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ANNUAL REPORT OF JMTR
FY1996
(APRIL 1, 1996 - MARCH 31, 1997)

February 1998

Department of JMTR

日本原子力研究所
Japan Atomic Energy Research Institute

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Annual Report of JMTR
FY1996
(April 1, 1996 - March 31, 1997)

Department of JMTR ※

Oarai Research Establishment
Japan Atomic Energy Research Institute
Oarai-machi, Higashiibaraki-gun, Ibaraki-ken

(Recieved January 29, 1998)

During FY1996, the JMTR was operated for 2 complete cycles (118th and 119th cycles) and was utilized for the research and development programs on the reactor technology for LWRs, HTTR and fusion reactor, as well as for basic research of fuels and materials, and for radio-isotope productions. With regard to technology development in the JMTR, improvement of evaluation technique for local neutron spectrum, development of new oxygen sensor for oxide fuel pellet are proceeded. A research on the blanket material for thermonuclear fusion reactor was also progressed.

Keywords : JMTR, Irradiation Facility, Hot Laboratory, Reactor Technology, Maintenance, Neutron Irradiation Studies, Post Irradiation Examinations

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材料試験炉　英文年報（1996年度）
(1996年4月1日～1997年3月31日)

日本原子力研究所大洗研究所
材料試験炉部※

(1998年1月29日受理)

1996年度、ＪＭＴＲは118、119サイクルの合計２サイクルの運転を行い、軽水炉及び核融合炉開発等に利用された。ＪＭＴＲの技術開発では、照射試料位置における中性子スペクトル評価精度を向上させるための技術開発、燃料棒内部の化学的基礎データ測定のための燃料棒用酸素センサの開発を進めた。核融合開発に関しては、ブランケット照射挙動に関する研究を進めた。

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Message from the Director

The Japan Materials Testing Reactor (JMTR) has been providing unique capability for the testing of materials and fuels since its beginning of the operation in 1968, as dedicated platform for irradiation testing. The reactor contributed to various R&D activities in the nuclear researches in Japan, such for the light water reactors (LWR), fast breeder reactor (FBR), advanced thermal reactor (ATR), High Temperature Engineering Test Reactor (HTTR), and fundamental researches on the fuels and materials. The production of radio-isotopes has been also carried out in JMTR.

During performing these activities, the Department of JMTR has been engaging in development of many features and techniques of irradiation tests and post irradiation examinations, through designing of capsules and instrumentation systems and of reactor control and measurement techniques.

Since 1994, the operation of JMTR has been fully employing the fuels with low enriched uranium according to the international collaboration effort following the non-proliferation policy of fissile materials.

This annual report outlines the activities in the Department of JMTR during the fiscal year 1996 (April 1996 to March 1997). It contains records of reactor operation, irradiation tests and post irradiation examinations in the JMTR Hot Laboratories, activities in designing of capsules and devices for irradiation tests, as well as the topics among the engineering development works in relation to the contribution of JMTR irradiation tests for nuclear power development programs in Japan.

It will be our great pleasure that this report helps a broad range of the people to understand the capability of the JMTR and the achievements in the department.

December 1997
Department of JMTR project
Director

O. Baba

BABA Osamu

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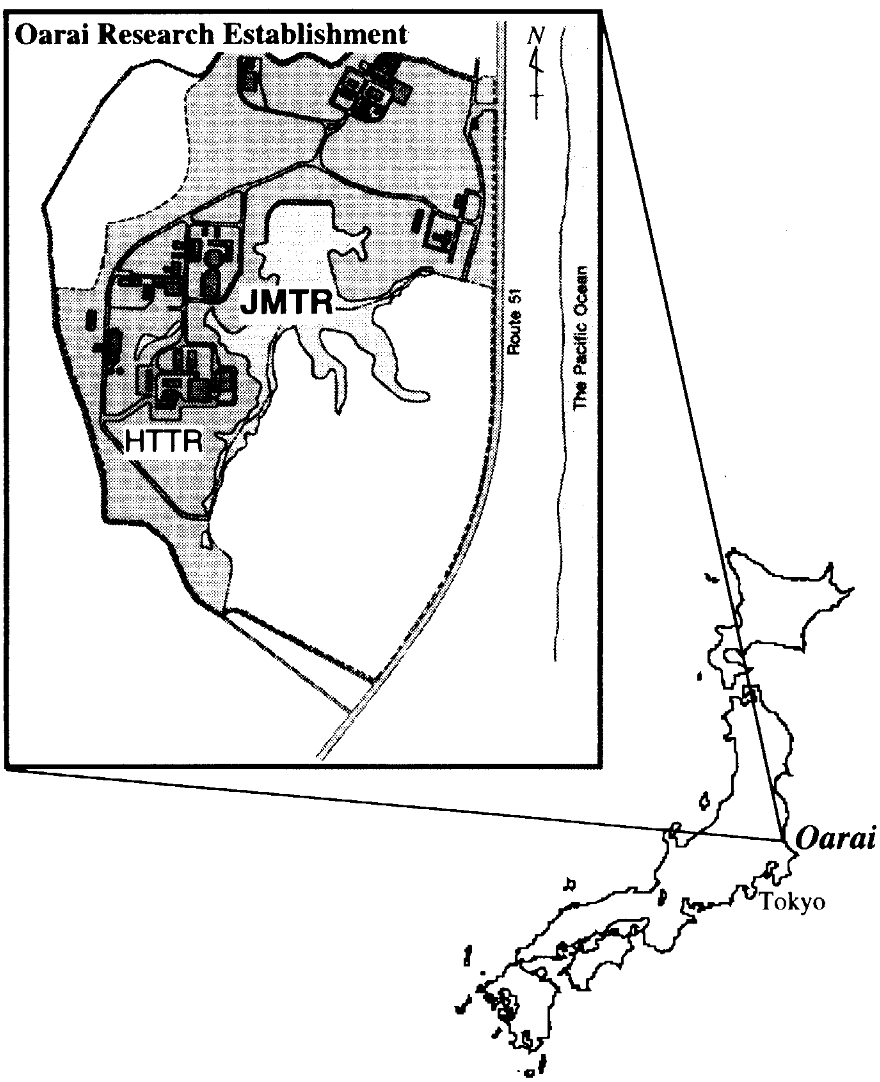
1. Introduction

Location of JMTR

The Japan Materials Testing Reactor (JMTR) is located in the Oarai Research Establishment, one of the five research establishments of the Japan Atomic Energy Research Institute (JAERI), about 100 km north-east from Tokyo (Fig. 1.1).

In the Oarai Research Establishment, the High Temperature Engineering Test Reactor (HTTR) is under construction. It will reach first criticality in 1998.

Fig. 1.1
JAERI Oarai Research Establishment



Historical background

In late 1950's, Japanese industries requested the government to construct a material testing reactor for irradiation tests of nuclear fuels and structural materials and for radioisotope production. The Japan Atomic Energy Commission (JAEC) approved the construction of materials testing reactor in JAERI as JMTR, and started the design of the reactor in 1962.

In the design concept, the main characteristics of JMTR were

(1) flexibility of the core configuration for irradiation tests, and

(2) adoption of the proven technologies which had been experienced in ETR, ORR and other high flux reactors, for safety and economic reason.

The construction of JMTR started in April 1965.

JMTR achieved the first criticality in March 1968 and started the first two operation cycles in December 1969 as test program. It accomplished 50 MW full power in January 1970. The cycle operation at 30 MW begun in August 1970 and the 50 MW operation started in October 1971. The cumulative power reached 100,000 MWd during the 108th operation cycle in March 1994.

Initially JMTR employed high enriched uranium (HEU) fuel elements. Under the international collaboration on the non-proliferation of fissile materials, JAERI started the program to employ reduced enrichment fuels for research reactors in 1979. Core configuration of JMTR was changed by totally employing medium enriched uranium (MEU) fuels in July 1986, 75th operation cycle. The fuel type was further changed to low enriched uranium (LEU) fuels since 108th operation cycle in Jan. 1994.

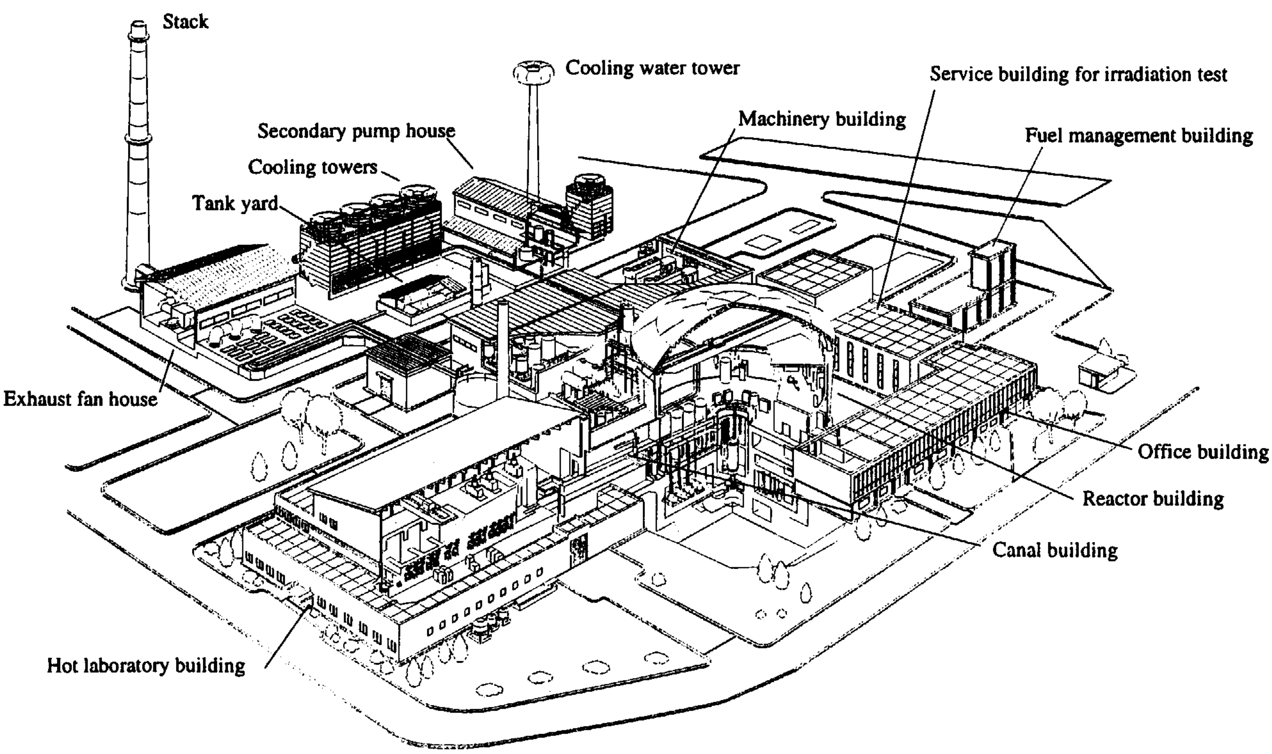
Table 1.1
History of JMTR

Aug.	1963	Construction of JMTR in Oarai was approved by Japan Atomic Energy Commission (JAEC)
Apr.	1965	Construction of JMTR started
Mar.	1967	Construction of Hot Laboratory started
Apr.	1967	Foundation of Oarai Research Establishment
Mar.	1968	JMTR achieved First criticality
Jan.	1969	Completion of the Hot Laboratory
Apr.	1972	Completion of Oarai Water Loop -2 (OWL-2)
Oct.	1972	Completion of Hydraulic Rabbit Irradiation Facility (HR-2)
Jan.	1977	Completion of Oarai Gas Loop (OGL-1)
	1979	International Collaboration on Non-proriferation of Fissile Material
Jan.	1983	Cumulative power reached 50,000 MWd
Jan.	1984	Completion of Oarai Shroud Irradiation Facility (OSF-1)
Jul.	1986	45% enriched (MEU) fuels were employed.
Jan.	1989	Replacement of OSF-1 in-pile stainless steal tube with zircaloy tube
Mar.	1992	JMTR achieved 100 operation cycles
Jan.	1994	20% enriched (LEU) fuels were employed.
Mar.	1994	Cumulative power reached 100,000 MWd
Dec.	1995	Removal of JMTRC was began.

2. Overview

JMTR facility consists of the reactor building, the machinery building, secondary pump house, cooling towers, exhaust fan house, tank yard, cooling water tower, main stack, service building for irradiation tests, canal building, hot laboratory building and office building. The arrangement of the buildings is shown in Fig. 2.1.

Fig. 2.1
Arrangement of the JMTR buildings

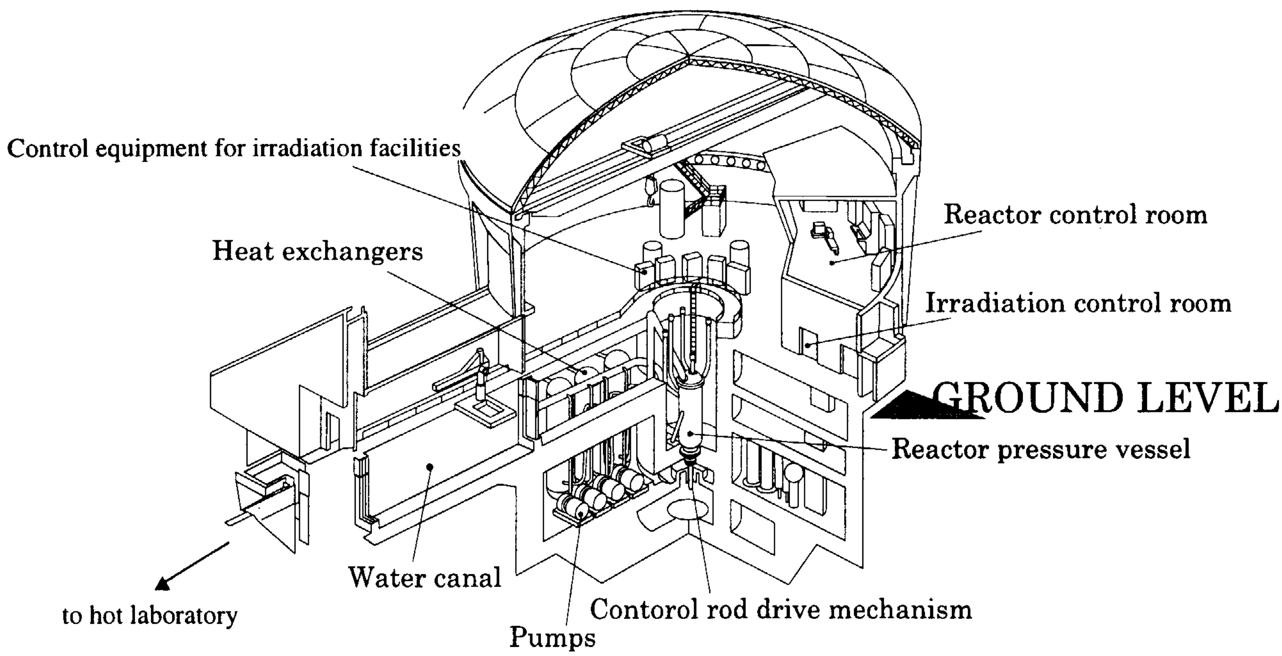


The reactor building has reinforced concrete structure. It's dimensions are 41 m in diameter, 20.4 m high above and 23.5 m deep below the ground level in height. The reactor pressure vessel is placed in the reactor pool near the center of the reactor building (Fig. 2.2).

The reactor control room and the control room for the irradiation facilities are located in the reactor building.

The hot laboratory is connected to the reactor building through a water canal. The irradiated specimens can be transferred to the hot laboratory underwater through the canal without any shielding devices for the post irradiation examination.

Fig. 2.2
Reactor building

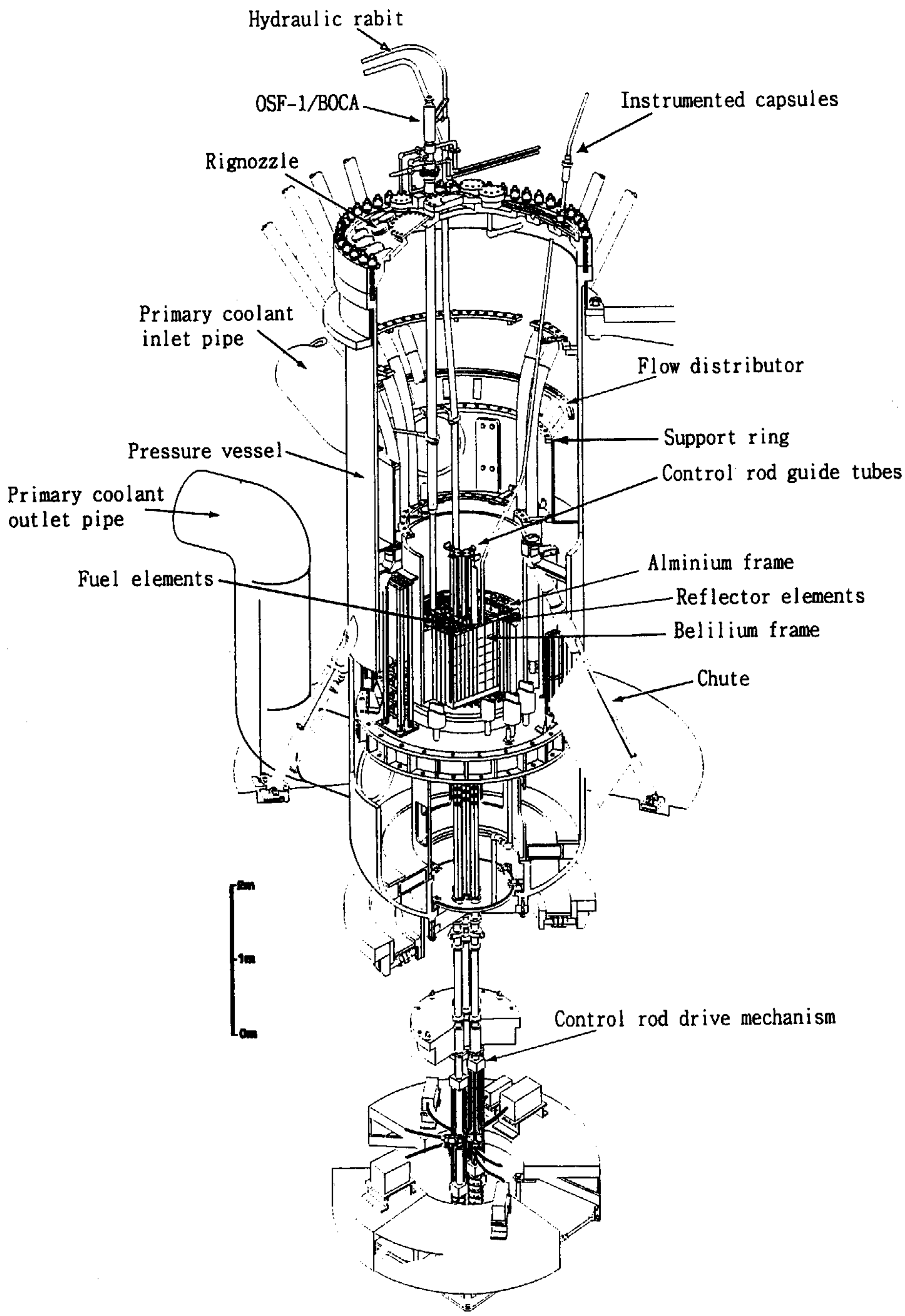


2.1. JMTR

Reactor

JMTR is a tank-in-pool type reactor cooled and moderated by light water with thermal power of 50 MW. JMTR is normally operated as five cycles a year, 26 days per cycle. Fig.2.1.1 shows the cross-section of the reactor. The engineering specifications of JMTR are described in Table 2.1.1. Schematic view of the reactor is shown in Fig. 2.1.2.

Fig. 2.1.1
Cross-section of JMTR



Pressure vessel

The pressure vessel, 9.5 m in height, 3 m in diameter and 34 mm thick, is made of stainless steel and its design pressure is 1.8 MPa. The pressure vessel is installed in the reactor pool which is 13 m in deep. The pressure vessel contains the reactor core, irradiation facilities and etc.. The top closure head of the pressure vessel provides many nozzles for in-pile loops, instrumented capsule irradiation facilities, and hydraulic rabbit irradiation facilities. Two chutes are attached on the side wall of the pressure vessel to

Fig.2.1.2 Cross-section of JMTR pressure vessel

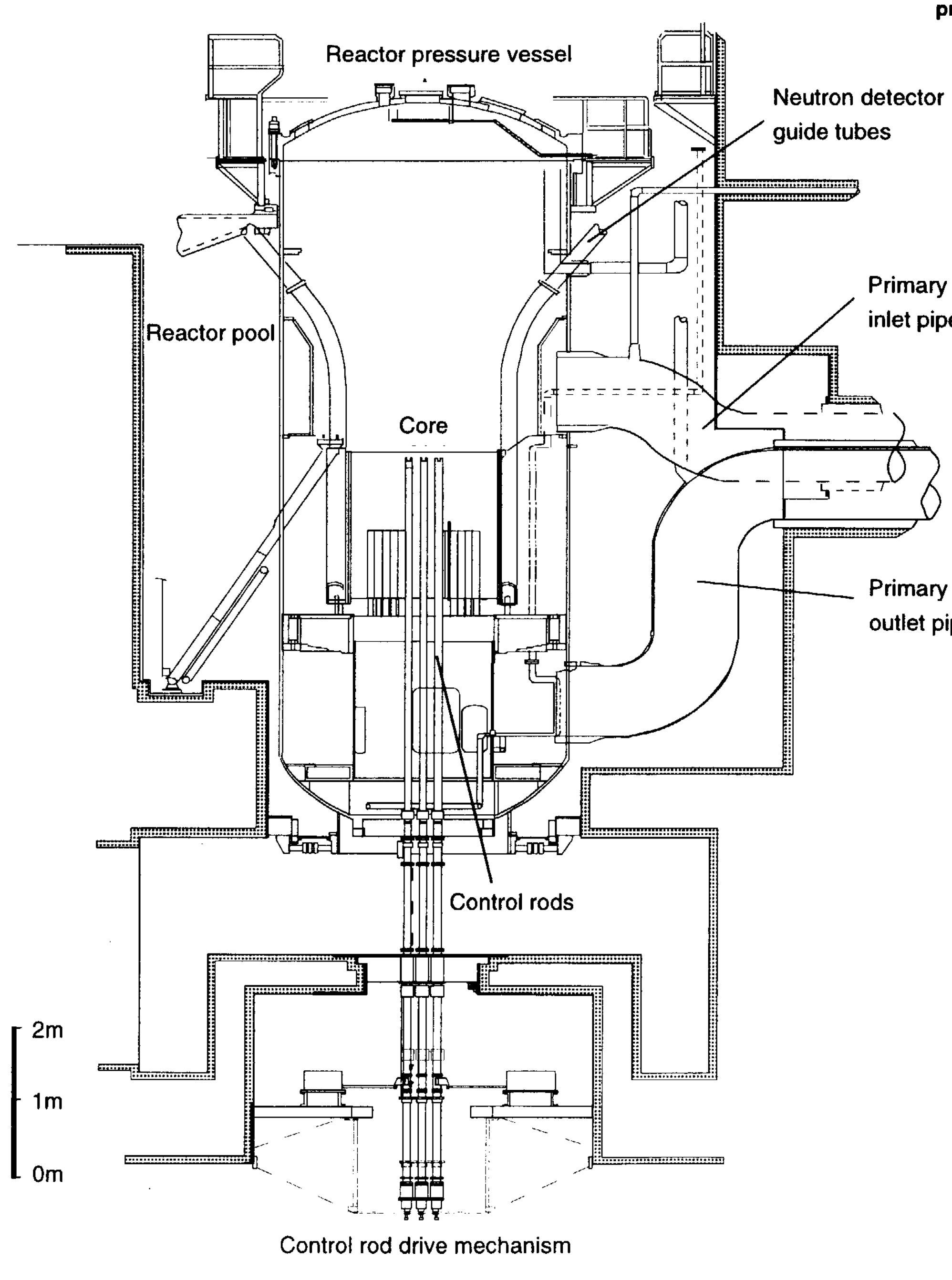


Table 2.1.1
Engineering specifications
of JMTR

Thermal power			50 MW	
Neutron flux	Fuel region	ϕ_{th} (<0.68 eV)	Max. 4×10^{14} [n/cm²·s]	
		ϕ_f (>1MeV)	Max. 4×10^{14} [n/cm²·s]	
	Reflector region	ϕ_{th} (<0.68 eV)	Max. 4×10^{14} [n/cm²·s]	
		ϕ_f (>1MeV)	Max. 1×10^{14} [n/cm²·s]	
Power density	Core average		Approx. 492 [kW/l]	
	Average heat flux		120 [W/cm²]	
Pressure vessel	Material		Stainless steel	
	Height		9.5 [m]	
	Diameter		3 [m]	
	Thickness		34 [mm]	
Core	Material of structure		Stainless steel	
	Effective hight		750 [mm]	
	Equivalent diameter		416 [mm]	
	Core loading(^{235}U)		Max. 11 [kg]	
Core components			Standard fuel element	Fuel follower
Fuel	Number of fuel elements		22	5
	Type		Plate	Plate
	Dimension		76.2□× 1,200 [mm]	63.6□× 890 [mm]
	Fuel plate	Number	19 per element	16 per element
		Thickness	1.27 [mm]	1.27 [mm]
		Total length	780 [mm]	769 [mm]
		Thickness of cladding	0.38 [mm]	0.38 [mm]
		Cladding material	Aluminum alloy	Aluminum alloy
		Thickness of meat	0.51 [mm]	0.51 [mm]
		Effective length	750 [mm]	750 [mm]
	Meat	Width	61.6 [mm]	49.7 [mm]
	^{235}U content	Per element	Approx. 410 [g]	Approx. 275 [g]
	Enrichment		Approx. 20 [%]	Approx. 20 [%]
	Burnable absorber	Number of Cadmium wires	18 (ϕ 0.3)	16(ϕ 0.3)
Control rod	Type	Top entry bottom mounted with follower		
	Number		5	
	Absober	Material	Hafnium	
		Dimension	63.0□× 800 [mm]	
Reflector	Material	Reflector element	Beryllium	
			Aluminum	
		H-shaped frame	Beryllium	
Grid plate	Material		Stainless steel	
Reactivity restviction	Max. excess reactivity		15 [%Δk/k]	
	Shutdown margin		≤0.9 [Keff]	
Primary cooling system	RPV inlet temprature		Max. 49 [℃]	
	RPV outlet temperature		Approx. 56 [℃]	
	Fuel surface temp		Max. 186[℃]	
	Fuel channel velocity		10 [m/s]	
	Fuel channel flow rate		125 [m³/h]	
	Total flow rate		6,000 [m³/h]	
	Core inlet pressure		1.4 [MPa]	
	Pressure difference between RPV inlet / outlet		0.32 [MPa]	

transfer the spent fuel and reflectors from the core to the reactor pool. The primary coolant, entering through the primary coolant inlet pipe, flows downwards through the reactor core for heat removal, passes the lower plenum and exits through the primary coolant outlet pipe.

Reactor core

The reactor core, 1,560 mm in diameter and 750 mm in effective height, consists of fuel elements, control rods, reflectors and the H-shaped beryllium frame as shown in Fig.2.1.3. The H-shaped beryllium frame separates the core into four regions. The reactor core consists of 204 lattice positions, 77.2 mm square each, arranged in a square matrix. The 22 standard fuel elements and 5 control rods are loaded in the lattice positions of center region.

The H-shaped beryllium frame and beryllium and aluminum reflector elements are equipped with irradiation holes. These irradiation holes are filled with solid plugs of the same material when they are not loaded with the capsules.

