targets in PIE facilities operation.

2. RECENT PRODUCTION OF PIE DATA

PIE facilities in KAERI have been executed hot cell and pool examinations for nuclear fuels, reactor core components irradiated in HANARO or operated in power plants. The capsules in HANARO, core components and PWR rods are transported from a site to facility pools by a cask. In a pool the PWR rods are executed of nondestructive tests. After tests they are loaded to a hot cell, and dismantled to be performed with various hot cell examinations. Fig. 2 shows the PIE flows for capsules and components from power plants.

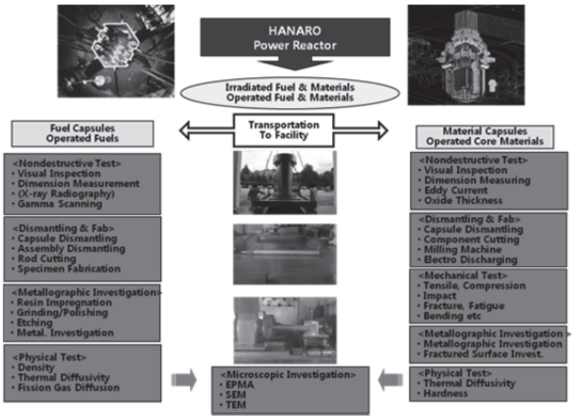

Fig. 2 Examination flows in PIE facilities

According to national R&D programs and power plant operation various PIEs are performed recently as shown in Fig. 3.

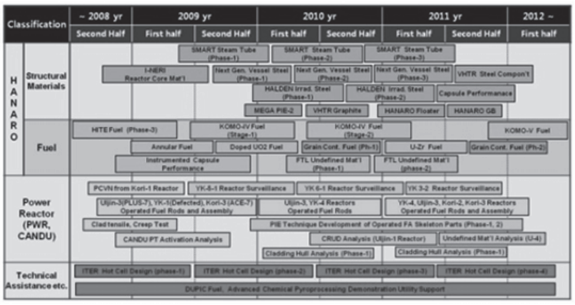

Fig. 3 Recent PIE activities in KAERI

Examinations related to newly developing nuclear fuels were performed such as HITE fuels, KOMO (Korean U-Moly) fuels, annular fuels and grain controlled fuels. From 2001 year the KOMO fuel examinations has been performed through five phases to verify irradiation performance of the developing atomized U-Mo fuels. The molybdenum and additional atomic composition were changed with each phase. The main PIE items were dimension measurements, gamma scanning, metallographic observations, density measurements, SEM and micro chemical analysis using EPMA. Through the examination silicon effects in fuels are verified to decrease the reaction layer in Al-Si base materials as shown in Fig. 4.

Examinations on reactor structural materials were performed such as steam generator (SG) tube materials for SMART (Small Modular Advanced ReacTor), Gen-IV reactor materials, PWR vessel materials for next generation, steel materials irradiated in Halden reactor and VHTR steel components etc.. During 3

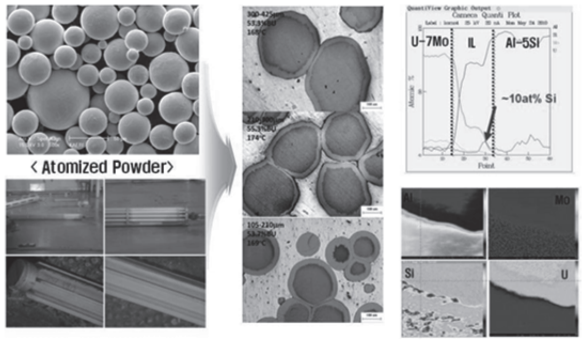

Fig.4 PIE examples of KOMO fuels

years the PIE on the SG tube materials had been performed through three phases according to irradiation fluences. The examination items were tensile, fracture toughness, hardness, thermal diffusivity and density measurements. Through the examinations the design license data for SG tubes were produced to verify excellent properties and the

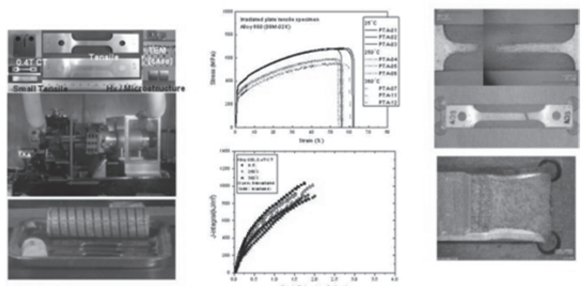

Fig. 5 PIE examples of SG tube materials in a SMART

integrity after irradiations. Fig. 5 shows example of test procedures and test data for SG tube materials.

On the other hand the technical supports of hot cell facility to ITER project were executed based on facility experiences for a long time. From 2009 year projects such as design reviews, feasibility studies and detailed conceptual designs related to the radwaste treatment hot cells were progressed. Fig. 6 shows the conceptual layout of hot cells.

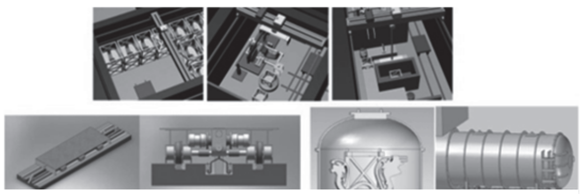

Fig. 6 Example of conceptual drawings for hot cell equipments in ITER project

Defected PWR assemblies and rods during operation in reactors were examined to analyze the causes and to set up the measures to prevent recurrence. Also the surveillance tests for reactor vessels in all domestic PWR's were performed periodically during operation.

3. DEVELOPMENT OF PIE TECHNIQUES

To supply PIE data to users on time the technique development plan were set up and promoted continuously. Fig. 7 shows the recent technique development plan.

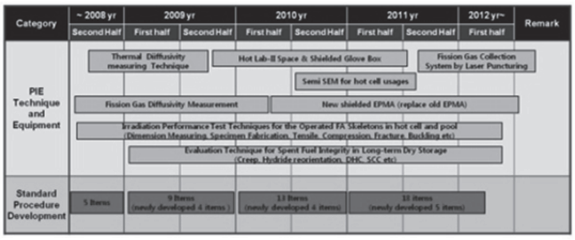

Fig. 7 Status of recent PIE technique and equipment developments

Recently eight PIE techniques and equipments were developed such as thermal diffusivity measurement, Hot Lab-II and shielding glove box, a fission gas collection system, a semi SEM, fission gas diffusivity measurement and a shielded EPMA etc..

The thermal diffusivity system was set up in a glove box at Hot Lab-II shown in Fig. 8. This system could examine irradiation fuels and structural materials. Thermal diffusivities for the irradiated SG tube materials for a SMART project were measured using system recently. For future this system will be useful equipment on the fuel developing projects for SFR and VHTR.

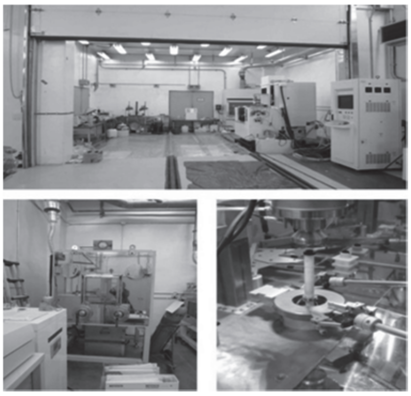

Fig. 8 View of HOT LAB-II and thermal diffusivity tester

Various test techniques of the fuel skeleton components were developed to evaluate operation reliability of newly developed assemblies. Fig. 9 shows examples of developed techniques.

A shielded type EPMA was developed to exchange a long-term operated machine by cooperation with JEOL Company. Many parts in general type machine were reconstructed to shield from gamma ray from radio activated materials according to WDS, specimen exchanged chamber to be operated by remote tongs, a specimen holder, a remote main body and a controller etc.. The developed machine will be installed in HOT LAB-I and started to normal operation in the middle of 2012 year. Fig. 10 shows the reconstructed components in a new machine.

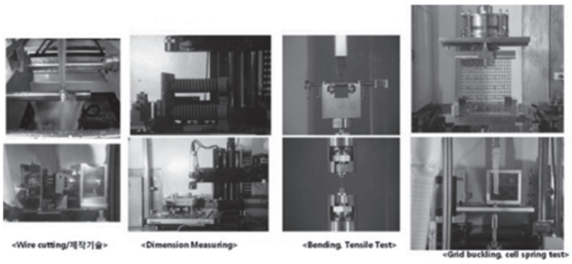

Fig. 9 View of fuel skeleton tests in hot cells

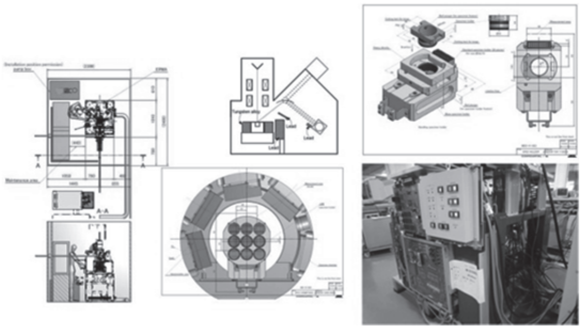

Fig. 10 Reconstructed components of newly developing shielded EPMA

To establish a quick and simple investigation tool for nuclear fuel and materials in a hot cell semi-SEM were developed and installed. Many parts of it were modified to be hot cell usages according to apply shielding techniques to main components located in a hot cell, and remote specimen installation by manipulators shown in Fig. 11.

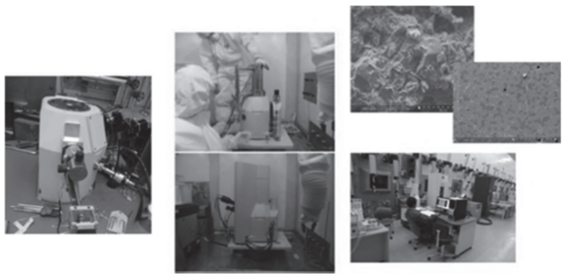

Fig. 11 Installation of SEM in a hot cell

4. MAINTENANCE OF FACILITY ARRANGEMENTS

The PIE facilities are complied to nuclear laws and regulations in Korea. They are subjected to various legal inspections by national and foreign institutes such as a periodic arrangements inspection, a QA inspection and a accounting inspection on nuclear fuels etc. To operate arrangements without troubles many activities were continuously executed such as periodic maintenance inspections, replace and repair works of equipments and compliance works to the laws and the regulations. Fig. 12 shows the work status of facility arrangement maintenance.

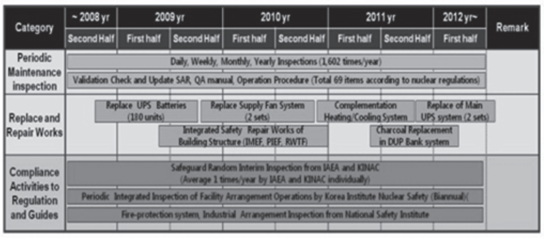

Fig. 12 Maintenance schedule of facility arrangements

Periodic maintenance inspections had been performed more 1,600 times annually such as daily, weekly, monthly and yearly shown in Fig. 13.


Fig. 13 View of periodic maintenance inspections of operating equipments

PIE facilities in KAERI had been aged from long life operation about 20 ~ 30 years. There were cracks, water leaking and paint delaminating in structural walls. Integrated safety repair works were performed during 18 months from 2010 year. The cracked inner and outer walls were repaired and painted, and the roofs were water proved as shown in Fig. 14.

Random interim inspections from IAEA and KINAC were performed a time annually, and industrial fire protection system and arrangement maintenance status were inspected from national safety institute.


Fig. 14 View of Repairing works for cracked wall and roofs.

5. UPGRADE PUBLIC ACCEPTANCE

To upgrade the public acceptance on nuclear industry many activities were performed to general citizens, university students and industrial staffs. Visiting events to the facility were always held to citizens, young generations and children. At the annual national science advertisement events the manipulators were displayed, and given operation experiences to students as shown in Fig. 15.


Fig. 15 View of facility advertising activities

The practical training courses of hot cell examinations were opened to university students who majored in a nuclear science and engineering as shown in Fig. 16.

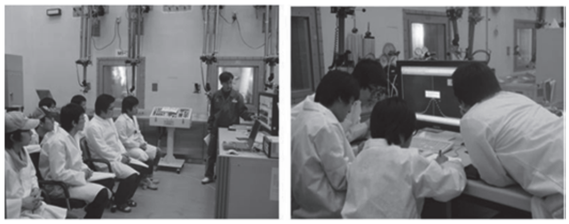
Fig. 16 Experience of hot cell testing for university students

1.

6. FUTURE PIE ACTIVITIES PLAN

An integrated PIE plan is set up regarding to a national nuclear R&D plan, an electric power supply plan for future. According to the fourth R&D plan five strategic targets and 20 core projects were set up. To acquire these targets effectively the facilities produce many PIE data. For the targets of a secure nuclear safety and a future nuclear core technique, PIE data according to material irradiation degradations, on-going operation safety evaluations, CANDU reactors, and SFR fuels, VHTR fuels and cores should be produced. For the targets of a fuel cycle technique and a innovation research, PIE data according to ACP performance, DFDF performance, double cooled fuels, and high strength steel, VHTR core steels, reactor core components should be produced. Furthermore considering the operation of nuclear power plants PIE data according to reactor surveillance tests, defected fuel analysis, new PWR fuel performance and CANDU pressure tube tests should be produced. Fig. 17 shows mid-term PIE plan integrated from national R&D projects and power plant operation plans.

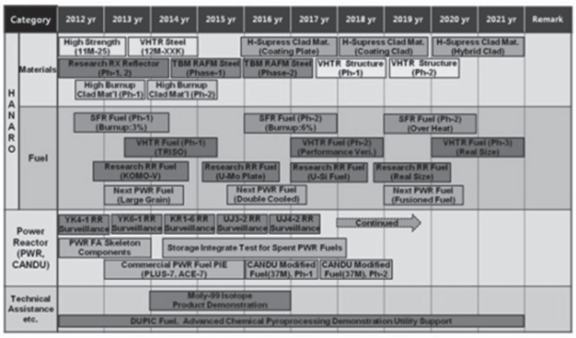

Fig. 17 Future PIE plan of facilities in KAERI

According to the future plan new techniques and equipments in hot cells should prepared to supply PIE data to the user on time. Especially the facilities are in need of fuel examinations in SFR fuel, VHTR fuel, research reactor fuel, new PWR fuel and existing power plant fuels. The required techniques comparing to the present PIE capabilities are shown in Fig. 18.

R&D Project Category	Present PIE capabilities	Required equipment and techniques	Remarks
SFR Fuel Development	Rig Inspection, Dimension Measuring, Gamma scanning, Cutting, Dismantling, Metallography, Density, Thermal diffusivity, Chemical burnup, Sodium-clad mutual reaction	✓Small Volume Fission Gas Collection ✓Atmosphere Controlled Hot Cells	✓ Laser type Puncturing Machine under developing ✓Steel hot cell facility construction under trying to budget
VHTR Fuel Development	Capsule dismantling, Rod visual inspection, Gamma scanning, Metallography, SEM, EPMA	✓ TRISO extraction and handling ✓Micro gamma scanning system ✓TRISO X-Ray machine ✓Electric ✓Fission gas collection in high temp. (KUEPA system)	✓ X-ray radiography system under trying ✓ Various machine establishment under try with R&D team
Research RR Fuel Development	FA visual inspection, dimension measuring, Gamma scanning, Rod/Plate cutting, Metallography, SEM, EPMA	✓ Plate type fuel dismantling machine ✓ Micro gamma scanning system ✓TEM specimen machining system	✓ Planned to develop by annual
New PWR Fuel Development	FA and rod visual inspection, Dimension measuring, Gamma scanning, Density, Metallography, SEM, EPMA	✓ Fission gas collection in very high temp. ✓Fuel meat micro cutting system	✓ Fission gas collection system under developing ✓ Various machine establishment under try with R&D team
Power Reactor Fuel	FA and visual inspection, Dimmension measuring, Gamma scanning, ECT, Oxide thickness, Gamma scanning, SEM EPMA, skeleton component mechanical test	✓ Thermal expansion measuring system ✓SIMS ✓Skeleton mechanical testing system	✓ Skeleton mechanical tester under developing

Fig. 18 Required techniques and equipments for PIE in the future.

According to it the facility are needed for the equipments to do in extreme environments such as very high temperature and controlled atmosphere. For this the facility has a plan to construct a new one equipped small sized hot cells.

7. CONCLUSIONS

Recent PIE activities in KAERI are reviewed according to production of PIE data, new technique developments, arrangement operations and upgrade public acceptance. Regarding the future of domestic R&D and a power reactor operating plan the future PIE demands are reviewed to prepare new techniques. Through this study conclusions are as follows.

1. PIE data had been produced to support more 10 R&D and power reactor operation projects every year , and new test techniques developed more 2 items annually.
2. Accident free operations of facility arrangements had been successfully achieved complying to nuclear laws and regulations, and various advertisement activities been performed to upgrade public acceptance.
3. The future PIE plans are set up on the basis of national R&D plans confirmed by the facility user groups.
4. In next 10 years the facility will develop new hot cell techniques to examine VHTR, SFR and research reactor fuels.

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3. Research Reactor Management

3.1 Safety Improvement Activities in HANARO

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Safety activities in HANARO have been continuously conducted to enhance its safe operation. Great effort has been placed on a normalization and improvement of the safety attitude of the regular staff and other employees working at the reactor and other experimental facilities. This paper introduces the activities on safety improvement that were performed over the last few years.

Keywords: Safety culture, Survey, E-learning program, Safety performance indicators, Peer review

1. INTRODUCTION

Safety of the operation and utilization of nuclear facilities is a matter of the highest priority. Since its first criticality, the safety improvement activities for HANARO have been implemented and the importance of safety management in nuclear activities has been also emphasized. The following activities and events were performed for plant safety improvement;
- Seminars and lectures on safety
- Peer review on the field safety
- Survey on the safety culture
- Preparation of e-learning education program
- Competition for safety improvement
- Development and application of the operational SPI (Safety Performance Indicator)
- Development a computerized business system for the safety information management

2. ACTIVITIES

2.1 Safety-Check-Day

HANARO carries out a Safety-Check-Day periodically. On this day, all the employees gather together and give lectures and seminars related to safety which cover various matters including experience propagation and technical education related to nuclear safety. The materials for Safety-Check-Day are useful for education and they also provide general knowledge and a practical sense of safety. The events of Safety-Check-Day are categorized into following groups;
- Technical lectures and seminars
- Educations of procedures and guides
- Education on general safety issues
- Propagation of experiences
- Improvement of safety culture

A systematic approach and an effective scheme for the safety matters enhance HANARO's safety attitude.

2.2 Peer Review

From 2009, a peer review on the field safety operated by the HANARO Safety Analysis team became a routine activity. A peer review is one of the safety culture activities used to improve safety through the inspection and evaluation of field status. The subjects of a peer review are a new facility, a first operation job, and works under radiation exposure possibility and safety risk. The specialists in charge of safety culture activity visit the working area and evaluate the safety related items. The check list for peer review includes an exit and entrance management, a protection from radiation exposure and contamination, a preparation of appropriate work procedure, a safety status, a safety culture attitude and a function of supervision management.

2.3 Safety Culture Survey

HANARO developed its own safety culture indicators to evaluate and enhance the safety attitude. The main frame consists of 3 organization groups as follows;
- Operating Organization
- Research Organization
- Design Organization

These 3 organization groups include 15 evaluation items. Table 1 shows the evaluation items for the HANARO safety culture [1].

In 2008 a questionnaire was developed based on the HANARO safety culture indicators for measuring safety attitudes. It consists of 68 questions composed of 55 objective questions, 8 subjective questions and 5 basic questions. The subjective questions are for the importance of the indicators, the frequency of training and field inspections, the safety culture

Table 1. Evaluation items for the HANARO safety culture

Evaluation items for HANARO Safety Culture	
A. Operation organization	1. Safety policy at the corporate level
	2. Safety practices at the corporate level
	3. Highlighting safety
	4. Definition of responsibilities
	5. Selection of managers
	6. Relations between plant management and regulators
	7. Review of a safety performance
	8. Training
	9. Local practices
	10. Field supervision by management
	11. Work-load
	12. Attitudes of managers
	13. Attitudes of individuals
B. Research organization	1. Research input to safety analyses
C. Design organization	1. Design review process

activities, the status of the organizational culture and the operational safety performance. The basic questions include the division of duty, age, position and experience. Using the questionnaire, the survey on the safety culture was conducted two times in 2008 and 2010.
The survey is helpful to understand the attitudes of employees and to set the safety culture activities necessary for an improvement of the reactor's safe operation. According to the survey result, the overall safety consciousness and the attitude of the safety culture in HANARO has been improved gradually.

2.4 E-learning education
An E-learning education program on safety was prepared to strengthen the safety attitude for the regular staff, users, external employees, and visitors.
It provides knowledge of the safety culture, the safety rules, radiation protection, laboratory safety, general industrial safety, emergency preparedness and its procedures, administration procedures, radiation safety, nuclear QA, nuclear laws, standards and codes. The course of e-learning includes 7 hours lectures and an examination. The materials for e-learning provide general knowledge and a practical sense of safety and they also are useful for the personnel working in radiation area.

2.5 Competition on safety improvement
Every year a competition for safety improvement is conducted. The purpose of competition is to lead the positive safety improvement activities and to propagate experiences like near accident and good practices. The evaluation basis include following items;
- Level of safety improvement
- Achievement of safety goal
- Clearness of responsibility
- Result of safety achievement
- Prevention of accident & incident
- Minimize dangerous condition
- Contribution on safety attitude
- Education effect
- Participation rate
- Propagation effect
- Application effect
- Reflection to procedure & system
- Leadership
- Driving intention for safety enhancement

According to the evaluation, a number of excellent presentations are selected as a good example.

2.6 Safety Performance Indicator (SPI)
SPI is a tool to assess the overall performance and the safety management status, in combination with other factors such as safety culture, human performance and operation status. HANARO has tried to develop a program for the establishment of safety performance indicators. The Korean Regulatory body (KINS, Korea Institute of Nuclear Safety) has an OPIS (Operational Performance Information System) program for nuclear power plant which suggests 15 performance indicators. The NRC regulatory framework for reactor oversight consists of three key strategic performance areas: reactor safety, radiation safety, and safeguards [2]. Referring to the SPI suggested by KINS OPIS and the NRC Reactor Oversight Process, HANARO has selected 12 indicators [3]. A set of SPI covers the plant's general performance: nuclear safety, environmental safety, utilization safety, and aspects of safety culture and management. Each indicator is estimated through a specific formula. Its safety grade can be evaluated into 4 grades for example excellent, good, average and warning according to the estimation results. The limit value of each grade was determined in consideration of the operating experience. Through reviewing these specific indicators, information of the plant's safety status, its safety parameter trends, and its radiation safety for an effective management of HANARO's safety can be obtained. Last year HANARO started to systematically gather the information on the operation/maintenance data for SPIs evaluation according to its own program. In 2011, the results of the SPI evaluation have shown mostly excellent performance except the indicator of 'Unplanned scram' and 'Emergency preparedness' as shown at Table 2. There was an unplanned reactor shutdown caused by a high radiation of the pool surface due to a malfunction of the NTD mechanism.

Table 2. Evaluation result of SPI

Indicator	Grade
Unplanned scram	Good
Emergency Water Supply System	Excellent
Emergency Ventilation System	Excellent
Radiation Monitoring System	Excellent
Fuel Integrity	Excellent
Reactor Coolant System	Excellent
Reactor Building Leakage	Excellent
Emergency Preparedness	Good
Occupational Exposure	Excellent
Expected Public Exposure	Excellent
Availability of RIPF	Excellent
Availability of Beam Facility	Excellent

The potential applications of the SPI are as follows:

- To identify the results of operation and utilization with quantitative values
- To evaluate the safety performance status
- To promote effective management through a trend analysis of performance
- To provide understandable safety performance information to the public

2.7 ANSIM (Advanced Nuclear Safety Information Management)

KAERI (Korea Atomic Energy Research Institute) has developed a computerized business system termed ANSIM (Advanced Nuclear Safety Information Management), which covers information of work processes, management, safety control, radiation monitoring, and quality control for all nuclear facilities in the site [4]. Some important factors for plant safety, for example 'SPI' and 'Safety culture survey result' are used as input data to the ANSIM system.

3. CONCLUSIONS

Many activities were performed for an improvement of the safety in HANARO since its first criticality. The major concern is how to harmonize the two axes, work achievement and safety. An effort was placed on a normalization and improvement of the safety culture attitude of the regular staff and other employees working at the reactor and other experimental facilities. HANARO will continuously pursue the trends of the safety attitude and the operational safety performance to enhance its safety.

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3.2 Support Required for Safety Management of JMTR in Extended Shutdown

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The Japan Materials Testing Reactor (JMTR) had been operated for 38 years with 165 cycles for various users since the first criticality in 1968. The JMTR has been in an extended shutdown period since August, 2006 for its refurbishment, and will restart in 2012. The JMTR is the only one testing reactor dedicated to the irradiation tests of materials and fuels in Japan. Long-term operation has been strongly requested by the various users. Then, Japan Atomic Energy Agency (JAEA) decided to refurbish the JMTR for long-term operation, and the work started in 2007. In order to improve the reliability and the safety of the reactor, kinds of reviews and inspections were conducted in parallel to the refurbishment. The experiences of the JMTR during the period could be shared with other organizations which currently or potentially have the similar situation. A Periodic Safety Review (PSR) was carried out to confirm the integrity of the JMTR facilities and the 10-year maintenance plan was made in 2004. Before the restart of the JMTR, equipment to be renewed were selected from evaluation on its damage and wear in terms of aging, significance in safety functions, and safety-related maintenance experiences in the past, in order to enhance the operational capability. Renewal work of all facilities was already finished by March 2011, however the 2011 off the Pacific coast of Tohoku Earthquake happened on March 11, 2011. For the influence of the earthquake, the investigation into integrity of JMTR facilities has been carried out until now. As for the safety management during the reactor operation, owner's periodical and daily inspections are supposed to be carried out in order to maintain integrity and reliability of the facility. The performance of the facilities and equipment is confirmed through the inspections. During the extended shutdown period, a special classification of the facilities were made from viewpoints whether their functions are required continuously even in refurbishment period. Maintenance work and periodical inspections, including those by the regulatory authority, are carried out based on the special classification. As for the radiation management, measures have been taken to reduce the workers exposure to a dose level well below the limit determined by the law, based on the concept of radiation protection of the International Commission on Radiation Protection (ICRP).

Keywords: Japan Materials Testing Reactor (JMTR), extended shutdown period, refurbishment, Periodic Safety Review (PSR), restart of the JMTR, renewal, maintenance, periodical inspection, regulatory authority, radiation management

1. INTRODUCTION

JMTR achieved the first criticality in 1968. The JMTR was operated for 38 years from the first criticality to the 165 cycle operation. Periodic Safety Review (PSR) was carried out to confirm the integrity of the JMTR reactor facilities. Moreover the 10-year maintenance plan was made in 2004. After that, the long-term operation has been strongly requested by various users as the only irradiation testing reactor in Japan. Then, Japan Atomic Energy Agency (JAEA) decided the refurbishment of the JMTR for long-term operation, and the refurbishment work was started in FY 2007. The JMTR is in long term shutdown for refurbishment. Before the restart of refurbishment, renewal facilities were selected from the result of the PSR and evaluation in which damage and wear were considered in terms of aging, significance in safety functions, past safety-related maintenance date and the enhancement of facility operation. Restart of the JMTR is planned in 2012 after the refurbishment. This report summarized the PSR, the refurbishment, and the safety management during reactor shutdown of the JMTR, including the periodical inspections by the regulatory authority.

2. OUTLINE OF JMTR

The JMTR is the materials testing reactor with thermal power of 50MW. The JMTR was built to develop domestic power reactors, whose utilization purpose is irradiation tests of materials and fuels, and RI production. It achieved the first criticality in 1968 and started utilization operation in 1970. The JMTR has various irradiation facilities for various irradiation tests. Moreover hot laboratory for the post irradiation examination is connected to the reactor building. Bird's-eye view of JMTR is shown in Fig. 1. As for the characteristic of the JMTR, fast and thermal neutron maximum fluxes are both 4×10^{18} $m^{-2}s^{-1}$, which is the highest neutron flux level in Japan. The irradiation in JMTR can be carried out in a wide variety of irradiation needs such as the acceleration irradiation of fuels and materials, the irradiation of large-sized specimen, the capsule irradiation under controlled temperature and load, the irradiation using the large-sized loop and so on. Moreover since the reactor connected to the hot laboratory with the canal, it is easy and safe to transfer samples to the hot laboratory to carry out the post irradiation examination. Furthermore, it is easy to re-fabricate the irradiated specimen into the capsule, and re-irradiation in reactor can be carried out. Major specifications of the JMTR are shown in Table 1.

3. PERIODIC SAFTY REVIEW (PSR)

"Regulations Concerning the Instalment, Operation, etc. of Research Reactors" of Japan was revised in February 2004. Based on the regulations, results of the PSR including the technical evaluation on aging and the maintenance plan of JMTR reactor facility were reported to the MEXT in March 2005. Considering the 33years of the JMTR operation, the monitoring of aging and evaluation method, etc. was investigated and evaluated based on the past maintenance record as for the technical evaluation on aging. Moreover, the facilities, which were difficult to be replaced and important for safety, were investigated and evaluated whether they can maintain long-term safety operation or not. As for the improvement and maintenance plans, improvement of maintenance activity and the 10-year maintenance plan concerning the maintenance for the reactor facility from results of the evaluation of the operation experience, investigation of the maintenance record and technical evaluation of equipment on aging was mentioned. According to the maintenance plan based on the PSR, it was confirmed that long term integrity of reactor facilities can be maintained by carrying out maintenance activity.

Table 1 Major specifications of JMTR

Reactor type		Light water moderated and cooled tank type
Thermal power		50,000 [kW] (50 MW)
Fuel element	U-235 enrichment	20 [wt%]
	Fuel meat	U_3Si_2-Al dispersion alloy
	Uranium density	4.8 [g/cm^3]
Neutron flux	Thermal neutron flux	4×10^{18}[/m^{2}s] (Max.)
	Fast neutron flux	4×10^{18}[/m^{2}s] (Max.)

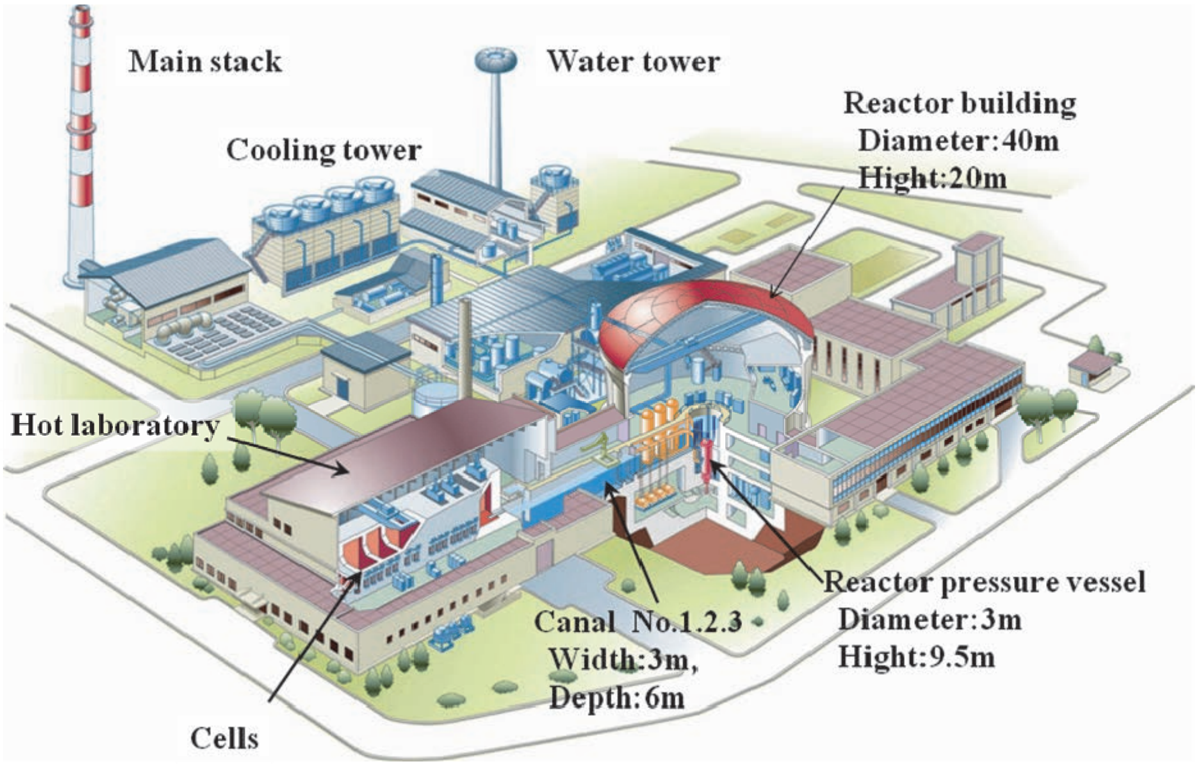

Fig. 1 Bird's-eye view of JMTR

4. REFURBISHMENT OF JMTR REACTOR FACILITY

The JMTR was once categorized as one of the facilities to be shutdown in the middle-term plan of Japan Atomic Energy Agency (JAEA) decided in October 2005. After that, the long term operation has been strongly requested by various users as it is the only irradiation testing reactor in Japan. Moreover it was judged that the refurbishment and re-operation of the JMTR should be carried out steadily by the Council for Science and Technology Policy (October, 2006). Finally, JAEA decided the refurbishment and restart of the JMTR in December 2006, and the refurbishment work has been carried out since FY 2007. The 40-year operation from the first criticality led to the aging of facilities. From results of the PSR, it was confirmed that the integrity of almost all facilities was maintained, and necessary refurbishment schedule for maintaining the stable operation of the facility was planned. Operation of the JMTR after restart is expected until 2030.

4.1.Selection of Renewal Facility

Renewal facilities were selected from evaluation on their damage and wear in terms of aging, significance in safety functions, past safety-related maintenance date, and enhancement of facility operation.

The selection concepts are:

- Replacement priority is given to facilities having aged and worn-out because the JMTR is expected to operate for about 20 years after restarting. Priority is decided with special attention to safety concerns. The availability of appropriate monitoring is an important factor in selecting facilities which will be used continuously after the restart.

- Appropriate maintenance of the facility will be important in considering the long-term operation of JMTR after restart. Facilities, whose replacement parts are no longer manufactured or not likely to be manufactured continuously in near future, are selected as renewal ones.

- Facilities that could be used without renewal (e.g., a reactor building, a reactor pressure vessel, a reactor pool lining, a grid plate and primary cooling pipes) will keep their integrity assessed by continuing current maintenance based on the periodical safety review in February 2005. The JMTR will be maintained after the restart according to maintenance plans based on the results of the PSR. The integrity of the JMTR will be also confirmed through the voluntary periodical inspection of facilities and other checkups. Before the restart of the JMTR, "the special task force for JMTR renewal plan" was established in JAEA Oarai R&D Center to discuss the fundamental idea on renewal, and it confirmed validity of the renewal plan.

4.2. Specifications of Replacing Equipment

Equipment to be renewed was designed to improve reliability and maintenance capabilities. Major renewal equipment is as follows;

(a) Equipment and parts to be renewed in the primary cooling system include the primary circulating pump motors and drive members of the main electric and electromagnetic valves. They are replaced with products having equivalent function.

(b) Equipment and parts to be renewed in the secondary cooling system include the circulating pumps (along with their motors), the auxiliary water pumps (with their motors), main electric valves and cooling tower fan motors. They are replaced with products having equivalent function.

(c) Equipment and parts to be renewed in the UCL system include the circulation pumps (with their motors), water pumps (with their motors), main electric valves and cooling tower fan motors. They are also replaced with products having equivalent function.

(d) Equipment to be renewed in the instrumentation and control system include the reactor control panel (complete renewal), the process instruments (complete renewal), the nuclear instruments (complete renewal) and part of the control rod drive mechanism. They are upgraded to improve operational efficiency and visibility in creating better man-machine interface.

4.3. Renewal Procedure of Reactor Facilities

Outline of the refurbishment of reactor facilities is shown in Fig. 2. Various facilities are to be renewed steadily during 4 years in refurbishment period, and the JMTR is to be restarted in 2012. Then renewal procedure must be planned considering the state of facilities. For example, the process instruments must be in a usable state when the pre-operational inspection of primary cooling system is carried out. Then, renewal of the process instruments must be scheduled before the primary cooling system is used, and each other's processes must not be piled up. The renewal work in the reactor building cannot be carried out when the feed and exhaust air system is under suspension. Then the feed and exhaust air system should be renewed earlier than the other systems. Renewal work outside of the reactor building is scheduled while the feed and exhaust air system is under renewal work. Thus, at the first step, renewal work of the boiler component, the power supply system, the feed and exhaust air system and so on were carried out. Then reactor control system, the control rod driven and the reactor cooling system were renewed. The refurbishment schedule is shown Table. 2

4.4. Renewed Equipment

For the power supply system, its design was started in FY 2007 and renewal works for a control board of the high voltage power supply, the transformer, cable and so on was carried out from April 2008 to January 2009. For the control board of the high voltage power supply, the following were improved from reliability and maintenance performance points; compaction of control board, digitization of the relay and the insertion of the dehumidifier in the control box for prevention of condensation. For the cable, the incombustible cable was selected for the countermeasure to the fire.

The refrigerator for cooling of the reactor building, the boiler for warming and the pure water production equipment were renewed. They have been used for 40 years. For the refrigerator, the absorption system of refrigerator using boiler heat was replaced with a turbo refrigerator, not requiring the boiler heat. The heavy oil cost by the operation of the boiler in summer is reduced largely, and the operation cost of the refrigerator can be reduced by about 40%. For the boiler, by reviewing the necessary heat capacity, and by increasing a heat transfer area per boiler, four boilers were replaced with two boilers.

For the feed and exhaust air system of the radioactive waste disposal facility, the exhauster, the blower and their control circuit were renewed. For the liquid waste disposal facility, the washroom, the drainage pump and their pipe, and the drainage pump of drainage tank were also renewed. The liquid waste disposal facility, washroom, drainage pump and their pipe are normally not contaminated with radioactivity, however they may be contaminated at accidents. The drainage pump of drainage tank treats the radioactive waste water in the tank yard. For the exhauster and blower, fan and motor of emergency exhauster were renewed. The driven mechanism of butterfly valve of feed air system, a part of the butterfly valve of exhaust air system and a part of the exhaust duct of the normal exhaust facility were renewed. For the motor of the blower for supplying air into the reactor building, its type was changed from winding type to squirrel cage type in order to improve the maintenance performance. For the renewal of exhauster, the approval procedure to the regulatory authority was carried out in May, 2008, and approval was obtained in June, 2008. After that renewal work was started, the pre-operational inspection was carried out by the regulatory authority in March, 2009 and passed. For the control circuit, the power relay unit in the control circuit was changed to the sequencer control device. From this change, number of component was reduced compared with previous device. Therefore, maintenance and operation management are simplified, and reduction of the trouble is expected. For the drainage facility, drainage tank and pipes to send the waste water of drainage tank in the reactor building were renewed. Renewal of the drainage pump of drainage tank treating the radioactive waste water in the tank yard was also renewed. Regarding the renewal of the drainage facility, the approval procedure was carried out in March, 2009 and the pre-operational inspection was passed in February, 2010.

Table 2 Renewal schedule of the JMTR reactor equipment

Items		2007	2008	2009	2010	2011	Status
Reactor	Beryllium frame, Gamma ray						Renewed
Instrumentation and Control System	Nuclear instruments, Process instruments, Safety guard circuit						Renewed
Reactor Cooling System	Primary cooling facility, Secondary cooling facility						Renewed
Radioactive waste disposal facility	Feed and exhaust air system, Drainage facility						Renewed
Power supply system	High voltage power supply control board, Transformer, Cables						Renewed
Boiler	Boiler, Refrigerator						Renewed
Pure water production equipment	Degassed pure water production equipment, Pure water production equipment						Renewed

: Design, Manufacture, Installation

For the Utility Cooling Line (UCL) system, renewal of the circulation pump including motor, lifting pump including motor, the main valves, motor of the cooling tower fan and so on was carried out. Regarding the renewal of the circulation pump and the lifting up pump, the approval procedure was carried out in May, 2008 and the pre-operational inspection was passed in March, 2009. These components have been used for over 40 years since the JMTR was constructed. From the view point of the safety operation and component procurement after restart of the JMTR, the existing equipment was renewed with equivalent performance ones. For the renewal of the transfer pump including the motor in the primary cooling system, the approval procedure was carried out in March, 2009 and the pre-operational inspection was passed in February, 2010. For the secondary cooling system, renewal of the circulating pump including the motor, auxiliary pump including motor, main electric valves, motor of cooling tower fan and so on was carried out. The existing equipment was also renewed with equivalent performance ones. Regarding the renewal of the circulating pump and auxiliary pump, the approval procedure was carried out in November, 2008 and the pre-operational inspection was passed in March, 2010.

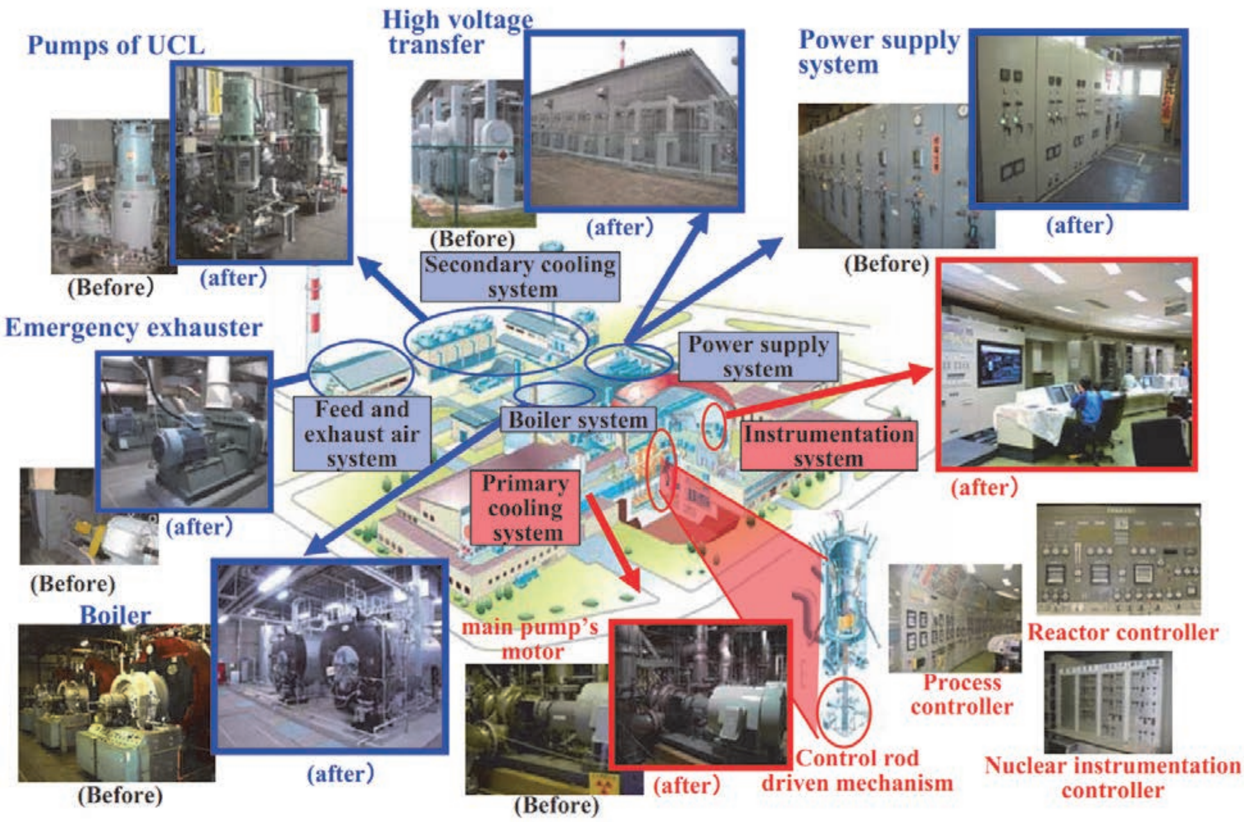

Fig.2 Outline of the refurbishment of reactor facilities

Equipment to be renewed is the instrumentation and control system including the nuclear instruments, the process instruments, the safety guard circuit and part of the control rod drive mechanism. Their reliability was improved by reinforcing with the provision for the noise mixture and the break in the circuit. The instrumentation equipment was classified by type (display, operation switch, etc.). Equipment will be upgraded to improve operational efficiency and visibility in creating better man-machine interface. For the renewal of the instrumentation equipment, the approval procedure was carried out in March, 2009.

For the equipment of primary cooling system, the main circulating pump motors, filler pump including motor, main electric valve's drive mechanism and so on were renewed. They will be replaced with equivalent products for improvement of the reliability. For the renewal of filler pump, the approval procedure was carried out in February, 2009.

5. SAFETY MANAGEMENT DURING REACTOR OPERATION PERIOD

As for the safety management in the reactor operating period, the operational management of the reactor facility, the radiation control and so on are important.

The purpose of the operational management of the reactor facility is to maintain the integrity and reliability of facilities and equipment for the safety and the steady operation of the reactor. In order to achieve this purpose, the owner's voluntary periodical inspection and daily inspection for facilities were carried out, and it is confirmed that the performance of facilities and equipment is maintained. An annual plan of maintenance for the JMTR was made in 1979. The annual plan includes inspection contents and frequencies of inspection required to maintain the performance of the equipment in consideration of the importance, the service conditions, the operating time, the structures, the past operating records of the equipment. The voluntary periodical inspection of facilities has been performed with the annual plan which has been reviewed and revised on a timely basis in accordance with the actual conditions.

The radiation dose exposed to the personnel engaged in radiation work is restricted by laws and regulations. Efforts are made to reduce the exposure based on the ALARA (As Low As Reasonably Achievable) spirit of the International Commission on Radiological Protection with the exposure limit observed in the radiation management as well. As for the management of working environment, the radiation control division measures the dose equivalent rate, the surface density and so on once a week based on the operational safety program. Moreover, the division continuously monitors the radiation, the radioactivity level, etc. in the working environment within the radiation control area by the monitoring device for radiation control. In addition, the radiation monitors and the survey monitors used for measurement of the working environment have been calibrated by the chief of radiation control division based on the operational safety program in every period of the facility's periodical inspection to ensure the reliability of the measurement values.

6. SAFETY MANAGEMENT DURING RENEWAL PERIOD

The safety management of the reactor renewal period is carried out in a way different from that of the reactor operation period. Classification of the facilities and inspections is reviewed and special measures are taken for the renewal work.

6.1. Facility Maintenance during Reactor Renewal Period

In facility maintenance during reactor renewal period, facilities are classified into two groups; (1) facilities needed to keep the function continuously during renewal period and (2) facilities not needed to keep the function continuously during renewal period. The voluntary periodical inspection is carried out on facilities needed for management in particular in group (1). It is confirmed that the function of the facility is maintained through the inspection. Under the renewal work, special attention should be taken into account work with the special measure.

6.2. Periodical Inspection

According to the law, any licensee of reactor operation shall, pursuant to the provision of the ordinance of the competent ministry, undergo an annual inspection by the competent minister concerning the performance of the reactor facilities specified by Cabinet Order. The periodical inspection was carried out on the JMTR from August to November in 2006, while the JMTR was in the extended shutdown period for the refurbishment. Therefore, the inspection is carried out on the equipment whose function must be kept continuously during the long-term shutdown period.

The following facilities were classified into those which shall keep its function even during the long-term shutdown period. These facilities are undergone an annual inspection by the regulatory authority, and are confirmed that their functions are maintained.

(1) The reactor pressure vessel and the main circulation system needed to keep the core covered by the cooling water.
(2) The storage facility of nuclear fuel materials which has storage ability and keeps non-criticality of fresh and spent fuel.
(3) The disposal facilities for gaseous waste needed to maintain the operational environment even during the reactor shutdown period.
(4) The reactor containment facility and ventilating installation needed to maintain confinement function of the radioactive materials.
(5) The diesel generator for power supply equipment needed in emergency when the commercial power supply goes out.
(6) Warning equipments related to the functions.

Concerning the facility classified into the group, its function is not needed to keep continuously during the renewal period, the regular inspection or maintenance is carried out to keep the function of the reactor facility from the viewpoint of the safety.

As for the facilities that need to keep the function continuously during long-term shutdown period, it was confirmed that the functions satisfied the standard of inspection by the regulatory authority in FY2006. The periodical inspection of the facilities during long-term shutdown period is carried out every period, not exceeding one year.

Major inspection items for the JMTR during a long-term shutdown period are as follows; the visual inspection of the new fuel storage facility, the non-criticality inspection and the storage ability verification inspection for the spent fuel storage facility, the leakage inspection in the reactor building and the leakage inspection of the pressure vessel, the leak check of the main circulation system and so on. The MEXT carries out the periodical inspection of research reactors in Japan directly based on the law.

6.3. Voluntary Periodical Inspection

Examinations of the facilities in a wider range are conducted voluntarily during the periodical inspection. Inspections on emergency shutdown for the instrumentation and control system facility are carried out. Furthermore, the performance examination for the emergency interception is carried out once or more in every month. Moreover, the following inspections are carried out to confirm the performance of facilities needed for management in particular in security; the leakage inspection of the pressure vessel and the visual inspection of the nuclear fuel storage equipment. The calibration inspection of the measuring gauges for the nuclear instrumentation, the process instrumentation, the radiation monitors of the cooling system is carried out. However, the calibration on the temperature difference indicator between reactor inlet and outlet, the thermal output meter, the primary cooling water monitor and so on is omitted when there is no operation plan of the JMTR and if it is clear that fuels are taken out of the reactor core during the reactor shutdown period. The inspections of the main pipe's relief valve and the safety valve of pressure surge tank are also omitted. Thus, the content of the inspection is rationalized. The calibration and the inspection on the necessary instruments and the equipment in the voluntary periodical inspection are carried out even during the renewal period of the reactor facility.

6.4. Notes of Work with Special Measures

In accordance with occupational safety and health laws, it is required to assume the special measures for particular work, such as the radiation exposure work, the high-place work, the hypoxia work in the closed place, the heavy component handling work and the asbestos removal work. The laws, regulations and

Table 3 Notes in work with renewal of main equipment and instrument

Details of work	Attention point	Measures
The exchange work of the Beryllium frame and the Gamma ray shielding plate	Radiation exposure and chemical pollution	Reduction in the amount of the radiation exposure of the worker in reference to the past results is planned.
The exchange work of the motor of the main circulating pump and the emergency pump	Heavy load handling	The safety sides such as the crane work and binding with the rope are noted.
The lining repair work of the secondary cooling system pipes inside	Tight place work High place work	・The oxygen density is measured with the oxygen meter. ・The safety belt is worn for prevention of falling.
The exchange work of high voltage control board, transformer and high-voltage cables	Electric shock	・The preventive measure of the electric shock is carried out.
The removal work of the thermal insulator of refrigerator's pipe and thermal insulator of the boiler steamy pipe	Asbestos absorption	・Wearing the dust mask and the measures of the dust prevention are carried out.

rules are observed when such work is carried out. Attention should be paid to the work with the renewal of the main equipment and the instrument is shown in Table 3.

7. SUMMARY

Japan Atomic Energy Agency (JAEA) decided to refurbish the Japan Materials Testing Reactor (JMTR) for long-term stable operation, in order to meet strong needs of various users for the only one irradiation testing reactor in Japan. The JMTR operated for 38 years from the first criticality to Aug. 2006 for 165 cycles. The refurbishment work started in 2007. The Periodic Safety Review (PSR) was carried out to confirm the integrity of the JMTR facilities. For the refurbishment of the JMTR, equipment to be renewed was selected based on the evaluation on its damage and wear in terms of aging, significance in safety functions, safety-related maintenance experiences, in order to enhance the operational capability. Renewal work of all facilities was already finished by April 2011. During the refurbishment period in the extended shutdown, facilities were classified into a special manner from viewpoints whether their functions are required even in the refurbishment period. Maintenance work and periodical inspections by the regulatory authority are carried out based on the special classification in the extended shutdown period. Long-term safety operation of the JMTR will be realized by the refurbishment on the appropriate components with the justified inspection and the renewal work. Until now investigation on integrity of reactor facilities are carrying out do to the 2011 off the Pacific coast of Tohoku Earthquake.

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3.3 Change of Tritium Concentration in Airtight Room by Tritium Removal System

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The High-flux Advanced Neutron Application Reactor (HANARO) uses heavy water as a reflector, sometimes called a moderator, and has an instrument room to control the flow of heavy water. The pumps, pipes, valves for circulating heavy water, various instruments for measuring temperature, pressure and flow rate, and heat exchangers for cooling heavy water are installed in the instrument room. Tritium concentration of the instrument room is higher than those in other areas because heavy water containing tritium is released from the connections in pumps, pipes, valves, instrument and heat exchangers. In 2005, to prevent the diffusion of tritium from the instrument room to other area, a part of instrument room, considering that the release of tritium was high, was isolated by a robust structure and the inside was closed tightly. And this area was named ‘the airtight room’.

Previously some researchers estimated the rates of tritium generation in and release from the airtight room. The change of tritium concentration was also evaluated by measuring and modeling on the tritium removal in the airtight room. According to their investigation, there was a difference between the result from measurement and that from modeling. The measured final tritium concentration was about 10 times higher than the one obtained by the model. They explained that the difference was due to the inhomogeneous mixing of the air in the airtight room during tritium removal.

It was re-estimated that the change of tritium concentration in an airtight room could be predicted well by using a model and equation proposed in the previous study. The tritium concentration and the dew point in the airtight room were continuously measured during the operation of the tritium removal system. The data were analyzed by using the previous model. It was confirmed that there was a definite difference between the measured tritium concentration and the one obtained by equation from the model. It is believed that the difference is not due to the inhomogeneous mixing of the air in the airtight room but due to the change of the generation rate of tritium which would increase as the dew point becomes lower. Based on this assumption, the generation rate of tritium was controlled to have higher value and the change of tritium concentration in airtight room could be more correctly predicted. By using the revised equation, the tritium removal system would be operated more effectively.

HANARO

A Change of Tritium Concentration in Airtight Room by Tritium Removal System

2012. 3.

KOREA ATOMIC ENERGY RESEARCH INSTITUTE

HANARO

View of HANARO

Design Feature

- Type — Open-tank-in-pool
- Power — 30MWth
- Coolant — Light Water
- Reflector — Heavy water
- Fuel Materials enriched — U_3Si, 19.75%
- Absorber — Hafnium
- Reactor Building Confinement
- Typical flux at port nose — $2x10^{14}$ n/cm2/s
- 7 horizontal ports & 36 vertical holes
- Vertical hole for cold neutron source
- Operation Cycle — 23 days@5 weeks

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HANARO

Instrument Room

- To control the flow of heavy water
- Pumps
- Pipes and Flanges
- Valves
- Instruments
  . Temperature
  . Pressure
  . Flow rate
- Heat exchangers
- Tritium concentration is higher than those in other areas

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HANARO

Airtight Room

- Constructed in 2005
- To prevent the diffusion of tritium from the instrument room to other area
- A part of Instrument room was isolated by robust structure.
- The inside was closed tightly.
- Connected with tritium removal system

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HANARO

Tritium Removal System

- Connected in airtight room
- To remove the moisture containing tritium in the air
- Consists of
  . Compressors
  . Condensers
  . Adsorption beds
- First, remove moisture by condensation via compression
- Then, by adsorption onto molecular sieves
- Final dew point of air from tritium removal system : less than - 70°C
- Flow rate of the system : 2.3 m^3/hr.

KOREA ATOMIC ENERGY RESEARCH INSTITUTE

HANARO

Tritium Removal System

Compressor & Condenser

Adsorption Beds

KOREA ATOMIC ENERGY RESEARCH INSTITUTE

HANARO

Prediction Model

- To predict the tritium concentration of air in the airtight room when the tritium removal system is operating.
- Developed by Jong Hak Park and Soo Yul Oh, originally.
  . Estimation of the rates of tritium generation in and release from the airtight room
  . Evaluation of tritium concentration in the airtight room by measuring and modeling
  . The difference between measured value and predicted one (About 6 times)
  . Attributed to Inhomogeneous mixing of the air in the airtight room during tritium removal
- Need to re-estimate the model

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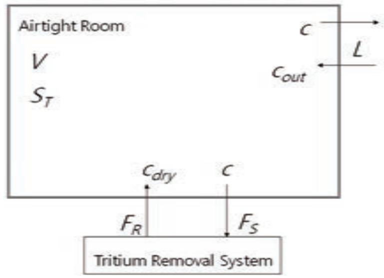
HANARO

Volumetric Radioactivity Balance



V : Volume of airtight room, 146 m^3
S_T : Tritium generation rate of the airtight room, Bq/hr
c : Tritium concentration of airtight room inside, Bq/m^3
c_{out} : Tritium concentration airtight room outside, Bq/m^3
c_{dry} : Tritium concentration of air from tritium removal system, Bq/m^3
F_S : Intake flow rate of tritium removal system, 138 m^3/hr
F_R : Discharge flow rate of tritium removal system, 138 m^3/hr
L : leakage rate of airtight room, m^3/hr

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Equation for a change of tritium concentration with time

$$V\frac{dc}{dt} = c_{out}L - cL + S_T - cF_S + c_{dry}F_R \qquad (Eq.\,1)$$

Since c_{dry} can be ignored,

$$V\frac{dc}{dt} = c_{out}L - cL + S_T - cF_S \qquad (Eq.\,2)$$

In a previous work [2],

S_T = 1.37 x 10^7 Bq/hr , L = 1.3 m³/hr, c_{out} = 8.25 x 10^3 Bq/m³

$$\frac{dc}{dt} = -0.936c + 93909 \qquad (Eq.\,3)$$

General solution :

$$c = 1.00\times10^5 + Ce^{-0.936t} \qquad (Eq.\,4)$$

When c_0, initial tritium concentration of air tight room, is 1.81 x 10^7 Bq/m³,

$$c = 1.00\times10^5 + 1.80\times10^7 e^{-936t} \qquad (Eq.\,5)$$

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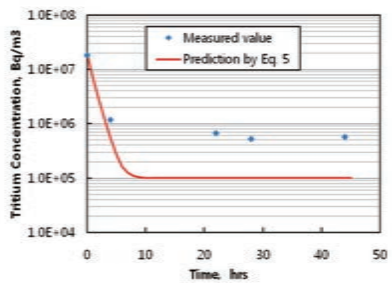
Comparison with Prediction and Measurement



Comparison of prediction by (Eq. 5) and measurement [Park and Oh's result]

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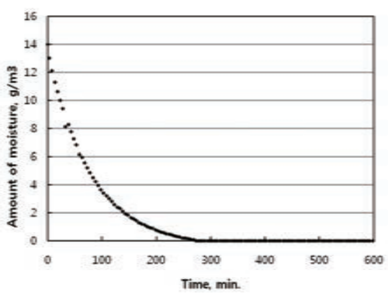
Measurement of Dew Point in Airtight Room



A Change of amount of moisture in airtight room with time

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Equation for a change of tritium concentration with time

$$V\frac{dc}{dt} = c_{out}L - cL + S_T - cF_S \qquad (Eq.\,2)$$

S_T = 1.37 x 10^7 Bq/hr → 9.0 x 10^7 Bq/hr

L = 1.3 m³/hr, c_{out} = 8.25 x 10^3 Bq/m³

General solution :

$$c = 6.59\times10^5 + 1.74\times10^7 e^{-0.936t} \qquad (Eq.\,6)$$

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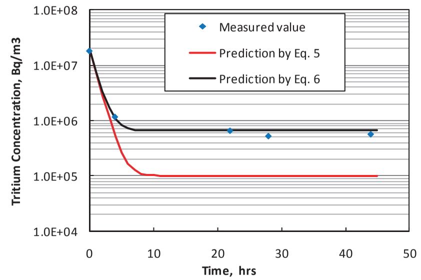
Comparison with Prediction and Measurement



Comparison of prediction by (Eq. 5 and 6) and measurement [Park and Oh's result]

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Equations

When c_0 = 1.81 x 10^7 /m^3, S_T = 1.37 x 10^7 Bq/hr

$$c = 1.00 \times 10^5 + 1.80 \times 10^7 e^{-936\,t} \qquad (Eq.5)$$

When c_0 = 1.81 x 10^7 /m^3, S_T = 9.0 x 10^7 Bq/hr

$$c = 6.59 \times 10^5 + 1.74 \times 10^7 e^{-0.936\,t} \qquad (Eq.6)$$

When c_0 = 1.83 x 10^7 /m^3 and S_T = 1.37 x 10^7 Bq/hr,

$$c = 1.00 \times 10^5 + 1.80 \times 10^7 e^{-0.936\,t} \qquad (Eq.7)$$

When c_0 = 1.83 x 10^7 /m^3 and S_T = 9.0 x 10^7 Bq/hr,

$$c = 6.59 \times 10^5 + 1.76 \times 10^7 e^{-0.936\,t} \qquad (Eq.8)$$

When c_0 = 1.83 x 10^7 /m^3 and S_T = 1.6 x 10^8 Bq/hr,

$$c = 1.17 \times 10^6 + 1.71 \times 10^7 e^{-0.936\,t} \qquad (Eq.9)$$

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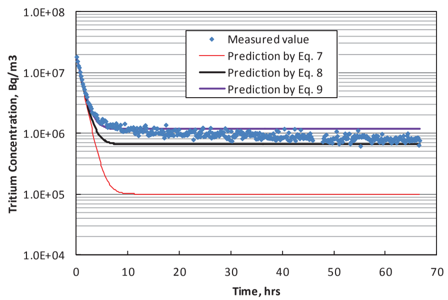
Comparison with Prediction and Measurement



Comparison of prediction by (Eq. 7, 8 and 9) and measurement

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Summary

Re-estimation of previous model

- Previous model
. A difference between predicted values and measured values

- Cause
. The generation rate of tritium was underestimated.
. The generation rate of tritium varies with time.
. It is high at initial stage.
. However, it decreases with time.

