3.8 QUANTITATIVE ANALYSIS OF GAMMA-RAY EMITTING RADIONUCLIDE IN REACTOR POOL WATER OF HANARO

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The species and concentrations of the radionuclide in the primary coolant of HANARO were analyzed by using the gamma-ray spectroscopy. The full-energy peak efficiency for the volume source was calibrated as a function of the photon energy for an HPGe detector. The primary coolant of HANARO was picked at the primary coolant purification system, and the water at the upper part of the reactor pool was taken at about 20 cm under the pool surface. In the primary coolant, the concentrations of Na-24, Mg-27 and Al-28 were much higher than those of other nuclide, and they were in $1\sim6\times10^6$ Bq/liter. Their origins were radiative reactions of aluminium used as the structure material and cladding of the nuclear fuel. The concentrations of Xe-138 and Xe-133 were relatively higher than those of other fission fragments. The source of the fission fragments in the coolant was the surface contamination of the nuclear fuel by uranium. Ar-41, Ce-141, Na-24 and Xe-133 were detected in the water at the upper part of the reactor pool. Na-24 was the main source of the pool top radiation level, and Xe-133 and Ar-41 were the main gaseous radionuclide released through the reactor pool surface.

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Origin of radionuclide in reactor pool water

- Activation of coolant water
  - $O^{16}(n,p)N^{16}$, $O^{17}(n,p)N^{17}$, $O^{18}(n,\gamma)O^{19}$
- Activation of dissolved materials in the coolant
- Emission of activated materials of reactor structure into the coolant
- From the nuclear fuel,
  - Uranium contamination on the fuel surface
  - Some defect of fuel (abnormal)
- Leakage of the irradiated target (abnormal)

In this work,

- ❑ Analysis on the origin of the radionuclide in the coolant water
- ❑ Measurements of the concentrations of the radionuclide in the coolant water by gamma-ray spectroscopy

- ❑ Determination of the contributions to the reactor pool water radiation in normal operation of reactor
- ❑ Exact and rapid detection of the abnormal situation of the reactor
- ❑ Important background data in research reactor operation

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Experimental condition

- ❑ Sample picking
  - Reactor power: 30 MW
  - Primary coolant: picking at the entrance to the primary coolant purification system
  - Water of reactor pool top: 30 cm under the pool surface
  - Volume of picked water: 100 cc
- ❑ Spectroscopy of the gamma-rays from the pool water
  - Detector: coaxial HPGe detector(15%)
  - Cooling time: 5 min. or several days
  - Efficiency calibration: Homogeneous radionuclide mixture in matrix - cylindrical bottle of outer diameter of 48 mm (100 cc, 0.97 g/cm^3)

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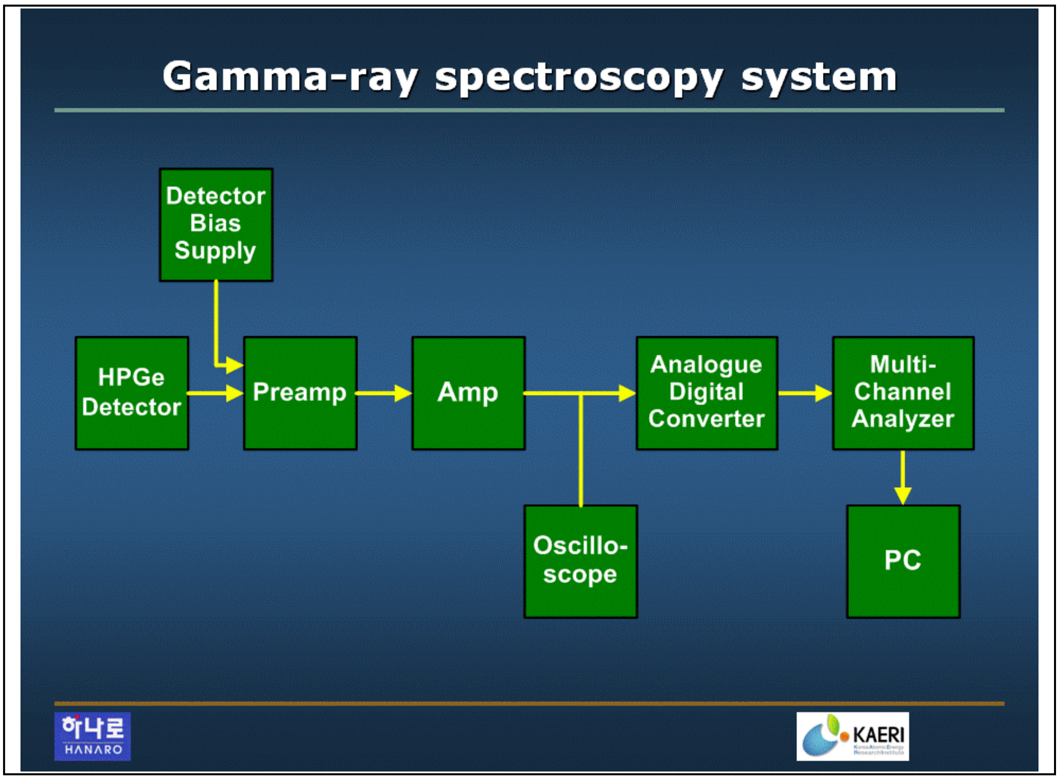
Gamma-ray spectroscopy system
Detector Bias Supply
HPGe Detector
Preamp
Amp
Analogue Digital Converter
Multi-Channel Analyzer
Oscillo-scope
PC
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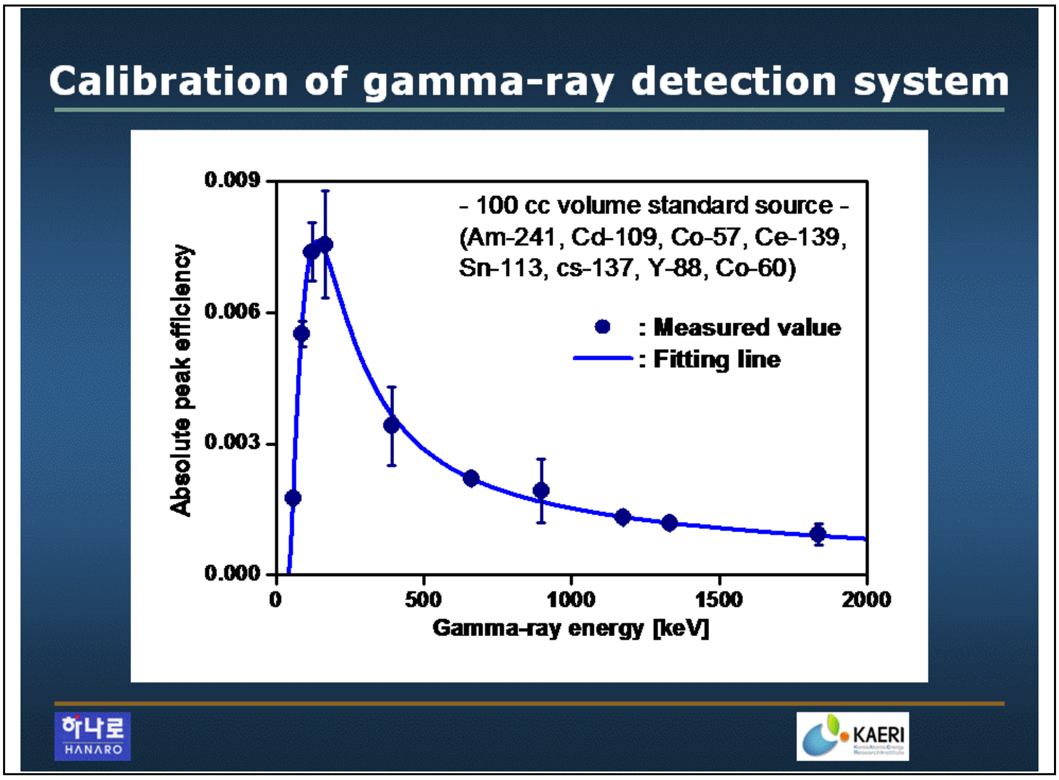
Calibration of gamma-ray detection system
- 100 cc volume standard source -
(Am-241, Cd-109, Co-57, Ce-139, Sn-113, cs-137, Y-88, Co-60)
: Measured value
: Fitting line
Absolute peak efficiency
0.009
0.006
0.003
0.000
0
500
1000
1500
2000
Gamma-ray energy [keV]
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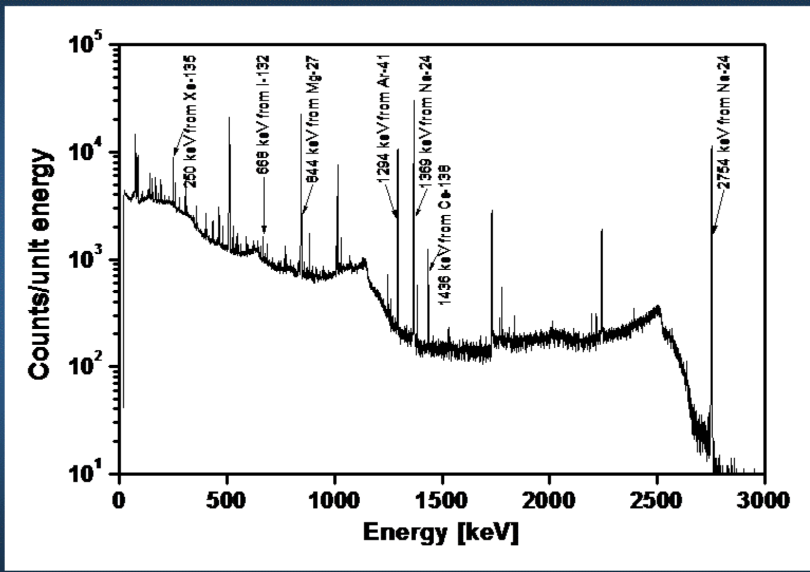
Gamma-ray spectrum from primary coolant



Concentration of radionuclide in pool water

Nuclide	Half-life (min.)	Main gamma-ray energy (keV)	Concentration (Bq/liter)	
			Primary coolant	Pool top water
Al-28	2.31	1778.8	1.75E+06	
Ar-41	109.8	1293.6	3.15E+05	2.51E+02-5.46E+02
Ba-139	83.06	165.8	1.38E+04	
Ba-141	18.267	190.22	1.58E+04	
Ce-141	46632.96	145.45		3.80E+02-9.12E+02
Cs-138	32.2	1435.86	2.56E+04	
I-131	11579.04	364.48		
I-132	142.8	667.69	7.17E+03	
I-133	1218	529.5	4.01E+03	
I-134	52.6	1072.53		
I-135	396.66	1260.41		
Kr-85m	268.86	151	1.52E+03	
Kr-87	76.4	402.7		
Kr-88	170.34	2392.11		
La-142	92.517	641.17		
Mg-27	9.462	843.73	5.25E+06	
Mn-56	154.56	1811.2	1.31E+04	
Mo-101	14.62	590.82		
Na-24	897.54	1368.55	1.54E+06	4.89E+03-7.83E+03

Concentration of radionuclide in pool water (cont.)

Nuclide	Half-life (min.)	Main gamma-ray energy (keV)	Concentration (Bq/liter)	
			Primary coolant	Pool top water
Nb-95	50364	765.82		~1.01E+03
Np-239	3391.68	103.7	6.13E+03	
Rb-88	17.8	1836	1.76E+04	
Rb-89	15.4	1031.88	3.09E+04	
Sr-91	580.14	555.57		
Sr-92	162.6	1383.9	7.48E+03	
Sr-93	7.3	590.9		
Tc-101	14.2	306.86	1.95E+04	
Tc-104	18.2	357.99	8.35E+03	
Tc-99m	361.14	140.51	4.73E+03	
Te-131	25	149.72	4.50E+03	
Te-131m	1800	852.21		
Te-132	4675.02	228.16	1.70E+03	~2.06E+02
Te-133	12.45	312.1	5.05E+03	
Te-134	41.8	565.99		
W-187	1434	685.74	3.20E+04	
Xe-133	7619.04	81	9.32E+03	6.19E+02~1.75E+03
Xe-135	544.98	249.79	6.90E+03	~3.36E+02
Xe-135m	15.6	526.81	1.19E+03	
Xe-138	14.13	258.31	2.45E+04	
Zr-95	92733.12	724.18		

Analysis on the origin of radionuclide in reactor pool water

- Activation of coolant water: not detected
  - $O^{16}(n,p)N^{16}$, $O^{17}(n,p)N^{17}$, $O^{18}(n,\gamma)O^{19}$
- Activation of dissolved materials in the coolant:
  - Ar-41: Activation of dissolved air
  - Na-24, Mg-27, Al-28: Activation of Al used as structure material and in the fuel
  - Mn-56: Activation of Fe in stainless steel used as structure material
  - Cr-51: Activation of Cr in stainless steel used as structure material
  - W-187: Activation of W used as welding rod
  - Zr related nuclide (Nb-95 etc.): Fission fragment or activation of Zr used in flow tube and fuel bundle
- Fission fragment (Iodine, Xenon, etc):
  - Uranium contamination on the fuel surface

Discussion

- In the primary coolant, the concentrations of Na-24, Mg-27 and Al-28 were much higher than those of other nuclide.
  - $1 \sim 6 \times 10^6$ Bq/liter
  - Used as the structure material and in cladding of the nuclear fuel
- The concentrations of Xe-138 and Xe-133 were relatively higher than those of other fission fragments.
  - Surface contamination of the nuclear fuel by uranium
- Ar-41, Ce-141, Na-24 and Xe-133 were detected in the water at the reactor pool top.
  - Na-24: main source of the pool top radiation level
  - Xe-133, Ar-41: main gaseous radionuclide released through the reactor pool surface

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Conclusion

- Determination of the species and concentrations of the radionuclide in the primary coolant of HANARO by using the gamma-ray spectroscopy
- Possibility of sensitive monitoring for fuel defect or leakage of irradiated sample

→ Utilization of this result as basic data for safe operation of the research reactor

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3.9 CURRENT STATUS OF IRRADIATION FACILITIES IN JRR-3, JRR-4 AND NSRR

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ABSTRACT

The Department of Research Reactor and Tandem Accelerator operates three research reactors, JRR-3 (Japan Research Reactor No.3), JRR-4 (Japan Research Reactor No.4) and NSRR (Nuclear Safety Research Reactor), which have been constructed in the Nuclear Science Research Institute. The reactors are utilized in the various fields and by in-house and outside users.

JRR-3 is a research reactor using low-enriched silicide fuel. The maximum thermal power is 20MW. It was operated for 180 days in Japanese Fiscal Year (JFY) 2007. The total numbers of irradiation capsules are 2,028. The neutron bender system was installed on the cold neutron guide tube. As a result, the cold beam intensity has increased by about 10 times.

JRR-4 is also a research reactor using low-enriched silicide fuel. The maximum thermal power is 3.5MW. It was operated for 93 days in JFY 2007. The total numbers of irradiation capsules are 521. Boron neutron capture therapy was carried out 25 times. The trouble of reflector occurred in December 2007. At present, the reactor has stopped to replace the reflectors.

NSRR is a TRIGA pulse reactor of annular core. It has been built for the investigation of light water reactor fuel behavior during off-normal conditions such as reactivity-initiated accident. The test fuel rod contained in an experimental capsule is loaded to the experimental cavity at the core center. Recently the investigation is performed for the behavior of high burnup fuel and mixed oxide fuel.

INTRODUCTION

The research reactors are used in neutron activation analysis, production of radioisotopes, neutron transmutation doping of silicon, medical irradiation, fuel and material irradiation, neutron beam experiments (neutron scattering experiment, neutron radiography, prompt-gamma-ray analysis), nuclear fuel safety research and others (training for reactor engineers, reactor engineering study, shielding experiments etc.).

The Department of Research Reactor and Tandem Accelerator operates three research reactors, JRR-3, JRR-4 and NSRR. These reactors were constructed in the Nuclear Science Research Institute. JRR-3 achieved the first criticality in 1962 as the first research reactor constructed with the homegrown technology. JRR-3 was modified for upgrade and resumed in 1990. JRR-4 was constructed to research the reactor shielding of the first Japanese nuclear ship “Mutsu” in 1965. JRR-4 was renewal for enrichment reducing in 1997. NSRR was constructed for nuclear fuel safety research in 1975.

UTILIZATION OF JRR-3

Purpose of JRR-3

- Neutron Beam Experiments
    - Neutron Scattering Experiment
    - Neutron Radiography
    - Prompt Gamma-ray Analysis
- Neutron Activation Analysis
- Production of Radioisotopes
- Neutron Transmutation Doping of Silicon
- Fuel and Material Irradiation
- etc.

Outline of JRR-3

JRR-3 is a high-performance, multi-purpose research reactor. The maximum thermal power is 20MW. JRR-3 is a light water moderated and cooled, pool type research reactor using low-enriched silicide fuel (LEU: approximately 20% enriched uranium).

The reactor core is situated at the bottom of the reactor pool, (8.5 m in depth, 4.5 m in diameter, the keyhole shaped horizontal cross section). A cylindrical reactor core is 60 cm in diameter and 75 cm in height. A heavy water tank, 200 cm in diameter and 160 cm in height, surrounds the reactor core.

A cold neutron source facility is installed in the heavy water tank at the horizontal tube 9C. In this facility, thermal neutrons pass through liquid hydrogen (~-200°C). The resultant neutrons have the energy reduced and are referred to cold neutrons. Cold neutrons have a long wavelength in comparison to thermal neutrons and are useful to study the structures of giant molecules.

Utilization facilities include irradiation facilities for using vertical irradiation tubes in the reactor core and the heavy water tank, and beam experimental facilities using horizontal experimental tubes in the heavy water tank. Both cold and thermal neutrons can be utilized as neutron beam. Neutrons are guided to the experimental building from the horizontal experimental tubes by the neutron guide tubes, which are rectangular glass tubes coated with a thin, nickel film. A variety of experimental devices are placed on the ports of neutron guide tubes.

One of the three cold neutron beam guide tubes has been upgrade by installing the neutron bender system. The system using newly developed neutron mirrors divides a neutron beam line into three beam lines. As a result, the cold beam intensity has increased by about 10 times.

A bird’s-eye view of the JRR-3's reactor room is shown in Figure 1, and JRR-3 reactor core is shown in Figure 2.

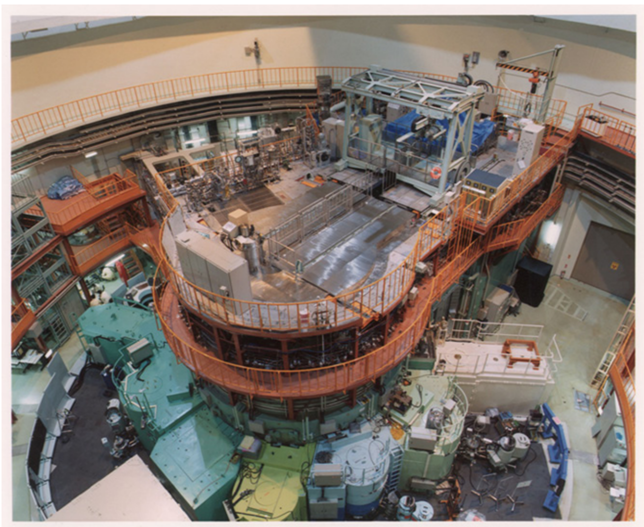


Figure 1 JRR-3's Reactor Room

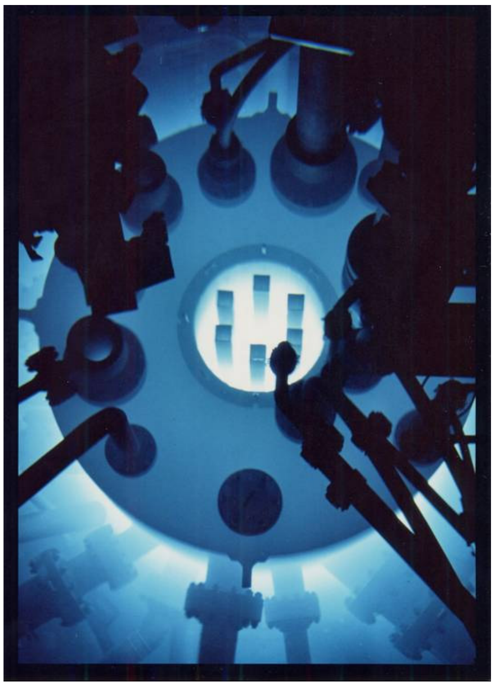


Figure 2 JRR-3 Reactor Core

Irradiation facilities and horizontal experimental tubes of JRR-3

The experimental tubes at JRR-3 are arranged as seen in the Figure 3. The specifications of JRR-3 irradiation facilities are listed in Table 1.

JRR-3 has nine vertical irradiation tubes (VT-1, RG-1, -2, -3, -4, BR-1, -2, -3, and -4) at the reactor core area. These are capsule irradiation facilities for a long-term irradiation (a month - several years). Instrumented and non-instrumented capsules are used for long-term irradiation tests of fuels and materials. One type of instrumented capsules can control the temperature of the specimen.

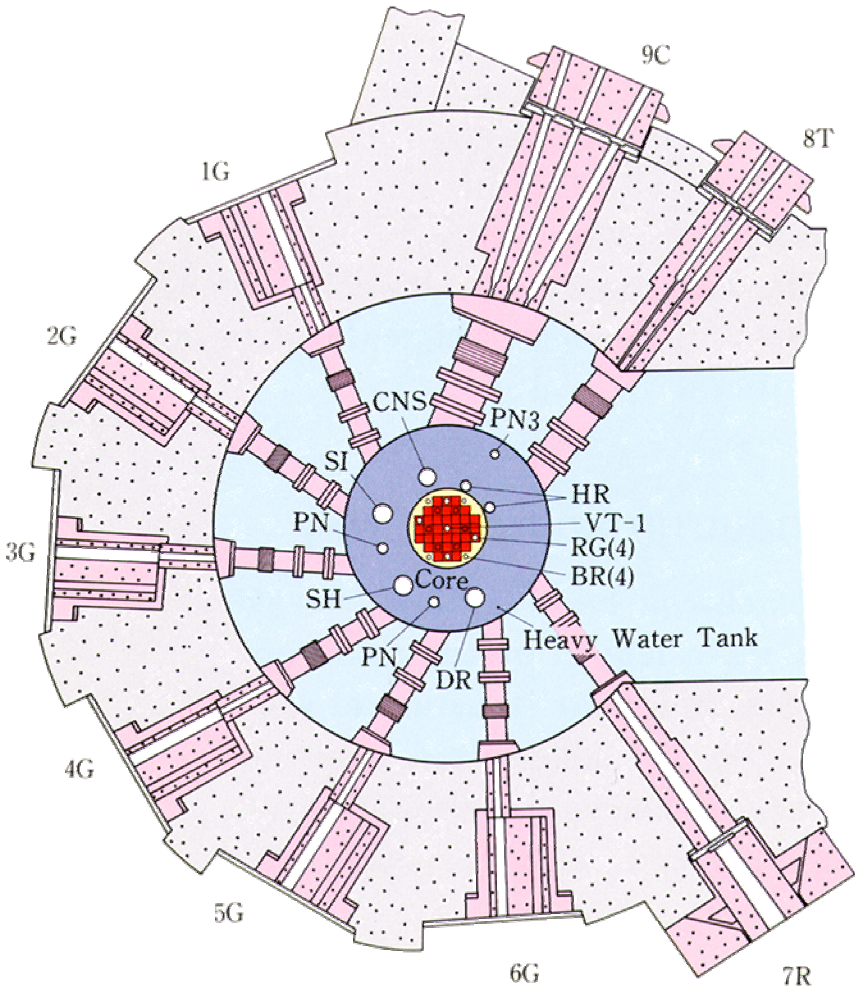


Figure 3 Arrangement of experimental tubes of JRR-3

Table 1 Specifications of JRR-3 irradiation facilities

Name	Neutron Flux ($m^{-2}s^{-1}$)		Application
	Thermal	Fast	
Capsule Irradiation (VT)	3.0×10^{18}	2.0×10^{18}	Exposure Test RI Production
Capsule Irradiation (RG)	2.0×10^{18}	1.0×10^{18}	
Capsule Irradiation (BR)	2.0×10^{18}	1.0×10^{18}	
Hydraulic Rabbit (HR)	9.8×10^{17}	1.7×10^{16}	RI Production, NAA
Pneumatic Rabbit (PN)	5.2×10^{17}	1.2×10^{15}	RI Production, NAA
Activation Analysis (PN-3)	1.5×10^{17}	4.4×10^{13}	NAA
Uniform Irradiation (SI)	2.0×10^{17}	-	Neutron Transmutation Doping of Silicon
Rotating Irradiation (DR)	3.0×10^{17}	-	Large Material Irradiation
Capsule Irradiation (SH)	4.0×10^{17}	-	Exposure Test RI Production
Horizontal Experimental Tube (1G – 6G, 7R, 8T, 9C)	1.2×10^{13} | 7.4×10^{13}	-	Neutron Scattering PGA, NRG

NAA: Neutron Activation Analysis
PGA: Prompt-Gamma-ray Analysis
NRG: Neutron Radiography

Irradiation tests together with post-irradiation examinations of fuels and materials contribute to the confirmation of their property change caused by neutron irradiation and to the research and development of fission and fusion nuclear reactor.

The irradiation tubes are two hydraulic rabbit irradiation facilities (HR-1 and -2), two pneumatic rabbit irradiation facilities (PN-1 and -2), one activation analysis irradiation facilities (PN-3), one uniform irradiation facility (SI-1: mainly using the production of semiconductors), one rotating irradiation facility (DR-1) and one capsule irradiation facility (SH-1). JRR-3 has nine horizontal experimental tubes (1G, 2G, 3G, 4G, 5G, 6G, 7R, 8T and 9C) that are needed for many kinds of neutron beam experimental facilities (neutron scattering experimental facilities, neutron radiography facilities , prompt-gamma-ray analysis devices etc.).

UTILIZATION OF JRR-4

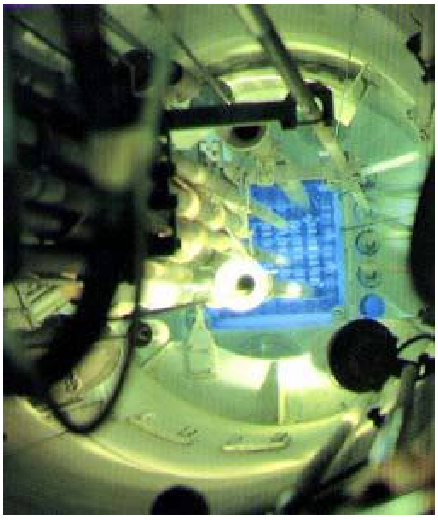
Purpose of JRR-4

- Neutron Activation Analysis
- Shielding Studies
- Nuclear Engineering Studies
- Irradiation Test
- Production of Radioisotopes
- Neutron Transmutation Doping of Silicon
- Education and Training
- Medical Irradiation
- Prompt Gamma-ray Analysis
- etc.

Outline of JRR-4

JRR-4 is the enriched uranium, light water moderate, coolant swimming pool type reactor. The maximum thermal power is 3.5MW. Fuel elements' enrichment of U-235 is about 20 wt%. Fuel core material is uranium-silicon dispersion alloy (U3Si2-Al). The reactor core is composed of twenty fuel elements, seven control rods (boron stainless steel) and is surrounded by reflector elements and tubes for irradiation experiments.

The reactor core and core configuration is shown in Figure 4 and Figure 5.



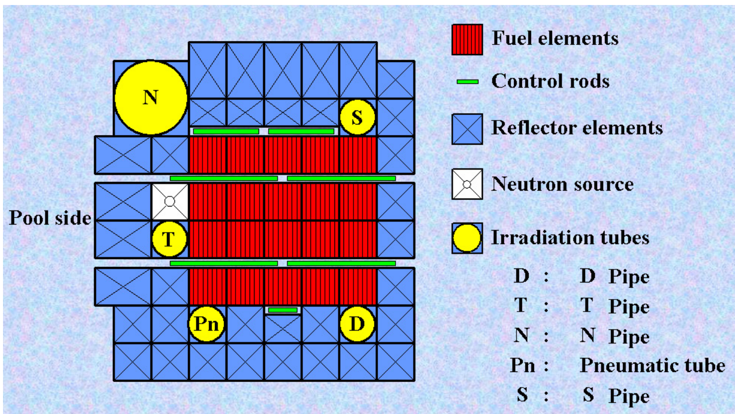


Figure 4 JRR-4's Reactor core

Figure 5 Reactor core configuration of JRR-4

Irradiation facilities of JRR-4

Specifications of JRR-4 irradiation facilities are tabled in Table 2. Medical neutron irradiation at neutron beam facility (NBF) is illustrated in Figure 6.

This reactor has five vertical irradiation tubes at the reactor core area, three capsule irradiation facilities (N pipe, S pipe, D pipe), one hydraulic rabbit irradiation facility (T pipe), and one pneumatic rabbit irradiation facility (Pn). There is a neutron beam facility, and it has used neutron beam experiments, irradiations for activation analysis and boron neutron capture therapy.

Boron neutron capture therapy was carried out 25 times in JFY 2007. Since the first medical irradiation for boron neutron capture therapy of JRR-4 was carried out on October 25th 1999, ninety nine times of boron neutron capture therapy irradiation were performed at JRR-4 until end of 2007.

The trouble of reflector was occurred in December 2007. At present, the reactor has stopped to replace the reflector.

Table 2 Specifications of JRR-4 irradiation facilities

Name	Neutron Flux Thermal ($m^{-2}s^{-1}$)	Application
Capsule Irradiation Facilities (N, D, S)	$(1.5 - 4.3) \times 10^{17}$	Neutron Transmutation Doping of Silicon Material Irradiation
Hydraulic Rabbit (T)	5.3×10^{17}	RI Production
Pneumatic Rabbit (Pn)	3.2×10^{17}	Neutron Activation Analysis
Neutron Beam Facility	2.0×10^{13}	Boron Neutron Capture Therapy (BNCT) Neutron Beam Experiment

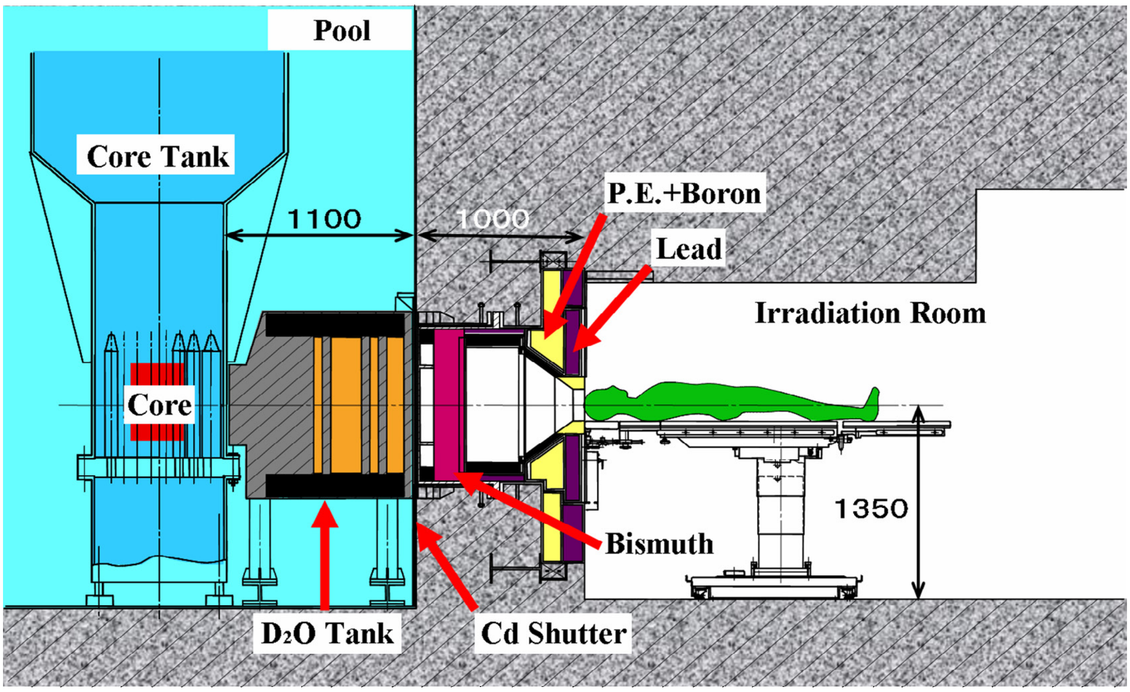


Figure 6 Medical irradiation at the neutron beam facility

IRRADIATION CAPSULES

Polyethylene Capsules

Polyethylene capsules are mainly used for short-term irradiation in the pneumatic tube (PN) in JRR-3 and the hydraulic irradiation facility (T pipe) in JRR-4. The maximum irradiation time is 20 min. in the PN and 40 min. in the T pipe. (See Figure 7)

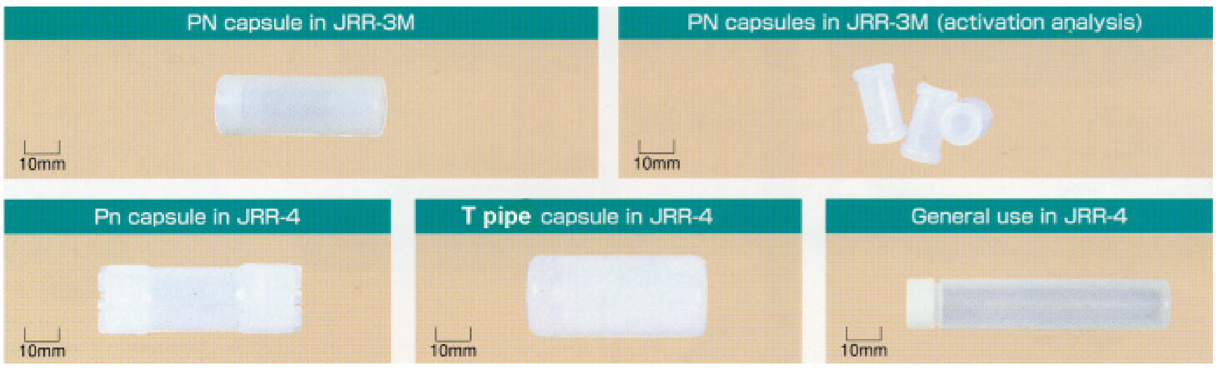


Figure 7 Polyethylene Capsules

Aluminum Capsules

Aluminum Capsules are mainly used for long-term irradiations in the HR in JRR-3 and the hydraulic rabbit irradiation facility (T pipe) in JRR-4. (See Figure 8)

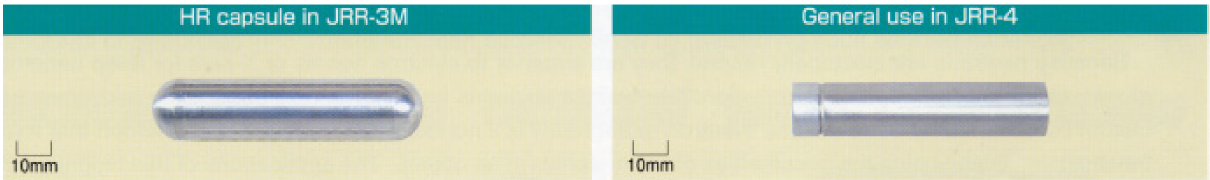


Figure 8 Aluminum capsules

STATUS OF UTILIZATION OF CAPSULES IRRADIATION

JRR-3 was operated for 180 days in JFY 2007. The total numbers of irradiation capsules are 2,028. JRR-4 was operated for 93 days in JFY 2007. The total numbers of irradiation capsules are 521. The status of utilization of capsule irradiation in JFY 2007 is presented in Figure 9. Irradiations for Neutron activation analysis are about 86% of the total capsule numbers of 2,546 capsules, Silicon Irradiations are about 8%.

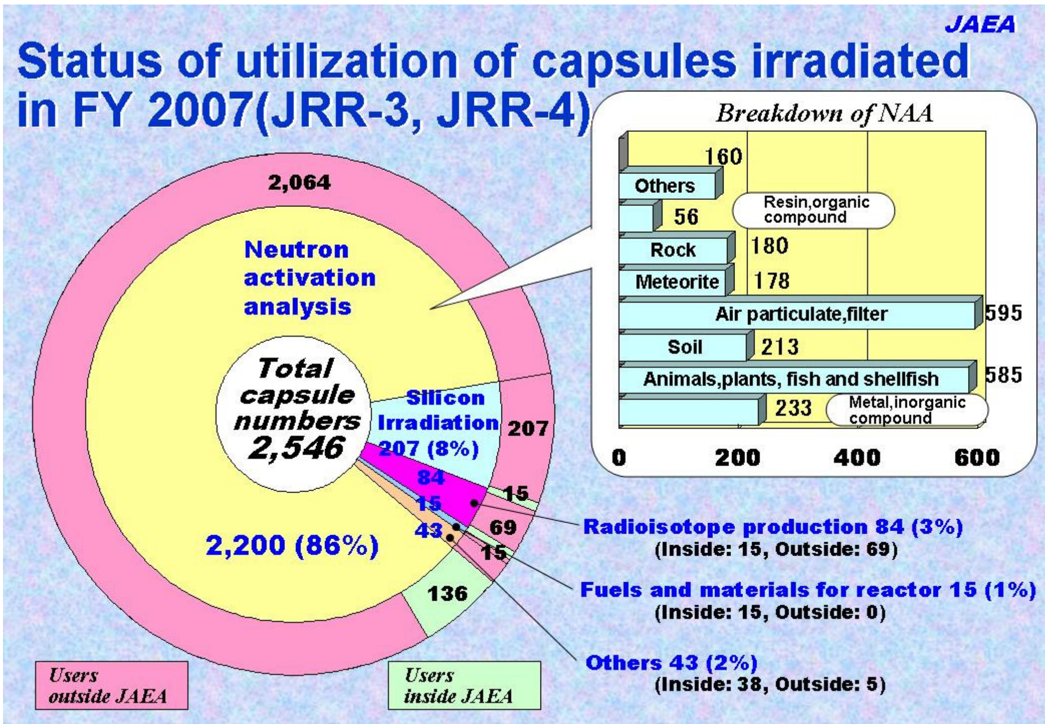


Figure 9 Status of utilization of capsule irradiation in JFY 2007

UTILIZATION OF NSRR

Purpose of NSRR

- Safety research of reactor fuel
- Fresh fuel experiments (Phase I Program) (1975~)
- Irradiated fuel experiments (Phase II Program) (1989~)
- High burnup fuel/ MOX fuel experiments (Phase III Program) (2002~)

Outline of NSRR

NSRR is a TRIGA pulse reactor of annular core. It was built for the investigation of light water reactor fuel behavior during off-normal conditions such as reactivity-initiated accident. The maximum reactor power is 23GW. Figure 10 and Figure 11 show vertical and horizontal cross section of the NSRR, respectively. The core structure is mounted at the bottom of open-top water pool (9 m deep), and cooled by natural circulation of the pool water. The NSRR core consists of 149 driver uranium-zirconium hydride (U-ZrH) fuel/moderator elements, six regulating rods with fuel follower, three transient rods and two safety rods with fuel follower, as shown in Figure 11. An experimental capsule containing test fuel rods is inserted to the experimental cavity located at the core center.

