PROFILES SUMMARY

The 30 profiles on the CD-ROM accompanying this publication provide a technical description of the research reactors, including their specific features for utilization (see Table 1). The profiles describe research reactors according to the template which was agreed among a broad group of international experts contributing to this publication. Most of the profiles were provided to the IAEA by Member State institutions in response to requests to complete the template. Although some profiles do not follow exactly the requested contents and format, they have been included for completeness.

The profile remains true to the original report submitted. The views expressed do not necessarily reflect those of the IAEA. The use of particular designations of countries or territories does not imply any judgement by the IAEA as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

TABLE 1. RESEARCH REACTORS INCLUDED IN THE PROFILES ON THE CD-ROM

Profile No.	Country	Research reactor	Readiness[a]	Table[b]
1	Argentina	Argentine multipurpose reactor (RA-10)	Planned	A–2
2	Belgium	Belgium Reactor 2 (BR-2)	Operational	A–1
3	China	China Experimental Fast Reactor (CEFR)	Operational	A–1
4	China	China Advanced Research Reactor (CARR)	Operational	A–1
5	Egypt	Experimental Training Research Reactor 2 (ETRR-2)	Potential	A–3
6	France	Jules Horowitz Reactor (JHR)	Planned	A–2
7	Hungary	Budapest Research Reactor (BRR)	Potential	A–3
8	India	DHRUVA	Operational	A–1
9	India	High Flux Research Reactor (HFRR)	Potential	A–2
10	Indonesia	Reaktor Serba Guna G.A. Siwabessy (RSG-GAS)	Operational	A–3
11	Japan	Japan Materials Testing Reactor (JMTR)	Operational	A–1
12	Japan	Experimental Fast Reactor (JOYO)	Potential	A–3
13	Kazakhstan	Impulse Graphite Reactor (IGR)	Operational[c]	A–4
14	Republic of Korea	High-flux Advanced Neutron Application Reactor (HANARO)	Operational	A–1
15	Netherlands	High Flux Reactor (HFR)	Operational	A–1
16	Norway	Halden Boiling Water Reactor (HBWR)	Operational	A–1
17	Poland	MARIA	Potential	A–3
18	Romania	TRIGA II Piteşti	Operational[c]	A–1, A–4
19	Russian Federation	Fast Pulse Graphite Reactor (BIGR)	Operational[c]	A–4
20	Russian Federation	IR-8	Operational	A–1

TABLE 1. RESEARCH REACTORS INCLUDED IN THE PROFILES ON THE CD-ROM (cont.)

Profile No.	Country	Research reactor	Readiness[a]	Table[b]
21	Russian Federation	IVV-2M	Potential	A–1
22	Russian Federation	MIR.M1	Operational	A–1
23	Russian Federation	SM-3	Operational	A–1
24	Russian Federation	BOR-60	Operational	A–1
25	Russian Federation	RBT-6	Operational	A–1
26	Slovenia	TRIGA Mark II	Operational	A–5
27	USA	Advanced Test Reactor (ATR)	Operational	A–1
28	USA	High Flux Isotope Reactor (HFIR)	Operational	A–1
29	USA	Massachusetts Institute of Technology Reactor (MITR)	Operational	A–1
30	Italy	TRIGA RC-1	Operational	A–5
	Italy	TAPIRO	Planned	A–5
	Russian Federation	BFS-1, BFS-2 (Fast Critical Assembly)	Operational	A–5

Note: The research reactors CABRI (France), Nuclear Safety Research Reactor (NSRR, Japan) and Transient Reactor Test Facility (TREAT, United States of America) did not provide information. However, it should be noted that they are important and powerful facilities, in particular for reactivity insertion accident tests. The research reactors ASTRID (France), MBIR (Russian Federation) and MYRRHA (Belgium) are projects to provide additional fast spectrum research capabilities, but they did not provide information. LVR-15 (Czech Republic) and OSIRIS (France) are not included in the profiles because the research reactors did not provide information. SS — steady state; TRIGA — training, research, isotopes, General Atomics.

[a] The readiness for materials testing research falls under three categories of research reactor: operational; planned or under construction; and those with the potential for materials testing research.

[b] Tables A–1 to A–5 can be found on the CD-ROM accompanying this publication.

[c] Operational pulsed research reactor.

Profile 1

ARGENTINE MULTIPURPOSE REACTOR — RA-10
ARGENTINA

1. GENERAL INFORMATION

The Argentinean Multipurpose Reactor (RA-10) is a 30 MW open-pool facility owned and will be exploited by the National Atomic Energy Commission. It will be placed at the Ezeiza Atomic Centre close to Buenos Aires city and will provide a replacement for the RA-3 reactor.

Nowadays the detailed engineering is being developed under a contract with INVAP S. E. The Preliminary Safety Report has been presented on September 2013. The beginning of the civil work is planned for September 2014; while the reactor commissioning is planned for the middle of 2018.

The RA-10 will fulfil national and region demand on radioisotope production, will support the national capabilities related to nuclear fuel production and will also offer new technological capabilities based on neutron thermal and cold beams. Up today, basic engineering is completed, the Preliminary Safety Report was presented to the National Regulatory Commission and it is beginning the detail engineering stage.

FIG. 1. The RA-10's architecture design.

The RA-10 has a compact core with MTR low enriched uranium fuel assemblies and a heavy water radial reflector contained in a zircalloy tank. The fuel elements are placed in a 5×5 grid with 19 fuel assemblies, two fast flux central positions (5E14 $n{\cdot}cm^{-2}{\cdot}s^{-1}$ with $E > 0.1$ MeV) and four adjacent positions (6E14 $n{\cdot}cm^{-2}{\cdot}s^{-1}$, total). The core is contained inside an open pool of demineralized water that provides both cooling and shielding against radiation from the core.

Reactivity control is performed by six hafnium control plates placed inside the core, constituting also the first shutdown system. It is distributed in two rows of three plates each, placed after the first and fourth row of fuel elements.

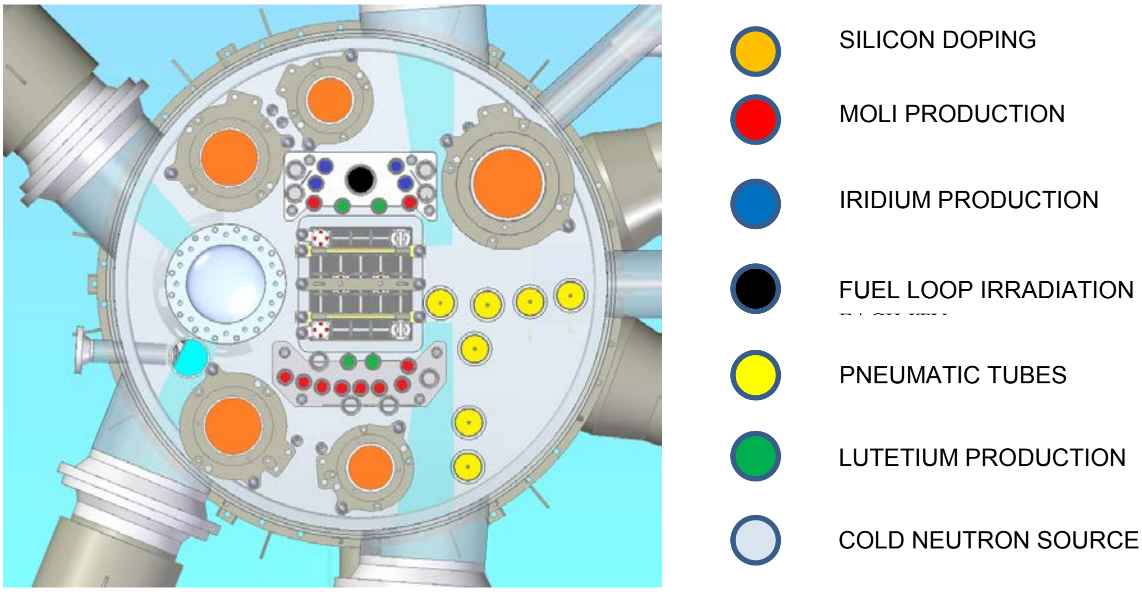


FIG. 2. The layout of RA-10 core.

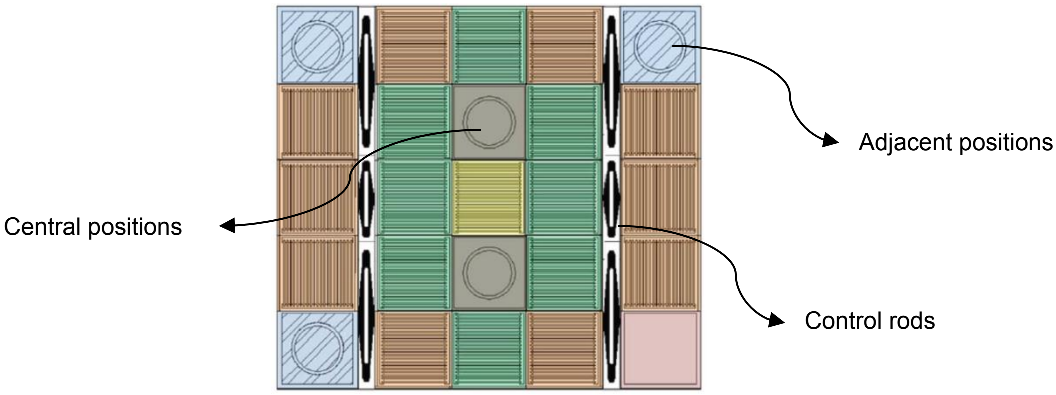


FIG. 3. The layout of RA-10 fuel assembly.

The heavy water reflector tank surrounds the core. It provides a high thermal neutron flux (1E13-2E14 n·cm^{-2} s^{-1} thermal) adequate to house irradiation facilities and perform a diverse and independent shutdown system by means of its drainage. A chimney rises above the core to guide the primary flow to the pump, a closure flow entering the top of the chimney prevents active particles from reaching the surface of the pool. The presence of the chimney also increases natural convection driving force improving cooling during shutdown state and provides a safety feature for loss of coolant beyond design basis events by keeping coolant inside the core. The irradiation rigs are independently cooled by means of the pools cooling system.

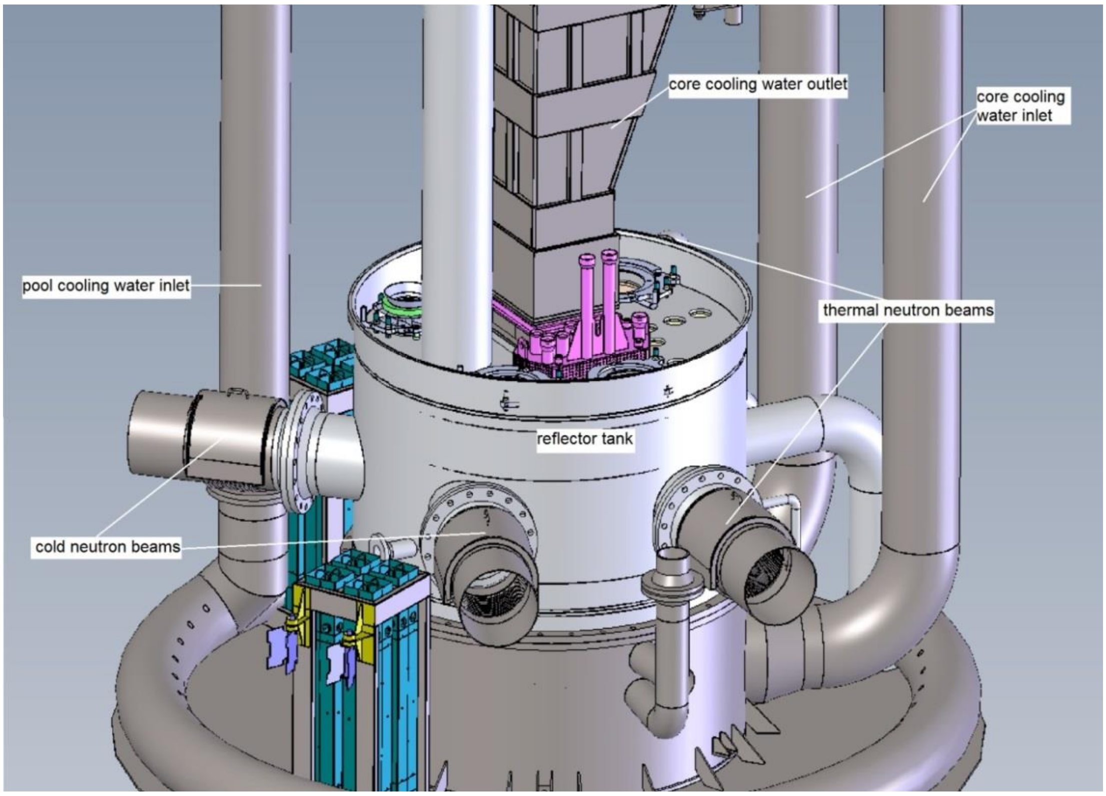


FIG. 4. RA-10 view.

Fuel assemblies are MTR LEU type, consisting of uranium silicide fuel plates, cladded in aluminium. Fuel plates are arranged in parallel forming channels in-between to remove fission heat by means of a forced upward water flow. The plates are of 1.45 mm thickness and the meat thickness is of 0.71 mm. Each fuel has 565 g of ^{235}U and 20 cadmium wires inserted in the aluminium fuel frames in order to improve core management. The cycle length will be of approximately 29.5 days of continuous operation.

2. PROSPECTIVE EXPERIMENTAL FACILITIES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

The core will have six internal positions for material and devices testing, two central with fast neutron flux, and four periferic for thermal and reactor neutron flux. Some of them could be instrumented. Nuclear power plant fuels could be tested in a loop (described below). Up today normal conditions are only considered (steady and transient states).

The reactor will have capabilities for production of radioisotopes such as ^{99}Mo, ^{192}Ir and ^{177}Lu. Also it will have neutron transmutation doping (NTD) positions for doped silicon production, neutron activation analysis pneumatic system, a deuterium cold neutron source, and thermal and cold beams.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

The RA-10 reactor will have a loop facility for nuclear power plant fuels testing, like PHWR and PWR, in steady state and transient conditions such as power ramps. It will be cooled by light water and pressurized up to 15 Bar. The maximum temperature will be 320ºC. It will be instrumented for experimental variables measurement. Power ramp will be produced by a neutron absorber surrounding the device at a maximum rate of 30 $W \cdot cm^{-1} \cdot s^{-1}$. The sample section considered is 12 cm^2 and the maximum length is 40 cm.

The expected performance of the loop will have a power of 500 W/cm and a maximum heat flux of 130 W/cm^2 in steady state conditions (three rods) and 600 W/cm and 150 W/cm^2 in ramp conditions (one rod).

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS

Not considered.

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

See below.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS

See below.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS

The RA-10 will have six in-core irradiation positions for material and devices. The objectives, position, size and flux (thermal (T), epithermal (E), fast (F)) for each kind of irradiation positions are presented in the following Table 1:

TABLE 1. THE OBJECTIVES, POSITION, SIZE AND FLUX FOR IRRADIATION POSITIONS

Type, position and number	Objectives	Size	Flux
High fast flux* Central 2	– corrosion under irradiation – hydrogen interactions – neutron damage synergy	5 cm (Ø) × 12 cm (long)	T: 1.4E14 E: 4.4E14 F: 5.0E14 (8 dpa/year)
Reactor spectrum Adjacent 2	– MTR Miniplates – Capsules	5 cm (Ø) × 12 cm (long)	T: 3.0E14 E: 1.7E14 F: 1.3E14
Reactor spectrum Adjacent 1	Neutron damage in reactor pressure vessel materials	5 cm (Ø) × 12 cm (long)	F: <1.0E13
Fuel position2 Adjacent 1	MTR fuel testing	8 cm × 8 cm × 65 cm	T: 3.0E14

* Fast flux: E> 0.1 MeV.

2.7. OTHER FACILITIES

Not considered.

3. RELATED ENGINEERING AND INFRASTRUCTURE

The reactor will have a hot cell for fresh and irradiated experimental material logistic and an in-pool neutron radiography facility for irradiated devices inspection.

Within the Ezeiza Atomic Center, there will be a Hot Cells Laboratory, which PIE instrumentation is still undefined.

4. RECENT ACHIEVEMENTS

Not applicable — planned research reactor.

Profile 2

BR2

BELGIUM

1. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

The BR2 reactor of the Belgian Nuclear Research Centre at Mol started routine operation in January 1963. BR2 is SCK•CEN's most important nuclear facility and was operated during the past 40 years within the framework of many international programmes concerning the behaviour under representative irradiation conditions of structural materials and nuclear fuels for the various types of nuclear fission reactors as well as for fusion reactor research. The particular features of the BR2 reactor also allow experiments aimed at assessing and demonstrating the safety of nuclear cores.

Today, the reactor plays an essential role in the national and international programmes related to:

— The safety of nuclear reactors, plant lifetime evaluations and ageing of components;
— The development and qualification of nuclear fuels, the increase of their burn-up and the use of MOX fuels;
— The evolution and assessment of safety problems.

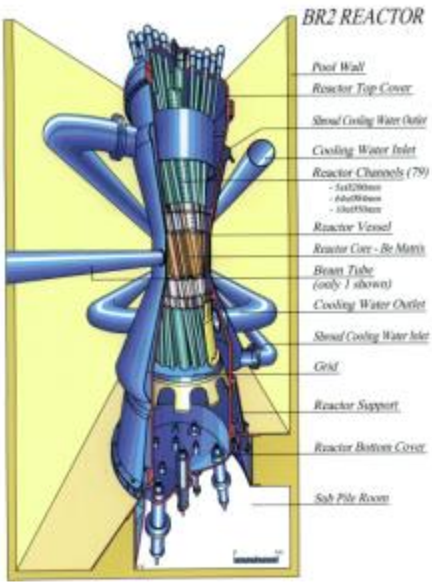


FIG. 1. View of BR-2 reactor.

Besides its scientific-technical R&D programmes, the BR2 reactor has a second mission: production activities for medical and industrial applications and Si-doping by neutron transmutation (NTD-Si). Reactor BR2's high thermal neutron flux (peak 10^{15} n·cm^{-2}·s^{-1}) is ideally suited for the routine production of many important radioisotopes that require a high specific activity. More than 50 radioisotopes are produced for medical and industrial applications.

The BR2 reactor is one of the best performing in the world, and is becoming increasingly more important, as the number of such research reactors is declining worldwide.

1.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

The capabilities and the design of the BR2 are particularly well suited to R&D options, offering:

- A core with a central vertical 200 mm diameter channel, with all its other channels (84 mm) inclined to form a hyperboloidal arrangement around it. This geometry combines compactness leading to high fission power density with easy access at the top and bottom covers, allowing complex irradiation devices to be inserted and withdrawn;
- A large number of experimental positions, including four peripheral 200 mm channels for large irradiation devices;
- Through-loop experiments can be installed via penetrations in the bottom cover of the vessel;
- A remarkable flexibility of utilization: the reactor core configuration and operation mode are continuously being adapted according to the experimental requirements;
- Irradiation conditions (temperature, pressure, environment, neutron spectrum, etc.) representative of various power reactor types;
- High neutron fluxes, both thermal and fast (up to 10^{15} $n{\cdot}cm^{-2}{\cdot}s^{-1}$).

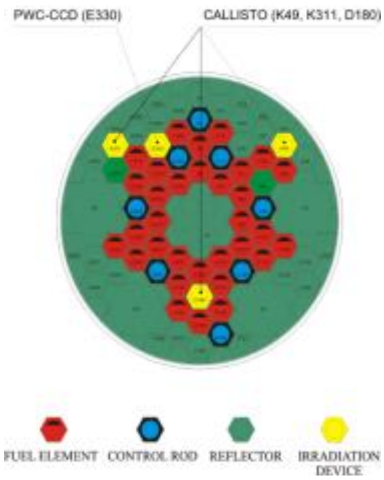


FIG. 2. BR-2 core cross-section.

1.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

1.2.1. CALLISTO: a PWR experimental facility for scientific in-pile studies

CALLISTO — *CA*pabi*L*ity for *L*ight water *I*rradiation in *S*teady state and *T*ransient *O*peration).

Performance and safe operation of nuclear power reactors are continuously improving. This process calls for predictive model validation and qualification testing under realistic power reactor operating conditions. The high pressure/high temperature experimental water loop CALLISTO in the BR2 reactor is used intensively for irradiation studies to cover these essential requirements.

CALLISTO consists of three experimental rigs, called in-pile sections (IPS), which are installed in three reactor channels to meet various irradiation conditions. They are connected to a common pressurized cooling loop, which can deliver a wide range of pressure and temperature working regimes. Each IPS can be equipped with specific instrumentation to unable to perform dedicated irradiation programmes:

— To investigate the behaviour of advanced fuel under representative PWR operating conditions;
— To qualify such fuel for safe, reliable and economical use in power reactors;
— To assess the irradiation assisted stress corrosion cracking (IASCC) phenomena in typical light water reactor materials;
— To study the corrosion process on candidate materials for future fusion reactors;
— To characterise the performances of high neutron dose irradiated materials for light water and fusion reactors as well as for ADS systems;
— To develop and to qualify new on-line in-pile detectors (like neutron and gamma flux detectors, dissolved hydrogen sensors, electrochemical potential reference electrodes, etc.) in a high neutron flux and in relevant thermal-hydraulic environment.

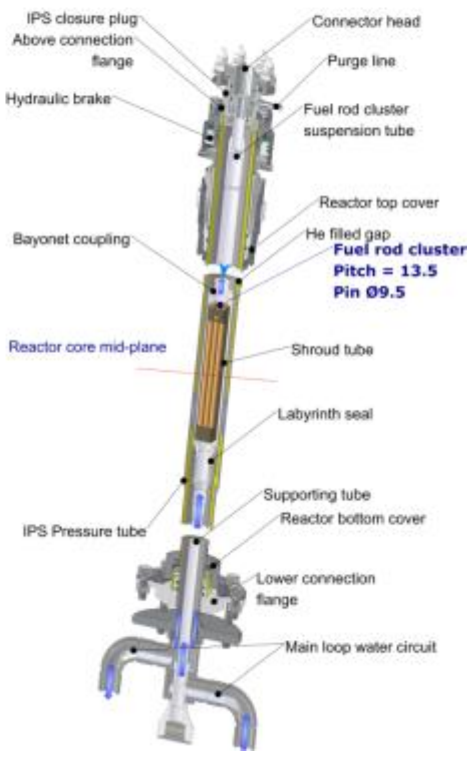


FIG. 3. View of CALLISTO experimental facility.

The basic layout of the CALLISTO facility permits the irradiation of clusters of nine fuel rods (fresh or pre-irradiated) in each of the three in-pile sections, thus a total of 27 rods. Standard the rods are about 1 m long, but variations are possible. Each in-pile section could also accommodate three full-length PWR rods, i.e. over 4 m long. Until now, fuel rods have been irradiated inside CALLISTO in nominal steady state regime in order to achieve their target burn-up. However the loop can also be operated, within certain limits, in off-normal and transient regimes, e.g. power ramping or cycling, cooling mismatch conditions, loss of flow, etc.

CALLISTO is also used for the irradiation of structural materials under representative PWR conditions. Each IPS is provided with a basket, the capacity of which is 37 mm × 37 mm × 1000 mm, and can contain fuel rods or any other target. Different basket geometries can also be accommodated.

The axial distribution enables a 400 mm long basket section (centred on the mid-plane) to be exposed to at least 80% of the maximum fluxes. Depending on the fuel composition and its enrichment, the maximum linear rod power (at peak pellet) ranges from 200 to 450 W/cm for commercial PWR fuel rods.

Two hundred fifty analogue and 350 digital signals are scanned and stored to collect measurements from the dedicated experimental devices loaded in the IPS, to generate diagnostics, to initiate corrective actions and to set alarms off. Integrated within the BR2 data acquisition system, CALLISTO data are remotely accessible from each user's office.

For non-stationary operation with regard to chemical conditions, the chemistry of the primary loop of CALLISTO is monitored both continuously by on-line measurements and by regular analyses of water samples.

1.2.2. MISTRAL

MISTRAL — *M*ultipurpose *I*rradiation *S*ystem for *T*esting *R*eactor *A*lloys.

MISTRAL is a reusable irradiation device for research on reactor materials exposed to a high fast neutron flux at temperatures below 350°C. The MISTRAL design characteristics are:

- Pressurised water capsule containing metallic specimens;
- Loaded inside a BR2 driver fuel element;
- Neutron flux (> 0.1 MeV) 2–3 × 10^{14} n·cm^{-2}·s^{-1};
- Temperature regulation in the range 160–350°C;
- 0.6 dpa per 21-day cycle at 60 MWth (nominal);
- Full instrumentation;
- Number of specimens and their dimension: typically, MISTRAL is designed to irradiate mini-charpy samples (4 mm × 3 mm × 27 mm) and round tensile (5 mm diameter & 27 mm long) specimens (see Fig. 1 here below).

FIG. 4. MISTRAL sample holder.

A maximum of 80 such specimens can be loaded over a 500 mm IPS length centred to the reactor mid-plane, the axial flux profile limited to 500 mm leads to a ratio maximum flux/minimum flux larger than 50%. Other type of specimens may be envisaged if they are compatible with the MISTRAL basket.

MISTRAL is equipped with electrical heaters. The targeted water temperatures (200°C and 300°C) are reached when the BR2 is at 60% nominal power thanks to the gamma heating. These temperatures can be maintained or adjusted at higher reactor power by controlling the electrical heater.

Reaching higher temperature into the specimens is possible by encapsulating the specimens into a matrix made of material that has a good thermal conductivity (Fig. 4). The targeted temperature (> 300°C) is fixed by the thickness of the gas gap between the matrix and the holder.

FIG. 5.

1.2.3. ROBIN

ROBIN — *RO*tating *B*asket with *I*nstrumented *N*eedles).

The aim of the ROBIN facility is to irradiate needles loaded with specimens (typically tensile or mini-charpy) in a GDG facility; GDG is a large thimble loaded in a standard BR2 channel, that is open to the reactor pool allowing devices to be loaded during reactor operation.

ROBIN can contain up to nine needles with 11 mm outside diameter (OD) similar to the ones that are loaded in CALLISTO. In addition, an instrumented needle containing, for example, thermocouples or a gamma-thermometer or a SPND or a fission chamber, can be loaded into ROBIN to measure these parameters on-line and in real time.

To compensate for the fast flux radial gradient through the selected irradiation position, this basket can be rotated during irradiation, leading to the same dose in all the specimens (at the reactor core level in the core) of the nine needles. The maximum fast neutron flux ($E > 1$ MeV) achievable in the facility will be around 3×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$ at the central basket position.

The temperature of the specimens could be adjusted by encapsulating them into a matrix made of material that has a good thermal conductivity and or a suited density. The design of the needles internal part (gas gap between the needle internal surface and matrix (holder) containing the specimen outer surface) determines the specimen temperature. Temperatures up to 300°C could be reached by defining the right thickness of the gas gap between the matrix and the holder and by selecting the appropriate holder material.

The irradiation temperature can be measured inside dummy specimens loaded in the basket central position and used to control the specimen temperature within a certain range by adjusting the basket cooling flow.

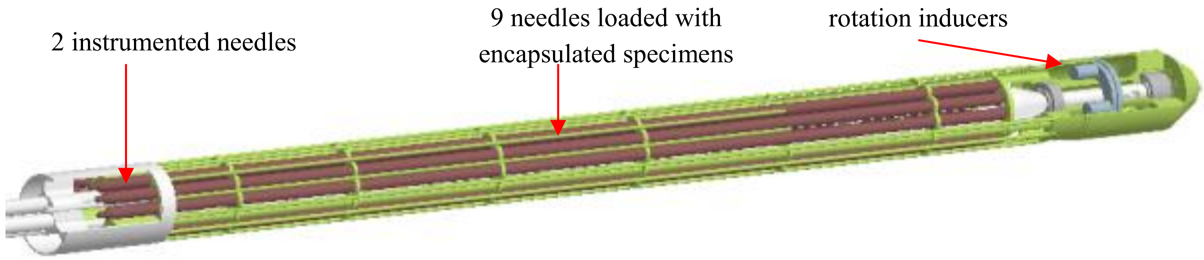


FIG. 6. ROBIN basket.

1.2.4. LIBERTY

LIBERTY — *LI*fting *B*asket in the *E*xperimental *R*ig for *B*R2 *T*himble tube s*Y*stem.

The LIBERTY rig is an improved version of the ROBIN design:
— Each needle can be independently lifted up (and down) above the reactor core level when the specified fluence is reached, while the other needles remain in the neutron flux;
— Each needle can be separately instrumented;
— Larger specimens like the mini CT-specimens (10 mm × 10 mm) can be tested,

— LIBERTY can be loaded into the GDG while BR2 is, in full operation;
— Some electrical heating wires could be put into the needles to control the temperature of the specimens;
— The specimens can be irradiated from 50°C up to 500°C and even higher (depending for instance on the needle filling material). Each of the 5 needles can have different temperatures.

1.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA, ETC.

These facilities are being built for specific programmes and depending on the severity of the simulated accident, not reusable (the MOL-7C and the PAHR programmes).

1.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Description of the hot cell autoclave and PWR loop:

Crack initiation and propagation tests of irradiated stainless steels can be performed in pressurized water reactor relevant conditions, in autoclaves installed inside hot cell facilities. The hot cell has a 1 m thick heavy concrete (barite) shielding (or equivalent lead shielding at penetration sites). The hot-cell is equipped with master-slave manipulators for handling of samples. There is a storage room for samples connected to the hot-cell to limit transfer operations.

Inside the hot-cell, three autoclaves are present with serial flow of water, so the chemical conditions required are always identical for all samples. The autoclaves are equipped with a servo-electric tensile loading device, which penetrates the autoclave lids. Each autoclave has one loading unit and one position for a CT specimen or a set of crack initiation specimens. That means that three (sets of) samples can be tested simultaneously, while each sample has its own loading unit and autoclave, but the same water chemistry as the autoclaves are connected in series. The loading devices can be operated in SSRT mode, constant load/displacement mode or cyclic loading. Each autoclave is equipped with an internal thermocouple and sample instrumentation wires, so that potential drop crack length monitoring could be carried out.

FIG. 7. The autoclaves in the hot cell.

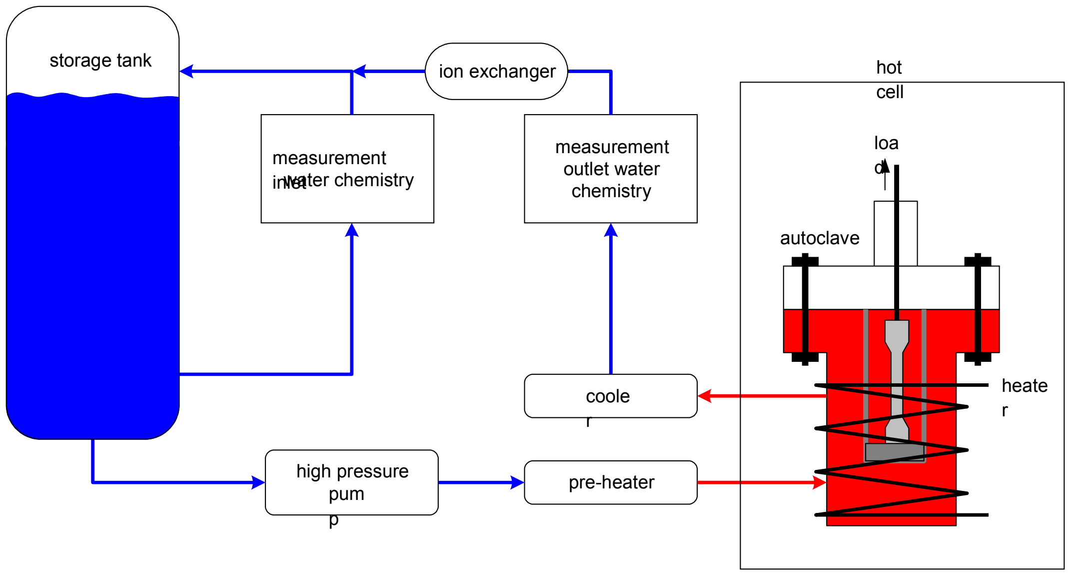


FIG. 8. Schematic of the hot cell installation.

The tests will be performed in serial flow, so all the autoclaves receive the same flow rate and chemical conditions. The on-line chemistry monitoring system records the solution conductivity, pH, dissolved oxygen and hydrogen in the water. In order to maintain PWR water chemistry, the ion exchange resins are pre-saturated with lithium and boron, so these additives are not removed during operation of the loop.

1.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES ETC.

BAMI — *BA*sket for *M*aterial *I*rradiation.

In BR2, a lot of standard 'capsules' are routinely used to perform several irradiation campaigns (testing of materials, production of radio-isotopes, etc.). These capsules are classified according to the requested flux or fluence:

(1) 'Capsules' loaded 'inside' the BR2 vessel;
- Cannot be unloaded during the cycle: 21 days irradiation;
- Cooled by the BR2 primary water: excellent cooling (T° of capsule wall ≈ T° of BR2 water).

TABLE 1. CAPSULES LOADED INTO A FUEL ELEMENT (typical values)

	Fast neutron flux (E >0.1 MeV) ($n{\cdot}cm^{-2}{\cdot}s^{-1}$)	Fast fluence (E >0.1 MeV) ($n{\cdot}cm^{-2}{\cdot}s^{-1}$)	Nuclear heating rate (W/g_{Al})
Max	5×10^{14}	9×10^{20}	10
Min	1×10^{14}	1.8×10^{20}	2

TABLE 2. CAPSULES LOADED INTO BE REFLECTOR CHANNEL (typical values)

	Fast neutron flux (E >0.1 MeV) ($n \cdot cm^{-2} \cdot s^{-1}$)	Fast fluence (E >0.1 MeV) ($n \cdot cm^{-2} \cdot s^{-1}$)	Nuclear heating rate (W/g_{Al})
Max	1×10^{14}	1.8×10^{20}	3
Min	1×10^{13}	1.8×10^{19}	0.6

(2) ‘Capsules’ loaded into a thimble tube open to the BR2 pool:
- — The tumble tube called DG (Doigt de Gant) is itself loaded into the BR2 vessel;
- — The capsules could be unloaded at any time during the cycle — the requested fluence determines the irradiation time;
- — Cooling only by free convection — poor cooling of the capsule.

TABLE 3. TYPICAL VALUES OF NEUTRON FLUX AND HEATING RATE

	Fast neutron flux (E > 0.1 MeV) ($n \cdot cm^{-2} \cdot s^{-1}$)	Nuclear heating rate (W/g_{Al})
Max	1×10^{14}	3
Min	1×10^{13}	0.6

(3) ‘Capsules’ loaded into the ‘pool tube’:
- — Very low flux (equivalent fission neutron flux < 3×10^{11} $n \cdot cm^{-2} \cdot s^{-1}$) — negligible fast neutron flux;
- — Only the upper part of the core length is available (lower part not accessible);
- — Very seldom used — very few data on the neutron characteristics of this channel.

Geometry of the capsules — at least, three capsules types are commonly used; the selection of the capsule type depends on its use (i.e. in which BR2 channel it will be irradiated).

TABLE 4. GEOMETRY OF THE CAPSULES

Type	Outer diameter (mm)	Inner diameter (mm)	Useful length (mm)
RH	15.0	13.4	66
RA	22.0	20.0	55
CSF	25.0	23.0	68

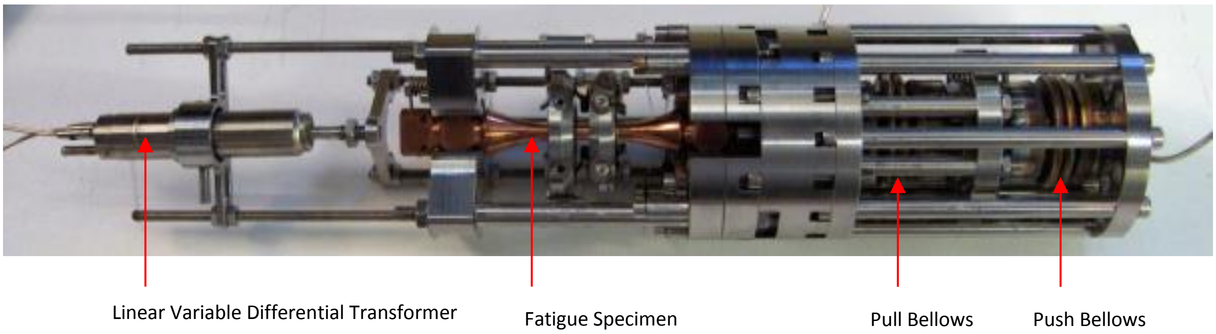
1.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (swelling, gas release, creep, long-term strength, relaxation resistance, etc.)

1.6.1. COFAT

COFAT — *CO*pper *FAT*igue.

The aim of the COFAT project is to investigate in-situ the dynamic effects of cyclically applied stresses in the copper components of ITER on the neutron damage accumulation and to assess the mechanical performance during neutron irradiation.

The COFAT rig contains the in-pile section and the required out-pile equipment (pool water cooling loop, high pressure helium supply loop); two creep-fatigue test modules allow performing *fatigue test* on specimens *during the irradiation*. Each module is made of two bellows (one to push and the other to pull) and of one LVDT (linear variable differential transformer) to measure the gage length variations.



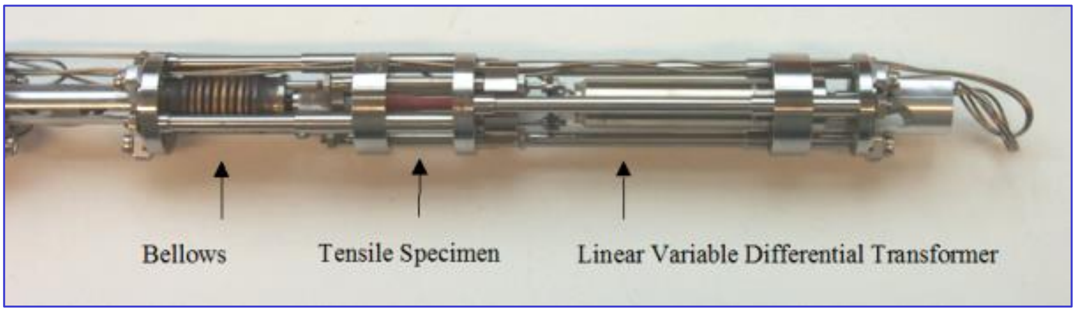


FIG. 9. COFAT: creep-fatigue test module.

1.6.2. The PWC/CCD irradiation rig

PWC/CCD — *P*ressurised *W*ater *C*apsule / *C*ycling and *C*alibration *D*evice.

The Pressurised Water Capsule (PWC) is an instrumented irradiation capsule for the testing of single fuel rod segments with a diameter of 8–15 mm and an active length up to 1000 mm, under steady-state or transient conditions. The target fuel segment, which can be equipped with a thermocouple, is placed into the stainless steel capsule filled with demineralized stagnant water.

The Cycling and Calibration Device (CCD) is a dedicated flow calorimeter which is designed to monitor the thermal performance of the coolant flowing through it, using a diaphragm flow meter and thermocouples placed at inlet and outlet. A helium screen, placed on the outer surface of the CCD, serves as a thermal shield to minimize heat leaks. The fuel rod heating is adjusted by varying the BR2 reactor power during dedicated short reactor cycles.

The PWC water can be pressurized in the range of 0.1 to 16 MPa. The heat generated in the rod is dissipated radially towards the outer surface of the pressure tube by natural convection, with or without boiling, depending on the irradiation programme. The PWC capsule is cooled by the reactor water flowing through the CCD calorimeter.

Thermocouples have been installed to monitor the temperatures on the outer surface of the fuel rod at its mid-plane and in the stagnant water of the PWC. Water samples of the PWC water can be taken before, during and after reactor operation for monitoring the fission product activity in order to detect a possible fuel rod failure.

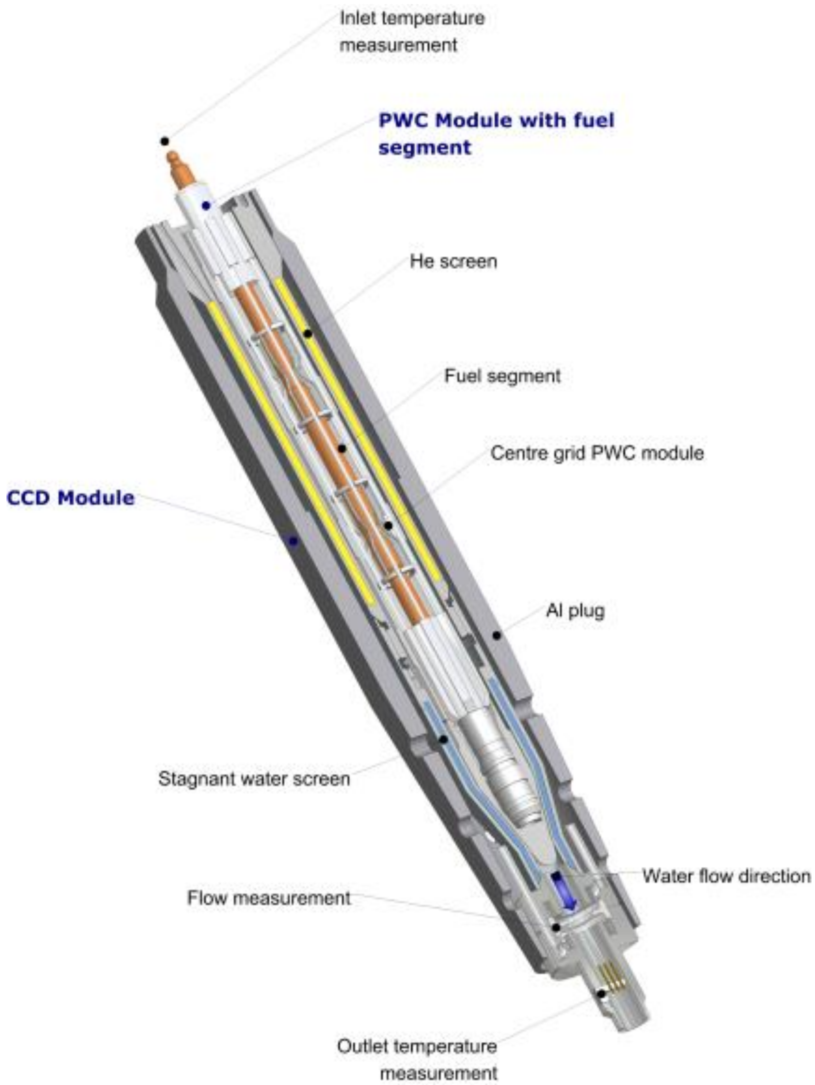


FIG. 10. The PWC/CCD irradiation rig.

The PWC/CCD device has been applied successfully in several fuel ramping tests during the last 30 years.

1.7. OTHER FACILITIES

1.7.1. SMIRNOF

This versatile in-pile reactor irradiation facility is used for radiation assessment of optical fibres in fission-reactor environment. The system fits any irradiation channel of BR2 and records in-situ the radiation response of irradiated optical fibres. This approach allows to cover a wide range of irradiation conditions accessible to BR2 and specified by the user.

The irradiation facility consists in an in-pile section (IPS) supporting a removable irradiation basket and an out-of-pile section (OPS) controlling the temperature condition during the irradiation. In the IPS the fibers can be coiled on the irradiation basket if desired. The fibers are protected in stainless steel tubes to ensure a safe operation during the whole experiment.

In transmission configuration up to four tubes can be irradiated simultaneously allowing to test a large number of fibers of various sizes. The irradiation basket can be removed and reloaded with new fibers during reactor operation. Typical lead length between the irradiation section and the data acquisition is 25 meters. As temperature is an important parameter governing the radiation response of optical fibers the irradiation facility is controlled in temperature by means of an air flow. For a given reactor power the temperature of the fibers is set to the desired value by varying the air mass flow and the pressure.

2.7.2. Gamma irradiations facilities

SCK•CEN is operating different gamma irradiation facilities, with gamma dose rates ranging from 1 Gy/h up to 35 kGy/h. The sources of gamma-radiation are Co^{60} or spent fuel elements. The gamma facilities cover most of the application areas dealing with ionizing radiation.

For dose rates below 1 Gy/h the bunker type irradiation facility CAL is used.

The main characteristics of the irradiation facilities are given in the Table 5. On-line instrumentation allows for performing electrical and optical (using optical fibres) measurements during irradiation. Elevated temperatures during irradiation are achieved by using dedicated ovens placed inside the irradiation container. The temperatures up to 300°C are routinely possible. Irradiation at cryogenic temperatures is possible by using liquid nitrogen. In this case the duration of an uninterrupted irradiation is limited to one day.

TABLE 5. THE MAIN CHARACTERISTICS OF THE IRRADIATION FACILITIES

Name	Rita	Geuse II	Brigitte		
			Pos A1/A2	Pos B	Kirsten
Source	Co-60	Spent fuel	Co-60	Co-60	Co-60
Dose rate (Gy/h)	10–1300	100–1400	100–35000	500–25000	16000
On-line instrumentation	available				no
Temperature control	available				measurable
Atmosphere control	inert gas, dry air				
Vacuum module	available		no	no	no
Available volume					
Diameter (mm)	380	360	80	220	35
Height (mm)	600	900	900	900	200
High dose rate volume					
Diameter (mm)	30	30	60	190	35
Height (mm)	200	400	60	100	100

2. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

2.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

SCK•CEN is authorized as an importer of radioactive materials and is licensed to receive fresh and irradiated materials such as powders, plates, rods and construction materials. These materials can be handled in multiple hot-cells, glove boxes and fume hoods as described in Section 3.2 or be irradiated in BR2. SCK•CEN is owner of approved containers for nuclear transports and also capable to receive and handle other types of containers. Material logistics are managed by certified persons using computerized systems to organize transports and the nuclear accountancy.

2.2. HOT CELLS, PIE FACILITIES

At the BR2 reactor, a hot cell for dismantling of experimental devices and first sample preparations is available. The hot cell is multi-purpose and allows installation of dedicated technology for individual projects if necessary.

The Laboratory for High and Medium Activity (LHMA) offers a full suite of post-irradiation examination (PIE) techniques for both non-destructive and destructive examination of irradiated and fresh alpha and beta-gamma active materials. It offers:

- Wide range of hot cells for multi-purpose application with different shielding and containment;
- Non-destructive fuel analysis: metrology, defect analysis, oxide measurement, radiography, gamma spectrometry, etc. of irradiated fuel rods up to 4 m in length;
- Microstructural analysis: five sample preparation lines (inactive, low active fume hood, low active glove box, high active alpha and high active beta-gamma) with optical microscopy, scanning electron microscopy (SEM) with EDX, electron probe microanalysis (EPMA) with WDX, transmission microscopy (TEM), x-ray diffraction (XRD) and X-ray photoelectron spectroscopy (XPS);
- Chemical analysis: radiochemistry with alpha and beta/gamma spectrometry, mass spectrometry (ICPMS and TIMS), total combustion analysis (TCA);
- Mechanical analysis: sample preparation technology (reconstitution, pre-cracking, side-grooving, etc.), Charpy testing, three point bending, tensile testing of both active and non-active specimens;
- Corrosion analysis: autoclaves with controlled water chemistry for mechanical testing of active and non-active specimens, in-cell setup for tensile testing in LLBE environment.

A description of the available technology is provided in the IAEA PIE database at http://infcis.iaea.org/.

2.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

SCK•CEN has a dedicated unit, the experimental rig design unit, for the design and manufacturing of experimental irradiation devices, including related measurement systems. The main mission of this unit is the design and construction of irradiation devices for the BR2 reactor to meet customer's requirements. Besides the main mission, they also perform design for other SCK•CEN institutes, for example for the future MYRRHA reactor,

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS

Link to the list of recent publication is recommended.

Profile 3

CHINA EXPERIMENTAL FAST REACTOR

CHINA

1. GENERAL INFORMATION AND TECHNICAL DATA OF RR (LINK WITH RRDB)

China Experimental Fast Reactor (CEFR) is the first pool type sodium cooled fast reactor in China, which firstly connected with net in 2011. The reactor is with (Pu, U)O_2 as fuel, but UO_2 as the first loading, Cr-Ni austenitic stainless steel as fuel cladding and reactor block structure material, bottom supported pool type, two main pumps and two loops for primary and secondary circuits respectively. The water-steam tertiary circuit is also two loops but the superheat steam is incorporated into one pipe which is connected with a turbine. Table 1 shows the major design features.

TABLE 1. THE MAIN DESIGN FEATURES OF CEFR

Parameter	Value
Rated thermal power	65 MW
Experimental electric power	20 MW
Fuel	UO_2 or MOX
Fuel SA/control rods SA	81/8
Equivalent reactor core diameter	600 mm
Core height	450 mm
Maximum neutron flux rate	3.2×10^{15} / 3.5×10^{15} n·cm^{-2}·s^{-1}
Maximum fuel burn up	60 000 MW·d/t
Core inlet/outlet temperature.	360-530°C
Core flow rate	301 kg/s

As shown in Fig. 1, the reactor core is composed of 81 fuel subassemblies, three compensation subassemblies and two regulation subassemblies, which all play a function of first shut down system, three safety subassemblies as secondary shut down system, then 336 stainless steel reflector subassemblies and 230 shielding subassemblies, and in addition 56 positions for primary storage of spent fuel subassemblies are included. Each fuel subassembly is composed of 61 fuel pins with a 6.0 mm outer diameter of fuel pin cladding and 0.95 mm wire warped in a hexagonal tube. The fuel column has a height of 450 mm and the total length of fuel subassembly is 2592 mm.

The CEFR core benefits from feedback effects providing the reactor with inherent safety characteristics. The core is designed with negative temperature coefficients, negative Doppler coefficients, and negative power coefficients.

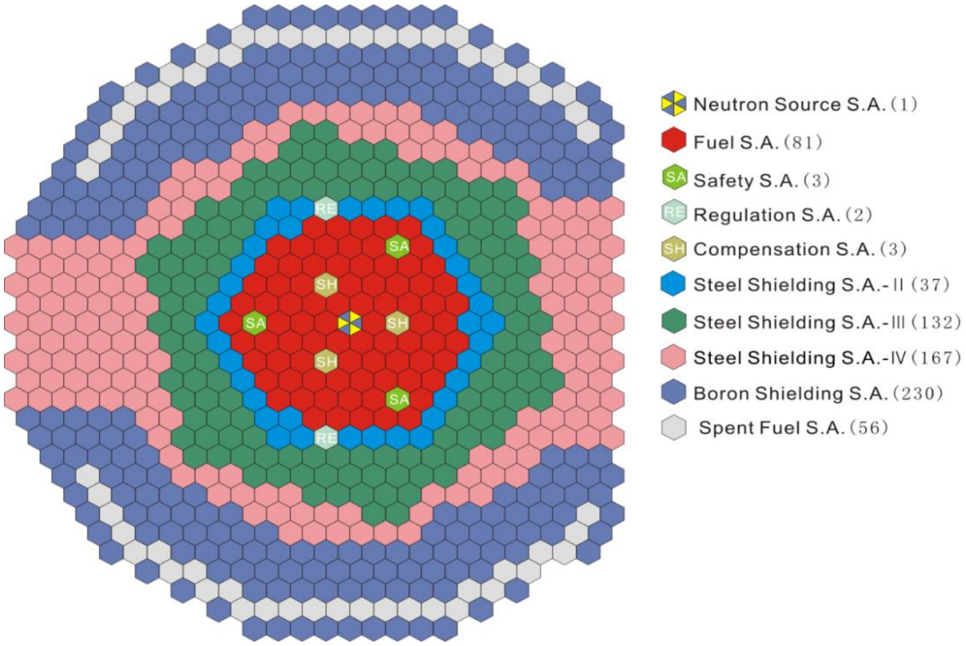
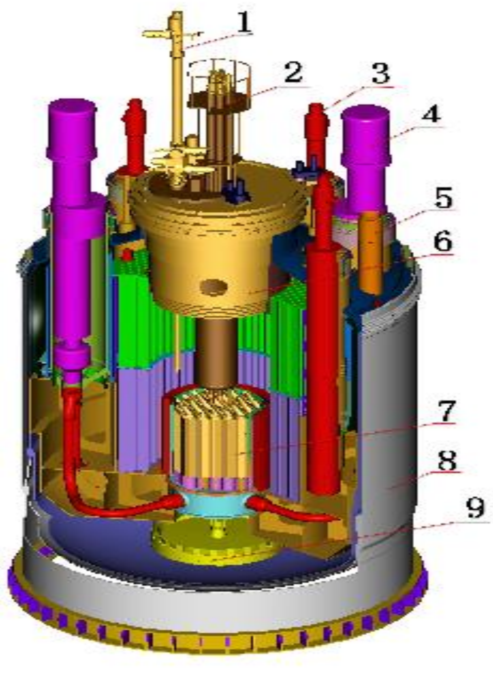


FIG. 1. The layout of CEFR core.



1. Fuel handling machine;
2. Control subassemblies;
3. IHX (4);
4. Primary sodium pump (2);
5. Independent heat exchanger (2);
6. Double rotating plug;
7. Core;
8. Main vessel and guard vessel;
9. Molten core catcher.

FIG. 2. CEFR reactor block.

The CEFR block is shown in Fig. 2, which is composed of main vessel and guard vessel supported from bottom on the floor of reactor pit. The reactor core and its support structure are supported on lower internal structures. Two main pumps and four intermediate heat exchangers are supported on upper internal structures. These two structures sit on the lower part of the main vessel. Two independent heat exchangers of the DHRS are hung from the shoulder of the main vessel. The double rotation plugs on which control rod driven mechanisms, fuel handling machine and some instrumentation structures are supported on the neck of the main vessel.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT CEFR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

For the closed nuclear fuel cycle, a main purpose of CEFR is to test the fast reactor technologies and to demonstrate the fast reactor power plant. So there is not some special test channel in CEFR, but it provides 251 positions, which can be used for the irradiation test (cooled by forced circulation), including 81 fuel subassemblies, 1 neutron source subassembly and 169 SS subassemblies, beside of 8 control rod subassemblies. All the irradiation test targets must be put into a special test subassembly, which has the same shape with fuel subassembly.

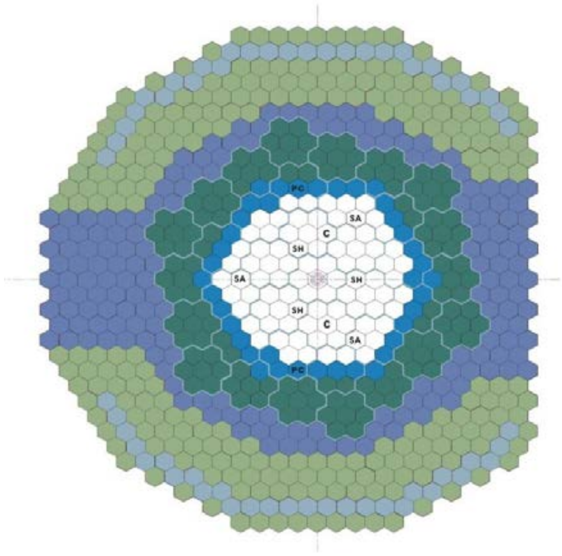


FIG. 3. The layout of CEFR core.

The max neutron flux of CEFR is 3.2×10^{15} n·cm^{-2}·s^{-1} with UO_2 fuel and 3.5×10^{15} n·cm^{-2}·s^{-1} with MOX fuel. The neutron flux in the SS subassemblies, which are forced cooled, is from 1.9×10^{15} n·cm^{-2}·s^{-1} to 4.4×10^{14} n·cm^{-2}·s^{-1}. The irradiation damage distribution and neutron flux are shown in Table 2.

TABLE 2. THE TOTAL NEUTRON FLUX OF CEFR

Position	Neutron flux, $n{\cdot}cm^{-2}{\cdot}s^{-1}$	Max irradiation damage, Dpa/80EFPDs
The very centre	3.2×10^{15}	8.6
2nd round	3.2×10^{15}	9.3
6th round	2.3×10^{15}	6.4
7th round	1.9×10^{15}	4.5
8th round	1.5×10^{15}	2.6
9th round	7.1×10^{14}	0.8
10th round	4.4×10^{14}	0.5
11th round	3.1×10^{14}	0.3
13th round	1.4×10^{14}	0.2
15th round	2.9×10^{13}	0.04

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE (fuel, control rods, structural materials, coolant technologies: lead, lead-bismuth, sodium, light and/or heavy water, molten salt, gas)

— At steady state conditions;
  There is not any special testing loop, so the conditions are same as the reactor;
— At transient conditions;
  There is not any special testing loop, so the conditions are same as the reactor;
— At accident conditions;
  There is not any special testing loop, so the conditions are same as the reactor;

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA, ETC.

Not considered.

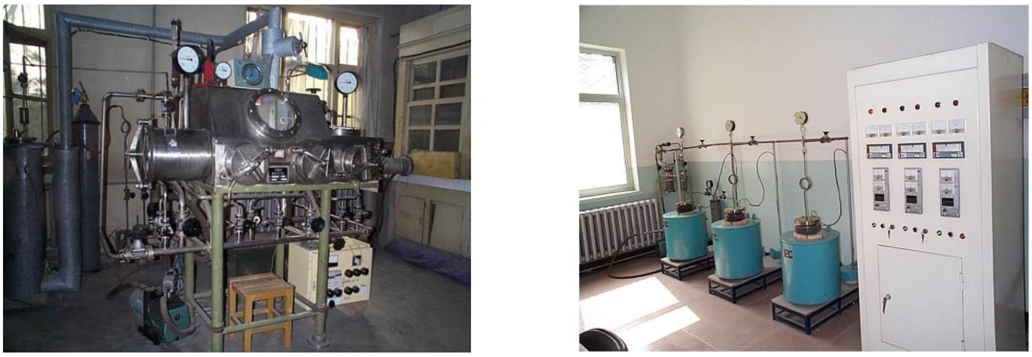
2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

See Section 2.5.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES ETC.

2.5.1. High temperature sodium static test facility (SSTF)

SSTF is one of the serials test out-of-core facilities for studying the compatibility of SFR structural materials with high temperature sodium (see Fig. 4). The facility consists of static test furnace where corrosion test is performed and glove box system where the test specimens were prepared. The main parameters of this facility are listed as following:
— Max. temp.: 800°C;
— Cover gas: Argon;
— Oxygen in sodium: 10 ppm;
— Carbon in sodium: 10 ppm.



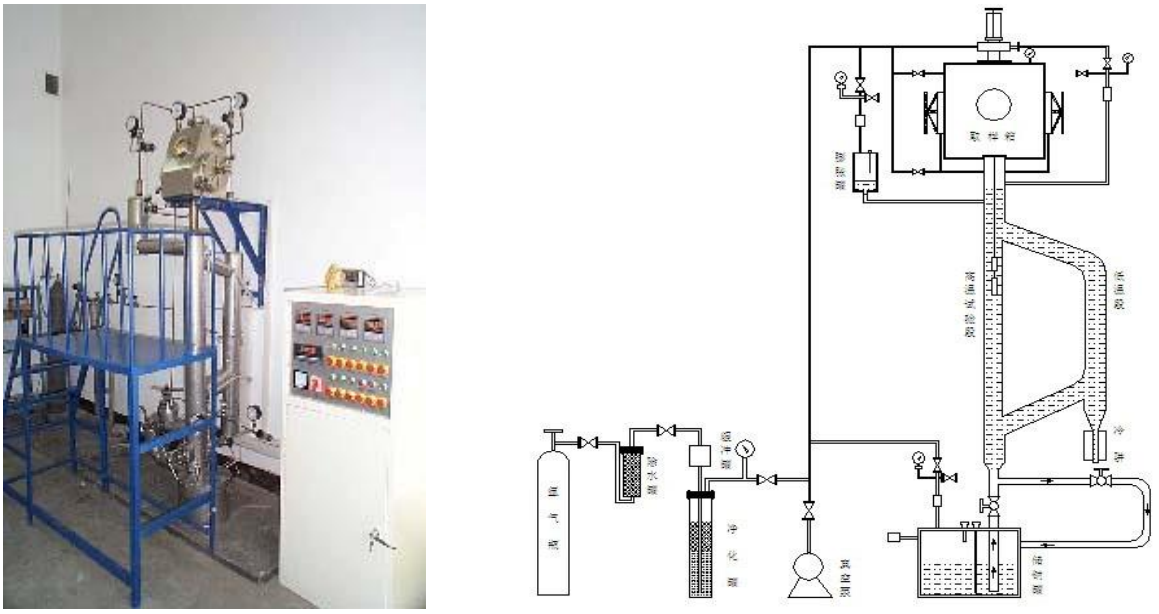
(a) (b)

FIG. 4. High temperature sodium static test facility; (a) glove box system; (b) static test furnaces.

2.5.2. High temperature sodium thermal convection test loop (STCTL)

STCTL is a quite important out-of-core loop which mainly used for studying the corrosion behaviour of SFR structural materials in the flowing sodium (the sodium velocity on the surface is ≤ 10 cm/s). The corrosion rates of structural materials could obtain after being tested in this loop, micro-structural of oxide layers formed on the surface of steels could be analysed and intergranular attack tendency is evaluated. Figure 5 is the schematic diagram and photo of facility. The main parameters of this loop are as following:

- Max. temperature: 550^{o}C;
- Max. temperature difference between hot-leg and cold leg: 100^{o}C;
- Flow speed: 4-10 cm/s;
- Oxygen in sodium: ≤ 25 ppm (min).



(a) (b)

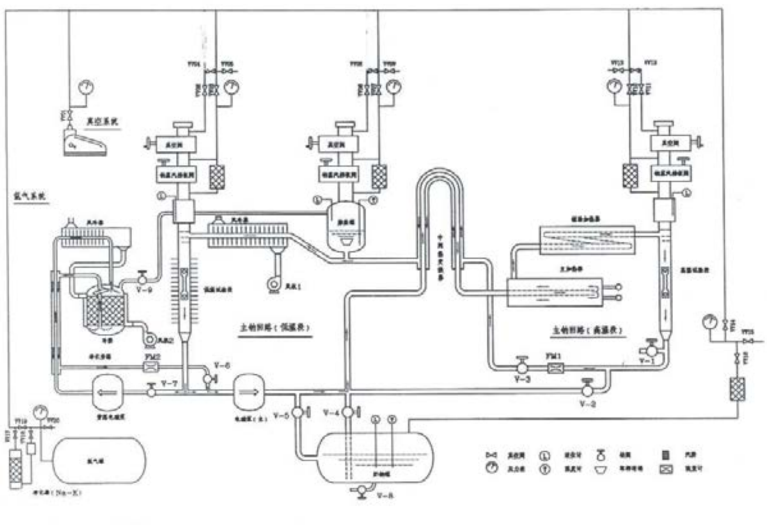
FIG. 5. High temperature sodium thermal convection loop; (a) photo; (b) schematic diagram.

2.5.3. High temperature sodium mass transfer test loop (SMTTL)

SMTTL is an un-isothermal sodium out-of-core loop system which is used for studying the elements of materials (such as carbon) transferring effect in main or secondary loop system of SFR. There are two test sections (with high temperature of 650°C and 450°C for low temperature section) in this loop. The main parameters of this loop are listed as following:

— Total electric power: ~ 85 kW;
— Gross capacity: ~ 0.125 m^3;
— Temperature of hot-leg: 550-650°C;
— Temperature of cold-leg: 350-450°C;
— Max. temperature difference: 250°C;
— Flow rate in main circuit: 1.2 m^3/h;
— Flow rate in by-pass: 0.22-0.125 m^3/h;
— Pressure: 0.03-0.06 MPa;
— Oxygen in sodium: 10-20 ppm.

(a)



(b)

FIG. 6. High temperature sodium mass transfer test loop (a) photo, (b) schematic diagram.

2.5.4. Fuel-cladding chemical interaction out-of-pile test facility (FCCITF)

Figure 7 shows the FCCIFT which is mainly used to simulate the interaction between SFR fuel cladding material with the fuel elements by out-of-pile simulation tests, while it could also serve for simulating chemical interactions between the control rod element B_4C pellets and the cladding of control sub-assembly. Main parameters of FCCIFT are listed as following:

- Test temperature (maximum): 900°C;
- Oxygen buffer: Cr/Cr_2O_3 & Ni/NiO;
- Cs, Te and Se mixture: simulating fission product.

FIG. 7. Photo of Fuel-Cladding chemical interaction out-of-pile test facility.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (SWELLING, GAS RELEASE, CREEP, LONG-TERM STRENGTH, RELAXATION RESISTANCE, ETC.)

The materials analysis lab (MAL) consists of electron microscopy lab and mechanical test lab which is located in the area of CEFR site.

The electron microscopy laboratory presently focuses on the non-radioactive materials, while both radioactive and non-radioactive materials will be permitted in the near future. A portion of the laboratory is dedicated to sample preparation (non-radioactive coupon), providing the researcher with facilities support, equipment, safety systems, and procedures to prepare samples of diverse materials for analysis. The primary instruments in MAL are a JEOL 2100F scanning transmission electron microscope (TEM) and a ZEISS Superra 55 scanning electron microscope (SEM) (see Figs 2.6 and 2.7). The TEM is capable of operating at 200 kV, and is capable of magnifications from 2000 X to 1 500 000 X. It is equipped with a JEOL Instruments energy dispersive X-ray spectrometer. Crystallographic information can be obtained by recording the diffraction patterns formed by electrons as they pass through the sample. The ZEISS Superra 55 SEM is a field emission instrument capable of operating at 30 kV, and is capable of magnifications from 15 X to 100 000 X. It is equipped with BRUKER Instruments energy dispersive (EDS) that can be used to obtain quantitative information about the elemental composition of a sample. In addition to the TEM and SEM, MAL also has several optical microscopes. Some of these are used to support sample preparation, and others are used for optical characterization of samples. Capabilities for sample preparation include cutting, grinding, and polishing, as well as specialized methods such as chemical milling to produce thin, electron-transparent samples, etching, and coating.

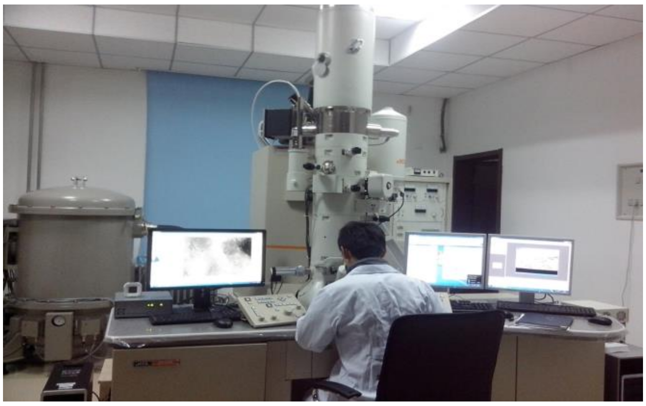


FIG. 8. TEM of materials analysis lab.

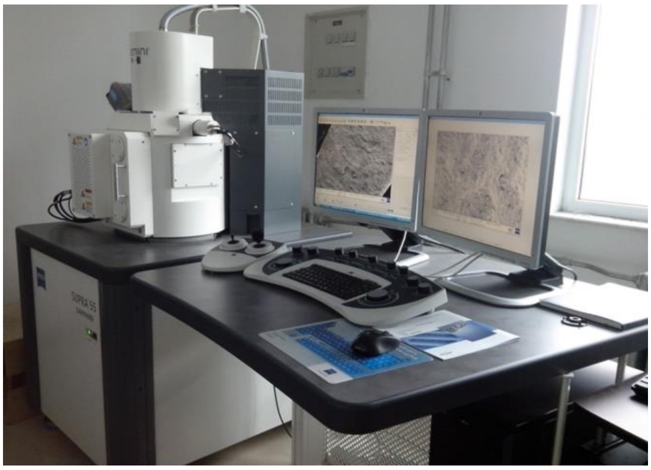


FIG. 9. SEM of materials analysis lab.

The mechanical test lab includes a 300J pendulum impact machines, several tensile test machines and a bunch of creep testing machines. Some machines are located in a semi-hot cell which could perform mechanical tests on irradiated samples (see Fig. 10).

FIG. 10. Semi-hot cell for mechanical testing.

2.7. OTHER FACILITIES (this section will be added only in the electronic version and it might include zero or low power facilities supporting innovative nuclear energy projects)

Not considered.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

The transport of 3.1. — Fresh and irradiated experimental material in CEFR is same as the fuel.

3.2. HOT CELLS, PIE FACILITIES (radiochemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

Post-irradiation examination (PIE) for irradiated assembly and specimens could perform inside the CIAE hot cells which include hot cell in CEFR site, 303 hot cell and Radiochemical laboratory. The capability of PIE works includes non-destructive analysis (for example dimensional measurements and neutron radiography) and destructive examination (such as mechanical testing or metallographic characterization). It can accept full-size PWR and SFR fuel assemblies.

FIG. 11. Radiochemical laboratory.

The main works are as following:

(1) Study on the reprocessing technology of SFR MOX spent fuel;
(2) Study on the process of separation of thorium uranium fuel cycle;
(3) Study on the dry reprocessing;
(4) Study on the process of separation of nuclear analysis and nuclear detection;
(5) Study on the process of separation of minor actinides and long-lived fission product elements.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

With a total staff of 3000, CIAE is composed of six departments, two technical divisions, six centres or key laboratories of national or ministerial level, five divisions for engineering projects, ten research centres and key laboratories. There are five research reactors, several zero power facilities here.

As the first base of reactor research and nuclear fuel production research, CIAE has been focusing on R&D of advanced reactor technology, fast neutron reactor, new fuel element, reactor physics, nuclear fuel reprocessing, nuclear fuel recycle, etc.