- To predict more correctly,
. The generation rate of tritium was adjusted.
→ Revised equation is proposed.

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3.4 Conceptual Design of Multipurpose Compact Research Reactor

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Conceptual design of the high-performance and low-cost multipurpose compact research reactor 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, ②progress of the science and technology, ③expansion of industry use, ④LWR materials and fuels safety research 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 multipurpose compact research reactor 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 materials testing reactors such as tank type, swimming pool type, etc. are operated in the world as shown in 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).

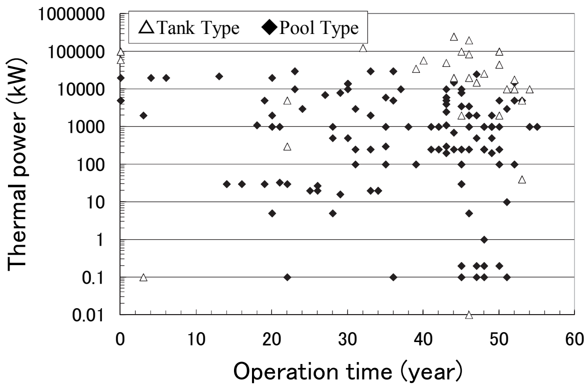

Fig.1. Relation between reactor type (tank type and swimming pool type) and thermal power in world test research reactor.

4. BASIC CONCEPT OF MULTIPURPOSE COMPACT RESEARCH REACTOR

Present technical specification of multipurpose compact research reactor is shown in Table 1.

Table 1 Technical specification of multipurpose compact research reactor

Reactor Type	Swimming Pool Type
Thermal Power	10 MW
Coolant	Light Water
Moderator	Light Water
Fast Neutron Flux	About 1×10^{18} n/(m^2・s) (E > 1 MeV)
Fuel Type	Plate Type
Control Rod	Hafnium
Reflector	High-Performance Reflector (Be, Al)
Coolant Flow Flow Direction in the Core	About 1,200 m^3/h Down Flow
Coolant Temperature	Inlet Temp. 42℃ Outlet Temp. 50℃
Availability Factor	About 70%
Main Use	Medical RI production Industrial RI production 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 8×10^{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 Fig. 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 shows calculation model for COREBN and Fig.4 shows effective multiplication factor (k_{eff}) as a function of reactor operation days, obtained by COREBN).

For the next step, study of core re-arrangement and burn-up calculation will be carried out to increase the performance.

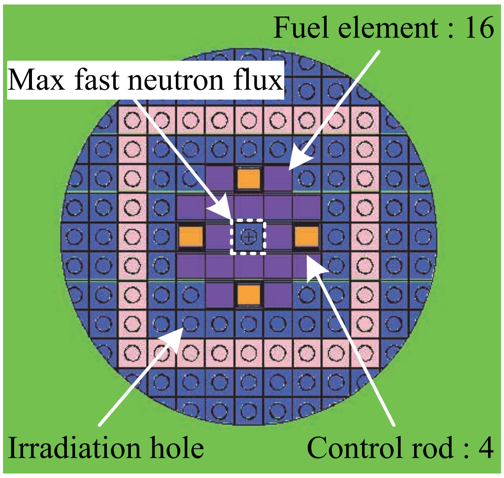

Fig.2. Calculation model by MCNP (standard core)

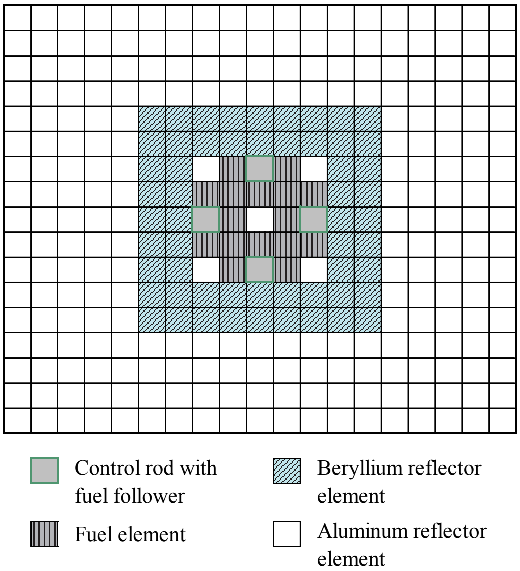

Fig.3. Calculation model for COREBN

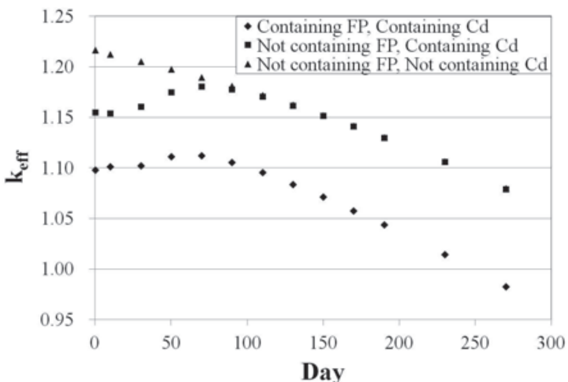

Fig.4. Effective multiplication factor (k_{eff}) as a function of operation days, obtained by COREBN

4.2 Preliminary Thermal Hydraulic Analysis Result 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). On the other hand, 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.7). 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.

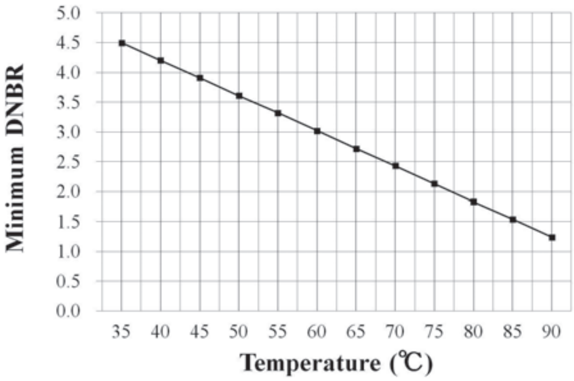

Fig.5. Change of minimum DNBR as a function of core inlet temperature.

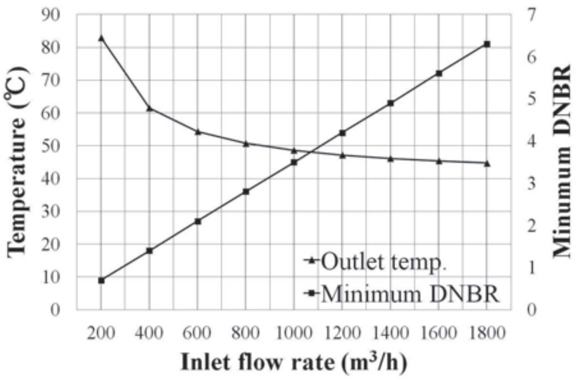

Fig.6. Change of minimum DNBR and reactor outlet temperature as a function of inlet flow rate.

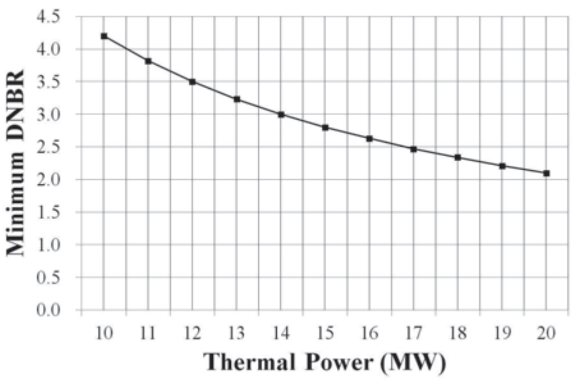

Fig.7. Change of minimum DNBR as a function of thermal power.

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 hydraulic 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 and 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 multipurpose compact research reactor 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 are 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 8×10^{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/h 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 multipurpose compact materials testing reactor should be examined with attention to these tendencies.

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3.5 Status of Ageing Management Program for HANARO

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HANARO is a 30 MW open pool type research reactor which has been operated for 16 years since its initial criticality in February 1995. It has been used for nuclear material testing, radioisotope production, neutron transmutation doping, nuclear activation analysis, and neutron scattering experiments. Recently, new facilities such as FTL (Fuel Transfer Loop) and CNS (Cold Neutron Source) were installed in the reactor. HANARO was originally designed to operate for at least 20 years under full power operating condition, but the actual life time is expected to be much more than the design lifetime by supporting with a safety reassessment based on realistic data and maintenance activities for an ageing management. The conducted inspections, maintenance activities, and the future plan of the ageing management for HANARO are presented in this paper.

1. INTRODUCTION

As key activities of our ageing management, three kinds of system inspections, which consist of Surveillance Inspection (SI), Periodic Inspection (PI) and In-Service Inspection (ISI), are performed to maintain the system, structures and components (SSCs) of the reactor in a safe condition. The SI is for the safety grade components and should be accompanied by a quality assurance procedure, while the PI is for non-safety grade and not necessarily mandatory. The ISI is carried out for the ASME Sec. III components such as reactor structures, reactivity control units, and safety related piping systems. According to the results of inspections, many kinds of maintenances have been fulfilled in appropriate ways. The corrective and preventive maintenances for the primary cooling system, the primary purification system, and the reflector cooling system were performed. The vibration monitoring system for the above 3 systems has been constructed for the predictive maintenance. The fission chambers were replaced and the electric system for the power supply was overhauled for the preventive maintenance. The safety diagnosis and reinforcement for an effective life management and the safety of the building structures such as the reactor building, stack, secondary cooling system equipment room and cooling tower were performed. The upgrade of the instrumentation and control system has been carried out gradually since 2001 to overcome ageing and obsolescence problem.

2. MAINTENANCE HISTORY

The SI and PI were done according to the fixed interval periodically. The ISIs were carried out for the ASME Sec. III components such as reactor structures, reactivity control units, and safety related piping systems. According to the results of inspections, many kinds of maintenances have been fulfilled in appropriate ways. Maintenance activities are for correction, prevention and ageing management program. The major activities for system maintenance were in Table1.

Year	Activities
1996	- Replacement of a secondary pump(P3)
1997	- Installation of a hot water layer system
1999	- Replacement of heat exchanger plate(HX1, HX2)
2000	- Replacement of filler for cooling tower
2002	- Installation of a Window-based Operator Work Station
2003	- Overhaul of diesel generator
2004	- Measurement of reactor vessel inner-shell straightness and visual inspection of SOR/CAR and fuel channels - Overhaul of a primary cooling pump(P2) - Removal of scale in the secondary side of primary heat exchangers(HX2) - Overhaul of a reflector pump(P2)
2005	- Installation of a steel compartment to confine D_2O reflector system components - Replacement of entrance doors to the reactor hall for physical protection - Extended endurance test of SOR for life extension - Overhaul of a primary cooling pump(P1) - Removal of scale in the secondary side of primary heat exchangers(HX1) - Overhaul of a reflector pump(P2) - Replacement of the UPS system
2006	-Replacement of NaI detectors with delayed neutron detectors for a failed fuel detection system -Installation of gamma ion chambers for power measurement and a trip signal replacing the thermal power measurement system -Overhaul of the compressed air system -Safety review and repair of the reactor building and cooling tower buildings
2007	- Upgrade OWS to integrate the FTL system - Installation of two large tanks in the reactor hall for the temporary storage of pool water - Re-structuring of the user rooms in the reactor hall for the improvement of fire-resistance

	- Installation of a voltage sag compensator to prevent the reactor trip due to a momentary interruption of electric power - Overhaul of the electrical system - Replacement of filler of cooling tower
2008	- Replacement of two neutron detectors(RPS-B, RRS-A) - Repair of most welding areas of service water supply piping line
2009	- Upgrade OWS to integrate the CNS system
2010	- Replacement of three neutron detectors((RPS-A,C, RRS-B)
2011	- Overhaul of a reflector pump(P1) - Replacement of two secondary pumps(P1,P2)

Table1 Major Maintenance activities

3. AGEING MANAGEMENT PROGRAM

Ageing management is performed in two categories, a normal scheduled maintenance for obsolesced structures, systems and components and a special maintenance for a physically aged SSCs. Problems by obsolescence can be expected by experiences and information from the markets and peers. And ISI and special inspection can monitor effects and performances of physical ageing of the SSCs.

Component	Inspection Item	Period
Reactor Structure & Beam tube	Visual inspection - reactor structure & beam tube	5 year
	Measurement of vertical straightness - inner shell of reflector tank	10 year
	Measurement of diameter - fuel flow tube	10 year
RCU	Visual inspection - CAR & SOR	5 year
	Wear inspection - carriage and track	10 year
	Measurement of diameter - shroud tube - CAR & SOR	10 year
Neutron detector housing	Torque test	5 year

Table 2 ISI plan for the reactor and RCU

terms of an ageing effect. Several mechanical structure and components in HANARO are classified into ASME code Section III components which require ISI during a reactor operation. The period of ISIs is in table 2. Hanaro is developing ageing management matrix based on the maintenance history and ISI results. And it will include results of special inspection and assessments such as periodic safety review which will be done in near future.

4. CONCLUSIONS

Many activities for a corrective maintenance and an ageing management including upgrading of the SSCs have been conducted during 16 years. And also the ISI program for HANARO was performed for the purpose of safe operation and lifetime assessment of the reactor. The ageing management is deployed as a management program already. And more systematic and effective program will be developed using an ageing matrix which includes comprehensive history data for operation and maintenance of HANARO.

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3.6 Core Cooling during Flow Reversal in Downward Flow Research Reactor

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For a high power research reactor with downward flow, thermal margin should be maintained even at flow stagnant condition during the flow reversal following reactor trip or shutdown. According to the reactor power level, it is determined whether auxiliary or emergency pumps are installed or not in the primary cooling system to avoid dangerous situation against a loss of flow transient due to pump off. Hence, a reactor power level which may cool the core without emergency pumps has been examined from the survey and reviews of existing reactors and relative studies. From the practical point of view, it seems probable that the reactor can be safely cooled without fuel failures during the flow reversal up to 10MW of a typical downflow research reactor with MTR type fuels. However, considering a licensing and a conservative safety design, continuous cooling by emergency pumps should be considered for more than 10MW reactor.

Keywords: Research reactor, Downward flow, Flow reversal, Emergency pump, Natural circulation

1. INTRODUCTION

In a high power RR (research reactor) with downward flow at the core, one concern of thermal hydraulic design is to maintain a safety margin while the downward forced flow changes its direction to upward flow for core cooling by natural circulation. Fig. 1 shows a typical fuel and coolant temperature behaviors during the LOFA (Loss of Flow Accident) in a downward flow RR, and it indicates a second peak of fuel temperature during the flow inversion which is higher than the first peak. Normally, the flow reversal is expected to occur at around 6~20s after pumps stop, depending on the power level and cooling system design. For a RR with high power density, the decay power during this period is still quite high for the fuel cooling in a stagnant flow condition. So, auxiliary or emergency pumps are usually installed in the primary cooling system (PCS) or other way to continuously supply coolant to the core is employed to secure thermal margin for shutdown cooling and against transient during the LOFA. First peak temperature is limited by the reactor trip by the reactor protection system and is not a significant problem.

If auxiliary or emergency pumps have such a safety function, a reliable safety class power such as a safety class diesel generator or uninterruptable power supply (UPS) should be connected to them. If flywheels attached to the pumps can manage this transient, auxiliary pumps supplied by a safety class power are not needed and it can simplify the facility design and reduce the construction cost as well.

From the survey of existing reactors, it is expected that the reactor power level which needs such emergency pumps against the LOFA are around 10MW. More exactly speaking, it depends on the fuel heat flux and geometry. So the possible reactor power level has been estimated from the reviews on existing reactors and related studies. It should be decided to satisfy both the economic and the safety aspects. This issue will be discussed here.

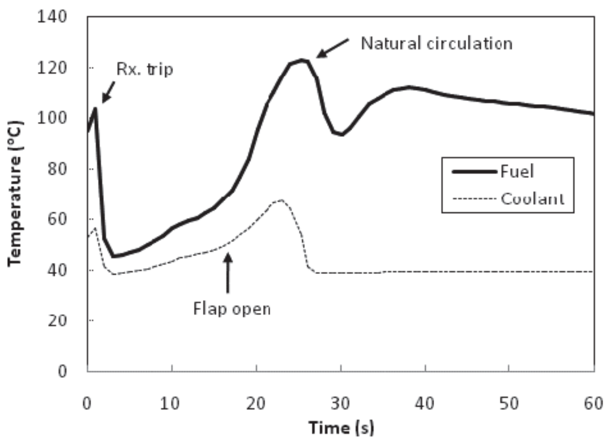

Fig.1. Typical temperature behaviors at LOFA

2. PRACTICES IN EXISTING REACTORS

Among existing RRs, 5 MW RRs with the downward flow, such as IEA-R1 in Brazil and MNR in

Canada, usually have no auxiliary or emergency pumps in the PCS to continue coolant flow against pump down. On the other hand, downflow RRs with more than 20MW reactor, such as JRR-3M, NBSR and FRM-II, have countermeasures against the LOFA such as emergency pumps.

For 10MW RRs, the tank type LVR-15 [1] has an emergency pump against LOFA while the pool type PKRR-1 with typical MTR fuels has no such emergency pump and the core is cooled by natural circulation after reactor trip. Flow reversal in the PKRR-1 was predicted to occur at 8.5s after loss of flow [2, 3]. IRT-1 in Libya has an emergency tank that acts similar to flywheel on pump, which can maintain downward flow for a longer time (around 60s) [4]. Table 1 shows thermal hydraulic parameters of those 10MW downflow RRs. Although fuels of the LVR-15 and the IRT-1 are concentric rectangular type, it can be regarded as the plate type from the thermal hydraulic point of view.

As the PKRR-1 is currently in operation at 10MW without emergency pump, it is expected that fuels may be cooled during the flow inversion if the peak heat flux during the normal operation is below 1000 kW/m^2.

Table 1 Thermal hydraulic parameter of 10MW RRs

Reactor	LVR-15	PKRR-1	IRT-1 (LEU)	IRT-1 (HEU)
Fuel type	ITR-2M	MTR	IRT-4M	IRT-2M
# of Ass.	28	28	28	16
Coolant gap Thickness (mm)	4.50	2.10	1.85	4.50
Active length (mm)	600	600	600	600
Press.(kPa)	120	125	125	125
In. temp(^{o}C)	46	38	45	45
Peak heat flux (kW/m^2)	890	940	998	1680
Power Density (kW/l)	120	125	125	125

3. STUDY BY R.S. Smith et.al [5]

3.1 PARET Calculations

They simulated flow inversions in a downward flow RR with typical MTR type fuels by using the PARET code. The application of PARET code to the flow inversion condition was assessed against a flow inversion experiment in BR2. The assumed reactor characteristics simulated by the PARET are as below. The assumed core has larger heat flux than the reactors in Table 1.

- No. of fuel assembly : 17
- No. of plates per a fuel assembly : 16
- Fuel meat/clad thickness : 0.51mm/0.51mm
- Core active length : 600mm
- Plenum height : 300mm
- Coolant channel gap : 2.39mm
- Pressure : 1.5 bar
- Inlet temperature : 20^{o}C
- Peaking factors: axial (1.31), radial (1.905)
- Average heat flux at 10MW : 460kW/m^2

Fig. 2 presents the calculation results of peak cladding temperature at a hot channel during the flow inversion with core power. Coolant coastdown time constant is 2s. At a given power, the results may depend on the coastdown time and plenum height. Assumed coastdown time constant of 2s is reasonable and can be easily realized by a flywheel in practical engineering. Plenum height is also reasonable comparing to existing reactors. From Fig. 2, the calculated peak clad and coolant temperatures are 114^{o}C and 96^{o}C, respectively. The maximum heat flux during the normal operation would be 1130kW/m^2 (=860kW/m^2 x 1.31). Hence, at the flow reversal moment (6.9s after initiation of coastdown), the peak heat flux is expected to be around 60kW/m^2 (5.3% of initial heat flux). For this condition, their conclusion was that nucleate boiling would be expected for flow inversion situation in a 10MW reactor, but excessive fuel temperature would not result as long as the bulk coolant remains subcooled (not saturated).

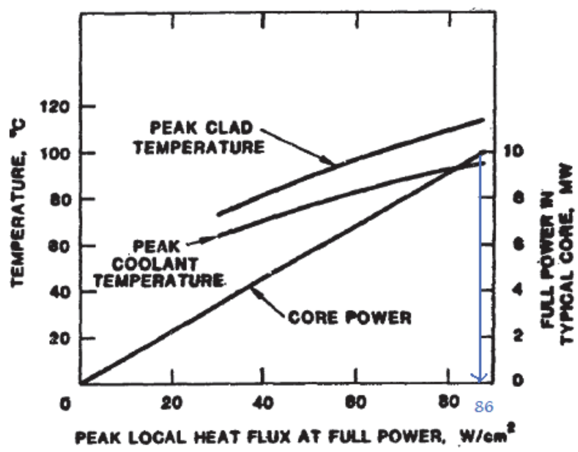

Fig.2. Temperatures and core power with heat flux

3.2 Flow Inversion Experiment in BR2

At the flow inversion test in the BR2, the core flow was decreased with a coastdown time constant of 7.2s just after the reactor trip, and the flow inversion was occurred at around 25s after initiation of the transient. As the maximum heat flux was 3700kW/m^2 at nominal condition, the maximum heat flux would be around 160kW/cm^2 (4.3% of initial power) near at the flow inversion condition (See Table 2). In this condition, fuels were safely cooled without CHF (Critical Heat Flux). In practice, there may be some margin since the measured cladding temperature is below ONB temperature and coolant temperature is also below saturation temperature. It is noted that the BR2 fuel is curved plate type and has a coolant channel gap thickness of 6~7mm which is around 2~3 times wider than that of typical MTR type fuel.

4. ESTIMATION OF COOLABLE POWER LEVEL WITHOUT EMERGENCY PUMP

For a given geometry, actual heat flux and CHF at stagnant flow condition are critical parameters to estimate thermal margin during the flow reversal. So a probable heat flux for an assumed 10MW reactor is compared with the CHF at stagnant condition in order to understand the coolable power level without emergency pumps.

4.1 Expected Maximum Heat Flux at 10 MW Reactor

With an assumption that 10MW reactor core consists of 16 standard and 4 fuel follower fuel assemblies of the JMTR fuel [6], then the average heat flux of the assumed core is 0.332 MW/m^2. If we assume a total peaking factor of 3, then the maximum heat flux is 0.996 MW/ m^2. However, as the power density of this core is 110kW/l, it can be increased by a more compact core, considering existing research reactors. And if we conservatively consider various uncertainties related to the design and analysis, then the local peak heat flux is expected to reach up to 1.5 MW/m^2 at maximum for 10MW reactor. Power density may depend on the utilization purposes and design, particularly, for the number of irradiation holes.

4.2 Power Variation(Decay Power)

If ANS 5.1-1973 [7] is applied to the core decay power calculation, the core power after the reactor trip will be varied as in Table 2. Heat flux variation for a 10MW reactor with a maximum local heat flux of 1500kW/m^2 is listed together. It is noted that the flow reversal would generally occur at around 6~20s during the LOFA in a downward flow research reactor depending on the flywheel size and reactor power.

Table 2 Decay power variation (ANS curve)

Time (s)	P/P0	Heat flux*(kW/m^2)
0.1	0.0675	102
1	0.0625	98
2	0.0590	88
4	0.0552	82
6	0.0533	80
8	0.0512	76
10	0.0500	75
20	0.0450	68
40	0.0396	59
60	0.0365	55
80	0.0346	51

* Heat flux variation in a RR with a maximum local heat flux 1500kW/m^2

4.3 Comparisons with CHF at Stagnant Condition

M.Kaminaga et.al [8] proposed a CHF correlation for a research reactor with plate type fuels based on the CHF experiments, which can be applied to the flow reversal conditions to establish a natural circulation. The CHF correlation proposed for stagnant or very low flow conditions is as below,

$$q^*_{CHF} = 0.7\frac{A}{A_H}\cdot\frac{\sqrt{W/\lambda}}{\left\{1+(\rho_g/\rho_l)^{1/4}\right\}^2}\cdot\left\{1+C.\Delta T^*_{SUB,in}\right\} \quad (1)$$

If we calculate by using eq.(1) the CHF at the stagnant condition for a typical MTR type reactor and BR2 experiment described in section 3, CHF are evaluated as 73kW/m^2 and 147kW/m^2, respectively. The PARET calculation results make sense because the heat flux of 60kW/m^2 at the flow reversal is lower than the predicted CHF of 73kW/m^2. But the actual heat flux of 160kW/m^2 in BR2 is higher than the estimated CHF of 147kW/m^2. This implies that the correlation of eq.(1) may give a little bit conservative prediction since fuels were maintained the integrity in the actual operational test. One of the reasons seems that eq.(1) was developed in blocked conditions while the coolant can actually be flowed into the flow channel from bottom in BR2. That it, when the slug generated in the flow channel moves upward, the coolant can flow into the channel from the bottom plenum in practice while the coolant can be flowed only from the top plenum and is limited by the flooding in the experiments for CHF correlation.

Fuel dimension of BR2 fuel is a little different from the typical MTR type fuel. So, if eq.(1) is assumed to be applicable, 160kW/m^2 in BR2 may be equivalent to 81kW/m^2 (=(2.4/6)/(600/762)*160 kW/m2) in a typical MTR fuel RR (i.e. assume that flow gap thickness changes 6mm to 2.4mm, active length from 762mm to 600mm). That means that the fuel integrity would be maintained up to 81kW/m^2 during the flow reversal in a typical MTR type reactor. If flow reversal occurs at 10s after the initiation of coastdown, heat flux of 81kW/m^2 corresponds to 10.8MW (=81/75*10MW) in Table 2. Considering that the integrity of BR2 fuels was maintained in the experiment and measured cladding and coolant temperatures are less than the ONB and the saturation temperature, it seems that fuels can be cooled during the flow reversal up to 10MW. However, if we have to consider the safety margin of 33% (= correlation uncertainty limit of 1.5), the allowable reactor power will be less than 7.2MW (=10.8/1.5) with a peak heat flux of 1180kW/m^2 during the normal operation.

On the other hand, Fig. 3 presents the analysis results of the DNBR (same meaning with CHFR) during LOFA in JRR-4 by using THYDE-W code. Flow reversal occurs at around 14s after the initiation of coastdown, and the minimum DNBR is predicted to be 2.19 by eq.(1). This implies that CHF will occur at the hot channel if the initial reactor power of JRR-4 is 7.7 MW (=3.5MW x2.19). [For reference, as the average heat flux of JRR-4 is 150kW/m^2, the local peak heat flux at flow reversal is expected to be 25.2kW/m^2 (=150 kW/m^2 x 3.5 x 0.048) with an assumption of total peaking factor of 3.5 (including hot channel factors) and the CHF is 55kW/m^2 (=25.2 kW/m^2 x 2.19) at this point.]. JRR-4 fuel has a flow gap thickness of 2.25mm and an active length of 750mm. If the JRR4 fuel is replaced with a MTR fuel of 2.4mm flow gap thickness and 600mm active length, the CHF at stagnant flow condition will be increased 33% (=(2.4/2.25)/(600/750)), i.e., 73 kW/m^2 and the reactor power can be increased to 10.2MW. However, if we consider the safety margin of 33%, the reactor power will be less than 6.8MW (=10.2/1.5) with a peak heat flux of 1120kW/m^2 during the normal operation.

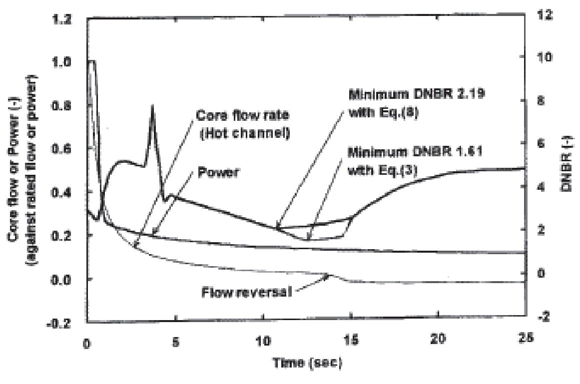

Fig.3. DNBR variation during the LOFA in JRR-4

4.4 Simulation by RELAP5/MOD3.3

To compare with above evaluations, a LOFA for a typical MTR fueled research reactor has been simulated by using RELAP5/MOD3.3. The given reactor characteristics are as below,

- No. of fuel assembly : 14(standard)+4 (follower)
- No. of plates per a fuel assembly : 21
- Fuel meat/clad thickness : 0.51mm/0.38mm
- Core active length : 600mm
- Coolant channel gap : 2.4mm
- Pressure : ~1.5 bar
- Inlet temperature : 35^{o}C
- Flow velocity : 5.4 m/s
- Coastdown time constant : 2s
- Total/Axial peaking factors : 3.5/1.35
- Peak heat flux : 1150kW/m^2

The calculation results on fuel and coolant temperature behaviors are shown in Fig. 4. Peak fuel temperature touched the ONB temperature (120^{o}C), but the coolant temperature at the hot spot is far below saturation temperature (116^{o}C) and there is no void in the flow channel. But, the minimum CHFR calculated by using the eq.(1) is 1.35 as shown in Fig. 5. As eq.(1) is not built-in RELAP5/MOD3, CHFR is calculated by using control variables. From the thermal hydraulic point of view, there would be considerable margin to bulk nucleate boiling, and the excessive fuel temperature is not expected. For a reactor with nominal peak heat flux of 1450kW/m^2 (25% higher than 1150 kW/m^2), hot channel coolant experiences bulk nuclear boiling as shown in Fig. 4 and the MCHFR by eq.(1) reaches 0.9 as shown in Fig. 5. Hence, the potential for the excessive fuel temperature increase become large. But, the RELAP5/MOD3 code predicted still considerable margin in the MCHFR. So, it is conservatively thought that the fuel integrity will be maintained during flow reversal if the nominal peak heat flux is less than 1150 kW/m^2.

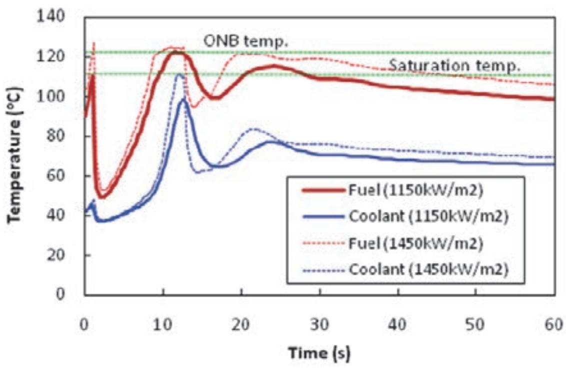

Fig.4. Temperature behaviors at LOFA

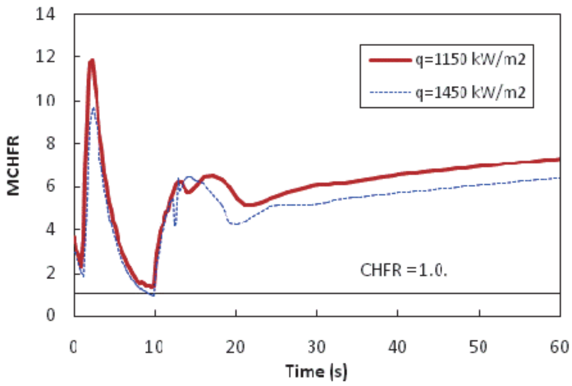

Fig.5. MCHFR during LOFA

5. CONCLUDING REMARKS

From the practical point of view, it seems probable that a downward flow research reactor with typical MTR fuels can be safely cooled during the flow reversal up to 10MW without emergency pumps. But, unfortunately, it is difficult to definitely confirm this since experimental studies on the fuel temperature behavior during the flow reversal are sparse.

On the other hand, if a licensing is concerned, the reactor power will be lower more than around 33% (=1/1.5) with the use of conservative assumptions such as peaking factors and various hot channel engineering factors. In this case, probable reactor power, which can be cooled without auxiliary pumps during flow reversal, will be around 7MW with a peak heat flux of 1150kW/m^2 during the normal operation. If a peak heat flux of 10MW reactor is less than 1150kW/m^2, it would also be probable.

Hence, if a reactor power is more than 10MW or up-rating power for more than 10MW is expected in future, cooling by emergency pumps should be considered in a downward flow research reactor.

It is noted that this remarks are based on the application of eq.(1) which seems to give conservative results for the flow reversal transient in a research reactor.

ACKNOWLEDGEMENTS

Authors would like to acknowledge to Dr. Tsuchiya, Dr. Nakamura, and Colleagues in irradiation engineering section of neutron irradiation and testing center of Oari, JAEA for the support of this work.

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3.7 Status of Utilization of Beam Ports and Irradiation Holes in HANARO (2009~2011)

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HANARO is a 30 MW_{th} multipurpose research reactor. The normal operation started from 1996 after commissioning test in 1995. HANARO has 7 beam ports including a cold neutron beam port, 7 irradiation holes in the core, 19 irradiation holes including two NTD (Neutron Transmutation Doping) holes, a LH(Large Hole)and 3 NAA facilities in the reflector tank. An irradiation hole in the core is used for fuel test loop.

HANARO has the facilities such as HRPD, HIPD, FCD, RSI, NRF, REF, 40 m and 18m SANS for beam utilization. The user has increased from 2010 after CNS (Cold neutron Source) facility was constructed and commission test was finished. The irradiation holes in the core are used for production of industrial radioisotope such as Ir-192 and irradiation for material or fuel test. For the material test, the structure materials of the PWR reactor vessel to extend life time were loaded several times in the core with instrumented capsule. The irradiation holes in the reflector tank are used for NTD silicon and medical radioisotopes production. The radio isotopes were produced about 1,416 Ci for medical and 224,000 Ci for industry in 2011. In case of NTD silicon irradiation, silicon ingots of 21 tons were irradiated in 2011. For the fuel test, FTL(Fuel Test Loop) facility completed in mid-2008, and system test was completed by the end of 2009. But the problem occurred during system test. To date, efforts to resolve the problem has been.

In this paper, the status of utilization of the neutron beam facilities, irradiation holes and NAA is described in HANARO.

Status of utilization of beam ports and irradiation holes in HANARO (2009~2011)

In Hyuk Kang,
Won Ho In, Kyeong Hwan Lim,
Sang Hyeen Lee, Tae Hwan Kim
KAERI
HANARO Management Division

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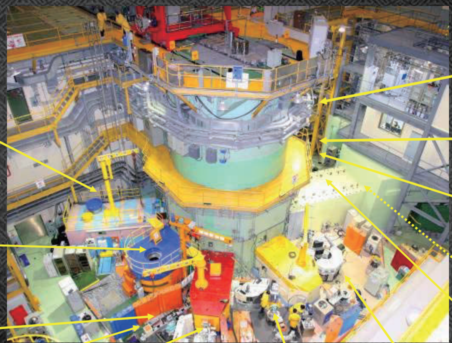
CONTENTS

- Introduction of HANARO's Beam Ports and Irradiation Holes
- Production of NTD
- Production of Radioactive Isotope
- Utilization of PTS
- Utilization of Capsule for Material and Fuel test
- Utilization of Beam Ports
- Future Utilization

2

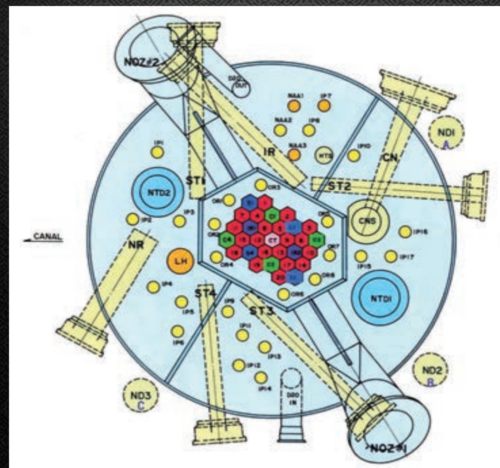
Beam Ports(1)



IR Port
Ex-Core Neutron Irradiation Facility (ENF)
ST1 Port
Prompt Gamma Neutron Activation Analysis(PGAA)
Residual Stress Instrument(RSI)
CN Port
Small Angle Neutron Scattering (SANS)
Cold Neutron Guide
NR Port
Neutron Radiography Facility (NRF)
ST4 Port
Triple Axis Spectrometer (TAS)
Neutron Reflectometer (REF-V)
Bio-Diffractometer (Bio-D)
Neutron Reflectometer (REF-H)
ST3 Port
High Intensity Powder Diffractometer (HIPD)
ST2 Port
High Resolution Powder Diffractometer (HRPD)
Four Circle Diffractometer (FCD)

3

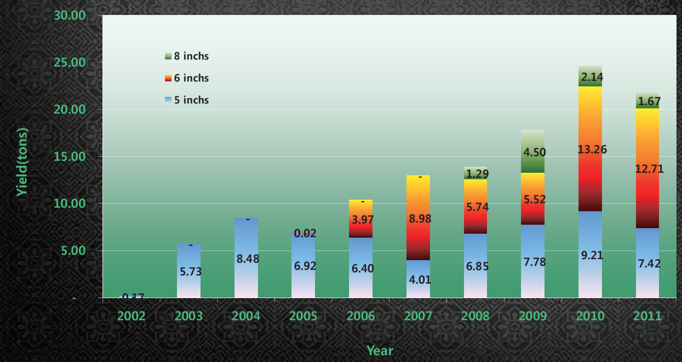
Irradiation Holes



- IR, CT, OR
  - Capsule, RI
- IP, LH
  - RI
- NAA
  - PTS
- NTD
  - NTD1 : 6, 8
  - NTD2 : 5, 6
- HTS
- CNS

Production of NTD

Production



5

Production of RI(1)

Utilization

- Medical
  - I-131, Ho-166
- Industry
  - Ir-192, Co-60
- Research/etc
  - Au-198, Cr-51, P-32, Tc-99m…

6

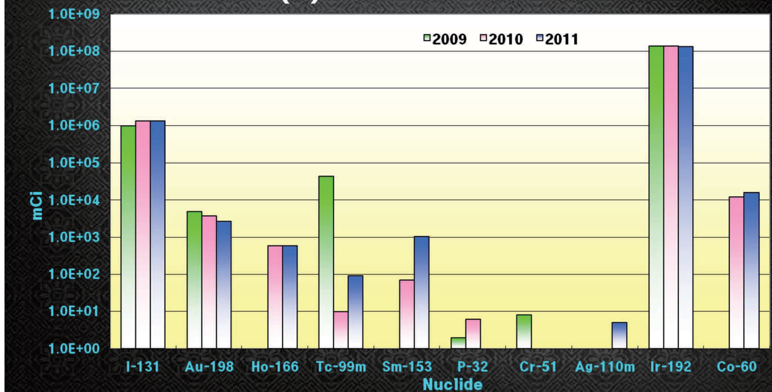
Production of RI(2)

Production(1)

7

Production of RI(3)

Production(2)



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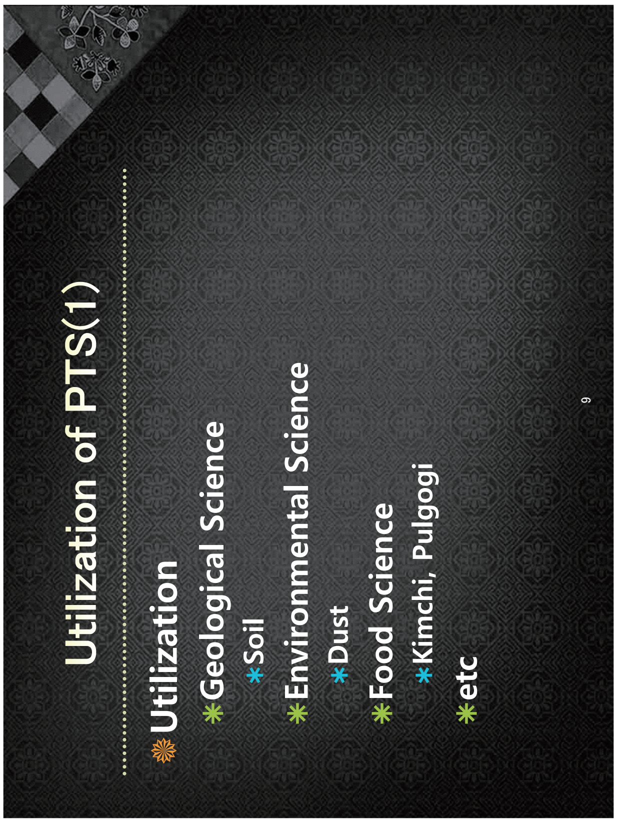
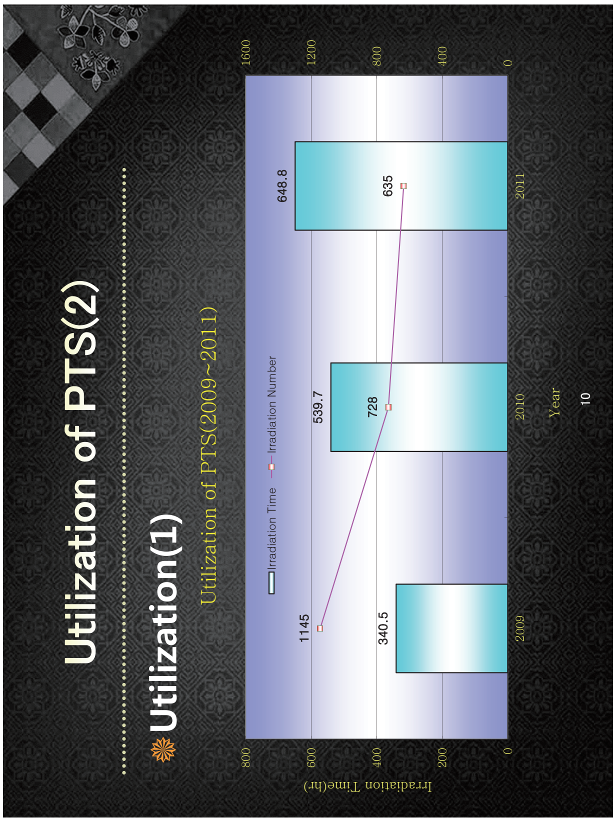

Utilization of PTS(1)
Utilization
Geological Science
Soil
Environmental Science
Dust
Food Science
Kimchi, Pulgogi
etc
9


Utilization of PTS(2)
Utilization(1)
Utilization of PTS(2009~2011)
Irradiation Time
Irradiation Number
Irradiation Time(hr)
Year
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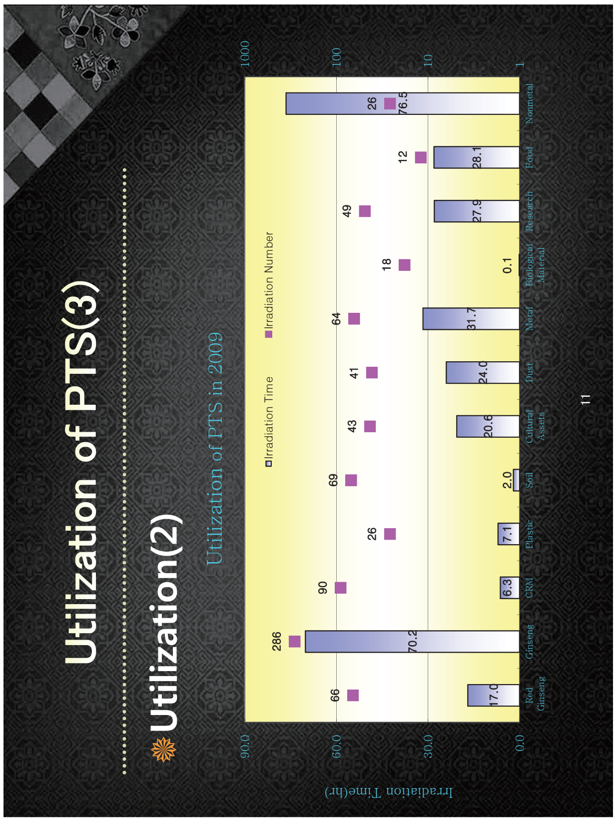
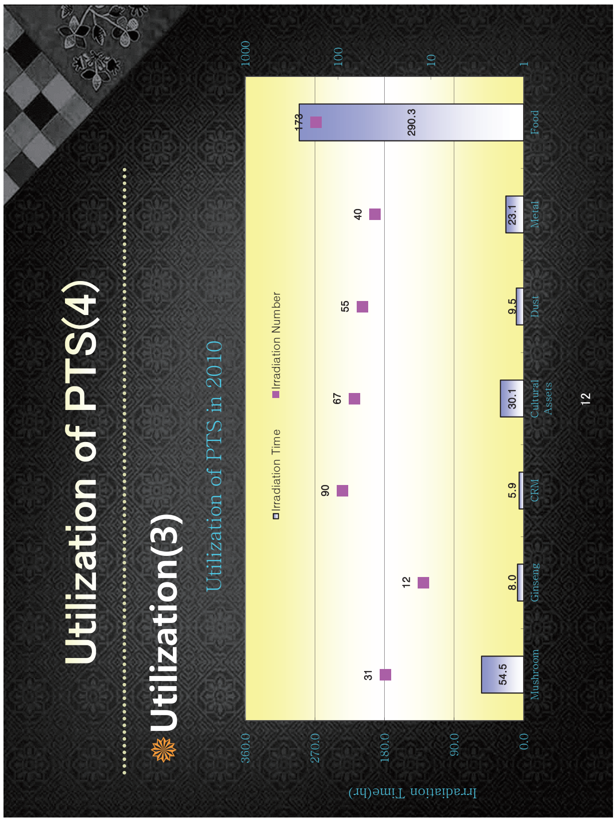
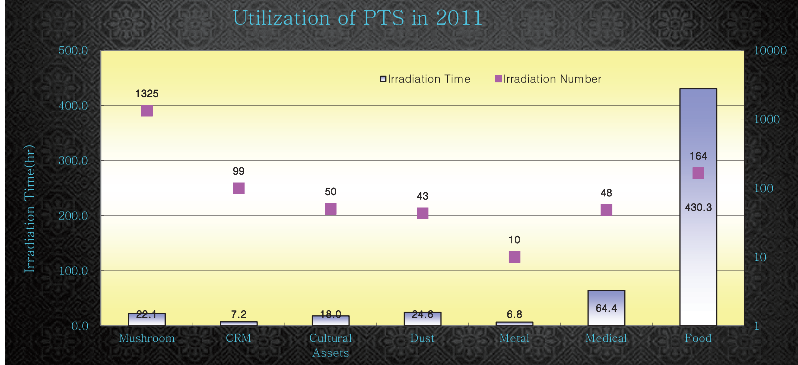
539.7
648.8
1145
728
635
2009
2010
2011
10


Utilization of PTS(3)
Utilization(2)
Utilization of PTS in 2009
Irradiation Time
Irradiation Number
Irradiation Time(hr)
17.0
70.2
6.3
7.1
2.0
20.6
24.0
31.7
0.1
27.9
28.1
76.5
66
286
90
26
69
43
41
64
18
49
12
26
11


Utilization of PTS(4)
Utilization(3)
Utilization of PTS in 2010
Irradiation Time
Irradiation Number
Irradiation Time(hr)
54.5
8.0
5.9
30.1
9.5
23.1
290.3
31
12
90
67
55
40
173
12

Utilization of PTS(5)

Utilization(4)



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Utilization of Capsule(1)

Purpose

- Investigation of Material Damage by Neutrons & Radiation
  - Safety Assessment of Nuclear Reactor Materials
  - Improvement of Reactor Safety & Efficiency

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Utilization of Capsule(2)

Capsule of Fuel

Fuel	Period by 30MW	Burn-up(%U235)	Use	Note
KH08C-001	127 days	47.1 %	RR	
KH09C-001	146.3 days	37.7 %	RR	Occupied
DUOCAP08	97.6 days	25.1 %	PWR	
APFCAP-A	724.6 days	92.2 %	PWR	
SMIRP1	221.6 days	20.7 %	SFR	
HYBR-001	85.7 days	17.6 %	PWR	Occupied
KF08E03A	6.5 days	Just testing	FTL	

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Utilization of Capsule(3)

Capsule of Material

Instrumented Capsule	Materials	Use
08M-02K	SA508, model alloys	Property evaluation of RPV model alloy
08M-09K	Fe-Cr-Mo alloy	Property evaluation of VHTR structural material
09M-02K	Alloy 690	Property evaluation of SMART material
09F-08K	UO2 pellet	R&D for high temperature irradiation technology of Fuel pellet
09M-09K	STS304	R&D for high temperature irradiation technology
10M-01K	Alloy 690	Property evaluation of SMART material
10M-15K	Modified Cr-Mo alloy	Property evaluation of VHTR structural material
11M-03K	Alloy 690	Property Evaluation of SMART material
11M-22K	STS304, C-C composite, Zircaloy-4	Property evaluation of high temperature material
11M-25K	Modified SA508	Property evaluation of RPV model alloy

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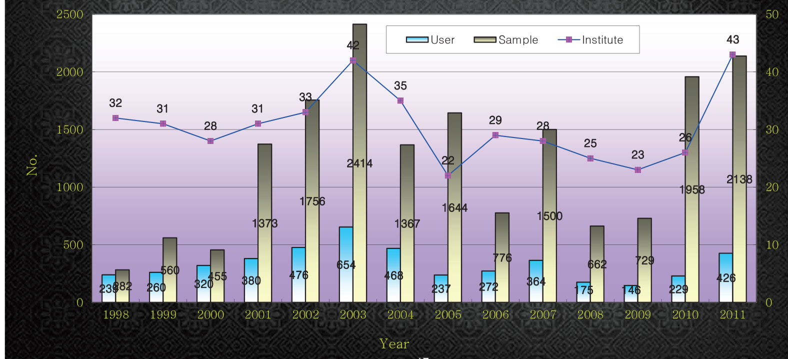
Utilization of Beam Ports

Utilization

Status of Utilization in Neutron Beam



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Future Utilization

- Irradiation Holes
  - RI, NTD : Increase of Production by The 2nd Research Reactor
  - PTS : Target irradiation for conventional NAA & radioactive tracer/RI production
  - Capsule :Development of New fuel and material for NPP & RR
- Beam Ports
  - Expansion of research related to the CNS

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Thank you

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4. Irradiation Technology (1)

4.1 Development of New In-pile Instrumentation at JMTR

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Development of instrumentation which can use under severe accident condition is important issue for the purpose to cope with severe accident at nuclear reactors. And also to improve the quality of irradiation tests data and to increase the reliability of safety management system of reactors, the development of new instrumentation is key issue. JAEA is developing several in-pile instrumentations to conduct irradiation tests at JMTR. This study includes the developments of three new instrumentations and describes the characteristics of the instrumentations. These are ECP sensor, new water level indicator and in-reactor observation system using Cherenkov light.

Keywords: ECP sensor, water level indicator, instrumentation Cherenkov light, in-situ experiment, IASCC, JMTR.

1. INTRODUCTION

It is strongly required to develop instrumentation technology for nuclear reactor vessel system and spent fuel pool system under severe accident condition in existing Light Water Reactors (LWRs). And in order to maintain and enhance safety of LWRs in long term operations, proper understanding of irradiation behavior of fuels and materials is important. And it also requires the development of high quality in-pile instrumentation. For these purpose, the development and improvement of instrumentation technology has been performed using Japan Materials Testing Reactor (JMTR) and JMTR Hot laboratory, the representative materials testing reactor facilities in Japan.

And the ECP sensor [1], the new water level indicator [2] and the in-reactor observation system using Cherenkov light [3] have been developed based on irradiation techniques improved at JMTR.

This study describes the current status of those instrumentations.

2. DEVELOPMENT OF THE ECP SENSOR

It is planned to perform in-pile IASCC growth tests of irradiated stainless steels by using the JMTR.[1] 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$\sqrt{m}$) 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) . Figure 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

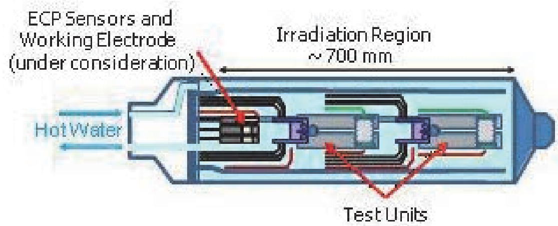
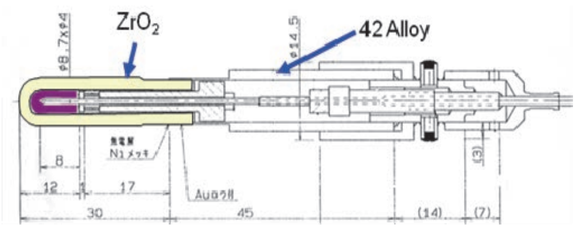


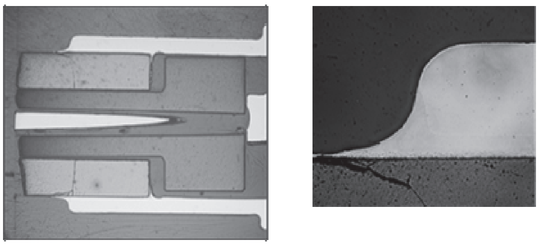
Fig.1 Schematic drawing of the in-pile test.

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. Figure 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.



(a) Structure of ECP sensor



(b) Details of cracks at the joint part

Fig 2 Observation of the crack in conventional ECP sensor.

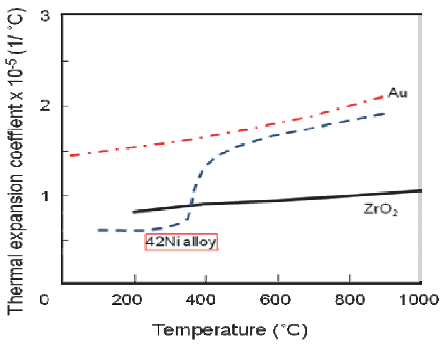


Fig. 3 Thermal expansion coefficient of component materials as a function of temperature.

Figure 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).

Figure 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 (230-288ºC) and high pressure (9MPa) out-pile water loop.

Figure 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 3000 hours were performed for the ECP sensors and integrity was confirmed.

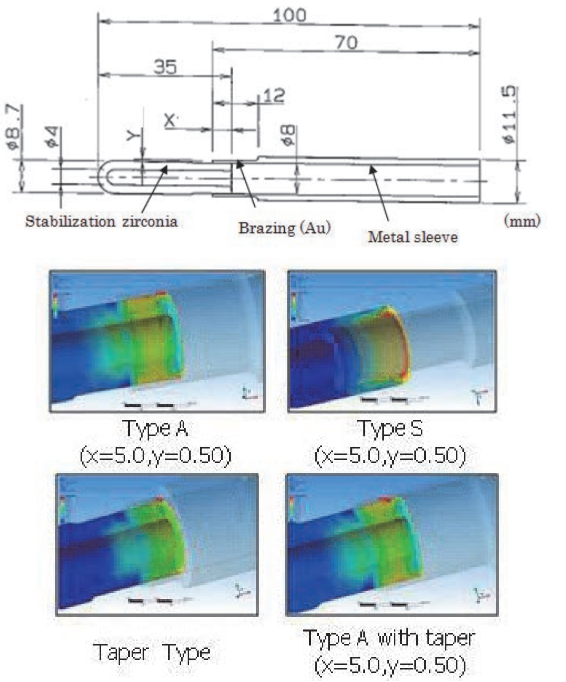


Fig. 4 Stress distribution of brazing juncture neighborhood.

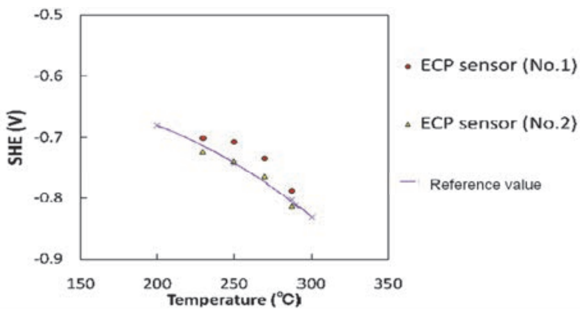


Fig.5 SHE value of ECP sensors as a function of temperature.

3. DEVELOPMENT OF THE WATER LEVEL INDICATOR

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 [4][5] and TRICOTH type [6] 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. Figure 6 shows schematic drawing of new water level sensor.

The demonstration test and characteristic evaluation of the water level indicator were performed and evaluated. Performance test was made with length of 200mm sensor and 100m MI lead cable. Test apparatus is depicted in Fig. 7.

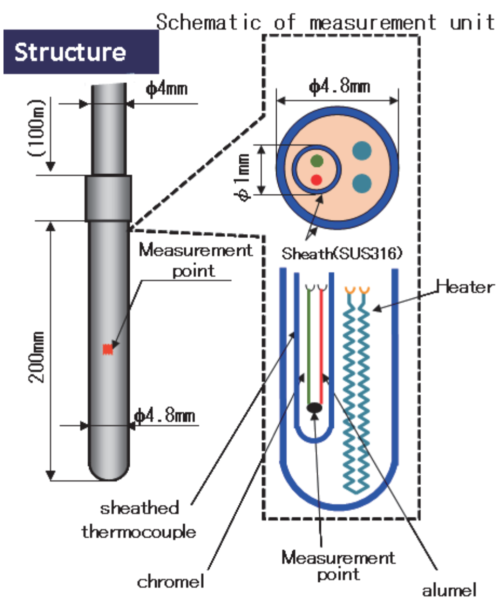


Fig.6 Structure of water level sensor.

The sensor unit is installed into water. Temperature of water was controlled to 27-90ºC to simulate spent fuel stack pool. Some additional thermocouples were attached to the surface of water level gauge to measure the temperature distribution along the heated thermocouples position. At various water levels, the output of the thermocouples was measured.

At the water surface level, temperature difference between in water and in air was observed. Figure 8 shows the results of the experiments. When the measurement point is above 20mm from water level, the temperature difference of 25ºC was observed at 27ºC water and the temperature difference of 50ºC was observed at 90ºC 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 20ºC 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 100 m distance to the measuring point at the spent fuel pool at Fukushima power plant. The manufacturing ability of such a water level sensor with long MI cables was demonstrated.

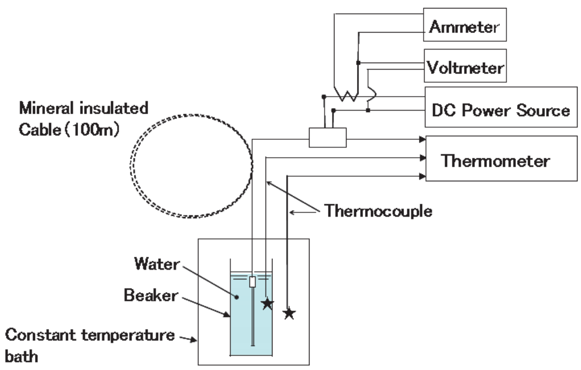


Fig. 7 Test apparatus of performance tests

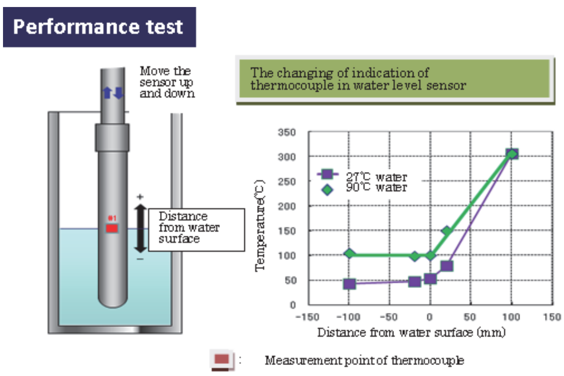


Fig.8 Results of performance tests.