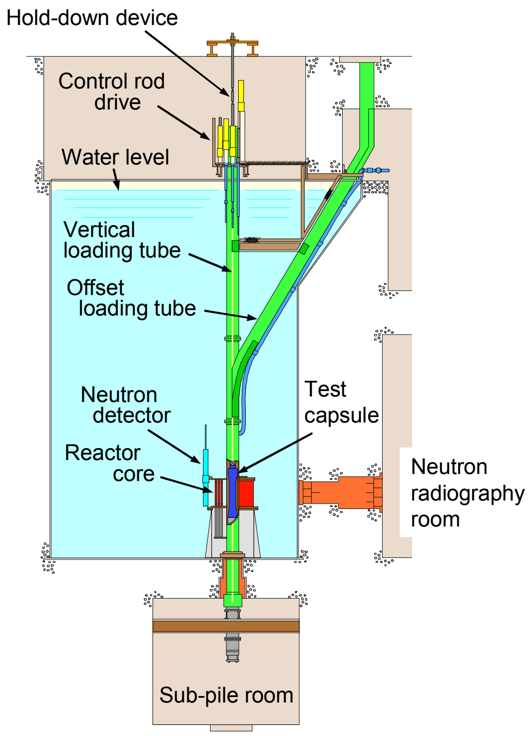


Figure 10 Vertical Cross section of the NSRR

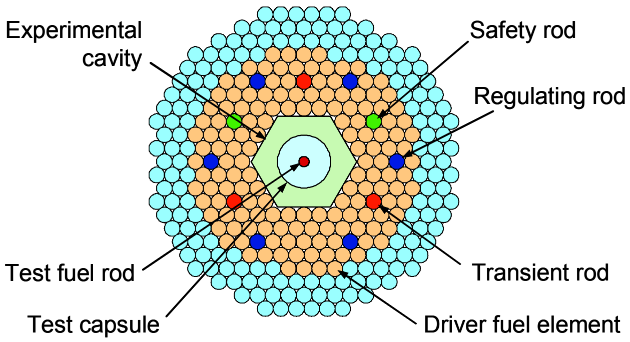


Figure 11 Horizontal Cross section of the NSRR core

High burnup fuel/ MOX fuel experiments

Behavior of high burnup fuels during off-normal conditions, such as reactivity-initiated accident, is being studied with the NSRR. Recent reactivity-initiated accident simulation experiments indicate that the occurrence of fuel failure at higher burnup is closely related to the cladding embrittlement due to the hydrogen absorption. The type of fuel failure, hydride-assisted PCMI (pellet/cladding mechanical interaction) failure, may be influenced by the initial temperature of cladding, since the ductility of cladding becomes high at a high temperature and such failure occurs before temperature rise of cladding due to the power burst. High burnup fuel/ MOX fuel experiments (Phase III Program) were started in 2002 to investigate the behavior of the high burnup fuel and mixed oxide (MOX) fuel which were transported from Europe. A new experimental capsule to achieve a high temperature and high pressure condition has been developed and NSRR experiments with the new capsule were started in 2006. At present, the NSRR experiments with the new capsule are continuing for high burnup fuels of 59, 67 and 71GWd/t. The schematic diagram of the new capsule is shown in Figure 12.

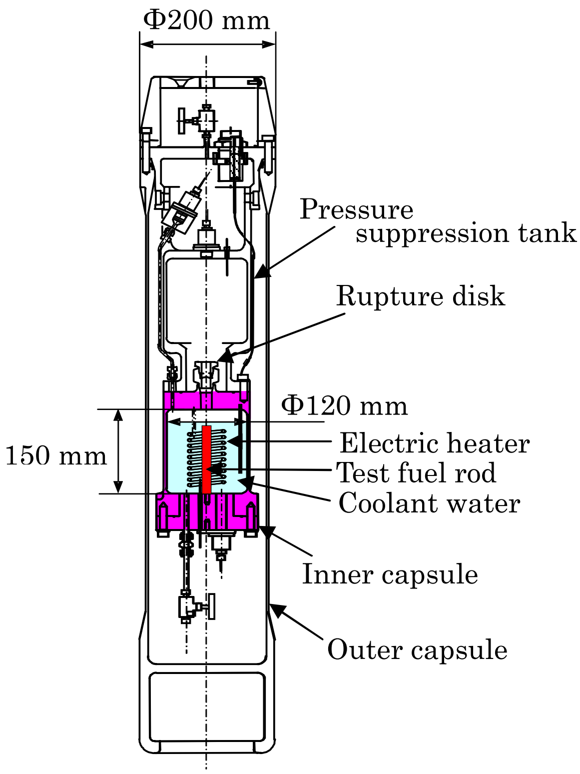


Figure 12 Schematic diagram of new capsule

REMARK

JRR-3 and JRR-4 were utilized capsule irradiation of 2,546 samples, for neutron activation analyses, neutron transmutation doping of silicon, etc in JFY 2007.

The neutron bender system at JRR-3 was installed on the cold neutron guide tube. As a result, the cold beam intensity has increased by about 10 times.

Boron neutron capture therapy was carried out 25 times using JRR-4 in JFY 2007. Since the first medical irradiation for boron neutron capture therapy of JRR-4 was carried out on October 25th 1999, ninety nine times of boron neutron capture therapy irradiation were performed at JRR-4. The trouble of reflector occurred in December 2007. At present, the reactor has stopped to replace the reflector.

A new capsule of NSRR which achieves high temperature condition was developed successfully. Experiments with this new capsule will clarify the temperature influence on the PCMI failure limit of high burnup fuels, of which cladding is embrittled due to hydrogen absorption.

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3.10 THE RESULTS OF AN ULTRASONIC EXAMINATION OF A FLYWHEEL ATTACHED TO A PRIMARY COOLING PUMP IN HANARO

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ABSTRACT

In HANARO, a multi-purpose research reactor of 30 MWth, the primary cooling system (PCS as below) is composed of two heat exchangers, two pumps, piping including valves and instruments for cooling the nuclear fission heat during a normal operation. At a loss of electric power, a flywheel attached to each pump motor shaft provides an inertia force to ensure a slow decrease in the coolant flow in order to prevent a fuel melting. During an operation at a normal speed, a flywheel has a sufficient kinetic energy to produce high-energy missiles, and excessive vibrations of the reactor coolant pump assembly. An excess speed of the pump rotor assembly during a transient increases both the potential energy for a failure and the kinetic energy of a flywheel. The safety consequences could be significant because of possible damage to the PCS, the containment, other equipments or systems important to its safety. This year we conducted an ultrasonic examination of a flywheel to verify its structural integrity during its second 10 years periodic in-service inspection period. This paper describes the results of the ultrasonic examination. It was confirmed that the structural integrity of each flywheel is maintained through the test results in that a marked defect was not found.

INTRODUCTION

In HANARO, a multi-purpose research reactor of 30 MWth, the PCS is composed of two heat exchangers, two pumps, piping including valves and instruments for cooling the nuclear fission heat during a normal operation. At a loss of electric power, the flywheel attached to each pump motor shaft provides an inertia force to ensure a slow decrease in the coolant flow in order to prevent a fuel melting.

During an operation at a normal speed, the flywheel has a sufficient kinetic energy to produce high-energy missiles, and excessive vibrations of the reactor coolant pump assembly. An excess speed of the pump rotor assembly during a transient increases both the potential energy for a failure and the kinetic energy of the flywheel. The safety consequences could be significant because of possible damage to the primary cooling system, the containment, other equipments or systems important to its safety. In order to maintain the structural and mechanical integrity of a flywheel, the guidelines require a 10 year periodic in-service inspection with a nondestructive examination method by qualified persons(1)(2).

This year we conducted an ultrasonic examination of a flywheel to verify its structural integrity during the second 10 years periodic in-service inspection period. This paper describes the results of the ultrasonic examination including the test requirements, test methods and results and considerations.

MAIN SUBJECT

Flywheel configuration

The PCS pump has a motor of 6600 volt, 60 Hz, three phase and 1170 rpm. A flywheel is horizontally connected to the backward end of the motor by a key. The flywheel is made with hot rolled steel as shown in Fig. 1.

A flywheel has an outside diameter of 1020 mm, a bore diameter of 110 mm, a thickness of 140 mm, sixteen 12 mm holes on a radius of 165 mm and eight 150 mm holes on a radius of 270 mm(3). A defect found out in this examination is

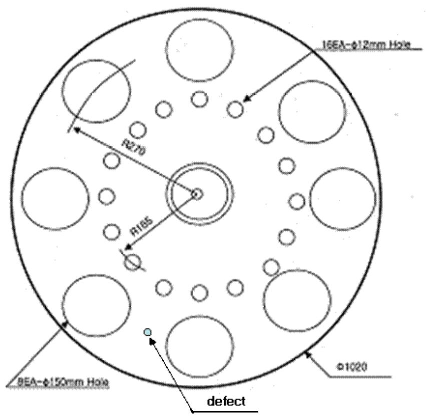


Fig. 1 Flywheel configuration

marked on the flywheel surface of no. 2 pump.

Examination requirements

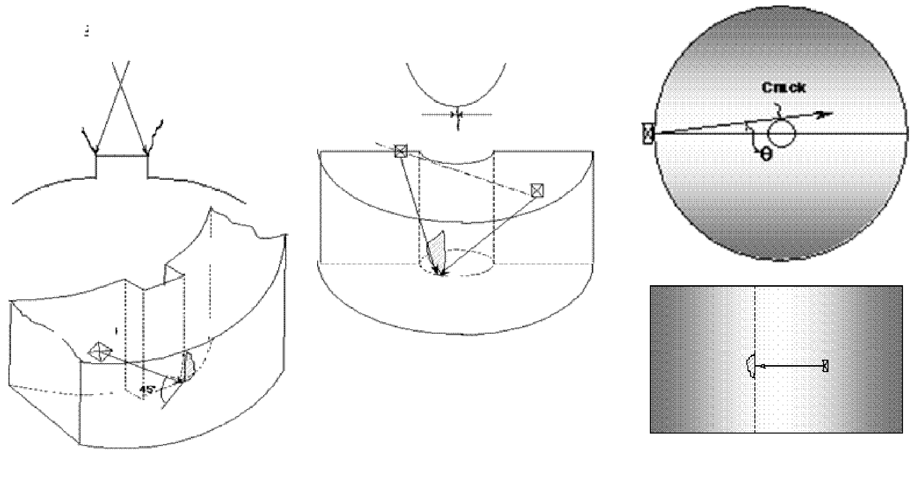
As the PCS pump including its flywheel is classified as safety class III, the pump should be designed, manufactured, installed and examined by the Class 3 components requirements of the ASME SEC. III Code. And it should also be examined by the 10 years periodic in-service inspection program of Article IWD of the ASME SEC. XI Code. According to the program, the keyway of a flywheel should be examined by an ultra sonic volumetric examination every three years and the whole surface of a flywheel should be examined by an ultra sonic volumetric examination every ten years to verify the structural integrity of a flywheel(4).

It is the limit to verify the integrity that a defect is below twenty percent (20 %) of a distance-amplitude correction(5). When a defect is larger than the limit, the status of the defect should be analyzed in detail whether it will developed or not(1).

Examination method

1) The keyway and bore surface of a flywheel

Under a normal operation, a severe induced flaw such as a crack can occur at the edge of a keyway due to a stress concentration as shown in Fig. 2. The crack can develop toward the inner body of a flywheel. And a crack can occur on the surface of a bore due to a structural flaw of a flywheel.



(a) (b) (c)

Fig. 2 Ultra sonic examination method

These cracks can be detected by a forty five degree (45^{o}) inclination angle of an ultra sonic wave as shown in Fig. 2. There are many small holes in the body of a flywheel in HANARO. When an ultra sonic wave passes through the examined surface, the holes can be interrupted with other flaws. To prevent an interruption of the holes, in parallel, the ultra sonic wave passes through the surface vertically.

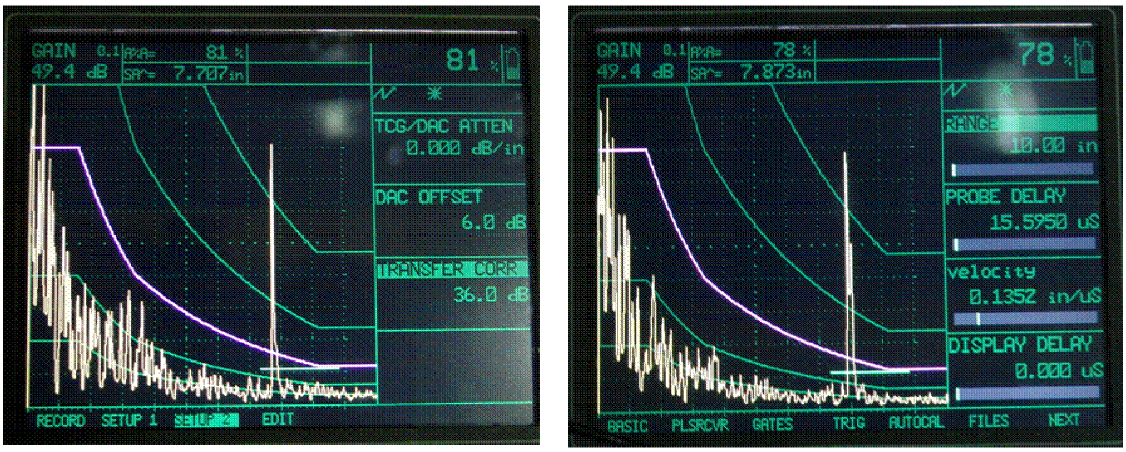
To detect a flaw on a bore surface, a vertical ultra sonic wave tangentially passes through the bore surface as shown in Fig. 2-(b). When an ultra sonic wave passes through the bore surface from the flywheel outside surface, the incidence angle (θ) is limited to six to seven degree for maintaining a vertical incidence on the surface as shown in Fig. 2-(c).

(2) Surface examination of a flywheel

A magnetic or a liquid penetration examination is generally applied for detecting a flaw on the surface. In HANARO, the flywheel surface is coated with an anti-corrosive paint. It is impossible to completely remove the paint. According to a reference, it is possible to avoid the paint with an 88 degree because an ultra sonic wave has an excellent delectability and resolution. An 88 degree angle of the surface ultra sonic wave is applied to this examination(6).

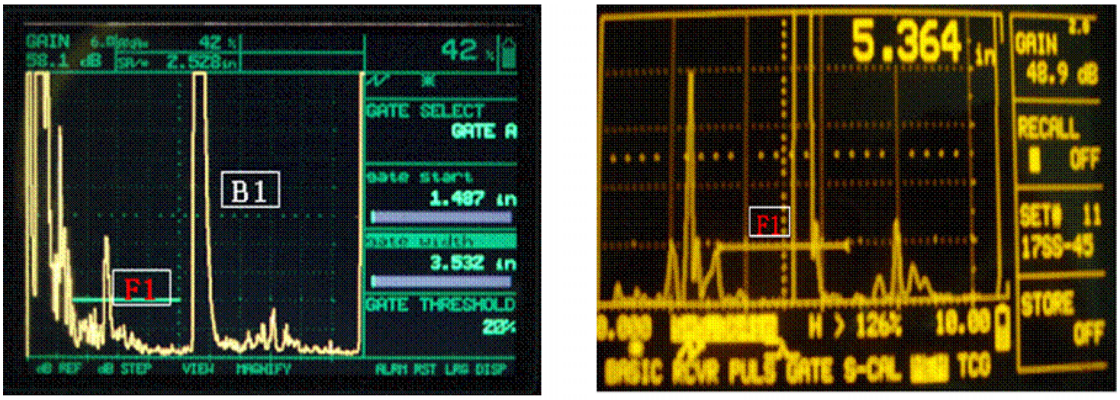
45 degree inclination angle ultra sonic wave examination results

Fig. 3 shows 45 degree inclination angle ultra sonic wave examination results in a clockwise and a count clockwise direction. As shown in the figure, there are no



(a) Clockwise direction (b) Count clockwise direction

Fig. 3 45 degree inclination angle ultra sonic wave examination results



(b) This examination **(b) Last examination**

Fig. 4 Vertical ultra sonic wave examination results

remarkable defect on the keyway and bore except for the bottom reflection wave. Therefore, it was confirmed through the results that the keyway and bore of the flywheels maintain their structural integrity.

Vertical ultra sonic wave examination results

Fig. 4-(a) shows a defect of NO. 2 flywheel, detected in this examination and the location is shown in Fig. 1. When the signal is converted to an actual size, the size is calculated as 6.14 % of the DAC. The defect is below the limit, 20 % of the DAC. It was confirmed through the examination results that the keyway and bore maintain their structural integrity.

Fig. 4-(b) shows the examination result in 2005. When the signal is converted to the actual size, 5.11 % of the DAC under a condition, an initial sensibility of 25 dB and an indicated sensibility of 48.9 dB. The size is similar to that of this examination. Therefore it was confirmed through the calculation results that the defect was not developed.

Surface ultra sonic wave examination results

Fig. 5 shows the results of the surface ultra sonic examination from this examination. As shown in the figure, there are no remarkable defects on the keyway and bore except for the bottom reflection wave. Therefore, it was confirmed through the results that the surface of the flywheels maintained their structural integrity.

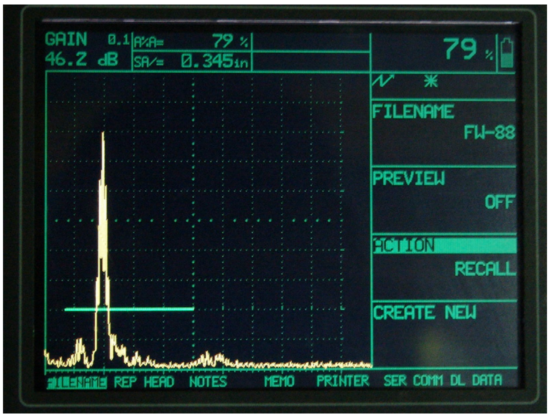


Fig. 5 Surface ultra sonic wave examination results

CONCLUSIONS

An ultra sonic examination was applied to verify the structural integrity of each flywheel attached to a primary cooling pump in HANARO. It was confirmed through this examination results that each flywheel maintained its structural integrity without a markable defect above the limit of 20% of the DAC. No. 2 pump has a defect of 6.14 % of the DAC which is below the limit. And the defect is similar to that detected in 2005 without any development.

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[5] ASME SEC. X Art 4, APP. K, "Recording Straight beam examination data for planner reflectors," 1998, p. 107.
[6] Nondestructive testing handbook, Volume 6, "Magnetic particle testing", Section 6, Part 3, Figure 12.

3.11 UTILIZATION OF REACTORS OVERSEAS FOR STUDY OF RADIATION EFFECTS IN MATERIALS

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ABSTRACT

This paper presents the current status of utilization of overseas reactors for study of radiation effects in materials as joint-use research at International Research Center for Nuclear Materials Science, Institute for Materials Research (IMR), Tohoku University (IMR-Oarai Center). Japan Materials Testing Reactor (JMTR) which had been mainly utilized for material irradiation at IMR-Oarai Center since 1970 was shut down in August 2006. Since then an overseas reactor, BR2 at SCK/CEN in Belgium, has been used in order to continue our irradiation programs and perform advanced material irradiation in collaboration with SCK/CEN for development of new irradiation techniques in BR2. In this paper, the features of irradiation facilities/devices in BR2 such as CALLISTO, MISTRAL, ROBIN and BAMI and the irradiation programs in 2006-2008 with use of each facility/device are described.

INTRODUCTION

International Research Center for Nuclear Materials Science, Institute for Materials Research (IMR), Tohoku University (IMR-Oarai Center), which is sited at Oarai in Japan Atomic Energy Agency (JAEA), has been open to researchers of universities and national institutes in Japan who are studying and will study the effects of neutron irradiation on mechanical and physical properties of advanced nuclear materials since 1969 and the physical and chemical properties of actinides since 1990.

For the study of radiation effects of materials, Japan Materials Testing Reactor (JMTR) and JOYO, a fast experimental reactor operated by JAEA, and several overseas reactors such as

FFTF, HFIR and ATR in U.S. have been utilized and then post irradiation examinations (PIEs) have been performed mainly at IMR-Oarai Center. Joint-use research based on such material irradiation and PIEs has accumulated a number of achievements and hence has been widely and continuously supported by users throughout Japan. On August 1st, 2006, however, JMTR, which had been used since 1970 as the major reactor for material irradiation as joint-use research, was shut down. We need a materials testing reactor (MTR) similar to JMTR in order to continue our irradiation program and perform advanced material irradiation. We surveyed currently operating MTRs in the world and decided to utilize BR2 at SCK/CEN in Belgium and perform advanced material irradiation in collaboration with SCK/CEN for development of new irradiation techniques in BR2. The BR2 reactor is a versatile high flux MTR and its channels are skewed to form a hyperboloid geometric arrangement which provides a very compact high-flux core with easy access to the Reactor Pressure Vessel (RPV) top for the control and instrumentation of the hardware needed to service the fuel, control rods, and channels that accommodate experimental and commercial programs [1)].

BR2 has been used since 2006 and greatly contributes to our irradiation programs for joint-use research. Explorative use of irradiation facilities/devices in BR2 provides new irradiation conditions that have been unavailable so far in our irradiation programs. In this paper, the features of irradiation facilities/devices in BR2 and the current status of the irradiation programs with use of each facility/device are presented.

BR2 AND ITS IRRADIATION FACILITIES/DEVICES [1-2)]

Studies of radiation effects in materials require a MTR with high flux of fast neutrons. In this respect, BR2 possesses slightly higher fast neutron flux than JMTR as shown in **Fig. 1**.

The BR2 reactor provides irradiation under PWR and BWR conditions identical to commercial power reactors by using irradiation facilities/devices of CALLIATO and MISTRAL, respectively. CALLISTO, a high pressure/high temperature experimental water loop facility, employs the standard CALLISTO basket designed for 9 needles (with rod specimens) without instrumentation (except dosimeters), where the maximum needle cross-section is 10 mm x 10 mm or 11 mm in diameter and the effective needle length is 600 mm. MISTRAL (Multipurpose Irradiation System for Testing of Reactor Alloy) is appropriate for irradiation that requires high neutron fluences, however, the space available to put specimens is restricted: the number of specimens and their dimension to be loaded are smaller than those in CALLISTO. In addition to these, we can utilize other two facilities/devices; ROBIN (ROtating Basket with INnstrumented Needles) and BAMI (BAsket for Material Irradiation).

ROBIN is effectively used for irradiation at relatively low fluences and low temperatures. The features of ROBIN are that loading and unloading of ROBIN are possible during reactor operation and that ROBIN is designed to rotate during irradiation for compensation of the fast neutron flux radial gradient through the selected irradiation position, i.e., at a fixed position (level in the needle), the fluence is the same in all the needles due to ROBIN rotation.

Irradiation below 100˚C is not possible in ROBIN, but BAMI allows irradiation at such low temperatures by using capsules that can be directly loaded in the BR2 vessel: in the fuel elements or in the beryllium plug (in the reflector region). The capsules can be located at different vertical levels in the core, so one can have a broad range of fluence. These capsules are directly cooled by the BR2 primary cooling water at about 50˚C. They have to stay during the whole cycle; unloading during the cycle is not possible.

The maximum dpa (displacement per atom) per one cycle and irradiation temperatures that can be attained by each irradiation facility/device are listed in **Table 1**. By decreasing the fuel elements around the channel containing the needles with specimens and adjusting the position of the specimens, the irradiation under much lower dpa than the maximum can be performed.

MISTRAL and ROBIN allow us to measure the irradiation temperature inside the dummy specimens loaded in the basket central position. The temperature can be controlled within a certain range by adjusting the basket cooling flow. Since no heaters to raise the irradiation temperature above the coolant temperature are equipped with the facilities/devices, higher irradiation temperatures rely on adjustment of γ heating controlled by the gas gaps between the holder and specimens. The coolant temperature for each facility/device is also shown in table 1.

IMR IRRADIATION PROGRAMS [2-3]

In 2006, the first year of our utilization of BR2, we decided to use CALLISTO in view of our users' demand for irradiation conditions, irradiation space availability, irradiation costs including holder manufacturing and PIEs (unloading of the irradiated specimens). **Table 2** shows the irradiation conditions in 2006, the program of which was named MICADO-1.

Many types of specimens for tensile tests, Charpy impact tests, 3-point bend tests, corrosion tests, 3 DAP (3 dimensional atom probe) analyses, positron annihilation measurements and transmission electron microscopy (TEM) observations were collected from all users in Japan. The specimens were carefully packed with Al foil every user (experimenter), specimen type and irradiation condition. The packed specimens were sent to SCK/CEN. In SCK/CEN all the packed specimens were encapsulated in order to prevent each specimen from any corrosion (except for corrosion test samples) and damaging due to vibration caused by the high water velocity, to make specimen identification easier, to keep them separate and to trace them during the handling operations (assembly in the needles, unloading after irradiation). **Fig. 2** shows holders after filling the packed specimens and welding covers on the holders under He control in MICADO-1.

The holders were loaded into several needles in which the positioning of each holder was made to meet the requested irradiation conditions. The needles were loaded into the standard CALLISTO basket and irradiated under the condition shown in table 2. After irradiation, each of the specimen packs was taken out from the holders and put into a small plastic box together with silica gel to prevent moisture pick-up. A cask with the plastic boxes containing the irradiated specimens were safely transported to IMR-Oarai Center for PIEs.

Aiming at explorative use of BR2 in 2007-2008, we had meetings with joint-use researchers in Japan three times in February 2007 and then visited SCK/CEN to discuss the explorative use of BR2. As a result, we decided the irradiation program in 2007-2008, which covers low temperatures (60~300˚C) and a wide range of neutron fluences (0.01~0.5dpa) as shown in **Table 3**. It should be noted that BAMI provides the irradiation condition of the temperatures as low as 60-90˚C and dose as high as ~0.5 dpa that have been unavailable so far in our irradiation[3)]. Such irradiation condition enables us to study the radiation effect for bulk metal glasses because the irradiation temperatures of 60~90˚C are well below the glass transition temperature or relaxation temperature in most cases.

In collaboration with SCK/CEN, we are developing new techniques needed for BR2 irradiation. **Fig. 3** shows an example of specimen holders developed for BAMI where packed TEM specimens were inserted and covers were joined to the holders by airtight mechanical sealing under He control. Airtight mechanical sealing without heating precludes microstructural changes and resultant property changes, which may be inevitable in weld sealing.

In addition, we held a small workshop on "Prospect of reactor irradiation and post irradiation examinations (PIEs)" in 2007 IMR-Oarai Research Meeting (9/6~9/7, 2007), inviting Dr. M. Weber who works for SCK/CEN to the workshop [2)]. He presented "BR2 irradiation and PIE capabilities at SCK-CEN". The workshop was very productive and fruitful.

PIEs AT IMR-OARAI CENTER

The specimens irradiated in BR2 and other reactors such as JOYO and JRR3 in JAEA are transported to the hot laboratory (HL) at IMR-Oarai Center. The irradiated specimens are sorted in HL and subjected to PIEs. A variety of facilities for PIEs are installed in the controlled areas in the four main buildings in IMR-Oarai Center. Those are for specimen cutting and polishing, mechanical testing (tensile, bending, Vickers micro-hardness, nano-indentation, instrumented Charpy impact tests, fatigue tests corrosion tests), specimen surface observations (SEM), microstructural observations (TEM), focused ion beam, three dimensional atom probe analyses, positron annihilation measurements, X-ray analyses, etc. **Fig. 4** shows photos of an electro-discharge machine for cutting PIEs' specimens from irradiated materials and an instrumented Charpy impact testing machine installed in HL. The latter machine allows us to measure the impact properties such as the absorbed energy, yield and maximum loads, deflection to brittle fracture and total deflection for full size and miniaturized Charpy V-notch specimens down to 1 mm by 1 mm by 20 mm before and after irradiation, at test temperatures from 77 to 1073 K and displacement rates from 1 to 5 m/s. Using the testing machine and these differently sized specimens, systematic studies have been made on the effects of notch geometry and material conditions such as chemical composition, heat treatment and neutron irradiation.

SUMMARY

1. JMTR which had been mainly utilized for material irradiation in joint-use research at IMR-Oarai Center since 1970 was shut down in August 2006 and hence BR2 at SCK/CEN in Belgium has been used in order to continue our irradiation programs and perform advanced material irradiation in collaboration with SCK/CEN for development of new irradiation techniques in BR2.
2. The BR2 reactor is a versatile high flux MTR and provides a wide range of irradiation conditions of 60-550˚C and 0.01~0.5dpa per one cycle by using four different irradiation facilities/devices of CALLISTO, MISTRAL, ROBIN and BAMI. The main features of each facility/device have been presented.
3. The BR2-irradiation programs for joint-use research in 2006 and 2007-2008 have been described. The program in 2007-2008, aiming at explorative use of BR2, includes the irradiation condition of the temperatures as low as 60-90˚C and dose as high as 0.5 dpa that have not been available so far in our irradiation.
4. Collaboration with SCK/CEN is in progress to develop techniques for irradiation in BR2.

A small workshop on "Prospect of reactor irradiation and post irradiation examinations (PIE)" was held in 2007 IMR-Oarai Research Meeting with Dr. M. Weber's (SCK/CEN) attendance.

ACKNOWLEDGEMENTS

The authors would like to express their gratitude to researchers/engineers at SCK/CEN and JAEA and IMR joint-use researchers for their continuous supports and collaborations for material irradiation and PIEs. They are also greatly indebted to Prof. M. Hasegawa, IMR at Tohoku University, because access to BR2 for joint-use research at IMR-Oarai Center owes to his collaboration with SCK/CEN since 2003.

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2. Report of 2007 IMR-Oarai Research Meeting, 2007.
3. Report of 2008 IMR-Oarai Research Meeting, 2008.

Table 1 Irradiation facilities/devices in BR2 and irradiation conditions available by using each facility/device. The dose (maximum) is per cycle (21 days).

Facility/Device	Temp. (˚C)	DPA in Fe (max.)	Fluence (max.) (n/m^2, E ≥ 1MeV)	Cooling water Temp. (˚C)
CALLISTO	300 ~ 500	0.25 (core) 0.1 (ref.)	2.2×10^{24} (core) 0.6×10^{24} (ref.)	250 ~ 300
MISTRAL	200 ~ 550	0.4 ~ 0.6	$(2.6 \sim 4) \times 10^{24}$	180 ~ 350
ROBIN	100 ~ 350	0.1	0.6×10^{24}	65 ~ 125
BAMI	60 ~ 100	0.1 ~ 0.5	$(0.6 \sim 3.5) \times 10^{24}$	~ 50

Table 2 BR2 irradiation conditions in 2006 (MICADO-1)

Facility/Device	Capsule	Channel (γ heating)	Fluence (n/m^2)	Temp. (˚C)	Atmo.
CALLISTO	06M-1BR	K (3W/g)	1×10^{23}	290	He/water
CALLISTO	06M-2BR	K (3W/g)	5×10^{23}	290	He/water
CALLISTO	06M-3BR	K (3W/g)	1×10^{24}	290	He/water
CALLISTO	06M-4BR	K (3W/g)	5×10^{24}	290	He/water

Table 3 BR2 irradiation condition in 2007-200 (MICADO-2, 3)

<table>
<tr><th>Facility/Device</th><th>Temp. (˚C)</th><th>DPA in Fe(max*)</th><th>Cycles</th><th>Position</th><th>Atmo.</th></tr>
<tr><td>ROBIN</td><td>100 ~ 200</td><td>~ 0.1</td><td>1</td><td>Reflector</td><td rowspan="3">He</td></tr>
<tr><td>BAMI</td><td>60 ~ 90</td><td>~ 0.5</td><td>2</td><td>Core</td></tr>
<tr><td>CALLISTO</td><td>290</td><td>~ 0.5</td><td>5</td><td>Reflector</td></tr>
</table>

＊ROBIN:~0.01dpa is possible by unloading the needles during irradiation.

＊CALLISTO: Half of the maximum dpa is possible by using axial distributions of the flux

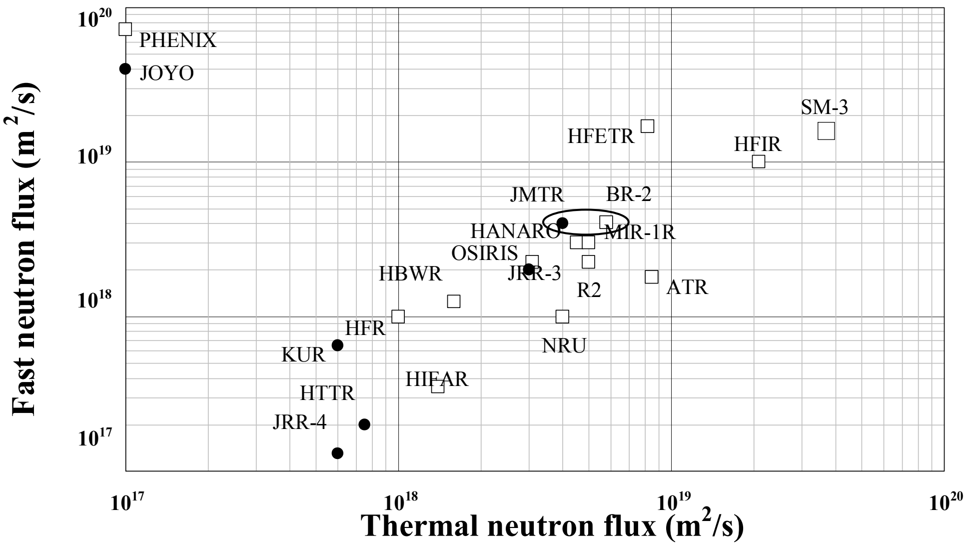

Fig. 1 Comparison of neutron flux between currently operating material testing reactors (MTRs) in the world.

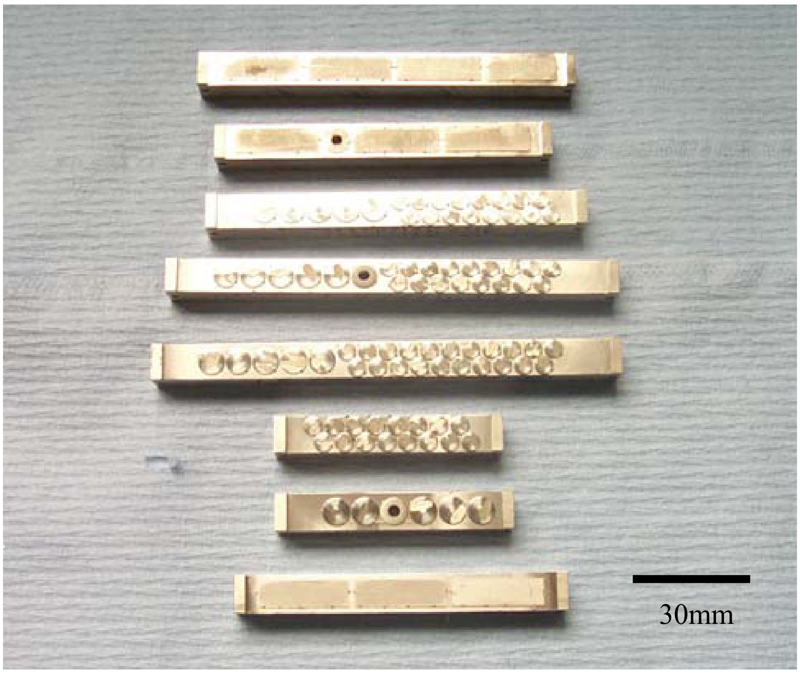

Fig. 2 Examples of specimen holders after filling the packed specimens and welding covers on the holders under He control in MICADO-1.