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS (link to the list of recent publication is recommended)

CEFR has realized first criticality on 21 July 2010, and some physical start up test has be done, including criticality test, control rod value measurement, some nuclear reaction distribution measurement and reactivity coefficient measurement. It will reach the full power in 2014, and begin the irradiation test, including MOX fuel, SS and MA transmutation.

Profile 4

CHINA ADVANCED RESEARCH REACTOR (CARR)

CHINA

1. GENERAL INFORMATION AND TECHNICAL DATA OF A RESEARCH REACTOR (RR) WITH A LINK TO THE IAEA RRDB

General information: located in Fangshan District of Beijing, China, the China Advanced Research Reactor (CARR), composed of reactors, auxiliary systems and testing facilities, is a safe, reliable and multifunctional research reactor with high performance. The construction area of CARR is approximately 18 000 m^2, about 2.3 hectares of ground area occupied. The CARR is a large-scale nuclear science project providing an important testing platform for nuclear science and research of China. For the aims of effective organization and management of the construction and commissioning of CARR, the China Institute of Atomic Energy (CIAE) established CARR Project Department and accredited it to control the quality and progress of the project, as well as its investment. CARR, started construction on 26th August 2002, successfully realized the first criticality on 13 May 2010. Initial full power operation for 72 h has been finished in April 2012. Up to now, all of the commissioning works on CARR have been finished.

TABLE. 1. TECHNICAL DATA

Reactor type	Tank in pool
Thermal Power	60 MW
Max Flux SS, thermal	8E14 (n/cm^2-s)
Max flux SS, fast	6E14 (n/cm^2-s)
Fuel material	U_3Si_2-Al
Cladding material	6061 Al
Moderator material	Light water
Coolant material	Light water
Reflector material	Heavy water
Control, safety, shutdown rods material/number	Hf/6
Nat convection/direction	<900 kW/Up
Forced cooling/direction	Yes/down
Cool velocity in core	10 m/s (nominal)

(http://nucleus.iaea.org/RRDB/RR/GeneralInfo.aspx?RId=613)

FIG. 1. CARR building.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

A set of in-pile irradiation testing facilities is projected to be constructed in CARR in 2014. The test loop can be used for steady state irradiation and transient test for nuclear fuel and material. It includes two parts: high temperature and high pressure test loop (HTHPTL) and ^{3}He pressure control loop (^{3}He loop). By using the in-pile irradiation testing facilities, the performance tests (including steady state and transient), high burn-up test, water chemistry activity transport and corrosion test, and fuel integrity and qualification test, etc., could be conducted. The PIE will be performed in the hot cell №303 and CARR hot cell. The PIE of PWR spent fuel can be performed in the hot cell №303.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE (fuel, control rods, structural materials, and coolant technologies: lead lead-bismuth sodium light and/or heavy water, molten salt, gas)

Loops for testing components at transient conditions are not considered. At present, two loops will be constructed, HTHPTL and He-3 Loop.

Test Apparatus of In-Pile Irradiation and Tritium In-situ Extraction (CIPITISE) of the Solid Breeder Pebble Bed in CARR for ITER and Future Fusion Reactors.

2.2.1. HTHPTL (under construction)

HTHPTL is a separate pressurized water loop, connected with an in-pile tube. By use of HTHPTL, the fuel or material testing can be performed at different pressures, temperatures, flow rates, and water chemistry. The HTHPTL is connected to a computer control system.

This control system controls, monitors, and provides emergency functions and alarms during operation of loop. HTHPTL consists of 10 sub-systems such as primary circuit system, safety injection system, purify and sampling system, component cooling system, etc. The sketch map of flow is shown in Figure 2.

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FIG. 2. Sketch map of flow of in-pile HTHP test loop.

The main parameters are as following:

- Design pressure: 17.2 MPa;
- Design temperature: 350°C;
- Cooling power: 300 kW;
- Design volume flow: 30 m^3/h;
- Diameter of channel: 260 mm.

The operation parameters such as the pressure, temperature and hydrochemistry can be adjusted to satisfy different irradiation test needs by the cooperation of relative sub-systems.

2.2.2. Helium-3 pressure control loop (proposed)

Helium-3 pressure control loop is used to change the pressure of helium around the test fuel assembly to adjust neutron flux, so that the power of the test fuel assembly is controlled. This loop adjusts the power rapidly, evenly and flexibly, and the irradiated parameters can be controlled accurately.

The main parameters are as following:

- Range of pressure change: 0.5-4.0 MPa;
- Design temperature of tritium trap: 400°C;
- Power ramping rate: 10 $kW \cdot m^{-1} \cdot min^{-1}$;
- Design volume flow: ~1-2 cm^3/s;
- Diameter of channel: 100 mm.

The loop can be used to carry out power ramp test and power cycling test. It can also be used to control local neutron flux of irradiation samples during steady state irradiation.

2.2.3. CARR In-Pile Irradiation and Tritium In-situ Extraction (CIPITISE) (under construction)

The CIPITISE system will be installed in 2015. It consists of the irradiation facility including the pebble bed assembly (PBA) in CARR irradiation hole, and the tritium analysis and monitoring system (TAMS) in CARR operation hall as well as the monitoring system of the irradiation operational parameters in CARR operation control room. After irradiation, the irradiation will be dismantled in CARR operational hot-cell, and then irradiated PBA will be transferred to the 303# hot-cell for PIE, and some irradiation performances will be also carried out.

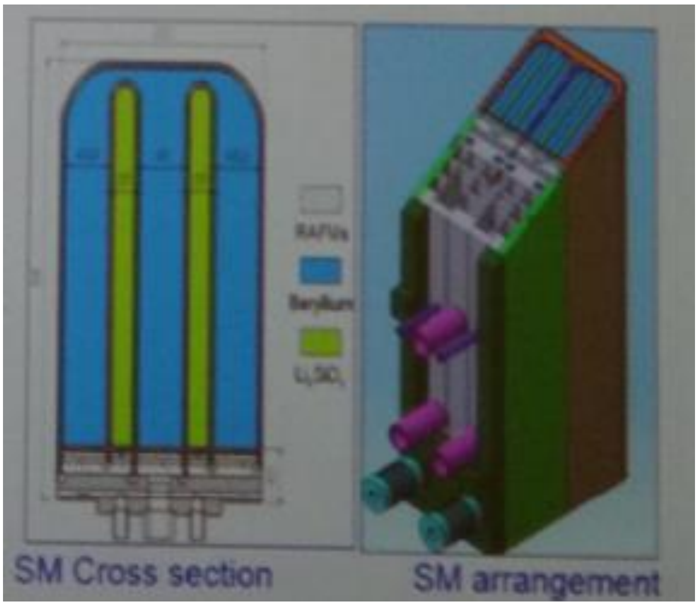

FIG. 3. CIPITISE.

The updated design was composed of twelve sub-modules in the form of six lines and two columns with the breeder out of tube in order to simplify the sub-module structure, reduce mass of the RAFM structure materials and improve TBR performance. The tritium breeder zones would be packing of the lithium ortho-silicate pebble with diameter of 1.0 mm, and isotope abundance of 80% ^{6}Li.

The main parameters are as following:

— Maximum temperature for lithium breeder pebble bed: 735°C;
— Maximum temperature for the RAFM structure steel: 538°C.

Its applying experiments and analysis are as following, which would be performed after 2016 one by one:

— Tests of effective thermal conductivity of the breeder pebble bed under In-pile irradiation;
— Experiments of the tritium release behaviours of the breeder pebble bed under the conditions of reactor power operation;
— Tests of the tritium release and retention performance of the breeder pebble bed by the electrically-assisted heating under the condition of reactor shutdown;app:addword:auxiliary
— Experiments of the tritium permeation behaviours of the RAFM structural steel and the tritium permeation resistance and stability of the barrier coatings on the RAFM surface under In-pile irradiation;
— Post-irradiation examination and study on irradiation swelling and/or damage performance of tritium breeder pebble after irradiation;
— Evaluation and validation of the neutronics and thermo-hydraulic analysis of the HCCB pebble bed;
— Comparison and assessment of the tritium production performance of the different lithium ceramics such as lithium ortho-silicate and meta-titanate pebble prepared by different methods.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFA, RIA, etc.

Until now, these facilities mentioned above have not been considered.

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Using HTHPTL and ^{3}He loop, corrosion test of some structural materials could be conducted.

2.5. DEVICES FOR CAPSULE/AMPOULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES etc.

Helium-3 loop could be conducted for capsule/ampoule tests of materials in different environment, at wide range temperature and dose rates etc.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (swelling, gas release, creep, long-term strength, relaxation resistance, etc.)

There are some devices in the hot-cell №303 could be conducted the investigation of fuel and structural materials behaviour and characteristics (swelling, gas release, creep, long-term strength, relaxation resistance, etc.

2.7. OTHER FACILITIES (this section will be added only in the electronic version and it might include zero or low power facilities supporting innovative nuclear energy projects)

Until now, no new facility could be planned to construct.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

Until now, these facilities mentioned above have not been considered.

3.2. HOT CELLS, PIE FACILITIES (radio chemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

3.2.1. Hot cell

(1) Description of the hot cell;
There are three hot cells in CIAE, including the hot cell of CARR. The hot cell of CARR is a large non-destructive examination hot cell, shielded with heavy concrete and lined with stainless steel. A slope hole is installed inside the side wall of the cell which is connected with storage pool of the reactor. This hot cell is the first one in China which can perform full size non-destructive examination to the fuel rods from nuclear power plant. This facility was designed in April 2002 and operated in 2012.
The hot cell of CARR is the first one in China which can perform full size

non-destructive examination to the fuel rods from nuclear power plant. Inside dimension of the hot cell is 7 m × 2.2 m × 4.1 m（L×W×H), which layout is shown in the Fig. 4. The walls are made of heavy concrete of 4.2 g/cm^3. Thickness of the front wall is 1.3 m, which allows a maximum activity of 3700 TBq (Ci × 10^5) for ^{60}Co. This facility was designed in April 2002 and operated in 2012. Figure 5 shows the front area of the hot cell. The main functions include non-destructive examination of fuel assembly, fuel rod and materials irradiated in CARR, full size fuel rods from PWR, dismantling of radioisotope targets.

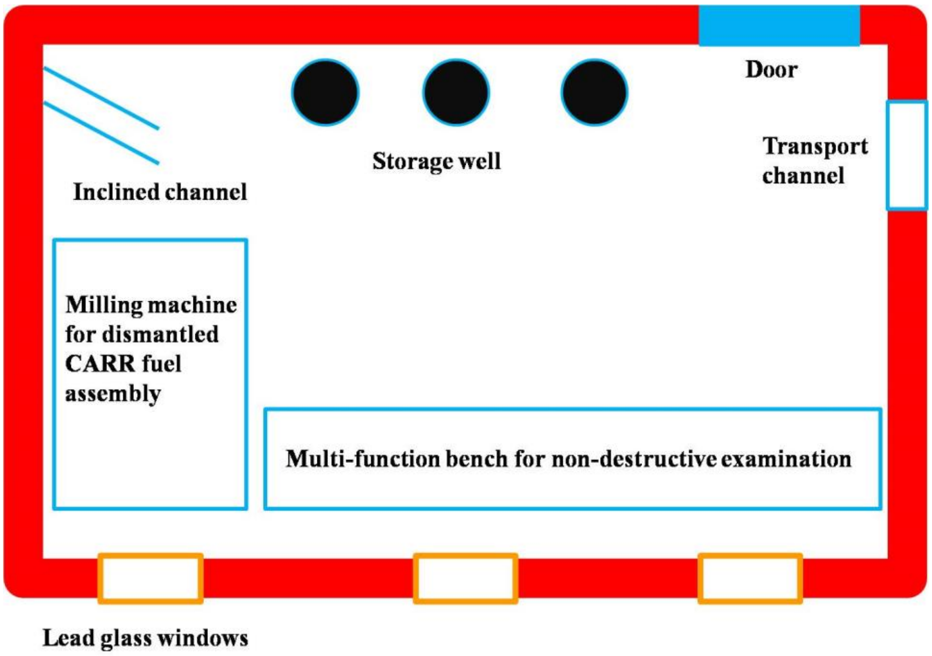

FIG. 4 Layout of CARR hot cell

FIG. 5 Front area of CARR NDT hot cell.

(2) PIE capabilities and main equipment;

CARR hot cell provides the PIE capabilities to perform the following automated and remotely operated examination:

- Visual inspection and photograph of CARR fuel assembly and fuel plate;
- Flow gap measurement of CARR assembly for the routine inspection;
- Dismantling of CARR fuel assembly, only Aluminium structure in the lower part will be cut off;
- Crud removal of fuel rod;
- Non-destructive examination of fuel rods: including visual inspection, dimension measurement, eddy current testing, gamma scanning for relative burn-up measurement and Real time scanning X radiography.

Main equipment is listed as follows:

- Cask for fuel rod and material transport;
- Video and camera for visual inspection of fuel assembly;
- Miller for dismantling of fuel assembly, only Aluminium structure in the lower part will be cut off;
- Multi-function bench for non-destructive examination of full size fuel rod and CARR fuel plate;
- Video and camera for visual inspection of fuel rod and CARR fuel plate;
- Dimensional measurement device for measuring diameter and length of fuel rod;
- Dimensional measurement device for measuring thickness of CARR fuel plate;
- Flow gap measurement bench for CARR assembly;
- Eddy current testing machine contains encircling coil and surface coil for measuring defects and oxide layer;
- Gamma detector and Collimator for measuring relative burn up distribution of fuel rod;
- Real time X radiography system for inspecting the defects, uniformity, structural integral of fuel rod.

3.2.2. Neutron beams facilities

(a) Description of the neutron beams facilities;

There are nine horizontal beam tubes on CARR shown in Fig. 6, including HT1: cold neutron source beam tube, HT2: multi-filtration neutron beam tube, HT3, HT4, HT6, HT8, HT9: thermal neutron beam tubes, HT5: long tangential beam tube, HT7: hot neutron source beam tube.

Horizontal channels with associated equipment and instruments are installed in the heavy water reflector and make it possible for full use of the strong neutron source created by the reactor for neutron scattering experiments, study on nuclear power development and neutron activation analysis and so forth.

(b) Capabilities and main equipment;

With newly installed equipment such as the advanced cold neutron source and neutron guide tubes, CARR will provide powerful capability for conducting a great deal of fundamental and engineering applied researches covering material science, life science, environment science, researches in physical-chemistry fields and in other important relevant areas.

(1) Current capabilities:

With newly installed equipments such as the advanced cold neutron source and neutron guide tubes, CARR will provide powerful capability for conducting a great deal of fundamental and engineering applied researches covering material science, life science, environment science, researches in physical-chemistry fields and in other important relevant areas.

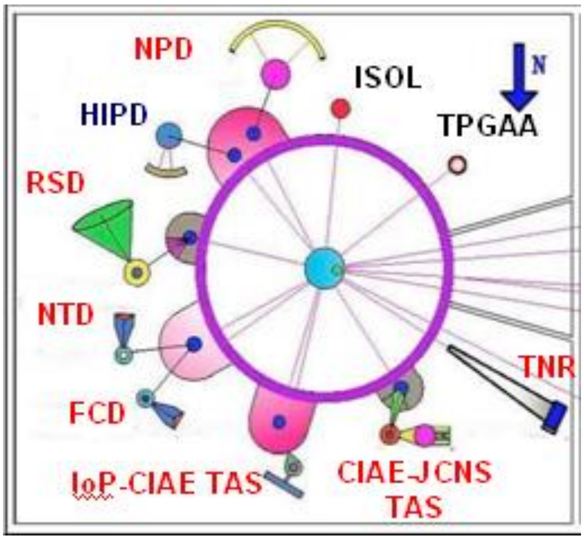

FIG. 6. Neutron beams facilities of CARR.

(2) Main equipment:

- — CTAS: cold trip-axes spectrometer;
- — USANS: ultra-small angle neutron scattering;
- — CTOF: cold time of flight;
- — PNR: polarized neutron reflectometer;
- — CNR: cold neutron radiography;
- — SANS: small angle neutron scattering;
- — NR: neutron Reflectometer;
- — TNR: thermal neutron radiography;
- — HIPD: high intensity powder diffraction;
- — TAS: trip-axes spectrometer;
- — FCD: four circle diffractometer;
- — NTD: neutron texture diffractometer;
- — RSD: residual stress diffractometer;
- — NPD: neutron powder diffractometer.

FIG. 7. TAS of CARR.

FIG. 8. FCD of CARR.

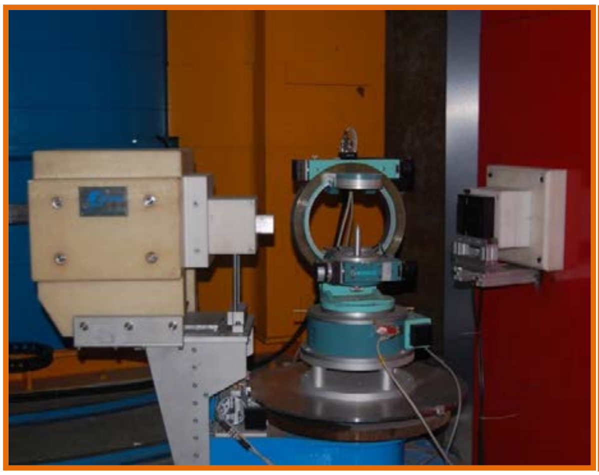

FIG. 9. NTD of CARR.

FIG. 10. RSD of CARR.

FIG. 11. NPD of CARR.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

There are four research reactors and ten zero power facilities in CIAE. Based on experiences of RR design, construction, operation and maintenance, a R&D group with rich experience has been developed. Now, one division in CIAE is in charge of experimental devices and measurement systems design including neutron physics, thermal-hydraulic, mechanism design and I&C design, the number of staff is about 60, most have attended the CARR design. The experimental factory in CIAE is in charge of manufacturing experimental devices, the number of staff is 100.

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS

During the last ten years, CARR had been under construction. Until last year, CARR reached full power, now is licensing for normal operation. Before normal operation, the work on CARR is to install devices and some measurement. Some R&D studies have been performed, but have not been completed until now.

Some information about our R&D studies is as follows:

(1) XU YONG, LI LONG BIN, Simulation Study on the Cold Neutron Guides in China Advanced Research Reactor, Atomic Energy Science and Technology, **37** 4 (2003).

Abstract:

The designs of the two cold neutron guides, CNG1 and CNG2, to be built in China advanced research reactor (CARR) are studied with Monte-Carlo simulation technique. The neutron flux density at the exit of the both guides can reach above 1×10^9 $n \cdot cm^{-2} \cdot s^{-1}$ under the assumed flux spectrum of the cold neutron source. The transmission efficiency is 50% and 42%, and the maximum divergence is about 2.2° and 1.9°, respectively for CNG1 and CNG2. Neutron distribution along horizontal direction is quite uniform for both guides, with maximum fluctuation of less than 3%. Gravity can affect neutron distribution along vertical direction considerably.

(2) HUANG XIN, ZHANG PEI-SHENG, TANG GUO-LIANG, ZHANG AI-MIN, ZHANG YING-CHAO, Research and Design of ^{3}He Pressure Control Loop, Atomic Energy Science and Technology, **42** 8 (2008).

Abstract:

In order to carry out power transient tests for PWR fuel element in CARR, the research and conceptual design of ^{3}He pressure control loop were completed. The working principle, design parameters and technological flow of the loop were described.

Profile 5

ETRR-2

EGYPT

1. GENERAL INFORMATION AND TECHNICAL DATA OF ETRR-2 RESEARCH REACTOR WITH A LINK TO THE IAEA RRDB

1.1. GENERAL INFORMATION[1]

- Owner: Atomic Energy Authority of Egypt[2];
- Operating organization: Atomic Energy Authority of Egypt;
- Regulatory Body: Nuclear and Radiation Regulatory Authority (NRRA);
- International safeguards: IAEA;
- URL: www.etrr2-aea.org.eg.

1.2. HISTORY

- Start of constriction: 1992-12-01;
- First criticality: 1997-11-27.

1.3. TECHNICAL DATA[3]

TABLE 1. TECHNICAL DATA

Reactor type	Pool
Thermal power	22 MW
Forced cooling	2000 m^3/h
Max thermal flux	2.8×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$
Max fast flux	7.6×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$
Moderator material	Light water
Coolant material	Light water
Reflector material / no of side	Be/4
Control, safety, shutdown rods material/ number	Ag-In-Cd/6
Maximum excess reactivity	2/3 of total control rod worth (BOC)
Minimum shutdown margin (pcm)	3000 (BOC)

[1] http://nucleus.iaea.org/RRDB/RR/GeneralInfo.aspx?RId=596.
[2] 13759, Abou Zabal, Egypt.
[3] http://nucleus.iaea.org/RRDB/RR/TechnicalData.aspx?RId=596.

FIG. 1. Outside view of the ETRR-2 and the radioisotopes production facilities (RPF).

The ETRR-2, owned and operated by the Egyptian Atomic Energy Authority (EAEA), is a Multi-Purpose Reactor (MPR), intended for radioisotope production, and is used for research activities, training and neutron radiography.

It is an open pool reactor; its core has 27–30 fuel elements with 19 plates each, made of 19.75% enriched uranium alloy, and a nominal power of 22 thermal MW. The core is at 10 m depth inside 4.5 m diameter pool and surrounded by a chimney through which a flow of demineralized light water is forced upwards. In-core six plates made of silver-indium-cadmium alloy are used to control the core power. This core design is an inherent safety feature (power reactivity feedback coefficient is negative). Furthermore, the reactor has a protection system (RPS) that monitors safety parameters to shut down the reactor if the setting is reached. The reactor includes several irradiation channels, hot cells and systems of neutron radiography for materials and components.

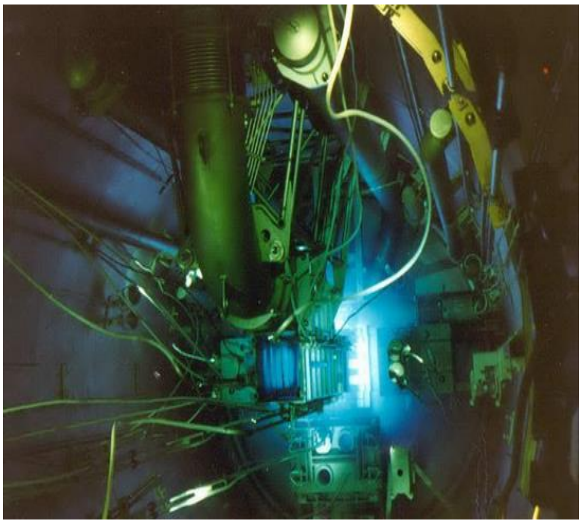

FIG. 2. Reactor pool.

FIG. 3. ETRR-2 hall.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

2.1.1. In-core and in-pool facilities

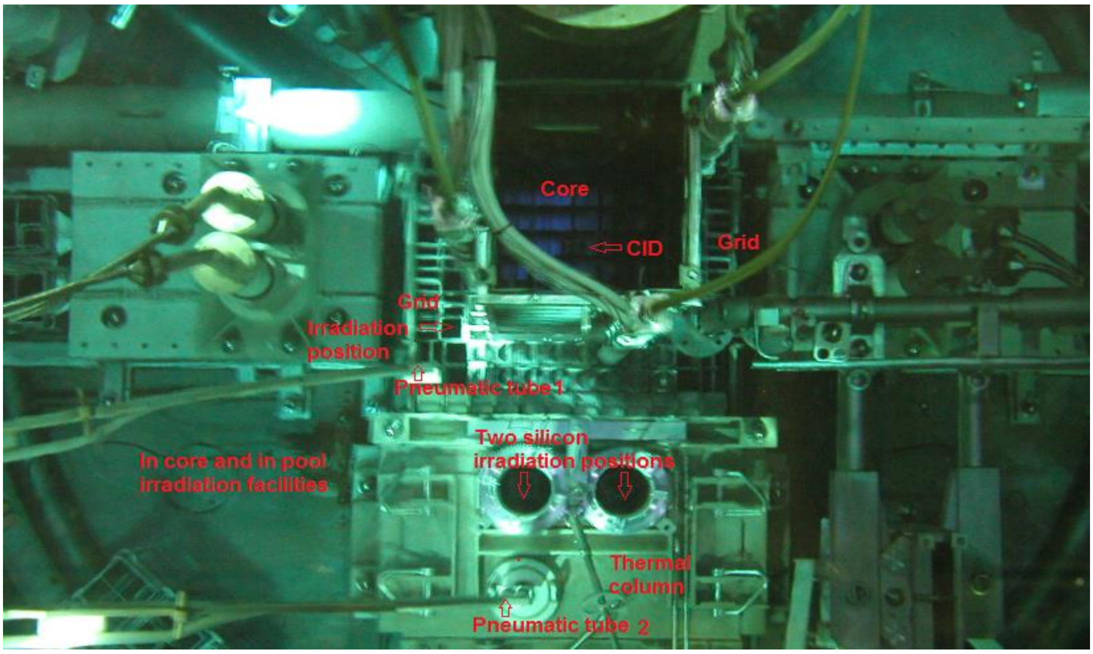

FIG. 4. In core and in pool facilities.

(1) Irradiation facilities:

— One cobalt irradiation device (CID) in the core centre position with neutron flux of 2.7×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$. The CID can accommodate two parts; each part has eight aluminium tubes with cobalt pelts inside. It is utilized for production of ^{60}Co sources. The cobalt cell in ETRR-2 is equipped to be used in cobalt sources processing and calibration.

— Two irradiation boxes can be loaded in two positions in the core with neutron flux of 2×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ to irradiate up to 24 uranium plates (130 mm × 35 mm × 1.4 mm for each) to produce ^{99}Mo. The testing cell and cobalt cell are equipped to be used for loading / unloading the plates in the irradiation box and loading in the dedicated shielded container.

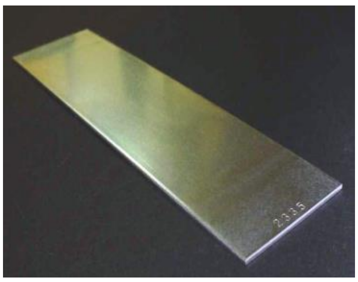

FIG. 4. Uranium plate.

— Twenty-three irradiation boxes at the irradiation grid including six boxes with relatively high neutron flux of 10^{14} n·cm^{-2}·s^{-1} and one box in thermal column of 10^{13} n·cm^{-2}·s^{-1}. Each of these boxes can accommodate a sample holder with 16 aluminium cans for sample irradiation for research and development or for irradiation of common targets for isotope production. Transfer hot cell are used for sample holder loading/ unloading and transfer to/from the pool. Universal and loading hot cells are used for irradiated samples visual inspection, can opening and loading irradiated samples in a shielded container for further transportation to research and development labs or RPF.

(2) Neutron transmutation doping (NTD) irradiation rigs:

— NTD at ETRR-2 consists of two irradiation rigs for silicon irradiation located in two positions at the thermal column. Theses rigs are capable of irradiating ingots up to 28 cm long and five inches diameter (127 mm) with axial resistively variation in the product less than 5% in the product resistivity. Thermal flux is 10^{13} n·cm^{-2}·s^{-1} and thermal to fast flux ratio is 67.7. NTD facility also includes labs for post irradiation tests and measurements.

— The existing silicon irradiation rigs have been upgraded to accommodate larger size ingots of six inches.

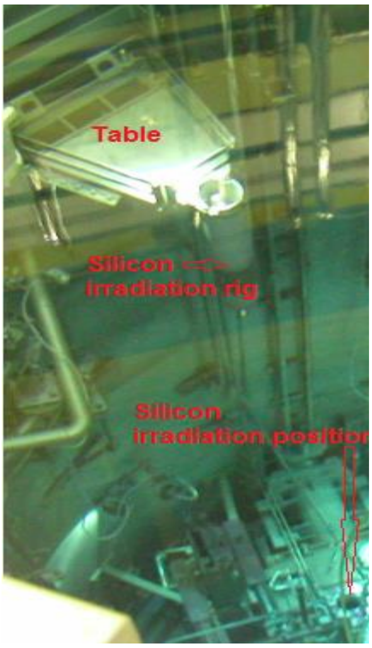

FIG. 5. Neutron transmutation doping (NTD) irradiation rigs.

(3) Pneumatic tubes;

There are two pneumatic tubes (1 and 2) available for fast irradiation for the use in neutron activation analysis applications. The neutron flux at the irradiation positions are 9×10^{13} n·cm^{-2}·s^{-1}, and 6.8×10^{11} n·cm^{-2}·s^{-1}, respectively.

FIG. 6. Sample send/receive stations 1 and 2.

(4) In-pool facilities that can be considered for future application:
- — Irradiation of material samples and specimens for material testing using *material testing hot cell. It is possible to design and manufacture a samples* and specimens *irradiation device*. The material testing cell is equipped with testing machine for performing destructive tests on irradiated samples.
- — A fast neutron irradiation (FNI) channel can be installed out of the core. Aluminium cylinder with boron inside its wall is ideal to be used as FNI channel. It is can be used to shield the thermal neutrons to minimize thermal neutron activation or used fast neutron experiments.
- — The outer positions in the irradiation grid can be made available for irradiation of more 6 inch diameter silicon ingots. A simple irradiation rig design similar to the one used can be adapted in the proposed positions.

2.1.2. Beam port facilities

(1) Neutron radiography facility:
- — ETRR-2 *neutron radiography facility* is mounted in front of a beam radial tube with proper shielding and samples handling mechanisms (thermal flux: 1.05×10^8 n·cm^{-2}·s^{-1}, epithermal flux: 2×10^8 n·cm^{-2}·s^{-1} and thermal flux to gamma exposure ratio: 5×10^5 n/cm^2·mR);
- — The facility (CCD camera, software, controlling system and PC) is operational showing capabilities in radiographic, tomographic and dynamic investigations of various samples.

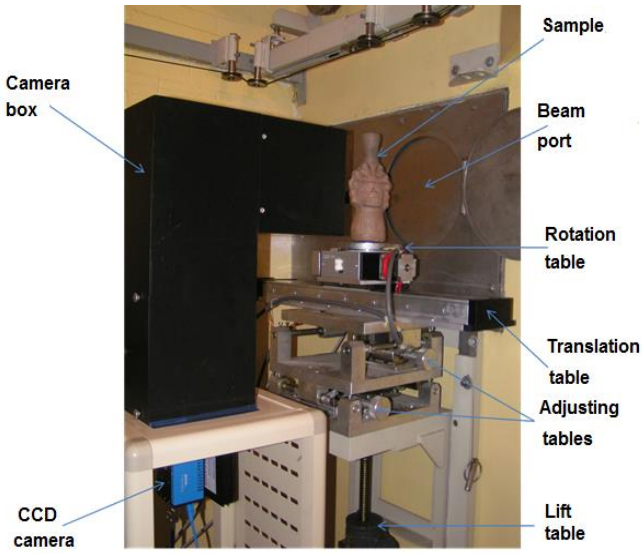

FIG. 7. ETRR-2 neutron radiography facility with its main components.

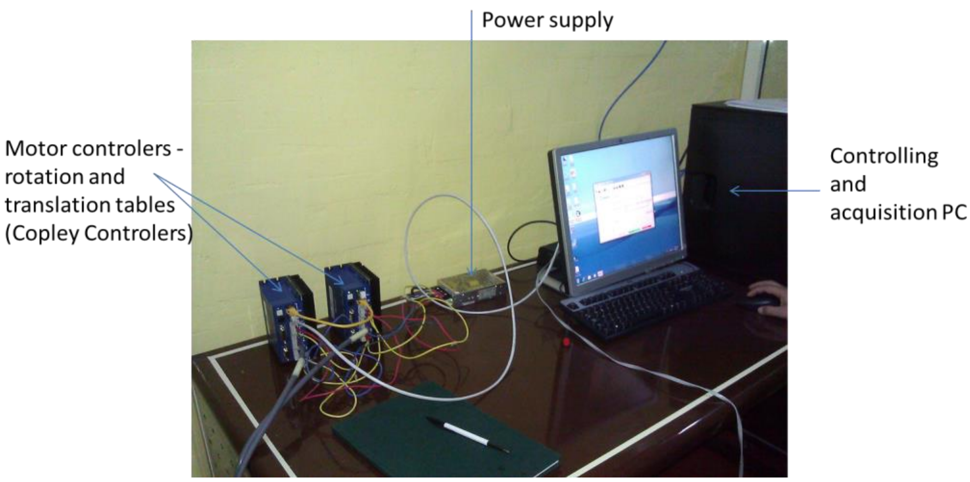

FIG. 8. Controlling and acquisition system of the neutron radiography.

(2) Existing non-operational beam ports and new development that is possible for consideration:

- *Radial beam tube* (thermal flux: 3.3 × 10^9 n·cm^{-2}·s^{-1}, epithermal flux: 7.7 × 10^9 n·cm^{-2}·s^{-1}, fast flux: 2.3 × 10^9 n·cm^{-2}·s^{-1} and gamma exposure rate: 2.2 × 10^6 R/h) is not utilized and research work could be carried out using this beam;
- *Underwater neutron radiography facility* is installed at radial beam tube housed at the main pool and commissioned to be used for irradiated samples non-destructive tests (thermal flux: 1.8 × 10^9 n·cm^{-2}·s^{-1}, epithermal flux: 3.4 × 10^8 n·cm^{-2}·s^{-1}, fast flux: 8.4 × 10^7 n·cm^{-2}·s^{-1}, gamma exposure rate: 3.4 × 10^4 R/h);
- *Tangential beam* (thermal neutron flux: 1.6 × 10^9 n·cm^{-2}·s^{-1}, epithermal neutron flux: 1.3 × 10^9 n·cm^{-2}·s^{-1}, fast neutron flux: 9.2 ×10^7 n·cm^{-2}·s^{-1} and gamma exposure rate: 7.3 × 10^5 R/h) has two ports; one can be used for installation of SANS instrument since it includes space and physical layout suitable for the installation. The design and installation of SANS is expected to enhance ETRR-2 capabilities for applications on material science and engineering.

— The *tangential beam* other port can be improved and adapted for implementation of prompt-gamma neutron activation analysis (PGAA) system. The implementation of PGNAA will strength the applications of ETRR-2 on elemental analysis and will add to the existing neutron activation analysis (NAA);
— The ETRR-2 thermal column port is still not utilized for neutron beam applications. The thermal column port includes special shielded room to be used for the installation of boron neutron capture therapy (BNCT) capabilities. The design and installation of BNCT will support research in use research reactors for the treatment and palliation of cancer patients.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

Not finalized installation of high pressure 500 kW test *loop* and 20 kW rig for testing fuel bundle or rod of nuclear power plant at steady state and transient conditions.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOFT AND RIE

2.3.1. Loss of flow transient (LOFT) experiment description

— LOFT experiment can be performed at ETRR-2 to measure the outlet temperatures profile inside the core chimney in addition to core inlet temperature during the LOFT. The outlet temperatures (T_0, T_1, T_2, and T_3) measurements are measured using thermocouples located vertically above one fuel element. The core inlet temperature is measured with PT-100 RTD specified in accordance to DIN 43760 Class A (permitted deviation better than ± 0.8 and response time is 5 s).
— The reactor is operated at steady state total power level (MW) when the core cooling pump and secondary pump are manually stopped and reactor is scram. The flows of primary pumps cost down and total secondary flow are measured. The redundant flappers (two valves) installed for natural cooling including devices to verify open/close status.

(1) Inputs:
- Core cooling flow rate (m^3/h);
- Secondary flow rate (m^3/h);
- Tower outlet temperature (^oC);
- Power peaking factor;
- Flapper valves opening time (s);
- Primary and secondary flow cost down;
- Core flow at scram time (m^3/h).

(2) Experiment results:

Time (sec)	Core inlet (^oC)	T_0 (^oC)	T_1 (^oC)	T_2 (^oC)	T_3 (^oC)

2.3.2. Negative reactivity insertion experiment (RIE) description

The reactor is operated at steady state power (MW) when one control rod dropped in the core from extraction level (%). The control rod worth vs extraction level is given as well as plant operation conditions. The results are the normalized reactor power and inlet and outlet core temperature with time. The temperatures are measured with PT-100 RTD specified in accordance to DIN 43760 Class A (permitted deviation better than ± 0.8°C and response time is 5 s).

(1) Inputs:
- Core cooling flow rate (m^3/h);
- Secondary flow rate (m^3/h);
- Tower outlet temperature (°C);
- Power peaking factor;
- Inlet pressure (mbar);
- Kinetic parameters;
- Feedback coefficients;
- Drop time of the rod (ms) from the extracted level;
- Control rod worth:

Extraction %	Control rod worth ($)

(2) Experiment results:

Time (s)	Normalized power %	Core inlet (°C)	Core outlet (°C)

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

(1) Monitoring the water quality by measuring:
- PH value and alkalinity;
- Chloride and sulphate;
- The electrical conductivity of the coolant and feed water (to calculate the concentration number);
- Total and calcium hardness of the coolant and feed water;
- The residuals of the added treatment chemicals (usually zinc and phosphate);
- Free chlorine residual.

(2) Monitoring corrosion behaviour in a semi-open system (secondary cooling system):
- The residual iron is analysed in the system monthly or whenever it is needed (higher iron content indicates active corrosion);
- There is a corrosion test rack mounted in the return line of the cooling tower system. A carbon steel coupon is inserted in the rack and left there for one-month period. The coupon is thus subjected to the same circumstances of the secondary system. After that it is removed, inspected visually, cleaned according to standards and weighed to estimate the corrosion rate from the weight loss of the coupon.

(3) *In closed systems* (primary cooling and water ventilation systems) the same tests can be carried out. In addition, copper coupons may be inserted as these systems contain copper as construction materials of some of its components. The residual copper is also analysed when necessary.

FIG. 9. Corrosion test rack.

FIG. 10. A carbon steel coupon after one-month insertion.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES etc.

— The sample holders (can accommodate 16 aluminium cans each) are used for sample and common targets irradiation;
— The irradiation boxes are used to irradiate uranium plated irradiation for ^{99}Mo and can be used to irradiate mini fuel plates;
— A specific device to irradiate specimens for material testing can be designed and manufacture at the ETRR-2 workshop.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS

The material testing hot cell is equipped with testing machine for performing destructive tests on irradiated samples and specimens of fuel and structural materials. Technical specifications of the installed machines and tests that could be carried out are:

— Universal testing machine;
  Fully instrumented ZWICK Z 15015 N5A floor type machine could carried out tensile, fracture mechanics and binding tests.
▪ Impact machine:
  Type: PW5 instrumented impact testing machine;
  Energy (max.): 25 J;
  Testing sub size charpy specimen of dimension 4 mm × 3 mm × 27 mm;
  Temperature range: –100 to 300°C;
  Test output: impact energy, load-time and energy-time curve and energy-temperature transition curve.
— Micro hardness tester:
  Type: AMH 3000;
  Load: 10, 25, 50, 100, 300, 500, and 1000 gf (one gram-force 1 gf = 0.00980665 N);
  Focusing, specimen movement and test are conducted.

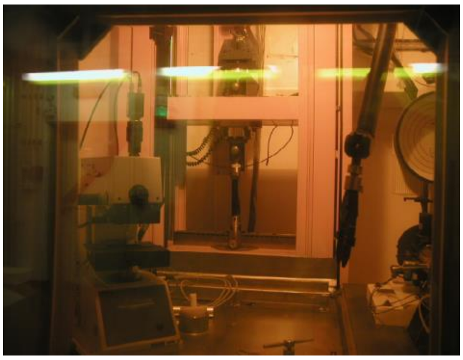

FIG. 11. Material testing hot cell.

FIG. 12. PC control and test results.

2.7. OTHER FACILITIES

2.7.1. NAA labs

Environmental, geological, and biological samples could be analysed for different applications at the NAA labs, which provided with the following measurements devices/systems:

— *Compton suppression system* consists of 100% efficiency HPGe detector, coincidence and anti-coincidence electronic modules, ultra-low back ground lead shield and complete qualitative and quantitative software;
— *HPGe spectroscopy systems* (There are *two* HPGe spectroscopy systems; one P-type and the other is N-one). Each of them is composed of 100% HPGe detector, ultra-fast counting rate electronic modules, ultra-low back ground lead shield and complete qualitative and quantitative software;
— *NaI (Tl) spectroscopy system* consists of 4 inch × 4 inch NaI(Tl) crystal, modular electronic circuit, ultra-low background shield and qualitative and quantitative analysis software.

FIG. 13. NAA lab.

2.7.2. Large sample neutron activation analysis (LSNAA)

— Irradiation and counting facilities are installed and tested for *LSNAA for inhomogeneous bulk archaeological samples and bulk objects*. The existing NTD irradiation positions and rigs are used for irradiating large samples with diameter up to 15 cm and length of 50 cm or larger (less than 80 cm). A ceramic bottle was analysed using LSANN facilities at ETRR-2;

— The counting facility consists of HPGe detector (100% relative efficiency and 2.1 keV resolution) and associated electronics, sample holder, horizontal stepper motor to rotate the sample around its vertical axis, vertical stepper motor to move the sample table in the vertical direction and control unit to control the motion of the motors. Collimator for sample scanning can be mounted between the sample and the detector.

FIG. 14. Counting facility for LSNAA at ETRR-2.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

- Specific and ordinary operational tools;
- Sample carrier to transfer the irradiated plates or samples from the pools to hot cells;
- Cranes;
- Auxiliary pool to carry out the activities:
  - Assembly and disassembly of CID and ^{60}Co storage loading in shielded container;
  - Assembly and disassembly of irradiation box of uranium plates and storage of irradiated plates;
  - Storage of irradiated sample and common targets.
- Under water tables and supporting structure;
- Shielded containers:
  - Irradiated samples shield;
  - The shielded container of ^{60}Co;
  - The shielded containers used for the transportation of irradiated uranium plates and targets of common isotopes:
    - CTTA shield is used for the transportation of uranium plates coming from ETRR-2 to the production hot cells in RPF;
    - SATE shielded container is used to transport irradiated common targets and waste management.

FIG. 15. CTTA.

FIG. 16. SATE.

3.2. HOT CELLS

— The reactor has five hot cells, namely cobalt cell, transfer cell, testing cell, universal cell and loading cell. Hot cells are provided with the associated services (ventilation, electrical and instrumentation, liquid waste disposal, fire extinction, and compressed air and dematerialized water):
  - Irradiated samples and targets are transferred from the reactor pool to the transfer cell using a mechanical vertical transport system (sample carrier). A conduit is provided to connect the transfer cell with the universal cell and loading cell to transfer the irradiated samples;
  - Irradiated samples to be tested and irradiated uranium plates are transferred from the auxiliary pool to the inside of the testing cell using a mechanical vertical transport system similar to that used by the transfer cell;
  - Irradiated ^{60}Co sample targets have a different circulation. Once the irradiated Co pencils have been loaded inside a shielded container in the auxiliary pool, the container is transferred top of the pool level to ground level where it is loaded onto a cart to be transported to the cobalt cell;
  - A conduit with diameter of 2.5 inches is provided to transfer irradiated samples/ plates from the testing cell to the cobalt cell.

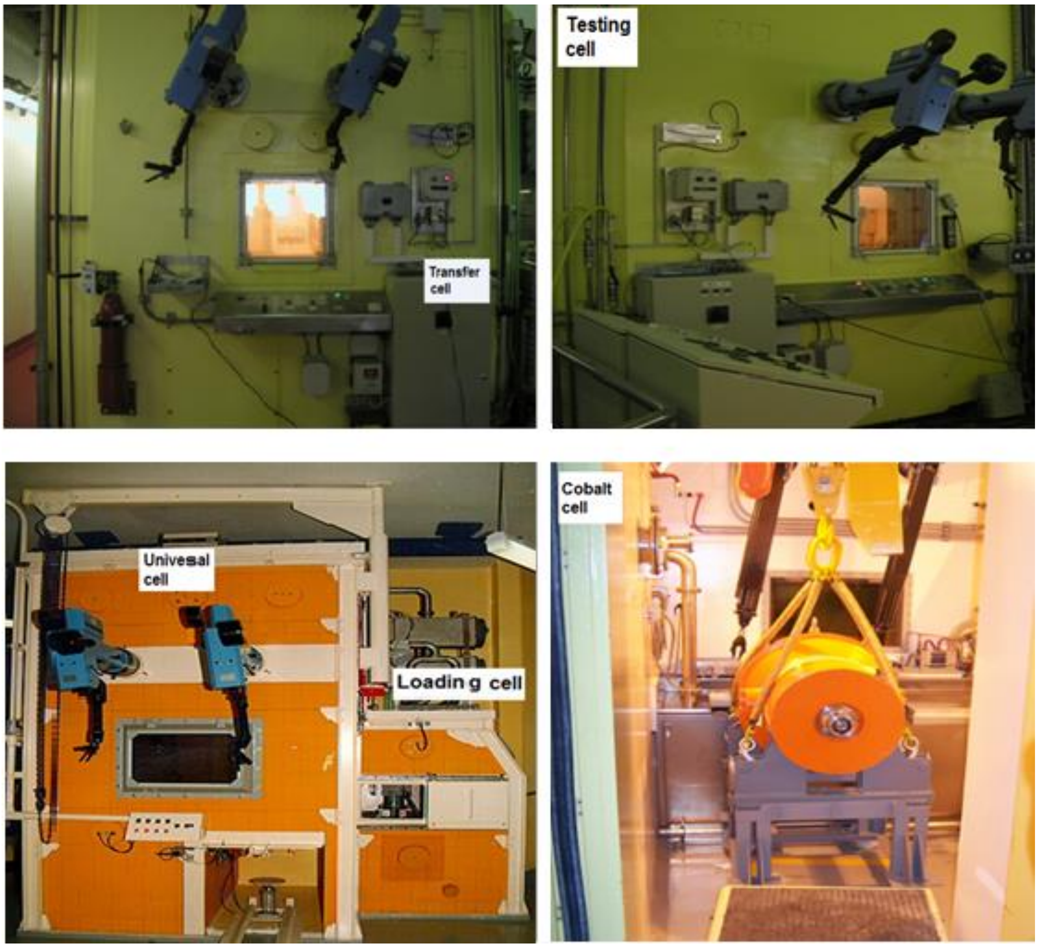

FIG. 17. Hot cells.

— The irradiated uranium plates from the ETRR-2 are processed in production hot cells for ^{99}Mo production, ^{99m}Tc loading and ^{131}I production. Also, the irradiated common targets are processed to produce ^{125}I, ^{192}Ir (wire), ^{192}Ir (foil) and ^{51}Cr.

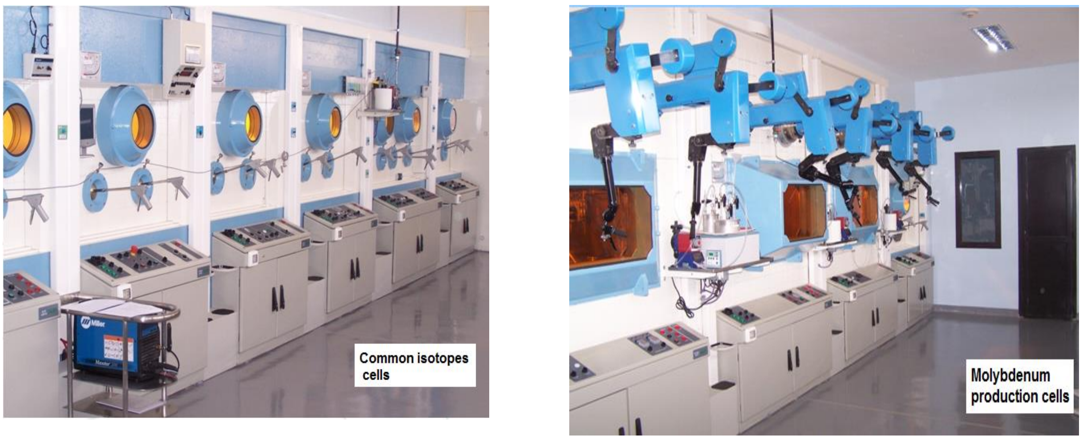

FIG. 18. Radio isotope production cells.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES

Capabilities to design and manufacture experimental devices and measurement systems are existing including human resources, workshops, electrical and instrumentation labs and services.

4. RECENT ACHIEVEMENTS (some examples of research and development studies performed during the last 10 years)

- Research and development:
  - Implementation of k0-standardization method of the INAA at ETRR-2;
  - Development of engineering, calculations and preparation of license documentation for irradiation of common isotopes ^{51}Cr, ^{192}Ir, ^{125}I;
  - Upgrading of the neutron transmutation doping irradiation facilities;
  - Development of NAA facilities at ETRR-2 for irradiation and detection of large samples;
  - Development of ETRR-2 neutron radiography facility to be real time in cooperation with IAEA.
- Participated in the IAEA coordinated research projects (CRP):
  - Developing techniques for small scale indigenous ^{99}Mo production using low enriched uranium (LEU) fission or neutron activation, (2007-09-15 to 2011-11-30) and participated in related IAEA activities;
  - Application of Neutron Activation Analysis for Investigation of Large Samples, 2008–1012 (related IAEA TECDOC is in press);
  - Benchmarking against experimental data of neutronics and thermal-hydraulics computer codes for research reactor analysis (2008–1012). Designed, using ETRR-2 experimental data, benchmark problems, namely steady state temperatures, loss of flow, insertion of negative reactivity, and core neutronic (related IAEA TECDOC is in press);
  - Application of 3D neutron imaging and tomography in cultural heritage research commenced in May 2012;
  - Development of an integrated approach to routine automation of neutron activation analysis (2012–2015);
  - Improved instrumentation and control maintenance techniques for research reactors using plant computer commenced in December 2012;
  - Establishment of material properties database for irradiated core structural components for continued safe operation and lifetime extension of ageing Research reactors to be commenced in November 2013.
- Publication of some examples of ETRR-2 studies and cooperation with Egyptian universities.

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Profile 6

JULES HOROWITZ REACTOR (JHR)
FRANCE

1. ABSTRACT

Material testing reactors (MTR) have always been of primary importance in France (SILOE, OSIRIS) for R&D in support to nuclear industry, public bodies and research institutes.

Some of the experimental capacities presented in this paper are on-going in the present OSIRIS MTR and their enhancements for the future JHR research reactor are described below. New developments are going-on to enlarge the existing OSIRIS experimental capacity and are also presented here. Transfer of *know-how* on scientific and technical skills from OSIRIS to JHR is of particular interest for establishing modern JHR tools.

The Jules Horowitz Reactor (JHR) is a new material testing reactor currently under construction at CEA Cadarache Research Centre in the south of France. It will represent a major research infrastructure for scientific studies dealing with material and fuel behaviour under irradiation (and is consequently identified for this purpose within various European road maps and forums; ESFRI, SNE-TP, etc.). The reactor will also be devoted to medical isotopes production.

The reactor will perform R&D programmes for the optimization of the present generation of nuclear power plants (NPP), support the development of the next generation of NPP (mainly LWR) and also offer irradiation possibilities for future reactors.

JHR is fully optimized for testing material and fuel under irradiation, in normal and non-normal conditions:

- With irradiation loops producing the operational condition of the different power reactor technologies;
- With high thermal and fast flux capacity to address existing and future NPP needs.

JHR is designed, built and will be operated as an international user-facility open to international collaboration. This results in several aspects:

- A partnership with the funding organizations gathered within a consortium;
- The preparation of the JHR International Programme (JHIP) as an OECD project open to non-members of the JHR consortium.

It will answer needs expressed by the scientific community (R&D institutes, TSO, etc.) and the industrial companies (utilities, fuel vendors, etc.).

Consequently the JHR facility will become a major scientific hub for cutting edge research and materials investigations (multilateral support to complete cost effective studies avoiding fragmentation of scientific effort, access to developing countries to such state of the art research reactor facilities, supra national approach, etc.).

This paper gives an up-to-date status of the construction and of the developments performed to build the future experimental capacity and will particularly focus on proposed operating rules of JHR as an International User Facility on Research Reactors.

2. INTRODUCTION

European material testing reactors (MTR) has provided an essential support for nuclear power programmes over the last 40 years within the European Community. However, these MTRs will be more than 50 years old in this decade and will face increasing probability of shut-down due to the obsolescence of their safety standards and of their experimental capability. Such a situation cannot be sustained long term since “nuclear energy is a competitive energy source meeting the dual requirements for energy security and the reduction of greenhouse gas emissions, and is also an essential component of the energy mix” [1].

Associated with hot laboratories for the post irradiation examinations, MTRs are structuring research facilities for the European research area in the field of nuclear fission energy.

MTRs address the development and the qualification of materials and fuels under irradiation with sizes and environment conditions relevant for nuclear power plants in order to optimize and demonstrate safe operations of existing power reactors as well as to support future reactor design:

— Nuclear plants will follow a long-term trend driven by the plant life extension and management, reinforcement of the safety, waste and resource management, flexibility and economic improvement;
— In parallel to extending performance and safety for existing and coming power plants, R&D programmes are taking place in order to assess and develop new reactor concepts (Generation IV reactors) that meet sustainability purposes;
— *In addition, for most European countries, keeping competences alive is a strategic cross-cutting issue*; developing and operating a new and up-to-date research reactor appears to be an effective way to train a new generation of scientists and engineers.

This analysis was already made by a thematic network of Euratom 5th FP, involving experts and industry representatives, in order to answer the question from the European Commission on the need for a new material testing reactor (MTR) in Europe [2].

This entire preparatory work leads to the fact that the *JHR research infrastructure has been identified on the ESFRI Roadmap since 2008.*

3. THE JHR PROJECT IN THIS CONTEXT

JHR will offer modern irradiation experimental capabilities to study material & fuel behaviour under irradiation. JHR will be a flexible experimental infrastructure to meet industrial and public needs within the European Union related to present and future nuclear power reactors.

JHR is designed to provide high neutron flux (twice as large as the maximum available today in MTRs), to run highly instrumented experiments to support advanced modelling giving prediction beyond experimental points, and to operate experimental devices giving environment conditions (pressure, temperature, flux, coolant chemistry, etc.) relevant for water reactors, for gas cooled thermal or fast reactors, for sodium fast reactors, etc.

These objectives require representative tests of structural materials and fuel components as well as in-depth investigations with ‘separate effects’ experiments coupled with advanced modelling.

For example, the JHR design accommodates improved on-line monitoring capabilities such as the fission product laboratory directly coupled to the experimental fuel sample under irradiation.

As a modern research infrastructure, JHR will contribute to the development of expertise and know-how, and to the training of the next generation of scientists and operators with a positive impact on nuclear safety, competitiveness and social acceptance. The JHR is mainly designed to meet these technical objectives.

As another important objective, *the JHR will contribute to secure the production of radioisotope for medical application. This is a key public health stake.*

JHR, as a future international User Facility, is driven by an international consortium gathering industry (utilities, fuel vendors, etc.) and public bodies (R&D centres, TSO, regulator, etc.). The economical model of JHR consortium is the following:

- CEA remains the owner and the nuclear operator of the nuclear facility with all liabilities;
- JHR consortium members are the owner of guaranteed access rights to the experimental capacities in proportion of their financial commitment to the construction and with a proportional voting right in the Consortium Governing Board;
- A member can use totally or partly his access rights for implementing proprietary programmes with full property of results and/or for participating to the Joint International Programmes open to non-members;
- JHR consortium membership is open to new member until completion of the reactor.

CEA is encouraging by this consortium to enlarge its membership and in 2013 a new member enter JHR consortium: the Nuclear National Laboratory from England (NNL).

Consequently, the present members list of JHR consortium is the following:
CEA (France), EDF (France), AREVA(France), Euratom, SCK.CEN (Belgium), UJV (Czech Republic), VTT(Finland), CIEMAT(Spain), Vattenfall (Sweden), DAE(India), IAEC (Israel), NNL (UK).

The JHR facility description and the development of the first experimental capacity is described in detailed in previous RRFM and IGORR conferences [3, 4, 5 and 6].

4. JHR UPDATE STATUS

The construction of JHR which was started in 2007 is going-on in a nominal way (foreseen operation by the end of this decade).

FIG. 1. View of the building site in 2013.

5. JHR SAFETY: AN INNOVATIVE APPROACH

The JHR differs from the previous generation of reactors by incorporating the safety analysis right from the design phase, based on a modern reference system and methodology; in particular those used in contemporary projects such as the EPR a GEN3 NPP under construction in Finland, China and France.

The methodological safety approach for the JHR is summarized, highlighting various innovative aspects and the specific design features of the new experimental reactor.

Then, some of the initial design choices and options are detailed, coming directly from this innovative approach and feedback from existing reactors.

The JHR Safety approach has been presented in detail at the IAEA General Conference on Research Reactors in Rabat last November 2011 and some examples of incorporating Safety from the design phase are described in [7].

6. DEVELOPING A MODERN EXPERIMENTAL CAPACITY

JHR is designed as a high performances MTR (thermal power up to 100 MW) and will be able to perform at the same time about 20 experiments. Characteristics are as follows:

- Thermal neutrons flux in reflector: up to 5.5 10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$;
- Fast neutron flux in the core: up to 5.5 10^{14} n/ $n{\cdot}cm^{-2}{\cdot}s^{-1}$ for E >1 MeV and/or up to 10^{15} $n{\cdot}cm^{-2}{\cdot}s^{-1}$for E>0.1 MeV;
- Material ageing: up to 16 dpa/y;
- Six displacement systems to adjust fissile power and perform power transients;
- Power transients for fuel limit to clad failure studies: up to 600 W/cm.

JHR will operate with 10 cycles a year (so about 260 EFPD).

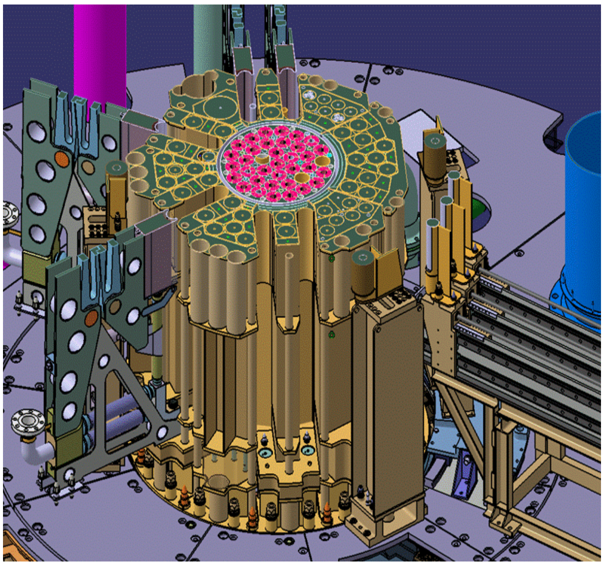


FIG. 2. View of JHR core and reflector where will be located experimental loops.

CEA with its partners is preparing the first experimental capacities by developing some modern experimental devices for fuel and material behaviour studies under irradiation described in detail in the two following paragraphs.

Compared to the existing experimental capacities worldwide, a great effort is on-going to improve the performances of such loops and to develop new devices with innovative concepts by:

— Better monitoring and follow-up of the irradiation conditions;
— Having a lot of *on-line instrumentation* to address key parameters (fast and thermal neutron fluxes, gamma heating, temperature, fission gas release for fuel investigation, material elongation, etc.);
— Having up-to-date post-irradiation exams either directly within JHR nuclear building (for non-destructive assay) or in Cadarache Hot Laboratory (for NDA and DA).

7. EXPERIMENTAL CAPACITY FOR FUEL INVESTIGATION

The design of the reactor (see Fig. 2) provides irradiation sites located within the reactor core with the highest ageing rate (up to 16 dpa/year) and irradiation sites located in the Beryllium reflector zone surrounding the reactor (with the highest thermal neutron flux). Numerous locations are implemented (up to 20 simultaneous experiments) with a large range of irradiation conditions:

— Seven in-core locations of small diameter (32 mm) with a high fast flux (up to 5.5E14 $n \cdot cm^{-2} \cdot s^{-1}$ perturbed flux above 1 MeV);
— Tree in-core locations of large diameter (80 mm) with a high fast flux (up to 4.0E14 $n \cdot cm^{-2} \cdot s^{-1}$ perturbed flux above 1 MeV);
— Twenty fixed positions (about 100 mm of diameter and one location with 200 mm) with a high thermal flux (up to 3.5E14 $n \cdot cm^{-2} \cdot s^{-1}$ perturbed flux);
— Six positions in water channels through the Beryllium reflector (on displacement systems).

A typical reactor cycle is expected to last 25 days, and the target is to operate the reactor 10 cycles/year.

7.1. USE OF DISPLACEMENT SYSTEMS FOR FUEL EXPERIMENTS

The LWR fuel test devices are located on displacement systems in the Beryllium reflector. These systems can move forwards and backwards from the core tank. Consequently they allow an accurate fuel sample linear heat rate control (see Fig. 3) and can reproduce complex linear power time histories. The total stroke of the displacement systems is up to 450 mm.

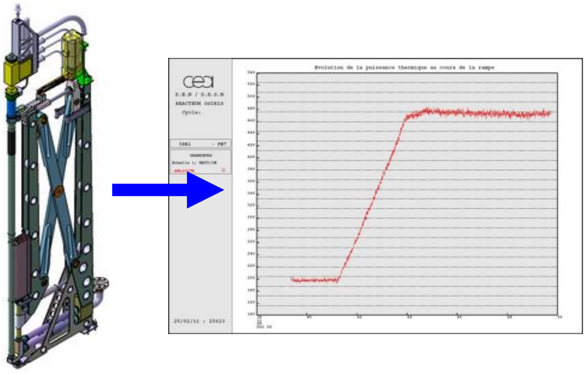


FIG. 3. View of a displacement system in the Beryllium reflector and an example of a realistic power ramp profile achievable in the JHR facility.

The maximum velocity tolerated towards the reactor core is 5 mm/s. This velocity allows carrying out high quality power ramp experiments up to 700 $W \cdot cm^{-1} \cdot min^{-1}$ even for very high burn-up fuels (120 GWd/t).

Such a concept is convenient for the operation (easy to handle, even during the operating cycle of the core) and for the safety (the safe back position and the off normal conditions of the device are not directly coupled to the core operations). In addition, the design of the displacement device allows a withdrawal velocity of 50 mm/s towards its safe back position.

7.2. NON-DESTRUCTIVE EXAMINATION BENCHES

The JHR experimental programmes will also take advantage from non-destructive examination (NDE) benches, implemented in the facility with the objective of significantly improving the scientific quality of the JHR irradiation process:

- A coupled gamma scanning-X tomography bench located in the reactor pool (designed to host entire test devices);
- A neutron radiography bench located in the reactor pool;
- A coupled gamma-X tomography bench, identical to the previous one and located in the storage pool of the nuclear auxiliary building;
- Non-destructive sample examinations in hot cells after extraction of the experimental sample out of the test devices.

These benches can be used at several steps of the experimental programme:

— For initial check of experimental load status just before the beginning of irradiation (after transportation, assembly in JHR hot cell or insertion in the device);
— For adjustment of experimental protocol or control parameters after a first short irradiation run;
— For a rapid examination of samples after an irradiation phase (e.g. geometrical changes after an off-normal transient, quantification of short half-life fission product distribution, etc.) when the examination is performed in the reactor pool with limited handling.

For a delayed detailed examination of samples after the end of an irradiation programme when the examination is performed in the storage pool, either for scientific data acquisition or for defining samples status before transportation.

7.3. HOT CELLS

One interest of the JHR facility is the possibility to use hot cells located in the auxiliary building (see Fig. 1). These cells are connected to the reactor pool thanks to a network of water channels going under hot cells and allow a rapid transfer from hot cells to reactor pool.

The hot cell systems are made of four large cells (of 12 m height) and three adjacent smaller ones. For test devices, the large cells allow the loading of an entire in-pile part and it is dedicated to:
— Loading and un-loading of irradiation rigs from its in-pile containment;
— Mounting and un-mounting of fuel samples and instrumentation from irradiation rigs.

The smaller cells are mainly dedicated to non-destructive examination of samples. Moreover, a so-called 'alpha cell' is devoted to host and handle degraded samples.

7.4. OTHER LABORATORIES IN SUPPORT OF THE EXPERIMENTAL CAPACITY

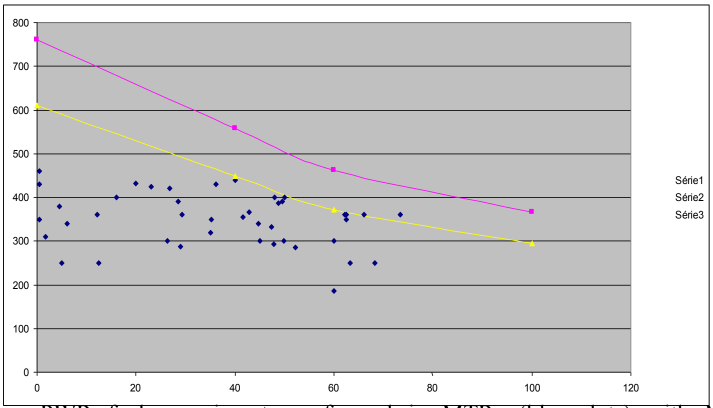
The JHR facility is equipped with several laboratories installed in the nuclear facility:
— A Fission Product Laboratory is implemented in the reactor building. It allows analysing radioactive and stable fission product releases from fuel experimental devices (mainly noble gases and volatile isotopes). For some fuel experiments, the fuel samples (under irradiation in a site of the reactor) can be directly connected to the fission laboratory (through under-water pipes and by means of a sweeping gas). This allows on line recovery and follow-up of fission gas release with a very short transit time. Sampling and temporary storage for delayed measurements are also possible;
— A Chemistry Laboratory is implemented in the auxiliary building. This facility allows performing chemical and radiochemical analyses dedicated to experimental purpose (chemical composition of water loops, chemical analyses of samples, etc.) or to support the JHR operation;
— A Radiochemistry and Dosimetry Laboratory.

7.5. THE MADISON DEVICE (NORMAL CONDITIONS) — NEW FOR JHR

This experimental device will carry out irradiations of LWR fuel samples (60 cm fissile stack) when no clad failure is expected. Consequently, the experimental conditions correspond to normal operation of power reactors (steady states or slow transients that can take place in power plants). IFE Halden is entrusted by CEA the detailed design, manufacturing and assembly in JHR site of most components from this experimental device to account for the long feedback of this Institute in the building of water loops, irradiation rigs and in-core instrumentation.

7.6. AN ABILITY TO REPRODUCE OPERATING CONDITIONS OF POWER REACTOR PLANTS

This experimental device is made of an in-pile part (holding the fuel samples) situated on a displacement system. This system allows on-line regulation of fuel linear power on the samples. Thanks to the high thermal neutron flux in the JHR reflector, this is possible to reach high linear power even on high burn-up samples (as an example, it is possible to reach 400 W/cm for a burn up of 80 GWd/t for a common UO_2 fuel of initial enrichment 4.95%, (see Fig. 4)).



Note: Comparison of former PWR fuel experiments performed in MTRs (blue dots) with MADISON calculated performances:

- on pink curve the results are best estimate;
- on yellow curve, a margin of 20% is taken on the performances.

FIG. 4. Fuel linear power (W/cm) as a function of burn-up (GWd/t) for a UO_2 fuel (4.95% enriched).

The in-pile part is connected to a water loop providing thermal-hydraulics conditions expected for a given experimental programme. The water loop (implemented in a dedicated cubicle (see Fig. 5)) allows reproducing the thermal-hydraulics conditions of nuclear power plants (PWR, BWR or WWER technologies) in terms of water loop pressure (up to 160 bar) and temperature (up to 320°C).

A specific chemical analysis system and a water treatment system allow a continuous regulation of chemical conditions. Usual chemical conditions of power plants can be reproduced as well as specific chemical conditions (depending on customer needs).

7.7. A LARGE AND FLEXIBLE HOSTING CAPACITY

In order to meet the large range of experimental needs expressed by the nuclear industry, the test section of the in-pile part has a large volume. This allows loading a large panel of sample holders from high embarking capacity (up to eight samples) with low instrumentation to low embarking capacity (up to one sample), but highly instrumented.

The design of MADISON irradiation rig allows re-using it as maximum. In particular, it is equipped with fast and reusable connectors that allow rapid connections of fuel samples instrumentation.

This design allows performing several types of experiments:

- Selection experiments: to irradiate numerous innovative fuel samples in identical conditions and select the most promising microstructures;
- Characterization experiments: to irradiate few samples (one or two) with a lot of instrumentation and measure on-line physical properties of the products: fuel temperature, clad temperature, clad elongation, clad diameter, fuel stack elongation, fuel plenum pressure, fission gas release, for fuel performance codes validation;
- Qualification experiments: to irradiate several samples under representative normal conditions of nuclear power plants (base and operating transients), in order to check if the products have adequate behaviour (compatibility with cycle length or with expected lifetime, chemistry, etc.).

7.8. HIGH PERFORMANCE INSTRUMENTATION

The MADISON device provides high performance instrumentations currently used in the nuclear industry. The first irradiation rig version has a carrying capacity of four fresh or pre-irradiated samples (with a maximum of two sensors per sample) and is flexible enough to operate with two samples (highly instrumented).

Some instruments are fixed on the water loop and measure on-line water loop thermal-hydraulics conditions (pressure, temperature, water flow, chemistry) and neutron flux conditions (thermal neutron flux, fast neutron flux). In addition, the design of the test section allows on-line measurement of thermal balance in the test section that provides a fuel samples linear power with a targeted 5% precision.

Some instruments are implemented on the fuel samples. For that purpose, five tight high temperature and pressure connectors are implemented on the sample holder to allow the plug-in of specific instruments. The following instrumentation can be easily used in the first MADISON sample holder manufactured for the JHR start up:

- Fuel central temperature;
- Clad temperature;
- Clad elongation;
- Fuel stack elongation;
- Fuel plenum pressure;
- Fission gas release composition based on acoustic measurement device.