4. DEVELOPMENT OF IN=REACTOR OBSERVATION SYSTEM USING CHERENKOV LIGHT

The Cherenkov light is a faint emission accompanying the passage of charged particles through a transparent medium at speeds faster than the speed of light in that medium [7]. Recently, the observation system of Cherenkov light was developed and preliminary tests were carried out with the halogen light [3]. In this study, Cherenkov light was observed by this system during KUR operation and the effect on transmittance of change of the diaphragm and ND-filters was investigated.

As the first experiment, the illuminance of Cherenkov light was measured by the digital illuminance meter (IM-5, TOPCON) during KUR operation. This meter was fixed by the jig and installed in the pipe for observation hole. The neutral density filters (ND-filters) were changed during the observation of Cherenkov light.

As the second experiment, Cherenkov light was observed by the observation system (Fig.9). The camera of observation system was fixed by the jig and installed in the pipe for core observation hole. The change of the diaphragm and ND-filters were carried out during the observation of Cherenkov light. After the observation, the obtained data were evaluated by the image processing software called “Image J”.

The relationship of transmittance between the catalogue value and measured value of the ND-filters was studied. The illuminance with no ND-filter was 0.230 lx and these values were corrected by the angle of incidence. From the result, the transmittance of Cherenkov light was almost the same tendency as that of the halogen light and the effect on the transmittance of the ND-filters was the same as that of the catalogue value.

The analysis of Cherenkov light was carried out with the fuel element in the “Ho-6”. The relative intensity of light decreased with increasing the diaphragm number. In Fig.11, the change of transmittance was evaluated. The relationship of transmittance between the calculated value and measured value of the diaphragm is shown in Fig.10. Here, the calculated values were determined to raise the rate of the diaphragm numbers to the second power and the measured values were calculated by the analysis results. From the result, the measured values did not agree with the calculated values in the range of high transmittance. On the other hand, the transmittance with ND-0.9 filter is a half of that with ND-0.6 filter. However, the transmittances between ND-0.6 and 0.9 were about 70 and 51% at the dia-phragms of F1.8 and F2.8, respectively. In future, it is necessary to evaluate this phenomenon including the halation and evaluation procedure by “Image J”.

The measurement of Cherenkov light was carried out with the observation system. Correlation between illuminance of Cherenkov light and ND-filter was evaluated and the transmittance of Cherenkov light almost agreed with the catalogue value. On the other hand, the transmittance with ND-filter and diaphragm by the camera was different from the calculated values. The measuring and analysis procedures will be improved in future.

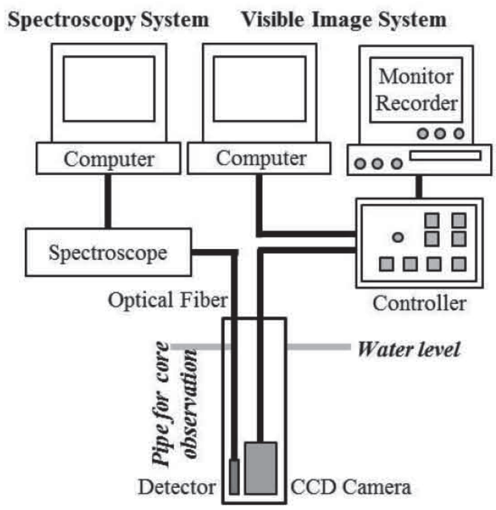


Fig.9 Cherenkov light measurement apparatus.

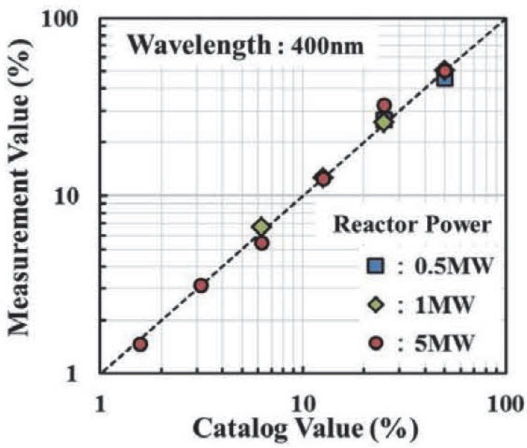


Fig.10 Characteristics of the ratio of transmitted light of the Cherenkov light by dimming filter.

6. SUMMARY

[1

JAEA is developing several in-pile instrumentations to conduct irradiation tests at JMTR. The results of each instrumentation are shown as below:

(1) 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.

(2) A new type water level indicator which composed of thermocouple and heater was developed. This sensor has more simple structure and the accuracy of measurement is about ±20mm. Inorganic materials such as metal and ceramics were chosen for structure material and it has high radiation resistance.

(3) By using simple system, proportional relation between the intensity of Cherenkov light and reactor power was observed. By using filters and aperture, a perspective to obtain in-reactor information by Cherenkov light was obtained.

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4.2 Operation Status and Prospect of Radioisotope Production Facility in HANARO

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At the RIPF at HANARO, Radioisotopes for industrial & medical purpose are produced and research & development for various radioisotopes are carried out. Major products include Ir-192 for NDT, I-131 for treatment and diagnosis of thyroid cancer, Mo-99/Tc-99m Generator for imaging diagnosis of cancer. Production of radioisotope and radiopharmaceutical is being increased every year. Due to world-wide unstableness in the supply of Mo-99, a technology to produce (n,γ)Mo-99 generator at HANARO had been developed as a short term countermeasure. It will be available by the end of 2012. As a long term countermeasure, we are trying to build a new fully dedicated isotope reactor that will produce Fission Mo-99. At present, utilization of RIPF at HANARO is being increased. However when the construction of a new dedicated isotope reactor is completed in 2016, the role of the existing facility and new facility should be established accordingly so that none of the facilities are idling. In the near future, when the prospect of a utilization plan is completed, we expect an opportunity to present the result.

Keywords: radioisotope, radiopharmaceutical, RIPF, Ir-192, I-131, Mo-99/Tc-99m generator, (n,γ)Mo-99 generator, fission Mo-99,

1. BRIEF HISTORY OF RI PRODUCTION IN KAERI

In 1960's, RI manufacturing & production license for Au-198 Colloid & I-131 was acquired when we were operating 100kw TRIGA MARK II research reactor I. Also we constructed 4 lead hot cells for RI production. RI production was launched during this period. In 1970's, production of P-32, Cr-51 & Tc-99m were started and we had constructed 2 concrete and 10 lead hot cells. Also Ir-192 for NDT was produced and supplied. In 1980's, RI sales license was acquired from MOST and the Solvent Extraction Equipment of Tc-99m was supplied to hospital. In this period, supply of Au-198 Colloid, I-131, Tc-99m, Mo-99, Cr-51&P-32, Tc-99m Cold Kits also began. In 1990's, RI production license for I-131 Capsule & Tc-99m with cold kit was acquired. Also RI production Facility was established and the supply of Ho-166 & Ir-192 began. In 2000's, the export of I-131, Ir-192 & Co-60 began and the supply of 99mTc Generator to the domestic market started.

2. RI PRODUCTION FACILITIES

KAERI has 4 hot cell banks. For example, Bank-1 has 4 heavy concrete hot cells to process highly active sources for industrial purpose. In bank 2, 11(eleven) lead hot cells are installed to develop and produce the P-32/33, Tc-99m, Cr-51, Lu-177, Sr-90/Y-90, W-188/Re-188 and Ir-192 for cancer treatment. In bank 3, 6(six) lead hot cells are installed to produce medical radioisotopes such as I-131, Ho-166 etc. In bank 4, 4(four) lead hot cells are installed to produce Mo-99/Tc-99m Generator. We have cold kits and radio-pharmaceutical production facility in 2nd floor of the RIPF building.

3. FACILITY OPERATION & MAINTENANCE

The control panel of RIPF was installed in 1994. It was outdated and most of the parts became no longer available in the market. In 2010, we removed

Fig. 1. Control Panel Replacement in 2010

old control panel and installed new control panel as shown in the Fig 1.

We installed new PLC and computer based HMI so that it became possible for operating staffs to be alarmed by a text message even at night and during holiday.

We are producing the I-131 for diagnosis and treatment of thyroid cancer. It is very important to maintain the I-131 concentration at the stack below the limited value. To enhance the quality of air, we installed new charcoal filter housing because the old filter housing had potential chances of leakage. In every 18 months, we replace the charcoal filters. When we removed the used charcoal filters, the remaining radioactivity is negligible because the filters were decayed for more than 6 months. After the installation of new filters, we perform the In-Place Leakage Test right after the system air flow rate measurement that should be within ±10% of the rated flow rate. The HEPA filters as well as the charcoal filters are tested every 18 months according to ANSI N510. We use the filter testing equipments such as the Halide Generator & Halide Detector for charcoal filter testing, DOP Generator & DOP Detector for HEPA filter testing. When the penetration is not greater than 0.05%, it is assumed that the filters are installed properly.

Fig. 2. RI Production up to the year 2012

In our RI production facility, we have a license to produce 200 kCi of Ir-192, 2k Ci of I-131 & 36.4 kCi of Tc-99m in a year. In 2011, we produced 222 kCi of Ir-192 that covered the 90% of domestic demand. In case of I-131, we produced 1,359 Ci, which covered 70% of domestic demand. Because of the crisis in the world Mo-99 supply market, the market shares of domestic product were drastically increased. In 2011, 5,787 Ci of Tc-99m Generator was supplied and is about 70% of the domestic demand. The amount of the cold kit supplied was 6,600 units.

As open sources, we produce ^{131}I for the medical diagnosis and treatment of thyroid cancer, ^{131}IMIBG and ^{99}Mo/^{99m}Tc for medical diagnosis by solvent extraction method. Also produced are ^{51}Cr and ^{32}P. All the open sources are commercialized.

As sealed sources, we produce ^{192}Ir source for NDT that has distinguished features such as an improved straightness and a mimic to a point source. It has been tested and passed through the 50,000 times loading and unloading cycles. It is also commercialized. For the sealed sources, ^{60}Co for gauge is commercialized and that for good irradiation is under development.

Low energy gamma radiography sources such as ^{169}Yb and ^{75}Se are under development. We are also producing and developing radio-labeling compounds such as ^{166}Ho chitosan complex for liver cancer treatment, ^{166}Ho patch for skin cancer treatment, RI-coated balloon and ^{166}Ho-DTPA filled balloon for the prevention of the restenosis of the coronary artery, ^{166}Ho stent for esophageal cancer treatment etc.

For diagnostic purpose, about 10 kinds of ^{99m}Tc cold kits are developed.

HANARO RI team is developing the therapeutic radioisotopes. The target radionuclides are Carrier Free radioisotopes such as ^{177}Lu and ^{149}Pm (promethium) and generators such as ^{188}W/^{188}Re and ^{90}Sr/^{90}Y. In the field of Sealed Sources, we are developing beta sources for industrial instruments as ^{85}Kr, ^{90}Sr, ^{147}Pm. Miniature ^{192}Ir Sources for NDT, and ^{125}I seed and ^{32}P Ophthalmic Applicator for brachytherapy.

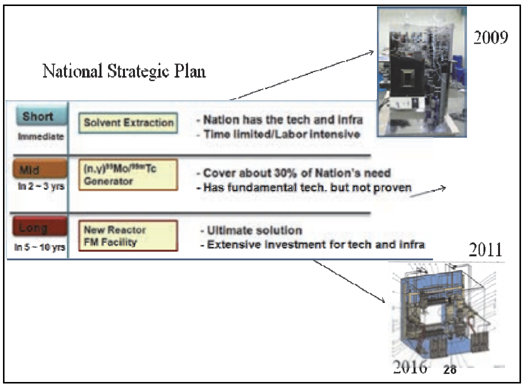

Fig. 3. National Strategic Plan for ^{99m}Tc Supply

In order to secure ^{99m}Tc, we have short, mid-term, long-term strategic plan. As a short term plan, Solvent Extraction unit is used. Our nation has the technology and infrastructure. But it is a time limited and a labor intensive method. As a mid-term plan, $(n,\gamma)^{99}Mo/^{99m}Tc$ generator is the one and it will cover about 30% of nation′s need. We have a fundamental technology but it is not proven yet.

As an ultimate solution, we are trying to build Fission Moly facility in the new reactor. We hope to finish this develop this facility by 2016..

4. FUTURE WORKS

We need to repair the existing RI production facility to enhance the performance. We will replace the inlet duct filter housing at hot cell bank 3 for ^{131}I production. It will be done to supply the air with better quality to hot cell and to prevent a contamination when we have a reverse flow in accident condition. Also we will have to repair RI production building and structure according to the result of the structural safety diagnosis that was done few years ago. In near future, we expect to build a new research reactor and RI production facility. When we build the RI production facility, we need to coordinate with the design team to build a better facility. When it is built, the role of the existing RIPF and new facility has to be established.

4.3 Present Status and Prospect of NTD in Korea

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Two vertical irradiation holes in the heavy water reflector region of HANARO have been utilized for the commercial NTD (Neutron Transmutation Doping) service since 2003. Now the service is concentrated on 5 and 6 inch silicon ingots and increasingly involved in 8 inch also. In spite of fixed NTD facility in HANARO, the production has steadily increased every year due to market demand change towards higher target resistivity and continuous improvements of the management. In 2011 total 23 tons of irradiated ingot was supplied to the NTD market. Because one irradiation hole is dedicated to the irradiation for 8 inch silicon ingot and its current market occupancy rate is less than 20%, the service volume from HANARO is expected to increase up to 40 tons per year in next few years.

In 2012, a national project for the construction of Korean New Research Reactor is starting. The new reactor will be sited in Ki-Jang, the south part of Korea, and plans to complete in 2016. One of the important missions of the new research reactor is to increase NTD capacity for the stable supply of NTD service. With the experiences in HANARO, the full scale study is started in order to achieve advanced neutron irradiation techniques and handling systems for the higher quality. The irradiation of larger diameter silicon more than 8 inch is included in the study for the future market demand.

1. INTRODUCTION

Neutron Transmutation Doping (NTD) is a method for the n-type silicon semiconductor production using a nuclear reaction. The NTD is based on a nuclear transmutation where a Si-30 atom absorbing a neutron is replaced into a phosphorus atom, which rolls as dopants for the n-type semiconductor materials in a silicon crystal. With the NTD technique, it is possible to obtain a much better uniform dopants distribution in a silicon ingot comparing with other conventional chemical doping methods [1, 2, 3]. For a long time, the silicon semiconductors have been used in various applications such as integrated circuits, thyristors, transistors, etc., and especially a better uniformity of the dopant distribution is required for high power operating devices such as IGBT1, IGCT2, GTO3, etc., in order to ensure the devices against a hot spot formations and a possible break down. Recently the demand for such high power devices using NTD-Si is increasing rapidly due to the rapid increasing needs for green energy techniques and the energy saving techniques for various industrial equipments.

Consequently, NTD is becoming a commercially applicable topic in the research reactor utilization fields and many research reactors under construction or being planned are adopting NTD application in their design.

Currently 5 and 6 inches take up the majority of the NTD-Si market and the world demand is estimated to be around 150 ~ 200 tons per year. The silicon wafer companies are forecasting a steady 7 % growth every year of the NTD-Si market for the time being.

2. DESIGN FEATURES OF NDT FACILITIES

2.1 Irradiation Holes for NDT

HANARO (High-flux Advanced Neutron Application Reactor) is an open-tank-in-pool type research reactor with a thermal power of 30 MW. It

[1] IGBT : Insulated Gate Bipolar Transistor
[2] IGCT : Integrated Gate Commutated Thyristor
[3] GTO : Gate Turn Off Thyristor

reached its first criticality in Feb. 1995. Since then, the power was raised to 15 MW in Jan 1996, 22 MW in Dec. 1999 and finally 30 MW, its design power, in Nov. 2004. It usually has 9 cycle operations per year and one cycle is accomplished by 23 days operation and 7 days shut down for refueling and maintenance services. HANARO has been used for the neutron beam utilizations, nuclear fuel and material tests, development and commercial supply of radioisotopes and radiopharmaceuticals, neutron activation analysis, boron neutron capture therapy study as well as NTD-Si production.

For the NTD works, HANARO has two vertical irradiation holes in the heavy water reflector tank surrounding the reactor core as shown in Fig. 1. They are around 67 cm distant from the core center in the opposite direction and the inner diameters are 22 cm (NTD1 hole) and 18 cm (NTD2 hole) respectively. The design maximum thermal neutron flux is around 5x1013/cm2sec in both locations, while the fast neutron flux and gamma level are low enough to prevent defects in a silicon crystal. The effective core length is 70 cm and the height of heavy water reflector tank is 120 cm.

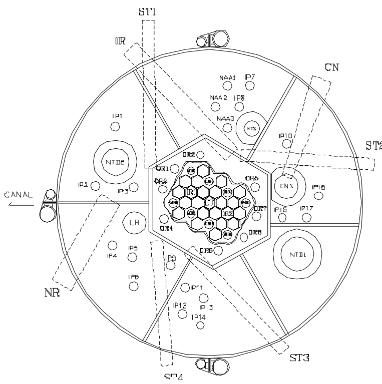

Fig.1 Reactor core of HANARO.

2.2 Irradiation Rigs

In the NTD-Si production, one of the most important factors is to irradiate the ingot very uniformly in both the radial and the axial directions. While a radial uniformity is normally achieved by rotating the ingot during the irradiation, some different methods have been practically used for the axial uniformity. A travelling method as in BR2 [4] is not proper in HANARO because the available space below the irradiation hole is not enough to move the ingot sufficiently downward unless by reducing the ingot length. And the technique of turning the ingot upside down after a half irradiation as in JRR-3M [5] is also not proper because the axial neutron profile in the irradiation hole is similar with cosine shape, and the region providing a linear flux change is not long enough. Besides, a flat neutron profile can be obtained using a neutron screen as in OPAL, of which filtering effect is different in the axial direction. The screen method is the best way practically applicable in HANARO considering higher irradiation capacity.

Usually nickel, titanium or stainless steel is used as the screen materials in many applications but they are disadvantageous to the radioactive wastes management and neutron economy because they are all relatively strong neutron absorbers. These materials were avoided at the first stage of our feasibility study for NTD. Instead, it was confirmed that a neutron screen with a high performance can be achieved using mainly water and aluminum and it will have great advantages of maximizing the neutron flux in the ingot as well as minimizing the long-lived radioactive wastes [6,7,8].

An aluminum irradiation rig (ingot container) was designed to have different inner diameters in the axial direction so that the outside water layer of the ingot is thicker at the middle region than at both ends of the ingot. In order to make a flat neutron flux from a cosine shape distribution, much more reduction of the neutron is needed at the middle region and consequently it makes the neutron flux in the ingot lower too. An effort was made to minimize the loss of neutron by the screen and it can be accomplished by placing cylindrical graphite blocks above and below the ingot as a reflector. This concept was adopted in developing our first 5 inch irradiation rig for NTD2 hole as shown in Fig. 2. The maximum thickness of the aluminum wall is 5 mm at the middle part and minimum thickness is 17.9 mm at the upper and lower part, and then the maximum axial difference of water gap is around 13 mm. The rig provides a flat neutron profile up to 605 mm long with the upper and lower graphite reflectors.

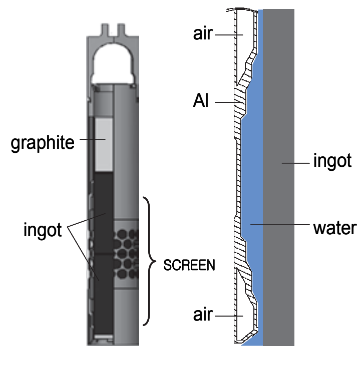

Fig. 2 Irradiation rigs and neutron screen

The next objective was to develop the irradiation rig to irradiate 6 inch ingot at the same hole. As the inner diameter of NTD2 hole is 180 mm, only 15 mm space remains inside the irradiation hole when 6 inch ingot occupies the hole. It is not enough to make a neutron screen with only water and aluminum. A substitute material of higher neutron absorption was needed and stainless steel was selected through the analysis of the several candidate materials. Fig. 3 shows an optimized neutron screen for the 6 inch ingot. Stainless steel is adopted only at the middle section of the rig and upper and lower sections are made by aluminum again. The rig has wall thickness varying from 2.2 to 5.1 mm at the screen section and also provides a flat neutron flux up to 605 mm with the upper and lower graphite reflectors. While an upper graphite block is loaded into the irradiation rig following the silicon ingot, the lower graphite is installed in the irradiation hole.

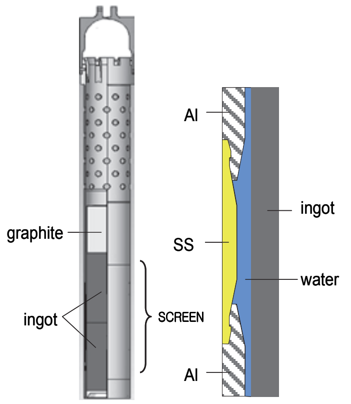

Fig. 3 Irradiation rigs and neutron screen

With the computational analysis for the fresh core, these rigs were expected to give an excellent axial uniformity with ±1.0% deviation of the average as shown in Fig. 4 [9]. The figure is rated with the normalized total reaction rate of Si30(n,γ)Si31. The RRG (radial resistivity gradient) was also expected not to exceed 2.5% for both 5 and 6 inch.

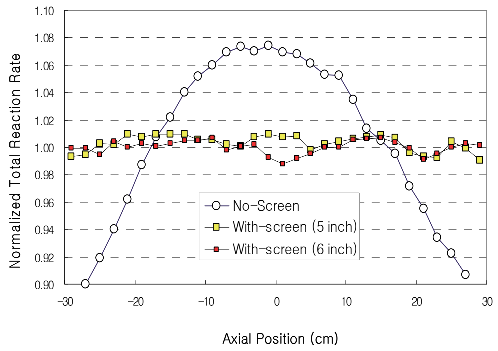

Fig.4 Axial uniformity of rig

2.3 Sleeve and Floater

An aluminum sleeve was installed in the irradiation hole for the guidance of insertion and withdrawal of the irradiation rig and for the mounting of the self-powered neutron detectors (SPND). The SPND monitors the neutron behavior in a real time and has a dimension of 5 mm in maximum diameter and 80 mm in effective length respectively.

The sleeve also guides a bottom graphite reflector and a void aluminum can. Insertion of the rig in the hole will increase the neutron flux due to the material change of the hole from water into the silicon ingots and withdrawal of the rig will decrease the neutron flux on the contrary. It can have a significant effect on the nearby experimental devices and neutron detectors for a reactor operation. The floating system was introduced to minimize the flux change by insertion or withdrawal of the ingots. When the irradiation hole is empty, the graphite is floated by buoyancy of the void aluminium can to the middle part of the hole as shown in Fig. 5 (a). When a rig is inserted into the hole, the rig pushes down on a graphite reflector and a void aluminium can as shown in Fig. 5 (b).

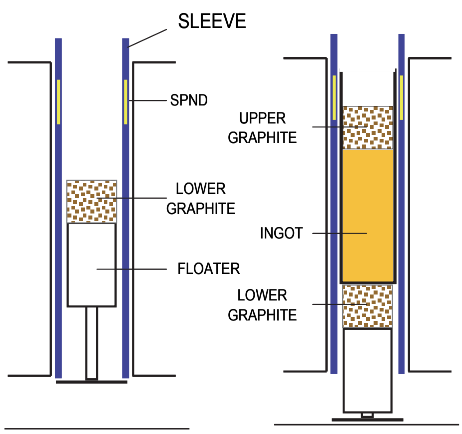

Fig. 5 Irradiation hole before and after loading a rig

3. SERVICE RESULTS AND MARKET PROSPECT

3.1 Annual Irradation Result

HANARO has been experiencing long-term shutdowns every year from 2005 to 2009 due to the new installations of a high-pressure high-temperature fuel test loop (FTL) and a cold neutron research facility (CNRF). As a result, HANARO was operated at a rate of 50 ~ 60 % of its normal operation capacity until 2009. Fig. 6 shows an annual record of NTD-Si production. In spite of a shortage of operation time of HANARO, the total volume was increased every year because 6 and 8 inches ingot were added and the target resistivity is becoming higher. As shown in Fig. 7. 6 inch demand shares almost 60% in 2011 and 8 inch shares only 10% but a rapid growth is expected from 2012.

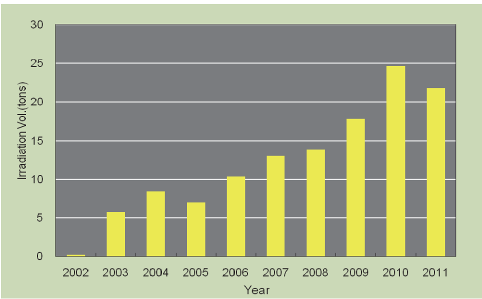

Fig. 6. Annual Production

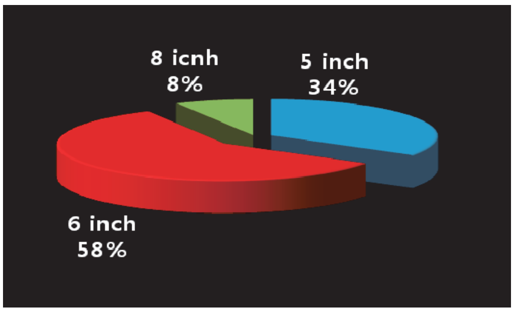

Fig.7. Production by Silicon Diameters (2011)

3.2 Market Prospects

The market of NTD-Si grew rapidly during the 1980s but stagnated from the early 1990s until mid-2000s due to an improved gas doping technique so called PFZ (Premium Floating Zone). The PFZ can offer a 15% deviation of doping uniformity and it was an outstanding result in comparison with the formal conventional chemical doping technique.

To prevent the global warming and save energy, the power semiconductor devices are widely using and their demands is also rapidly increasing. According to a market research report by Yano Research Institute in 2010, the world power device market size is around 145 hundred million US$ in 2010 but expected to be 230 hundred million US$ in 2015. The PFZ is most widely used for power device but they have technical limits for higher power applications comparing NTD-Si. And epitaxial wafer or compound semiconductors such and SiC or GaN have outstanding advantages comparing to the silicon based semiconductor but the cost is too high and mass production is not easy. Therefore NTD-Si is the only reasonable solution for now and the future.

The current world market need of NTD-Si is around 200 tons per year. It is very closely related with supplying capacity from the research reactors.

4. CONCLUSION

KAERI has been conducting commercial irradiation services for the NTD-Si production since 2003 using HANARO. The irradiation facility in HANARO has a very unique feature that a neutron screen for the axial flatness of the neutron flux is incorporated in the irradiation rig. This rig-screen assembly, which is fabricated with aluminum only and sometimes incorporated with stainless steel, has a great advantage for a high accuracy and a high productivity of NTD-Si. It can give a quite flat neutron flux with less than ± 1.5% axial deviation over 60 cm long and a very high neutron flux of more than 3.5x1013 n/cm2 sec for all sizes of silicon ingots. In 2011 total 23 tons of NTD-Si was produced using HANARO. Excepting 8 inch silicon, 5 and 6 inches more than 20 tons were produced using only one irradiation hole. If consider the increase of 8 inch demand, the maximum capacity of HANARO is more than 40 tons per year.

In 2012, a national project for the construction of Korean New Research Reactor is starting. The new reactor will be sited in Ki-Jang, the south part of Korea peninsular, and plans to complete in 2016. One of the important missions of the new reactor is to increase NTD capacity for the stable supply to the market. With the experiences in HANARO, the full scale study is started in order to achieve advanced neutron irradiation techniques and handling systems for the higher quality. The irradiation of larger diameter silicon more than 8 inch is included in the study for the future demand.

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4.4 International Standardization of instruments for Neutron Irradiation Tests

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The JMTR in JAEA and HANARO in KAERI are the foremost testing/research reactors in the world and these are expected to contribute to many nuclear fields. As a part of instrument development in irradiation field, information exchange of instruments started from 2010 under the cooperation agreements between KAERI and JAEA. The instruments developed in JMTR and HANARO are introduced and cooperation experiments as future plan are discussed for international standardization.

Keywords: JMTR, HANARO, Measurement instruments, Thermocouple, LVDT, SPND

1. INTRODUCTION

The JMTR in JAEA and HANARO in KAERI are the foremost testing/research reactors in the world and these are expected to contribute to many nuclear fields. As a part of instrument development in irradiation field, information exchange of instruments started from 2010 under the cooperation agreements between KAERI and JAEA. To get measurement data with high accuracy for fuel and material behavior studies in irradiation tests, many types of measuring instruments have been developed; these are the K-type thermocouple and multi-paired thermocouples, the Linear Voltage Differential Transformer (LVDT) type gas pressure gauge and Self-Powered Neutron Detector (SPND). In future, the irradiation tests of these developed instruments will be carried out to establish the international standard of worldwide Material Testing Reactors.

In this paper, the instruments developed in JMTR and HANARO are introduced and cooperation experiments as future plan are discussed for international standardization of the instruments in the Material Testing Reactors.

2. TERMOCOUPLE

2.1 JMTR

It is necessary to measure temperatures during neutron irradiation tests of fuels and materials. Thermocouples are used for temperature measurement in the neutron irradiation tests. However, the number of thermocouples installed in an irradiation capsule is limited because the diameter of an inner tube in the capsule is from 20 to 30 mm. Therefore, a type-K multi-paired thermocouple was developed to evaluate the axial temperature distribution in irradiation samples. The main specifications and structure of the type-K multi-paired thermocouple are shown in Fig. 1. The axial temperature distribution in the irradiation samples has been successfully measured by the type-K multi-paired thermocouple. It was accurately measured for 17,000h at 400˚C or higher (maximum 700˚C) in the irradiation test of tritium breeder pebbles bed for a fusion blanket [1].

The higher temperature measurement at above 1,000˚C is needed in irradiation tests with LWR fuels. However, the type-K thermocouple has a short lifetime at above 600˚C. Therefore, the development of a new type multi-paired thermocouple has been carried out. First, a type-N (Nicrosil-Nisil) thermocouple which is usable at 1,000˚C or higher under neutron irradiation has been developed. The trial fabrication test was carried out in the same specifications of the type-K multi-paired thermocouple [2] and the accuracy of hot junction axial positions in the type-N multi-paired thermocouple was ±1 mm in the trial fabrication. The temperature characteristics of the type-N multi-paired thermocouple were also evaluated through demonstration tests under un-irradiation conditions [3]. The demonstration test system of the type-N multi-paired thermocouples is shown in Fig.2. Reference thermocouples of type-R were set at positions of the hot junctions of the type-N multi-paired thermocouple, and temperatures were measured from room temperature up to 1100°C. The comparison of the temperatures measured by the type-N multi-paired thermocouple and the reference thermocouple is shown in Fig.3. It is clear that the accuracy of the measured temperatures is ±1 % up to 1100˚C based on the comparison with the reference temperatures.

Then, endurance tests of type-N multi-paired and type-N thermocouple were performed with the infrared furnace. The type-N multi-paired thermocouples were heated up to 1000˚C for 3000h. The type-N thermocouple was heated up to 1200˚C for 1000h. Figure 4 shows the results of endurance tests in the infrared furnace. Type-N multi-paired thermocouples have excellent durability which is able to stand at 1000˚C with 3000h. On the other hand, as for the Japanese type-N thermocouple, the decrease of electric potential with the increase of duration in the measurement temperature was not seen at 1200˚C compared with US type-N thermocouple. Japanese type-N thermocouple is obtained a good characteristic.

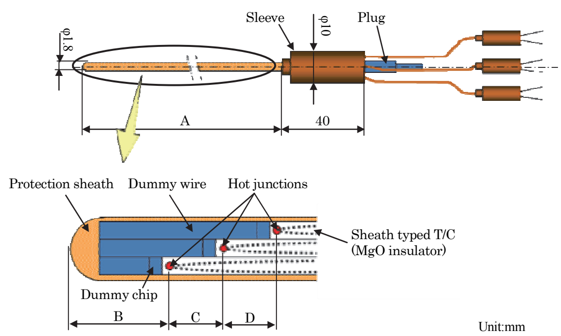


Fig.1 Structure of multi-paired thermocouple.

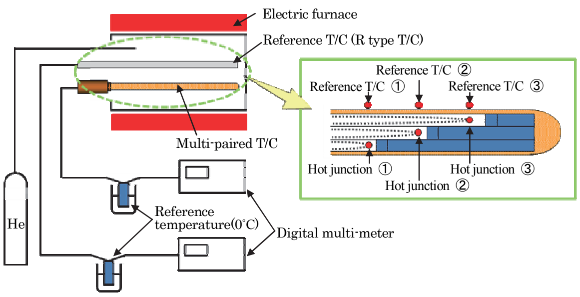


Fig.2 Demonstration test of type-N multi-paired T/C.

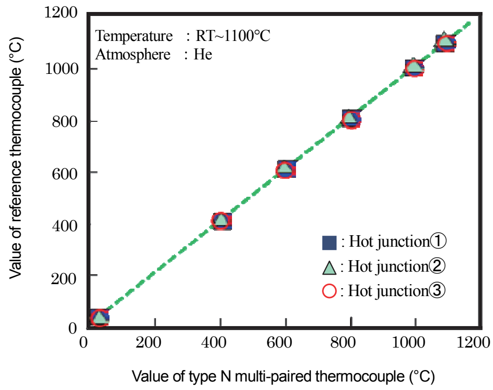


Fig.3 Result of Demonstration test with type-N multi-paired thermocouple [3].

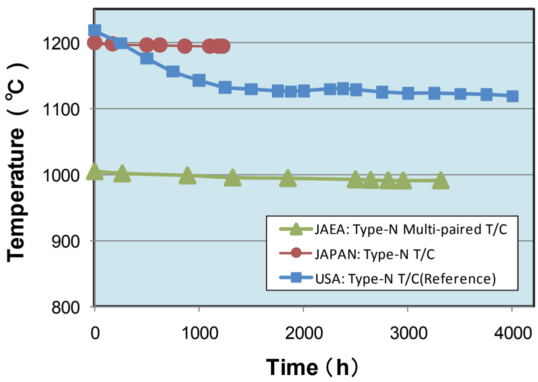


Fig.4 Result of the type-N multi-paired and type-N thermocouple in endurance tests.

2.2 HANARO

In KAERI, K-type thermocouples made by Thermocoax Co. of France have been used in the capsule of HANARO until now. The type-K thermocouple can be used in temperatures up to 1150˚C normally and up to 1250˚C for short periods depending on sheath material and diameter. The acceptable deviations of the Thermocoax thermocouple are 2.5˚C up to 333˚C and ±0.75% above 333˚C in accordance with I.E.C. standards. Recently, a thermocouple made by a Korean company was used in an in-reactor test for the future use in HANARO. Fig.5 shows result of the compared with between Thermocoax and Sentech thermocouple under neutron irradiation. Its sheath material is STS 304L and the outer diameter is 1mm. It was installed with the Thermocoax thermocouple in a capsule, and irradiated at 700˚C for 2 months. It was operated within ±3˚C deviations when compared with the Thermocoax thermocouple. The accuracy of the thermocouple was

±2˚C up to 1000˚C in the fabrication shop test, which is in the range required in KS standards (0.4% up to 1000˚C).

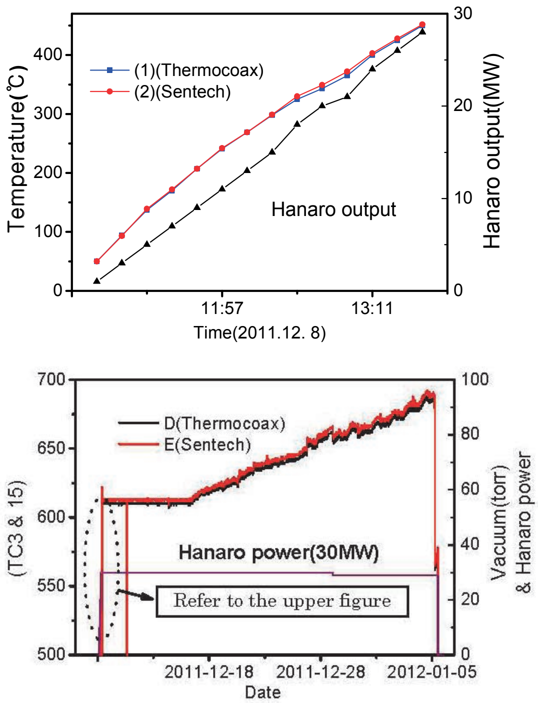


Fig.5 Result of the Thermocoax and Sentech thermocouple in HANARO test.

3. LINEAR VOLTAGE DIFFERENTIAL TRANSFORMER

3.1 JMTR

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 new JMTR has a plan of the power ramp tests of the fuels. 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 [4].

The rod inner pressure gauges for 5 and 10 MPa have been developed to meet 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, wire material for 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.

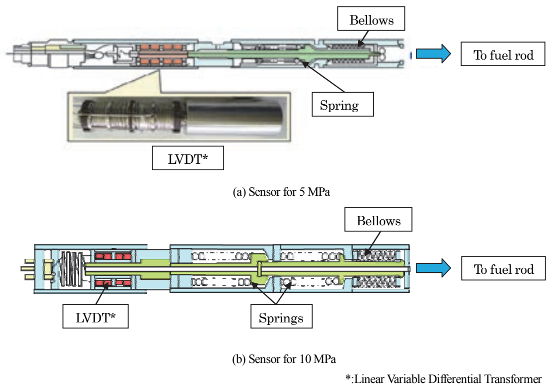


Fig.6 Structure of rod inner pressure gauges.

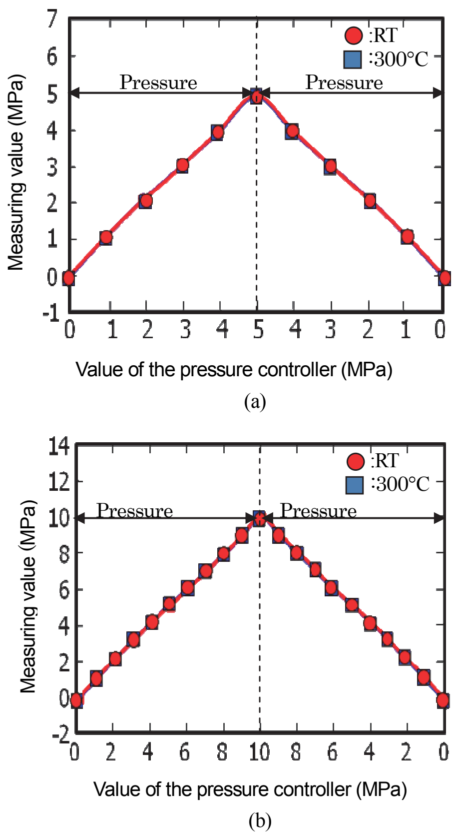


Fig.7 Results of performance test with rod inner pressure gauge:(a) 5 MPa, (b) 10 MPa.

Performance tests of the rod inner pressure gauges were carried out [3]. 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. 7. The relationship between the pressure and the axial displacement of bellows was linear and there were 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 at 200 through 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.

3.2 HANARO

In KAERI, an elongation LVDT for HANARO irradiation test was newly designed and fabricated (see Fig.8). The designed elongation LVDT uses STS-316L material and one MI cable. The LVDT itself and its core assembly were welded by a fiber laser welding system. Pressure LVDT was also designed for measuring a fission gas pressure during irradiation tests in HANARO. The pressure LVDT has the same design like an elongation LVDT but the lower part has a metal bellows. The fabricated LVDT showed good linearity and sensitivity. The elongation LVDT will be irradiated in HANARO in May, 2012 for performance test.

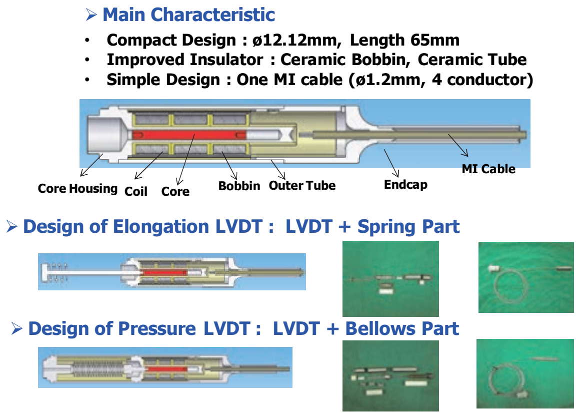


Fig.8 Development of KAERI's LVDT.

4. SELF-POWERED NEUTRON DETECTOR

4.1 JMTR

The SPNDs were applied for measurement of thermal neutron flux change in the reactor. In the SPNDs, Co, Rh or Pt are generally used as the emitter material. As for the Co or Pt type SPNDs, sensitivity is low, but the response speed is fast. On the other hand, Rh type SPNDs is high sensitivity but the response speed is slow. In order to correct the defect, Hybrid SPNDs with high sensitivity and fast response have been developed. Two types of Hybrid SPNDs were developed, Co-Rh type and Pt-Rh type. The response characteristics of Co-Rh type were higher than that of Pt-Rh type. However, the manufacture of Co-Rh alloy is difficult, and cladding structure, in which Rh emitter is surrounded by Co emitters, is difficult to miniaturize to use in capsule in JMTR. Therefore, Pt-Rh type SPNDs, which is easy to miniaturize is developed. Figure 9 shows new Hybrid SPNDs transient response when the thermal neutron flux was changed.

The Pt-Rh type SPNDs were irradiated in JMTR at 30 cycles (from 136cy to 165cy). As a result, the durability of the Hybrid SPNDs was temperature 400°C to 700°C under 17,000 h that was examined.

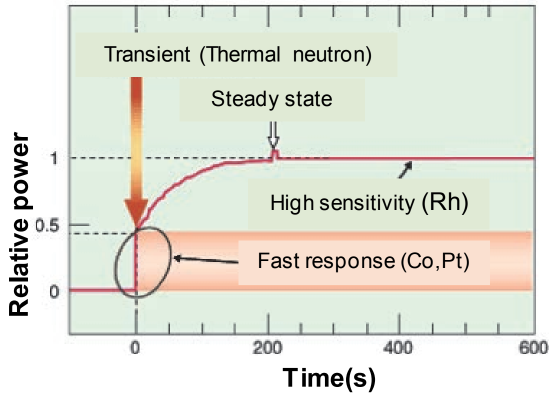


Fig.9 Hybrid SPNDs transient response when the thermal neutron flux was changed.

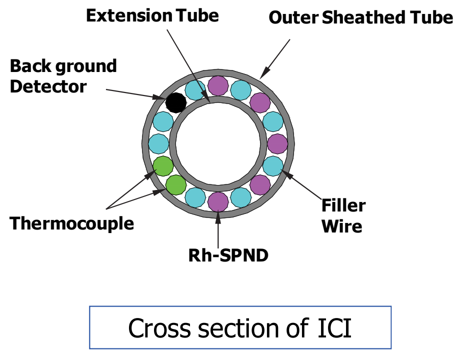
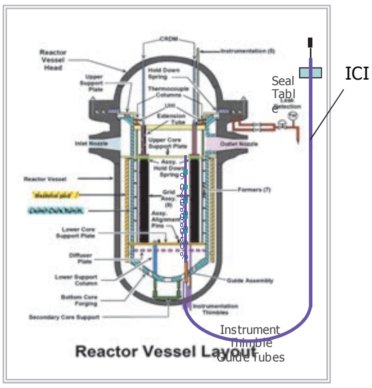


Fig.10 In-Core-Instrument containing several localized SPNDs.

4.2 HANARO

Korean Standard Nuclear Power Plants are using Rh-SPNDs as in-core neutron detectors. A private Korean company, Woo-Jin INC succeeded in localization of Rh-SPND production in 2008. In this development, the external neutron facility (ENF) of HANARO was utilized for the performance test of the products and their absolute sensitivity evaluation. Figure 10 shows a few ICIs (In-Core-Instrument containing several localized SPNDs) were successfully examined in the different power reactors during last two years. In 2012, Woo-Jin INC will make the first delivery of their Rh-SPNDs to some power reactors in Korea, and HANARO will provide testing and sensitivity measurement of some samples.



45 ICI assemblies per plant for 10 power plants

5. CONCLUSIONS

(1)Thermocouples
- -The N-type multi-paired thermocouples were developed, and stable temperature measurements were confirmed at 1000˚C with about 3500h in the out-pile performance tests.
- -KAERI compared the in-reactor performances of the K-type thermocouples made by Thermocoax and a Korean company.

(2) LVDT
- -The rod inner pressure gauges were developed with the irradiation capsule of the LVDT. Two kinds of LVDTs are irradiated for evaluation of irradiation effects in the WWR-K reactor at KZ.
- -KAERI developed LVDTs for the measurement of elongation and pressure of a fuel rod and will irradiate them in reactor sooner or later.

(3) SPND
- -The Hybrid-type SPND remained intact during irradiation test in JMTR (136CY-165CY).
- -Rh-SPND for the use of power rectors was localized by a Korean company in cooperation with HANARO.

In future, the irradiation tests of these developed instruments will be carried out for the international standard of worldwide Material Testing Reactors.

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4.5 Design and Fabrication of LVDT for Irradiation Test of Nuclear Fuel and Material

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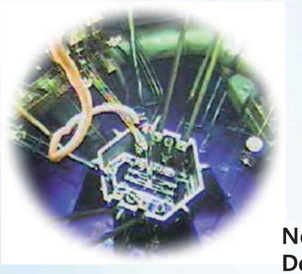
ABSTRACT

The LVDT (Linear Variable Differential Transformer) is used to measure the elongation and pressure of a nuclear fuel rod, or the creep and fatigue of the material during an irradiation test. This device must be a radiation-resistant LVDT for use in a research reactor. The LVDTs for an irradiation test have been used Halden's LVDTs that is developed in Norway at KAERI; however KAERI has been developing a new LVDT since 2007 because of high costs. The elongation LVDT for HANARO irradiation testing were newly designed and fabricated. The designed elongation LVDT uses STS-316L material and one MI cable. Also, the LVDT itself and its core assembly are welded by a laser. The pressure LVDT is designed for measuring the fission gas pressure during irradiation tests. The pressure LVDT has the same design like an elongation LVDT but the lower part has a metal bellows. The material of the pressure transducer and the LVDT itself is the same STS-316L. Welding is a very important factor for the fabrication of an LVDT. We are using a 150W fiber laser welding system that consists of a welding head, monitoring vision system and rotary index. The fabricated LVDT shows good linearity compared to the Halden LVDT. The developed LVDT for the irradiation test is utilized for the elongation measurement of a pellet and pressure measurement of a fuel rod. Until now, the use has been limited because of high price, but developed LVDT can be utilized for fuel and material testing and in a high temperature environment.

K-J Joint Seminar

The Design and Fabrication of LVDT for the Irradiation Test of Nuclear Fuel and Material



2012. 03. 29

Neutron Utilization Technology Division
Development of Irradiation Technology

1

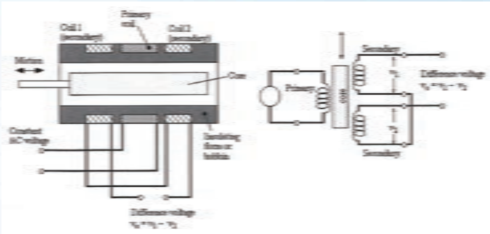
Contents

2

1. LVDT

➢ Introduction of LVDT (Linear Variable Displacement Transducer)

- **LVDT is senor for the measurement of displacement.**
- **An LVDT consists of one primary coil and two secondary coils.**
- **When the core moves , Output voltage is developed in secondary coil.**



➢ Purpose of LVDT use

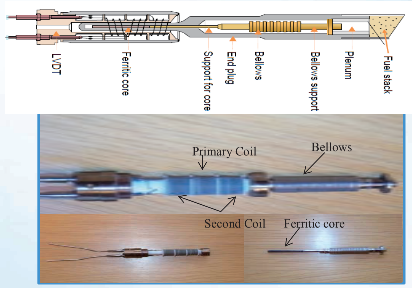
- **Fission gas pressure, Fuel temperature measurement, Fuel/Cladding Elongation, Creep/Fatigue of Material**

3

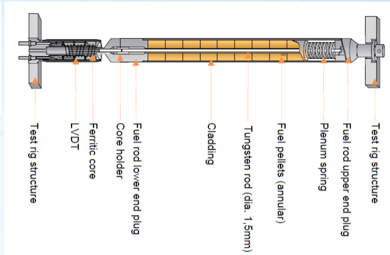
➢ Halden LVDT for Fuel Irradiation Test

– Measurement Applications : Fuel rod pressure measurement, Fuel stack elongation, Cladding elongation, Pellet average temperature measurement, Cladding diameter for oxidation layer

– LVDT & LVDT Signal Conditioner are high price.



Halden Pressure LVDT



Halden LVDT for Average Temp

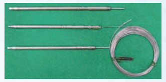
4

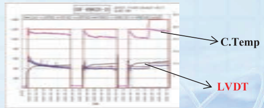
2. KAERI Experiences about LVDT

➤ Halden LVDTs were used two times in KAERI Capsule irradiation test .

Name	Instrumenta -tions	Fuel	Pellet	Conditions	Period	LHR	Burnup	LVDT
03F-05K	Temp, SPND, LVDT	STS316L Fuel Rod 3	UO_2 2.42w/o	OR5, 24-30M	60FPD	Max 38kW/m Avr 28kW/m	3,461 MWd/t	Pressure, Elongation LVDT
05F-01K	Temp, SPND, LVDT	STS316L Fuel Rod 3	UO_2 2.42w/o	OR5, 30M	45FPD	Max 38kW/m Avr 28kW/m	4,069 MWd/t	Pressure LVDT

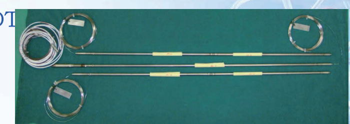

Instrumented Capsule


LVDT Instrumented Fuel Rod



➤ KAERI FTL used a Halden LVDT

– Pressure LVDT

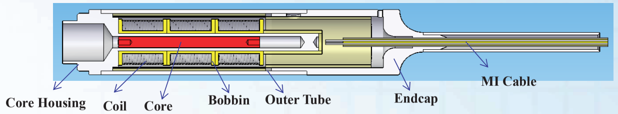

FTL Instrumented Fuel Rod

5

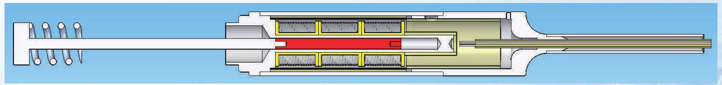
3. LVDT Design

➤ Main Characteristic

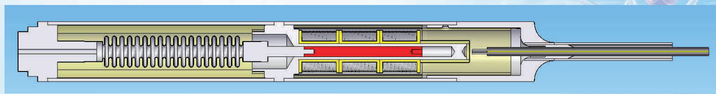
- Compact Design : ø12.12mm, Length 65mm
- Improved Insulator : Ceramic Bobbin, Ceramic Tube
- Simple Design : One MI cable (ø1.6mm, 4 conductor)



➤ Design of Elongation LVDT : LVDT + Spring Part



➤ Design of Pressure LVDT : LVDT + Bellows Part



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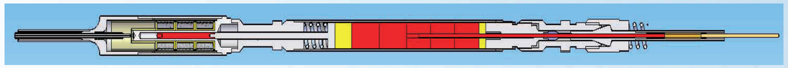
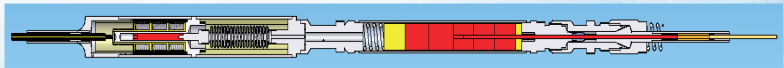
3. Application of LVDT

➤ Application of Elongation LVDT : Elongation LVDT + Irradiation Fuel Rod



➤ Application of Pressure LVDT : Pressure LVDT + Irradiation Fuel Rod



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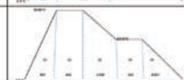
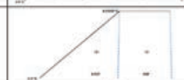
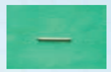
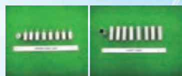
4. LVDT Fabrication

➤ LVDT Parts Heat Treatment

– Objectives : Stress relief, Recrystallization

Part	Program number	program	Reference
Core STS410	1		750℃, 1h
Bellows STS300	2		350℃, 3h
Magnetic shield Ni-Fe-Coil	3		800℃, 1h → 350℃, 1h
Spring Inconelx-718	4		1000℃, 1h
Endcap/LVDT Tube STS316L	5		1050℃, 1h





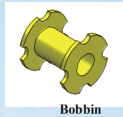

Furnace (Max. 1700℃)

8

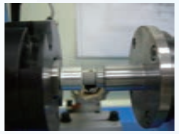
4. LVDT Fabrication

➢ Coil Winding

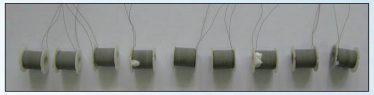
– Coil : Ceramic coated wire (Copper clad with Nikel, ø0.113mm)
– Ceramic Bobbin
– Semi Auto Winding Machine



Bobbin



Coil Winding



Bobbin with coil

Winding Conditions : Turns 620, Pitch 0.128, Speed 800rpm

Coil Winder

➢ MI cable splicing & coil connection

– MI cable : Manual splicing
– Laser welding

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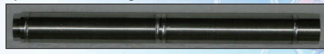
➢ LVDT Welding

1) The welding is very important to perform various instrumentations for the irradiation test.
2) A fiber laser welding system with a 150W and 50µm optical fiber transmission can be weld from endcap welding to thermocouple sheath welding.
3) Also, other dissimilar materials welding is available(for examples ; Inconel to Zir-4, SUS to Inconel).

- ❑ Fiber laser source
  - ➢ 150W average power, pulsed type
  - ➢ Peak power 1.5kW with air cooled
  - ➢ Feeding fiber couplers: QBH connectors (fiber core dia.=50um)
- ❑ Addition device
  - ➢ Welding head + Spot size adjusting unit
  - ➢ Vision system (He-Ne laser)
  - ➢ Rotary index + Work table with X-drives

Thermocouple sheath welding

Bellows welding



LVDT Out tube welding

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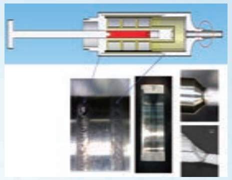
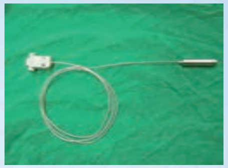
➢ LVDT Welding

– Elongation LVDT

Before assembly

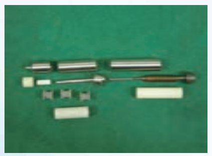


Laser Welding

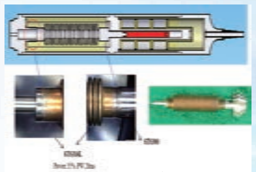


After Welding

– Pressure LVDT



Before assembly



Laser Welding

After Welding

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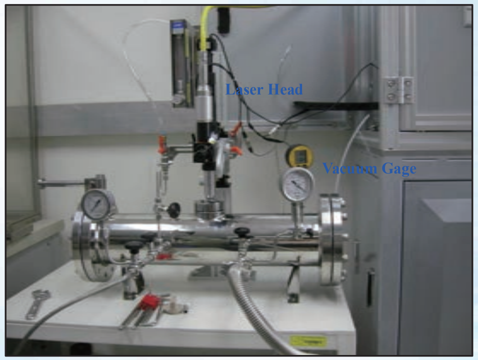
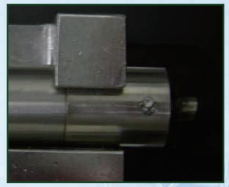
➢ Chamber for the Pin-Hole Laser Welding

- We designed a pin hole welding chamber using the welding head of a fiber laser. The design value of the chamber is pressure of maximum 150 bar and a vacuum of maximum 10^{-3} torr.



10^{-3} Torr, 150bar Chamber



Pin Hole Welding

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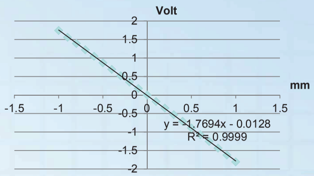
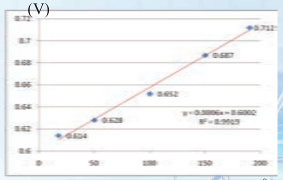
➢ Test for LVDT Performance

✓ **Out-of-Pile Test**

- LVDT Signal Conditioner : Macro Sensors
  3kHz source power
- Tool : Micrometer Header
- Measure Length : ±2mm
- Sensitivity=1,800mV/mm



LVDT Test at RT

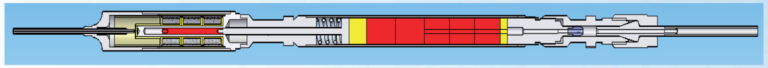


LVDT Test at High Temp.

13

➢ In-Pile Test for LVDT Performance

- **Purpose : LVDT In-Pile Test in HANARO IP5 Irradiation Hole**
- **Simulated Fuel Rod with Elongation LVDT**



Part	LVDT	Cable	Cladding	Pellet
Material	STS316L	STS316L	STS316L	Aluminum


Before Assembly


After Assembly


Final Assembly

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➢ LVDT Performance Test

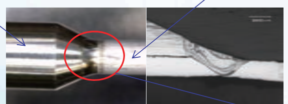
✓ **Result of In-Pile Test**

- **After the LVDT was installed Irradiation Hole, the LVDT output showed abnormal value.**
- **We decided to stop irradiation test.**
- **There was leak in the LVDT & MI cable welding part.**

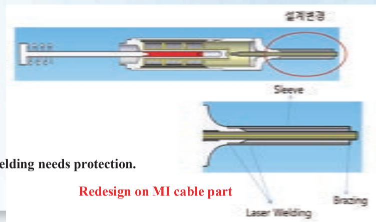
At early, R decreased and V increased rapidly.

Endcap

MI cable


STS316L/STS316L Welding

This welding needs protection.



Redesign on MI cable part

15

5. Conclusions

- ✓ KAERI has been developing a new LVDT since 2007 and we are in development completion step almost.
- ✓ The Elongation and Pressure LVDT for HANARO irradiation test were newly designed and fabricated.
- ✓ We are using a 150W fiber laser welding system, Coil Winder and Vacuum Chamber.
- ✓ The fabricated LVDT shows good linearity in the out-of-pile test.
- ✓ Although the first LVDT was stopped, the redesigned LVDT will be fabricated and irradiated sooner or later.

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4.6 ^{99}Mo-^{99m}Tc Production Development by (n, γ) Reaction

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The renewed JMTR will be started from the later half of JFY2012, and it is expected to various fields. Supply of ^{99}Mo in Japan depends only on imports from foreign countries. JAEA has a plan to produce ^{99}Mo, a parent nuclide of ^{99m}Tc. JMTR will contribute to produce ^{99}Mo by (n, r) method as one of effective uses of the JMTR. In this paper, outline of the technical study items for production method of ^{99}Mo-^{99m}Tc in JMTR will be described.

Keywords: ^{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 using JMTR about 20% (about 37 TBq/week (1,000Ci/week)) of the amount of ^{99}Mo imported from overseas [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.

In this paper, the status of R&D for the production of ^{99}Mo-^{99m}Tc is introduced.

2. R&D ITEMS FOR ESTABLISHMENT OF Ite ^{99}Mo PRODUCTION BY (n, γ) METHOD

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. R&D 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. 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 target quantity about 1000Ci/week of (n, γ)^{99}Mo production. The preliminary fabrication tests of MoO_3 pellets were carried out in JAEA [7]. Table 2 shows the result of fabrication summary of high-density MoO_3 pellets. MoO_3 pellets were fabricated by the cold pressing and sintering, the Spark Plasma Sintering (SPS) and Plasma Activated Sintering (Ed-Pas) methods. As a result, high density MoO_3 pellets were fabricated by the SPS and Ed-Pas methods and it was promising to fabricate the high density MoO_3 pellets by the plasma sintering methods such as SPS and Ed-Pas. In future, the characteristics such as chemical, thermal and dissolving properties will be evaluated.

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)

Table 2 Result of fabrication summary of high-density MoO_3 pellets

Items \ Method	CS	Plasma Sintering	
		SPS	Ed-Pas
Mass Fabrication	F	P	P
Fabrication Time	F	F	P
Fabrication Cost	E	P	P
Density Control	P	E	E
Product Impurity	F	E	E

E : Excellent, P : Possible, F : Failure

CS : Cold press & Sintering
SPS : Spark Plasma Sintering
Ed-Pas : Plasma Activated Sintering

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 by Mo-Tc extraction devices using methyl ethyl ketone (MEK) or through Mo-Tc generators such as Mo absorbent.

Table 3 Comparison with properties on ^{99}Mo separation from Mo solution, extraction of ^{99m}Tc and concentration of ^{99m}Tc

Items \ Method	Solvent Extraction	Column Method	
		PZC	Al_2O_3
Separation of Mo	P	E	P
Extraction of ^{99m}Tc	E	E	P
Concentration of ^{99m}Tc	E	F	F
Purity of Solution	P	P	E
Operation Time	E	P	P
Operation Control	E	P	P

E : Excellent, P : Possible, F : Failure

MEK : Methyl Ethyl Ketone, PZC : Poly Zirconium Compound

Table 3 shows comparison with properties on ^{99}Mo separation from Mo solution, extraction of ^{99m}Tc and concentration of ^{99m}Tc. Methyl Ethyl Ketone (MEK) is used in the solvent extraction method. Mo-Tc generators such as Poly Zirconium Compound (PZC) and Al_2O_3 are used in the column method. From the results, it is promising to develop the solvent extraction method as Master-Milker and Colum method of PZC as Mo-Tc generator.