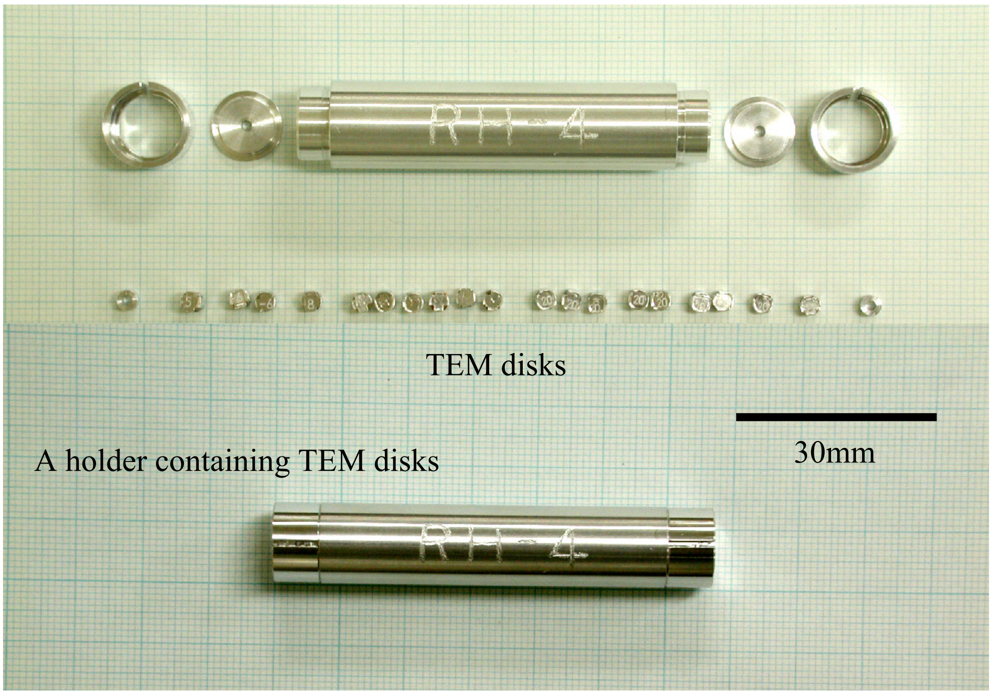

Fig. 3 A specimen holder developed for BAMI irradiation where packed TEM specimens are inserted and covers were joined to the holders by airtight mechanical sealing under He control. Airtight mechanical sealing without heating precludes microstructural changes and resultant property changes, which may be inevitable in weld sealing.

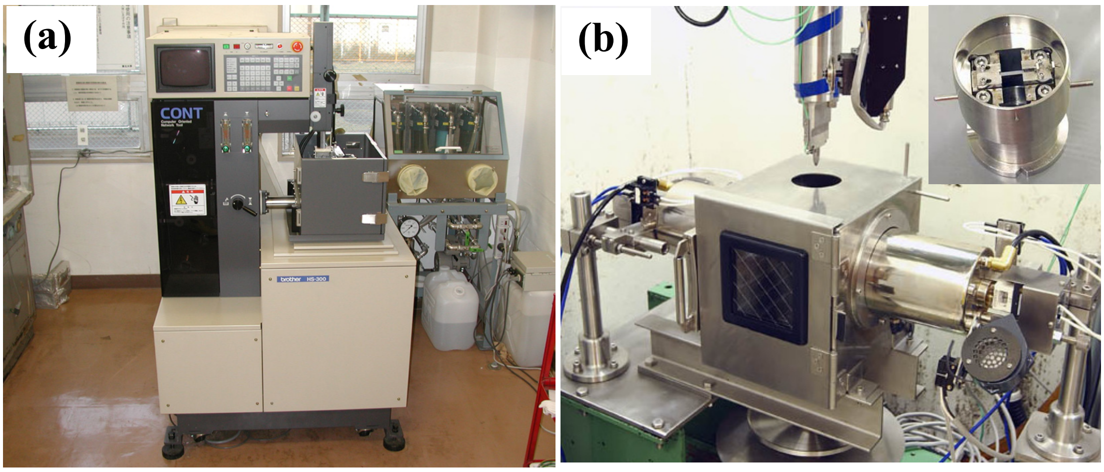

Fig. 4 (a) Electro-discharge machine and (b) instrumented Charpy impact testing machine with a specimen holder (upper right) installed in the hot laboratory in IMR-Oarai Center for PIEs.

3.12 IN-SERVICE INSPECTION OF ZIRCONIUM COMPONENTS OF HANARO

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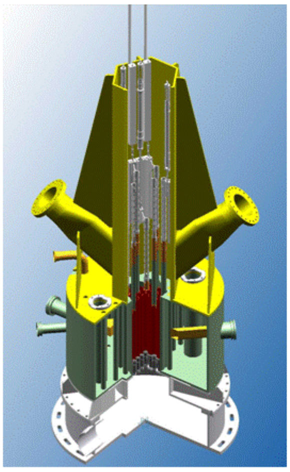
This article reports on the in-service inspection of zirconium components such as the inner shell, flow tubes, absorbers and the shroud tubes. All of them are located in the core and related to the structural integrity of the reactor core and the safe shutdown of the control rods or shutoff rods. Therefore their physical dimensions should be monitored periodically to confirm the structural integrity and certain clearances between neighbor components.

The inner shell of the reflector vessel surrounding the core is the most critical part from the viewpoint of a neutron irradiation. The periodic measurement of the dimensional changes in the vertical straightness of the inner shell is considered as one of the in-service inspections. The measurement of the straightness for the inner shell in Aug. 2004, the first measurement ever after 9 years reactor operation showed a deformation of a maximum of 0.26 mm toward the core center, which is smaller than the prediction by a design analysis.

During the a few times the removal of the absorbers and flow tubes, in 1997 and 2004, we found brindle marks around the bottom of the absorbers, the top of flow tubes and the inner shell. These parts are located just above the fuel zone. The conclusion for the brindle marks by discussions with metallurgists was an oxidation on the surfaces due to the temperatures.

The control absorber or shutoff rod moves up and down by two guides; a shroud tube at the top and a flow tube at the bottom. In 2006, it was confirmed by a remote measurement for the diameters at the sliding parts that the clearances are properly maintained between the neighbor components.

KAERI HANARO
High-flux Advanced Neutron Application Reactor



In-service Inspection of Zirconium Components of HANARO

Nov. 2008

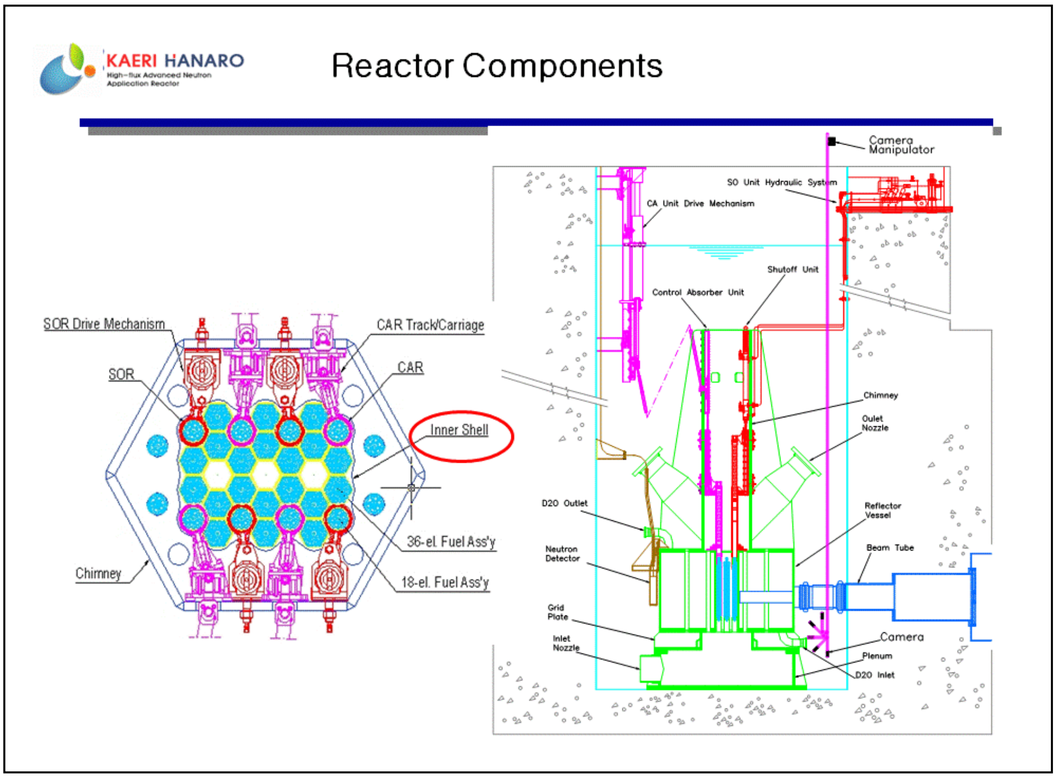
YeongGarp Cho, HoanSung Jung, SangJun Park, InCheol Lim, SungJae Park, BongGoo Kim and YoungSuk Kim

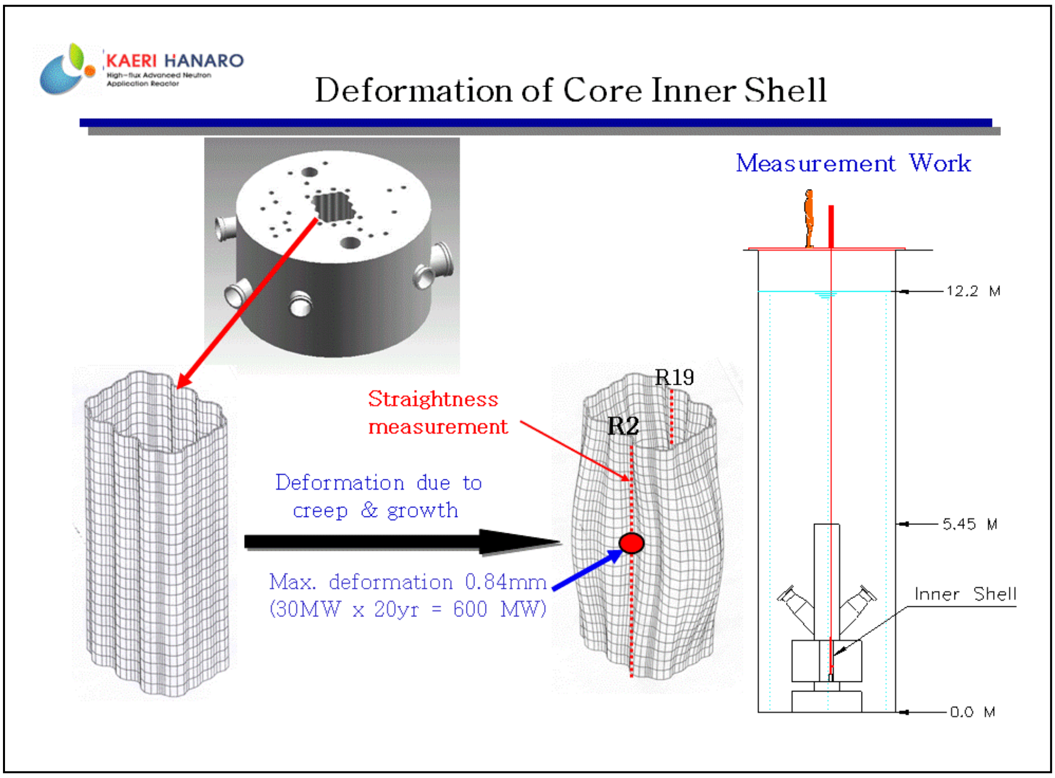
KAERI

KAERI HANARO
High-flux Advanced Neutron Application Reactor

Contents

- Deformation of Core Inner Shell
- Physical Clearances around SOR/CAR
- Surface Color Changes on SOR/CAR, Flow Tubes and inner Shell
- Wear of Fuel Channels
- Future Irradiation Test Plan

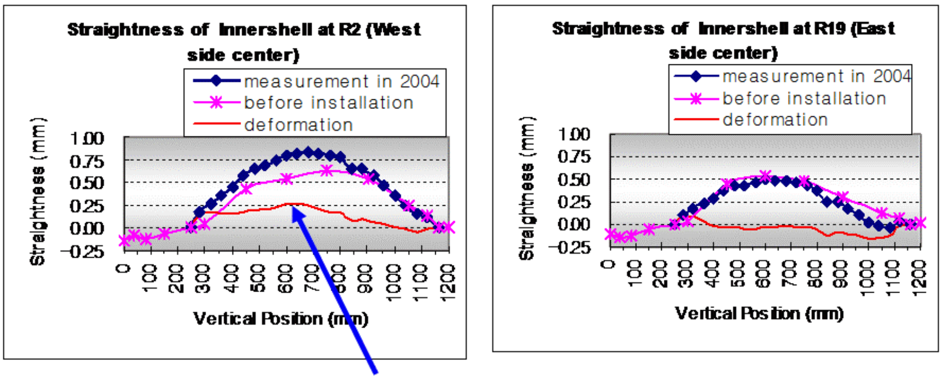

KAERI HANARO
Reactor Components
SOR Drive Mechanism
CAR Track/Carriage
SOR
CAR
Inner Shell
36-el. Fuel Ass'y
18-el. Fuel Ass'y
Chimney
Camera Manipulator
SO Unit Hydraulic System
CA Unit Drive Mechanism
Shutoff Unit
Control Absorber Unit
Chimney
Oulet Nozzle
D2O Outlet
Reflector Vessel
Neutron Detector
Beam Tube
Grid Plate
Inlet Nozzle
Camera
Plenum
D2O Inlet


KAERI HANARO
Deformation of Core Inner Shell
Measurement Work
12.2 M
Straightness measurement
R19
R2
Deformation due to creep & growth
5.45 M
Max. deformation 0.84mm
(30MW x 20yr = 600 MW)
Inner Shell
0.0 M

KAERI HANARO

Deformation of Core Inner Shell

- In-service inspection required at every 10 years
- 1st measurement of inner shell straightness in 2004



Max. deformation 0.26 mm

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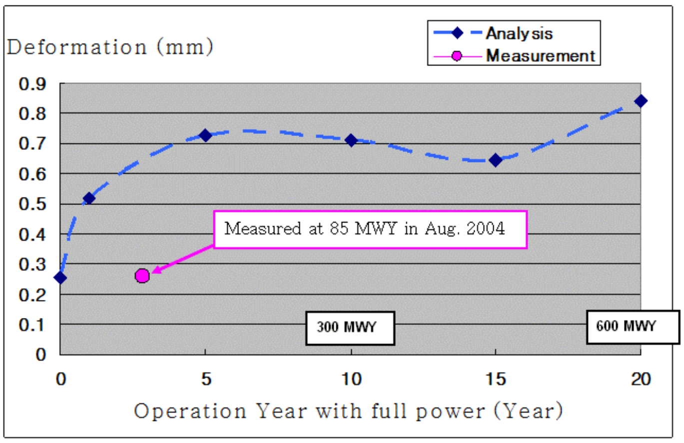
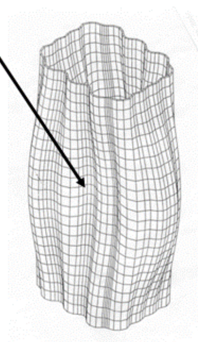
Deformation of Core Inner Shell

Analytic estimation and Measurement Value at Center Point





Reference: B.Leitch, KMRR Creep and Growth Analysis. 37-31200-220-006, 1991

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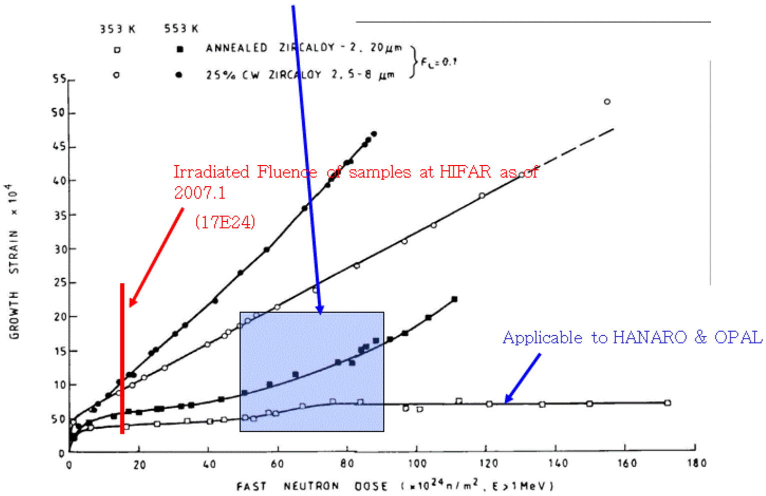
Neutron Dose vs Growth Curve

- Fast neutron fluence at inner shell (E> 1MeV) , as of August 2004 = 50E24 – 93E24 n/m²



Ref: Young Suk Kim1, Kang Soo Kim and Yong Moo Cheong "Evaluation of Degradation of the Zircaloy-4 Reflector Vessel with Time", Hanaro 2005

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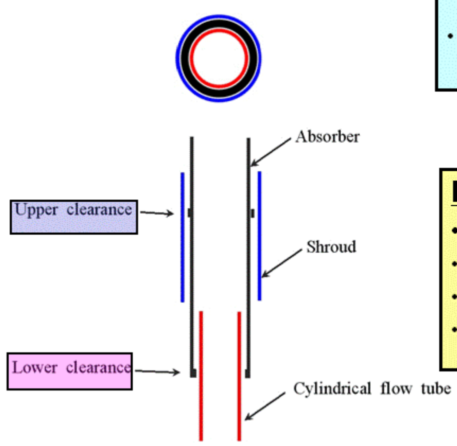
Physical Clearances around SOR/CAR



Motivation

- Welding deformation on as-built shroud ~ 1.2mm in diameter
- Diameters may change by neutron irradiation and/or residual stress relieving
- In-service inspection requirement for clearance

Diameter measurement for full stroke

- Maximum OD for cylindrical flow tube
- Minimum ID for absorber lower bearing
- Maximum OD for absorber upper bearing
- Minimum ID for shroud

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Physical Clearances around SOR/CAR

Summary for clearances in diameter (mm)

Location	Design Clearance	Estimated minimum clearance due to shroud deformation	Minimum clearance measured
Upper Clearance	1.21-1.81	SOR1 : 0.2 - 0.3 SOR2 : 0.45 – 0.55 SOR3 : 0.45 – 0.55 SOR4 : 0.45 – 0.55 CAR1 : 0.45 – 0.55 CAR2 : 0.55 – 0.65 CAR3 : 0.15 – 0.25 CAR4 : 0.0 – 0.1	SOR1 : 0.68 SOR2 : 0.68 SOR3 : 0.59 SOR4 : 0.76 CAR1 : Not measured CAR2 : Not measured CAR3 : Not measured CAR4 : 0.68
Lower Clearance	0.75-1.25	0.75-1.25	SOR1 : 1.01 SOR2 : 1.01 SOR3 : 0.81 SOR4 : 1.01 CAR4 : 1.01

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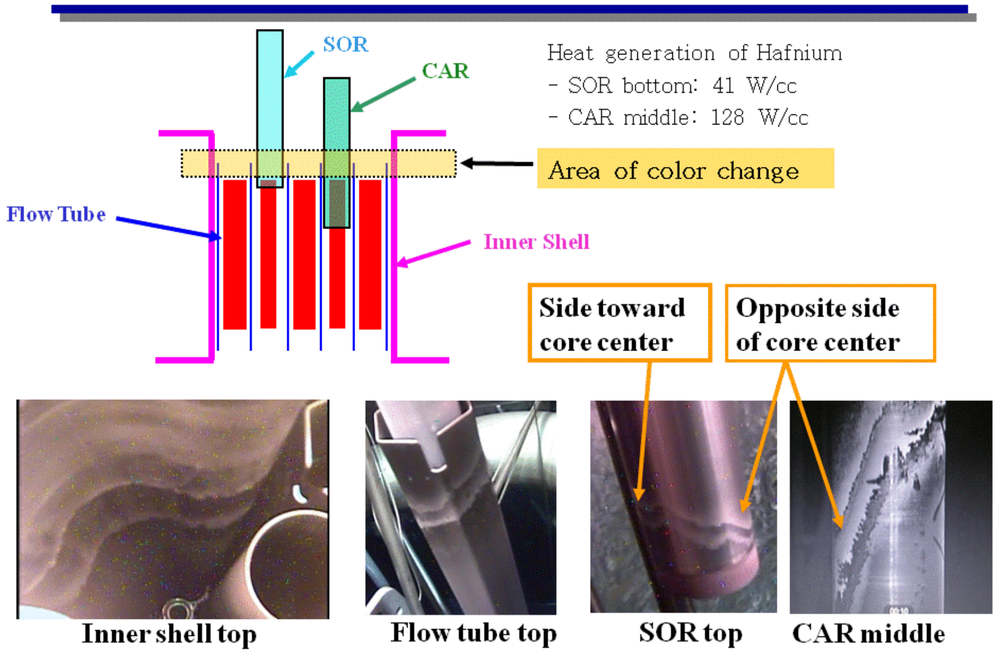
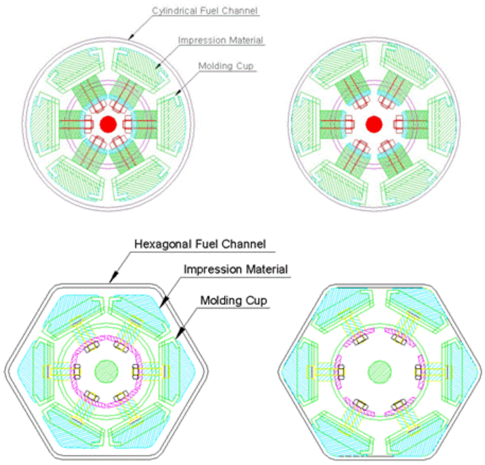
Visual Inspections in 2004



Wear Inspections of Fuel Channels

- Development of Inspection Technology & Tools
  - Impression material
  - Inspection tools
- Flow channels being inspected
  - Slight local wear

  (max. wear depth: 0.2mm for 1.6mm thick tube), end of 2001



(Insertion of the inspection tool with impression material

(Contact of impression material with the inside wall

Future Irradiation Test Plan

- Joint research with ANSTO
  - 2004-2006 : irradiation test at HIFAR
  - 2007-- : irradiation test at OPAL
- Irradiation test plan at HANARO
  - 2008 : Planning and design of sample cartridge
  - 2009 : Design and fabrication of the irradiation capsule and cartridge assembly
  - 2010 : Start of aging
- Cooperation with international researchers
  - France
  - Canada
  - India
  - etc

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Summary and Conclusions

In-service Inspections for 12 Years

- 1st measurement of core inner shell in 2004
  : Deformation is far less than estimation
- 1st measurement of clearance around SOR/CAR in 2006
  : Clearances are bigger than estimation
  : Enough clearance around SOR/CAR for safe shutdown
- Visual inspection in 2004 & 2006
  : Surface oxidation on absorbers, flow tubes and inner shell
- Wear Inspection of Fuel Channels in 2001
  : No considerable wear marks

3.13 INTRODUCTION TO THE COLD NEUTRON SOURCE AT HANARO

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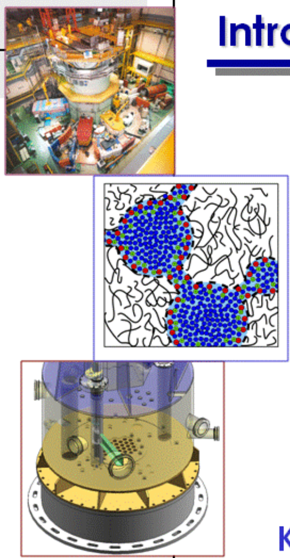
This article aims to introduce the development of a Cold Neutron Source (CNS) at HANARO. The reactor has been safely operating for 13 years since its initial criticality in February of 1995. The reactor was designed for a multi-purpose use in the science research and engineering field such as a neutron scattering, a material irradiation, a nuclear fuel test loop, a neutron transmutation for silicon, and etc. All the aforementioned purposes except for the CNS have been successively developed for their inherent aims. In order to complete the remaining aim, a development project for the CNS has been underway since 2003. Also, it is intended to enhance the utilization capacity of HANARO as a research reactor. The development was divided into two stages for 7 years depending on the characteristics of the stage such as a design and implementation. It was intended to effectively achieve its final goal on schedule. The first stage has been finished in 2006 and the second stage should be finished in 2009.

The HANARO CNS basically consists of a hydrogen system, a vacuum system, a gas blanketing system, and a helium refrigeration system because it has adopted liquid hydrogen as a moderator. The liquid hydrogen contained in the moderator cell evaporates due to a gamma heating. The hydrogen vaporizes up to the condenser, where it is re-liquefied then it returns down to the moderator cell. This thermo-siphon loop can only be established under a very low temperature environment, which requires a method for a thermal insulation. All of the CNS equipment and systems except for the IPA (In-Pile Assembly) will be installed in September 2008 and it will start its commissioning at the end of 2008. The IPA will be installed in the first half of 2009. On the other hand, the cold neutron laboratory building, which was designed for a guide hall to accommodate cold neutron scattering instruments, has been constructed in May 2008. At the beginning of 2010, it is hoped that HANARO will successively provide a cold neutron source including a research facility for international users who want to use a good experimental environment to do their researches by using the cold neutron source.



Introduction to the CNS at HANARO

2008. 11. 5

Sang-ik Wu and Young-ki Kim

Development of Systems for CNS Project
Korea Atomic Energy Research Institute

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High-flux Advanced Neutron Application Reactor

Contents

- Objectives
- Organization & Milestone
- Safety Design Requirements
- Schematics of IPA
- System Compositions
- System Design Status
- Construction of CNL Building
- Conclusions and Future Plan

KAERI HANARO High-flux Advanced Neutron Application Reactor

Objectives

- Aims
  - To enhance the research & utilization of HANARO
  - To complete a final remaining purpose of HANARO
- Basic idea
  - To assure the reactor safety
  - To avoid any accidental H_2-O_2 reaction
  - To apply the depth-in-depth approach to design

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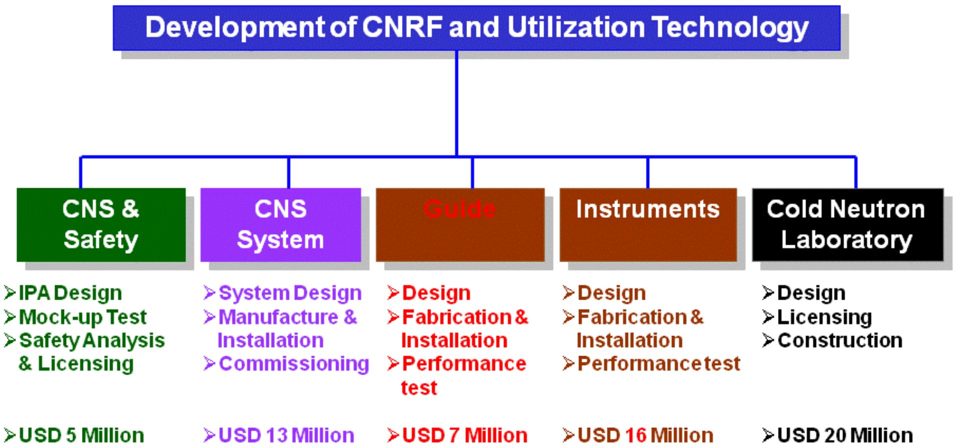
Project Organization



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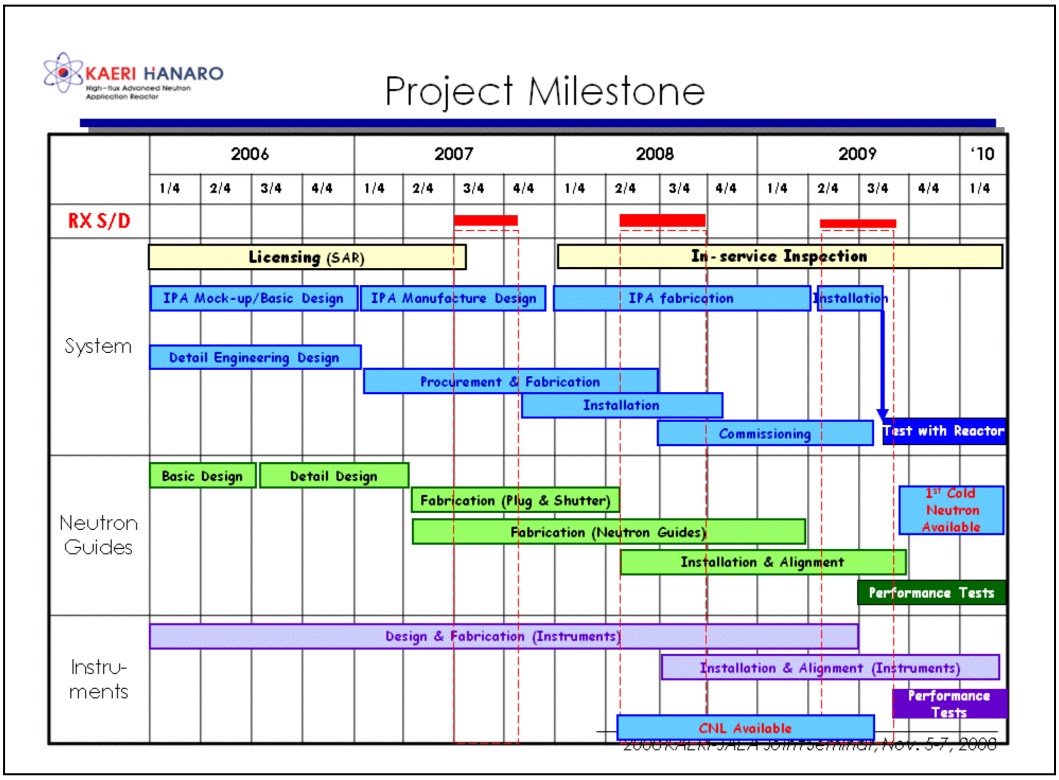

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Project Milestone
2006
2007
2008
2009
'10
1/4
2/4
3/4
4/4
RX S/D
System
Licensing (SAR)
In-service Inspection
IPA Mock-up/Basic Design
IPA Manufacture Design
IPA fabrication
Installation
Detail Engineering Design
Procurement & Fabrication
Installation
Commissioning
Test with Reactor
Neutron Guides
Basic Design
Detail Design
Fabrication (Plug & Shutter)
Fabrication (Neutron Guides)
Installation & Alignment
1st Cold Neutron Available
Performance Tests
Instru-ments
Design & Fabrication (Instruments)
Installation & Alignment (Instruments)
Performance Tests
CNL Available
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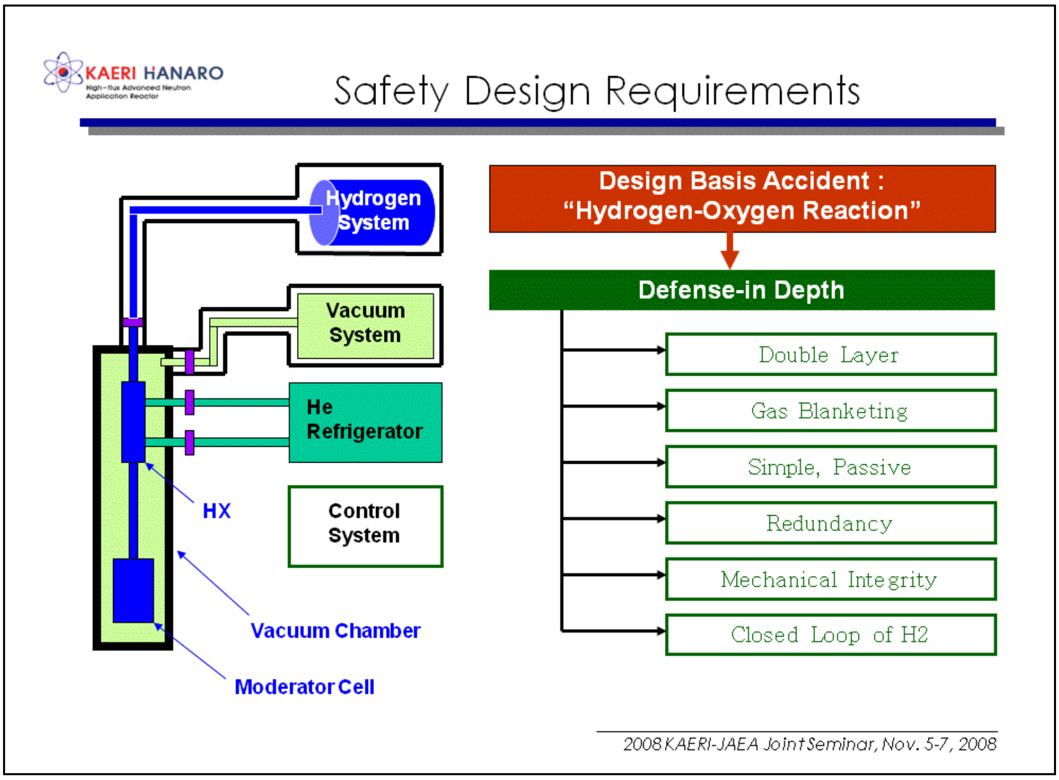

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Safety Design Requirements
Hydrogen System
Vacuum System
He Refrigerator
Control System
HX
Vacuum Chamber
Moderator Cell
Design Basis Accident : "Hydrogen-Oxygen Reaction"
Defense-in Depth
Double Layer
Gas Blanketing
Simple, Passive
Redundancy
Mechanical Integrity
Closed Loop of H2
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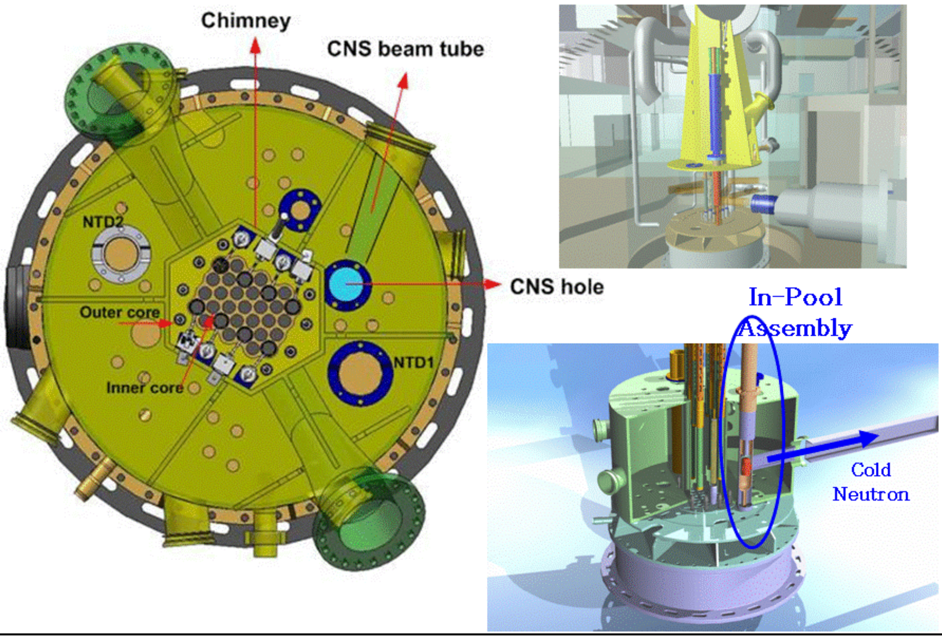
Schematics of IPA [1]



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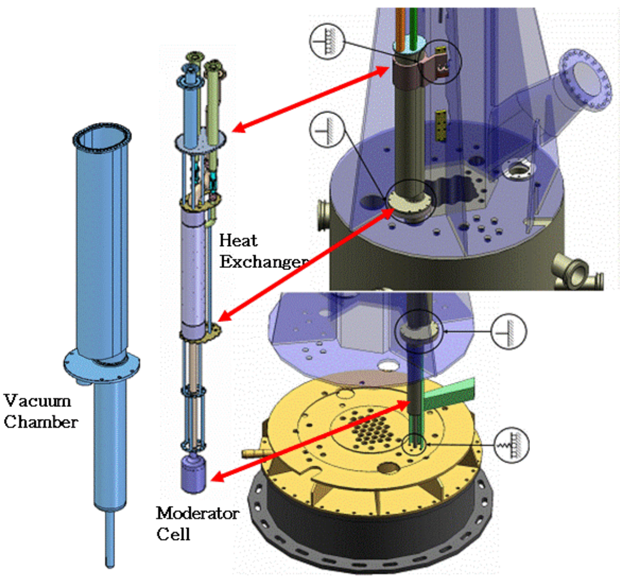
Schematics of IPA [2]



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System Compositions

- Hydrogen system
  - To supply hydrogen gases for moderator in IPA
- Vacuum system
  - To act as thermal insulation for the cold part of IPA
  - To act as safety barrier against the interruption of liquids and/or gases from outside the vacuum chamber
- Gas blanket system
  - To insulate the hydrogen gas from eventual contact with air and/or water
  - To prevent air / water entry to the IPA
- Helium refrigerator system
  - To supply low temperature helium to heat exchanger in IPA
- Electrical and I&C system

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System Design Status (1)

- Hydrogen System

  - Closed Loop Concept
  - Metal Hydride Unit for Hydrogen Removal from the System
  - Reactor Trip Signal at Hydrogen Pressure High/Low
  - Operation Modes :

Mode / System	Shutdown (SD)	Start-up (SU)	Normal (NO)
Reactor	Shutdown	Shutdown	Operation
Hydrogen	Vapor	Liquid	Liquid
Helium Refrigerator	Stop	Operation	Operation

- Helium Refrigeration System

  - Cooling Capacity : 2.0 kW
  - Cold parts → Foreign company, Warm Parts → Domestic company)
  - Automatic Control Scheme for Heat Load Fluctuations

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System Design Status (2)

- Vacuum System
  - Dual Redundant Configuration for System Reliability
  - 30 bars of Design Pressure for Outer Containment
  - Reactor Trip Signal at Vacuum Failure
- Gas Blanket System
  - Helium Blanket + Nitrogen Blanket
  - Comprehensive Pressure Monitoring Concept for Leak Detection
- Instrumentation and Control System
  - Dual Train of Trip Logic System
    . Digitalized Logic by PLC
    . Analog Logic by Hard-wired Relays
  - Digital Network to Plant Computer Systems
    . Integrated HANARO + CNS + He Refrigerator Computer

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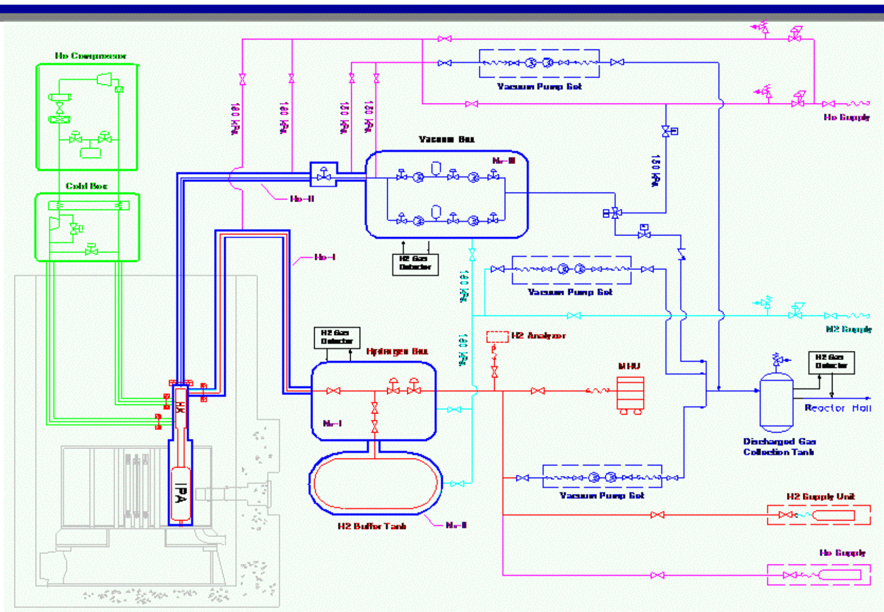
Flow Sheet of CNS System



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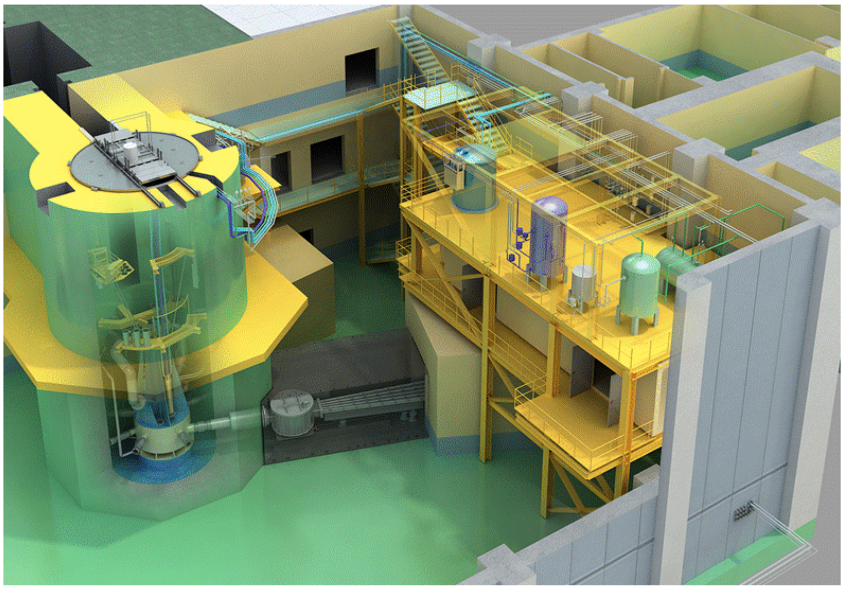
Final view of CNS equipment island



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Construction of CNL Building

- Construction
  - ✓ Detail design was completed in May 2005.
  - ✓ Construction permit was awarded by MOST in April 2006.
  - ✓ Construction was successively completed in May 2008.
- Future Plan
  - ✓ Licensing approval is expected in December 2008.
  - ✓ Scattering Instruments will be installed from the end of 2008.