A second version of the MADISON sample holder is under investigation and is based on the previous one, but it will be limited to two rod samples when equipped with a diameter gauge.

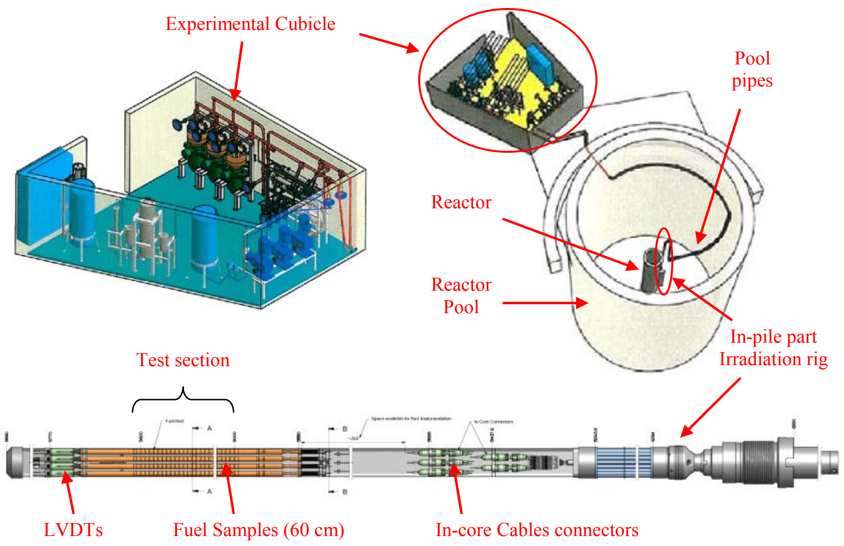


FIG. 5. View of the MADISON experimental device in the JHR facility with focus on the water loop in the experimental cubicle and on the first irradiation rig.

7.9. THE ADELINE DEVICE (OPERATING UP TO LIMITS CONDITIONS) —OSIRIS AND JHR

7.9.1. Objectives of the ADELINE irradiation loop

The ADELINE device is able to hold a single experimental fuel rod from all LWR technologies to reproduce various experimental irradiation scenarios where clad failure is either a risk or an experimental objective. Similarly to the MADISON experimental device, this experiment is made of an in-pile part (see Fig. 6) and an out-of-pile water loop. Fresh or pre-irradiated fuel rods can be used to perform:

- Power ramp tests;
- Rod internal over-pressurization ('lift-off');
- Rod internal free volumes gas sweeping;
- Power to melt approach margin mastering.

A first version is mainly dedicated to power ramps testing. The objective is to continue the service offer and to operate with an experimental quality at least as good as the one currently offered by the ISABELLE 1 loop in the French material testing reactor OSIRIS. In particular, the design of this device is optimized to provide a qualified thermal balance and a good accuracy on the clad failure instant and consequently a good knowledge of the linear power inducing the failure. A quantitative gamma spectrometry system allows quantifying the radiological source term released in the coolant when a rod fails.

Some enhancements are added in order to make on-line quantitative clad elongation measurement during power transients and to manage several successive experiments during one reactor cycle. In addition, this device can be easily upgraded in order to manage highly instrumented experiments with fuel and clad temperature measurement and fission gas release measurement by gas sweeping.

In a longer term, a second version will be dedicated to the study of the long-term post-failure behaviour in normal conditions (failure evolution, secondary hydrating, release of fission products and of fissile material, etc.) coupled with the fission product laboratory.

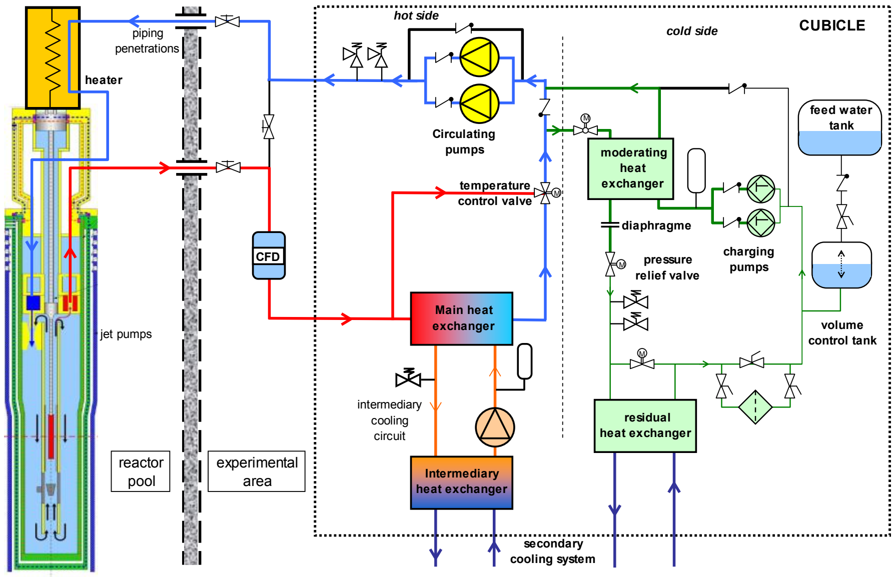


FIG. 6. Schematic diagram of the ADELINE loop.

7.9.2. ADELINE typical experimental power ramp scenario and performances

The device accepts different fuel samples types:

— PWR (including WWER type) and BWR fuel pellets from 5.5 mm up to 14 mm of diameter;
— UO_2 fuels up to 12% enrichment of ^{235}U;
— MOX fuels up to 20% ratio of Pu/(U+Pu);
— Rod length up to 600 mm fissile stack;
— Fresh fuel as well as high burn-up fuel up to 120 GWd/t.

A typical PWR power ramp sequence is made of the following phases (see Fig. 7):

— A low power plateau (from 0.5 up to 7 days) with control of clad surface temperature at 330°C (±10°C) while the sample linear heat rate is controlled between 50 and 250 W/cm, depending on customer's request;
— A linear power ramp at a continuous rate ranging between 100 $W{\cdot}cm^{-1}{\cdot}min^{-1}$ and 700 $W{\cdot}cm^{-1}{\cdot}min^{-1}$. During this phase, clad surface temperature is stable at saturation condition, as soon as the sample reaches 350 ·W/cm at its peak level;
— A high power plateau that may last 24 hours at a linear heat rate up to 620°·W/cm·at the sample peak level.

If a clad failure occurs during one of the last two phases, it does not necessarily lead to a stop of the experiment. Indeed, such decision is based on a pre-determined threshold (depending on customer specifications) of fission product contamination in the loop.

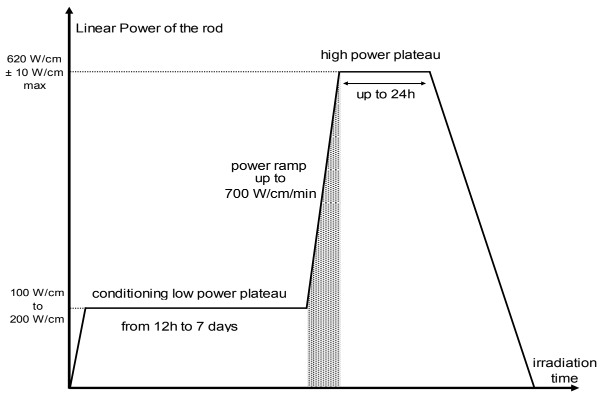


FIG. 7. Standard power ramp test time history.

Description of the loop — the loop is made of two parts (see Fig. 6):

(a) *An in-pile section,* loaded on a displacement system, features the following components:
- The containment made of a pressure flask and a surrounding tube, both of them in Zircaloy;
- The instrumentation holder containing the environment sensors and the so-called 'jet pump' flow amplification system;
- The sample holder including the instrumented test rod.

The internal structures and the process are regulated thermally by the fluid in the loop. The fluid is injected from the out-of-pile section by high pressure circulation pumps operating simultaneously to obtain an overall flow rate of 50 g/s (called 'inducing flow rate'). The fluid flows through a flow rate injection module used to re-entrain part of the main flow in the test line and thus amplify the inducing flow rate. The amplification factor is around 4–5, resulting in a fuel cooling flow rate (also called 'induced flow rate') in the test channel of 200 g/s.

(b) *An out-of pile section* is located in a 30 m² metallic liner covered cubicle close to its piping penetration. It includes the fluid circuit and the equipment needed to reproduce the thermal hydraulic conditions of the in-pile section.

The pressurization control is based on a simultaneous use of charging pumps and relief valve. The temperature control is done by the use of a 30 kW heater in the device head and a 40 kW heat exchanger in the cubicle. This main heat exchanger is cooled down by an intermediate cooling system located in the cubicle and its convenient location (instead of inside the reactor pool) makes it possible to totally by-pass it, in order to carry out energy saving during low power plateau phases.

A low pressure and low temperature system is installed in parallel of a part of the main loop. It mainly includes the final cooling, the filtration and the loop chemistry modules.

It is worthwhile to notice that the inducing circuit is designed to provide a 100 g/s inducing flow rate (and consequently an about 400 g/s induced flow rate), in order to be able to implement future experiments requesting a higher linear power on the sample (e.g. fuel central melting approach study).

7.9.3. Key features

The design of the ADELINE device is designed in order to allow an easy and fast recovery of the samples, what allows performing several power ramps during a unique JHR cycle. For that purpose, an underwater transfer station is under design in order to load and unload sample holders without disconnecting the ADELINE device. It avoids periods of unavailability of the experimental device due to its transfer toward the hot cell. This new tool makes possible to manage 3 to 4 ramp tests during a reactor cycle (about 25 days long).

For experimental purpose, this experimental device allows performing highly instrumented experiments. The instrumented sample holder design allows implementing various sensors such as fuel centreline thermocouple, cladding thermocouple and also a quantitative measurement of clad elongation thanks to a system using two LVDT-type sensors. Moreover one or two capillary tubes connected to the top and the bottom ends of the rod may be used either to apply a controlled internal pressure, or to sweep the gases (fission product and He) released by the fuel and to route them to the fission product laboratory in the JHR experimentation area.

For a continuous follow-up of the irradiation conditions on the fuel samples, the in-pile instrumentation provides a good accuracy of the thermal balance, based on a differential temperature measurement greater than 20°C between upstream and downstream from test section. With an inlet temperature higher than 250°C at 160 bar and a fluid velocity of about 0.8 m/s, a 5.7% uncertainty (2σ) on the final value of the linear power during the ramp upper plateau is expected. In addition, during a power ramp, the displacement device and the control system will guarantee that the targeted final measured linear power will have ± 10 W/cm accuracy at 620 W/cm.

7.9.4. The LORELEI device (accidental situations) — new for JHR

The purpose of LORELEI device is investigating the behaviour (thermal-mechanical and radiological consequences) of LWR-type pre-irradiated fuel rods under 'Loss Of Coolant Accident' conditions. The thermal-hydraulic phenomena does not reproduce all phases of a realistic LOCA-type power reactor sequence (in particular the first clad temperature peak), but the thermal-mechanical conditions (clad temperature, clad over-pressure, steam environment) will be representative (see Fig. 8).

This equipment consists in an integrated water capsule that can be operated as a thermo-siphon able to cool and re-irradiate a single pre-irradiated fuel sample, and to produce a short half-life fission product (FP) inventory. For the first version of the test device, the re-irradiation power is low and adapted to the production of a detectable short half-life FP inventory (versus long half-life radionuclides already present in the fuel material). Next version will allow reproducing thermal conditions representative of current LWR power reactors and performing a re-irradiation of samples at higher power in order to simulate the effects of the local peak power ('core hot spot') and to produce a representative FP inventory and distribution at the accidental sequence start-up.

It is equipped with a gas injection able to rapidly empty the test device in order to simulate the dry-out phase of the fuel rod during LOCA transient. A neutron shielding can be used to flatten the axial neutron flux profile. An electrical heater implemented in the sample holder allows getting a homogeneous temperature azimuthal distribution and acts as a dynamic thermal insulation in order to get representative adiabatic conditions (initial heat-up rate depending on customer request and typically ranging between 10 to 20°C/s).

The high temperature phase (up to about 1200°C) will be monitored by adjusting the rod nuclear power with the displacement system. During this phase, the electrical heater will be switched-off in order to increase heat losses and to prevent any temperature escalation (e.g. due to steam — zirconium reaction). At last, low temperature water can be re-injected in the device to simulate the quenching process.

This device allows investigating ballooning and burst of the fuel cladding (the inner pressure of the fuel rod can be monitored to that purpose), clad corrosion phenomena (oxidation and hydrating), thermal-mechanical behaviour, quenching, post-quench behaviour and fission product release. To that purpose, the device will be connected to the fission product laboratory. Some additional components (e.g. grid springs, surrounding rod array simulation, etc.) can be added to get more representative solicitations applied to the tested rod or the environment.

The design and manufacturing of the test device are made in collaboration with the Israel Atomic Energy Commission (IAEC).

8. EXPERIMENTAL CAPACITY FOR MATERIAL INVESTIGATION

8.1. MICA — OSIRIS AND JHR

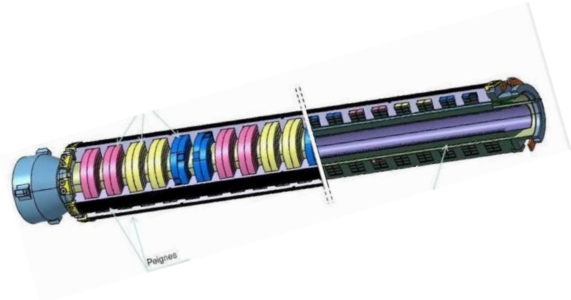
The MICA device (material irradiation capsule) has the same performances than the current CHOUCA test device widely used in OSIRIS reactor, i.e. irradiation of various geometries of samples in NaK (up to 450°C) or gas (up to 1000°C). These test devices are mainly foreseen for in core irradiations were fast flux can reach up to 9 dpa a year (at 70 MW). Since adaptation studies where necessary to fit to JHR and a couple of years were available before the manufacturing of first batch of MICA, additional studies have been launched in two main directions:

— The specificities of JHR, in terms of test devices outline dimensions, lead to an advanced integrated head of device. The main defined constraints are the handling procedure and the co-activity in the reactor pool during the few days of refuelling (inter cycles). The nowadays design embeds the gas circuits control components (valves, pressure sensors, connection) and the current electrical connections (instrumentation and electrical heating). These numerous modifications, compared to former CHOUCA device, lead to manufacture a prototype of a MICA head completed late 2011 and tested in 2012: easiness of changing sensors, plugging/unplugging actions, tests with remote manipulator arms, tightness of circuits, etc.
— The multipurpose carrier that MICA represents leads to keep most of widely former concepts that made CHOUCA devices successful but improve their thermal behaviour in order to meet the requirements, particularly in temperature precision and gradients mastering. The previous technological solutions chosen for the CHOUCA electrical elements have been reassessed to make the additional electrical heating more predictable in terms of modelling. Moreover, a special effort will be done in the determination of reactor gamma heating with qualification measurement during the start-up phase of JHR.

8.2. CALIPSO — NEW FOR JHR

The CALIPSO device (in-Core Advanced Loop for Irradiation in Potassium SOdium) meets the original need of a low temperature axial gradient (a maximum of 8°C) all along the sample holding, in liquid metal coolant (NaK), up to 450°C for a first step of development, and up to 600°C in a second phase. The locations of such devices are the same than MICA devices. The design is based on an embedded thermos-hydraulic loop, including a heater, an electromagnetic pump and an exchanger. The setting of each parameter (power of heater, flow of the pump and efficiency of exchanger) leads to a full control of the thermal conditions inside the test device and in particular in the sample location. The most difficult component turns out to be the pump, mostly because of dimensional and density of integration. Different technical solutions have been tested at the late 2011, and for by the end of 2012 the final prototype was operational.

The layout of such a liquid metal coolant loop has been done in the past, particularly for fast reactor samples irradiations. Nevertheless, theses former devices had an external loop very restrictive for both handling and safety point of view. Because safety rules became more rigorous and handling in JHR more time-consuming, CALIPSO with its embedded coolant loop represent a real innovative test device.



Design

Prototype

FIG. 8. CALIPSO pump.

8.3. OCCITANE –OSIRIS AND JHR

In the field of pressure vessel steels of NPPs, irradiations are carried out to justify the safety of this second containment barrier and to improve its lifetime. CEA is designing a hosting system named OCCITANE (Out-of-Core Capsule for Irradiation Testing of Ageing by Neutrons), which will allows irradiations in an inert gas at least from 230°C to 300°C. It will be implemented in the JHR reflector and reach damage rate about 100 mdpa/year (E >1 MeV). The associated instrumentation will include at least thermocouples and dosimeters as close as possible to the samples. OCCITANE is based on IRMA device of OSIRIS. The design studies consist mainly in decreasing thermal gradient in the sample area (see Fig. 9) and in integrating the capsule to the JHR environment.

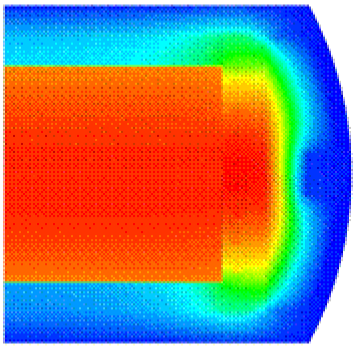


FIG. 9. Thermal calculation in OCCITANE device at 0.6 W/g.

8.4. CLOE: CORROSION LOOP — NEW FOR JHR

Due to ageing of the NPPs, stainless steel core components undergo increasing radiation doses, which enhance their susceptibility to local corrosion phenomena, known as irradiation-assisted stress corrosion cracking (IASCC). Cold laboratories can study and model SCC phenomena; but to really be representative of LWR environments, these studies need integral tests to take into account irradiation effects (radiation dose and flux) in MTRs. To answer to these industrial needs and in collaboration with DAE teams (India), CEA has just begun the design of a LWR corrosion loop, which will be located in the JHR reflector close to the tank. Its design will integrate the operational experience accumulated by the existing corrosion loops in cold laboratories at CEA and of course by the existing MTRs. A special attention will relate to the instrumentation associated with this device and will be based on the conclusions of the European programme MTR+I3.

9. JHR AS AN INTERNATIONAL USER FACILITY

In parallel to the construction of the reactor, the preparation of an international community around JHR is continuing; this is an important topic because, as indicated in the introduction, building and gathering a strong international community in support to MTR experiments is a key-issue for the R&D in nuclear energy field. Consequently, CEA is welcoming scientists, Engineers (called Secondee) from various organizations/institutes who are integrated within the JHR team for a limited period of time (typically one year) for various topics such as physics studies for the development of the experimental devices (neutron physic, thermo-hydraulic, etc.) and/or for support to the future Operator (Safety Analysis, I-C&C, etc.).

This Secondment programme is an important topic for countries willing to go on nuclear technology helping them to create and sustain key-competences.

Actually, between the academic training and the 'commercial training linked to a product' there is a need for *setting-up a framework for Nuclear Education 'in the field'* using modern high-performances infrastructures dedicated to the training of future seniors scientists, Engineers, etc. or the benefits of decision-makers in countries wishing to develop nuclear energy. These scientists invited to an International User Facility such JHR for getting this nuclear education are called Secondee as described above.

The JHR Secondee Programme is giving Nuclear Education 'in the field' that offer direct experience of working in nuclear facilities and provide training opportunities that fill the gap between academic education and commercial-product specific training. This will allow

Secondee to learn about best practices in Material and Fuel Science, Nuclear Safety, Reactor Operation, Nuclear Technologies, Fuel Management, Wastes Management, ancillary Hot Cells and Laboratories and so on.

10. CONCLUSION

As indicated, JHR building is going-on in a nominal way and its first criticality is scheduled for the end of 2016. This facility — regarding the experimental capacity — is already open and will be more and more so to international collaboration. It is clear that regarding the age of the existing fleet of research reactors worldwide, the JHR will be a key infrastructure in the European and international research area for R&D in support to the use of nuclear energy during the next few decades.

11. REFERENCES

[1] EUROPEAN COMMISSION, Towards a European Energy Security Strategy, European Commission Green Paper (2000).

[2] FEUNMARR, Future European Union Needs in Material Research Reactors, 5th FP thematic network, Nov. 2001 – Oct. 2002.

[3] Bignan, G., Iracane, D., "The Jules Horowitz Reactor Project: A New High Performances European and International Material Testing Reactor for the 21st Century", Nuclear Energy International Publication, NEI (2008).

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[5] Bignan, G., Lemoine, P., Bravo, X., "The Jules Horowitz Reactor: A new European MTR (Material Testing Reactor) Open to International Collaboration: Description and Status", RRFM (2011), Roma (2011).

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Profile 7

BUDAPEST RESEARCH REACTOR[1] (BRR)

HUNGARY

1. GENERAL INFORMATION AND TECHNICAL DATA OF A RESEARCH REACTOR

The Budapest Research Reactor (BRR)[2] is a tank-type reactor, moderated and cooled by light water. The reactor, which went critical in 1959, is of Soviet origin. The initial thermal power was 2 MW. The first upgrading took place in 1967 when the power was increased from 2 MW to 5 MW, using a new type of fuel and a beryllium reflector. A full-scale reactor reconstruction and upgrading project began in 1986, following 27 years of operation since initial criticality. The upgraded 10 MW reactor received the operation license in November 1993.

TABLE 1. MAIN TECHNICAL DATA

Reactor type	Light-water cooled and moderated tank-type reactor with beryllium reflector
Fuel	VVR-SM (-M2), transition to 20% fuel has been finished
Nominal thermal power	10 MW
Mean power density in the core	61.2 kW/L
Neutron flux density in the core	2.5×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ (thermal in the flux trap) 10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ (approx. max. fast flux in the fast channel)

Operation record of the upgraded reactor:

- From the time of start-up the upgraded reactor has been operating on average ≈ 3500 h/year without any significant problem;
- A typical operation cycles is 234 hours on 10 MW nominal power and a campaign consists of 9-10 operation cycles.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT THE BUDAPEST RR INCLUDING INSTRUMENTATION

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

The Hungarian Academy of Sciences, Centre for Energy Research (MTA EK) runs Budapest Research Reactor. The BRR is used for various purposes among other things for irradiation and neutron research, the latter being the main utilization (to serve as a neutron source). Irradiations are performed in vertical channels. The reactor has more than 60 channels, including six flux traps, which can be used for isotope production, material testing, in one of these channels a pneumatic rabbit system is working, serving for neutron activation analysis, while experiments are made at the horizontal neutron beam ports.

[1] Contact person: Belgya T., e-mail: tamas.belgya@energia.mta.hu.

[2] Link to the Budapest RR at the IAEA RR Database: http://nucleus.iaea.org/RRDB/RR/HeaderInfo.aspx?RId=196.

The reactor has ten beam ports (eight radial and two tangential) and nearly all of them are in use. At one of the tangential beam ports a cold plug containing the moderator cell has been installed ensuring a cold neutron source (CNS) with three neutron guidelines. The measurement facilities of CNS are placed in a neutron guide hall built adjacent to the reactor hall. The beam ports with the research facilities installed in the reactor hall and neutron guide hall of CNS can be seen in Fig. 1.

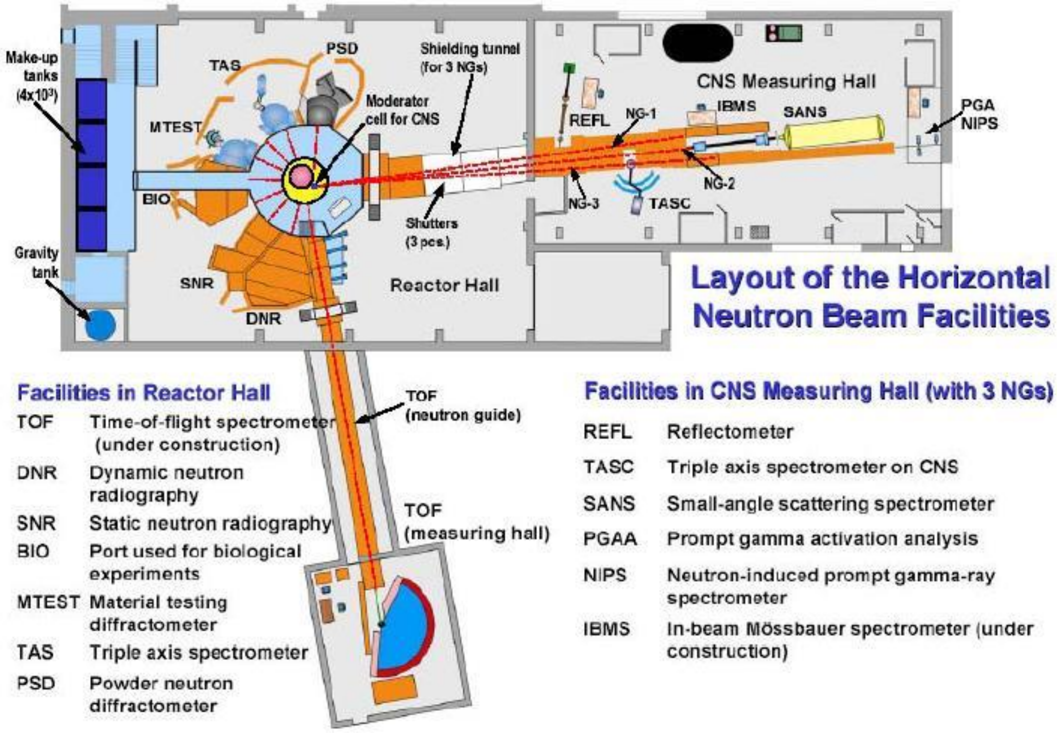

FIG. 1. Layout of the horizontal neutron beam facilities at the BRR.

Research instruments at the BRR have been offered to the entire international user community, and in particular, for EU and associated countries of the European Union in the ‘Access to Research Infrastructures’ action of the sixth and seventh Framework Programmes. The owners of these research instruments (currently MTA EK and Wigner Research Centre for Physics, Hungarian Academy of Sciences (MTA Wigner FK)) have formed the Budapest Neutron Centre (BNC) to coordinate the neutron research activities[3].

2.2. LOOPS FOR TESTING COMPONENTS

2.2.1. BAGIRA3 — reactor irradiation loop

(1) Introduction;

At the Budapest Research Reactor, two gas cooled irradiation rigs (BAGIRA 1 and 2) have been operated since 1998. Twenty-four different irradiation researches have been performed, testing irradiation ageing of reactor and fusion devices, structural materials, as low alloyed and stainless steels, Al, Ti and W alloys, ceramics etc. The devices served more than 12 years. The material aged by irradiation and corrosion, and their

[3] More information on the BNC can be found at http://www.bnc.hu/.

capacity cannot satisfy the up-to date requirements of the newly developing materials. Presently the main interest of the nuclear industry is the development of fusion reactors and Generation IV reactors. To increase the efficiencey and decreasse the impact on the environment, high operation temperature will be used. Consequently high temperature irradiation combined with in pile creep and fatigue testing are the future tasks of the irradiation devices.

Nowadays, there are only a few research reactors in Europe. With the new device in connection with the hot laboratory and with the existing know-how, we are capable of participating in international research projects, as well as customized irradiation and post irradiation examination in cooperation with several institutes and nuclear power industries worldwide.

(2) Description of the new device;
The new device is called Budapest Advanced Gas-cooled Irradiation Rig with Aluminium structure 3, BAGIRA 3.

The main features are:

— The rig capacity is 36 Charpy size specimen (approximetely 1200 g steel) or similar. The specimen sizes and shape can be varied according to the requirements, since only the target simple holder has to be changed.

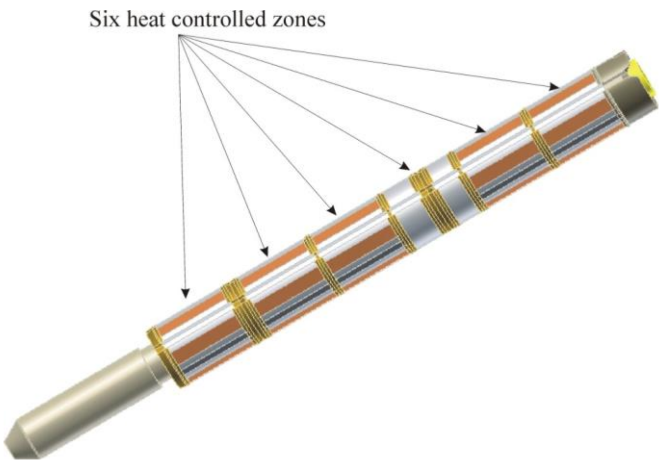

FIG. 2. Target holder, filled with specimens.

— At each of the new six zones, the electric heating can be separately controlled, ensuring to keep the required irradiation temperature within ±5°C. Irradiation temperature can be controlled between 150-650°C with gamma and electric heating and helium-nitrogen gas mix cooling.
— The maximum fluence rate is 1-5 × 10^{13} $cm^{-2}\cdot s^{-1}$ at E > 1 MeV (approximately 0.5 dpa/year). The irradiation rig is shielded with boron carbide, to filter the thermal neutrons, reducing the activity of the irradiated specimens and the nuclear heating. Reduced target activity decreases the cost of the test or transportation of the irradiated specimens.
— The target holder is separated from the thermocouples and electric heating system. This way the cost of the heating elements and thermocouples decreasing, and only minor quantity of aluminium or titanium heat removal material goes into the radioactive waste.

— The new target pick up and eject system allows the quick target change during the operation brake of the reactor, and active target also can be used (e.g. irradiated and annealed material can be re-irradiated).
— The target can be rotated during irradiation to ensure the same irradiation of the specimens located on the same level.
— The rig design allows irradiation creep or irradiation-low cycle fatigue study too.
— The device is designed for automatic operation, programmable, and it has several safety features, (including emergency passive cooling system, automatic reset in case of any malfunction of the control system, etc.).

FIG. 3. Assembled Bagira 3 in the reactor channel with the view of the rotating engine.

The equipment is ready and the control software is tested several hundred hours. Twelve different safety tests were performed successfully, and the Hungarian National Safety Authority permitted the installation into the reactor. The test run in August 2012 was also successful. The rig operates from January 2013 in several research projects.[4]

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA, etc.

None

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

None

2.5. DEVICES FOR CAPSULE/AMPOULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES, etc.

None

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (swelling, gas release, creep, long-term strength, relaxation resistance, etc.)

None

[4] Gillemot, F., Study of irradiation effects at the research reactor, Strengths of materials, **42** 1 (2010) Springer Science 78-83.

2.7. OTHER FACILITIES (this section will be added only in the electronic version and it might include zero or low power facilities supporting innovative nuclear energy projects)

None

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

NA

3.2. HOT CELLS, PIE FACILITIES (radiochemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

The BRR has hot cells to investigate irradiated structural materials as indicated at the description of BAGIRA 3. It also has a radiochemistry facility related to neutron activation analysis of various materials for moderated activities. The MTA EK is also capable to perform radioactive tracing, and uses tracers for studying chemical processes. The MTA EK has scanning electron microscope equipped with energy dispersive spectrometer for elemental analysis; it also has several XRF equipment and an ICP MS facility for studying low amount of trans uranium samples.

Beside this equipment the following reactor neutron facilities are available for research (see Fig. 1). Here brief description is given[5].

3.2.1. Biological irradiation facility[6]

An irradiation facility existed at the BRR from 1968 for 18 years. During the reconstruction of the reactor a new system for biology and dosimetry research was designed and completed in 1995. The final tests and the investigation of the beam quality were performed in early 1996. Since that time the system is in continuous operation and improvement.

The channel lock consists of three steel and heavy-concrete segments turnable by an eccentrical axis to open and close the channel. There is an internal remotely controlled filter holder at a distance of 262 cm from the core which has six windows with the following materials: four Bi disks of 5 cm, 10 cm, 15 cm and 20 cm thick and one Pb disk of 20 cm, the sixth one is an open hole. At the orifice of the beam tube two cylindrical tanks were constructed of alumina to serve as a water shutter and its emergency water storage, respectively. The water can be pumped up from and released to a larger buffer tank located outside of the reactor shielding block by pressurized air. A micro-processor controlled electronic unit connected to a PC operates the two shutters and the internal filter systems. The construction materials inside the beam tube work as internal, not removable filters with total thickness of 18 mm Pb and 15 mm Al.

The irradiation cavity is situated outside of the shielding block of the reactor in a distance of 1400 mm, thus its surface-to-reactor core distance is 3100 mm including the exchangeable core window (65 mm) made either of beryllium (rolling as the fast neutron reflector, too) or of aluminium. This window can be changed only during the maintenance or refuelling period. The use of the aluminium window results in a hard neutron spectrum. Between the shielding

[5] For pictures visit the website http://www.bnc.hu/.

[6] For *BIO* instrument responsible: B. Zábori.

surface of the reactor and the cavity there is a borated water shielded collimator with a useful diameter of 10 cm. It is possible to use this collimator as a holder for outer filters of about 800 mm length. Presently, filters of plexiglas, polyethylene, iron, aluminium and lead are available to decrease the gamma and neutron intensity or to modify the neutron spectrum and the neutron-to-gamma ratio. There are two changeable filter disks of boron-carbide working as thermal and epithermal absorbers. The collimator is movable on a rail. The samples to be irradiated can be rotated to achieve a uniform, homogeneous irradiation. Cadmium or boron carbide filters are used, if required, for decreasing the thermal neutron contribution. A large variety of irradiation geometry can be configured inside or outside of the collimator depending on the state, shape, weight of the material to be exposed.

The cavity is surrounded by a borated water shield which can be moved on a rail, as well. The whole construction is covered and surrounded by shielding elements, like a bunker, made of borated water and paraffin wax, heavy concrete and lead.

Three levels of the dosimetry system were developed: real time, active beam monitors; passive activation, track-etch and TL detectors and computer codes for spectrum and dose calculations. Each exposure is individually planned and continuously monitored during the procedure. Some typical dose and flux values are presented in Table 2.

TABLE 2. PRESENTLY EXISTING MINIMUM AND MAXIMUM DOSE AND FLUX VALUES VARIABLE BY INTERNAL AND EXTERNAL FILTERS

Quantity	Energy	Unit	Min	Max
Neutron dose rate	E >0.5 eV	mGy/s	0.23	14
γ dose rate	-	μGy/s	1.5	2570
Flux	E >0.5 eV	$n{\cdot}cm^{-2}{\cdot}s^{-1}$	$2{\cdot}\times 10^6$	$2{\cdot}\times 10^9$
Thermal flux	E <0.5 eV	$n{\cdot}cm^{-2}{\cdot}s^{-1}$	$5{\cdot}\times 10^4$	$3{\cdot}\times 10^8$
Intermediate flux	0.5 eV	$n{\cdot}cm^{-2}{\cdot}s^{-1}$ per unit lethargy	$8{\cdot}\times 10^3$	$2{\cdot}\times 10^6$

3.2.2. Dynamic radiography station[7]

Neutron radiography utilizes transmission to obtain information on the structure and/or inner processes of a given object. It is used for various non-destructive test measurements. A dynamic radiography station has been built to visualize and analyse the flow of fluids, the evaporation and the condensation processes in closed metal objects, tube systems and other types of dynamic events.

Main parameters of the dynamic radiography station at the thermal channel No. 2:

‘A’ In the conventional arrangement:

— Complex pin-hole type collimator for neutron and gamma radiation with a collimation ratio of L/D = 170;
— Neutron flux at the objects position: 6×10^7 $n{\cdot}cm^{-2}{\cdot}s^{-1}$;
— behind of CD an In filter: 3×10^6 $n{\cdot}cm^{-2}{\cdot}s^{-1}$;
— Gamma intensity: ~ 8,5 Gy/h;
— X-ray energy: 50-300 keV; 5 mA;

[7] For DNR instrument responsible: L. Horváth.

— Variable beam diameter, with a maximum of 150 mm at the object position;
— Maximum surface for investigation: 700mm × 1000 mm;
— Maximum weight of the investigated object: 250 kg.

'B' In the extended inspection area (for study of helicopter rotor blades):
— Maximum beam diameter: 185 mm;
— Maximum surface: for investigation 9750 mm × 700 mm;
— Maximum weight of the investigated object: 200 kg;
— Practicable to study, the efficiency of the moisture condition of the inspected objects, by a moistening module is driven by a high pressure water pump;
— Converters (radiation into light): for neutron radiography NE 426 scintillation screen with resolution of 100 mm; for gamma and X-ray radiography NaCs single crystal with resolution of 200 mm, or ZnS screen with resolution of 100 mm;
— Variable filters: Cd, In;
— Detection of the radiography image: low-light-level TV camera with a light sensitivity of 10^{-4} lx, imaging cycle is 40 ms, and a double cooled CCD camera (756 × 580 pixel), 10 bit.
— Radiography image is visualized on monitor, stored by S-VHS video recorder and DVD recorder and for further quantitative analysis a Quantel image processing system is used with Sapphire V.0.5 software, and an Iman b version software;
— Photo-luminescent Imaging Plates technique used by X-ray radiation or by neutron radiation with transfer method BAS IP-SR 20×25 and IP-SR 20×40 (In and Dy (100 mm) foils). The evaluation of exposed IP-s are by BAS 2500 reader unit used an AIDA picture reconstruction software.

Unique feature of the dynamic radiography station:

Our radiation sources give a possibility to study semi-simultaneously or simultaneously the investigated objects by neutron-, gamma- and X-ray radiography to use all advantages of the complementary features of the different radiations. Simultaneously, other non-destructive inspection as vibration diagnostics and acoustic emission can be used.

Main parameters of the static radiography station at thermal channel No. 3:
— A primary collimator consists of a pin-hole type conical beam collimator and of changeable iron, lead and bismuth filters; they absorb fast neutrons and gamma radiation. The collimation ratio is L/D = 100;
— Diameter of the beam is 75 mm;
— Neutron flux is 3 × 105 $n{\cdot}cm^{-2}{\cdot}s^{-1}$;
— The Cd ratio is 56;
— Direct and transfer radiography methods are used by classical film technique. For transfer neutron radiography imaging dysprosium foil, (100 μm) and indium foil (100 μm) for direct neutron radiography imaging gadolinium foil (75 μm) and NE 426 converter screens are used;
— Direct and transfer radiography methods are used by Photo-luminescent Imaging Plates technique. Direct method is used by BAS IP-ND 20×25 and the transfer radiography imaging is used by BAS IP-SR 20×25. The BAS 1800 equipment services the reading of exposed IP-s by the AIDA picture reconstruction software;
— Real time neutron radiography is performed by Peltier cooled CCD camera. Its pictures are recorded by S-VHS videomagno and by a PC with Photo Lite version 3 RC5 software.

3.2.3. Dynamic radiography station neutron diffractometer with a position sensitive detector system[8]

The PSD neutron diffractometer is suitable for atomic structure investigations of amorphous materials, liquids and crystalline materials where the resolution requirements are not high. It is a 2-axis diffractometer equipped with a linear position sensitive detector system. The detector assembly is mounted on the diffractometer arm and it spans a scattering angle range of 25° at a given detector position. The entire diffraction spectrum can be measured in five steps. During the year of 2002, the detector system of the PSD has been upgraded: instead of the original analogue design, a new system has been purchased from Studsvik NFL (Sweden) with a digital electronics technology. The detector system is based on three ^{3}He filled linear position sensitive Reuter-Stokes detectors (610 mm in length, 25 mm diameter), similarly as the previous ones, but a more novel type (P4-0824208). Three detectors are placed in the scattering plane above each other. Data transfer and instrument control has been done by PC-AT (Master PC) with Eagle I/O card. A Windows based — user friendly — instrument software program package has been developed.

Recently the interface electronics has been fully upgraded. A new dedicated electronic device has been constructed, which serves for the electronic control of the movements and data transfer of the diffractometer.

TABLE 3. CHARACTERISTIC FEATURES OF THE PSD NEUTRON DIFFRACTOMETER FOR TWO ACTUAL ARRANGEMENTS

Channel	Thermal, 9T tangential	
Primary collimation	Soller-type: 20’	
Take-off monochromator angle facility	-5° <$2\Theta_M$ <45°	
Monochromator and mosaicity	Cu (111), 16’	Cu(220), 20’
Monochromatic wavelength	1.069 Å	0.66 Å
Resolution, Δd/d	$1.2 \times \cdot 10^{-2}$	2.4×10^{-2}
Flux at the sample position	10^6 n·cm^{-2}·s^{-1}	10^5 n·cm^{-2}s^{-1}
Beam size at the specimen	10 mm × 50 mm	
Scattering angle, 2θ	5°<2Θ <110°	
Momentum transfer interval, Q	0.6-9.2 Å^{-1}	0.8-15.8 Å^{-1}
Monitor counter	Fission chamber	
Detector system	Three linear position sensitive ^{3}He detectors The detector assembly spans 25° scattering angle at a given position	
Data collection	Xilinx pre-programmed unit	
Data transfer and control	PC-AT with Eagle I/O card and a dedicated electronic device	
Remote control and file transfer	Windows program package	

[8] For PSD instrument responsible M. Fábián.

3.2.4. Material test diffractometer[9]

The material test (MTEST) neutron diffractometer was originally designed for studying internal stresses in alloys. Recently, the diffractometer has been upgraded by a position sensitive detector and by a monochromator changer; this way, a more efficient use of the available beam time, with various sample environments, can be achieved. At the present status, the instrument allows for performing total (Bragg and diffuse) scattering measurements on powder, liquid and amorphous materials. The four-circle goniometer maintains also the chance for texture measurements.

The MTEST diffractometer is installed on the sixth axial thermal channel of the reactor. The maximum flux can be obtained at a wavelength of 0.144 nm. A sapphire single crystal is used, deep inside the beam shutter, to filter out epithermal neutrons. The neutron flux at the sample table is 2×10^6 $n \cdot cm^{-2} \cdot s^{-1}$ at a wavelength of 0.133 nm.

In order to produce monochromatic beams, various single crystals are available (see Table 4).

TABLE 4. AVAILABLE MONOCHROMATORS

Monochromator	Ge (111)	Cu (111)	Ge (220)	Cu (220)
Wavelength at 40° take-off angle (nm)	0.223	0.143	0.137	0.087
The corresponding Q-range ($Ĺ^{-1}$)	0.24~5.35	0.4~8.35	0.4~8.7	0.65~13.7

Changing the wavelength is easy and doable in five minutes, by using the newly designed monochromator changer that can use three crystals: (Ge (111), Cu (111) and Cu (220)). The diffractometer is equipped with air cushions to achieve the necessary flexibility for a continuously variable wavelength. Thus, monochromator take-off angles between 28° and 54° may be set, without removing any elements of the current setup (the upper limit increases to 90° if some elements are removed). This allows obtaining neutron beams of wavelengths between 0.065 nm and 0.35 nm.

From the monochromator to the sample various Soller-type collimators can be installed (1°, 40', 30' and 12').

A low efficiency fission chamber monitor and an Ordela position sensitive detector (with two sample/detector positions) serve data collections. For high resolution measurements, a BF_3 point-detector (with various receiving collimators) is available. The diffraction spectra can be measured up to 144º by a single detector, up to 141° by using ‘near position’ and up to 151° by ‘far position’ of the position sensitive detector. The whole angular range can be covered by six (‘near’) or ten (‘far’) angular positions of this detector. The current level of background in ‘near’ position equals to the scattered intensity from a 6 mm diameter, 0.05 mm thick, 40 mm long vanadium sample holder.

At the sample stage, the following options are available by the four-circle goniometer:
— Automatic X, Y sample displacements, manual Z displacement;
— Automatic sample changer is available for four samples (only with four-circle goniometer);

Sample environment: vacuum-furnace (RT to 1000ºC); liquid N_2 cryostat (scheduled to be

[9] For MTES instrument responsible L. Temleitner.

installed in 2013 and commissioned in the first half of 2014);

Our group is ready to help users, starting from measurements, through data evaluation, simulation, and publication.

3.2.5. Thermal neutron tree-axis spectrometer and neutron holographic instrument[10]

The thermal neutron three-axis spectrometer (TAST) at BNC is installed on the thermal neutron channel No. 8 at the Budapest Research Reactor.

The TAST instrument provides moderate resolution (~1.0 meV) with sufficient intensity for use in a wide range of problems. It is ideally suited for the study of phonon and magnon dispersion curves in single crystals, phonon density of states for that large class of materials which contain hydrogen. Independent control of the momentum (Q) and energy transfer (E) is routine if required as opposed to the time of flight spectrometer in which Q and E are related by the instrumental configuration. Because of the limited number of other operational equipment the triple axis spectrometer is used in a multi-purpose regime, e.g. for high-resolution diffractometry, strain analysis, quasielastic and inelastic scattering as well as for thermal beam irradiation and transmission tests.

The monochromatic beam is provided by a 90 mm high focusing multi-blade Cu monochromator. In order to suppress the intensity of fast neutrons 15 cm long sapphire single crystal is inserted in the primary shutter. For higher order filtering in the incident monochromatic beam a Ge analyser can be used. The beam divergence is determined by thin film Soller type Mylar collimators coated with Gd_2O_3.

A highly efficient (90% at 1 Å) ^{3}He single counter of 1 inch diameter is applied as detector. A two dimensional position sensitive detector in medium resolution mode is also available. Using this detector the efficiency of data collection rises about 40 times in quasi-elastic mode. For energy analysing a focusing pyrolitic graphite crystal assembly is used.

The spectrometer can be equipped by an Eulerian Cradle, or a goniometer that can hold various sample environment devices up to a mass of 100 kg. TAST is also used as a dedicated instrument for atomic resolution neutron holography measurements both in neutron or gamma-ray detection modes.

TABLE 5. MAIN PARAMETERS OF THE SPECTROMETER

Beam tube	Channel No. 8 (radial, sapphire single crystal filter)
Monochromator	Cu 200 (doubly focusing)
Analyser	Pyrolitic graphite 002 (horizontally focusing)
Detector	^{3}He single
Range of monochromator angle	$14° < 2\Theta < 90°$
Range of analyser angle	$-100° < 2\Theta < 120°$
Range of crystal orientation	$0° < 2\Theta < 360°$
Angular resolution	0.01°
Flux at specimen at 1 Å	2×10^6 n·cm^{-2}·s^{-1}
Beam size	50 mm × 50 mm / 10 mm × 15 mm (depends on focusing)
Momentum transfer	0.2–10 Å^{-1}
Energy transfer	1-60 meV

[10] For TAST Instrument responsible A. Szakál.

3.2.6. Reactor-neutron activation analysis[11]

Combined with computerized high resolution gamma-ray spectrometry, reactor-neutron activation analysis (RNAA) offers mostly non-destructive, multi-element routine analysis needed in such areas as environmental monitoring, geochemistry, nutrition, archaeology and material science. Among its favourable characteristics negligible matrix effects, excellent selectivity and high sensitivity are worth mentioning, for about 75 elements less than 0.01 mg can be determined.

(1) Instrumentation;
Besides more than 40 vertical channels, a pneumatic sample transfer system is also available at the BRR. The control and data acquisition electronics and software of the fast rabbit system of the BRR has been upgraded recently. In the new system Field Point modular based I/O was implemented (National Instruments, USA). In order to extend the irradiation period (up to 20 minutes) a new sample holder capsule made from a high purity polymer (DuPont™ Vespel^{Ò} SP-1) is used. The cleanliness of this new material has been measured by INAA as well as the surface contamination of the capsule during irradiations and the sample temperature inside the capsule. The concentrations of the Al, As, Cu, Mg, Mn, and Na producing short half-life isotopes with impurities < 0.7 ppm, have no limiting effects on the usage of these capsules in several irradiation cycles per day.

In the 'B' vertical pneumatic tube, thermal neutron flux variation along the axis of the irradiation capsule is less than 5%. Neutron flux parameters have been measured with the 'Bare Triple-Monitor' method using Zr, Al-0.1% Au and Fe foils (see Table 6).

TABLE 6. MAIN SPECIFICATIONS FOR CHANNEL 'B' AND ROTATING CHANNEL No. 17

Inner dimensions of the polyethylene rabbit	14 mm dia. × 42 mm
Thermal neutron flux	4.45×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$
f (thermal to epithermal flux ratio)	34.8
α (representing the epithermal flux with $1/E^{1+\alpha}$)	0.029

For long cycle irradiations, samples together with flux monitors are irradiated in one of the vertical, rotating irradiation channels (No. 17) of the reactor at a thermal neutron flux density of 1.86×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$, a thermal to epithermal flux ratio (f) of 42 and $\alpha = 0.031$.

(2) Gamma-ray spectrometry;
High resolution gamma-ray spectrometric measurements are performed with a Canberra HPGe detector (energy resolution of 1.82 keV and rel. efficiency of 36% for the 1332.5 keV ^{60}Co line) and TRP preamplifier, CI 2026 amplifier (4 μs shaping); ACCUSPEC/B MCA (2×8K), 8715 8K/800 ns ADC and Genie 2000 program for data acquisition. Counting losses are corrected with a Loss-Free Counting module.

(3) Gamma spectra evaluations and calculation of element concentrations;
For gamma-ray spectrum evaluation, the Hypermet-PC (ver. 5) is applied involving

[11] For RNAA instrument responsible I. Sziklai-László.

automatic peak search, energy calibration, net peak counts computation with the NonLin and Dual Spectrum Loss-Free Counting (LFC) Option. For the quantitative evaluation, an in-house program RNAACNC, based on the k_0 standardization, is used. This program supports the following features: absolute activity, alpha value, element concentration, detector efficiency, isotope identification, thermal and fast neutron flux and flux ratio (f), nuclear data library, specific activity computation.

(4) Developments and applications;

Selenium and other trace element levels were measured in basic food ingredients, human milk, and formula milk from Hungary. Se intake, blood status, and urinary Se excretion of healthy, diabetic, and asthmatic children were also measured.

The activity concentrations of characteristic fission and corrosion products (Cr, Mn, Co, Cu, Fe, etc.) in the primary cooling water and the chemical concentrations of different impurity components in various water systems of the BRR are measured to monitor the water quality.

Epithermal neutron activation analysis (ENAA) was developed, boron activation ratios (RB) and improvement factors (IFB) for 23 nuclides were determined. Using the boron shield a very effective suppression of strongly activating 1/v and low resonance target isotopes (i.e. ^{24}Na, ^{42}K, ^{38}Cl, ^{46}Sc, etc.) can be achieved and number of important nuclides (^{75}As, ^{197}Au, ^{111}Cd, ^{121}Sb, ^{124}Sn, ^{238}U) can be determined in geological and biological samples with minimum delay.

A radiochemical method for the selective separation of Cs has been adopted and tested. NAA and ICP-MS were applied parallel for the accurate determination of ^{135}Cs in nuclear power plant wastes. Acceptable low detection limits (10-50 ng/L) and high accuracies (5-20%) were achieved by both techniques. Results of ^{135}Cs determination by ICP-MS and NAA agreed well.

Using NAA complementing with PGAA in the fields of materials material science, archaeometry and geology.

3.2.7. Prompt gamma activation analysis[12]

The prompt gamma activation analysis (PGAA) target chamber is at 1.5 m distance from the end of the guide. The sample chamber can be evacuated or filled up with gases to decrease beam-induced background. To prevent scattering of neutrons to the PGAA sample from the lower beam, a layer of neutron absorber is placed below the sample. The targets are mounted on thin Al frames by Teflon strings. Optionally, an automated sample changer with a capacity of 16 samples can be used. A neutron absorber after the PGAA target chamber stops the upper beam.

The detector system of the PGAA facility consists of an n-type high-purity germanium (Canberra HPGe 2720/S) main detector with closed-end coaxial geometry, and a BGO Compton-suppressor surrounded by a 10 cm thick lead shielding. The sample-to-detector distance is adjustable, but it is typically 230 mm. By removing the front detector shielding the HPGe main detector can be placed as close as 12 cm to the target.

The BGO annulus and catchers around the HPGe detect most of the scattered gamma photons. If the events from the HPGe and the BGO are collected in anticoincidence mode, Compton-suppressed spectra can be acquired. An analogue spectroscopy amplifier combined with an ADC and an Ethernet-based multichannel analyser (Canberra AIM 556A) collects the counts.

[12] For PGAA Instrument responsible Z. Kasztovszky.

TABLE 7. SPECIFICATIONS OF THE PGAA FACILITY

Beam tube	NV1 guide, end position
Distance from guide end	1.5 m
Beam cross section (computer selectable)	2 cm × 2 cm, 1.4 cm × 1.4 cm, 1 cm × 1 cm, 42 mm^2, 23 mm^2, 10 mm^2, 5 mm^2, 1/30 attenuator
Thermal-equivalent flux at target	≈ 7.7 × 10^7 cm^{-2}·s^{-1} (in vacuum)
Vacuum in target chamber (optional)	≈ 1 mbar
Target chamber Al-window thickness	0.5 mm
Form of target at room temperature	Solid, powder, liquid; gas in a pressurized container
Target packing at atmospheric pressure	sealed FEP Teflon bag or vial
Sample chamber dimensions	4 cm × 4 cm × 10 cm
γ-ray detector	n-type coax. HPGe, with BGO shield
Distance from target to detector window	230 mm
HPGe window	Carbon epoxy, 0.5 mm
Relative efficiency	27% at 1332 keV (^{60}Co)
FWHM	2.1 keV at 1332 keV (^{60}Co)
Compton suppression factor	≈ 5 (1332 keV) to ≈ 40 (7000 keV)

3.2.8. Neutron induced prompt gamma-ray spectroscopy[13]

The Neutron-induced prompt gamma-ray spectroscopy (NIPS) facility is located 1 m downstream of the PGAA facility at the end position of the neutron guide NV1. The NIPS/Neutron Optics and Radiography for Material Analysis (NORMA) facility has been designed for a large variety of experiments involving nuclear reaction-induced prompt and delayed gamma radiations, including γ-γ-coincidences, large-sample PGAA, Prompt-Gamma Activation Imaging (PGAI), as well as neutron radiography (NR) and tomography (NT).

The beam arrives through a flight tube of 5 cm × 5 cm cross section. If multiple detectors are to be placed close to the sample, a narrow aluminium tube with a 5 cm × 5cm × 5 cm sample chamber is available. Alternatively a sample chamber with dimensions of 20 cm × 20 cm × 20 cm is available for large-sample PGAA and position-sensitive applications. It is made of AlMgSi alloy, and lined from inside with ^{6}Li-enriched polymer. By removing one or more side panels, larger objects up to 5 kg weight could also be analysed (such as a sword, vase, stones, etc.). Samples can be loaded manually from the top, or placed onto an XYZ-w motorized sample stage with a travel distance of 200 mm and a guaranteed precision of 15 µm, which is introduced from the bottom. If custom devices are to be built into the beam, a short flight tube without a sample chamber is the proper choice.

An n-type coaxial HPGe detector (Canberra GR 2318/S) equipped with a Scionix BGO Compton suppressor is used for the routine prompt gamma measurements. This latter can accommodate HPGe detectors with larger crystals (up to end cap diameter of 76 mm). The passive shielding is made of standard lead bricks.

A digital signal processor combined with an Ethernet-based multichannel analyser module

[13] For NIPS instrument responsible L. Szentmiklósi; for NORMA instrument responsible Z. Kis.

(Canberra AIM 556B) collects the counts. Alternatively, a four-channel, all-digital XIA Pixie 4 spectrometer can also be used.

The NORMA setup also comprises an imaging system. It consists of a 150 µm thick $^6Li/ZnS$ scintillator, a quartz mirror with Al layer and a cooled, black-and-white, back illuminated Andor iKon-M CCD camera with 1024 × 1024 pixels and 16-bit pixel depth, mounted in a light tight aluminium housing. The custom optics projects a 48.6 mm × 48.6 mm area (in which the beam spot is about 30 mm × 35 mm) onto the 13 mm × 13 mm sensitive surface of the CCD chip. The resolution of the imaging system is about 300 mm. The measured L/D ratio, characteristic to the neutron beam's divergence, was found to be 342. The specifications of the facility are listed in Table 8.

TABLE 8. SPECIFICATIONS OF NIPS/NORMA FACILITY

Beam tube	NV1 guide, end position
Distance from guide end	2.6 m
Beam cross section for PGAA/PGAI	2 cm × 2 cm, 1.4 cm × 1.4 cm, 1 cm × 1 cm, 42 mm^2, 23 mm^2, 10 mm^2, 5 mm^2, 1 mm vertical slit
Beam cross section for imaging	up to 3.0 cm × 3.5 cm
Thermal-equivalent flux at target	$\approx 2.7 \times 10^7 \cdot cm^{-2} \cdot s^{-1}$
Vacuum in target chamber	Not available
Form of target at room temperature	Solid, powder, liquid; gas in a pressurized container
Target packing at atmospheric pressure	Sealed FEP Teflon bag or vial
Sample chamber dimensions	5 cm × 5 cm × 5 cm / 20 cm × 20 cm × 20 cm
Distance from target to detector window	minimum 25 mm, typical 280 mm
γ-ray detector	n-type coax. HPGe, with BGO shield
HPGe window	Al, 0.5 mm
Relative efficiency	23% at 1332 keV (^{60}Co)
FWHM	2.2 keV at 1332 keV (^{60}Co)
Compton-suppression factor	≈ 3.5 (1332 keV) to ≈ 30 (7000 keV)

3.2.9. The low-level counting facility (very heavy measuring equipment)[14]

A permanent low-level counting facility (very heavy measuring equipment (DÖME)) has recently been established to assist the in-beam activation measurements, to perform off-line counting of samples on a routine basis, and to enable the measurement of environmental samples of low activities during the reactor shutdown periods. This consists of an iron chamber manufactured from pre-World War II steel, and is therefore free of any man-made radioactivity. The chamber with an internal dimension of 800 mm × 800 mm × 800 mm accommodates the Canberra GR1319 HPGe detector with a Big MAC cryostat along its horizontal diagonal. This geometry allows sample-to-detector distances up to 250 mm. The wall of the chamber is 155 mm thick and has a graded shielding inside (Cd and Cu layers). As an option for measurement of low-energy lines with a better energy resolution, a Canberra

[14] For DÖME instrument responsible Z. Kis.

Low Energy HPGe (GL1018) detector is also available. A Canberra DSA-2000 digital gamma spectrometer is used for the data collection. It has a background of about 1.3 cps over the energy range of 7-3150 keV.

Data acquisition and data processing:

(a) A user-friendly facility control program '*Budapest PGAA-NIPS Data Acquisition Software*', has been written for manual, semi-automatic, and unattended automatic batch measurements. This program controls the beam shutters, the sample changer of the PGAA setup, the motorized sample stage of the second station and the gamma acquisition.

(b) The gamma detector systems are regularly calibrated for counting efficiency [1] and non-linearity [2]. This procedure results in a precision of about 0.5% for the relative efficiency curve, 1% for the absolute efficiency curve and a precision of 0.005-0.1 keV for energy determination of peaks. The complex γ-ray spectra are evaluated with the spectroscopy program Hypermet-PC [3, 4] and see Refs 1 and 2.

(c) A series of measurements has been completed earlier at our PGAA facility to establish a library of prompt gamma lines for qualitative and quantitative analysis. These energy and intensity data are accurate enough for the routine analysis [5, 6]. An Excel macro (ProSpeRo) is used for the chemical analysis which compares the spectroscopic data library and the measured areas of the characteristic peaks [7].

(d) The radiography images taken at NORMA require several steps of data treatment. The spatial inhomogeneity of the beam and the thermal noise of the camera should be removed. These are called 'beam image correction' and 'dark image correction', respectively. In tomography, the goal is to determine a measure of the interaction probability between the material and the neutron as a function of spatial coordinates. This quantity delivers the structural information about the interior of the sample. The reconstruction codes, such as the OCTOPUS reconstruction software [8], apply the inverse Radon-transformation and filtered back projection algorithms. The visualization of the dataset in 3D space (volume rendering) is carried out using VGStudio 2.1 [9].
Recently, applications of the PGAA, NIPS and NORMA facilities, including nuclear data measurements has been summarized in [10].

3.2.10. Neutron reflectometer[15]

The reflectometer is situated on the guide No. 1. Due to the geometrical restrictions the neutron path had to be aligned parallel with the guide No.1. This solution was performed by introduction of a double-pyrolytic graphite (PG) monochromator. The slit system, the sample holder table and the control electronics remained as it was in the previous position. The main technical novelty is the installation of a 2D position sensitive detector (PSD).

The reflectometer operates at wavelength of $\lambda = 4.28$ Å. The maximum scattering angle is $\Theta_{scatt} = 5^o$. The angular resolution is variable by changing of the slit widths using micrometer screws down to 0.0055 grad. The flux of the neutrons at the samples is 200 $n \cdot cm^{-2} \cdot s^{-1}$ and the background for the whole detector is 5 n/s.

In the current configuration with the relatively low flux and increased background, the applicability of the reflectometer is mostly limited to the investigation of large area industrial samples.

The double monochromator system is being replaced by a focusing phase space compressing

[15] For REF instrument responsible T. Veres.

geometry using high quality PG crystals. Due to the improvement in the mosaicity and focusing geometry, according to Monte-Carlo simulations, a tenfold increase of beam intensity is expected to be achieved at the sample position. The PSD detector is also to be upgraded for the 2D analysis by a new data acquisition system. It is also essential to get rid of the epithermal neutrons causing the dominant part of the background. For that both the shielding of the guide No.1 and that of the reflectometer will be improved. In this way the reflectometer will be attractive for the users by offering much better intensity/background ratio.

The reflectometer is regular used for quality control of supermirror multilayers. After upgrade it can also be used for standing-wave observation and application for nanometer thickness measurements and biological investigations (e.g. bio-membranes).

3.2.11. Triple-axis spectrometer ATHOS[16]

A triple-axis spectrometer (ATHOS) has been designed for structural and dynamical studies of condensed matter. Because of the limited number of other operational equipment the triple axis spectrometer is used in a multi purpose regime, e.g. high resolution diffractometry, strain analysis, reflectometry, quasielastic and inelastic scattering. We have a polarization setup on this spectrometer too.

The spectrometer is installed on a curved (4200 m) neutron guide at a 19 m position from the beam port exit. The guide is 1.5 Qc boron glass and is coated NiTi multilayer.

The monochromatic beam is provided by a 90 mm high focusing multi- blade pyrolythic graphite monochromator.

The movable part of the monochromatic shielding has a chain type construction. Changing the incident wavelength the whole chain is driven by the monochromator-sample arm. This construction automatically provides the most effective shielding near the detector area (see Fig. 1). We could achieve very low background conditions (1 neutron/300 s).

The beam divergence is determined by thin film Soller type Mylar collimators coated with GdO.

A two dimensional position sensitive detector in medium resolution mode were installed .and the efficiency of data collection has been raised 40 times in quasielastic mode.

The spectrometer is to be developed into the RITA-type spectrometer. For the polarization option, we use a mirror assembly as a polarizer and analyser. Heussler-crystals are foreseen in a future.

[16] For ATHOS instrument responsible G. Török.

TABLE 9. THE MAIN PARAMETERS OF THE SPECTROMETER

Beam tube	Neutron guide No. 1
Monochromator	Pyrolitic graphite 90 mm x 80 mm (24 min mosaicity)
Analyser	Pyrolitic graphite 50 mm x 90 mm (24 min mosaicity) or Ge mono crystal (15 min mosaicity)
Collimation	Interchangeable 45', 30', 15'
Range of monochromator angle	36º<2Q<126º
Range of scattering angle	-120º<2F<70º
Range of crystal orientation	0º<2Θ<360º
Angular resolution	0.01º
Flux at specimen	2×10^6 $n \cdot cm^{-2} \cdot s^{-1}$
Momentum transfer	0-2.7 Å
Energy transfer	0-9 meV
Characteristic resolution at 3.3 Å	120-150 meV
Sample environment	Cryostat (liquid N_2) Magnet up to 2 T, max scattering angle 100° Furnace up to 1000ºC Thermostat -20ºC-100ºC

3.2.12. Small angle neutron scattering diffractometer[17]

Small angle neutron scattering (SANS) diffractometer Yellow Submarine located on cold neutron guide No. 2. It covers a Q-range 0.003-0.7 $Å^{-1}$ allowing probing structures at length scales from 5 Å to 1500 Å. It has a wide range of applications from studies of defects and precipitates in materials, surfactant and colloid solutions, ferromagnetics, magnetic correlations, alloy segregation, polymers, proteins, biological membranes. The instrument is installed on the curved neutron guide No.2, with 4 cm x 4 cm cross-section, made of (1.5 θc) supermirrors. The beam is monochromatized by a multidisc type velocity selector, (L. Rosta: Physica B 174 (1991) 562) the rotation speed can be tuned between 700 and 7000 rot/min (wavelengths between 3.5 and 25 Å). The width $\Delta\lambda/\lambda$ of the transmitted wavelength distribution can be varied between 12% and 30% by changing the tilt angle between the selector axis and the direction of the neutron beam. Five metres and 1 m collimation distances allow the optimization of flux and resolution for different sample-to-detector distances.

(1) Sample environment;
In most of the experiments an automatic sample changer with six positions is used. It can be thermostated from an external bath between –10 ºC and 100ºC. Eleven positions sample changer can be used for ambient temperature experiments. Liquid nitrogen cryostat, or closed cycle refrigerator can be used (from 10 K to 300 K). Electromagnets can also be mounted on the sample table (field 1.4 T in the gap 25 mm).

(2) Detector;
The scattered neutrons are detected by a 64×64 pixels (1 cm × 1cm pixel size) two dimensional position sensitive LETI (Grenoble, France) detector filled with BF_3 gas.

(3) Data acquisition;
The control and data acquisition electronics and software have been made by Laboratoire Léon Brillouin, Saclay, France, and ANTE Ltd., Budapest, Hungary.

[17] For SANS instrument responsible L. Almásy; instrument scientists — A. Len, R. Ünnep.

TABLE 10. MAIN CHARACTERISTICS

Beam tube	Cold neutron guide No.2/1
Monochromator	Multidisc velocity selector
Detector	2D position sensitive, 64 cm × 64 cm, filled with BF_3 gas
Collimations	1 m or 5m
Flux at the guide exit	5×10^7 n·cm^{-2}·s^{-1}
Sample-to-detector distance	Continuously adjustable between 0.92 and 5.6 m
Incident wavelength	3-25 Å
Wavelength spread	Adjustable between 12 and 30%

3.2.13. Neutron reflectometer with polarization option[18]

The GINA reflectometer is a constant-energy angle-dispersive, vertical-sample instrument. The setup is displayed in Fig. 1 and the operation parameters are summarized in Table 11. The focusing graphite monochromator MONO provides neutrons with wavelengths within the range of 3.2-5.7 Å and $\Delta\lambda/\lambda$ ~ 1%. The polarized neutron beam is produced by using a magnetized supermirror (P1) and an adiabatic radio-frequency (RF) spin flipper (SF1). The beam scattered on the sample may undergo spin analysis by an identical setup of a spin flipper and a spin analyser (P2), and finally it is detected by a two-dimensional position sensitive neutron detector (DET). The incident intensity is monitored by a low efficiency (~ 0.1% at λ = 4.6 Å) beam intensity monitor (IM). The components of the reflectometer are mounted on two heavy-load optical benches. The first one supports the beam shutter (BS), the IM, the beryllium filter (BF), the slit (S1) and the SM polarizer (P1), the adiabatic RF spin flipper (SF1) and the slit (S2). The downstream end of the bench is fixed to the central sample tower ST and supports the various sample environment components (electromagnet, cryostat, etc.). The incident angle on the sample surface is set by the major (Θ) goniometer of ST. The second bench, the 2Θ-arm of the reflectometer, supports the slit S3, the spin flipper SF2, the spin analyser P2, and the detector along with its electronics and dedicated control PC mounted underneath. The slit S4 in front of the detector is optionally used when data collection is restricted to specularly scattered neutrons. The 2Θ-motion is driven by a wheel running on the marble surface while the corresponding air pads are activated. The wavelength may be changed by manually rotating the entire GINA setup around the turntable under the monochromator while air pads are activated and both arms float over the marble floor. At present, the available wavelengths are restricted to 3.2, 3.9, 4.6, 5.2 and 5.7 Å by the respective channels through the cylindrical concrete shielding (SH) around the monochromator assembly.

The monochromator MONO is located in a gap of the curved Ni/Ti SM guide 19 meters downstream the cold source, and comprises five highly oriented pyrolytic graphite crystals on small motorized 2-circle cradles for horizontal alignment and vertical focusing. Vertical focusing of the beam to the sample position doubled the intensity reflected by a 20×20 mm^2 sample at grazing incidence as compared to the non-focused case of parallel graphite crystals. Higher harmonics intensity is efficiently filtered by a Be block. The transmission of the filter is 41% and 87% for λ = 4.6 Å, without and with liquid nitrogen cooling, respectively.

Fine definition of the beam is maintained by the four slits. The blades are operated with a

[18] For GINA instrument responsible L. Bottyán.

precision of 0.05 mm. Slit S1 defines the beam on the polarizer P1 to decrease the divergence thus increasing the polarization ratio. Slit S2 decreases the beam divergence on the sample and absorbs the neutrons scattered by the polarizer. With these optical elements the setup exhibits a relative Q-resolution of 10% to 2% for the available Q-range of 0.005 to ~ 0.25 $Å^{-1}$.

Polarized neutrons are produced by an Fe-Co/Si magnetic SM in transmission geometry. Spin analysis of the specularly reflected beam is performed by a single magnetic SM analyser (P2) of identical construction with P1. The spin flippers are of adiabatic RF type [11]: the flipper coil is placed in a longitudinal gradient field of 20-40 mT/m, with a centre field of 5.6 mT. The flipper coil is part of a resonant circuit, with typical values of effective RF current and bandwidth of 4 A and 4.5 kHz at the resonance frequency of 166 kHz.