The 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 [8-10]. 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 cold tests are carried out with Re elements [11]. Form the results of the cold test, the concentrated specification activity of ^{99m}Tc is about 10-50Ci/ml.

The 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 [12]. ^{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.

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 4 shows pharmacopoeia in USA.

Table 4 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%

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%. [13].

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.

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. As other technical study item, 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. Figure 2 shows concept of Mo recycling methods with PZC. There are three subjects for Mo recycling.

(a) Recovery of Mo resource
 Mo recovery from the Mo solution and used ^{99}Mo-^{99m}Tc generators (PZC, etc.) are developed and the recovered Mo will be re-used.

(b) Reduction of Wastes
 Treatment and of waste reduction of used ^{99}Mo-^{99m}Tc generators (PZC, etc.) and Mo solution

(c) Recovery of PZC
 The reusable ^{99}Mo-^{99m}Tc generators are developed.

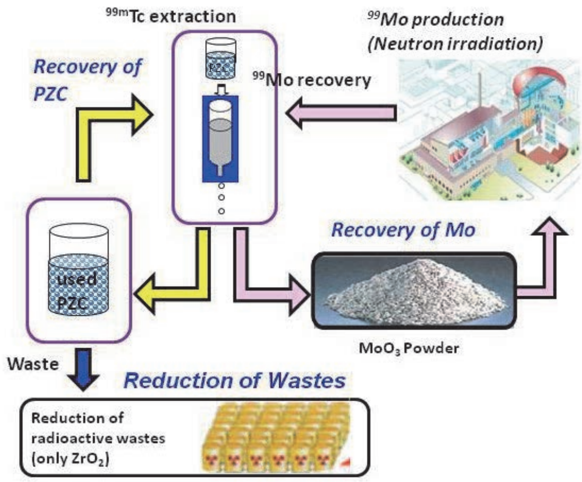

Fig.2 Concept of Mo recycling (in case of PZC).

In JAEA, the preliminary tests with PZC absorbed un-irradiated Mo were carried out [14]. 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.

4. STATUS OF CONSTRUCTION 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.

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 2012. 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

4. CONCLUSIONS

^{99}Mo production by (n, γ) method has been determined from viewpoints of nuclear nonproliferation, reduction of radioactive wastes and utilization of Mo resource in JMTR.

At present, New hydraulic rabbit irradiation facility is constructed as irradiation facility of ^{99}Mo production. JMTR will be able to provide about 20% (about 37 TBq/week (1,000Ci/week)) of the amount of ^{99}Mo imported from overseas.

Moreover, the sintered MoO_3 pellet with high density will be used as irradiated target material for ^{99}Mo production. And fabrication development of high density MoO_3 pellet is carried out, the high density MoO_3 pellets can be produced by SPS method and density will be reached over 90%T.D.

Development of ^{99}Mo-^{99m}Tc extraction and concentration methods by MEK is carried out and useful data are accumulated. The prospect for the concentration method of ^{99m}Tc solution (concentration over 1Ci/ml) is bright from results of preliminary cold tests with Re elements.

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5. Irradiation Technology (2)

5.1 Instrumentation of Sensors on DCF Test Rig Fabrication

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ABSTRACT

DCF (Dual Cooled Fuel) was proposed for use in gas-cooled reactors during the 1960s, and was considered for PWRs (pressurized water reactors) at the joint research of KAERI (Korea Atomic Energy Research Institute) and IFE (Institute For Energiteknikk). DCF has an annular type geometry, which allows coolant to pass through the inner and outer surfaces of the fuel rod at the same time. During to its increased heat transfer surface compared to standard fuel, the maximum fuel temperature during irradiation test can be decreased. Therefore, it can decrease the generation of fission gas and increase the margin against a fuel melt down or DNB (Departure from Nucleate Boiling). To measure the irradiation characteristics of DCF, 16 K-type thermocouples, 1 LVDT (Linear Variable Differential Transformer), and 4 SPNDs (Self-Powered Neutron Detector) were installed in a rig. Measuring the temperature of the center of the fuel and that of the coolant are important factors for the evaluation of irradiation behavior. The flow rate is calculated based on a noise analysis of the data, which is obtained from the inlet and outlet of the coolant through thermocouples. LVDT is used to accurately measure the internal pressure of the fuel rod during the irradiation test. Its Type 5 product fabricated by IFE was mounted on top of a DCF rod end plug. The thermal neutron flux is measured during an irradiation test through the SPND instrumented in the test rig. The SPND is a widely used device in the core of a nuclear reactor to measure a high neutron flux. SPNDs are installed outside the insulated flow channel of the DCF test rig. Three of them are a vanadium type and the other is a cobalt type. The final check of the test rig was achieved by testing in the ATL. The ATL is used for testing a test rig with the same pressure and temperature as the in-core operating conditions.

Keywords: Dual Cooled Fuel (DCF), Sensor instrumentation, Thermocouple (TC), LVDT, SPND, Assembly Test Loop (ATL), Irradiation test rig.

1. INTRODUCTION

Recently, a DCF test rig was designed and fabricated through the joint research of KAERI and IFE to evaluate the irradiation characteristics and validity of the annular type fuel. The DCF has a geometry allowing water to be cooled on the inner and outer surfaces of an annular fuel rod. In addition, the main merits of a DCF are an increased heat transfer surface area and a reduced fuel temperature, which result in less fission gas release and increased margins against fuel melting and DNB compared to a standard fuel [1]. Another advantage is lower stored energy and lower peak cladding temperatures during a LOCA (Loss of Coolant Accident). In this work, the DCF rod consists of coaxial inner and outer cladding tubes in which donut-shaped pellets are stacked, and thus it has internal and external coolant passages. The intrinsic purpose of the DCF is to increase the heat transfer area, which results in a fuel temperature decrease so that a power up-rate can be achieved together with a safety and effective enhancement [2]. The main objective of this test is to analyze the validity of the concept regarding the operation and the heat transfer split between the inner and outer surface of the DCF rod under PWR conditions. The DCF test rig is fabricated according to the design drawings and technical specifications to perform a noise analysis during the ATL-test. The final check-out of the test rig is performed by testing in the ATL. The ATL is a test loop used for testing the test rig at the same pressure and temperature as the in-core operating conditions (305, 155bar). A noise analysis has been used to provide information on the flow rate for the coolant, and fuel fill gas effects, fission gas release, PCMI (pellet cladding mechanical accident), pellet cracking and relocation, and fuel failure for the fuel rod. Under steady power conditions, the reactor coolant temperature and power level show small fluctuations lasting one second and about one minute, respectively. These short-term fluctuations are what are referred to when using the term noise [3].

In this paper, the instrumentation process and test results of a DCF test rig designed to assess the heat transfer characteristics and reliability of the fuel rod are introduced. In addition, the test results performed during the ATL-test are summarized.

2. DESIGN AND FABRICATION

2.1. Test Rig

The DCF irradiation test rig designed to assess the heat transfer characteristics and reliability of DCF is shown in Fig. 1. It consists mainly of five parts: an insulated channel part, DCF rod part, rod support part, elongation and top seal part, and an outer wire protection part. The insulated channel part includes a DCF test rod installed with in-pile instruments, and separates the inlet and outlet coolant. SPNDs are instrumented on its outside surface to detect a neutron flux. The DCF rod part includes one fuel rod which is an axial cluster and is installed with thermocouples and a LVDT. The rod support part includes the support part of a LVDT and a fuel centerline thermocouple, and is covered with a removal type flow separator. Also, the elongation and top seal part includes the sealing signal cables which pass through an elongation tube and top seal feedthrough part which forms a pressure boundary. The outer wire protection part is composed of flexible protection tube and hose to protect the signal cables coming from the seniors instrumented the DCF rod. The flexible protection tube and hose cover and protect the signal cables over the distance from the top seal assembly to the permanent signal cable junction cabinet in the ATL test hall or reactor hall. They were made of a doubly braided stainless steel mesh and fixed to the sleeves by silver soldering.

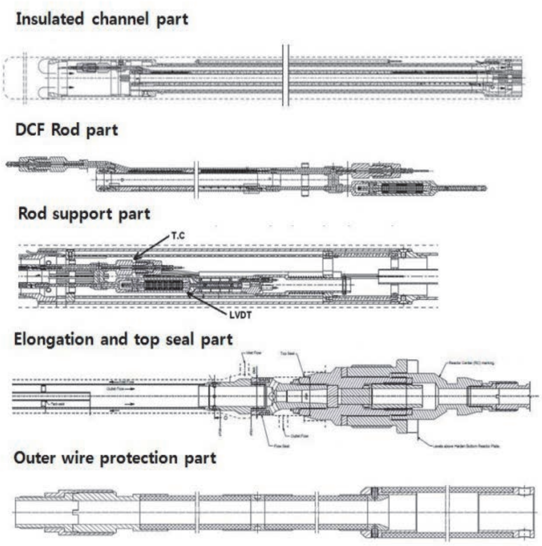

Fig 1. Configuration of the DCF test rig.

2.1. Test Rod

The cladding material used for the DCF rod and end plug is Zircaloy-4. The Zircaloy-4 tubes have been verified as having an OD/ID of 17.88 /14.22 mm. This tube type is machined from 17.88 to 15.9 mm on the OD to fit the DCF outer tube dimension. The pellet fitted with this outer/inner tube size is an OD/ID of 14.1 /9.6 mm. Some composition, dimensional, and mechanical characteristics are fitted for the purpose of the proposed tubes. For the inner tube, the use of a standard Zircaloy-4 17x17 PWR cladding with a nominal OD/ID of 9.5 /

8.36 mm was proposed. The active length of the fuel rod will be 70 mm. The rod was pressurized with helium gas. The filling pressure was 50bar. The fuel rod was instrumented with a K-type thermocouple to measure the temperatures of the center of the fuel and the coolant. The LVDT is instrumented to measure fission gas release and pressure variation in the inside of the test rod during the irradiation test. A DCF test rod consists of the inner and outer cladding tubes, top and bottom end caps, a plenum spring, spacer disks, and UO_2 pellets. In a conventional fuel rod, the spacer is usually made of alumina (Al_2O_3) and is located between the plenum spring and pellets as well as the bottom end cap and pellets. Annular dummy pellets of the test rod were made of ASNI 316 L and aluminum, while those of the cladding and end plug were made of Zr-4.

2.3. Assembly and Inspections

The welding methods used in this fabrication were electron beam welds (EB), LASER welds, and TIG welds. All critical welds are specified with the depth and given distance from a fixed point. A WPQ (Weld Procedure Qualification) was issued for all critical welds and qualified with weld tests. The welding of the test pieces is needed for the approval of the WPS (Weld Procedure Specification) and has to be carried out on the basis of a preliminary WPS [4]. If standard requirement is fulfilled, the WPS becomes a document providing in detail the required variables for a specific application to assure repeatability. The critical weld points of an insulated test channel to weld real products in a test rig are shown in Fig. 2.

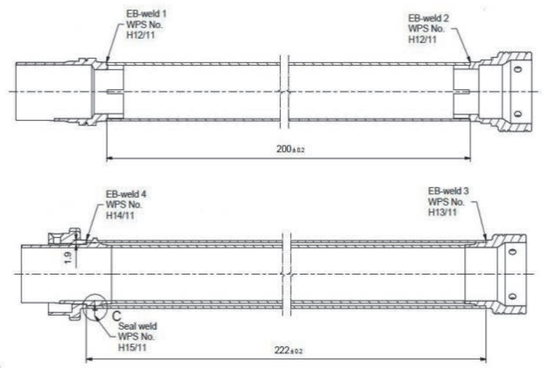

Fig 2. Critical weld points of an insulated test channel welded for WPS documents.

The procedure qualification of round weld for the insulated channel and fuel rod was carried out using a Hamilton Standard model EB welder. The qualification of the spot seal weld for the insulated channel chamber was also carried out using a Millar Syncowave 300P TIG welder. The microstructures of the fuel rod for the EB round weld and the insulated channel for the TIG spot seal weld are shown in Fig 3. The other normal weld parts were welded using a LASER and TIG welder without WPS issues.

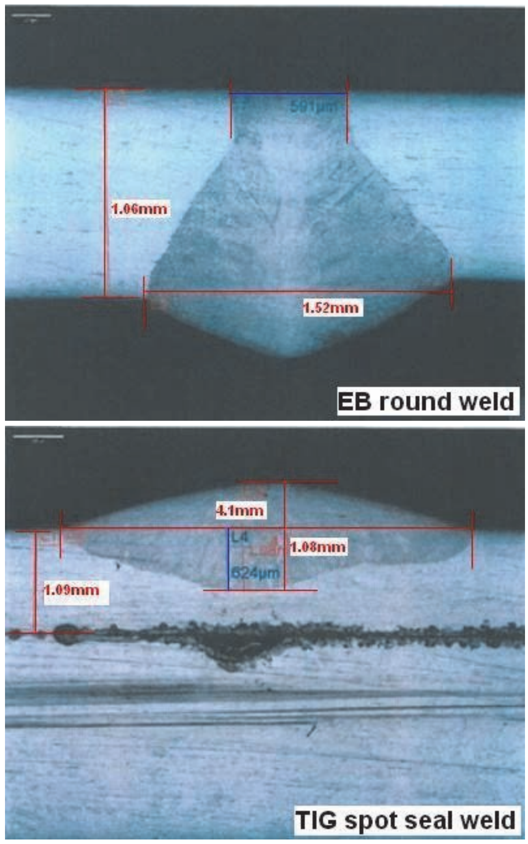

Fig 3. The microstructures of samples welded in the qualification test.

3. INSTRUMENTATION

The position of the sensors instrumented in the test rig is shown in Fig 4. To measure the irradiation characteristics of the DCF, 16 K-type thermocouples (4 sets at the inlet coolant part, 4 sets at the outlet coolant part, four at the mixed outlet coolant part, two at the downcommer coolant part and 2 sets at top and bottom parts of the fuel rod), 1 LVDT, and 4 SPNDs are installed in the test rig. Measuring the temperature of the center of the fuel and coolant is important for the evaluation of irradiation behavior.

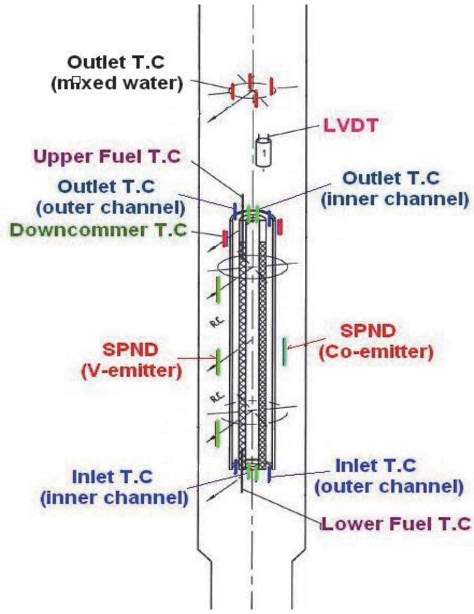

Fig 4. Position of sensors instrumented in the test rig.

3.1. Thermocouple

The temperature of the DCF pellets during the tests was acquainted from thermocouples instrumented at the top and bottom of the DCF rod. The TC instrumented on the top of a fuel rod is shown in Fig 5. To measure the fuel centerline temperature of the DCF rod, K-type TCs are installed at the top and bottom of the end plugs.

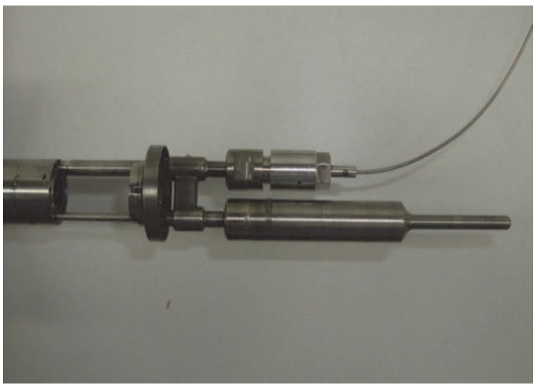

Fig 5. TC mounted on the top of DCF rod.

The fuel centerline TC is sealed with a Swagelok because the MI cable and fuel plug cannot be welded with hetero-metal.

The fuel centerline T.C is sealed with a Swagelok because the MI cable and fuel plug cannot be welded with hetero-metal. The structure of the Swagelok part consists of a seal tube, a nut for the Swagelok, a ring for the Swagelok, and a guide sleeve. The DCF rod is composed of an inner-cladding, an outer-cladding and annular dummy pellets. Fig 6 shows the coolant TC installed in the test rig. Upper right picture is downcommer TCs used to measure the outside coolant temperature of the test rig, and they were mounted at the top of an insulated channel. Lower right picture is an inlet inner channel thermocouple and these are mixed outlet TCs. The sensors used to monitor and acquire the temperature were well satisfied using Type K (Chromel/Alumel) thermocouples, with alumina (Al_2O_3) insulation and a 0.5 mm outer diameter sheath made of INCONEL alloy 600. The thermocouples instrumented in the test rig demonstrated excellent reliability and signal stability under irradiation, even for a very high thermal neutron.

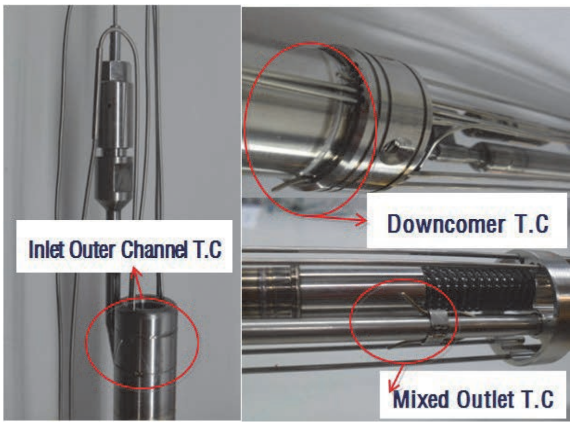

Fig 6. Coolant TCs installed in the test rig.

3.2. LVDT

The LVDT is used to accurately measure the internal pressure of the fuel rod during the irradiation test. Fig 7 shows a LVDT mounted next to a fuel centerline thermocouple on the top of end plug in a DCF rod. This LVDT is a Type 5 model fabricated by IFE and consists of a pressure sensor part to measure the gas pressure, a transformer with first and second coils, and a single channel signal unit able to read data values from a transformer.

Fig 7. LVDT mounted between DCF rod and cable tube

3.3. SPND

The thermal neutron flux is measured during an irradiation test through the SPND instrumented in the test rig. The SPND is widely used in the core of a nuclear reactor to measure the high neutron flux. Fig. 8 shows the insulated channel equipped with SPNDs. The SPNDs were installed outside the insulated flow channel in this test rig. Three of them were vanadium types and the other was a cobalt type.

Fig 8. SPND mounted on the insulated test channel assembly

3.4. Principal Processes during Instrumentation

The MI (Mineral Insulated) cable is helpful for the instrumentation of nuclear fuel and material irradiation because of their high electrical insulation, heat resistance, and mechanical strength. The sealing process of the end tip for the MI cable is as follows: Above all, the sheath of the MI cable is set up 5-8 mm from the end. The strip part of the sheath is squeezed gently at 0^o and 90^o several times and is removed by twisting it gently from side to side until it breaks using pliers with flat edges. At this time, the end of the MI cable must be confirmed to have no contact between the wire conductors and sheath. Also, the wire conductors should be clean. A striped MI cable is then closed using plastic polymer powder at the ends to seal the insulator inside the MI cable tube from moisture. For sensor instrumentations, the sealing of the end tip of the MI cable is very important and is performed by a striping process and sealing process to seal the insulator inside the MI cable tube from moisture. MI cables are used to connect the measuring device and the sensors such as a thermocouple, LVDT and SPND, which are used to measure various irradiation characteristics of nuclear fuels and materials. A top seal assembly drawing is shown in Fig 9. This top seal assembly is at the upper part of the test rig and is the main pressure boundary between the environment inside the pressure flask and the environment outside the pressure vessel. The seal between the top seal assembly and high pressure flask was made up of a cylindered sealing surface and its material is flake graphite powder. The top seal process is performed on the top seal assembly forced against the cylindered seat which is filled with graphite powder in the high pressure flask by means of a threaded nut.

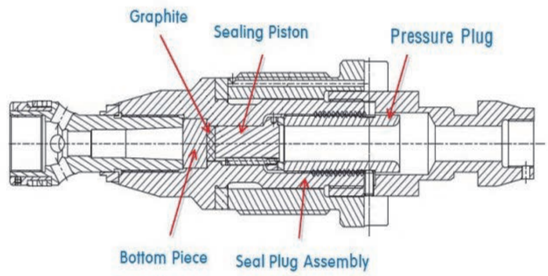

Fig 9. Top seal assembly (Graphite Sealing)

3.5. Soundness confirmation of the instrumented sensors

A final instrument position, polarity, labeling, and function check was performed before ATL testing. All checks were carried out for each individual instrument. Thermocouples installed in the test rig were confirmed through a functional test carried out on all sensors accessible by heating. LVDT was carried out a functional test by inserting its core and SPNDs were identified for instruments and labels by heating. And all the instrumented sensors were checked against drawing and data cable document. In these checks and functional tests, all sensors were contented for the requirement of the test.

4. ATL TEST

The ATL test results for the test rig are shown in Fig 10. The out-pile performance test of the test rig was performed for about 2 days in the ATL. Every temperature value of the TCs installed in the test rig is similar with temperature values of the ATL facility. The ATL is used for testing the test rig with the same pressure and temperature as the in-core operating conditions, 305 degrees centigrade and 155 bars.

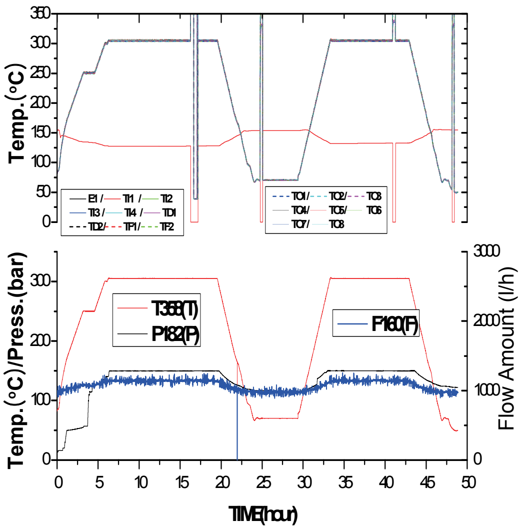

Fig 10. ATL test results for a test rig

The electrical characteristic for signal cables is shown in Fig 11. During the ATL test of the rig, the insulation resistance and loop resistance values were measured at each fixed temperature region. The insulation resistance values in this graph are almost over 10G ohm, but reduced at over 300 degrees centigrade. The insulation resistance values of every SPND were over 10G ohm, but that of the upper fuel TC was poor. The loop resistance values in this graph are different owing to the position of the TC. A noise analysis performed during the ATL test is shown in Fig 12. To fulfill the noise analysis during the ATL test, thermocouples were instrumented on the top and bottom of the test fuel. The noise analysis was performed through processes, such as signal conditioning, storage, cross correlation, time difference, and flow rate calculation using the signal data of these thermocouples. The flow rate and flow amount for the coolant are dependent on the mechanical characteristics of the dual cooled fuel.

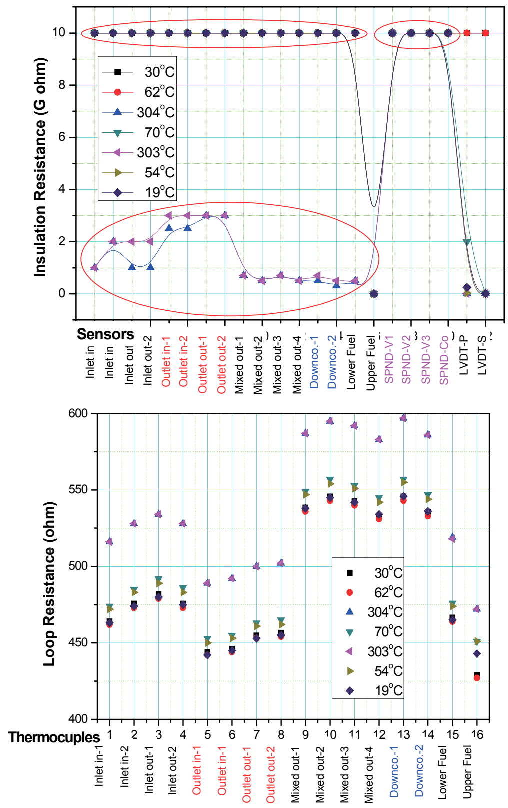

Fig 11. Insulation resistance and loop resistance values measured for signal cables

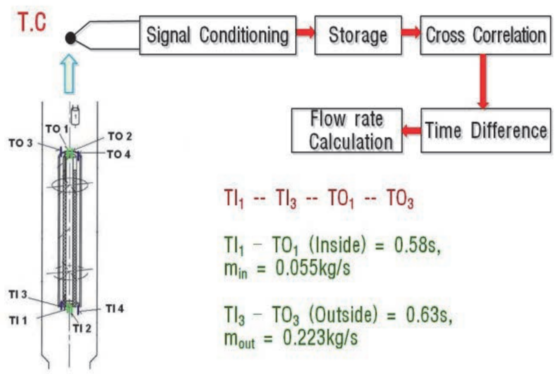

Fig 12. The noise analysis process performed during the ATL test

5. CONCLUSIONS

In this study, the DCF test rig was designed and fabricated through the joint research of KAERI and IFE to evaluate the irradiation characteristics of the annular type fuel. A final instrument position and function check was performed before the ATL testing. The final check-out of the test rig at the same pressure and temperature was implemented as the in-core operation conditions (305, 155bar). A position check was carried out for each individual instrument. The instrument positions were checked against the drawing and data cable document. In this study, the experimental results will be applied to the development of the irradiation test rig in HANARO.

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5.2 Thermal-Hydraulic Tests with Out-of-Pile Test Facility for BOCA Development

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The fuel transient test facility was prepared for power ramping tests of light-water-reactor (LWR) fuels in the Japan Materials Testing Reactor (JMTR) under a contract project with the Nuclear Industrial Safety Agent (NISA) of the Ministry of Economy, Trade and Industry (METI).

It is necessary to develop high accuracy analysis procedure for power ramping tests after restart of the JMTR. The out-of-pile test facility to simulate thermal-hydraulic conditions of the fuel transient test facility was therefore developed. Applicability of the analysis code ACE-3D was examined for thermal-hydraulic analysis of power ramping tests for 10×10 BWR fuels by the fuel transient test facility.

As the results, the calculated temperature was 304°C in comparison with measured value of 304.9~317.4°C in the condition of 600 W/cm. There is a bright prospect of high accuracy power ramping tests by the fuel transient test facility in JMTR.

Keywords: High burn-up LWR fuel, Fuel integrity, JMTR, Power ramping test, Boiling water capsule, DNB, Out-of-pile test, ACE-3D

1. INTRODUCTION

The fuel transient test facility was prepared for power ramping tests of LWR fuels in the JMTR under a contract project with the NISA of the METI [1-3]. The facility is utilized for power ramping tests of LWR fuels after restart of JMTR.

It is necessity to predict or evaluate irradiation conditions accurately such as the neutron flux, the temperature and the liner power of the fuels.

The out-of-pile test facility was therefore designed and manufactured for simulation of thermal-hydraulic conditions. And applicability of the thermal-hydraulic analysis code ACE-3D was examined for power ramping tests of 10×10 BWR fuels in JMTR.

2. FUEL TRANSIENT TEST FACILITY.

Fuel transient test facility has been developed to evaluate the integrity for the high burn-up LWR fuels [1, 3].

The facility consists of a Boling Water Capsule (BOCA), a BOCA pressure control unit, a He-3 gas pressure control unit and a BOCA cooling unit (OSF-1). Figure 1 shows the schematic configuration of the facility. Figure 2 shows the schematic configuration of the core part of BOCA/OSF-1. Table 1 shows the major specifications of the facility. The BOCA is used to irradiate a fuel sample under the condition simulating the BWR and PWR coolant. The pressurized water adjusted by the BOCA pressure control unit is injected in the BOCA through a connection box. The pressurized water flows around the fuel sample, and is boiled by heat generation of the fuel sample under irradiation. The BOCA inserted into the in-pile tube of the OSF-1 is capable to be exchanged in the reactor operation using the capsule handling device. The in-pile tube of the OSF-1 is penetrating into the reactor core through the lid of the pressure vessel and forms independent pressure boundary. OSF-1 coolant is injected from the OSF-1 to remove the generated heat of the BOCA. In power ramping tests, the linear power of the fuel is controlled by changing He-3 gas pressure in the He-3 gas screen installed in the in-pile tube of the OSF-1[4]. Figure 3 shows the typical transients of the He-3 gas pressure, the linear power of the fuel and the surface temperature of the fuel during a power ramping test.

Table 1 Major specifications of fuel transient test facility.

BOCA pressure control unit	
	Coolant : light water
	Pressure :Max. 17MPa
	Temperature : Max. 375°C
	Flow rate : 1~10cm^3/s
OSF-1	
	Coolant : light water
	Pressure : 0.4MPa
	Temperature : Max. 90°C
	Flow rate : 1.9m^3/h

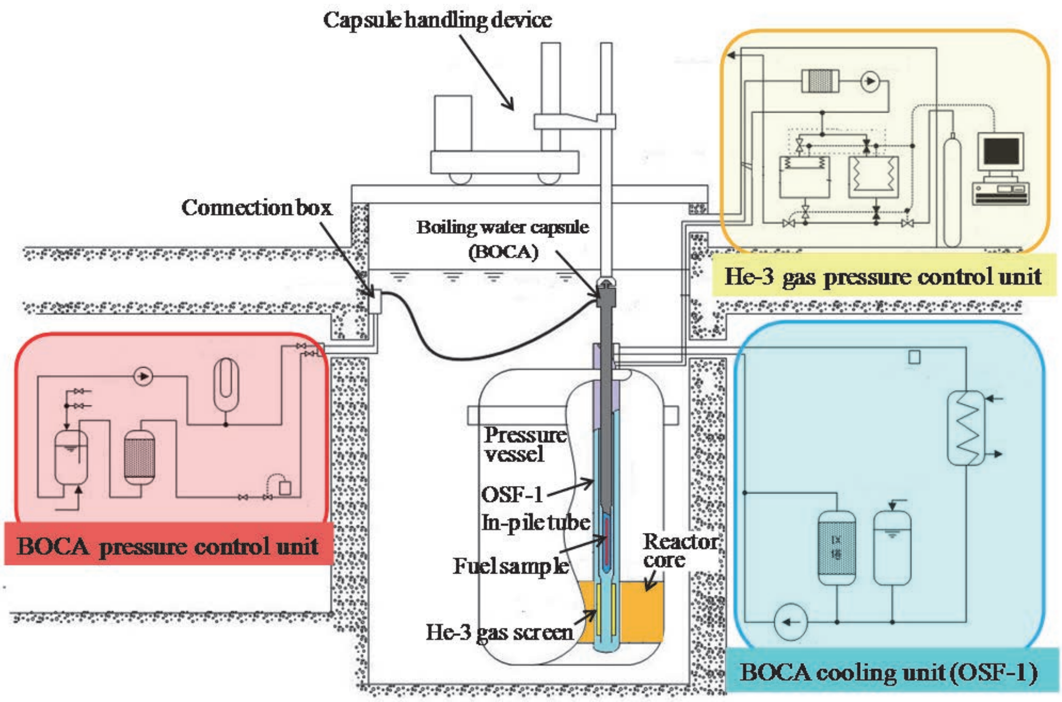

Fig.1Schematic configuration of fuel transient test facility

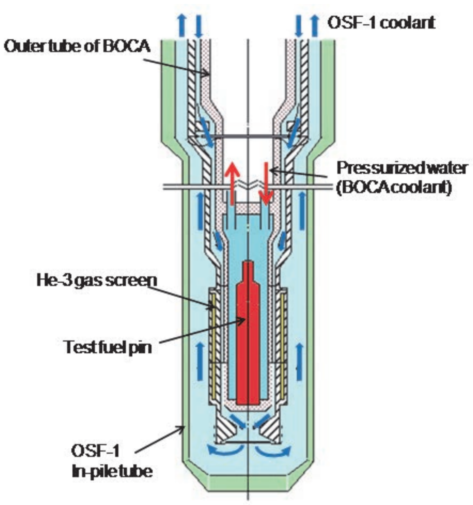

Fig.2 Schematic of the core part of BOCA/OSF-1

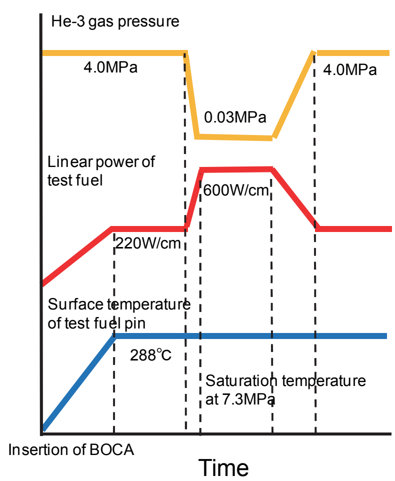

Fig.3 Example of power ramping test of JMTR

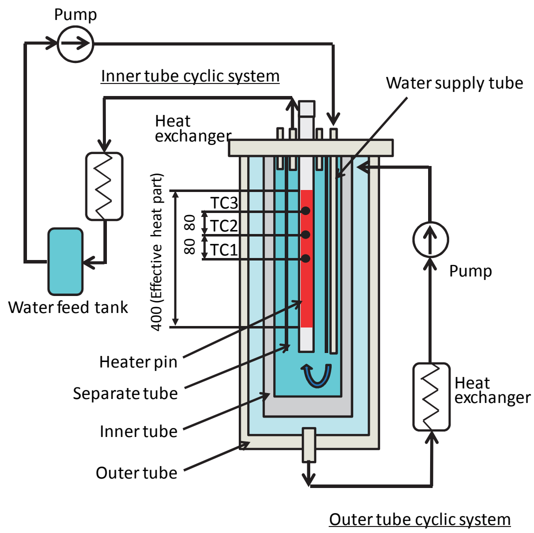

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Fig. 4 Schematic configuration of the out-of-pile test facility.

3. DEVELOPMENT OF OUT-OF-PILE TEST FACILITY

Figure 4 shows the schematic configuration of the out-of-pile test facility. Table 2 shows design conditions of the out-of-pile test facility.

The out-of-pile test facility consisted of a heater pin instead of the fuel, an inner tube simulated the outer tube of the BOCA, an outer tube simulated the in-pile tube of the OSF-1 and cooling systems for these tubes.

The design of the heater pin was 9.5 mm in outer diameter and 400 mm in length of effective heat part and the maximum liner power of 750 W/cm. And type K thermocouple of 0.5 mm in outer diameter was set up at every 80 mm from the center of the effective heat part on surface of the heater pin. The inner tube and the cooling system were designed that coolant is light water, BWR conditions (10MPa in the pressure and 320°C in the temperature), PWR conditions (17.3MPa in the pressure and 375°C in the temperature) and the maximum flow rate is 10cm^3/s. The outer tube and the cooling system were designed that coolant is light water, coolant conditions (0.6MPa in the pressure and 100°C in the temperature) and the maximum flow rate is 1.9m^3/h.

Table 3 shows major specifications of the out-of-pile test facility. The inner tubes were 32mm in the outer diameter and can be exchanged for a tube of 27 mm in inner diameter under the BWR condition and a tube of 25 mm in inner diameter under the PWR condition. The outer tube was 38 mm in outer diameter and 34 mm in inner diameter. The separate tube, which consisted of two pieces of support plates (4 mm in width and 1.5 mm in thickness), was used for fixing the water supply tube. The pressurized water of the inner tube was injected from a water supply tube which is 5 mm in outer diameter and 4 mm in inner diameter [5].

Table 2 Design conditions of the out-of-pile test facility.

Inner tube	
Coolant : light water	
Pressure : 10 MPa (BWR)	
17.3 MPa (PWR)	
Temperature : 320°C (BWR)	
375°C (PWR)	
Flow rate : Max.10cm^3/s	
Outer tube	
Coolant : light water	
Pressure : 0.6 MPa	
Temperature : Max.100°C	
Flow rate : 1.9 m^3/h	

Table 3 Major specifications of the out-of-pile test facility.

Item	Major specification	
Inner tube	O.D.×I.D. (mm)	32×27 (BWR condition) 32×25 (PWR condition)
Outer tube	O.D.×I.D. (mm)	38×34
Separate tube	O.D.×width ×thick (mm)	16×4×1.5 (2 plates)
Heater pin	O.D.(mm)	9.5

4. EXPERIMENTAL

Table 4 shows the test conditions of experiments. In order to evaluate on the safe side, the pressure conditions in the inner tube were higher than the coolant conditions of BWR (7.3 MPa) and PWR (15 MPa). The flow rate was 1 cm^3/s simulated the flow rate of the pressurized water on the power ramping test. The coolant condition of the outer tube simulated the OSF-1. In the experiments, the surface

Table 4 Test conditions of experiments.

Inner tube
Coolant : Water
Pressure : 7.5 MPa (BWR condition)
15.5 MPa (PWR condition)
Flow rate : 1 cm^3/s
Outer tube
Coolant : Water
Pressure : 0.4 MPa
Flow rate : 1.9 m^3/h
Temperature of the feeding water : 20°C
Liner power of heater pin : Max.600 W/cm

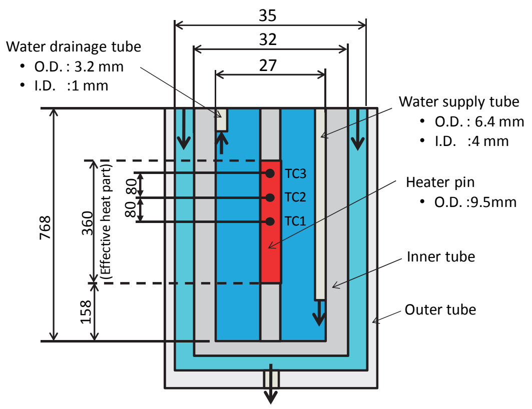

Fig.5 Schematic configuration of analysis model.

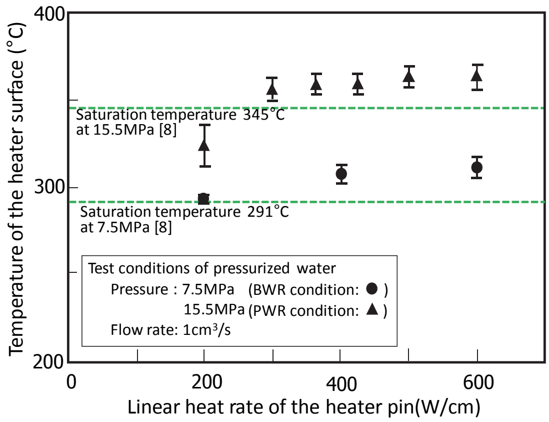

Fig.6 Results of thermal-hydraulic tests.

Table 5 Calculation conditions.

Inner tube
Coolant : Water
Pressure : 7.33MPa
Flow rate : 0.9cm^3/s
Outer tube
Coolant : Water
Pressure : 0.4MPa
Flow rate : 1.9m^3/h
Heat transfer coefficient : 2.33W/cm^2/K
Liner heat rate of heater pin : 600W/cm

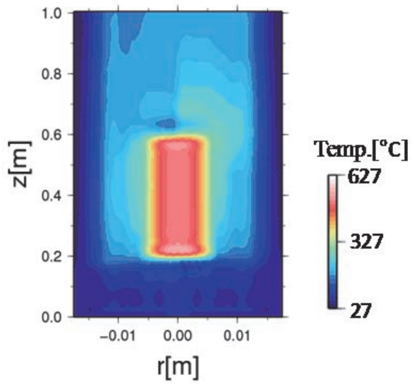

Fig.7 Calculation results under BWR conditions on the liner power of 600 W/cm.

temperature of the heater pin was measured when the linear power of the heater pin was heated up to 600 W/cm, which is the maximum linear power of the power ramping test.

5. CALCULATION

The ACE-3D is three-dimensional two-fluid model analysis code used by the finite element method. The convection term is applied to upwind difference method and the diffusion term is applied to central difference method. Time marching method is applied to semi-implicit method. The two-phase flow turbulent model is used to standard k-ε model. The fluid is capable to simulate water-vapor or water-air two-phase flow [6, 7].

Figure 5 shows schematic configuration of the analysis model. The analysis model consisted of a heater pin (9.5 mm in outer diameter and 360 mm in length of effective heat part), a inner tube (32 mm in outer diameter and 27 mm in inner diameter), flow channel of outer tube (35 mm in outer diameter and 32 mm in inner diameter), water supply tube (6.4 mm in outer diameter and 4 mm in inner diameter) and water drainage tube (3.2 mm in outer diameter and 1 mm in inner diameter).

Table 5 shows the calculation conditions. The coordinate system of the analysis model was cylindrical, and three-dimensional non-stationary analysis was carried out.

6. RESULTS AND DISCUSSION

Figure 6 shows the results of the experiments. In the BWR condition, the maximum temperature was about 317°C on the liner power of 600 W/cm, which was 26°C higher than the saturation temperature 291°C at 7.5 MPa [8]. In the range of 200~600 W/cm, the surface temperature was a little higher than the saturation temperature. In the PWR condition, the maximum temperature was about 370°C on the liner power of 600W/cm, which was 25°C higher than the saturation temperature 345°C at 15.5 MPa [8]. In the range of 300~600 W/cm, the surface temperature was 4°C higher than the saturation temperature.
From these results, it was confirmed that DNB is not occurred when the heater pin is heated up to 600 W/cm under coolant conditions at 7.5 MPa and 15.5 MPa. Therefore, there were prospects that the linear power of a fuel sample with an outer diameter of 9.5 mm is capable to achieve to 600 W/cm without DNB in the power ramping test.

Figure 7 shows the calculation results under BWR conditions of the liner power of 600 W/cm. The surface temperature of the heater pin was about 304°C.And, the calculated surface temperature of the heater pin agreed with the experimental results, which were 304.9~317.4°C, within -4.2~-0.2% error. From these results, there were prospects that the ACE-3D has capability to evaluate the surface temperature of fuels with the power ramping tests.

7. CONCLUSIONS

The fuel transient test facility was prepared in order to carry out power ramping tests of LWR fuels.

It is necessary to predict and evaluate the irradiation conditions accurately such as the neutron flux, the temperature and the liner power of the fuels.

The out-of-pile test facility to simulate thermal-hydraulic conditions of the fuel transient test facility was therefore developed, and the applicability of the analysis code ACE-3D was examined for evaluation of surface temperature of LWR fuels with the power ramping tests of 10×10 BWR fuels.

As the result, calculated temperature is good agreement of measured one within -4.2~-0.2% error under BWR conditions of the liner power of 600 W/cm. Therefore, it was confirmed that the ACE-3D code had capability to evaluate surface temperature of LWR fuels with the power ramping tests in the JMTR.

ACKNOWLEDGEMENTS

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5.3 Improvement and Utilization of Irradiation Capsule Technology in HANARO

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Several improvements of irradiation capsule technology regarding irradiation test parameters, such as temperature and neutron flux/fluence, and regarding instrumentation have progressed at HANARO since the last KAERI-JAERI joint seminar held in 2008. The standard HANARO capsule technology that was developed for use in a commercial power plant temperature of about 300℃ was improved to apply to a temperature range of 100-1000℃ for the irradiation test of materials of new research reactors and future nuclear systems. Low-flux and long-term irradiation technologies have been developed at HANARO. As a beginning step of the localization of capsule instrumentation technology, the irradiation performance of a domestically produced thermocouple and LVDT will be examined at HANARO. The accuracy of an evaluation of neutron fluence and precise welding technology are also being examined at HANARO.

Based on these accumulated capsule technologies, a HANARO irradiation capsule system is being actively utilized for the national R&D programme on commercial nuclear reactors and nuclear fuel cycle technology in Korea. HANARO has recently started the irradiation support of R&D relevant to future nuclear systems including SMART, VHTR, and SFR, and HANARO is preparing new support relevant to new research and Fusion reactors.

Keywords: HANARO, Irradiation capsule, Low-flux irradiation, Instrumentation, Future nuclear system

1. INTRODUCTION

The High Flux Advanced Neutron Application Reactor (HANARO) is an open pool type multipurpose research reactor with 30MW thermal power located at the Korea Atomic Energy Research Institute (KAERI) in Korea. Its general design features and detailed information are available at the HANARO home page (http://hanaro.kaeri.re.kr). In an effort to boost the nation's research capability, HANARO was conceived and constructed in the 1980s using domestic reactor technology from KAERI [1]. HANARO has been operated as a platform for basic nuclear research in Korea and the functions of its systems have been improved continuously since its first criticality in February 1995. It is now being successfully utilized in such areas as fuel and material irradiation tests, neutron beam research, radioisotope production, neutron activation analysis, and neutron transmutation doping to meet industrial, academic, and research demands.

To support the national research and development programs for nuclear reactors and nuclear fuel cycle technology in Korea, various neutron irradiation facilities such as rabbit (small non-instrumented capsule) irradiation facilities, capsule irradiation facilities, and fuel test loop facilities have been developed and actively utilized for the irradiation tests requested by numerous users [2,3]. Among the irradiation facilities, the capsule is the most useful device for coping with the various test requirements at HANARO. Therefore, it has played an important role in the integrity evaluation of reactor core materials and the development of new materials through precise irradiation tests of specimens such as a reactor pressure vessel, reactor core structural materials, fuel assembly parts, and high technology materials at HANARO [4].

In this paper, the improvement and utilization of irradiation capsule technology progressed at HANARO since the last KAERI-JAERI joint seminar held in 2008 are described.

2. HANARO AND CAPSULE

2.1 HANARO

HANARO is an open pool type multipurpose research reactor located at the Korea Atomic Energy Research Institute (KAERI). There are several vertical test holes such as CT, IR1, IR2 (hexagonal type) and OR (cylindrical type) in the core of HANARO, and additionally, Large Hole (LH), Hydraulic Transfer System (HTS), Neutron Transmutation Doping (NTD) and Irradiation Position (IP) positions in the reflector region of the reactor for nuclear fuels and materials irradiation testing, RI production and Si doping, as shown in Fig.1.

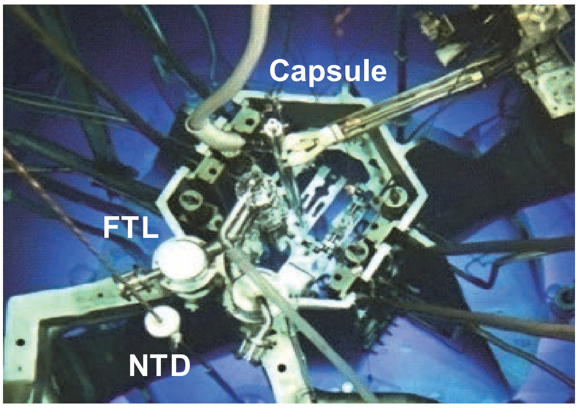

Fig.1 Reactor core of HANARO.

At present, another research reactor that will specialize in radioisotope production and demonstrations of reactor designs is under construction in Korea. Therefore, HANARO will specialize more on the irradiation research of nuclear fuels and materials.

2.2 Irradiation Rabbit and Capsule

Various neutron irradiation facilities such as the rabbit irradiation facilities, the loop facilities and the capsule irradiation facilities for irradiation tests of nuclear materials, fuels and radioisotope products have been developed at HANARO [2,3]. Among the irradiation facilities at HANARO, the capsule and rabbit systems as shown in Fig.2 have been used for the irradiation of nuclear materials. The FTL was installed in IR1 by the end of 2008 and commissioning of the system was performed up to September 2009. At present, a cold test is being performed for long-term utilization. The FTL will be used for the irradiation tests of advanced fuels after completion of the commissioning.

The rabbit was originally designed for isotope production, but it can be used for the irradiation test of fuels and materials. It is very useful for numerous irradiation tests of small specimens at a low temperature, below 200°C, and neutron flux conditions.

The instrumented and non-instrumented capsules have been developed at HANARO for new alloy and fuel developments and the lifetime estimation of nuclear power plants (NPPs). For the development of an instrumented capsule system, the capsule related systems such as a supporting, connecting and controlling system were also developed. After locking the capsule in a test hole, the instrumented capsule is fixed by a chimney bracket and robotic arm supporting systems. Two sets of cantilever type robotic arm systems for the CT and IR2 test holes were installed at the location of the platform level of the reactor, which is 5.5 m in height from the bottom of the capsule, but the in-chimney bracket is temporarily installed on the top of the reactor chimney for capsule irradiation tests. At the junction box system, heaters and thermocouples can be easily connected to and separated from the capsule controlling system before or after an irradiation test. The capsule temperature control system consists of three subsystems: a vacuum control system, a multi-stage heater control system and a man-machine interface system. After an irradiation test, the main body of the instrumented capsule is cut off at the bottom of the protection tube with the cutting system, and it is transported to the Irradiated Materials Examination Facility (IMEF) by using a HANARO fuel cask.

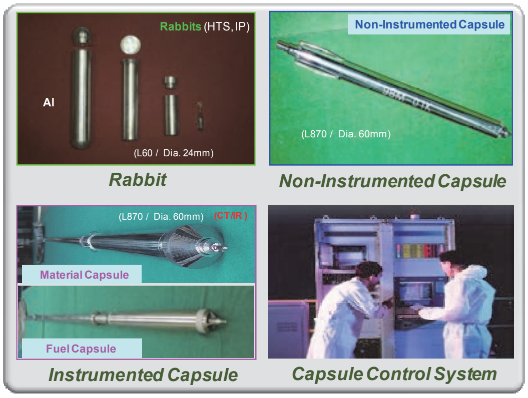

Fig.2 Irradiation rabbit, non-instrumented capsule and an instrumented capsule.

3. IMPROVEMENT OF CAPSLUE TECHNOLOGY

The standard HANARO capsule technology was developed for use in a commercial power plant temperature of about 300 ℃ and continuously improved to apply for the irradiation test of materials of new research reactors and future nuclear systems. In the previous KAERI-JAEA joint seminar held in 2008, capsule technologies for a more precise control of the irradiation temperature and fluence of a specimen, irrespective of the reactor operation, OR/IP capsule technologies for an irradiation test in the HANARO OR and IP test holes with a relatively lower neutron flux than the CT and IR test holes, and high temperature irradiation technology were introduced as new developing capsule technologies in KAERI (see Fig.3).

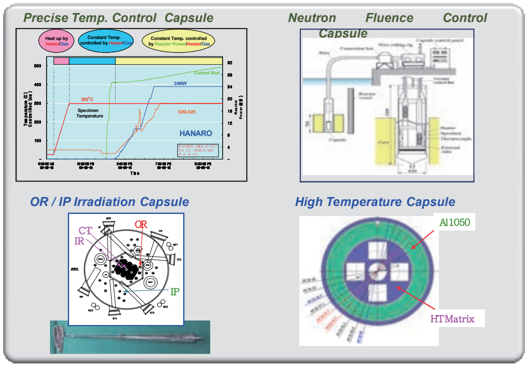

Fig.3 Irradiation technologies introduced in the previous 2008 K-J seminar presentation.

The OR/IP capsule technologies were successfully developed and applied recently for the irradiation tests of an International Nuclear Energy Research Initiative (I-NERI) Project for VHTR, Project for lifetime extension of PWR RPV, and System-integrated Modular Advanced Reactor (SMART) project.

To effectively support R&D relevant to new nuclear systems of the very high temperature reactor system (VHTR) and sodium cooled fast reactor system (SFR), the development of high temperature irradiation technologies is being preferentially developed at HANARO. Several capsules with double layered thermal media, in which the outer one is aluminum and the inner contains high temperature resistant materials such as Ti, Mo, Fe, Zr, Gr as shown in Fig.4, were designed and in-pile tested at HANARO. The structural integrity and irradiation safety of the capsule with double layered structure were confirmed up to 800℃.

The capsule technologies for a more precise control of the irradiation temperature and fluence of a specimen, irrespective of the reactor operation, were successfully out-pile tested but were not applied in reactor. There was no user's need for those technologies up to now.

Recently, there is a great need for precise inspecting safety status of nuclear power reactor, especially for the life-extended old reactor.

The development of future nuclear systems such as VHTR, SFR, and Fusion reactors, is one of the most important projects planned by the Korean government. The environmental conditions for these reactors are generally beyond present day reactor technology, especially as regards the combinations of operating temperatures, reactor coolant characteristics, and neutron flux and spectra.

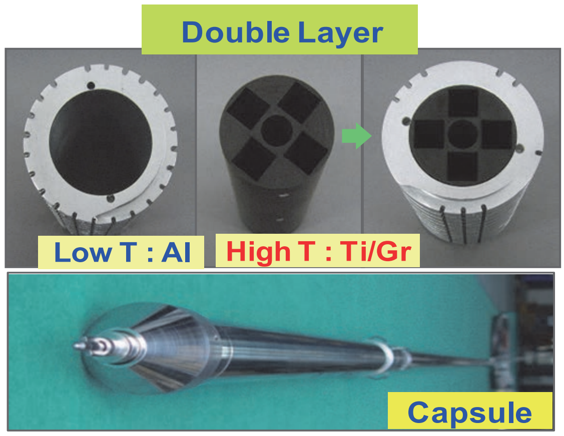

Fig.4 Double layered thermal media and high temperature irradiation capsule.

The Korean government hopes to be a crucial reactor vendor in the global nuclear market through the construction of a 5MW multipurpose research reactor, called the Jordan Research and Training Reactor (JRTR), and another research reactor in Gijang. The new research reactor in Gijang will also become the most up-to-date research reactor available in the world and will specialize in radioisotope production and demonstrations of reactor designs. Therefore, HANARO will specialize more on the irradiation research on the irradiation research of nuclear fuels and materials.

To effectively support these national R&Ds relevant to present NPP, research/SMR reactors, and future nuclear systems, the development of advanced irradiation technologies concerning irradiation temperature and instrumentation is being preferentially developed at HANARO. Based on the accumulated experience and users' sophisticated requirements, advanced capsule technologies are being recently developed for more precise control and analysis of neutron irradiation effects at HANARO.

The standard HANARO capsule technology that was developed for use in a commercial power plant temperature of about 300℃ will be improved to apply to a temperature range of 100-1000℃ for the irradiation test of materials of new research reactors (low temperature irradiation) and future nuclear systems (high temperature irradiation). Low-flux and long-term irradiation technologies will be developed for the precise evaluation of the irradiation characteristics of the structural materials of commercial and future nuclear systems.

Several improvements of irradiation capsule technology regarding instrumentation have also progressed at HANARO since the seminar. As a beginning step of the localization of capsule instrumentation technology, the irradiation performance of a domestically produced K-type thermocouple and Linear Variable Differential Transformer (LVDT) for elongation and pressure have been examined and compared to foreign products at HANARO. Fig.5 shows the design characteristics and final product of the elongation and pressure LVDTs made by KAERI. The accuracy of an evaluation of neutron fluence and precise welding technology are also being examined at HANARO.

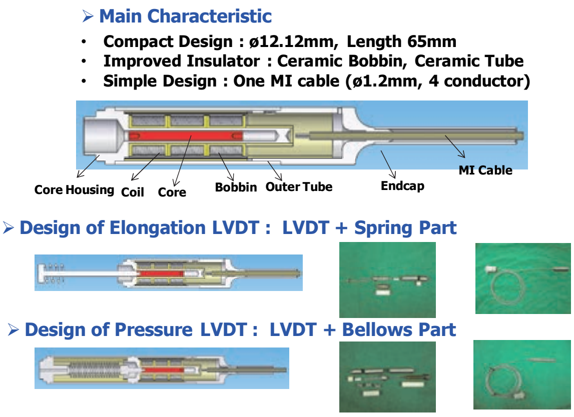

Fig.5 Development of KAERI's LVDT.

4. UTILIZATION OF CAPSULE

The irradiation facilities of HANARO have been actively utilized for various nuclear fuel and material irradiation tests requested by users from research institutes, universities, and industries. Fig.6 shows the trends of the irradiation specimens and the time requested by users. The increasing trends of the irradiation tests were recently disturbed by the installation of the CNRF and FTL. Since 1995, 11000 specimens from research institutes, nuclear industry companies and universities have been irradiated at HANARO for 113000 cumulative hours using the developed capsule and rabbit irradiation systems.

The national R&D programme on nuclear reactors and nuclear fuel cycle technology in Korea requires numerous in-pile tests at HANARO. Most irradiation tests at HANARO have been related to R&D relevant to commercial nuclear power reactor ageing management and the safety evaluation of its components. The capsules were mainly designed for the irradiation of a reactor pressure vessel, reactor core materials, and Zr-based alloys.

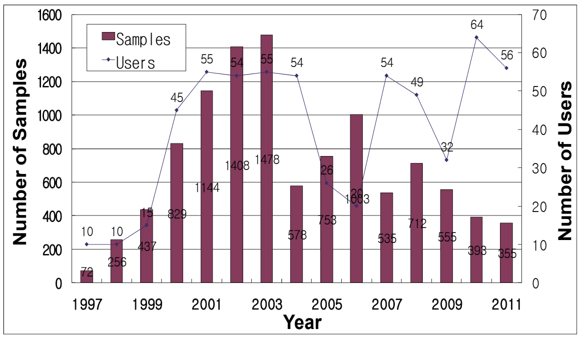

Fig.6 Annual trends of utilization of HANARO.

Most capsules were made for KAERI R&D, but some capsules have been applied to several commercial-based irradiation tests relevant to the lifetime extension of the current nuclear power reactor Kori-1, new alloy and fuel developments with Doosan Heavy Industry Company (DHI) and KEPCO Nuclear Fuel Company (KNF). The archive material of the reactor pressure vessel of the Kori-1 reactor, which is the first NPP in Korea, was irradiated and evaluated to support the lifetime extension of the reactor, and the neutron irradiation performance of the Korean-made commercial RPV materials was also evaluated at HANARO. Several fuel irradiation capsules were designed and irradiated at HANARO to improve the nuclear fuel cycle technology of power and research reactors, and were also irradiated for an evaluation of the neutron irradiation properties of the nuclear fuel assembly parts fabricated by KNF.

HANARO has recently started new support of R&D relevant to new nuclear systems including SMART and future nuclear systems of VHTR and SFR.

The SMART is one of the most advanced small and medium sized reactors (SMRs) in the world [5]. There has also been a growing interest in small and medium sized reactors in developed countries that have deregulated their electricity market, under a call for flexibility in power generation. One of the most important material performance issues is fracture toughness for which an engineering database is necessary to design a steam generator. Because the SMART steam generators are located inside the reactor vessel, the degradation of the fracture toughness of the Alloy 690 heat exchanger tube should be clearly determined for a design lifetime neutron fluence. However, the neutron irradiation characteristics of the alloy are barely known. Therefore, irradiation tests of the Alloy 690 materials to obtain the neutron irradiation characteristics of the alloy were successfully performed at HANARO.

The Generation IV (GEN-IV) International Forum, or GIF, was chartered in July 2001 to lead the collaborative efforts of the world's leading nuclear technology nations to develop next generation nuclear energy systems to meet the world's future energy needs. Among the six GEN-IV systems, Korea has participated in the VHTR and SFR R&D programmes.

The VHTR is one of the leading reactor designs, with participation between Korea and the US. VHTR technology addresses the advanced concepts for a helium gas cooled, graphite moderated, thermal neutron spectrum reactor with a core outlet temperature greater than 900 °C. The VHTR environment is unique, and little data exists on the behaviour of materials under irradiation and in the temperature and pressure ranges of interest. At present, no candidate alloy has been confirmed for use as either the cladding or structural material in VHTRs. To meet these challenges, a GEN-IV R&D plan for the structural materials in VHTRs was initiated as an I-NERI Project, which is a bilateral research agreement between the Ministry of Science and Technology (MOST) of Korea and the Department of Energy of the US [6]. 9Cr-1Mo and 9Cr-1Mo-1W steels were selected as candidate materials of a reactor pressure vessel of the VHTR, and were successfully irradiated at HANARO.

KAERI seeks to develop and demonstrate the technologies needed to transmute the long-lived transuranic actinide isotopes in spent nuclear fuel into shorter-lived fission products, thereby dramatically decreasing the volume material requiring disposal and the long-term radio-toxicity and heat load of high level waste sent to a geological repository [7]. Metallic fuel has advantages such as simple fabrication procedures, good neutron economy, high thermal conductivity, excellent compatibility with a Na coolant and inherent passive safety. U-Zr-Pu alloy fuels have been used for SFR related to the closed fuel cycle for managing minor actinides and reducing high radioactivity levels since the 1980s. An irradiation test of U-Zr and U-Zr-Ce fuels at HANARO has been planned to validate the in-reactor performance. The reduced fuel elements of U-Zr and U-Zr-Ce fuels were fabricated and successfully irradiated at HANARO.