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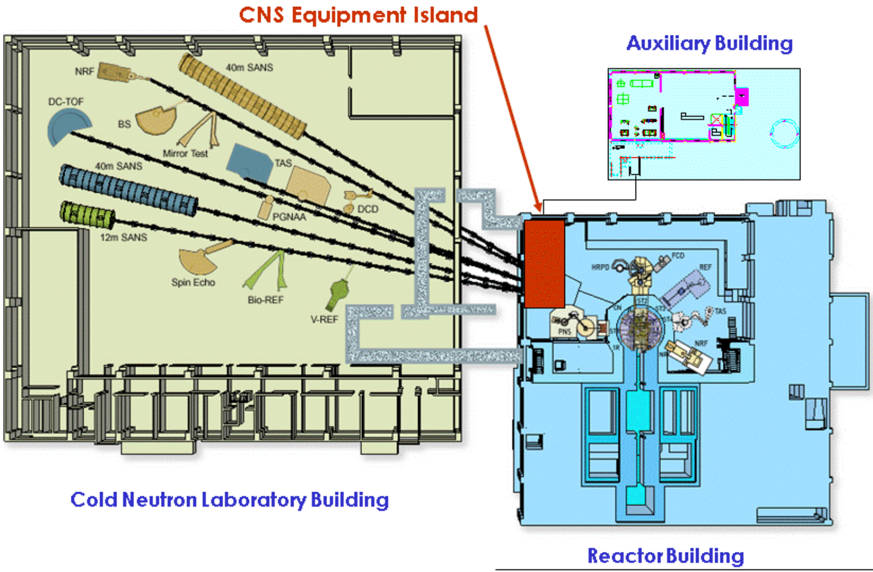
CNRF Building Layout



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Conclusions and Future Plan

- The CNS design should be guided as much by the safety aspects as by the needs of the user's group.
- The design objective is to make an efficient system, which can be operated safely with regards to the reactor and personnel.
- The Systems for CNS including the CNL building were successively installed in September 2008.
- The CNS commissioning will start at the end of 2008.
- The IPA will be installed in the first half of 2009.
- At the beginning of 2010, HANARO will provide initially a cold neutron source for international users.

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4. Advanced Irradiation Technology

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4.1 JOINING TECHNIQUES DEVELOPMENT FOR NEUTRON IRRADIATION TESTS AND POST IRRADIATION EXAMINATIONS IN JMTR-HL

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ABSTRACT

The hot laboratory (JMTR-HL) associated with the JMTR (Japan Materials Testing Reactor) was founded to examine the material and fuel specimens irradiated mainly in the JMTR. Through the post-irradiation examinations (PIEs) have been developed and performed, the JMTR-HL contributes not only to research of materials for light water reactors, fast reactors, high temperature gas reactor, and fusion reactor, but also to production of domestic industrial radio isotopes like ^{192}Ir. As the part of PIE technology development, several kinds of welding techniques have been systematically developed. These research and development of welding techniques such as circumference and sealing for irradiation capsules and rewelding with irradiated materials were implemented under the remote-controlled conditions in the hot cells. These techniques are very indispensable for the neutron irradiation tests and PIEs to be conducted in the JMTR.

INTRODUCTION

The hot laboratory (JMTR-HL) associated with the JMTR was founded to examine the material and fuel specimens irradiated mainly in the JMTR, and has been operated since 1971 [1]. The JMTR-HL is directly connected with reactor core by a water canal. Hence, irradiated radioactive capsules are efficiently transported under water through the canal in a short time. Through the post-irradiation examinations (PIEs) have been developed and performed, the JMTR-HL contributes not only to research of materials for light water reactors (LWRs), fast reactors, high temperature gas reactor, and fusion reactor, but also to production of domestic industrial radio isotopes like ^{192}Ir. As the part of PIE technology development, several kinds of welding techniques have been systematically developed [2]. These techniques are as follows;

(1) re-instrumentation of FP gas pressure gauge and thermocouple to an irradiated fuel rod,

(2) welding procedure development for re-capsuling of irradiated materials,

(3) joining technique and PIEs development of different materials with friction welding for new typed irradiation capsules,

(4) rewelding with irradiated and un-irradiated materials and fabrication of test specimen with the rewelding for fusion reactor development.

In this paper, these welding techniques are introduced and summarized.

RE-INSTRUMENTATION OF FOR AN IRRADIATED FUEL ROD

Generally, it is understood that FP gas release from UO_2 pellet fuels consists mainly of two processes, diffusional movement of gas atoms from grain interior to boundary and gas release from grain boundary to fuel rod free space through the formation of grain boundary bubbles and interlinked porosities. There is a growing demand for nuclear power plants to contribute to electric power supplies not only in conventional base load operations, but also in more flexible patterns like frequency control and higher burnup of fuel [3].

Therefore, power ramping tests in JMTR using BOCA (Boiling Water Capsule) were planned and performed for the purpose of safety research of load following operations on LWR fuels. The JMTR-HL developed an installing apparatus for irradiated fuel pins into the BOCA, an assembling apparatus for capsules, and a dismantling apparatus of capsules after irradiation [4-5]. In the power ramping tests, it is essential to comprehend the fuel-rod behavior under various neutron irradiation conditions. Especially, FP (Fission Product) gas release may be listed as one of the most important phenomena in this regard.

In this study, re-instrumentation devices of FP gas pressure gauge and thermocouple to an irradiated fuel rod were developed in the hot cell. Flow chart of re-instrumentation of FP gas pressure gauge and thermocouple and photograph of the apparatus are shown in Fig.1. From the development of remote-handling TIG (Tungsten Inert gas) welding machine, it is easy to clamp the tube and plug for instrumentation devices onto the machine, to adjust the welding position and to adjust the gap distance between the torch and the capsule.

The fuel rod re-instrumented FP gas pressure gauge and thermocouple was installed in the BOCA, and the BOCA was re-irradiated in the OSF-1 (Oarai Shroud Irradiation Facility) [6-7]. Figure 2 shows the detail of in-core part of BOCA/OSF-1 and results of power ramping test in JMTR. These tests were performed over the course of about 20 years from 1981 to 1999, and they contributed to safety research and achievement of high burn-up of LWR fuels. After refurbishment of JMTR, these ramping tests will be also examined following the FP gas pressures and welding techniques developments.

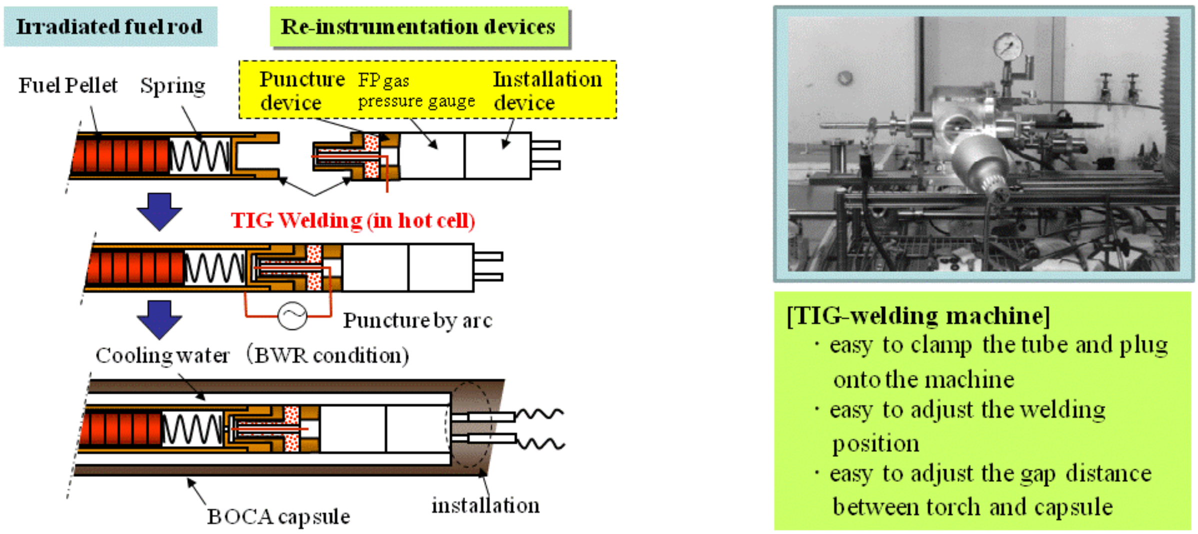


Fig.1 Flow chart of re-instrumentation of FP gas pressure gauge and thermocouple and photograph of the apparatus.

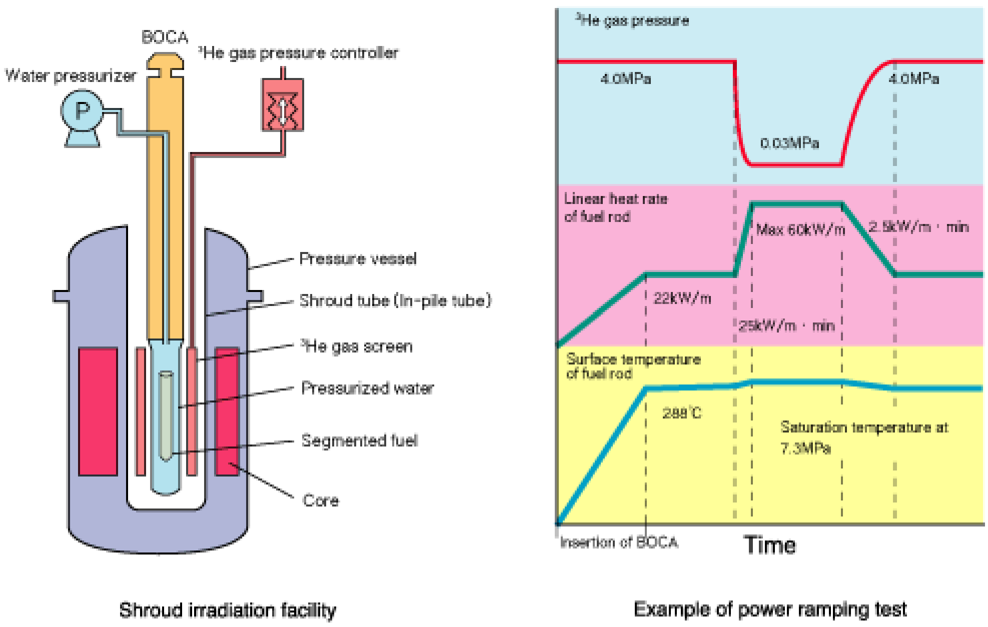


Fig.2 Detail of in-core part of BOCA/OSF-1 and results of power ramping test in JMTR.

WELDING PROCEDURE DEVELOPMENT FOR RE-CAPSULING

Irradiation assisted stress corrosion cracking (IASCC) occurred in stainless steel is considered to be one of the key issues from a viewpoint of the life management of core components in the aged LWRs. To simulate IASCC behavior in LWRs for PIEs, tensile tests are performed under high temperature and high pressure water conditions on specimens irradiated up to a neutron fluence higher than the so-called IASCC threshold fluence in a test reactor. Figure 3 shows the flow chart of re-capsuling for IASCC experiment in the hot cell [5,8]. In-pile crack growth and crack initiation studies in the JMTR were performed to evaluate factors affecting IASCC behavior. These studies were performed by in-pile IASCC test capsules that simulate LWR water conditions under irradiation. The results were compared with these of PIEs.

There were, however, some technical hurdles to overcome for the experiments. To perform in-pile IASCC tests, pre-irradiated specimens were relocated from pre-irradiation capsules to an in-pile test capsule in a hot cell by remote handling. Hence, a remote TIG welding technique was developed for assembling the in-pile test capsules [9-10]. Figure 4 shows the photograph of TIG welding equipment. The welding conditions of inner and outer tubes were obtained through a lot of cold mock-up tests. Figure 5 shows the result of cold mock-up welding test of the model tube with the same diameter and thickness of the inner tube. After the cold mock-up tests, the in-pile SCC capsule was assembled in the hot cell (Fig.5).

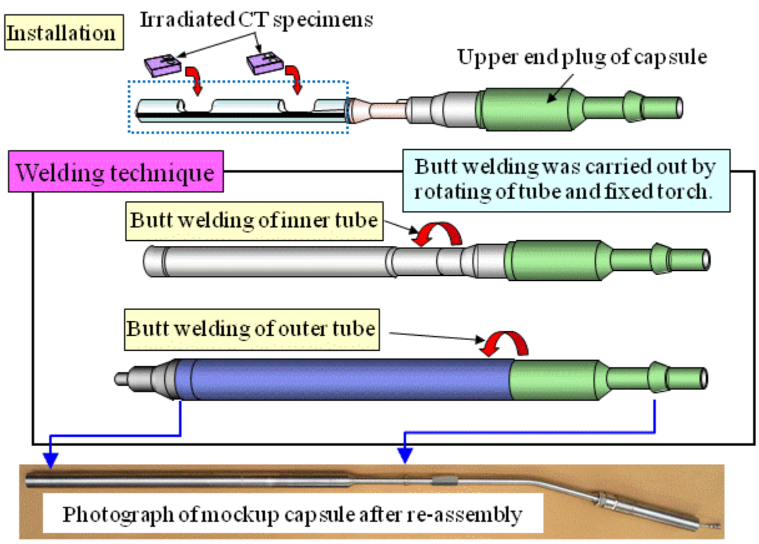


Fig.3 Flow chart for re-capsuling for IASCC experiment in the hot cell.

[TIG-welding machine]
· easy to clamp the tube and plug onto the machine
· easy to adjust the welding position
· easy to adjust the gap distance between torch and capsule

Fig.4 Photograph of TIG welding equipment.

The helium leak test, the liquid penetrant test and the visual observation of the in-pile IASCC capsule assembled in the hot cell were remotely performed remotely as the final inspection before the re-irradiation test in JMTR. From the results and experiences, eight in-pile IASCC test capsules were assembled in the hot cell.

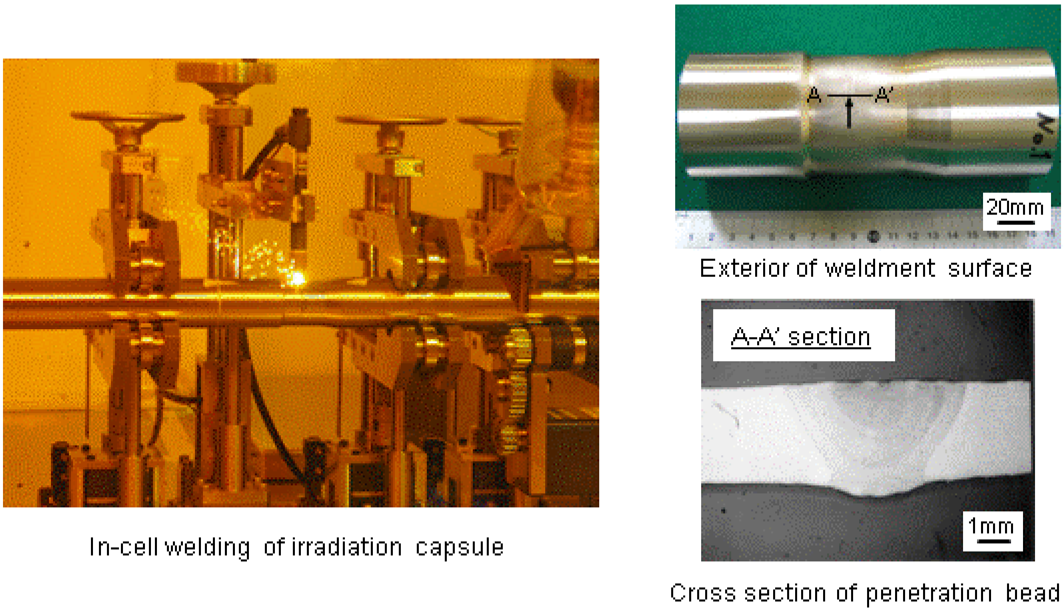


Fig.5 Result of cold mock-up welding test of the model tube.

JOINING TECHNIQUE OF DIFFERENT MATERIALS

Aluminum alloys, niobium alloys such as Nb1%Zr and zirconium alloys such as Zry-2 and Zry-4 are selected as the advanced materials because of heat resistance and low activation material. On the other hand, copper alloys such as Al_2O_3-dispersed coppers and CuCrZr, and titanium alloys such as Ti6Al4V are candidate structural materials for fusion reactors [11-12].

Technology development for joining advanced materials and stainless steel has been investigated for diffusion bonding, brazing, roll bonding, explosive bonding and hot isostatic pressing (HIP). On the other hand, a friction welding is one of most popular welding procedures for the joint of different materials used as a tube [13]. In the JMTR, joints of various materials and stainless steel fabricated by the friction welding were developed and used for structural components as pressure boundaries of an irradiation capsule against the primary coolant of the JMTR [14].

A break-typed friction welding machine was employed throughout. A description of the friction welding procedure is shown in Fig.6. The welding conditions, i.e. friction pressure, friction time, upset pressure and upset time, were determined by preliminary fabrication tests since these conditions influence mechanical properties of the joints.

In this study, the welding technique and characteristics of Nb1%Zr/SS304 joints fabricated by a friction welding were introduced. Nb1%Zr and SS304 rods of 15mm diameter were used and welding conditions were determined. The welding was made with a rotational speed of 2000rpm under a friction pressure of 1.5-2.0MPa. Upset pressure was the range from 2.5 to 3.0MPa and welding time was 0.1 sec. After friction welding, the burr glowed at welding was removed for fabrication of the test specimens by the lathe. Characteristics of the Nb1%Zr/SS304 joints were evaluated before/after neutron irradiation. For example, Nb1%Zr/SS304 joints broke at the part of base material (Nb1%Zr) in the tensile tests before neutron irradiation test. Each tensile strength of the joints was almost the same as that of Nb1%Zr at 20, 300 and 500˚C. In the torsion fatigue test, Nb1%Zr/SS304 joints also broke at the part of base material. From the results of these tests, the strength of Nb1%Zr/SS304 joints was almost similar to that of Nb1%Zr alloy within the error of 10%. On the other hand, the effect of neutron irradiation on mechanical properties of Nb1%Zr/SS304 joints was evaluated. Irradiation temperatures of Nb1%Zr/SS304 joints were 50 and 500˚C, and fast neutron fluence (E>1.0MeV) was the range from 2.6×10^{24} to 3.8×10^{24} /m^2. The tensile strength of Nb1%Zr/SS304 joints as a function of test temperature is shown in Fig.7. All irradiated specimens broke at the part of base material (Nb1%Zr) and the tensile strength of the irradiated joints was almost similar to that of the irradiated Nb1%Zr [11, 15-16].

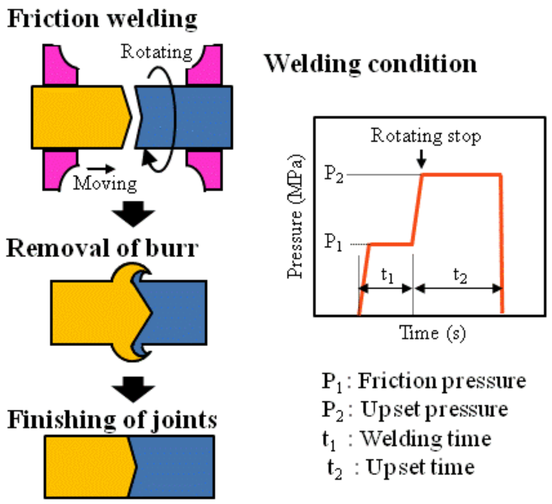


Fig.6 Description of the friction welding procedure.

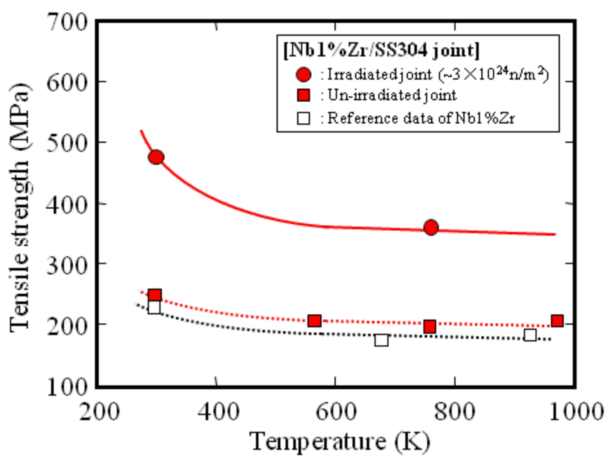


Fig.7 Tensile strength of Nb1%Zr/SS304 joints as a function of test temperature.

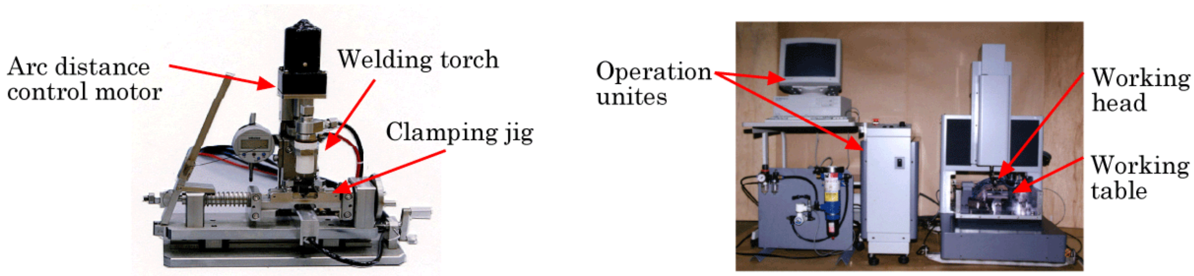
It was obvious that the Nb1%Zr/SS304 joints by friction welding were preferable joints for the capsule structure and cooling tube for the special irradiation capsules. From the development of friction welding with different materials, these joints are used in the irradiation capsules of JMTR and the Cu alloys/SS joints by friction welding have been proposed in the cooling tubes of the fusion blanket.

REWELDING WITH IRRADIATED AND UN-IRRADIATED MATERIALS

On the maintenance and/or replacement of the fusion blanket, new cooling pipes of the blanket will be joined by the welding to existent cooling pipes which were irradiated by high energy neutrons [17-19]. Therefore, it is necessary to evaluate the effect of helium generation by neutron irradiation on mechanical properties of the weldment. In this study, all work from the welding and processing of test specimens to the installing of the specimens into the irradiation capsule is carried out in the hot cell by remote handling operation because of the irradiated materials. The remote-handling type welding and processing devices and assembling procedure of irradiation capsules were developed based on the experiences of the PIEs in the “JMTR-HL”.

Using the austenitic stainless steel of type SS316-LN-IG (IG means “ITER” grade) which was irradiated up to the ITER irradiation condition in the JMTR, the TIG welding between the irradiated and un-irradiated materials was tested in the hot cell. The remote-handling type Tungsten inert gas (TIG) welding machine (see Fig.8(a)) was developed. In order to evaluate correctly the mechanical properties of the weldment, the plate with the tab where the Crater part was able to be removed was adopted as shape of the test specimens. On the other hand, the processing machine (see Fig.8(b)) with the end-milling type numerical control (NC) lathe was developed. It is possible to fabricate the weldment specimens with high dimension accuracy up to 50μm. These developments of remote-handling techniques enabled a systematic evaluation on mechanical properties of the weldment specimens with irradiated and un-irradiated materials.

From the results of the preliminary tests, it was obvious that the sealed gas and the welding heat input affected to the welding defects such as a crack at the surface and the cross section of weldment, an under cut and an insufficient welding. Figure 9 shows the dependence of the character of weld-induced defects on helium content and weld heat input [20]. It was found that when the welding heat input was chosen a value between 1 and 2 kJ/cm, good weldments without a crack was obtained.



(a) TIG-welding machine **(b) Processing machine**

Fig.8 Technical development of a remote-handling type welding and processing machines.

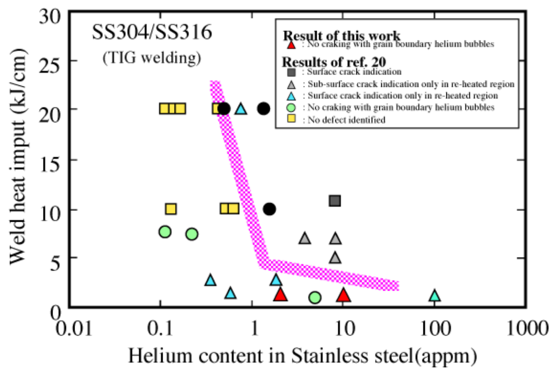


Fig.9 Dependence of the character of weld -induced defects on helium content and weld heat input.

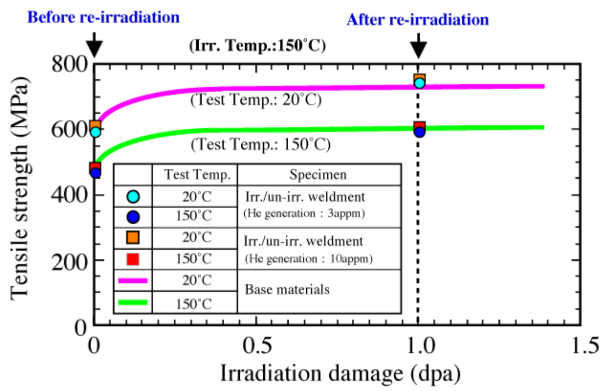


Fig.10 Results of tensile tests of weldments before and after the re-irradiation tests.

Furthermore, their welding test specimens were processed as the tensile-type weldment specimens and they were installed onto the re-irradiation capsule in the hot cell. After the re-irradiation test, PIEs of these weldments were carried out. Results of tensile tests of weldments before and after the re-irradiation tests are shown in Fig.10. The neutron re-irradiation test succeed in “JMTR” and the mechanical properties of re-irradiated weldments were clarified for the first time in world.

CONCLUSIONS

These research and development of welding techniques such as circumference and sealing for irradiation capsules and rewelding with irradiated materials were implemented under the remote-controlled conditions in the hot cells. These techniques are very indispensable for the irradiation tests and PIEs to be conducted in the JMTR. In the development of re-instrumentation for the irradiated fuel rods and welding procedure for re-capsuling of irradiated materials, the prospects are bright for new typed-irradiation tests and PIEs in JMTR using the irradiated materials and fuels for LWRs and advanced fission reactors. In the development of joining with different materials and re-welding with the irradiated materials, the prospects are bright not only for new typed-irradiation tests in JMTR, but also for research and development of maintenance for LWRs and fusion reactors.

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4.2 DEVELOPMENT OF FUEL TEST LOOP IN HANARO

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ABSTRACT

KAERI have developed a fuel test loop facility to conduct the fuel irradiation test at HANARO. Maximum 3 pins of fuel can be tested in the IR-1 irradiation hole of HANARO under commercial power plant operating conditions. The integral system performance test with mock-up fuels under a high temperature is being performed. The FTL will be used for an advanced fuel irradiation test and could maximize the usage of HANARO.

INTRODUCTION

The FTL (Fuel Test Loop) is a facility which can conduct a fuel irradiation test at HANARO. The FTL simulates commercial NPPs' operating conditions such as their pressure, flow, temperature, neutron flux levels and chemical conditions to conduct the irradiation and thermo-hydraulic tests. The conceptual design of the FTL was started at the end of 2001 and the detailed design had been finished by March 2004. The equipment installation of the FTL was successfully completed in March 2007. The commissioning of the FTL is being performed since April 2007. The FTL will be used for the irradiation test of advanced PWR fuels after its commissioning is completed. In this paper, the characteristics and commissioning of the FTL facility are introduced.

CHARACTERISTICS OF FTL FACILITY

Process systems [1,2,3,4]

The FTL provides the test conditions of a high pressure and temperature similar to those of commercial PWR and CANDU reactors. The FTL is composed of an OPS (Out Pile system) and an IPS (In-Pile test Section). The OPS is composed of a process system and an I&C (Instrumentation and Control) system. The IPS is to be loaded into the IR-1 position in the HANARO core. The FTL coolant is supplied to the IPS at the required temperature, pressure and flow conditions that are consistent with a test fuel. The nuclear heat added within the IPS is removed by the main cooling water. The main system of the FTL is designed by ASME Boiler & Pressure Vessel Section III code and its subsidiary system is designed by ASME Boiler & Pressure Vessel Section VIII code and the ASME B31.1 Power Piping code.

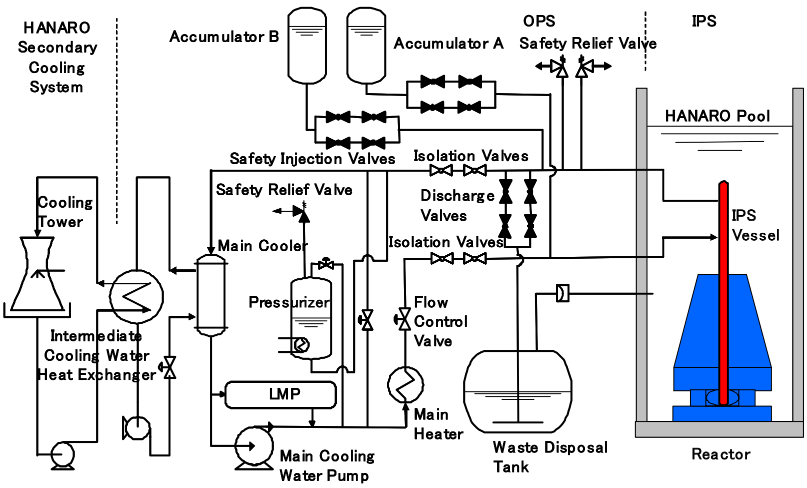

Fig. 1. Schematic diagram of the FTL.

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

- Main cooling water system
- Emergency cooling water system

- Penetration cooling water system
- Letdown, makeup, and purification system
- Waste storage and transfer system
- Intermediate cooling water system
- Sampling system
- IPS inter-space gas filling and monitoring system
- Miscellaneous systems

The main cooling water system controls and regulates the system pressure, temperature and flow rates of the coolant. It removes the fission and gamma heat of the IPS for a normal operation. The emergency cooling system is provided to maintain the experimental fuel cooling conditions in the event of an anticipated operational occurrence or the design basis accidents. The emergency cooling water system will inject the emergency cooling water directly from an accumulator to the main cooling water system if a line break accident occurs. The emergency cooling water system consists of two accumulators, safety injection valves, depressurization vent valves, and associated pipes. Accumulator A and B are connected by the safety injection pipes to the hot and cold legs of the main cooling water system respectively. The waste storage tank is connected by the depressurization vent pipe to the hot leg. As shown in figure 1, the pipes are designed as two trains for a redundancy. Each train has two valves in series because the accumulators and the waste storage tank should be completely isolated from the main cooling water system for a normal operation. All the equipment of the emergency cooling water system is designed as nuclear safety class 2. Seismic and electrical design is associated with the nuclear safety classification. The coolant of the accumulators is pressurized with nitrogen gas. The accumulators supply emergency coolant to the IPS for about 30 minutes. To prevent an injection of nitrogen into the IPS, the water level is measured with 3 sensors and the safety injection valves are closed if a low level trip occurs. The low level trip is actuated by the 2 out of 3 logic. The safety injection and depressurization valves are a solenoid-operated valve and the stroke time is 0.2 second.

The penetration cooling water system circulates the HANARO pool water to cool down the concrete penetration parts. The let down, make-up & purification system controls the volume, purification and chemical quality of the main cooling water. The waste storage and transport system collects the waste water and gas from the OPS and transports them to either the HANARO liquid radwaste system or the HANARO ventilation system for a normal operation. The waste storage tank also receives the discharges from the safety relief valves and the emergency coolant from the accumulators for a suppression of all the design basis events. The intermediate cooling water system transfers the fissile and pump heat to the HANARO secondary cooling system. The test loop sampling system monitors the water quality periodically. The IPS inter-space gas filling and monitoring system provides neon gas to the pressure vessel gap and provides air gas to the in-pool pipe gap to insulate them from the pool water. The hydrogen control system supplies hydrogen gas to the de-gasfier to remove the solved oxygen in the cooling cooler. The test conditions for the fuel rest loop are given in Table 1.

Table 1. Test conditions in the FTL.

Test conditions	Value
Reactor operation cycle	9 cycles/year
Operation cycle length (EFPD/cycle)	28 days
Test rods	3 rods
LHGR	≤ 320 W/cm
Peak to average heat rate	≤ 1.16
Fast neutron flux in cladding surface	1.2×10^{14} n/cm^2·sec
Coolant temperature	300 ~ 308 °C
Coolant pressure	150 ~ 159 kg/cm^2
Coolant velocity	1.37 ~ 1.84 kg/s
B concentration	≤ 1500 ppm
Dissolved oxygen concentration	≤ 0.1 ppm
pH at 300 °C	5.5 ~ 8.0

I&C systems [5,6]

The I&C system for the FTL is divided into a safety control system and a non-safety control system. The I&C system has the following functions;

- Maintaining the irradiation test conditions by an automatic control,
- HANARO trip and a FTL safe shutdown during transient or accident conditions,
- Satisfaction of the safety design requirements such as a redundancy, independency and single-failure criterion,
- Simultaneous operation of the FTL with HANARO.

The safety control system is used for controlling the safety related process systems of the FTL and a shutdown of the HANARO reactor from abnormal operating conditions. The non-safety control system consists of a computer control system and a data acquisition system. The digitalized computer control system controls and monitors all the field signals from the process systems such as the main cooling water system, the intermediate cooling water system, etc. The safety related control panels are classified as Quality class “Q” and Seismic category “I”. The safety related control panels were designed with the following safety regulation of IEEE std-603 to ensure a system’s reliability such as single failure criterion, redundancy, independence, diversity, fail-safety design, manual initiation, channel checks, channel bypass, identification of protective action, interface with a non safety related system, equipment qualification, etc. The FTL protection panels composed of three channels receive the signals from the corresponding field instruments, and generate the HANARO trip signal and the FTL shutdown signal if the measured signal exceeds the trip setpoint. The HANARO trip signals from the protection panels are interfaced with the corresponding channels of the HANARO RPS (Reactor Protection System) panels which generate the reactor trip signal. The HANARO RPS panels have a '2 out-of 3' local coincidence logic for a reliability assurance. The FTL safety control panels are composed of two independent panels, and they have some manual switches and relays in each panel to control the safety related process systems. The main purpose of the safety control panels is to supply the emergency cooling water to remove the heat from the test fuels after a reactor shutdown. Fig. 2 shows the overall control system configuration for the FTL.