Neutrons scattered by the sample are registered by a delay line type multi-wire proportional chamber with active area of 200 mm × 200 mm and spatial resolution of 1.6 mm (FWHM). The detector is filled with a gas mixture of 2.5 and 3 bar partial pressures of ^3He and CF_4, respectively, and it is encased in a boron-containing shielding for background suppression. A DASY TDC module (produced by ESRF, Grenoble) is installed in a slot of the PC dedicated exclusively to the detector data-acquisition and mounted on the 2Θ-arm of the reflectometer. If no spin analysis is required, for further background suppression, an evacuated flight tube is mounted along the entire length of the 2Θ-arm.

TABLE 11. THE OPERATION PARAMETERS OF GINA

Parameter	Range
Wavelength	3.9-5.1 Å in five steps
Present wavelength	4.6 Å
Max. scattering angle	≥θ = 35°
Angular resolution ($\Delta\theta$)	0.003°
$\Delta\lambda/\lambda$	~ 1%
Background level	0.01 cps cm^{-2}
Detector	2D PSD, 200 mm × 200 mm
Detector spatial resolution	1.6 mm × 1.6 mm
Neutron flux at the monochromator position	4×10^5 $n{\cdot}cm^{-2}{\cdot}s^{-1}$
Background reflectivity	$<7 \times 10^{-5}$
Overall polarization efficiency	0.895

The GINA hardware and control software are designed for maximum flexibility and remote controllability. In its full configuration, GINA comprises more than 30 remotely controllable stepping motors. Θ- and 2Θ-angles and precision slit positions are encoder-controlled. Hardware control is established via a USB multi-function data acquisition module to control the air compressor, the air pads, the temperature of the Be-filter, the beam shutter and control lights, the beam intensity monitor and various modular DC power supplies. The high voltage power supplies, the amplifiers, the discriminators, and the rate meters are of NIM standard. The control and detector PCs communicate via Ethernet and with the stepping motor indexers as well as with the temperature controller via RS232 under the supervision of GINASoft written in LabView 2009. The program user interface is highly configurable and performs experiments including alignments, polarization and the sample environment control (flipper current and frequency, temperature, magnet current, etc.). Detector pictures and reflectivity data are efficiently viewed and manipulated during the data acquisition. Data and log information are saved in a clearly structured database format. Human control is facilitated by a web camera. Using remote desktop option, most operations can be performed remotely via internet from outside the experimental hall or even from a distant continent.

GINA is dedicated to magnetic heterostructures, for studies requiring different environmental parameters, such as low temperature and occasionally high external magnetic fields. For room-temperature reflectivity measurements, the sample is held in position by vacuum. Two cradles and two perpendicular translators position the sample in the vertical plane and set the sample surface orientation. A closed-cycle cryostat (range 12-300 K) can be mounted on the sample tower ST with or without the electromagnet. At GINA an air-cooled electromagnet is available, which generates magnetic fields up to 0.55 T for the pole distance of 40 mm that accommodates the 1.5” diameter cryostat housing. The optional water-cooled air core coil pair provides fields up to approximetely 35 mT.

In summary, the GINA reflectometer is a versatile dance-floor-type, vertical sample, constant energy angle-dispersive instrument. Reflectivity ranges above four orders of magnitude have been measured. Further developments including an environmental cell for membrane studies, a supermirror fan analyser and further background suppression elements will be installed within the next two years.

3.2.14. Time-of-flight diffractometer[19]

The high resolution time-of-flight powder diffractometer (TOF) at BNC has been installed to a radial thermal neutron beam in a new guide-hall in collaboration with the Hahn-Meitner-Institut. According to Monte-Carlo simulation results it was expected that this type of instrument can outperforms a conventional crystal monochromator powder diffractometer at continuous reactor source in the resolution range of $\Delta d/d = 1–5 \times 10^{-3}$. The other advantage to apply TOF monochromatization to neutron diffractometry on a continuous source is the variable resolution and intensity. A full diffraction spectrum can be gained within a variable bandwidth with ultrahigh resolution or with high intensity at conventional resolutions.

The monochromator system consists of a fast double and the two single choppers and a straight neutron guide with 2.5 cm × 10 cm cross section at the end. The double chopper is designed for a maximum speed of 12000 rpm. While in high resolution mode the very short — 10μs — neutron pulse and the 25 m total flight path allow us to obtain a diffractogram with an accuracy of 10^{-3}Å (at back scattering mode) in a single measurement on polycrystalline materials, in low resolution mode liquid diffraction can be performed at good neutron intensity up to 15 $Å^{-1}$ scattering vector. As it was expected, the beam was contaminated with epithermal and fast neutrons because the straight guide is directed on the centre of the zone and the gadolinium coated chopper disks are transparent for them. Temporary silicon filters was applied with which the signal-noise ratio had been increased by a factor of 5-10.

The double disk chopper (Ch1 and Ch2) has two windows: a 1.5° opening for short pulses (10 μs) and a 15° window for long variable pulses (20–200 μs), and can be operated in parallel or counter rotating mode. The latter option is used to produce very short pulses at high speed. To minimize the opening time the neutron beam is reduced from 25 to 10 mm

[19] For TOF instrument responsible G. Káli.

width at the position of the pulse choppers using a 4.5 m compressor neutron guide section before and a same decompressor after them (see Fig. 1). Ch3 limits crosstalk between different pulses and Ch4 prevents frame overlap.

The instrument is working in back scattering mode to reach the best possible resolution. Until the planned detector (a 60 cm × 100 cm 2D detector) reach completion, a box of four ^{3}He tube is used with a 2.5 MHz event recording board. Because of the much smaller surface, the box is placed closer (2 m) to the sample opposite to the designed (3m). To achieve the maximum resolution the 2D position sensitive detector will be applied in combination with a bank of 32 pieces 6 mm thick pressed ^{3}He tubes. The data are acquired in so called list or time stamping mode: all the event on the detector, the chopper signs and optionally changes in the sample environment are registered with the time passed since the starting of the experiment. In this mode many uncertainties can be filtered out during the treatment and re-treatments.

TABLE 12. MAIN PARAMETERS

Total flight path from chopper 1	L = 25 m
Wavelength range	0.07-0.45 nm
Bandwidth in single experiment $\Delta\lambda$	from 0.4 nm to 0.08 nm (200 Hz)
Resolution $\Delta d/d$	1×10^{-3} at d = 0.15 nm
Straight neutron guide cross section	25 mm ×100 mm
Coating	Supermirror NiTi, m = 2
Beam flux at opened windows	4×10^7 n·cm^{-2}·s^{-1}
Pulse length	10-1000 µs
Max. speed for the double chopper	12000 rpm

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

Most of the instrumentations under Section 3.2. were designed and manufactured locally including the human resources developments.

4. RECENT ACHIEVEMENTS

Our last two year references can be found at http://www.energia.mta.hu/content/publications.

Individual person’s publication list is collected at the data base of the Hungarian Academy of Sciences https://vm.mtmt.hu/www/index.php?scid=21 (use translate to English of the web page).

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Profile 8

DHRUVA

INDIA

1. GENERAL INFORMATION AND TECHNICAL DATA

DHRUVA reactor, which attained its first criticality on 8 August 1985, is a 100 MW (th) research reactor with a maximum thermal neutron flux of 1.8×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$. The reactor utilizes heavy water as coolant, moderator, reflector and metallic natural uranium as fuel. DHRUVA, with its high neutron flux and large irradiation volume meets the growing demands for radioisotopes, research in basic sciences and engineering and other research reactor based requirements in India. The thermal, epithermal and fast neutron flux at various locations of the reactor is given in Table 1.

TABLE 1. NEUTRON FLUX AT VARIOUS POSITIONS

Position	Thermal (0.625 eV) $n \cdot cm^{-2} \cdot s^{-1}$	Epithermal (0.625 eV-1 MeV) $n \cdot cm^{-2} \cdot s^{-1}$	Fast (>1 MeV) $n \cdot cm^{-2} \cdot s^{-1}$
Isotope assemblies	1.8×10^{14}	3.32×10^{13}	2.7×10^{12}
Pneumatic carrier facility (PCF)	9.5×10^{13}	1.75×10^{13}	1.42×10^{12}
In pile loop (IPL)	1.4×10^{14}	2.58×10^{13}	2.1×10^{12}
Radial beam tube (near core region)	9.5×10^{13}	1.75×10^{13}	1.42×10^{12}
Beam tube (20 cm away from active core)	6×10^{13}	–	–

The reactor is housed in a rectangular concrete confinement building. The equipment connected with heavy water system, cover gas system and in-pile loops (IPL) are located inside the reactor building. A fuel storage block with facilities for temporary storage of spent fuel is also housed in this building. Two fuelling machines traverse on top of the reactor block and storage block to facilitate handling of the fuel and experimental assemblies.

A service building located at one side of the reactor building houses equipment for secondary cooling water and other conventional systems. Spent fuel storage building is located adjacent to the reactor building and is connected to the reactor building through a water filled trench to facilitate transfer spent fuel from reactor building to spent fuel storage building.

The spent fuel storage building houses water filled bays for the storage and handling of the spent fuel. General arrangement of DHRUVA reactor is presented in Fig. 1.

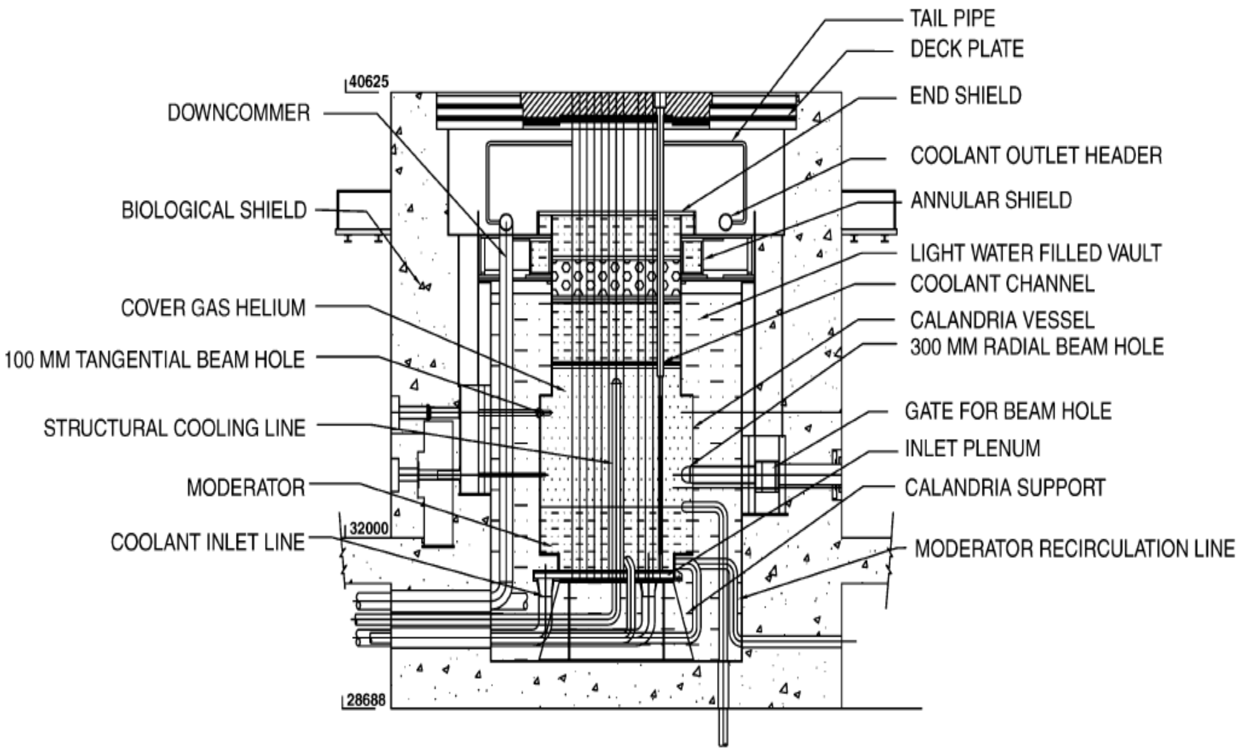


FIG. 1. General arrangement of DHRUVA reactor.

The reactor core is contained in a cylindrical stainless steel vessel placed in a light water filled vault. There are 146 lattice positions in the reactor vessel arranged in a square lattice pitch. Normally, 127 lattice positions are used for fuel assemblies, nine for shut off rods and the remaining for isotope production and experimental purposes. There are two dedicated positions for in-pile creep and corrosion test loops and two for in-pile loops.

In order to provide shielding against neutrons in upper service space an end shield is provided above the reactor vessel. The end shield is a cylindrical structure made of stainless steel and consists of composite sections of steel balls and water. It also provides support to the coolant channels. Water in the end shield is circulated through heat exchanger to remove the heat generated in the shield.

The end shield is supported on an annular shield, which provides shielding in the upward direction in the annular region around the end. It is a composite structure made up of mild steel girders, concrete blocks, carbon steel shielding plates, etc.

The biological shield houses the reactor vessel and is made of 2.4 m thick high density concrete. It is internally lined with the stainless steel. The annular space of about 1.22 m between the reactor vessel and the biological shield is filled with de-mineralized water. The light water filled vault has simplified the engineering design of the biological shield.

Coolant channel consists of components such as fuel channel cup, guide tube, extension tube, stump tube, seal cap and dust cap. The portion of guide tube in the reactor vessel region is made of zircaloy-2 whereas upper portion, above the reactor vessel, is made of stainless steel. The guide tube is a semi-permanent structural component and is easily replaceable. This provision has given the requisite flexibility to alter the core configuration, if required.

The reactor is provided with a confinement to confine the radio-nuclides that might get released in an unlikely event. The building is made of reinforced concrete and is designed for an internal pressure of 3.0 kPa and a vacuum of 1.0 kPa. The building is provided with two vacuum relief dampers. Automatic confinement isolation takes place in case of detection of any high gamma activity in the reactor building exhaust air or on a condition of reactor building pressure becoming off-normal.

The optimized core design provides a maximum thermal neutron flux of 1.8×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ and an excess core reactivity of about 20 mk exclusively to cater to various in-core experiments and irradiations at the rated power level of 100 MW. The optimized core design has been achieved by using heavy water as coolant, which has provided better neutron economy than light water. This has also helped in reducing the positive void coefficient of the reactor, which is desirable in limiting the peak power being reached during loss of coolant accident.

The reactor power regulation is achieved by varying the moderator level with constant inflow and variable outflow. Three independent channels of instrumentation are provided for every important parameter. Reactor trip signal is generated on two-out of -three global co-incidence logic. Nine cadmium shut off-rods constitute the primary shut down system of the reactor. Fast shut down of the reactor is achieved by gravity insertion of the shut off- rods fully into the core within three seconds of the reactor trip signal. The instrumentation with adequate redundancy covers monitoring and recording of all-important nuclear and process parameters and provides audio-visual alarm annunciations.

To maintain long-term sub-criticality, moderator dumping from the reactor vessel to dump tank, to a pre-determined level, is provided. This is independent of the primary shut down system. Moderator dumping is achieved by opening three dump and three control valves, which form part of this moderator system. All the six valves fully open on a completed reactor trip signal. These valves have quick opening characteristics and they attain the fully open position in two seconds.

The heat generated in the fuel assemblies is removed by primary coolant flowing from bottom to top through the coolant channels. Heat from the primary coolant is removed in the heavy water/process water heat exchangers. The process water is in-turn cooled by sea water in tertiary side. Reactor trip is provided on tripping of the main coolant pumps and shut-down coolant pumps come into operation automatically providing shut down cooling to the core. The shut-down coolant pumps are provided with two prime movers operating on diverse principles, one operating on electric power and other on hydraulic power. One prime mover operates at a time. The water to the turbine is supplied by gravity flow from an overhead storage tank; this water subsequently passes through the primary coolant heat exchangers for decay heat removal.

Helium is used as cover gas for venting of the moderator and the primary coolant system. Helium gas provides an inert blanket for the heavy water system to avoid isotopic and chemical degradation. Helium purification flow is provided to carry away the products of radiolytic decomposition of heavy water and other gaseous impurities. In the purification system, deuterium and oxygen present in the cover gas are recombined. The recombined D_2O is removed subsequently in the freezer dryers. Any other gaseous impurity present in the cover gas is removed in activated charcoal beds. Purified helium flows to a low-pressure gasholder located at the suction side of the compressors.

An engineered safety feature in the form of emergency core cooling system (ECCS) is provided to mitigate the consequences of significant loss of coolant from the pressure boundary. In the event of a loss of coolant accident (LOCA), the reactor will automatically trip. The heavy water leaking out from the system gets collected into tanks under gravity flow. The collected heavy water is pumped back to the core by the emergency core cooling pumps.

Fuel assembly consists of seven pin cluster of metallic natural uranium, cladded in aluminium and placed in an aluminium flow tube; through which the coolant flows upwards. The fuel cluster assembly is pinned at the top to an aluminium seal and shield assembly. The seal assembly provides pressure boundary for the coolant and the locking mechanism for locking the fuel assembly inside the coolant channel. The fuel pin configuration and the fuel pin size are mainly based on optimization of core physics and thermal hydraulic considerations. The maximum linear heat generation rate in fuel pin is restricted to 1 kW/cm with adequate thermal safety margins. The flow through individual fuel channel is monitored with three independent flow instruments. Low flow through any channel would bring in the reactor trip. Apart from the flow, coolant water activity and the temperature are monitored for each fuel channel. Prior to installation in pile each fuel assembly is tested in the flow test station to ensure that the flow through the assembly is normal.

Fuel handling operation involves replacement of an irradiated fuel with a fresh one using a fuelling machine. The irradiated fuel is transported under water to spent fuel storage bay for long term storage.

2. EXPERIMENTAL AND ISOTOPE PRODUCTION FACILITIES IN DHRUVA

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

India has a fleet of PHWRs of both 220 MWe and 540 MWe capacity. It is proposed to setup many 700 MWe PHWRs in future. Besides, India also has BWRs. In future, it is proposed to set-up, imported as well as indigenously developed PWRs. Data related to long-term behaviour of materials, under irradiation environment, is needed for studying behaviour of existing materials as well as for residual life prediction of reactor components and systems. Detailed study of degrading mechanisms would help in developing new materials and alloys and studying their irradiation behaviour before their use in reactors. Some of the common materials of interest are Zr-2.5%Nb, low alloy pressure vessel steel and austenitic steels. Irradiations conditions required for these materials span in the region of fluence between 10^{20} n/m^2 to 10^{26} n/m^2 and temperature of 250-300°C. The important properties to be studied include tensile strength, elasticity, toughness, irradiation creep and growth, impact properties, fatigue strength, hardness, thermal conductivity and coefficient of thermal expansion. The sample size would be based on the property to be studied.

2.2. MATERIAL AND FUEL TESTING

2.2.1. Controlled temperature irradiation facility (CTIF)

Controlled temperature irradiation facility (CTIF) is designed for study and development of fuel, clad and reactor vessel materials (pressure tube, calandria tube and end fittings of PHWRs). The irradiation of samples is carried out at 300±5ºC. The thermal neutron flux at the irradiation location is 7×10^{13} n·cm^{-2}·s^{-1}. The irradiated samples will be assessed for changes in their mechanical properties (fracture toughness, impact and tensile properties) and metallurgical properties.

FIG. 2. CTIF assembly.

2.2.2. In-pile loop (IPL)

The in-pile loop is for carrying out irradiation testing of fuel bundles under simulated condition of pressure and temperature existing in power reactors. The loop mainly consists of a zircalloy test section at core region. The pressure and temperature in the loop is maintained with the help of a steam Pressuriser and immersion type electric heaters. Adequate instrumentation and controls have been provided for the safe operation of the loop.

The rated power of the loop is 1.6 MW. The out-off pile systems has been commissioned.

FIG. 3. In pile loop.

2.3. NEUTRON ACTIVATION ANALYSIS FACILITY

This facility is specially meant for the irradiation of short-lived samples which require minimum transit time between the completion of irradiation and counting. Neutron flux at the irradiation location is of the order of 5×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$. This facility has provision for shooting the sample into the core for irradiation and receiving back the same into a laboratory. The samples are encapsulated in 25 mm diameter and 38 mm long ethylene propylene capsules and are pneumatically transported to the reactor.

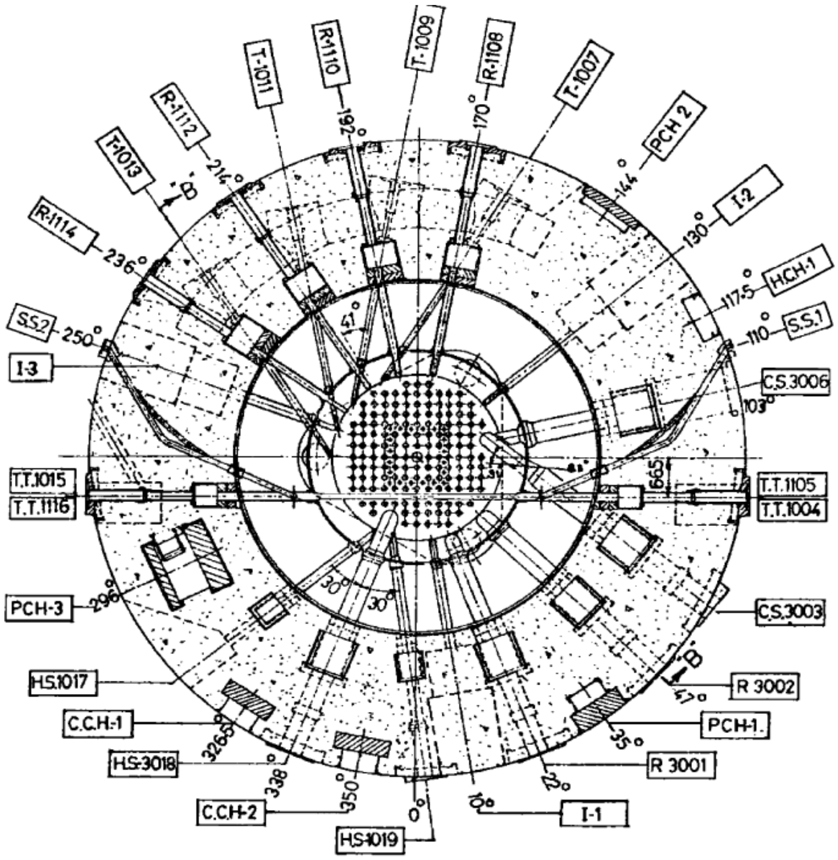
FIG. 4. Sample handling in PCF facility.

All the operations involving irradiation of samples is fully automated. Samples are received back after the required period of irradiation. Neutron activation analysis technique provides not only rapid quantitative analysis down to ppb level or below but also provides critical validation support to other techniques. A wide variety of samples were irradiated in PCF for application in material sciences, environmental and life sciences, forensic science and archaeology. PCF has also been used for determination of uranium by solid-state nuclear track detector (SSNTD) using fission track analysis (FTA).

2.4. NEUTRON BEAM TUBE RESEARCH

Thermal neutrons are used in a variety of ways to investigate engineering, metallurgical, chemical, biological and other material properties, like crystallinity, magnetic structures, nature of atomic motions etc.

The thermal neutrons from reactors make it possible to carry out a variety of fission physics experiments to obtain new data on both the de-excitation process of the fission fragments and on the dynamics of the fission process. It is achieved through the measurement of prompt neutrons, gamma rays, K-X-rays and light charged particle emitted in fission and through the study of fragment mass, charge and energy distributions and their inter-correlations. The beam tubes around DHRUVA are indicated in Fig. 5.



I	- Ion chamber basket	SS	- Self-serve facility for short term radiation
R300	- 300 mm diameter radial beam tube	T100	- 100 mm diameter tangential beam tube
PCH	- Pile chase	R100	- 100 mm diameter Radial beam tube
CS300	- 300 mm diameter cold neutron source	HS	- Hot neutron source
TT	- Through tube	CCH	- Cable chase
HCH	- Helium chase		

FIG. 5. Cross section of reactor.

Fourteen number of beam tube facilities are provided in DHRUVA reactor for neutron beam experiments as also for sample irradiations as follows:

— Four 100 mm diameter tangential beam tube;
— Four 100 mm diameter radial beam tubes;
— Two 300 mm diameter radial beam tubes;
— Two through-tubes of 100 mm diameter providing four experimental ports;
— One 300 mm diameter beam tube, designed for installation of a cold neutron source, with a rectangular satellite port for installation of neutron guides;
— One 300 mm diameter beam tube designed for installation of a hot neutron source with a provision of two 100 mm diameter satellite ports for beam extraction.

The provision of tangential beam tubes, through tubes and facilities for re-thermalization of neutron beams are new features, built into the experimental facilities at DHRUVA. Cold neutron beams can be transported using guides, to the laboratory located adjacent to the reactor building for conducting experiments in low gamma and neutron back ground condition.

Other applications of beam tubes:

(a) Accelerated life testing of ion-chambers;
Neutron detectors of various types and sensitivities are be tested for their performance

under simulated conditions. One of the DHRUVA beam tubes is being utilized for accelerated life testing of newly developed ion chambers.

(b) Characterization of neutron detectors;
Beam tubes are utilized to evaluate various design parameters of self-powered Inconel neutron detectors (SPND) such as gamma and neutron sensitivity, linearity etc. These types of SPNDs will be used for regulation and protection of PHWRs.

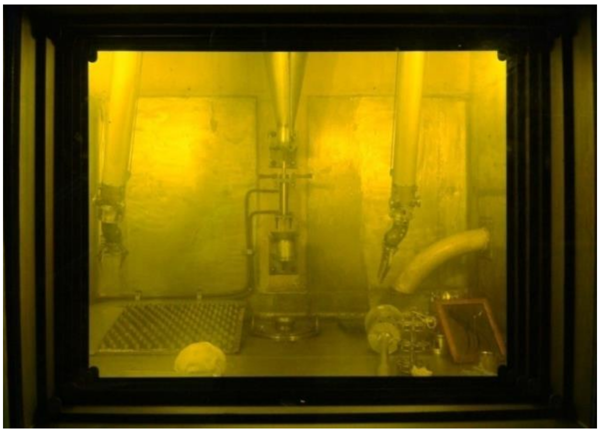


FIG. 6. Hot cell facility.

2.5. OTHER FACILITIES (radio-isotope production)

(a) Tray rods
DHRUVA reactor provides facility for producing fairly large quantities of radioisotopes by irradiation of target materials for required period of irradiation at desired flux level. Each isotope tray rod can carry ninety capsules containing targets enclosed in cold welded standard aluminium capsules. Provision for installing three tray rods at a time exists in the reactor. Tray rods are handled on-power using fuelling machine.

The hot cell is partly lined with 6 mm SS sheet and can handle a maximum activity of 2×10^5 Ci of Cobalt-60. The ventilation provided in the cell ensures a negative pressure of about 4 mm of WC. The cell is equipped with through-the-wall type master slave manipulator (MSM) having slave arm in hot cell and master arm in the control station. Radiation shielding window (RSW) has been provided in hot cell for handling various radio-isotopes with MSM. The window has one alpha protection glass and five shielding glasses of different thickness, size and density for protection from gamma. The Table 2 gives target materials and specific activity of radio-isotopes.

TABLE-2. SPECIFIC ACTIVITIES OF MAJOR RADIOISOTOPES PRODUCED AT DIFFERENT REACTOR POWERS

Isotope	Half life	Natural abundance (%)	Target material	Typical irradiation (RMWDs)	Specific activity (Ci/gm)
Mo-99	66.02 h	24.13	MoO_3	500	0.49-0.56
Lu-177	6.73 d	2.59	$LuCl_3$ (82%)	1500	14000
Ir-192	74.2 d	37.30	Ir	4000	420
Sm-153	1.93 d	26.75	Sm_2O_3 (98.7%)	550	1050
I-131	8.0 d	34.08	TeO_2	1000	125000
Co-60	5.2 year	100.00	Co	25000	20-100

(b) Self-serve facility;
Self-serve facility is for producing small quantity of radioisotopes at relatively lower flux levels. DHRUVA provides a set of two self-serve units located at upper through tubes. Each unit has five irradiation locations. The neutron flux at the irradiation location is of the order of 5×10^{12} $n{\cdot}cm^{-2}{\cdot}s^{-1}$. An aluminum capsule containing target material is enclosed in a spherical ball. The ball is rolled into the irradiation location under gravity and after irradiation the ball is rolled out into a lead shielded flask. Further extraction and handling of sample is carried out in the hot cell. This irradiation can be carried out without affecting the reactor operation and is ideally suited for the production of short-lived radioisotopes.

FIG. 7. Self-serve ball with capsule.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTICS

The organizational structure of the plant management composed off various divisions and sections. Dedicated division (s) is involved in the design, modification, safety review and commissioning of all experimental facilities. Post irradiation, the assemblies are removed from the pile with proper radiation protection and stored in shielded rooms under continuous radiation monitoring. The organization has developed expertise in safe disposal of such irradiated assemblies to the waste management facility.

3.2. HOT CELLS, PIE FACILITIES

A hot cell facility for handling irradiated radio-isotopes exists in the reactor. There is no provision of post irradiation examination facility in DHRUVA reactor complex. However the research centre has adequate facilities for PIE.

a) Aging studies and data collection for residual life assessment of SSCs;
Dhruva has seen more than three decades of service life. Hence a detailed action plan for aging management of DHRUVA has been initiated. As a part of this action plan the residual life of system structures and components (SSC) affecting safety will be assessed. Based on the aging studies refurbishment of plant SSCs to extend the operating life of the reactor will be formulated.

b) Setting up of Prognostics and Health Management (PHM) and Reliability Laboratory;
Plant equipment and components indicate different characteristic prior to their failure. These characteristics get reflected in the measured parameters. Assessing the deteriorating condition of an equipment or component in advance to predict the required maintenance before a component fails saves significant man hour required in maintenance as well as component outage time. To characterize the failure pattern and develop the requisite model for different components (mechanical, electrical, and micro-electronic) a prognostics and health management and reliability analysis laboratory is being setup.

c) Probabilistic safety analysis of DHRUVA;
A level 1 probabilistic safety assessment (PSA) of DHRUVA reactor has been performed for full power operation considering internal initiating events to evaluate core damage frequency (CDF) of the reactor and the work on PSA levels 2 and 3 has been initiated. Additionally deterministic and probabilistic fire risk assessment of plant has been started. As a part of PSA application, development of risk monitor and risk based In Service inspection has been completed.

d) Human resource management;
All the persons who shall manage the reactor operation or maintenance are imparted training in their respective field. To develop a quality manpower strong emphasis is given on the formal training and licensing of personnel in research reactors. The organization has a structured training programme. As a part of training class room lectures are delivered by senior and well qualified operation and maintenance staff members. On-the-job training in different plant areas is also provided for better exposure. In parallel the learning is assessed by respective system checklists (a set of questions) which are signed by the authorized plant personnel.

e) Development of DHRUVA simulator;
Towards improving the quality of human resource and reduce human error, development of DHRUVA simulator has been started. The simulator will be utilized for familiarization of the trainees with the reactor start-up, shutdown and normal operation. Additionally it will be used to observe and model the human response during anticipated operational occurrences. The output obtained from the simulated environment will be utilized to further enhance human reliability.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES

The organization has sufficient expertise and capabilities to design and manufacture experimental devices and measurement systems. A well-established human resources development system is also available.

4. RECENT ACHIEVEMENTS AND FUTURE PLAN

a) *Irradiation of Zircaloy calandria tube samples*;

The calandria tubes (CT) of Indian PHWR were manufactured from Zr^2 cold rolled strips by forming and seam welding operations. A manufacturing route for fabricating seamless CT by hot extrusion and cold pilgering process has been developed. Samples of welded and seamless Zircaloy calandria tubes were test irradiated in DHRUVA reactor to study their comparative In-pile growth behaviour. These studies along with subsequent studies done at fast breeder test reactor at higher fluence, established the manufacturing route for the PHWR calandria tubes.

b) *Irradiation of Thoria assembly*;

A number of thorium assemblies were irradiated in DHRUVA for generating data on thorium based fuel cycle, especially on U^{233} production, and contamination level of U^{232} in U^{233}.

c) *Advanced heavy water reactor (AHWR) fuel pin irradiation*;

It is proposed to irradiate a cluster of six AHWR fuel pins in a regular fuel position of DHRUVA. The proposed cluster consists of three pins of Th-Pu MOX and three pins of Th–U MOX.

d) *Validation of thermal hydraulic codes*;

The steady state and transient temperature of DHRUVA fuel pin clad were evaluated theoretically using different thermal hydraulic models and codes. In order to validate the design, an instrumented fuel assembly was fabricated and irradiated at DHRUVA to measure temperature of fuel pin clad during steady state and transients.

The steady state and rate of change of clad temperature of fuel pin clad were monitored during reactor operation and power changes. A flow coast down experiment was conducted at high reactor power by tripping all main coolant pumps simultaneously. The results matched reasonably well with the results obtained by COBRA-IV-I code.

e) *Establishing applicability of neutron noise measurement technique for diagnostics of in-core components for heavy water reactors*;

The neutron flux signal in a reactor is composed of a steady direct current component produced by power operation of reactor and a very small fluctuating AC component 'noise'.

Analysis of neutron noise from suitably located sensors is a proven technique to monitor the vibration and hence condition of in-core components of light water reactors. However, the use of neutron noise has been rare for PHWRs as it was generally felt that the unfavourable transform function characteristics of the reactor would limit its applicability. To access the applicability of this technique in heavy water moderated and cooled reactor, an experiment was conducted in DHRUVA using in-core and out-of-core sensors.

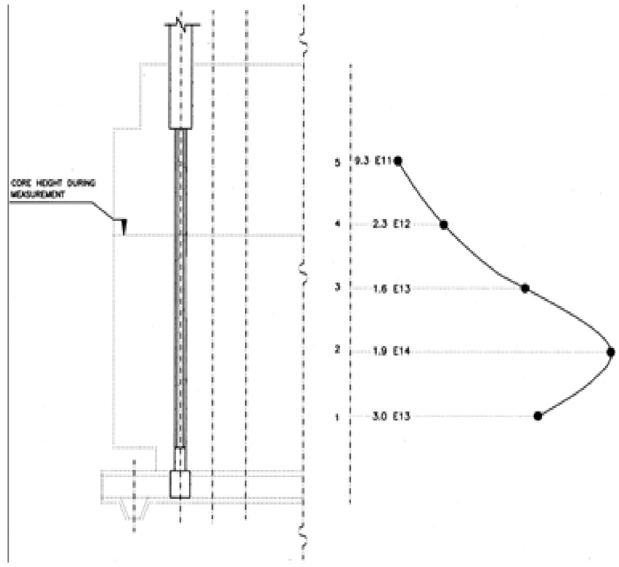


FIG. 8. Axial flux pattern.

A specially designed assembly consisting of five neutron sensors with integral cable assembly, mounted at different elevations was installed in one of the vertical experimental positions in reactor core.

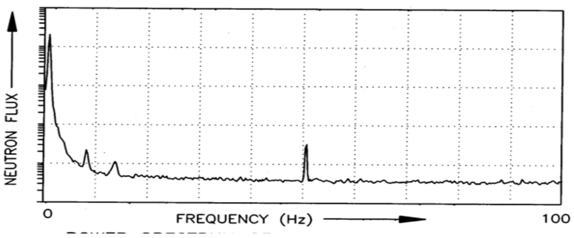


FIG. 9(a). Power spectrum of in-core neutron noise.

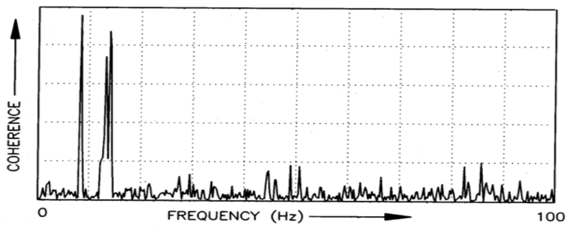


FIG. 9(b). Coherence of in-core neutron noise and mechanical vibration.

The AC component of the signal was electrically separated and recorded simultaneously with vibration signal tapped by accelerometers mounted on nearby core structure extensions. The recorded signal was analysed by FFT analyser. Prominent distinct nearby frequency peaks could be identified both in the AC output of the neutron sensors of the special assembly and the mechanical vibration of in-core structures. This experiment indicated that neutron noise could be effectively utilized as an early diagnostic technique for in-core components in heavy water reactors.

Profile 9

HIGH FLUX RESEARCH REACTOR (HFRR)

INDIA

1. GENERAL INFORMATION AND TECHNICAL DATA

Construction of a new 30 MW (Th) High Flux Research Reactor (HFRR) is proposed at Vizag, India mainly to meet the projected demand for radioisotopes in the country and requirements of reactor based facilities for basic and applied research. The proposed reactor with its high neutron flux and irradiation volume will provide the appropriate facilities for research in reactor fuels and reactor materials, condensed matter research for study of microscopic structure and dynamics of materials and dynamic radiography and time-of-flight refractrometry.

HFRR is an open pool type research reactor, fuelled with plate type fuel assemblies with Low Enriched Uranium (LEU) in the form of U_3Si_2 dispersed in Al matrix with a clad of Al-alloy. Demineralized water is used as coolant and moderator. The physics design of the High Flux Research Reactor aims at:

— High thermal and fast neutron flux for experimental/irradiation positions;
— Low fuel inventory;
— Negative temperature, power and void coefficient of reactivity;
— Two independent shut down systems and high reactivity safety margins;
— Sufficient excess reactivity to cater to experimental and irradiation reactivity loads along with operational reactivity requirements.

The maximum thermal and fast neutron flux in the core is about 6.7×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ and 1.8×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ respectively. The thermal, epithermal and fast neutron flux at various irradiation positions are shown in Table 1.

TABLE-1. NEUTRON FLUX AT DIFFERENT IRRADIATION POSITIONS

Irradiation position	Thermal neutron (<0.621 eV)	Epithermal (0.625 eV-821 keV)	Fast neutron (>821 keV)
In-core water hole	6.7×10^{14}	3.4×10^{14}	1.8×10^{14}
In-core peripheral water holes	4.4×10^{14}	2.4×10^{14}	1.3×10^{14}
Irradiation holes in D_2O, 7 cm away from core edge	3.7×10^{14}	6.0×10^{13}	1.2×10^{13}
Irradiation holes in D_2O, 20 cm away from core edge	2.9×10^{14}	5.0×10^{13}	1.7×10^{12}
NTD-Si irradiation holes	$2.5\text{-}4.0 \times 10^{13}$	-	-
Pneumatic carrier facility	2.0×10^{13}	-	-
Flow test loop	2.0×10^{14}	-	-
Beam tubes	3.0×10^{14}	5.0×10^{13}	1.0×10^{13}

The reactor core is surrounded by an annular heavy water reflector tank which houses most of the irradiation positions. The maximum thermal neutron flux available in the reflector tank is 3.7×10^{14} n·cm^{-2}·s^{-1}.

A compact reactor core is required for achieving high neutron flux but it restricts the number of in-core experimental/irradiation positions in the core. Hence the heavy water reflector tank around the reactor core helps in:

(i) Sustaining high thermal neutron flux over a large radial distance/volume;

(ii) Achieving large core excess reactivity.

The estimated temperature coefficient of reactivity of fuel, coolant and power at BOC are: -0.013 mk/^{o}C, –0.05 mk/^{o}C and –0.12 mk/MW respectively.

Reactor is designed to have negative void coefficient of reactivity, both in fresh & irradiated conditions of the fuel assemblies. Hence, formation of steam voids, if any, during operation is not expected to lead to unsafe reactivity transients. The estimated value of void coefficient will be about –1.2 mk/% void.

The core is designed to operate at a thermal power of 30 MW without refuelling for about 25 days of operating cycle.

The core is mounted on a fixed grid plate having 25 locations arranged in square lattice. Two types of fuel assemblies i.e. Standard Fuel Assembly (SFA) and Control Fuel Assembly (CFA) are used. A SFA consists of 20 fuel plates and 2 Al-alloy end plates. The minimum amount of ^{235}U in a fuel plate will be about 22.7 g and in a standard fuel assembly is about 454 g. The fuel and the end plates are swaged into two Al-alloy side plates to maintain a uniform water gap between the plates. The CFA is basically a standard assembly with eight fuel plates removed to create two adequately sized gaps to accommodate two hafnium absorber elements. Each water gap (to accommodate hafnium blade) is about 9.7 mm. Refer Fig. 1 for core layout. The nominal core consists of:

— Seventeen standard fuel assemblies;
— Four control cum shut off rods, two shut-off rods;
— One central and one peripheral irradiation facility for high thermal/fast neutron flux experiments/irradiations.

The reflector tank is designed to accommodate horizontal beam tubes and vertical irradiation positions of different diameters. It is an annular tank with a square cross section in the inner region and a circular cross section in the outer region. The reflector tank is filled with heavy water which acts as reflector. It is supported with a skirt support at the outer cylindrical shell location on the top of the Inlet plenum. The skirt is an extension of the outer shell of the reflector tank with a flanged end so as to connect the tank to the inlet plenum through a flange joint. The overall height of the reflector tank is about 1000 mm with a reflector height of about 920 mm. It is designed to contain heavy water with the required leak-tightness and to accommodate the neutron beam tubes, irradiation rigs, and pneumatic irradiation facilities, material irradiation facilities, flow test loop, NTD silicon irradiation facilities and other experimental facilities. Core chimney is connected with the reflector tank but supported on an independent support structure. Neutron beam tube segments are connected to the reflector tank (refer Fig. 2).

The reactor block consists of a reactor pool and a service pool, filled with de-mineralized water. The pools are open tanks connected to each other by a transfer canal. The reactor pool contains the reactor core, the reflector tank with irradiation facilities & neutron beam tubes, core chimney & its support structure, shut-off and control rods & their support structures, ion-chambers & their support structures, process piping, natural circulation valves and various experimental facilities (refer Fig. 3).

The service pool provides storage space for irradiated fuel elements, temporary storage of irradiated material, space for handling irradiated materials and has provision to lower the fuel transfer shielding flask at a specified location.

Hot cells for handling isotope assemblies are located at the top of the service pool which house facilities to handle & transport irradiated isotopes. This room is connected to the service pool. Tray Rod containing the isotope capsules is taken out of the reactor core and stored in service pool storage racks for cooling. From storage rack location tray rod is transported to the hot cell. The rooms are provided with master slave manipulators to load and unload the isotope capsules.

The C&I system consist of reactor protection system, reactor regulating system, alarm annunciation system, plant information system, etc. Inputs to these systems are the nuclear parameters and the process parameters of the reactor. C&I system architecture, encompasses all the above systems and are categorized into Class IA, IB and IC, as per the IEC classification. The reactivity is controlled by vertical movement of four control cum shut-off rods (CSR) which move in two banks.

With the help of a pre-determined reactivity control program the withdrawal of the four CSRs in a bank is effected and the reactor is made critical. The withdrawal procedure of the CSRs by the RRS is continued till sufficient power level is reached.

First Shutdown system consists of four control cum shut-off rods along with two shut–off rods (SOR) located in the core. Partial dumping of heavy water from the reflector tank is provided as a back-up shut down system.

The reactor core is cooled by demineralized water flowing from bottom to top through the core. The water leaving the core exits through chimney provided at the core top. The average fuel channel velocity of coolant is about 8.5 m/s. The coolant system consists of two independent loops, each consisting of two delay tanks, two Main Coolant Pumps (MCP), two shut-down cooling pumps, two heat exchangers and the associated piping and valves. In each loop, hot water from the chimney outlet after passing through the delay tanks joins the pump suction and is passed through heat exchangers where heat is transferred to the secondary coolant. Cold water from heat exchanger outlet is fed back to the core inlet plenum. A part of the flow from the heat exchanger outlet is discharged into service pool and reactor pool and an equivalent amount of flow is drawn from pool through the chimney, which along with the coolant water from the reactor core is circulated through the loop. For reducing the radiation field due to the N^{16} activity, two delay tanks, sized to obtain a delay of about one minute, are provided at the core outlet in each loop. To ensure adequate coast down cooling flow during the transient between pump trip and establishment of core cooling by shut-down cooling pumps, flywheel is provided on each MCP. Siphon break lines are provided on coolant outlet lines from the reactor pool and on the coolant inlet lines to reactor and service pools to prevent complete drainage of pool water in an unlikely event of a pipe failure outside the pool.

The reactor heat from the secondary cooling water system is ultimately rejected to the atmosphere through a set of cooling towers. Long term shutdown cooling of the core is achieved by natural convection flow of pool water. The reactor block is housed in a confinement building with once-through circulation of ventilation air. Air is exhausted to atmosphere through a HEPA filter bank and a stack. Main design features of HFRR are given in Table 2.

The reactor has facilities for radioisotope production, beam tube research and neutron activation analysis. List of major isotopes produced in the reactor is given in Fig. 3. Facilities for production of NTD silicon and for installation of a Flow Test Loop (FTL) exist in the reactor. The reactor would also be equipped with two in-core irradiation facilities (Irradiation Location, IL-1 & IL-2) for carrying out fuel and material irradiation (refer Fig. 1).These facilities, besides carrying out irradiation of fuel and materials for current generation reactors, are being designed to cater to similar needs for future thermal reactors proposed under Indian nuclear power programme.

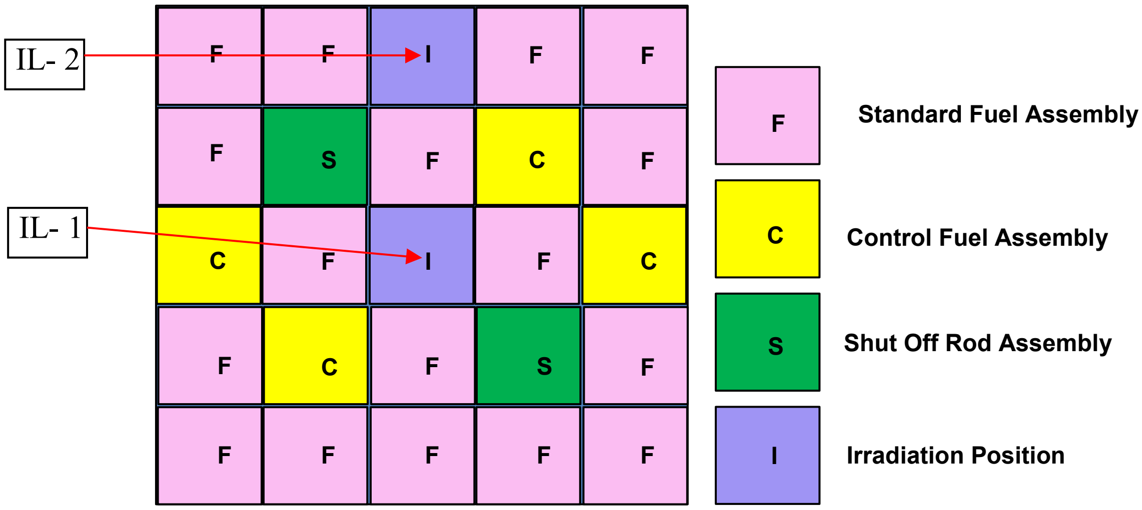


FIG. 1. Reactor core layout.

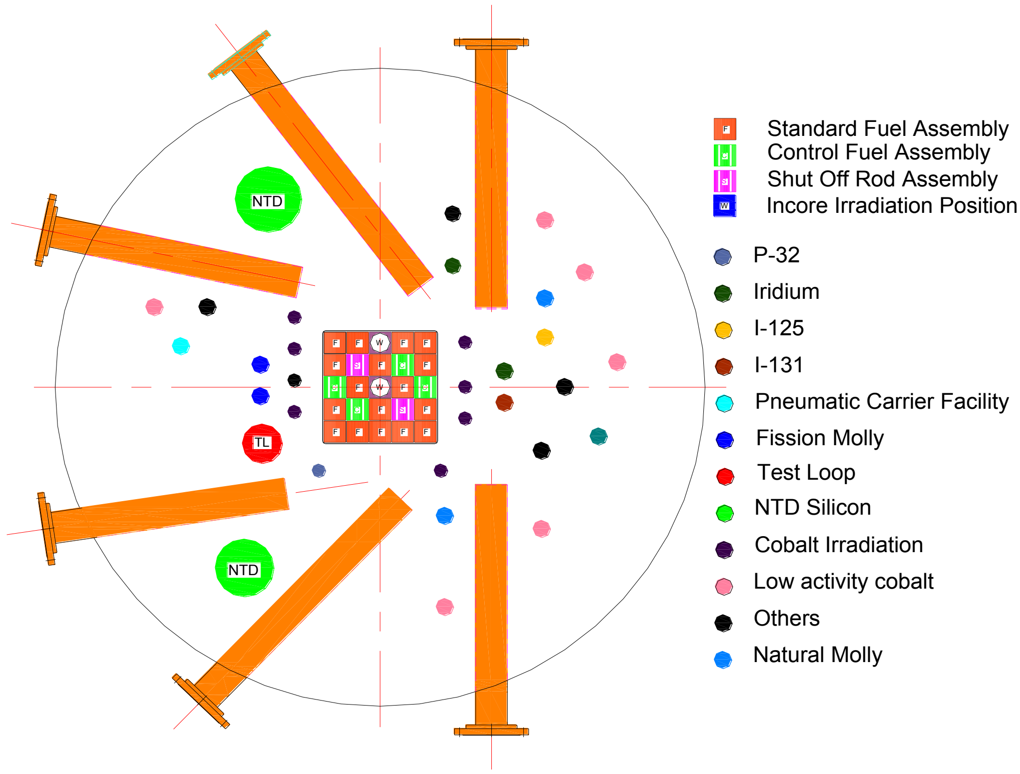


FIG. 2. Reflector tank.

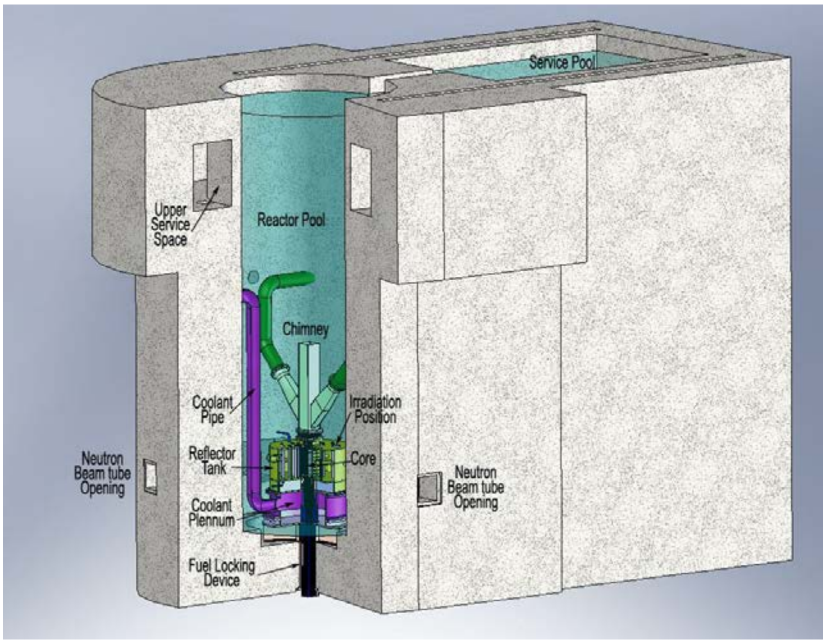


FIG. 3. Reactor pool block.

TABLE 2. MAIN DESIGN FEATURES OF HFRR

Reactor type	Open tank in pool type
Thermal power	30 MW
Coolant	Light water
Reflector	Heavy water in annular tank
Fuel (LEU)	Plate type U_3Si_2 , 19.75% enrichment
Loading density	4.3 g/cm^3
Moderator	Demineralized water
Core	Fixed core with 25 lattice positions
Maximum thermal flux	6.7×10^{14} n·cm^{-2}·s^{-1}
Maximum fast neutron flux	1.8 x 10^{14} n·cm^{-2}·s^{-1}
Operation cycle	25 days
Shutdown systems	– 4 hafnium Control-Cum-Shut Off Rods + 2 Shut off rods – Partial reflector heavy water dumping
No. of beam tubes	6
In core irradiation positions	2
Irradiation/experimental positions in reflector region	– Irradiation positions: 26 – Pneumatic carrier facility: 1 – Neutron transmutation doping: 2 – Flow test loop: 1

TABLE-3. RADIO ISOTOPE PRODUCTION

Radio isotopes	Batch irradiation time	Flux ($n \cdot cm^{-2} \cdot s^{-1}$)	No of position
Fission Molly	Weekly	1.8×10^{14}	2 (65 mm)
Co-60	2 years	3.0×10^{14}	4 (65 mm)
Co-60	5 years	3.8×10^{14}	1 (50 mm)
P-32	2 months	1.8×10^{13} (fast)	1 (50 mm)
Co-60	40 days	3.0×10^{13}	2 (65 mm)
Co-60	6 months	1.0×10^{14}	4 (65 mm)
Mo-99	1 Week	1.8×10^{14}	1 (65 mm)
Ir-192	1 month	2.0×10^{14}	2 (65 mm)
Ir-192	4 months	2.0×10^{14}	
I-131	1 month	2.0×10^{14}	1(65mm)
$I\text{-}^{125}$	2 weeks	2.0×10^{14}	1(65mm)
Se-75	2 months	5.0×10^{13}	1(65mm)
Yb-169	1 months	5.0×10^{13}	1 (65mm)
NTD-Si		$2.0\text{-}4.0 \times 10^{13}$	1 (250 mm) 1 (200 mm)
PCF		5.0×10^{13}	1 (65 mm)
Test Loop		2.5×10^{14}	1 (150 mm)
Others			4 (65 mm) + 1 (50 mm)

2. PROSPECTIVE EXPERIMENTAL FACILITIES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

India has a fleet of PHWRs of both 220 MWe and 540 MWe capacity. It is proposed to setup many 700 MWe PHWRs in future. Besides, India also has BWRs. In future, it is proposed to set-up, imported as well as indigenously developed PWRs. Data related to long-term behaviour of materials, under irradiation environment, has long been one of the most important requirements for the structural designers. Such data is needed for studying behaviour of existing materials as well as for residual life prediction of reactor components and systems. Detailed study of degrading mechanisms would help in developing new materials and alloys and studying their irradiation behaviour before their use in reactors. Some of the common materials of interest are Zr-2.5% Nb, low alloy pressure vessel steel and austenitic steels. Irradiations conditions required for these materials span in the region of fluence between 10^{20} n/m^2 to 10^{26} n/m^2 and temperature of 250-300°C. The important properties to be studied include tensile strength, elasticity, toughness, irradiation creep and growth, impact properties, fatigue strength, hardness, thermal conductivity and coefficient of thermal expansion. The sample size would be based on the property to be studied.

The future nuclear programme of India envisages, setting up of many high temperature reactors for hydrogen production as well as molten salt coolant based reactors, including Molten Salt Breeder Reactors (MSBR), attractive for thorium utilization during the third stage of Indian nuclear power programme. The reactors currently being designed include Compact High Temperature Reactor (CHTR), Innovative High Temperature Reactor (IHTR) and Molten Salt Breeder Reactor. The structural materials in these reactors would be subjected to high temperatures varying from 600°C to 1000°C and the reactor will utilize wide range of coolants such as heavy liquid metals (lead-bismuth eutectic) and molten salts. The proposed incore materials include BeO, high density isotropic graphite, carbon-carbon composites and other ceramics such as SiC and SiC-SiC composites. Metallic materials for reactor shells, coolant plenum and other components include alloys of refractory metals such as Niobium based alloy with 1% zirconium and 0.1% carbon, tantalum based alloy with 10% tungstun, Inconel-617, Hastelloy-N (modified), Incoloy-800H as well as various oxidation resistant coatings. It is proposed to qualify these materials as well as coatings for actual reactor environment before they can be utilised in actual reactor applications. Besides, India is a partner of ITER group. Irradiation of candidate materials for this reactor is also proposed to be carried out.

2.2. FUEL TEST LOOP

A fuel test loop (FTL) facility is planned in High Flux Research Reactor (HFRR). This facility simulates nuclear power reactor operating conditions for conducting various irradiation experiments on fuel and reactor materials. The FTL is used:

- To investigate the irradiation behaviour of candidate fuel for new reactors under the representative PWR operating conditions;
- To study irradiation damage of clad material;
- To carry out material irradiation for characterization of structural steel for advanced reactors.

The pressurized water loop consists of an In-Pile test Section (IPS) and Out of Pile System (OPS). The IPS will be inserted in the irradiation thimble in the reflector region and accommodates the test fuel pins in the pressure flask for fuel irradiation experiments. Out of pile system consists of various process systems i.e. Main circulation system, Emergency cooling system (ECS) and other auxiliary systems. Main coolant is supplied to the IPS at the required temperature, pressure and flow conditions consistent with a test fuel. The nuclear heat added within the IPS is extracted by the main cooling water.

Salient design features of PWL in HFRR are as follows:

- The loop is designed for maximum temperature of 350°C and pressure of 17.5 MPa in the test section;
- PWL has the capability to produce the various environments based on the need of fuel irradiation experiments;
- For fuel irradiation experiment, the power generated/deposited in the fuel is monitored.
- Water chemistry is maintained and adjusted by on-line purification systems or by addition of chemicals;
- The sample holder houses instrumentation for monitoring temperature, fluence, and water chemistry.

Feasibility of making additional provision in the loop to cater to the needs of performing material irradiation experiments, corrosion experiments and study of complex phenomena of Irradiation Assisted Stress Corrosion Cracking (IASCC) is also being explored.

2.3. MATERIAL IRRADIATION FACILITIES

2.3.1. Overall irradiation requirements

An overall list of requirements which are to be addressed by the material irradiation facilities in HFRR has been compiled keeping in mind the irradiation needs for the present as well as future reactors. These are listed below:

(a) Study of post irradiation behaviour with respect to the material properties such as biaxial creep, thermal conductivity, coefficient of thermal expansion, swelling (irradiation growth), Young's modulus, and other mechanical properties such as tensile strength, fracture toughness, low cycle fatigue, corrosion studies, hydrogen pickup etc.;
(b) Possibility of irradiating materials up to a temperature of 1000 °C;
(c) Measurement of small strains (0.1%), with precision of measurement of the order of few microns;
(d) Accurate temperature measurements and stability during the entire irradiation process with an aim to control within ± 10°C;
(e) Dose up to few tens of dpa and irradiation for few years;
(f) Integrated neutron fluence (E >1 MeV) up to 10^{26} n/m^2;
(g) Environment: Inert as well as under molten salt coolant;
(h) Measurement of fluence seen by the samples.

2.3.2. Irradiation devices planned

The irradiation facility is planned to consist of two types of devices based on different schemes of control:

- In-pile instrumented loop, IL-2 (precise control of temperature and coolant environment);
- Tray-rod capsule train, IL-1.

(1) In-Pile instrumented loop based facility:
The In-Pile instrumented loop based facility is installed in the outer irradiation location (IL-2) of the reactor core. This type of device is proposed to achieve high temperature range during irradiation as well as precise control of the sample temperature. The device is pre-mounted with various accessories required for temperature measurement and control such as pumps, heaters, heat exchangers, thermocouples, etc. The required instrumentation for precision dimensional measurement would also be part of the device. The irradiated samples would be replaced with a fresh batch of samples, inside a hot-cell, after completion of each irradiation campaign. Geometric setup of the device can be summarised as:
  (a) The multiple material samples would form a longitudinal train. The number of samples and the length of the train would be based on the properties for which the samples are being irradiated. For example samples for bi-axial creep would be in the form of train of pressurised capsules.
  (b) The size of the samples would be decided based on the space constraints as well as limitations on mass of the samples so as to limit heat generation for achieving better temperature control.
  (c) This train of samples is assembled in the device along with the necessary instrumentation like sensors for dimensional measurement, neutronics as well as temperatures.
  (d) Although the reactor coolant is light water, the device would have coolant/ environment as per irradiation requirement. Currently irradiation under two environments viz. inert gas and molten salt has been proposed. The temperature and flow of the coolant would be regulated to achieve the required temperature as well as precise control on its variations within samples.
  a) The complete device would be attached to the reactor by a suitable interface.

Thus in-pile loop based device is a dedicated system with its own coolant, auxiliary devices, instrumentation, etc. for maintaining the required irradiation temperature of the samples within the allowable variation of ±10 °C. The device would interact with the reactor for fulfilment of only irradiation flux requirement. The In-Pile loop based device would be customised according to the type of coolant being used taking into considerations the pressure, temperature and the safety requirements of the various coolants. It is proposed to design the device for two temperature ranges:

(a) Temperature Range 1: 200°C - 450°C with inert gas environment such as helium
(b) Temperature Range 2: 450°C – 1000°C with molten salt environment

The structural material of the loop would be selected from the consideration of temperature as well as environment.

(2) Tray-rod capsule:
The tray-rod capsule based device is based on conventional capsule design for irradiation of the sample material. This device would be located at the central irradiation location (IL-1) of the reactor core. The capsules would be in direct contact with the reactor coolant. The peak temperature of the samples kept in the capsules is regulated to some extent with the use of insulation, wrapped around the samples within the capsules.

Geometric setup of the device can be summarized as:

(a) Standard capsules would be made up of suitable material to hold the samples inside them;
(b) Size of the samples would be limited by the heat generation due to irradiation and the heat removal capacity of the reactor coolant;
(c) Based on the temperature requirement of the samples, insulation would be provided inside the capsules;
(d) Instrumentation for monitoring the flux (irradiation strips) as well as temperature (temperature foils) would also be accommodated inside the capsules;
(e) Based on the environment conditions required during the irradiation, the capsule might be filled with various media (inert gas, liquid metals, molten salts, water etc.).

Subsequent to the irradiation campaign, the samples would be transferred to hot cell for carrying out post-irradiation examination so as to determine the changes in the material properties. This device would be used for comparative studies between irradiated and non-irradiated samples for nano-indentation experiments, studies for understanding microstructure evolution, small punch tests on miniature samples, small scale specimen testing such as mini tensile, bending tests, etc.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL LOGISTIC

Required logistic for safe and secure management of fresh and irradiated assemblies is planned at site. Sufficient in-house expertise is available for this. The organizational structure of the plant management is composed off various divisions and sections, based on their area of expertise. The organization has also facilities for disposal of radio-active wastes to waste management facility.

3.2. HOT CELL, PIE FACILITIES

Hot cells: The reactor is provided with two hot cells for handling irradiated samples. Separate hot cell is planned for radiochemistry analysis and for other requirements. Facilities for neutron beam research, including provision for a Cold Neutron Source also exists in the reactor.

PIE facilities: It is planned to provide post irradiation examination facilities at HFRR, Vishakapatnam for catering to the requirements of performance evaluation of irradiated nuclear fuels, develop, quantify and ensure improved performance of new materials, surveillance programme for condition monitoring and ageing management of critical reactor components.

The state-of the- art PIE laboratory is provided with following facilities:

- Drop Tower Impact Testing Machine for measuring impact energy and DBTT;
- Quadrapole mass analyser for fission gas analysis;
- Remotized-shielded metallograph for micro structural studies;
- SEM/EDAX for micro structural and compositional studies;
- GDOES for getting chemical composition;
- HVQMS for compositional studies;
- X-Ray diffractometer with Texture and Stress Goniometer for texture studies;
- Shielded Differential Scanning Calorimetry equipment;
- Laser Flash Thermal Diffusivity Measuring Instrument;
- Direct Current Potential Drop (DCPD) system;
- Electro Magnetic Resonant (EMR) Machine for fatigue pre-cracking;
- Screw Driven Machine (TIRA) for mechanical testing;
- Screw Driven Machine (STAR) for ring tension test;
- Servo hydraulic UTM (BISS) for fracture toughness studies;
- Small Punch Test (SPT) for getting fracture and mechanical properties;
- Automated Ball Indentation (ABI) test for miniature sample studies.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES

The organization has sufficient expertise and capabilities to design and manufacture experimental devices and measurement systems. A well-established human resources development system is also available.

4. RECENT ACHIEVEMENT

Not applicable

Profile 10

RSG-GAS

INDONESIA

1. GENERAL

The RSG-GAS is a 30 MW open pool type reactor. General description of the RSG-GAS facility is presented in Table 1.

TABLE 1. GENERAL DESCRIPTION OF THE FACILITY

Type of reactor	Open pool
Nominal thermal power	30 MW
Maximum neutron flux: – Thermal – Fast	 2.51×10^{14} n·cm^{-2}·s^{-1} 2.28×10^{14} n·cm^{-2}·s^{-1}
Fuel type	Material testing reactor (MTR)
Enrichment	19.75%
Reference pressure of the facility	1.997 kg/cm^2
Coolant (type and flow direction)	H_2O (top-down)
Moderator	H_2O
Reflector	Be
Maximum heat flux	221.7×10^4 W/m^2
Nominal flow rate	3200 m^3/h

1.1. REACTOR CORE

The reactor parameters are given in 0.

TABLE 2. GENERAL REACTOR PARAMETERS

Parameter	Value & tolerance
Core shape	cube
Core lattice size (x)	100 cm
Core lattice size (y)	100 cm
Active core height	60 cm
Moderator type	H_2O
Coolant type	H_2O
Recirculation	Forced convection
Nominal thermal power	30 MW
Nominal moderator temperature	40°C
Nominal moderator density	1 g/cm^3
Nominal coolant inlet temperature	40°C
Nominal coolant outlet temperature	49°C
Nominal Reflector temperature	40°C
Operating coolant outlet pressure	1.497 bar
Operating coolant inlet pressure	1.997 bar
Fuel enrichment	19.75%
Control system	AgInCd Fork type

1.2. REACTOR GEOMETRY

FIG. *1* shows a plan view of the reactor.

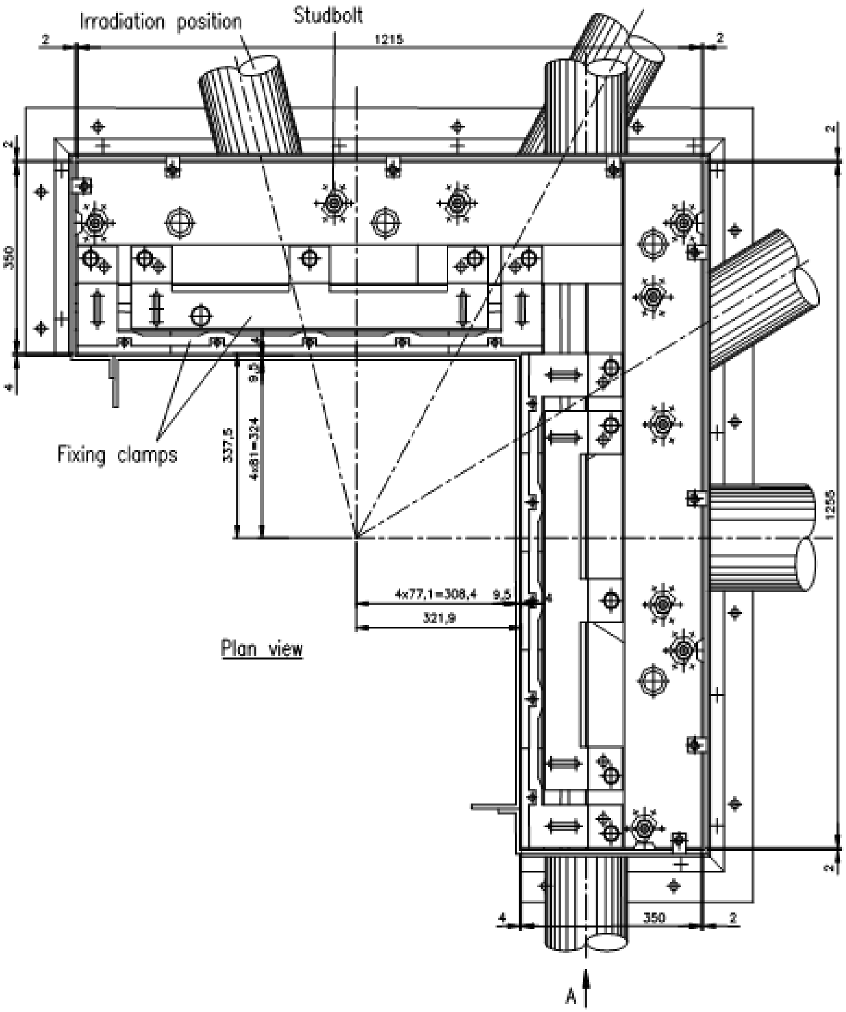


FIG. 1. Plan view.

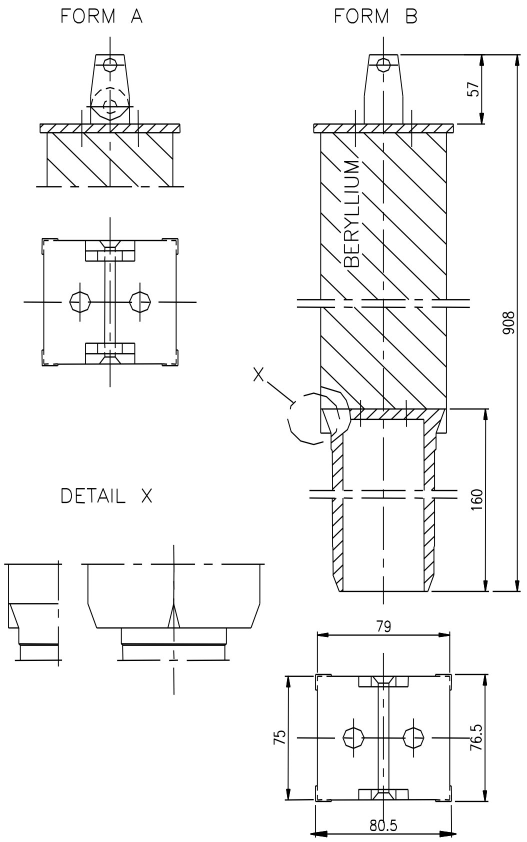


FIG. 2. Beryllium reflector.