HANARO is also preparing new support relevant to a R&D of Fusion reactor. Irradiation tests of structural materials of a Fusion reactor are being technically analyzed in relation to the International Thermonuclear Experimental Reactor (ITER) Project.

As a national platform for basic and nuclear research, HANARO irradiation facilities have been actively used for irradiation tests of basic research requested by users from research institutes and universities. Various specimens such as nuclear fuels and materials, and new functional materials including conductors and optical/electrical materials have been irradiated using capsule and rabbit systems. The electro-magnetic and optical properties of the materials are closely dependent on the size and density of their internal defects, and neutron irradiation is a very effective method to produce micro-defects in these materials. Therefore, neutron technology can be applied effectively toward the development of new materials. Recently, several efforts have been made at HANARO to evaluate the effect of neutron irradiation on the physical properties of various functional materials [8].

5. CONCLUSIONS

As a national platform for basic and nuclear research, the irradiation technology has been continuously improved in HANARO. An irradiation capsule technology regarding irradiation test parameters, such as temperature and neutron flux/fluence, and regarding instrumentation has progressed at HANARO since the last KAERI-JAERI joint seminar held in 2008. The capsule technology will be improved to apply to a temperature range of 100-1000℃ for the irradiation test of materials of new research reactors and future nuclear systems. Low-flux and long-term irradiation technologies, localization of capsule instrumentation technology, and improvement of the accuracy of an evaluation of neutron fluence are also being examined at HANARO.

The HANARO irradiation technology has been actively utilized for the national R&D programme on commercial nuclear reactors and nuclear fuel cycle technology in Korea and has recently started the irradiation support of the National R&Ds relevant to various nuclear systems including SMART, new research reactor, VHTR, SFR, and Fusion reactors.

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5.4 Out-pile Tests for Improved Type Rabbits in JMTR

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Irradiation tests by hydraulic rabbit are expected to increase the demand of medical use radioisotope productions and nuclear human resource development after restart of Japan Materials Testing Reactor (JMTR). Therefore, it is necessary to shorten fabrication period of hydraulic rabbits.

In this study, applicability of the pressure welding type hydraulic rabbit, which was used in Japan Research Reactor-3 (JRR-3), was examined.

As the results, it was confirmed that the fabricated hydraulic rabbit kept airtightness of the external pressure of 2.45 MPa, impact of the fall from 5m in height and every 5 times of thermal cycle under conditions such as 110°C ,200°C and 300°C. Furthermore, it was clear that the fabricated hydraulic rabbit had the endurance up to the inner pressure of 1 MPa. Therefore, there were bright prospects that the pressure hydraulic type rabbit has the applicability of the irradiation test in JMTR.

Keywords: JMTR, Irradiation test, Hydraulic rabbit, Shorten fabrication, Pressure welding type hydraulic rabbit, Design and fabrication, Out-pile-tests.

1. INTRODUCTION

Irradiation tests by hydraulic rabbit are expected to increase the demand of medical use radioisotope productions (RIs) and nuclear human resource development after restart of Japan Materials Testing Reactor (JMTR) [1].

Concerning the domestic radioisotope production of molybdenum 99 (^{99}Mo), irradiation tests by hydraulic rabbits are planned [2]. For the nuclear human resource development, as part of the teaching, irradiation tests of hydraulic rabbits are planned.

Therefore, it is necessary to shorten fabrication period of hydraulic rabbits in the JMTR. In this study, applicability of the pressure welding type hydraulic rabbit, which was used in Japan Research Reactor-3 (JRR-3), was examined.

2. HYDRAULIC RABBIT IRRADIATION FACILITY

The hydraulic-rabbit-irradiation-facility (HR) is a water loop system to transfer the small sized (150mm length) capsule so called rabbit, into and take out from the core by the water flow in the loop. This facility is widely utilized mainly for basic researches and for the production of short-lived RIs [2]. Figure 1 shows the flow chart of the HR.

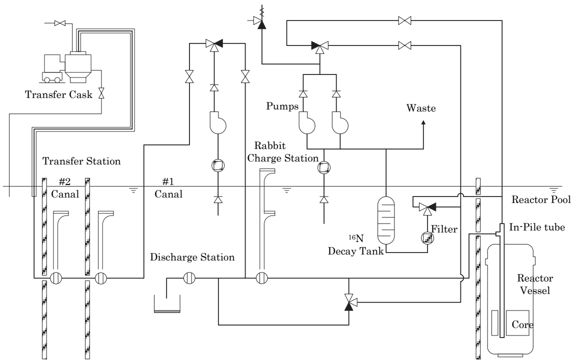

Fig.1 Flow chart of the HR

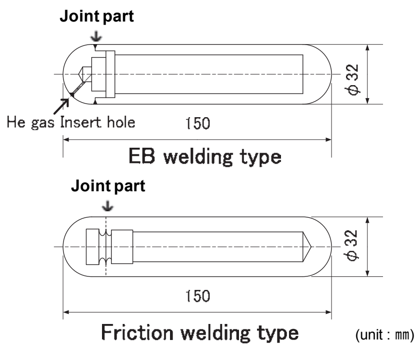

Fig.2 Structures of EB welding type hydraulic rabbit and friction welding type rabbit in JMTR

3. DESIGN OF PRESSURE WELDING TYPE RABBIT

3.1 Concept

Figure 2 shows structures of electron beam (EB) welding type rabbit and friction welding type hydraulic rabbit in JMTR. EB welding method and friction welding method have been used in JMTR [4]. These welding type hydraulic rabbits are made to order, and it has taken production period more than ten days. On the other hand, pressure welding type hydraulic rabbit, which is fabricated easily by such as communization of the parts, has been used in JRR-3[5], and the assembly takes about 2~4 hours. Therefore, the communization of the parts of the hydraulic rabbit using the JMTR was carried out, and the hydraulic rabbit fabricated in a short term was developed.

3.2 Design Fabrication

The design conditions of the pressure welding type hydraulic rabbit for JMTR assumed in external pressure of 1.96 MPa, inner pressure of 0.19 MPa, temperature of 150°C and the inserting impact of 300G. Figure 3 shows the structure of the pressure welding type hydraulic rabbit. The hydraulic rabbit consisted of a body and a cap and a plug. Considering the insertion of the hydraulic rabbit into the in-pile-tube of HR, the size of the body was determined. The body and the cap have thickness joint part and screw part to prevent scatter of parts such as specimen at the time of jointing. The plug has a role to protect the jointed part.

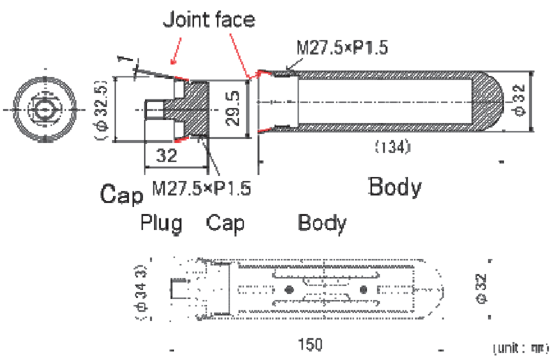

Fig.3 Structure of pressure welding type hydraulic rabbit

4. FABRICATION

Figure 4 shows the process of pressure welding of the hydraulic rabbit. The assembly tightened a screw of the body and the cap was hold by the assembly jig and press from the top, then the joint parts were squeezed tightly and welded. Fabrication tests were carried out using the press machine of 5 ton. As the results, the helium leakage rate was 1.3×10-10 Pa•m^3/s (<1.0×10-6 Pa•m^3/s :acceptance of JMTR [7]). Figure 5 shows the metallographic observation

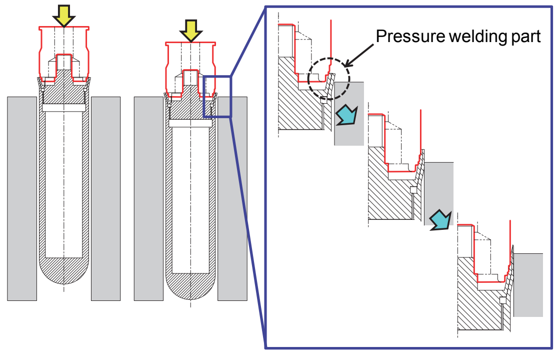

Fig.4 Process of pressure welding of the rabbit

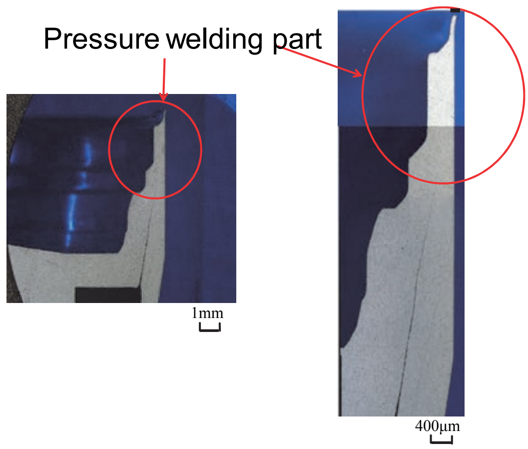

Fig.5 Metallographic observation of the rabbit

of joint part. It was confirmed the connection length of about 2 mm. From these result, it was found that the joint of the fabricated hydraulic rabbit was good.

5. PERFORMANCE TESTS

Fabricated hydraulic rabbits were prepared with the weight of 300g simulated the irradiation test and the weight of 510g simulated the maximum weight of the hydraulic rabbit. The performance tests such as external pressure test, impact test, thermal cycle test and test were carried out using the fabricated hydraulic rabbit

The performance tests were carried out from (1) to (4) in a turn. In order to confirm the sealing performance of the rabbit, the helium leakage rate was carried out in the external pressure test, impact test and thermal cycle test.

(1) External pressure test

External pressure tests were carried out during 12 minutes at 2.45 MPa. As the results, the appearance was good and the helium leakage rate was 1.3×10-10 Pa•m^3/s.

(2) Impact test

The impact tests were performed by dropping the hydraulic rabbit from 5 m high by free fall. As the results, the hydraulic rabbit was shortened, and the deformation was about 1 mm. the helium leakage rate was 1.3×10^{-10} Pa・m^3/s.

(3) Thermal cycle test

Figure 6 shows photograph in the electric furnace. The temperature in the hydraulic rabbit was measured using a mock-up hydraulic rabbit inserted a thermoelectric couple. Every 5 times of thermal cycle were carried out under conditions such as 110°C, 200°C and 300°C. As the results, the helium leakage rate was 1.3×10-10 Pa•m^3/s.

(4) Endurance test

Figure 7 shows photograph of endurance test. Endurance tests were carried out up to the inner pressure of 1 MPa connected a hydraulic pump. As the results, the appearance was good, and it was confirmed that the fabricated hydraulic rabbits had the endurance for the design inner pressure of 0.19 MPa.

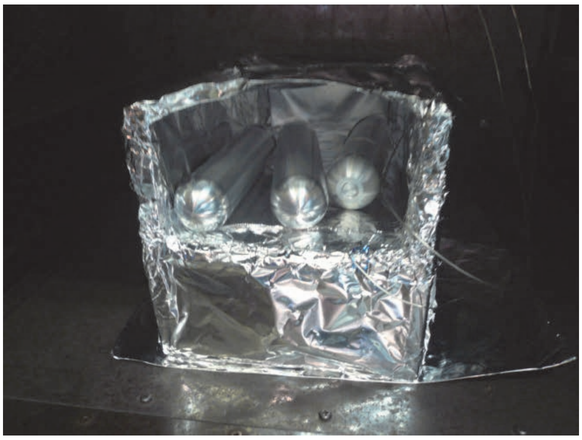

Fig.6 Photograph in the electric furnace

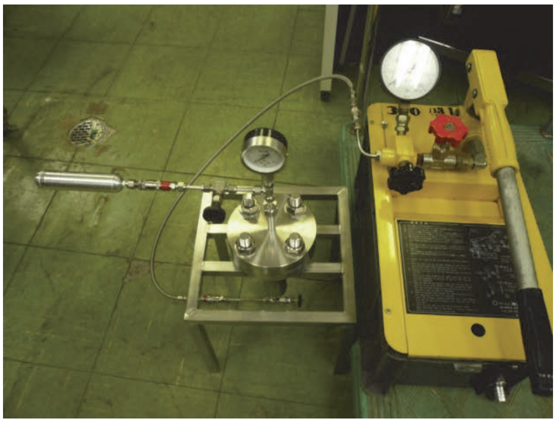

Fig.7 Photograph of endurance test

6. CONCLUSIONS

It was studied of shortening of the production period of the hydraulic rabbit in order to respond to the increased use of the hydraulic rabbit irradiation in JMTR. In this study, the pressure welding type hydraulic rabbit used in Japan Research Reactor-3 (JRR-3) was examined the applicability of the irradiation test in JMTR.

The pressure welding type hydraulic rabbit used in Japan Research Reactor-3 (JRR-3) was examined the applicability of the irradiation test in JMTR. From these results, it was confirmed that the fabricated hydraulic rabbit kept airtightness of the external pressure of 2.45 MPa, impact of the fall from 5m in height and every 5 times of thermal cycle under conditions such as 110°C ,200°C and 300°C. Furthermore, it was clear that the fabricated hydraulic rabbit had the endurance up to the inner pressure of 1 MPa. Therefore, there were bright prospects that the pressure type hydraulic rabbit has the applicability of the irradiation test in JMTR

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5.5 Measurement and Evaluation of Fast Neutron Flux of CT and OR5 Irradiation Hole in HANARO

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ABSTRACT

The irradiation test has been conducted to evaluate the irradiation performance of many materials by a material capsule at HANARO. Since the fast neutron fluence above 1 MeV is important for the irradiation test of material, it must be measured and evaluated exactly at each irradiation hole. Therefore, a fast neutron flux was measured and evaluated by a 09M-02K capsule irradiated in an OR5 irradiation hole and a 10M-01K capsule irradiated in a CT irradiation hole. Fe, Ni, and Ti wires as the fluence monitor were used for the detection of fast neutron flux. Before the irradiation test, the neutron flux and spectrum was calculated for each irradiation hole using an MCNP code. After the irradiation test, the activity of the fluence monitor was measured by an HPGe detector and the reaction rate was calculated. For the OR5 irradiation hole, the radial difference of the fast neutron flux was observed from a calculated data due to the OR5 irradiation hole being located outside the core. Furthermore, a control absorber rod was withdrawn from the core as the increase of the irradiation time at the same irradiation cycle, so the distribution of neutron flux was changed from the beginning to the end of the cycle. These effects were considered to evaluate the fast neutron flux. Neutron spectrums of the CT and OR5 irradiation hole were adjusted by the measured data. The fluxes of a fast neutron above 1 MeV were compared with calculated and measured value. Although the maximum difference was shown at 18.48%, most of the results showed good agreement.

1. INTRODUCTION

Recently, the need for irradiation testing on materials used in the nuclear industry is increasing because of the following reasons;
(1) Demand of safety improvement for operating and aging nuclear power plants due to the Fukushima accident,
(2) Development and performance verification from in-pile test for future nuclear system and fusion materials,
(3) To produce the design data for small and medium size reactor materials such as research reactor and SMART.

Therefore, it is expected that many irradiation tests will be conducted at HANARO by the material irradiation capsule.

The fast neutron fluence is an important factor because it can affect the integrity of material related to safety. Therefore, a fast neutron fluence for the reactor pressure vessel is periodically evaluated and regulated at commercial nuclear power plants[1]. Also a fast neutron fluence must be accurately evaluated for the irradiation-tested materials at HANARO.

Fast neutron fluence is generally calculated by computer code before an irradiation test to design a material capsule. After the irradiation test, a comparison between the calculated and measured data is conducted to evaluate the exact value and the error. Although many evaluations for the irradiation tests have been conducted, the calculated and measured values are over 20% different. Moreover, the dosimetry method for the fast neutron has yet to be established at HANARO.

In this study, the measurement and evaluation of the fast neutron flux was conducted for a 09M-02K capsule irradiated in a OR5 irradiation hole and a 10M-01K capsule irradiated in a CT irradiation hole. The calculated and measured values were compared. These results will be the basic data to evaluate the neutron dosimetry exactly at HANARO.

2. METHODS AND RESULTS

2.1 Calculation of Neutron Flux

During the irradiation test at HANARO, the reactor power is nearly constant. If the neutron flux is directly proportional to reactor power, the neutron fluence can be approximated as follows[2]:

$$\Phi = \left(\frac{\phi}{P}\right)\sum_{i=1}^{n} P_i t_i$$

Where,

ϕ = the neutron flux,

P = the reference power level,

t_i = duration of the i-th operating period,

P_i = reactor power level during that operating period.

Because the neutron fluence is dependent on the irradiation time, it can't be measured directly. Therefore, the neutron flux was calculated from the MCNP code to compare to the measured value.

Fig. 1 shows the calculated axial neutron flux by MCNP code for a 10M-01K capsule irradiated in a CT hole depending on the position of control absorber rod (CAR).

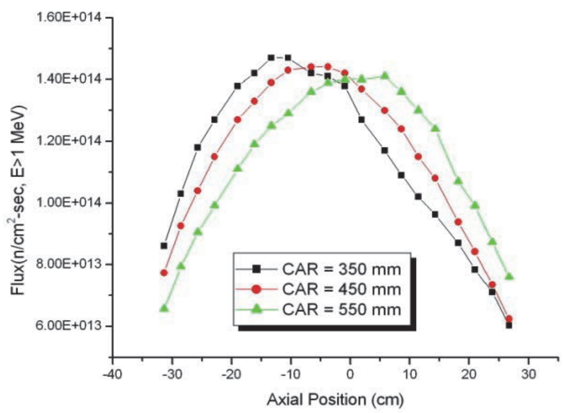

Fig. 1. The calculated axial neutron flux distribution for a 10M-01K capsule irradiated in a CT hole

2.2 Measurement of the Neutron Flux

To measure the neutron flux, the method for applying the dosimeter activation analysis has been used for the material capsule. It is known as an accurate measuring method, therefore, many countries used this method for the measurement of neutron flux[3].

Pure metal foils and wires were used as the fluence monitor (F/M). In this study, Fe, Ni, and Ti wires were used as the F/M due to an active reaction with fast neutron. These were installed in a capsule at each stage. Fig. 2 shows F/Ms and containers used in this study. The schematic diagrams of installed F/M in a capsule were shown fig. 3. Before the installation in a capsule, the weights of the F/Ms were measured. After the irradiation test, these F/Ms were disassembled from the capsule and the absolute activities of the F/Ms were measured by HPGe detector. This data was used for computing the measured value.

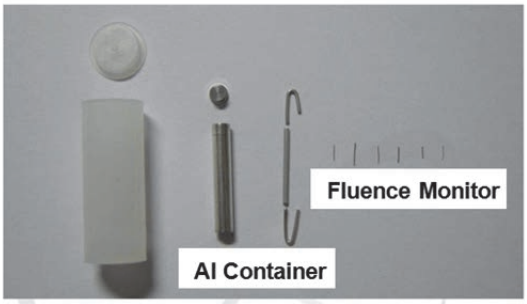

Fig. 2. F/Ms and containers used in this study

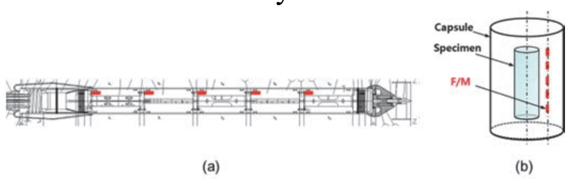

Fig. 3. The schematic diagrams of installed F/M in a capsule
(a) Installed F/M in each stage, (b) Installed position in a capsule

2.3 Evaluation of Fast Neutron F lux

Before computing the measured data, the CAR operating history must be analyzed because the neutron flux distribution is very different according to the position of the CAR shown in Fig. 1. In the same irradiation cycle of HANARO, CAR is increasingly withdrawn from the core due to the combustion of the fuel. Fig. 4 and fig. 5 show the CAR operation history of the 63rd and 64th operation cycle of HANARO that was same with the irradiated cycle of a 09M-02K and a 10M-01K capsule. At the 63rd operation cycle of HANARO, many reactor trips occurred. At the 64th operation cycle, a once-in-a-cycle reactor trip occurred. The average position of the CAR was calculated from this histories were about 458.4 and 509.3 mm. The calculated fast neutron flux was linearly considered from the CAR position of 450 to 550 mm.

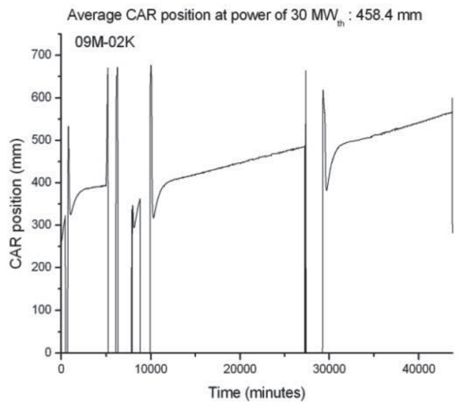

Fig. 4. The CAR operation history of 63rd operation cycle of HANARO

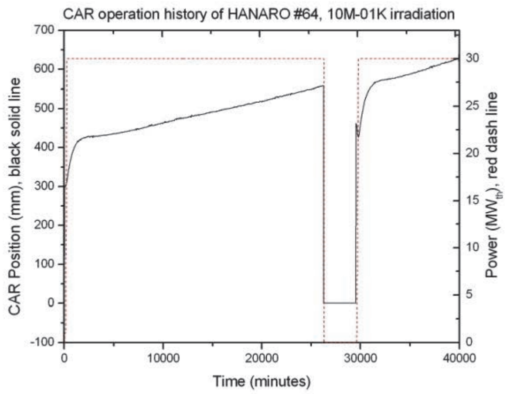

Fig. 5. The CAR operation history of 64th operation cycle of HANARO

To calculate the measured value, many computer codes and methods were used in many countries. SAND-II that is a semi-iterative code to compute the spectrum adjustment was used in this study. Although the neutron reaction cross section was originally provided by the SAND-II code, it was revised by an updated cross section data from ENDF/B-VII library using an NJOY code. To compute the measured fast neutron flux and the neutron spectrum, the initial spectrum calculated from an MCNP code and reaction rates of F/Ms were used as input data for the SAND-II code. The SAND-II code adjusted the initial neutron spectrum and computed the fast neutron flux.

Fig. 6 shows the comparison results of the fast neutron flux between the calculated and the measured data of a 09M-02K capsule irradiated in OR5 irradiation hole. In OR5 irradiation hole, large difference of fast neutron flux was calculated according to the position of F/Ms because it is located outside of the core. The difference of fast neutron flux with average value was showed up to 24.8%. Therefore, in OR5 irradiation hole, the consideration concerning F/M installation position was necessary. Although the maximum difference was observed up to 18.48%, all of results showed within ± 20% in 09M-02K capsule irradiated in OR5 irradiation hole shown in fig. 6..

Fig. 7 shows the comparison results of the fast neutron flux between the calculated and the measured data of a 10M-01K capsule irradiated in a CT hole. Most of the comparison data showed good agreement within ± 10%. However, at the height of 27.25 cm, the comparison result showed 13.03%. On most of the region of the capsule, the fast neutron flux was linearly increased according to the increase of the CAR position shown in Fig. 1. However, at the upper region over the height of 18 cm, the fast neutron flux was sharply increased from the CAR position of 450 to 550 mm. In this study, the calculation data was considered only with linear increase of flux by the increase of the CAR position. It might affect the difference of the comparison between the calculated and measured value.

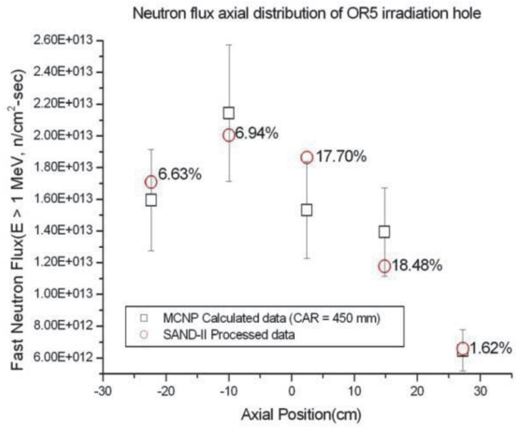

Fig. 6. The comparison fast neutron flux between the calculated and measured value of 09M-02K capsule

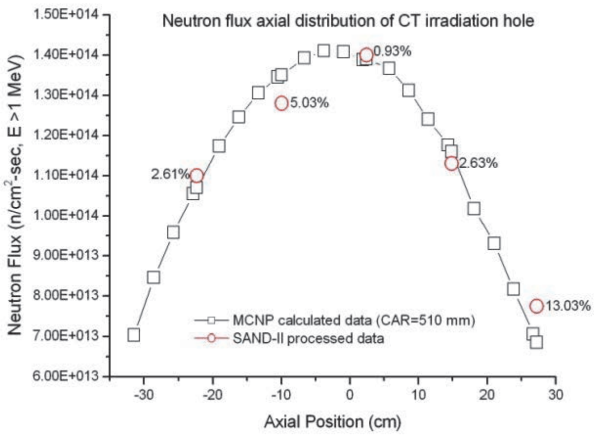

Fig. 7. The comparison fast neutron flux between the calculated and measured value of 10M-01K capsule

3. CONCLUSIONS

The measurement and evaluation of a fast neutron flux was conducted for 09M-02K and a 10M-01K capsule irradiated in OR5 and CT irradiation hole. The fast neutron flux was calculated using MCNP code. The measured data was used via SAND-II code. A CAR operation history was analyzed for accurate data comparison. Although the maximum difference between the calculated and measured value showed 18.48%, most of the comparison results showed a good agreement within ±20%.

ACKNOWLEDGEMENT

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5.6 Neutron Irradiation Tests for Beryllium Material Selection of Neutron Reflector in JMTR

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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. As a part of the reactor upgrade, the Japan Atomic Energy Agency (JAEA) 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 tests for evaluation of the various beryllium grades are carried out in JRR-3 and prepared in JMTR. In this paper, material selection, irradiation tests and PIE development for lifetime expansion of beryllium are described for material testing reactors.

Keywords: Beryllium, Reflector, Lifetime expansion, Material Testing Reactor, JMTR, JRR-3

1. INTRODUCTION

Reactors with Beryllium (Be) as a moderator and/or reflector 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 (^{3}H) is produced in the Be by a well known reaction sequence. It is difficult to reprocess irradiated Be because of high induced radioactivity [1]. Thus, it is necessary to reduce the used beryllium reflector elements [2].

In this paper, material selection, irradiation tests and development of post-irradiation examinations (PIEs) for lifetime expansion of Be are described for material testing reactors.

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} /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.

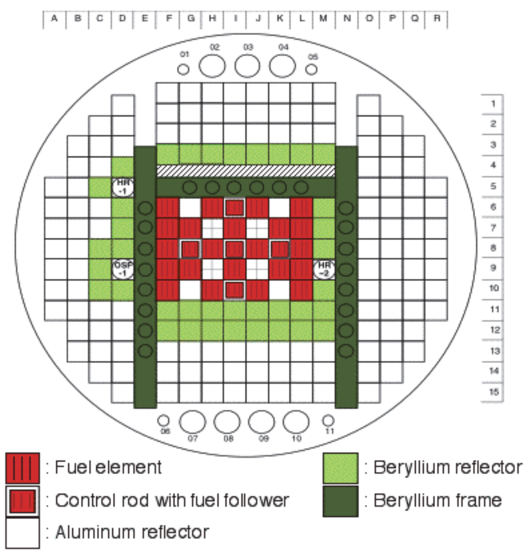
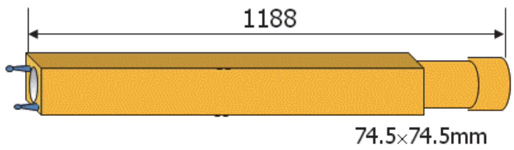
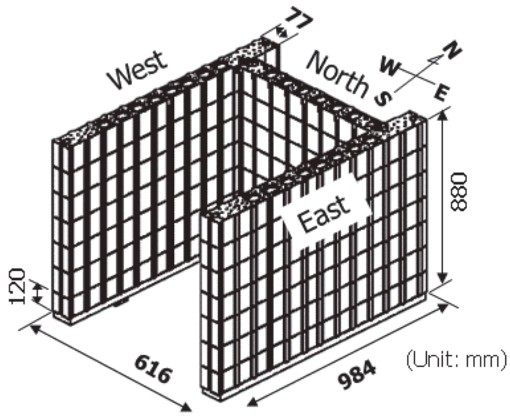


Fig. 1 Core arrangement of JMTR.



(a) Be Reflector Element



(b) Be Reflector Frame

Fig. 2 Configuration of Beryllium reflectors in JMTR.

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 and the frames were exchanged six times up to the JMTR operation periods of 165 cycles.

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 MBBe&C'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

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]. 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.

For the irradiation tests, five kinds of irradiation samples (see Figure 3) were prepared for tensile test, bending test, impact test, H/He release test and observation in MBBe&C.

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

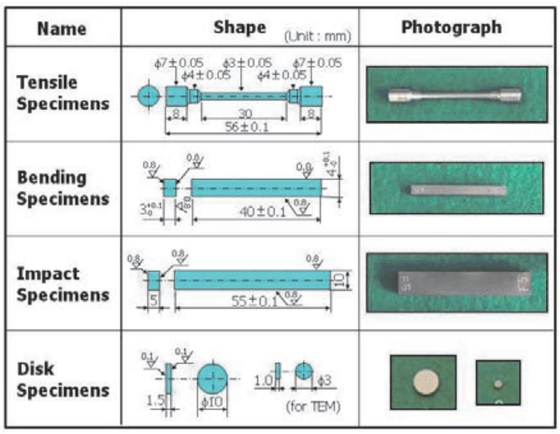


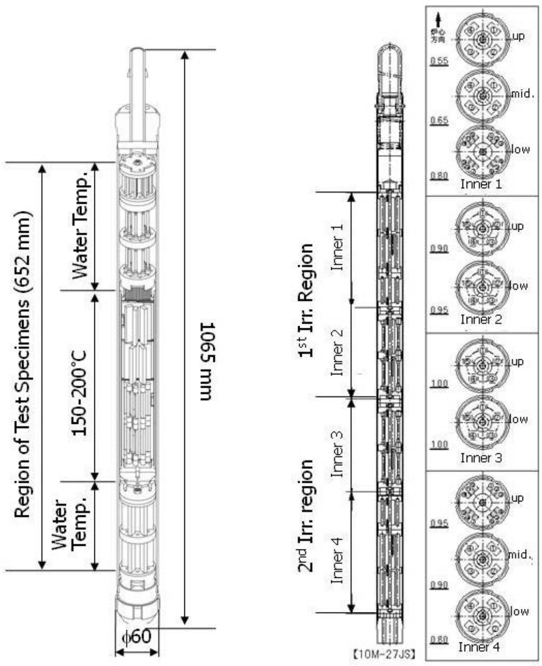
Fig.3 Beryllium specimens for irradiation tests.

4. 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 were finished in JRR-3 and the irradiated Be specimens will be transported from Tokai to Oarai. Configuration of irradiation capsule in JRR-3 is shown in Figure 4(a). 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 4,200h. Irradiation temperatures of Be samples were about 50˚C and 150˚C in cooling water and He gas atmosphere, respectively. Irradiation temperature of the specimens in the inner capsule was evaluated by NISA (Numerically Integrated elements for System Analysis). The result of irradiation temperature is shown in Figure 5.

On the other hand, the irradiation tests in JMTR will be started at the JMTR re-start (Oct., 2012). For the irradiation tests in JMTR, the two irradiation capsules (see Figure 4(b)) were fabricated. In JMTR, the Be specimens will be irradiated at four fast neutron fluences (E>1MeV) of 1, 2, 3.5 and 5×10^{25} /m^2. Irradiation temperature of Be samples was about 50˚C in cooling water.



(a) JRR-3 (b) JMTR

Fig.4 Configuration of irradiation capsule in JRR-3and JMTR.

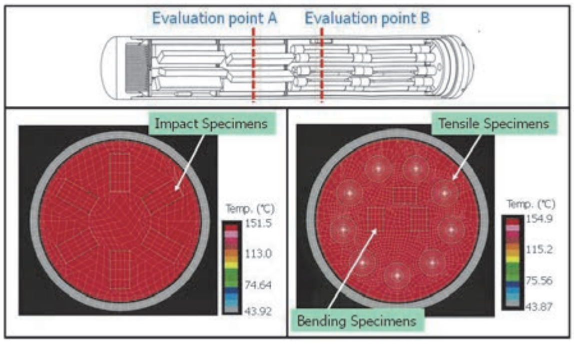


Fig.5 Result of irradiation temperature calculated by NISA.

5. DEVELOPMENT OF PIE TECHNIQUES

As the development of new PIE techniques, non-destructive inspections such as Electromagnetic Transducer (E-MAT) and electrical resistor, surface analysis such as X-ray Photoelectron Spectroscopy (XPS) 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 [5]. The surface analysis of corroded Be specimens was examined by XPS [6]. The preparation method of Be samples and TEM observation results was being examined as a evaluation method of the un-irradiated Be samples [7].

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

6. CONCLUSIONS

Material Selection of Beryllium Grades as MTR Reflectors was discussed in the Be Specialist Meetings. The reference (S-200F), isotropy (S-65H) and strength (I-220H) were selected as candidate beryllium grades and irradiation specimens were prepared for the irradiation tests. The irradiation tests have been carried out in JRR-3 and prepared in JMTR. The properties of the irradiated S-200F, S-65H and I-220H will be measured by the developed techniques such as non-destructive inspections, XPS and TEM after irradiation test.

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6. Post Irradiation Examination Technology (1)

6.1 Experimental validation of transmutation behavior for U and Am samples irradiated under fast neutron spectra based on chemical analysis

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Concepts of effective use and recycling of minor actinides (MAs) in fast reactor were proposed to reduce environmental burden. Evaluation of MAs transmutation behavior plays important roles for this purpose. However, there are a few experimental data. Transmutation behavior of Am and U samples irradiated in the experimental fast reactor "Joyo" were evaluated experimentally by chemical analysis. High purity Am oxides (^{241}Am 99.9 % purity), as the Am sample, were irradiated at different two axial position in the 3rd row of Joyo Mk-II core for 275 effective full power days (EFPDs). High purity U metals (^{238}U 99.8 % purity), as the U sample, were irradiated at different six axial positions in the core center of Joyo Mk-II core for 117 EFPDs. The irradiated samples were dissolved and separated into U, Pu, Am and Cm fractions. Each isotopic compositions and elemental compositions in the irradiated Am and U samples were obtained. Fission to capture ratio, Pu production rate per fluence and Cm production rate per fluence in the Am sampels were evaluated. Their dependence on neutron energy was evaluated. Difference of ^{239}Pu production in the U samples between the core region and the reflector region was evaluated.

Keywords: Isotopic composition, transmutation, Irradiation, Chemical analysis, Americium-241, Uranium-238, experimental fast reactor, Joyo, burnup

1. INTRODUCTION

Minor actinides (MAs) were generated by neutron capture reactions of uranium (U) and plutonium (Pu) in UO_2 and mixed oxide (MOX) fuels. They have radioactive toxicity and long half-lives. They have impacts on environment thorough underground disposal of high-level radioactive waste. Concepts of effective use and recycling of MAs in fast reactors are proposed to reduce environmental burden. Irradiation of MAs containing MOX fuels in core region is proposed. Irradiation of blanket fuels with MAs is also proposed. The latter concept aims to contribute to enhancement of proliferation resistance of produced Pu from U blanket.

Concepts of irradiation of MAs containing MOX fuels are proposed as the Fast Reactor Cycle Technology Development Project (FaCT) in Japan [1,2]. Americium (Am) is one of the important nuclides in MAs and recovered from spent nuclear fuels. Initial Am in MAs containing MOX fuels decrease by transmutation. However, U and Pu newly produce Am. Consequently, total amounts of Am do not change thorough irradiation according to initial Am content. Thus, considering the total Am mass in nuclear fuel cycle, Am should be isolated in the nuclear fuel cycle.

Addition of ^{241}Am to ^{238}U was proposed from view points of proliferation resistance of Pu [3]. Irradiation of ^{238}U produces Pu with high ^{239}Pu ratio. The Pu with high ^{239}Pu content is inferior in proliferation resistance. On the other hand, ^{238}Pu is produced by neutron capture reaction of ^{241}Am. Plutonium-238 is a strong neutron source and a high heat generating nuclide with half-life of 87.7 y. A quality of Pu as nuclear explosive materials is degraded by increasing the ^{238}Pu isotopic composition. Irradiation of ^{238}U with ^{241}Am produces Pu with appropriate ^{238}Pu/^{239}Pu ratio for proliferation resistance.

Transmutation behavior of all actinide elements must be fully understood for evaluation of Am accumulation in nuclear fuel cycle and for evaluation of proliferation resistance previously described. However, there are a few experimental data [4-6]. Therefore, high purity ^{241}Am and ^{238}U samples (Am samples and U samples, respectively) were irradiated in the experimental fast reactor "Joyo". Transmutation behavior of Am and U samples were evaluated experimentally by chemical analysis.

2. EXPERIMENTAL

2.1. Sample Irradiation History

Two ^{241}Am samples and six ^{238}U samples were irradiated in the experimental fast reactor "Joyo" Mk-II core.

About 0.1 mg of ^{241}Am oxide powder with 99.9 % purity was encapsulated in a vanadium capsule (φ = 1.5 mm, length = 8 mm). The two vanadium capsules were loaded into the subassembly number PFB090 of the Joyo MK-II core. The PFB090 subassembly was located in the 3rd row and was irradiated from the 29th to 33rd cycles for 275 EFPD until August 31th, 1999. The two Am samples were set at the core center and in the upper reflector region (+350 mm from the center), respectively. The two Am samples were irradiated by the fast neutron flux of 1.08 x 10^{15} (reflector region) and 3.25 x 10^{15} (core center) n•cm^{-2}•s^{-1} ($E \geq 0.1$ MeV), respectively. Neutron spectra at the Am samples irradiation position were shown in Fig. 1.

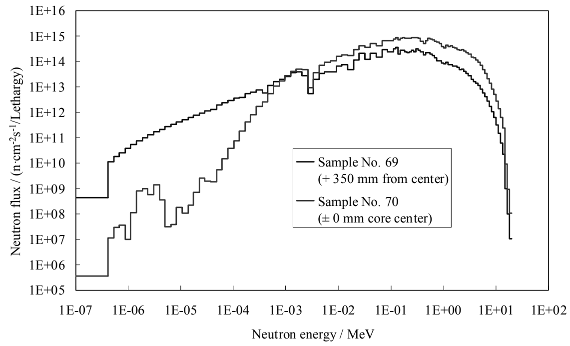

Fig. 1. Neutron spectra at the Am samples irradiation position

About 1 or 10 mg of ^{238}U metal were encapsulated in a vanadium capsule (φ = 1.5 mm, length = 8 mm). Their isotopic purities were above 99.83 at% (^{235}U : ^{238}U = 0.17 : 99.83 or ^{235}U : ^{238}U = 0.04 : 99.96). The six vanadium capsules were loaded into the subassembly number PSV010 of the Joyo MK-II core. The PSV010 subassembly was located in the address 000 (the vertical center of the core) and was irradiated from the 34-1th to 35-2rd cycles for 117 EFPD until June 1st, 2000. The six U samples were set at the different axial positions, that is, the lower reflector region (-627 mm from the center), the core region (-257 mm, 0 mm and +257 mm from the center) and the upper reflector region (+627 mm and +924 mm from the center). The six U samples were irradiated by the fast neutron flux of 0.02 x 10^{15} (reflector region) and 2.28 x 10^{15} (core center) n•cm^{-2}•s^{-1} ($E \geq 0.1$ MeV), respectively. Example of neutron spectra at the U samples irradiation position were shown in Fig. 2.

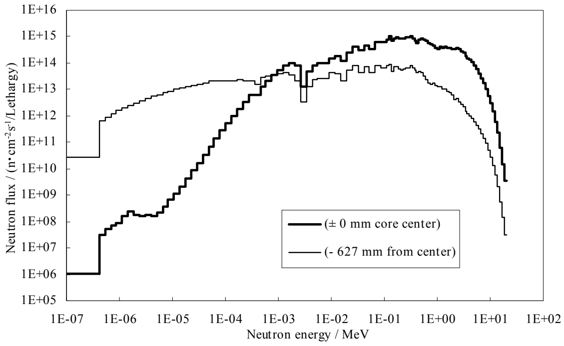

Fig. 2. Neutron spectra at the U samples irradiation position

2.2. Chemical Analysis

The sequential radiochemical separation method (SRCS) was developed by JAEA to analyze U, Np, Pu, Am and Cm in irradiated fuels [7]. This method is composed of samples dissolution by nitric acid, elemental separation by anion exchange methods, isotopic analysis by thermal ionization mass spectrometry, alpha spectrometry by solid state silicon detector and gamma-ray spectrometry by high purity germanium detector. This method was applied to chemical analysis of the Am and U samples. The analysis of the Am and U samples followed the flow sheet shown in Fig. 3. The number of atoms of the U and Pu in the irradiated samples were evaluated by isotope dilution mass spectrometry with of ^{233}U and ^{242}Pu spike solution. The atomic ratio of the Cm/Am and Pu/Am in the Am samples were evaluated by alpha spectrometry of the sample solutions.

All the reagents used for the above procedures were analytical grade.

A multi-channel pulse height analysis system (ORTEC Model 7600-000), coupled with a desktop computer, was used to accumulate and analyze alpha and gamma-ray spectra. A solid state silicon detector, (ORTEC Model BU-017-200-300) was used to get alpha spectra. A coaxial thin window type high purity detector (ORTEC Model GMX-20P4) was used to take gamma-ray spectra. The counting sample for alpha spectrometry was prepared as follows. A 10–100 μL aliquot of sample solution was dropped on a niobium plate (25 mm in diameter) or a stainless plate (25 mm in diameter). The sample was dried by gently heating on a hot plate.

The specifications of the thermal ionization mass spectrometer are shown in Table 1. A 1 μL of aliquot was dropped on rhenium filament. Then, the samples were dried by heating.

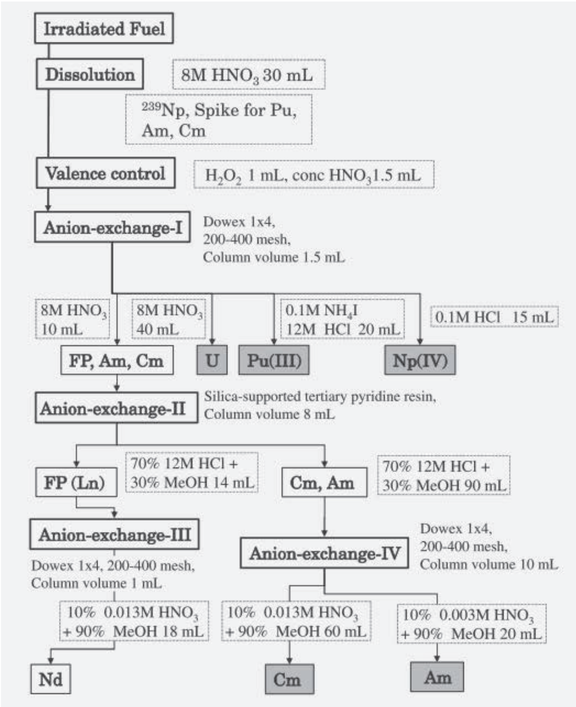

Fig. 3. Schematic flow sheets of chemical analysis

Table 1 specifications of the thermal ionization mass spectrometer

Type	Finnigan MAT Institute MAT 262
Voltage	10 kV
Element examined	U, Pu, Nd, 0.05 – 2 μg (13 samples)
Detector used	Faraday cups : 9, SEM : 1
Detection limit	< 2 ppm (Uranium)
Accuracy	< ±0.03 % (Uranium)
Resolution	> 500 (M/ΔM)

3. RESULTS AND DISCUSSION

3.1. Am Sample

The $^{241\text{-}243}$Am, $^{242\text{-}248}$Cm and $^{238\text{-}242}$Pu in the irradiated Am samples were successfully determined in chemical analysis. The analytical results of Am samples are summarized in Table 2. Isotopic compositions of ^{242m}Am in Am are 1.0092 and 1.4844 at. %. ^{243}Cm and ^{238}Pu are dominant isotopes in Cm and Pu, respectively. The main transmutation chains for ^{243}Cm and ^{238}Pu are shown in Fig. 4. It seemed that the difference of the isotopic compositions of Am, Cm and Pu between the two Am samples was small.

The numbers of fission events were evaluated by ^{137}Cs as burnup monitor nuclide. Usually, the burnup was measured by using the ^{148}Nd monitor method. In this study, the total amount of ^{148}Nd atom generated in the Am samples was very low, and therefore ^{137}Cs was used for burnup determination. The amounts of ^{137}Cs were determined by gamma-ray spectrometry. The details of procedure for determining burnup were described in Ref. [8].

Fission reaction of ^{241}Am was preferred than neutron capture reaction of ^{241}Am in MAs recycling concepts. Fission to capture ratios were evaluated for the two Am samples. The numbers of capture reactions are defined as the sum of all the nuclides except for ^{241}Am. These data are summarized in table 3. As results, fission to capture ratio under fast neutron spectrum is superior to that under softened neutron spectrum.

Table 2 The results of chemical analysis of the irradiated Am samples

Sample position	**350**		**0**	
Neutron Flux	1.08×10^{15}(n·cm^{-2}s^{-1})		3.25×10^{15}(n·cm^{-2}s^{-1})	
Nuclide	Isotopic composition (at%)			
^{241}Am	98.9260	±0.0005	98.4330	±0.0004
^{242m}Am	1.0092	±0.0004	1.4844	±0.0002
^{243}Am	0.0648	±0.0001	0.0826	±0.0005
^{242}Cm	10.73	±0.01	8.08	±0.07
^{243}Cm	77.68	±0.10	83.02	±0.22
^{244}Cm	10.61	±0.11	8.44	±0.04
^{245}Cm	0.31	±0.05	0.17	±0.01
^{246}Cm	0.19	±0.15	0.07	±0.08
^{247}Cm	0.16	±0.11	0.18	±0.18
^{248}Cm	0.312	±0.003	0.04	±0.03
^{238}Pu	81.6	±2.7	81.3	±2.7
^{239}Pu	1.02	±0.03	1.1	±0.0
^{240}Pu	0.232	±0.008	0.149	±0.005
^{241}Pu	0.088	±0.003	0.0286	±0.0009
^{242}Pu	17.06	±0.56	17.42	±0.57
Cm/Am ratio (%)	0.030		0.059	
Pu/Am ratio (%)	7.3		8.1	

Table 3 Fission to capture ratio, Cm production rate and Pu production rate of the Am samples

Axial position from core center	+350	0
Fission to capture ratio	0.129	0.482
Cm production rate per fluence	0.290	0.139
Pu production rate per fluence	2.428	0.828

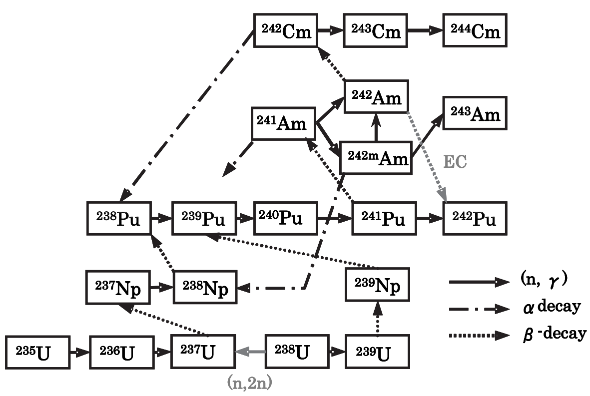

Fig. 4. Main transmutation chains of actinides in fast reactor

Accumulations of higher mass Pu and Cm have to be also inhibited in MAs recycling concepts. Pu and Cm production rates per fluenece of the Am samples were also evaluated. These data are also summarized in table 3. As results, Pu and Cm productions in fast neutron spectrum were more inhibited than those in softened neutron spectrum.

These data will contribute to improvement of nuclear data.

3.2. U Sample

The $^{238\text{-}241}$Pu in the irradiated U samples were successfully determined in chemical analysis. The analytical results of the U samples are summarized in Table 4. Plutonium-239 is produced from ^{238}U passed through ^{239}U and ^{239}Np. The following neutron capture reaction produces higher mass plutonium.

Neodymium-148 was selected for burnup monitor nuclide for determination of the burnups of the U samples. The amounts of ^{148}Nd in the U samples were determined by isotope dilution mass spectrometry. The details of procedure for determining burnup were described in Ref. [9].

Table 4 The results of chemical analysis of the irradiated U samples

Sample position	942	627	258	0	-258	-627
Neutron Flux ($n \cdot cm^{-2}s^{-1}$)	0.02×10^{15}	0.11×10^{15}	1.16×10^{15}	2.28×10^{15}	1.34×10^{15}	0.13×10^{15}
Nuclide	Isotopic composition (at%)					
^{238}Pu	0.01	0.01	0.05	0.05	0.06	0.01
^{239}Pu	99.62	99.47	98.89	99.13	98.58	98.43
^{240}Pu	0.29	0.49	0.99	0.80	1.28	1.58
^{241}Pu	0.08	0.03	0.02	0.02	0.03	0.05
^{242}Pu	N.D[b)]	N.D[b)]	0.05	N.D[b)]	0.05	N.D[b)]
Pu production ratio (%)[a)]	0.22	0.42	0.60	1.01	0.55	1.34
Burn up (%FIMA)	0.002	0.007	0.149	0.353	0.179	0.040

a) ^{239}Pu atoms / initial U atoms x 100
b) Not detected

Axial distribution of the ^{239}Pu production ratio, the burnup in the U samples and neutron flux in the subassembly SVIR-1 are shown in Fig. 5. The axial distribution of the burnups almost corresponds to the neutron fluence distribution. While, the axial distribution of the ^{239}Pu production ratios does not corresponds to the neutron fluence distribution, especially, far from core center.

The relation of the ^{239}Pu production ratios and the neutron fluences is shown in Fig. 6. The ^{239}Pu production tendencies were different between core region and reflector region.

Neutron energy dependence of Pu production from U and Am has to be considered in order to propose U-Am blanket fuel for enhancement of proliferation resistance. These data will contribute to design of U-Am blanket fuel concepts.

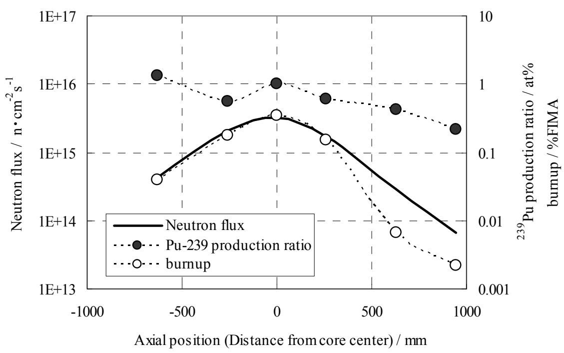

Fig. 5. Axial distribution of the ^{239}Pu production ratios, the burnups in the irradiated U samples and neutron flux in the subassembly SVIR-1

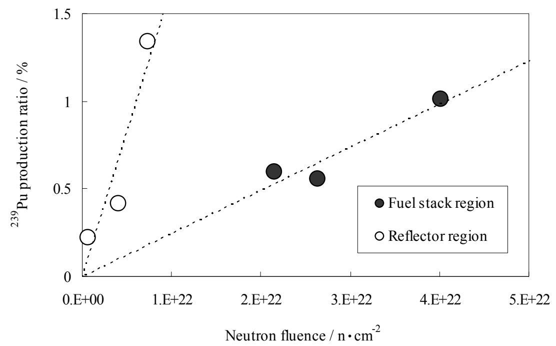

Fig. 6. ^{239}Pu production ratio as a function of neutron fluence

4. SUMMARY

The high purity Am and U samples were irradiated in the experimental fast reactor "Joyo". Transmutation behavior of Am and U samples were evaluated experimentally by chemical analysis.

As results, $^{241\text{-}243}$Am, $^{242\text{-}248}$Cm and $^{238\text{-}242}$Pu were successfully determined in chemical analysis of irradiated the Am samples. Neutron energy dependence of fission events and Pu, Cm production were evaluated. Fission to capture ratio is higher under fast neutron spectrum than softened neutron spectrum. Pu and Cm production is lower under fast neutron spectrum than softened neutron spectrum.

Plutonium-238-241 were successfully determined in chemical analysis of irradiated the U samples. Neutron energy dependence of Pu production was evaluated. Pu production is higher in softened neutron spectrum than in fast neutron spectrum.

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6.2 Development of Mechanical Test Techniques for Structural Components of Irradiated PWR Fuel Assembly

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An increase of fuel burnup and duration of fuel life remains one of the main methods for a nuclear power engineering enhancement. Properties of structural materials providing corrosion resistance, mechanical strength, and dimensional instability of the components of a fuel assembly (FA) are of great importance for fuel operational reliability in such fuel life cycles. Generally, PWR fuel assemblies consist of a top nozzle, spacer grid, bottom nozzle, and guide/instrumentation tubes. The top and bottom nozzle are fixed to the guide tubes using a screw or bulge method. The spacer grid fixed to the guide/instrumentation tubes using a spot weld or bulge method. To understand the in-reactor performance of PWR FA, several devices and test techniques have been developed for mechanical property tests. Among the structural components of PWR FA, a spacer grid, a hold down spring of a top nozzle and a connecting part of FA were considered. Experimental works were carried out for the unirradiated and irradiated components of advanced nuclear fuel assemblies for KSNPs and Westinghouse type PWRs at IMEF (Irradiated Materials Examination Facility) at KAERI. The developed techniques were verified through a hot cell tests.

Keywords: Fuel Assembly, Irradiation, PIE, Spacer Grid, Top Nozzle, Cell Spring, Static Buckling, Hold Down Spring

1. INTRODUCTION

Since the PWR was operated in the middle of 1970's in Korea, the fuel related technologies had been developed continuously. In the early 1990's the core design and fuel fabrication technologies were almost established the level of advanced foreign vendors. Korea Nuclear Fuel Company (KNF) has accumulated a high level technology for a fuel design and fabrication up to now. However, it is necessary to have a self-reliance capability for the performance evaluation of a spent fuel assembly (FA), so that we expand our market in the world. To establish our own technology for developing nuclear fuel assembly, it is very important to possess the fundamental technologies such as mechanical and structural analyses and test capabilities.

PWR fuel assembly consists of fuel rods, top nozzle, spacer grids, bottom nozzle, guide tubes and instrumentation tube(s) in general. The spacer grid is one of major components of a PWR fuel assembly. It supports and locates fuel rods properly in the fuel assembly. It also provides a flow channel between the fuel rods, which assists in transferring the heat from the fuel pellet into the coolant in a reactor. It shall be designed in shape to have a function of enhancing the thermal margin and maintaining the fuel rod integrity without fuel failure due to fuel rod fretting and flow-induced vibration. The top nozzle guides the control rods, applies a hold down force to fuel assembly, and is used to handle and transfer a fuel assembly. It shall be designed in shape to reduce its height to accommodate the fuel rod growth for high burnup and to have a function for easy reconstitution of the fuel assembly.

The mechanical tests of a newly designed or irradiated PWR structural components offer a basic essential data to verify the design and to evaluate the performance of the fuel assembly in core. However, the mechanical test techniques in a spent fuel assembly components had not been fully established up to now in Korea. If such tests are conducted in abroad, it is evitable to open our designs and test results to the competing foreign fuel vendor. Therefore, we have developed several mechanical test techniques to produce the performance evaluation data for the new structural components that might be developed in the future and irradiated structural components in use. Finally, we will have capabilities not only to achieve a self-reliance in fuel development but also to compete with foreign vendors in the international market.

2. SPACER GRID

Spacer grid supports and locates fuel rods properly in the fuel assembly. We developed two kinds of test techniques; a cell spring test and a static buckling test.

The purpose of a cell spring test is to obtain a force versus deflection curves of an inner and outer cell spring. From this curve, the maximum spring force at the specified deflection values and the spring stiffness are calculated. The changes of cell spring properties of spacer grid after irradiation are evaluated by this test. Fig. 1 shows the conceptual drawing of a cell spring test. The spacer grid is tightly fixed at the grid fixture of lower grip, and a loading rod is inserted into the test cell through upper grip. And then, the axial compressive load is applied to the cell spring until the displacement reaches to the specified displacement. The test fixture of a cell spring test is shown in Fig. 2.

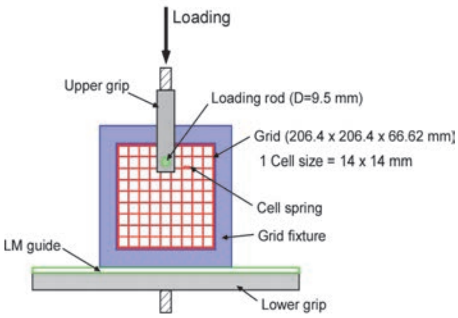


Fig.1. Conceptual drawing of a cell spring test

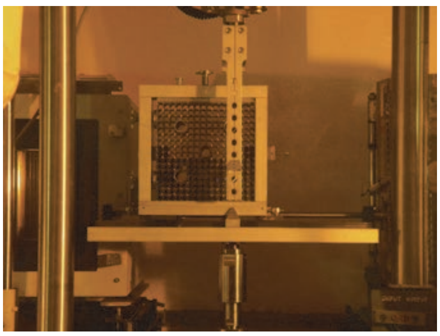


Fig.2. Test fixture of a cell spring test

The purpose of a static buckling test is to obtain a load versus displacement curve of a spacer grid. From this test, the static buckling load and the stiffness of spacer grid are determined. The response of a static buckling after irradiation is examined by a buckling test. Fig. 3 is the conceptual drawing of a static buckling test. The spacer grid is installed on the lower grip, and the axial compressive load is applied to the upper surface of the spacer grid until the static buckling occurs. The upper grip is designed so that the applied load is transferred uniformly to the spacer grid by using a spherical bearing, and the LM guide is installed to allow a lateral deformation of the spacer grid during the test. Fig. 4 is the test fixture of a static buckling test of a spacer grid.

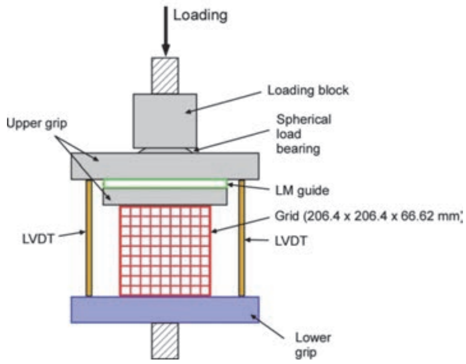


Fig.3. Conceptual drawing of a static buckling test

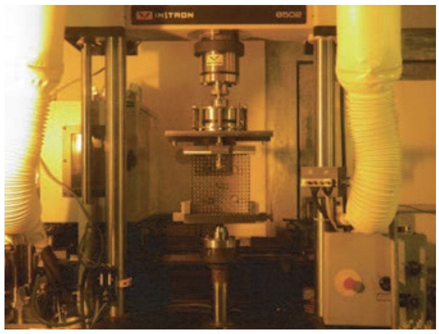


Fig.4. Test fixture of a static buckling test

3. TOP NOZZLE

The top nozzle guides the control rods and applies a hold down force to fuel assembly in a reactor core. A hold down spring test technique has been developed for two types of top nozzle; coil spring and leaf spring types. The purpose of a hold down spring test is to obtain a load versus displacement curve of a hold down spring. The changes of the maximum spring force after irradiation is estimated. Fig. 5 is the conceptual drawing of a hold down spring test. The top nozzle is installed on the lower grip, and a compressive load is applied to the hold down spring by a upper grip until the displacement reaches to the specified displacement. The test fixture of a hold down spring test is shown in Fig. 6.

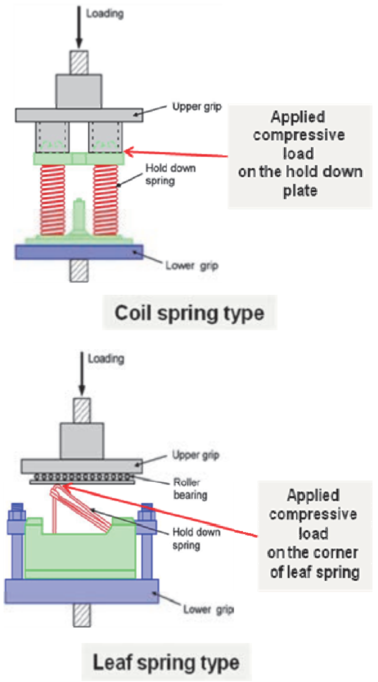


Fig.5. Conceptual drawing of a hold down spring test

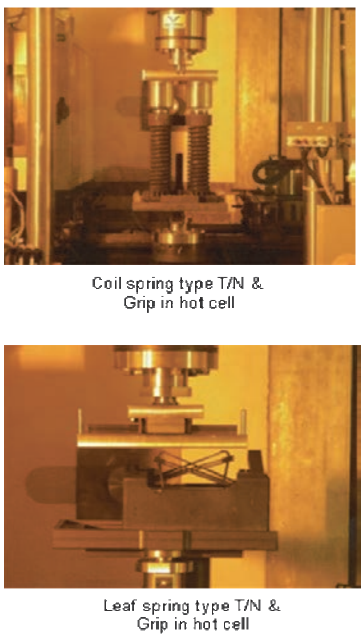


Fig.6. Test fixture of a hold down spring test

4. PERFORMANCE TEST

We have done the performance tests of the test fixtures developed in this study.