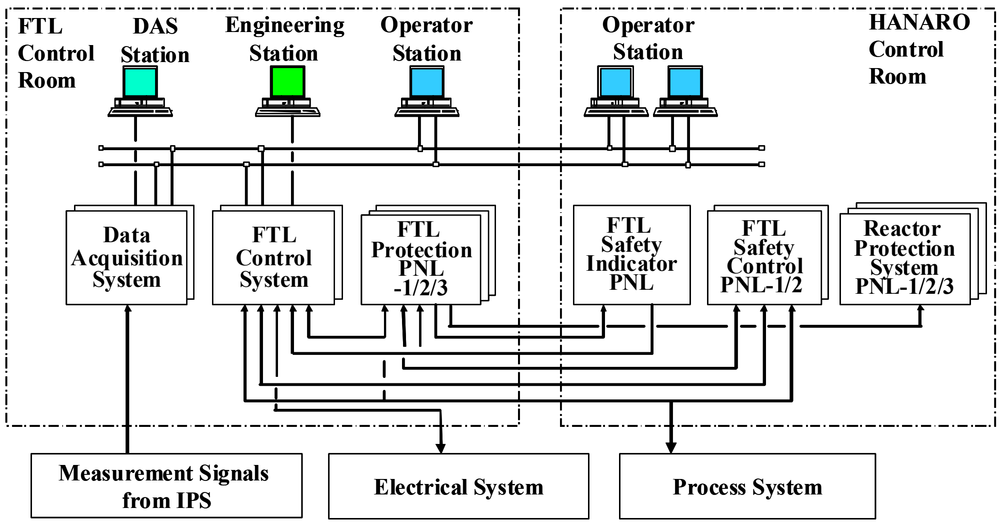

Fig. 2. Overall control system configuration.

The data acquisition system collects and stores the signals from the in-pile instruments (SPND, Thermocouple, LVDT, etc.) installed in the IPS. The main measurement parameters are the centerline temperature of a test fuel, the neutron flux, the coolant temperature, the fission gap pressure, etc. The irradiation data can be monitored in office building located in reactor outside on a real time basis through the network.

IPS(In-Pile test Section)

The IPS including the test rig is to be loaded into the IR-1 position in the HANARO core. This implies that the environment around the IPS is subjected to a high neutron flux (Thermal neutron flux: 1.2×10^{14} n/cm^2·sec, Fast neutron flux: 1.6×10^{14} n/cm^2·sec). The IPS can accommodate up to 3 pins of fuel and has instruments such as a thermocouple, LVDT and SPND to measure a fuel's performance during a test. The IPS is composed of the IPS head, the outer pressure vessel, the inner pressure vessel, a flow divider and a test fuel carrier. Inlet nozzle and outlet nozzle for the main cooling water are located in the IPS head and insulated from the HANARO pool. Neon gas is filled into the gap between the outer pressure vessel and the inner pressure vessel to insulate the IPS from the HANARO pool. A flow divider divides the outlet cooling water from the inlet cooling water. The test fuel carrier is composed of a fuel carrier support stem (with 6 slots for the hot cooling water injection), a fuel carrier leg (3 legs are arranged through the 120° angles) and a fuel carrier head. Fig. 3 shows a schematic diagram of the IPS. The outer pressure vessel is a 321 stainless steel of a 5.0 mm thickness and has 9 SPNDs. Inner pressure vessel is a 321 stainless steel of a 4.0 mm thickness and its upper part is designed as a collar form to prevent a cooling water stagnation.

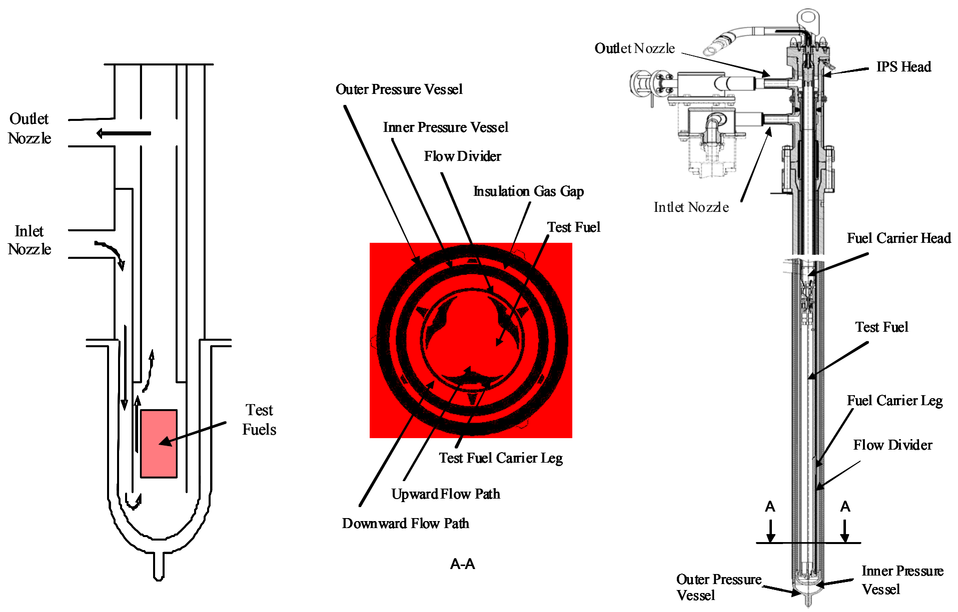

Fig. 4. Schematic diagram of the IPS.

Commissioning

The equipment of the FTL was installed from July 2006 to March 2007. The commissioning was started from April 2007. The commissioning is performed with three stages. An individual system performance test under room temperature is performed in the first stage, and the integral system performance test with mock-up fuels under a high temperature is performed in the second stage, and finally the integral system performance test with test fuels under a high temperature is performed in the third stage. The individual system performance test had been successfully completed. The integral system performance test with mock-up fuels under a high temperature is being performed. The passivation operation was performed at the starting point of the FTL operation under the high temperature condition. The integral system performance test with test fuels under a high temperature will be performed from December 2008. Figure 4 shows the pictures for the commissioning.

Fig. 4. Pictures for the commissioning.

CONCLUSIONS

KAERI have developed the FTL to conduct the fuel irradiation test at HANARO. The IPS which shall be loaded in the IR1 irradiation hole has a double pressure vessel and is designed to accommodate up to 3 pins of fuel. The application fields of the FTL are as follows;

- Nuclear fuel irradiation behavior test at the operating conditions of a commercial power plant,
- Fuel burn-up and mechanical integrity verification,
- Irradiation data generation for an analysis model,
- Technical improvement of a design and a fabrication for an advanced fuel development.

The FTL will be used for an irradiation test of advanced PWR fuels after its commissioning is completed in the end of 2008. The R&D for the irradiation test technologies will be progressed in the future.

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4.3 SAFETY RESEARCH PROGRAM OF LWR FUELS AND MATERIALS USING THE JAPAN MATERIALS TESTING REACTOR

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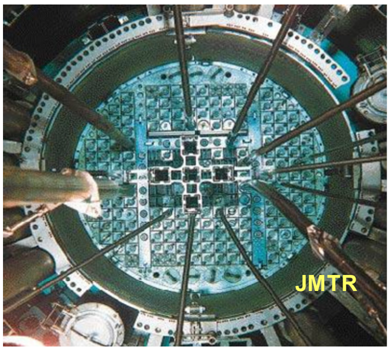
ABSTRACT

Power up-rates, burn-up extension and long term operation enable us to utilize Light Water Reactors efficiently. This will have the fuels and structural materials exposed to severe operational condition for a longer period, which can affect their integrity. Continuous researches for solving irradiation-related issues on the fuels for high-duty uses and the plant aging are essential in order to realize the up-graded uses of LWR safely. Japanese regulator has decided to install new irradiation test facilities in the Japan Material Testing Reactor (JMTR) at the Japan Atomic Energy Agency (JAEA). For the fuels tests, transient tests facility is being constructed for the power transient tests of new design BWR fuels. For the materials tests, the irradiation test loops under well controlled environment simulating BWR water chemistry condition and a large irradiation capsule, which can accommodate 1 inch-thickness compact tension specimens in an inert gas environment, are being prepared for the researches on stress corrosion cracking and irradiation embrittlement, respectively. These fuels and materials irradiation tests will be started in 2011 after refurbishment of JMTR.

Safety Research Program of LWR Fuels and Materials Using the Japan Material Testing Reactor



A part of the program is being conducted under a contract with the Nuclear and Industrial Safety Agency of the Ministry of Economy, Trade and Industry.

Nov. 5-7, 2008
2008 KAERI/JAEA Joint Seminar

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Back ground

For the up-graded use of LWRs, transition to burn-up extension, up-rating, modified water chemistry, etc. for life time extension of plants are essential.

This causes

Fuels and materials are exposed to severe operational condition which can affect their integrity, due to excess corrosion, embrittlement of materials, hydrogen absorption of cladding, and so on.

Therefore

A new safety research program will be performed to answer the regulatory and developing requirements for reliable and economical uses of LWRs in the long term and up-graded operations using the JMTR.

3

Fuels and Materials Irradiation Plan

Fuel Irradiation Tests

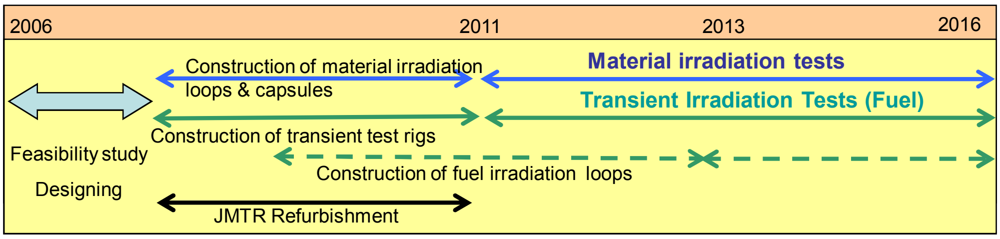
- Transient tests under power ramping conditions (Facility preparation in progress)
- Irradiation tests under high duty conditions, such as high rod internal pressure, uprated power and high burnups (Design finished and proposed)

Material Irradiation Tests

(Facility preparation in progress)

- Stress Corrosion Cracking (SCC) tests under irradiation conditions
- Material irradiation tests of hafnium under high temperature water condition for corrosion, etc.
- Fracture toughness tests of reactor pressure vessel (RPV) materials with 1T-CT specimens

Schedule



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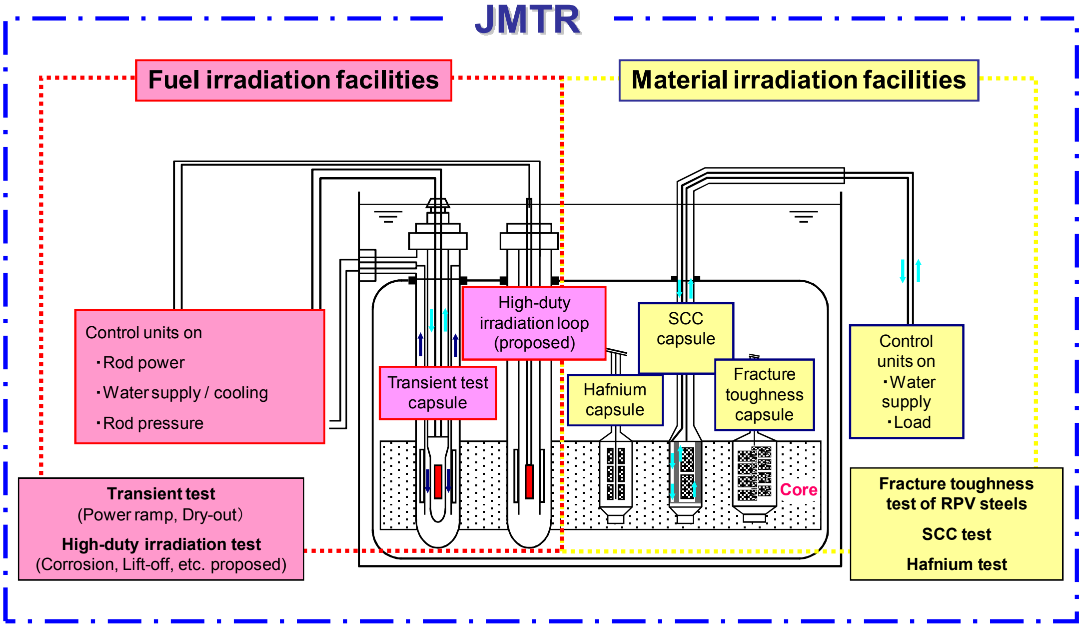
Fuels and Materials Irradiation Plan



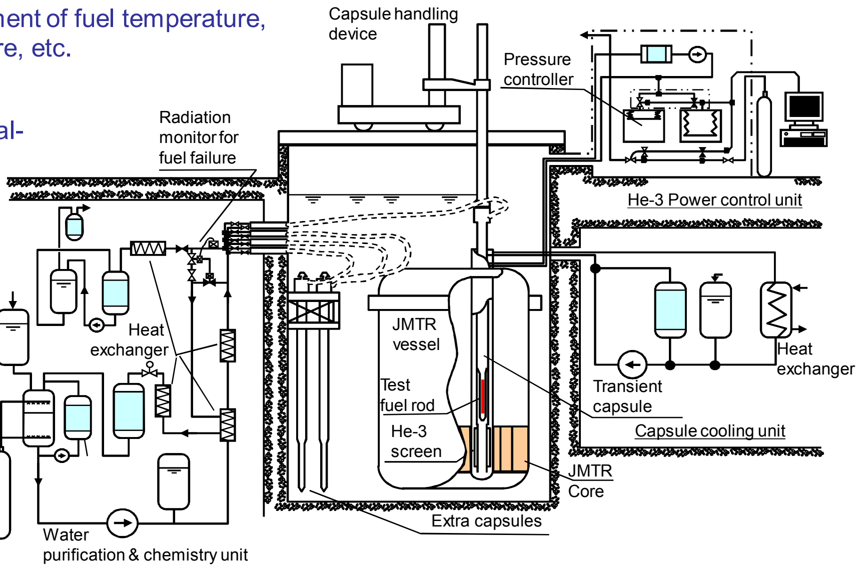
Transient Tests (Preparation on-going)

5

Types of capsules for power and cooling transients are being designed.
(Natural and forced convection capsules for power transients etc.)

On-power handling and independent power control system for effective operation

- Online measurement of fuel temperature, rod internal pressure, etc.
- Effective capsule handing in the canal-connected hot laboratories followed by detailed pre/post-test examination
- Power transient tests of high burnup BWR fuel to be started in year 2011.

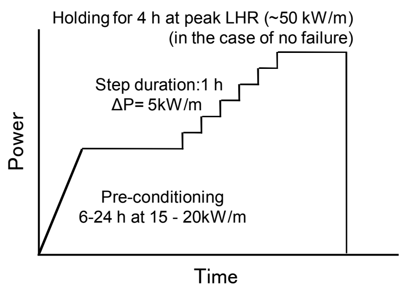


Power transient tests under LWR conditions

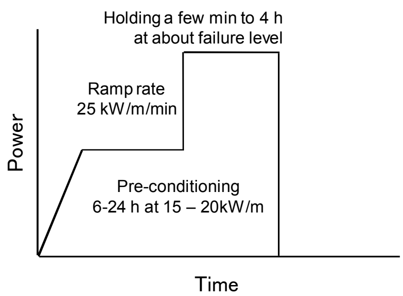
6

[Cooling conditions]
- ✓ Natural circulation capsule, well experienced for BWR tests (preparation on going)
- ✓ Forced convection capsule (proposed for future tests)

[Power ramping mode]
- ✓ Power ramping 1 : Multiple step ramp mode for scoping tests
- ✓ Power ramping 2 : Single step ramp mode to obtain the fuel failure threshold.



Multi step ramp



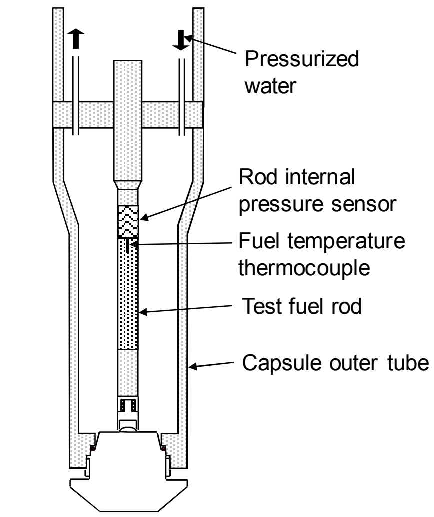
Single step ramp

Expected max. liner heat rates for transient tests
about 55kW/m for BWR & PWR fuel rods at 70GWd/t

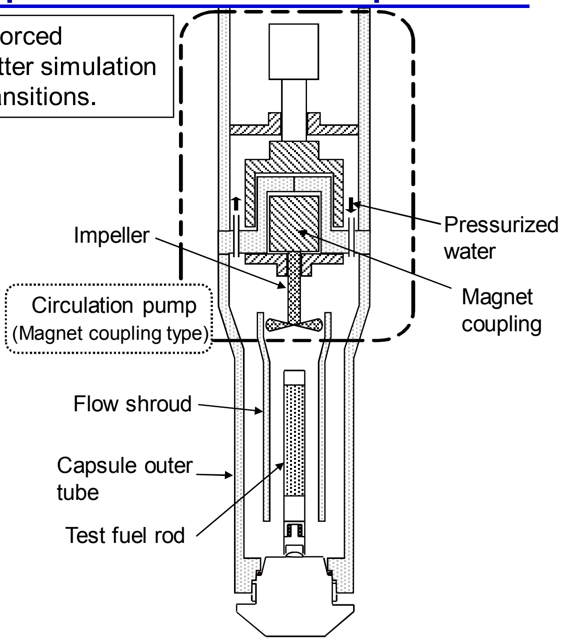
Schematic configuration of power transient test capsules

7

In addition to natural convection capsules, a forced convection capsule is being developed for better simulation of LWR cooling conditions avoiding boiling transitions.



Natural convection capsule



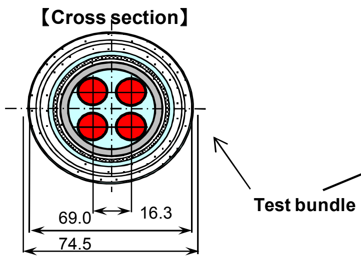
Forced convection capsule (proposed)

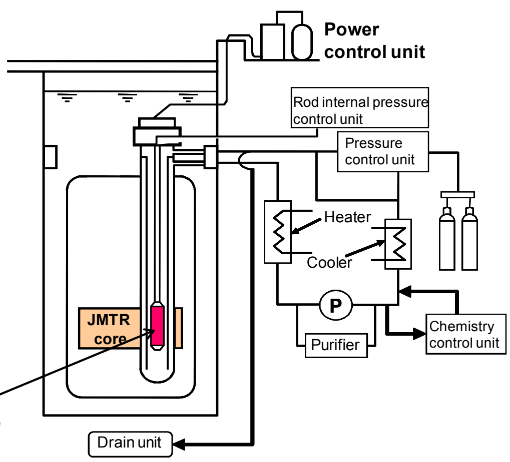
High duty Irradiation Test (proposed)

8

System designing has finished for UO_2 and MOX fuel bundle irradiation test rigs under high duty uses. Temperature and deformation behavior of fuel rods would be studied at high internal pressures, high burnups and high rated powers in the irradiation loop. The loops were designed to have chemistry units added for further tests, in order to simulate combined water chemistry effects e.g. NMCA: noble metal chemical addition, Zinc addition, hydrogen water chemistry).

- Online measurement of fuel temperature, rod internal pressure, etc.
- Long term irradiation of new fuel rods and lift off tests to examine rod internal pressure limits are being proposed.
- Burnup extension for further tests



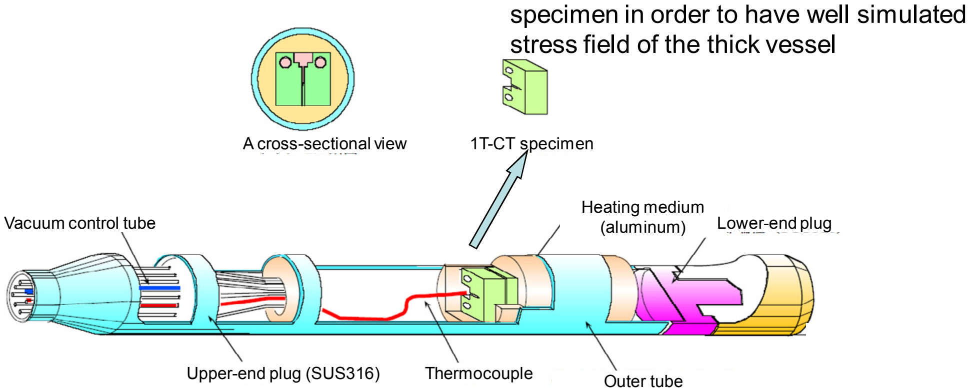


Fracture toughness tests

9

To evaluate irradiation embrittlement of the reactor pressure vessel steels for long term uses of LWRs, fracture toughness of the vessel materials will be investigated with larger size specimens.

Tests with thicker 1T-CT 25mm (←10mm) specimen in order to have well simulated stress field of the thick vessel



Test capsule to be installed in the JMTR

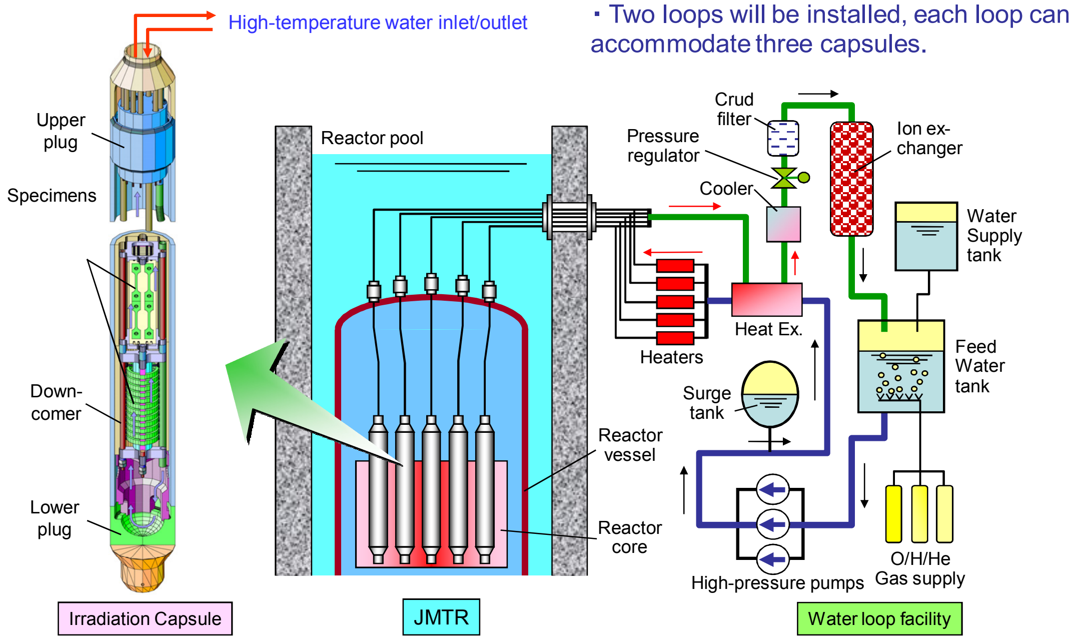
- Fast neutron (E>1MeV) flux : about 4 x 10^{16} n/m²/s
- fluence : above 5 x 10^{23} n/m² for 1T-CT

Material Irradiation Water Loops

- For SCC and corrosion tests -

10

- Two loops will be installed, each loop can accommodate three capsules.



In-pile SCC Tests

11

To obtain SCC data under simulated LWR conditions with irradiation effects. They should support laboratory–SCC data for evaluation in actual plants at various locations under various conditions.

(1) In-pile SCC growth test under simulated BWR condition

Materials*1	Fluence Level *2 (E>1MeV)	Size of Specimen	Range of Stress Intensity Factor, K ($MPa \cdot m^{1/2}$)	Irradiation Temperature	Water Condition*3
316L SS	Low ($<1\times10^{25} n/m^2$) High (1×10^{25} - $3\times10^{25} n/m^2$)	0.5T-CT (B=12.7mm) ／ 0.4T-CT (B=5.6mm)	10 - 30	288℃	NWC／HWC

*1 Other materials such as 304 SS and 304L SS are under consideration.

*2 Flux during in-pile tests is ~1×10^{16} - 3×10^{16} $n/m^2/s$, which corresponds to that for core shroud in BWR power plant.

*3 NWC: Normal Water Chemistry, HWC: Hydrogen Water Chemistry

(2) In-pile SCC initiation test under simulated BWR condition

Under consideration

Corrosion & irradiation behavior tests

12

To obtain integrity evaluation data of core components made of various materials, basic characteristics of the materials will be studied. Corrosion and irradiation growth of control rod material, Hf, irradiation tests are planned, which is a cause of SCC troubles in the Japanese BWR control rod blades.

Proposed test parameters

	Test range				
Material	Heat treatment	Impurity			
Corrosion test duration	200days	400days	600days		
Irradiation growth Fluence (n/m^2, >1MeV)	0.2×10^{26}	0.4×10^{26}	0.6×10^{26}	0.8×10^{26}	1×10^{26}

13

Summary

- ✓ Fuel safety research at JAEA will be started in FY2011 to examine fuel integrity under transient conditions using new test rigs designed for JMTR. Preparation of the test facilities and fuel transportation is in progress. In addition, fuel irradiation test loops are proposed for development of next-generation LWR fuels with high duty uses.
- ✓ The new tests would provide fuel failure criteria under the power transients and data to examine evaluation models on the fuel integrity.
- ✓ Material irradiation study will be performed to examine,
  - ･Fracture toughness of reactor pressure vessel steels
  - ･Stress Corrosion Cracking under simulated LWR irradiation field
  - ･Corrosion and irradiation growth of core components

  The irradiation studies would contribute not only to solve the current problems but also to identify possible seeds of troubles and to make proactive responses.

4.4 A BASIC DESIGN OF A DOUBLE CLADDING FUEL ROD TO CONTROL THE IRRADIATION TEMPERATURE OF NUCLEAR FUELS

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An instrumented capsule for a nuclear fuel irradiation test(hereinafter referred to as “instrumented fuel capsule”) has been developed to measure fuel characteristics, such as a fuel center and surface temperature, the internal pressure of a fuel rod, a fuel pellet elongation and neutron flux, during an irradiation test at HANARO. And six types of dual instrumented fuel rods, which allow for two characteristics to be measured simultaneously in one fuel rod, have been developed to enhance the efficiency of an irradiation test using an instrumented fuel capsule at HANARO. In the future, nuclear fuel irradiation tests under a high temperature condition are expected from users. To prepare for this request, we have continued developing the technologies for high temperature nuclear fuel irradiation tests at HANARO. The purpose of this paper is to control the temperature of nuclear fuels during an irradiation test at HANARO. Therefore we basically designed a double cladding fuel rod and an instrumented fuel capsule basically. The basic design of a double cladding rod was based on out-pile tests using mockups and the thermal analyses using some relevant codes. This paper presents the design and fabrication of the double cladding fuel rod mockups, the results of the out-pile tests, the results of the temperature calculation and the basic design of a double cladding fuel rod and an instrumented fuel capsule.

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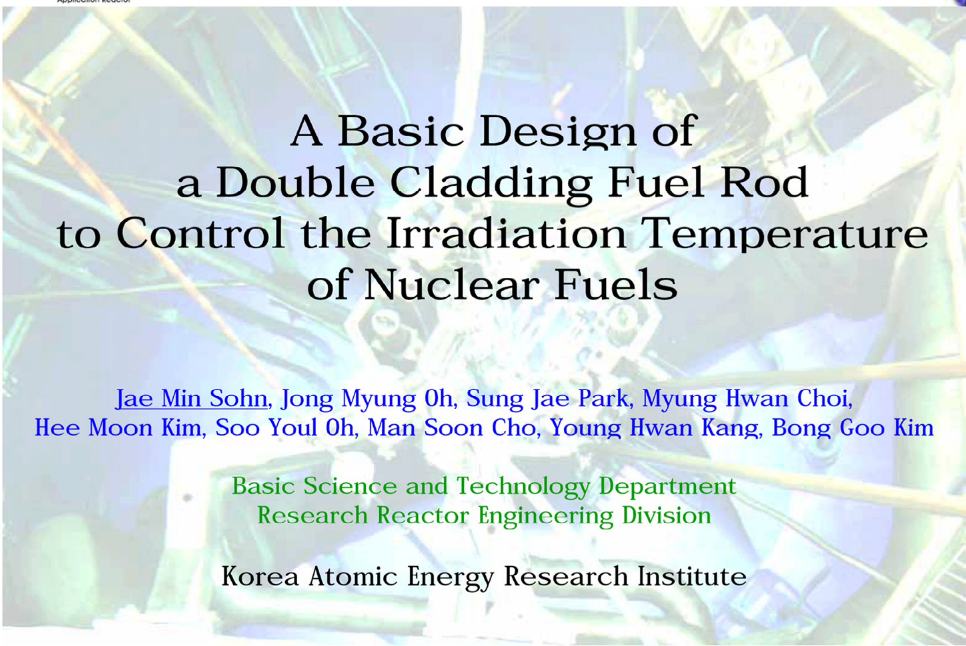

A Basic Design of a Double Cladding Fuel Rod to Control the Irradiation Temperature of Nuclear Fuels

Jae Min Sohn, Jong Myung Oh, Sung Jae Park, Myung Hwan Choi,
Hee Moon Kim, Soo Youl Oh, Man Soon Cho, Young Hwan Kang, Bong Goo Kim

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Introduction

- **Purpose**
  - ✓ To control the temperature of nuclear fuels during an irradiation test at HANARO
- **Major studies**
  - ✓ Out-pile tests using the mockups of the double cladding fuel rods
  - ✓ Basic design of a double cladding fuel rod
  - ✓ Design of an instrumented fuel capsule(07F-06K)
    => to install the two double cladding fuel rods
  - ✓ Nuclear properties and thermal analysis

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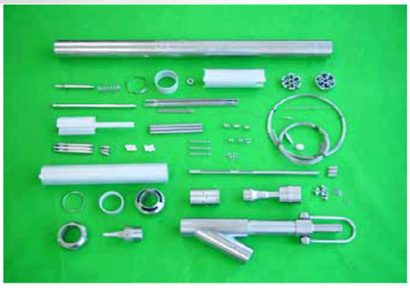
Standard Instrumented Fuel Capsule

- Test Hole : OR vertical experimental hole
- Total length : 4,939 mm(L)
- Materials & Dimensions
  - ✓ Outer tube : STS 316L
    726 mm(L), 56.0 mm(OD)
  - ✓ Protection tube : STS304
    3,679 mm(L), 42.5 mm(OD)
  - ✓ Cable Guide Tube(Spring Hose) :
    10,152 mm(L), 24 mm(OD)
- Instrumentations
  - ✓ Centerline Temp. of nuclear fuels
  - ✓ Surface Temp. of nuclear fuels
  - ✓ Internal pressure of nuclear fuel rod
  - ✓ Elongation of nuclear fuels
- Fuel rods
  - ❖ Three single or duel instrumentation fuel rods can be installed in a capsule



Components of Instrumented Fuel Capsule

Instrumented Fuel Capsule

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Experiences of Irradiation Tests at HANARO

◆ using the standard Instrumented fuel capsules

Irradiation Test Subjects	02F-11K	03F-05K	05F-01K
HANARO Power (MW)	24	24 ~ 30	30
Experimental Vertical Hole	OR5	OR5	OR5
Maximum Linear Power (kW/m)	53.2	50.1	47.43
Average Linear Power (kW/m)	49.2	46.3	44.73
Average Burn-up (MWD/MTU)	5,930	5,556	4,069
Effective Full Power Days	53.84	59.5	45.84
Maximum Temperature (℃)	1,375 (center)	1,316(center)	385 (surface)

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Instrumentations of Fuel Capsules

	Fuel Rod #1	Fuel Rod #2	Fuel Rod #3
02F-11K	Centerline Temp. of Nuclear Fuel	Centerline Temp. of Nuclear Fuel	Centerline Temp. of Nuclear Fuel
03F-05K	Elongation of Nuclear Fuel	Centerline Temp. of Nuclear Fuel	Internal Pressure of Nuclear Fuel Rod
05F-01K	Internal Pressure of Nuclear Fuel Rod & Surface Temp. of Nuclear Fuel	Centerline Temp. of Nuclear Fuel & Surface Temp. of Nuclear Fuel	Elongation of Nuclear Fuel & Centerline Temp. of Nuclear Fuel
06F-01K (Assembling)	Internal Pressure of Nuclear Fuel Rod & Centerline Temp. of Nuclear Fuel	Centerline Temp. of Nuclear Fuel (upper & lower)	Elongation of Nuclear Fuel & Surface Temp. of Nuclear Fuel

◆ 02F-01K & 03F-05K capsules contain three instrumented fuel rods.

◆ 05F-01K & 06F-01K capsules contain three dual instrumented fuel rods.

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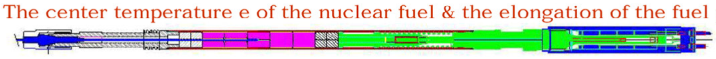
Dual Instrumented Fuel Rods

◆ Six different types of the dual instrumented fuel rods were developed to enhance the efficiency of the irradiation test using an instrumented fuel capsule

1. The center temperature of the nuclear fuel & the internal pressure of the fuel rod

2. The surface temperature of the nuclear fuel & the internal pressure of the fuel rod

3. The center temperature e of the nuclear fuel & the elongation of the fuel pellet



4. The center temperature e of the nuclear fuel & the elongation of the fuel pellet

5. The center & surface temperature of the nuclear fuel

6. The center temperature of the nuclear fuel of the upper & lower part

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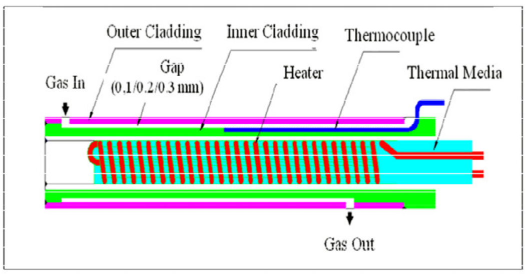
Out-pile Tests (1)

◆ Design and fabrication of mockups of double cladding rods
- ✓ Major components : outer & inner cladding, gas in & out sleeves, heater (heat source), thermocouple
- ✓ Three types of gap (between an outer cladding and an inner cladding) were designed => 0.1 mm, 0.2 mm, 0.3 mm
- ✓ Each mockups has a thermocouple to measure the temperature of an outside surface of an inner cladding
- ✓ The mixture of helium and neon gas flows through the gap between an outer and an inner cladding



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Out-pile Tests (2)

◆ Condition of the out-pile tests

- ✓ The coolant conditions was same as OR5 at HANARO : 40 degrees ℃, 12.7 kg/s
- ✓ The gas flow was fixed 50 cc/mim
  => because the gas flow has only a little effect on temperature
- ✓ The GSF-2002 control system was used
  => to control the heater power and the mixture ratio of helium and neon gas

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Out-pile Tests (3)

◆ Data was acquired during the out-pile tests every 1 sec.

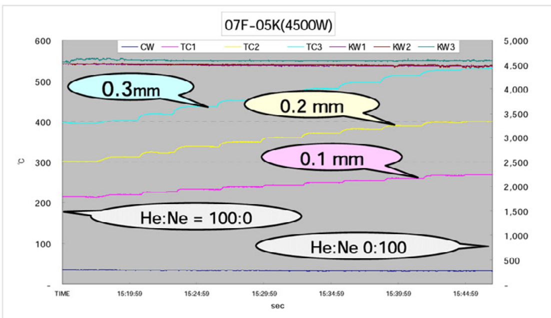
- ✓ Data contents an outside surface temperature of an inner cladding, a heater power, a coolant temperature

◆ Method of out-pile tests

- ✓ The heater powers of the three mockups were fixed at same level &
- ✓ Mixture ratio of helium and neon gas was changed in 10% increments from 0% to 100%
- ✓ Out-pile tests were performed at various temperatures

◆ Results

- ✓ The effects of a gap size and a gas mixture ratio on the temperature were analyzed.
- ✓ The measured maximum temperature was 531 degrees ℃ at 0.3 mm gap with 100% neon gas condition.