1.3. REFLECTOR ELEMENTS AND REFLECTOR BLOCK

Beryllium is used as reflector material to surround the active core by an optimum reflector. Two sides of the active core are covered by two rows of beryllium elements to provide sufficient flexibility for the insertion of irradiation devices as needed. FIG. *2* shows the beryllium reflector.

The other two sides of the core are connected to a beryllium block reflector specifically designed to have an optimum beam tube coupling to the core.

All beryllium parts are made from nuclear grade beryllium. The beryllium elements consist of a lower end fitting which fits into the grid plate bore-holes, a square-shaped beryllium bar within a height of 683 mm and 79 mm × 75 mm outer cross section dimensions. Some of the beryllium reflector elements are provided with one bore-hole of 50 mm diameter for the insertion of irradiation samples. These elements are also suitable for the acceptance of fixed absorbers as an option for the present core design as well as for the acceptance of the start-up source. If necessary the bore holes can be filled with beryllium plugs.

The position of the reflector elements can be chosen at the grid plate wherever the suitable flux level for the specific irradiation purpose exists to provide maximum flexibility for the users.

Furthermore, all elements can be turned around regularly to avoid any bowing effect of the beryllium elements due to different elongation caused by swelling under radiation on different sides of the element. Only small elongation effects will occur that do not affect the intended use of the elements.

In order to keep the cooling gaps between the elements in the specified limits, the elements have to be turned round after the first year of full power operation and then each two years of full power operation.

The beryllium block main design features are:

— Two square-shaped blocks are arranged to form the two legs of a rectangular angle. The outer/inner lengths of the two legs are 1215 mm/865 mm and 1255 mm/905 mm, respectively. The height of the blocks is 800 mm, the width 350 mm.
— The overall beryllium block is divided into three layers with widths of 40 mm, 82 mm and 222 mm respectively. The axial separation is in accordance with the demands resulting from the incoming beam tubes. Azimuthally the block is separated into parts taking into account manufacturing aspects and available material block sizes.

The beam tubes are guided into the central part of the reflector. A gap is left between the core shroud and the reflector for cooling purposes. On the outside, the reflector is surrounded by an aluminium shroud, which is also positioned to leave a cooling gap. In addition, cooling gaps are provided between certain layers of the beryllium block and around the beam tubes that remove the heat released in the reflector. The cooling water flows down through the cooling gaps and enters the plenum below.

In addition to the beam tubes for irradiation purposes there are five vertical bore-holes over the whole height within the beryllium block reflector, one bore-hole with 35 mm diameter in the second layer. If they are not used they can be filled with plugs.

1.4. FUEL COMPONENT

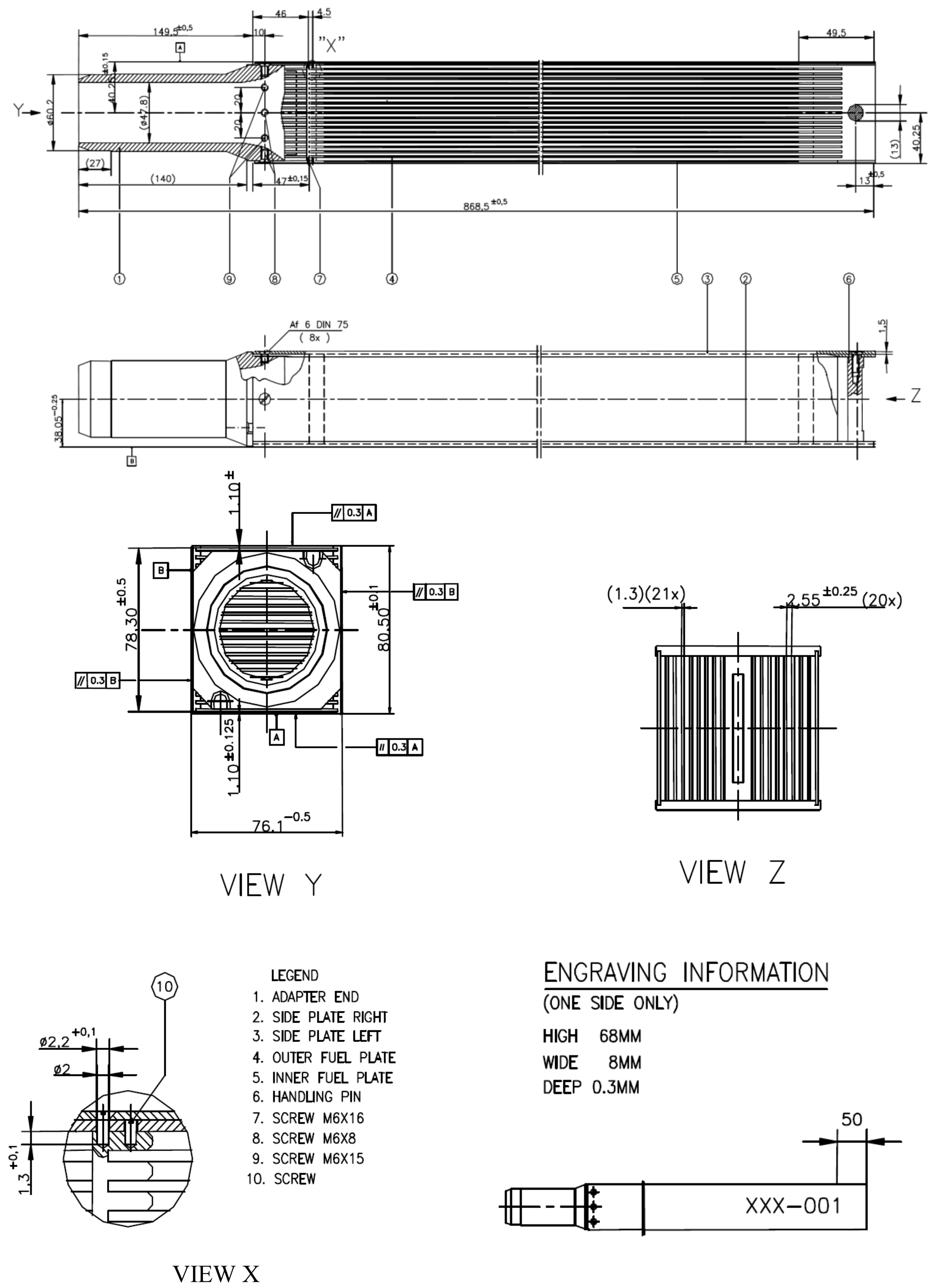


FIG. 3. Fuel components.

The fuel assembly specification is given in 0.

TABLE 3. FUEL ASSEMBLY SPECIFICATION

Fuel assembly specifications	Value	Tolerance
Assembly geometry	Plate	
Enrichment	19.75%	+0.2
Uranium density	2.96 g/cm^3	
Plate geometry		
Number of plates / rods per assembly	21	
Assembly x dimension	80.52 mm	
Assembly y dimension	868.9 mm	
Fuel element type	MTR/ U_3Si_2-Al	
Axial assembly dimension		
Axial plate dimension	70.75 mm	0.15 mm
Active height	625 mm	0.2 mm
Meat thickness	0.54 mm	
Clad thickness	0.38 mm	
Coolant gap size	2.5 mm	
Clad-fuel gap	N/A	
Clad-fuel gap material	N/A	
Burnable absorber type	N/A	

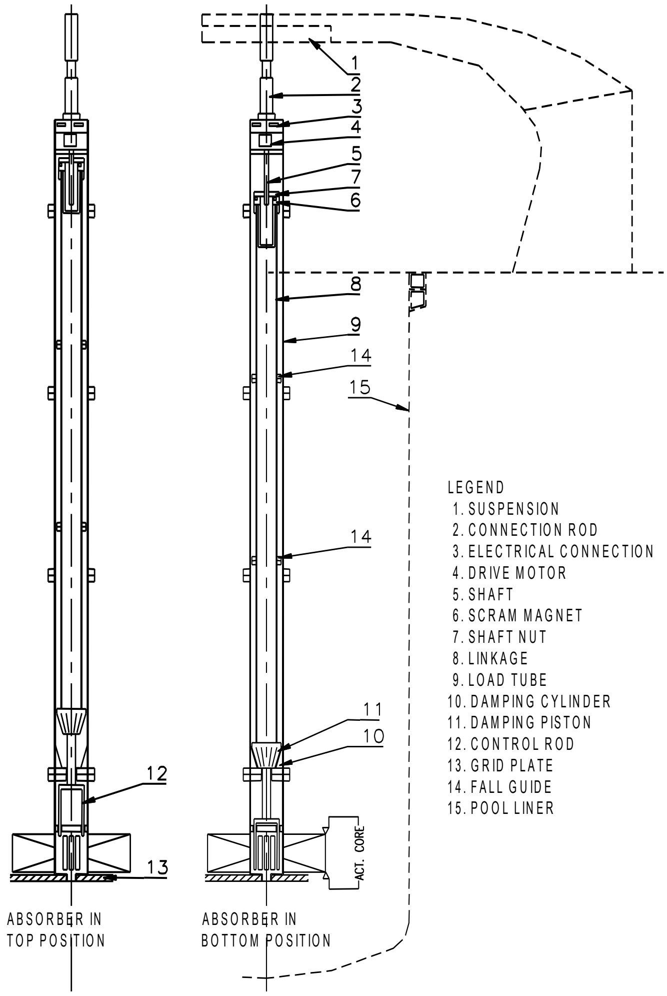


FIG. 4. Absorbers.

1.5. CONTROL COMPONENT

The control assembly specifications are given in Table 4.

TABLE 4. CONTROL ASSEMBLY SPECIFICATIONS

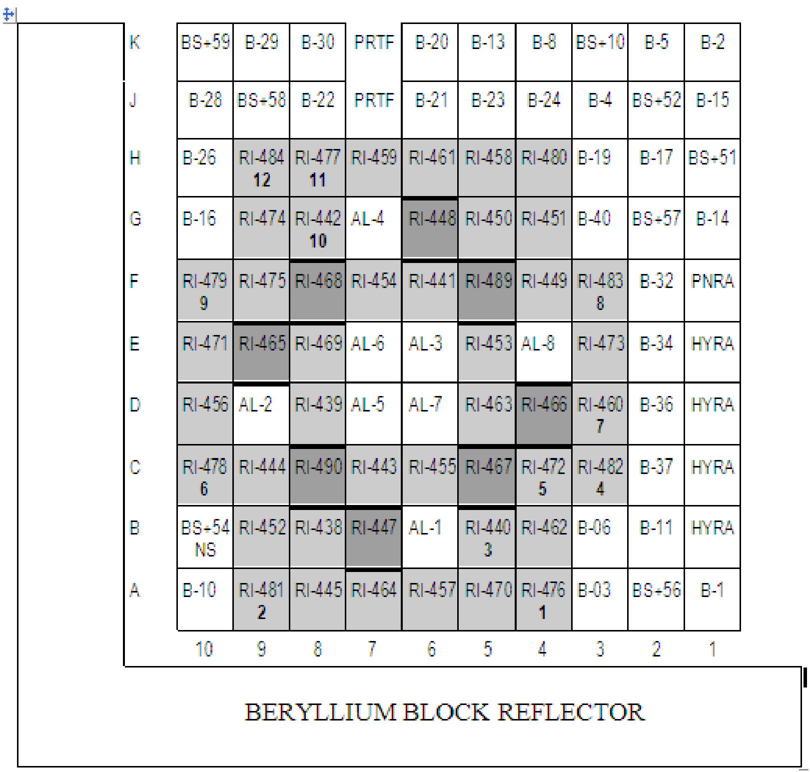
Parameter	Value	Uncertainty
Control type	Fork	
Full travel distance	600 mm	
Direction of travel	Vertical	
Number of control rods	8	

The material specification is given in Table 5.

TABLE 5. MATERIAL SPECIFICATION

Material	Concentration
Ag	80%
In	15%
Cd	5%
Impurities	
Cr	17.4%
Ni	11.18%

1.6. REACTOR MATERIAL/INVENTORY DISTRIBUTION



B — beryllium; BS — beryllium stopper with plug; Al — aluminium stopper without plug; Rl — fuel; NS — neutron source.

FIG. 5. Core composition.

1.7. FUEL ASSEMBLY

The fuel assembly specification is given in Table 6.

TABLE 6. FUEL ASSEMBLY SPECIFICATION

Type (parallel plates/rods/concentric rings)		
Meat	Material	Enriched uranium
	Composition	U_3Si_2-Al
	Enrichment	19.75%
	Thickness	0.54 mm
	Length	600 mm
	Width/inlet diameter	62.75 mm
	Conductivity	1.07 W/cmK
	Density	2.96 g/cm^3
	Heat capacity	0.484 J/gK
Cladding	Material	$AlMg_2$
	Thickness	0.38 mm
	Length	625 mm
	Width/inlet diameter	70.75 mm
	Conductivity	2.16 W/cmK
	Density	2.70 g/cm^3
	Heat capacity	0.961 J/gK
Oxide layer	Thickness	N/A
	Conductivity	N/A
Number of fuel assemblies	Standard	40
	Control	8
Number of plates/rods/rings	Standard	21
	Control	15
Coolant channel	Gap	2.55 mm
	Width	67.1 mm

1.8. CONTROL AND SAFETY ASSEMBLIES

The control and safety assemblies specification is given in Table 7.

TABLE 7. CONTROL AND SAFETY ASSEMBLIES SPECIFICATION

Number of control assemblies		8
Number of safety assemblies		N/A
Absorber	Material	AgInCd
	Thickness	3.41 mm
	Width/internal diameter	63.09 mm
	Length	625 mm
Cladding	Material	SS-316L
	Thickness	1.56 mm
	Width/internal diameter	64.75 mm
	Length	638.3 mm

1.9. CORE GRID

The core grid specification is given in Table 8.

TABLE 8. CORE GRID SPECIFICATION

Material	$AlMg_2$
Array X-Y dimensions	99.99 mm × 95.22 mm
Lattice	81 mm × 77.1 mm
Thickness	0.125 m
Passing holes dimensions	60 mm

1.10. POOL

The pool and internal components specification is given in Table 9.

TABLE 9. POOL AND INTERNAL COMPONENTS SPECIFICATION

Diameter		5 m
Total height		13 m
Water column above the top of the core		12 m
Effective water volume		250.22 m^3
Core elevation above the pool bottom		1.2 m
Inlet pipe	Quantity	1
	Internal diameter	600 mm
	Length	4200 mm
Outlet pipe	Quantity	1
	Internal diameter	600 mm
	Length	3800 mm
Inlet plenum	Diameter	690 mm
	Height	40 cm
Outlet plenum	Diameter	690 mm
	Height	80 cm
Natural circulation device	Type	Flap
	Quantity	2
	Elevation from the core bottom	20 cm
	Opening condition	15% of nominal primary coolant flow rate

1.11. COOLING SYSTEMS

Table 10 and Table 11 summarize the specifications for the main components.

TABLE 10. HEAT EXCHANGER SPECIFICATION

Type	Shell & tube
Number	2
Pressure drop (primary side)/secondary	48 kPa (0.48 bar) / 43 kPa (0.43 bar)
Heat transfer area	780 m^2
Heat transfer coefficient	1.9627 W/m^2K

TABLE 11. PUMP SPECIFICATION

Number	3 × 50%
Power per pump	Pri. = 160 kW, Sec. = 220 kW
Nominal flow rate	Pri. = 1570 m^3/h, Sec. = 1950 m^3/h
Efficiency	87%
Rotation speed	1470 rpm (rotation per min)
Elevation height	Pri. = ± 0.00 m, Sec. = - 6.5 m
Torque	1019 Nm
Moment of inertia	80 kg m^2

Pump coast down data is provided in Table 12.

TABLE 12. THE COAST DOWN FLOW RATE OF THE RSG-GAS PRIMARY COOLING SYSTEM

Time (second)	Flow rate (%)	Time (second)	Flow rate (%)
0	100	60	17
5	75	70	15
10	58	80	13
15	48	90	11
20	38	100	10
30	30	110	8
40	23	120	7
50	19		

1.12. NEUTRONIC INPUT DATA

The neutronic data are given in Table 13.

TABLE 13. NEUTRONIC DATA

Average power density		87.569 W/ cm^3
Power density in each FE		87.569 W/ cm^3
Total peaking factor of the core		2.391
Radial peaking factor of the fuel		1.27
Axial peaking factor of the fuel		1.77
Axial power profile		See figure 6
3D power distribution		N/A
Direct heat deposition in the coolant		
Beta effective (if available, 6 group fraction and decay constants)		0.00765
Neutron lifetime		64.5 μs
Feedback coefficients	Doppler	(value or table)
	Coolant temperature (constant density)	$-1.14 \times 10^{-4} \Delta\rho/K$
	Coolant temperature (variable density)	
	Void	$-1.34 \times 10^{-3} \Delta\rho/K$
	Fuel expansion	

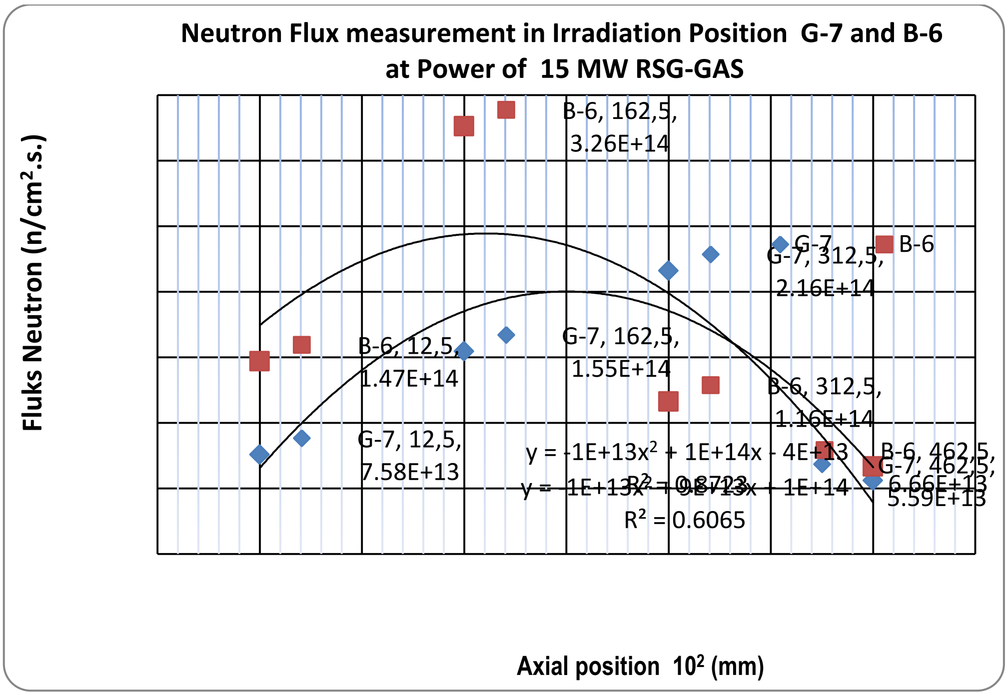


FIG. 6. Axial power distribution represented by flux neutron measurement (done at the 64th silicide reactor cycle).

1.13. SENSORS

Table 14 indicates the parameter being measured and the associated data.

TABLE 14. SENSOR SPECIFICATION

Measured parameter	Pressure
Accuracy	>99.85%
Response time	± 3 ms
Range	0–160 kPa (1.6 bar), 0–160 kPa (23.2 psi)
Units	>20

1.14. SHUTDOWN SYSTEM

The shutdown system specification is given in Table 15.

TABLE 15. SHUTDOWN SYSTEM SPECIFICATION

Total worth	13.8%
Actuation time	<0.4 s
Travelling time	<0.4 s

1.15. TRANSIENT CONDITIONS

Experimental measurement results of coast down flow of primary coolant during transient condition are summarized in Table 16.

TABLE 16. MEASUREMENT RESULTS OF THE RSG-GAS LOFA EXPERIMENT AT 30MW

Reactor system parameter	Measurement results
1. *Steady-state condition, before scram*	
Primary coolant system	
Primary coolant flow	3269 ± 48 m^3/h
Core inlet temp	39.75°C
Core outlet temp	48.03°C
Reactor power	30.045 MW
Instrumented fuel element temperature	
Clad. temp. RI-11/T1	84.4°C
Clad. temp. RI-11/T2	74.21°C
Clad. temp. RI-11/T3	75.05°C
Clad. temp. RI-11/T4	76.61°C
Coolant inlet temp RI-11/T6	Fail
Coolant outlet temp RI-11/T5	51.88°C
Secondary coolant system	
Heat exchanger	
Inlet temp.	42.60°C
Outlet temp.	35.40°C
HE-1Flowrate	2000 m^3/h
HE-2Flowrate	2050 m^3/h
2. *Fuel temperature and coolant at scram condition*	
Reactor power, MW	
Instrumented fuel element temperature	
Clad. temp. RI-11/T1	85.65°C
Clad. temp. RI-11/T2	74.81°C
Clad. temp. RI-11/T3	Not recorded
Clad. temp. RI-11/T4	77.45°C
Coolant inlet temp RI-11/T6	52.37°C
Coolant outlet temp RI-11/T5	Fail
3. *Minimum fuel and coolant temperature (± 6 s after scram)*	
Instrumented fuel element temperature	
Clad. Temp. RI-11/T1	42.41°C
Clad. Temp. RI-11/T2	42.66°C
Clad. Temp. RI-11/T3	Not recorded
Clad. Temp. RI-11/T4	46.41°C
Coolant inlet temp RI-11/T6	41.44°C
Coolant outlet temp RI-11/T5	Fail
4. *Maximum fuel and coolant temperature after scram*	
Instrumented fuel element temperature	
Clad. temp. RI-11/T1	74.81°C (112 s after scram)
Clad. temp. RI-11/T2	72.17°C (111 s after scram)
Clad. temp. RI-11/T3	Not recorded
Clad. temp. RI-11/T4	78.41°C (115 s after scram)
Coolant inlet temp RI-11/T6	48.73°C (105 s after scram)
Coolant outlet temp RI-11/T5	Fail

Some of important time signal:

— Minimum flow' occurred at 84.8% of minimum flow, at 4.4 s after switch-off of primary pump;

— Reactor scram at 4.18 s after switch-off of primary pump due to signal of 'minimum flow';
— Natural circulation flap open at 104.5 s after switch-off of primary pump.

1.16. POSITION OF THERMOCOUPLES

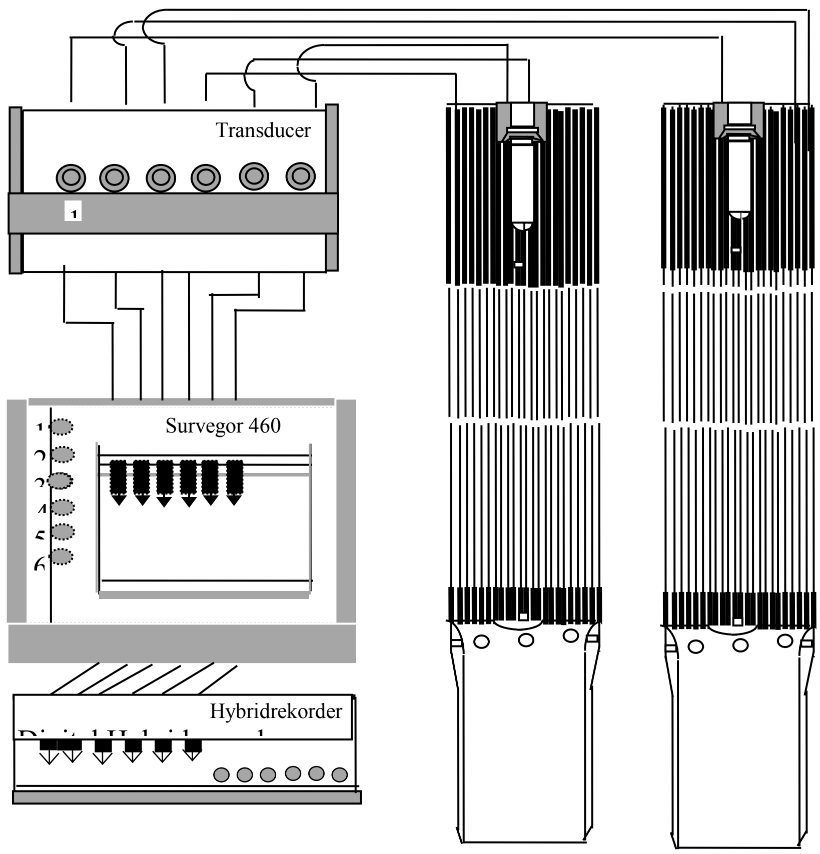


FIG. 7. Thermocouple located in instrumented fuel for temperature measurement.

Location of six units thermocouples within each instrumented fuel is shown on Table 18.

TABLE 18. THERMOCOUPLE LOCATION ON INSTRUMENTED FUEL

Instrumented fuel	Thermocouple ID	Plate number	Position on the plate
RI-10	T-4	11	50 mm below middle part of the fuel meat
	T-5	11	31mm below bottom part of fuel meat
	T-6	11	31 mm above upper part of fuel meat
RI-11	T-1	1	50 mm below bottom part of fuel meat
	T-2	10	150 mm below middle part of fuel meat
	T-3	10	50 mm above upper part of fuel meat

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

2.1.1. Power ramp test facility (PRTF)

This is capsule type experiment. This facility is used to carry out the experiment of power ramping in reactor fuel. The sample is held in a rechargeable capsule and surrounded by simulated environment like a power reactor. Power ramps involve rapid alteration of fuel power from low to higher output or vice versa. For these purpose, the fuel rods are moved up to the core in an irradiation capsule mounted on a trolley. Horizontal displacement serves to simulate load changes and power ramps in the fuel rod without having to intervene in the reactor power control system.

Inside the irradiation capsule, the pressurized conditions for the fuel rod are maintained in a pressure vessel with practically stagnant water. The heat generated is transferred to the outer wall via natural convection, and this wall is subsequently cooled by pool water flowing in forced circulation in the outer channel of the irradiation capsule.

Operational specification:
- Operating pressure of the primary circuit 160 bar and temperature 350°C;
- Water flow of the primary circuit is 3.6 L/h;
- Linear power max. 900 W/cm.

During irradiation, fuel pins are subjected to continuous checking of cladding perforation by means of continuous monitoring of primary water activity in the irradiation capsule. Post irradiation test investigation will cover axial burn-up distribution, eddy current, ultrasonic, geometry measurement and neutron radiography.

2.1.2. Experimental beam tube

A total of six beam tubes are provided with the RSG-GAS. One of them is for isotope production while the remaining is for research on material science. Those are primarily used in research of magnetic alloys, polymers and minerals of nuclear and Industrial importance. Various aspects of these materials will be investigation such as the structure, texture defects, precipitates, dynamics, etc.

The technique selected for carrying out the investigations included small angle neutron scattering, neutron diffraction and polarized neutron scattering, etc.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

This facility has been installed inside the reactor pool, but due to some reason it has never been operated.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA, etc.

The existing facilities are no longer in operation.

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Investigation of corrosion of reactor materials is conducted manually by putting several coupon assemblies at certain place and certain period in primary pool cooling. The coupon assemblies are hung individually inside the spent fuel storage pool adjacent to the hot cell wall. After having been exposed by radiation those coupons are analysed to investigate what type of corrosion are encountered. Materials of coupons represent material of the reactor primary structures.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES etc.

There are several capsules with different function so as different environment. They include of rabbit system capsules made of Al; and poly ethylene operated at about 42^{o}C; FPM target capsule (^{99}Mo) made of stainless steel operated at higher temperature than that of rabbit system; iridium (Ir) capsule made of aluminium operated at high temperature as FPM.

2.5.1. Devices for investigation of fuel and structural materials behaviour and characteristics (swelling, gas release, creep, long-term strength, relaxation resistance, etc.).

Irradiated fuel and structural behaviour are investigated by non destructive test and destructive test inside hot cell.

Irradiated fuel is verified visually using binocular camera and its thickness is measured by micrometer.

To investigate blister on the fuel surface, fuel is heated using a proper furnace and followed by measurement using micrometer.

Radiochemistry facilities is used to conduct spent fuel burn-up measurement using chemical agent (destructive test analysis) while content of radioactive fission products within fuel plate is determined by gamma scanning. Other testing conducted are metallographic test and hardness test.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

Fresh fuel is checked for thickness, blister and homogenous of ^{235}U distribution within fuel plate using X-ray.

3.2. HOT CELLS, PIE FACILITIES (radiochemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

Hot cell is used to investigate irradiated fuel characteristic including fission product distribution within fuel plate done by gamma scanning and burn-up analysis done using spectrometer.

3.2.1. Neutron beams facilities

— Research and development in materials science and technology:
  • Crystal structure study in various crystalline materials;

- Size, structure and conformation of mol;

— Materials characterization and inspection:
- Residual stress and texture measurement
- Neutron radiography for materials inspection

3.2.2. Equipment for neutron beam scattering

(1) DN1 — Neutron diffractometer for residual stress measurement. Application on: residual stress measurement on tungsten reinforced titanium; Aluminium samples welded using TIG method and three-crystal aluminium sample.

FIG. 8. DN1.

TABLE 19. INSTRUMENT CHARACTERISTICS

Beam source	Thermal neutron (S6)
Location	Reactor experimental hall
Beam collimation	Parallel collimator 40'
Monochromator	Doubly focused bent Si (311)
Monochromator take-off-angle	0–90°
Diffraction angle range (2θ)	10–120°
Detector	BF_3 point detector
Counting system	Canberra
Control system	Labo + PC Windows
Aquisition software	RESA (JAEA)
X-Y-X Goniometer	X = 140 ± 0.025 mm, Y = 140 ± 0.025 mm, Z = 50 ± 0.025 mm
Delta Theta/Theta	5×10 e^{-5}
Texture evaluation	Full circle goniometer
Tensile rig	1 kN + strain gage reader
Cryostat	15 K

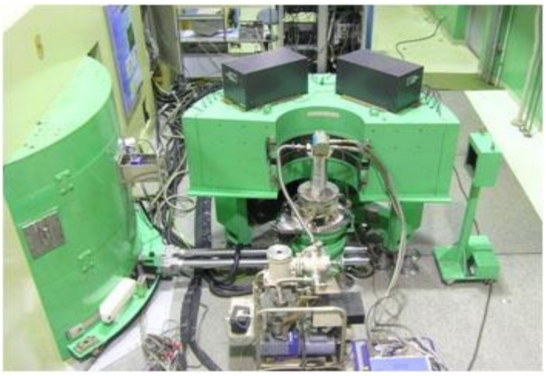
(2) DN2 — four-circle diffractometer/texture diffractometer. Application on: texture measurement in industrial products: rolled, pressed samples.

FIG. 9. DN2.

TABLE 19. INSTRUMENT CHARACTERISTICS

Location	Beam tube S5
Take off angle	$2\theta_m = 46°$
Monochromator	Bent Si (311)
Wavelengt	1.28 Å
Flux at sample position	3.81×10^5 n·cm^{-2} s^{-1}
Beam slit	Min. (5 mm × 5 mm); max. (30 mm × 30 mm)
	Detector
Monitor	BF_3 (side windows)
Main	BF_3 (end windows)
Soller collimator	20’ atau 30’
2θ angle range	$0° \leq 2\theta \leq 110°$ and $0° \leq \theta \leq 90°$
Eulerian Craddle	$-90° \leq \omega \leq 40°$; $-180° \leq \phi \leq 180°$; $-100° \leq \chi \leq 40°$

(3) DN3 — high resolution powder diffractometer. Application on crystal structure study incrystalline materials (powder and bulk of metal, ceramic, magnetic material).



Instrumental Parameters

Beam port	: NG2-3
Take off angle	: 41.5°, 89° (now) and 130°
Monochromator	: hot-pressed Ge(331), (100mm x 45mm x 10mm) mosaicity FWHM β = 23.4′ (0.39º)
Wavelength	: 2θM = 89°, λ = 1.8223 Å
Beam narrower	: Max. 40mm (horizontal) x 100 mm (vertical)
Collimators	: α1 (NG2) = ±16.2′; α2 = 20′, 40′; α3 = 6′ , 10′
Scattering angle	: -170° < 2θs < +10°
Main detectors	: 32 units of ³He detectors, XERAM 30NH15
Data acquisition	: PC compatible

FIG. 10. DN3.

(4) SN1 — triple axis spectrometer. Application: crystal structure and magnetic structure determination (elastic scattering mode); investigation of lattice dynamics and spin dynamics of condensed matters (inelastic scattering mode).

FIG. 11. SN1.

TABLE 20. INSTRUMENT CHARACTERISTICS

	Monochromator
Type	Roatating-shield type
Reflections	Pyrolytic graphite (002)
Size	75 mm (H) × 50 mm (V)
Scattering angle range	$15° < 2\theta_m < 75°$
Ki range	$5.99 < ki < 2.33$ Å^{-1}
Wave length range	$1.1 < \lambda_i < 2.7$ Å
Filter	PG filter (removable)
	Sample table
Goniometer	Translational x, y, Tilt, Rx, Ry
Scattering angle range	$-5 < 2\theta_s < 110$ degree
Analyser	same as monochromator
Collimator	#1, #2, #3, #4: (20', 40')
Control data acquisition software	nsl ver.1.0, bataco ver.1.0 (for elastic and inelastic experiment)

(5) SN2 — small angle neutron scattering spectrometer. Application: study of soft and hard matters, i.e. colloids, polymers, ceramics, alloys, magnetic materials, micellar solutions, protein solutions & virus.

FIG. 12. SN2.

TABLE 21. INSTRUMENT CHARACTERISTICS

Neutron source	Thermal neutron S5 neutron guide
Monochromator	Multidisk mechanical velocity selector
Incident wavelength	~ 3–6 Å
Wavelength resolution	~10–20%
Effective Q-range	$0.003< Q <0.3 Å^{-1}$(relevant to size of ~ 1–250 nm)
Max. flux at sample position	7×10^6 $n \cdot cm^{-2}$ s^{-1}
Beam size	min. 5 mm in diameter
Detector	2D-PSD
Sample to detector distance L1	1.0 and 1.3–18 m
Collimator length L2	1.5 m, 4 m, 8 m, 13 m, 18 m
Pinholes	30 mm, 20 mm, 14 mm, 10 mm, 7 mm, 5 mm (diameter)
Software for data processing	GRAPS, Igor NIST, SASfit, ATSAS

(6) SN3 — high resolution small angle neutron scattering spectrometer. Application: soft and hard matter: size, structure, conformation of molecules (silica powder sample, etc.).

FIG. 13. SN3.

TABLE 22. INSTRUMENT CHARACTERISTICS

Monochromator	
Type	Roatating-shield type
Reflections	Pyrolytic graphite (002)
Size	75 mm (H) × 50 mm (V)
Scattering angle range	$15<2\theta_m<75°$
Ki range	$5.99 < k_i < 2.33 Å^{-1}$
Wave length range	$1.1< \lambda_i <2.7$ Å
Filter	PG filter (removable)
Sample table	
Goniometer	Translational x, y, Tilt, Rx, Ry
Scattering angle range	$–5<2\theta_s<110$ degree
Analyser	Same as monohromator
Collimator	#1, #2, #3, #4: (20’, 40’)

Control data acquisition software	nsl ver. 1.0, bataco ver. 1.0 (for elastic and inelastic experiment)

4. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

NA

5. RECENT ACHIEVEMENTS

Some examples of R&D studies performed during the last ten years.

(1) R&D studies relate to neutron beam scattering:

- Residual stress measurement on tungsten reinforced titanium; aluminium samples welded using TIG method and three-crystal aluminium sample;
- Texture measurement in industrial products: rolled, pressed samples;
- Crystal structure study in crystalline materials (powder and bulk of metal, ceramic, magnetic material);
- Crystal structure and magnetic structure determination (elastic scattering mode). Investigation of lattice dynamics and spin dynamics of condensed matters (inelastic scattering mode);
- Study of soft and hard matters, i.e. colloids, polymers, ceramics, alloys, magnetic materials, micellar solutions, protein solutions & virus;
- Soft and hard matter: size, structure, conformation of molecules (silica powder sample, etc.).

(2) R&D related to irradiated fuel element behaviour:

- Corrosion resistance on ALMg2 cladding structure material;
- Swelling behaviour of irradiated fuel U_3Si_2-Al;
- Development and characteristic of future dispersion MTR fuel element of U-Mo;
- Various mass of U loading in the fabrication of U-7Mo/Al fuel to development of porosity within fuel meat;
- Separation process and analysis of radionuclide of ^{137}Cs within irradiated U_3Si_2-Al plate.

Profile 11

JAPAN MATERIALS TESTING REACTOR[1] (JMTR)

JAPAN

1. GENERAL INFORMATION

Information of the Japan Materials Testing Reactor (JMTR) in Japan Atomic Energy Agency (JAEA) is as follows:

- Light water cooled;
- Tank type reactor;
- Thermal power of 50000 kW (50 MW);
- Temporary shutdown[2].

The JMTR is a testing reactor dedicated to the irradiation tests of materials and fuels. It achieved first criticality in March 1968. Currently the JMTR is being operated at thermal power of 50 MW at about seven operation cycles a year, with about 30 operation days a cycle. Figure 1 shows outline of the JMTR, and cross section of the core is shown in Fig. 2. Specifications of the JMTR are shown in Table 1. Outline of the JMTR and JMTR hot laboratory (JMTR-HL) is shown in Fig. 3.

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

In August 2006, operation of the JMTR was terminated. Then, there were user's strong requests for the JMTR reoperation from various fields, such as nuclear power industries, universities, radioisotope production companies. As a result of the national discussion, the JMTR was decided to be restarted after necessary refurbishment works. The refurbishment started from the beginning of Japanese fiscal year (JFY) 2007, and replaced were the primary and secondary cooling pump motors, nuclear instrumentation system, process control system, safety protection system etc. The refurbishment was finished after four years on schedule in March 2011 (JFY 2010).

The JMTR has some expected roles after refurbishment. Firstly, the JMTR contributes to aging management of LWRs and safety measure development for LWRs, and will be utilized to develop higher burn-up fuels and to evaluate soundness of materials. Also, contribute to solving the Fukushima Dai-ichi NPPs accident. Secondly, the JMTR is utilized to innovate in the field of science and technology, such as nuclear fusion research, materials/fuels research for high temperature gas cooled reactors and damage mechanism research of materials by neutron irradiation in the field of basic research for the nuclear energy. Thirdly, the JMTR expected to expand industrial use including production of ^{99}Mo medical radioisotope used for diagnosis, or production of large dimension Si semiconductor. Finally, the JMTR contributes to development of nuclear human resources not only in Japan but also in Asian countries. It is expected to educate next generation practical engineers through on the job training.

[1] Neutron Irradiation and Testing Reactor Centre, Oarai Research and Development Centre, Japan Atomic Energy Agency (JAEA), 4002 Narita, Oarai-machi, Higashiibaraki-gun, Ibaraki, 311-1393, Japan.
[2] http://nucleus.iaea.org/RRDB/RR/HeaderInfo.aspx?RId=248.

TABLE 1. SPECIFICATIONS OF THE JMTR

Reactor power (thermal)	50 MW
Fast neutron flux (max.)	4×10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Thermal neutron flux (max.)	4×10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Flow primary coolant	6000 m^3/h
Coolant temperature	49-56℃
Core height	750 mm
Fuel	Plate type, 19.8% U-235
Irradiation capability (max.)	60 (20*) capsules
Fluence/y (max.)	3×10^{25} $n \cdot m^{-2} \cdot y^{-1}$
dpa of stainless steel (max.)	4 dpa
Diameter of capsule	30-110 mm
Temperature control (max.)	2000℃

* Capsule with in-situ measurement.

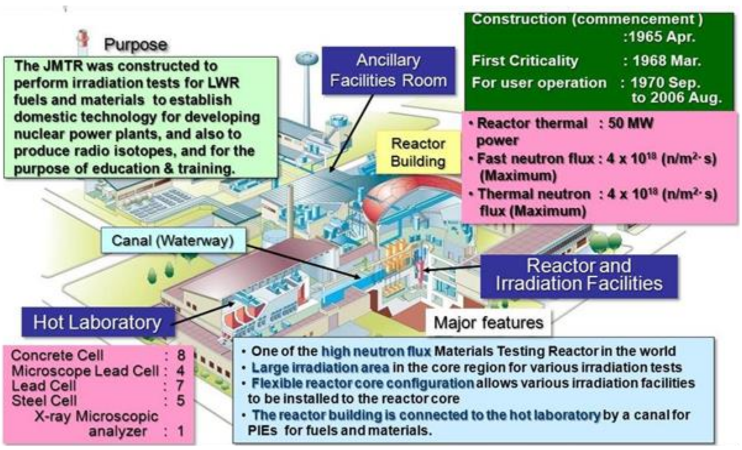


FIG. 1. Outline of the JMTR.

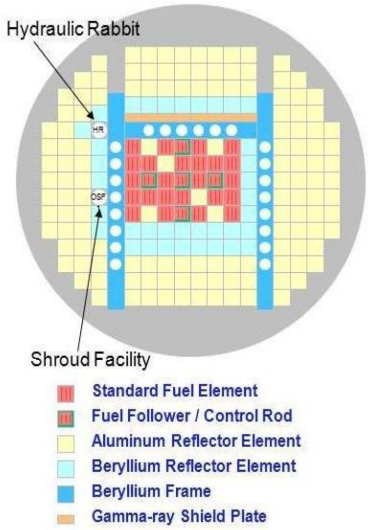


FIG. 2. Cross section of the core.

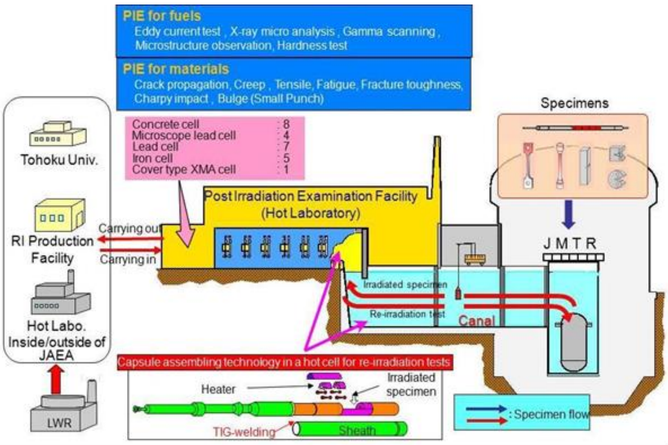


FIG. 3. Outline of the JMTR and JMTR-HL.

2. REACTOR AND FACILITIES

The reactor pressure vessel, 9.5 m high with 3 m in inner diameter, is made of low carbon stainless steel, and is located in the reactor pool, which is 13 m in depth. The control rod drive mechanisms are located under the pressure vessel, for easy handling of the irradiation facilities and fuels in the core. The core of the JMTR is in a cylindrical shape with 1.56 m in diameter and 0.75 m high. It consists of 22 or 24 (for high burn-up core) standard fuel elements, five control rods with fuel followers, reflectors and H-shaped beryllium frame.

Cooling water in the primary cooling system is pressurized at about 1.5 MPa to avoid local boiling in the core during rated power operation. The heat generated in the core is removed by the cooling water in the primary cooling system. The cooling water flows downwards in the core and transfers the heat from the core to the secondary cooling system through heat exchangers. The heat transferred to the secondary cooling system is removed away into the atmosphere in cooling towers.

2.1. EXPERIMENTAL AND TESTING FACILITIES

The JMTR provides many kinds of irradiation facilities such as the capsule irradiation facilities, the shroud irradiation facility and the hydraulic rabbit irradiation facility for the irradiation tests of nuclear fuels, materials and radioisotopes production. Each capsule facility is installed into an irradiation hole where the neutron flux is suitable for irradiation purpose. Locations of the hydraulic rabbit irradiation facility and the shroud irradiation facility are fixed in the core; however the neutron fluence is controlled by moving the specimen into or take out from the core during reactor operation. Main irradiation facilities are follows:

(1) Capsule irradiation facilities — non-instrumented, instrumented, special (such as IASCC test capsule);

(2) Shroud irradiation facility — BOCA (boiling water capsule)/OSF-1 (Oarai shroud facility-1) irradiation facility for using power ramping test;

(3) Hydraulic rabbit irradiation facility — the hydraulic rabbit irradiation facility is a water loop system to transfer the small sized (150 mm length) capsule, so called rabbit, into and take out from the core by the water flow in the loop. This facility is widely utilized mainly for the basic researches and for the production of short-lived radioisotopes;

(4) Advanced water chemistry controlled facility — this facility is used to carry out the irradiation test of materials under controlled conditions of temperature, radiation, and water chemistry aiming at irradiation tests for the IASCC (irradiation assisted stress corrosion cracking) research. This facility is consists of SATCAP (saturated temperature capsules) and water control unit;

(5) Irradiation temperature controlled facility — irradiation temperature controlled facility is composed of 14 temperature control devices and an operator station, and is possible to control simultaneously temperatures about 28 capsules in automatically, see Fig. 4. Namely, constant temperature control during the reactor power change, cyclic temperature control with wide range and other complicated temperature control are possible to carry out in automatically.

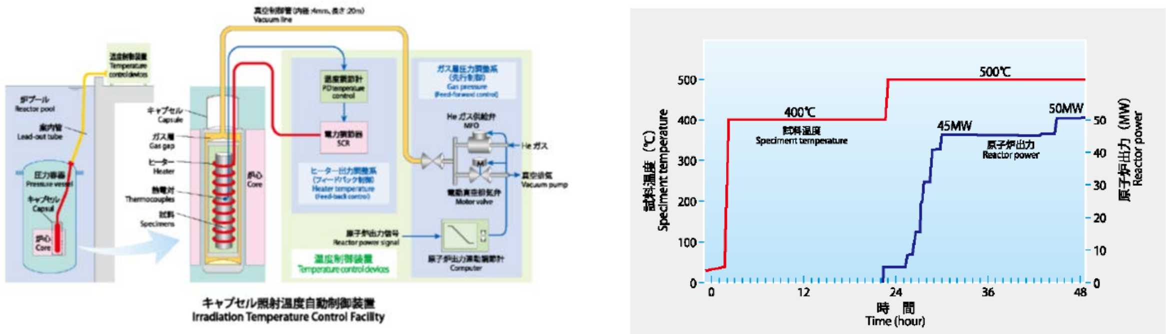


FIG. 4. Irradiation temperature control facility.

2.2. COMPONENTS OF REACTOR CORE

(1) Fuel;

Fuel element is an assembly of flat fuel plates with cooling channels between plates. Each plate contains a layer of uranium-silicide (U_3Si_2) - aluminium dispersion alloy covered with aluminium alloy cladding. Number of fuel plates per fuel element is 19 for standard element and 16 for fuel follower which is connected with the control rod. The size of the fuel element is 76.2 mm^2 square for standard element and 63.6 mm^2 square for fuel follower in horizontal cross section, and about 1200 mm and 890 mm in height, respectively. Enrichment of the uranium in the fuel is slightly less than 20wt%. Fuel element contains thin cadmium wires as burnable absorbers.

(2) Control rods;

Control rods are used to control fission chain reaction in the core. There are five control rods in the JMTR, each consists of the neutron absorber (square-tube of hafnium), fuel follower and shock section. During the operation of the control rod, the electromagnet is activated by the current to the coil, and then the link-latch is fixed in open-shape and holds the control rod. When coil current is cut off at reactor scram, the electromagnet is released and the link-latch is closed. The control rod detaches from drive mechanism and quickly drops by own weight and downward cooling water flow in the pressure vessel. Then the neutron absorber is inserted into the core and fission chain reaction is terminated.

2.3. EXPERIMENTAL FACILITIES FOR RAMPING TEST

OSF-1 provides irradiation environments for the BOCA which is used to irradiate LWR fuel samples in the condition of BWR coolant. In-pile tube of the OSF-1 is penetrated into the core of the JMTR through the top lid of reactor pressure (about 7.3 MPa) capsule made of stainless steel in which instrumented segment fuel is loaded, and is cooled by the pressurized water. The BOCA is inserted into the in-pile tube of the OSF-1, and power ramping test is performed by controlling ^{3}He gas (acting as neutron absorber) pressure in the ^{3}He gas screen of the in-pile tube. BOCA and OSF-1 are extensively used for the study on the integrity of the high performance fuels and high burn-up fuels of LWRs. These facilities are shown in Fig.5.

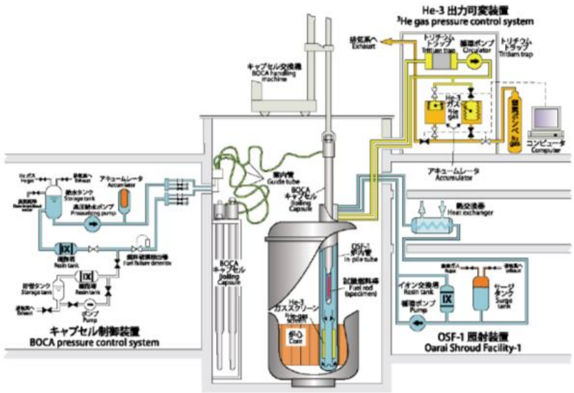


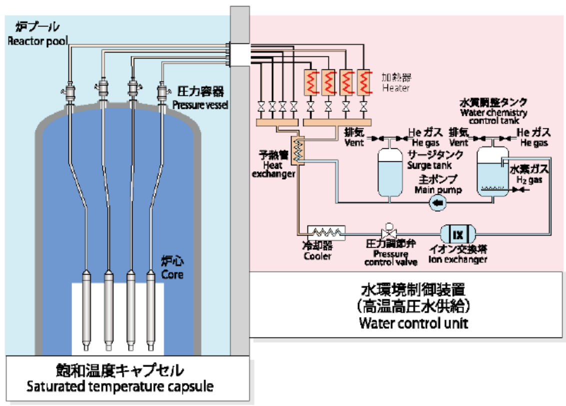
FIG. 5. Shroud irradiation facility.

2.4. INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Material specimens are irradiated by the SATCAP in the high-temperature and high-pressure water, and temperatures of all specimens are controlled to be constant by the saturated boiling phenomena at the specimen surface. This system is shown in Fig. 6.

(1) SATCAP;
The SATCAP is designed to irradiate material specimens in the high-temperature and high pressure water. Temperatures of all specimens are kept almost equal over the inner loading space due to nucleate boiling of the cooling water.

(2) Water control unit;
Water control Unit can supply the high- temperature and high-pressure water for saturated temperature capsule. According to the purpose of testing, feeding water’s temperature, flow, concentrations of dissolved oxygen and hydrogen, and electrochemical potential are controlled.



FIF. 6. SATCAP and water control unit.

2.5. DEVICE FOR CAPSULE TESTS

(1) Capsule irradiation facilities;

Specimens are inserted in a capsule, and the capsule is inserted into an irradiation hole. About 60 irradiation holes are available in the reactor core, and about 30 irradiation holes can be used for instrumented capsules. Schematic drawing of capsule and capsule controller are shown in Fig. 7. Requested specimen's temperature during irradiation is achieved by the selection of suitable irradiation holes as well as capsule design according to the irradiation purpose.

Consequently, it is possible to select specimen's temperature with the range from 45 to 2000°C corresponding to the irradiation purpose.

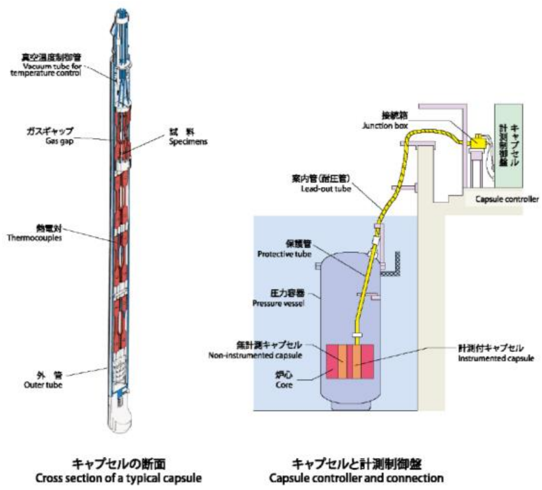


FIG. 7. Capsule and capsule controller.

(2) In-pile creep capsule with spectrum conditioning;

Creep strain under irradiation condition, which is very important for nuclear materials to evaluate its lifetime, is influenced by not only fast neutron but also thermal neutron which generates He production in the material. This capsule enables to measure creep strain directly with controlled thermal neutron flux conditions. The capsule contains thermal neutron absorption material or breeder material.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

New JMTR is preparing its restart to be used as an important infrastructure for a safety research of fuels and materials for nuclear power plants, basic research for nuclear science, industrial utilization, and human resource developments of nuclear engineers as well as operators. In June 2010, the project named 'Birth of the nuclear techno-park with the JMTR' was selected as one of projects of the Leading-edge Research Infrastructure Programme by Japanese government. In the new project, new irradiation facilities and PIE facilities is installed in the JMTR and JMTR-HL in order to promote basic as well as applied researches. This new installation of irradiation facilities and PIE facilities are shown in Figs 8 and 9, respectively. In these facilities, such as BOCA capsules and manipulator in hot cell can be treated until 110 MWd/kg. Moreover, in the project, development of user-friendly environment especially for young and female researchers is highlighted.

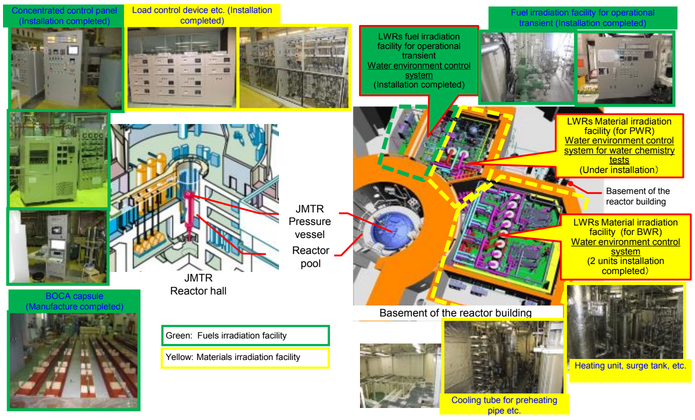


FIG. 8. Installation of new irradiation facilities.

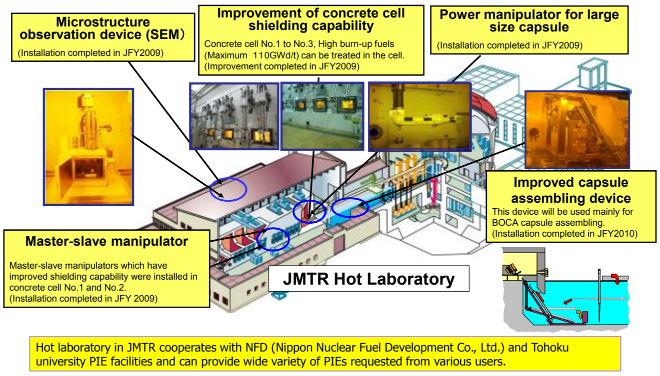


FIG. 9. Installation of new PIE facilities.

3.1. EXPERIMENTAL MATERIAL LOGISTIC

The refurbishment project works of the JMTR were started from the beginning of JFY 2007. The refurbishment project was promoted two subjects; the one is the replacement of reactor components, and the other is the construction of new irradiation facilities.

The replacement work was finished at the end of February 2011, according to the schedule. The construction of new irradiation facilities is in progress as scheduled as shown in Fig. 10.

Before the replacement of reactor components, an investigation of aged components (aged-investigation) was performed in order to identify integrity of facilities and components to be used for re-operation of JMTR. The equipment which needs replacing before the restart of the JMTR was selected after evaluated on its damage and wear due to aging significance in safety functions, past safety-related maintenance date, and the enhancement of facility operation. The replacement work of power supply system, boiler, radioactive waste facility, reactor control system, nuclear instrumentation system etc. was already carried out as scheduled [1].

On the other hand, corresponding to the user's irradiation request, new irradiation facilities, such as irradiation test facilities for LWRs materials/fuels with a purpose of long-term and up-graded operations, production facilities for medical radioisotopes for industrial purpose, were planned to install in the JMTR.

(1) Replacement of reactor components;

Based on criteria for selecting components, following items were reviewed and studied:

— Aging during 20 years during reoperation;
— Importance grade of reactor facilities/equipment;
— Conditions of facilities/equipment;
— Stable supply of spare parts in the maintenance activities during 20 years.

Replacement and renewal of the components were selected from evaluation on their damage and wear in terms of aging as shown in Table 2. Facilities whose replacement parts are no longer manufactured or not likely to be manufactured continuously in near future, were selected as renewal ones. Furthermore, replacement priority was decided with special attention to safety concerns. A monitoring of aging condition by the regular maintenance activity is an important factor in selection of continuous using after the restart. Taking also account of a continuous operation with safety, reactor facilities/equipment to be renewed was decided [2].

As a result, aged or old-designed components of the control rod drive mechanism, primary cooling system, secondary cooling system, electric power supply system etc., were to be replaced by present-designed ones. Furthermore, the replacements and renewal were possible to carry out within the range of licensing permission of the JMTR.

For facilities which are not replaced, e.g. heat exchangers, pressure vessel, secondary cooling towers and so on, their safety was evaluated from a view point of aging. The long-term operation in future will be possible by maintaining the present condition in accordance with the periodic safety review of the JMTR [3].

Renewal of the feed and exhaust air system is carried out at first, and also renewal of utility facilities of electric power supply system, boiler component, etc. is carried out at the same time. Then, facilities in the reactor building are to be finally renewed. Renewal of the JMTR had been on schedule, and completed.

After restart of the JMTR, the maintenance activity will be carried out by the maintenance program based on the periodic safety review of the JMTR. By the replacement of reactor facilities, the failure possibility of each component will decrease, and this leads the improvement of the higher reactor availability-factor in future as shown in Fig. 11 [4].

(2) Construction of new irradiation facilities;
Corresponding to the user’s irradiation request, new irradiation facilities, such as irradiation test facilities for materials/fuels, production facilities for medical radioisotopes, were planned to install in the JMTR as shown in Fig. 12.

(3) New material and fuel irradiation tests (facility for fuel development);
An irradiation facility of fuel behaviour test at transient condition has been developed to evaluate the safety for the high burn-up light-water reactor fuels, uranium and MOX fuels in the JMTR. The facility is capable of carrying out power ramping and boiling transition tests on light-water reactor fuels. The fuel irradiation test facility consists of shroud irradiation equipment, capsule control equipment and ^{3}He power control equipment [5].
The material irradiation test facility is developed to study the stress corrosion cracking (SCC) under neutron irradiation for the light-water reactor in-core materials. This facility consists of a water environmental control system in the BWR material irradiation facility simulating the BWR environment and the water chemical test facility simulating the broad water environment such as the BWR, PWR. The BWR material irradiation facility consists of a water environment control system, weight-loading control unit and capsules.

(4) New irradiation facility for industrial purpose;
One of irradiation facilities is intended to provide the ^{99m}Tc for medical use. A hydraulic rabbit irradiation facility, which is well developed and already used for irradiation in the JMTR, can be applied to the production.

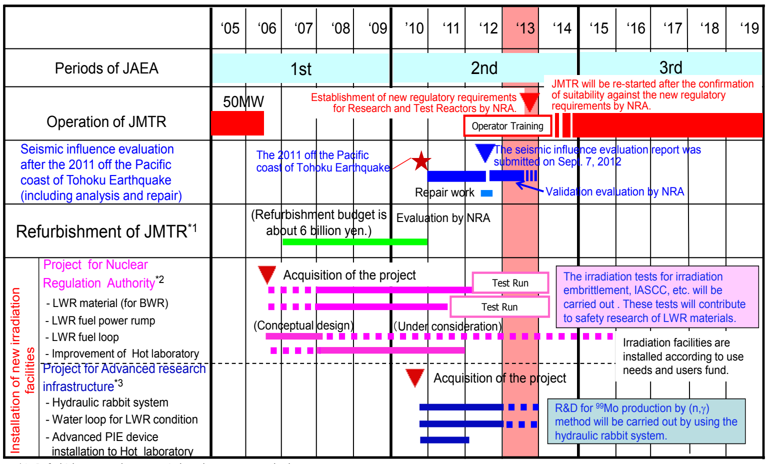
TABLE 2. SELECTION OF COMPONENTS TO BE REPLACED

Facility, system	Components	Criteria 1	2	3	4	
Reactor and process control system	Reactor control panel	o		o	o	Repl.
	Process control panel	o	o	o	o	Repl.
	Neutron instruments	o	o	o	o	Repl.
	CRDM	o	o	o	o	Repl.
Reactor cooling system	Maim pump motors	o	o	o	o	Repl.
	Maim heat exchangers		o			Cont.
	UCL circulation pump	o	o		o	Repl.
	Secondary cooling system main pump					Cont.
Radiological waste disposal system	Emergency blowers	o	o		o	Repl.
	Regular blowers					Cont.
Power supply system	Power supply units	o			o	Repl.
	High voltage transformer	o		o	o	Repl.
Other system	Water demineralizer	o			o	Repl.
	Boiler units	o			o	Repl.

Notes: criteria for selecting components to be replaced:

(a) Safety point of view:
- (1) Aging of components; o — there is possibility;
- (2) Importance of safety feature; o — importance is high level;
- (3) Maintenance experience; o — high trouble frequency or short service life time etc.;

(b) Improvement of availability:
- (4) Affordability of spare parts; o — difficulty.



*1: Refurbishment works are carried out by government budget.
*2: IASCC, Irradiation embrittlement, Hafnium irradiation and fuel ramp tests are being prepared by the NRA (Nuclear Regulation Authority) project.
*3: Hydraulic rabbit system, Water loop for LWR condition and Advanced PIE device installation to hot laboratory are being prepared by the MEXT project.

FIG. 10. JMTR refurbishment work schedule.

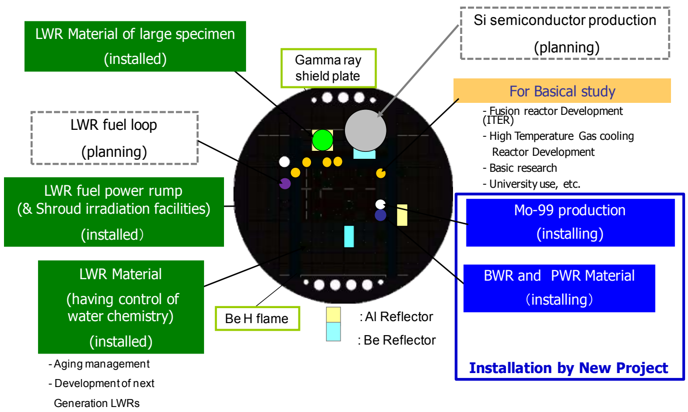


FIG. 12. Outline of JMTR core irradiation facilities.

3.2. HOT CELL, PIE FACILITIES

The JMTR-HL was founded to examine the specimens irradiated mainly in the JMTR, and has been operated since 1971. Post irradiation examinations (PIE) for research and development in wide variety of nuclear fields such as nuclear fuels and materials are carried out in the JMTR-HL. The JMTR-HL is located adjacent to JMTR, and is connected to JMTR by a water canal, 6 m in depth with 3 m width. Irradiated radioactive capsules are transferred speedily and safely through the canal with the sufficient shielding capability of the water. High radioactive materials can be handled in the JMTH-HL, and various PIEs are performed treating high dose specimens.

The JMTH-HL consists of eight concrete cells which attached four microscope lead cells, seven lead cells and five steel cells. Dismantling of capsule, X-ray radiography, dimensional measurement etc. can be performed in the concrete cells as shown in Fig. 13. The lead cells and steel cells are used for PIEs of materials, and PIEs for fuels can be carried out only in the concrete cells. In the lead cells, PIEs are carried out such as tensile test, instrumented impact test, Irradiation Associated Stress Corrosion Cracking (IASCC) tests, visual inspections, dimensional measurements. In the steel cells, PIEs for high temperature tensile/compression tests, fracture toughness tests, creep tests, fatigue tests, etc. are carried out. The specimens are transferred from the concrete cells using transport casks. Structure of the concrete cell and lead cell is shown in Fig. 14.

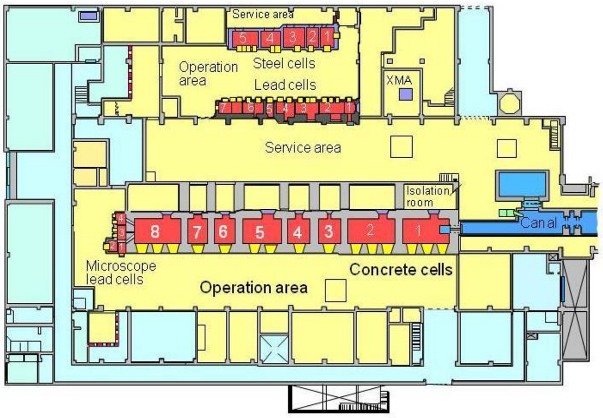


FIG. 13. Horizontal view of JMTR-HL.

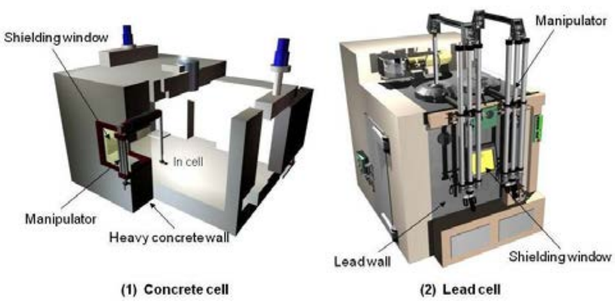


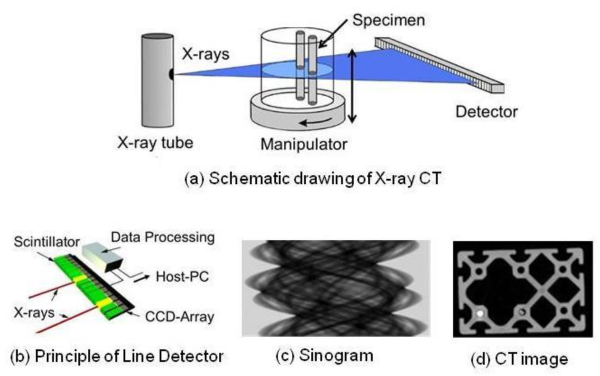
FIG. 14. Structure of the concrete cell and lead cell.

Taking four years from the beginning of JFY 2007 to March 2011, the JMTR had carried out refurbishment works as mentioned in the previous section [6].

During this period, advanced equipment/facilities have been developed and installed in the JMTR-HL. Obtaining high valuable technical data is requested as PIEs in order to contribute to safety management as well as lifetime expansion management of nuclear power plants and progress of science and technology. For that purpose, a three dimensional X-ray radiography system is developed and installed in the JMTR-HL [7]. Figure 15 shows the principle of the X-ray CT. A cone shaped radiation beam is emitted by the X-ray tube, and its intensity distribution, named 'sinogram', is measured by the detector, which is composed of scintillator, CCD-array, data processing and host-PC. The radiation intensity is reduced depending on density and thickness of the inspection target in front of the detector. The CT image is obtained by restructuring of the sonogram. Figure 16 shows the three dimensional X-ray inspection system installed in JMTR-HL. The system consists of an X-ray generator, an X-ray detector, a machine for specimen movement, a control board and a data processing unit. Its specifications are summarized in Table 3. To get clear CT image, it is necessary to reduce the noise signal caused by the gamma-ray emitted by radioactive specimens. Then, the GOST (Gamma-ray Offset Scanning Technique) program has been developed, and installed in the system. Resolution performance test was carried out, and result is shown in Fig.17. X-ray CT measurements were carried out using platinum double wire specimens (diameter: 0.05-0.8 mm, distance between wires: 0.05-0.8 mm), and spatial resolution is confirmed up to 0.16 mm. Performance test results using irradiated fuel rod, burn up at 25 MWd/kg, is shown in Fig. 18. From the test, we can see that the clear image is possible to obtain by the developed system.

TABLE 3. SPECIFICATIONS OF X-RAY CT INSPECTION SYSTEM

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



(a) Schematic drawing of X-ray CT

(b) Principle of Line Detector (c) Sinogram (d) CT image

FIG. 15. Principle of X-ray CT.

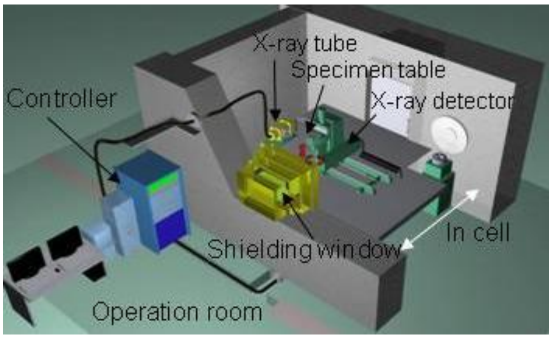


FIG. 16. Schematic drawing of 3D X-ray CT.

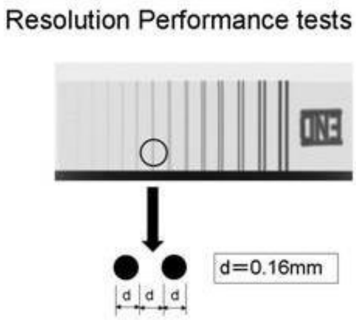


FIG. 17. Resolution performance test.