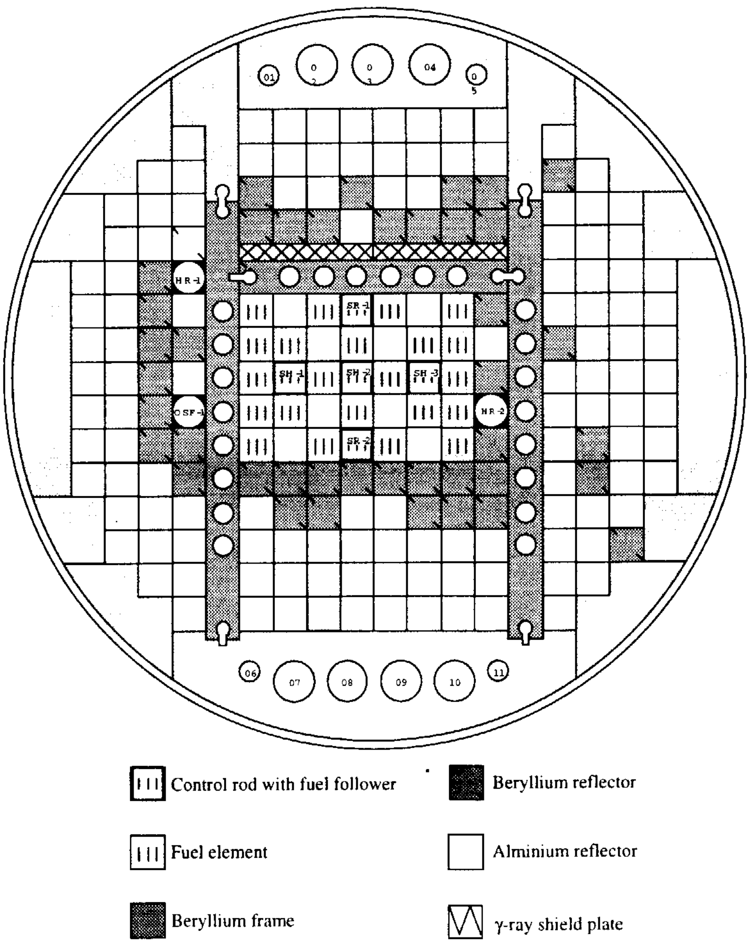


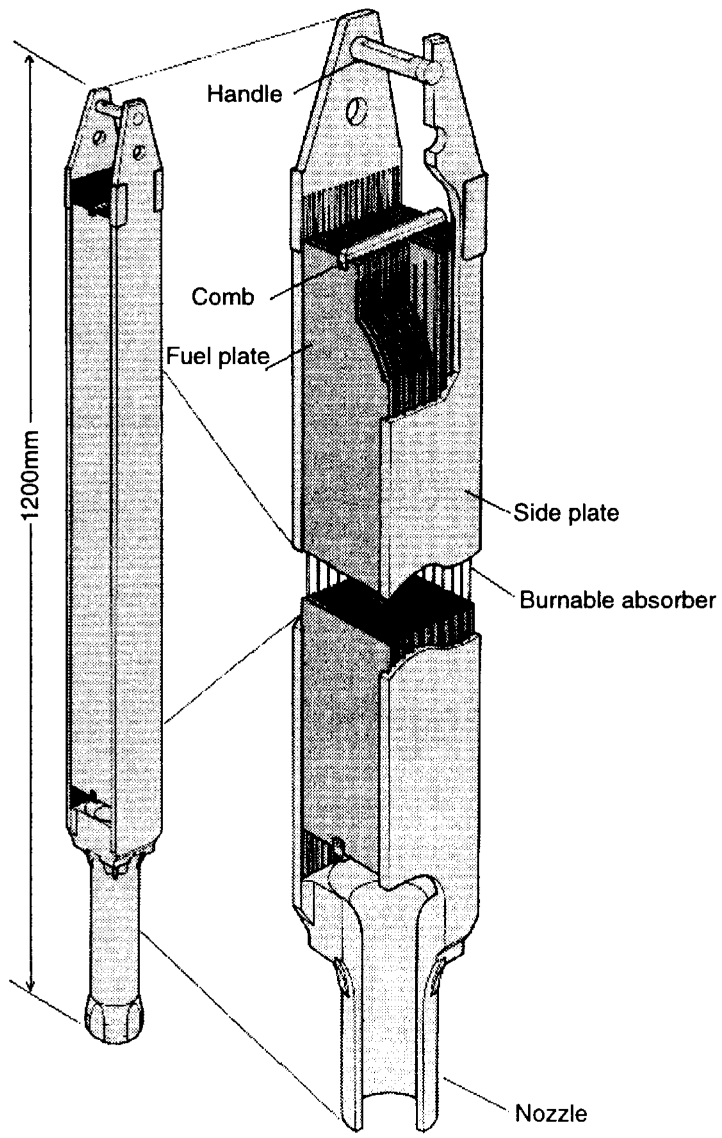
Fig. 2.1.3
Core configuration

Fuel element

The fuel elements are modified ETR type, flat-plate assemblies. They are classified into standard fuel elements and fuel followers.

Each standard fuel element consists of 19 fuel plates which are 1.27 mm thick, 70.5 mm wide, and 780 mm long (Fig.2.1.4). The fuel plates are attached to the aluminum alloy side plates by a "roll-swagging" technique. As burnable neutron absorbers, cadmium wires with aluminum cladding are inserted into grooves where fuel plates are attached. A fuel plate consists of a 0.5 mm thickness layer of uranium-silicon-aluminum fuel (U_3Si_2-Al) and aluminum cladding.

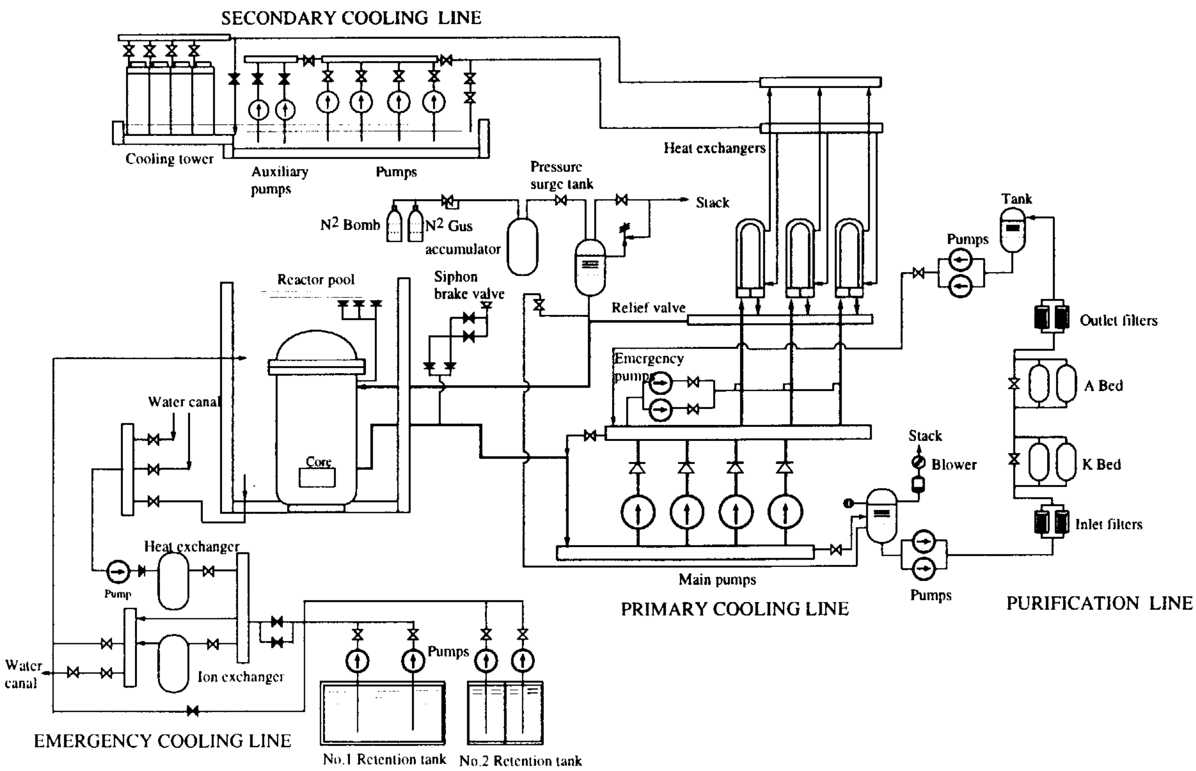
Fig. 2.1.4
Fuel element
(standard element)



Cooling systems

Flow diagram of the reactor cooling system is shown in Fig.2.1.5. The primary cooling system consists of four main pumps, two emergency pumps, an emergency cooling system and three heat exchangers. Three main pumps and an emergency pump operate during the normal operation. In case of abnormal reactor conditions such as LOCA or loss of primary coolant flow, one main pump and an emergency pump are powered by a diesel engine generator for decay heat removal. The primary coolant flows downward through the reactor core at the flow rate of 6,000 m^3/h. Nominal coolant inlet temperature is 49°C and the corresponding outlet temperature is 56°C. The heat removed from the core is transferred to the secondary coolant in the heat exchangers and is dissipated into the atmosphere in the cooling tower. (Table 2.1.1)

Fig. 2.1.5
Flow diagram of the reactor cooling systems



Instrumentation and control system

The nuclear instrumentation of JMTR is composed of start-up channel, log-power channel and linear-power channel. Each channel is composed of identical and independent three channels. Mean value of the three channels is used to measure nuclear power level and period. The trip signals are taken through the 2-out-of-3 circuits.

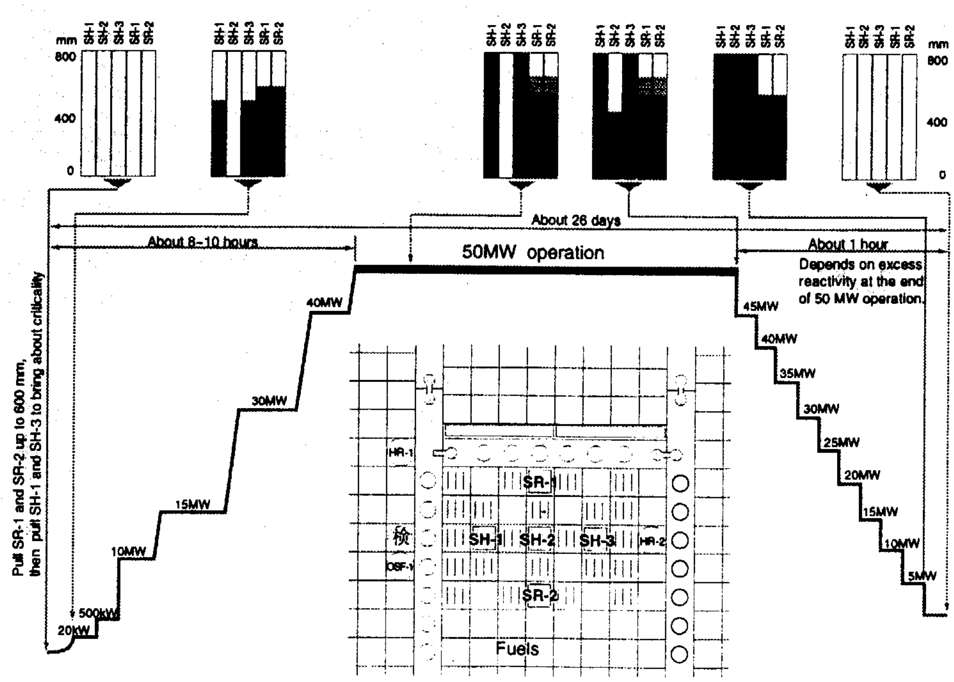
There are five control rods among which three rods are used for shim control and the other two rods are used for regulating. One of the regulating control rods is used for automatic control of the reactor power.

Each rod consists of a box-type hafnium absorber and a fuel follower attached to the bottom of the absorber. The drive mechanism is installed beneath the bottom of the pressure vessel. The control rods move vertically along the guide tubes. When a control rod is withdrawn upwards, the fuel follower displaces the hafnium absorber of the control rod in the core.

Operation procedure

Typical operation procedure of the reactor is shown in Fig.2.1.6. It takes about 8 hours from the start-up to the full power of 50 MW. The reactor power is increased stepwise up to 50 MW with safety checks of reactor and irradiation experiment facilities.

Fig. 2.1.6
Operation pattern and position of control rods during operation



Operation support system: ARGUS

Two computer systems are installed, one for the control of reactor operation and the other for the control of the irradiation facilities. ARGUS is the computer system for the reactor operation and has the following functions:

(1) monitoring the state of the feed back control circuits,

(2) logging the major process data and indicating trend,

(3) checking the data over/under limited value,

(4) displaying the result of monitoring sequence control under abnormal conditions,

(5) diagnosing unusual status and indicating the result of diagnosis,

(6) indicating operation guidance.

ARGUS is also used for the following calculations.

(1) neutron fluence and irradiation deformation of core components,

(2) estimation of the primary coolant leakage,

(3) possible restart time after the scram

The schematic diagram of ARGUS is shown in Fig. 2.1.7.

The support system for the control of irradiation facilities (LOOCAS, IDASS) is described in Chapter 2.2.

Radiation monitoring system

Radiation monitoring in JMTR is carried out continuously by the central radiation monitoring system as shown in Fig. 2.1.8. The central radiation monitoring system and radiation monitoring panel are located in the reactor control room and central radiation monitoring system is located at the radiation management room in the JMTR.

Fig. 2.1.7
Operation support system: ARGUS

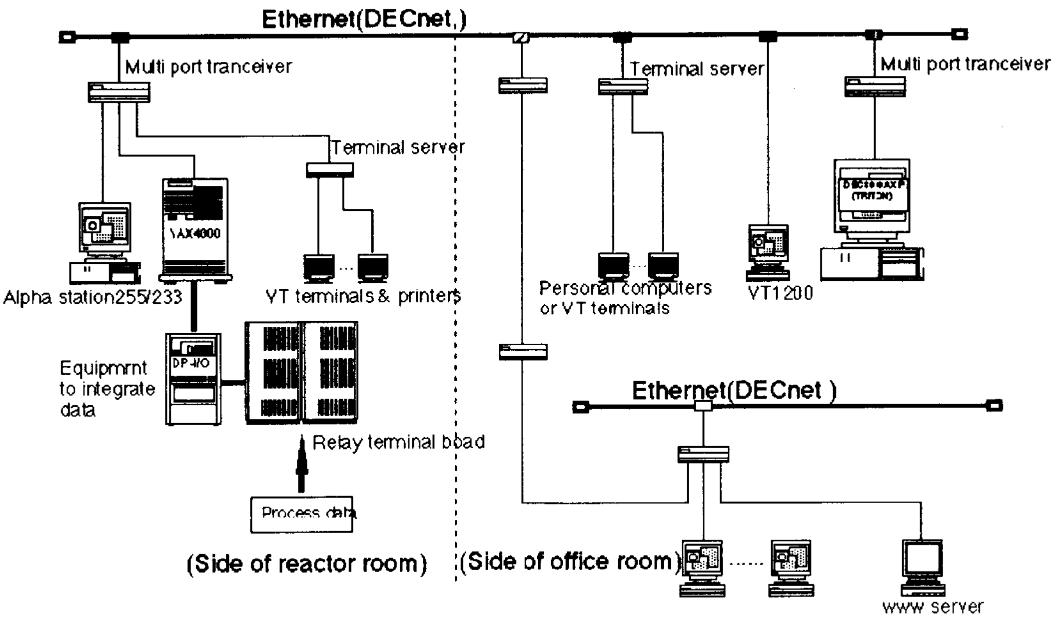
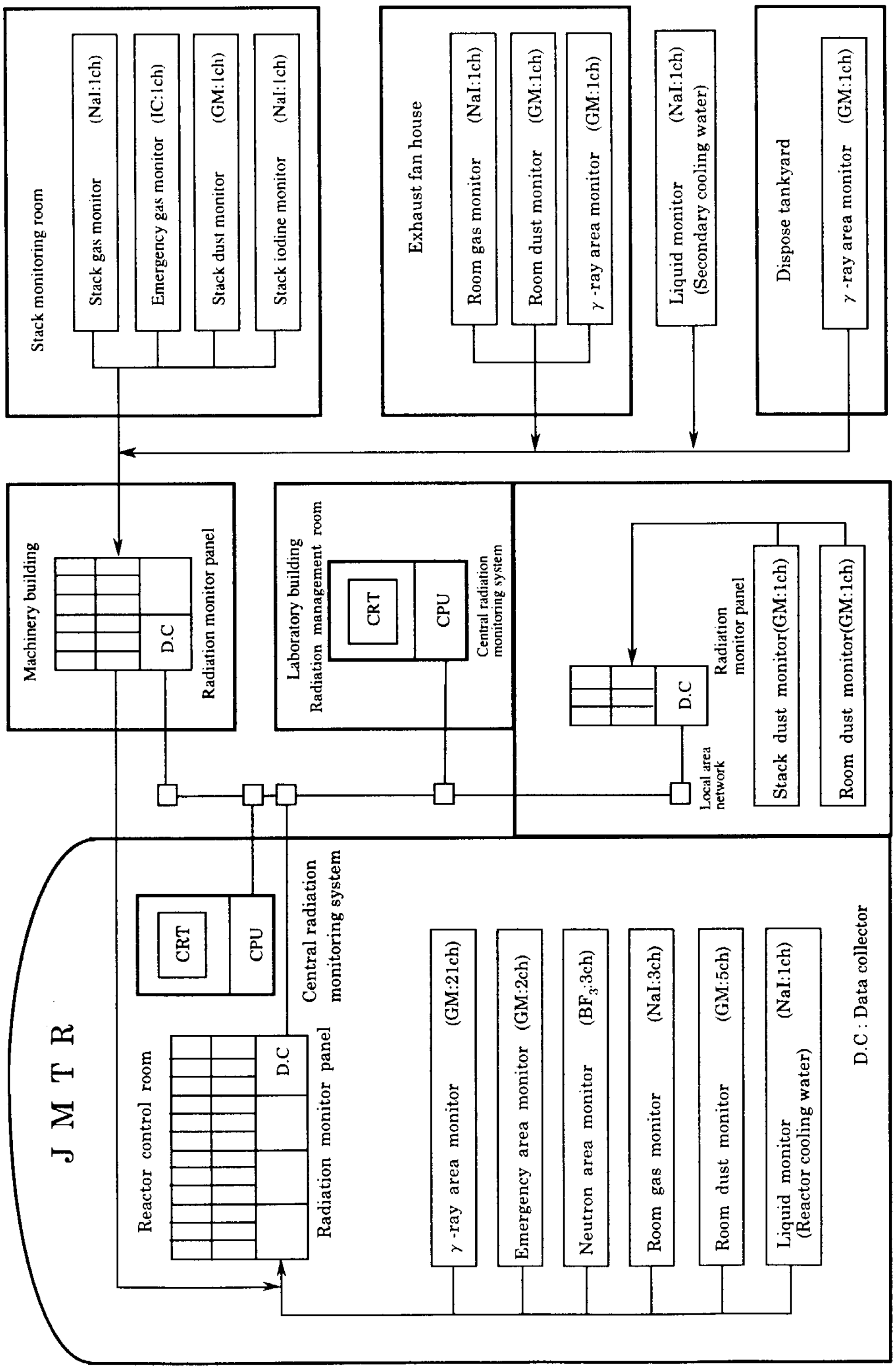


Fig.2.1.8
Central radiation monitoring system in the JMTR



2.2. Irradiation Facilities

JMTR is providing a variety of irradiation facilities in order to meet the requirements of irradiation tests for research and development of nuclear fuels and materials, and radioisotope production.

The capsules are inserted into suitable irradiation holes in the reactor core according to required irradiation conditions. The two Hydraulic Rabbit irradiation facilities (HR-1 and HR-2) and the power ramping test facilities are installed in fixed positions in the reactor core.

Capsule irradiation facilities

The capsule irradiation facilities are used mainly for the irradiation tests of nuclear fuels and materials as well as for radioisotope production. The capsules are classified into three types;

(1) non-instrumented capsule,

(2) instrumented capsule, and

(3) advanced type capsule.

The capsules are cooled by the reactor primary coolant. As a standard configulation of the core, up to 60 capsules can be loaded a cycle, of which 20 capsules are the instrumented capsules. The characteristics of irradiation field for the capsules are shown in Table 2.2.1. The types of the capsules are listed in Table 2.2.2 and schematically shown in Fig.2.2.1.

Fig. 2.2.1
Capsule structure

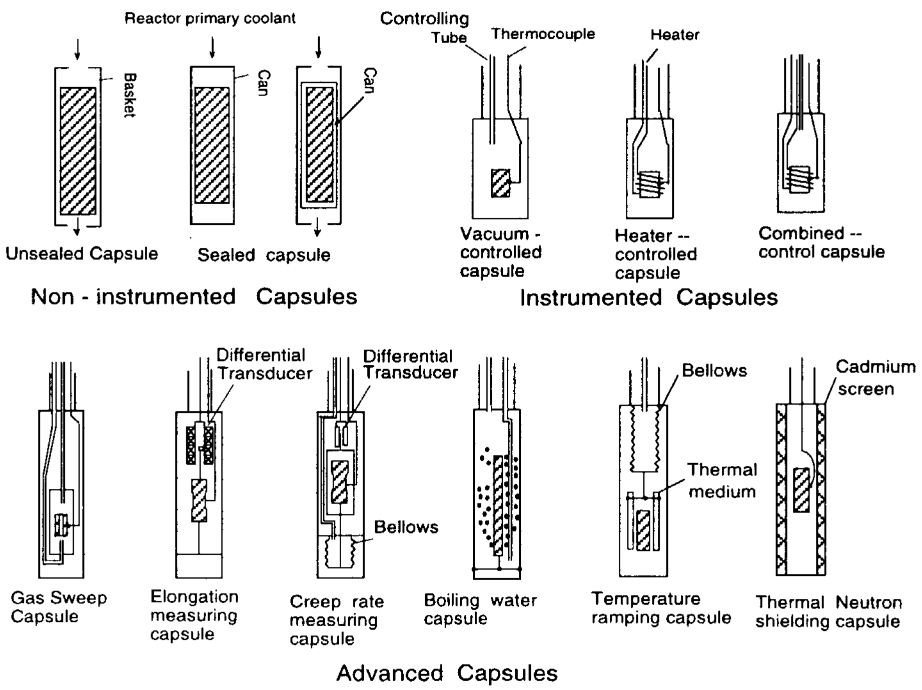


Table 2.2.1
Characteristics of irradiation feild for capsules

Thermal neutron flux	$1.0 \times 10^{13} \sim 3.0 \times 10^{14}$ n/(cm^2·s)
Fast neutron flux (>1MeV)	$1.0 \times 10^{13} \sim 2.0 \times 10^{14}$ n/(cm^2·s)
γ heating rate	0.5 ~ 10 W/g
Heat generation of heater	Max. 100 kW

Table 2.2.2
Types of capsules

Type of capsule	Structure and function
Non-instrumented Capsule	
Unsealed capsule	Test specimens are held in an open-type basket, and are directly cooled by the reactor primary coolant.
Sealed capsule	Test specimens are contained in a sealed vessel which is cooled by the reactor primary coolant.
Instrumented Capsule	
Non-control capsule	Temperature of test specimen is not controlled. The temperature and/or neutron fluence around the specimens are measured by thermocouples and self-powered neutron detectors (SPND) or fluence monitors (FM) .
Vacuum-control capsule	Temperature of test specimen is controlled by changing thermal conductivity of depends on helium gas layer between inner and outer vessels by regulating gas pressure.
Heater-control capsule	Temperature of test specimens is controlled by electric heaters wound around the specimens.
Combined-control capsule	Temperature of test specimens is controlled by combination of the vacuum-control and the heater-control.
Advanced type capsule	
Fission gas sweep capsule	This capsule is designed for measuring activity of FP gas released from coated particle fuels for HTTR fuel. FP gas is swept out to the sweep gas measuring equipment by carrier gas.
Elongation measuring capsule	Elongation of the specimen is measured by a differential transducer or a helium gas micrometer.
Creep rate measuring capsule	Test specimens are stressed by a pressurized bellows and its creep rate is measured by a differential transducer.
Boiling water capsule	Surface temperature of specimens is controlled by the pressure of boiling water which surrounds the test specimens.
Temperature ramping capsule	The temperature of test specimens is rapidly increased by removing a solid thermal medium which has high thermal conductivity and surrounds the test specimens.
Thermal neutron shielding capsule.	Test specimens are surrounded by thermal neutron absorber such as cadmium or hafnium for cutting thermal neutrons.

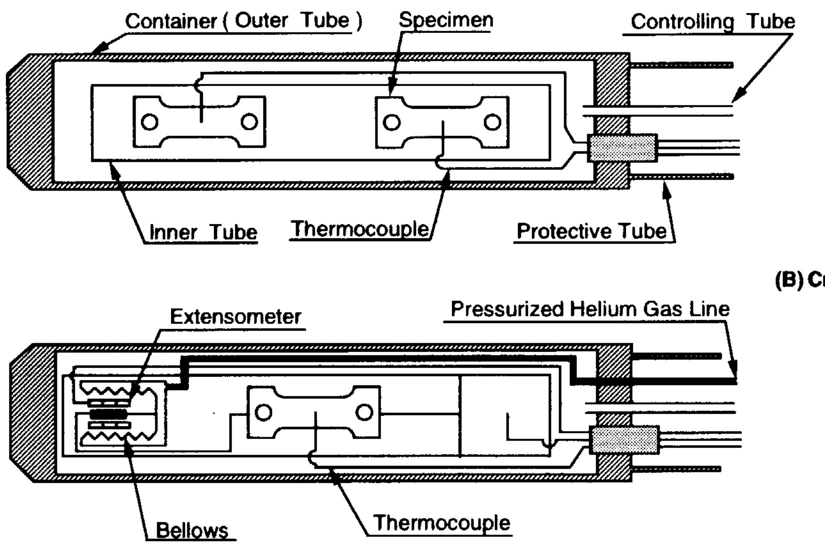
(1) Non-instrumented capsules

In an unsealed capsule, the reactor primary coolant flows between specimens and the outer tube, and directly cools the specimens. In a sealed capsule, specimens are held in an aluminum or stainless steel can, and cooled indirectly.

(2) Instrumented capsules

Using instrumented capsule, temperature of specimens and pressure can be controlled and measured on line. The instrumented capsule irradiation facilities consists of a capsule, a protective tube, a guide tube, a connection box and a control panel. The structure of a typical instrumented capsule is shown in Fig.2.2.2, and a typical configuration of the control system is shown in Fig.2.2.3. The capsule consists of a container (outer tube) which is made of stainless steel, an inner tube for holding specimens, a protective tube and a guide tube including the vacuum control tubes and the instrument cables. The guide tube is joined to the connection box around the reactor pool. The vacuum control tubes and the instrument cables are joined to the control panel through the connection box.

Fig. 2.2.2
Vacuum-controlled capsule and creep rate capsule



Currently, three methods are available to control temperature of specimen during irradiation.

1) Vacuum-control method

Thermal conductivity of the helium gas in the annular space between inner and outer tubes is altered by changing pressure in order to control temperature. Range of the helium gas pressure is normally between 1.33 x 10^2 Pa and 1.37 x 10^5 Pa.

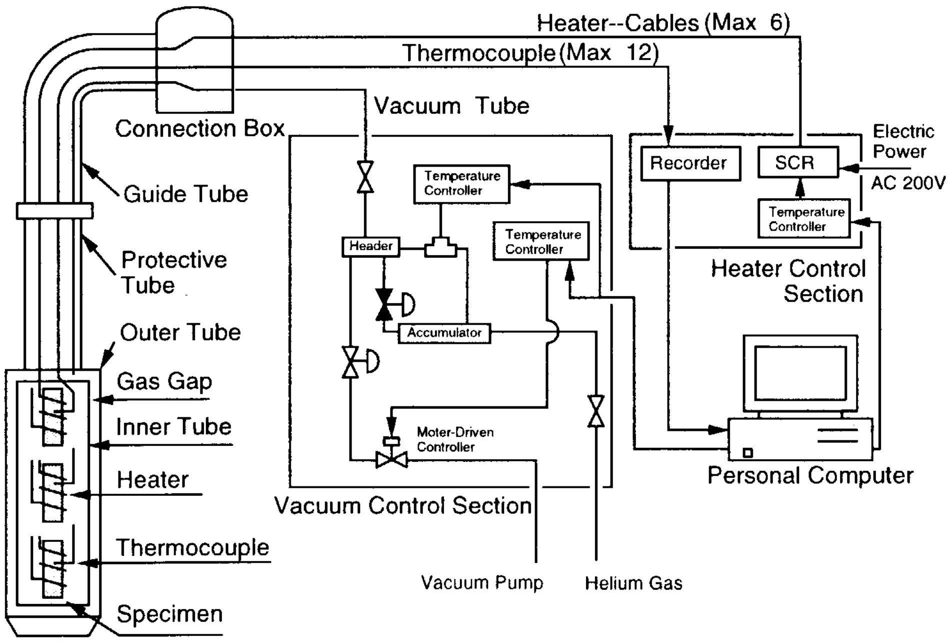
2) Heater-control method

The temperature of specimens is controlled by electric heaters around specimens.

3) Combined-control method

The temperature of the specimen is controlled using combination of the two methods, vacuum control and temperate control, mentioned above. This method provides a fine control of the temperature of the specimen.

Fig. 2.2.3
Capsule control system



(3) Advanced type capsules

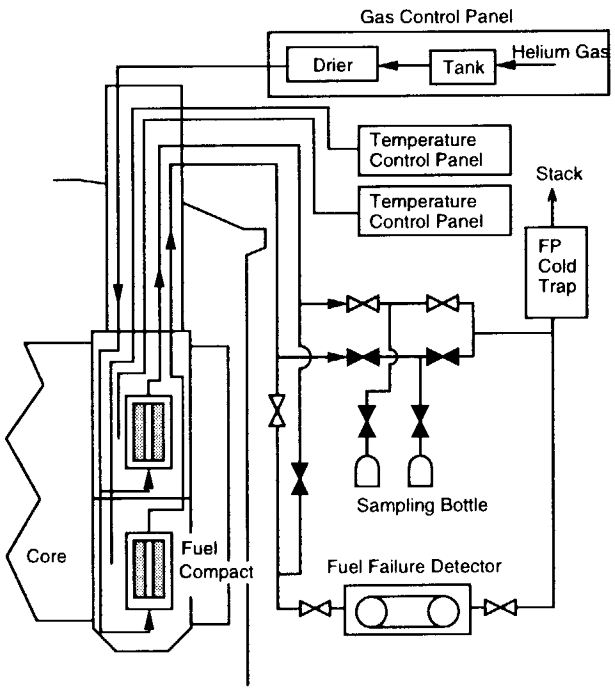
Various advanced type capsules have been developed to accomplish required irradiation conditions (see Fig.2.2.1 and Table 2.2.2).

As a typical example of advanced type capsule, the specification and the system configuration of the Fission Gas Sweep (FGS) capsule are shown in Table 2.2.3 and Fig.2.2.4.

Table 2.2.3 Characteristics of fission gas sweep capsule

Type of fuel	Coated particle fuel
Diameter of coated particle fuel	920×10^{-6} m
Total amount of ^{235}U	Max. 6 g
Heat generation of the heater	Max. 41.2 kW
Temperature of the specimen	Max. 1600 °C
Flow rate of helium gas	1 ℓ/min
Pressure of helium gas	0.3 MPa

Fig. 2.2.4 Configuration of fission gas sweep capsule



Hydraulic rabbit irradiation facilities

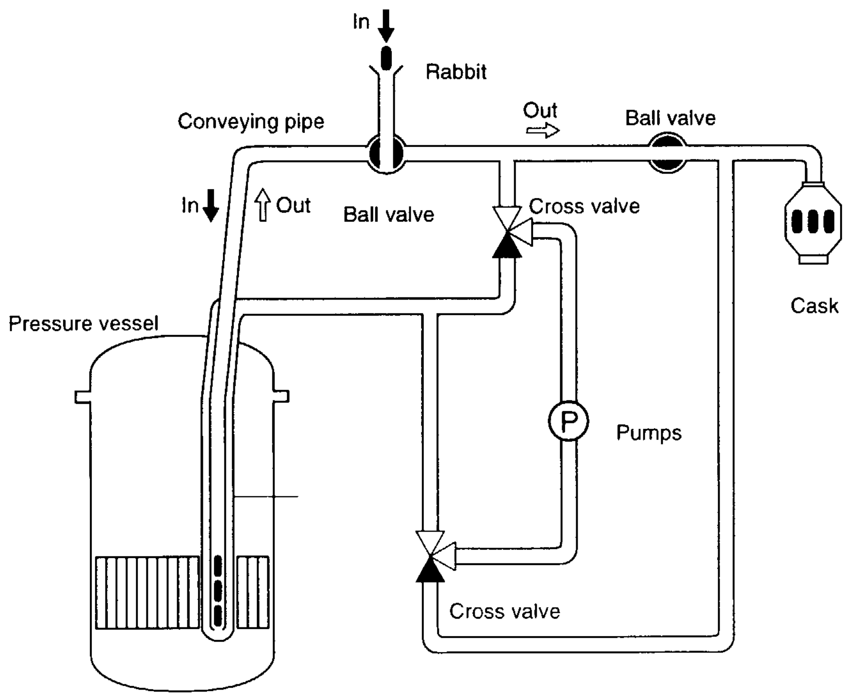
Two hydraulic rabbit irradiation facilities (HR-1 and HR-2) are installed for short term irradiations. The major specifications are shown in Table 2.2.4. The conceptual diagram of hydraulic rabbit is shown in Fig.2.2.5.