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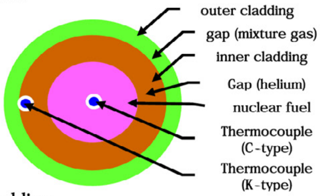
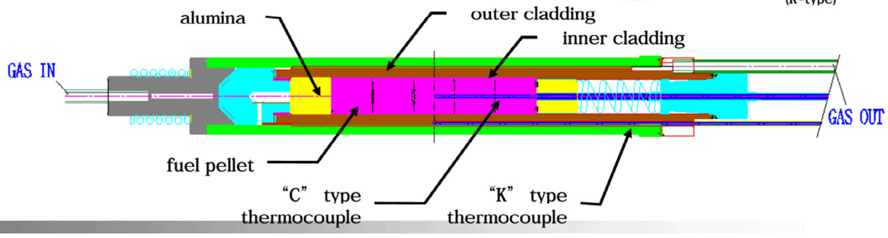
Basic Design of a Dual Cladding Rod

- Total dimensions : 18 mm (OD), 154 mm(L)
- Outer cladding : STS316L, 18 mm(OD), 109 mm(L), 1.72 mm(T)
- Inner cladding : STS316L, 14 mm(OD), 106 mm(L), 3.0 mm(T)
- Gap (between claddings) : 0.1 mm with helium
- Instrumentations
  - ✓ Centerline temperature of nuclear fuels
  - ✓ Outside temperature of an inner cladding
- Nuclear fuels (5 pellets/rod) characteristics
  - ✓ PWR 17x17 type, 2.42 w/o UO_2
  - ✓ Dimension : Φ8.19 mm X 10.2 mm





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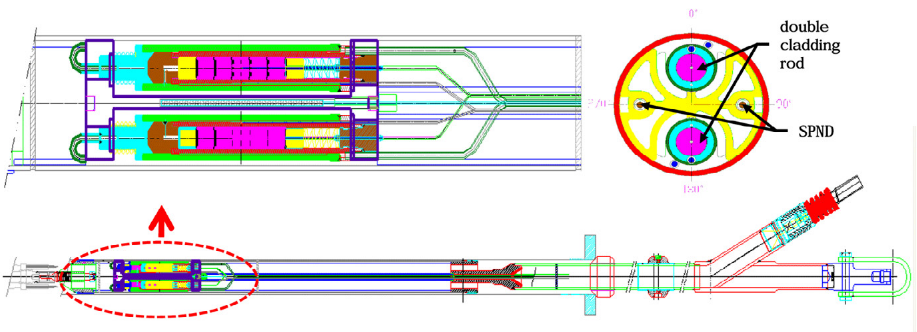
Design of a Fuel Capsule (07F-06K)

- 07F-06K capsule contains two double cladding rods & two SPNDs
- Test Hole : OR5 vertical experimental hole of HANARO
- Total length : 4,939 mm(L)
- Materials & Dimensions are same as standard instrumented fuel capsule



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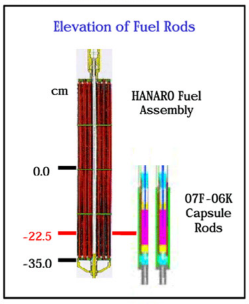
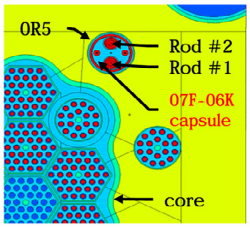
Nuclear Properties

- Calculation results of the linear power using MCNP
  - ✓ Elevation of the nuclear pellets of capsule
    : -22.5 cm from the center of HANARO fuel assembly

Pellet No. (from top)	Elevation (cm)		Rod #1 (W/cm)			Rod #2 (W/cm)		
	from	to	Fuel	Inner Cladding	Outer Cladding	Fuel	Inner Cladding	Outer Cladding
#1	20	21	339.8	21.64	16.37	303.2	19.43	14.72
#2	-21	-22	333.7	21.03	16.76	301.2	17.83	14.94
#3	-22	-23	333.2	21.06	16.61	283.1	17.48	14.45
#4	-23	-24	333.6	20.04	16.18	267.0	17.48	13.41
#5	-24	-25	328.7	17.95	15.41	286.3	16.34	14.00





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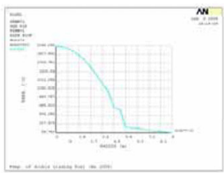
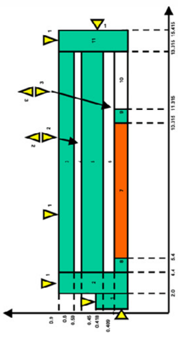
Thermal Analysis

- Calculation results of the temperature of fuel and claddings using ANSYS & Heating7.2

		ANSYS	Heating7.2
Nuclear fuel	Centerline	2,089 ℃	2,190 ℃
	Surface	977 ℃	1,043 ℃
Inner Cladding	Inside	622 ℃	690 ℃
	Outside	458 ℃	540 ℃
Outer Cladding	Inside	118 ℃	237 ℃
	Outside	59 ℃	64 ℃





- The calculation results of ANSYS are similar to Heating
- The calculation results of centerline temperature are less than the melting point of UO_2 (2,827 ℃)

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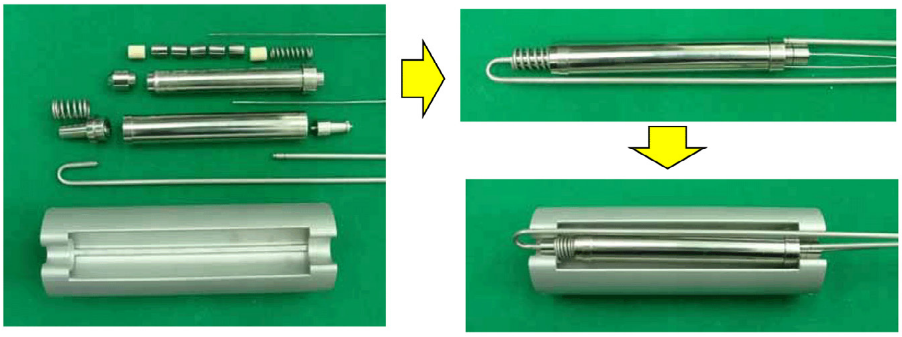
Assembly and Parts

◆ Fabricate the parts of the double cladding rod to test

- ✓ assembly procedure
- ✓ welding or brazing conditions



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Summary

◆ A double cladding fuel rod has been successfully designed.

◆ The double cladding fuel rods will be irradiated at the end of the 2009 at HANARO.

◆ The double cladding fuel rods will be used a high temperature irradiation test of nuclear fuels.

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4.5 DESIGN AND FABRICATION OF A CAPSULE FOR A MATERIAL IRRADIATION IN AN OR HOLE OF HANARO

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ABSTRACT

This paper focuses on the design of a capsule for a material irradiation that is supposed to be loaded in an OR test hole in HANARO. Nuclear characteristics for the test holes of HANARO were analyzed for this purpose. A test for a pressure drop and a vibration of this capsule was performed before manufacturing a material capsule suitable for an OR hole in HANARO. In the basic design, capsules with an outside diameter of 50, 52, 54, 56mm were reviewed theoretically to establish if they met the hydraulic requirements in HANARO. It was estimated that the diameter of a capsule for an OR holes should be more than 49mm by an evaluation of the flow rate and pressure drop in theory. After a pressure drop test, the capsules with a diameter of 54 and 56mm were selected and the flow rates were measured in the HANARO operation conditions. Using the velocity data measured at the single channel test loop of the outer core test facility, the heat transfer coefficient was reviewed and the temperature on the surface of the capsule was evaluated to confirm that it was less than the ONB temperature. Finally, a thermal performance test was performed using a mock-up capsule made with the out-pile test results.

TEST HOLES AND IRRADIATION EXPERIENCE

Characteristics of HANARO test holes were surveyed and analyzed to establish the best fit for the irradiation test of a high temperature and low neutron flux. To obtain a required fast neutron fluence of 1×10^{19} n/cm^2 (E>1.0 MeV), OR 4, 5 and IP 9, 10, 11 test holes are preliminarily selected as candidate test holes. Fig. 1 shows the layout of the HANARO core and the location of the selected test holes, and table 1 shows the nuclear characteristics of the holes.

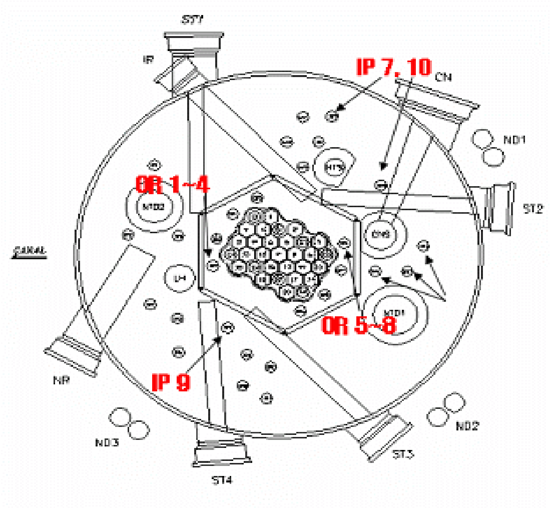

Fig. 1. Layout of the HANARO core and candidate test holes

Table 1 Neutron flux and Fluence (E>1.0 MeV)
after 1 cycle irradiation (23 days)

Test Hole	Neutron flux	Fluence(m/cm^2)
IP 10	3.7E10	7.4E16
IP 11	6.6E10	1.3E17
IP 9	2.5E11	4.9E17
OR(4/5)	1.0E13	2.1E19
CT	1.1~1.7E14	2.2~3.4E20

Although there are eight OR holes in the HANARO, only OR 4 and 5 can be used for a capsule irradiation due to a limitation of the chimney fixing system. There is a forced coolant flow in the OR holes of 60 mm in diameter and 1,200 mm in height. Among the 17 IP holes, IP 9, 10, 11 were selected for the irradiation of high temperature materials in consideration of their interactions with other reactor facilities and the properties of the holes. There is no coolant flow in the IP holes with the same dimensions as the OR hole. The typical gamma heating rates of the OR and IP test holes are known to be 1.0 and 2.5 W/g, respectively [1].

Most irradiation tests of nuclear materials have been performed in the CT and IR holes of HANARO with a relatively higher neutron flux [2, 3]. Therefore, the possibility and safety problems for an irradiation in the OR and IP holes were examined thoroughly based on a standard material irradiation test system. Based on a nuclear fuel capsule system irradiated in the OR holes[4] and a creep capsule system irradiated in an IP hole[5], a basic design for a irradiation capsule of high temperature materials was performed. As a case study on the design parameters, the cross sections of a fuel capsule were investigated. It was found that a material capsule with a diameter more than 50 mm satisfies the coolant flow condition in the OR hole. However, several properties such as the pressure drop, thermal conductivity on a capsule surface and vibration behavior should be examined more carefully to secure an in-reactor safety of a capsule. Because of a possibility for a violation of the ONB condition in the IP holes, a capsule for the OR holes will be developed first and then a capsule for the IP holes will be developed.

CONCEPTUAL DESIGN

The conceptual design was established to design a capsule to irradiate materials using the OR and IP holes. The OR capsule is fixed by using a clamp device pressing against the protection tube, on the other hand the IP capsule is inserted in the hole without a special clamping device. In the basic design of the OR capsule, capsules with outside diameters of 50, 52, 54, 56mm were reviewed theoretically to see if they met the hydraulic requirements in HANARO. An array of specimens was reviewed for the type of dispersion or concentration to high temperatures. The greater the outside diameter of a capsule is, the more advantageous a capsule is for an economical point of view, but an evaluation of the flow rate is necessary as it becomes decreased as the diameter becomes bigger. To evaluate a cooling, the pressure drop was evaluated for various diameters of the capsules. It was estimated that the diameter of the OR capsule should be more than 49mm to satisfy the requirement for a flow and pressure drop in HANARO.

The temperatures for the capsule components were calculated using the GENGTC code using nuclear and thermal characteristics for some representative OR and IP holes. As the GENGTC code is for a one dimensional thermal calculation, modeling work was performed according to the arrangement of the specimens. The temperatures of the specimens were adjusted to 400˚C on a target by controlling the gap between the outer tube and the thermal media of a capsule in consideration of the outside diameter of a capsule and the arrangement of the specimens as in Table 2. But, the surface temperature of the IP capsule becomes more than 100˚C, so it does not satisfy the requirements for the ONB condition in HANARO.

Even though it was assumed that there was no flow of cooling water in the IP hole in the beginning, a buoyant force is generated by a temperature difference between the upper and lower positions in a practical manner, so the cooling water would flow by a pressure difference generated as a result. If the temperature at the upper position is assumed to be 100˚C, the flow rate would be 2.82 kg/s. The generation of a flow by a temperature difference will raise the heat transfer coefficient greatly, the ONB condition at the surface needs to be reviewed again.

Table 2 Temperatures of the OR/IP capsules

Test hole	Specimen Arrangement	Capsule O.D	Specimen Temp.(˚C)	Surface Temp(˚C)	Gap (mm)
OR	1 hole Centered	52	396	41	0.48
		54	396	42	
		56	398	42	
	4 holes Scattered	52	396	41	0.34
		54	406	42	
		56	398	45	
IP	1 hole Centered	52	387	100	0.32
		54	387	100	
		56	386	100	
	4 holes Scattered	52	344	99	0.48
		54	343	100	
		56	343	101	

The capsules in the OR and IP holes will be designed with the same shape. As the OR capsule will be supported by a clamp and a bottom guide tip is inserted into the spider hole, an evaluation for its structural integrity is necessary with a vibration by cooling water. The stability when loading and unloading a capsule in the IP hole was reviewed and so the guide structures used in the capsule irradiated in the CT hole were proved to be unnecessary.

Fig. 2 shows the temperature distribution according to a distance in the radial direction. The calculation was performed using a gamma heating of 2.5 and 1.1 W/g respectively in the OR and IP holes and a degree of vacuum of 0.4 $K_{He,1atm}$.

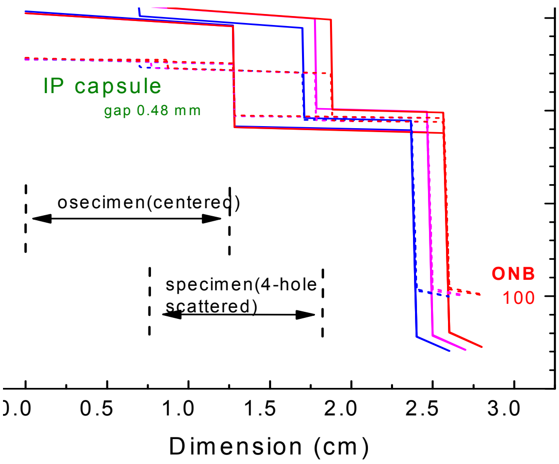

Fig. 2 Temperature distribution in the OP and IP holes

FABRICATION AND OUT-PILE TESTS OF A MOCK-UP CAPSULE

The test for a pressure drop and vibration was performed to determine the diameter of a material capsule for an irradiation in the OR hole. It was estimated that the diameter of the OR capsule should be more than 49mm by an evaluation for the flow rate and pressure drop theoretically. According to this estimation, 3 kinds of mock-up capsules with a diameter of 52, 54, 56mm were made and used for the pressure drop test. As a result of the pressure drop test, the requirement for a pressure and flow rate in HANARO was confirmed to be satisfied for all three diameters. The heat transfer coefficient and temperature on the surface of a capsule was estimated on the basis of the flow rate as a result of the pressure drop test. As the temperature on the surface of the capsule was calculated to be 43.7˚C, the ONB condition in HANARO was satisfied.

Pressure Drop Test

The pressure drop test for a OR capsule was performed in the single channel test loop of the HANARO out-pile test facilities. The capsules used for the pressure drop test were made with a protection tube connected to 3 different sizes of capsules as in Fig. 3. The results for the pressure drop test are shown in Fig. 3. The 3 kinds of mock-up capsules with a diameter of 52, 54 and 56 mm satisfied the HANARO requirement for a flow rate[6].

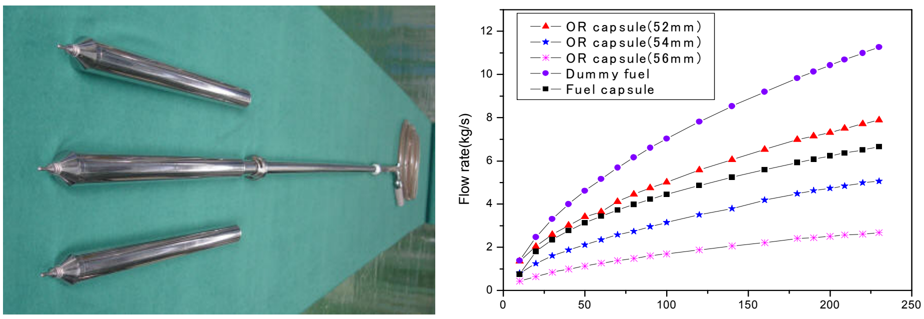

Fig. 3 mock-up capsules and pressure drop test

Vibration Test

The capsules with a diameter of 54 and 56mm were used in the vibration test by taking into consideration a receptive capacity of the specimens. The test was performed in the HANARO out-pile test facilities. The flow rate was 10% more than the normal one, and the pressure drop was 220kPa during the test. 4 underwater and 2 inner accelerometers were installed on the protection and in the capsule.

The analyses for a time-region and a frequency-region were performed for the measured vibration signals. The sampling frequency was 4096Hz, and the time for obtaining the data was more than 120 seconds, and the displacements were calculated by integrating two times the acceleration signals. In the analysis for a time-region, the RMS(root mean square) and the maximum amplitude were obtained. In the analysis for a frequency-region, the frequency components were obtained in detail and the co-relations between the measured signals were analyzed.

The maximum amplitude of an acceleration for the capsule of 54mm diameter was in the range of 40.67~41.93m/s^2. These values are not suitable because the allowable value for an amplitude of an acceleration in HANARO is 18.99m/s^2[7]. The maximum amplitude of an acceleration for the capsule of 56mm diameter was in the range of 10.40~12.14m/s^2, and this capsule is suitable because this is less than the allowable value in HANARO. As the displacement of this capsule was 0.41~0.47mm and less than the gap between the OR flow tube and the capsule, this capsule does not bump into the wall of the OR flow tube.

A capsule of a diameter 56mm satisfies the requirement for an allowable limit of a vibration acceleration applied in HANARO and its maximum amplitude of a displacement is less than the gap (2mm) between a flow tube and a capsule. Therefore, the capsule of a diameter 56mm will maintain a structural integrity by a vibration and not interfere with the adjacent structures.

ANALYSIS OF A HEAT TRANSFER AND SURFACE TEMPERATURE

If the dimensions of a capsule are determined, a heat transfer coefficient should be calculated and then the temperature on the surface of the capsule should be evaluated to confirm it is less than the ONB temperature. When the diameter of a capsule is 56mm, the flow rate was 2.57kg/s at a pressure difference of 209kPa according to the results of the test. The heat transfer coefficient was evaluated as 33,011 W/m^2 ˚C. A boiling should not occur on the surface of a capsule installed in HANARO. The temperature on the surface of a capsule should be less than 124˚C so as not to be boiled. The rate of a heat generation should be found to obtain the surface temperature of a capsule. The heating rate of the OR capsule may be estimated to be 17,418W. The surface temperature of the OR capsule is evaluated as 43.7˚C, and this is less than the ONB temperature.

THERMAL PERFORMANCE TEST

Based on a previous technical examination and on the out-pile test results by using a mock-up capsule, a capsule for a thermal performance test for high temperature materials was finally designed and fabricated. The capsule with a diameter of 56mm is composed of five stages with a separated thermal medium, specimens and an electric heater at each stage. The contained thermal medium has 4-holes to contain the specimens of STS 304 material with a dimension of 10x10x100mmL. Other thermal media like Al, Fe, Zr, Ti and Mo were used. These materials are candidate ones to be used as a substitute of Al thermal media for an irradiation of high temperature materials in the future. The length of the main body is 813mm and the total length including the main body and the protection body is 4934mm.

The thermal performance test was applied at the heater power of 1800 and 2850W in a He environment of 760 and 100 torr. The results are indicated in Fig. 4. The temperature of the specimens in the Fe thermal medium at the 2nd stage was 400°C at a 760 torr, and 527°C at a 100 torr. The temperatures were distributed in the order of a high temperature at Fe, Ti, Mo, Zr and Al. The temperatures are inversely proportional to the values of the thermal conductance.

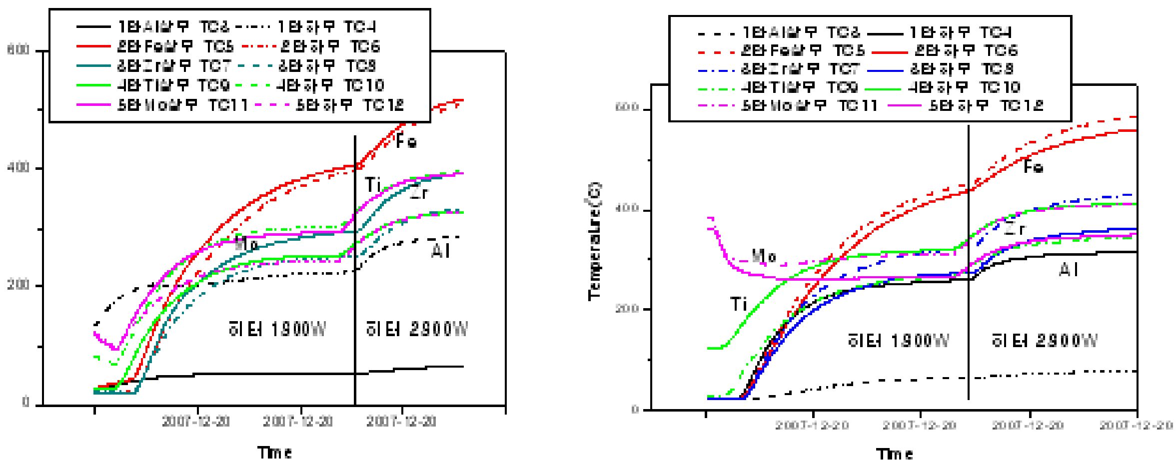

Fig. 4 Temperatures in heater powers

CONCLUSIONS

Out-pile tests were performed to determine the size of a capsule for a material irradiation in the OR holes of HANARO. A capsule with a diameter of 56mm satisfied all the requirements for a pressure drop, a vibration, etc. After a thermal performance test for a mock-up capsule, the first capsule for an irradiation at the OR holes was fabricated and irradiated successfully.

ACKNOWLEDGEMENTS

The authors would like to express their appreciation to Korea Science and Engineering Foundation (KOSEF) and the Ministry of Education, Science and Technology (MEST) of the Republic of Korea for the support of this work through the Nuclear R&D Project.

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4.6 ADVANCES IN MATERIAL CAPSULE TECHNOLOGY IN HANARO

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A material capsule system has been developed for irradiation tests of non-fissile materials in HANARO. This capsule system has been actively utilized for various material irradiation tests requested by users from research institutes, universities, and the industries. Based on the accumulated experience and the user's sophisticated requirements, several advances in material capsule technologies were obtained recently for a more precise control and analysis of the neutron irradiation effect in HANARO. New instrumented capsule technologies for a more precise control of the irradiation temperature and fluence of a specimen, irrespective of the reactor operation, have been developed and out-pile tested. The 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 have also been developed and in-pile tested, successfully. A high temperature irradiation technology up to 1000°C is under development. An evaluation of the DPA (Displacement Per Atom) and activation of irradiated specimens was attempted by using the SPECTOR and ORIGEN2 codes, respectively. A new fluence monitor with a decreased activity was designed to measure the thermal and fast neutron fluences of the irradiated specimens. A friction welded tube using STS304 and Al1050 alloys was introduced to prevent a coolant leakage into a capsule during a capsule cutting process after an irradiation.

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2008 KAERI-JAEA Joint Seminar

Advances in Material Capsule Technology in HANARO

Kee-Nam CHOO, Young-Hwan KANG, Man-Soon CHO, Yoon-Taek SHIN, Seng-Jae Park, Bong-Goo KIM, Young-Ki Kim

Research Reactor Engineering Division

Korea Atomic Energy Research Institute

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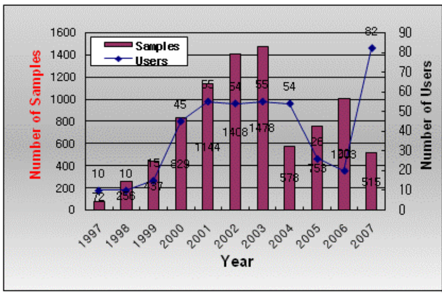
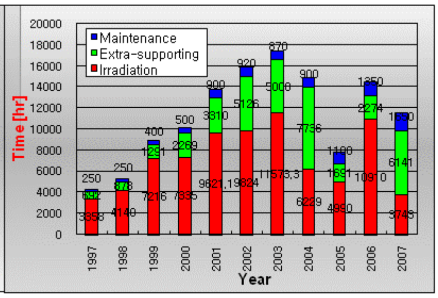
HANARO Irradiation Capsule

- R & D of Instrumented Capsule (1995~)
- Irradiation Tests for Users
  - Research Institute, Industry, University, International Projects
  - 7,000 specimens : Nuclear, Advanced/Basic Researches
- Advances in Irradiation Capsule Technology





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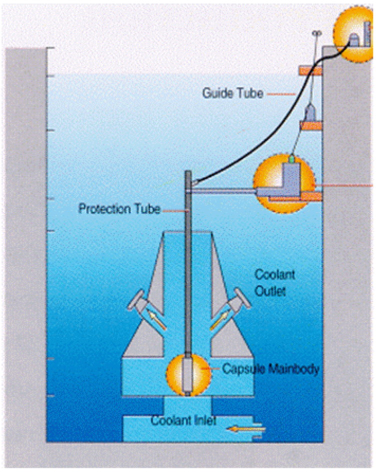
Schematic View of HANARO

HANARO	Open-tank-in-pool Type
Max. thermal power	30 MW
Coolant	Light water
Reflector	Heavy water
Fuel material	U_3Si in Al matrix, 19.75w/o enriched
Absorber material	Hafnium
Secondary cooling	Cooling tower

- **Capsule / Rabbit System**
- **Capsule Related Facilities**
  - Temp. Control System
  - Supporting System (Robot Arm)
  - In-Chimney Bracket
  - Capsule Cutting System
  - Cask for Transfer to IMEF



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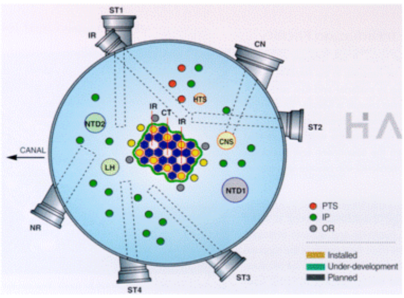
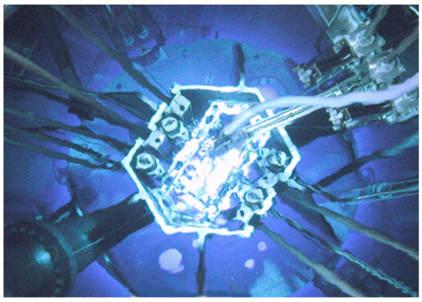
Test Holes in HANARO





Location	Hole Name	Hole No	Inside Dia. (cm)	Neutron Flux (n/cm² . sec) Fast Neutron (E>1.0 Mev)	Neutron Flux (n/cm² . sec) Thermal Neutron (<0.625 ev)	Remarks
Core	CT	1	7.44	1.54×10^{14}	4.39×10^{14}	Fuel/material isotope production
	IR	2	7.44	1.50×10^{14}	3.93×10^{14}	
	OR	4	6.00	2.07×10^{13}	3.36×10^{14}	
Reflector	LH	1	15.0	6.62×10^{11}	9.77×10^{13}	Fuel/material Isotope, semi-conductors
	HTS	1	10.0	9.44×10^{10}	47.97×10^{13}	
	IP	17	6.0	1.45×10^{9} – 2.20×10^{12}	2.40×10^{13} – 1.95×10^{14}	

(E>0.82 MeV for Fast Neutrons of the Holes in Reflector)

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Irradiation Facilities in HANARO

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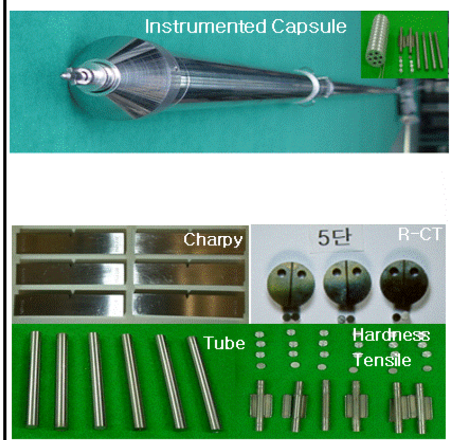
Instrumented Capsule

- **Test Hole :** CT, IR2, OR, IP
- **Dimension :** 5900mm
  - 5 Stages with each Heater
  - Main body : 56/60mm D x 870mm H)
- **Materials**
  - External/Protection Tube : STS 316L/304
  - Guide Tube : PVC Spring Hose
- **Instrumentations**
  - Micro-heater(263W/cm) & He Control
  - T/C : K-type, 14 sets
  - Fluence Monitor : Fe-Ni-Ti-Ag-Nb
- **Specimens**
  - RPV, Fuel Assembly, Basic Materials
  - Charpy, Tensile, SP, TEM, PA, CT etc.



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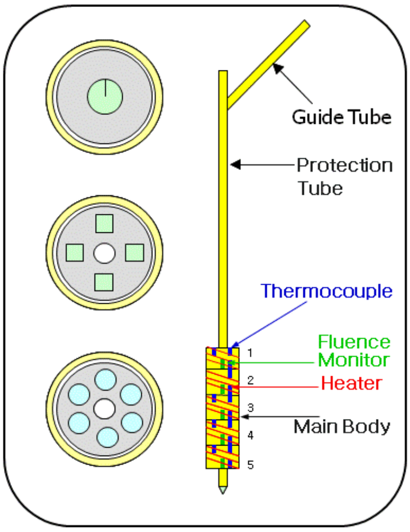
Capsule Design and Analysis

- **Basic Design of Capsule**
  - Specimen & Instrumentation Design
- **Nuclear Property** (MCNP,VENTURE)
  - Reactor Reactivity Analysis
  - Gamma Heatings & Neutron Flux
- **Thermal / Structural Analysis**
  - GENGTC / ANSYS
- **Post Irradiation Analysis**
  - Fluence Monitor Analysis (SAND II)
  - DPA Analysis (SPECTOR)



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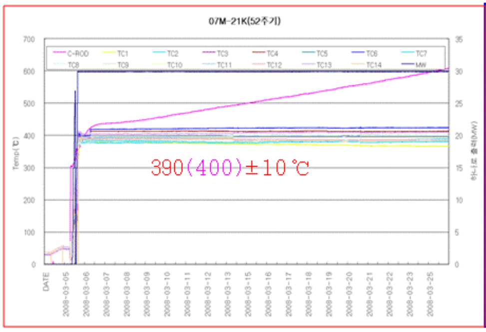
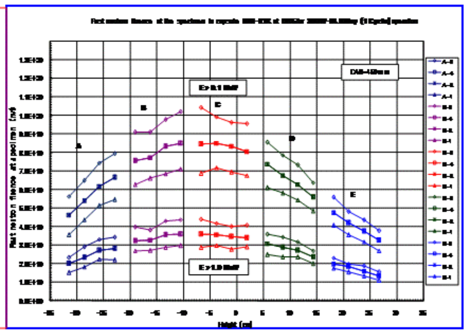
07M-21K Capsule Irradiation





- **HANARO** (OR5 hole (30W), 24.6 days (731 MWD))
- **Irradiation Temperature Control** (Gamma heating > He gas > Heater Control)
- **Temperature (CTCS) :** 14 K-types T/C, Controlling & Monitoring
- **Fast Neutron Fluence** (E>1.0MeV : 1.1 ~ 4.4x10^{19} n/cm^2)
  (E>0.1MeV : 0.27 ~ 1.04x10^{20} n/cm^2)
- **DPA (Displacement Per Atom) :** 0.034 ~ 0.07

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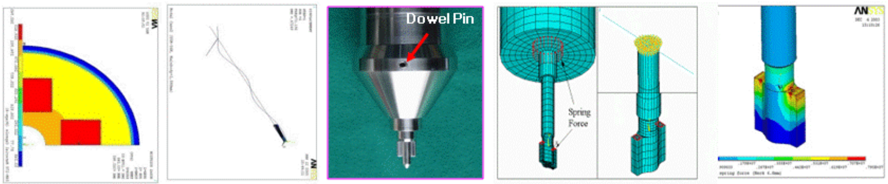
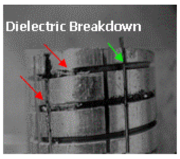
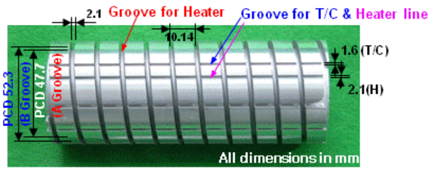
Design Improvement of Capsule

- **Structural Design Improvements**
  - **Up-Flow Coolant ⇒ Vibration Induced Seismic Safety & Fatigue Failure**
  - **Design Improvement : Structural Design Change in Bottom Parts**



- **Procedural Improvements**
  - **Heat Failure during Irradiation**
  - **Limited Operating Condition (He, Heater Control)**





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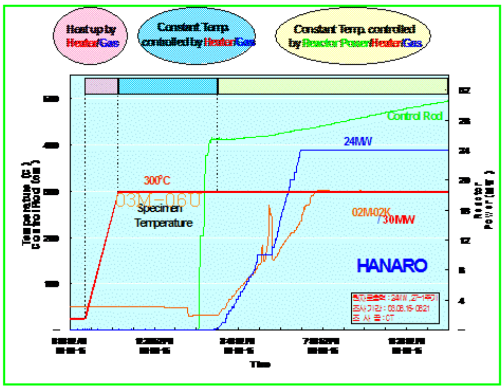
Precise Temp. Control Capsule



- **Specimen Temp. Control**
  - Temp. Control irresp. of Reactor
  - Valuable for Researches on Mechanism of Irradiation Damage
- **Out-Pile Test**
  - 03M-06U Capsule
  - 290°C before Reactor Operation (by Vacuum/Heater Control)
- **In-Pile Test** (03M-06U)
  - CT in HANARO at 30MW
  - 42days irradiation

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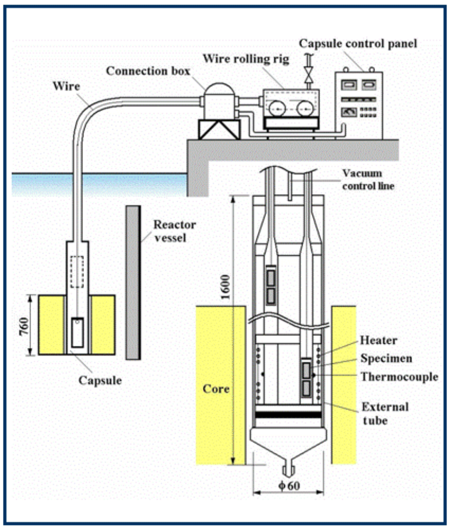
Neutron Fluence Control Capsule



- **Needs for Capsule**
  - 24days' HANARO Operation
  - Needs for Short Time Irradiation
  - Needs for Neutron Fluence Control during Rx. Operation
- **Out-pile Test**
  - 05M-06K Capsule
  - Design & Fabrication Technologies
  - Out-pile Performance Tests
  - Integrity Analysis in HANARO

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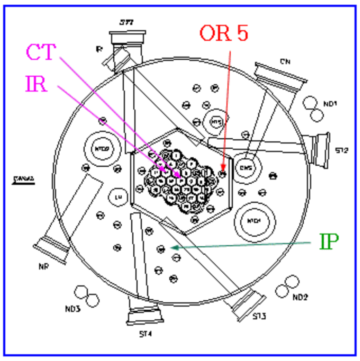
OR / IP Irradiation Capsule



- **Needs for Capsule**
  - Need for Different Neutron Flux
  - Different Flux in CT>IR>OR>IP
- **Out-Pile Test**
  - 07M-09K Capsule
  - Nuclear & Thermal Designs
  - Reactor Safety Analysis (Seismic Safety, Pressure Drop)
- **In-Pile Test**
  - 07M-21K in OR5 for 24.6 days (I-NERI Projects (VHTR))

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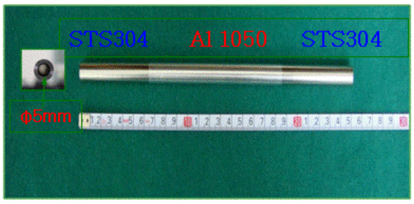
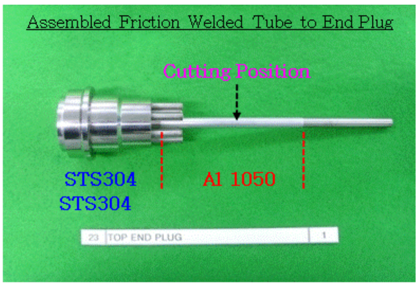
Friction Welded Tube





- **Coolant Leakage into Capsule**
  - During Capsule Cutting after Test
  - Corrosion of Al Parts & Specimens
  - Error Sources on F/M Analysis
- **Development of FWT**
  - STS-Al-STS Friction Welded Tube
  - Safety Analysis : Tensile/Sealing Tests
- **Applications**
  - 07M-13N/07M-21K Capsules
  - Protection of Specimens & F/M

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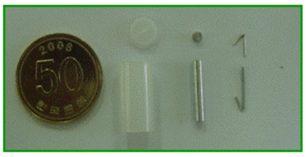
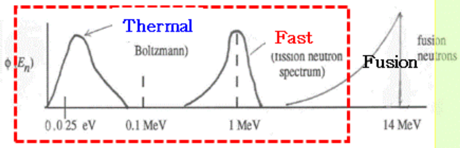
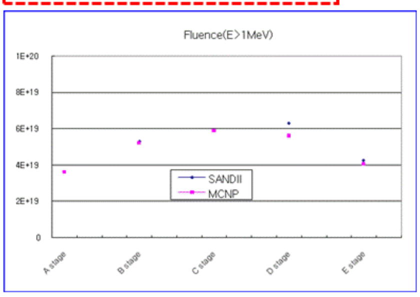
Neutron Fluence Analysis







- **Fast/Thermal Neutron Fluence**
  - MCNP & SANDII Comparison
  - ±15% Error Range
  - FWT Tube against Water Corrosion
  - Detailed Spectrum & Reaction
  - High Purity Wires (Fe,Ni,Ti) and Al
  - Thermal F/M Wires (Ag, Nb, Co)
  - High Activity Problems
- **DPA by SPECTOR**
  - Irradiation Damage Study
- **Material Activation by ORIGEN2**

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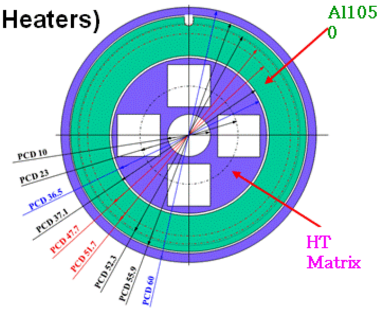
High Temp. Irradiation

- Irradiation up to 500℃
  - Standard Capsule Design (Al Thermal Media)
    - 522℃ for 4h Out-pile Test with 07M-09K Capsule
  - Application : VHTR Materials (Cr-Mo-W Alloys)

- Irradiation up to 1000 ℃
  - Instrumentations (Inconel Sheathed T/C & Heaters)
  - Thermal Media for High Temperature
    - Candidates : Mo, Nb, Zr, Fe, Ti
  - Double Layered Structure
  - Application : VHTR, FBR, Fusion Materials



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Summary

- *A material capsule system has been developed & utilized for irradiation tests of non-fissile materials in HANARO.*
- *Based on experience and user's needs, advances in capsule technology were progressed for Precise control of temp. & fluence, Wide range flux test, and High temperature irradiation.*
- *Technologies for DPA, Activation, Precise thermal/ fast neutron fluence were also progressed recently in HANARO.*
- *HANARO irradiation facilities are freely available for international research programs.*

4.7 DEVELOPMENT OF A CAPSULE ASSEMBLY MACHINE FOR THE RE-IRRADIATION TESTS IN HANARO

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ABSTRACT

A capsule assembly machine (CAM) for the long-term irradiation tests in the HANARO reactor has been designed, developed and demonstrated at the Korea Atomic Energy Research Institute (KAERI). The CAM will provide a technical base for viable re-irradiation services. This machine will be installed in the reactor service pool of the HANARO reactor. The new assembly technique by using a mock up of the CAM in air demonstrated its suitability for an assembly operation, and for an application of this technique to a reactor. The technique will be upgraded after a commissioning test under water environments. This technique would be expected to be recommended for a country where an underwater canal for transporting irradiated devices and enough space of a hot cell for assembling capsule components are not available.