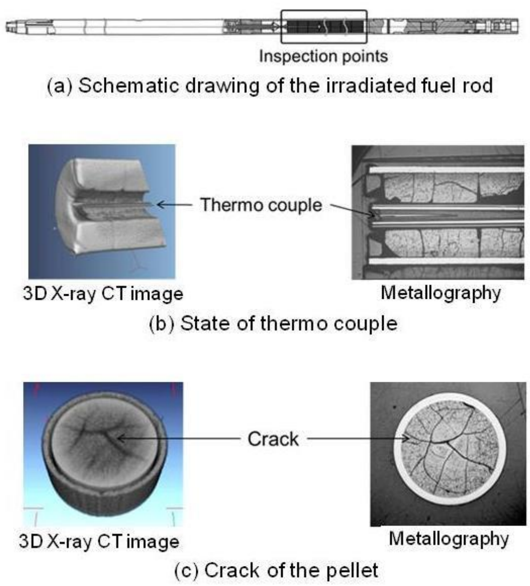


FIG. 18. Performance test s using irradiated fuel rod.

One key funding is the 'Leading-edge Research Infrastructure Programme' from the Japanese government in June 2010. From this funding, complex-type microstructure analysis equipment, which consist of TEM (transmission electron microscope), FIB (focused ion beam processing equipment) and XPS (X-ray photoelectron spectrometer), have been installed in the JMTR-HL.

(1) TEM;

By electron beam irradiation to the sample and analysing the scattered and transmitted electrons of the sample, an atomic level structural analysis is possible. The analysis will be applied to the research on irradiation damage from observation of irradiation defect and/or changes in metal structure. Three types of observation modes are available:

— TEM mode: at 200 kV accelerating voltage, it is possible to observe up to maximum magnification of 20 000 000 times, and the resolution is 0.1nm.
— STEM (Scanning transmission electron microscopy) mode: by combination with EDS (Energy dispersed spectroscopy) and EELS (Electron energy Lloss spectroscopy), it is possible to carry out element analysis, element mapping, etc.;
— SEM (Scanning electron microscopy) mode: it is possible to find the initial observation point, and to analyse the structure of sample which cannot transmit the electron beam;
— Figure 19 shows the photograph of TEM and typical image obtained by the TEM. Moreover, specifications of TEM are summarized in Table 4.

TABLE 4. SPECIFICATIONS OF TEM

Item		Specifications
Accelerating voltage		200 kV
Resolution	TEM	0.1 nm
	STEM	0.2 nm
Magnification	TEM	20 000 000 times
	STEM	20 000 000 times
Sample stage	Moving range	X,Y: from –1 to 1 mm, Z: from –0.2 to 0.2 mm
	Inclination angle	X: from -25 to 25° Y: from 130 to 30°

FIG. 19. Photo of TEM and typical obtained image.

(2) FIB;

It is possible to prepare observation samples in micrometer order from irradiated bulk specimens by the Gallium (Ga) ion beam sputtering to the bulk specimen. From this, TEM samples can prepare directly from bulk irradiated specimens in the JMTR-HL. Moreover, by combining a Ga ion gun having a large current of 60 nA with a scanning electron microscope (SEM), it is possible to prepare TEM samples in high-precision at a high speed. In addition, since the Electron backscatter diffraction (EBSD) is also installed in this system, it is possible to obtain the crystal orientation distribution of the sample surface by analysing the Kikuchi patterns which are generated by inelastic scattered electrons when electrons are incident on the material. Specifications of FIB are summarized in Table 5.

TABLE 5. SPECIFICATIONS OF FIB

Item	Specifications
Current	Max. 60 nA
Resolution	10 nm
Beam current	0.5 pA to 60 nA
Accelerating voltage	1-30 kV
Accelerating voltage	30 times (observation) 100-300 000 times (processing and observation)
Ion source	Gallium liquid metal ion source
Maximum processing range	1.28 mm x 0.96 mm
Irradiation time	0.4 μs/point-400 ms/point

(3) XPS;

Using the XPS equipment, it is possible to clarify the elemental analysis and chemical states of the sample surface by analysing the emitted photoelectrons from an X-ray irradiated sample. This is useful for the assessment of the oxide layer composition in the Stress corrosion cracking (SCC) and/or analysis of the film formed on the surface of reactor materials. Measured photoelectrons are very near surface (at few nm) of the sample; however it is possible to carry out the high accurate analysis in depth direction (about ~ 1μm) in combination with Ar sputtering. Moreover, the surface analysis is possible at small area (max. diameter 7.5 μm) by the focused X-ray beam, and also possible maximum at 1.4 mm × 1.4 mm area by scanning of the X-ray beam. Specifications of XPS are summarized in Table 6.

TABLE 6. SPECIFICATIONS OF XPS

Item	Specifications
X-ray source	Monochrome X-ray source
Resolution	Below 0.60 (eV) for X-ray beam diameter 10 μm to 20 μm
Specimen size	70 mm (X) × 70 mm (Y) × 20 mm (Z)
Sample stage	X: -40 mm to 40 mm Y: -40 mm to 40 mm Z: -25 mm to 25 mm Inclination: -45° to 45° Rotation: 360°

3.3. SIMULATOR OF MATERIALS TESTING REACTORS INCLUDING HRD

In June 2010, 'Birth of the nuclear techno-park with the JMTR' was selected as one of projects of the Leading-edge Research Infrastructure Programme by Japanese government. In this project, new irradiation facilities, post irradiation examination facilities, etc. will be installed up to JFY 2013 to build international research and development infrastructure. As a part of this project, a real-time simulator for operating both reactor and irradiation facilities of a materials testing reactor, 'Simulator of materials testing reactors', was developed by (JAEA) with ITOCHU Techno-Solutions Corporation and GSE Power Systems.

It can simulate the behaviour of the materials testing reactor under normal operation condition, anticipated operational transients and accident condition in order to utilize it for a nuclear human resource development and to promote partnership with developing countries which have a plan to introduce nuclear power plant and/or research reactor.

The JMTR of the JAEA had been stopped the reactor operation from August 2006, and then the refurbishment works were started in order to ensure safety and increase operating efficiency at the beginning of JFY 2007. In the refurbishment works, especially, the instrumentation and control system were completely upgraded. Therefore, the simulator is designed basically based on the refurbished JMTR, and treats systems/facilities of the JMTR, e.g. reactor core, primary cooling system, secondary cooling system, emergency cooling system, instrumentation and control systems, safety protection system and electrical system. The simulation model is developed by using the newest and various optimum techniques, codes and tools for real-time simulation with high precision. Furthermore, the simulation model can be modified for future demand by a developed support tool. The simulator had been designed in JFY 2010, fabricated from JFY 2011, and was successfully completed on May 2012 [8].

From JFY 2012, two courses will be held, and 20 trainees will be accepted in each course in the same way as JFY 2011. From JFY 2012, training of simulated operation of nuclear reactor will start using a simulator of irradiation test reactors [9]. The simulator was designed and fabricated based on the JMTR, and it can simulate operation, irradiation tests and various kinds of accidents in an irradiation test reactor. The hardware construction of the simulator is shown in Fig. 20.

FIG. 20. Overview of the simulator.

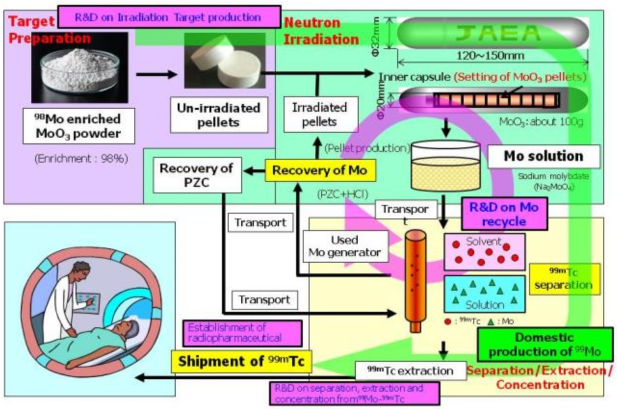


FIG. 21. Flow chart for the domestic production of ^{99}Mo-^{99m}Tc in JMTR.

4. RECENT ACHIEVEMENT

As one of effective uses of the JMTR, JAEA has a plan to produce ^{99}Mo by (n, γ) method ((n, γ)^{99}Mo production), a parent nuclide of ^{99m}Tc. ^{99m}Tc is most commonly used as a radiopharmaceutical in the field of nuclear medicine. In case of Japan, the supplying of ^{99}Mo depends only on imports from foreign countries. Therefore, the (n, γ) ^{99}Mo production was adopted from viewpoints of safety, nuclear proliferation resistance and waste management in JMTR. Advantages of JMTR are high neutron flux and direct connection to hot laboratory for RI productions. The (n, γ)^{99}Mo has several advantages compared to the fission Mo, but the extremely low specific activity makes its uses less convenient than the fission Mo. Figure 21 shows flow chart for the domestic production of ^{99}Mo-^{99m}Tc in JMTR. The R&D has been carried out with foreign organizations and relevant Japanese manufacturers under the cooperation programmes and original R&D promotion programme in Japan. The main R&D items for the (n, γ) ^{99}Mo production are as follows:

(1) Fabrication development of irradiation target as the high-density MoO_3 pellets;
(2) Separation and concentration development of ^{99m}Tc by the solvent extraction from Mo solution;
(3) Examination of ^{99m}Tc solution for a medicine; and
(4) Mo recycling development from Mo generator and solution.

Especially, the study of fabrication of high-density MoO_3 pellets is giving good results since it was started. As before, MoO_3 pellets produced by the Hot press were not able to obtain in high density owing to sintering temperature of the MoO_3 is at above 700°C whereas low sublimation point (700°C). Therefore, the plasma sintering methods were selected because of lower sintering temperature with less time consumption for the production of high density pellets. Then, prototypes of high-density MoO_3 pellets were tested taking account of 'Mass production of pellets' and 'Pellets solution for workability in hot laboratory'. As a result, production technique of MoO_3 pellets was developed by plasma sintering method, and the method would be applicable to fabricate the high-density MoO_3 pellets.

The restart of JMTR will be achieved considering the safety as well as stable operation, then the in-pile test for ^{99}Mo production by the (n, γ) method will be carried out aiming at the domestic production of ^{99}Mo to realize so called 'Life-innovation' safety and security of national health. This programme is carried out under the Strategic Promotion Programme for Basic Nuclear Research by the Ministry of Education, Culture, Sport, Science and Technology of Japan (MEXT).

5. STATUS AND FUTURE

At the end of the JFY 2010 on March 11, 'the off the Pacific coast of Tohoku Earthquake and Tsunami' and the subsequence accident of Fukushima Dai-ichi NPPs occurred. The JMTR was damaged by the earthquake. After the earthquake, integrity of the reactor building and equipment instrument was confirmed by equipment inspections and seismic response analysis. The repair work on places, such as cracks in concrete that were identified by the inspections has been completed.

However, the gigantic earthquake and tsunami and the accident of Fukushima Dai-ichi NPPs made a serious impact for both commercial use and research and development of nuclear power in Japan. Regulatory system was revamped, and the Nuclear Regulation Authority (NRA) was established in September 2012. In 2013, NRA determined new regulatory standards both for commercial plants and testing reactors. JAEA needs to certify that JMTR is conforming to the new standard. Although the seismic influence evaluation report was submitted to the NRA, timing of the restart of the JMTR is not clear.

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

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Profile 12

THE EXPERIMENTAL FAST REACTOR JOYO[1]

JAPAN

1. GENERAL INFORMATION

The experimental fast reactor JOYO of the Japan Atomic Energy Agency (JAEA) is the first sodium-cooled fast reactor (SFR) in Japan. JOYO attained its initial criticality as a breeder core (MK-I core) in 1977. During the MK-I operation, which consisted of two 50 MWt and six 75 MWt duty cycles, the basic characteristics of plutonium (Pu) and uranium (U) mixed oxide (MOX) fuel core and sodium cooling system were investigated and the breeding performance was verified. In 1983, the reactor increased its thermal output up to 100 MWt in order to start the irradiation tests of fuels and materials to be used mainly for other SFRs. Thirty-five duty cycle operations and many irradiation tests were successfully carried out using the MK-II core by 2000.

The core was then modified to the MK-III core in 2003 [1, 2]. In order to obtain higher fast neutron flux, the core was modified from one region core to two region core with different Pu fissile contents. Accordingly, the reactor power increased up to 140 MWt together with a renewal of intermediate heat exchangers (IHXs) and dump heat exchangers (DHXs). The rated power operation of the MK-III core started in 2004. The MK-III core has been used for the irradiation tests of fuels and materials for future SFRs and other R&D fields like innovative nuclear energy systems and technologies as well. This powerful neutron irradiation flux has an advantage especially for high burn-up fuel irradiation and material irradiation with high neutron dose.

This paper shows the outline of the irradiation capabilities and capacities to develop innovative nuclear energy systems and technologies.

2. EXPERIMENTAL FACILITIES IN JOYO

2.1. GENERAL DESCRIPTION OF JOYO AND RELATED FACILITIES

2.1.1. Core

Figure 1 shows the configuration of the MK-III third operational cycle core in 2006 as an example. The fuel region is divided into two radial enrichment zones to flatten the neutron flux distribution.

The MK-III driver fuel is a MOX fuel. The fissile Pu content ((^{239}Pu + ^{241}Pu) / (U + Pu)) is about 16 wt% in the inner core fuel and about 21 wt% in the outer core fuel. The fuel region is surrounded by a radial stainless steel reflector region, which is about 25 cm thick. Shielding subassemblies including B_4C are loaded in the outer two rows of the reactor grid.

The main parameters of the MK-III core are shown in Table 1.

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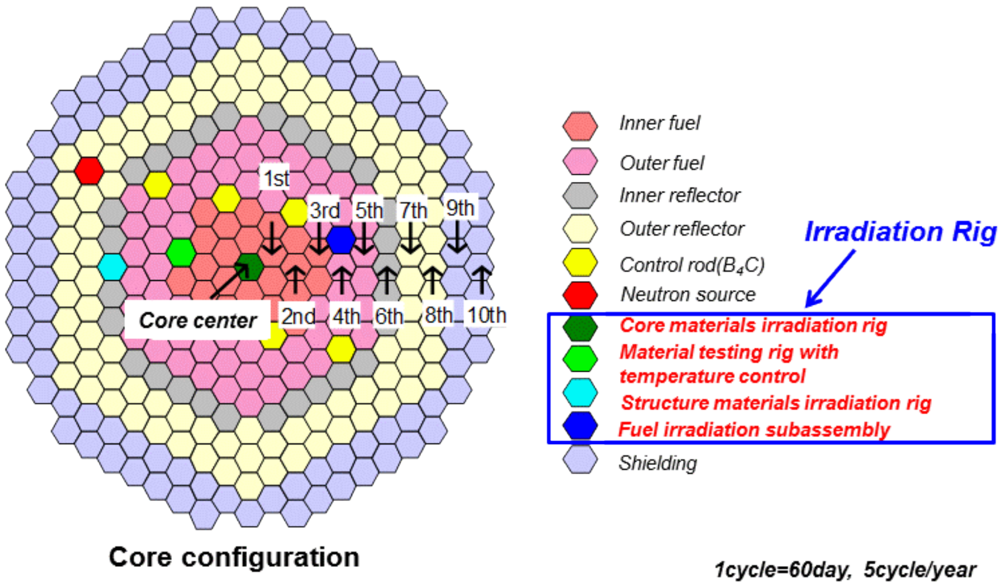


	Flux (n/cm²·s)						
	Core center	1st row	2nd row	3rd row	4th row	5th row	6th row
E≧0.1MeV	3.9×10^{15}	4.0×10^{15}	3.8×10^{15}	3.4×10^{15}	2.9×10^{15}	2.2×10^{15}	1.4×10^{15}
E≧0.414eV	3.9×10^{6}	7.6×10^{6}	8.5×10^{6}	6.6×10^{6}	7.2×10^{6}	2.7×10^{7}	6.2×10^{9}
Total	5.6×10^{15}	5.6×10^{15}	5.4×10^{15}	4.7×10^{15}	4.1×10^{15}	3.2×10^{15}	2.3×10^{15}
dpa/year(Fe)	45	49	47	42	36	27	15

FIG. 1. Example of JOYO core configuration (MK-III third operational cycle).

TABLE 1. MAIN CORE AND PLANT PARAMETERS OF JOYO

Items	MK-III core
Reactor thermal output	140 MWt
Max. number of irradiation test assembly	21
Core diameter	~ 80 cm
Core height	50 cm
^{235}U enrichment	~ 18 wt%
Pu content	≤ 30
Pu fissile content (inner/outer core)	~ 16/21 wt%
Neutron flux total	5.7×10^{15} n·cm^{-2}·s^{-1}
Neutron flux fast (E > 0.1 MeV)	4.0×10^{15} n·cm^{-2}·s^{-1}
Primary coolant temperature (inlet/outlet)	350-500°C
Operation period	60 day/cycle
Reflector/shielding	SUS/B_4C

2.1.2. Cooling system

JOYO has two primary loops, two secondary loops and an auxiliary cooling system as shown in Fig. 2.

Approximately 200 tons of sodium is charged for the cooling system. In the MK-III core, the sodium enters into the core at 350°C at a flow rate of 1350 $t{\cdot}h^{-1}{\cdot}loop^{-1}$ and exits from the reactor vessel at 500°C.

An intermediate heat exchanger (IHX) separates radioactive sodium in the primary system from non-radioactive sodium in the secondary system. The secondary sodium loops transport the reactor heat from the IHXs to the air-cooled dump heat exchangers (DHXs).

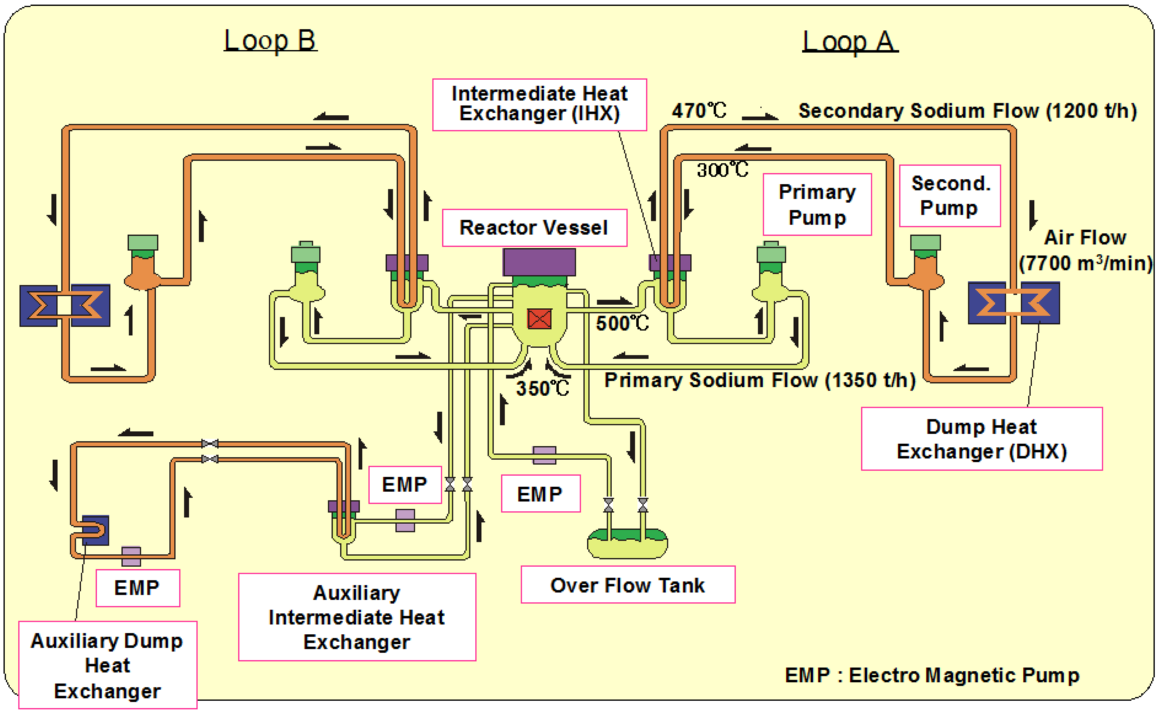


FIG. 2. JOYO MK-III heat transport system.

2.2. TESTING COMPONENTS OF REACTOR CORE

Basically, irradiation tests are carried out with irradiation rigs in the core and reflector region of JOYO.

2.2.1. Fuel irradiation test

Major fuel irradiation tests are performed by off-line irradiation rigs as shown in Fig. 3. In JOYO, four types of fuel irradiation subassemblies are prepared to conduct fuel irradiation tests as follows:

- Type-A: a few experimental fuel pins are placed within a small duct (flow channel) in the centre of JOYO driver fuel pin bundle;
- Type-B: experimental fuel pins are loaded to six compartments which enable to set flow conditions respectively. This type of rig is also suitable for reassembling in the hot-cell and reloading to the reactor, after interim examination of fuel pins;
- Type-C: experimental fuel pin-bundle is wrapped by double-duct to demonstrate the

MOX fuel subassembly design and investigate the irradiation behaviour as MOX fuel pin bundle. The size of inner duct is adjusted to an optional dimension of the pin diameter and pitch;

— Type-D: compartment type rig which improved on the Type-B design. Up to 18 compartments can be placed in this rig. Each compartment holds one single test fuel pin and forms insulated channels of coolant flow enabling to set each individual pin cladding temperature parameter.

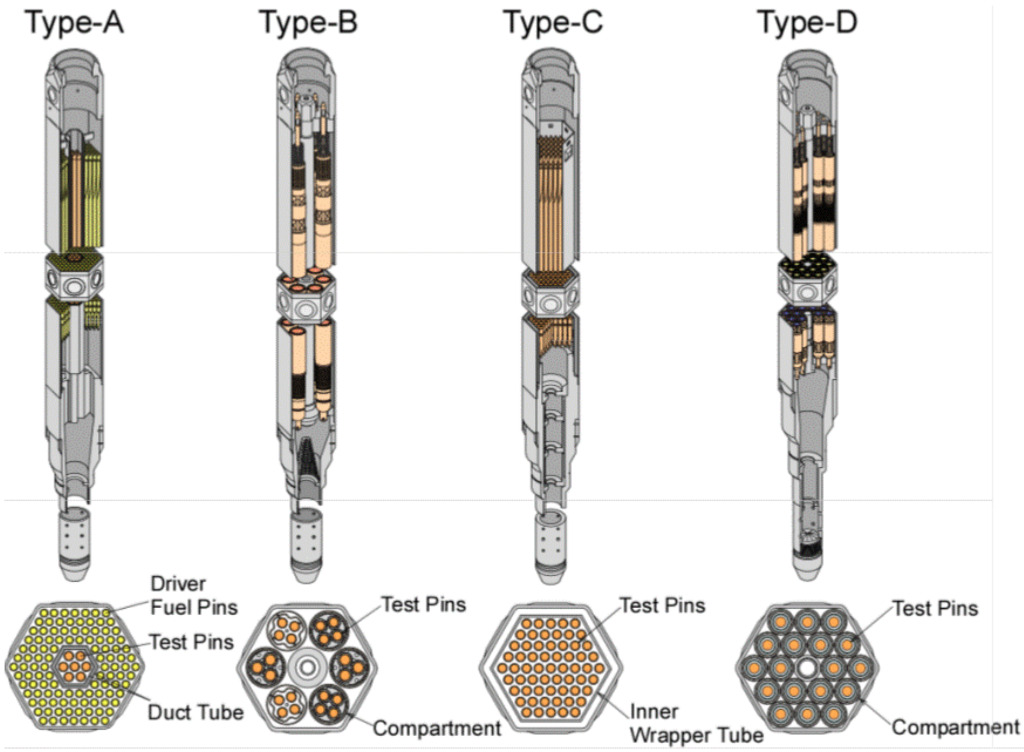


FIG. 3. Fuel irradiation rigs lineup in JOYO.

All fuel test pins are irradiated in the sodium environment. Although there is no special in-pile testing loop in JOYO, in-core tests such as the power-to-melt (PTM) test for MOX fuel simulating transient overpower and the run-to-cladding- breach (RTCB) test are licensed in the Type-B rig. The maximum melt fraction in the PTM test under normal operation is 20%. And maximum burn up in RTCB test license is 200 GWd/t. RTCB test can be carried out using the existent system of fuel failure detection (FFD) and primary coolant cleaning system.

Besides the capsule-type rig is available for irradiation tests of various fuel forms including oxides, carbides, nitrides, metals and those bearing minor actinides (MAs) and FPs. A test fuel pin is installed to a capsule which can withstand the pressure increase in the event of fuel failure. The features, license and structure of capsule type irradiation rig are shown in Fig. 4.

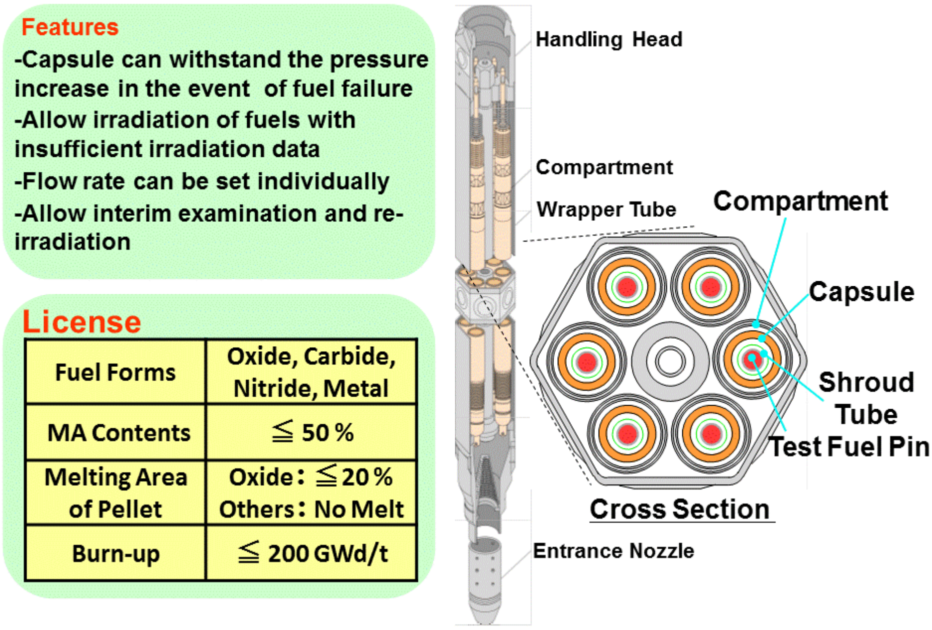


FIG. 4. Capsule type fuel irradiation rig.

2.2.2. Material irradiation test

A test material is installed at unsealed holder or sealed capsule. The test material in holder is immersed in sodium. The material irradiation capsule is filled with sodium or inert gas (helium and/or argon). The structure materials irradiation test rigs are shown in Fig.5.

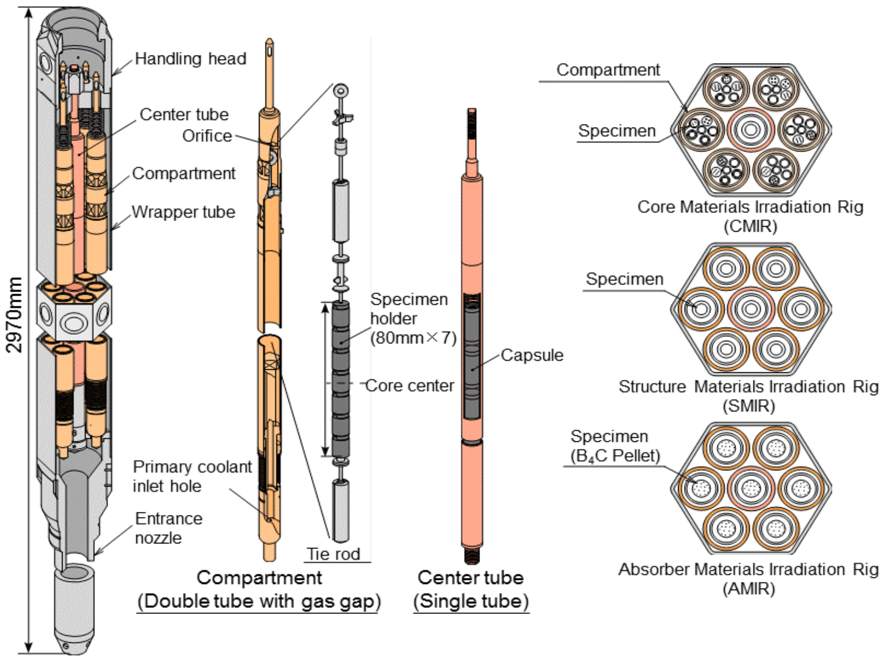


FIG. 5. Materials irradiation test rig.

2.3. EXPERIMENTAL FACILITIES FOR IN-PILE INVESTIGATION OF ACCIDENTAL CONDITIONS

Currently there is no in-pile investigation facilities and installation planning.

2.4. LOOPS FOR IN-PILE INVESTIGATION OF CORROSION RESISTANCE OF REACTOR MATERIALS

Currently there is no in-pile investigation facilities and installation planning.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES etc.

Dose rate is 3–10 dpa/cycle (1cycle = 60 EFPD) in the core fuel region. And wide range temperature conditions can be set in a test rig according to required conditions.

2.5.1. Condition with sodium bonded irradiation

The primary coolant sodium enters into the core at 350°C and exits from the reactor vessel at 500°C and the coolant temperature in an irradiation rig can be raised up to about 750°C by regulating coolant flow rate. Therefore, temperature of irradiated material specimen can be set 350-750°C. Peak cladding temperature of test fuel pin can be set 500-750°C in standard design.

In addition, lowered temperature operation mode is licensed. In this mode, the reactor inlet temperature can be lower than 300°C. Assuming that the inlet temperature is 290°C, a fast neutron flux of about 2×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$ can be obtained at 370 °C (specimen temperature).

2.5.2. Condition with gas bonded irradiation

Material irradiation capsule is insulated by inert gas and specimens are heated by gamma ray heating. Consequently, material specimen temperatures in gas condition are higher than those in sodium bonded irradiation.

Besides a high-temperature irradiation capsule was developed. That is equipped with a tungsten inner tube to produce higher gamma ray heating. In the preliminary temperature calculation, capsule design with a higher temperature than 1000°C is feasible as shown in Fig. 6.

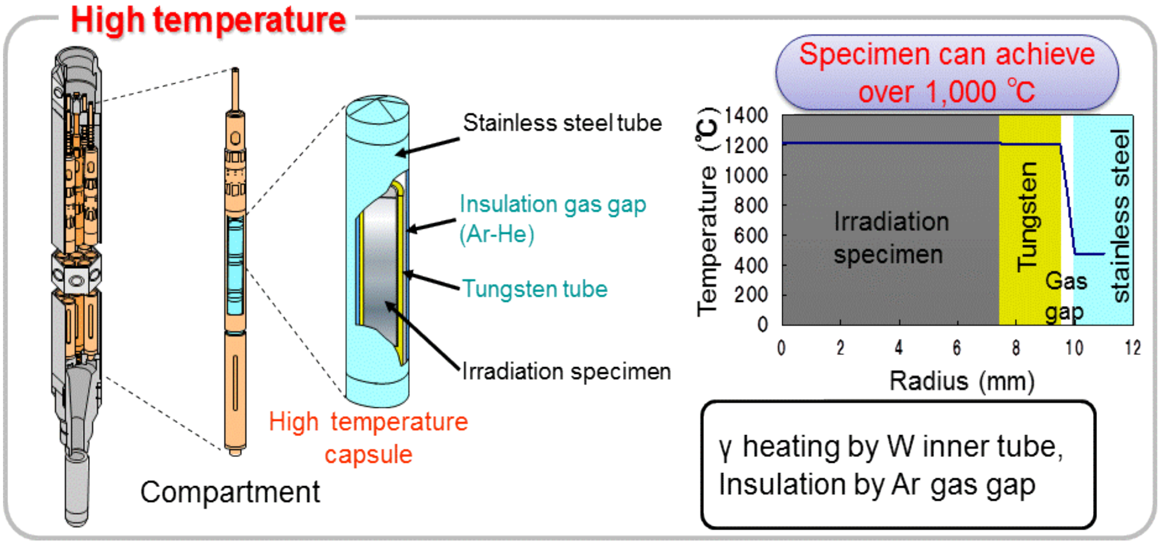
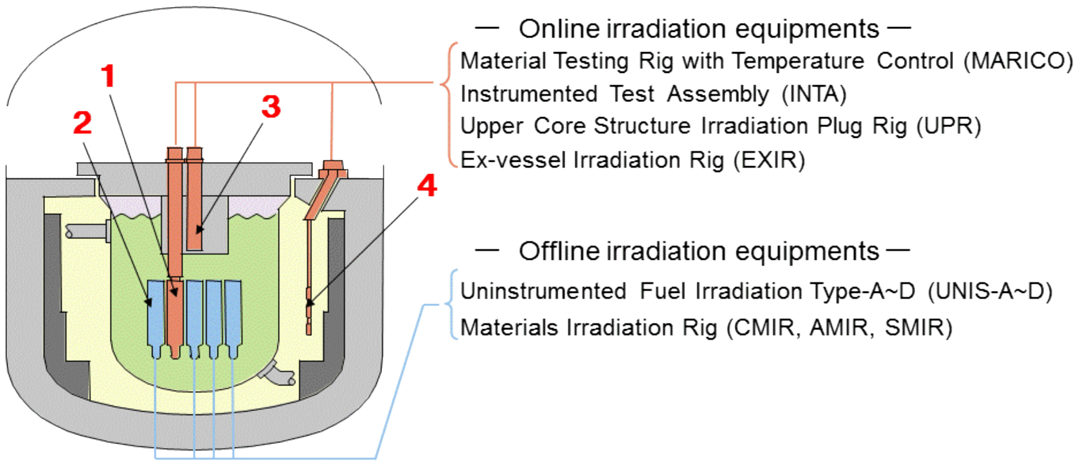


FIG. 6. Preliminary temperature calculation of high temperature irradiation capsule.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS

Most of irradiation tests are carried out in an off-line irradiation rig such as a capsule type irradiation rig and a material irradiation rig described in Section 2.2. On the other hand, JOYO has on-line irradiation techniques. Outline of irradiation devices is shown in Fig. 7.



	Region	Temp. Unit:℃	Neutron flux Unit: n/cm²·s		Environment
			Total	Fast (E≧0.1MeV)	
1	Fuel region	400～1200	$(4\sim5)\times10^{15}$	$(3\sim4)\times10^{15}$	Sodium
2	Reflector region	350～1200	$10^{14}\sim3\times10^{15}$	$3\times10^{13}\sim2\times10^{15}$	Sodium, inert gas
3	Upper core region	500～	$10^{11}\sim10^{12}$	$10^{10}\sim10^{11}$	Sodium, inert gas
4	Irradiation hole outside reactor vessel	200～	$\sim10^{12}$	$\sim10^{10}$	Sodium, inert gas, Water

FIG. 7. Irradiation field and devices in JOYO.

2.6.1. Off-line irradiation devices

In an off-line rig, irradiation behaviour and characteristics of materials are investigated in the post irradiation examination (PIE). RTCB test will contribute to investigate the mechanical lifetime of a test fuel pin using the off-line rig.

2.6.2. On-line irradiation devices

Several on-line rigs are available to obtain real-time data.

(1) Instrumented test assembly (INTA);
The INTA is an on-line test rig equipped with thermocouples, gas pressure gauges, etc. As shown in Fig. 8, the INTA-1 and INTA-2 were fuel bundle subassemblies. Additionally the INTA-S was the subassembly for installing material capsules. Instrumentation cables, which extend from the core and penetrate the upper core structure (UCS), can transmit in-pile data.

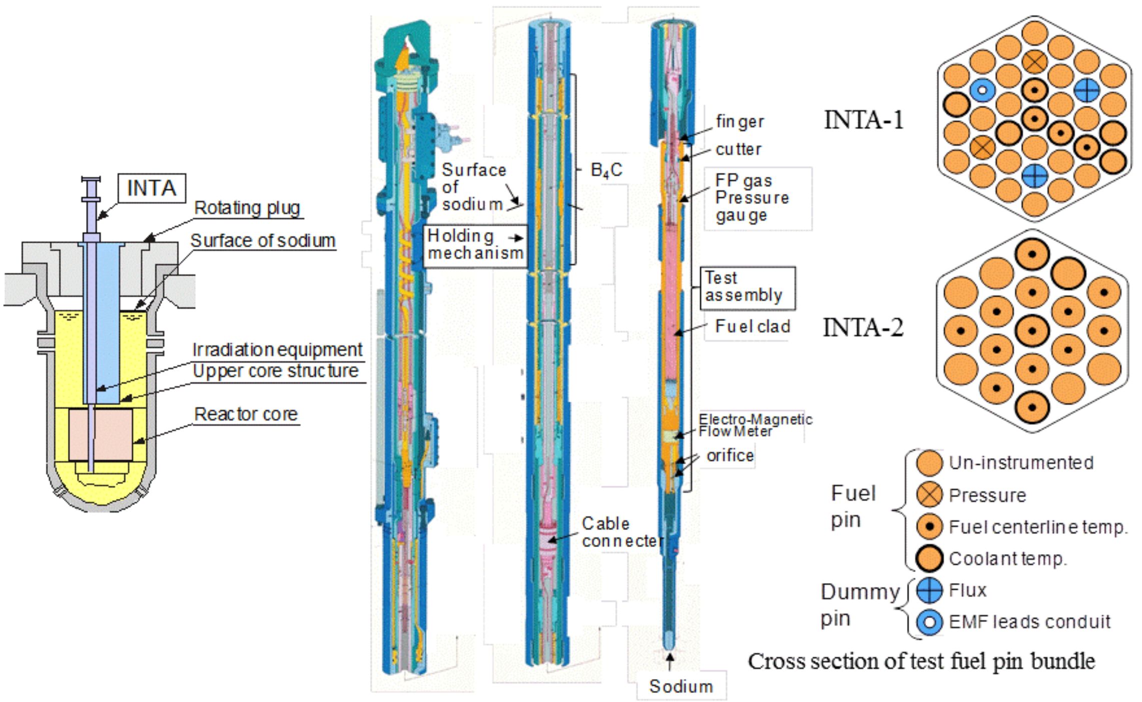


FIG. 8. Instrumented fuel test subassembly (INTA-1and INTA-2).

(2) Material testing rig with temperature control (MARICO);
MARICO was developed for the irradiation test under the controlled temperature within ±4°C by means of changing the mixture gas composition of argon/helium as shown in Fig. 9. The first MARICO-1 was used in the MK-II core to obtain the in-pile data of austenitic cladding materials such as creep rupture strength and swelling under the accurate temperature. MARICO-2 was mainly used for the in-pile creep rupture test

of Oxide Dispersion Strengthened (ODS) ferritic steel cladding in the MK-III core as shown in Fig. 9. Although the main structure is same with the MARICO-1, the MARICO-2 contained an electrical heater capsule for temperature control, a larger irradiation space and a function to reload irradiated specimen in the Fuel monitoring facility.

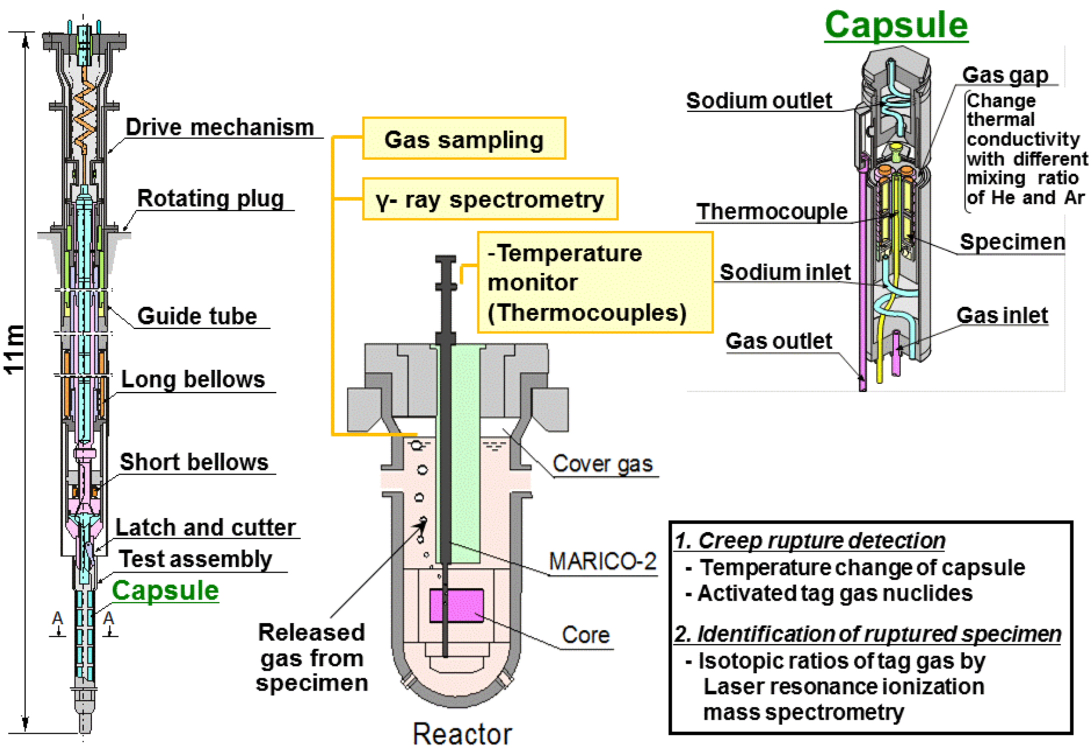


FIG. 9. Material testing rig with temperature control for in-pile creep investigation.

(3) Upper core structure plug rig (UPR);
The UPR is installed at the upper core structure. Although the fast neutron flux is low (10^{10}-10^{11} $n \cdot cm^{-2} \cdot s^{-1}$), the temperature of the specimens can be controlled by an electric heater. In the MK-III core, this rig is used for a test of electromagnet material to measure changes of magnetic characteristics under the irradiation.

(4) Ex-vessel irradiation rig (EXIR);
The EXIR is the test device for reactor structural materials by utilizing the space between the reactor vessel and the safety vessel. Although the fast neutron flux is low (10^{10} $n \cdot cm^{-2} \cdot s^{-1}$), the temperature of the specimens can be controlled by an electric heater. Uniaxial creep test pieces loaded by gas pressurization were irradiated in the EXIR.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

In the Oarai research and development centre, there are some facilities necessary for PIEs as shown in Figs 10 and 11. The irradiated test subassemblies have been transferred to the fuel monitoring facility (FMF) adjacent to JOYO. The FMF has the large stainless steel-lined, nitrogen-gas-tight hot cells with the remote maintenance systems for conducting PIEs. Additionally, non-destructive PIEs have been conducted using the X-ray computer tomography (CT) device in the FMF. The test subassemblies are disassembled in FMF and the capsules are retrieved which contain several sets of multiple activation foils and helium accumulation fluence monitors (HAFMs) to be used for the neutron dosimetry [3]. The activation foils are finally transferred to the irradiation rig assembling facility (IRAF) adjacent to JOYO, where the gamma-ray measurement system is installed to analyse the reaction rate of the irradiated foils.

The retrieved fuel samples are transferred to the Alpha-gamma facility (AGF) to conduct the destructive PIE to determine the fuel burn-up by means of ^{148}Nd method and to take the metallography of cut cross section of the samples. The AGF was originally established as the PIE facility of the irradiated fuels. Later, a small-scale fuel fabrication unit was equipped in the hot cell to develop a remote fabrication technology for MA-MOX fuel.

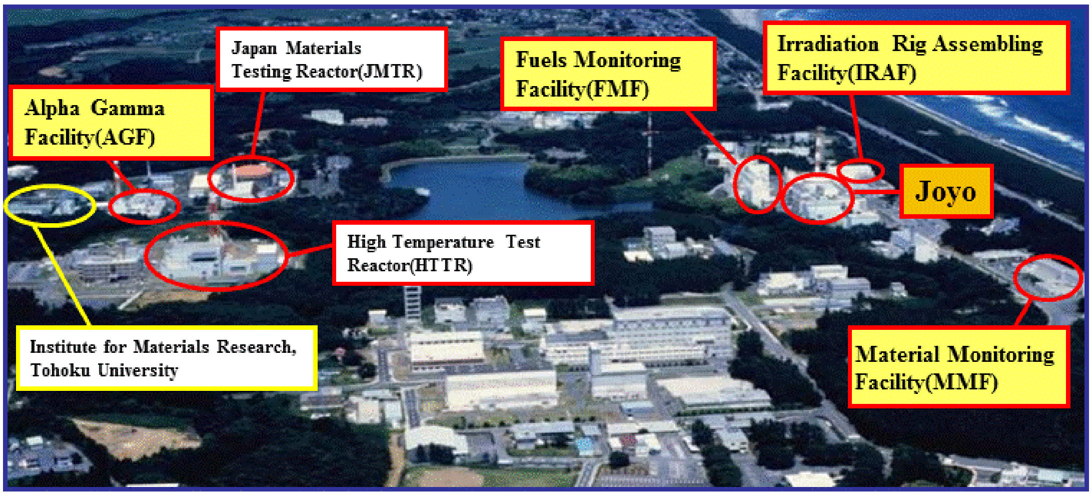


FIG. 10. Reactor and related facilities in the Oarai R&D centre.

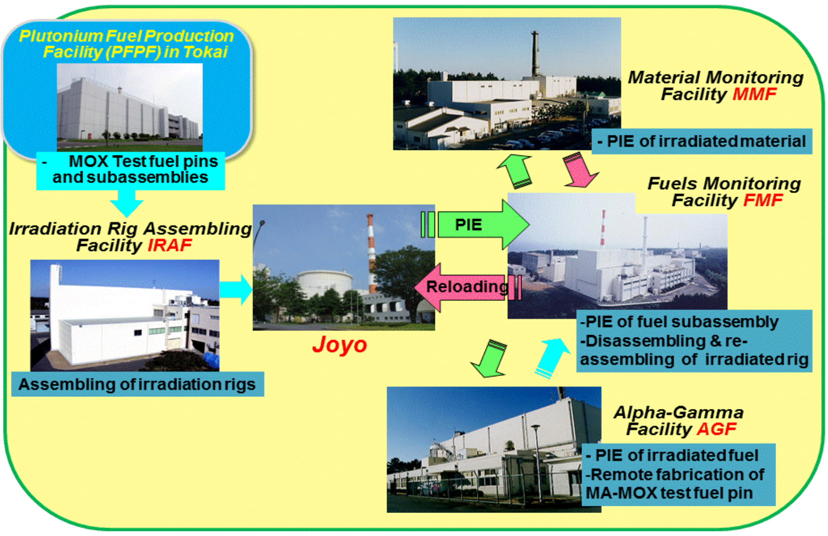


FIG. 11. Test fuel fabrication and hot laboratories related to JOYO.

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

There are two test fuel fabrication facilities related to JOYO in Japan Atomic Energy Agency:

(1) Plutonium fuel development facility (PFDF);
The main supplier of the fresh experimental MOX fuel pins for JOYO irradiation tests is the plutonium fuel development facility (PFDF) in the Tokai research and development centre. This is a glove box facility handling pin scale MOX fuels (including fuels with low MA content).

(2) Alpha-gamma-facility (AGF);
As mentioned later, AGF is a hot-cell facility for the PIE of irradiated fuels in Oarai R&D centre. This facility was later equipped with a small-scale fuel fabrication unit in the hot-cell to develop a remote fuel fabrication technology for fuel containing MAs. Am-MOX fuel pins were fabricated, and those were irradiated in the JOYO MK-III core.

3.2. HOT CELLS, PIE FACILITIES

There are three PIE facilities related to JOYO in Oarai R&D centre:

(1) Fuel monitoring facility (FMF);
The FMF was constructed to examine the fast reactor fuels and materials irradiated in JOYO. There equipped α-γ sealed type hot cell and a β-γ type hot cell in the FMF. The atmosphere of these cells is maintained as high purity nitrogen.

Non-destructive examination or the basic destructive examinations for irradiated fuel subassemblies and fuel pins are conducted in the FMF. For detailed non-destructive inspection, an X-ray computed tomography (CT) device is installed in the FMF. A part of fuel pin segments or material samples are transferred to the AGF or the MMF for further detailed examinations.

The FMF is located adjacent to JOYO, and it is possible to re-install irradiated fuel pins and materials into a new subassembly after the interim examination of irradiated components. This reassembling technique enables to conduct the interim inspection of a part of fuel pins or material samples and also enables the long-term irradiation test up to high fluence or high burn-up beyond the life-time of irradiation rig by renewal the irradiation rig.

(2) Materials monitoring facility (MMF);
PIEs for fuel claddings, wrapper tubes, structural materials, control materials etc. are performed in the MMF. The MMF has several hot-cells such as concrete cells of α-γ sealed type, β-γ type, and lead cells. In the α-γ sealed type, fuels are removed from the segmented fuel pin, and the mechanical tests are conducted such as tensile tests, creep tests and transient burst tests and so on. In the β-γ type cells, a mechanical test, a metallurgy test, dimensional and density measurements can be conducted. The field-emission transmission electron microscopy (FE-TEM) and TEM are equipped to observe the microstructure of irradiated material.

(3) Alpha-gamma facility (AGF);
The main role of this facility is to evaluate the fuel burn-up, physical property,

chemical composition of trans-uranium nuclides in MOX fuel and so on. One characteristic of the AGF is the inner box-type cell, in which movable stainless steel boxes to make air tight are installed in radiation shieldings such as concrete, lead, etc.

PIE services in these hot facilities related to JOYO are shown in Table 2.

TABLE 2. PIE SERVICES

Facility	FMF	MMF	AGF
PIE item	– Visual inspections – Detailed visual inspections of the fuel pin – Profilometry of the subassembly – Weight measurement of the fuel pin – Eddy current inspection – Profilometry of the fuel pin – γ-scanning of the fuel pin – Fuel pin puncture – X-ray radiography – X-ray CT test – Optical microscope – SEM observation – FE- SEM observation	– Tensile test – Transient burst test of cladding – Density measurement – Charpy impact test – Magnetization measurement – Uniaxial creep test – Optical microscope – EF/TEM observation – Gas analysis – TEM observation – Hardness test – Thermal conductivity measurement – Thermal expansion coefficient measurement – X-ray diffraction analysis	– Optical microscope – Melting point measurement – FP release examination – O/M ratio measurement – X-ray diffraction analysis – EPMA analysis – ICP emission spectrometric analysis – Halogen analysis – Moisture analysis – MA analysis – Burn-up measurement – ICP-MS
Other functions	– Sodium removal equipment – Disassembling of the subassembly – Sample preparation to transport to the MMF or AGF	– Manufacture of specimen from irradiated steel parts – Gas enclosing (only for non-irradiated specimens)	– Remote TRU fuel pellet production and inspection – Remote fuel pin welding and inspection

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS

Irradiation test rigs are assembled in the irradiation rig assembling facility (IRAF). A lot of knowledge and experience of design are accumulated. And a small workshop for parts of material irradiation capsule is placed in the IRAF.

4. RECENT ACHIEVEMENTS

Operational histry of JOYO is shown in Fig. 12.

During the operation with the MK-II and MK-III core, a large number of fuel and material irradiation tests were conducted. Five on-line irradiation test rigs (two for fuel irradiation and three for material irradiation) and 84 off-line rigs (28 for fuel irradiation and 56 for material irradiation) were irradiated and PIEs were carried out in the hot cell facilities. Concerning the fuel irradiation, the highest burn-up of 144 MWd/kg was achieved by the irradiation rig named C4F, which was designed and fabricated in the collaboration with CEA. Other tests included power to melt tests (PTM) of MOX fuel, performance tests of failed fuel detection system (FFD) and failed fuel detection and location system (FFDL) and natural circulation tests were also conducted.

Irradiation tests for cladding material were conducted by using off-line and on-line compartment type test rigs. For structure, absorber and shielding material irradiration tests, several off-line compartment type test rigs were loaded in the core and radial reflector region. By using those rigs, over 40 000 pieces of material specimen, which were proposed by universities, have been also irradiated.

In 2004, JOYO started the operation with the MK-III core. Several irradiation tests of fuel and material were successfully conducted. Recent R&D activities including irradiation test experience and enhancement of irradiation test capability are listed with reference as follows:

(1) Irradiation fields characteristic test;
Irradiation field of JOYO were characterized by calculations and experiments of which reliability were evaluated. This paper described the detailed characterization of the neutron irradiation field including neutron energy spectrum in JOYO MK-III Core [4].

(2) Fuel failure simulation test;
Performance of JOYO's FFD & FFDL system in the case of fuel failure were required to confirm after the plant modification in 2003. Therefore a fuel failure simulation test with MK-III core was carried out [5].

(3) Test for self-actuated shutdown system;
Self-actuated shutdown system (SASS) with a curie point electromagnet has been developed for use in a future SFR in order to establish the passive shutdown capability. As the final stage of the development, the stability of SASS needs to be confirmed under the actual reactor-operational environment. For this purpose, the demonstration test of holding stability using the reduced-scale experimental equipment of SASS was conducted in JOYO [6].

(4) Short term irradiation test for MA-MOX fuel;
Two short-term irradiation experiments for MA-MOX fuel were conducted in JOYO. Six fuel pins included MOX fuel containing 3% or 5% americium (Am-MOX), MOX fuel containing 2% americium and 2% neptunium (Np/Am-MOX), and reference MOX fuel were prepared. The first test was 10 minutes test in order to confirm whether fuel melting occurred. After 10 minutes test, an Am-MOX pin and Np/Am-MOX pin were discharged to conduct the PIE. The remaining four fuel pins were re-irradiated for 24-hours to obtain MA re-distribution data. The test results are expected to reduce uncertainties in the thermal design for MA-MOX fuel [7, 8, 9].

(5) In-pile creep rupture test of ODS ferritic steel;
ODS ferritic steel has been developed as a most promising fuel cladding materials for the next generation fast reactors. An in-pile creep rupture experiment was conducted in JOYO to evaluate the creep rupture strength of ODS ferritic steel under neutron irradiation. The measured strength will contribute to the development of the ODS cladding tube for SFR and so on [10].

(6) Development of a high resolution X-ray CT technique for irradiated fuel pellets;
A non-destructive method making use of X-ray computer tomography (X-ray CT) has

been applied to PIEs of fast reactor fuel subassemblies. In the X-ray CT system, a 12 MeV X-ray pulse was used in synchronization with switching on the detector to minimize the effects of gamma ray emissions from the irradiated fuel subassemblies. This technique enables a clear cross section CT image of the irradiated fuel subassembly to be obtained [11].

(7) Recent R&D to improve irradiation test capability;

For the multi-purpose utilization, the following concepts to enhance irradiation capabilities of JOYO are considered.

(a) Computerized reactor control system by automatic control rod operation;

(b) Neutron spectrum tailoring for creating versatile irradiation field;

(c) Expansion of temperature range by lowering inlet coolant temperature and high temperature irradiation capsule;

(d) Flexible transient experiment using sample movable device;

(e) Fast neutron beam hole for multi-purpose utilization;

The licenses for (a), (b), (c) have already been permitted, and the conceptual design of all item are completed [12].

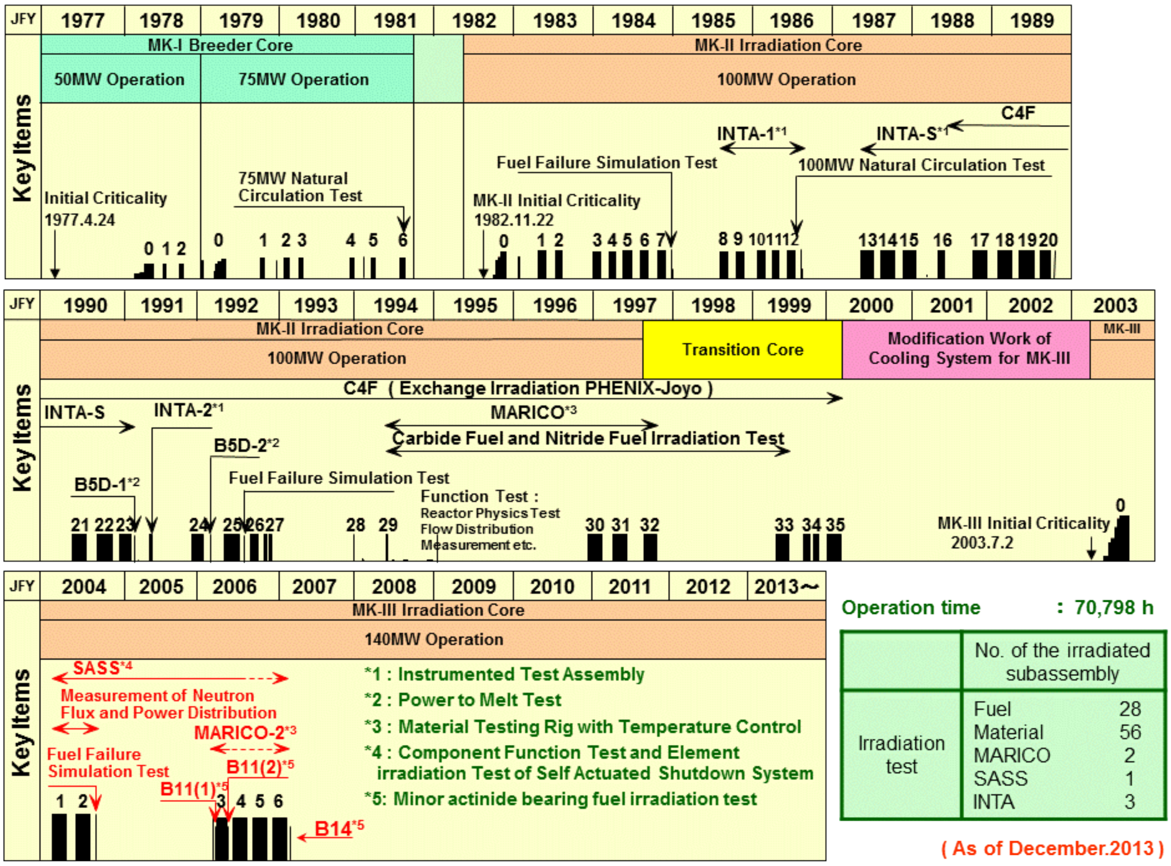


FIG. 12. Operational history of JOYO.

5. POTENTIAL OF JOYO FOR THE USE OF INNOVATIVE NUCLEAR ENERGY SYSTEMS AND TECHNOLOGIES

As explained in the former chapters, JOYO has a large potential for not only the SFR development but also a variety of users ranging from fusion and other concepts of next generation reactors (Gen-IV) as well.

JOYO is the only experimental reactor in Japan, which has licenses to conduct the open core PTM or RTCB test and to irradiate fuel pins of which physical or material property is not well known. The irradiation temperature is now able to extend for both lower and higher values from the normal operation temperatures of SFR. Furthermore, the highest neutron flux level in the world as much as 4.0×10^{15} $n \cdot cm^{-2}$ s^{-1} provides the accelerated irradiation to meet the requests of many users.

Currently JOYO has not been operating due to the incident, which occurred in 2007 [4]. However, the resumption work has started and it is planned to complete in 2014.

JOYO is expected to be used as a powerful irradiation facility after the restart.

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Profile 13

IGR

KAZAKHSTAN

1. GENERAL INFORMATION AND TECHNICAL DATA OF IMPULSE GRAPHITE REACTOR RESEARCH REACTOR

1.1. GENERAL INFORMATION

Information of the impulse graphite reactor (IGR) is as follows:
- On thermal neutrons;
- Homogeneous uranium-graphite core;
- On heat-capacity (has no coolant system);
- Thermal power up to 10000 MW (10 GW) at non-controlled pulse mode;
- Thermal power up to 1000 MW (1 GW) at controlled mode.

Impulse graphite reactor IGR was created in 1960 year to research fast passing processes in the nuclear reactors and to obtain short-time, but rather high, integral fluxes of neutron and gamma radiation to irradiate the tested objects. Later on the reactor was used to conduct dynamic tests of the fuel pins and fuel assemblies (FA) according to the development of start-up modes of the land prototypes of nuclear rocket engine (NRE) and nuclear facilities safety, including conduction of the experiments on destructing and melting of fuel and design materials.

Experimental studies and tests have been conducted and are conducted on IGR reactor, almost always associated with obtaining experimental information about fast physical and thermal processes in nuclear reactors, test objects efficiency in normal and emergency operating conditions, the behavior of fuel and construction materials for safety confirmation of systems and components of nuclear power plants.

The IGR reactor is Impulse Research Reactor on thermal neutrons with homogeneous uranium-graphite core (see Figs 1 and 2). IGR reactor is a self-quenching reactor by a principle of shutdown of any impulse.

1.2. TECHNICAL DATA

Structurally the reactor is a pile from graphite blocks jointed in columns (see Figs 1 and 2, position 3), which placed in waterproof steel cylindrical casing (position 1, hereafter referred as vessel) with helium medium. Total mass of graphite is 33870 kg. Cylindrical thermal shields are installed between the graphite pile and reactor vessel (see Figs 1 and 2, position 2). The reactor vessel is placed inside the tank with cooling water (see Figs 1 and 2, position 12).

Graphite reactor core blocks (see Figs 1 and 2, position 4) are saturated with uranium salts with concentration 3.1 grams of uranium for one kilogram of graphite with enrichment on isotope U^{235} — 90%, and total mass U^{235} — 9.0 kg. Graphite is a good moderator and has a high heat capacity, high temperature-proof and thermal shock-resistant, and removes and absorbs the heat from uranium fuel particles. Owing to these graphite features the special core cooling system (coolant loop) is not required. The energy released inside the core during reactor startup is accumulated in graphite and then removed with water which cools reactor vessel.

Reactor core is approximately cubic-shaped, consists of immovable and movable parts:

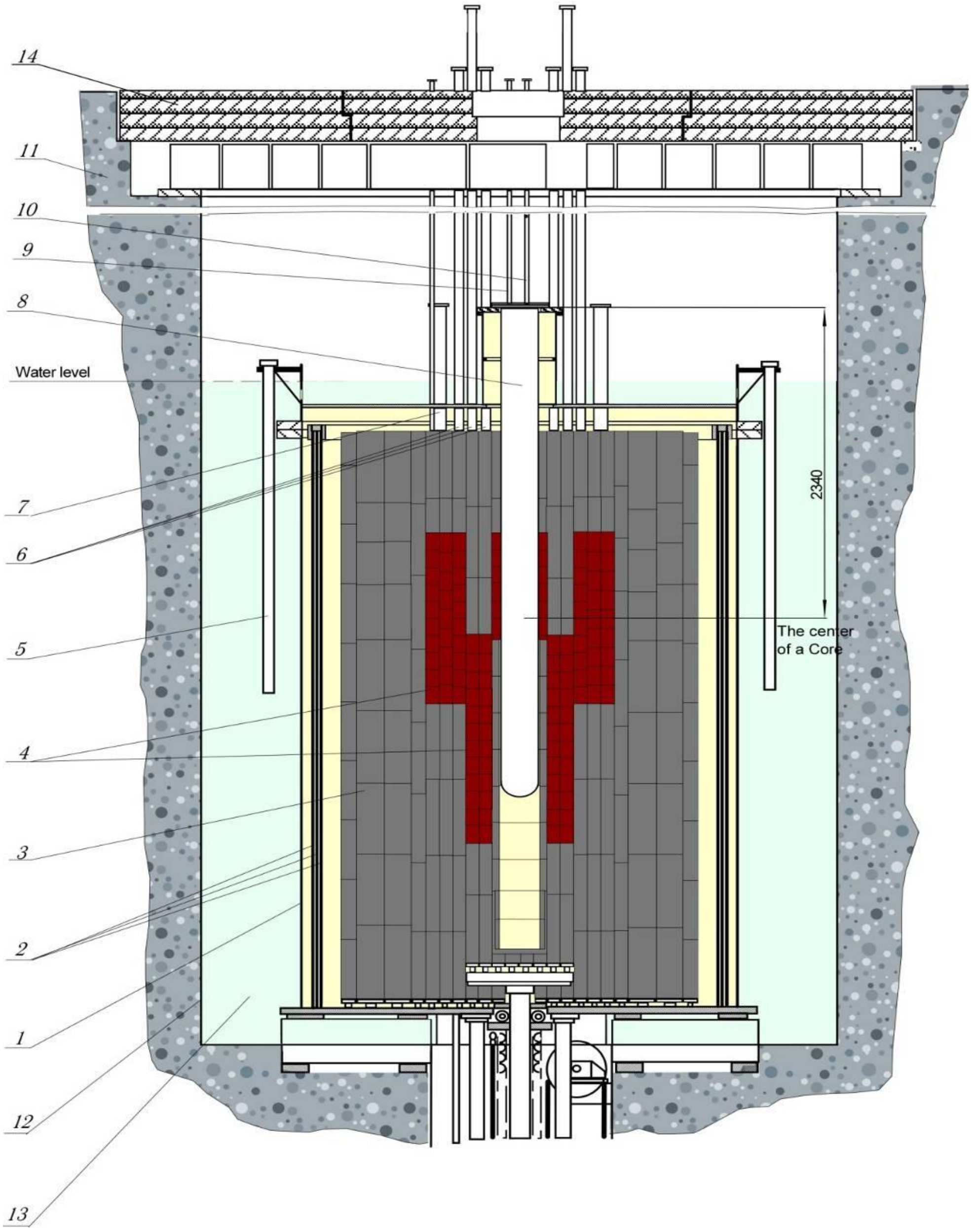
dimensions in cross section are 1400 mm × 1400 mm; height in movable part is 1463 mm; height in immovable part is 1332 mm. Horizontal section of movable part is 800 mm × 800 mm.

Reactor has central experimental channel CEC (see Figs 1 and 2, position 8) and lateral one LEC (see Figs 1 and 2, position 7) equipped with water-cooled experimental devices (immovable ampoules) pressurized gaseous cavity and prevented test objects against the thermal influence of the core. Internal diameter of experimental cavities: CEC — 228 mm and LEC — 82 mm, and the length of: CEC — 3825 mm and LEC — 3440 mm. Axial non-uniformity coefficient of thermal neutron flux is 1.15 (on core height — 1400 mm).

Control and adjustment of reactor power is provided by means of ionization chambers (IC) installed in standpipes on vessel perimeter at the level of core center.

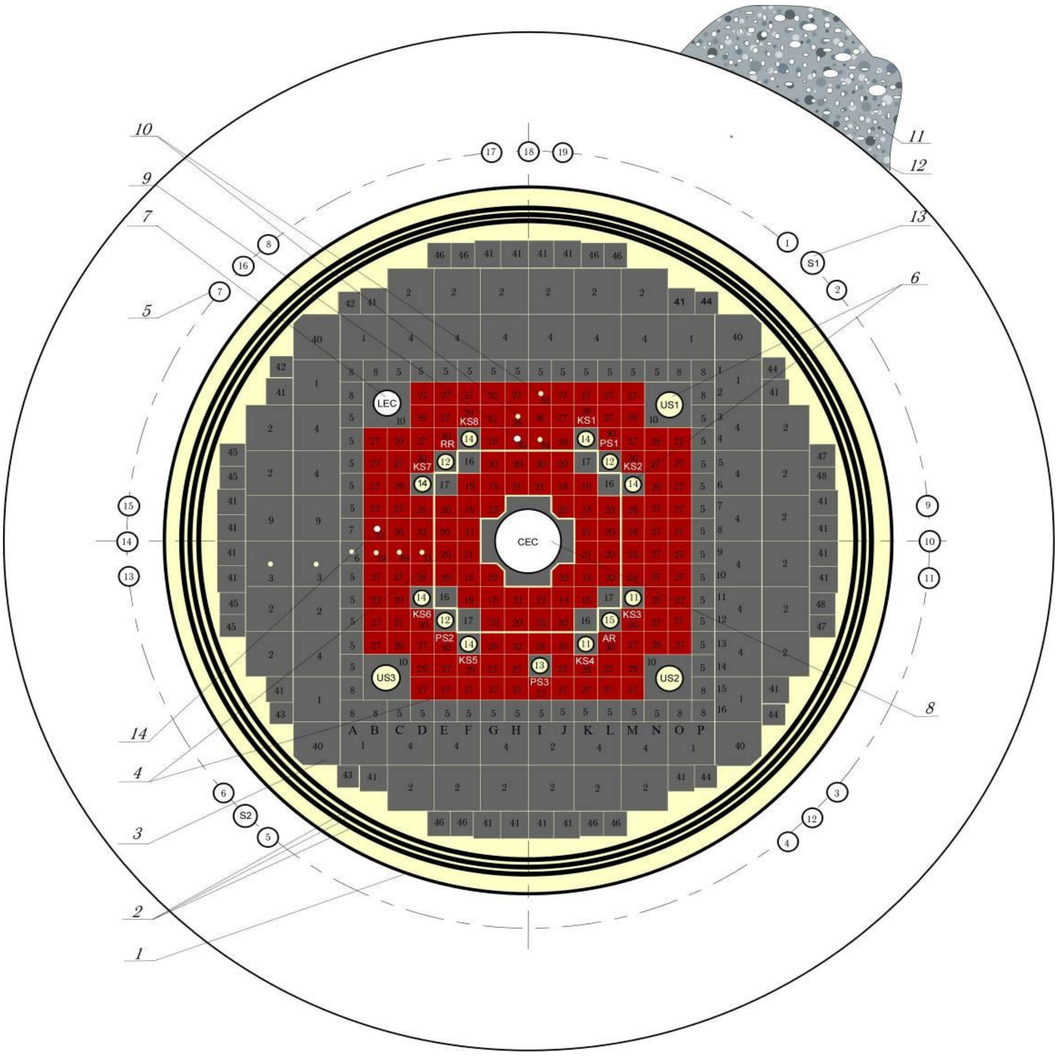
Control rods (16 pcs) of the control and protection system of the reactor with gadolinium oxide neutron absorber are mounted in the immovable part of core (see Figs 1 and 2, position 6).