Each facility consists of an in-pile tube, two pumps, a loading station and an unloading station. The specimen is put in a metallic holder called rabbit. The rabbit is transferred into the reactor core and withdrawn from the core by water flow, during the reactor operation by changing the direction of the cooling flow in the facilities.

Table 2.2.4
Major specifications of two hydraulic rabbit irradiation facilities

	HR-1	HR-2
Location in the core	D - 5	M - 11
Thermal neutron flux	1.1×10^{14}n/(cm^2·s)	1.3×10^{14}n/(cm^2·s)
Fast neutron flux (>1MeV)	8.8×10^{12}n/(cm^2·s)	2.1×10^{13}n/(cm^2·s)
γ heat rate	1.1 W/g	2.2 W/g
Coolant	Light Water	Light Water
Temperature of the coolant	Max. 50 °C	Max. 50 °C
Pressure of the coolant	2.0 M Pa	2.0 M Pa
Flow rate of the coolant	Max. 11.0 m^3/h	Max. 8.4 m^3/h
Dimension of the holder	φ32 x 150 mm	φ32 x 150 mm
Specimen dimension	Max. φ26 x 120 mm	Max.φ 26 x 120 mm
Number of charged rabbit	Max. 3	Max. 3
Time of irradiation	Min. 1 min	Min. 1 min

Fig. 2.2.5
Conceptual diagram of hydraulic rabbit irradiation facility



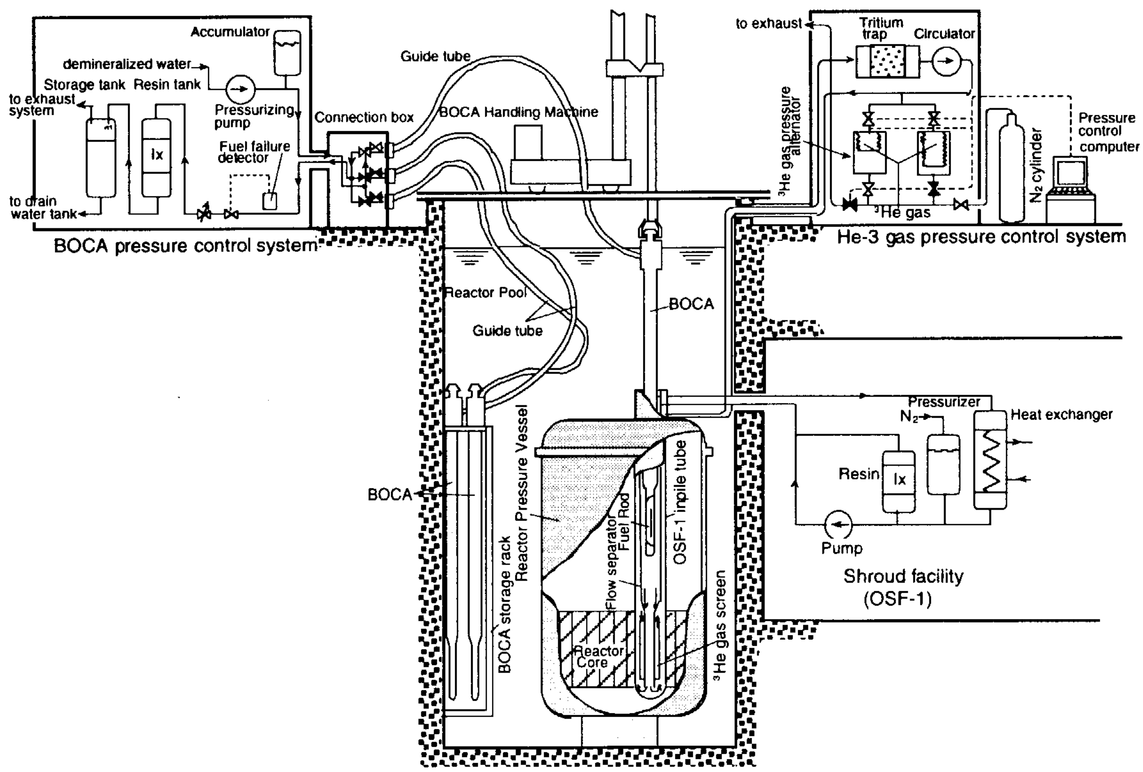
Power ramping test facility

The power ramping test facility consists of the Boiling Water Capsule (BOCA), the Oarai Shroud Facility-1 (OSF-1), BOCA pressure control system, a He-3 gas screen pressure control system and a BOCA handling machine. Arrangement of these components are shown in Fig.2.2.6. Major specifications are shown in Table 2.2.5.

Table 2.2.5
Major specifications of the power ramping test facility

BOCA	Coolant	Light water
	Press. of the coolant	7.3 M Pa
	Flow rate of the coolant	1.0 cm^3/s
	Linear heat rate	Max. 60 kW/m
	Thermal neutron flux	Max. 2.6 $\times 10^{14}$ n/($cm^2 \cdot s$)
	Fast neutron flux (>1MeV)	Max. 2.2 $\times 10^{13}$ n/($cm^2 \cdot s$)
OSF-1	Location in the core	D-9
	Coolant	Light water
	Pressure of the coolant	0.4 M Pa
	Flow rate of the coolant	1.9 m^3/h
	Temp. of the coolant	Max. 90 °C
	γ heating rate	Max. 2.5 W/g

Fig. 2.2.6
Power ramping test facility



BOCA is used to contain a fuel rod test element in a coolant condition simulating core channel of the boiling water reactor (BWR). Main structure of BOCA is cylindrical tube of stainless steel with 800 cm long and 6.9 cm in maximum outer diameter. Its inside configuration is explained in Fig.2.2.7.

Each BOCA is connected to the pressure control system through connection box. Cooling water pressurized at 7.3 MPa is injected from the control system. It flows around the fuel rod test element in the lowest part of the capsule and is boiled due to heat generated in the fuel during irradiation. Four BOCAs can be connected to the connection box in maximum, and one capsule of these is inserted at a time into the in-pile tube of OSF-1 to perform power ramping test.

The in-pile tube of OSF-1 is penetrating into the JMTR core through the top head of the reactor. OSF-1 has a penetration at the elevation near reactor top head, through which a BOCA is inserted into the in-pile tube of OSF-1. Inserted BOCA can be exchanged even in the reactor operation using BOCA handling machine.

OSF-1 has an independent cooling circuit to remove heat generated in BOCA. In pile tube has He-3 gas screen, an aluminum annular cylinder containing He-3 gas. Its pressure can be changed in order to control screening effect on thermal neutron, therefore to control fission rate in the fuel rod installed in BOCA.

Typical transients of the He-3 gas pressure, linear heat rate and surface temperature of fuel rod test element during power ramping test are shown in Fig 2.2.8. Heat generation rate is estimated by calorimetric method based on measured inlet/outlet temperature difference and flow rate of the in-core tube of OSF-1.

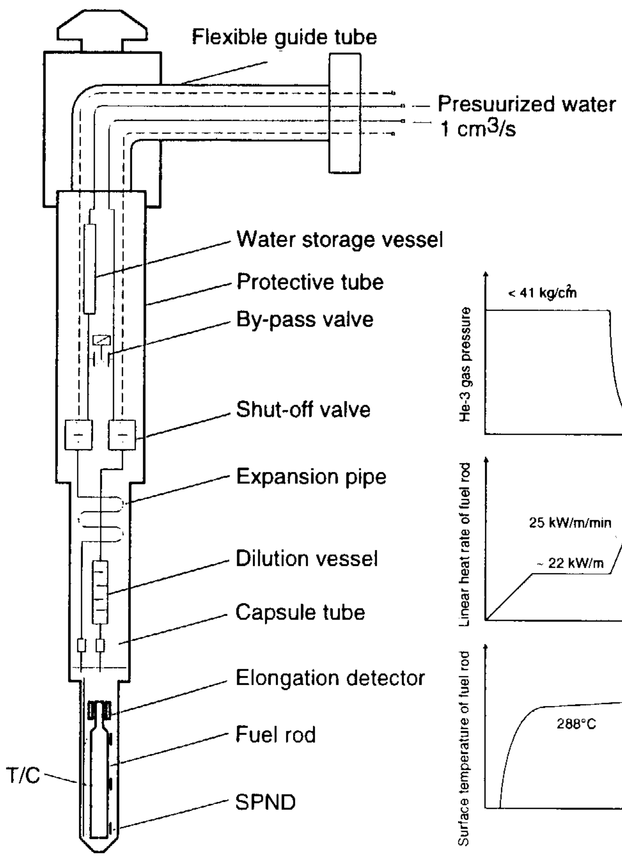


Fig. 2.2.7 (left) Boiling water capsule (BOCA)

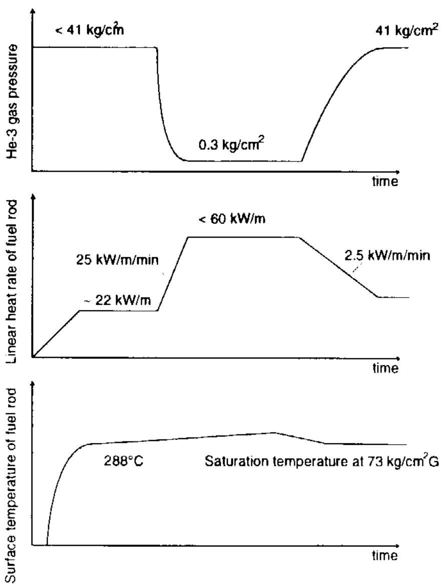


Fig. 2.2.8(Lower right) Change of linear heat rate and surface temperature for fuel rod by He-3 gas pressure change

Data acquisition system : IOSS

IOSS (Irradiation Facility Operation Support System) is a computer system for monitoring operating condition and collection of the irradiation test data of all irradiation facilities. The configuration of this system is shown in Fig.2.2.9.

IOSS is capable to collect and record the data from 1024 measurement points in the capsules and other facilities every 2 seconds. Collected data are used to prepare irradiation reports for the users.

Removed irradiation facilities

There were several irradiation facilities in JMTR other than the facilities mentioned above. OWL-1 was an in-pile water loop for testing fuels and structural materials of BWR and of PWR. The OWL-2 was other in-pile water loop for the irradiation tests of fuels and materials for PWR, BWR and advanced thermal reactor (ATR). These irradiation facilities simulate PWR, BWR or ATR condition.

NCF was constructed for controlling the total neutron flux or the temperature of the specimen by moving the capsule vertically.

OGL-1 was in-pile gas loop, and built for the irradiation tests to develop the HTTR fuels and materials.

These irradiation facilities were already removed after competition of their original purposes.

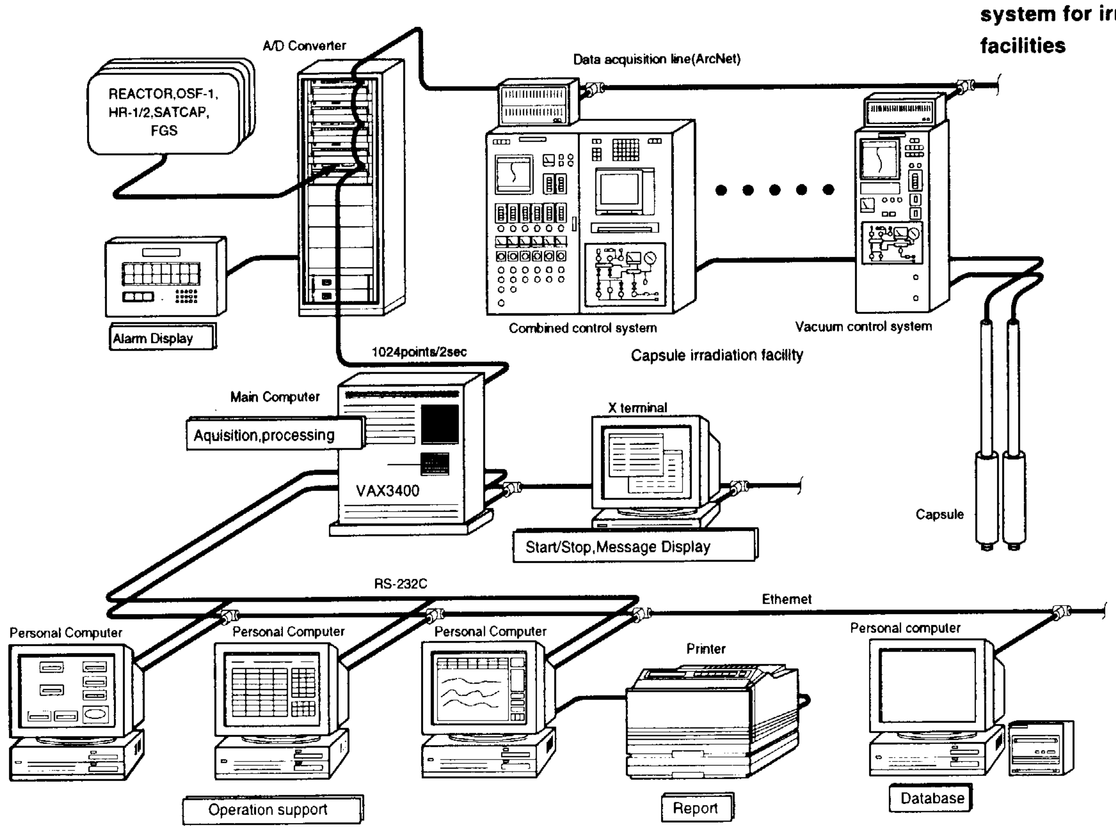


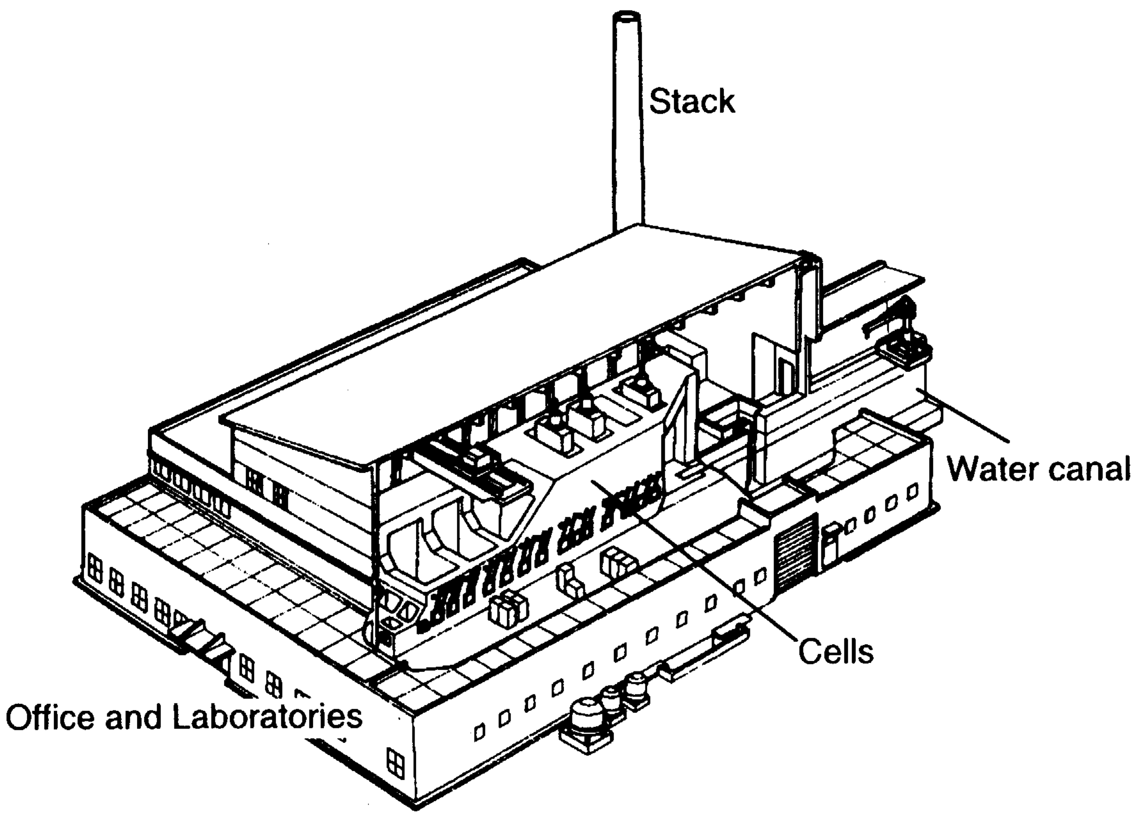
Fig. 2.2.9 Data acquisition system for irradiation facilities

2.3. Hot Laboratory

Specimens irradiated in JMTR are usually transferred to the Hot Laboratory (see Fig. 2.3.1) for post irradiation examination (PIE). The Hot Laboratory is connected to the reactor through a water canal.

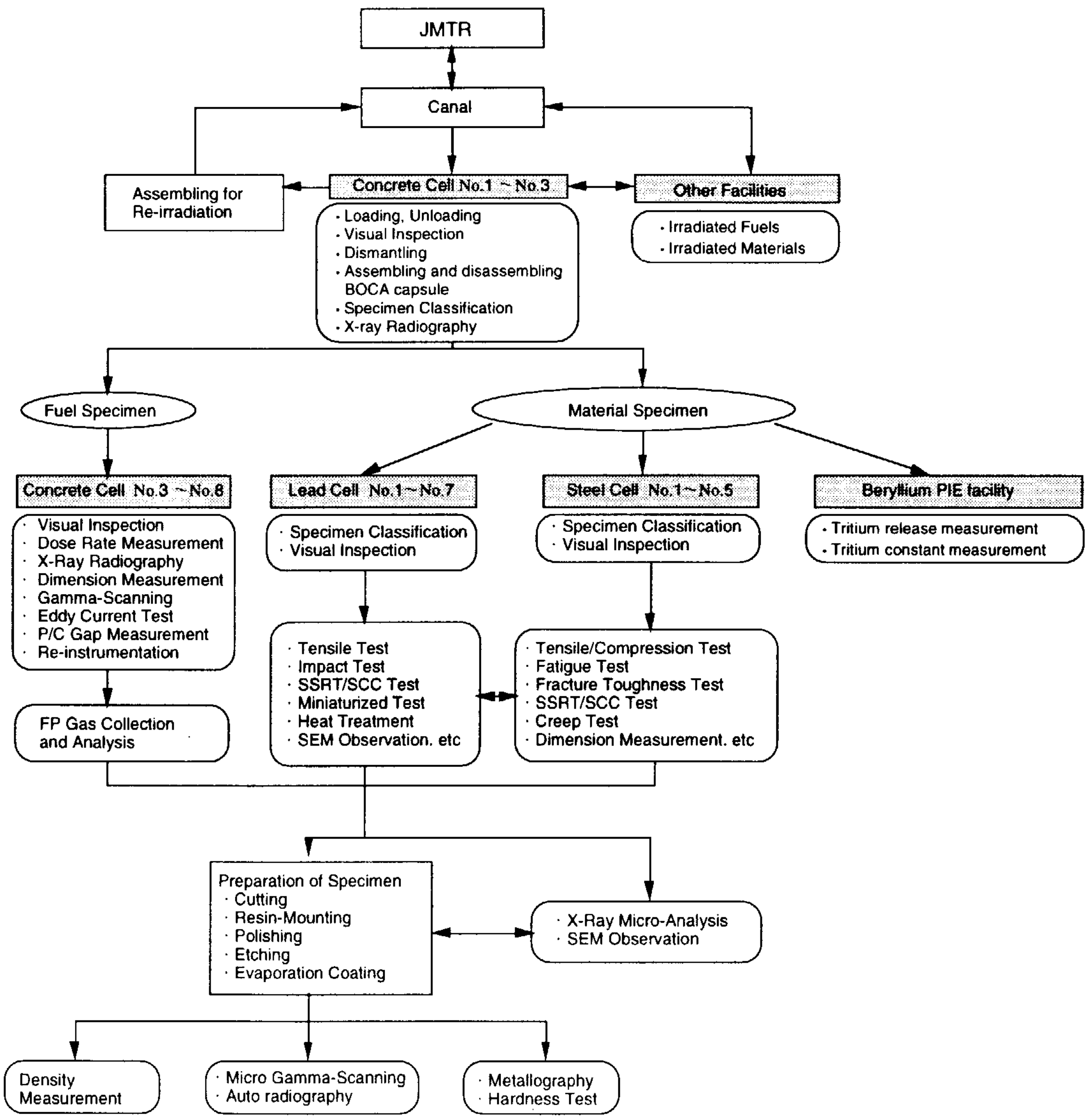
Construction of the hot laboratory started in 1967 and its service started in 1971. Additional steel cells were equipped in the hot laboratory in 1982.

Fig. 2.3.1
Hot laboratory building



The hot laboratory provides the data of post irradiation examinations (PIE) on the fuels and materials irradiated in the JMTR and/or other reactors. The concrete cells, the lead cells and the steel cells are located in the hot laboratory. After preparatory works for the post irradiation examination in the C-1 through C-3 concrete cells, nondestructive examinations and destructive examinations on irradiated fuel specimens are carried out in other concrete cells. Mechanical property test on irradiated material specimens are carried out in the lead cells and/or the steel cells. The flow diagram of the post irradiation examinations in the hot laboratory is shown in Fig. 2.3.2.

Fig. 2.3.2
Flow diagram of post irradiation examination



Arrangement of hot laboratory

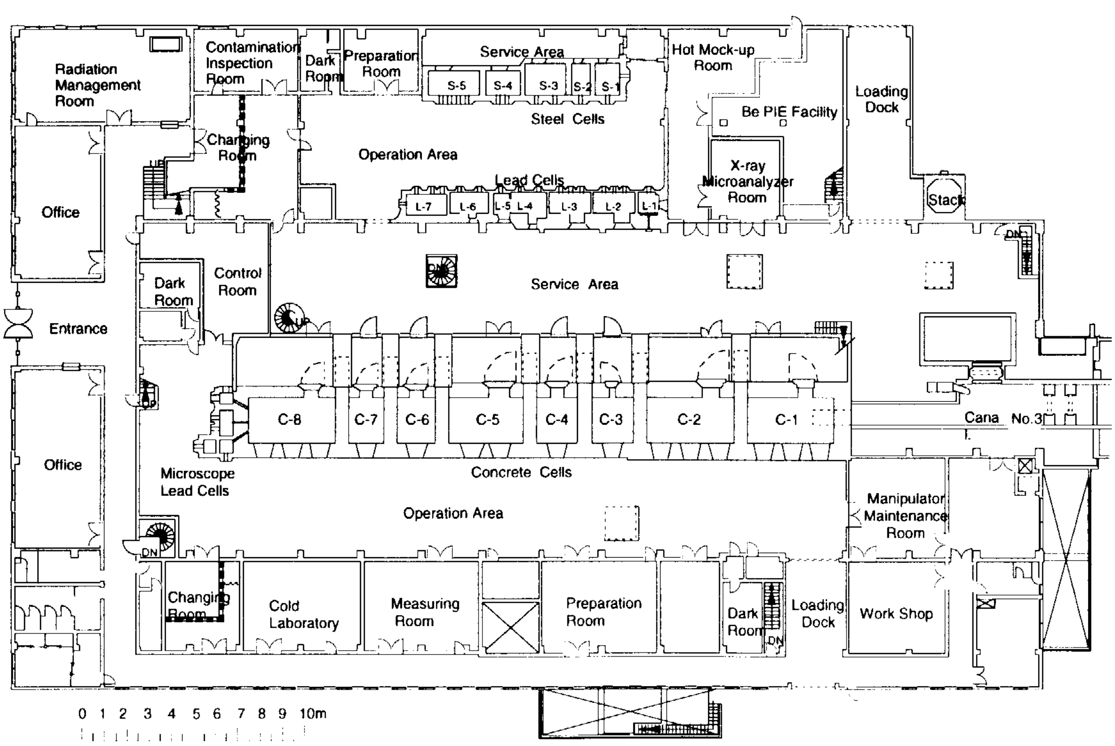
The hot laboratory is connected to JMTR through the water canal which is used to transfer irradiated capsules from JMTR and to JMTR for re-irradiation. The building houses the concrete cells with the microscope lead cells, the lead cells and the steel cells. Each cell is designed as so called β–γ cell. Figure 2.3.3 shows the arrangement in the ground floor of the hot laboratory. Auxiliary facilities such as the ventilation system, the power supply system and the liquid waste disposal system are located on the basement.

Concrete cells and microscope lead cell

Using 8 concrete cells and 4 microscope lead cells, following works can be carried out:

(1) loading and unloading capsules
(2) disassembling irradiated capsules
(3) re-capsuling
(4) re-instrumentations of specimens
(5) nondestructive examinations of fuel specimens
(6) destructive examinations of fuel specimens

Fig. 2.3.3
Hot Laboratory



The maximum length of the fuel specimens to be handled in the hot laboratory is about 1 m. The irradiated capsules in JMTR are transferred into the C-1 cell from the canal after cooled. The capsules irradiated in the other reactors are transported by the transport casks and loaded into the C-1 cell through the posting port installed on the ceiling. The re-instrumentations on the fuel specimens are carried out in the C-7 cell, then re-instrumented fuel specimens are installed into the Boiling Water Capsule (BOCA) in the C-1 cell.

Among the PIEs performed in the concrete cells, NDEs (nondestructive examinations) are visual inspection, X-ray radiography, dimensional measurement, gamma scanning, eddy current test and pellet/ cladding (P/C) gap measurement. As DEs (destructive examinations), fission gas collection and analysis, density measurement of fuel specimens and sample preparation for metallography, are performed.

Metallography, hardness measurements, autoradiography and micro-gamma scanning for fuels and materials are carried out in the microscope lead cells attached to the concrete cells. Preparatory works for composition analysis and fractography with the X-ray Micro-Analyzer (XMA) installed in another room are carried out in the concrete cells.