4.1 Cell Spring Test

Cell spring test was carried out on the unirradiated and irradiated spacer grids by INSTRON 8562 (+/- 0.1 kN) test machine. Before a test, the compressive load (= -5 N) is applied initially. Test was performed at a constant loading rate, 0.5 mm/min, and the compressive load was applied to the specified displacement (= 0.4 mm) at room temperature. Dummy claddings (Fig. 7) are inserted into cells around the test cell.

Fig.7. Dummy cladding for a cell spring test

The test cells are indicated in Fig. 8. Fig. 9 shows the typical load versus deflection curve of an irradiated spacer grid. In this figure, the deflection values are converted from the original values by considering the compliance of a test machine, grip and loading rod.

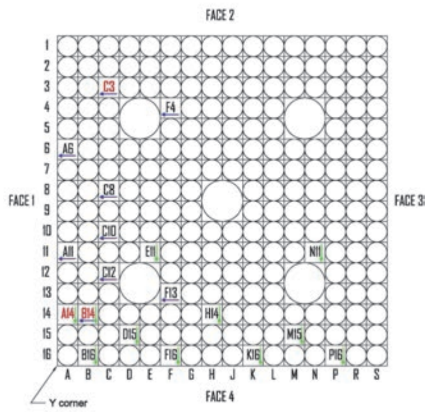


Fig.8. Location of test cells

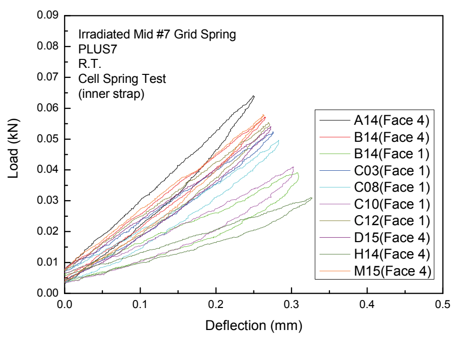


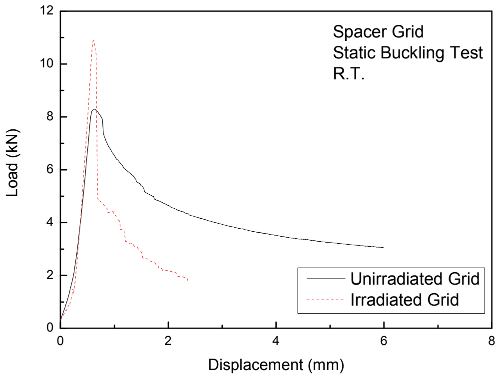
Fig.9. Typical test results of a cell spring test

4.2 Static Buckling Test

Static buckling test was carried out on the unirradiated and irradiated spacer grids by INSTRON 8502 (+/- 50 kN) test machine. Before a test, the compressive load (= -0.5 kN) is applied initially. Test was performed at a constant loading rate, 0.5 mm/min, and the compressive load was applied until a buckling occurs at room temperature. Dummy claddings are not inserted into cells.

Fig. 10 shows the typical load versus displacement curve of an unirradiated and irradiated spacer grid. In this figure, the maximum buckling load is increased due to an irradiation, but the stiffness of irradiated grid is similar to that of unirradiated grid.

Fig.10. Typical test results of a static buckling test



4.3 Hold Down Spring Test of Top Nozzle

Hold down spring test of a top nozzle was carried out on the unirradiated and irradiated top nozzles by INSTRON 8502 (+/- 50 kN) test machine. Before a test, the compressive load (= -0.5 kN for a coil spring type, = -0.1 kN for a leaf spring type) is applied initially. Test was performed at a constant loading rate, 0.5 mm/min, and the compressive load was applied the specified displacement (= 32 mm, less than the solid height of a spring) at room temperature. Fig. 11 and Fig. 12 show the typical load versus displacement curve of an coil spring and leaf spring type top nozzle respectively. In this figure, the maximum spring force is increased due to an irradiation, but the stiffness of irradiated top nozzle is similar to that of unirradiated one.

5. CONCLUSIONS

To establish the PWR spent fuel assembly performance test system, we have developed the mechanical test techniques for the several components of fuel assembly. The following conclusions are obtained. The mechanical test equipments were developed for the cell spring and static buckling test of spacer grid, and for the hold down spring test of top nozzle. Those test techniques were verified by testing a uniradiated and irradiated structural component specimens. The test equipments produce the performance evaluation data for the irradiated components of spent fuel assembly and the new structural components that might be developed in the future. The mechanical test techniques of the irradiated structural components developed in this paper will be utilized not only for developing our own fuel assembly but also for analyzing the characteristics of proven components.

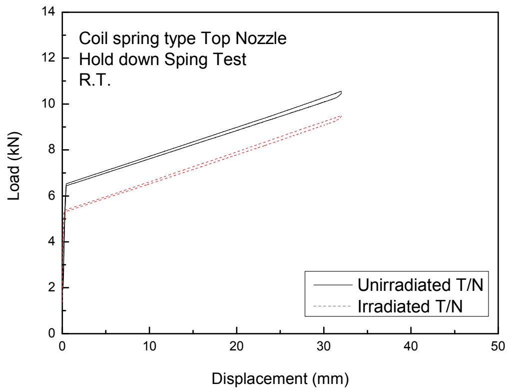


Fig.11. Typical test results of a hold down spring test for the coil spring type top nozzle

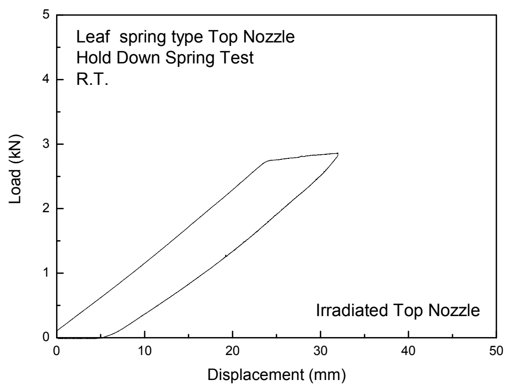


Fig.12. Typical test results of a hold down spring test for the leaf spring type top nozzle

6.3 Measurement Techniques of Magnetic Properties for Evaluation of Neutron Irradiation Damage on Austenitic Stainless Steels

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The remote-controlled equipment for measurement of magnetic flux density has been developed in order to evaluate the irradiation damage of austenitic stainless steels. Magnetic flux densities by neutron irradiation in austenitic stainless steels, SUS304 and Fast Breeder Reactor grade type 316 (316FR), have been measured by the equipment. The results show that irradiation damage affected to magnetic flux density, and indicate the measuring method of magnetic flux density using a small magnetizer with a permanent magnet of 2 mm in diameter is less affected by specimen shape.

Keywords: nondestructive evaluation, neutron irradiation, magnetic property, austenitic stainless steels, magnetic flux density

1. INTRODUCTION

Although the radiation environments which reactor vessels and core internals in nuclear power plants are exposed are relatively mild, the exposure durations are long and these structures are difficult to be replaced. Therefore, proper management of neutron irradiation damage on structural materials is important to ensure the reliability of long-life nuclear power plants. Development of a nondestructive evaluation method of neutron irradiation damage will contribute to it. Irradiation dose is known to be related to irradiation hardening [1]. Several studies reported that ferrite phase was induced in austenitic stainless steels, which are typical structural materials of nuclear power plants, by neutron irradiation and also magnetization in materials increased with fast neutron fluence, although the mechanism of ferrite formation has not been completely revealed [2-5]. However, the measurement methods employed in most of studies are limited the maximum sample size, which cannot be applied to nondestructive evaluation methods for plants in operation. Measuring magnetic flux density by a Flux Gate (FG) sensor which does not have limit sample size is suitable for nuclear power plants.

On the other hand, magnetic flux densities depend on sample shapes. Therefore, measured values are required to correct based on magnetostatic field analysis. In this study, the measuring technique of magnetic flux density which is less affected by sample shape is introduced, and results measured are reported.

2. EXPERIMENTS

2.1 Materials

Two types of austenitic stainless steels, SUS304 and Fast Breeder Reactor grade type 316 (316FR) were investigated in this study and the chemical compositions are listed in Table 1. SUS304 and 316FR were solution-annealed by heat treatments at 1100 ºC for 36 min, and at 1050 ºC for 30 min followed by water quench, respectively. Samples were irradiated in the Fast Reactor Joyo. The ranges of dose and temperature of the neutron irradiation in this study are calculated to be 0.4-18.4 dpa, 430-560 ºC, respectively.

Specimens were picked up from two positions of irradiated tensile samples, which were grip and gauge section (See Fig.1). Specimens were mounted by epoxy resin and mechanical polishing was conducted. The surface finish of the measuring planes was achieved by polishing with diamond paste with the grain diameter of about 1 μm.

Table 1 Chemical composition of samples (wt%)

Alloy	C	Si	Mn	P	S	Cu
SUS304	0.05	0.60	0.87	0.026	0.002	0.09
316FR	0.01	0.59	0.84	0.026	0.003	0.26
Alloy	**Ni**	**Cr**	**Mo**	**V**	**N**	**Fe**
SUS304	8.94	18.59	-	0.05	0.022	Bal.
316FR	11.19	16.87	2.23	0.08	0.08	Bal.

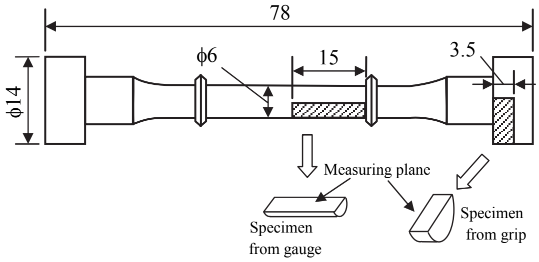


Fig.1. Sample dimensions

2.2 Measurement of Magnetic Flux Density

The equipment for measurement of magnetic flux density was developed to evaluate magnetic properties of irradiated materials. It consisted of two separated units, remote-controlled measurement apparatus in a hot cell (See Fig.2) and controller unit outside hot cell. The measurement apparatus contained the FG sensor measuring magnetic field and a motorized sample stage with sample holder. Measurement of magnetic flux density was performed scanning of the sample surface by moving the sample relatively to the sensor, and the maximum absolute value in the magnetized area was adopted as the measurement result.

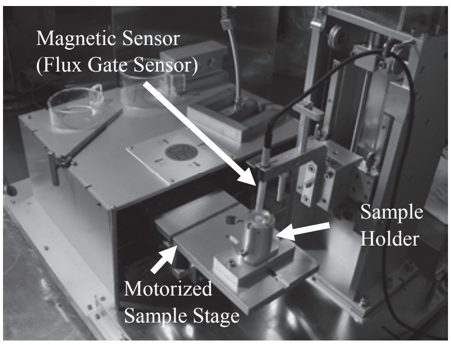
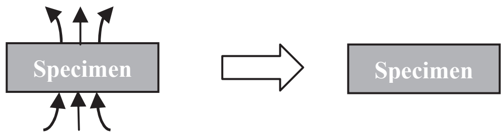


Fig.2. Remote-controlled apparatus for measurement of magnetic flux density.

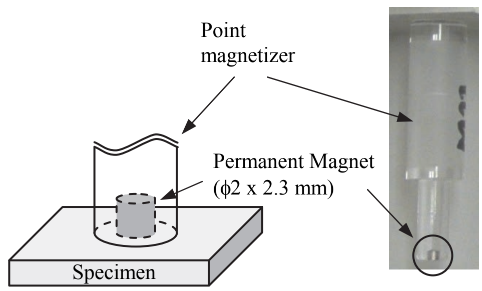
The schematic view of measuring procedure of magnetic flux density is shown in Fig.3. Because specimens have a certain level of residual magnetization, they need to be demagnetized by an alternating magnetic field before measurements (Fig.3 (a)). After verifying that residual magnetization was below the detection limit of the sensor, the magnetic field was applied to specimen by a permanent magnet. In the previous study, magnetization was conducted with a pair of large magnets, which formed a wide range of magnetic field to magnetize the whole specimen [6]. The magnetic flux density of specimen was affected to its shape and required correction of measurement value. In order to reduce influence of specimen shape on magnetic flux density, the point magnetizer with a small permanent magnet was invented (Fig.3 (b)). The magnetic field at the center of magnet is about $0.28/\mu_0$ A/m, where μ_0 is the magnetic permeability in a vacuum. The parallel component of magnetic flux density to the magnetizing direction was measured at the center of the sample using the FG sensor at room temperature. The distance between the sensor and the sample surface was called "lift-off" and it was about 1 mm in normal measurement (Fig.3 (c)).

The magnetic flux density resolution of the FG sensor employed was about 1 μT. The FG sensors have a simple composition suitable for neutron irradiation environments. The magnetic flux density resolution of the FG sensor is worse than that of a Superconducting Quantum Interference Device (SQUID) sensor, but better than that of Hall sensors. In addition, The FG sensors do not need to be cooled at 77 K unlike the SQUID sensor. It enables easier measurement and positioning of sensors close to measurement objects.

(a) Erase residual magnetization



(b) Magnetize specimen by point magnetizer



(c) Measure magnetic flux density by FG sensor

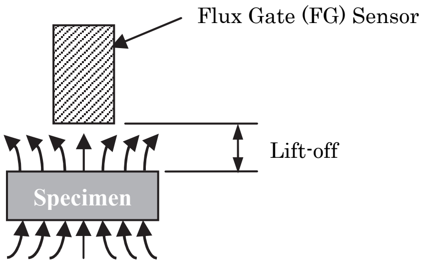


Fig.3. Schematic view of measuring magnetic flux density.

3. RESULTS AND DISCUSSIONS

The measurement results of magnetic flux density of neutron irradiated SUS304 and 316FR are shown in Fig.4. In both types of steels, magnetic flux densities tended to increase with increasing irradiation dose. However, variations of magnetic flux density caused by irradiation were different between SUS 304 and 316FR. Although further experiments were needed to clarify the reason why, microstructure change due to displacement damage or thermal aging during irradiation could be affected.

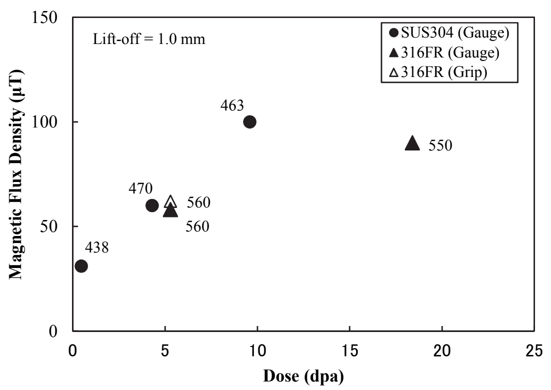


Fig.4. Relationship between irradiation dose and magnetic flux density for SUS304 and 316FR.
Values in the plot mean irradiation temperatures (ºC).

In Fig.4, measurement values of the gauge section and the grip section specimens could be compared to 316FR at around 5 dpa. This data and other measurement values in SUS304 with the 3mm lift-off for the reference were shown in Fig.5. In both the steels, no significant difference was seen in magnetic flux density between gauge and grip section. The results indicated the measuring method in this study was less affected by specimen shape used in this study.

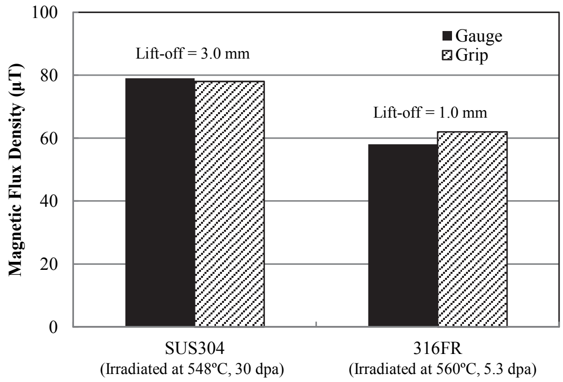


Fig.5. Comparing magnetic flux density between two sample shapes.

4. CONCLUSIONS

In the present work, a new magnetized technique with a small permanent magnet has been developed in order to reduce influences of specimen shapes on magnetic flux density. Measurement results of magnetic flux densities have indicated that irradiation damage of austenitic stainless steels has correlation with magnetic property. However, the cause of magnetic property change on irradiated steels has not been revealed. Although the data obtained in this study is preliminary, but would be important, the future work should be conducted with focusing on microstructures related magnetic phases.

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6.4 Development of X-ray CT Scanner System

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To contribute to the aging countermeasure of LWR and other science researches, three-dimensional X-ray system was installed in the hot laboratory of Japan Materials Testing Reactor.

The equipment was installed within concrete No.3 cell in FY 2009, and put into service from FY 2011. With this system, Gamma-ray Offset Scanning Technique (GOST) was developed to reduce the influence of Gamma-rays that emitted from radioactive specimen itself to the CT image.. Therefore, clear image was obtained for the irradiated test sample with high radio activity.

Keywords: JMTR, hot laboratory, post irradiation examination, non-destructive inspection, X-ray inspection, X-ray CT.

1. INTRODUCTION

The X-ray inspection is a non-destructive inspection method that makes it possible to examine crack and structure inside specimen without breaking specimen. It is used in various fields such as industry, medical, and research, etc. In the Japan Atomic Energy Agency (JAEA), the X-ray inspection is used for integrity evaluation of the spent nuclear fuels, irradiated specimens and the instrumentation devices of internal capsule used in a reactor. In the Hot Laboratory of the Japan Materials Testing Reactor (JMTR-HL), the X-ray inspection system of transmission type had installed in a hot cell, and had been used for integrity evaluation of fuels that are used for the irradiation examination in JMTR. However, JMTR-HL is requested high-value technical data that contribute to integrity evaluation for the aging countermeasure of light water reactor and other science researches. [1]

Therefore, JMTR-HL installed the X-ray inspection system of Computerized Tomography type (X-ray CT system) in a hot cell. The CT method is possible to obtain the internal information of inspection object in addition to the positional information. Furthermore, the development to reduce effect of Gamma-ray that irradiated from radioactive specimen itself was conducted to obtain high resolution image. [2]

In this paper, the outline of JMTR-HL and X-ray CT system is explained first and then the development points are explained. Finally the results of performance test is explained.

2. OUTLINE OF JMTR-HL

JMTR-HL started operation since 1971 as a facility that carries out the Post Irradiation Examination (PIE) of the irradiated specimen in JMTR and other facilities.

JMTR-HL has eight concrete cells, four microscope lead cells, seven lead cells and five steel cells, and are used for PIEs of strength test, metallography, X-ray inspection, Gamma-ray scanning, etc. And furthermore, JMTR-HL is connected with JMTR by water canal, and it is possible to transport capsules with fuels and materials between the hot cell and the reactor through the canal.

Figure 1 shows outline of JMTR-HL. X-ray CT system is installed in the third concrete cell of JMTR-HL. The third concrete cell is made with heavy concrete, and is possible to handle fuels of maximum 3.7 pBq. [3]

3. X-RAY CT SCANNER SYSTEM OF JMTR-HL

Figure 2 shows principal of X-ray CT. The X-ray tube emits cone shaped radiation beam. Its distribution of intensity is measured by a detector that is composed of scintillator, CCD-array, data processing and host-PC. This distribution is named sinogram. The radiation intensity is reduced depending on density and thickness of the inspection object in front of the detector. CT image is obtained by restructuring of sinogram.

The X-ray CT system installed in JMTR-HL is composed of X-ray system, detector system and manipulator system. Figure 3 shows the X-ray CT system installed in the hot cell. The specifications of these systems are as follows;

1) X-ray system
-Target material : W (Tungsten)
-Usable tube voltage : 20 - 450 kV
-Max. tube current : 1.55 mA (at 450 kV)
2) Detector system
-Detector type : Line detector array (LDA)
-Scintillator crystal : $CdWO_4$
-Pixel size : 0.254 mm
-Number of Pixel : 1984 pixel
-Effective detector length : approx. 504 mm
3) Manipulator system
-Max. movement : 1000 mm (Vertical)
-Min. movement : 0.1 mm (Vertical)
-Min. rotation angle : 0.025 degree

4. ATTEMPT TO ACQUIRE HIGH RESOLUTION CT IMAGES

This system of a collimator and a restructuring program has been developed to acquire high resolution CT image.

4.1 Collimator

This collimator is installed in front of the scintillator for fining of the inspection area by narrowing X-ray. Figure 4 shows schematic drawing of the collimator.

This collimator is made of tungsten of 40mm in thickness, and it has a adjustable slit in width between 0.1 to 3 mm.

This collimator is possible to narrow X-ray to minimum 0.1 mm.

4.2 Restructure Program

This program is made to reduce the effects of Gamma-ray that emitted from radioactive specimen itself. The name of this program is GOST (Gamma-ray Offset Scanning Technique).

The influence factors on the X-ray scan inspection caused by the Gamma-ray of specimen itself are as follows;

-Energy of the Gamma-ray.
-Intensity of the Gamma-ray.
-Distance between object and detector.
-Location of the Gamma-ray source in the scan object.
-Rotational-symmetry distribution of the source relating to the rotation axis.
-Distribution of the sources along the longitudinal axis.
-Irradiation angles of the source aside the scan height.

This program is possible to reduce the effects of gamma-ray by the follow methods.

-Analysis of the gamma radiation area and obtain the offset information.
-Computed tomography scan with X-rays and gamma-rays.
-Suppress of the gamma-rays from the scan data.
-Reconstruction of Scan data.

5. PERFORMANCE TEST

5.1 Spatial Resolution Test

This test was carries out for the evaluation of spatial resolution of X-ray CT system. Figure 5 shows outline of specimen and result of spatial resolution test. This test is performed to confirm resolution performance of tomography using platinum double wire which is used as center line of thermo couple. X-ray CT measurements are performed to platinum double wire specimens (Diameter: 0.05 to 0.8 mm, Distance between wires: 0.05 to 0.8 mm) and it is confirmed that spatial resolution is recognized up to 0.16 mm.

5.2 Transmissivity Test

This test was carries out for the evaluation of transmissivity with X-ray CT system. Figure 6 shows outline and result of the transmissivity test. The shielding object of stainless steel of 100mm in thickness was set between the X-ray tube and the inspection object for this test.

This system is confirmed to have a transmissivity of at least 100 mm at stainless steel.

5.3 Synthesis Test

This test is synthesis test that used the irradiated fuel rod to evaluate the capacity of developed restructuring program. Figure 7 shows outline of specimen and result. The specimen is irradiated fuel rod that was used in commercial reactor and then shorten, and used for ramping test in JMTR. The burn up of the fuel rod is 25GWd/t and output of that ramping test is 400W/cm. The pellet in this irradiated fuel rod is inserted with thermo couple.

This system observed thermo couple that was inserted in pellet and cracks of the pellet's surface. Furthermore, this result shows good agreement with the result of metallography. It is confirmed that this non-destructive testing is equally evaluatable as destructive testing.

6. CONCLUSION

X-ray CT system has installed into the hot cell in JMTR-HL. Results of performance tests are as follow,

1) Spatial resolution is 0.16 mm and transmissivity is more than 100 mm for stainless steel.
2) By developments of collimator and innovative restructuring program, it is possible to eliminate influence of gamma-rays.

From the above results, it is possible to perform X-ray inspections of high burn-up fuels those are planned to perform after restart of JMTR.

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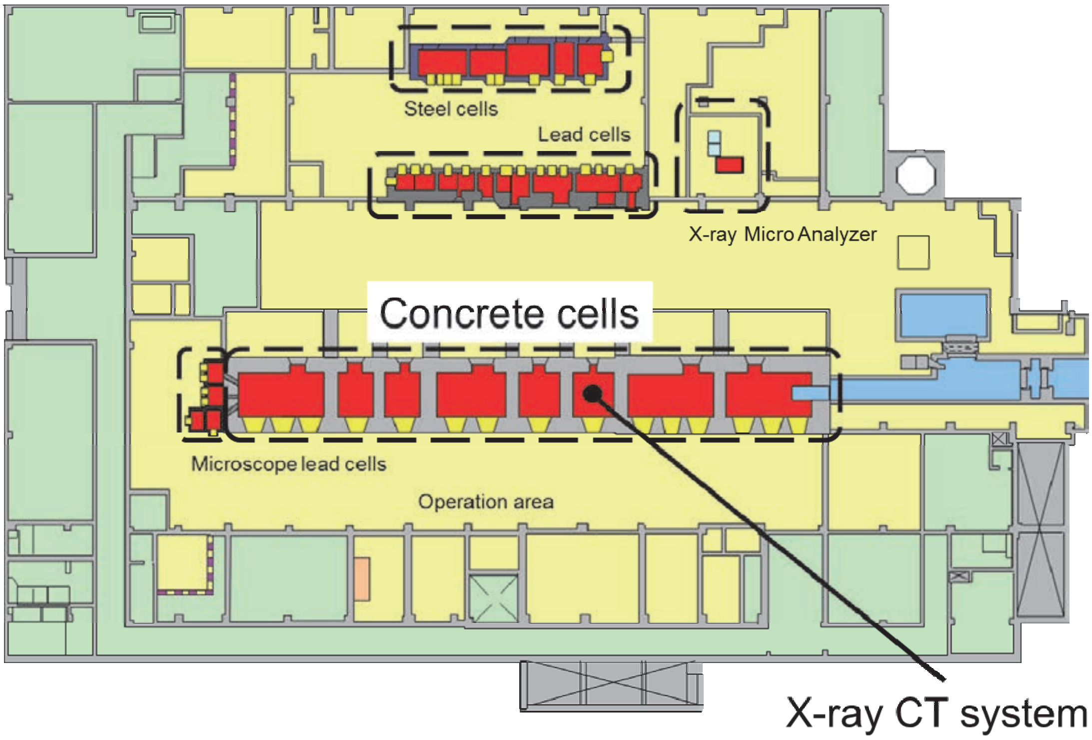
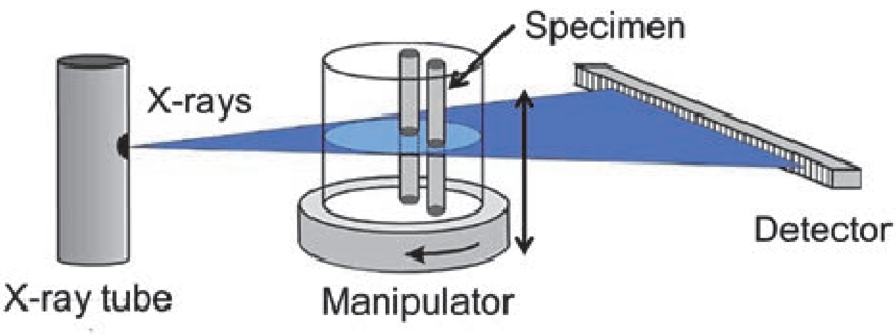
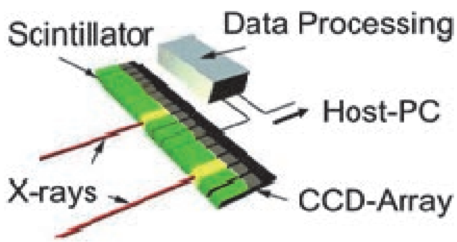


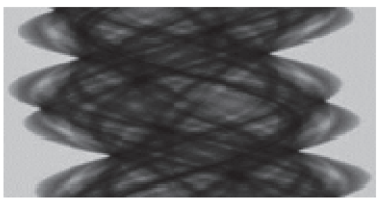
Fig.1 Plans of JMTR-HL.



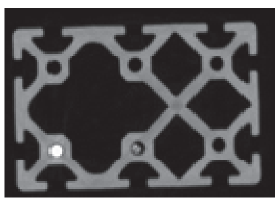
(a) Schematic drawing of X-ray CT



(b) Principle of a Line Detector



(c) Sinogram



(d) CT image

Fig.2 Principal of X-ray CT.

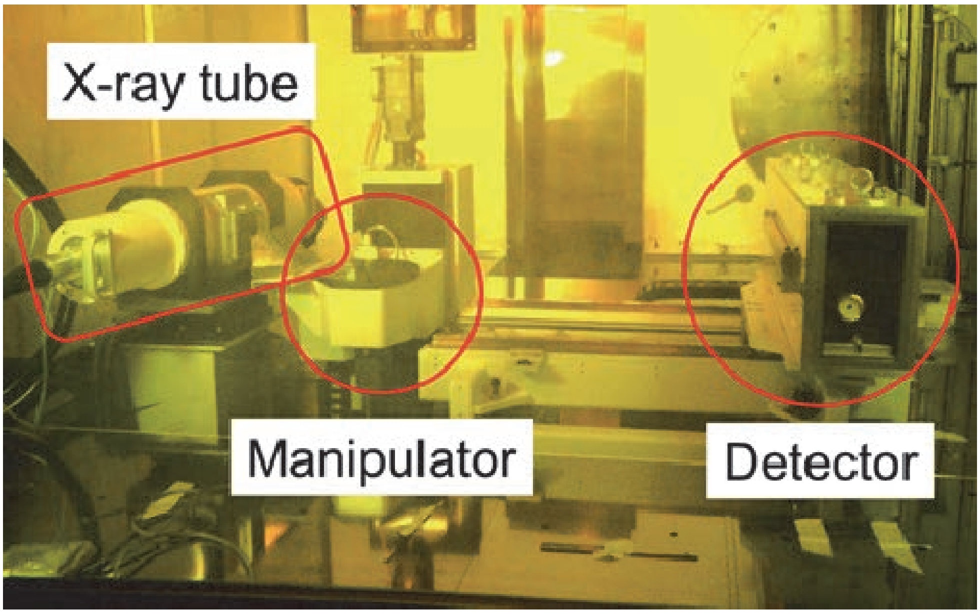


Fig.3 The X-ray scanner system in hot cell.

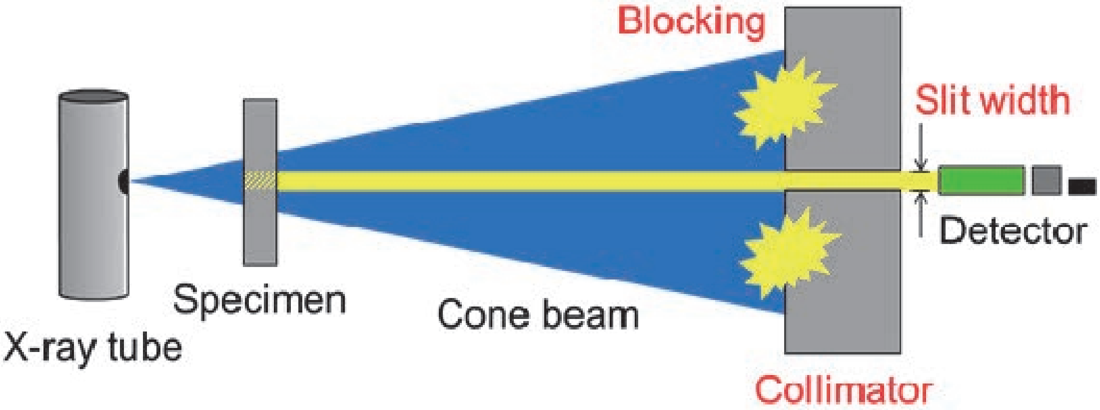


Fig.4 Schematic drawing of the developed

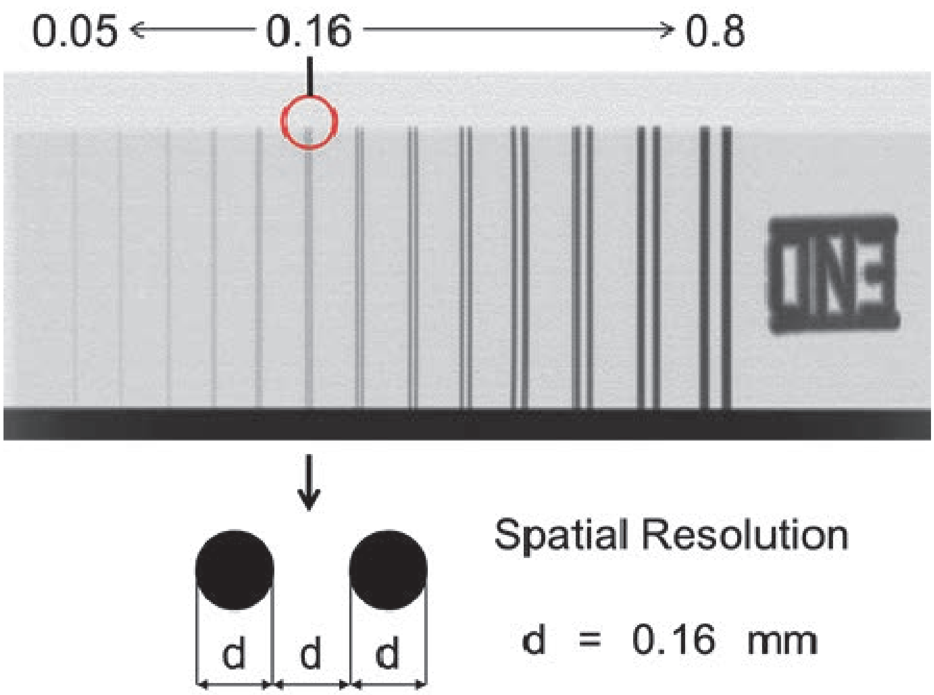
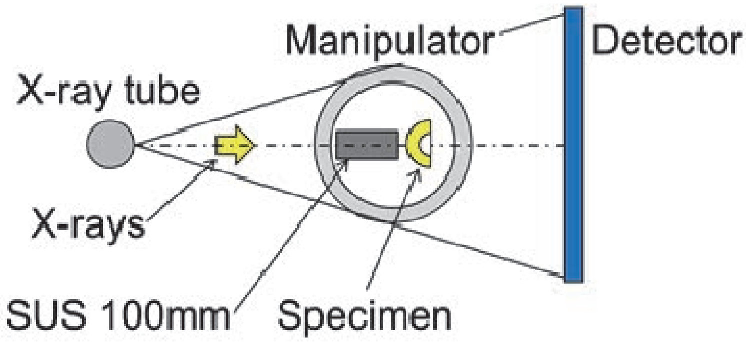
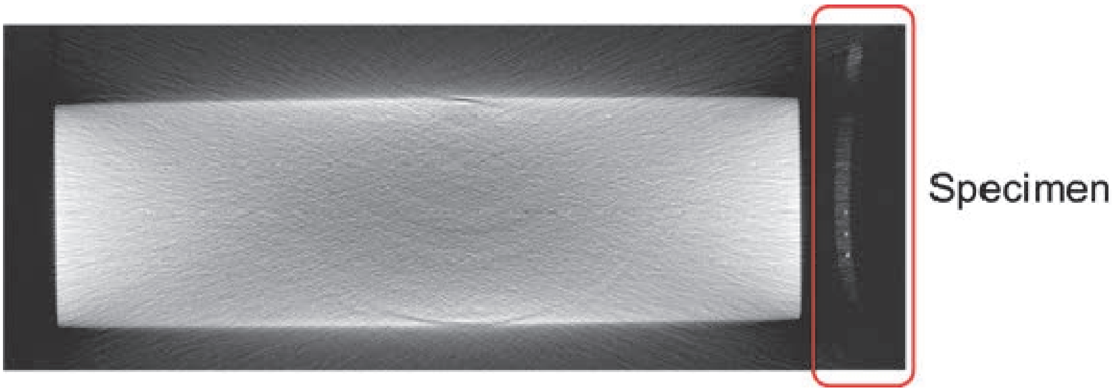


Fig.5 Spatial resolution test

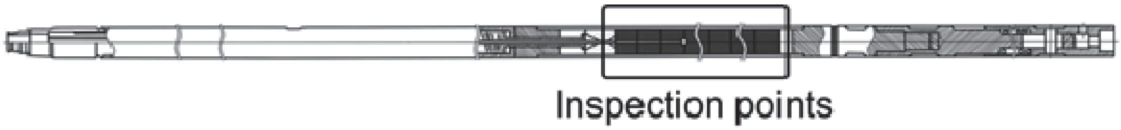


(a) Schematic of transmissivity test

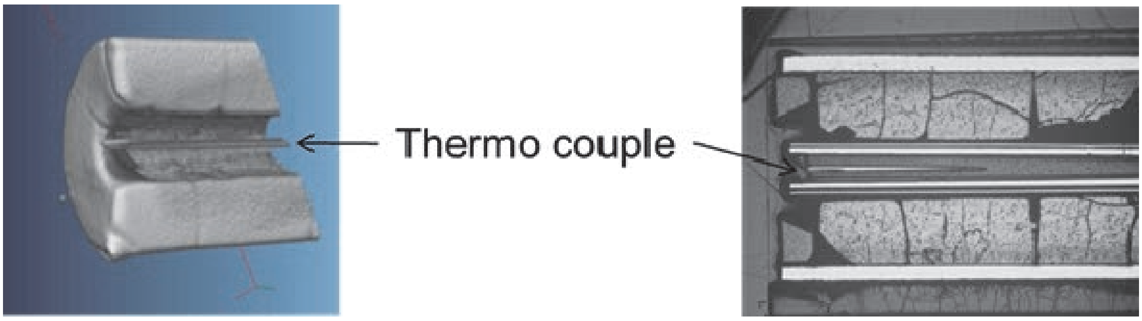


(b) the results

Fig.6 Transmissivity test.



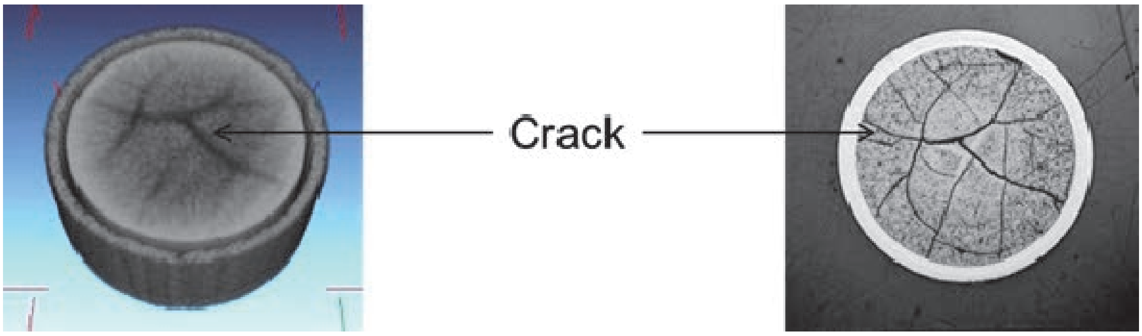
(a) Schematic drawing of the irradiated fuel rod



3D X-ray CT image

Metallography

(b) State of thermo couple



3D X-ray CT image

Metallography

(c) Crack of the pellet surface

Fig.7 Synthesis test.

6.5 Analysis of CRUD flake by shielded EPMA

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Crud specimens, which were a scraped from twice-burned fuel cladding in Korean PWRs, were intensively analyzed using Shielded-EPMA. Also, we changed the power conditions of this study. Even though the general power application conditions for an EPMA analysis is about 20 kV and 10 nA, the power condition applied to this analysis was 20 kV and 1200 nA, which was applied for 5 to 30 seconds, and created by opening an adjustable aperture device for gun alignment adjustment. As a result, we found that part of the crud was evaporated, and the main metal composition material, such as Iron, remained.

Keywords: Crud flakes, Shielded-EPMA, Beam current change, Boling chimney hole, Crud thickness.

1. INTRODUCTION

This Unidentified materials, known as CRUD (Chalk River Unidentified Deposit), which are activated corrosion products from BWR and PWR reactors, are primarily deposited on the outer surfaces of fuel rods [1]. These deposits are distinguishable from the oxide corrosion products dissolved from structural materials and piping which are transported into the core by the primary coolant. The fuel for the PWR core power can change the distribution of a study while in progress due to crud deposition [2, 3].

Chemical analysis of crud with ICP-MS, including ICP-AES, is a well-known analysis tool, where as other techniques like SEM and TEM can give information on the morphology and local composition of crud materials [4,5]. However, the analysis of crud, using shielded EPMA, can produce specific results.

This study summarizes the exam result of crud scraped in a Korean PWR. The microstructure and composition of twice-burned crud samples were characterized by EPMA to determine the chemical form and oxidation state of the deposit components.

2. SAMPLES

Crud specimens, which a scraped from the twice-burned fuel cladding in Korean PWRs, were transported to an Irradiated Materials Examination Facility (KAERI, IMEF) for analysis. The observation results of the surface of each specimen were determined using 350-times magnification on an HIROX device. The results determined by HIROX were used as a basic date to analyze the use of devices such as EPMA and SEM. To prepare the EPMA and SEM specimens, the area found by HIROX was cut by scissors, attached to carbon tape with good adhesion, and deposited using a carbon evaporator.

3. RESULTS AND DISCUSSION

3.1. Cruds Flake Analysis using SEM

Crud flakes were scraped from the twice-burned fuel cladding in a Korean PWR. The specimen was prepared by cutting a filter with scissors after identifying each area of the filter by HIROX to identify the presence of the specimen.

As shown in Fig. 1. Crud was shaped as W/L/t

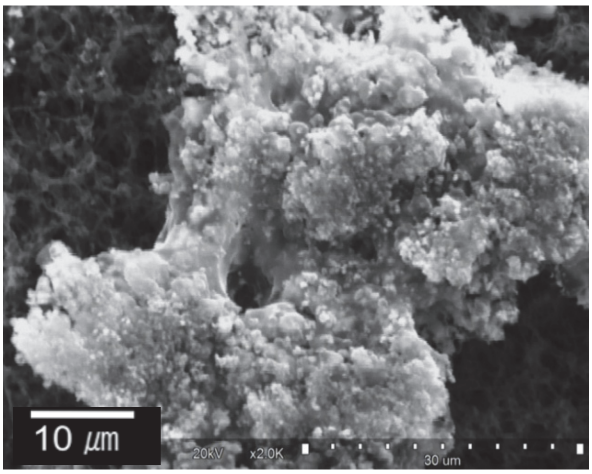

Fig.1. SEM of a crud flake.
- Boling chimney hole size: ≒ 6 ㎛
- Crud thickness: ≒ 12 ㎛.

≒ 50/120/12 ㎛, and the boiling chimney hole size was observed to be: ≒ 6 ㎛. As for the surface shape of the area contacting the coolant, crud materials dissolved in the coolant were shown to be deposited in precipitated form. Precipitation growth is made as very small particles gradually grow, rather than being deposited as a big flake. Other crud flakes are considered to be crud in areas that have contacted the cladding as the surface is flat as it contacts the cladding, and the boiling chimney hole is not clear.

3.2. Crud Flake before and after Burning

Even though the general power application conditions for EPMA analysis are about 20 kV and 10 nA, the power conditions applied to this analysis were 20 kV and 1200 nA, which were applied for 5 seconds, and created by opening an adjustable aperture device for gun alignment adjustment. This 1200 nA was the value shown on the screen, and the actually applied current was estimated to be 1,500 - 2,000 nA. It was confirmed that the crud flake disappears, shown in figure 2.

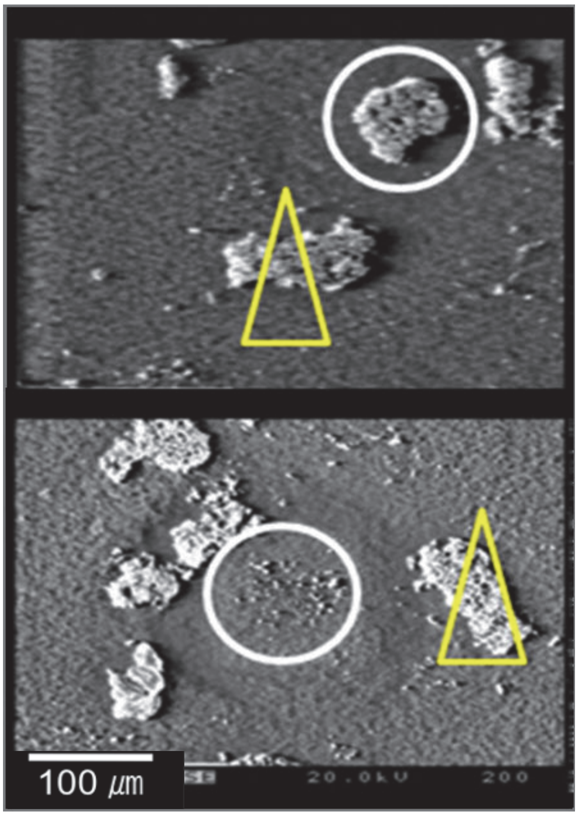

Fig.2. BSE of crud before and after burning.
- Beam conditions: 20 kV, 1200 nA,
- Beam exposure time: 5 sec

During charging, heat is created from the interaction when a large amount of electrons temporarily collides with a specimen with poor conductivity. It is considered that part of the crud is evaporated and the main metal composition such as iron remains.

3.2.1 After Burn Precipitate No.1, 2, 3, 4

No.1 shows a typical nickel ferrate like the composition shown in table1. The WDS analysis is obtained by quantitatively analyzing at 3,000, 5,000 and 10,000-times magnification, which cover the entire specimen No. 1, and then averaging the values. In particular, Iron shows a higher composition in the lower part of the specimen. Oxygen shows a similar concentration distribution as iron.

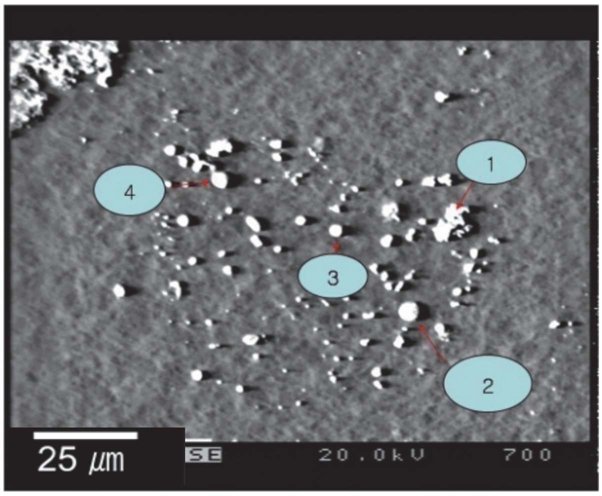

Fig.3. BSE of crud flake after burning.
- Beam conditions: 20 kV, 1200 nA,
- Beam exposure time: 5 sec

Table1. Quantitative analyses of crud flake after burning

	Fe	Ni	Zn	Cr	O
No 1					
wt %	30.36	16.52	0.37	0.84	17.74
At %	27.80	14.39	0.21	0.83	56.69
No 2					
wt %	28.39	20.89	0.42	1.66	0.00
At %	56.33	39.43	0.70	3.54	0.00
No 3					
wt %	25.54	20.10	0.43	0.70	0.38
At %	54.20	40.58	0.78	1.61	2.82
No 4					
wt %	19.04	18.28	0.39	1.62	5.73
At %	32.56	29.73	0.57	2.97	34.18

According to M. Haginuma, The formation of nickel ferrite, which is known to be the main composition of fuel crud, is an important subject in relation to reactor coolant chemistry, which is also formed by an exchange reaction between Ni^{2+} in the coolant and Fe^{2+} in the Fe_3O_4 lattice [6].

No.2 has a ratio of Ni/Fe=40/56 and there is zero oxygen. The Zirconium mass shown in the lower part of the picture is considered to be either scraped off the surface of the cladding or precipitated from being present as colloid in the coolant. Therefore, it is considered that this is not ZrO_2, which has been present as oxide on the surface of cladding, but is zirconium colloid dissolved in coolant. Zirconium is present both as a precipitated condition and independently since it does not have metal affinity with nickel or iron.

No.3 shows the same characteristics as No. 2. In terms of iron and nickel traces, the same distribution curve is shown at the same location under exactly the same analysis conditions. This means that this is a full employment condition. The melting temperature at this point is considered to be around over 1,000℃, however, to make a more accurate analysis, it is necessary to collect the specimen in a new way, or to find another appropriate action using devices such as a TGA.

No.4 has a ratio of Fe/Ni/O=1/1/1. It has an oxide condition unlike the other masses. That is, when it comes to the distribution of composition, shown in Fig.1, of the same flake, it was determined that it has various compositions depending on the location, or it is composed or oxygen or other materials. Here, the material that has been dissolved as a particulate or ion phase in the coolant is present in the crud with an exchange reaction between Ni^{2+} in the coolant and Fe^{2+} in the Fe_3O_4 lattice = Ni/Fe material and in the exchange reaction between Ni^{2+} in the coolant and $Fe_2[OH]_3$ in the $Fe_3O_4 + 3H_2O$ = NiFeO. More research is required to determine the composition and distribution of such materials. When an appropriate method of specimen collection and research is developed, it would be useful data like the above analysis data in identifying the basic component element of crud.

3.2.2. Crud Flake 2nd Burn & 3rd Burn

EPMA beam condition were changed from 20 kV and 1200 nA for 5 seconds to 30 seconds under the same conditions. However, it was determined that the phenomenon after the power 5 seconds application only partially disappears. Like the analysis results already obtained under the same conditions, it is considered that the compositions of the material disappeared at this time and that the remaining small masses are the same as the previous analysis result.

The power was projected to the same location under the same application conditions for 30 seconds. As seen in the figure 4, crud flakes disappeared, In addition, the figure shows damage on the paper filter, unlike when it was projected once. Damage like this seems like to be made when the crud flake material is ripped or melted down while it is forming another

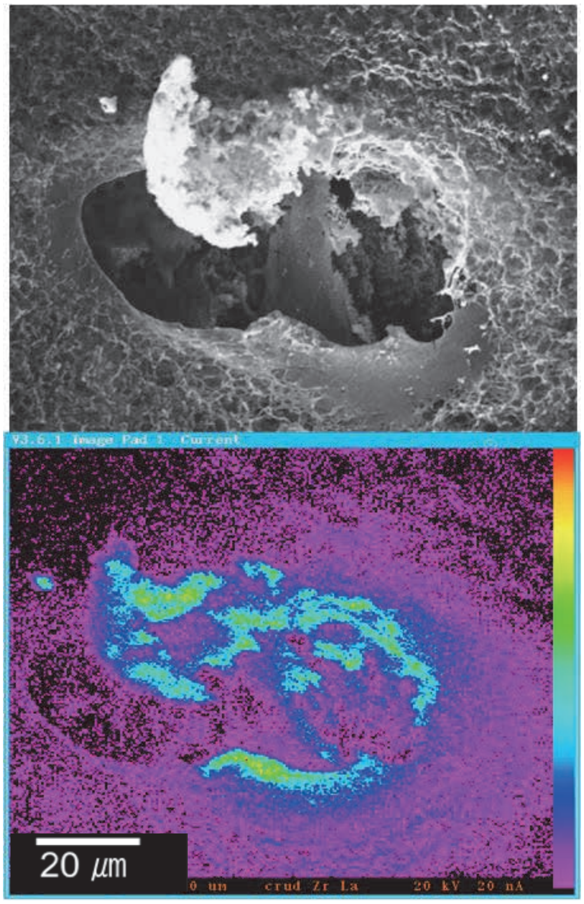

Fig.4. BSE and X-ray map of crud flake after burning.
- Beam conditions: 20 kV, 1200 nA,
- 2nd beam exposure time: 5 sec
- 3nd beam exposure time: 30 sec

Shape, as shown in the figure 4. However, the paper itself was not burned in a high vacuum environment.

As shown in figure 4, while the composition of zirconium can be clearly seen, it was determined that oxygen is not present. This conforms to the results that zirconium was not present as an oxide when the primary burn was performed.

As the melting temperature of zirconium is higher than that of iron and nickel, no change was observed when the power was applied for 5 seconds under the same conditions, and a change was then observed when it reached a higher temperature. Even though the power was projected unilaterally for 30 seconds, if the crud material can be divided based on temperature using TGI and the like, research on the essence of crud will be possible.

It is unique that Zirconium also has the shape of general crud. Generally, it is dominantly thought that zirconium will not be dissolved by coolant since it is a stable material; however, zirconium is not only present as a crud flake form, but it is also present as a particulate, not as an oxide.

Also, it was determined that the crud was not extracted from the oxide layer of the cladding during crud collection. If there is ZrO_2 on the oxide layer of the cladding, it would not have a crud flake form, and the composition of oxygen would have been clearly identified.

In conclusion, it is determined that the composition of dissolved material present in coolant is made with various materials including Zirconium.

4. CONCLUSIONS

A detailed analysis was performed for crud flakes scraped from twice-burned fuel cladding in Korean PWRs by EPMA. The crud was shaped as W/L/t ≒ 50/120/12㎛ and the crud flake thickness was observed as≒4~25㎛. We also observed the boiling chimney hole size as ≒4~6 ㎛.

We changed the power conditions for this study. Even though the general power application conditions for EPMA analysis are about 20 kV and 10㎁, the power conditions applied to this analysis were 20 kV and 1200㎁ for 5 to 30 seconds, created by opening an adjustable aperture device for gun alignment adjustment. As a result, we found that part of the crud was evaporated and the main metal composition, such as iron, remains.

The power was projected to the same location under the same application conditions for 30 seconds. It showed damage on the paper filter, unlike when it was projected for 5 seconds. Damage like this seems like to be made when the crud flake material is ripped or melted down while receiving a high current beam. Nickel ferrites formed by an exchange reaction between Ni^{2+} in the coolant and Fe^{2+} in the Fe_3O_4 lattice. The results of the after burning x-ray map, it was well agree with M. Haginuma's paper, which we confirmed.

Also, we found Zirconium crud flake. it was determined that the crud was not extracted from the cladding oxide layer during crud collection. If it is ZrO_2 on the oxide layer of the cladding, it would not have a crud flake form, and the composition of oxygen would have been clearly identified.

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6.6 Improvement of center boring device for irradiated fuel pellets

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The power ramp test will be performed at JMTR in Oarai R&D Center to study the safety margin of high bumup fuels. The commercial fuel rods irradiated in Europe (approx. 70 GWd/t) will be refabricated as the test rods with the several instrumentations to observe the fuel behavior under the transient condition. One of the important parameters to be 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 upgrade 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 improved boring device was installed into the hot cell in 2010, and the mock-up test was performed with the dummy pellets to confirm the benefit of the improvements.

Keywords: Power ramp test, 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. This study was conducted under a contract with the Nuclear and Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI). 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.

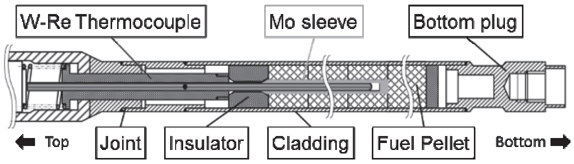

Fig. 1 Fuel rod for Power ramp test

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 (Step 1). 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 (Step 2) and the cleaning (Step 3) 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 (Step 4). 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.

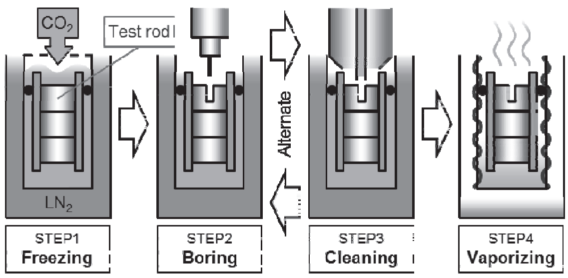

Fig. 2 Processing steps of center boring device

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.

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.

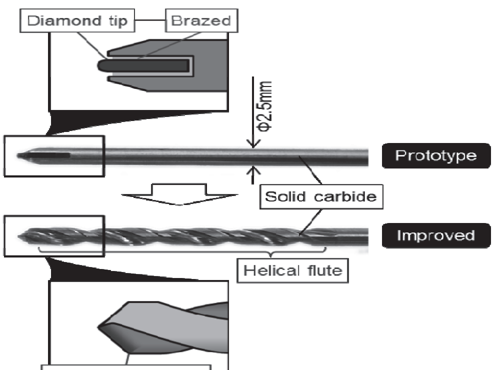

Fig. 3 Comparison of drill bits

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.

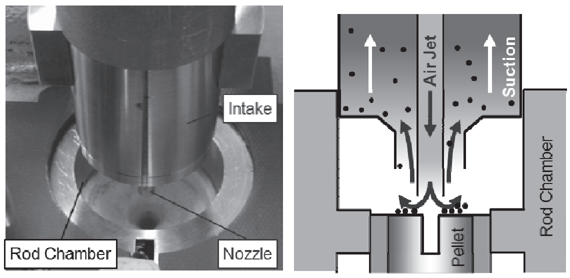

Fig. 4 Turnings cleaner

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.

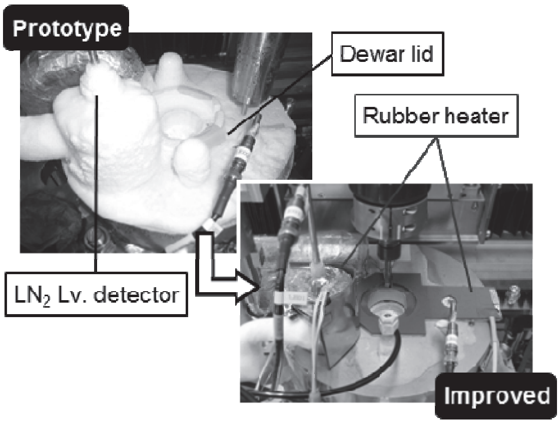

Fig. 5 Rubber heater on the Dewar lid

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, turnings 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.

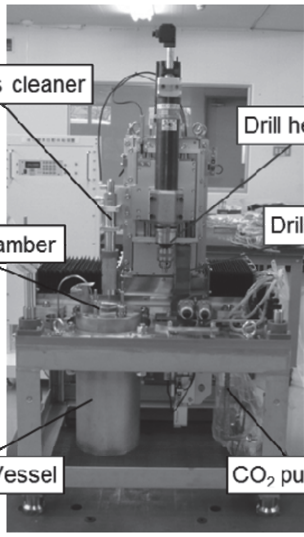

Fig. 6 Improved center boring device

3. PERFORMANCE TEST

The cold tests were performed with the dummy pellets to confirm the efficiency of the improvements of the center boring device. Based on the hardness of the irradiated pellets, the dummy pellets were made of barium ferrite and the target dimension of the boring is 40 mm in depth, 2.5 mm in diameter.

Fig. 7 shows the test result of the cold test to compare the boring speed between by the prototype device with the improved device. In case of the prototype device, the boring speed was slowed down repeatedly due to the blockage by the remained turnings. As a result, it took approximately 19 hours to bore a single hole. On the other hand, the boring speed by the improved device was kept constant during the whole process and the total boring time can be reduced 3 hours, less than 16% of prototype. It means that the boring blockage by the remains of turnings could be avoided by the ejection function of twist drill bit. And it was confirmed that the twist drill bit has the enough strength to bear against the rotation torque, because no breakage of the twist drill bit occurred.

Regarding the newly designed turnings cleaner, its collection rate achieved over 95% with the air filter of 5 μm mesh and the mist separator of 0.3 μm mesh. Therefore, it is suggested that the turnings cleaner will be able to keep the contamination minimum also in the hot cell environment. With the inching motion, the hole was bored precisely at the pellet center without the unfixing of the pellets and the rubber heater made it possible to ensure the correct maneuvering of the driving system for the accurate boring.

With the X-ray radiography, it was confirmed that the hole was bored correctly at the pellet center and there is no movement of the pellets during the boring.

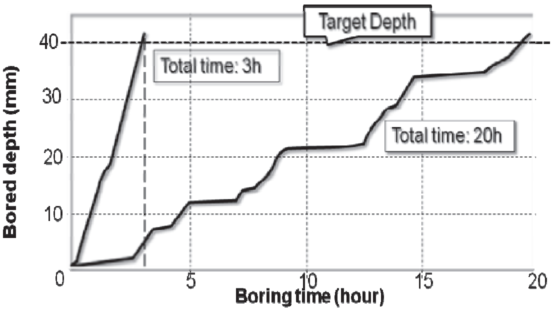

Fig. 7 Comparison of boring speed between improved and prototype drills

4. 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.

With the performance test using the dummy pellet, it was confirmed that the new drill bit could reduce the total boring time as less than 16% of prototype without the breakage of the drill bit, and the newly designed turnings cleaner could collect over 95% of the turnings. Furthermore, the inching motion and the rubber heater made it possible to ensure its processing reliability with the enough boring accuracy. Additionally, the other improvements were beneficial for the remote operation, and the utilization of the hot cell.