Vapour-vacuum and water loops of reactor technological system vacuumize and fill gaseous reactor cavity with helium as well as circulate the water in immovable ampoules of experimental central and lateral channels and in tank.



1 —	casing;	8 —	central experimental channel;
2 —	lateral shield (three shells);	9 —	channel for physical measurements;
3 —	reflector;	10 —	channel for thermo-electrical converter;
4 —	core (immovable and movable parts;	11 —	biological shield;
5 —	channel of ionization chamber;	12 —	tank;
6 —	channel of control rods;	13 —	cooling water cavity;
7 —	lateral experimental channel;	14 —	upper cover.

FIG. 1. IGR reactor, vertical section.



1 — casing;
2 — lateral shield (three shells);
3 — reflector;
4 — core (immovable and movable parts;
5 — channel of ionization chamber (19 pcs);
6 — channel of control rods (16 pcs);
7 — lateral experimental channel;
8 — central experimental channel;
9 — channel for physical measurements;
10 — channel for thermo-electrical converter (9 pcs);
11 — biological shield;
12 — tank;
13 — channel of neutron counter (2 pcs);
14 — channel of neutron source.

FIG. 2. IGR reactor, horizontal section.

Main physical and technical data of the IGR are shown in Table 1.

TABLE 1. MAIN PHYSICAL AND TECHNICAL DATA OF THE IGR

Parameter	Value
1. Power at the pulse mode (peak)	10 GW
2. Minimum pulse half-width	0.12 s
3. Max. energy release	5.2 GJ
4. Core temperature	1400 K
5. Max. neutron fluence, thermal/fast	3.7×10^{16} / 1.1×10^{15} n/cm^2
6. Max-density of thermal neutron flux	7×10^{16} n·cm^{-2}·s^{-1}
7. Core fuel	U-saturated graphite
– enrichment	90%
– U-235 loading	9 kg
8. Moderator and reflector	graphite
9. Life period of prompt neutrons	0.965×10^{-3} s
10. Effective fraction of the delayed neutrons	0.00685 β_{eff}
11. Multiplication coefficient	1.275 K_{eff}
12. Temperature coefficient of reactivity	-0.03 β_{eff}/K
13. Experimental cavity, diameter/length	
– central vertical channel	0.228 m/3.825 m
– lateral vertical channel	0.082 m/3.440 m

The main reactor operation modes are non-controlled pulse mode – mode of self-quenching neutron burst (see Fig. 1), and controlled mode (see Fig. 2). The reactor is imposed certain positive reactivity (CPS), which determines the form, amplitude and half-width of burst to realize the mode of self-quenching neutron burst; burst is quenched as a result of reactivity negative temperature effect. The controlled mode is realized by displacing of the CPS working bodies, compensating reactivity negative temperature effect according to the given law. The form, amplitude (power level) and duration of the controlled mode can be various and, they are determined on the basis of test objectives in terms of non-increasing the core temperature 1400 K (operating limit). Therewith energy release in the core is ≈ 5.2 GJ and corresponds to thermal neutron fluence 3.7×10^{16} cm^{-2} in central experimental reactor channel.

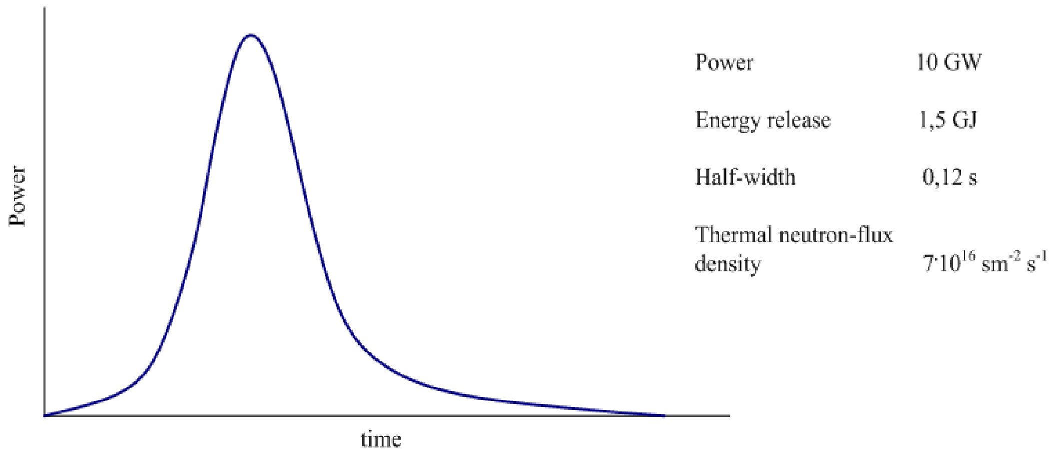


FIG. 3. Non-controlled pulse mode.

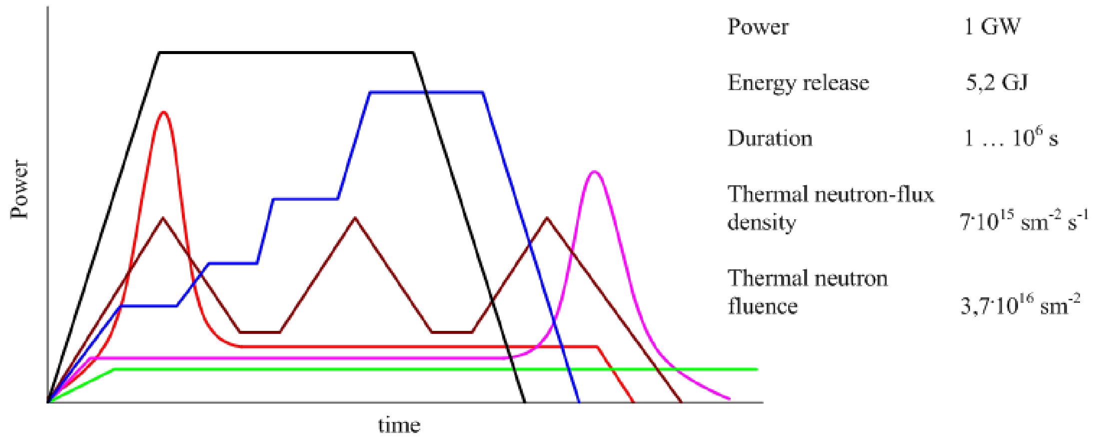


FIG. 4. Controlled mode.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

In the reactor building a recognized procedure to develop fuel and structures supposes using experimental loop stands to provide required neutronic, power and thermal-hydraulic modes in experimental devices. It makes possible to receive data to study efficiency of structural materials for nuclear reactors under conditions much closer to real ones and to solve a lot of test issues under easier, safer and cheaper conditions.

The Pneumohydraulic test bench (PHTB) was constructed at IGR reactor thereto. Initially the pneumohydraulic test bench was made for testing of nuclear rocket engine (NRE) gas-cooled fuel assemblies (FA), and provided performance of dynamic reactor tests of loop FA with hydrogen heating up to 3070 K and specific energy release in fuel of 25 MW/dm^3. At present pneumohydraulic test bench provides performance of loop reactor tests using different coolants — hydrogen, nitrogen, carbon dioxide, helium, water and others during different diagrams of flow, pressure and temperature.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE (fuel, control rods, structural materials, coolant technologies: lead, lead-bismuth, sodium, light and/or heavy water, molten salt, gas)

— At steady state conditions;
— At transient conditions;
— At accident conditions.

In order to realize experiment (test) experimental device with test sample is placed inside the CEC or LEC of IGR reactor and connected with PHS to provide specified technological parameters (coolant flow, pressure, temperature). Experiments (tests) are possible to conduct both in steady-state energy release in the reactor and in transient power state.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA, etc.

Combination of dynamic characteristics of the reactor and Pneumohydraulic test bench provides broad experimental features and test conditions of nuclear equipment at various design and beyond design accidental conditions LOCA, LOFT, RIA, etc.). Simultaneous simulation of different accidental conditions is possible. For example, both decrease in coolant flow rate and delay of emergency protection or accidental entry of positive reactivity due to spontaneous recovery of CPS control rods, either because of the phase state change of the samples when working at nominal power level are simulated.

2.3.1. Pneumohydraulic test bench

The PHTB is an extensive system of constructions, equipment, shut-off and control valves and safety valves, pipelines, control devices with the sites of the open loops of test objects cooling (irradiators), in which the coolant is pumped through the test object one time and is drown from the loop into the discharge (for gases) and discharge (for liquids) hermetic systems.

TABLE 2. THE BASIC SPECIFICATIONS OF PHTB

Parameter	Value
Pressure of the source working environments:	
– hydrogen	35 MPa
– nitrogen	32 MPa
– helium	20 MPa
– distillate	20, 40 MPa
Working environments flow:	
– hydrogen	6 kg/s
– nitrogen	22 kg/s
– helium	4 kg/s
– distillate	25 kg/s
Number of gas paths of high pressure supply (>10 MPa)	11
Number of water paths of high pressure supply (>10 MPa)	7
Capacity of closed discharge system for gas at the pressure of 0.5 MPa	950 m^3
Capacity of closed discharge system for water at the pressure of 0.8 MPa	25 m^3

2.3.2. Pulse pressure control system in in-pile experiments

Parameters of the system provide a measure of static and pulse pressure in corrosive environments (including liquid sodium) under neutron and gamma radiation at a temperature in controlled point to 1000°C. Bandwidth controlled pressure is to 60 MPa when the pressure pulse lasts 5 - 10 ms.

2.3.3. Technological monitoring system

Technological monitoring system (TMS) including facility-based monitoring system ‘National Instruments’ provides measuring and recording various technological parameters of experimental device during experiments.

TABLE 3. THE MAIN TECHNICAL DATA OF THE SYSTEM

Parameters	Value
Number of analogous-digital input channels	160
The frequency range of the measurement channel	≤1 kHz
Reduced error of measurement channel	≤0.1%
The number of channels of tolerance control	24

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

No.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES etc.

Various ampoules are available, providing testing of materials in a wide range of energy (dose), temperature, etc. Hinges for mounting the test object in the ampoule are developed and manufactured under the test task.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (swelling, gas release, creep, long-term strength, relaxation resistance, etc.)

No.

2.7. OTHER FACILITIES

No.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

CRR IGR has storage for fresh and IGR-irradiated nuclear fuel. Uranium dioxide ^{235}U pellets enriched to 17% is used as fresh fuel in experimental devices. The Institution has licenses and special vehicle to transport nuclear fuel including irradiated fuel and radioactive waste (RW).

Irradiated nuclear spent fuel including as a part of experimental devices is placed to age it in CRR IGR fuel storage. After being aged (falling activity) NSF is moved to CRR Baikal-1 and placed in long-term storage after post-test examinations in radiation-protection (hot) chamber.

Gaseous RW is accumulated in exhaust system ($V \cong 1000$ m^3) and when activity is fallen (lower than background radioactivity) it discharges in the air through the filter system.

Liquid RW is accumulated in the tank (~ 3 m^3) for liquid RW and then moved to CRR Baikal-1 to utilize.

Solid RW is gathered and moved to CRR Baikal-1 to long-term store.

3.2. HOT CELLS, PIE FACILITIES (radiochemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

- *Hot chamber*;
  Hot chamber located in CRR Baikal-1 is designed to treat safely with ionizing irradiation sources and post-reactor examine irradiated experimental devices and SNF. It is placed in CRR Baikal-1.
- *Non-destructive X-ray installation*;
  The digital X-ray system on the basis of X-ray apparatus RAP 150/300-14 (300 kV X-ray tube 1,2-3 BPM5-300) is used for nondestructive testing of experimental devices applying in various physical out-of-pile and in-pile experiments.

FIG. 5. Apparatus RAP 150/300-14.

The system allows studying the high resolution and contrasting sensitivity about 0.5% different constructive elements. Radiographic layer thickness for steel is 70 mm, for aluminum — 250 mm. High parameters of conversion and signal recording system allow to use effectively Introscope used to determine and compare the spatial location of nuclear fuel and other structural elements in the experimental devices before and after testing. The results of X-ray scanning are represented in a graphic file format.

— *Gamma-scanning facility*;

Gamma-scanner is designed to study distribution of spent nuclear fuel at an altitude of experimental devices by non-destructive gamma-ray spectrometry method. Experimental determination of SNF distribution through the height of experimental device is based on the gamma-spectrometric method for measuring the intensity of gamma radiation fission ^{235}U. The intensity of gamma radiation is measured by gamma-lines of ^{235}U (^{95}Zr, ^{95}Nb) fission products against gamma irradiation products of activation of ED structural elements.

Scanning is performed using gamma spectrometry of the complex, which includes Canberra Inspector gamma-ray spectrometer with semiconductor GL 0515R detector and slit collimator combination of steel and lead.

— *Material testing investigations*;

— *Sample production room*;

All operations are performed with radioactive materials in protective glove boxes, equipped with exhaust ventilation.

FIG. 6. Sample production room.

— *X-ray phase analysis*;

To make X-ray phase analysis we use diffractometer DRON-3, modernized and equipped with an automatic system Roentgen-Master-4.2.

Elements that are part of the device allow you to:

- Produce research of a general nature (qualitative and quantitative phase analysis, study of solid solutions, definition of macro-and micro strain, study of short-range order, etc.);
- Get a full set of integrated intensities of reflections from single crystals;
- Determine orientation of slices of single crystals;
- Examine texture.

— *Metallographic examinations*;

Two horizontal grinding and polishing machine NERIS and LaboPol-25 (manufactured by Struers) are used to make thin sections.

Test specimens (including spent fuel) are recorded in a special holder using compound (epoxy resin, sulfur) and crosscut is made by transverse cutting machine. Next, one surface receives metallographic grinding and from the second half the samples are taken for phase analysis or determination physical properties.

To rapidly assess quality section preparation and analyse surface of particles we use viewing stereoscopic microscope MBS-2 with a magnification range from 3.5 to 88 and a field of view from 39 to 2.6 mm. To fix the result of observing we use a digital camera with a special adapter installed in place of one of the eyepieces.

To carry out optical metallography of the samples tested we use different metallurgical direct microscopes MMU-3, BX-41M and inverted MMP-2. Each microscope is capable to take photographs with digital camera and adapter.

The sample microhardness is measured by microhardness tester PMT-3M by indentation of diamond pyramid (Vickers). To measure microhardness is an indispensable support method for the identification of sample phase composition.

— *SEM*;

In laboratory a raster scanning electron microscope JSM-6390 is used for electron microscopy. At present, the microscope is not equipped with microanalysis system (EPMA), but it is possible to do it.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS

The company has a license to design and manufacture equipment, which is subject to the requirements for atomic energy codes as well as structural divisions including design and research and engineering departments, technical laboratory diagnosis and monitoring that carry out the work. There is production area for thermoelectric converters including high-temperature ones.

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS (link to the list of recent publication is recommended)

Over the last ten years the main theme of IGR tests is experimental support for the concept of controlled movement of the melt in the perspective fast reactor to prevent the re-criticality in severe accidents with core melting. During this period, under the contract with Japan Nuclear Centre (JNC) and Japan Atomic Energy Agency (JAEA), eight full-scale (credit) experiments have been implemented at IGR reactor. Note that the development of design documentation, manufacturing (except power vessels) and assembling experimental devices, equipping them with primary transducers (sensors) are made by specialists of the Centre.

5. RECENT PUBLICATION

KENSUKE, K., LUNICHI, T., et al, EAGLE-project: Experimental study on elimination of the re-criticality issue during CDAs, 18; The result of the second in-pile integral test, Theses '2007 Fall Meeting of the Atomic Energy Society of Japan (Autumn)', Kita-Kyushu International Conference Centre, September 27-29, 2007, Proceedings of Annual/Fall Meetings of Atomic Energy Society of Japan, (2007).

VASSILIEV, Y., The result of a wall failure in-pile experiment under the EAGLE project, Nuclear Engineering and Design, **237** 22 (2007), 2165–2174.

TOVOOKA. J., KONISHI. K., et al, EAGLE-project: Experimental study on elimination of the re-criticality issue during CDAs, 20; Evaluation of molten fuel-pool heat transfer in the first integral test, Theses 'AESJ Annual Meeting (Spring), Osaka University, Japan, AESJ (2008).

Profile 14

HANARO
KOREA

1. GENERAL INFORMATION AND TECHNICAL DATA

FIG. 1. View of HANARO building.

The HANARO (High-flux Advanced Neutron Application Reactor) is a 30 MW thermal power multi-purpose research reactor generating high neutron flux. It was designed by Korea Atomic Energy Research Institute (KAERI) as a unique domestic facility for research and development on the neutron science and its applications. The industrial, academic, and research demands on the utilization of the HANARO have increased in various fields such as nuclear in-pile tests, production of key radioisotopes, neutron transmutation doping, neutron activation analysis, neutron beam research, radiography, environmental science, health science, agriculture, bio-engineering, etc. A cold neutron source installed in 2009 and cold neutron scattering instruments are opening a new era for HANARO to become a regional and an international facility. In addition, several countries asked KAERI to involve in their feasibility studies, staff training for safe operation, and development of utilization technology for their future reactors.

The maximum thermal neutron flux is 5×10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$ and the maximum fast neutron flux reaches 2×10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$. These enable HANARO to accommodate experimental devices in core and reflector for various purposes. The irradiation holes in core region are used for material/fuel irradiation and isotope production. In the reflector region, seven beam tubes, three irradiation holes for NAA and several holes for RI production are provided. The cycle length is about four weeks and the operation day per year is normally 200 days.

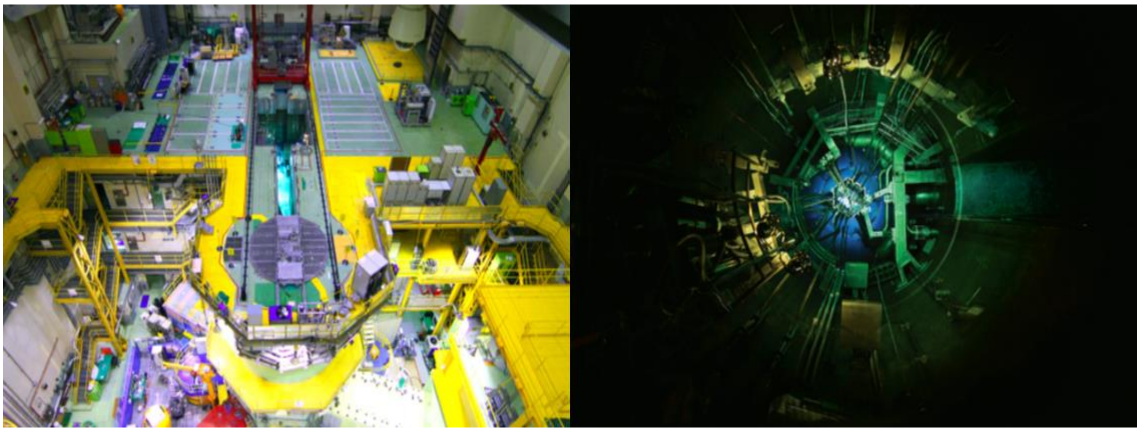


FIG. 2. Reactor hall and pool.

1.1. REACTOR

The reflector tank provides a large region of high thermal neutron flux and the excess reactivity is sufficient for experiment. The average burn-up of the discharged fuel assemblies meets the design objective, greater than 50 atom % burn-up of initial fissile material. The reactor operation cycle is 28 days. The reactor has inherent safety characteristics such as heat removal by natural circulation and reactor trip by gravity drop of shut-off rods.

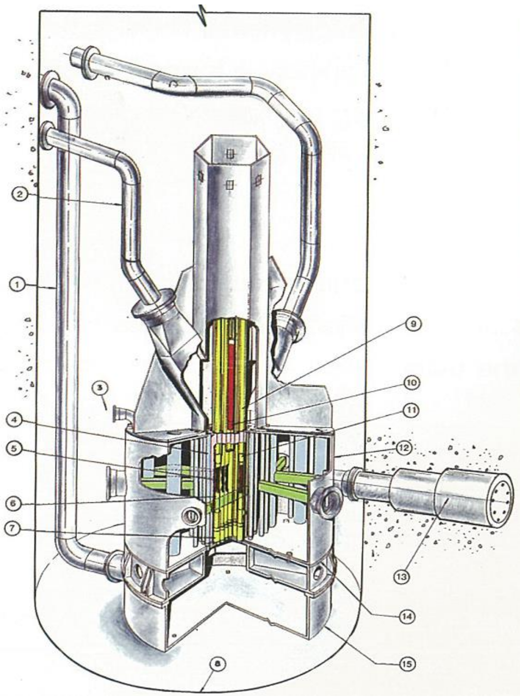
The core features a combination of light-water cooled/moderated inner core and light-water cooled/heavy-water moderated outer core. The inner core has 28 fuel sites and 3 test sites. Among them, 20 fuel sites have hexagonal shapes, and 8 fuel sites are for circular fuel assemblies, which are enclosed by 4 control absorber rod (CAR) shrouds and 4 shut-off rod (SOR) shrouds made of natural hafnium. Three test sites are also in hexagonal shapes and used for capsules. The outer core consists of four fuel sites and four test sites, which are embedded in the reflector tank.

TABLE 1. REACTOR SPECIFICATION

Type	Open-tank-in-pool
Maximum thermal Power	30 MW
Coolant, moderator	Light water
Reflector	Heavy water
Absorber material	Hafnium
Maximum thermal neutron flux	In-core – 5.0×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ Out-core – 2.0×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$
Safety features	Large heat sink Passive cooling system by natural convection Negative reactivity coefficient Two shutdown systems

TABLE 2. REACTOR CORE

Fuel loading holes	36-element fuel assembly: 20 18-element fuel assembly: 12
Vertical holes for irradiation	In-core: 3 Out-core: 4 In reflector: 25



No.	Description
①	Primary Coolant Inlet Pipe
②	Primary Coolant Outlet Pipe
③	Heavy Water Outlet
④	Outer Fuel Site
⑤	Fuel Bundle
⑥	Flow Tube
⑦	Fuel Receptacle
⑧	Pool Liner
⑨	Shroud Tube
⑩	Shutoff Rod
⑪	Control Absorber Rod
⑫	Reflector Tank
⑬	Beam Tube
⑭	Grid Plate
⑮	Inlet Plenum

FIG. 3. Reactor structure assembly.

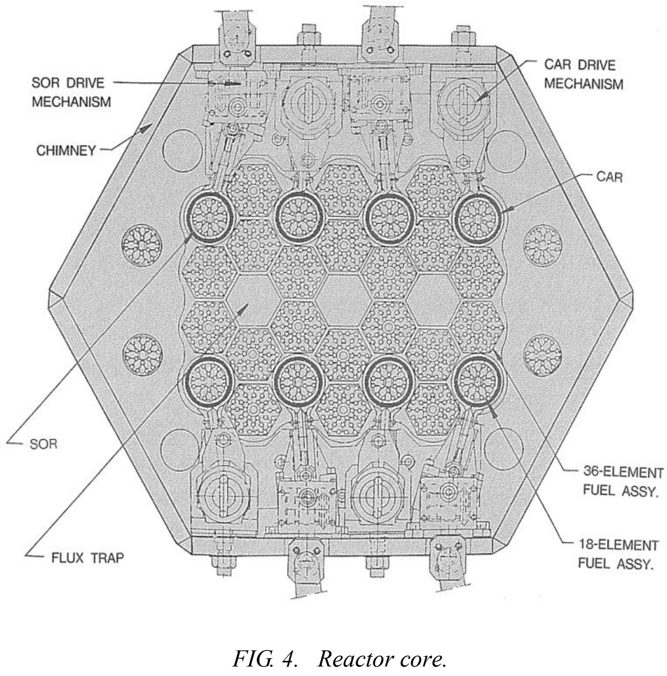


FIG. 4. Reactor core.

1.2. FUEL

The fuel used in HANARO is low enriched uranium (19.75% U-235) in uranium silicide compound which is dispersed in the aluminium matrix. The aluminium cladding, which is finned to increase the heat transfer surface area, protects the uranium from corrosion and prevents radioactive fission products from escaping. The quantity of uranium in one fuel element is 69.1 g in a standard core element and 51.4 g in a reduced core element. The reduced core elements are induced to have a uniform power distribution within a fuel assembly and located at the outermost ring of the hexagonal fuel assembly. There are two types of fuel assemblies required for HANARO — hexagonal fuel assemblies having 36 elements and circular ones having 18 elements.

TABLE 3. FUEL ASSEMBLY

	18-Elements	36-Elements
Type	Circular assembly	Hexagonal assembly
Material	U_3Si-Al (58.6 w% U, 2.4 w% Si)	
Enrichment	19.75 w% U-235	
Outside diameter	59.5 mm	73 mm
Fuel meat length	700 mm	700 mm
Fuel pin diameter	6.35 mm	6.35/5.49 mm
Fuel fin per bundle	8 numbers	8 numbers

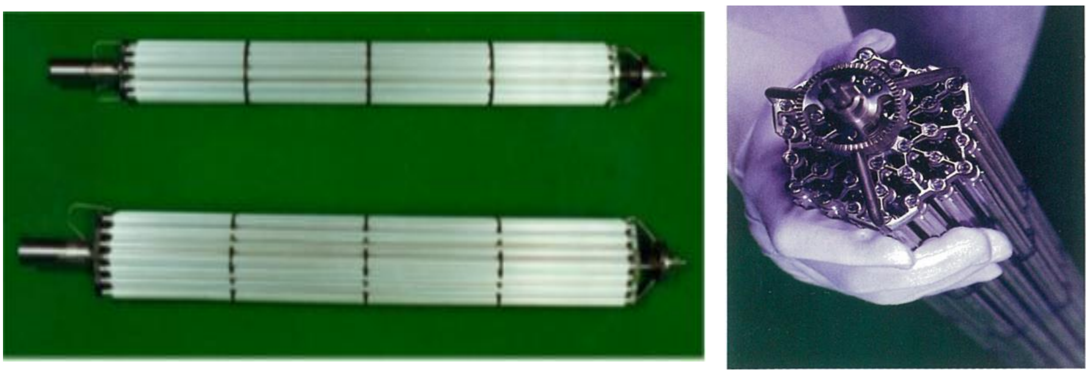


FIG. 5. HANARO fuel assemblies.

2. EXISTING AND PERSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

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

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

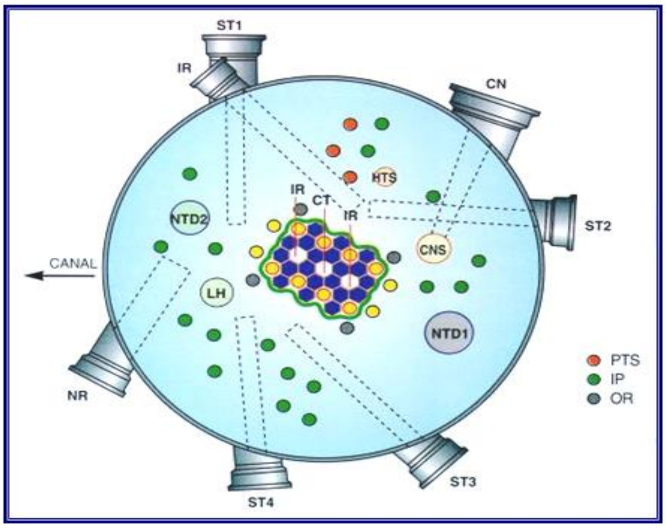


FIG. 6. Experimental holes of HANARO.

2.1.1. Neutron beam application

Neutron radiography facility (NRF), High resolution power diffractometer (HRPD), Four circle neutron diffractometer (FCD), Residual stress instruments (RSD), Vertical neutron reflectometer (REF-V), Vertical neutron reflectometer (REF-V), Horizontal neutron reflectometer (REF-H), High intensity powder diffractometer (HIPD), Ex-core neutron irradiation facility (ENF) and Prompt gamma neutron activation analysis system (PGAA), are operating at the HANARO reactor hall. In addition there is a Neutron activation analysis system (NAA) consisting of the irradiation facility and the radiation measurement equipment in HANARO reactor. A 9 m SANS instrument installed at CN horizontal port was dismantled and moved to the cold neutron laboratory and currently the cold neutron guide system is installed at CN port area. The Bio-diffractometer (Bio-D) at ST-3 port and Triple axis spectrometer (TAS) at ST4 are scheduled to be installed at the reactor hall. Three new cold neutron scattering instruments will be developed and installed in the cold neutron guide hall and three neutron instruments currently existing in the reactor hall will be moved after upgrades. Three new instruments are 40 m small angle neutron scattering Instruments (40 m SANS), cold triple axis spectrometer (Cold-TAS) and disk chopper time of flight (DC-TOF) and three upgrade instruments are 12M-SANS, REF-V and Bio-REF.

TABLE 4. HORIZONTAL HOLES FOR BEAM INSTRUMENTS

Beam tubes	Form	Size (cm)	Neutron characteristics at the nob of tube		Use
			Thermal flux (<0.625 eV)	Cd ratio	
ST1	Rectangle	7 × 4	$1{,}89 \times 10^{14}$	8.24	Neutron scattering Experiment
ST2	Rectangle	7 × 4	2.22×10^{14}	11.32	
ST3	Rectangle	7 × 4	2.24×10^{14}	11.81	
ST4	Rectangle	7 × 4	2.53×10^{14}	13.97	
CN	Rectangle	6 × 15	9.68×10^{13}	33.82	
IR	Circular	10	2.74×10^{14}	6.72	
NR	Circular	10	4.32×10^{13}	60.36	Neutron radiography

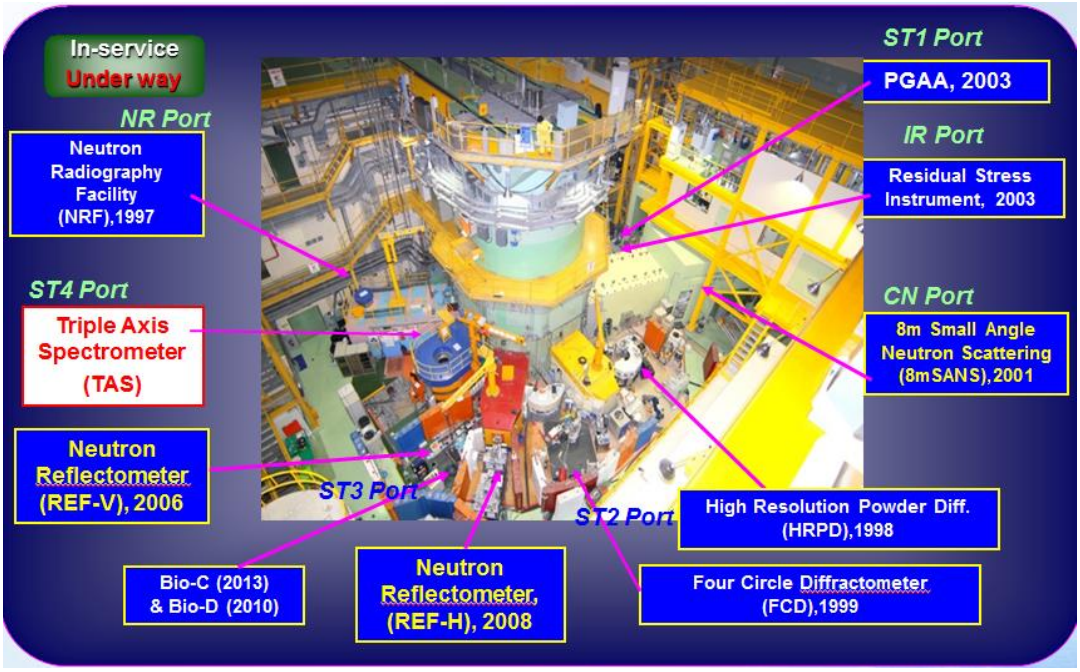


FIG. 7. Neutron scattering instruments in the HANARO reactor hall.

2.1.2. Capsule irradiation test facility

The national research and development programme on nuclear reactors and nuclear fuel cycle technology requires numerous in-pile tests in HANARO. Extensive efforts have been made to establish the design and manufacturing technology for irradiation facilities. Since HANARO is one of the world's most powerful multipurpose reactors, this reactor provides a variety of irradiation tests that benefit from the exceptionally high neutron flux available. The main activities of capsule development and utilization programmes are focused on the in-reactor material tests, new and advanced fuel research and development, safety-related research and development for nuclear reactor materials and components, and basic research. Now, capsules have been developed and being utilized for the irradiation test of materials and nuclear fuels in HANARO, and the advanced capsules for the research of the irradiation features of creep and fatigue have been developed. Also, in order to study the high burn-up nuclear fuel in nuclear fuel in HANARO, re-instrumentation and re-irradiation technology is under review.

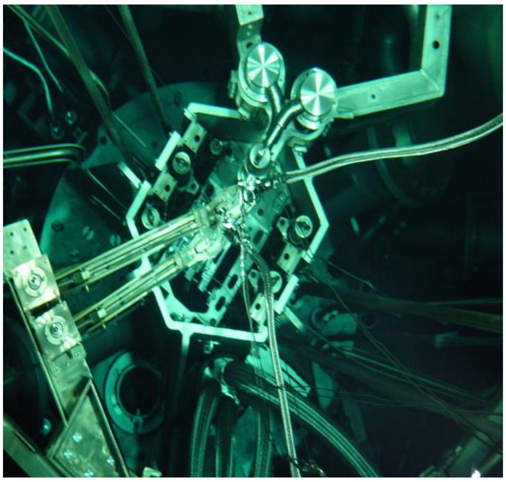


FIG. 8. Capsule irradiation in HANARO.

TABLE 5. VERTICAL HOLES FOR IRRADIATION

Location	Hole		Inside Dia. (cm)	Neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)		Use
	Name	Number		Fast (>0.82 MeV)	Thermal (<0.625 eV)	
Core	CT	1	7.44	2.10×10^{14}	4.39×10^{14}	Material/fuel irradiation test
	IR	2	7.44	1.95×10^{14}	3.93×10^{14}	Material/fuel irradiation test
	OR	8	6.00	2.23×10^{13}	3.36×10^{14}	Material/fuel irradiation test
Reflector	NTD	2	22.0 18.0	6.98×10^{10} 1.11×10^{11}	3.93×10^{13} 6.2×10^{13}	NTD
	CNS	1	16.0	1.32×10^{12}	1.73×10^{14}	Cold neutron source
	LH	1	15.0	6.62×10^{11}	9.77×10^{13}	RI production
	HTS	1	10.0	9.84×10^{10}	8.17×10^{13}	RI production
	NAA	3	6.0	1.34×10^{12}	1.62×10^{14}	NAA
	IP	17	6.0	1.51×10^{12}	2.44×10^{13}	RI production

2.1.3. Radioisotope production

The radioisotopes produced from HANARO are being supplied for medical and industrial purpose. Twenty-two target positions, which consist of 4 ORs, 17 IPS and one HTS in the reactor core, are used to produce radioisotopes. Four concrete hot cells and 17 lead hot cells are equipped in the radioisotopes building. New approaches are being developed for automatic processing of variable radioisotopes of current interest in industry and nuclear medicine. ^{131}I and ^{192}IR are the major products and therapeutic radioisotopes like ^{165}Dy, ^{166}Ho, ^{153}Sm, ^{186}Re and ^{188}Re, ^{99m}Tc, etc. are being produced.

2.1.4. NAA

The automatic and manual pneumatic transfer system have been installed in three irradiation holes for neutron activation analysis, and one of the irradiation holes is lined with cadmium. Available thermal neutron flux with each irradiation site is in the range of 3.9×10^{13}–1.6×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ and cadmium ratios are 50–250. One system is designed for the analysis of short-lived nuclides and the other system is designed for analysis of medium and long-lived nuclides. All systems are operated by the electronics and PC controller.

2.1.4. NTD

The HANARO has two vertical holes for NTD in the heavy water reflector region. They are very good for the NTD of silicon from the viewpoints of neutron quality and size. The doping service at a small hole (NTD 2) was started from the end of 2002 for a 5-inch silicon ingot. In 2005, the service for a 6-inch ingot also began at the same hole. In 2008, doping for 6 and 8-inch ingots began at a large hole (NTD 1). The irradiation rigs with a flux screen to make neutron irradiation inside an ingot uniform have been developed. Additionally, the burn-up following irradiation technology has been developed.

2.1.4. Others

In addition, there are the neutron radiography and tomography facility (NRF), and the ex-core neutron facility (ENF), cold neutron research facility (CNRF) and the facility manufacturing neutron guides etc. in HANARO.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

Not available

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, RIA etc.

Not available

2.4. LOOPS FOR INVESTIGATION OF CORROSION OF REACTOR MATERIAL

Capsule is utilized for investigation of corrosion properties.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS

The safe availability of nuclear power is dependent on the material performance of the structures and components of the nuclear facility. Therefore, a safety evaluation and in-reactor design data of the reactor core materials are crucial for a structural integrity evaluation of a nuclear reactor. Numerous material tests are required in the national research and development programme on nuclear reactors and nuclear fuel cycle technology. Extensive efforts have been made to establish the design and manufacturing technology for irradiation facilities. Capsules are being developed for the irradiation testing of materials and nuclear fuels. In addition, capsules were developed to study the creep and the fatigue behaviours of materials under irradiation. The main activities of the capsule development and utilization programmes are as follows:

- Development of irradiation technology at high temperature;
- Development of evaluation technology for fast and thermal neutron fluencies;
- Development of long-term irradiation and re-irradiation technology;
- Establishment of instrumentation and evaluation technologies for characterizing nuclear fuel and material performance;
- Development of irradiation technology for future nuclear reactors.

In the near future, irradiation devices and technologies will be developed to irradiate materials and nuclear fuel to support the development of Gen-IV reactor systems such as VHTR (Very High Temperature Reactor) and SFR (Sodium Fast Reactor).

2.5.1. Capsule for material irradiation test

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

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

nuclear power reactor;

— Basic researches of the irradiation effects in materials.

(1) Non-instrumented capsule;

A non-instrumented capsule is typically 1 m in length and 60 mm in diameter. Specimen temperatures are controlled by varying the widths of the gas-filled gaps between the specimens and specimen holders, and monitored with passive fluence and temperature monitors. A variety of specimens can be included in five stages of the capsule.

(2) Instrumented capsule;

An instrumented capsule is a cylindrical shape and its main-body is 60 mm in diameter and 870 mm in length. The basic instruments of the capsule consist of the thermocouples, fluence monitors, and micro-heaters to fulfil the user requirements. The temperatures of the specimens in five stages are independently controlled by a capsule temperature control system.

FIG. 9. Instrumented capsule for material test.

2.5.2. Ampule for material irradiation test

The followings receptacles are used for a small volume of material specimens. They are shown in Fig. 10.

FIG. 10. Small irradiation receptacles.

2.5.3. Support facilities for material/fuel irradiation test

— *Half core test loop*:

The half core test loop has the same outside appearance and the inside structure cut down by half of the HANARO reactor, and is also utilized to verify the performance of capsules through the vibration, endurance, and pressure drop test, etc.

— *Single channel test loop*:
The single channel test loop is used for the loading/unloading test, the performance test of the control system and the safety assessment for the irradiation devices such as a capsule and a rig by a thermal hydraulic and vibration test.
— *Capsule supporting system*:
Each capsule supporting system for CT, IR test holes in HANARO is used to grip the capsule during an irradiation test.
— Capsule cutting system:
After irradiation test, the capsule's main-body is separated from the protection tube in service pool by the cutting system.

2.6. CAPSULE FOR FUEL IRRADIATION TEST

The fuel capsule is applicable to research into the irradiation characteristics of the fuel pellet and to obtain the in-core performance and the design data of the nuclear fuel in HANARO. The non-instrumented capsule was developed in 1999, and has been utilized for the irradiation characteristics test of the DUPIC fuel and advanced PWR fuel pellets. The design verification test of the instrumented capsule was completed in HANARO's test hole in 2003. Now, the instrumented capsule can be used to measure the fuel temperature, internal pressure of the fuel rod, the fuel deformation and the neutron flux during a fuel irradiation test. The application fields are as follows:

— Irradiation of a nuclear fuel for the DUPIC, advanced PWR and CANDU;
— Study on the in-core characteristics of the UO_2 pellet and UO_2 pellet including additives;
— Basic researches of the fission gas release, etc.

(1) Non-instrumented fuel capsule:
A non-instrumented capsule is utilized in studies on the irradiation characteristics of a nuclear fuel in the OR test hole of HANARO. Its main body is 960 mm in length and 56 mm in diameter, and it can hold 3–6 test fuel rods, length of about 200 mm, in the capsule.
(2) Instrumented fuel capsule:
An instrumented capsule can be installed in the OR4 and/or OR5 of HANARO, its main body is 56 mm in diameter and about 720 mm in length and the total length including the protection tube is about 5 m. The capsule includes several test fuel rods instrumented with thermocouple, pressure transducer and elongation detector to measure the fuel temperature, internal pressure of the fuel rod, and the fuel deformation, respectively, and SPNDs to detect the neutron flux.

FIG. 11. Fuel capsule.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTICS

Not available.

3.2. HOT CELLS, PIE FACILITIES

The IMEF (Irradiated Materials Examination Facility) has been conducting post-irradiation examinations (PIE) on irradiated materials used in the HANARO research reactor and power plant reactors to achieve the following objectives:

- Integrity and life estimation of the structural parts in an operating reactor;
- Irradiation behaviour evaluation of developing fuels and structural materials for next generation and future reactors;
- Back end fuel cycle demonstration tests.

The IMEF is devoting to supply high-quality PIE data to R&D projects on nuclear fuel and materials. It has contributed toward the Republic of Korea becoming a world-leading nuclear technology nation. IMEF construction began in 1988, and the facility has operated regularly since 1996. The DUPIC facility for the development of the direct use of spent PWR fuel in CANDU reactors was established in 1999. Furthermore, the ACP facility for the development of an advanced spent fuel conditioning process was constructed in the basement in 2005.

IMEF has three stories and one basement with section areas of 29.0 m x 51.0 m and about 4000 m^2 of total floor space. The hot cells, which are the main facilities, had 26 work units with a total length of 60 m at the beginning stage. That was increased to 71 m in total length, with 31 work units after the construction of new hot cells in the basement in 2005. The maximum wall thickness of the hot cells is 1.2 m to shield a radiation source with a maximum radioactive level of 3.7×10^{16} Bq. In addition, a pool with the depth of 10 m is located in the service area to handle transporting casks.

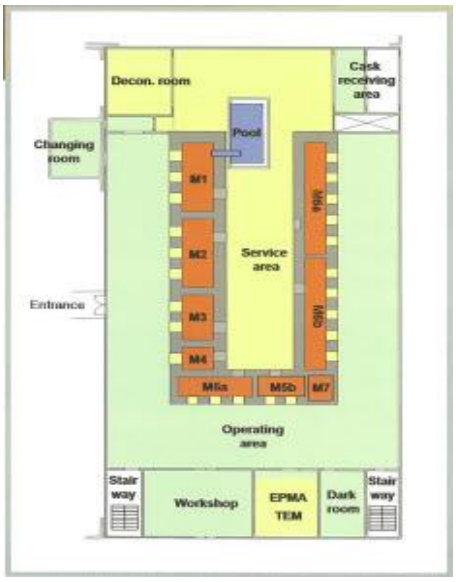


FIG. 12. IMEF hot cell layout.

The hot cells in the ground floor consist of six concrete cells (M1–M6) and one lead cell (M7) with a ‘U’ shape line arrangement. One hot-cell (M8) was placed in the basement of ACPF.

— M1–M4 line: non-destructive tests, capsule dismantling, specimen preparation, storage;
— M5 line: mechanical tests (impact, tensile, fracture, fatigue), dimension measurement;
— M6 line: DUPIC experiments (powdering, oxidation-reduction, sintering);
— M7 line: microscopy, hardness, density, SEM;
— M8 line: pyro-processing demonstration tests.

TABLE 6. HOTCELL SPECIFICATION AND EQUIPMENTS

Pool/hot cell	Main function	Dimensions (L × W × H) (m)	No of working units	Equipment
Pool	Cask handling	6.0 × 3.0 × 10.0	-	Bucket elevator Purification system Cask handling tool
M1 hot cell	Non-destructive tests	7.0 × 3.0 × 6.0	3	Gamma scanning system Dimension measuring system Blistering system
M2 hot cell	Dismantling & cutting	7.0 × 3.0 × 6.0	3	CNC milling M/C Capsule cutting M/C Electric discharge M/C De-cladding M/C
M3 hot cell	Specimen preparations	4.7 × 3.0 × 6.0	2	Micro cutting M/C Mounting Press Polishing M/C Periscope
M4 hot cell	Specimen storage	2.3 × 3.0 × 6.0	1	Storage rack
M5a hot cell	Mechanical tests	7.1 × 2.0 × 4.0	3	Impact tester, UTM Dimensional measuring M/C
M5b hot cell	Mechanical tests	4.8 × 2.0 × 4.0	2	Static tensile tester N\Dynamic UTM
M7 lead hot cell	Microscopy	2.5 × 1.4 × 4.65	2	Optical microscope Hardness tester Density measuring equipment SEM
M6 hot cell	DUPIC experiments	23.4 × 2.0 × 4.0	10	Mechanical slitter OREOX furnace Rotary ball miller Sintering furnace Compaction press Laser welder
M8 hot cell	Advanced fuel cycle experiments	10.3 × 2.0 × 4.3	5	De-cladding M/C Powdering M/C Furnace
Hot Lab.- 1	Micro-analysis	5.8 × 7.5 × 8.4	-	Shielded EPMA TEM, EDS
Hot Lab.- II	Physical test		-	Thermal diffusivity Fission gas diffusivity

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEM INCLUDING HUMAN RESOURCES DEVELOPMENTS

HANARO user training is performed for surveying and cultivating users. The training programmes consist of neutron beam application, irradiation, and NAA. The training and education are performed

The programmes are divided as the introductory and advanced course. The introductory course consists of 1–2 days classes, and the advanced course has 3–4 days classes. In the course of irradiation test, 20–30 persons participate in from universities and industries.

4. RECENT ACHIEVEMENTS

4.1. IRRADIATION FIELDS

The irradiation technology has been utilized to the irradiation tests of materials and fuels for currently operating NPP, the export research reactors and the future nuclear systems. The irradiation tests performed in the some last years are listed in the Fig. 12. Recently, through the construction of a 5 MW multipurpose research reactor, called the Jordan Research and Training Reactor (JRTR) and another domestic research reactor in Gijang, the Korean government hopes to be a crucial reactor vendor in the global nuclear market. The new research reactor in Gijang is under construction, and is due to start up in 2017. The new research reactor will become the most up-to-date research reactor available in the world and will specialize in radioisotope and NTD production and demonstrations of reactor designs. Therefore, HANARO will specialize more on irradiation research of nuclear fuels and materials.

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

To effectively support the national R&D relevant to the present NPP, research reactors, and future nuclear systems, the development of advanced irradiation technologies concerning irradiation temperature and instrumentation is being preferentially developed at HANARO. After operation of the new research reactor in Gijang, HANARO is supposed to be specialized more on irradiation research. The available irradiation temperature will be extended up to 100–1000°C, and several key instruments such as thermocouple, LVDT, and SPND are being localized at HANARO. New irradiation technology needed to test advanced nuclear fuels and materials for the VHTR, SFR, and fusion systems are being developed recently. In addition, the research status and possibility of new electro-magnetic materials using neutron irradiation will be surveyed to ascertain a utilization of neutron irradiation technology in high-tech material industries.

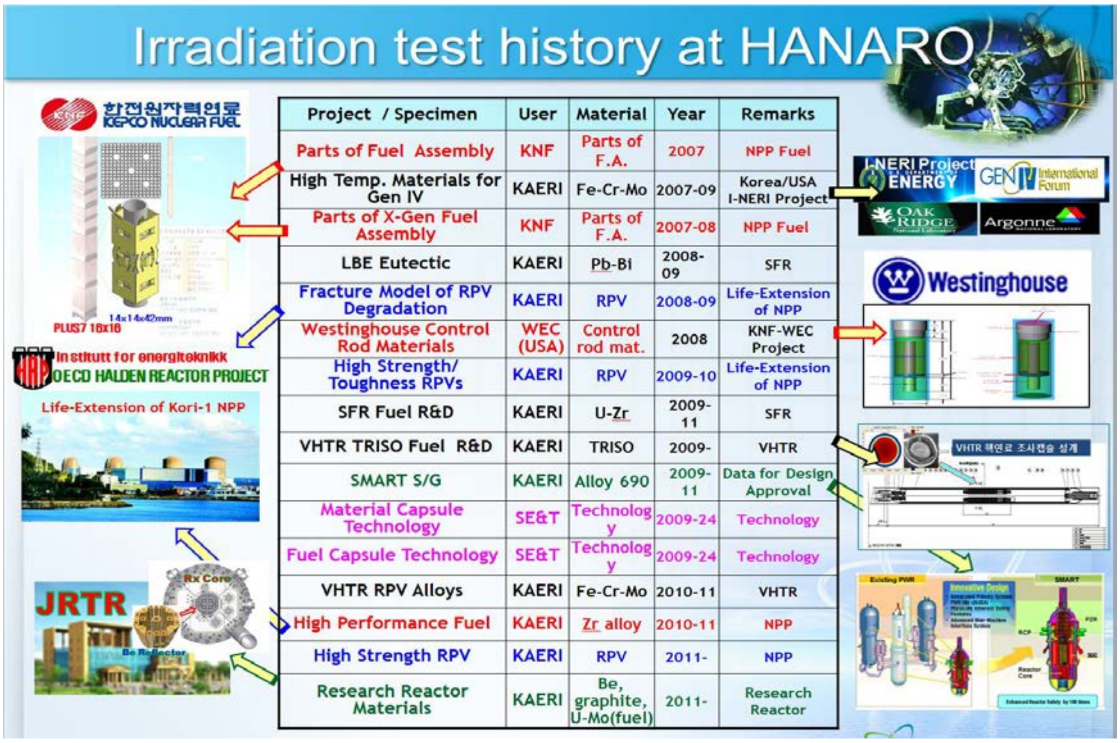


FIG. 13. Irradiation test history at HANARO.

4.2. OTHER FIELDS

4.2.1. Cold neutron activation station

Cold neutron activation station (CONAS) was developed recently. CONAS includes three neutron activation analysis instruments measuring PGAA and NIPS and charged particles (NDP) after neutron capture. PGAA and NIPS are placed at the end position of the CG2B neutron guide and are applied for a nuclear structure and decay study and chemical analysis. NDP is located at the end position of CG1 neutron guide and gives elemental concentration profiles up to a few micrometers in depth for the industrially important elements such as He, Li, B, O and so on as a near surface probe.

Design feature of instruments:

— PGAA and NIPS instruments:
  - True integrated neutron flux at the end of CG2B guide: 8×10^8 n·cm^{-2}·s^{-1};
  - Gamma-ray spectrometers;

— NDP instrument:
  - True integrated neutron flux at the end of CG1 guide: 9×10^7 n·cm^{-2}·s^{-1};
  - Charged particle spectrometer.

4.2.2. NTD

A few years ago, doping services for 8-inch ingots was started. Doping facility using only fast neutron is now being developed.

4.2.3. Radioisotopes production

The mass production of radioisotopes became possible after the operation of HANARO.

Recently, the RI production at HANARO has reached near its limit. Currently a new RR at Gijang is being constructed to meet the national and international demands of radioisotopes, where Fission Moly will be produced. In addition, a transportable gamma-ray CT system using ^{137}Cs and ^{60}Co is being developed for diagnose without interruption of factory operation. And thermo-electric and micro RI battery is being developed, which can supply the long-term electric power source in the extreme condition.

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Profile 15

THE HIGH FLUX REACTOR (HFR)
THE NETHERLANDS

1. GENERAL INFORMATION AND TECHNICAL DATA

The High Flux Reactor (HFR), located in Petten in The Netherlands, is one of the most powerful multi-purpose research and test reactors in the world. Together with the hot cells of NRG at the Petten site, it has provided for over five decades, an integral and full complement of irradiation and post-irradiation examination services as required by current and future R&D for nuclear energy, industry and research organizations. The HFR has 17 in-core and 12 poolside irradiation positions, plus 12 horizontal beam tubes.

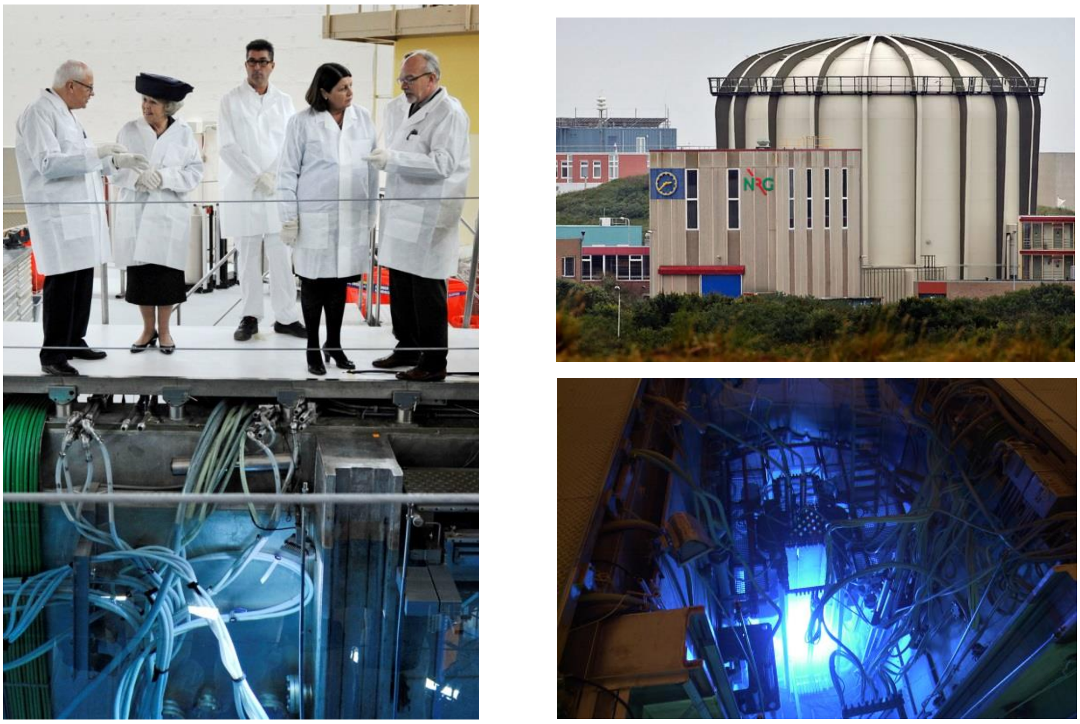

FIG. 1. Visit of H.M. Queen Beatrix of the Netherlands and Eurocommissioner M. Geoghegan–Quinn on the occasion of 50 years HFR, 22 November 2011.

The HFR uses low-enriched uranium U_3Si_2 fuel. The reactor has a total of 33 fuel rods and six control rods. The current cycle length is 31.5 days with 28 full power days (ten cycles per year). The HFR operates at a constant power of 45 MW and uses light water both for cooling and moderation. The reactor is of the tank in pool type with a rectangular aluminium vessel. The core is surrounded by beryllium reflector elements at three sides. At the fourth side the pool side facility (PSF) is located (outside the reactor vessel).

With a variety of dedicated irradiation devices and with its long-standing experience in executing small and large irradiation projects, the HFR is particularly suited for fuel, materials and components testing for all reactor lines, including thermo-nuclear fusion reactors. In addition, processing with neutrons and gamma rays, research with neutrons and inspection services are employed by industry and research, such as neutron radiography and neutron diffraction.

The HFR is one of the world most important producers of radio-isotopes, both for medical and industrial applications. In radioisotope production the HFR has attained the European leadership in production volume.

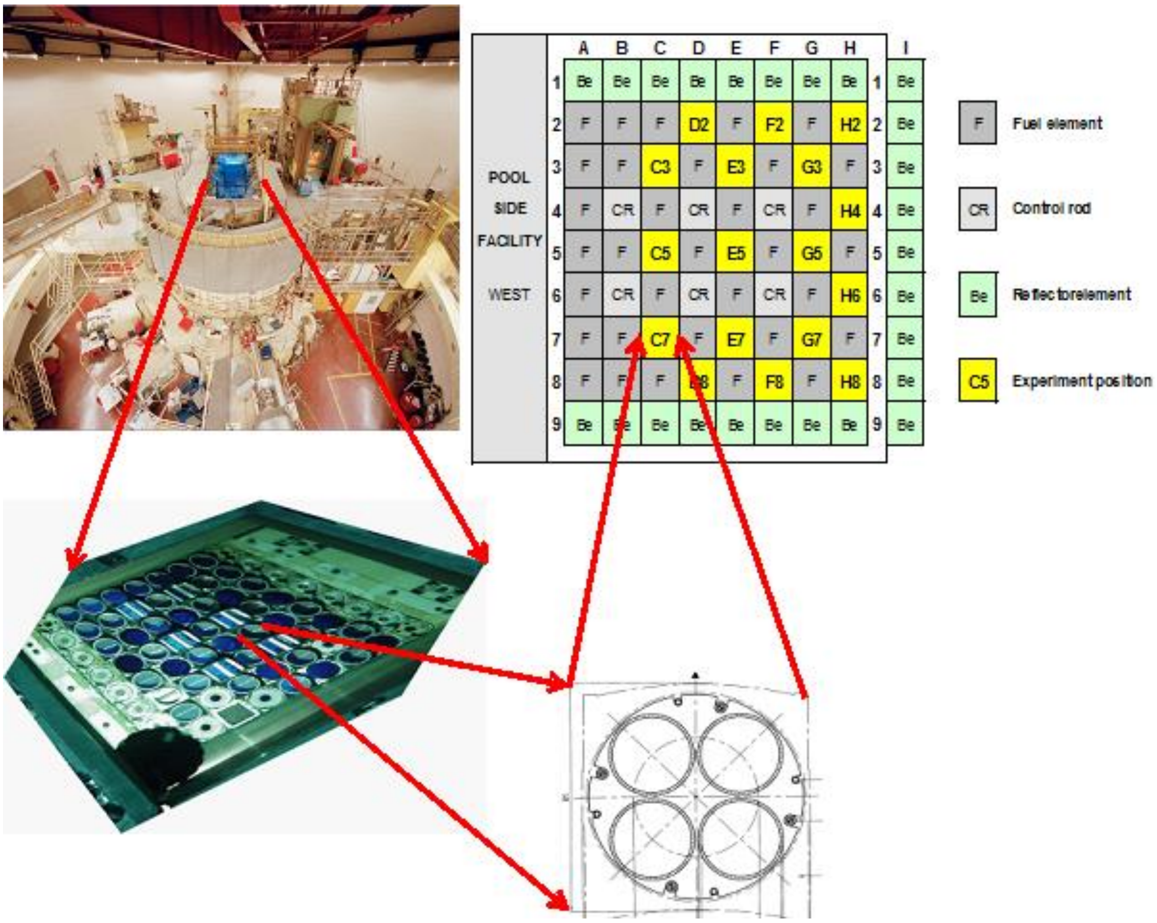

FIG. 2. Overview of the reactor hall, the basin, the core and the positioning of irradiation experiments.

The current irradiation programmes at the HFR address areas in:

- R&D for nuclear fission energy, i.e. materials irradiation in support of nuclear plant life extension, transient testing of pre-irradiated LWR fuel rods containing UO_2 or MOX fuel, testing of fuel and materials for innovative reactors;
- Transmutation studies of actinides and long-lived fission products for the heterogeneous and direct cycle and studies in support of incineration of plutonium;
- Support of R&D for thermo-nuclear fusion energy within the European Fusion Technology Programme, i.e. irradiation testing of candidate materials for the first wall and diverter protection materials, and testing of breeding blanket materials and sub-modules for reference blankets with tritium analysis;
- Processing with neutrons and gamma rays, investigation with neutron-based research and inspection services, i.e. neutron radiography and neutron diffraction.

The HFR is owned by the Institute for Energy and Transport (IET) of the Joint Research Centre (JRC) of the European Commission (EC). Its operation has been entrusted since 1962 to the Netherlands Energy Research Foundation and later the Nuclear Research and consultancy Group (NRG). Since February 2005, NRG became also the licence holder of the HFR.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT THE HFR

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

The HFR has three main types of irradiation locations:

- In-core irradiation positions;
- Pool Side Facility irradiation positions;
- Neutron beam facilities.

These three types of irradiation positions are described hereafter in more detail:

(1) In-core irradiation positions;
Inside the aluminium reactor vessel, the irradiation devices may be installed in a lattice position inside the fuel region. In total 17 irradiation positions with a 60 cm effective height are available (typical diameter 70 mm). These positions can be subdivided in smaller diameter (typical diameter 30 mm) irradiation positions. Maximum flux values are in the range 2.6×10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$ (thermal flux) and 1.8×10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$ (fast flux, E > 1 MeV).

(2) Pool side facility (PSF) irradiation positions;
Directly outside the aluminium reactor vessel (2.5 cm from the corebox), 12 irradiation positions are located, each with a width of 65 mm. The PSF locations can be merged into broader locations. The neutron spectrum of the PSF is softer than that in-core with a thermal flux values in the range 1.5×10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$. Irradiation facilities can be placed on a trolley located on the PSF and then moved to and away from the core. PSF locations are amongst others suitable for:
    - Irradiations that last a shorter period than an HFR reactor cycle (e.g. ^{99}Mo production irradiations, other short lived isotopes);
    - Irradiations of large objects (e.g. 6 inch silicon ingots);
    - Irradiations that require a variable neutron flux (e.g. fuel ramp testing).

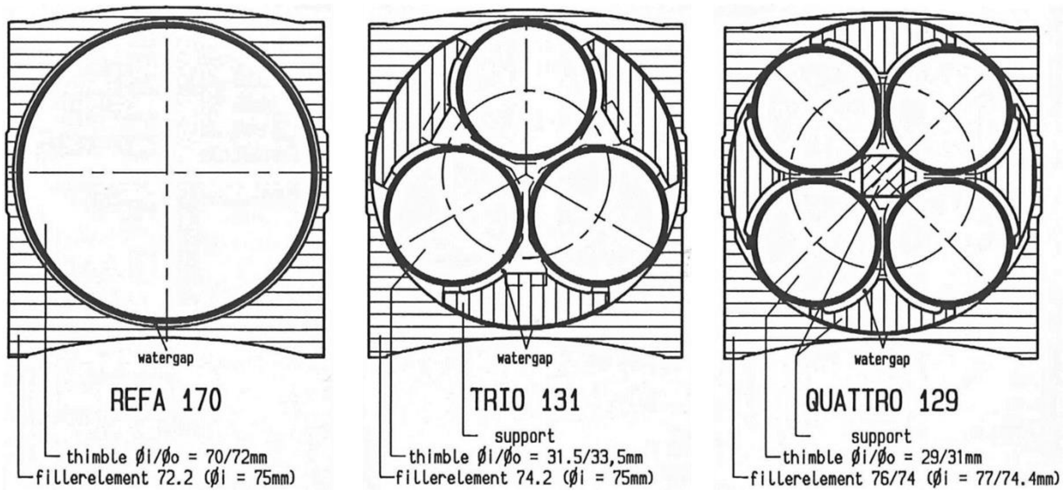

FIG. 3. Standard HFR irradiation rigs; outside water cooled, inside gas swept (mixtures of helium, neon, nitrogen); customisation possible.

(3) Neutron beams.
The HFR is equipped with 12 horizontal beam tubes.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

A large variety of nuclear components is being irradiated in the HFR. Irradiations are performed in standard irradiation facilities suited for a range of cooling agents including sodium; sodium is used since it has a high thermal conductivity and therefore homogenizes the temperature distribution.

Currently, no loops are present at the HFR, but existing facilities allow for irradiations under stagnant or semi-stagnant conditions. Examples include:

- Irradiation of metals in lead-bismuth to study their corrosion behaviour, followed by post irradiation examination at the NRG hot cells with special attention for specific safety actions to manage the radiological risks of ^{210}Po;
- Irradiations of lead amongst others in the form of lithium-lead compounds and as the above-mentioned lead bismuth;
- Various types of gas sweep-loop facilities are being used at the HFR:
  - A facility for the irradiation of high temperature gas-cooled reactor fuel. For this purpose a small gas flow is continuously purged along the HTGR fuel pebbles. Any release of gas, both xenon and krypton, from the fuel is measured using an HPGe gamma detector;
  - A facility for detection of tritium production and tritium release from lithium compounds and beryllium;
  - A facility for the determination of the degree of corrosion of graphite compounds under various irradiation conditions. Both the inlet and the outlet of the gas-purge system are connected to a gas chromatograph, thereby allowing quantitative determination of the corrosion rate. Inlet gasses with various compositions can be used.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA, etc.

From the 1970's until 1990's the HFR exploited various irradiation facilities for studies on the behaviour of fast reactor fuel under LOCA and under overpower conditions. Power control was performed by varying the distance to the reactor core using the PSF trolley system in combination with a BF_3-gas screen. The cooling was achieved either with stagnant sodium or flowing sodium or NaK. In the same period the behaviour of iodine under LWR conditions has been studied. The facilities are still available for (above described) safety studies in the HFR.

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

NRG is planning to construct a hot cell facility for corrosion tests under LWR conditions on irradiated samples.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT, AT WIDE RANGE TEMPERATURE AND DOSE RATES etc.

A large number of irradiation tests has been performed either in irradiation facilities with semi-stagnant gas or in closed capsule/ampules. Maximum irradiation temperatures may be in the 1000–1100^{o}C range can be achieved.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS

Neutron radiography is available for inspection and characterization purposes. In between two cycles neutron radiography on fuels or structural materials can be performed to study the impact of the irradiation on the dimensions (swelling, relocation etc.). Gas release can be measured as described under Section 2.2.

Irradiation facilities with bellows that apply a fixed force on the samples under irradiation are used to determine creep under irradiation conditions.

In the NRG hot cells, various facilities are present in which the impact of irradiation on the mechanical and physical properties can be determined including swelling (see Section 3.2).

On the Petten site facilities for the handling and interim storage of nuclear waste are available. The site has the disposal of a well-equipped mechanical workshop including facilities for sodium filling and high energy (3 MeV) X-ray inspection.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

The NRG hot cells are well-equipped with facilities to handle various types of transport containers. NRG owns several types of containers and has access to a network of transport companies in order to facilitate radioactive transports.

3.2. HOT CELLS, PIE FACILITIES (radiochemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

NRG has several facilities for PIE and fabrication and processing.

— The hot cell facilities (HCL) are used for handling highly radioactive objects such as radioisotopes and nuclear materials, including fuel. The HCL is amongst other equipped with alpha-tight cells for destructive examination of irradiated fuel using amongst others: optical microscopy, SEM with WDS, EDS and EBSD and autoradiography. Available non-destructive PIE techniques include: fuel rod-puncturing and gas analysis, gamma spectrometry, profilometry, X-ray photography and neutron radiography (at the HFR). Other facilities for characterization purposes include equipment for tensile, bending, compressive and Charpy impact testing, fatigue crack propagation, fracture toughness and creep as well as equipment for the determination of Dynamic Young's modules (ToF and resonance), thermal conductivity/diffusivity (laser flash) and expansion and electrical resistivity.

FIG. 4. Overview of part of the NRG hot cell facilities.

FIG. 5. Determination of the irradiation-induced stress relaxation behaviour of a bolt specimen.

— The ‘Jaap Goedkoop’ radiological laboratory has been designed for the characterization, fabrication and/or processing of medium and low radioactive materials. The building houses a series of dedicated labs for radiochemistry, solid source mass spectrometry (TIMS), XRD, XRT, neutron tomography, mechanical testing, tritium release measurement and synthesis/fabrication of (nuclear) materials. The laboratory also includes a facility for the fabrication of monitor sets for neutron dosimetrie including a series of dedicated detectors and equipment for data processing.

FIG. 6. Aerial photograph of the ‘Jaap Goedkoop’ radiological laboratory.

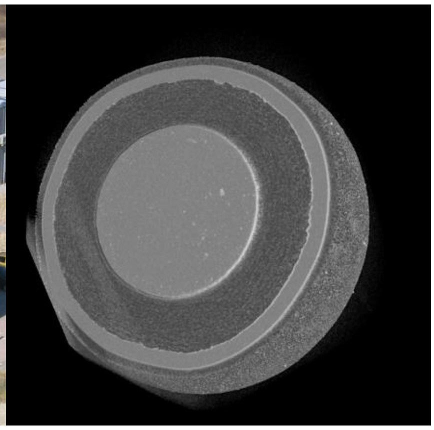

FIG. 7. 3D XRT (X-ray tomography) visualization of a TRISO particle (PYCASSO experiment).

— The actinides laboratory is located in the HCL for activities involving uranium, plutonium and small amounts of americium. This laboratory is used for amongst others the production of fuel pellets and fuel rodlets and characterization of (non-irradiated) material properties using. X-ray diffraction, thermo gravimetric analysis (TGA), differential scanning calorimetry (DSC) and microscopy.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

Almost all irradiation facilities for the HFR are designed and fabricated on the Petten site. All design steps and fabrication is performed by NRG’s mother company ECN, which has a well-equipped workshop.

FIG. 8. Overview of NRG's actinides laboratory; inset shows examples of experimental fuels for incineration of plutonium for the OTTO and FP6 experiment LWR Deputy, respectively.

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS

Tables 1 and 2 give an overview over (non-commercial) R&D irradiation experiments over the past 15 years. By far most experiments have been conducted as part of EU FP projects. The experiments in Table 1 concern material irradiations; the experiments in Table 2 irradiations on innovative fuels. Note that in some cases one experiment comprises multiple irradiations (up to 12, see SUMO in Table 1).