The wall of the concrete cells is constructed with the high density magnetite concrete and it's thickness is 1 m or 1.1 m. The shielding windows (dry type) with equivalent shielding thickness to the wall are installed in the wall. In the microscope lead cells, lead bricks with thickness of 17.8 cm and shielding windows were built into the wall. Table 2.3.1 shows the specifications of the concrete cells and the microscope lead cells. Figure 2.3.4 shows the major apparatus and functions of the concrete cells and the microscope

Table 2.3.1
Specification of concrete cells and microscope lead cells

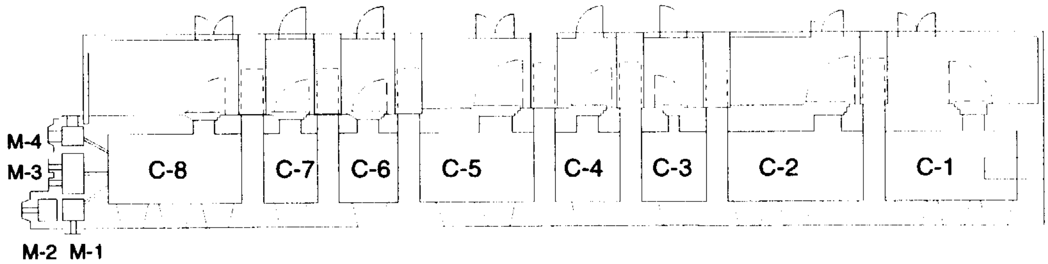
(A) Concrete cells

Cell No.	Inside dimension W x D x H	Shielding wall thickness	Shielding wall density	Number of windows	Maximum activity (1MeV)
C-1	6 x 3 x 5.5 m	1.1 m	>3.8	3	33 pBq
C-2	6 x 3 x 5.5 m	1.1 m	>3.8	3	33 pBq
C-3	3 x 3 x 5.5 m	1.0 m	>3.8	1	3.7 pBq
C-4	3 x 3 x 4.5 m	1.0 m	>3.8	1	1.1 pBq
C-5	5 x 3 x 4.5 m	1.0 m	>3.8	2	1.1 pBq
C-6	2.5 x 3 x 4.5 m	1.0 m	>3.1	1	85 TBq
C-7	2.5 x 3 x 4.5 m	1.0 m	>3.1	1	85 TBq
C-8	6 x 3 x 5.5 m	1.0 m	>3.1	3	85 TBq

(B) Microscope lead cells

Cell No.	Inside dimension W x D x H	Shielding wall thickness	Number of windows	Maximum activity (1MeV)
M-1	1 x 1 x 1.1 m	17.8 cm	1	3.7 TBq
M-2	1 x 0.85 x 1.1 m	17.8 cm	1	3.7 TBq
M-3	1.59 x 1 x 1.1 m	17.8 cm	2	3.7 TBq
M-4	1.5 x 1 x 0.95 m	17.8 cm	2	3.7 TBq

Fig. 2.3.4
Functions and major apparatus in concrete cells



Cell no. Functions	Main Apparatus & specifications
C-1	
Lifting up irradiated capsules from water canal	Loading lift Capacity:500kg
Irradiated specimens loading and unloading	Loading cask Shielding wall Pb 150mm
BOCA capsule assembling and dismantling	End-plug tightener Torque:7kg・m
Visual inspection	Periscope Mag. ×10
Dose measurement	Dose measuring device 3mR～10,000R, 0.1mR/min,35kev～1.2Mev 0mR～1000R, 0.1mR/min,40kev ～3MeV
C-2	
Dismantling of capsule	Diamond cutter Size:Max. φ60, Blade width 1.0mm
Cutting of capsule	Guillotin cutter Size: φ120 ×5t Power:Max.200ton
Processing after reweld for irradiated test piece	Milling machine
C-3	
X-ray radiography	X-ray radiography system Cap:150kVp, 300kVp, Film size 140×990
Gamma scanning	Gamma scanning system Scanning speed:5,10,20,40mm/s, Collimator:0.2 ×20,1×15, φ0.75, φ1.5
C-4	
Eddy current test	Eddy current testing machine Size: φ4～ φ17, 100 ～1000mm Feed speed: 5～30mm/s
Pellet/Clad gap measurement	Gap measuring apparatus Size: φ6～ φ18, 50～1000mm Gap range:1mm ±5 μm
Dismantling of NaK capsule	NaK capsule dismantling machine Size: φ15～ φ50, 150 ～800mm Medium:Kerosen (Max. Nak 200cc)
C-5	
Dimensional measurement	Dimension measuring apparatus Diameter: φ5 ～ φ30 ±0.005mm Bowing : ±3.0mm ±0.02mm Length : 0 ～1000mm ±0.02mm
FP gas volume and pressure measurement	Puncturing device and gas collector Size: φ6.0 ～ φ17, 30～1000mm
FP gas analysis	Mass spectrometer Detection limit:Kr,Xe,H2,He,Ar,CH4,O2, O2,N2＋CO,CO2 0.01v%
Preparation for XMA sample	Vacuum evaporetor Size: φ30×30H Vacuum (3×10-4Pa)
(C-5)	
Leak test of fuel rod	Leak locator Size: φ50～ φ20, 100 ～800mm Medium:White spirit ρ=0.794 Size: φ9～ φ40
Welding for re-fabrication	Remote welding machine Size: φ9～ φ40 Welding method:Tig welding
Machining of end plug	End plug machining device Size: φ8～ φ20±0.05mm,100～800mm
Weight measurement	Mettler balance Cap:Max. 160g ±0.0001g Max.1200g ±0.01g
Density measurement	Densimeter Cap:Max. 150g ±0.01g/cc Medium:Metaxylene
Electromotive force measurement	Electromotive force measuring appratus Furnace size: φ40, 150mm Temp range:100～1000℃ in Inert gas
C-6	
Electoron beam heating facility	Beam power Max.50kW Beam current Max.1.7A
C-7	
Making center hole of UO2 pellets to insert thermocouple	Drilling machin Size: φ2.0, 54mm Frozen CO_2(-78℃) machining:-160℃
C-8	
Preparation for metallography	Micro cutter(Diamond cutter) Test piece: Max. φ30, Cutting width 1mm Grinding polisher:Abrasive paper #180 ～#100 Ultrasonic cleaner:100W, 28kHz Resin impregnating machine:Vacuum(10-2P) Periscope: Magnification ×4.5 ～×45
M-1	
Metallography	Magnification: x 50～900
M-2	
Metallography	Magnification: x 50～900
M-3	
Metallography	Magnification: x 5～10(zoom)
Hardness test	Load: 50, 100, 200, 500, 1,000 g Measuring mag. : x 400
M-4	
Micro-gamma scanning	Step scanning: 0.1 mm Collimater : φ0.2～0.5 Detector: Ge-NaI anticoincidence Ge 50cc, 2.5 keV

lead cells.

Master-slave manipulators, power manipulators and in-cell hoists are installed in the concrete cells. Ball socket manipulators are installed in the microscope lead cells.

An isolation room is attached to the each concrete cell to prevent splashing of radioactive materials from the concrete cell to the service area. The isolation room are lined with steel plates on the floor and the lower part of wall for easier decontamination. Each cell has a shielding access door. The access door is a hinge type and manually opened with mild steel shielding wall in the rear side. It is used as entrance for the personnel for decontamination and maintenance works.

Lead cells

The lead cells consist of 7 cells for post irradiation examinations on material specimens. The irradiated specimens to be examined are brought in/out through a wall type posting port installed in the rear wall of the L-1 cell. The L-1 cell is used for interim storage of specimens which require relatively long cooling interval after irradiation.

The post irradiation examinations carried out in the lead cells are tensile test, SCC (Stress Corrosion Cracking) test, instrumented impact test, visual inspection, dimensional measurement, and others.

The post irradiation examinations on miniaturized specimen used in the study of fusion reactor materials are carried out in the lead cells, including the development of testing techniques.

The front wall of the each cell is built up with the lead bricks with a thickness of 15 cm or 20 cm. It has shielding windows (dry type) which have an equivalent shielding

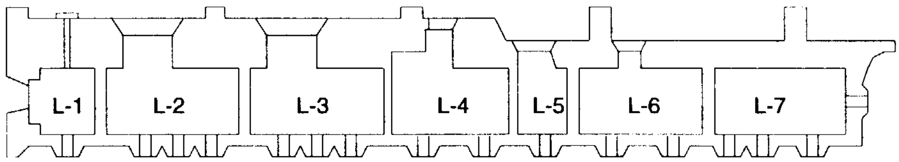
Table 2.3.2
Specification of lead cells

Cell No.	Inside dimension W x D x H	Shielding wall thickness	Number of windows	Maximum activity (1MeV)
L-1	1.5 x 1.3 x 1.95 m	20 cm	1	1.2 TBq
L-2	2.9 x 1.75 x 2.25 m	15 cm	3	37 GBq
L-3	2.9 x 1.75 x 2.25 m	15 cm	3	37 GBq
L-4	1.2 x 1.75 x 1.8 m	15 cm	2	37 GBq
	1.2 x 1.25 x 1.8 m			
L-5	1.2 x 1.25 x 1.8 m	15 cm	1	37 GBq
L-6	2.8 x 1.25 x 1.8 m	15 cm	2	37 GBq
L-7	2.75 x 1.25 x 1 m	15 cm	4	37 GBq

thickness to the wall. Table 2.3.2 shows the specifications of the lead cells. Figure 2.3.5 shows the major apparatus and functions of the lead cells. Each lead cell is provided with a pair of master-slave (M/S) manipulators and some of them are provided with ball socket manipulators.

The rear wall of the each cell is constructed with the ordinary concrete with necessary

Fig. 2.3.5
Functions and major apparatus of lead cells



Cell no.	Functions	Main Apparatus	Specifications	
L-1	Specimen identification	Periscope	Mag.:x10	
	Specimen storage	Storage pits	60 pits	
L-2	SSRT/SCC test	SSRT/SCC testing machine	Capacity: Test temp.: Atmospher:	30kN -320 Purified water, Inert gas
	Visual inspection	Periscope	Mag.:	x10
L-3	Tensile test	Tensile testing machine (Instron-type)	Capacity: Test temp.: Atmospher:	10kN Max.1500 Vacuum
	Charpy impact test	Instrumented Charpy impact testing machine	Capacity: Test temp.:	300J -120~200°C
	Visual inspection	Periscope	Mag.: x10	
L-4	Miniaturized specimen test	Small punch testing machine	Capacity: Test temp.: Atmospher:	5kN -160 ~750 °C Vacuum
	Heat treatment	Heat treatment fumace	Test piece volume:Max. ϕ30x100l Atmospher: Test temp.:	 Vacuum or Ar gas Max.1000°C
	Visual inspection	Periscope	Mag.:	x10
L-5	Miniaturized specimen machining	Electrical discharge machining device	Atmospher:	Oil
L-6	Miniaturized specimen test	High speed punch testing machine	Capacity: Test temp.: Atmospher:	1kN -150°C Vacuum
	Miniaturized specimen handlling	Micro-manipulator		
L-7	Fractography	Scanning electron microscope	Mag.: Resolving power:0.5~3kV	x15~200,000 5~30kV 40nm

shielding thickness. Each cell is provided with a shielding access door and a transfer port for the transportation of specimens to the next hot cells. The access door is hinge type and manually opened with mild steel shielding wall in the rear side. The doors are used for entrance and exit of personnel for decontamination and maintenance works of the cells.

Steel cells

The steel cells consist of 5 cells which are mainly used for testing the property of material specimens. Major items of the post irradiation examinations carried out in the cells are high temperature tensile/compression test, fracture toughness test, creep test, fatigue test, SCC test, visual inspection and dimensional measurement.

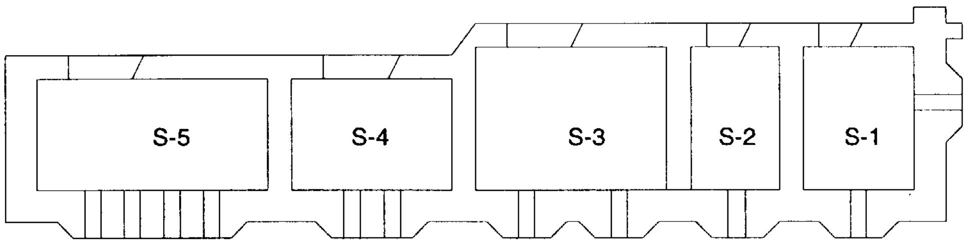
Cell walls of operation side are built with the mild steel with a thickness of 35 cm or 40 cm and the shielding windows (dry type) which have the equivalent shielding thickness to the wall. Table 2.3.3 shows the specifications of the steel cells. Figure 2.3.6 shows the apparatus and functions of the steel cells.

A pair of master-slave (M/S) manipulators are installed in each cells. Each cell has a shielding access door made of mild steel. The access doors are hinge type built in the rear side of the shielding wall and used for entrance and exit of personnel for decontamination and maintenance works in the cell. Transfer ports installed on the partition walls of the cells are used for transfer of specimens to the adjacent hot cells.

Table 2.3.3
Specification of steel cells

Cell No.	Inside dimension W x D x H	Shielding wall thickness	Number of windows	Maximum activity (1MeV)
S-1	2 x 1.7 x 2.4 m	35 cm	2	66 GBq
S-2	1.3 x 1.6 x 2.35 m	40 cm	1	5.9 TBq
S-3	3.2 x 1.7 x 2.4 m	35 cm	2	66 GBq
S-4	2.2 x 1.25 x 2.4 m	35 cm	2	66 GBq
S-5	4 x 1.25 x 2.4 m	35 cm	4	66 GBq

Fig. 2.3.6
Functions and major apparatus of steel cells



Cell no.	Functions	Main apparatus	Specifications	
S-1	Low cycle fatigue test	Fatigue testing machine	Capacity: Atmospher: Test temp.: Cyclic speed:	100kN Vacuum ~900°C Max 100Hz
	Dimensional measurement	Dimensional measuring device	Max. Length: Precision:	100mm +-10μ m
S-2	Specimen storage	Storage pits	42 pits	
	Visual inspection	TV Monitor		
S-3	Tensile/compression test	Tensile/compression testing machine	Capacity : Atmospher : Test temp.:	50kN (Instron type) Air or Ar gas -150~900°C
	Fracture toughness test	Fracture toughness testing machine	Capacity : Atmospher : Test temp.:	63kN Air -150~500°C
	Visual inspection	Periscope	Mag.:	X 10
S-4	SSRT/SCC test	SSRT/SCC testing machine	Capacity : Atmospher : Test temp.:	20kN Purified water or Ar gas ~320°C
	SCC test	UCL/SCC testing machine	Capacity : Atmospher : Test temp.:	5kN Purified water or Ar gas ~320°C
S-5	Creep rupture test	Creep rupture testing machines(3)	Capacity : Atmospher : Test temp.:	10kN Vacuum or Ar gas Max:1000°C
	Creep deformation test	Creep testing machine	Capacity : Atmospher : Test temp.:	5kN Vacuum or Ar gas Max:1000°C

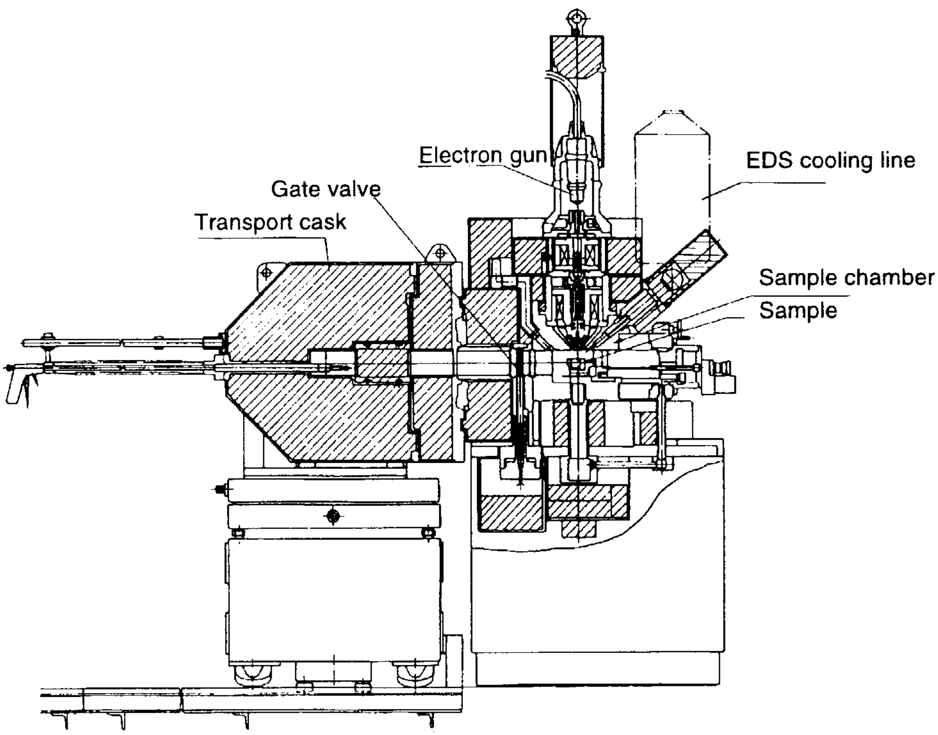
SSRT: Slow Strain Rate Tensile
SCC: Stress Corrosion Cracking
UCL: Uni-axial Constant Load

X-ray Micro-Analyzer

The X-ray micro-analyzer (XMA) consists of a sample chamber, an electron gun, an EDS cooling system, a gate valve and a transportation cask (Fig. 2.3.7). The sample chamber is surrounded with a tungsten shielding wall for radiation protection.

The XMA is used for an element analysis of micro sphere on specimens and observation of dispersion status of specified element and observation of surface status by secondary electron, backscattered electron and absorbed electron.

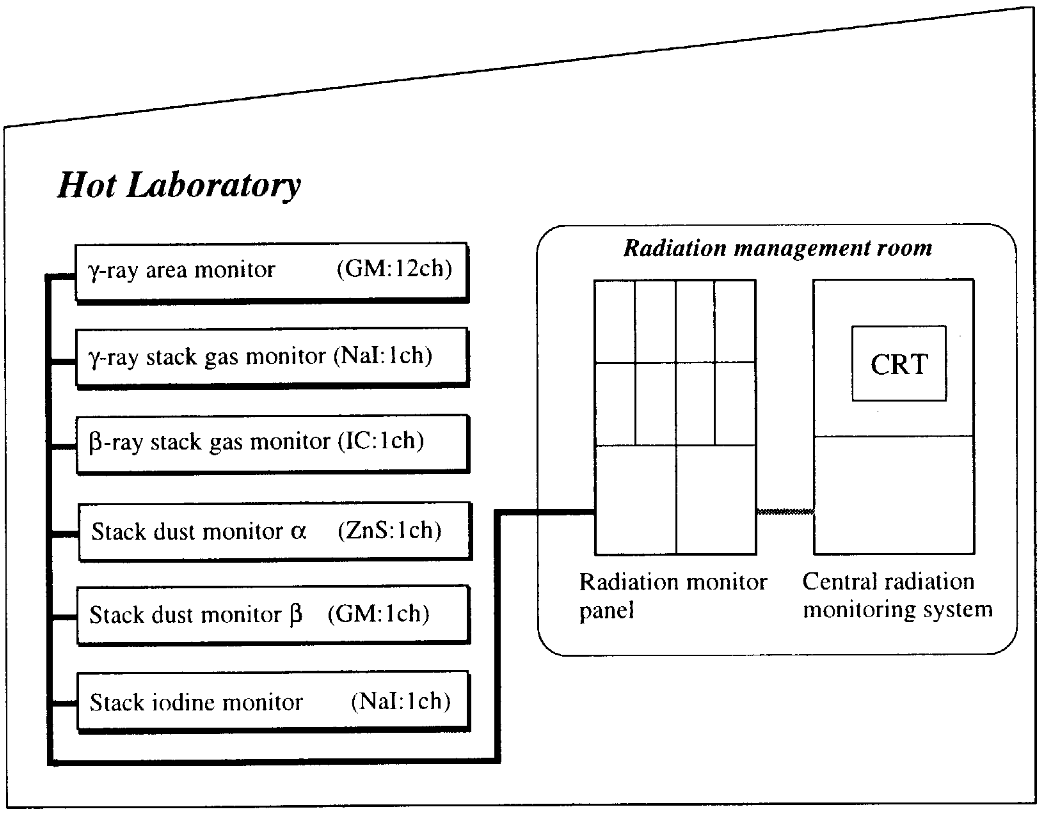
Fig. 2.3.7
X-ray micro-analyzer (XMA)



Radiation monitoring system

Radiation monitoring in the hot laboratory is carried out continuously by the central radiation monitoring system as shown in Fig. 2.3.8. The central radiation monitoring system and radiation monitoring panel is located at the radiation management room in the hot laboratory.

Fig. 2.3.8
Central radiation monitoring system in the hot laboratory



3. Activities in FY1996

3.1. Reactor Operation

JMTR has been operated with a Low Enriched Uranium (LEU) fuel core since its 108th operation cycle in 1994. For FY 1996, four operation cycles, 118th through 121st, were planned. However, minute cracks were found in the pressure surge tank during annual maintenance after reactor operation of two cycles. The tank was decided to be replaced with new one before the 120th operation cycle. According to this schedule change, 120th cycle was postponed in to next fiscal year. Operation summary of FY 1996 is shown in Table 3.1.1. The integrated reactor power of 118th and 119th operation cycles attained 112,905.3 MWD. The nuclear characteristic values of each operation cycle were measured at a low-power level of 20 kW prior to the high-power operation for irradiation. Values of excess reactivity, shutdown margin and the worth of the two regulating control rods (SR-1, SR-2) are listed in Table 3.1.2.

During reactor operation, chemical analysis of the reactor cooling water is regularly performed, which detects ^{24}Na, ^{27}Mg, some radioactive isotopes of iodine, etc. The concentration of the radioactive isotopes are shown in Fig.3.1.1 and Fig.3.1.2 for the interval from 113rd to 119th operation cycles. No significant change took place during these operation term. Detected radioactive iodines are considered to be produced by the fission of impure uranium included in the beryllium reflector elements. 3,925 m^3 of high-purity demineralized water was supplied to the primary cooling channel, the pool-canal and the cooling system of the irradiation facility during FY1996. 3,217,750 m^3 of filtered water was supplied to the secondary cooling channel and to some other utility systems.

Table.3.1.1
Operation summary of FY 1996

Operation cycle No.	Period*1	Integral power (MWd)	Operation time*2 (hours:minutes)	Operating efficiency (%)*3	Cumulative integral power (MWd)
118	96.5. 8～96.6. 2	1236.3	602:27	99.6	111667.7
119	96.6.22～96.7.17	1237.6	605:33	99.7	112905.3

Note *1 Period indicates the startup and shutdown dates for 50MW operation.
*2 Operation time excludes the time to get the nuclear properties of the reactor.
*3 Operation efficiency = Actual integral power / Scheduled integral power (1241MWd).

Table. 3.1.2
Nuclear characteristics of each operation cycle

Operation cycle No.	Excess reactivity (%Δ k/k)	Shutdown margin (%Δ k/k)	keff	Rod worth of automatic control rods SR-1 (%Δ k/k)	SR-2 (%Δ k/k)
118	11.87	34.3	0.75	0.23	0.19
119	12.10	29.7	0.77	0.24	0.17

Fig.3.1.1
Radioactive iodine concentration in primary coolant (From 113rd to 119th operation cycles of JMTR)

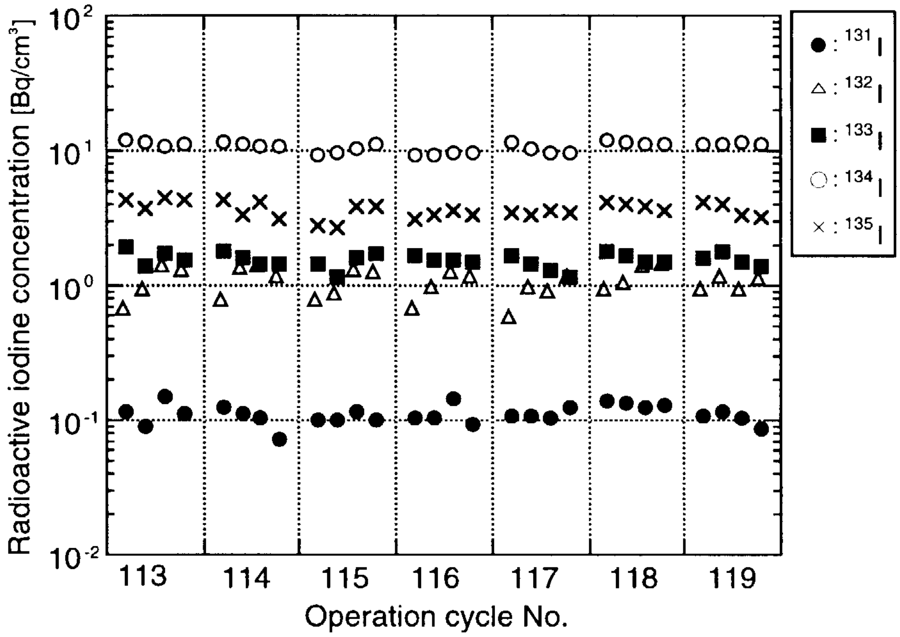
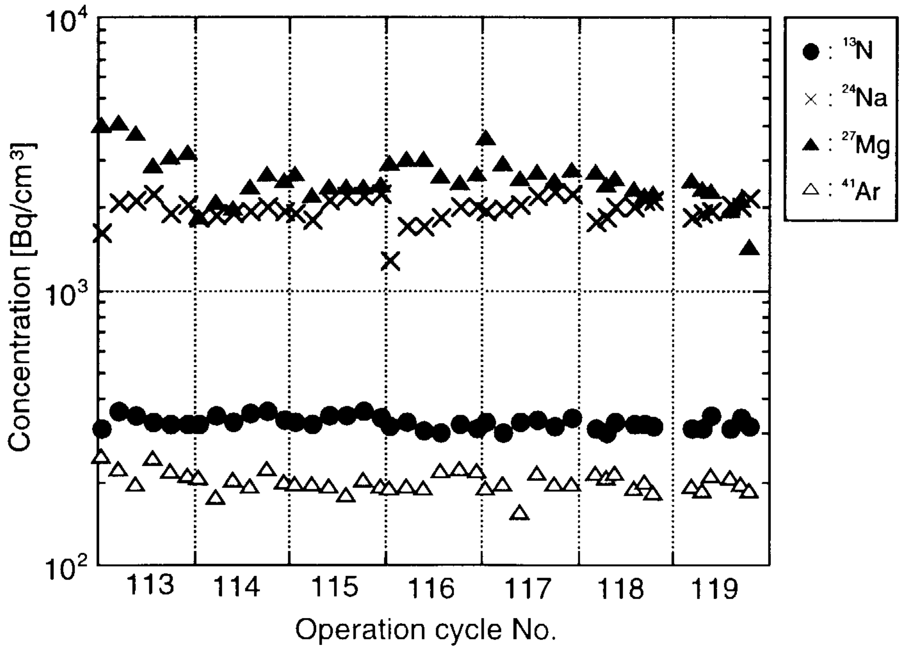


Fig.3.1.2
Radioactive nuclide concentration in primary coolant (From 113rd to 119th operation cycles of JMTR)



Personnel exposure and release of radioactive waste

(1) Personnel exposure

Records in JMTR during FY1996 are in Table 3.1.3. There has been no occupational exposure exceeding the prescribed effective dose equivalent limit.

(2) Release of radioactive gaseous nor liquid waste

Records in JMTR during FY1996 are in Table 3.1.4. Neither radioactive gaseous nor liquid waste released exceeding the release limit specified in the relevant regulations.

Table 3.1.3 Annual effective dose equivalent of personnel at JMTR

Contents	No. of personnel	No. of personnel with accumulated dose (mSv) of each range			collective dose equivalent
		Undetectable	$0.2 \leqq D \leqq 1$	$1 < D \leqq 5$	(person・mSv)*1)
JAERI personnel	136	133	3	0	1.3
Visiting researchers	15	15	0	0	0
Consigned employee	318	323	8	0	1.9

*1) Not detect according to internal exposure

Table 3.1.4 Gaseous and liquid waste released from JMTR

Gaseous, dust wastes		
Gaseous	(Bq)	1.8×10^{13} (^{41}Ar)
Gross α -emitter		Undetectable
Gross β -emitter		Undetectable
Iodine-131		Undetectable
Liquid wastes *2)		
Volume	(m^3)	2.2×10^3
Main nuclide		^3H, ^{51}Cr, ^{60}Co

*2) Those liquid wastes were transported to Radioactive Waste Management Facilities and there treated.

3.2. Fuel and Reflector Management

Acceptance

The fuel elements and follower elements using LEU were manufactured by CERCA (France) and B&W (U.S.A). 44 standard fuel elements and 20 fuel follower elements were supplied for JMTR in FY 1996.

Delivery of spent fuel

Though reprocessing facility of DOE (department of energy) still has not started in the operation, U.S.A government started the program to take back spent fuels used in the test reactor overseas in May, 1997. According to this program, 60 spent fuel elements containing HEU would be shipped from JMTR to U.S.A in June 5,1997.

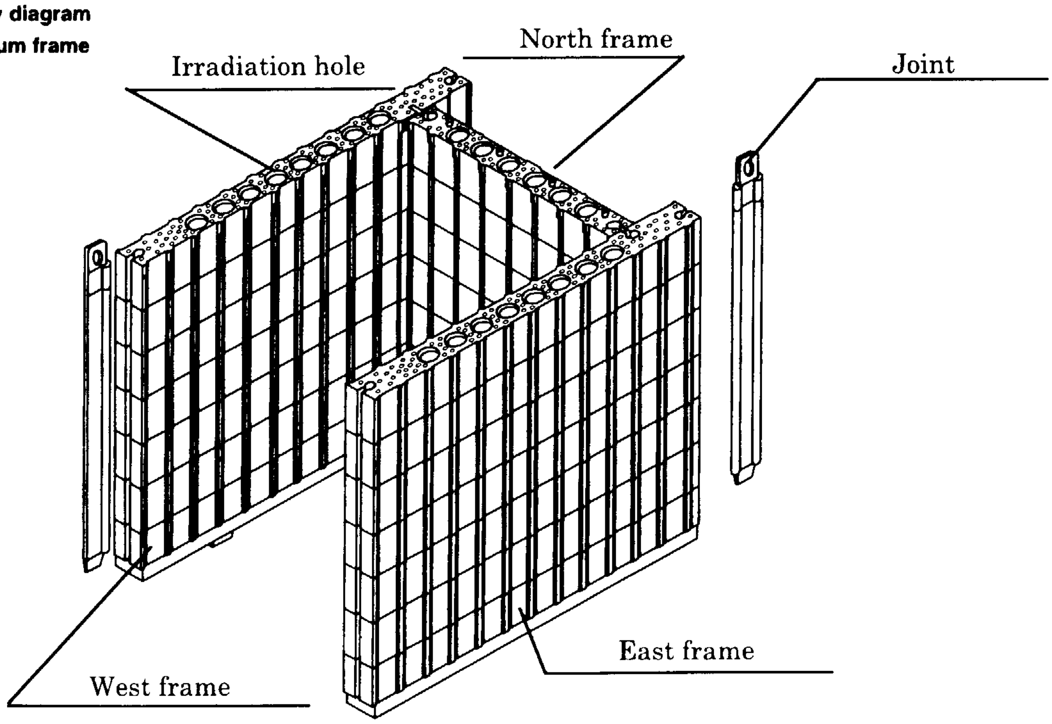
Inspection

12 regular inspections on the fissile material and one special inspection on the spent fuel were carried out by IAEA according to the inspection agreement between Japanese Government and IAEA.

Renewal of beryllium frame

The beryllium frame is one of the in-core structure of JMTR, which is installed in order to enhance thermal neutron flux in the core. It consists of beryllium metal blocks which are stacked in seven stages and connected by the joints vertically and arranged in H-shape horizontally (see Fig.3.2.1). There are many vertical irradiation holes in the beryllium blocks. During the reactor operation, beryllium frame become gradually distorted due to non-uniform distribution of neutron irradiation over the structure. For this reason, the frame is necessary to be replaced about every 5 years in order to prevent degradation of the cooling condition of the capsules inserted in the frame and to keep clearances between the frame and adjacent fuel elements. During the interval reported in this document, distortion of the frame was reached to the prescribed limit, and replacement work was carried out in October 1996. New beryllium frame was successfully installed and its use was approved by the regulatory authority on November 18 after the inspection.

Fig.3.2.1 Summary diagram of beryllium frame



3.3. Utilization of Irradiation Facilities

JMTR has several kinds of irradiation facilities. These are the equipment for capsule irradiation tests, OFS-1 shroud facility and hydraulic rabbit driving devices. Using these facilities, various irradiation tests, the radioisotope productions and irradiation technique development have been performed. Irradiation tests are conducted by the requests of research staffs of JAERI, of universities and of other research organizations. Irradiated materials are mostly fuels and structural materials, which are being used in nuclear facilities or under development to be used.

Capsules irradiated during FY1996 are summarized in Table 3.3.1, Table 3.3.2 and Table 3.3.3, as breakdowns by the type of capsules, related research field and user category, respectively. About the hydraulic rabbits, total irradiation time (sum of the irradiation time for each rabbit in each cycle) are shown in Table 3.3.4. Breakdowns according to the related research field and user category are shown in Fig.3.3.1 and Fig.3.3.2.