The improved device was installed into the hot cell to perform the hot test with the irradiated fuel pellet for the confirmation of its boring reliability and also its remote operability. After this confirmation, the test rods will be fabricated to provide them to the power ramp test.

ACKNOWLEDGMENT

The improvement of the center boring device for the irradiated fuel pellets has been sponsored by the Nuclear and Industrial Safety Agency, the Ministry of Economy, Trade and Industry of Japan. The authors are indebted to the engineers, technicians, researchers in JAEA working on the RFEF.

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6.7 Advanced Disassembling Technique of Irradiated Driver Fuel Assembly for Continuous Irradiation of Fuel Pins

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ABSTRACT

It was necessary to carry out continuous irradiation tests in order to obtain the irradiation data of high burn-up fuel and high neutron dose material for FaCT (Fast Reactor Cycle Technology Development) project. There, the disassembling technique of an irradiated fuel assembly was advanced in order to realize further continuous irradiation tests. Although the conventional disassembling technique had been cutting a lower end-plug of a fuel pin needed to fix fuel pins to an irradiation vehicle, the advanced disassembling technique did not need cutting a lower end-plug. As a result, it was possible to supply many irradiated fuel pins to various continuous irradiation tests for FaCT project.

1. Introduction

In the fast reactor cycle technology development project (FaCT Project), development of fuel corresponding to high burn-up is one of the important issues, and for this it is necessary to obtain the irradiation data of high burn-up fuel and high neutron dose material. To obtain these irradiated data, it is very effective to irradiate the fuel pins irradiated in the experimental fast reactor JOYO again. But re-irradiation is very difficult due to the relationship between the structure of the driver fuel assembly and the conventional disassembling technique. Therefore, a new disassembling technique of the driver fuel assembly was established so that the re-irradiation of the fuel pins irradiated in JOYO might become possible.

2. Consideration of disassembling technique

2. 1 Conventional disassembling technique

The conventional disassembling technique for the irradiated JOYO driver fuel assemblies in FMF (Fuel Monitoring Facility) is cutting the lower end plugs of fuel pins from the relation on the structure of the JOYO driver fuel assembly. As shown in Figure 1, the JOYO driver fuel assembly is composed of 127 fuel pins. The arrangement of fuel pins in the JOYO driver fuel assembly is 13 rows. As for each rows, two knock bars (upper knock bar, lower knock bar) are inserted into piercing hole of the end plug. The fixed method of row is welding the both ends of the knock bars to a wrapper tube. Figure 1 shows the conventional disassembling procedure technique in FMF. Since rows of fuel pins are being firmly fixed by welding, the simplest possible method is adopted in the disassembling of the JOYO driver fuel assemblies. The conventional disassembling technique procedure in FMF is as follows. The six sides of the wrapper tube are cut by an end-mill, and the wrapper tube is pulled out. The rows of fuel pins are separately cut by a band-saw. A cutting plane is about 5mm from an upper knock bar. The fuel pins taken out from the JOYO driver fuel assembles are supplied to various kinds of PIE (Post Irradiation Examination).

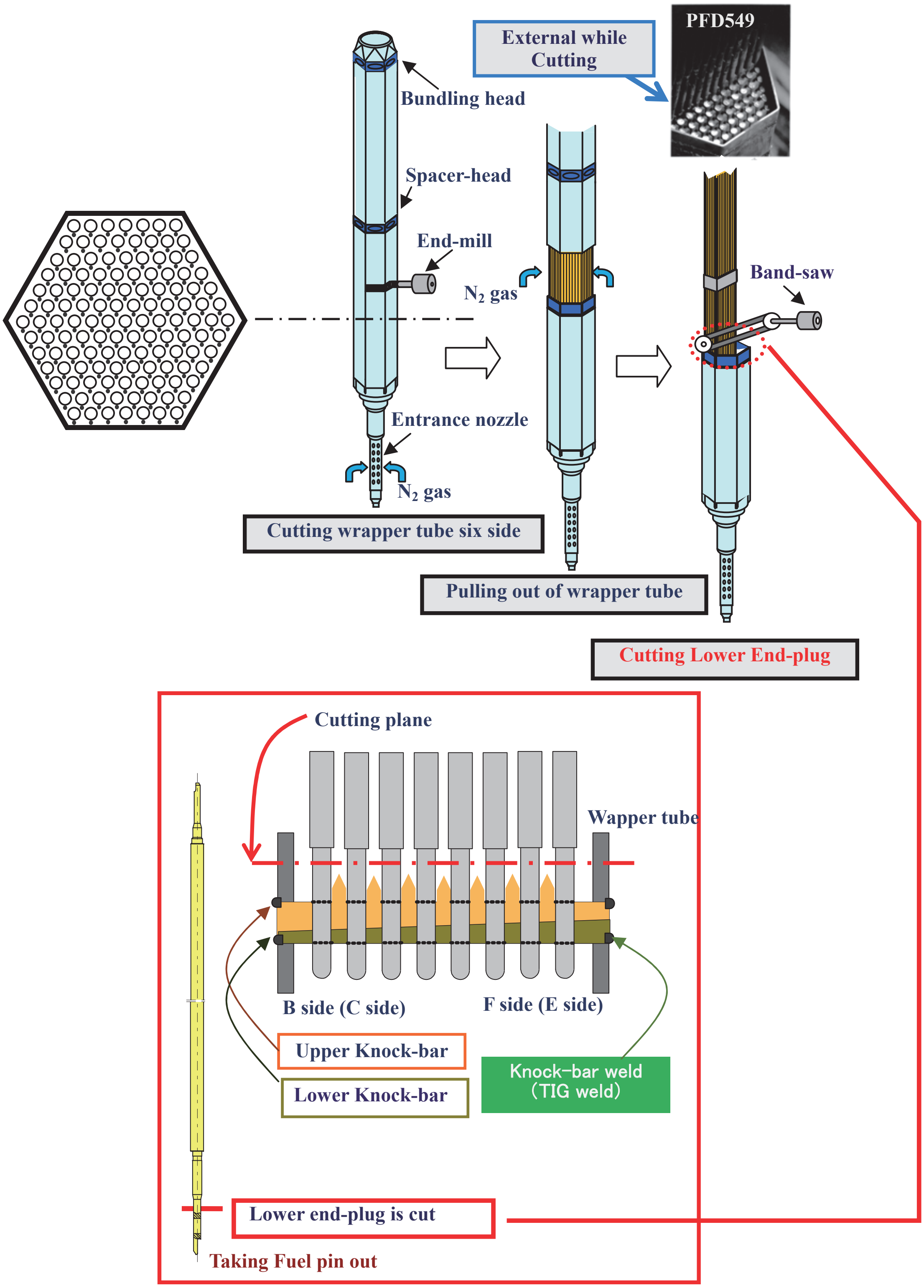

Figure 1 Conventional disassembling technique

2. 2 Irradiation Vehicle

There are several kinds of irradiation vehicles in JOYO and the choice is made by considering the irradiation conditions and purpose, etc. The irradiation vehicle used for re-irradiation will be a capsule type, the Uninstrumented Fuel Irradiation Subassembly Type-B (UNIS-B). The use of the capsule type UNIS-B is considered as a countermeasure for possible fuel failure during irradiation.

Figure 2 shows the method of loading the fuel pins to the UNIS-B. The fuel pins are put in a three-part container consisting of a shroud tube, a capsule, and a compartment, and the container is loaded into the UNIS-B. When the fuel pin is loaded into the shroud tube, the fuel pin is fixed by the support spring through the piecing hole of the lower end-plug. Threfore, it is necessary to leave the lower end-plug of fuel pin in place when disassembling the driver fuel assembly if the irradiated fuel pin is to be irradiated again.

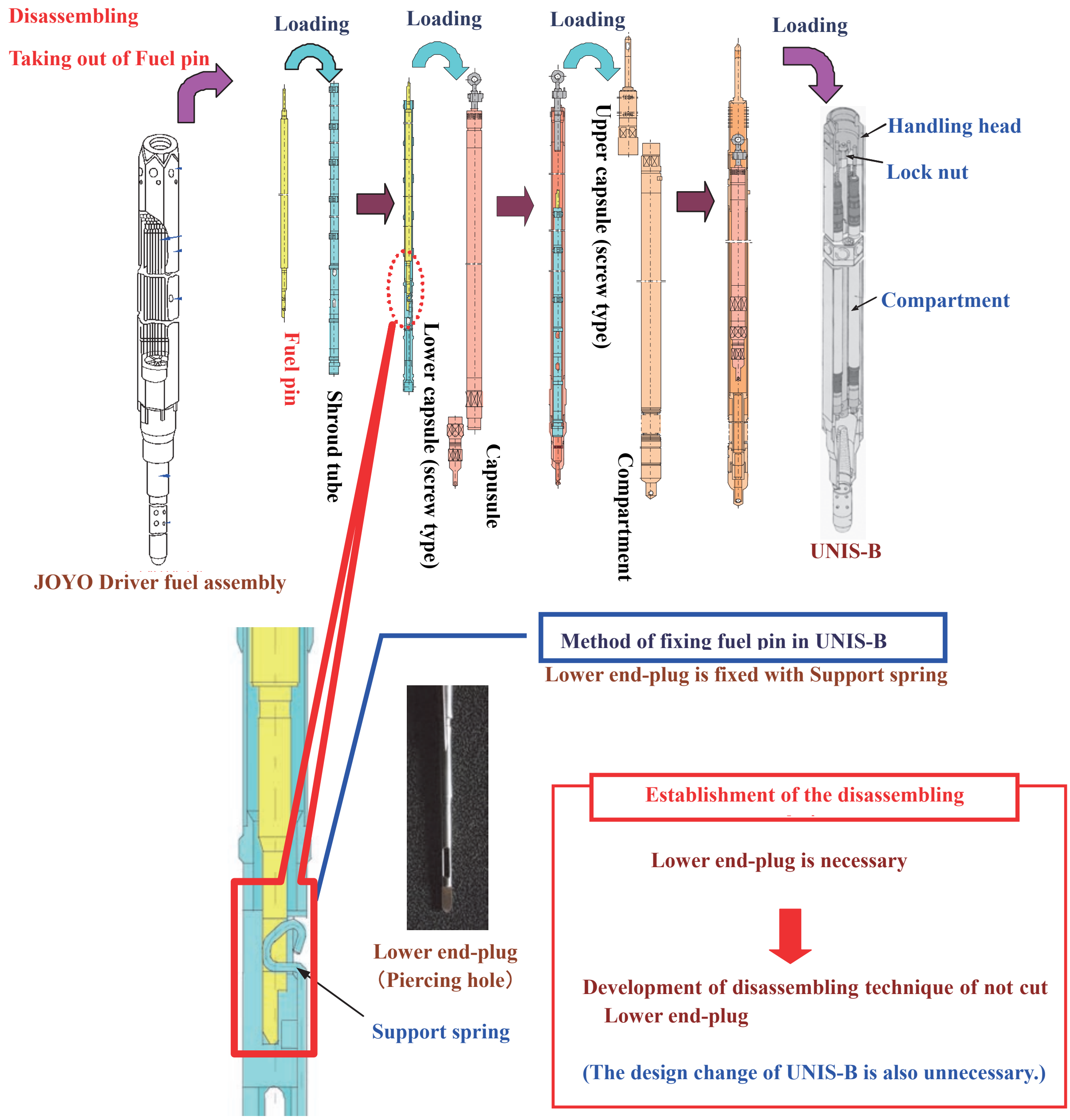

Figure 2 Method for loading UNIS-B

2. 3 Establishment of the disassembling technique

The newly designed disassembling technique is shown in Figure 3. The work consists of cutting knock-bar weld and taking out the knock-bar. The weld of the wrapper tube and the knock-bar to the fuel pin is cut by the end-mill separately. The knock-bar is taken out with a newly developed extrusion tool. This method does not cut the lower end-plugs of the fuel pins. Moreover, the shape etc. of the extrusion tool was devised to allow for remote operability in all work.

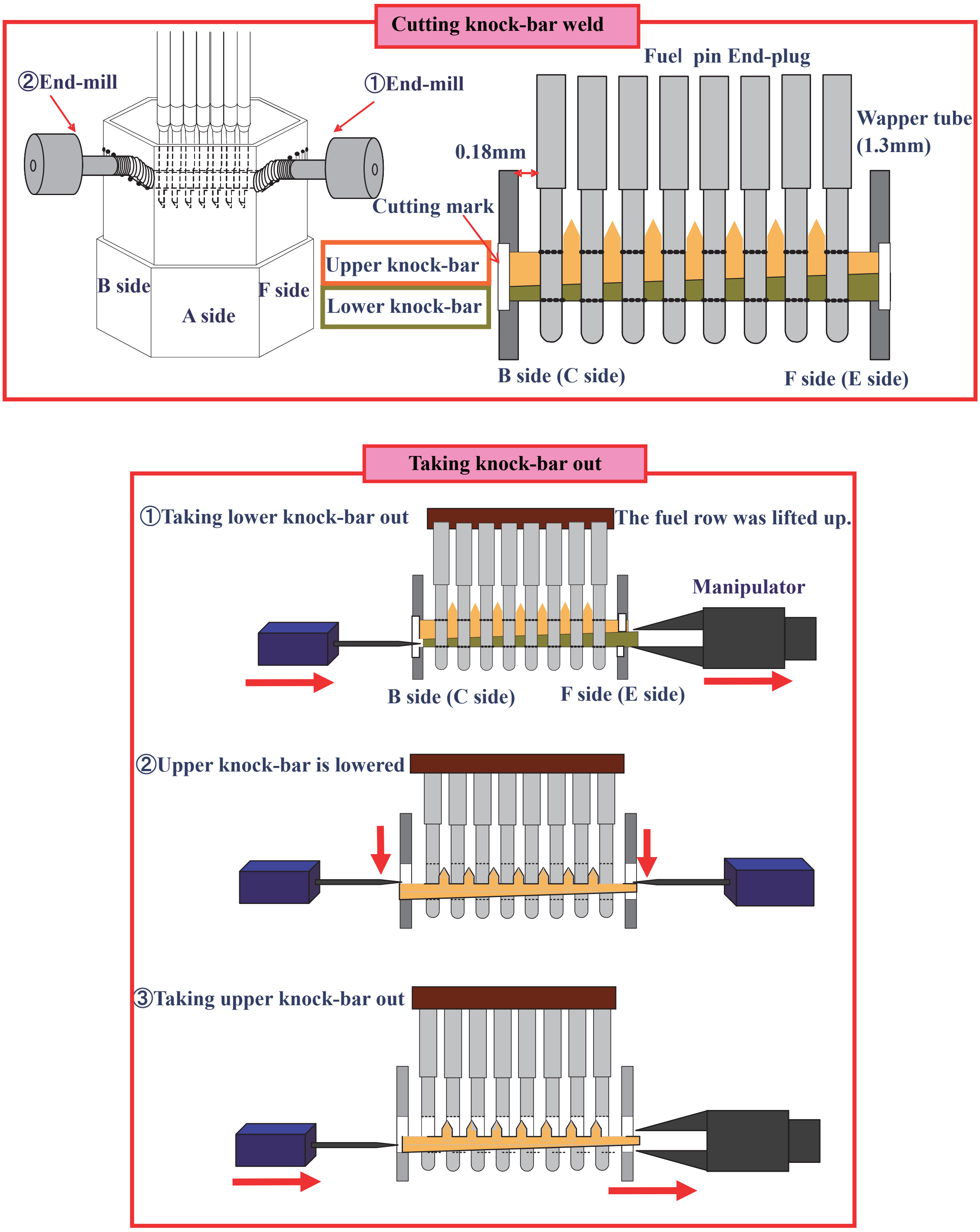

Figure 3 Caption correction work procedure

3. Application to JOYO driver fuel assembly

3. 1 Outline of the JOYO driver fuel assembly

This disassembling technique was applied to the JOYO MK-Ⅱ driver fuel assembly. The irradiation conditions of this driver fuel assembly were as follows.

*Burn-up: Average 6.09×10^4MWd/t,
Max 9.49×10^4MWd/t

*Fluence: 8.83×10^{22}n/cm^2 (E>0.1MeV)

Moreover, this driver fuel assembly had been kept in water for about 17 years after irradiation. After disassembling, part of the fuel pins taken will be provided for the re-irradiation test and part for the Post Irradiation (PIE).

3. 2 X-Ray Computer Tomography Observation

As for the structure of the driver fuel assembly, the space between the wrapper tube and the lower end-plug is very narrow. Therefore, it is necessary to know the internal situation. Then, the driver fuel assembly was observed by X-ray Computer Tomography (CT) before disassembling. Figures 4 and 5 show the CT image of the driver fuel assembly. As a result, it was confirmed that neither the fuel pin lower end-plug nor the knock-bar were changed by the irradiation. Moreover, it was confirmed that there was a space (0.18mm) between the wrapper tube and the fuel pin lower end-plug. It was judged to be possible to apply this disassembling technique to the driver fuel assembly.

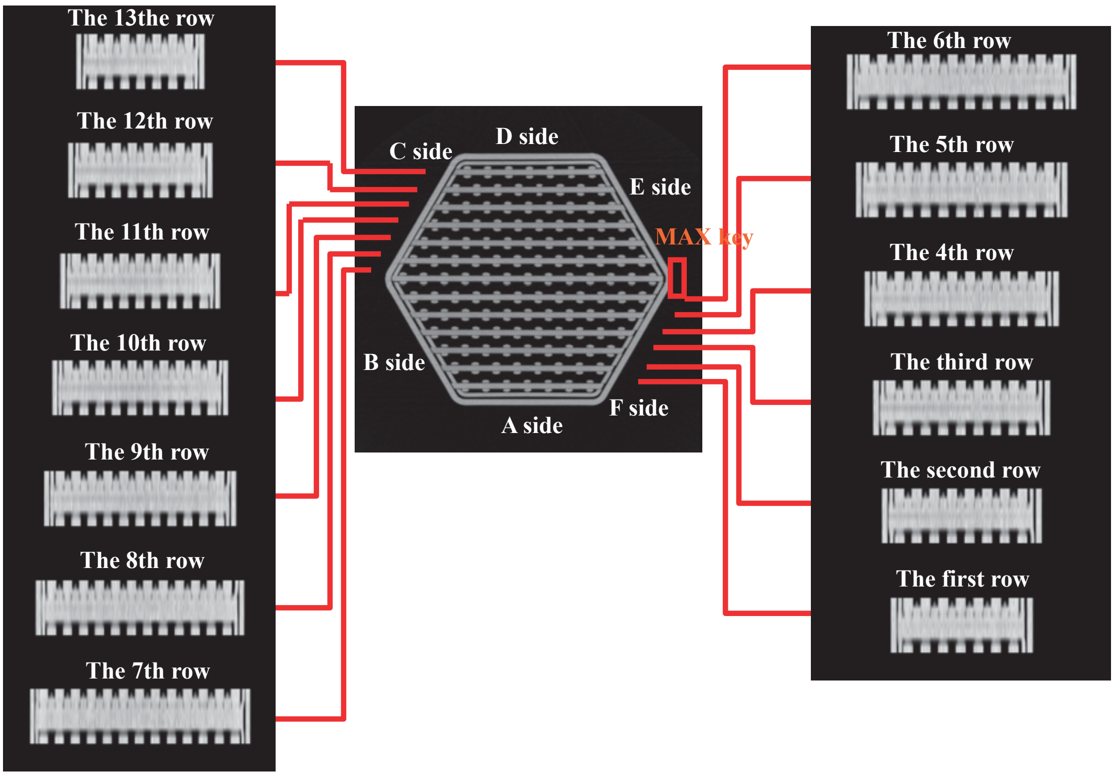

Figure 4 Cross-section observation (Range of taking picture: 1249.6～1267.6㎜ from bottom of assembly)

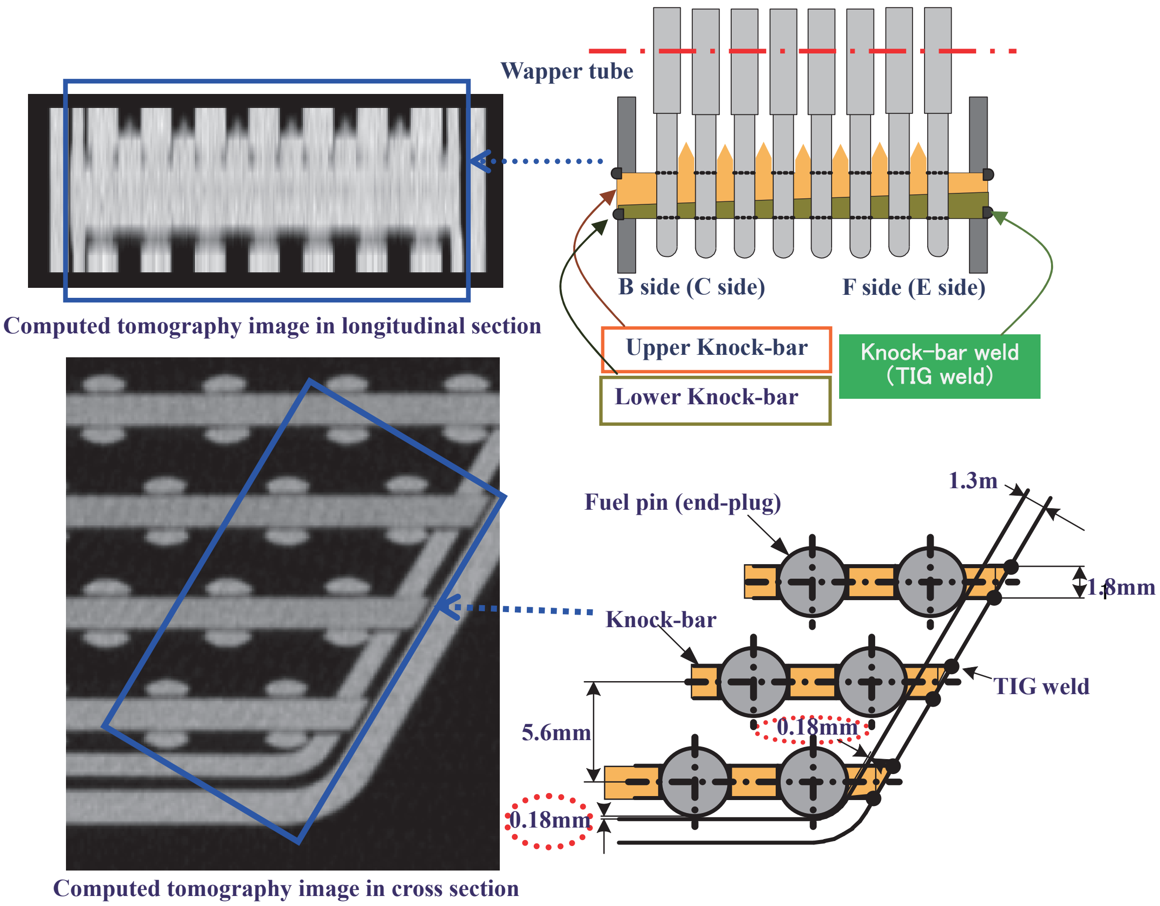

Figure 5 State of the driver fuel assembly

3. 3 Cutting knock-bar weld

Figure 6 shows cutting of the knock-bar weld. The knock-bar weld on the wrapper tube (thickness: 1.3mm) was cut by the end-mill. The cutting conditions of the end-mill were as follows. Max amount of cutting was 0.3mm per one time, sending speed was 3mm/s, and cutting position was 1,2567mm from the bottom of the driver fuel assembly. A row of fuel pin was cut separately for each row in order to take out the knock-bar out. Cutting the knock-bar was in the order of the F side and B side then the E side and C side. To avoid touching the fuel pin lower end-plug by the end-mill, the amount of cutting in the end-mill was controlled in detail.

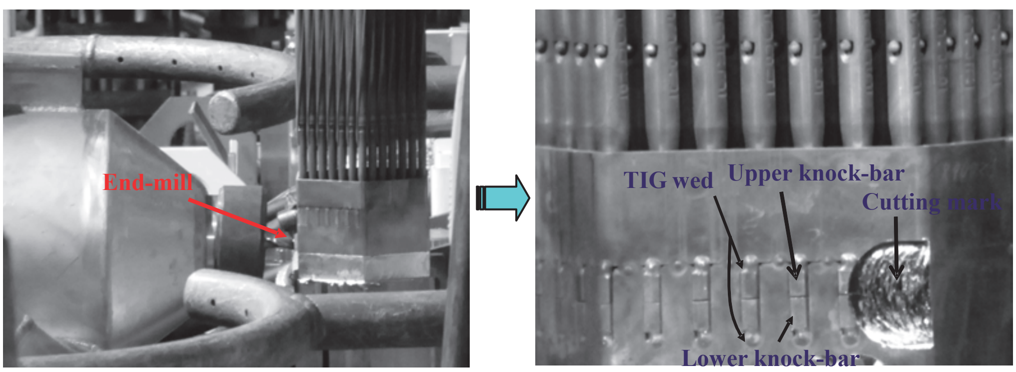

Figure 6 Cutting knock-bar weld

3. 4 Taking the knock-bar out

Figure 7 shows taking the knock-bar out. The knock-bar is pushed out of the B side or C side by the extrusion tool, and taken out of the F side or E side by a manipulator. The knock-bar was taken out in the order of lower knock-bar then upper knock-bar. The upper knock-bar was pushed up by the lower knock-bar in a fuel pin row. Therefore, after the low knock-bar had been taken out, the upper knock-bar was lowered to the position of the piercing hole by using the extrusion tool.

Figure 8 shows the disassembling procedure of the first row. All fuel pins could be taken out without cutting the lower end-plugs. Some of these fuel pins will be loaded to UNIS-B and re-irradiated in JOYO.

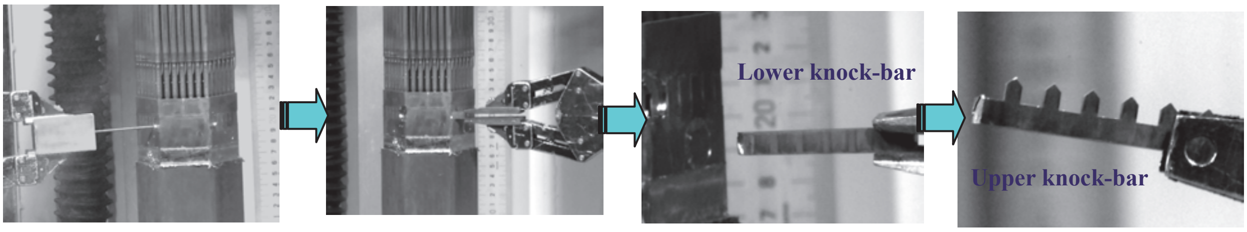

Figure 7 Taking knock-bar out

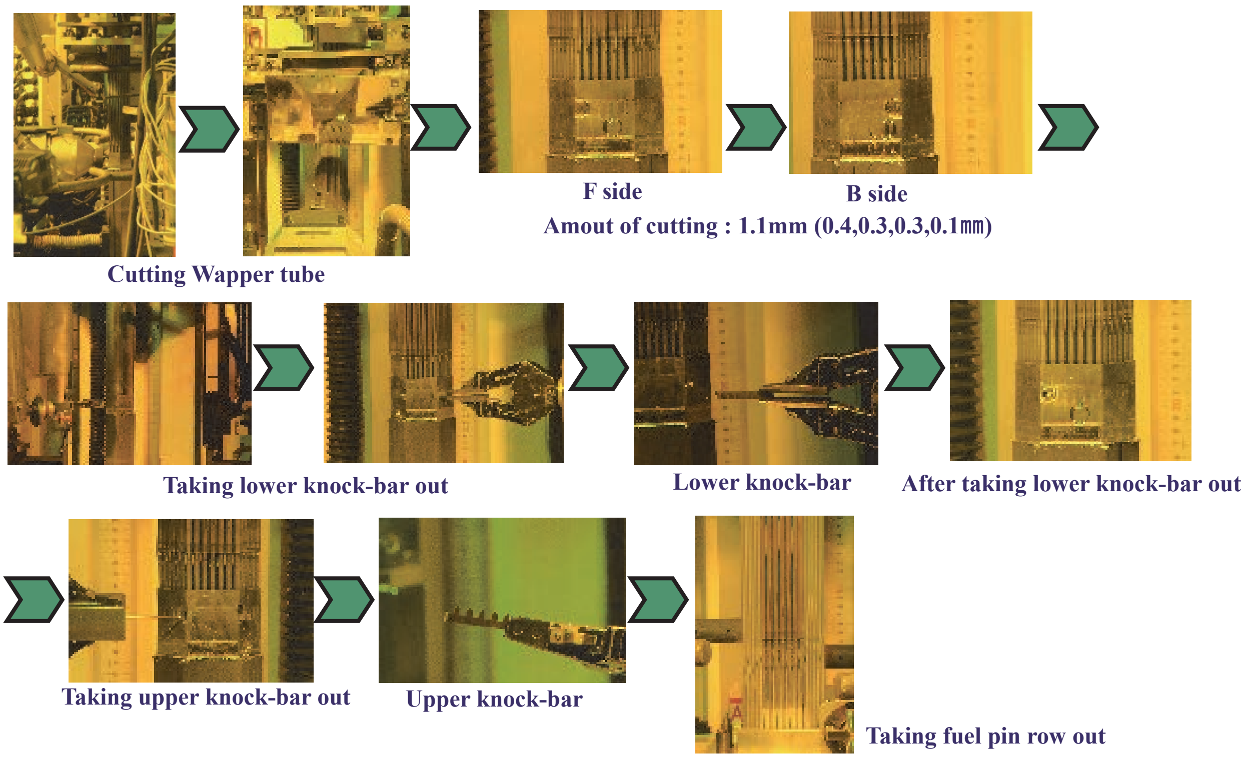

Figure 8 disassembling procedure of the first row

4. Application of the disassembling technique

This disassembling technique can also be applied to disassembling the Uninstrumented Fuel Irradiation Subassembly Type-C (UNIS-C). UNIS-C is an irradiation vehicle similar to UNIS-B, and the method of fixing fuel pins in UNIS-C is the same “knock-bar fixation type” as the JOYO driver fuel assembly. UNIS-C is a double wrapper tube structure, and continuous irradiation of the fuel pins is possible by exchange of the outside wrapper tubes. A further continuous irradiation using UNIS-B becomes possible by applying this disassembling technique after the continuous irradiation of UNIS-C ends.

5. Conclusion

The re-irradiation of the fuel pin of the driver fuel assembly (knock-bar fixation type) irradiated in JOYO has become possible. As a result, it will be possible to obtain high burn-up fuel for the FaCT Project and to reuse irradiated fuel in JOYO. This disassembling technique can get many fuel pins for the continuous irradiation from the same driver fuel assembly. It will be possible to carry out important irradiation examinations such as the Power-To-Melt (PTM) examination using the previously irradiated pins.

Reference

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7. Post Irradiation Examination Technology (2)

7.1 Preliminary Test of Thermal Diffusivity for Fuel Samples by Laser Flash Method

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To measure the thermal diffusivity of high burnup UO_2 fuel, some fuel samples were measured using laser flash apparatus with sapphire holder made in this study. The thermal diffusivity was measured with non-irradiated UO_2 and SIMFUEL as a preliminary test. Then, the effects of the sample holder and the cracked sample shape were studied with reference to the data. The samples were made in disk and cracked shapes. Three SIMFUEL samples were made as burnup. The holder with the sapphire tray gave data of up to 700℃, which is lower than the standard holder, but at a higher temperature, both data were coincident in measurement with the UO_2 disk sample. The thermal diffusivities of the disk and cracked samples were measured and compared with each other at 100℃ ~ 1600℃. The results of the UO_2 cracked sample agreed with those of the UO_2 disk samples. To observe the thickness effect, four UO_2 samples with different thicknesses were measured. Thicker samples had higher data at all temperature ranges, but data of 1～3 mm of thickness showed little difference. In the measurement of the SIMFUEL disk sample, the results normally showed a burnup effect. The higher the burnup sample, the lower the diffusivity was shown at all temperature ranges. Unlike UO_2 samples, the results of the SIMFUEL cracked sample were higher than those of the SIMFUEL disk samples. Consequently, the new holder with the sapphire tray showed lower results at 100℃ by 15%, but as the temperature went up to 700℃, this difference was reduced even at higher temperature where the results agreed with the reference data.

Keywords: Thermal diffusivity, LFA, Laser Flash, SIMFUEL, Sapphire, UO_2

1. INTRODUCTION

The thermal conductivity of nuclear fuel is important factor to predict the thermal behavior in reactor. It consists of the thermal diffusivity, the specific heat and the density. Especially, measurements of thermal diffusivity have been studied and apparatus being developed as well.

But, our country does not have experience with thermal diffusivity test yet. So far, not even the system has been prepared. The LFA(Laser Flash Apparatus) was installed in IMEF(Irradiated Materials Examination Facility) recently. To measure the thermal diffusivity of an irradiated fuel sample such as UO_2 pellet, some of the considerations are as follows; sample shape is not a disk but a cracked shape because the pellet cracking occurs in the beginning of reactor operation and some chips or parts of the sample would be dropped under measurement in addition to easy sample loading and withdrawing not by hand but by tools.

In this study, the new sample holder was designed using the sapphire tray for the arbitrary sample shape and the thermal diffusivity was measured with the non-irradiated UO_2 and SIMFUEL as a preliminary test. Then, the effects of the sample holder and the cracked sample shape were studied with reference and standard data.

2. EXPERIMENTAL

2.1 Laser Apparatus

The LFA consists of a laser generator, a furnace and an IR detector as shown in Fig.1. The laser is generated by Nd:YAG and wavelength is 1.064μm with 40 J of pulse energy. Diameter of laser beam is about 13 mm. the laser is contacted on the bottom surface of a sample in the furnace and IR detector is activated to measure the thermal signal on the upper surface of it. The furnace is available to heat up to

2,000 oC by graphite heat source. S-type thermocouple was used to measure the sample temperature. Crystal of IR detector is In-Sb. The LFA system in the facility is carried out in only gas flowing atmosphere not in vacuum state. Argon was used as the flowing gas under heating and measurement.

In this study, thermal diffusivity was measured at every 100 oC from room temperature to 1,600 oC and three laser shots were performed at every temperature point. Argon flowing was set at 150 ml/min. under measurement. Laser in every test was shot with 450 V of energy and 0.6 msec of duration time.

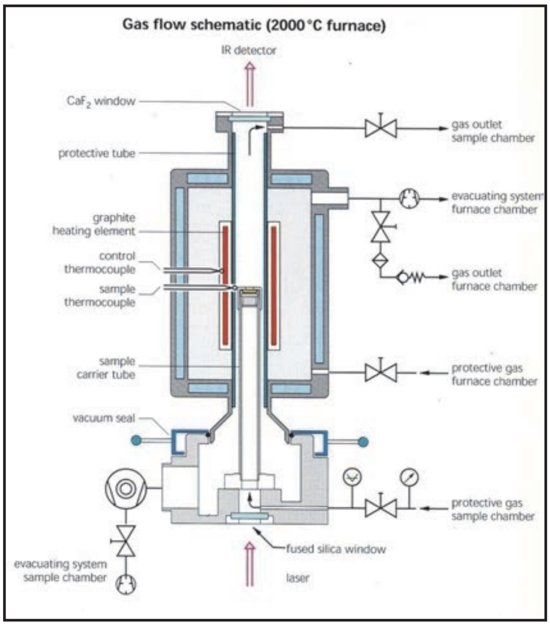

Fig. 1 Diagram of LFA

2.2 Sample Preparations

The samples in this test were fresh UO_2 and SIMFUEL with 97%~98% of theoretical density and natural enrichment. The SIMFUEL(SS1,SS2,SS3) was made with several compositions with burnup in Table. 1. Samples were disk type with 8 mm of dia. and about 2 mm of thickness as well as the crack type with 2 mm of thickness as shown in Fig. 2 and Fig. 3.

To observe the thickness effect, four UO_2 disk samples were prepared with about 1mm, 2mm, 3mm and 4mm of thickness as shown in Fig. 4.

Table 1 Compositions of SIMFUEL with burnup

Index	SS1	SS2	SS3
SrO	0.055	0.11	0.22
Y_2O_3	0.039	0.078	0.156
ZrO_2	0.23	0.46	0.92
TeO_2	0.041	0.082	0.164
La_2O_3	0.139	0.278	0.556
CeO_2	0.646	1.292	2.584
Nd_2O_3	0.477	0.954	1.908
SUM	1.627	3.254	6.508
Burnup	3%	6%	12%

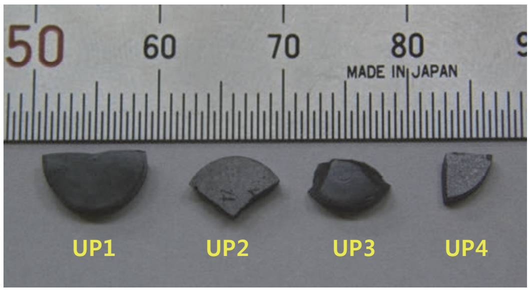

Fig. 2 Crack samples of UO_2

Fig. 3 Crack samples of SIMFUEL

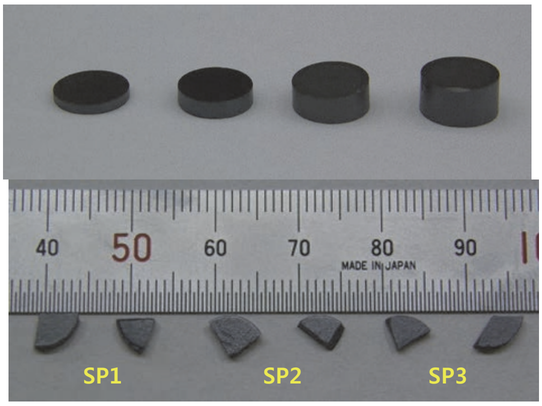

Fig. 4 UO_2 samples with different thickness

2.3 Sample Holder Design

Sample holder consists of an alumina holder and a SiC cap. A cap is used to stop the laser leakage which causes the thermal signal noise of IR detection. A standard sample holder was designed as three tips to reduce the contact area of sample when a sample was loaded on it as shown in Fig.5. It is convenient in a cold-lab test but not easy to load sample in a hot-cell or glove box. Moreover, in the case of the irradiated ceramic fuel, sample would be dropped by the cracking under test. With considerations of dropping accident and easy sample loading, new holder was designed with a sapphire tray as shown in Fig.5. The sapphire has high melting point(2,040 oC) and good hardness. The new holder including the cap was made with high purity alumina.

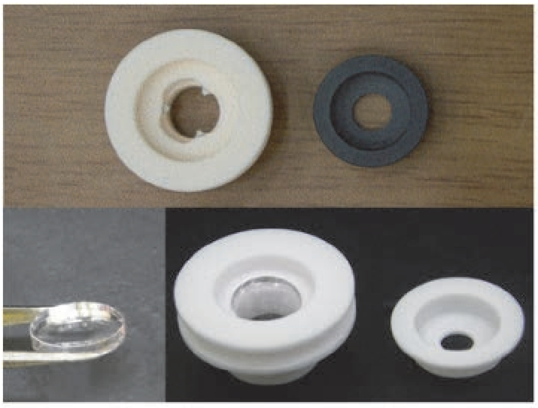

Fig. 5 Standard holder(up) and new holder with sapphire tray(down)

3. RESULTS

Thermal diffusivity is generally measured in vacuum state. So, the test with flowing argon gas was verified to observe atmosphere effect. Fig.6 showed no difference between reference data[1] in vacuum state and the argon atmosphere condition using UO_2 disk sample with ignorance of porosity effect.

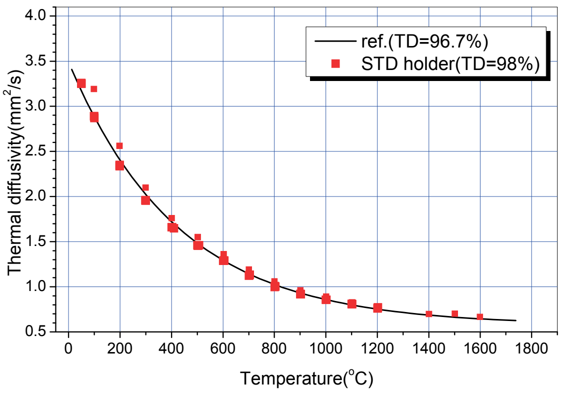

Fig. 6 Comparison of thermal diffusivity between vacuum state and flowing gas state.

Two types of sapphire trays were made; one was flat type and the other was concave type of one side as shown in Fig.7. Test for the effect of the sapphire tray shape was carried out with UO_2 disk sample using new alumina holder.

In the Fig.8, the flat type of sapphire tray showed higher data than concave type due to heat loss by the sample contact. So, the flowing gas state and the concave sapphire tray were applied to all of the following tests.

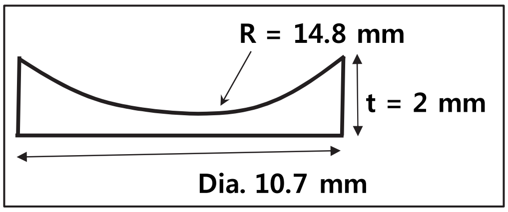

Fig. 7 Geometry of concave sapphire tray

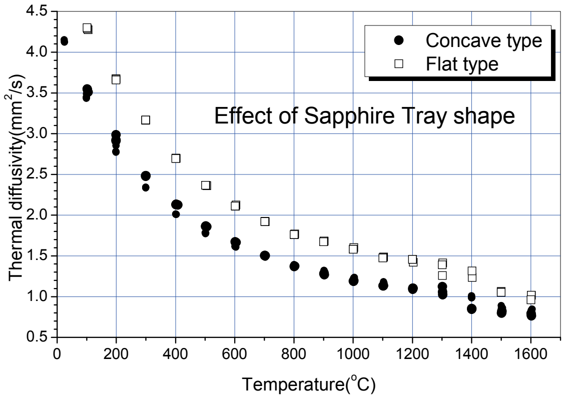

Fig. 8 Effect of sapphire tray shape

Cape-Lehman model[2] is generally applied for thermal signal analysis in LFA system but laser leakage from new holder occurred as soon as laser was contacted sample, which raised up the thermal signal at the beginning point as shown in Fig. 9. That was reason why data with new holder were higher than those of standard holder as shown in Fig. 10. To fit this behavior, the radiation model was applied and the results were shown in Fig. 11.

So, the radiation model was applied to all of the following tests.

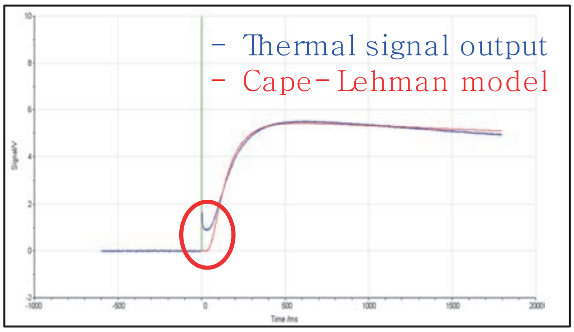

Fig. 9 The thermal signal output and Cape-Lehman model

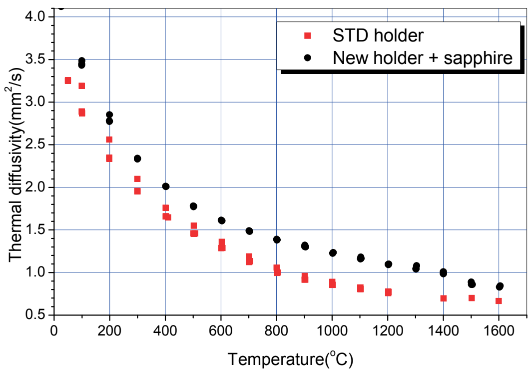

Fig. 10 Comparison of data between STD holder and new holder (Cape-Lehman model)

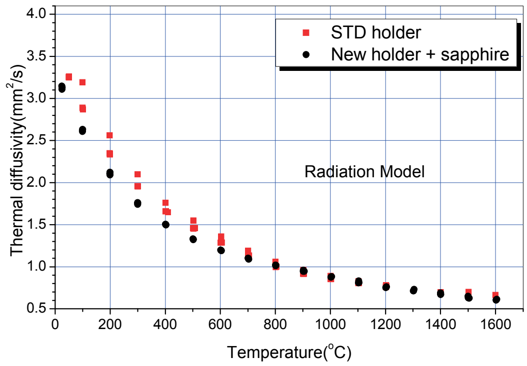

Fig. 12 Comparison of data between STD holder and new holder (Radiation model)

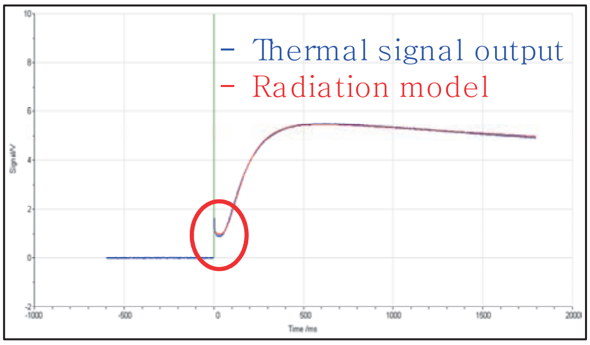

Fig. 11 The radiation model behavior

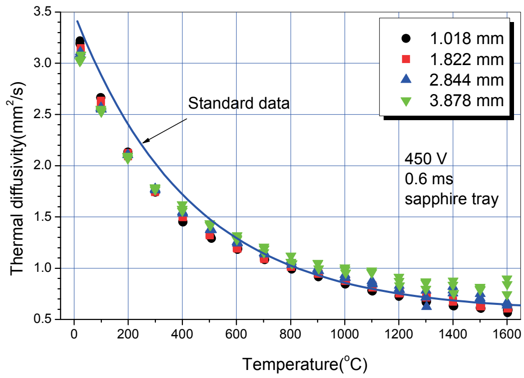

Fig. 13 The thickness effect (UO2 disk samples)

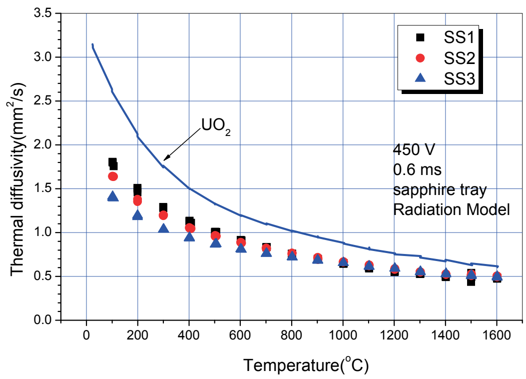

Fig. 14 The thermal diffusivity of SIMFUEL

In the case of UO_2 disk sample, 15% of difference was observe at 100 oC but it decreased as temperature went up then, there was no difference over 800 oC as shown in Fig. 12.

To estimate the thickness effect, UO_2 disk samples with four different thicknesses were measured in the same conditions(laser voltage : 450V, duration time : 0.6 ms). Fig. 13 showed the thicker sample was higher data at high temperature due to the heat loss of sample surface. At the low temperature below 600oC, all data were lower than data with standard holder.

Appropriate thickness is seemed to be 2 mm as considerations of the dimension error on thin sample and the heat loss on thick sample.

The tests for SIMFUEL disk samples(SS1,SS2,SS3) were carried out as burnup. Fig. 14 showed all samples were lower than UO_2 and the different data were generally shown at low temperature due to burnup effect. But amount of the additives in SIMFUEL were not significant at high temperature as almost same data at high temperature.

Four UO_2 cracked samples(UP1, UP2, UP3, UP4) were measured and those data with disk and standard data were shown in Fig. 15.

The data of cracked samples agreed with those of disk sample in spite of the data scattering from different shape and size.

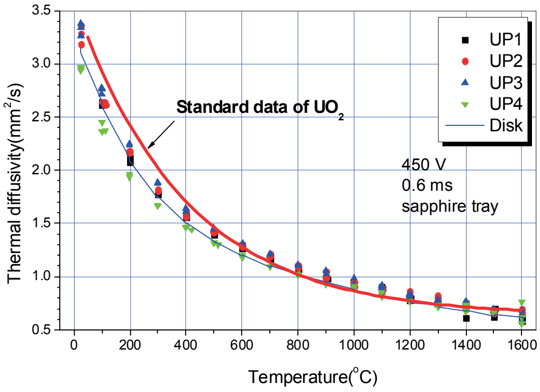

Fig. 15 The thermal diffusivity of UO2 crack samples

In the case of the SIMFUEL cracked samples(SP1, SP2, SP3), the data difference by burnup effect was observed at the same as disk data as shown in Fig. 16. All data were higher than disk data unlike the case of the UO_2 disk and cracked sample data. The data of high burnup(SP3) showed little difference with disk data.

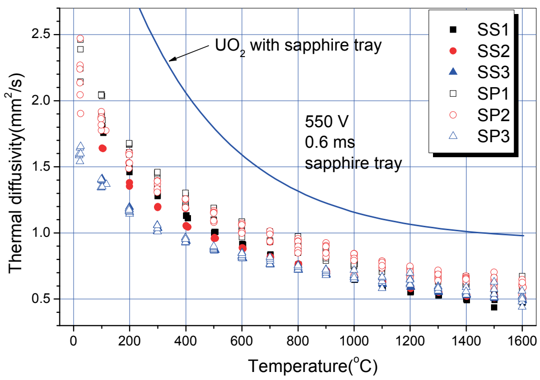

Fig. 16 The thermal diffusivity of SIMFUEL cracked samples

The cracked sample showed wide scattering at the low temperature due to non-standard shape and size of sample. Therefore, sapphire tray was made bigger in size than cracked sample to reduce error of laser contact, but the thermal signal detection area was too small, which made the data scattering in various sample condition.

4. CONCLUSIONS

To measure the thermal diffusivity of irradiated fuel(ceramic fuel), the cracked samples were used. As considerations of the system safety and sample shape, new sample holder with sapphire tray was designed and tested with non-irradiated UO_2 and SIMFUEL for preliminary test.

- Flowing gas atmosphere in this system showed no different data with those in vacuum state.
- Thickness effect was observed and 2 mm is recommended in this study; the thinner sample has dimension measurement error and the thicker one has heat loss on surface.
- In the case of UO_2, data at low temperature would be underestimated, so it is needed to be calibrated while no difference was observed at high temperature.
- In the case of SIMFUEL, as higher burnup sample, lower data were obtained due to burnup effect. Unlike UO_2 test, SIMFUEL disk samples showed lower data than the cracked ones. But there was little difference at the sample with highest burnup.

The scattering data of cracked samples were observed due to sample shape and size. The holder design with sapphire must be studied more because various sample shapes and thermal detection method would not be decided.

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[4] Expert advice for the sapphire tray, ITU, Germany

7.2 Development of Capsule Assembling Apparatus

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The service of JMTR hot laboratory, associated with the Japan Materials Testing Reactor, was started on 1971 to examine specimens irradiated mainly in the JMTR. A wide variety of post irradiation examinations for research and development of nuclear fuels and materials are available in the JMTR hot laboratory. This laboratory has an advantage that its hot cell is connected with JMTR by a canal directly, and it is easy to transport irradiated capsule and specimens. New power ramping test for the high burn-up fuels by using the JMTR has been planed. The power ramping test using a boiling water capsule facility needs a re-capsuling of fuel rods for re-irradiation, and a modification of the facility up to about 100 GWD/t were necessary. This report introduces the new handling techniques and capsule assembling apparatus for the boiling water capsule facility.

Keywords : JMTR, Hot laboratory, Post irradiation examination, Power ramping test, Boiling water capsule facility, Handling techniques and capsule assembling apparatus

1. INTRODUCTION

New Post Irradiation Examination (PIE) facility for the high burn-up fuels up to about 100 GWD/t has been planned for the purpose to perform "Light-water reactor fuel and material detailed integrity investigation" contracted with Nuclear and Industrial Safety Agency of Ministry of Economy, Trade and Industry. Because of the maximum handling capability of UO_2 fuels burn up in the JMTR hot laboratory is 55 GWd/t, (1) enforcement of neutron shield, (2) development of new capsule assembling apparatus, and (3) the design change of transporting cask are necessary. In this report, development of the capsule assembling apparatus is described.

The capsule assembling apparatus assembles the instrumented fuel rod into the Boiling Water Capsule (BOCA) which used for fuel transient tests. BOCA has 7892 mm in total length and about 100 kg in weight. Specimens are installed into reused BOCA using water canal and Concrete cell No. 1 (C-1). The neutron shielding capacity for the former capsule assembling apparatus was not enough. Therefore, the assemble method was changed, and new capsule assembling apparatus was developed. The assembling for high burn up fuel into the BOCA was performed safely and efficiently by this development.

2. FUEL TRANSIENT TESTS PLAN

To confirm the integrity of high burn up fuels under power ramping condition, the fuel transient test using the BOCA/OSF-1 is scheduled in JMTR. The BOCA which high burn up fuel rods installed is to be irradiated in this test. The BOCA/OSF-1 consists of capsule cooling system, power output control system for test fuel rod and fuel failure detecting system, and is possible to perform power ramping test for fuel rods.

The target fuels of this plan are high burn up 10x10 type BWR fuel. Which is used in EU for industrial development, and fuel pellets are normal UO_2 pellets. The fuel irradiated in European commercial reactor will be processed to short length, and transported to Japan. End plugs and instrumentations (Extensometer and FP gas pressure indicator) are equipped in the fuel test facility in JAEA. Finally, it is transported to JMTR hot laboratory, and installed into BOCA.

3. HANDLING OF HIGH BURN UP FUELS

Former capsule assembling apparatus required about 50 persons to assemble per a BOCA. Only two BOCAs were assembled in a JMTR operation cycle. Four BOCAs irradiation is planned in new project, and it is necessary to shorten the assemblage period and to reduce the operating cost.

The former capsule assembling apparatus was installed to assemble and dismantle capsules used for power ramping tests in 1980. This apparatus consists of shielding body and end plug tightening apparatus. The shielding body was used to transport fuels installed in capsules. The shield of former capsule assembling apparatus is about 1.2 m in diameter, 9 m in length and 20 ton in weight, and was difficult to handle in the water canal (6 m in depth and 3 m in width). Former capsule assembling apparatus is shown in Fig.1.

1) The shielding body was a tube type lead casting shield and the upper lid separate to two pieces. The BOCA and a moving bed were installed into shielding body under water canal with 6 m in depth. Shielding body moving mechanism is installed on by crane.
2) The transporting mechanism moves electric motor on the rail, and connects the shielding body with the horizontal γ gate of the C-1. BOCA was installed into the C-1 by this mechanism.
3) The capsule end plug tightening apparatus was installed on the rack in the C-1. The apparatus keep the end of outer tube of the BOCA, and fuel rod are installed and picked up from the BOCA. The capsule end plug tightening apparatus also has function of sealing performance test.
4) The capsule turning apparatus and the moving bed turn the BOCA to horizontal or from horizontal in the water canal. It is arranged near the water canal wall.

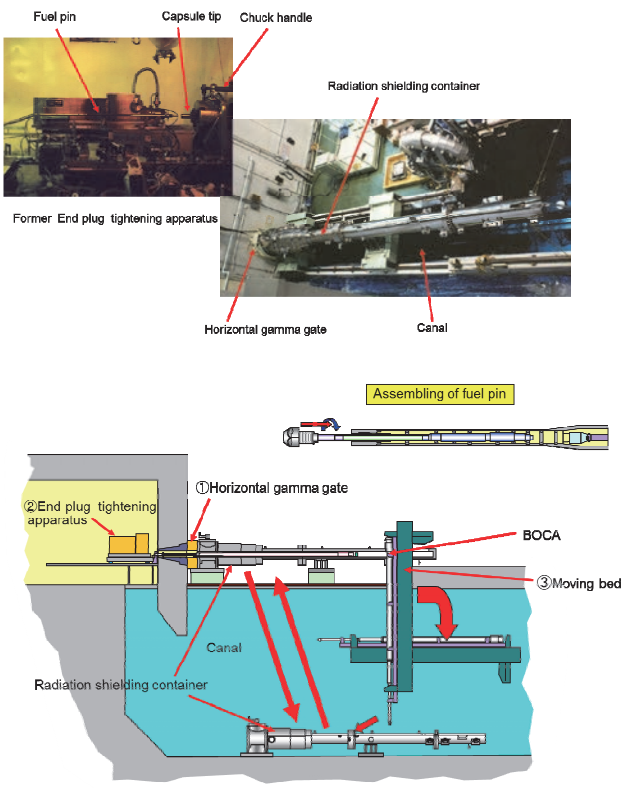

Fig.1 Schematics of the former capsule assembling apparatus.

4. NEW CAPSULE ASSEMBLING APPARATUS

To design new capsule assembling apparatus, the following points were studied.

1) Assembling shielding body

Additional neutron shield is required in this new project because of the increasing radiation dose. However, additional neutron shield is impossible to use former type shielding body. Therefore, the canal water is designed to use as the neutron shield. New capsule transporting apparatus which carry the BOCA from the canal water directly move to the C-1. The driving mechanism of this apparatus was decided to the chain method in consideration of durability, stability, and maintainability.

2) Capsule end plug tightening apparatus.

The capsule end plug tightening apparatus was designed for the new capsule transporting apparatus which adapt the vertical (diagonal) carrying method. New capsule end plug tightening apparatus is possible to perform the control of tightening of end plug (torque and length) and helium leakage test as same as former apparatus.

New capsule assembling apparatus consist of the capsule carrying apparatus in water canal and the end plug tightening apparatus in cell is shown in Fig.2. This apparatus carries the BOCA into the C-1. About 500 mm of lower outer tube is lifted on the rack in the C-1 with inclination of 50 degrees through the canal water. New capsule end plug tightening apparatus was changed to the tilt setting type from former horizontal setting type.

The details of the newly designed capsule specimen installing apparatus are as follows;

1） Capsule keeping bed

The capsule keeping bed is 7 m in length to keep the straightness of the BOCA, and set on the moving track of capsule carrying apparatus.

2） Capsule carrying apparatus

The capsule carrying apparatus consist of guide rail settled on canal wall. The rail frame settled on bottom of the canal, and moved by the driving chain. The driving chain unit is driven by the electric motor settled out of the canal.

3） Capsule end plug tightening apparatus

The capsule end plug tightening apparatus keeps the end of the BOCA, and inserts instrumented fuel from the lower end plug of the BOCA. This apparatus consists of screw tighten apparatus, tightening control apparatus and helium leakage test apparatus. The connecting for instrumentation is confirmed by the continuity test of the electric signal.

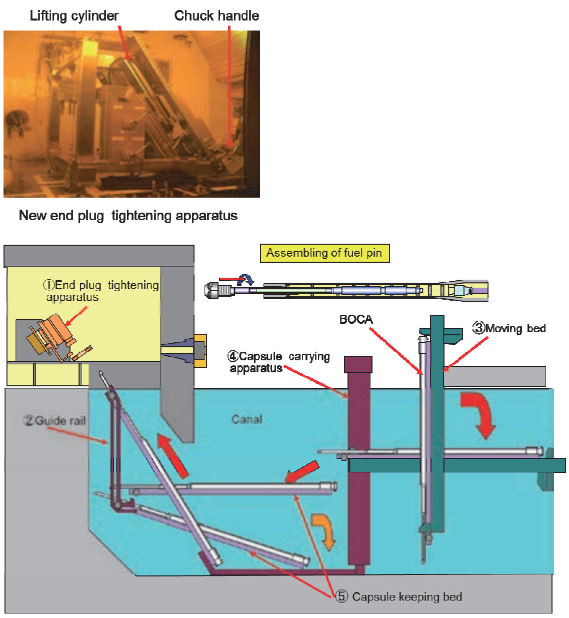

Fig.2 Schematics of the new capsule assembling apparatus.

5. MOCK UP TEST BY NWE APPARATUS

After manufacturing in a factory, new apparatus was transported to the JMTR hot Laboratory, and the total performance tests using mock up capsule was performed. The simulated BOCA capsule used mock up test is shown in Fig. 3. The installation of apparatus and mock-up test were performed as follows;

1) Installation of the capsule assembling apparatus

Capsule carrying apparatus was moved to bottom of the water canal with whole parts of apparatus were assembled. This apparatus was fixed by anchor bolts to the concrete of canal to endure earthquake.

2) Installation of capsule end plug tightening apparatus

The capsule end plug tightening apparatus was designed to be able to move by crane settled in cell for the purpose to avoid the conflict with other works. Four guide blocks are settled in cell for the purpose to decide position of setting.

3) Capsule mock-up test of capsule assembling apparatus

A capsule mockup test performed by simulated BOCA capsule. It was confirmed that the capsule was moved into cell as to the design. The end plug tighten apparatus can assemble and disassemble end plug by the specified tightening torque.