INTRODUCTION

New capsule assembly technology has been developed to meet the demands for a high burnup test at HANARO since 2003. The main problems in the development of the required techniques in HANARO are a space limitation of the hot cell to manufacture a full size (~6m) capsule and no canal between the reactor and the post irradiation examination facility. Furthermore, for a long term irradiation, the lifetime or total fluence of the capsule structural materials would be limited (for example, ~10^{21} n/cm^2). The re-assembly method reported in the literature for a capsule manufacturing where most of the major steps are progressed in a hot cell is as follows: The first step is transferring an irradiated capsule to the hot cell to dismantle and separate the specimens. The second step is loading the irradiated specimens into a fresh new capsule and attaching various types of sensors to the capsule for a main body preparation. The third step is welding the capsule's main body and the protection tube to obtain a full size of the capsule for a re-irradiation test.

Therefore, the need to develop new techniques that can assemble a capsule's components such as a capsule's main body and a protection tube was recognized as one of the key challenges remaining for the re-irradiation test techniques in the HANARO reactor. In 2003, a mockup of the CAM was designed and fabricated [1, 2]. The performance test which started in 2004 was undertaken to determine and present the main performance characteristics of the CAM including the special tools. With these improvements, the tools were fully qualified by February 2006. In addition, these techniques will be applied to fabricate a fuel capsule for a re-irradiation test.

This paper presents a summary of the latest results from the development, the design improvements and the performance tests. The details are presented in several other papers [1], [2], and [4].

DESIGN FEATURES OF THE CAPSULE ASSEMBLY TECHNIQUE AT HANARO

The basic idea was based on the principal that most of the assembly processes could be carried out in a hot cell except for the last connecting process between a capsule's main body and a protection tube. As the protection tube has a role to protect the instrumented cables and gas tubes from the reactor coolant and to link a capsule main body to an instrumentation and control system, the present idea makes it possible to solve the aforementioned problems. Two options for the connecting process were considered; one is a bolting with four bolts and the other is a joining with a specially designed joint.

In the case of an assembly process by a bolting, each of the flanges is welded to both sides of the bottom of a protection tube and the top of a capsule's main body, and the two flanges are connected with four stainless steel bolts by using a remote working tool. Sufficient access to the top of a capsule's main body and space for an in-situ bolting by a remote tool must be considered in the assembly process and a structural integrity of the components in service is required. As described in more detail in reference [2], for operation efficiency, two guide pins at the flange of a capsule's main body were introduced in the process, while the feasibility of the bolting process, using a remote tool, has been demonstrated through a series of mock up fabrications [2] and performance tests [4].

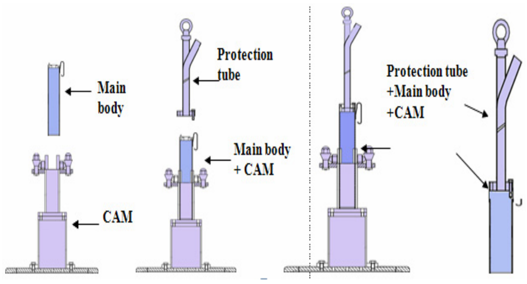

Fig. 1 Conceptual figure of the connecting work

In the case of an assembly process by a joining, a specially designed joint was selected from the market to meet the structural integrity of a capsule during an irradiation. However, this concept was tentatively abandoned because the position of the capsule stopper could be changed due to the flexible feature of the joint.

The conceptual figuration of the connecting work is shown in Fig. 1. The reactor service pool, 3 m wide and 6 m deep, has sufficient a water depth for a gamma shielding and enough space for the connecting tasks.

FABRICATION OF THE CAPSULE ASSEMBLY MACHINE MOCK-UP

For the design of the CAM to meet the proposed assembly processes, two main criteria were set up: the machine must be as simple as possible, and it must be designed and manufactured to provide a safe handling and assembly of a capsule's components within a reasonable time. To demonstrate the performance of the design concept of the CAM, KAERI has fabricated a mock-up to carry out an integral test in air and under water.

The CAM mock-up has been developed at KAERI. In the present design of the CAM, a number of interface structures, such as a chuck and housings to support a capsule's main body, have been introduced as shown in Fig. 2, in order to simplify the operation procedures [2]. As a stopper of the HANARO fuel capsule, which is located at a position of 300 mm from the bottom of a protection tube, the accessibility of a remote tool for a bolting from the main bridge of the HANARO reactor is heavily restricted due to the highly curved outer walls of the stopper. The critical issue in the present design is therefore to confirm the feasibilities of a tool fabrication for a bolting.

There are limits on the diameter and the length of the CAM. The diameter of the clamping chuck of the CAM must be less than 73 mm according to the size limitation of a capsule. The length must be less than 1800 mm for adopting a working space in the reactor service pool. After the assembly work, instrument lines must be connected for an irradiation condition control and measurement.

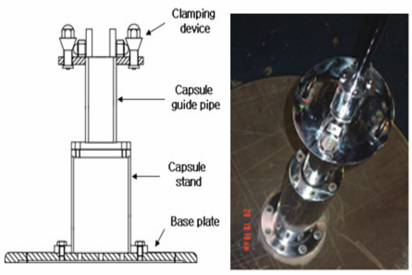

Fig. 2 Schematic and real view of the new capsule assembly machine

To accommodate a reduction of the space assigned to the reactor service pool of HANARO, the dimensions of the designed mockup are 1m in outer diameter, 1.8m in height and 136kg in weight. Key components were all made of stainless steel. Especially, the clamping device to grasp a capsule's main body for an assembling step, has a manually operated two-jaw chuck of the guide pipe.

The assembly procedure by using the CAM consists of three steps. The first step is to orient a capsule's main body vertically inside the guide pipe. The second step is to turn the clamping screw with the special tool until the jaw meets a capsule's main body to be properly clamped. The third step is to insert the protection tube with holes into the guide pin, which is located at the top end of a capsule's main body, and fasten them together with bolts.

OUT-PILE TESTS FOR DESIGN IMPROVEMENTS AND AN OPTIMIZATION

In this assembly procedure, the potential parameters that affect the safety or environmental effects associated with a capsule's use can be addressed as a source of concern. This situation requires additional care in the design and operation of the machine. Thus, a series of pre-operation tests is required for a better understanding and design optimization of the machine. In addition, the basic data necessary for preparing the technical specifications of the CAM is also required through fabrication and performance tests of a mock-up. A prerequisite for an increase of the flexibility and reliability of the clamping device is a monitoring of the clamping force during an assembly operation. As the calibrated torque wrench(model: Kanon-230QLK) was attached at the top of the tool handle for a measurement of the clamping load, the clamping forces vary with the number of turns of the locking bolt head clockwise. The method for determining the desirable clamping force, that is acceptable for assembling a capsule's components, is based on the fact that although the accuracy and consistency of the clamping loads are not reliable, the achieved clamping force depends on the tightening torque selected with the torque wrench. The test data shows that the optimum clamping force is 350 kgf·cm for preventing a rotation or shaking of a capsule's main body during an assembly of a capsule's main body and a protection tube [4].

In addition, a clamping torque test for preventing a deformation of a capsule tube was also carried out to ensure a long operational life and an optimal performance. The measurement of the diameter change of a capsule's main body was done by increasing the applied load to the clamping bolt screw. This test data [4] shows that the applicable load due to the total mass of the tool and an operator during an assembly of the parts should be smaller than 700 kgf/cm as shown in Fig. 3.

The performance test has been successfully performed under the simulated conditions, which correspond to the working condition of the reactor service pool. No degradation of the performance of the mock-up has been observed in these tests. These results show that the CAM with a bolting concept can be one of the promising options for an assembly of a capsule's components.

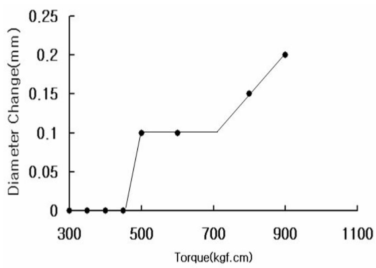

Fig. 3 Diameter change of the capsule tube with an applied load

Initially, assembly techniques in air were demonstrated by using a mock up of the CAM. Although in our previous study, we confirmed that the present concept is suitable for an assembly operation, it will be modified for an application of this technique to HANARO.

CONCLUSION

A new capsule assembly technique was proposed in which specially fabricated bolts are used to assemble a capsule's components. A mock up of the CAM was developed through a series of performance tests to meet the proposed assembly process. The assembly technique by using a mock up of the CAM in air demonstrated its suitability for an assembly operation and for an application of this technique to a reactor. The technique may be upgraded after a commissioning test under water environments. Parallel to the development of the re-irradiation techniques including the instrumentation, the design and manufacturing of an assembly machine for a fuel capsule manufacturing will be continued for a commissioning in the reactor service pool.

Furthermore, this developed technique that could be most easily applied for manufacturing a capsule for re-irradiation tests would be expected to be suitable for a country where an underwater canal for transporting irradiated devices and enough space of a hot cell for assembling capsule components are not available.

ACKNOWLEDGEMENTS

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4.8 DEVELOPMENT AND DESIGN FOR Mo-PRODUCTION FACILITY IN JMTR

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ABSTRACT

At Oarai Research and Development Center, Japan Atomic Energy Agency (JAEA) being advanced is the plan of refurbishing Japan Materials Testing Reactor (JMTR) to start the operation in FY 2011. As one of effective use of the JMTR, JAEA has a plan to product ^{99}Mo, a parent nuclide of ^{99m}Tc. ^{99m}Tc is most commonly used as a radiopharmaceutical in the field of nuclear medicine. Currently the supplying of ^{99}Mo depends only on imports from foreign countries, therefore JAEA is aiming at domestic production of a part of ^{99}Mo in cooperation with industrial users. As JAEA's activities, mentioned are the process, the selection and fabric of the irradiation facilities for ^{99}Mo production, the technical study of commercializing equipment after irradiation, and the cost evaluation for ^{99}Mo production.

INTRODUCTION

In the refurbishing of JMTR that has been performed since 2007, new irradiation utilization facilities will be installed in order to correspond to user's requirement. After the refurbishment, JMTR operation will be started from FY 2011. As one of the new irradiation utilization facilities, the production of ^{99m}Tc used as radiopharmaceutical was studied.

At present, radiation and radioisotope (RI) are indispensable for a sick diagnosis and treatment in the field of medical treatment. Demand of Technetium-99m(half life time 6h) used as a radiopharmaceutical increases up year by year. Moreover, the expansion of demand will be expected in the future.

However, the supply of ^{99}Mo of Japan depends fully on the import from foreign countries. Therefore, it is needed to supply ^{99}Mo stably by the domestic production. ^{99}Mo (half life time 66.7h) production has two methods; the one is the nuclear fission (*n,fiss*) method and the other is ^{98}Mo target (*n,γ*) method [$^{98}Mo(n,\gamma)^{99}Mo \xrightarrow{\beta^-} {}^{99m}Tc$].

^{99}Mo production adopted a simple (*n,γ*) method in JMTR was studied, and evaluated. As a result, in case ^{99}Mo is produced with the JMTR hydraulic rabbit irradiation facility, it has been understood to be able to stably supply ^{99}Mo of the partially amount of demand.

OUTLINE OF IRRADIATION METHOD

Hydraulic rabbit irradiation facility is a water loop system to transfer the small sized (150mm length) capsule, so called rabbit, into and take out from core by water flow.

Molybdenum trioxide (MoO_3) built as a pellet is enclosed among the aluminum rabbit. After irradiation for a regular term, the rabbit take out from core. Irradiation capsules or specimens are transferred to the hot laboratory, which is connected to the reactor building through the water canal, for post irradiation examinations. Owing to the shielding capability of the water, irradiated radioactive capsules or specimens are safely transferred underwater through the canal.

The irradiation rabbit enclosed the Molybdenum-pellet can also promptly transfer to hot laboratory cell through the canal. Moreover, ^{99}Mo is shipped from the hot laboratory cell through activities as the rabbit dismantlement, dissolution, and ^{99}Mo extraction. These processes can be efficiently carried out by using the JMTR hot laboratory.

The hydraulic rabbit irradiation facility and the properties of Molybdenum trioxide (MoO_3) are shown in Fig.1. Flow chart of ^{99}Mo production by the rabbit on JMTR is shown in Fig.2.

FLOW OF PRE- AND POST-IRRADIATION PROCESS

(1) Pre-irradiation and irradiation

The high purity Molybdenum trioxide (MoO_3)powder as the raw material is sintered, and the irradiation target pellet is prepared.

After preparation of the pellet, the dimension inspection, the weight measurement, the visual inspection, and the impurity analysis, etc. are performed. The pellet is enclosed with the inner tube and the rabbit holder, and the irradiation rabbit is prepared. The irradiation rabbit is set in core and take out from core by water flow.

(2) Post-irradiation process

The irradiation rabbit is handed over with underwater basket to the JMTR hot laboratory through the Canal.

The rabbit outer tube is opened, after that, the inner tube is taken out. The inner tube is opened, the MoO_3 pellet to which the inside is sealed is taken out, and it is commercialized with the chemical treatment process in the hot lab.

The MoO_3 pellet taken out is dissolved with the sodium hydroxide (NaOH) , and ^{99}Mo is extracted. Impurities are analyzed with the product inspection (pH measurement, γ ray measurement, capacity, capacity measurement, and weight measurement) and a small amount of sampling in the state of the PZC (Poly Zirconium Compound)) absorbent -^{99}Mo or ^{99}Mo-solution.

After inspection, Mo is distributed to the product container according to the amount of the radioactivity, and the container is loaded to transfer cask. The surface contamination test and 1m dose rate measurement, etc. for the transfer cask are carried out.

Because Half life time of ^{99}Mo is 66.7hour and ^{99m}Tc is 6h, the working hours to treat ^{99}Mo from the irradiation to the shipment should be short.

STUDY FOR AMOUNT OF ^{99}Mo PRODUCTION

(1) Study of irradiation position

There are two main factors to decide the amount of ^{99}Mo production. The one is a thermal neutron flux at the irradiation position in the reactor core, and the other is a number of rabbits that can irradiate it at the same time.

Irradiation holes D-5 and M-9 were studied as a reactor core position in which the rabbit is able to be irradiated. Because the hole size of the reactor core grid plate is different according to the irradiation hole, the structure of the hydraulic rabbit tube is different. Therefore, the irradiation hole D-5 enables 3 rabbits and the irradiation hole M-9 enable 5 rabbits irradiations in maximums.

The structure of the hydraulic rabbit tube applied to the reactor core irradiation position and each irradiation hole is shown in Fig.3.

(2) Evaluation of ^{99}Mo production

When one irradiation cycle assumed to be 30 days, and also one rabbit irradiation needs 6 days, it is possible to irradiate 4 batches by one irradiation hole for one irradiation cycle.

Moreover, necessary time from the irradiation to the shipment was evaluated as 2.5 days. In this case, ^{99}Mo production with two (D-9 and M-9) tubes is effective. Therefore, ^{99}Mo of 18.5TBq or less/week (500Ci/week) can be shipped.

The amount of ^{99}Mo for each case and total estimation cost ratio are shown in Table 1.

FUTURE PLAN FOR IRRADIATION FACILITY AND HOT LABORATORY EQUIPMENTS

(1) Irradiation facility

Existing hydraulic rabbit irradiation facility (D-5 irradiation hole) will be used after a little maintenance. As for new hydraulic rabbit irradiation facility, a concrete adjustment is scheduled with the maker of medicine manufacture.

Moreover, the approval for its design of Japanese Government for the maintenance and the new establishment of hydraulic rabbit irradiation facilities will be prepared.

(2) Hot laboratory Equipments

^{99}Mo extraction equipment and the dismantlement equipment of the rabbit outer tube will be installed to the hot laboratory cell. ^{99}Mo production flow in the hot laboratory cell is shown in Fig.4.

After shipping ^{99}Mo to the medical institution with a top-open type transport cask, the medical institution extracts ^{99m}Tc from ^{99}Mo (^{99m}Tc generator).

CONCLUSION

Molybdenum trioxide (MoO_3)was selected to the raw material as ^{99}Mo production process in JMTR. Moreover, the ^{98}Mo(n ,γ) reaction is selected, and the sintering pellet of Mo will be used as target material.

Hydraulic rabbit irradiation facility was selected as irradiation facility of ^{99}Mo production, because the facility is possible to irradiate the Mo target from short time to long time without reactor shutdown. After irradiation, ^{99}Mo is dissolved from the MoO_3 pellet by an easy chemical treatment. ^{99}Mo absorbent or the solution will be commercialized, and shipped promptly from JMTR hot laboratory.

As the result, JMTR will be able to provide about 20% for amount (88.8TBq:2400Ci) of

^{99}Mo imported to Japan.

Moreover, it is studying that amount of ^{99}Mo production is raised up due to increasing the MoO_3 density, and the number of rabbits irradiated by the M-9 irradiation hole will be increased from 3 to 5 rabbits. In this case, the amount of ^{99}Mo production will be reached about 37TBq/week (1,000ci/week).

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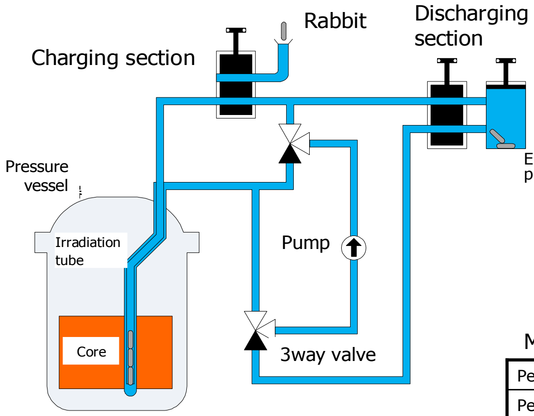
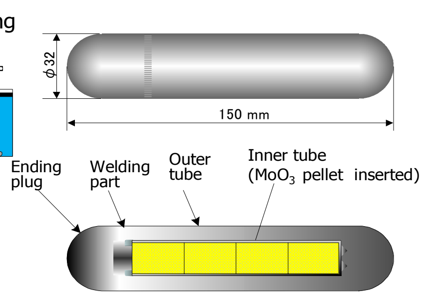
Table 1:
Estimation of the amount of ^{99}Mo production using hydraulic irradiation facility

Items		CASE 1	CASE 2	
Hydraulic irradiation facility		Existing hydraulic irradiation facility	Existing hydraulic irradiation facility and added a hydraulic irradiation facility	
Rabbit Specification	**Irradiation hole**	D-5	D-5	M-9
	Thermal Neutron Flux (max.)	1.1×10^{18} ($m^{-2}\cdot s^{-1}$)	1.1×10^{18} ($m^{-2}\cdot s^{-1}$)	3.5×10^{18} ($m^{-2}\cdot s^{-1}$)
	Number of irradiation rabbit (max.)	3 rabbits	3 rabbits	5 rabbits
Ratio of estimated cost		1	6	
Amount of ^{99}Mo production		3.7 TBq/week (100 Ci/week)	18.5 TBq/week (500Ci/week)	
Percentage for the amount of import*		4%	20%	

* estimation of the amount of import: 88.8 TBq/week (2,400 Ci/week)



Hydraulic rabbit irradiation facility



A typical rabbit and its cross section

Molybdenum trioxide (MoO_3) properties

Pellet type	Sintered type
Pellet density	3.25-3.4g/cm³
Sublimation temp.	~750℃
MoO_3 Molecular weight	143.95 (Mo:66.6% O:33.3%)
MoO_3 Density	4.692g/cm³
MoO_3 Melting point	780℃

Fig.1 Hydraulic rabbit irradiation facility and Molybdenum trioxide(MoO_3) properties

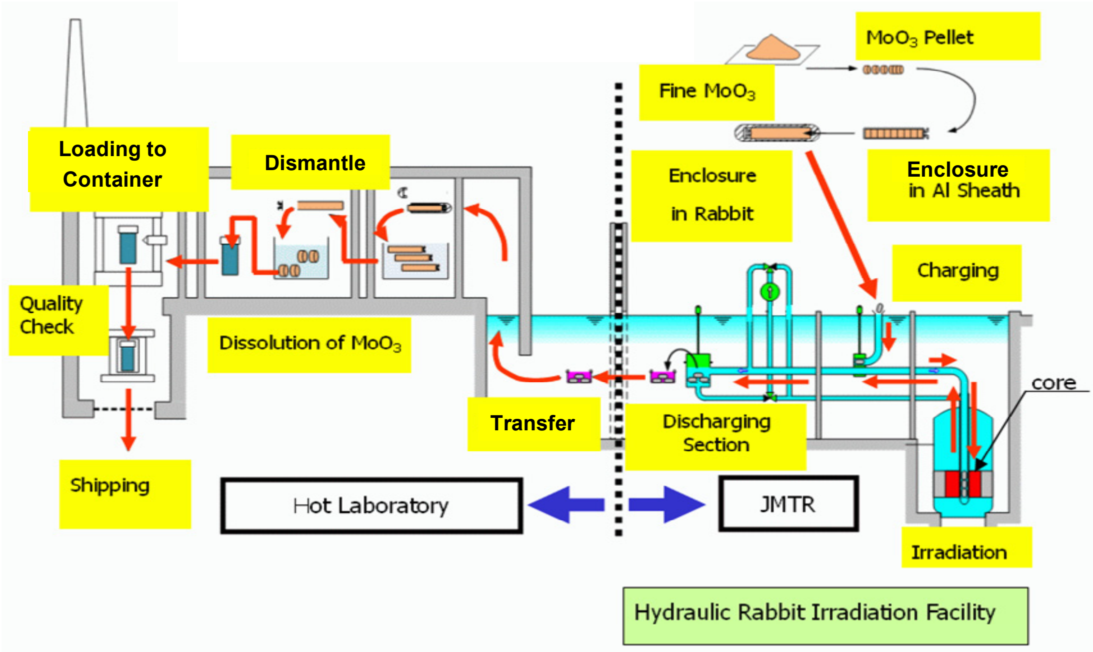


Fig.2 ^{99}Mo Production by Rabbit on JMTR

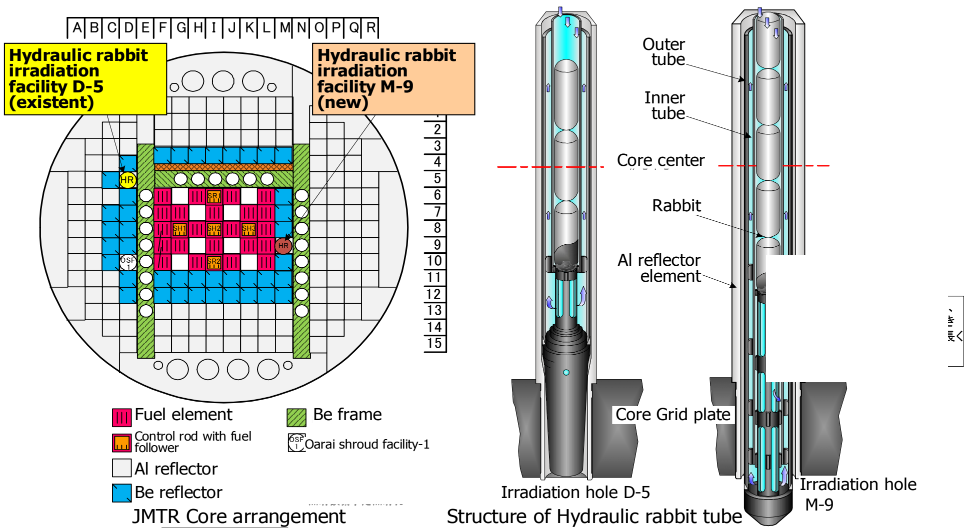


Fig.3 Irradiation position and the structure of the hydraulic rabbit tube in core

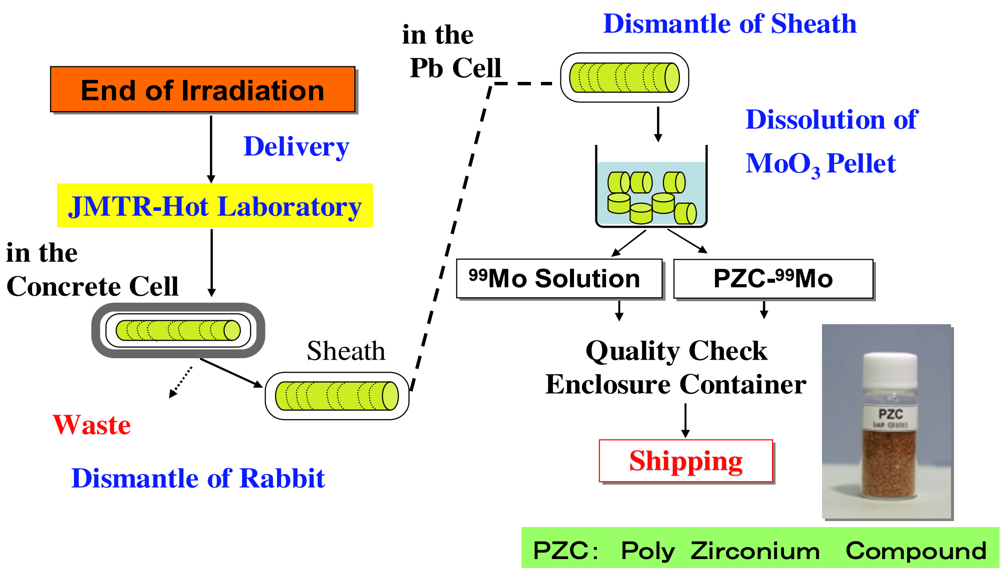


Fig.4 Flow of ^{99}Mo Production in JMTR Hot Laboratory

4.9 DEVELOPMENT ON ^{99}Mo PRODUCTION TECHNOLOGY BY MOLYBDENUM SOLUTION IRRADIATION METHOD

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ABSTRACT

^{99m}Tc for the medical diagnosis is the most widely used radioisotope in the world and its demand is growing up year by year. Importance of the domestic production in Japan is pointed out from a viewpoint of the stable supply of ^{99}Mo, parent nuclide of ^{99m}Tc, because of some troubles on transportation or the research reactor, and so on. Therefore, the (n, γ) method using solid molybdenum target by the hydraulic rabbit irradiation facility has been planned in JMTR.

In order to increase the amount of ^{99}Mo production, the molybdenum solution irradiation method was proposed as a new production method having several advantages such as large irradiation volume, high efficient and low cost comparing with conventional enriched ^{235}U(n, f)^{99}Mo method or solid molybdenum target (n, γ) method.

Aiming at the realization of the molybdenum solution irradiation method, un-irradiated and γ ray irradiated tests of the molybdenum solution were carried out, and the combination of potassium molybdate solution and stainless steel was selected as the promising materials from the result of the compatibility and the chemical stability.

INTRODUCTION

Status of ^{99}Mo use in Japan

^{99m}Tc is the most widely used radioisotope in the world for the medical diagnosis such as the checks of cancer, bowels disease, and brain facult, and the demand is continuously growing up year by year. In case of Japan, all ^{99}Mo, parent nuclide of ^{99m}Tc, has been imported. However, the importance of the domestic production is pointed out from a viewpoint of the stable supply of ^{99}Mo because of some troubles of the research reactor, and so on [1,2]. Therefore, the (n, γ) method using solid molybdenum target by the hydraulic rabbit irradiation facility has been planned in JMTR. However, it will be difficult to satisfy domestic demand because of the limited irradiation volumes of rabbit. Status of ^{99}Mo supply in Japan is shown in Fig. 1. To increase the amount of ^{99}Mo production, the molybdenum solution irradiation method (M-SUIMIT) was proposed as a new production method [3-5]. This method is based on the neutron irradiation of circulating molybdenum solution.

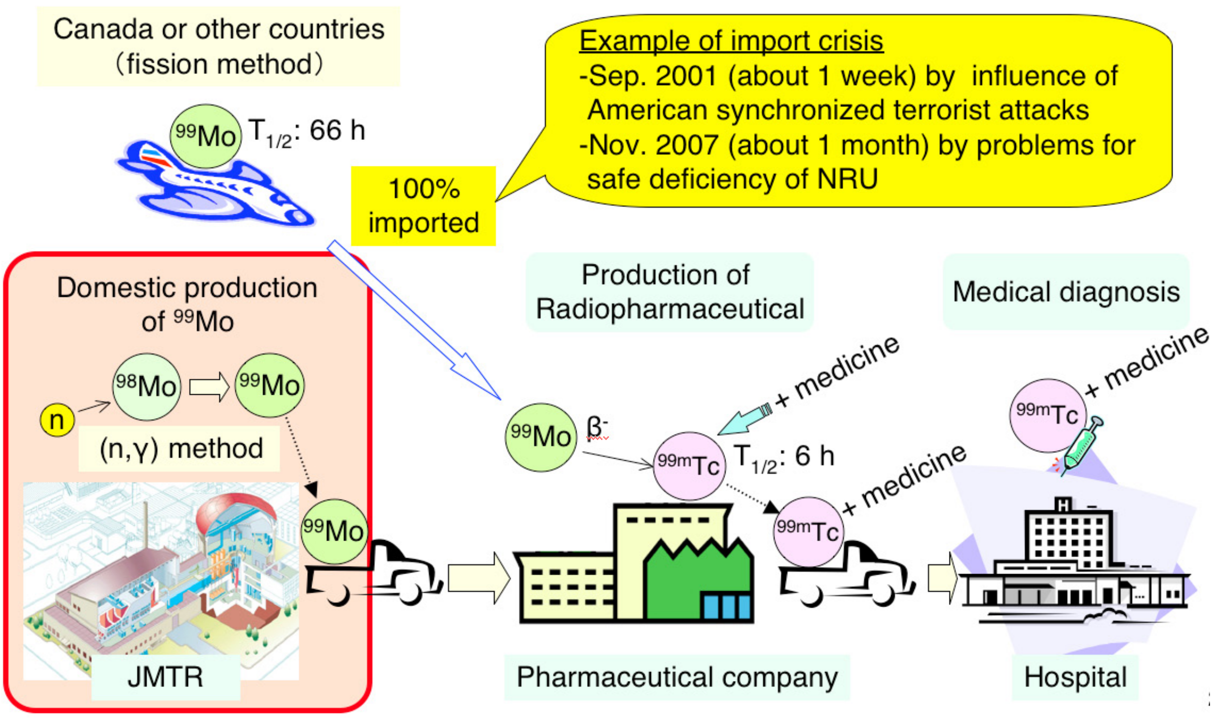

Fig. 1 Status of ^{99}Mo supply in Japan.

M-SUIMIT

Outline of the M-SUIMIT is shown in Fig. 2. In the M-SUIMIT, the collection of ^{99}Mo under the reactor operation is simple and increasing the irradiation volume is easy. The separation and dissolution processes after the irradiation are not necessary, and the amount of generated radioactive waste is smaller than that by the conventional methods. Comparison between ^{99}Mo production methods in JMTR is shown in Table 1.

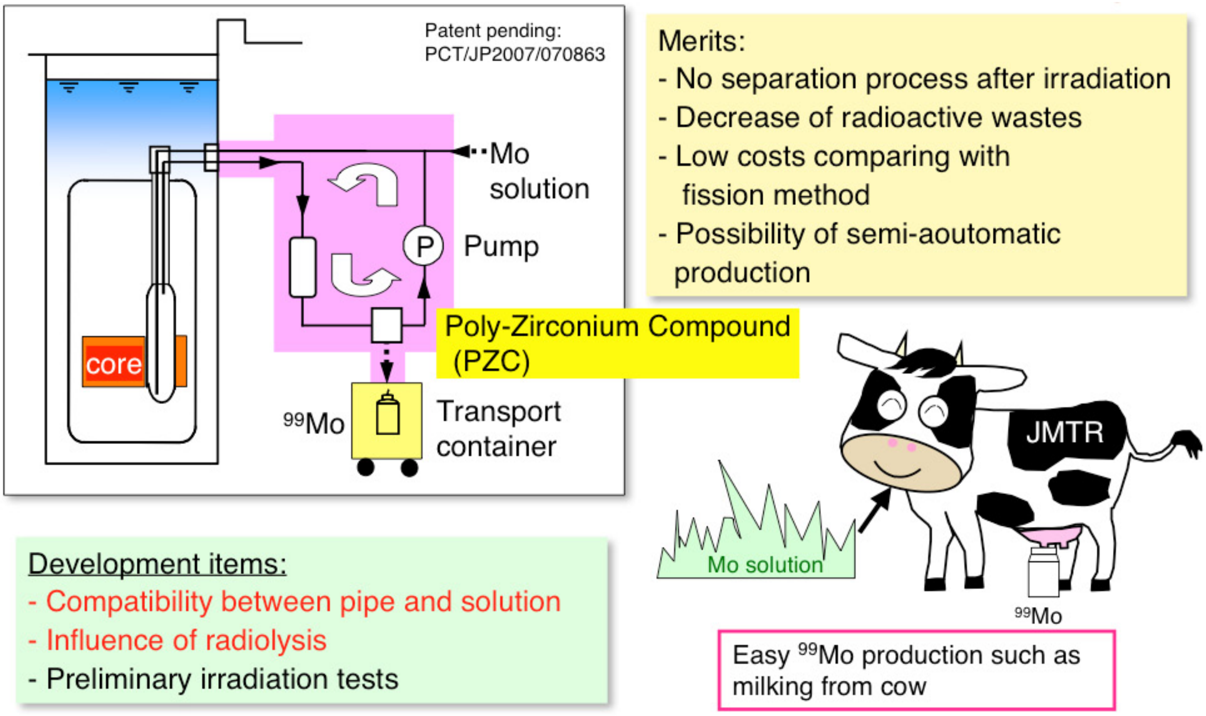

Fig. 2 Outline of molybdenum solution irradiation method (M-SUIMIT).

In the M-SUIMIT, the use of Poly-Zirconium Compound (PZC) is the one of the keys technologies. PZC had been developed as a high performance molybdenum adsorbent by JAEA and KAKEN Inc. in 1995 [6]. The molybdenum adsorbent performance of the PZC is over 100 times in comparison with the conventional molybdenum adsorbent of alumina. The PZC make chemical bond with ^{99}Mo selectively, and release ^{99m}Tc only after converted from ^{99}Mo. These characteristics mean that ^{99m}Tc generators may produce directly by use of the PZC, because the PZC can play the role of a separation for other nuclide and collection of ^{99}Mo in the M-SUIMIT system, simultaneously.

Table 1. Comparison of ^{99}Mo production methods in JMTR

Production method	Fission method (n, f)	Neutron capture method (n, γ)	
		Solid target	M-SUIMIT
Production in Japan	Difficult	Possible	Possible
Irradiation target	U-Al alloy	MoO_3 etc.	Molybdate solution
^{99}Mo radiation fraction	About 370 TBq/g(Mo)	37～74 GBq/g(Mo)	37～74 GBq/g(Mo)
Irradiation volume	30cm^3 (Rabbit)	30cm^3 (Rabbit)	~1,700cm^3 (Capsule)
Treatment as PIE	Isolation in cell (Complex)	Dissolution in cell (Relatively simple)	No special treatment
Expected production cost	1	1 /200	< 1/200
Radioactive waste	Quite a number of α, γ nuclides	A few γ nuclides	A few γ nuclides
Automation	Difficult	Difficult	Semi-automatic

However, in order to realize the M-SUIMIT, the following subjects should be investigated as a first step:

- Selection of irradiation target solution
- Compatibility between molybdenum solution and structural material under un-irradiated and irradiated conditions
- Stability of irradiation target solution
- Effect of radiolysis, γ ray irradiation, and neutron irradiation

In this study, the selection of the irradiation target, the compatibility between the molybdenum solutions and the structural materials, and the stability of the solutions were investigated under un-irradiated conditions. To study of the influence of radiolysis, γ ray irradiated tests of the molybdenum solution were also carried out.

PRELIMINARY SELECTION OF IRRADIATION TARGET

Comparison of molybdenum solutions as irradiation targets is shown in Table 2. Requirements for the molybdenum solution as the irradiation target in the M-SUIMIT are as follows:

(1) High molybdenum content for the efficient production of ^{99}Mo
(2) A few activation by-products for the prevention of radioactive contamination
(3) Good compatibility with the structural materials of pipes and capsules for the prevention of the corrosion and radioactive contamination
(4) Chemical stability

Based on (1) and (2), the potassium molybdate solution and the ammonium molybdate solution were selected as the candidates for the irradiation target. Especially, the potassium molybdate solution has an attractive character of high molybdenum content. It is considered that the generation of ^{24}Na with high radiation dose or ^{3}H with removal and isolating difficult property should be some demerit as the irradiation target.