Table 3.3.1 JMTR irradiation results in FY1996

Device		118cy	119cy	Total
Capsules		24	27	51
(type)	Instrumented	13	14	27
	Non-Instrumentated	10	12	22
	FGS	1	1	2
(material)	Fuel	4	4	8
	Material	20	23	43
BOCA		1	1	2
Rabbit		26	23	49

Table 3.3.2 Purpose of JMTR utilaization in FY1996 (Capsule, BOCA and Hydraulic Rabbit)

Purpose of the utilization	Capsule, BOCA		Hydraulic Rabbit	
	item	cycle · item	item	hour · item
Fusion R&D	8	12	–	–
LWR R&D	7	13	–	–
RI Production	9	10	6	52.1
Technical Development	5	10	3	17.7
Researches at Universities	3	3	40	2366
FBR fuel	1	2	–	–
HTTR fuel	1	2	–	–
ATR fuel	1	1	–	–
Total	35	53	49	2435.8

Table 3.3.3 Users of JMTR Utilaization in FY1996 (Capsule, BOCA and Hydraulic Rabbit)

Users of the utilization	Capsule, BOCA		Hydraulic Rabbit	
	item	cycle · item	item	hour · item
JAERI(Laboratory)	14	22	2	17.7
JAERI(JMTR)	9	17	–	–
JAERI(Isotope)	9	11	4	52.1
Universities	3	3	43	2366.0
Total	35	53	49	2435.8

Table 3.3.4
List of capsules and Rabbits irradiated for radioisotope production in FY1996

Radioisotope	Target	Reaction	Uses	Production (TBq)
^{32}P	S powder	$^{32}S(n,p)$	Tracer in medical and agricultural field	0.85 0.63×10^{-3} 0.35×10^{-3}
^{51}Cr	Cr_2O_3 powder	$^{50}Cr(n,\gamma)$	Diagonostic of blood disease in medical field	0.20 0.19
^{35}S	KCl powder	$^{35}Cl(n,p)$	Tracer in agricultural and industrial field	0.08
^{192}Ir	Ir pellet (ϕ2×2mm)	$^{191}Ir(n,\gamma)$	Radiation source of non-destractive inspectio	194.70
	Ir pellet (ϕ1.1×1.2mm)		Radiation source for treatment of cancer	25.20 20.80
^{169}Yb	Yb_2O_3 pellet (ϕ1×1mm)	$^{168}Yb(n,\gamma)$	Radiation source of non-destractive inspectio for thin welded tube	0.96 1.00
^{60}Co	Co wire (ϕ0.91×15mm)	$^{60}Co(n,\gamma)$	Radiation source for treatment of cancer	0.07 0.03
^{188}Re	$^{186}WO_3$ powder	$^{186}W(n,\gamma)^{187}W(n,\gamma)^{188}W(\beta^-)$	Radiation Source for treatment of cancer	0.08

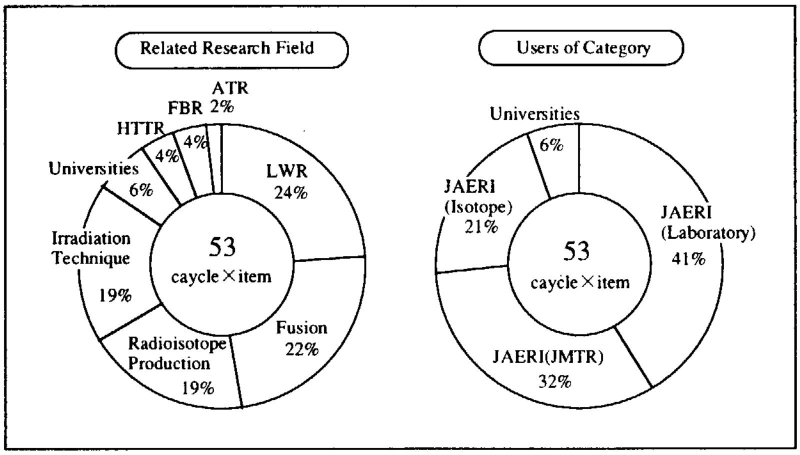


Fig. 3.3.1
Capsules irradiated in JMTR during FY1996

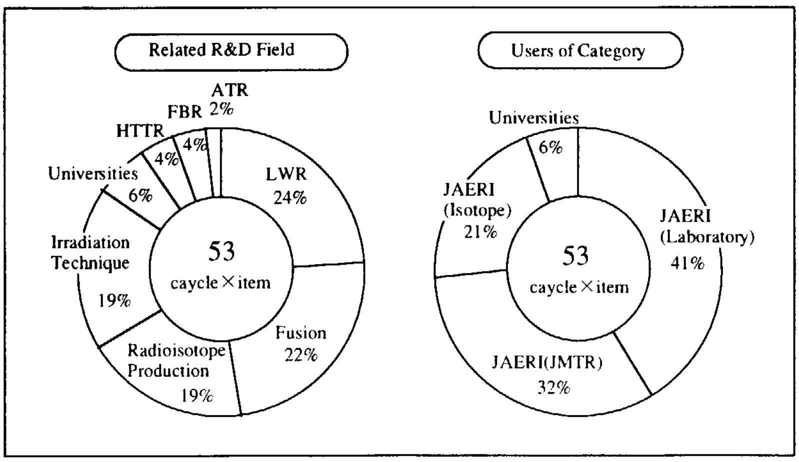


Fig. 3.3.2
Hydraulic Rabbits irradiated in JMTR during FY1996

3.4. Maintenance Engineering

Database of maintenance records

In JMTR, a computerized data management system for maintenance activity has been developed in 1994. This system (see Fig.3.4.1), called as MES (Maintenance Engineering System), has been utilized as;

i) to establish database containing the data about periodical checking, inspection and trouble-shooting since the beginning of power operation,

ii) to collect new data about periodical checking, inspection and trouble-shooting,

iii) to support the preparation of maintenance reports.

Using this system, trend analysis on the troubles of devices and the evaluation of adequate maintenance intervals have been performed. These activities are very helpful for effective planning of the maintenance. A typical example is the analysis of deterioration of the equipments in the control rod drive mechanism.

A total of 970 new records were added to MES during FY1996, then accumulated number of records became 2,370.

Computerized support system for routine patrol

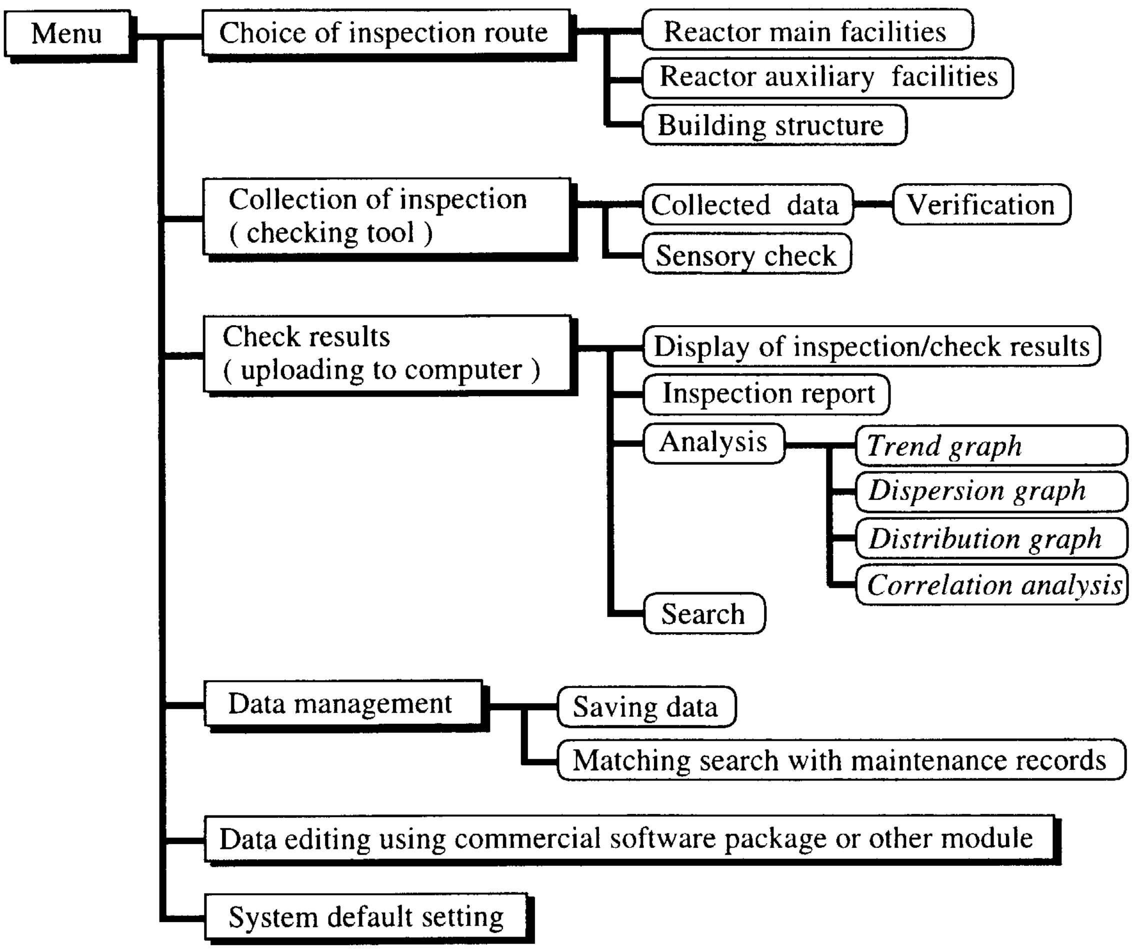
A data management system for daily patrol in the reactor facility has been developed. The system is to help systematic collection of the data obtained from device checking results during the patrol and to arrange the data as a database. Data analyses including quick search of related information among the past maintenance records and trend analysis, will be possible by using this system. Main software has been developed and the listing up of the facilities and devices, to be focused by this system, has been started in FY1996.

Individual maintenance strategy development

Main trend of the facility maintenance in the nuclear industry is being shifted recently from "preventive maintenance" to "predictable maintenance". The latter means a predicting renewal period method by using the sympton parameter-trend evaluation without by tied to actual results of the period.

In such maintenance strategy, individual management of each device/facility will become more important. In JMTR, enforcement of individual management is under discussion by taking into account of the development of support systems described in the items above. It will enable early detection of the failures, quick resolution of the troubles and more efficient scheduling of the maintenance.

Fig.3.4.1
Functional outline graph
of daily patrol system



3.5. Post Irradiation Examinations

Post irradiation examinations (PIEs) performed in the hot laboratory during FY 1996 were on LWR fuels subjected to power ramping test, NSRR test fuels, structural materials of LWR, HTTR and fusion reactor, shape memory alloys, and others. Remote recapsuling of BOCAs for the power ramping tests in JMTR and their remote dismantling after the tests were also performed.

Nuclear fuels

(1) Fuel rods for power ramping test

A fuel rod segment irradiated in power rectors was installed into and take out from the BOCA in the C-1 concrete cell. After assembling of the BOCA, the fuel rod was shipped to the Nuclear Fuel Development Co. (NFD) hot laboratory for PIEs. Concerning the fuel rods irradiated for the tests by JAERI researchers, their PIEs were carried out in the concrete cells.

(2) Nuclear Safety Research Reactor (NSRR) test fuel rods

Destructive tests on the test fuel rods of NSRR were carried out such as FP gas collections and analyses, and metallography after the nondestructive tests. These fuels were also examined by micro-gammascanning, autoradiography and density measurements.

Structural materials

1) Structural materials of LWR

Test pieces of stainless steel for core internals were irradiated to develop the life estimation of LWR. The PIEs on these test pieces were performed as tensile test, slow-strain-rate tensile test, metallography and scanning electron microscopy.

2) Structural materials of HTTR

Test pieces of materials of pressure vessel and control rod cladding were irradiated to study their characteristics. The PIEs on test pieces of pressure vessel materials were carried out by small punch test. PIEs on the specimens of control rod cladding material were performed as metallography and scanning electron microscopy.

3) Structural materials of fusion reactor

Test piece of stainless steel (SUS316 (JIS)) and Inconel 625 for vacuum vessel of fusion reactor were irradiated to study the re-welding characteristics. The PIEs such as weldment test, milling machining, dimension measurement, visual inspection, tensile test ,metallography and scanning electron microscopy were carried out.

4) Structural materials of capsules

Test pieces of shape memory alloys were irradiated to study the effect of neutron irradiation. The PIEs such as Charpy impact test, high temperature tensile test, constant-load deformation test, fracture toughness test, fractography and metallography were carried out. Test pieces of friction welding materials (Nb-1Zr/SUS304 (JIS)) for capsule were also irradiated to study irradiation effect on the material. The PIEs such as tensile test, hardness test, metallography and X-ray microanalysis were carried out.

5) Treatment of radioisotope production capsules

During FY 1996, 11 radioisotope production capsules were treated in the hot laboratory and dismantled in the C-2 concrete cell. Produced radioisotopes were 3.12×10^{14}Bq of ^{192}Ir, 2.5×10^{12} Bq of ^{33}P, 9.9×10^{11} Bq of ^{51}Cr, 6.6×10^{11} Bq of ^{35}S and 6.1×10^{12} Bq of ^{169}Yb, 3.7×10^{6} Bq of ^{89}Sr and 1.0×10^{9} Bq of ^{188}Re. There were transported to the Department of Radioisotope Production in the Tokai Establishment of JAERI.

Personnel exposure and release of radioactive waste

(1) Personnel exposure

In the hot laboratory, there was no occupational exposure exceeding the prescribed effective dose equivalent limit as shown in Table 3.5.1.

(2) Release of radioactive gaseous and liquid waste

There was no release of gaseous and liquid wastes from the hot laboratory exceeding the release limit specified by relevant regulation. Amount of gaseous and liquid wastes released from the hot laboratory are shown in Table 3.5.2.

Table 3.5.1
Annual effective dose equivalent of personnel in the hot laboratory

Contents	No. of personnel	No. of personnel with accumulated dose (mSv) of each range			collective dose equivalent
		Undetectable	$0.2 \leqq D \leqq 1$	$1 < D \leqq 5$	(person・mSv)*1)
JAERI personnel	25	16	6	3	8.5
Visiting researchers	4	4	0	0	0
Consigned employee	145	122	22	1	10.2

*1) Not detect according to internal exposure

Table 3.5.2
Gaseous and liquid wastes released from the hot laboratory

Gaseous, dust wastes		
Gaseous	(Bq)	1.5×10^{9} (^{85}Kr)
Gross α-emitter		Undetectable
Gross β-emitter		Undetectable
^{131}I		2.2×10^{4}
Liquid wastes *2)		
Volume	(m3)	6.3×10^{1}
Main nuclide		^{60}Co, ^{134}Cs, ^{137}Cs

*2) Those liquid wastes were transported to Radioactive Waste Management Facilities and there treated.

4. Topics in FY 1996

4.1. Evaluation of Neutron Field for Irradiation

With progress of nuclear material developments, the needs on irradiation test shifts to the one with high quality and high precision. Since effect of neutron energy spectrum on the irradiation damage is regarded to be important from the viewpoint of irradiation correlation, detailed spectrum information and neutron spectrum adjustment are needed in irradiation tests for R&D of light water reactor or fusion reactor materials. The technology development has been conducted since 1992 to positively respond to these requirements.

Neutron energy spectra which have been provided for JMTR users are the data evaluated for each layer of the core (fuel region, reflector layer no.1,2, etc.) based on the multi-foil measurement using JMTRC (JMTR Critical Facility). These layer-wise spectra were evaluated as sufficiently precise for fast neutrons, but their accuracy were not confirmed in thermal neutron energy region yet. It is difficult to evaluate the thermal neutron at local position like at sample position due to strong dependence on capsule structure materials. The development of whole energy spectra evaluation technology is therefore proceeded. The spectrum measurement experiments with multiple activation wires were started from 1994, and the irradiation of the wires in JMTR was finished. Wires of Ag, Au and Co were used as thermal neutron monitors, and wires of Ti, Mn, Cu, Ni, Fe and Nb were used as fast neutron monitors. Schematic of the capsule used for irradiation and capsule location in the JMTR core are shown in Fig.4.1.1. Activation measurements for irradiated wires after dismantling work was started and preparation of NEUPAC code for spectrum adjustment was proceeded. With regard to improvement of the precision of the calculated neutron spectrum at the irradiation position, preparation of MCNP code for detailed spectrum calculation of whole core in the JMTR was proceeded. The data on the adjustment of neutron spectrum and the evaluation of the calculated spectrum will be obtained at an end in FY 1997.

Evaluation of calculated neutron flux was conducted on the first step of improvement of the precision of the calculated neutron spectrum using MCNP code in FY 1996. As a result of verifying by using measured data, it was confirmed that calculation could estimate fast neutron flux within about 10% on large irradiation sample (about 300cm^3). Accuracy of thermal neutron flux calculated on large irradiation sample and (fast and thermal) neutron flux calculated on small irradiation sample (about 0.3cm^3) are being verified.

In order to respond to the requirement of JMTR users, the calculated neutron spectrum at irradiated sample position was started to be offered in February 1997. An example of calculated neutron spectrum is shown in Fig.4.1.2. Calculated neutron spectrum at an optional neutron energy group within 137 Groups (MGCL group structure) can be provided for JMTR users.

Fig. 4.1.1
A schematic of the capsule used for neutron energy spectrum evaluation in the JMTR

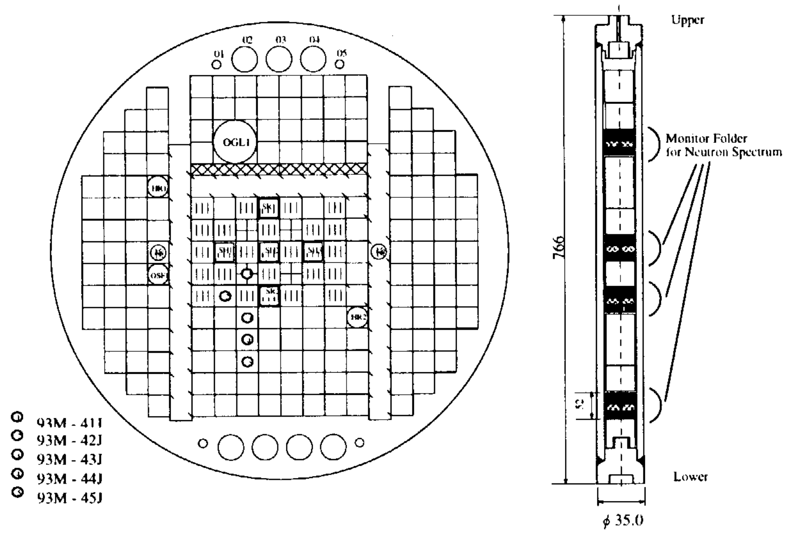
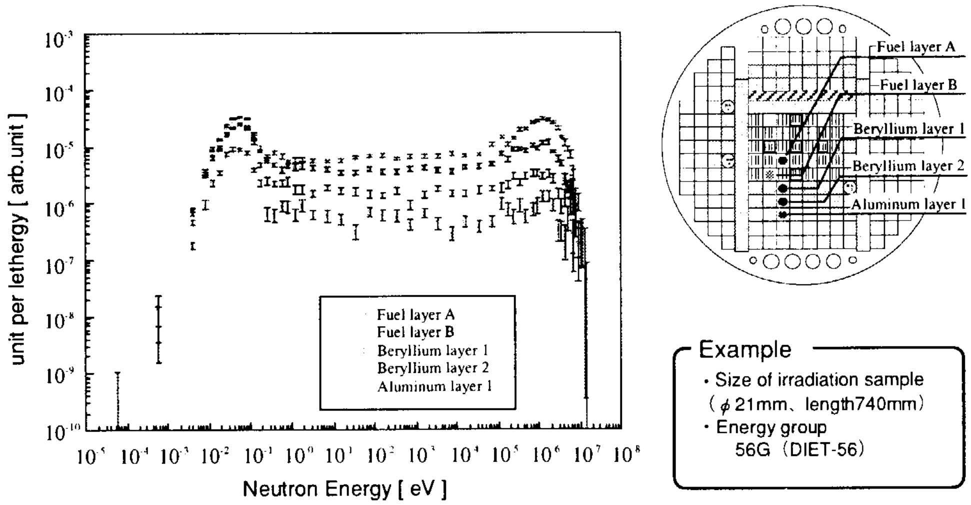


Fig.4.1.2
Analitical results of Helium production in 304 stainless steel



4.2. Design of Capsules and Irradiation Facilities

Development of the oxygen sensor

Development of the new oxygen sensor attachable to the fuel sample, is being developed to be used for examining mechanism of chemical behavior of the fuel under irradiation. In FY 1996, the solid normal mode oxygen sensor using the zirconia (solid electrolyte), was fabricated and irradiated in order to examine the effect of the neutron irradiation. Also a hydraulic rabbit with the oxygen sensor was irradiated. The structure of the oxygen sensor is shown in Fig.4.2.1. For the oxygen sensor in the capsule, the electromotive force was measured in-situ. For the sensor in the hydraulic rabbit were measured electromotive force and the a.c. impedance after irradiation. The measurement of a.c. impedance was to clarify the effect of neutron irradiation of the zirconia solid electrolyte which permits the oxygen ion conduction. The data were used to evaluate the warmth sum behavior and activation energy of the dipole. The results are shown Fig.4.2.2.

Fig. 4.2.1 Oxygen sensor

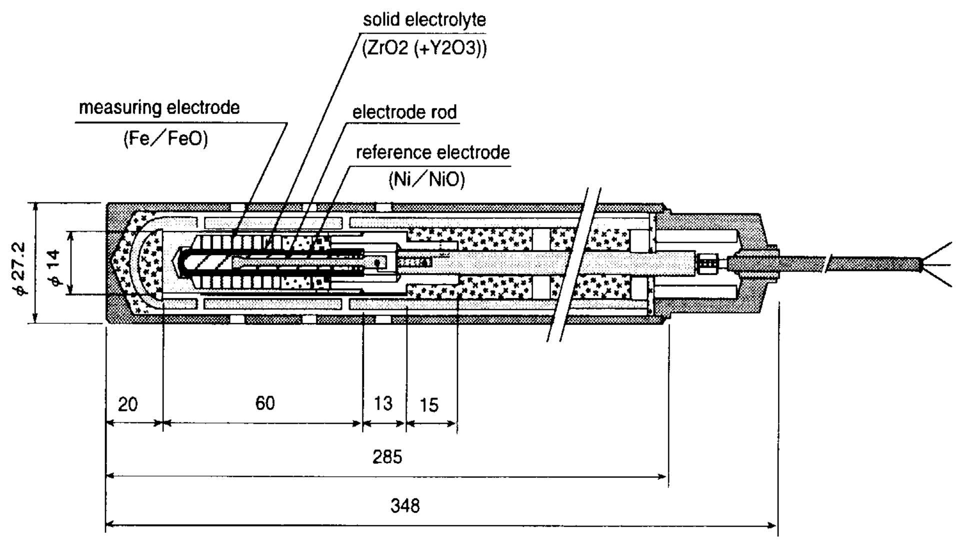
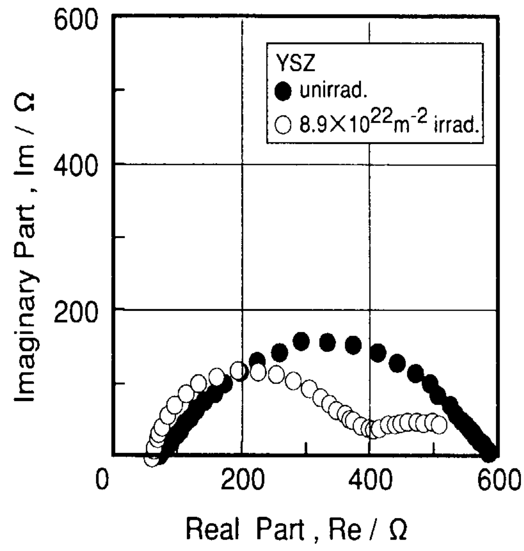


Fig. 4.2.2
Measurement results of a.c. impedance of the oxygen sensor at 1123K



The migration technique of irradiation sample

As a part of activities to establish practical application of the accelerated irradiation method, two types of capsule were developed in FY1996, i.e., pull-up type capsule and vertical migration type capsule. Both capsule are to be used to study the effects of accelerated irradiation by evacuating samples to the out of core after desired irradiation time for each sample.

[Pull-up type]

This capsule is designed to be used in relatively high neutron flux region. Conceptual figure of the structure is shown as Fig.4.2.3. Inner capsules containing samples are hanged using coil springs and initially fixed in irradiation positions by Zn tension wires connected to the terminals at the bottom. When a desired irradiation time for a sample pass, tension wire is melted by controlled heater and the inner capsule is pulled up to the out of core by the spring. Mock-up tests for basic function of this capsule are currently being carried out.

[Vertical migration type]

In this capsule, coil spring made of recently developed shaped memory alloy (SMA) is employed to migrate samples and measuring devices. The feasibility on the application of shape memory alloy to irradiation capsule has already been examined in previous year. The coil is capable to move a sample unit in long span. This capability is specially important to shelter radiation sensitive measurement device such as optical fiber from the core when it is not used. In FY1996, out-pile functional test using SMA coil element has been carried out. Material of SMA is $Ti_{50}Ni_{50}$ with reversible transformation onset temperature of 368K. Wire diameter is 2 mm, coil outer diameter is 28 mm and free length is 172 mm. Using electric heater, driving distance of 430 mm was achieved under the condition of 12N load. Structure of the vertical migration type capsule is shown in Fig.4.2.4. Irradiation test using this capsule is scheduled to start at 120th operation cycle in 1997.

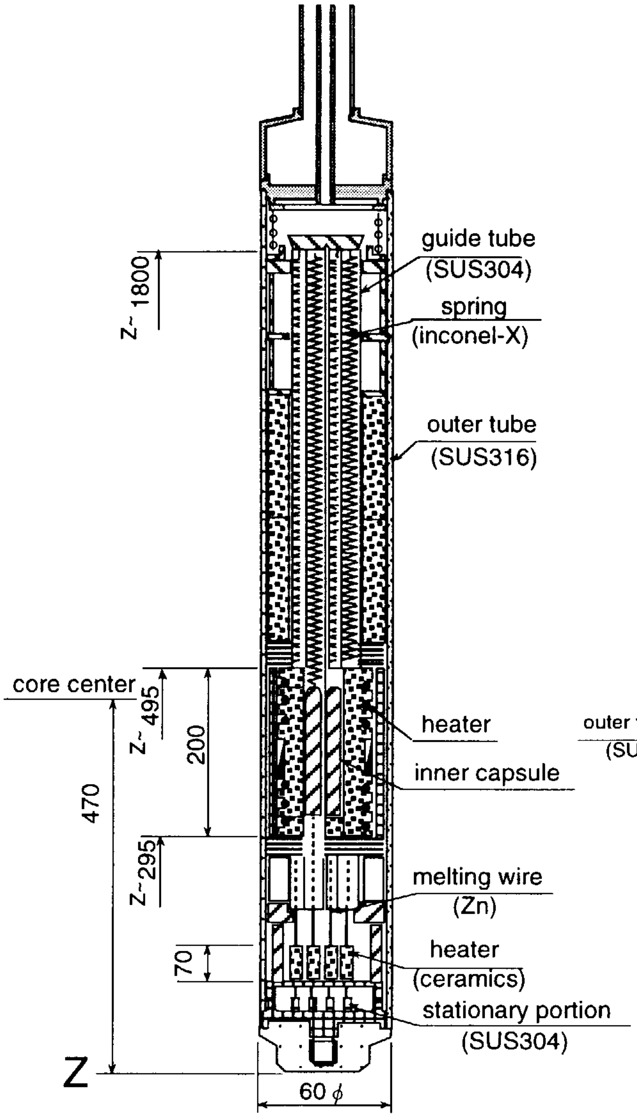


Fig. 4.2.3
Pull-up capsule

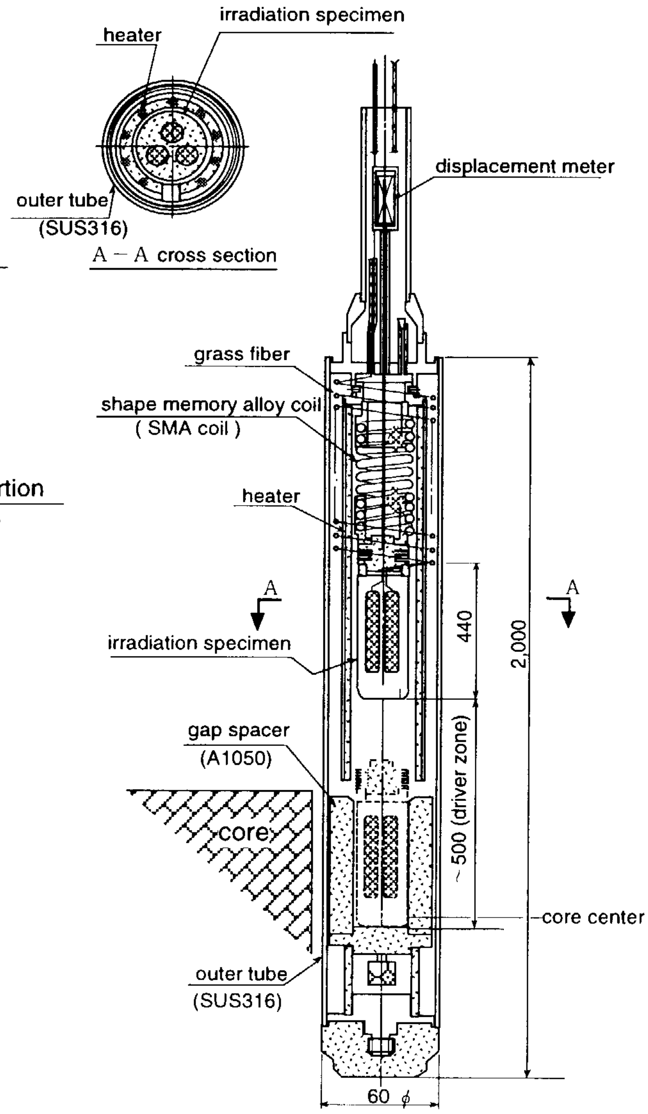
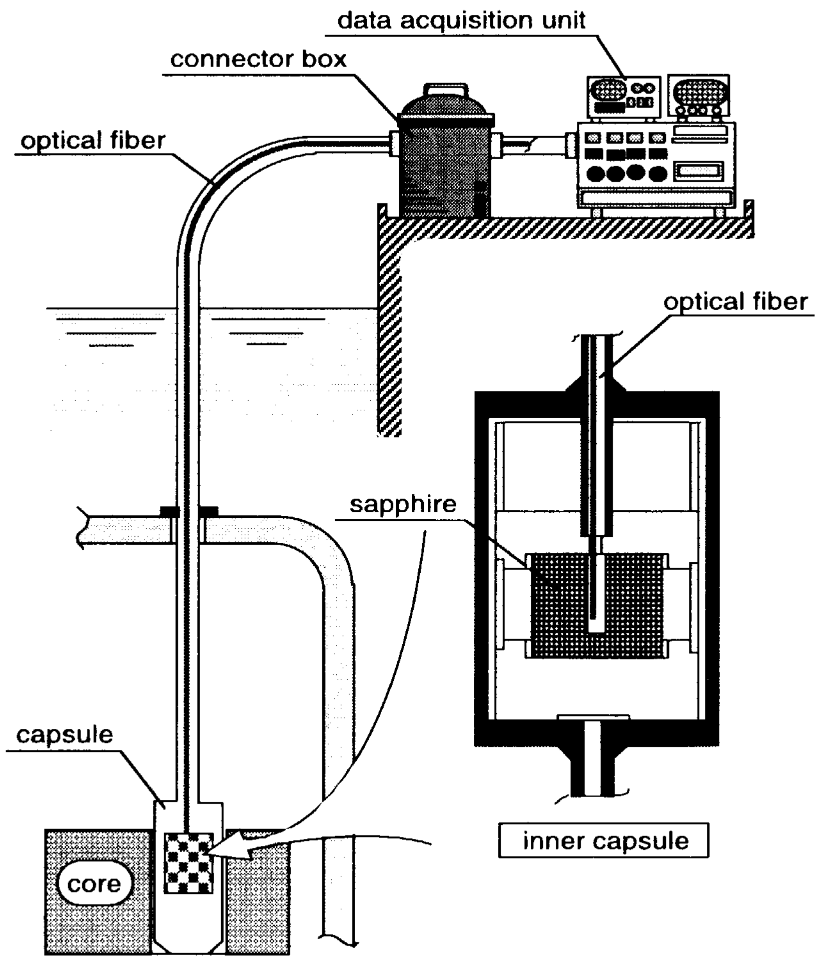


Fig. 4.2.4
Vertical migration capsule

In-situ measurement technique using optical fiber

Among in-situ measurement technologies for neutron irradiation tests, the instrumentation using optical fiber has not succeeded to measure the light signal from samples yet due to degradation problem of the optical fiber by neutron and gamma ray irradiation. Recently, the radiation tolerant fiber has been developed. Applying this type of optical fiber, a technique to measure the light emission of irradiated sapphire has been developed. Sapphire is a candidate of the material for optical measurement window of fusion reactor. The outline of optical measurement instrument is shown in Fig. 4.2.5. The sapphire emits the light by neutron and gamma ray irradiation. One tip of the 30 m long optical fiber was inserted in the sapphire, and another tip was connected to the data acquisition unit. During the power operation of the JMTR the light emission from the sapphire was continuously measured up to 3.5×10^{22} n/cm^2 of fast neutron fluence at 923K.