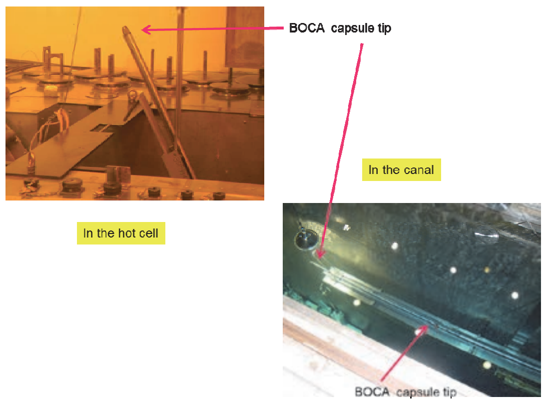

Fig. 3 BOCA capsule during mock-up test

6. SUMMARY

In this development, to handle the high burn up fuel in the JMTR hot laboratory, the capsule carrying apparatus and specimen installation apparatus were newly developed and installed without larger neutron shield. Main development results are shown as follows;

1) The assembling and setup work of shielding body above the canal was omitted, and BOCA is installed into cell from water canal directly. It make possible to decrease working cost, number of working persons and working period.
2) By using the canal water as the neutron shielding, increased neutron flux and the radiation dose of capsule assembling workers will be decreased.
3) New capsule assembling apparatus make it possible to install high burn up fuel into the BOCA safely and efficiently.

7.3 Integrated Test for Evaluating PWR Spent Fuel Integrity in Dry Storage Condition

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The PWR spent fuel dry storage project was started by Korea Radioactive Waste Management Corporation (KRMC) established in 2009 with Korea Atomic Energy Research Institute (KAERI) in order to prepare the commercial PWR SF dry storage. KAERI are designing the integrated test apparatus to verify spent fuel integrity in an accelerated dry condition. The apparatus will be placed inside pool with 15 m depth at Post Irradiation Examination Facility (PIEF) of KAERI in 2014. 12 spent fuel rods, subject to the integrated test are planned to be transported from a Korea commercial nuclear power plant to PIEF of KAERI in May 2012. Six rods will be selected among them for the integrated test, after the non-destructive hotcell test is carried out for all rods. KAERI has a plan to do the integrated test with 6 rods for 3 ~ 6 years in the accelerated dry condition of He atmosphere with about 400 °C to support a cladding degradation modeling and estimate SF integrity engineering margin.

Keywords: PWR spent fuel, radioactive waste management, dry storage, integrated test, hotcell, post irradiation examination

1. INTRODUCTION

The dry storage of the PWR spent fuel is considered as an alternative of the wet storage, which has limited storage capacity in KOREA. KRMC kicked off the spent fuel dry storage project with KAERI in 2009. KAERI designing the integrated test apparatus in order to evaluate the integrated fuel performance in an accelerated dry condition, validate cladding degradation models, and develop technologies related in the demonstration test. The apparatus will be placed inside pool with 15 m depth at PIEF of KAERI in 2014.

The integrated test has different concept with the demonstration test. Table 1 summarizes a comparison of the major features of them. In this study, the integrated test uses the spent fuel rod as a target material in the accelerated dry condition, whereas the demonstration test uses the spent fuel assembly in the actual dry storage condition.

Table 1. Comparison between the integrated test and the demonstration test

Characteristic	Integrated Test	Demonstration Test
Target Material	SF rod	SF assembly
Duration	2~5 years	actual storage period
Test Temperature	accelerated	real condition
Heating	heater	decay heat
Purpose	V&V	to demonstrate dry storage system

2. SELECTION OF TARGET SPENT FUEL RODS

In order to select target spent fuel rods for the integrated test, we started from 11,121 assemblies of KRMC database as shown in figure 1. Several selection bases were used: burnup of 45 to 48 GWd/tU, Zircaloy-4 cladding, decay time of 7 to 10 years, specific fuel model and worst case of power history. Finally, 12 spent fuel rods were selected. 6 rods among them will be used for the characteristic test and 6 rods for the integrated test. Pre-test before the integrated test and post-test for selected rods are performed by the general post irradiation examination procedure for PWR spent fuel rod as shown in the block diagram of figure 2. Pre-test has only non-destructive tests. Post-test includes non-destructive test, destructive test and mechanical test for the spent fuel.

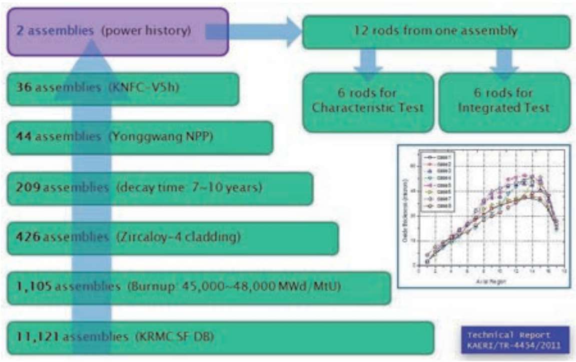

Fig.1. Selection of target spent fuel rods.

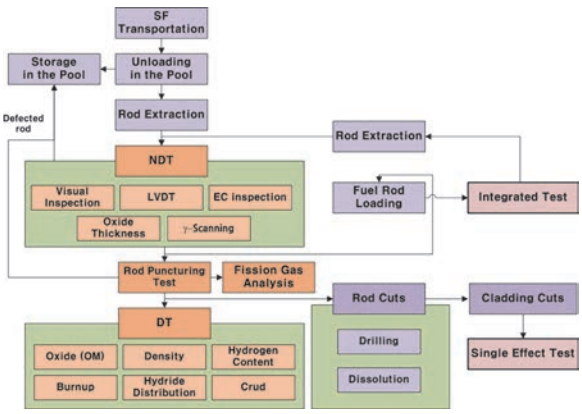

Fig.2. Characteristic test of the spent fuel rod.

3. BASIC DESIGN OF THE INTEGRATED TEST APPARATUS

Figure 3 shows basic design of the integrated test apparatus. It has key design concepts as follows;

- Installation in the pool
- Long-term high temperature operation
- Several meter remote handling
- Minimum radioactive shielding
- Inspection of rod integrity (cover gas sampling)
- Precise temperature control
- Vessel drying in the pool
- Seismic resistance
- Emergency stop system
- Helium environment

And also this apparatus will be installed inside the pool with 15 m depth. Because of that, remote handling is very important issue.

In order to measure and control temperature inside vessel about 50 thermocouples are installed. Dispersive and indirect heating method was adopted. Allowable pool water temperature of PIEF is 50 °C. CFD analysis was carried out to meet the criterion and to calculate the cladding temperature.

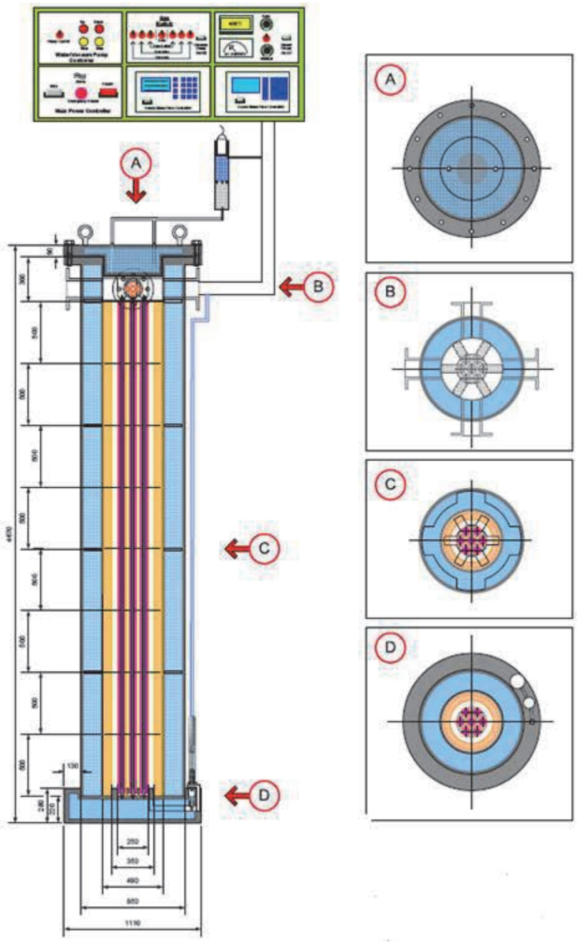

Fig.3. Basic design concept of the integrated test apparatus.

4. INSTALLATION PLAN

We have a plan to install the apparatus in the pool of PIEF. Advantages of installation in the pool are to reduce shielding thickness of the vessel, to secure the thermal safety by pool water and to utilize the space of PIEF. Technical issues of the installation are as follows;

- Underwater remote handling
- Water proof
- High temperature underwater instrumentation
- Structural integrity of the installation structure

Underwater performance test will be done before the installation performance test. It includes endurance and airtight verification, remote handling test, assembling and inspection. We will construct the test facility attached to the LWR fuel compatibility test facility, PLUTO of KAERI. The PLUTO can supply enough electric power, compressed air and demineralized water. In the test facility, the pool with about 5 m depth will be set up for underwater performance test.

5. RESEARCH SCHEDULE

This research project consists of 3 phases. Now is the first year of second phase. In the first phase, we reviewed former cases and set up the test plan. In the second phase, we selected target spent fuel and we are doing the basic and detailed design of the integrated test apparatus. In addition, second phase includes manufacture of core and instrument devices, performance test, thermal analysis and licensing. In third phase, the integrated test apparatus will be operated with six spent fuel rods. Table 2 summarizes the research schedule.

Phase 1	Phase 2			Phase 3
2009 2010	2011	2012	2013	2014 2015 2016
Review of Former Cases	SF Rods Selection	Core Devices Manufacture	Performance Test	
Feasibility Study of Integrated Test	SF Transportation	Instrument Devices Setting	Thermal Analysis	Integrated Test with 6 SF rods
Test Plan	Basic/Detailed Design	System Installation License		

Table 2. Research schedule

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7.4 Application of FE-SEM with Elemental Analyzer for Irradiated Fuel Materials

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ABSTRACT

It is important to study the irradiation behavior of the uranium-plutonium mixed oxide fuels (MOX fuels) for development of fast reactor fuels. During irradiation in a fast reactor, the changes of microstructures and the changes of element distributions along radial direction occur in the MOX fuels because of a radial temperature gradient. In order to make detailed observations of microstructure and elemental analyses of fuel samples, a field emission scanning electron microscope (FE-SEM) equipped with a wavelength-dispersive X-ray spectrometer (WDX) and an energy-dispersive X-ray spectrometer (EDX) were installed in a hot laboratory.

Because fuel samples have high radioactivities and emit α-particles, the instrument was modified correspondingly. The notable modified points were as follows.

1) To prevent leakage of radioactive materials, the instrument was attached to a remote control air-tight sample transfer unit between a shielded hot cell and the FE-SEM.

2) To protect operators and the instruments from radiation, the FE-SEM was installed in a lead shield box and the control unit was separately located outside the box.

After the installation, the microscopy and elemental analyses were made on low burnup fuel samples. High resolution images were obtained on the fuel sample surface. The characteristic X-rays (U, Pu) emitted from the fuel sample surface measured along radial direction successfully. Thereby, it was able to grasp the change of U, Pu radial distribution after irradiation. The technique has the great advantage of being able to evaluate the changes of microstructures and the changes of element distributions of MOX fuels due to irradiation. In future work, samples of even higher radioactivity will be observed and analyzed.

1. INTRODUCTION

It is important to study the irradiation behavior of the MOX fuels for design advancement of nuclear reactor fuels. The microstructure change of MOX fuels irradiated in a fast reactor occurs because of a radial temperature gradient[1]-[4]. Typical MOX fuels irradiated in a fast reactor, voids were sweeped towards the fuel center, formation of a central void, and columnar grain structure.

The distribution changes of fuel elements and fission product elements occur during irradiation. Thus, even more detailed observations and elemental analyses of the irradiated MOX fuel surfaces are required than ever before[3] [5] [6].

A FE-SEM, which uses a field emission gun, is capable of imaging at much higher magnification than typical SEM[7]. In addition, FE-SEM has an electron beam current of high stability which is needed for surface elemental analyses.

It was necessary to shield operators and devices from radioactivity of samples and to prevent leakage of radioactive materials (especially uranium and plutonium) when irradiated fuel samples are handled for examination. Thus, until now, irradiated fuel samples were handled in hot cells which were air-tight and shielded against radioactivity.

In this study, an overview of the installed FE-SEM which was modified for use with irradiated fuel samples and the results of observations and elemental analyses for irradiated fuel samples are reported.

2. EXPERIMENTAL INSTRUMENT

The new FE-SEM (JEOL JSM-7001F) was modified and installed in a shielded box. This FE-SEM provides high resolution observations (3.0nm at 15kV) and a highly stable electron beam current because it uses a thermal Schottky type field emission gun. In addition, the FE-SEM is equipped with WDX (Oxford Instruments INCA WAVE) and EDX(Oxford Instruments INCA X-act) for surface elemental analysis. Figure 1 shows a schematic diagram of the modified FE-SEM. So it can be used to observe fuel samples having high radioactivities and emitting α-particles, the instrument was modified as follows

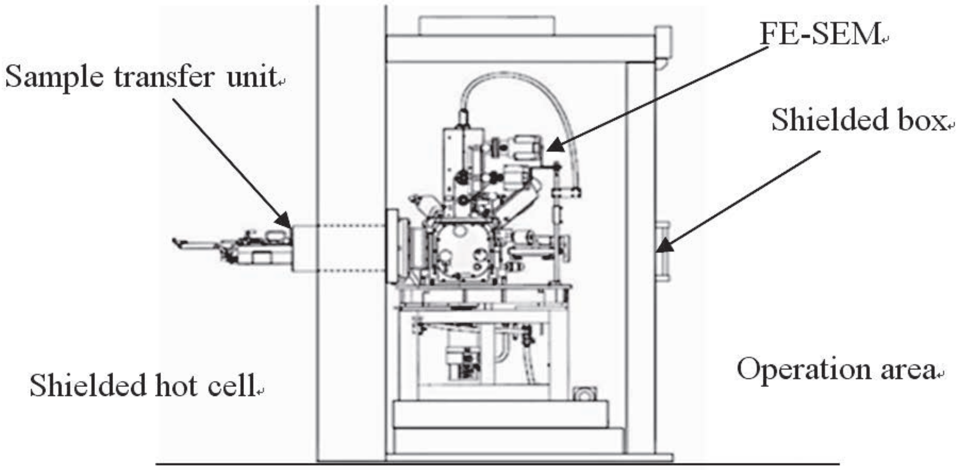

Figure 1 Schematic diagram of the modified FE-SEM.

1) Attachment of a remote control air-tight sample transfer unit

To prevent leakage of radioactive materials, the instrument is attached to a remote control air-tight sample transfer unit between the shielded hot cell and the FE-SEM. This unit is electric-powered and able to transfer a radioactive sample with a maximum size of Φ20mm×10mm. The boundary between the operation room and the shielded hot cell is made airtight by an O-ring seal. To maintain air-tightness, the vibration-free damper is removed from the FE-SEM. And the boundary between the sample chamber in the FE-SEM and the shielded hot cell is made air-tight by a gate valve which is capable of operating at a high vacuum. This unit can be operated manually to remove the irradiated fuel sample from the sample chamber by a remote handling jig and manipulator when the FE-SEM experiences some problem.

In addition, to prevent leakage of the radioactive materials, the exhaust pipes of this FE-SEM are connected with the exhaust pipes of the shielded hot cell. And FE-SEM is purged using pure nitrogen gas supplied from outside.

2) Installation of a lead shielded box and separate control unit of the FE-SEM

To protect operators and the instruments from radiation, the FE-SEM is installed in a lead shielded box. The thickness of the box walls is about 120mm. This lead shielded box has doors attached for maintenance of FE-SEM. These doors are locked when the radiation dose inside the shielded box is over 100μSv/h at inner dosimeter.

The control unit of the FE-SEM is separately installed outside the shielded box. Thus, the acceleration voltage, magnification, astigmatism, and sample stage position can be changed from outside the shielded box. A CCD camera is installed in the sample chamber of the FE-SEM to monitor the operation of sample exchange.

3. RESULTS OF EXAMINATION AND DISCUSSION

The short-term irradiated fuel samples were examined by the modified FE-SEM. These samples were MOX fuels, included 27wt% Pu. Maximum burnup of these samples was 0.332MWd/t (0.05Atomic %), maximum sample dosage rate of these samples was 16mSv/h. These samples were cut about 5mm length and mounted on holder (Φ20mm×10mm) with epoxy resin. After mounting, these samples were polished by emery papers and diamond pastes in the shielded hot cell, ultrasonic cleaned with the kerosene and the ethanol, and coated with carbon. After preparation, the sample was transferred from the shielded hot cell to the FE-SEM. The sample was observed and analysed by the FE-SEM. After observation and elemental analysis, the sample was transferred from the FE-SEM to the shielded hot cell. During their transfer, the dose rate outside of the shielded box was under 1μSv/h (the dose limit of the operation area is 20μSv/h). The air-tightness of the shielded hot cell and the FE-SEM were maintained. There were no leaks of radioactive materials from the shielded hot cell and the FE-SEM.

Figure 2 shows secondary electron images of the fuel sample surfaces. Good images of the fuel sample were taken at high magnification (2000X, 50000X) by the FE-SEM. It was possible to observe the face of grain boundary in detail(Dark area is crack area). The acceleration voltage, magnification, astigmatism, and sample stage position could all be changed by remote control during the fuel sample observation.

As an example of the elemental analysis, Figure 3 shows the results of the fuel sample obtained by WDX in near cladding tube area and near central void area. The characteristic X-ray peaks of U and Pu were successfully detected. In near central void area, the Pu peaks were higher than in near cladding tube area. On the other hand, the U peaks in near central void area were lower than in near cladding tube areas. It was able to grasp the change of distribution of U, Pu after irradiation.

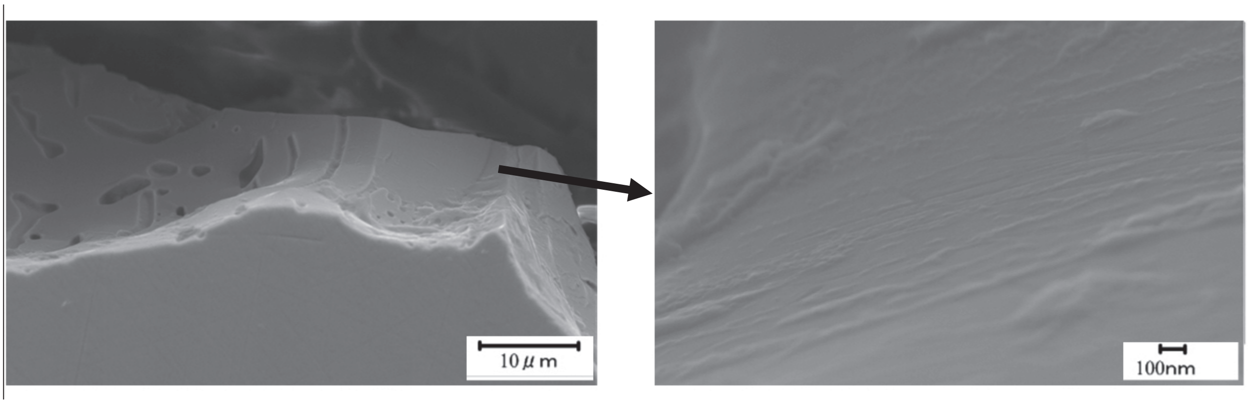

Figure 2 Secondary electron images of the fuel sample surface (Accelerating voltage, 15kV).

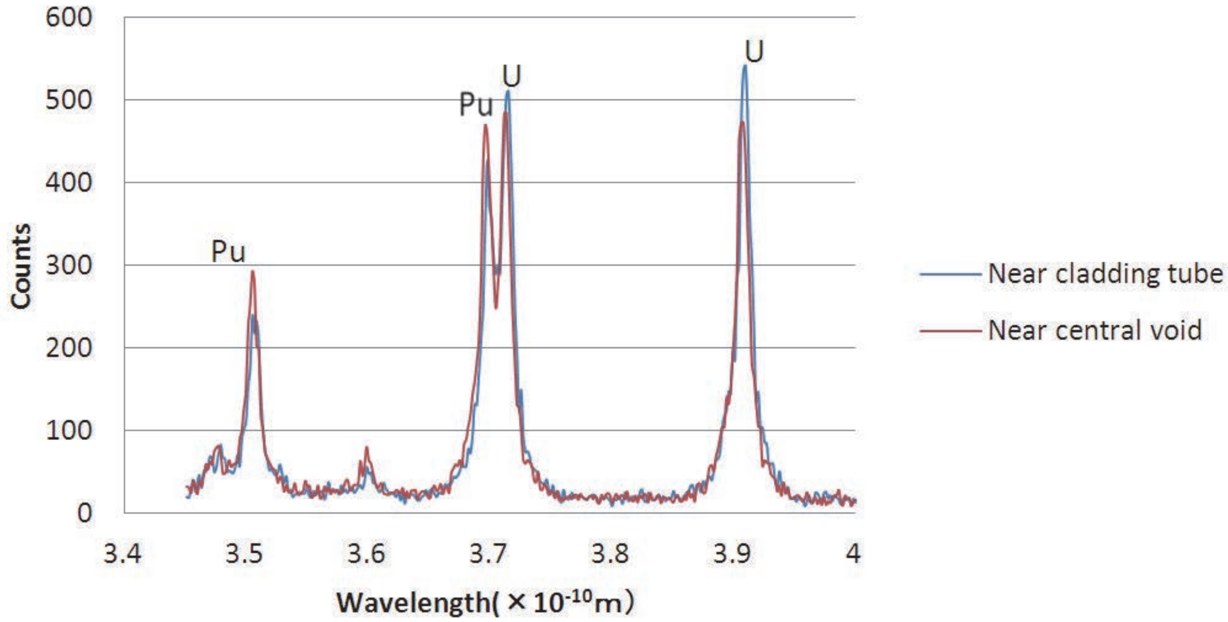

Figure 3 Elemental analysis results of the fuel sample by WDX (Beam current: about 30nA, Accelerating voltage: 20kV).

4. CONCLUSION

The newly modified FE-SEM could be used to carry out both observations of the micro structure and elemental analysis of irradiated fuel sample surfaces. The technique has the great advantage of being able to evaluate the irradiated fuels in detail. In future work, samples of even higher radioactivity will be observed and analyzed.

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7.5 Development of Single Effect Test Equipment for Integrity Evaluation of Spent Fuel Cladding during Long-term Dry Storage

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Degradation of fuel cladding integrity during dry storage is mainly caused by cladding creep and mechanical strength degradation with hydride reorientation. Two single effect tests (creep test, hydride reorientation test) were selected to evaluate the spent fuel integrity during dry storage. In order to evaluate creep behavior, internal pressure type multi-channel creep test equipment with 250mm length spent fuel cladding specimens was designed and is being manufactured. And for hydride reorientation test, a servo motor type tensile test machine with high temperature furnace was installed to apply a hoop stress and temperature to ring specimens. These two single effect test equipments are supposed to be installed at PIEF(Post Irradiation Examination facility) Hot Laboratory.

Keywords: Dry storage, spent fuel cladding integrity, single effect test, creep, hydride orientation

1. INTRODUCTION

Many spent fuel pools at nuclear power plants in Korea are near capacity, utility have to consider interim dry storage of spent fuel as an option for increasing spent fuel storage capacity. Therefore, the dry storage concept is becoming a major technical consideration for intermediate spent fuel storage at present in Korea. Among technical considerations, degradation of fuel cladding integrity during dry storage is mainly caused by cladding creep and mechanical strength degradation with hydride reorientation, where these factors are in turn affected by cladding overpressure, fuel decay heat, and storage time.

2. DEVELOPMENT OF MULTI-CHANNEL IRRADIATED CLADDING TUBE CREEP TESTER

One criterion for consideration of cladding integrity during dry storage is creep. To avoid degradation of cladding, the strain calculated to occur in storage should be less than the creep strain to failure. The two principal factors in the creep behavior of irradiated cladding are the hoop stress and the temperature. The hoop stress results from the rod internal pressure, and the temperature results from the decay heat of the fuel assemblies.

However Korea has not yet provided regulations of licensing limits for interim dry storage such as peak cladding temperature, internal pressure, strain limit, etc.

In order to make a cladding tube creep model for interim dry storage and validate the tube creep model, it is necessary to study the detailed correlations among creep strain(up to rupture of cladding tube), temperature, and hydrogen content.

Therefore internal pressure type multi-channel creep test equipment with 250mm length spent fuel cladding tube specimens was designed and is being manufactured to validate the tube creep model. This equipment is shown in figure 1.

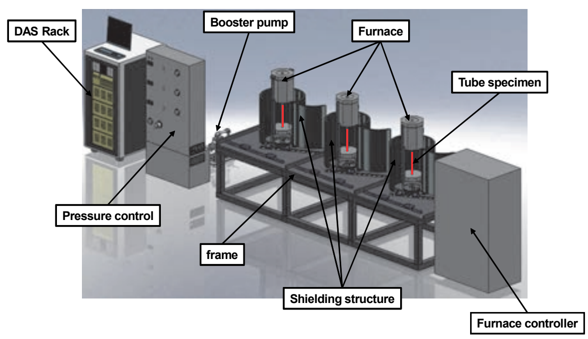

Figure 1. Multi-channel irradiated clad tube creep tester

This equipment consists of several parts. In order to provide a constant internal pressure using inert gas, booster pump and other mechanical fitting are equipped. Electric furnace to maintain a certain temperature is equipped. Pressure and temperature control system are also installed.

Especially, the shielded laser extensometer is equipped to measure a diametral creep strain of the irradiated cladding tube real time. Finally, a cylindrical lead shielding wall around the furnace is located to protect a person from radiation. The concept and schematic diagram of the shielded laser extensometer system are shown in figure 2.

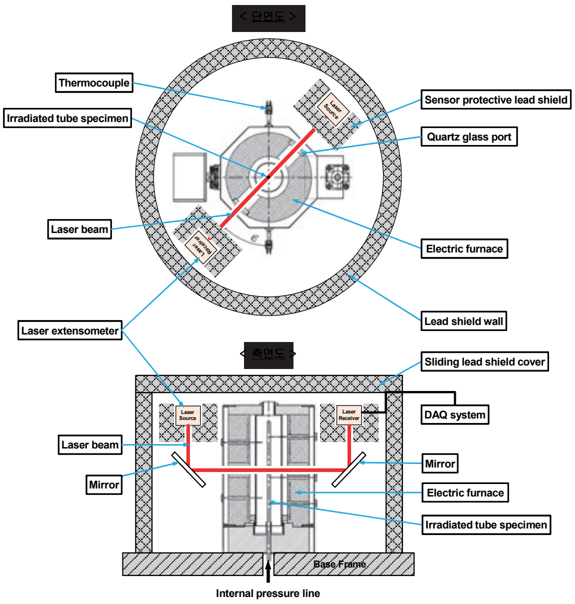

Figure 2. Measurement concept of diametral creep strain of irradiated cladding tube

3. HYDRIDE REORIENTATION TESTER

As a result of corrosion during irradiation the hydrogen concentration can increase to values in excess of 300 ppm (for higher burnup fuels, the concentrations may be considerably higher) and hence result in hydride formation and precipitation. In general, commercial spent fuel is manufactured with hydrides predominately oriented in the circumfer ential direction. However under sufficient stress during dry storage, hydrides will reorient to the radial direction. The radial hydride orientation is more detrimental than circumferential hydrides. Hydride reorientation may be an important mechanism adversely affecting spent fuel cladding integrity for spent fuel with higher burnup levels.

The effects of hoop stress and temperature in hydride reorientation are shown in figure 3. The stress limits for hydride reorientation appear to decrease with an increasing temperature.

In this study, a servo motor type tensile test machine with high temperature furnace was installed to evaluate the effect of temperature and hoop stress on hydride reorientation of irradiated cladding specimen.

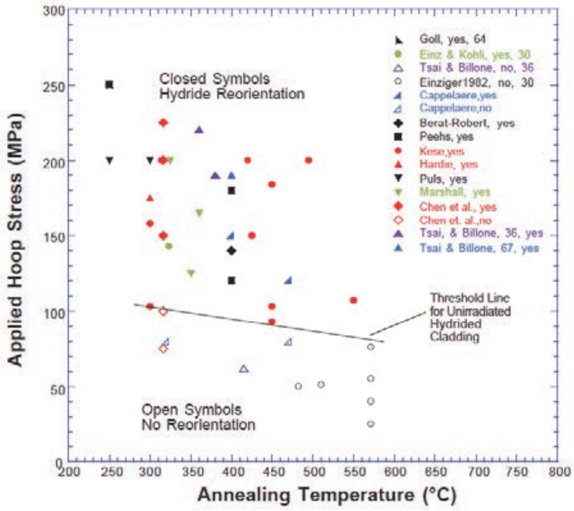

Figure 3. NRC Evaluation of hydride reorientation, Einziger et al., 2005.

Hydride reorientation tester is already installed in PIEF hot laboratory and performance test is being carried out now. This equipment is shown in figure 4.

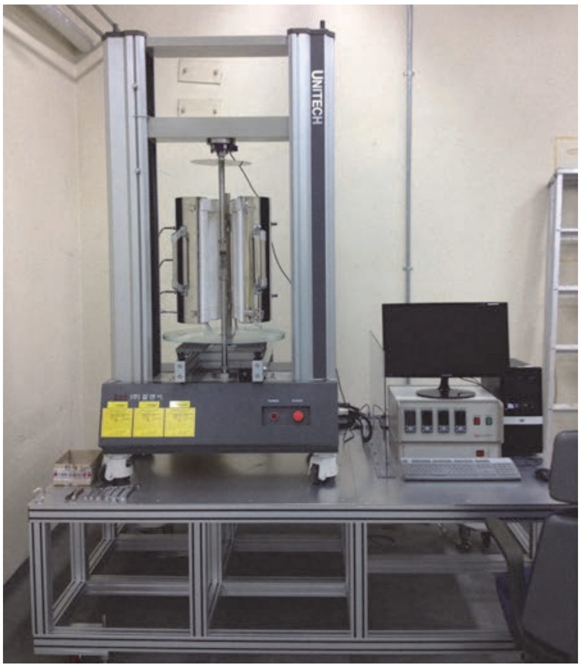

Figure 4. Hydride reorientation tester in PIIEF hot Lab.

4. CONCLUSIONS

Two single effect test equipments are supposed to be installed at PIEF(Post Irradiation Examination facility) Hot Laboratory. The preliminary data necessary for determining the criteria of spent fuel cladding integrity during dry storage will be provided through the single effect tests

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7.6 Post Irradiation Examination Technology Exchange

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Under the KAERI and JAEA agreement, in a part of the program 18 (Post Irradiation Examination (PIE) and Evaluation Technique of Irradiated Materials), an eddy current test was proposed as a round robin test, and it has been being progressed in both organizations in order to enhance the post irradiation examination technology. Up to now, several data are obtained by both PIE facilities. In this paper, the round robin test program is shown, and also shown obtained data with discussion from applicability as a nondestructive test in the hot cell.

Keywords: Eddy current test, Fatigue crack, Post irradiation examination, Internal defect, External defect, Zircaloy tube, Encircling coil, Probe coil

1. INTRODUCTION

Under the JAEA and KAERI agreement, technical information has been exchanged on Post Irradiation Examination (PIE) technologies such as recycling technique of an irradiated capsule, nanometer-level analysis technique, which is necessary to understand the irradiation defect behavior, telemetry technique.

For above activity, in a part of the program 18 (PIE and Evaluation Technique of Irradiated Materials), an eddy current test was proposed as a round robin test, and has been being progressed in both organizations in order to enhance the post irradiation examination technology. In the test, a standard sample was prepared, and the eddy current data of the sample were obtained by instruments in both hot laboratories. Then, the obtained data were compared. In this paper, the round robin test program is introduced, and shown obtained data with discussion from applicability as a nondestructive test in the hot cell.

2. TEST PROCESS

The examination program for the eddy current test is shown in **Table 1**.

As for the schedule, after discussion by both sides, manufacture of the test samples was carried out in JAEA side until early in 2010. Defects were introduced in test samples in both external and internal surfaces of the Zircaloy tube with 9.5mm outside diameter by machining.

Furthermore, in addition to the machined defect, the artificial fatigue crack was manufactured by the JAEA side.

We started the inspection by JAEA and KAERI using the prepared sample. At first, the inspection was carried out in the JAEA from the second half of 2010 to the beginning of 2011 using the prepared test sample. Then, the sample was transferred to KAERI, and then the inspection was carried out in 2011.

3. TEST TECHNOLOGY

3.1 Test Equipment

The test equipment in JAEA is made by HARA Electronics Co. Ltd., and model is FD-2203.

Equipment consists of a sample transfer part, a controlling element, and a record part. The sample transfer part is installed in a concrete cell, and the controlling element and the record part are installed out of the cell.

A test sample is perpendicularly held in the sample chuck mechanism arranged at the upper part of a sample transfer part, and also is led to the coil central part with a guide roller for centering of a coil and an test sample, and examined at the time of the rise of a sample chuck mechanism .

Moreover, a sample chuck mechanism can be rotated where a test sample is held, and is used as carried out to the direction of circumference defective distribution of the test sample using a probe coil.

The detector coil can choose an encircling type coil and a probe type coil according to the outside diameter of an examination sample, and the experimental purpose.

An encircling coil is used in order to detect the defective position and defective kind of test sample, and on the other hand, a probe coil is used in order to detect the direction of circumference defective distribution of a test sample.

Control panel is constituted from group of inspection frequency ch1 by groups of 16, 32, 64 and 128kHz, and ch2 by 128, 256, 512 and 1024kHz, and the inspection by single or double frequency is possible for it. The appearance of equipment is shown in **Fig. 1**.

3.2 Test Conditions

Though the general inspection using single frequency, this equipment can combine two frequencies of 16-1024kHz, and can inspect simultaneously.

Moreover, setup arbitrary in about 0.4 to the 25mm/sec range is possible for a test rate. The optimum condition of the test is shown below:

1) Used frequency
   Channel:1 64KHz
   Channel:2 256KHz
2) Scanning speed
   20mm/sec.
3) Used coil
   Encircling coil (inside diameter: 10.5 mm)

3.3 Manufacture of Test Sample, and Test Method

3.3.1 Test Sample Manufacture

In the test sample, external defects were introduced in three positions and internal defects were introduced in also three positions into a Zircaloy tube, outer diameter of 9.5 mm with 8.36 mm inner diameter, in the direction of a circumference with width at 3mm.

The defects were introduced in the Zircaloy tube samples at depth to thickness ratios of 5%, 10%, and 15%, respectively.

Table 2 shows inspection data of defects, and **Fig. 2** shows the test sample photograph.

Preparing fatigue cracks were targeted as more natural form generated by fatigue crack growth phenomenon. Cracks were produced by fracture toughness measuring device in a hot laboratory. Cracks were prepared changing the conditions of fatigue crack generation. Crack preparing conditions are shown in **Table 3**.

The crack test sample was manufactured by the special implement which imposes load for a Zircaloy tube up and down by maximum load at 1.5kN with about 65,000 times of fatigue cycle.

In **Fig.3,** the outside photograph of a fatigue crack manufactured in the Zircaloy tube is shown.

3.3.2 The Test Method of a Test Sample

The test sample was made by the connection with the Zircaloy tube, which has a fatigue crack in the length of manufactured, and another Zircaloy tube, which has machined defect with an insulator, as one test sample.

All measurements can be carried out in the same conditions by above so that the evaluation after the test can be performed easily.

The drawing of sample is shown in **Fig.4**.

Moreover, in KAERI, measurements were carried out using the sample with a penetration hole (0.5 mm of outside diameters) in equivalent Zircaloy tube, and connecting with these. Here, this penetration hole is simulated the penetration of the fatigue crack.

3.3.2 The Distinguish Method of Defects

The defects were judged by comparison with a standard defective sample. Phase analysis (X-Y pattern figure creation) of the remarkable signal portion on a data chart by contrast with the data of a standard defective sample is carried out to distinguish the kind of defect. The distinguish method of a defect is shown in **Fig.5**.

4. RESULT

4.1 Defect Detection Position

Inspection data and detection signals of the test sample were compared, and the detection position of the defect was evaluated.

1) The test chart of a test sample is shown in **Fig.6**. For external defects ①, ② and ⑤, they were detected at the manufactured position. The sample of X-Y patterns for the external defect ① and for the internal defect ③ are shown in **Fig.7**. For the internal defect ③, it was detected at the manufactured position. However, for the internal defect ⑥, distinction of defect is difficult. For the internal defect ④, it was detected far from the manufactured position.

2) For the fatigue crack signal, it has detected in manufactured position.

4.2 The Form and the Size of Defects

1) For the external defects ① and ②, and the internal defects ③ and ④, they had clear detection signals. These defects are more than 10% d/t ratio, which is the ratio of depth to Zircaloy tube thickness. However, for internal defect ④, although the d/t-ratio is 10%, it is not detected clearly at the manufactured position; it is presumed that the reason is the processed defect position, which was near the target position.

2) For the external defect ⑤ and internal defect ⑥, they had defective signals, however, it is difficult to judge to be a defect owing to slightly larger signal than the noise level.

3) For the fatigue crack defect ⑦, the crack length from visual measurement is about 8mm. However, it was not able to be identified whether the crack penetrated to the inside of a Zircaloy tube or not. In order to distinguish the kind of defect from the

detected signal, phase analysis was carried out by the X-Y pattern processing. The X-Y pattern of a crack part is shown in **Fig. 8**. From result of a test, crack is identified as a penetration hole, and it is guessed that the crack has a penetration form in the Zircaloy tube. Moreover, it was guessed from the X-Y pattern that the crack form is complex such as overlapped one. The metallography of center part of the crack part is shown in **Fig. 9**. The photograph shows that the penetrated crack and the crack from inside have arisen. As for the size of a crack, it is difficult to presume by the obtained data, because the standard crack defect as a the penetrated crack does not existing.

5. CONCLUSION

Using the Zircaloy tube, thickness of 0.57mm with outside diameter of 9.5mm, the followings were confirmed as a result of the eddy current test.

1) By the equipment of KAERI and JAEA, defects were detected clearly at d/t-ratio (depth to Zircaloy tube thickness ratio) 10% and 15% of the external defect, 10% and 15% of the internal defect.
2) For 5% d/t-ratio of the internal defect, detection was difficult by the equipment of KAERI and JAEA.
3) In detection of crack, both have detected clearly. Since the manufactured crack had penetrated to inside, it was identified as a defective signal of the penetration hole.

These examinations were the first trials. For future investigation, nondestructive examinations are being planned to conduct for the purpose of the advancement and standardization of the PIE testing technology between KAERI and JAEA.

REFERENCE

[1] Takashi IWAI, shizuo SOUZAWA, Seiichi MIYATA, Haruyuki SAKAI and Atsushi SAKAKURA, “Result of eddy current test for second round robin by HALDEN reactor project”,Aug.1986, JAERI-M (86-126).

[2] P.S.Fraser, J.A. Simpson, “The Eddy Current Testing of Zircaloy Tubes Containing Defects Prepared by OECD at Halden,Norway”, AEEW -M1897, United Kingdom Atomic Energy Authority.

[3] Takashi IWAI, Masahiro Outi, Isao OWADA, Kazuaki YANAGISAWA, “Report of the workshop meeting on eddy-current testing of Zircaloy tubes”, JAERI-memo59-015.

Table 1. Eddy current test program as round robin test.

Item/F.Y	2009	2010	2011	2012
1. Plannning				
2. Discussion by both side				
3. Specimen manufacture				
4. Fatigue crack manufacture				
5. Testing by JAEA				
6. Testing by KAERI				
7. Evaluation of results				
8. Making report				

Table 2. Defect of eddy current inspection data for Zircaloy tube.

Defect No.	Type	Dimension(l×w)(mm)	depth/thicness
1	External defect	0.22×3.08	0.082(15%)
2	External defect	0.21×3.04	0.059(10%)
3	Internal defect	0.22×3.01	0.081(15%)
4	Internal defect	0.22×3.02	0.053(10%)
5	External defect	0.22×3.00	0.027(5%)
6	Internal defect	0.21×3.02	0.026(5%)
material:Zircaly-4		Tube:OD:9.5mm ID:8.36mm	

Table 3. Crack manufacture condition.

Test temperature	Room temperature
Load(maximum)	1.5kN
Load(minimum)	0.5kN
The number of times	64,500

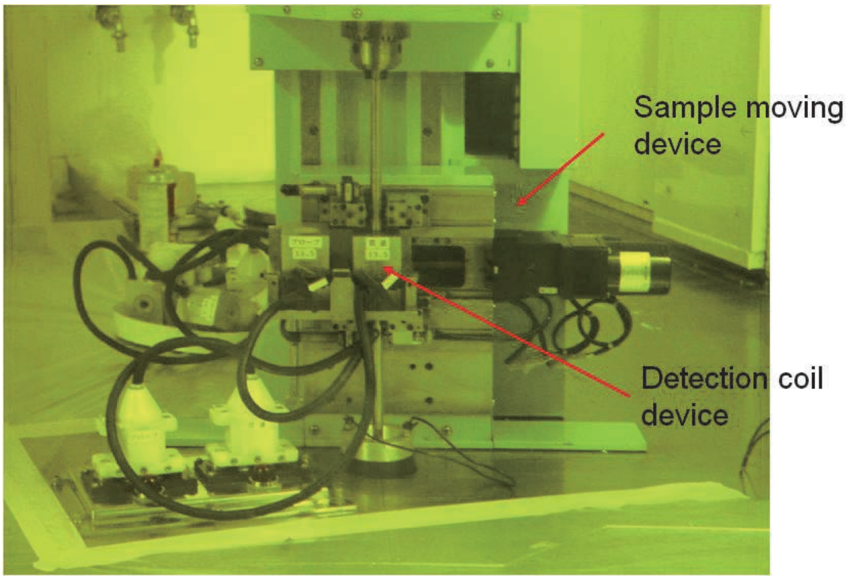

Fig.1 In cell device of eddy current test equipment.

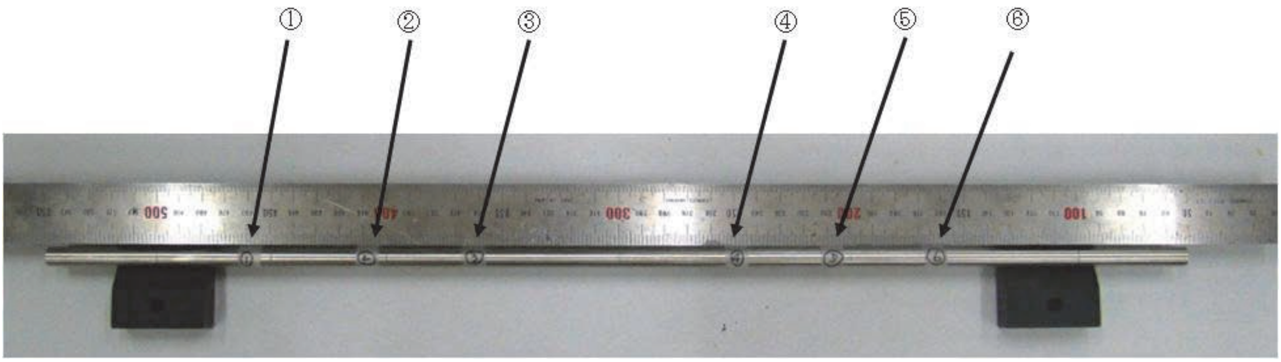

Fig.2 Photograph of Zircaloy tube.

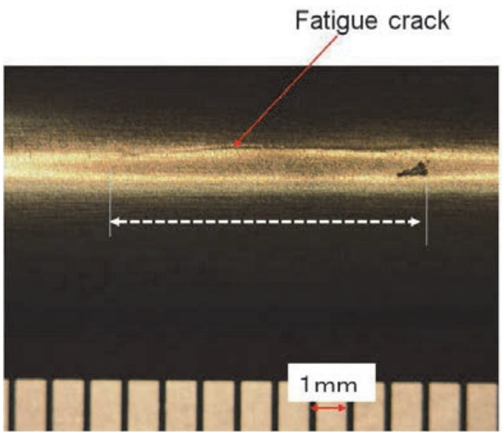

Fig.3 Photograph of fatigue crack Appearance.

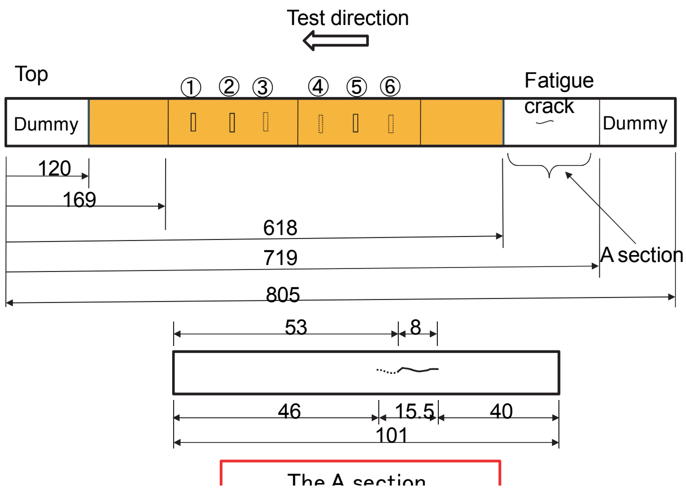

Fig.4 The induced defect of induced test sample.

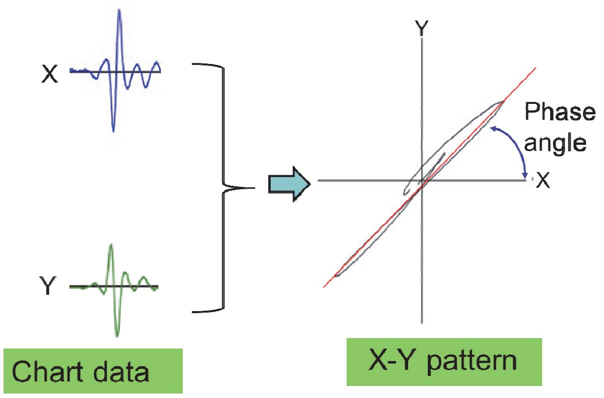

Fig. 5 The relation of data and X-Y pattern.

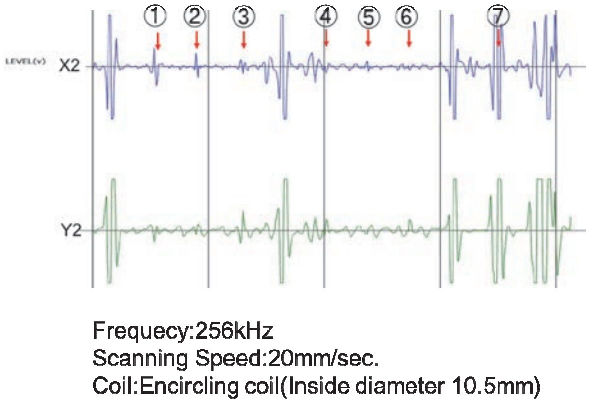

Fig. 6 A sample of output chart data.

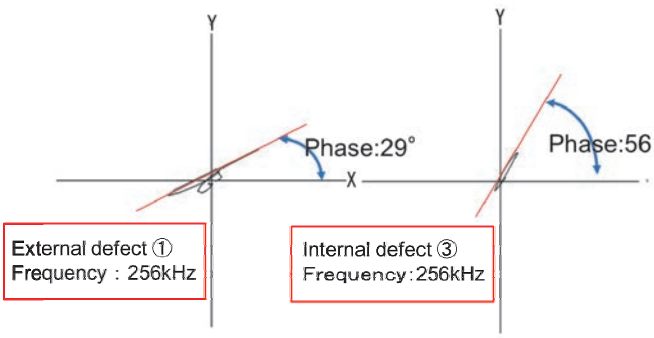

Fig. 7 Pattern of standard defect.

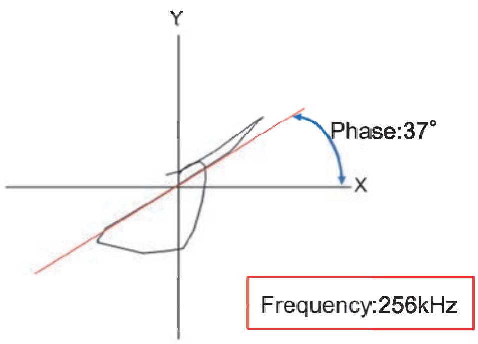

Fig. 8 Fatigue crack of X-Y pattern.

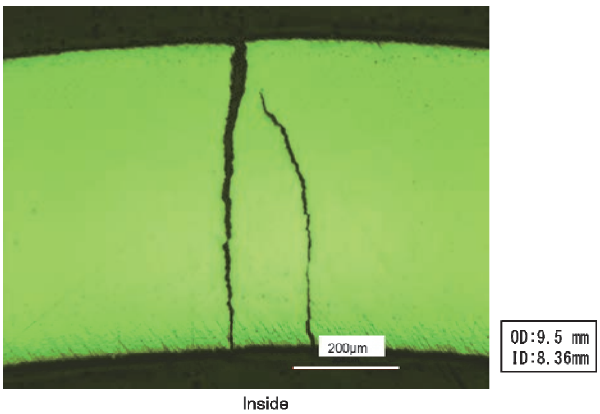

Fig. 9 Fatigue crack of Metallography.

8. Closing Address

Masahiro Ishihara
Deputy Director,
Neutron Irradiation and testing Reactor Center,
Japan Atomic Energy Agency

At the closing of the "2012 JAEA/KAERI Joint Seminar on Advanced Irradiation and PIE Technologies", I would like to express my gratitude to many attendees from KAERI as well as JAEA.

In this joint seminar total 37 presentations, 17 presentations from KAERI and 20 from JAEA, were made in the field of research reactor management, advanced irradiation technology and post irradiation technology, and active information exchanges were carried out with 56 participants, 21 persons from KAERI and 35 from JAEA.

In the final session, we made a discussion on future cooperation activities divided into three fields with facing discussion among counter-persons, and detail activities are proposed as a result. By conducting these proposals with closer cooperation, I think that the more active research & developments (R&Ds), which are useful as well as valuable for nuclear programs in both organizations, will be carried out.

As mentioned opening address, the JMTR is under the safety evaluation process after the 3.11 great-earthquake, which was occurred at off the Pacific cost of Tohoku, and its restart will be planned in 2012. After restart, the JMTR will operate at least twenty years around 2030. In the next joint seminar, which will be planned to hold 2014 in KAERI, we will report the experiences how to restart after the great-earthquake disaster.

Finally, I must mention that this seminar was supported by chairpersons, track leaders, secretaries and concerning persons in KAERI as well as JAEA. Here, I express my sincere gratitude to all contributed persons for their assistance of this joint seminar.

I expect that every participant will succeed in R&D activities, and also I wish every participant's good health. In the next seminar, 2014, we would like to meet again in KAERI, and we will make an active information exchange again.

付　録　　会議プログラム

2012 JAEA/KAERI Joint Seminar on Advanced Irradiation and PIE Technologies, March 28-30, 2012, Mito, Japan

PROGRAM

Wednesday 28th Mar., 2012			
9:30 - 10:00	**Registration**		
10:00 - 10:10	**Opening**	S.Suzuki H.S.Jung	JAEA-Oarai, Director general KAERI, Director
Session 1 : Plenary Session		**Chairman:**	**M.Suzuki**
10:10 - 10:30	Current Status of JMTR	N.Hori, M.Kaminaga, T.Kusunoki, M.Ishihara, Y.Komori, M.Suzuki	Neutron Irradiation and Testing Reactor Center, JAEA
10:30 - 10:50	Current Status of HANARO	C.S.Lee, K.H.Lim, H.S.Jung	HANARO Reactor Management division, KAERI
10:50 - 11:10	Current Status of JRR-3	M.Arai, Y.Murayama, S.Wada	Department of Research Reactor and Tandem Accelerator, JAEA
11:10 - 11:30	Recent and Future PIE Activities in KAERI	S.Ahn, W.S.Ryu, K.Hong, Y.Jeon, D.Kim, Y.Choo	PIE & Radwaste Division, KAERI
11:30 - 12:30	COMMEMORATIVE PHOTOGRAPH and LUNCH		
Session 2 : Research Reactor Management		**Chairman:**	**N.Hori** **C.S.Lee**
12:30 - 12:50	Safety Improvement Activities in HANARO	J.S.Wu, H.S.Jung, S.T.Hong, G.H.Ahn	HANARO management Division, KAERI
12:50 - 13:10	Support Required for Safety Management of JMTR in Extended Shutdown	S.Watahiki, T.Yamaura, T.Kusunoki	Neutron Irradiation and Testing Reactor Center, JAEA
13:10 - 13:30	Change of Tritium Concentration in Airtight Room by Tritium Removal System	S.P.Yim, J.H.Lee, S.J.Kim, H.K.Kim, J.W.Shin, H.S.Jung	KAERI
13:30 - 13:50	Conceptual Design of Multipurpose Compact Research Reactor	H.Nagata, T.Kusunoki, N.Hori, M.Kaminaga	Neutron Irradiation and Testing Reactor Center, JAEA
13:50 - 14:10	Status of Ageing Management Program for HANARO	S.J.Kim, J.W.Shin, H.K.Kim, H.Su.Jung	HANARO Reactor Management division, KAERI
14:10 - 14:30	Core Cooling during Flow Reversal in Downward FlowRR	C.Park, M.Kaminaga	Neutron Irradiation and Testing Reactor Center, JAEA
14:30 - 14:50	Status of Utilization of Beam Ports and Irradiation Holes in HANARO (2009~2011)	I.H.Kang, W.H.In, K.H.Lim, S.H.Lee, T.H.Kim	HANARO Reactor Management division, KAERI
14:50 - 15:10	BREAK		
Session 3 : Irradiation Technology (1)		**Chairman:**	**K.Tsuchiya** **K.N.Choo**
15:10 - 15:30	Development of New In-pile Instrumentation at JMTR	A.Shibata, N.Kimura, M.Tanimoto, J.Nakamura, T.Saito, K.Tsuchiya	Neutron Irradiation and Testing Reactor Center, JAEA
15:30 - 15:50	Operation Status and Prospect of Radioisotope Production Facility in HANARO	M.Kim, H.S.Jung	Radioisotope Production Facility Team, KAERI
15:50 - 16:10	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	Nuclear Safety Research Center, JAEA
16:10 - 16:30	Present Status and Prospect of NTD in HANARO	S.J.Park, K.D.Kang, H.S.Jung	HANARO Management Division, KAERI
16:30 - 16:50	International Standarization of Instruments for Neutron Irradiation Tests	M.Tanimoto, M.Cho, A.Shibata, C.Lee, J.Nakamura, S.Park, K.Tsuchiya, K.Choo	Neutron Irradiation and Testing Reactor Center, JAEA
16:50 - 17:10	Design and Fabrication of LVDT for Irradiation Test of Nuclear Fuel and Material	C.Y.Lee, S.W.Yang, K.N.Choo	HANARO Development Division, KAERI
17:10 - 17:30	^{99}Mo-^{99m}Tc Production Development by (n, γ) Reaction in JMTR	H.Izumo, K.Nishikata, A.Kimura, M.Tanimoto, K.Tsuchiya, M.Ishihara, M.Kaminaga	Neutron Irradiation and Testing Reactor Center, JAEA

Thursday 29th Mar., 2012

Session 4 : Irradiation Technology (2)			Chairman:	K.Tsuchiya K.N.Choo
9:30 - 9:50	Instrumentation of Sensors on DCF Test Rig Fabrication	C.Y.Joung, J.T.Hong, H.Y.Jeong, S.H. Ahn		Department of Research Reactor Utilization and Development, KAERI
9:50 - 10:10	Thermal-Hydraulic Tests with Out-of Pile Test Facility for BOCA Development	S.Kitagishi, M.Aoyama, M.Tobita, Y.Inaba, T.Yamaura		Department of JMTR Operation, JAEA
10:10 - 10:30	Improvement and Utilization of Irradiation Capsule Technology in HANARO	K.N.Choo, M.S.Cho, B.G.Kim, C.Y.Lee, S.W.Yang, Y.T.Shin, S.J.Park, H.S.Jung		Research Reactor Engineering Division, KAERI
10:30 - 10:50	Out-pile Tests for Improved Type Rabbits in JMTR	S.Kitagishi, F.Isozaki, K.Takita, M.Aoyama, Y.Matsui		Department of JMTR Operation, JAEA
10:50 - 11:10	Measurement and Evaluation of Fast Neutron Flux of CT and OR5 Irradiation Hole in HANARO	S.W.Yang, K.N.Choo, S.K.Lee, Y.K.Kim		HANARO Management Division, KAERI and Hanyang University
11:10 - 11:30	Neutron Irradiation Tests for Beryllium Material Selection as Neutron Reflector in JMTR	K.Tsuchiya, M.Ito, S.Kitagishi, Y.Endo, T.Saito, Y.Hanawa, C.Dorn		Neutron Irradiation and Testing Reactor Center, JAEA

11:30 - 12:50 LUNCH

Session 5 : Post Irradiation Examination Technology (1)			Chairman:	S.Sozawa S.B.Ahn
12:50 - 13:10	Experimental Validation of Transmutation Behavior for U and Am Samples Irradiated under Fast Neutron Spectra Based on Chemical Analysis	T.Ohnishi, S.Koyama		Fuels and Materials Department, JAEA
13:10 - 13:30	Development of Mechanical Test Techniques for Structural Components of Irradiated PWR Fuel Assembly	D.Kim, S.J.Baik, B.O.Yoo, I.C.Kwon, K.H.Kim, Y.J.Sohn, W.S.Song, Y.G.Jin, S.B.Ahn, Y.B.Chun, Y.S.Choo		PIE & Radwaste Division, KAERI
13:30 - 13:50	Measurement Techniques of Magnetic Properties for Evaluation of Neutron Irradiation Damage on Austenitic Stainless Steels	I.Yamagata, S.Konno, T.Hayashi , S.Takaya		Fuels and Materials Department, JAEA
13:50 - 14:10	Development of X-ray CT Scanner System	Y.Kato, M.Ito, S.Sozawa, M.Yonekawa		Department of JMTR Operation, JAEA
14:10 - 14:30	Analysis of CRUD Flake using Shielded EPMA	Y.H.ung, B.Yoo, S.Baik, S. Ahn, Y.Choo		PIE & Radwaste Division, KAERI
14:30 - 14:50	Improvement of Center Boring Device for Irradiated Fuel Pellets	H.Shiina, K.Usami, Y.Kimura, N.Sakuraba, A.Harada, A.Onozawa, M.Nakata		Department of Hot Laboratories and Facilities, JAEA
14:50 - 15:10	Advanced Disassembling Technique of Irradiated Driver Fuel Assembly for Continuous Irradiation of Fuel Pins	S.Ichikawa, H.Haga, K.Katsuyama, K.Maeda, K.Nishinoiri		Fuels and Materials Department, JAEA

15:10 - 15:30 BREAK

Session 6 : Post Irradiation Examination Technology (2)			Chairman:	S.Sozawa S.B.Ahn
15:30 - 15:50	Preliminary Test of Thermal Diffusivity for Fuel Samples by Laser Flash Method	H.Kim, D.G.Park, S.H.Na, S.Ahn, Y.S.Choo		PIE & Radwaste Division, KAERI
15:50 - 16:10	Development of Capsule Assemble Device	Y.Tayama, Y.Kanazawa, S.Sozawa, K.Hayashi, K.Kawamata, Y.Sizuoka, S.Onizawa, T.Nakagawa, S.Iwamatsu		Department of JMTR Operation, JAEA
16:10 - 16:30	Integrated Test for Evaluating PWR Spent Fuel Integrity in Dry Storage Condition	H.M.Kwon, H.S.Seo, H.K.Lee, J.S.Ju, Y.Z.Son, D.H.Kook, Y.S.Yang, J.W.Choi, Y.S.Choo		PIE & Radwaste Division, KAERI
16:30 - 16:50	Application of FE-SEM with Elemental Analyzer for IrradiatedFuel Materials	S.Sasaki, K.Maeda, A.Yamada		Fuels and Materials Department, JAEA
16:50 - 17:10	Development of Single Effect Test Equipment for Integrity Evaluation of Spent Fuel Cladding during Long-term Dry Storage	J.Jang, W.Oh, H. Lee, Y. Hwang, Y.B.Chun, D.Kook, J.Choi, Y.S.Choo		PIE & Radwaste Division, KAERI
17:10 - 17:30	Post Irradiation Examination Technology Exchange	S.Sozawa, M.Ito, T.Nakagawa, T.Taguchi, H.K.Lee		Department of JMTR Operation, JAEA

Friday 30th Mar., 2012

Discussion of Arrangement		**Chairman:** M.Suzuki H.S.Jung Y.S.Choo
9:30 - 12:00	Discussion for future cooperation (1) Research reactor management (2) Irradiation Technology (3) Post irradiation Examination	All

12:00 - 13:00 LUNCH

13:00 - 17:00 **Technical Tour (JMTR, AGF)**

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	$1L=11=1dm^3=10^3cm^3=10^{-3}m^3$
トン	t	$1t=10^3$ kg

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/(m² 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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