Table 2. Comparison of molybdenum solution as irradiation target

Items \ Name	Potassium molybdate	Ammonium molybdate	Sodium molybdate	Lithium molybdate	Molybdenum oxide
Chemical formula	K_2MoO_4	$(NH_4)_6Mo_7O_{24}$	Na_2MoO_4	Li_2MoO_4	MoO_3
Solubility ($g/100g$-H_2O)	182	44	66	81	0.05
Mo content (g/100g-SWS*)	74	24	26	45	0.03
Liquid characteristics	Weak alkaline	Weak acid	Alkaline	Alkaline	-
Main activities of by-product ($T_{1/2}$)	^{42}K(12h), ^{92m}Nb(10d)	^{14}C($5.7x10^3$y), ^{92m}Nb	^{24}Na(15h), ^{92m}Nb	^{3}H(12y), ^{92m}Nb	^{92m}Nb

COMPATIBILITY WITH STRUCTURAL MATERIALS

Compatibility tests between the selected two molybdate solutions and structural materials (aluminum alloy and stainless steel) were carried out. Example of the test results is shown in Fig. 3. For the compatibility tests with stainless steel, it was found that the obvious corrosion was not observed on the stainless steel immersed in the two molybdate solutions. For the compatibility tests with aluminum alloy, it was found that the obvious corrosion was observed on aluminum alloy immersed in two molybdate solutions. Corrosion in ammonium molybdate solution was increased, and the precipitation was observed in the ammonium molybdate solution. As a results, the molybdenum concentration decreased by 40% in comparison with the initial value. It was also found that the corrosion on aluminum alloy in the ammonium molybdate solution can be prevented fairly by alumite treatment, and that chemical stability of ammonium molybdate solution can be controlled by the pH adjustment from about 6 to 8. However, pH control to increase the chemical stability of the ammonium molybdate solution is disadvantageous, because the molybdenum concentration will decrease.

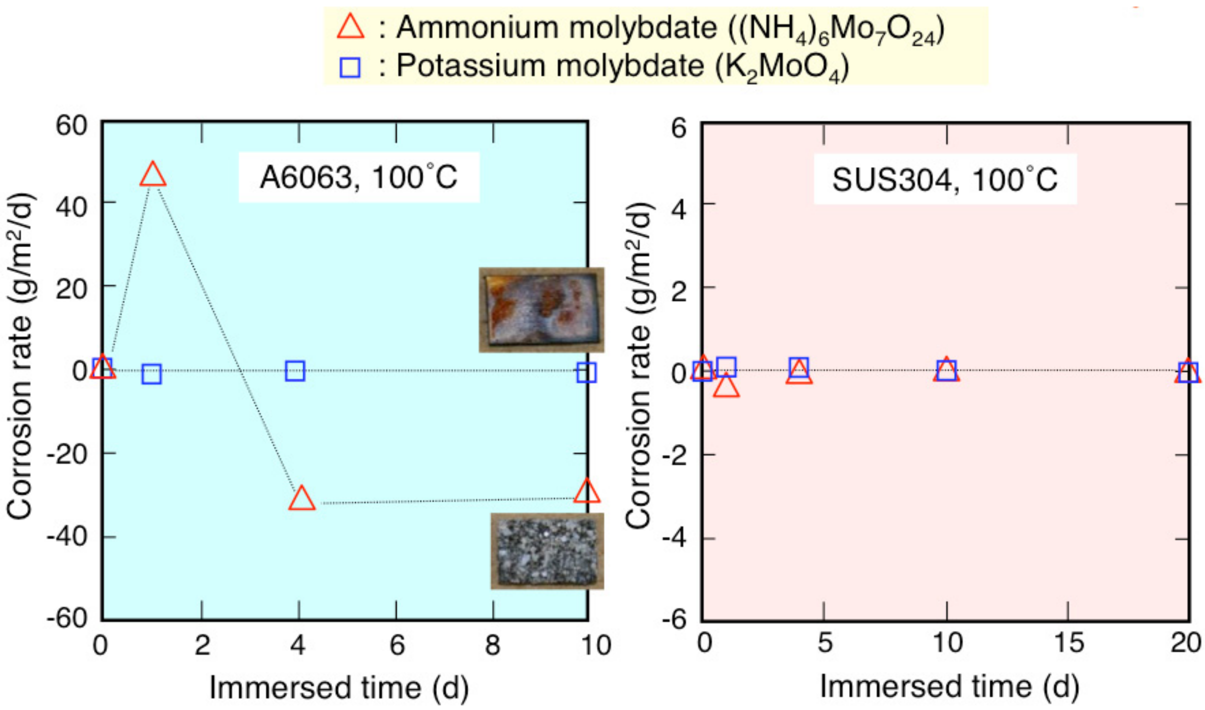

Fig. 3. Example of the results for compatibility tests.

GAMMA RAY IRRADIATION TEST

γ ray irradiation tests were carried out to investigate the effect of radiolysis, because large quantity of gasifications and flow blockage by precipitation due to chemical form change would affect the safety issues. Equipment and results of γ ray irradiation tests are shown in Fig. 4. Dose rate of γ ray irradiation was about 8×10^{3} Gy/h. Gas generation of molybdate solutions was one order higher than that of pure water as reference, and no flow blockage was observed under the γ ray irradiation. From the experience of water loop irradiation facilities, the generated gas will be possible to remove by ventilation of tank in the M-SUIMIT circulating system.

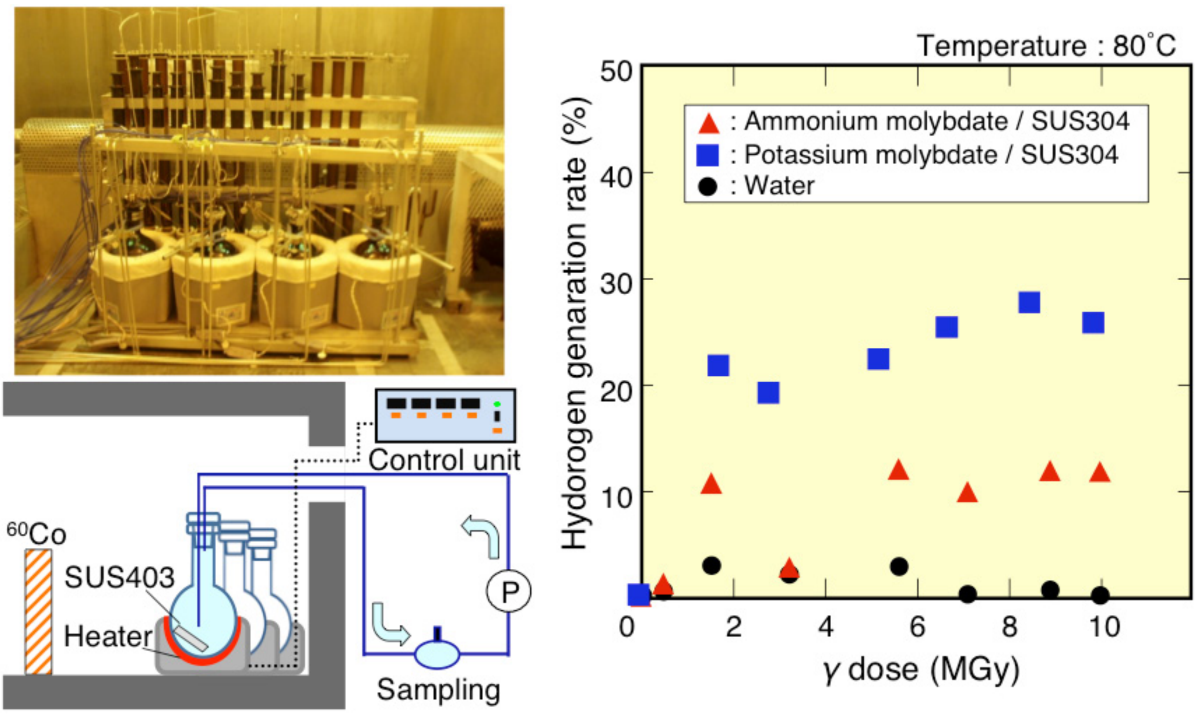

Fig. 4. Equipment and results of γ irradiation tests.

CONCLUSIONS

To increase the amount of ^{99}Mo production, the M-SUIMIT was proposed as a new production method. Evaluation on selection of the irradiation target, the compatibility between the molybdenum solutions and the structural materials, γ ray irradiation tests were carried out to realize the M-SUIMIT, and the following results were obtained (see Table 3).

Table 3. Summary of compatibility and γ irradiation tests

Items \ Name		Ammonium molybdate $(NH_4)_6Mo_7O_{24}$	Potassium molybdate K_2MoO_4
Stability of solution (at 100˚C)		Unstable (pH6-->pH8)	Stable (pH10)
Corrosion	SUS304	No	No
	A6063	Big	Small
γ Irradiation Test	H_2 generation	Small* (10 times of water)	Small* (20 times of water)
	Flow blockage	No	No

*: Removal of H_2 will be possible by ventilation of tank

- Potassium molybdate solution and ammonium molybdate solution were selected as the candidates for the irradiation target from the view point of high molybdenum content and few activation.
- For the chemical stability, the potassium molybdate solution is stable, ammonium molybdate solution is not stable. However, the chemical stability of ammonium molybdate solution can be controlled by the pH adjustment from about 6 to 8.
- From the results of compatibility tests, the obvious corrosion of stainless steel was not found, whereas the obvious corrosion of aluminum alloy was observed.
- From the results of γ ray irradiation test, gas generation of molybdate solutions was one order higher than that of water, and no flow blockage was observed.

From above results, the combination of potassium molybdate solution and stainless steel is considered as the highest possibility to realize the ^{99}Mo production by the M-SUIMIT.

ACKNOWLEDGMENTS

The authors greatly appreciate the helpful comments on this paper by Dr. M. Ishihara, deputy director of Neutron Irradiation and Testing Reactor Center.

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[4] Y. Inaba, K. Ishikawa, T. Ishida, K. Kurosawa, Y. Hishinuma, Y. Tatenuma and E. Ishitsuka, "Highly Efficient Production of Natural-Mo (n, γ) ^{99}Mo for Medical Use", Proc. of Nuclear Power of Republic Kazakhstan, Kurchatov, September 3-5 (2007).
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4.10 PRACTICE OF ADDING VALUE TO MATERIALS BY A NEUTRON IRRADIATION IN RESEARCH REACTORS

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Adding more value to materials by a neutron irradiation is one of the utilization areas of research reactors. A neutron transmutation doping (NTD) of Si, a gemstone coloration, and a track-etched membrane production are the three important subjects and these are being serviced on a commercial basis. Nevertheless, a further enhancement of the technologies is necessary, and the IAEA/RCA has supported the development and dissemination of the relevant technologies.

This article reports on what was achieved during the IAEA/RCA Regional Training Course on the Design and Operation of Neutron Irradiation Facilities which was held in Korea in April 2008 for two weeks. The Course, which was open to the Asian RCA member states, consisted of lectures, presentations from each country, and design and experiment exercises concerning the above three subjects. The lectures covered not only the principles but also the practice in every detail. The experience of Korea on the NTD and those of Indonesia and Thailand on the gemstone coloration drew concentrated attention of the participants. Meanwhile, as one of the exercises the participants designed a device for the NTD or the gemstone irradiation and analyzed its performance from the neutronics point of view. The Course was successful in deepening the understanding on the practice of value-adding technologies and also in sharing some ideas for their enhancement.

2008 KAERI-JAEA Joint Seminar on Advanced Irradiation & PIE Technologies
Nov. 5-7, 2008, Daejeon, Korea

Practice of Adding Value to Materials by Neutron Irradiation in Research Reactors

OH, Soo-Youl and JUN, Byung Jin
Korea Atomic Energy Research Institute

KAERI

A report on the achievements of the IAEA/RCA Regional Training Course on the Design and Operation of Neutron Irradiation Facilities, 7-18 April 2008, Daejeon, Korea

(*KAERI/GP-275/2008*)

KAERI

Contents

1. RCA's Research Reactor Project in 2007-2008
2. Methods of Value-Adding to Materials by Neutron Irradiation
3. Achievements of IAEA/RCA RTC
4. Concluding Remarks

KAERI

1.1 What is the RCA?

- Regional Cooperative Agreement for research, development and training related to nuclear science and technology for Asia and the Pacific
- RCA is one of the regional co-operative frameworks under the aegis of IAEA
(RCA 1972, AFRA 1990, ARCAL 1998, ARASIA 2002)
- Member states: AUS, BAN, CHN, INA, IND, JPN, KOR, MAS, MGL, MYA, NZL, PAK, PHI, SIN, SRI, THA, VIE (17 states)

KAERI

1.2 RR Collaborations in the Region

- RCA
  - ✓ Research reactor (RR) project
  - ✓ RI application projects
  - ✓ NAA application project for environment
  - ✓ Radiation protection project
- FNCA
  - ✓ RR projects
  - ✓ Safety culture project
  - ✓ RI application projects
- Others
  - ✓ IAEA activities such as CRP's, Non-RCA, etc.
  - ✓ Bilateral collaborations

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1.3 RCA RR Projects

- 1999-2004
  Improving RR operation & utilization
- 2005-2006
  RI production & neutron beam applications with assured safety
- 2007-2008
  Adding value to materials through irradiation with neutrons
  - ✓ Expanding the utilization of RR's with a specific focus on material irradiation to increase its value
  - ✓ Creating additional revenue

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2.1 Value-Adding: NTD

- Neutron Transmutation Doping (NTD) of Si
- Principle
  - ✓ Creation of impurities in Si ingot by neutron irradiation
  - ✓ $Si^{30}(n, \gamma)\ Si^{31} \rightarrow P^{31}$
- Issues
  - ✓ Design of irradiation facility for high doping uniformity and accurate irradiation time
  - ✓ Determination of the irradiation factor, K

    Neutron Fluence = $K\ (1/\rho_{after} - 1/\rho_{before})$
  - ✓ Demand for larger ingot (up to 8"), etc.

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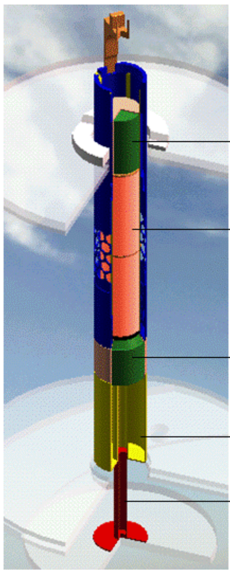
2.1 Value-Adding: NTD cont.

Measuring the straightness



An irradiation rig at HANARO

KAERI

2.2 Value-Adding: Gem Coloration

- Coloration of (colorless) topaz stone by neutron irradiation
- Issues
  - ✓ Design of irradiation facility for reducing/ eliminating thermal neutron irradiation and (thermal and radioactivity) cooling the stones:
    Out-of-core vs. In-core irradiation
  - ✓ How to achieve the color of "London blue"?

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2.2 Value-Adding: Gem Coloration cont.

Courtesy of TINT, Thailand

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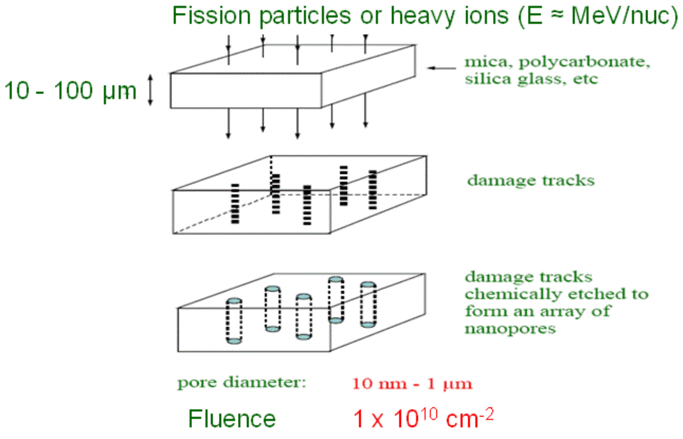
2.3 Value-Adding: TM

Principle of Track-etched Membrane



Courtesy of Dr. A. N. Nechaev

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2.3 Value-Adding: TM cont.

- **Issues**
  - ✓ Reactor TM vs. Accelerator TM
  - ✓ Track design (track length, pore size and shape, angle, etc.)
  - ✓ Film material (polymeric, inorganic etc.)

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3.1 Lectures at RTC

NTD

- Overview of NTD History and Activities in Several Countries
- Neutronics Design Procedure for the NTD Facility
- Cross Sections of Single Crystal Si for the NTD Neutronics Analysis
- Structural Defects of Materials by Neutron Irradiation
- Neutron Transmutation Doping Facility in HANARO
- Procedure of Silicon Irradiation for NTD in HANARO
- Design for Heat Removal from a NTD Device using Computational Fluid Dynamics
- SPND and Dynamic Compensation, and
- so on

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3.1 Lectures at RTC (cont.)

Track-etched Membrane

- Main principle of Track-etched Membrane Technologies
- Physico-chemical Properties of Track-etched Membranes and Characterization Technologies
- TM Applications and New Concepts to Produce Innovative Polymeric TM
- Application of Track-etched Membrane in Nanotechnology: Basic Concept and Examples
- Plasmapheresis and Therapeutic Apheresis Market
- Membrane Plasmofilters PFM-500 & PFM-800, and
- so on

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3.2 Case Studies at RTC

- NTD
  - ✓ Silicon NTD in CIAE, China
  - ✓ Neutron Transmutation Doping Silicon in Indian Research Reactors
  - ✓ Neutron Transmutation Doping Facility in Indonesia
  - ✓ Neutron Transmutation Doping in HANARO
- Gemstone Coloration
  - ✓ Enhancement of Gemstone in Indonesia
  - ✓ Gemstone Coloration in Thailand

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4. Concluding Remarks

- **Details on the practice of NTD, gemstone coloration and TM were open and discussed.**
- **A database of research reactors will be available soon: It will provide an overall picture of usefulness and effectiveness of irradiation facilities available among the member states.**
- **There are chances of collaboration in the region.**
  - ✓ Backup in NTD irradiation
  - ✓ Share of technology and ideas for enhanced service,…

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5. Post Irradiation Examination Technology

5.1 IRRADIATED FUEL EXAMINATION PROGRAM FOR ADVANCED PWR FUELS IN KOREA

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YongBum Chun[b], PyoungWi Moon[c], Teukki Choe[c]

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[b]*KAERI, 1045 Daedukdaero, Yuseong-gu, Daejeon 305-353,*
[c]*KHNP, 411 Yongdongdaero Gangnam-gu SEOUL 135-791*

In Korea, sixteen PWRs are in operation and eight PWRs are to be built by 2016 and eleven more PWRs' construction is planned by 2030. Sixteen operating PWRs comprise of 8 Optimized Power Reactors (OPRs) and eight Westinghouse type reactors (17x17, 16x16 and 14x14 types). Korea Nuclear Fuel (KNF) is fuel designing and manufacturing company, and has been supplying nuclear fuels for all operating PWRs in Korea since 1989 using imported fuel technology. From 1999, advanced fuels, PLUS7™, 16ACE7™ and 17ACE7™, have been jointly developed with foreign vendors. After successful in-reactor performance verification (PSE) for lead test assemblies, PLUS7 has been commercially supplied to 8 OPRs from 2006. 16ACE7™ was supplied to Kori-2 in 2008, and 17ACE7™ will be supplied to 6 Westinghouse 17 type reactors from 2009. Now, additional fuel development program and fuel code development program are on-going. Also, several operation strategies to increase fuel economy have been applied or scheduled such as high burnup, longer cycle operation, power uprating etc. Therefore, irradiated fuel's examination is indispensible to get more reliable fuel performance data, which are used in verification of advanced fuel, developing performance model, and evaluating quantitative design margin etc,. This paper describes the fuel examination program for the advanced PWR fuels in Korea.

Irradiated Fuel Examination Program for Advanced PWR Fuels in Korea

2008. 11.6.

KNF: JI Kim, JC Shin, YK Jang, KL Jeon
KAERI: YB Chun
KHNP: PW Moon, TK Choe

Introduction of Korea PWR plants

2

Nuclear Power Plants in Korea

Type	In Operation	Under Construction or Planning	Total
OPR1000	8	4	12
APR1400	0	4	4
WH14x14	1	0	1
WH16x16	1	0	1
WH17x17	6	0	6
CANDU	4	0	4
Total	20	8	28

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Nuclear Fuel Operation History & Plan

NPP type		Plant	FA type	1990 – 2015
PWR	WH Type	Kori-1	14x14	Fuel Import → KOFA → OFA
PWR	WH Type	Kori-2	16x16	Fuel Import → KOFA → STD → 16ACE7
PWR	WH Type	Kori-3/4	17x17	Fuel Import → KOFA → V5H → RFA → 17ACE7
PWR	WH Type	YGN-1/2	17x17	Fuel Import → KOFA → V5H → RFA → 17ACE7
PWR	WH Type	UCN-1/2	17x17	Fuel Import → KOFA → V5H → RFA → 17ACE7
PWR	OPR1000 (CE Type)	YGN-3/4	16x16	STD → Guardian → PLUS7
PWR	OPR1000 (CE Type)	YGN-5/6	16x16	STD → Guardian → PLUS7
PWR	OPR1000 (CE Type)	UCN-3/4	16x16	STD → Guardian → PLUS7
PWR	OPR1000 (CE Type)	UCN-5/6	16x16	Guardian → PLUS7
PWR	OPR1000 (CE Type)	S-Kori-1/2	16x16	Gu → PLUS7
PWR	OPR1000 (CE Type)	S-WS-1/2	16x16	Gu → PLUS7
PWR	APR1400	S-Kori-3/4	16x16	PLUS7
PHWR		WS-1/2/3/4	-	Standard CANDU Fuel

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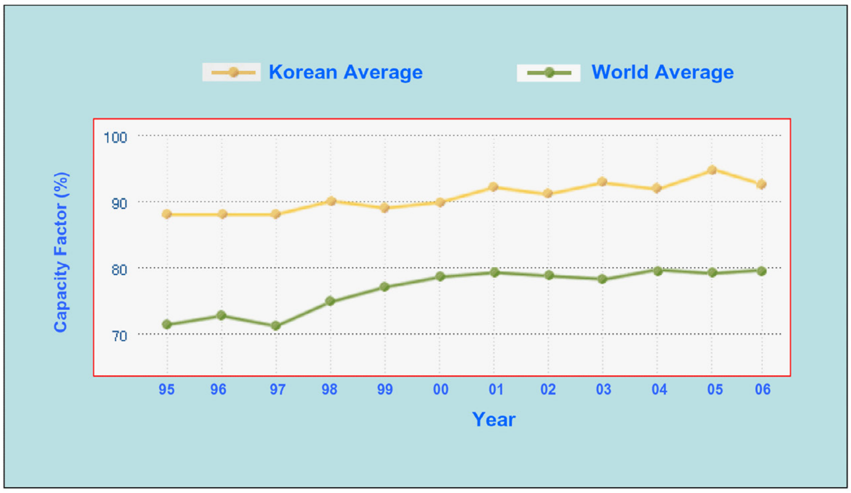
Nuclear Plants Capacity Factor



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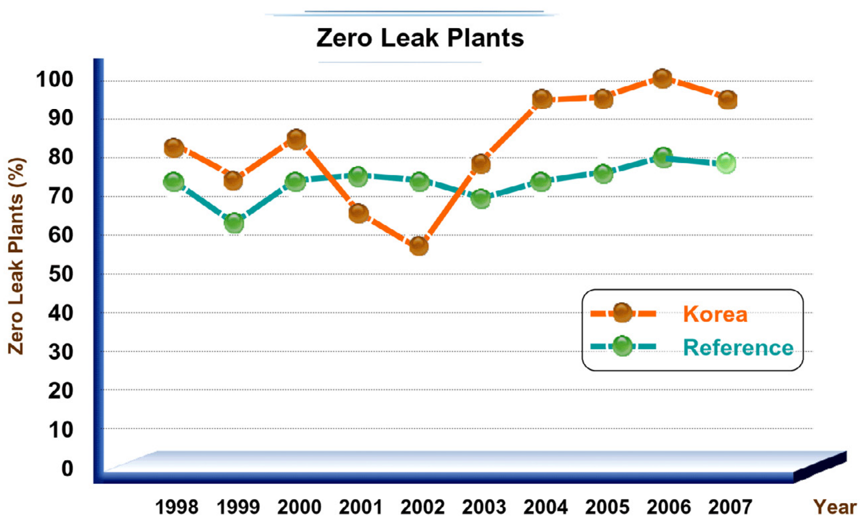
5

Fuel Performance Record in Korea



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Irradiated Fuel Examination Program

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PIE Needs on Korea PWR Plants

- Continuous Advanced Fuel Development Fuel Program
  - To achieve more higher fuel reliability and economics
    - ✓ PLUS7 in OPR 1000, APR1400
    - ✓ 17ACE7 in 17 type W plants
    - ✓ 16ACE7 in 16 type W plant etc.,
- Fuel Performance Code Development Program
  - the current model's validation in the condition of Korea plants
- More Severe Operation Strategy
  - Quantitative Design Margin Evaluation
    - ✓ Power Uprating
    - ✓ Longer Cycle Operation
- Root Cause Analysis on Fuel Failure

8

PLUS7 Fuel for OPR1000 & APR1400, ACE7 Fuel for W 16&17

TARGET (PLUS7, ACE7)	**Improve the Current Fuel (Guardian, RFA, 16STD) Performance** - Batch Avg. Discharge BU > 55 GWD/MTU - Overpower Margin Increase > 10% - Meet 0.3g Seismic Requirements(PLUS7) - Minimize Fuel Assembly Bowing (16 &17 ACE7) - Zero Defect against Debris & Grid-to-Rod Fretting Wear
SCHEDULE (PLUS7)	• Project Initiation : April 1999 • LTA Irradiation Initiation : January 2003 • Region Application : June 2006
SCHEDULE (ACE7)	• Project Initiation : August 2001 • LTA Irradiation Initiation : Jan. 2005(16ACE7), Jun. 2005(17ACE7) • Region Application : Aug. 2008(16ACE7), Feb. 2009(17ACE7)

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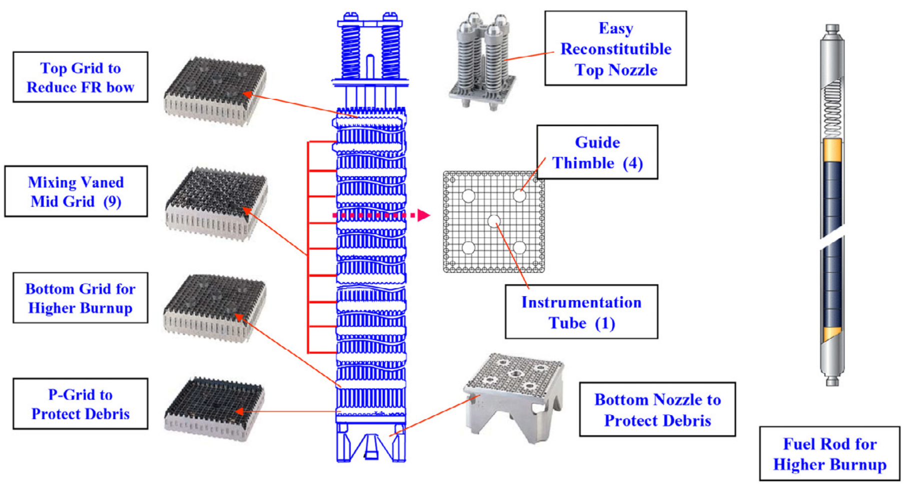
PLUS7 Design Feature



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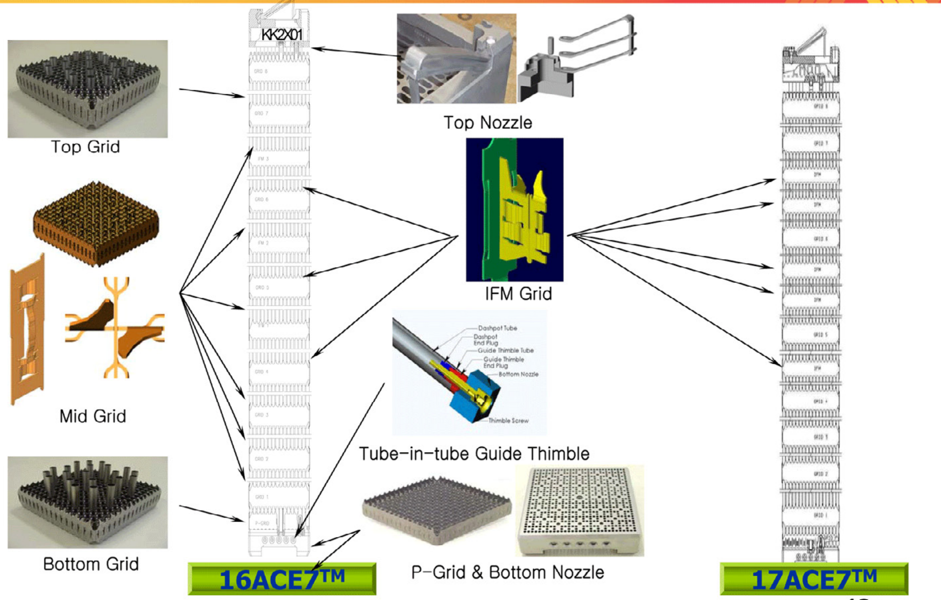
ACE7 Design Feature



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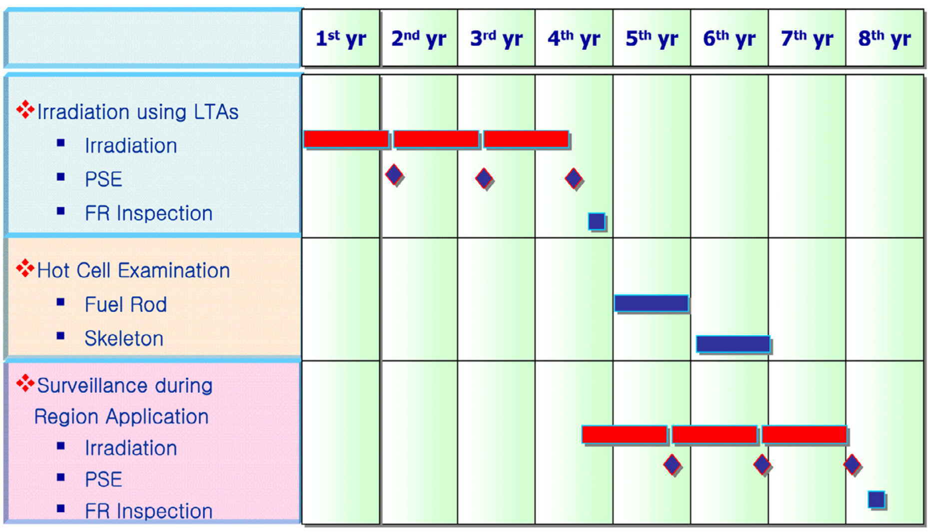
Procedure of In-Reactor Verification of Advanced Fuels



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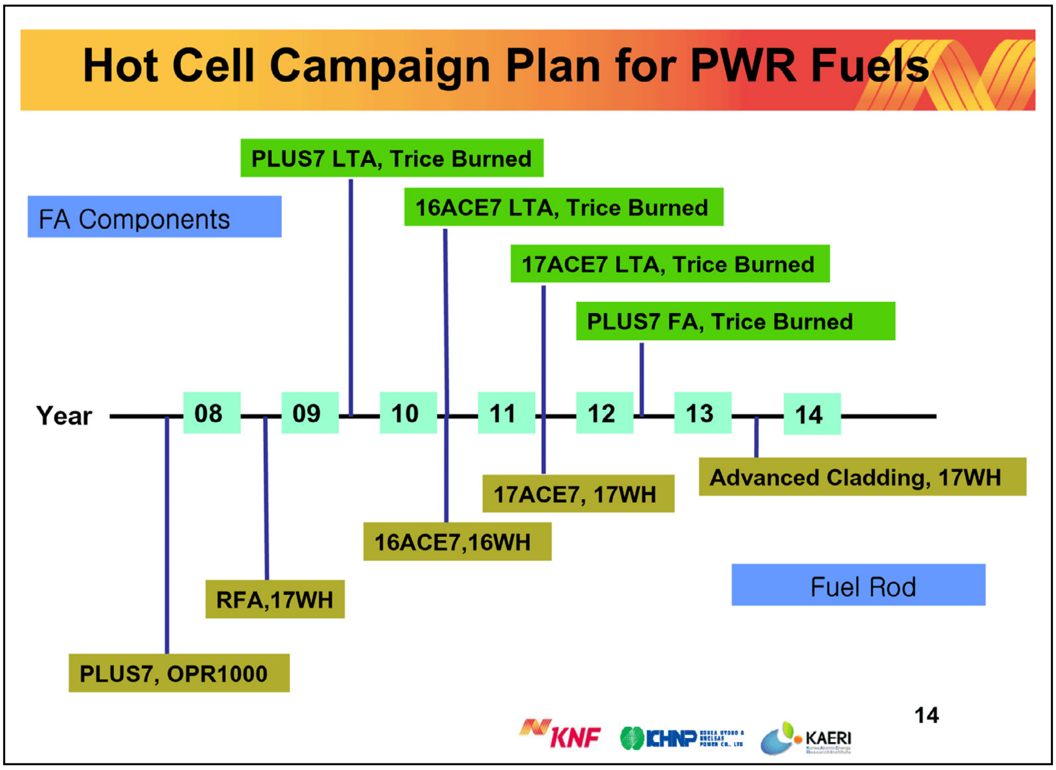
Hot Cell Campaign Plan for PWR Fuels
FA Components
PLUS7 LTA, Trice Burned
16ACE7 LTA, Trice Burned
17ACE7 LTA, Trice Burned
PLUS7 FA, Trice Burned
Year
08
09
10
11
12
13
14
Advanced Cladding, 17WH
17ACE7, 17WH
16ACE7,16WH
RFA,17WH
PLUS7, OPR1000
Fuel Rod
KNF
KHNP
KAERI
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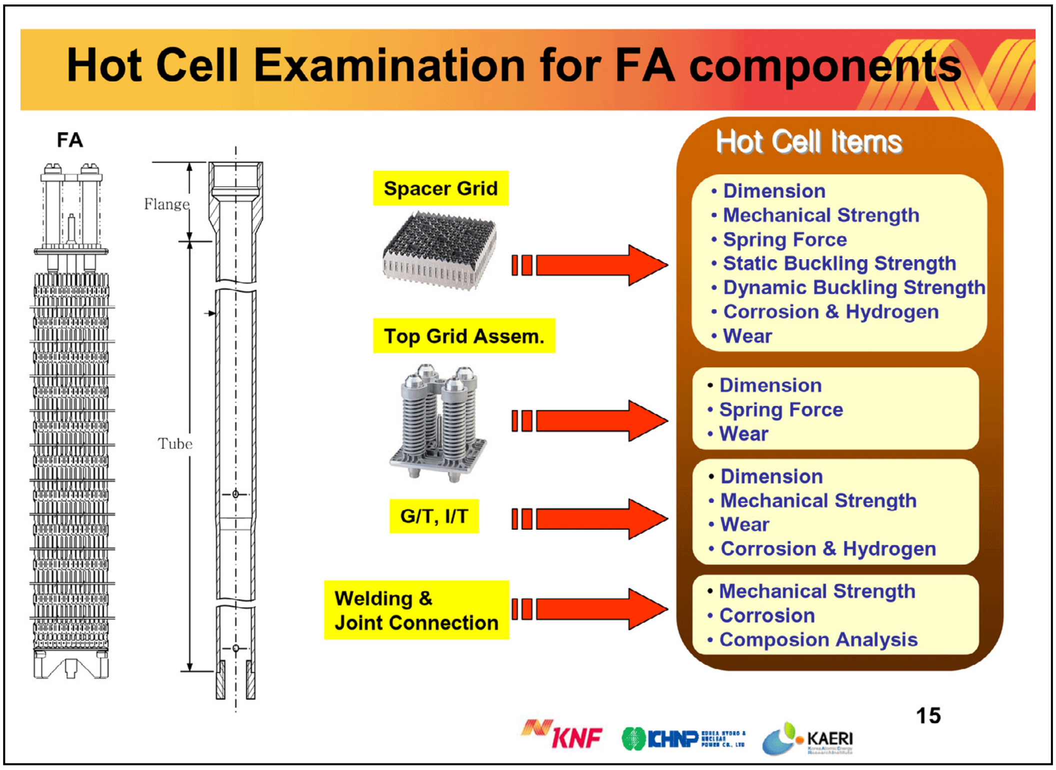
Hot Cell Examination for FA components
FA
Flange
Tube
Spacer Grid
Top Grid Assem.
G/T, I/T
Welding &
Joint Connection
Hot Cell Items
• Dimension
• Mechanical Strength
• Spring Force
• Static Buckling Strength
• Dynamic Buckling Strength
• Corrosion & Hydrogen
• Wear
• Dimension
• Spring Force
• Wear
• Dimension
• Mechanical Strength
• Wear
• Corrosion & Hydrogen
• Mechanical Strength
• Corrosion
• Composion Analysis
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Summary

- Korea has 20's operating NPPs and they show very good fuel performance.
- Several PIE campaigns are scheduled to cope with continuous advanced fuel development, code development program and increase fuel reliability in Korea.
- Establishing hot cell infrastructure and hot cell exam is performed cooperatively with KAERI/KNF/KHNP
- The results of irradiation fuel exam will be feedback to design, manufacturing and power operation areas, which finally will contribute for zero leaking fuel failure.

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