Fig. 4.2.5
Outline of optical measurement instrumentation for irradiation test capsule



Coupling irradiation techniques

Coupling irradiation utilizing different neutron sources is considered as an important technique to simulate various damage characteristics of the materials such as defect growth under accelerated condition. In this technique realized in the JMTR, material or fuel irradiated in the LWR, FBR or accelerator is installed in the capsule with instrumentation at the hot laboratory and re-irradiated in the JMTR. As the first stage of the technique development, non-instrumented capsule for re-irradiation was assembled in the hot laboratory and irradiated in the JMTR in FY1995. In FY1996, instrumented capsule for re-irradiation was designed and fabricated, and irradiated samples were installed into assembled capsule. Assembling of the capsule for coupling irradiation is schematically shown in Fig.4.2.6. This capsule will be irradiated in FY1997 after completing installation of data transmission cables, vacuum temperature control guide tube and protection tube.

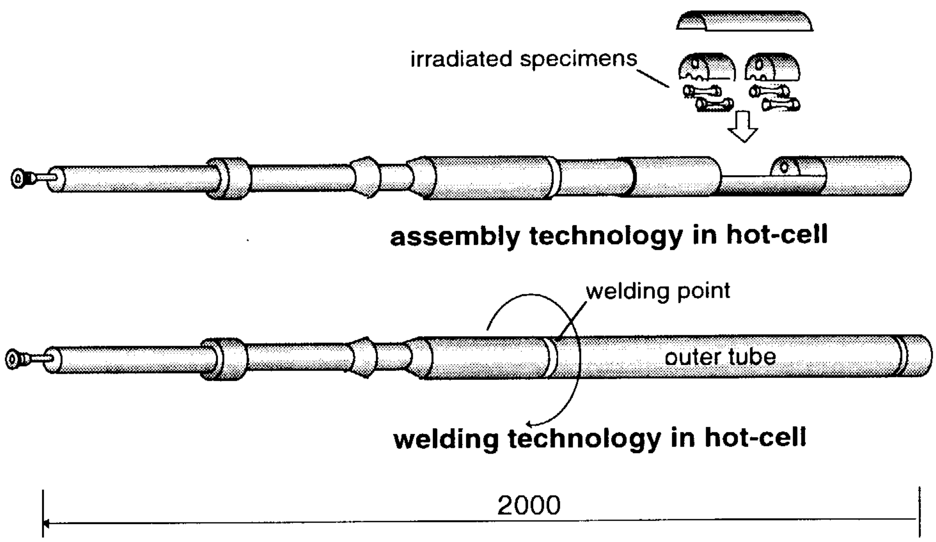


Fig. 4.2.6
Assembly technology in hot-cell

4.3. Renewal of HR-2

In-pile tube of Hydraulic Rabbit irradiation facility No.2 (HR-2) has been used for 24 years and irradiated neutron fluence reached design limit after the 119th JMTR operating cycle. Therefore, it was renewed in the 1996 fiscal year. At the renewal, the irradiation performance improvement was also achieved by changing installed location among the core to enhance the neutron flux in it. The new in-pile tube was started to be fabricated in the 1995 fiscal year, and finished installing in JMTR in the 1996 fiscal year. A flow diagram of the facility is shown in Fig.4.3.1.

Characteristics of In-pile tube

The purpose of the performance improvement is to enhance the irradiation neutron flux of the facility. Characteristics of new in-pile tube is shown in Table 4.3.1 in comparison to that of the previous one. With regard to the neutron flux, the fast neutron flux is about 4 times higher than previous, and the thermal neutron flux is about doubled. The irradiation experiences of the former HR-2 are shown in Fig.4.3.2. The proportion of the rabbits which irradiated over 100 h reaches about 50%, however, the exposure times for such rabbits become possible to be shorter by using renewed HR-2.

Fig. 4.3.1
Flow diagram of HR-2

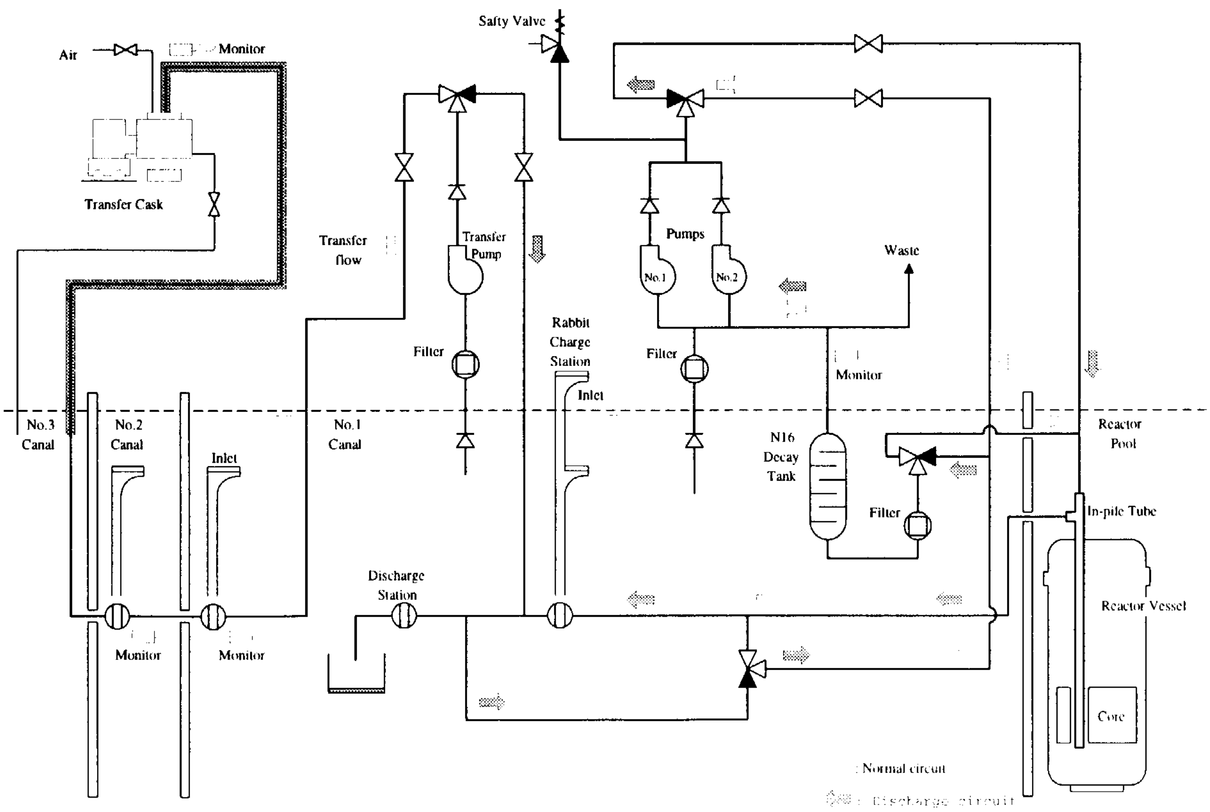


Table 4.3.1
Characteristics of HR-2

		New in-pile tube	Previous one
Location in the core		M-9	M-11
Thermal neutron flux	Max.	3.5×10^{14} (n/cm^2 · s)	1.3×10^{14} (n/cm^2 · s)
[>1Mev]	Ave.	2.6×10^{14}	9.9×10^{13}
Fast neutron flux	Max.	9.3×10^{13}	2.1×10^{13}
[<0.68eV]	Ave.	7.0×10^{13}	1.6×10^{13}
γ heat rate		6.0 (W/g)	2.2 (W/g)
Coolant		Light-water	
Flow rate of the coolant		8.4 (m^3/h)	
Temperature of the coolant		40 (℃)	
Pressure of the coolant		Max. 2.0 (MPs)	
Dimension of the holder		Φ 32×L150 (mm)	
Material of the holder		Al or SUS	
Dimension of the specimen		Φ 26×L120 (mm)	
Heat generation		9 (kW)	
Number of charged rabbits		Max. 3	
Time of irradiation		Min. 1 min	

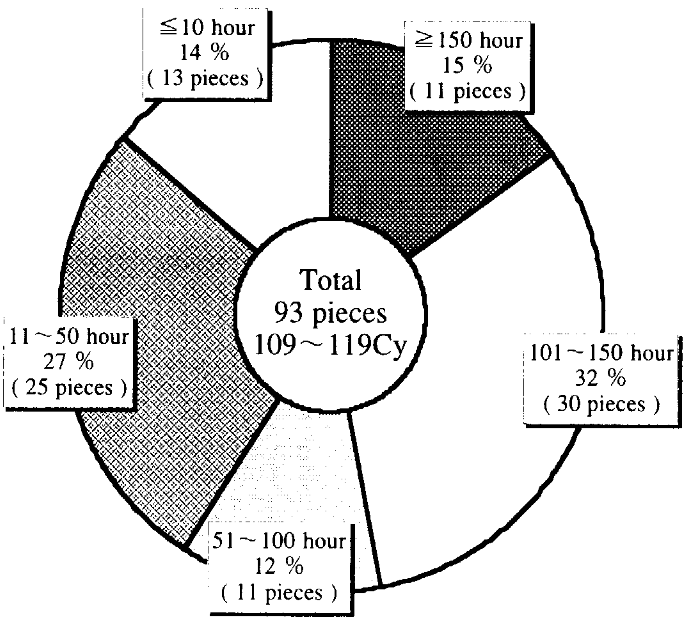


Fig. 4.3.2
The utilization proportion according to the irradiated time in HR-2

Matter concerned on the performance improvement

It is possible to enhance the irradiation neutron flux of the facility by installing the new in-pile tube into the irradiation hole where the neutron flux is higher. As the result, however, the radiation fluence for the in-pile tube itself is accelerated, and the duration of service of the in-pile tube will be shorten. The radiation shielding for the facility loop must be also considered, because the radioactivity of HR-2 circulating water may increase by enhancing the irradiation neutron flux. Considering above subjects, the possibility of performance improvement were discussed on condition that out-pile part of the facility was not modified. Main points are as follows:

a) Change of the structure of in-pile tube and selection of the irradiation hole

The strength of the welded joint is usually inferior than that of the base metal under neutron irradiation. From the viewpoint of the life time of the in-pile tube, it was decided to design new in-pile tube by shifting welding joint to the elevation out of active core to reduce neutron irradiation. The M-9 irradiation hole was selected where new HR-2 in-pile tube penetrate the core of the JMTR. The structures of new In-pile tube and former one are shown in Fig.4.3.3.

b) Radiation-resistant life time of new in-pile tube

The radiation-resistant life time of the in-pile tube was limited to be $3x10^{21}$ n/cm^2 for fast neutron fluence on the basis of post irradiation examination results of the SS316 irradiated in the JMTR. On the other hand, the duration of service of the in-pile tube will be shortened by enhancing the irradiation neutron. As additional PIE, the mechanical properties test of irradiated SS316 specimen was carried out in the JMTR hot laboratory in order to collect data for neutron fluence range from $3x10^{21}$ n/cm^2 to $1x10^{22}$ n/cm^2.

As an example of the test results, the relationship between the neutron fluence and the rupture elongation is shown in Fig.4.3.4. The total neutron fluence of the new in-pile tube will reach $1x10^{22}$ n/cm^2 after JMTR operation of 50 cycles. For this fluence, the residual elongation is assume to be about 37%, and this value is larger than 10% which is a standard residual ductility. Therefore, it is possible to enhance the irradiation neutron flux without shortening the duration of service of the in-pile tube.

Evaluation of the radiation dose of the circulation system

It is expected the radioactivity of the HR-2 circulating water will increase by enhancing neutron flux. The main radioactivity sources are ^{16}N which is created by $^{16}O(n,p)^{16}N$ reaction, and activated corrosion products which are included in the water. It is confirmed by the shielding analysis that the ability of the ^{16}N decay tank installed in the canal is sufficient to decay the radio nuclide. Although the dose equivalent rate of the area is expected to increase, it was also confirmed by the analysis within the acceptable level to remain.

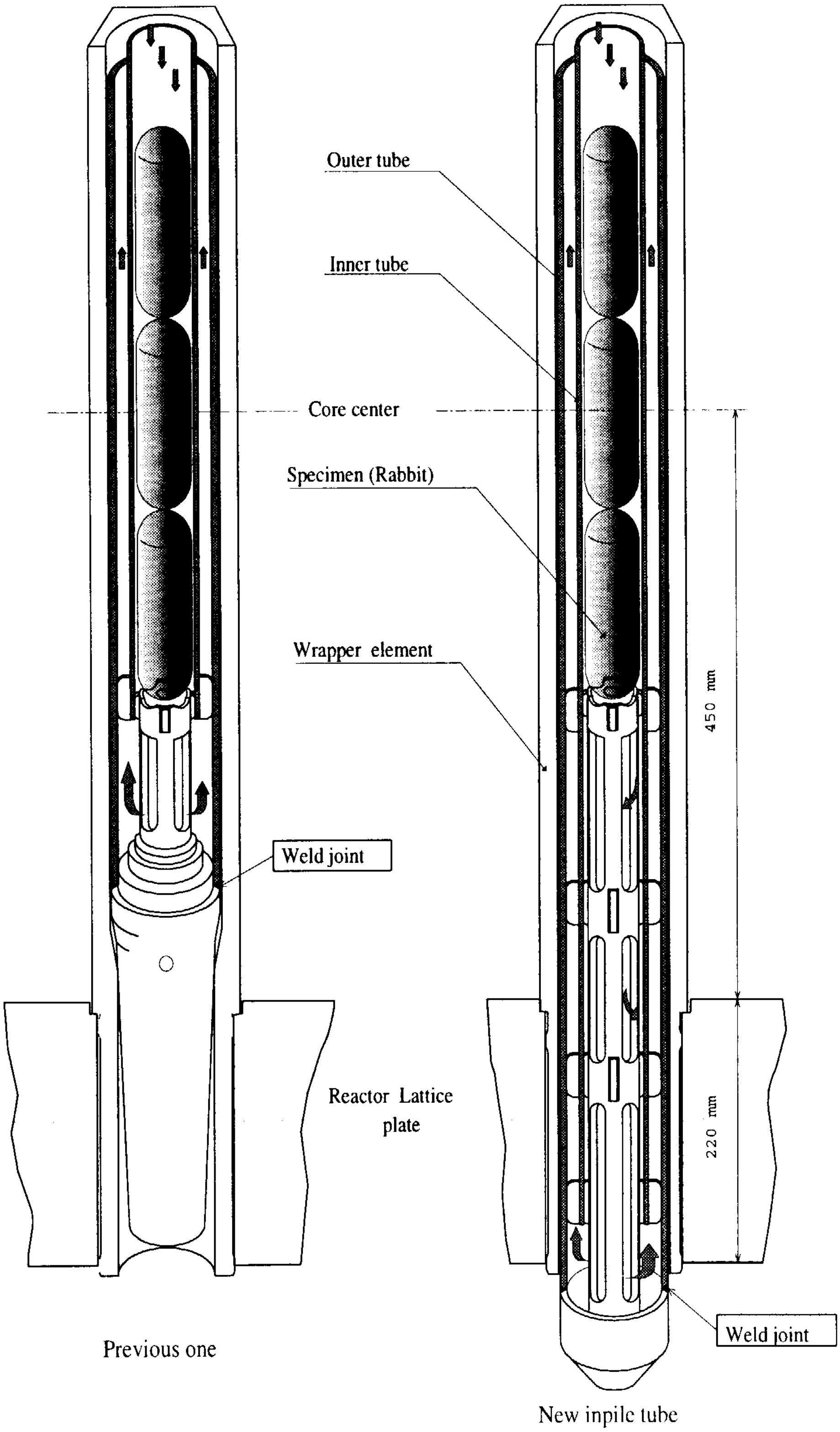


Fig. 4.3.3
Structure of new in-pile tube and previous one

Nuclear effect for the reactor

It is anticipated that reactivity disturbance by inserting or taking out a rabbit will increase, because the new HR-2 is installed in the irradiation hole near the fuel region of reactor core. However, as a result of the calculation, it was confirmed that the value of the reactivity disturbance would remain as 0.03%Δk/k or less, within prescribed limit of reactivity disturbance.

The expected γ-heating rate changes from 2 W/g to 6 W/g, and the heat input to the circulating water will increase. In this case, it was confirmed that the local boiling which cause a strong reactivity disturbance will not occur.

Irradiation utilization

The facility utilizations are expected as following.

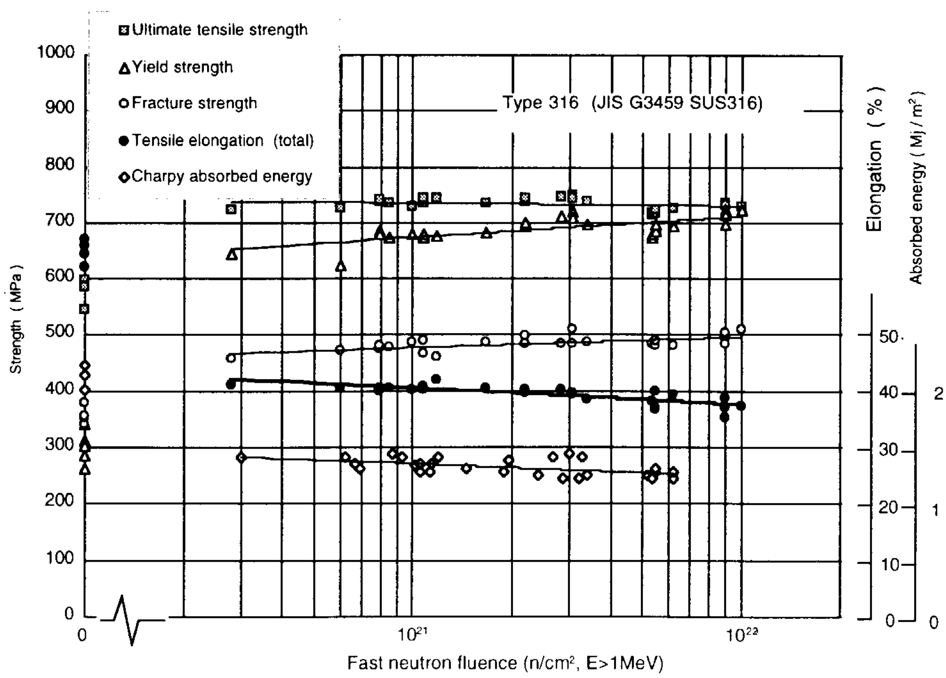
a) The efficiency of the facility utilization including flexibility of irradiation scheduling will be increased by shortening irradiation time of each rabbit.

b) Production and development of ^{67}Cu, ^{33}P and ^{35}S become possible using increasing fast neutron flux.

c) The production control of ^{186}Re is currently impossible because it needs to use capsule and irradiated neutron fluence is too high. It becomes possible by using new HR-2.

d) The irradiation test for the extremely minute radioactivation analysis in the basic research become possible, because the fast neutron flux increases.

Fig. 4.3.4 Mechanical properties of stainless steel at a exposure by neutrons



4.4. Replacement of Partition Tube in OSF-1

Power ramp test facility

In JMTR, the power ramp test facility was founded for the power ramp test of the light water reactor fuel. This facility has been composed of boiling water capsule (BOCA) and pressure control system which retain the fuel sample in the BWR operating condition, He-3 pressure control system which changes an power of the fuel sample, Oarai shroud irradiation facility (OSF-1) and capsule handling machine which enables the capsule exchange during nuclear reactor operation. Using this facility, power ramp tests of 76 fuel samples has been carried out since fiscal year 1981 and many useful results elucidating behavior of the light water reactor fuel has been obtained.

In 1988 in-pile tube, originally made of stainless steel, was replaced with Zircaloy made in-pile tube. At the same time gas screen portion of the flow partition tube was replaced with one made of aluminum, which was also made of stainless steel. These replacement were aiming to enhance thermal neutron flux in the BOCA and to realize power ramping test for high burn-up BWR fuels.

The fatigue life of this aluminum gas screen was estimated to be about 10,000 times of He-3 gas pressure change. Operation cycles of the gas screen reached nearly this life limit. Therefore its replacement was decided.

History of removed partition tube

Removed partition tube has been used from 1988 to 1996. During this period, power ramp tests of 50 elements on BWR high-performance fuel of BWR high burn-up fuel elements and safety study fuel elements were carried out. It experienced fuel failure of 9 fuel elements and 2 times of 1,000 cycle test, etc.. Inside surface of the gas screen was inspected using fiber scope every year, and the integrity of the gas screen inside surface has been confirmed even many insertion and ejection of the BOCAs were experienced.

Design of new partition tube

The partition tube replacement program was conducted for 5 years from fiscal 1992 to 1996. The design was carried out for first 2 years, and the preparation of new tube took 3 years. The schematics of the OSF-1 in-pile tube and the partition tube are shown in Fig.4.4.1.

(1) The design of the new partition tube

The design for acquiring "authorization of the design and method of construction" considered the data obtained through qualification tests such as out-core fatigue test using the gas screen test specimen, metalography of the fatigue test sample, strength confirmation test of the weld zone. The following points were especially considered on the design of the partition tube replacement.

a) Based on ministerial ordinance issued by the Prime Minister's Office on technical standards of the design and method of construction of nuclear reactors served in purposes of test and study, etc., the evaluation on the fatigue life in the gas screen was carried out.

b) With re-examining the earthquake resistant design code of the Oarai district, the earthquake-resistant design calculation was carried out.

c) The stress evaluation was carried out which will arise in the cross section of the gas screen vessel of the partition tube, because it has welded structure of double cylindrical walls.

By these results, it was confirmed that the fatigue life of 10,000 times was a sufficiently conservative safety limit, and that there is no difference between the results and conclusions about earthquake-proof and stress evaluation for new partition tube and those for old tube.

(2) Fabrication and exchange of the partition tube

The partition tube was fabricated using special processes of production such as pressure welding of the aluminum alloy and stainless steel, electron beam welding and boring and trepanning association (BTA) processing. The delay occurred due to Great Hanshin Earthquake on January 17th, 1996 in the fabrication but recovered among entire schedule. Among the performance tests, cold run test was completed as scheduled, but hot run test was delayed in to FY 1997, according to the change of JMTR operation schedule which caused by the trouble of the surge tank for JMTR.

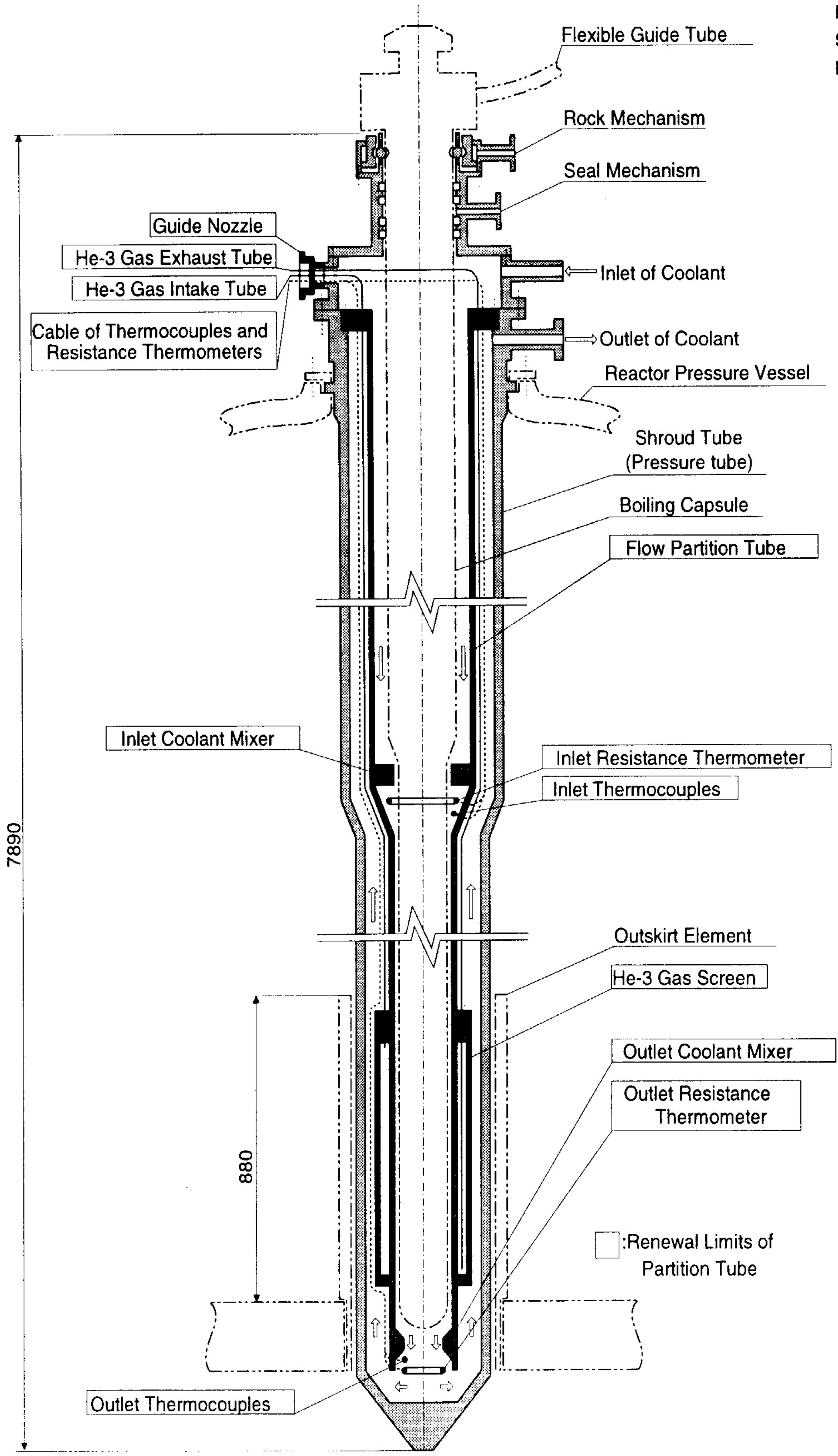


Fig. 4.4.1 Schematics of OSF-1 In-pile Tube

4.5. Neutron Irradiation Studies for Fusion Blanket Development

In 1996, in-pile functional test facility, a beryllium PIE facility and an electron beam heating facility (OHBIS) were installed in the JMTR and JMTR hot laboratory in order to collect the engineering data.

Development of irradiation facilities

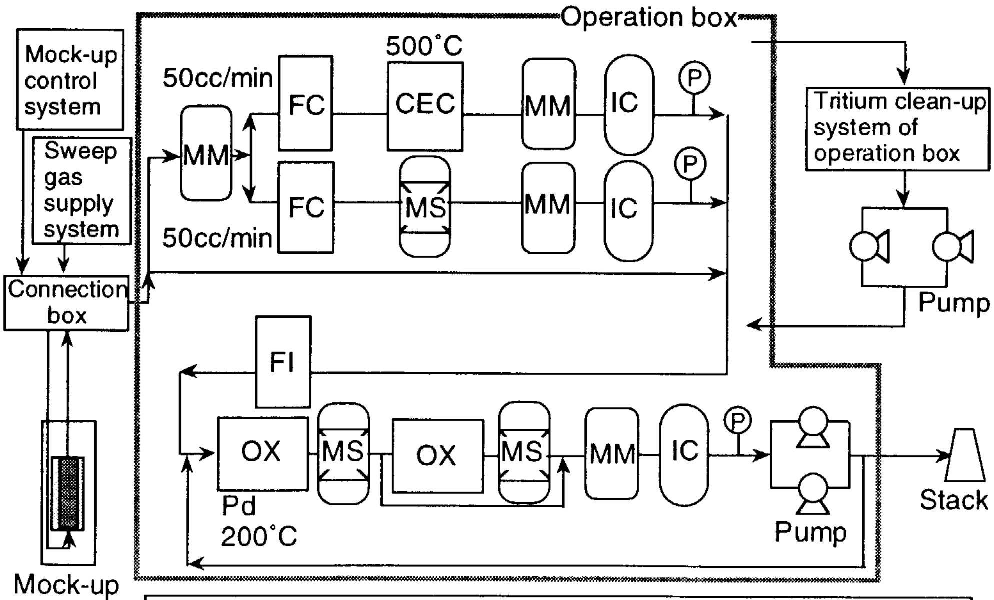
1) In-pile Functional Test Facility

An in-pile functional test facility has been considered to obtain engineering data since 1991. The facility consists of a blanket mock-up and a sweep gas system.

With regard to the sweep gas system, the installation to the JMTR building was finished. This system can measure total tritium and tritium gas concentration in the sweep gas released from a blanket mock-up on line. The flow rate of sweep gas can be controlled from 10cc/min to 1,000cc/min. Moreover, the hydrogen can be mixed in the sweep gas in the concentration of 10 to 10,000 ppm. The flow diagram of the sweep gas system are shown in Fig.4.5.1.

With regard to the blanket mock-up, the dummy mock-up in which stainless steel pebbles and copper pebbles are loaded instead of tritium breeder pebbles and beryllium pebbles respectively was fabricated in order to confirm functions of the sweep gas system integrally.

Fig. 4.5.1
Flow diagram of sweep gas system



MM	: Moisture meter	P	: Pressure gauge
FC	: Flow controller	FI	: Flow indicator
CEC	: Ceramic electrolytic cell	OX	: Oxidation bed
IC	: Ionization chamber	MS	: Molecular sieve bed

2) Beryllium PIE facility

Outline of the beryllium PIE facility is shown in Fig.4.5.2. This facility consists of the five glove boxes, dry air supplier, tritium monitoring and removal system and storage box for neutron irradiated beryllium. Maximum handling amount is restricted by the amount of tritium; 7.4 GBq/day (200 mCi/day) and γ-emitting radionuclides; 7.4 MBq/day in ^{60}Co equivalence.

In this facility, tritium release apparatus and laser flush apparatus of thermal properties measurement had installed in the glove box No.1 (GB-1) and No.4 (GB-4). During fiscal this year, a mechanical test apparatus was installed in glove box No.5 (GB-5). Rreprocessing test apparatus will be installed in glove box No.2/3 (GB-2/3) in near future.

Apparatus for the mechanical test consists of tensile/compression machine, electric furnace, vacuum system and tritium removal system. It is possible to measure the mechanical properties such as the tensile strength (shape of specimen : $^{\phi}3\times^{l}56$ mm) and bending strength (shape of specimen :4x3x40 mm and pebbles : <$^{\phi}$3 mm) under the condition of R.T.~800°C in vacuum.

Each apparatus include the mechanical test are summarized in Table 4.5.1.

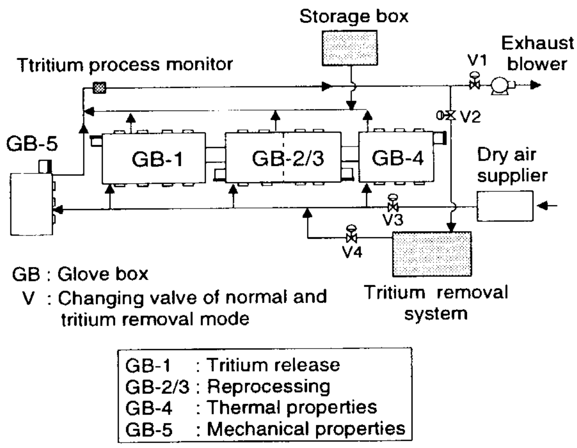


Fig.4.5.2 Beryllium PIE facility.

Table 4.5.1 Outline of Each apparatus in Beryllium PIE facility.

Apparatus	Test temperature	Test condition	Remarks
Tritium release	~1300°C	Helium sweep gas or vacuum	Measurement of gaseous tritiated water and tritium gas
Laser flush	R.T.~1200°C	Vacuum 2.7×10^{-5} Pa	Measurement of thermal diffusivity and heat capacity at 700 °C
Mechanical test	R.T.~800C	Vacuum 2.7×10^{-3} Pa at 800 °C	Tensile test Bending test Compression test

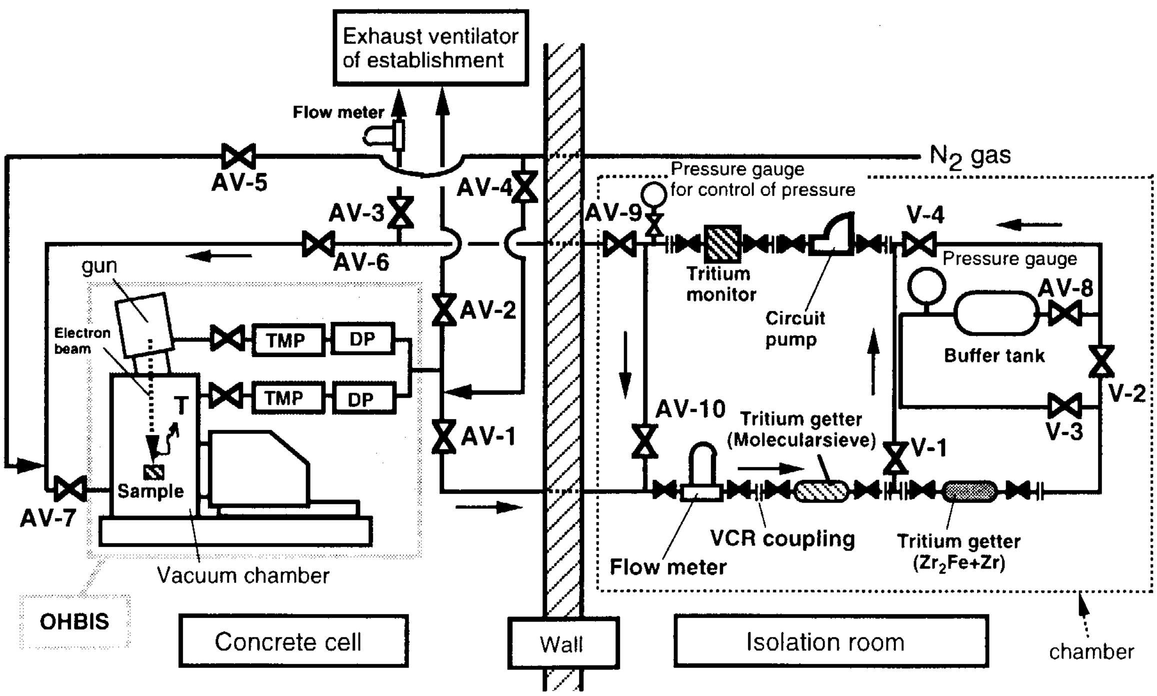
3) Electron beam heating facility

OHBIS (Oarai Hot-cell electron Beam Irradiation System) has been developed for thermal-shock test and steady state heat loading test of plasma facing materials to be used in fusion reactor (beryllium, carbon fiber composite, tungsten, etc.). Among these materials, especially, neutron irradiated beryllium will release tritium gas or tritium water during such thermal tests. Therefore, the tritium removal system was installed through OHBIS. The schematic diagram of the tritium removal equipment is shown in Fig.4.5.3.

The system has two kinds of tritium getters, i.e. Zr_2Fe + Zr getter for removal of tritium gas, and a molecular-sieve getter for removal of tritium water. Each getter has a capacity which can recover tritium of about 1 g. In the nomal operation, exhaust gas through these getters is released in the ventilator from vacuum pump in OHBIS.

On the other hand, a licence concerning tritium handling was got. From next year, heat loading tests using samples irradiated in JMTR will be conducted in OHBIS.

Fig.4.5.3 Schematic diagram of the tritium removal system.



Development of the instruments for in-pile functional test

It is necessary to develop a sweep gas sensor which can directly measure the tritium concentration in a blanket mock-up in order to obtain the engineering data for a design of a fusion blanket. So far, the sweep gas sensor with perovskite-type oxide was fabricated on a trial basis (see Fig.4.5.4), and verification tests such as an electromotive force test, responsibility test and a durability test were carried out. During this fiscal year, considering installation into the blanket mock-up, the smaller sized and highly sealed sweep gas sensor was studied and fabricated on trial. From the results of estimating the length of electrode so as not to affect the measurement and of estimating the heat affect zone in Au-brazing between a proton conductor and a protection tube, the length of proton conductor could be decreased from 60mm up to 40mm. Moreover, the diameter of protection tube could be reduced from ϕ18mm to ϕ13mm by placing the proton conductor eccentrically in the protection tube. The structure of newly fabricated sweep gas sensor is shown in Fig.4.5.5. After that, the verification tests with the newly fabricated sensor will be conducted including an electromotive force test with tritium gas.

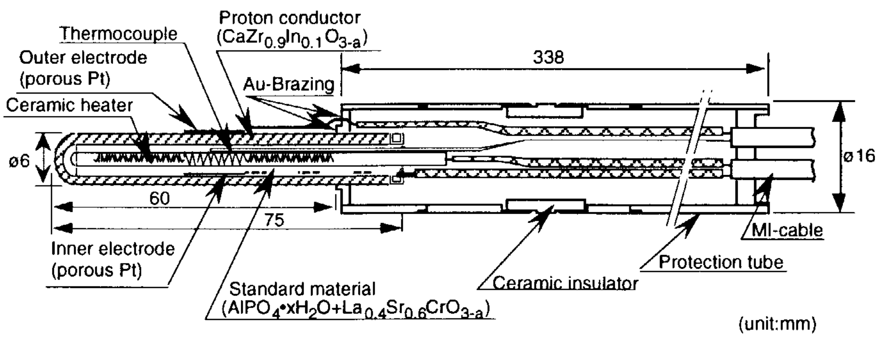


Fig.4.5.4
Structure of previous sweep gas sensor.

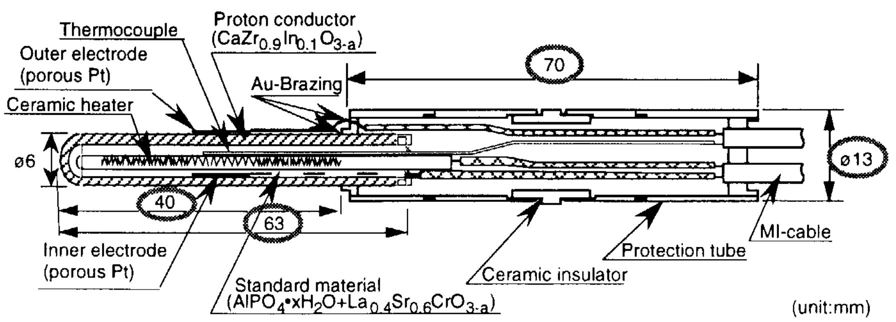


Fig.4.5.5
Structure of newly fabricated sweep gas sensor.

Research and development on tritium breeders

1) Fabrication development of lithium ceramics

Recently, lithium titanate (Li_2TiO_3) has attracted the attention of many researchers from a point of good tritium recovery at low temperature, chemical stability, etc.. As the shape of Li_2TiO_3, a small pebble is selected in Japanese design study of fusion blanket. As the fabrication method of Li_2TiO_3 pebbles, the sol-gel method is most advantageous from a view point of mass production, etc.. However, fabrication process of Li_2TiO_3 pebbles by sol-gel method was not established. Therefore, fabrication and characterization tests of Li_2TiO_3 pebbles by sol-gel method were performed, noting the aging temperatures in the fabrication process of the gel-spheres.

Resulted productivity were described as follows. The content of Li_2TiO_3 in gel-spheres increased with decreasing the aging temperature. However, crack was observed in the gel-spheres, when the aging temperature was below -40°C. Therefore, about -30°C was considered to the best temperature for aging processing. In the sintering tests, density of Li_2TiO_3 pebbles increased with increasing the sintering temperature and sintering time. Density of Li_2TiO_3 pebbles became about 81% at 1400°C for 4 h.

The results of characterization tests were described as follows. Sphericity and crushing strength of Li_2TiO_3 pebbles were 1.11±0.07 and 4.7±0.9 kgf, respectively. And only X-ray diffraction peaks corresponding to Li_2TiO_3 appeared from the Li_2TiO_3 pebbles by XRD analysis. Summary of fabrication tests of Li_2TiO_3 pebbles is shown in Table 4.5.2. As the results of these tests, excellent prospects were obtained concerning the fabrication of Li_2TiO_3 pebbles with a target density (80-85%T.D.) by sol-gel method.

Table 4.5.2
Summary of fabrication tests of Li_2TiO_3 pebbles

Items \ Process	Mixture Ratio	Aging Temperature	Calcination Temperature	Sintering Time	Sintering Temperature	Density
Preliminary Fabrication Test	Li_2TiO_3:43wt% PVA :4wt%	0°C	650°C 6 h	10 h	1000°C	40%T.D.
1st Improvement Test	Li_2TiO_3:46wt% PVA :4wt%	0°C	650°C 6 h	4 h	1150°C	60%T.D.
2nd Improvement Test	Li_2TiO_3:46wt% PVA :4wt%	-20°C	650°C 6 h	4 h	1400°C	76%T.D.
3rd Improvement Test	Li_2TiO_3:46wt% PVA :4wt%	-30°C	650°C 6 h	4 h	1400°C	81%T.D.

2) Characterization of lithium ceramics

The characterization tests of Li_2TiO_3 have been performed for the material selection and fusion blanket design. Three kinds of Li_2TiO_3 pellets with different density, i.e., 73%T.D., 83%T.D. and 93%T.D., were prepared, and density dependence on thermal conductivity, specific heat and thermal expansion, were investigated. Temperature dependence on thermal conductivity of Li_2TiO_3 pelletsis shown in Fig.4.5.6. Experimental equations on specific heat, thermal conductivity and thermal expansion are shown in Table 4.5.3. Density dependence on thermal conductivity was expressed by the modified Maxwel-Eucken equation. Thermal hysteresis of Li_2TiO_3 was not observed at heating and cooling cycle in thermal diffusivity and thermal expansion, and Li_2TiO_3 may be stable to temperature between room temperature and 1100K.

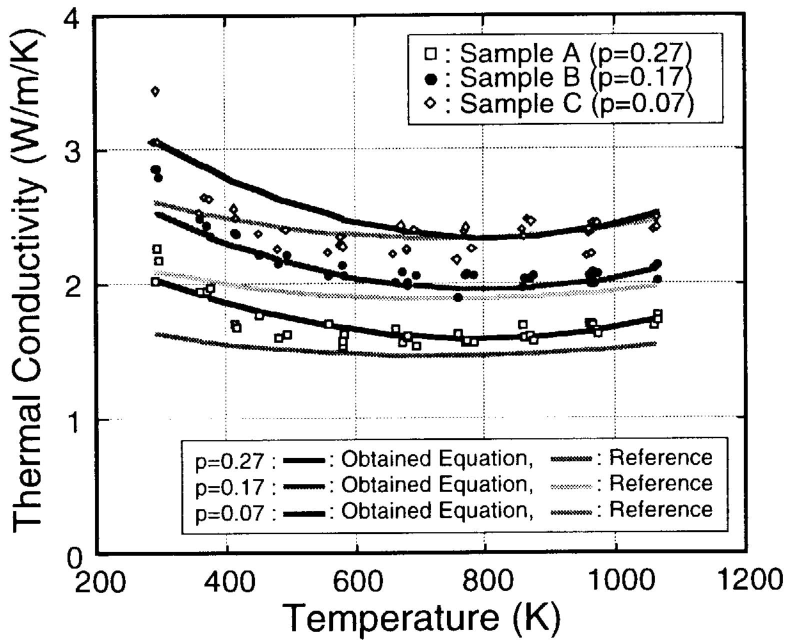


Fig.4.5.6
Temperature dependence on thermal conductivity

Specific Heat
$C_p(J/g/K) = 0.73 + 8.44 \times 10^{-4} T - 1.67 \times 10^{-7} T^{-2}$
Thermal conductivity
$k = \frac{(1-p)}{(1+\beta p)}\left(4.77 - 5.11 \times 10^{-3} T + 3.12 \times 10^{-6} T^2\right)$
$\beta = 1.06 - 2.88 \times 10^{-4} T$
Thermal Expansion
$L = -0.578 + 1.71 \times 10^{-3} T + 1.867 \times 10^{-7} T^2$

Table 4.5.3
Experimental equations on specific heat, thermal conductivity and thermal expansion

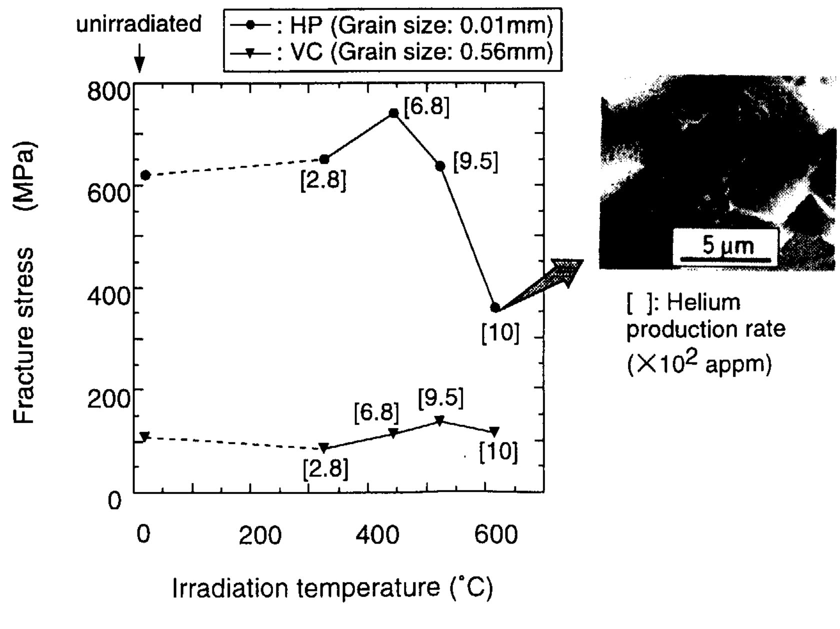
Research and development on neutron multiplier materials

1) Characterization of beryllium

For the neutron multiplier, production technology of beryllium pebbles are almost complete by using the rotating electrode method, and the evaluation of characteristics under the neutron irradiation and reprocessing is carrying out.

In this year, mechanical test was carried out. Fracture stress of the two kinds for beryllium bending specimens (4x3x40 mm) which produced by the Hot-Press (HP, grain size: 0.01 mm, purity: 99.1 %) and Vacuum Cast (VC, grain size: 0.56 mm, purity: 99.8 %) method were measured by the mechanical test apparatus. Irradiation conditions of the beryllium bending specimens were the total fast neutron fluence of 1.3~4.3×10^{21} n/cm^2 (E>1 MeV) at 327~616°C. Fracture stress of beryllium bending specimens as a function of irradiated temperature is shown in Fig.4.5.7. Obvious decreasing of the fracture stress was observed in high temperature irradiated HP specimen (4.3×10^{21} n/cm^2, 616°C, swelling: 3.3 $\Delta V/V$ %). However, decreasing of the fracture stress was not observed in VC specimens (1.1 $\Delta V/V$ %) although same irradiation conditions with HP specimens. SEM photographs of fracture surface for each specimens are shown in same figure. Helium bubbles were observed on grain boundary, and larger grains kept on larger helium bubbles on the grain boundaries. Decreasing of the fracture stress was presumably caused by helium bubbles on grain boundaries because fracture mechanism changed to intergranular fracture for the high-temperature irradiated specimens from intragranular fracture for the low-ltemperature irradiated specimens.These data suggest dependence of the grain size by neutron irradiation. However, impurities and other effects must be also studied in details.

Fig.4.5.7
Results of bending specimens.



Technology Development Related to the Fusion Blanket

1) Development of joint technology

Al_2O_3-dispersed copper alloys are attractive candidate and exhibit the best combination of swelling resistance, retention of strength, and electrical conductivity of any copper alloy examined to data. Recently, joint technology development of Al_2O_3-dispersed copper alloys and stainless steel (SS316 (JIS : SUS316)) have been carried out for the solid state diffusion bonding, the brazing joints, and so on. A friction welding is one of most popular welding method and can be applied to joints of this copper alloy to SS316. In the present work, the effects of neutron irradiation on mechanical properties of the Cu-alloys/SS316 joints fabricated by the friction welding are described in this section.

The Cu-0.3%Al_2O_3 alloy (Al-15) and oxygen free copper (OF-Cu) were prepared as Al_2O_3-dispersed copper alloys and the reference copper, respectively. The Al-15/SS316 and OF-Cu/SS316 joints were irradiated in Japan Materials Testing Reactor (JMTR). The fast neutron fluence (>1MeV) was maximum 9.3×10^{20} n/cm^2 and the irradiation temperatures were 598K, 693K and 773K. After irradiation, tensile test, hardness test and metallographical observation of the Al-15/SS316 and OF-Cu/SS316 joints have been carried out and metallographical observation has been performed as the post irradiation examination (PIE).

Tensile test were carried out at room temperature and each irradiation temperature. Results of tensile tests on Al-15/SS316 joints is shown in Fig.4.5.8. In the tensile tests of Al-15/SS316 and OF-Cu/SS316 joints, all irradiated specimens have broken at the copper sides. Yield strength and tensile strength of the Al-15/SS316 joints irradiated at 693K were equal to those of the base material (Al-15) irradiated in the same condition.

From these results, the effect of neutron irradiation on mechanical properties of the Al-15/SS316 and OF-Cu/SS316 joints fabricated by the friction welding was discussed for the cooling tubes of fusion blanket.

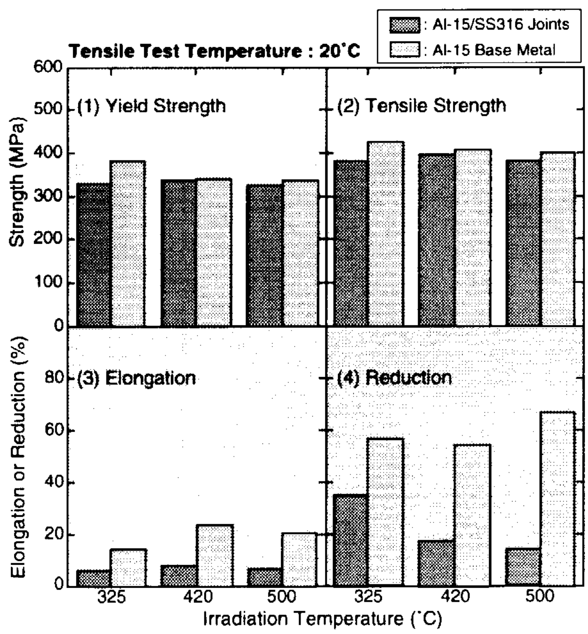


Fig.4.5.8
Results of tensile tests on Al-15/SS316 joints

2) Reweldability test of irradiated materials by TIG welding method

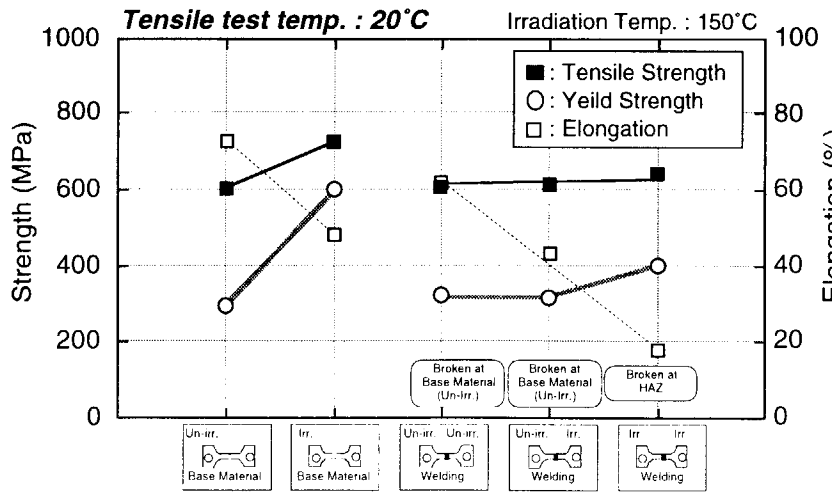
Rewelding of irradiated materials has large impact on the design and the maintenance scheme of in-vessel components for a fusion reactor. Recently, joining technology with irradiated structural materials such as stainless steels and inconel alloys (Inconel 625) was investigated in the way of tungsten inert gas (TIG) welding, laser welding and so on.

On the other hand, helium is one of the most prominent transmutation products generated in these materials because of the high cross section for (n, α) nuclear reactions of high energy neutrons of fusion reaction. And helium is essentially insoluble in metals. The generation of helium is known to degrade the properties of materials. The formation of grain boundary (GB) bubbles can ultimately lead to drastic changes in microscopic properties , including severe embrittlement at elevated temperature. At high temperature, these bubbles will grow under the influence of stress and temperature. Welding processes produce internal stresses and elevated temperatures. In this study, systematic evaluation of weldments was not been performed.

Weldments of un-irradiated and/or irradiated type 316 stainless steel (SS316LN) were performed by the TIG welding method, and mechanical properties of their weldments were systematically evaluated. Results of tensile tests on rewelding joints of un-irradiated and/ or irradiated stainless steel is shown in Fig.4.5.9. From this results, weldments of unirradiated/unirradiated and irradiated/unirradiated SS316LN broke at the part of the unirradiated base materials. On the other hand, weldments of irradiated/irradiated SS316 broke at the part of heat affect zone (HAZ).

In the future, hardness, metallographical observation and SEM/XMA analysis will be carried out. Additionally, re-irradiation effects of these weldments must be investigated.

Fig.4.5.9
Results of tensile tests on rewelding joints of un-irradiated and/or irradiated stainless steel



3) Development of ceramic coating technology

Ceramic coating on the surface of structural materials such as SS316 have been considered for an electrical insulator and tritium permeation barrier in fusion reactor. Y_2O_3 is one of most promising materials as coating from a point of high electrical resistivity, etc. And SS410 was selected as the undercoating for preventing the crack and peeling of coating. The undercoating of SS410 was coated about 160μm thick by the vacuum plasma spraying method. The Y_2O_3 (purity : 99.95wt%) was coated about 50μm thick by the atmospheric plasma spraying method. The Y_2O_3 coating was densified by impregnation with $Y(NO_3)_3$ solution and fired in order to close open pores. Neutron irradiation specimen of SS316 with Y_2O_3 coating was irradiated at 300°C in He atmosphere.

The results of in-situ measurement of electrical resistivity on Y_2O_3 coating is shown in Fig.4.5.10. From the results of in-situ measurement of electrical resistivity on Y_2O_3 coating, electrical resistivity of Y_2O_3 coating was decreased with JMTR power at power-up. This phenomenon was named as the Radiation Induced Conductivity (RIC). Electrical resistivity of Y_2O_3 coating before neutron irradiation and at JMTR operation were about $1 \times 10^{12}\Omega\cdot cm$ and about $1 \times 10^{9}\Omega\cdot cm$, respectively. Electrical resistivity of Y_2O_3 coating was constant under neutron irradiation at JMTR operation. Electrical resistivity of Y_2O_3 coating after neutron irradiation was recovered that before neutron irradiation. The Radiation Induced Electrical Degradation (RIED) was not recognized for Y_2O_3 coating. It was clear that the Y_2O_3 coating had good electrical resistivity under neutron irradiation up to about $6 \times 10^{20} n/cm^2$(>1MeV).

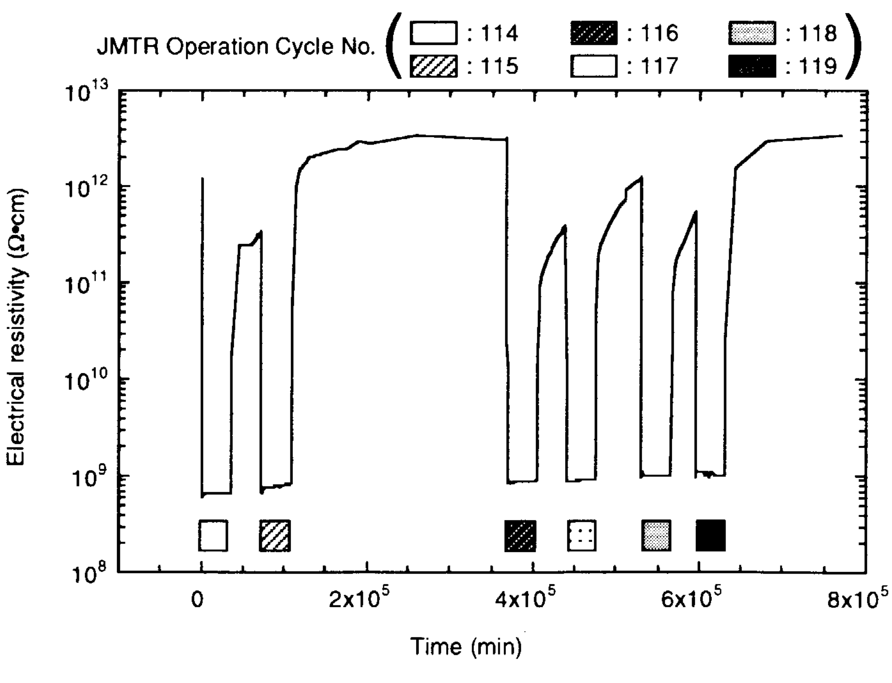


Fig.4.5.10 Results of in-situ measurement of electrical resistivity

4) Irradaition tests of plasma facing materials

Since plasma facing components such as the first wall and the divertor are exposed to high heat load and high neutron flux generated by the plasma, it is urgent to develop plasma facing components which stand them.

The electron beam heating facility ("OHBIS", Oarai Hot-cell electron Beam Irradiating System) had been established in a hot cell in JMTR hot laboratory in order to estimate thermal shock resistivity of plasma facing materials and heat removal capabilities of divertor element under steady state heat load. In this facility, un-irradiated / irradiated plasma facing materials (beryllium, carbon based materials and so on) and divertor elements can be treated. These samples are irradiated by electron beam with the maximum power of 50kW, and exposure time of the beam is not less than 0.1ms.

Thermal shock test has been started in OHBIS from this year. At first heat flux which was generated by irradiation of electron beam, was estimated. As a result of the test, it is appeared that the heat flux which are absorbed into samples, are different between the kind of materials (see Fig.4.5.11). And heat flux profile which was measured by beryllium chip calorimeter, is shown in Fig.4.5.12. Half band width of the profile was about 3 - 4mm diameter.

Fig.4.5.11
Heat flux vs. beam current

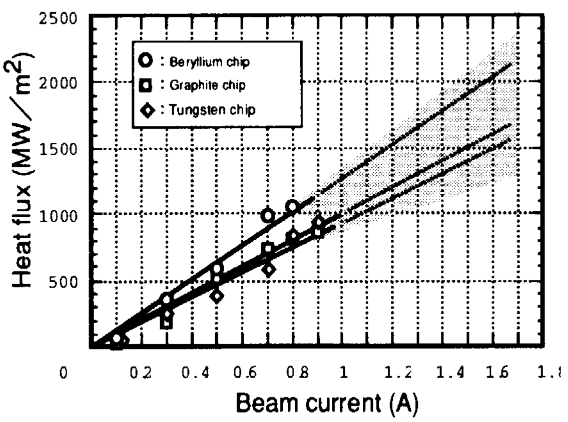
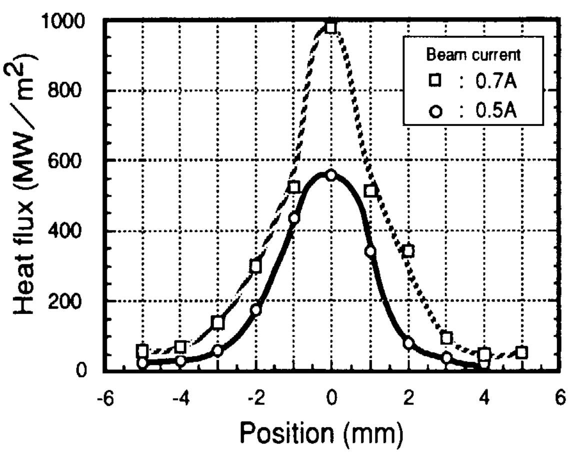


Fig.4.5.12
Profile of electron beam (Beryllium)



4.6. Development of Apparatus for PIE

Test piece manufacturing from irradiated components

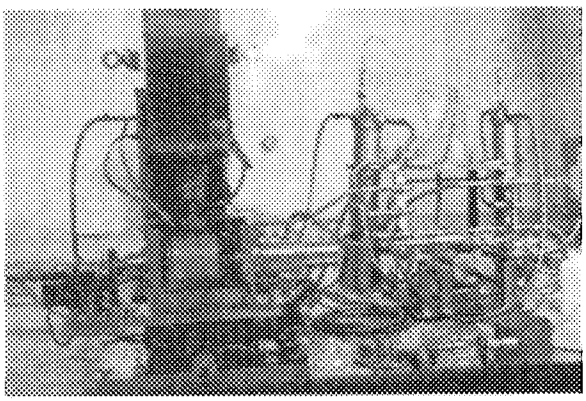
In the case of materials irradiation studies in the JMTR, the specimens machined to test pieces are usually irradiated in helium atmosphere in a capsule. After these specimens have been taken out from the capsule in a hot cell, they are examined by using various kinds of mechanical testing apparatus. Recent irradiation studies, such as IASCC (Irradiation Assisted Stress Corrosion Cracking) test of reactor core structural materials, require the irradiation of long term in reactor to obtain the fast neutron fluence of 10^{21} ~10^{22} n/cm^2.

On the other hand in the JMTR, the outer tubes of capsule used for long term irradiation and used in-pile loops are stored under the condition finished the missions. Some of these components can be estimated to have been irradiated up to 1 ~9 x 10^{21} n/cm^2. It had been planned to take out the test pieces of mechanical testing from these components. According to the purpose, a pipe cutter and an electro-discharge machining apparatus were developed to remotely machine to test pieces, and are being installed in a hot cell.

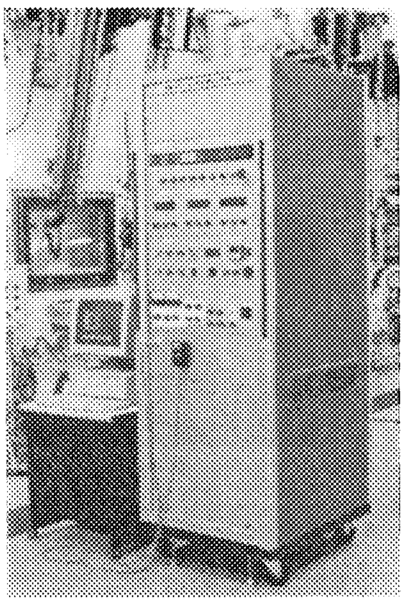
An in-pile tube (SUS 316 : max. diameter 145 mm, thickness 4.8 mm) was already cut to 1650 mm long by the other cutter in reactor canal. This tube will be cut to about 100 mm long by the pipe cutter. This cutter was improved from commercial guillotine saw pipe cutter to be handled with remote control especially on the pipe chucking mechanism, the saw exchange mechanism and the waste collecting mechanism.

The electro-discharge machining apparatus is used to machine from above 100 mm long tube to the test pieces of tensile test (plate and rod), impact test and compact tension test. Photographs of the electro-discharge machining apparatus are shown in Fig4.6.1. This apparatus has not only numerical control machining function but a compact body and the design considered easy to maintain. Test piece is brewed water during machining and this water is circulated through filters. After machining by this apparatus, test piece is polished to 10 μm in smoothness by air grinder which is replaced with machining electrode.

Fig.4.6.1
Electro-discharge machining apparatus



Main body in a hot cell



Control panel in operation room

Fatigue testing apparatus

Fatigue characteristics of reactor structural material at high temperature are necessary to be evaluated for ensuring the safety of the advanced thermal reactor (ATR). Especially, the high temperature test data on safety research such as low cycle fatigue property and crack propagation property for reactor pressure vessel are important for the development of the ATR. Responding to these needs, a remote controlled type fatigue testing machine has been developed and installed in a hot cell to get the fatigue data of irradiated materials.

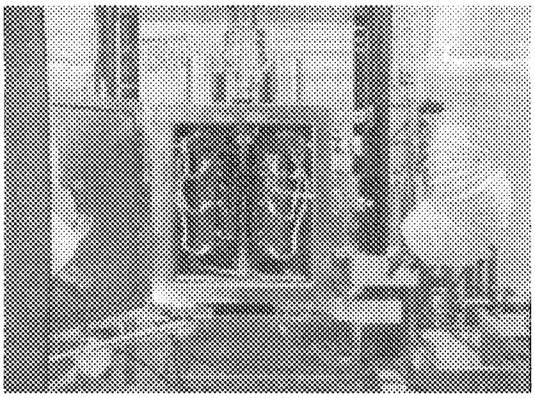
The machine was developed modifying a commercially by available mechanical servo type fatigue testing machine to withstand radiation and be remotely operated.

The fatigue testing machine is shown in Fig.4.6.2 .

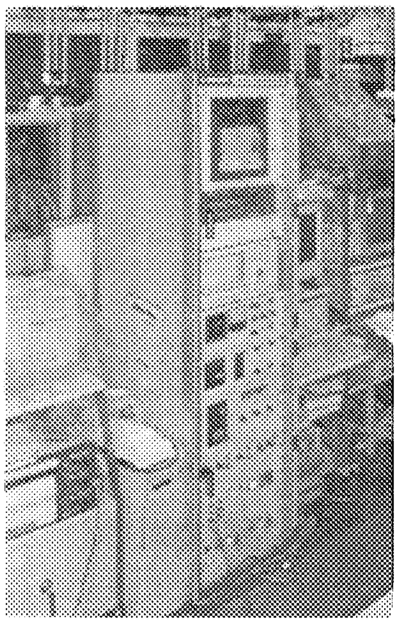
The major specifications are as follows:

Type of testing machine	: Electro-mechanical servo type
Load capacity	: 10 kN
Stroke	: 70 mm
Frequency	: 1.25 ～ 1.25^{-5} Hz
Temperature	: 350 ～ 750℃
Environment	: Vacuum (Max. 1.3 x 10^{-3} Pa) or inert gas

Fig.4.6.2
Fatigue testing
Apparatus



Fatigue testing machine



Operation console

Small specimen test techniques

An accelerator-driven deuterium-lithium (d-Li) stripping reaction-type neutron source, such as the International Fusion Materials Irradiation Facility (IFMIF) planned by the International Atomic Energy Organization is recognized as one of the most promising facility to obtain test environments of high-energy neutrons for fusion reactor materials development. The limitation on the available irradiation volume of the irradiation facility requires the development of the small specimen test techniques (SSTT). Application of SSTT to evaluate the degradation of various components in the light water reactor for the life extension is expected to be also quite beneficial.

A testing machine for the Small Punch (SP) and miniaturized tensile tests was developed at the hot laboratory of the Japan Materials Testing Reactor (JMTR). The machine is designed for testing at temperatures ranging between 93 and 1123 K to evaluate the temperature dependence of the strength of materials including the embrittlement at low temperatures and the softening at elevated temperatures. The tests are performed in a vacuum or in an inert gas environment.

Fig.4.6.3 shows the SP testing machine installed in a hot cell, a rig for the tensile test of small specimens, schematic of the tensile test for the small specimen and the configuration of a small tensile specimen.

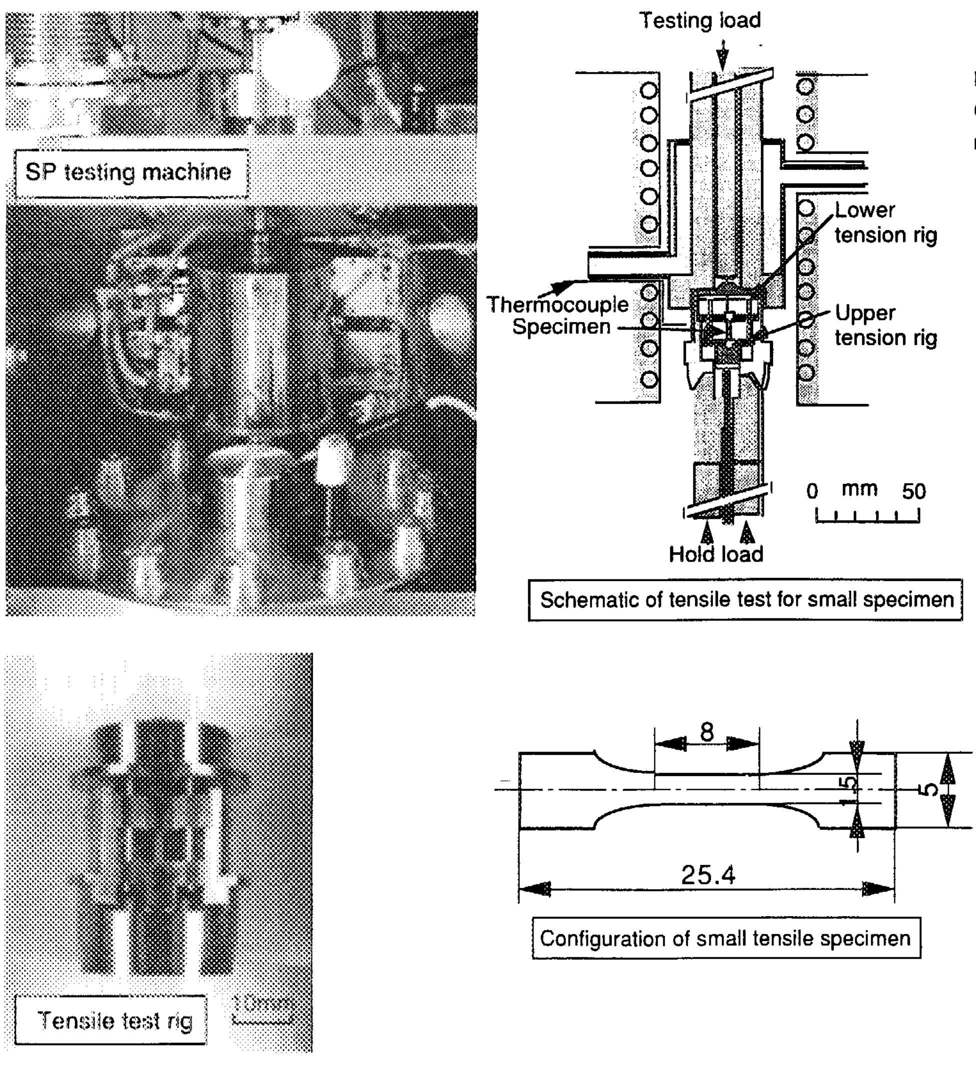


Fig.4.6.3 Outline of small specimen test techniques

5. Summary

This report describes the activities in the JMTR performed in the fiscal year of 1996. JMTR was operated for 2 cycles. During the annual maintenance, renewal of beryllium frame and renewal of HR-2 were performed.

The irradiation service was routinely conducted mainly for satisfying the requirements from the laboratories inside JAERI. Irradiations for RI production were also performed.

R&D works on irradiation and PIE technologies have been extensively performed. They are:

- Power ramping tests for LWR fuels.
- The assessment on the irradiation damage of an in-pile tube of OSF-1 loop.
- Improvement of the re-instrumentation technique adopting a dual (FP gas pressure gauge and thermocouple) re-instrumentation technique for BOCA.
- Development of remote controlled testing apparatus for characterization of irradiated materials.
- Irradiation facility for fusion reactor development installed in JMTR.

6. Publications

Reports

1. Jan.-97, "Development of a Remote Controlled Small Punch Testing Apparatus", M. Oumi, et al., Journal of the Atomic Energy Society of Japan.
2. Apr.-97, "Date Base for Tritium Breeding Blanket of Fusion Reactor (1) - Li_2O Solid Breeding Material -", H. Kawamura, et al., Annual Report of Hydrogen Isotope Research Center, Toyama University.
3. May-96 , "Dose Evaluation of External Exposure by Direct and Skyshine Gamma Rays in Postulated Accident of Nuclear Fuel Handling Facilities at JMTR", N. Tsuchida, et al., JAERI-Tech 96-020.
4. Jan.-97, " Report of Minutes of the Workshop of In-pile Irradiation Tests Department of JMTR", JAERI-Conf 97-006.
5. Feb.-97, "A Proposal of Revised Method for Determination of Large Positive Reactivity", Y. Kaneko, et al., JAERI-Research 97-003.
6. May-97, "Neutron Irradiation Characteristic Tests of Oxygen Sensors Using Zirconia Solid Electrolyte", N. Hiura, et al., JAERI-Research 97-028.
7. Mar.-97, "Annual Report of JMTR, No.10 Department of JMTR", JAERI-Review 97-006.

Contributions to conferences

1. May-96, " Irradiation-coupling Techniques with Different Neutron Spectra of MTR/FBR or MTR/LWR Using JMTR Capsule", Y. Matsui, et al., 5th ASRR (Korea).
2. May-96, "Development of JMTR Irradiation Rigs for Improved Irradiations with Cyclic Temperature, Neutron Flux and Neutron Fluence Variations", T. Yamaura, et al., 5th ASRR (Korea).
3. May-96, " Status and Future Plan of Utilization in JMTR", N. Tsuchida, et al., 5th ASRR (Korea).
4. Jul.-96, "Statues of Fusion Blanket Irradiation Study in JAERI", H. Kawamura, et al., Int. Tritium Workshop on Present Status and Prospect of Tritium-material Interaction Studies.
5. Aug.-96, "Irradiation Study for Fusion Blanket Development with JMTR", H. Kawamura, et al., Int. Workshop on interfacial Effects in Quantum Eng..
6. Aug.-96, "Characterization of Y_2O_3 Coating for Liquid Blanket", M. Nakamichi, et al., Int. Workshop on interfacial Effects in Quantum Eng.
7. Sep.-96, "Beryllium Irradiation Study with JMTR", E. Ishitsuka, et al., 4th Int. Symp. on Fusion Nuclear Technology.
8. Sep.-96, "Modifications of Electron Beam Facility (OHBIS) for Irradiated Divertor Element with Cooling Water Channel", N. Sakamoto, et al.., 4th Int. Symp. on Fusion Nuclear Technology.
9. Sep.-96, "Reactivity between Beryllium and Copper Alloys", N. Sakamoto, et al., 4th Int. Symp. on Fusion Nuclear Technology.
10. Sep.-96, "Dissolving and Recovering Properties for Reprocessing Technology of Lithium Titanate", K. Tsuchiya, et al., 4th Int. Symp. on Fusion Nuclear Technology.
11. Sep.-96, "Trial Fabrication of Tritium Breeders for Fusion Blanket with Lithium Recovered from Seawater", K. Tsuchiya, et al., 4th Int. Symp. on Fusion Nuclear Technology.

12. Sep.-96, "Preliminary Nuclear Design of Pulse Operation Simulating Mock-up for In-pile Functional Test of Fusion Blanket", Y. Nagao, et al., 4th Int. Symp. on Fusion Nuclear Technology.
13. Sep.-96, "Material Design of Ceramic Coating by Plasma Spray Method", M. Nakamichi, et al., 4th Int. Symp. on Fusion Nuclear Technology.
14. Sep.-96, "Thermal Properties of Neutron Irradiated Beryllium", E. Ishitsuka, et al., 5th Int. Workshop on Ceramic Breeder Blanket Interactions (Italy).
15. Sep.-96, "Status of Fusion Blanket Irradiation in JAERI", H. Kawamura, et al., 5th Int. Workshop on Ceramic Breeder Blanket Interactions (Italy).
16. Sep.-96, "Fabrication Development of Ceramic Tritium Breeders by Sol-gel Method", K. Tsuchiya, et al., 5th Int. Workshop on Ceramic Breeder Blanket Interactions (Italy).
17. Sep.-96, "Reactivity Test between Beryllium and Copper Alloys", N. Sakamoto, et al., 5th Int. Workshop on Ceramic Breeder Blanket Interactions (Italy).
18. Sep.-96, "Materials Testing Reactor and Irradiation Techniques - Performance and Future Prospect of JMTR in JAERI - ", I. Kondo, et al., AESJ.
19. Sep.-96, "Neutron Flux Evaluation of Pulse Operation Mock-up for Fusion Reactor (2) -Preliminary Nuclear Design- ", Y. Nagao, et al., AESJ.
20. Sep.-96, "Characterization of Temperature under Neutron Irradiation of Self-Powered Neutron Detector", M. Nakamichi, et al, AESJ.
21. Sep.-96, "Development of Irradiation Technique on the Controlled Helium Generation -Development of the Creep Capsule Irradiated under Tailored Neutron Spectrum-", Y. Itabashi, et al., AESJ.
22. Sep.-96, "Development of Irradiation-coupling Techniques -Design of the Coupling Capsule with Instrumentation in JMTR-", Y. Matsui, et al., AESJ.
23. Sep.-96, "Observation of Optical Luminescence from Sapphire Crystal during Irradiation in JMTR", T. Sagawa, et al., AESJ.
24. Sep.-96, "Development of Welding and Specimen Machining Technologies for Irradiated Materials", M. Shimizu, et al., AESJ.
25. Sep.-96, "High Efficiency of ZrNi Alloy for Its Tritium Gettering Properties (1) -Equilibrium Pressure Properties-", K. Tsuchiya, et al., AESJ.
26. Sep.-96, "High Efficiency of ZrNi Alloy for Its Tritium Gettering Properties (2) -Breakthrough Properties of the Particle Packed Bed-", K. Tsuchiya, et al., AESJ.
27. Sep.-96, "Calculation of the JMTR Core with Monte Carlo Code MCNP", Y. Nagao, et al., AESJ.
28. Sep.-96, "Preliminary Fabrication Test for Li_2TiO_3 Pebbles by Wet Process", K. Tsuchiya, et al., AESJ.
29. Sep.-96, "Development of Functionally Gradient Materials of Beryllium and Copper (2) -Measurement of thermal Expansion Coefficient of Mixed Powder-", S. Saito, et al., AESJ.
30. Sep.-96, "Development of Irradiation Technique with Extraordinary Precision in Pressurized Water -Irradiation Test using the Hybrid Type Saturated Temperature Capsule in JMTR-", M. Niimi, et al., AESJ.
31. Sep.-96, "Characterization of Self-powered Neutron Detector at High Temperature under Neutron Irradiation", M. Nakamichi, et al., 19th Symp. on Fusion Technol. (Portugal).
32. Sep.-96, "Reactivity Test between Beryllium and Dispersion Strengthened Copper", N. Sakamoto, et al., 19th Symp. on Fusion Technol. (Portugal).

33. Sep.-96, "Reprocessing Technology Development for Irradiated Beryllium", H. Kawamura, et al., 19th Symp. on Fusion Technol. (Portugal).
34. Sep.-96, "The Effect of Neutron Irradiation on Mechanical Properties of Nb-1%Zr/SS304 Joints Fabricated by Friction Welding", K. Tsuchiya, et al., 19th Symp. on Fusion Technol. (Portugal).
35. Sep.-96, "Breakthrough Properties of Hydrogen with Zr_9Ni_{11} Particle Packed Bed", K. Tsuchiya, et al., 19th Symp. on Fusion Technol. (Portugal).
36. Sep.-96, "Design Study of In-pile Blanket Mockup Simulated Neutron Pulse Operation of Fusion Reactor", M. Nakamichi, et al., 19th Symp. on Fusion Technol. (Portugal).
37. Sep.-96, "Thermal Properties of Neutron Irradiated Beryllium", E. Ishitsuka, et al., 19th Symp. on Fusion Technol. (Portugal).
38. Sep.-96, "Density Improvement of Li_2O Pebble Fabricated by Sol-gel Method", K. Tsuchiya, et al., 19th Symp. on Fusion Technol. (Portugal).
39. Sep.-96, "The Core Characteristics Evaluation of JMTR with Low-enriched Uranium Fuel", S. Shimakawa, et al., Int. Conf. on the Phys. of Reactors ; Physor 96.
40. Sep.-96, "Neutron Dosimetry for Materials Irradiation Tests in JMTR", S. Shimakawa, et al., 9th Int. Symp. on Reactor Dosimetry (Czechoslovakia).
41. Oct.-96, "Preliminary Characterization of Interlayer for Be/Cu Functionally Gradient materials - Thermophysical Properties of Be/Cu Sintered Compacts -", S. Saito, et al., FMG '96 4th Int. Symp. on Functionally Graded Materials.
42. Mar.-97, "Application of Correction Factor Method on Shutdown Margin Determination for JMTR", S. Shimakawa, et al., AESJ.
43. Mar.-97, "Neutron Irradiation Behavior of Beryllium Pebbles -Change of Microstructure and Mechanical Properties-", E. Ishitsuka, et al., AESJ.
44. Mar.-97, "Evaluation on Thermal Properties of Ceramic Tritium Breeders", S. Saito, et al., AESJ.
45. Mar.-97, "Development of Joining Technique between Sapphire and SUS316 by HIP Method (1) -Fabrication Trial Test on Sapphire/Nb1%Zr Joint-", N. Sakamoto, et al., AESJ.
46. Mar.-97, "Preliminary Characterization of Ceramic Coating as Tritium Permeation Barrier", M. Nakamichi, et al., AESJ.

7. Organization

- ***Oarai Research Establishment***
  - **Department of JMTR**
    - JMTR Administration Division — Administrative services of JMTR utilizations
    - Project Engineering Division — Planning for operation and utilization of JMTR; Reactor physics for JMTR utilization
    - Reactor Division I — Reactor operation; Fuel management
    - Reactor Division II — Reactor maintenance
    - Irradiation Division I — Design and fabrication of irradiation devices; R/D of irradiation devices and instrumentation
    - Irradiation Division II — Operation of irradiation facilities; Maintenance of irradiation facilities
    - Engineering Testing Division — Capsule inspection; Water and gas analysis
    - Hot Laboratory — Post-irradiation examinations; R/D of post-irradiation examination technique
    - Blanket Irradiation and Analysis Laboratory — Irradiation study on blanket materials for fusion reactor; Irradiation study on fusion reactor components
  - **Department of Administrative Services**
  - **Department of HTTR Project**

国際単位系(SI)と換算表

表1　SI基本単位および補助単位

量	名　称	記　号
長　さ	メートル	m
質　量	キログラム	kg
時　間	秒	s
電　流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光　度	カンデラ	cd
平面角	ラジアン	rad
立体角	ステラジアン	sr

表2　SIと併用される単位

名　称	記　号
分，時，日	min，h，d
度，分，秒	°，′，″
リットル	l，L
トン	t
電子ボルト	eV
原子質量単位	u

1 eV = 1.60218×10^{-19} J
1 u = 1.66054×10^{-27} kg

表5　SI接頭語

倍数	接頭語	記　号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

表3　固有の名称をもつSI組立単位

量	名　称	記号	他のSI単位による表現
周波数	ヘルツ	Hz	s^{-1}
力	ニュートン	N	$m \cdot kg/s^2$
圧力，応力	パスカル	Pa	N/m^2
エネルギー，仕事，熱量	ジュール	J	N·m
工率，放射束	ワット	W	J/s
電気量，電荷	クーロン	C	A·s
電位，電圧，起電力	ボルト	V	W/A
静電容量	ファラド	F	C/V
電気抵抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁束	ウェーバ	Wb	V·s
磁束密度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	℃	
光束	ルーメン	lm	cd·sr
照度	ルクス	lx	lm/m^2
放射能	ベクレル	Bq	s^{-1}
吸収線量	グレイ	Gy	J/kg
線量当量	シーベルト	Sv	J/kg

表4　SIと共に暫定的に維持される単位

名　称	記　号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

1 Å = 0.1 nm = 10^{-10} m
1 b = 100 fm^2 = 10^{-28} m^2
1 bar = 0.1 MPa = 10^5 Pa
1 Gal = 1 cm/s^2 = 10^{-2} m/s^2
1 Ci = 3.7×10^{10} Bq
1 R = 2.58×10^{-4} C/kg
1 rad = 1 cGy = 10^{-2} Gy
1 rem = 1 cSv = 10^{-2} Sv

(注)

1. 表1－5は「国際単位系」第5版，国際度量衡局 1985年刊行による。ただし，1 eVおよび1 uの値はCODATAの1986年推奨値によった。
2. 表4には海里，ノット，アール，ヘクタールも含まれているが日常の単位なのでここでは省略した。
3. barは，JISでは流体の圧力を表わす場合に限り表2のカテゴリーに分類されている。
4. EC閣僚理事会指令ではbar，barnおよび「血圧の単位」mmHgを表2のカテゴリーに入れている。

換　算　表

力	N($=10^5$dyn)	kgf	lbf
	1	0.101972	0.224809
	9.80665	1	2.20462
	4.44822	0.453592	1

粘　度　1 Pa·s(N·s/m^2) = 10 P(ポアズ)(g/(cm·s))

動粘度　1 m^2/s = 10^4St(ストークス)(cm^2/s)

圧力	MPa(=10 bar)	kgf/cm^2	atm	mmHg(Torr)	lbf/in^2(psi)
	1	10.1972	9.86923	7.50062×10^3	145.038
	0.0980665	1	0.967841	735.559	14.2233
	0.101325	1.03323	1	760	14.6959
	1.33322×10^{-4}	1.35951×10^{-3}	1.31579×10^{-3}	1	1.93368×10^{-2}
	6.89476×10^{-3}	7.03070×10^{-2}	6.80460×10^{-2}	51.7149	1

エネルギー・仕事・熱量	J($=10^7$erg)	kgf·m	kW·h	cal(計量法)	Btu	ft·lbf	eV
	1	0.101972	2.77778×10^{-7}	0.238889	9.47813×10^{-4}	0.737562	6.24150×10^{18}
	9.80665	1	2.72407×10^{-6}	2.34270	9.29487×10^{-3}	7.23301	6.12082×10^{19}
	3.6×10^6	3.67098×10^5	1	8.59999×10^5	3412.13	2.65522×10^6	2.24694×10^{25}
	4.18605	0.426858	1.16279×10^{-6}	1	3.96759×10^{-3}	3.08747	2.61272×10^{19}
	1055.06	107.586	2.93072×10^{-4}	252.042	1	778.172	6.58515×10^{21}
	1.35582	0.138255	3.76616×10^{-7}	0.323890	1.28506×10^{-3}	1	8.46233×10^{18}
	1.60218×10^{-19}	1.63377×10^{-20}	4.45050×10^{-26}	3.82743×10^{-20}	1.51857×10^{-22}	1.18171×10^{-19}	1

1 cal = 4.18605 J (計量法)
= 4.184 J (熱化学)
= 4.1855 J (15 °C)
= 4.1868 J (国際蒸気表)

仕事率　1 PS (仏馬力)
= 75 kgf·m/s
= 735.499 W

放射能	Bq	Ci
	1	2.70270×10^{-11}
	3.7×10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58×10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

(86年12月26日現在)

ANNUAL REPORT OF JMTR, FY1996 (APRIL 1, 1996 - MARCH 31, 1997)