TABLE 1. OVERVIEW OF MATERIAL IRRADIATIONS EXPERIMENTS IN THE HFR OVER THE PAST 15 YEARS

Experiment	Application area
SUMO-1 to -12	9 Cr steels & joints for fission/fusion
STROBO-1 to -7	Stress-relaxation of bolt materials
CIWI	BWR core shroud welds
SOSIA-1 to -5	Creep & creep fatigue of 9Cr steels

IBIS	Structural material in lead-bismuth
INNOGRAPH-1A, -1B, -2A, -2B	HTR graphite irradiations
EXTREMAT-1, -2	High temperature materials
BODEX	Transmutation targets
POSITIVE	ITER first wall components
LYRA-1 to -10	RPV steel irradiations
PYCASSO-I, -II	HTR surrogate particles
HICU	Breeder material for fusion
EXOTIC-1 to -9	Solid tritium breeder materials
LIBRETTO-1 to -5	Liquid tritium breeder materials
HIDOBE-1, -2	High dose beryllium irradiation
Pebble Bed Assembly	Integrated fusion blanket experiment

TABLE 2. OVERVIEW OF FUEL IRRADIATIONS EXPERIMENTS IN THE HFR OVER THE PAST 15 YEARS

NRG name	Application area
OTTO	Once Through Then Out, Pu-transmutation
THORIUM CYCLE	Thorium fuel experiment
EFFTRA-T4, T4 bis, T4ter	Transmutation experiments under EFFTRA
HELIOS	Minor actinide fuels and targets
CONFIRM	Nitride fuels for fast reactors
MARIOS	SFR minor actinide fuel irradiation
HFR-EU1, HFREU1bis	HTR pebble irradiations
SMART	Nitride for advanced fuels
TRABANT	Fast reactor annular MOX fuel irradiation
SPHERE	Minor actinide bearing sphere-pac fuel
MARINE	Fast reactor minor actinide bearing fuel

A selection of peer-reviewed publications highlighting the above mentioned experiments is listed below.

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Profile 16

HALDEN BOILING WATER REACTOR (HBWR)

NORWAY

1. GENERAL INFORMATION AND TECHNICAL DATA

The Halden Boiling Water Reactor (HBWR) is located in Halden, a coastal town in the south-eastern part of Norway. The reactor hall and the reactor are situated within a rock hillside. The reactor was built as a prototype of a power generating reactor, and the location was chosen such that steam could be delivered to a near-by paper factory. This is done as of today when the reactor is in operation, but the main purpose is to serve as a materials testing facility for the OECD Halden Reactor Project and for bilaterally and multilaterally sponsored research.

FIG. 1. View of Halden Reactor with entrance to reactor hall in the mountain.

The HBWR is a boiling heavy water reactor operated in natural circulation with 14 tons of heavy water as moderator and coolant. The core consists of driver fuel assemblies as well as different fuel and material test assemblies. The heat removal system is designed for 25 MW, but the normal operating power is 18–20 MW (thermal), and the maximum water temperature is 240°C corresponding to an operating pressure of 33.6 bar. Prototypical water reactor thermal-hydraulic conditions as required for experiments can be simulated in loop systems.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES

After the initial phase of studying the characteristics of the HBWR as a power reactor, the use of the reactor became focused on fuels and materials irradiation experiments. Many ancillary systems were developed in support of these experiments of which many hundreds have been executed over the years. Twenty–thirty experiments can be active at the same time in the reactor core.

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

The HBWR is a versatile tool for nuclear fuels and materials investigations. It offers the following irradiation capacity:

- More than 300 positions individually accessible;
- About 110 positions in the central core (light blue in core cross section);
- About 30 positions for experimental purposes (any of 110/300);
- Experimental channel diameter is 70 mm in HBWR moderator and 35–45 mm in pressure flask;
- Height of active core 80 cm;
- Usable length within moderator about 160 cm.

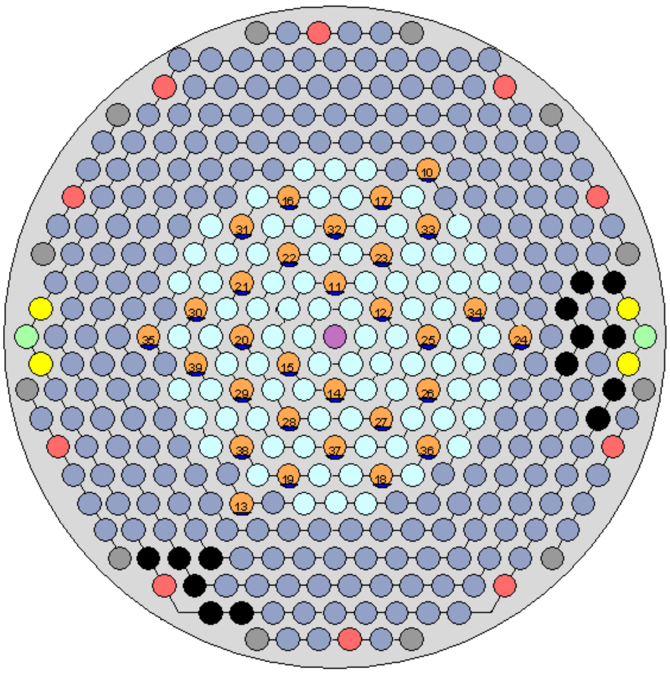

FIG. 2. Cross-section of Halden reactor core (light blue positions may contain experiments with fuel).

The fast neutron flux can be increased locally with booster fuel rods surrounding the test material.

Experiments are carried out using rigs which are inserted into the reactor core and connected to a loop and other outer systems as required for the experimental conditions and test objectives:

- Loop systems for simulation of BWR/PWR/WWER/CANDU conditions;
- Pressurisation system for imposing up to 500 bar pressure on fuel rods under operating conditions;
- Gas flow system;
- Gas analysis system;
- Hydraulic drive system.

The rigs are designed for the specific objectives of an experiment or series of experiments. In the latter case, fuel rods and specimens can be exchanged, thus facilitating multipurpose use and economy. A number of heavily instrumented rigs to suit different test objectives have been developed, and a handling compartment is available for exchange of fuel rods and for interim inspection and measurements.

Instrumentation for in-core measurements is an essential part of experiments in the Halden reactor. It provides information on a number of phenomena, separate and combined, which are essential for fuel performance assessments and safety analyses. For fuels testing, instrumentation typically includes:

- *Fuel thermocouple* which gives direct information on the fuel temperature and indirectly on fuel thermal conductivity and the changes induced by increasing burn-up;
- *Rod pressure transducer* providing data on the kinetics and amount of fission gas release under different operating conditions and how the threshold for fission gas release is reduced with burn-up. Fuel densification and swelling can also be deduced prior to the start of fission gas release;
- *Cladding extensometer* which gives data on the axial stress induced by pellet-cladding interaction and the relaxation with time. Fuel swelling can also be deduced in the case of prevailing contact between fuel and cladding;
- *Fuel stack elongation detector* measuring dimensional changes caused by fuel densification and fuel swelling;
- *Moveable diameter gauge* for scanning the cladding response to power changes and for detecting long-term diameter changes.

For materials testing, instrumentation can encompass:

- *DC potential drop measurement* which is a method to determine crack growth in materials over a long time;
- *Electrochemical potential sensor* which provides information important for irradiation assisted stress corrosion cracking experiments;
- *Water conductivity cell* for measurement of the electrical conductivity of the coolant water;
- *Electrochemical impedance measurement* for on-line assessment of parameters having an influence on corrosion.

Experiments with fuel are power calibrated. For fuels testing, the distribution of the total power is determined by means of neutron detectors mapping the neutron flux in the rig. Fuel experiment can contain one to eight fuel rods in one cluster, and twice these amounts in two-cluster arrangements.

Materials testing experiments can contain a large number of specimens in contact with the coolant or encapsulated for dry irradiation. Positions above and below the active length of the core are available where specimens can be exposed to the same chemistry and thermal-hydraulic conditions as in the core, but without the fast neutrons and gammas.

Data are acquired and displayed in real time with a standard frequency of two measurements per second. Data are usually stored permanently once per minute, but on request the frequency can be increased to up to two per second for some time. For noise analysis purposes, selected signals can be sampled and stored with 10 ms interval.

2.2. LOOPS FOR TESTING COMPONENTS OF A REACTOR CORE

The HBWR is a heavy water moderated and cooled reactor with natural conditions similar to those of commercial water moderated and cooled reactors. If more prototypical conditions are required by experiments, they are provided by about ten loop systems installed and in operation in the HBWR. Most experiments require steady state conditions. Special loops and/or rigs are being used for fast power ramp testing and transient and accident conditions such as dry-out and LOCA (see next section).

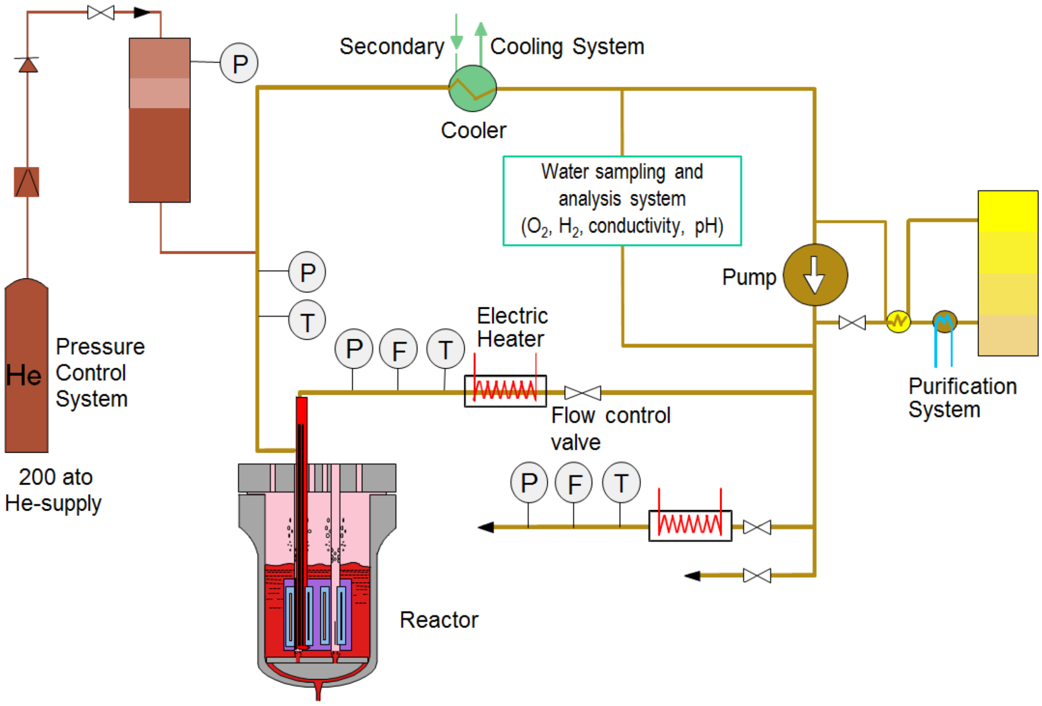

FIG. 3. Schematics of components in HBWR water loops.

Rigs for fuel and material testing under simulated water reactor conditions are inserted into in-core pressure flasks connected to light or heavy water circulation systems. These systems, which are completely separated from the reactor cooling systems, are designed for operation at pressures and temperatures of 165 bar and 340°C. Most of them simulate thermal-hydraulic and chemistry conditions of light water moderated and cooled reactors while one loop is operated with heavy water providing prototypical CANDU reactor conditions.

The maximum heat removal capacity from a pressure flask is approximately 200 kW, and operation with normal or hydrogen water chemistry is possible. The boron and lithium concentrations can be varied over a wide range, and it is also possible to operate with controlled additions of water impurities such as chromium, zinc, sulphuric acid etc. The loop systems are used both for fuel testing as well as for core material studies, e.g. IASCC.

In addition, an out-of-pile loop is available for commissioning of instrumented fuel assemblies and for evaluating sensors for water chemistry monitoring under typical water reactor conditions.

The main circuit of a loop system has instrumentation for monitoring the water temperature, pressure and flow. The operating conditions are automatically controlled within specified limits by means of a Digital Logic Control (DLC) Unit.

For loop water purification, a controlled flow is continuously drawn from the main circuit through the high pressure purification circuit, which includes a particle filter and a mixed bed ion exchanger. PWR loops will have two additional ion exchangers for control of the boron and lithium content of the water. A particle filter is also installed after the ion exchangers.

The required loop pressure is obtained by helium pressurizing of a tank partly filled with water, connected to the loop main circuit. The pressure is controlled and automatically kept stable by a programmable logic controller (PLC).

Each loop has a feed water system with continuous circulation through a filter and mixed bed for purification of the water and for adding controlled amounts of hydrogen, boron, lithium and other additives to the water, if required.

Each loop has a sample line with continuous flow to the Water Chemistry Laboratory. Water conductivity, oxygen and hydrogen content are continuously monitored on this sample line. Analysis of sample water particle concentration and analysis of impurities are performed by inductively coupled plasma mass spectrometry (ICPMS), atomic absorption spectrometry, spectro photometry and capillary electrophoresis.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS

The experimental programme executed in the HBWR has included or includes fuels and cladding testing in dry-out and LOCA conditions. LOCA testing has utilized 3x3 rod bundles (fresh fuel with reduced diameter corner rods) and single rods with high burn-up fuel.

The on-going series of LOCA tests that started in 2003 with the commissioning of the test facility is using a single fuel rod (length up to 50 cm) inserted into a pressure flask connected to a water loop. A low level of nuclear power generation in the fuel rod simulates decay heat, whereas the electrical heater surrounding the rod simulates the heat from neighbour rods.

In general, the rod instrumentation consists of two cladding thermocouples at the upper part of the rod, a cladding thermocouple at the lower part, two heater thermocouples at different elevations, a cladding extensometer and a rod pressure sensor. The rig contains coolant thermocouples, a loop pressure sensor, and three axially distributed vanadium neutron detectors to measure the axial power distribution.

The loop has three modes of operation. Initially, forced circulation is maintained through the in-core part (pressure flask) and the entire loop. Prior to blow-down, the pressure flask is isolated from the rest of the loop, and the fuel rod is cooled by natural circulation in the pressure flask. To start the LOCA, valves in the line to the blow-down tank are opened. At end of blow-down, the pressure in the system is typically 2–3 bar due to non-condensable gases.

Changes like fuel rod heat-up, ballooning and burst go their course according to the dynamic behaviour of the system without interference by experimenters apart from injecting small amounts of water/steam during the high temperature phase after burst. This is done in order to maintain a sufficient amount of steam for the cladding oxidation.

The system can be re-filled at hot conditions (quenching).

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Investigations of corrosion of reactor materials are a standard part of the experimental programme in the HBWR. They comprise cladding corrosion studies and irradiation assisted stress corrosion cracking (IASCC) of reactor internals materials.

A corrosion experiments requires a rig inserted into a pressure flask which is connected to a loop producing the desired thermal-hydraulic conditions. Water chemistry control is an essential element of such testing, e.g., to provide oxygen or hydrogen water chemistry for IASCC studies or elevated pH in conjunction with high boron and lithium content of the coolant in conjunction with cladding corrosion investigations.

For cladding corrosion experiments, test materials can have the shape of strips, plates, curved coupons and closed tubes with fuel inside. For IASCC experiments, a common specimen type is the Compact Tension (CT) specimen with width 16 mm and nominal thickness 5 mm. The specimens are equipped with small pressurized bellows which enable the applied stress intensity to be varied during testing. Crack growth is measured in-pile with the potential drop technique and both internal and external current and voltage probe attachments may be employed.

The specimens are prepared from unirradiated and irradiated stainless steel and Ni-based core component materials. They are used to evaluate the effects of applied stress intensity and water chemistry on rates of crack growth.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENT

Encapsulated specimens are employed for dry irradiation of materials at controlled temperatures. The desired temperature level can be achieved in various ways, e.g., passively by gamma heating in combination with an insulating gap surrounding the specimen. The achievable temperature is determined by the gap size, the fill gas (He/Ar mixture) and the gamma heating.

An active control of the temperature level uses gas lines into and out of the capsule in order to adjust the gas mixture and thus to adjust the temperature, or electrical heating of the filler bodies around the test specimens.

If required, capsules can contain actively loaded specimens as described for specimens used in IASCC, stress relaxation and creep studies.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS

Various well-proven methods are available for studying fuel behaviour with respect to swelling, densification, creep, gas release and pellet-clad interaction. Standard fuel behaviour testing uses instrumented fuel rods starting with fresh fuel or re-fabricated and instrumented pre-irradiated fuel segments from commercial nuclear power stations. The rods are mounted in an irradiation rig and inserted into the reactor core. Typical instrumentation comprises fuel centre thermocouple, fuel stack elongation detector, cladding elongation detector, and rod pressure sensor. The instruments provide data as indicated in the general overview Section 2.1.

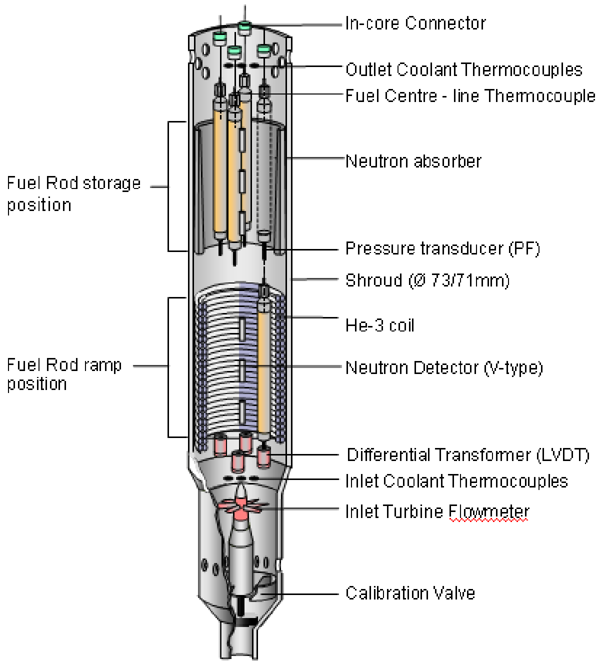

FIG. 4. Examples of irradiation rig for ramp testing.

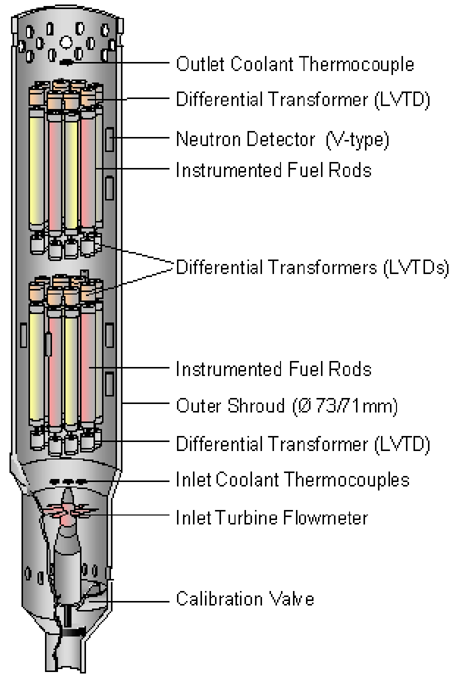

FIG. 5. Irradiation of instrumented fuel rods.

Specialized devices/rigs for addressing certain phenomena separately are available. They are used for studies on:

(a) *Cladding creep under variable loading conditions;*
This type of testing requires a rig where the fuel rods can be subjected to varying internal pressure such that rod under- and over-pressure and thus inwards and outwards creep can be obtained. The rods are connected to a gas pressurisation system that can deliver up to 600 bar pressure. The creep response of the cladding to the pressure is measured with a moveable diameter gauge scanning the cladding surface along three lines. The movement of the gauge is effected by a hydraulic drive system.

(b) *Rod overpressure and clad lift-off;*
The objectives of this type of test are to determine the maximum pressure above system pressure to which fuel rods of different designs and types of fuel can be operated without causing lasting/continuous fuel temperature. To this end, the fuel rod is connected to the high pressure gas supply system which can produce more than 400 bar overpressure in PWR conditions. In addition to measuring the fuel temperature, the cladding elongation is also monitored allowing the state of PCMI during the test to be determined. The tests are also designed to produce data on axial gas communication within high burn-up fuels. The influence of filler gas (Ar/He) and gas pressure on steady state and dynamic fuel thermal response can also be studied.

(c) *Fuel creep;*
For this phenomenon, the separate influences of fission rate and temperature on fuel creep are of interest. The proven experimental setup employs fuel disks between refractory metal disks which produces a relatively uniform temperature in the fuel disks. A variable load is applied with bellows pressing on the stack of disks. The pressure in the bellows and thus the load can be controlled and changed from the outside. The creep is measured on-line with a fuel stack elongation detector.

(d) *Gas flow and fission gas analysis;*

Using the gas flow system, radioactive krypton and xenon isotopes can be swept out of fuel rods during operation and collected in cold traps. The analysis of their decay provides information on the effective diffusion coefficient and on the surface-to-volume ratio of the fuel being studied.

Structural materials (stainless steels, nickel-base alloys) are studied under prototypical conditions in irradiation rigs where the load on the specimens can be controlled and creep and failure can be measured on-line. Typical investigations include:

(e) *Creep and stress relaxation;*

Tensile specimens are housed in test units that are equipped with bellows for load application, gas lines for on-line temperature control and LVDTs for specimen elongation measurements. The specimens may be prepared from austenitic stainless steels, Ni based alloys or Zircaloy. Creep is determined by measuring sample elongation under constant stress conditions while stress relaxation data are obtained when constant displacement is maintained on the specimens by reducing the applied stress on-line.

(f) *Susceptibility to crack initiation;*

Miniature tensile specimens, which are prepared from unirradiated and irradiated stainless steel or Ni based core component materials are used to measure the susceptibility to crack initiation under various operating conditions. The small diameter of the specimens ensures failure soon after stress corrosion crack initiation, thus providing integrated time-to-failure (i.e. initiation, propagation and failure) data.

2.7. OTHER FACILITIES

A feasibility study has shown that it is possible to install instrumented supercritical water loops into the Halden reactor (up to 250 bar, 600°C) for materials and fuel studies. Work on the design of such a loop is on-going. The external loop system will be similar to a PWR-loop system and with possibility for hydrogen addition. For materials studies, a flow rate of 0.1 kg/s is sufficient, while for fuel irradiations a flow rate of 1 kg/s is required. The useable inner diameter of the in-pile section will be 35 to 43 mm.

Instruments such as electrochemical potential sensors and linear voltage differential transducers (LVDT) for supercritical water conditions are presently under development. An LVDT suitable for liquid NaK and supercritical water with operating temperature up to 700°C and 250 bar pressure has been tested successfully.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

The operation of the Halden reactor comprises more than 50 years of experience with design, fabrication, irradiation and examination of fuels and materials. Transportation and interim storage are covered as well.

Important components for fuels and materials testing, in addition to the HBWR, are:

- Workshop for design and fabrication of irradiation devices;
- Instrument development, testing and qualification;
- Hot laboratory for fuel fabrication and re-fabrication, post irradiation examination;
- Neutron radiography;
- Storage space for spent fuel;
- Transportation of radioactive fuels and materials.

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

(a) *UO_2-Pellet production;*
IFE's hot laboratory has a complete line for UO_2-pellet production. It includes equipment for powder control, milling and blending. Sixty ton hydraulic powder press and sintering furnaces are available for pellet fabrication. For geometrical pellet adjustments, a centreless grinding machine and tools for chamfering, drilling, etc. are used. Fuel rod assembly is performed in pressurization chambers utilizing electron beam and TIG-welding equipment. The laboratory is equipped for assembly of MOX-containing fuel rods and drilling of holes for TF in MOX pellets, but not for production of MOX-fuel pellets (this capability is currently being implemented). UO_2 with enrichment up to 20% can be handled.

(b) *Transportation;*
Transportation between the Halden reactor and the hot laboratory at IFE-Kjeller is done on a routine basis using casks owned by IFE. Cost-effective transportation is also established between IFE's sites at Kjeller and Halden and the Swedish Studsvik research centre. The co-operation with Studsvik gives access to a wide range of irradiated fuels and materials.

3.2. HOT CELLS, PIE FACILITIES

The IFE Material Technology Department with its hot laboratory is equipped to undertake most types of examination work, i.e., dismantling, visual inspection, photography, dimensional measurements, isotopic analysis, gamma scanning, fission gas analysis, pressure measurements, location of leakage in case of fuel failure, burst tests of irradiated canning tubes, metallography, and neutron radiography.

The department comprises a nuclear fuel/hot laboratory, an electron beam welding laboratory and a mechanical workshop. The nuclear fuel/hot laboratory is supported by an analytical chemistry laboratory performing fission gas analysis, burn-up analysis, and various chemical analyses. The services at the hot laboratory include:

— Post irradiation examination (PIE) of fuel rods, including non-destructive and destructive testing, neutron radiography, mechanical testing, chemical analysis, optical and electron microscopy, electronic image treatment;
— Re-fabrication and instrumentation of irradiated fuel rods with remotely handled cutting, drilling, machining, grinding and welding operations;
— UO_2 pellet production.

The laboratory has three concrete hot cells and several lead shielded alpha tight cells. The cells are located in a row, with partition walls movable by remote control. The hot laboratory has auxiliary installations such as an unloading bay for shipping flasks, storage pits, decontamination and maintenance rooms for active components, etc. The laboratory is furnished with partly shielded equipment for tensile testing (ring shaped samples), hardness measurement, and active and inactive SEM with EDX analyser.

FIG. 5. Shielded cells for re-fabrication of irradiated fuels and post-irradiation examinations.

(a) *Post irradiation examination (PIE) of fuel rods;*

The hot cells are used for visual inspection, profilometry, γ-scanning, eddy-current testing, puncturing for fission gas analysis, burst testing, stress corrosion testing, sampling for burn-up tests and retained fission gas analysis, residual fuel/clad gap measurements, and density measurements.

Neutron radiography of irradiated fuel rods is done using the JEEP II reactor on the IFE-Kjeller site as neutron source.

Lead cells are used for cutting, defueling of irradiated rods, ultrasonic cleaning of defueled cladding samples, metallography/ceramography, optical microscopy, α and β/γ autoradiography, preparation of fuel replica for SEM, and specimen preparation for retained fission gas analysis.

Retained fission gas analysis and burn-up analysis are done by mass spectrometry. The concentration of hydrogen in cladding samples is determined by hot vacuum extraction.

The hot cell-periscope and all microscopes can be attached to an electronic quantitative image analyser. The laboratory is fully equipped for electronic image treatment and image storage.

(b) *Re-fabrication and instrumentation of irradiated fuel rods;*

The hot cells are also used for instrumentation of irradiated fuel rods for tasks such as drilling holes in irradiated fuel pellets for centre fuel-thermocouples, mounting the thermocouples and welding the end plugs, welding of instrumented end plugs containing e.g. a bellow fission gas pressure transducers, or attachment of cladding extensometers. Re-fabrication of irradiated fuel rods may include operations such as refilling of pre-irradiated cladding with fresh pellets, mounting of plenum spring and instrumented end plug assembly, etc. The services include rod refilling with He (pressurizing), welding of vent hole, and He leak testing.

The different fuel rod instruments developed at the Halden Reactor can be attached to both un-irradiated and pre-irradiated fuel rods. This means that it is possible to retrieve pre-irradiated fuel rods from power reactors and re-instrument them before the irradiation is continued in the HBWR.

To be able to install a re-instrumented fuel rod with a thermocouple into a test rig, an in-core connector has been developed. The in-core connector connects the cable from the fuel rod (thermocouple) to the designated cable in the test rig. The in-core connector can be used in PWR conditions.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS

IFE's division for *Test Rig Design and Production* is taking care of the development of experiments before their irradiation in the HBWR. The division's multi-disciplinary, highly skilled staff has unique experience and competence in:

- Design and fabrication of experimental rigs;
- Design, fabrication and testing of in-core instrumentation;
- Re-instrumentation of pre-irradiated fuels and materials in co-operation with hot lab staff.

The division's advanced workshop facilities include CNC-machines (lathes, 5-axis milling machine, wire electrical discharge machine), TIG welding, electron beam and laser welding equipment. For the development, testing and qualification of instrumentation, advanced test measuring equipment and autoclaves are available.

The rig instrument production and cables installation follow strict quality control procedures, including calibration and checking during subsequent production phases. At the end of fabrication and prior to loading in the reactor, each rig undergoes a test in an out-of-pile hot loop at water temperature and pressure conditions comparable to the ones specified for the in-reactor operation.

While primarily intended for serving the experimental programme in the Halden reactor, the division for *Test Rig Design and Production* also delivers advanced instruments, equipment and test rigs for other research institutes and customers.

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS

Publications can be searched in http://www.ife.no/en/publications.

An excerpt of recent years' publications relevant to fuels and materials testing in current reactors and Gen IV reactors is given below (see section Bybliography).

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Profile 17

MARIA REACTOR
POLAND

1. GENERAL INFORMATION

The MARIA reactor is Poland's second research nuclear reactor and the only one still in use. It is located at Świerk-Otwock, near Warsaw and named in honour of MARIA Skłodowska-Curie. It is the only reactor of Polish design.

MARIA is a multifunctional research tool, with a notable application in production of radioisotopes, research with utilization of neutron beams, neutron therapy, and neutron activation analysis. It operates about 4000 hours annually, usually in blocks of 100 hours. The MARIA reactor is mainly used to produce radiopharmaceuticals, isotopes for physics research (mainly in the condensed matter physics) for industrial applications (e.g. neutron activation analyses), as well as for training group.

MARIA is a pool type reactor with a power of 20 to 30 MW (thermal). Despite it being a pool reactor, it contains channels (aluminium tubes) individually connected to the primary coolant. The water pool provides cooling for elements (e.g., fuel elements) that are not otherwise cooled, and also acts as radiation shielding. MARIA uses enriched uranium as fuel (80% enrichment in ^{235}U till year 1999; 36% since). The fuel elements and channels are vertical but arranged conically. Water and beryllium blocks serve as the moderator (70% and 30% of the moderation, respectively). Elements of boron carbide sheathed in aluminium are utilized for control, compensation, and safety. The use of beryllium blocks results in a comparatively large fuel lattice pitch, and consequently large volume for payload targets. There is also a graphite reflector (aluminium sheathed). MARIA reactor supplies a neutron flux of 4×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ (thermal neutrons) and 2×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ (fast neutrons). There are six horizontal channels for controlled use of neutron beams. There is also a window of lead-containing glass through which the core can be viewed. The reactor is housed in a sealed containment.

Following preparation, which started in 2004, MARIA was converted to use low-enriched uranium (LEU) fuel by 2012. With assistance from the US Department of Energy's National Nuclear Security Administration (NNSA), the MARIA civilian research reactor in Poland was converted from operation with highly enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel on 24 September 2012. NNSA assisted this reactor conversion through its Global Threat Reduction Initiative (GTRI), whose mission is to reduce and protect vulnerable nuclear and radiological materials located at civilian sites worldwide. The GTRI Reactor Conversion Programme was established in 2004 by NNSA as a continuation of the Reduced Enrichment for Research and Test Reactors (RERTR) Programme that was established by the Department of Energy in 1978. Argonne National Laboratory has provided technical support since the inception of RERTR and among its current GTRI responsibilities provides technical leadership for the conversion of foreign reactors to LEU fuel.

2. DESCRIPTION OF REACTOR MARIA CORE

The reactor core, fuel and loop channels, headers, and connections between the headers and fuel channels are all submerged in the pool under a layer of water ensuring sufficient radiation shielding above the core.

The characteristic design feature of the core is a conical arrangement of fuel channels. The fuel channels are situated in beryllium matrix made of blocks of 110 cm high and enclosed by a lateral reflector made of graphite blocks in aluminium cans. Some of the blocks contain horizontal holes for extraction of neutron beams from the reflector to the horizontal beam tubes penetrating the reactor lateral shield.

2.1. FUEL ELEMENTS

Fuel assembly contains six tubes with uranium enriched to 36% ^{235}U. Fuel assemblies are placed in the pressurized channels. Each channel is individually connected to the primary cooling circuit.

In order to reduce the excess reactivity at the beginning of each cycle and to economize fuel, the reactor contains a number of movable fuel assemblies, which remain in fuel channel extensions and can be hoisted into the core using driving mechanisms mounted on the upper support plate. The operation of fuel hoisting may be performed during reactor operation at reduced power. This reactor feature reduces requirements for compensation system, facilitates control of neutron flux distribution in the core and improves neutron balance by eliminating useless neutron absorption in the compensating rods.

2.2. MODERATOR

The main moderating element in the core is water, which also provides cooling of fuel channels and core materials. Its volumetric fraction in the core is only 20%, but its contribution to neutron slowing down is 70%. The other moderating material is beryllium, constituting the matrix, which fills up the space between the fuel channels. The insertion of beryllium has made it possible to expand the lattice pitch according to the requirements of large loop experiments.

2.3. REFLECTOR

The reflector is assembled of graphite blocks canned in aluminium and cooled by water flowing in the separate pool cooling circuit. Besides basic graphite blocks the reflector contains multiple square blocks with beam holes forming the extensions of horizontal beam tubes, and aluminium blocks filling the reflector space around the horizontal channel passages.

2.4. CONTROL RODS

The control rod units are divided into two sections, connected with disconnectable joints. The upper sections are supported on the upper support plate above the water level and contain drive mechanisms; the lower sections, containing active portions of the rods as well as connections and control rod followers, move inside the tubes placed in 28 mm dia. holes in the beryllium matrix blocks. The control rods are cooled with pool water flowing down the core along the gaps in the control rod channels.

3. GENERAL DISCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

Research reactor MARIA is a multipurpose reactor and remains only one high flux research reactor in Poland it has been designed to operate with several experimental devices in parallel. The most important of them are:

— Vertical irradiation channels for radioisotope production;

— Reactor test rigs for structural materials and reactor fuel studies under stationary conditions;
— Horizontal experimental channels for neutron beam studies.

3.1. INFRASTRUCTURE FOR IRRADIATION OF TARGET

In the area of reactor core and reflector there is a number of vertical irradiation channels, made of aluminium tubes with conical enlargements in the upper part, facilitating loading of target cans into the channels. Almost all channels are so called long ones, i.e. channels reaching above the upper support plate.

The following channels are available:
— Channels of 23 mm diameter in Be blocks;
— Channels of 28 mm diameter in graphite reflector;
— Channels of 38 mm diameter in graphite reflector;
— Channels of 18 mm diameter inside modified fuel element;
— Channel located under safety rod;
— Channels equipped with hydraulic transport system.

The reactor core layout is shown in Fig. 2, gives an example of core management, which fulfils the needs of radioisotope production and several experimental programmes.

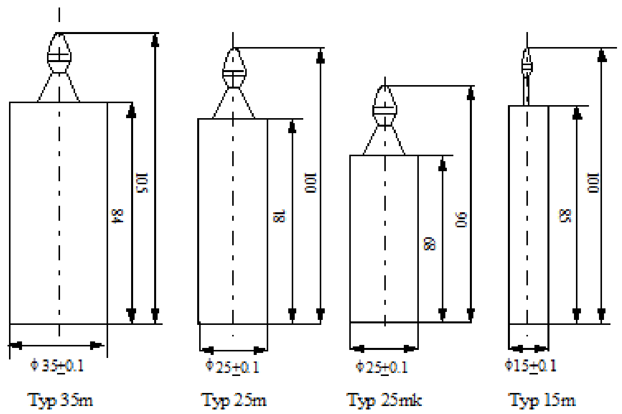
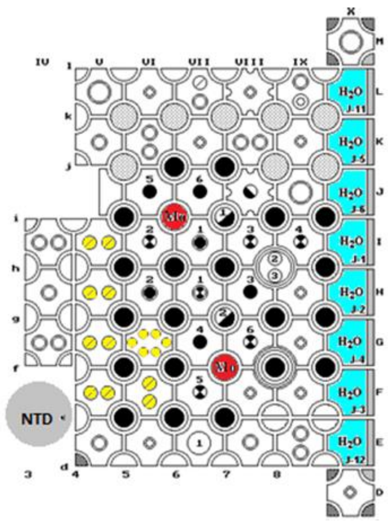


FIG. 1. Standard cans for irradiation target materials.



The yellow color of channels position — irradiation position;
The red position — molybdenum rigs.

FIG. 2. Reactor core layout.

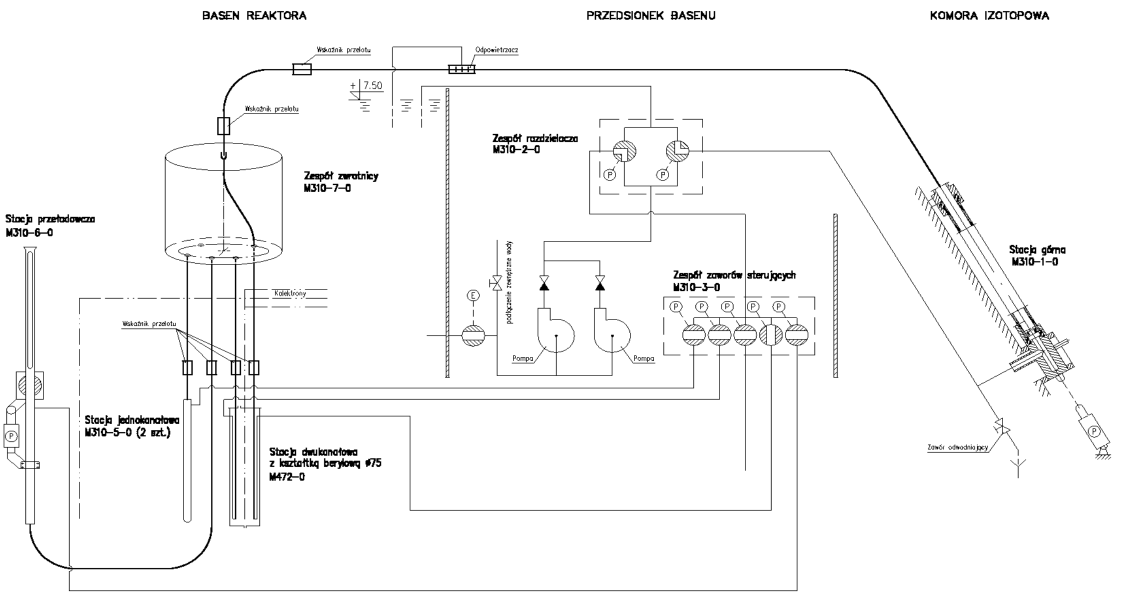


FIG. 3. Irradiation channel equipped with hydraulic system.

3.2. HOT CELLS FOR RADIOISOTOPE PRODUCTION

There are two hot cells for radioisotope production: one for 1000 Ci of ^{60}Co activities (closer to the reactor pool) and one for 100 Ci activity (further from the reactor pool). The access to the hot cells is only from above, through the opening covered with shielding blocks, but the radionuclides are transported with a loading machine reaching into the cells through curvilinear channels. Below the cells there is a transport corridor for inter-cell transport and for transport outside the reactor.

Besides these two hot cells the reactor is provided with a dismantling cell for spent fuel element handling, with heavy shielding allowing handling elements of 20 kCi of ^{60}Co activity.

3.3. REACTOR FACILITY FOR NEUTRON TRANSMUTATION DOPING OF SILICON SINGLE CRYSTALS

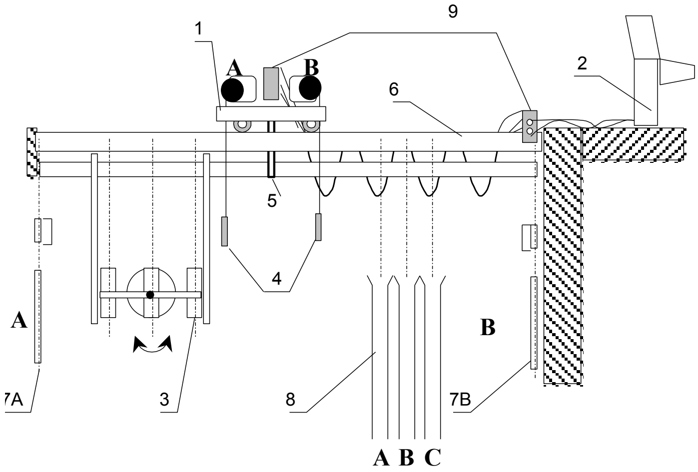
The innovation of technical solution is featured by advantage well-known formula of neutron transmutation doping and implementary it at production facility where specific physical characteristics of MARIA research reactor have been adopted. The vertical neutron flux distribution to be available in graphite reflector together with additional shaping by means of titanium absorber screens has been adopted for achieving high degree of axial and radial doping distribution uniformity. The above solution allowed to achieve the simplest technical construction and very high quality level of doped silicon crystals.

In case of large silicon diameters (> 5 inches) to be subjected to neutron transmutation doping (NTD) the most crucial parameter is the doping uniformity. In the accepted and realized doping conceptual outline the requirements for radial uniformity of doping was achieved by rotating the silicon container around the channel axis, however, the demand of axial uniformity — by the bipositional irradiation cycle with additional shaping of thermal neutron field by means of specially formed absorption screens. It is an innovating solution enabling to achieve high doping uniformity. The height of the irradiated crystals is 2 × 250 mm.

The irradiation device is composing of two parts: installation for irradiation located in reactor pool under base plate of the reactor and outside pool — the part for the preparation and reception of the silicon crystal. There are three channels: A, B, C for silicon irradiation. The dimension of each irradiated crystals is six inches. The channel is constructed with aluminium casing and rotated tube. The container with silicon crystal is loading to the channel by the hoist with lifting magnet bucket. The hoist is installed on movable car together with the electric components. The motion of the rotated is causing by pneumatic motor and gear transmission.

TABLE 1. MAIN PHYSICAL AND TECHNICAL DATA OF THE MARIA REACTOR

Parameter	Description
Objective	NTD of silicon (*neutron transmutation doping of single crystals)*
Irradiation material	Single silicon crystals
Position	Channel located in graphite reflector zone
Name of hole or facility	NTD-1
Thermal neutron flux (with or without target)	$6.0 \times \cdot 10^{12}$ $n \cdot cm^{-2} \cdot s^{-1}$
Fast neutron flux	$4.5 \times \cdot 10^{10}$ $n \cdot cm^{-2} \cdot s^{-1}$
Gamma heat generation	<0.1 W/g
Longitudinal length of hole or facility	500 mm
Diameter or size	157 mm
Diameter of the accessible irradiated material	5 inches or 6 inches
Way of uniform irradiation	Neutron flux flattering: rotation plus flipping with flux profile linearization
Way of reduction for background radiation	Well moderated neutron spectrum in graphite reflector
Irradiation time (max., min. and average)	Dependent on target resistivity (7.5 Ωcm–100 Ωcm)
Way of rotation	Electric drive
Way of cooling for irradiation object	Forced convection by pool cooling flow
Way of monitoring the irradiation fluence	SPND — detector system
Way of decontamination	Ultrasonic bath
Way of residual radioactivity survey	Gamma & beta ray counter
Exemption criteria for residual radioactivity	^{32}P: <2.5 Bq/g
Cadmium ratio for gold or other material	Epithermal index r = 0.02
Capacity per year	6200 kg/y (for ingots diameter 5 in) or 9500 kg/y (for ingots diameter 6 in)
Present status	Ready for operation



1 - car of container transportation device;
3 - turn-around table;
8 - irradiation channels with rotation drive (3-channels: A, B, C).

FIG. 4. Three channels NTD of silicon facility during out reactor tests.

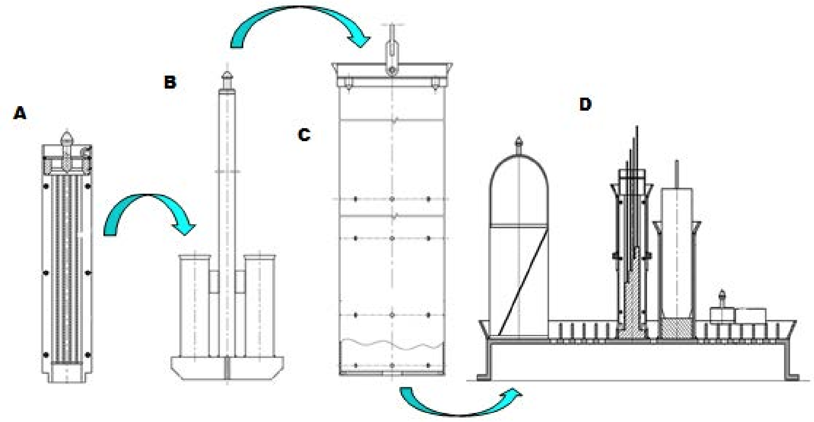
3.4. FACILITY FOR IRRADIATION URANIUM TARGETS FOR ^{99}MO PRODUCTION

Receiving the ^{99}Mo form the fission products is one of the most effective methods for gaining this isotope. It enables to gain the isotope of high specific activity, which is a very important property from the viewpoint over the production molybdenum-technetium generators commonly used in on oncological diagnostics.

The technology proposed to irradiate uranium targets in the reactor in relatively simple way and at same time a very effective utilizes from one side the MARIA reactor structured possibilities and from the other side comes up to the fundamental relationships between the fission power generated in the targets and the activity of the created ^{99}Mo to be one of the ^{235}U fission products.

The technology implemented in MARIA reactor employs a specific construction of fuel channels with the cooling system, mechanical hardware and measuring system. Linked to the measuring system of thermal-hydraulic parameters of the reactor fuel elements.

Irradiation of uranium plates set takes place in the adopted for this goal fuel element of MARIA reactor. An adaptation of the reactor fuel channel to irradiate uranium plates relies on such its modification to provide multifold loading and discharging of uranium sets with targets into and out of installation without necessity of evacuation of the whole molybdenum channel from the position socket in the reactor core. The construction of the internal fuel structure allows after reactor shutdown and removing the fuel channel head to take out and transportation of the irradiation container together with uranium targets beyond the hoop of reactor core. It is in line with a standard procedure to be binding for the reactor fuel channel. The obligatory technology of uranium target's irradiation gives the possibility to use the standard reactor channel cooling system for removal of the heat from molybdenum channels as well as all reactor control systems.



(A) – irradiation container in molybdenum channel;
(B) – two-positional transport container;
(C) – inside pool transport facility;
(D) – reloading stand inside disassembling (hot) cell.

FIG. 5. Procedure of reloading operation with irradiated uranium targets in MARIA reactor.

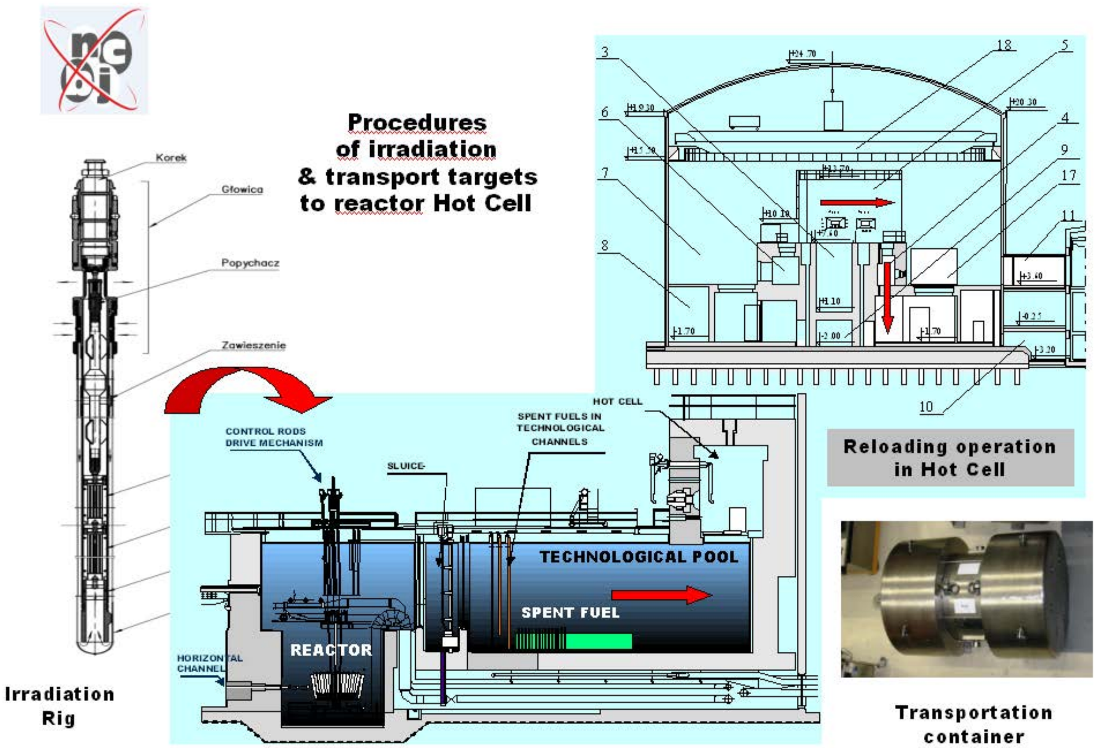


FIG. 6. Technology irradiation uranium targets (HEU) for ^{99}Mo production.

3.5. EXPERIMENTAL FACILITIES FOR FAST NEUTRON IRRADIATION

The MARIA reactor is equipped with out-of-core irradiation facility enable to irradiate the large-size (up to 90 mm in diameter) target materials/devices with fast neutron flux density up to 1.7×10^{12} $n{\cdot}cm^{-2}{\cdot}s^{-1}$ and well reduced thermal neutron flux limited down to

3.4×10^{10} n·cm^{-2}·s^{-1}. The facility consists of four assemblies with four irradiation channels each of them and is currently used to minerals irradiation.

Purpose-build channel for irradiation by 14 MeV neutrons is prepared to be installed in MARIA reactor core. The facility consists of thermal to 14 MeV neutron converter based on lithium-6 and deuterium compounds. The principle of its operation based on two-step reactions induced by thermal neutrons. It is expected to achieve the 14 MeV neutron flux density up to 5×10^{10} n·cm^{-2}·s^{-1} inside the converter with cut-off thermal neutrons at the same time. The neutron energy spectrum depending on converter location in reactor core and is similar to one in thermonuclear devices. Its loading capacity is ca. 60 cm^3 (four containers, Ø 15 mm × 90 mm each). The converter is devoted to material research, i.e. irradiation tests of new structure materials for generation IV nuclear reactors and fusion facilities. It can be also used to obtain radiopharmaceuticals produced by fast neutrons in threshold reactions.

It is predict to construct new in-pile fast neutron irradiation facility. It is expected to achieve fast neutron flux density over 1×10^{14} n·cm^{-2}·s^{-1} (fission spectrum) with thermal neutron flux density limited down to 3×10^{14} n·cm^{-2}·s^{-1}. The irradiation channel should be ca. 30–45 mm in diameter. It could be used to material research and produce radiopharmaceuticals in reactions with fast neutrons, e.g. ^{47}Sc, ^{89}Sr.

3.6. HORIZONTAL NEUTRON BEAM CHANNELS

Reactor is equipped with six horizontal channels for neutron beams with an output thermal neutron flux of order 3–5 10^{13} n·cm^{-2}·s^{-1}. The horizontal channels for investigations of internal structure of condensed matter are operated by Regional Laboratory of Neutronography NCBJ.

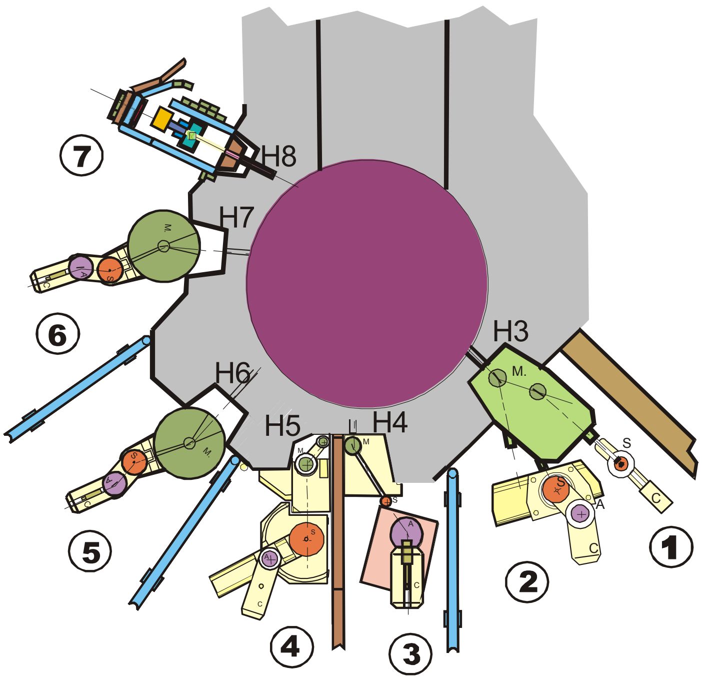


FIG. 7. Horizontal neutron beam channels.

FIG. 8. The base instrumentation of H3-H8 channels.

Profile 18

TRIGA II PITESTI-SS CORE

ROMANIA

INSTITUTE FOR NUCLEAR RESEARCH, PITESTI-14MW TRIGA RR AND ACPR-TRIGA-RESEARCH REACTORS[1]

INTRODUCTION

The 14MW TRIGA research reactor was commissioned in 1980 being extensively used for materials and nuclear fuel testing. Through Institute continuous activity and support of IAEA and DOE NNSI the reactor core was full converted from HEU-93.3 enriched to LEU — in May 2006 and HEU spent fuel was shipped in the country of origin, last shipment in 2008 so the issues of non-proliferation is not any more of concern for Institute.

A large project of modernization of safety and control system and of other systems of reactor physical protection system upgrading due to of ageing and obsolescence was achieved between 2006 and 2009 so the issues of safety, security and availability of reactor is not any more of concern for Institute.

The 14 MW TRIGA RR and ACPR RR are currently licensed by Romanian Regulatory Authority — CNCAN.

The reactor operation is subject to an Integrated Quality Management System.

By initial design 14MW TRIGA RR-SSR core and ACPR core was designed for nuclear fuel and materials testing for nuclear energy development in Romania.

The Institute structure was designed to ensure research, design, and construction of experimental equipment for nuclear reactors utilization. The experience of 33 years of operation of research infrastructure of institute is available for new projects in the area of innovative systems for energy. The international cooperation activity of institute in area of nuclear energy and medical radioisotopes is a part of efforts for reactor utilization.

Some recent activities in this area concern the Institute participation in competition for material irradiation for future ITER design, MAXSSIMA — *M*ethodology, *A*nalysis and *eXperiments* for the '**S**afety *I*n *M*YRRHA *A*ssessment', CANDU6 nuclear fuel behaviour in power cycle regime, irradiation testing of neutron absorber element form ACR nuclear fuel, ALFRED-Advanced Lead Fast Reactor European Demonstrator, a potential GIV project sustained by a national research programme.

The capacity of research reactor utilization depend on sustainability of research infrastructure, availability of qualified human resources, preservation of nuclear knowledge and mix founding by state and international programmes promoted by IAEA and EU which foster the accessibility and intellectual properties share.

Recent update of TRIGA RR reactor strategic safety utilization plan outlines and analyses in details the above issues.

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1. GENERAL INFORMATIONS

1.1. TECHNICAL DATA OF RESEARCH REACTORS

1.1.1. Reactor type, short description of 14 MW TRIGA research reactor

Reactor pool type 14MWTRIGA research reactor contained in IAEA RRDB as TRIGA II PITESTI-SS Core, IAEA code RO0002. The reactor was commissioned and started the continuous operation from 1980, the reactor was fully converted from HEU to LEU in 2006 and refurbished in 2009 and planned operational life is estimated till 2030.

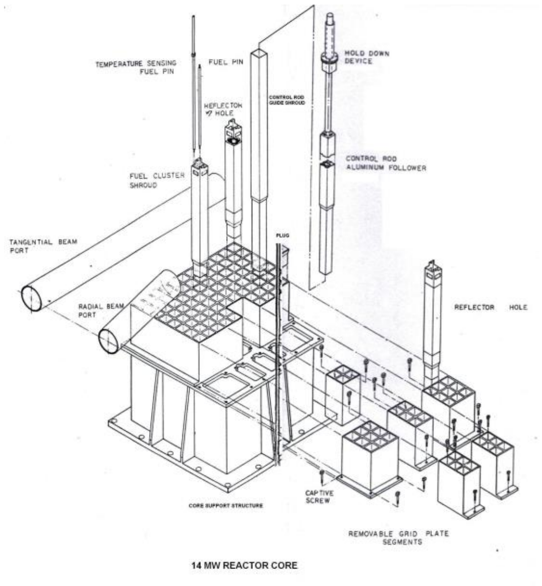
The design of core incorporate features which ensure maintainability, testability and inspectability for the lifetime of reactor. Fuel rods are manufactured from a hydrated alloy of uranium, zirconium and erbium, cladded in Incoloy 800 tubes with 13 mm outer diameter (O.D).

Fuel assemblies contains 25 fuel rods assembled in a 5 × 5 square lattice in a square Al 6061 tube with Inconell 600 spacers. The fuel assembly is provided with top and down cast Aluminium alloys fittings. The top fitting allow the handling of assembly and control the water flow. The fuel assembly and core elements are placed into lower reactor gird with a lattice pitch of 90 mm.

This design allows a large flexibility for core configuration and in core install of irradiation devices.

The fuel area is surrounded by vertical blocks of beryllium reflectors sustained also by reactor grid. The core contains also eight square guide tubes for reactor control rods sustained by reactor grid. All irradiations in core devices are arranged inside of aluminium alloy guide tubes and sustained from the top to prevent the inadvertent reactor grid loading. All other free spaces in core are covered with square plugs.

The reactor grid call for a cross beams design manufactured from cast aluminium and precisely machined. The spatial structure of reactor grid is bolted on top part of core shroud which sustain entire core. In order to allow the installation of in core experiments or irradiation devices, the reactor grid contains several removable segments bolted in the grid frame, see Figs 1 and 2.



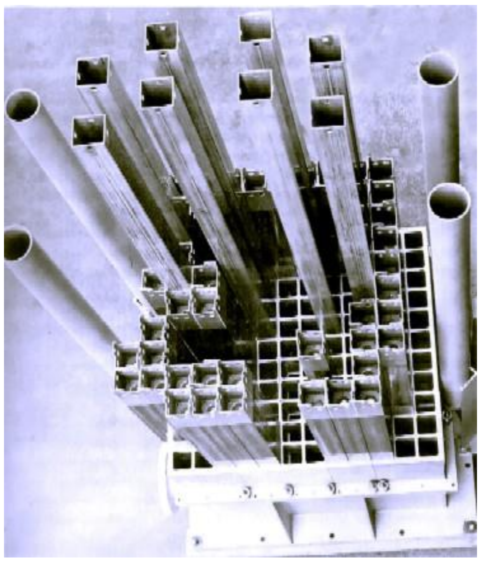


FIG. 1. Schematic reactor core structure . *FIG. 2. Reactor core structure.*

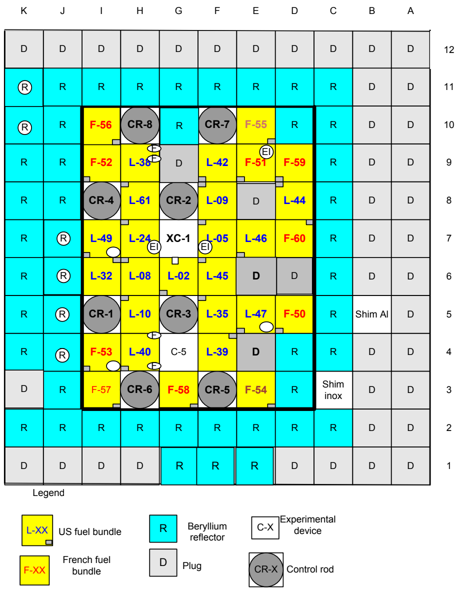


FIG. 3. Layout of TRIGA14MW reactor core.

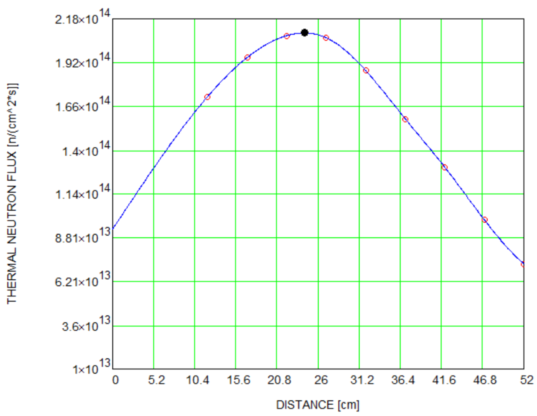


FIG. 4. Thermal neutron flux axial distribution in XC-1 channel (P = 11 MW).

The flux density per unit lethargy spectrum output of the SAND2 code for 621 energy groups in the XC-1 water filled irradiation channel is shown in Fig. 5.

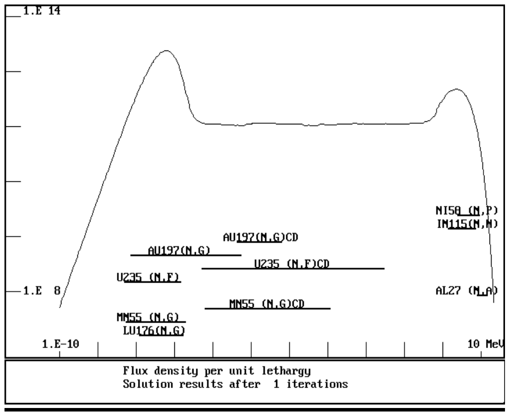


FIG. 5. Flux density per unit lethargy in XC-1 channel water filled.

Thermal and fast neutron flux spectrum was recently determined in the irradiation vertical channel XC1 at nominal power of reactor i.e.14 MW.

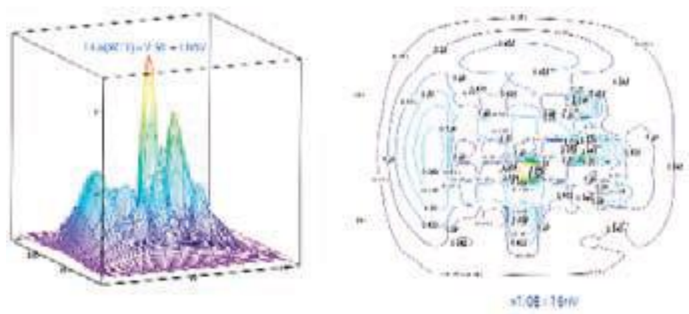


FIG. 6. Thermal flux distribution at 10 MW.

The absolute neutron flux-spectrum in XC-1 irradiation channel (water filled, see Fig. 5) is presented in Table 1.

TABLE1. FLUX SPECTRUM VALUES

Neutron energy range (MeV)	Neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)	Average energy (MeV)
10^{-10}–18	4.22×10^{14}	0.41
10^{-10}–5.5×10^{-7}	2.63×10^{14}	
5.5×10^{-7}–1.0	1.34×10^{14}	
1.0–18	6.89×10^{13}	2.46

The fast neutron contribution represents 16.33% from the integral flux density (E >1 MeV).

The Cd ratio for the Au reaction is 2.45.

Those data where measured with all irradiation position (vertical channels) filled with water i.e. non perturbed flux.

Following each core configuration and experimental loading (in core irradiation devices) the neutron fluxes for each irradiation device is computed and verified by in core measurements.

It worth mentioning that INR staff has professional abilities for the following:

— Neutron modelling and design of the irradiation conditions for in core material testing;
— Irradiation devices design, manufacture and instrumentation to conduct the experiment with the support of the on-line data acquisition;
— To ensure the irradiation environment according to the customer's requirements;
— To ensure an effective irradiation volume of 1.500 cm^3 with axial peak to average flux factor of 1.1–1.3.

1.1.2. Reactor type, description of ACPR reactor

Annular core pulsed reactor (ACPR) is contained in IAEA data base as TRIGA II PITESTI-Pulsed, IAEA code RO0004. The reactor is operated by design with LEU fuel. The reactor was commissioned and started the continuous operation from 1980 and planned operational life is estimated till 2030.

The ACPR is completely independent of the steady-state reactor with two exceptions:

(1) Both reactors share a common reactor pool; and
(2) Both reactors depend on the same cooling and water purify-cation systems.

The reactor utilizes a close-packed array of TRIGA stainless steel-clad $UZrH_{1.6}$ fuel-moderator elements, which contain 12% uranium enriched to 20% in U-235. The elements are arranged on a triangular spacing to form a hexagonal pattern of fuel surrounding the hexagonal experiment tube. The operational loading of the reactor comprises 153 fuel-moderator elements, 6 fuelled follower control/safety rods, and 3 transient rods. With this core size, the predicted energy release for a reactivity insertion of 6.85$ is about 100 MW-s with a reactor period of 1.2 ms.

The general arrangement of the Annular core pulse reactor (ACPR) is shown in Fig. 7. The main characteristics of TRIGA ACPR research reactor are presented in Table 2.

TABLE 2. TRIGA ACPR MAIN CHARACTERISTICS

Reactor type	Annular core pulsing reactor
Reactor vessel	4.5 m (width), 9 m (length), 10 m (depth), open pool
Fuel	Metal alloy 12% U-ZrH weight
Enrichment	19.75% U-235 weight
Clad material	Stainless steel
Pellet diameter	3.56 cm
Clad diameter (external)	3.76 cm
Active section length	38 cm
Number of fuel elements	146 + 6 fuel followers
Control rods	
Number	8
Type	With active section and fuel follower
Poison material	Natural B4C
Drive mechanism	With pinion and rack
Transient control rods	
Number	2 rods with fixed motion and 1 rod adjustable
Drive mechanism	Pneumatic cylinder for full stroke

Type	With air follower
Poison material	B_4C natural
Drive mechanism for transient rod	With pinion and rack, and pneumatic cylinder for partial stroke
Reactor power	
Steady state	500 kW maximum power
Pulsed	20.000 MW maximum power

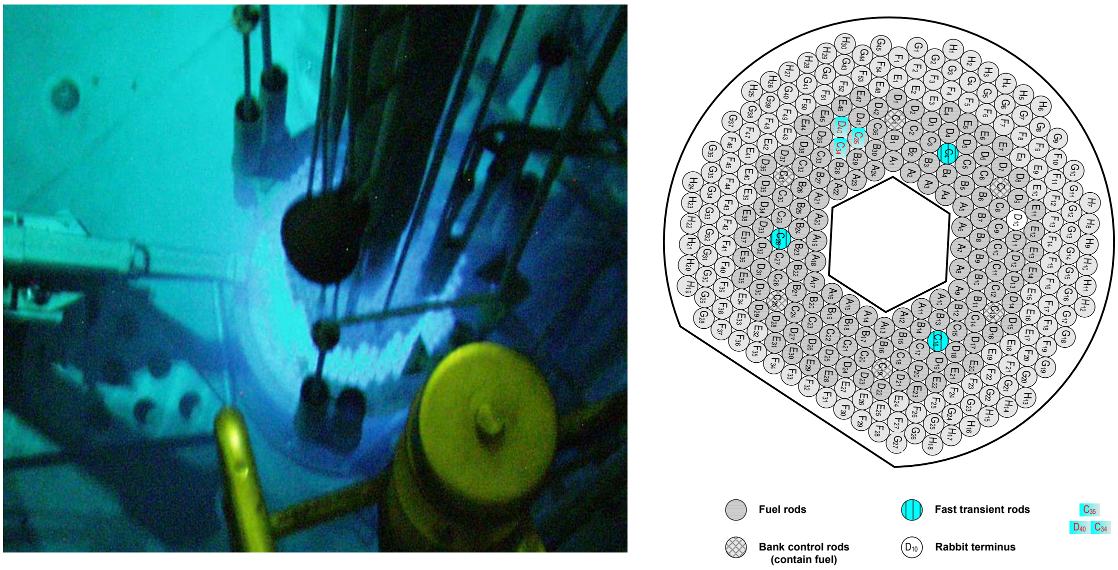


FIG. 7. ACPR reactor core in operation and core map.

1.2. TECHNICAL DATA OF POST IRRADIATION EXAMINATION LABORATORY (PIEL)

The INR PIEL was commissioned in 1984 being the newest and modern laboratory in Europe for fuel and material development research, study of in core irradiation effects, oriented to sustain new or recent stringent safety requirements for safe operation of existing nuclear power plant.

The INR PIEL is an essential part of Institute research infrastructure which was refurbished and modernized between 2000 and 2002 in order to answer to new demands on advance research tools concerning nuclear fuel and materials for GIV, materials for ITER under development, some issues of existing and future radioactive waste.

The PIEL play an essential roll in the irradiation activity being used for loading/unloading of experiments into in core part of irradiation devices. This operation has the following advantages:

— Interim examination of samples and fuel;
— Shuffling and/or replacing samples between irradiation campaign;
— Reducing to minimal the amount of medium radioactive waste per irradiation.

PIEL is also used for ^{192}Ir and ^{60}Co sealed sources production.

PIEL is located adjacent to the TRIGA reactor and is interconnected to the TRIGA reactor pool by a water canal. Thanks to this design, it is possible to examine in PIEL hot cells in short times samples irradiated of fuel from the reactor. The transit time between the two facilities is only a few minutes.

PIEs of experimental CANDU fuel elements manufactured by INR and irradiated in the 14 MW TRIGA reactor have been performed since 1984.

The basic PIEL infrastructure comprises:
— Two large heavy concrete hot cells;
— Three steel shielded hot cells; and
— One lead shielded hot cell.

The general arrangements of hot cells are presented in Fig. 8.

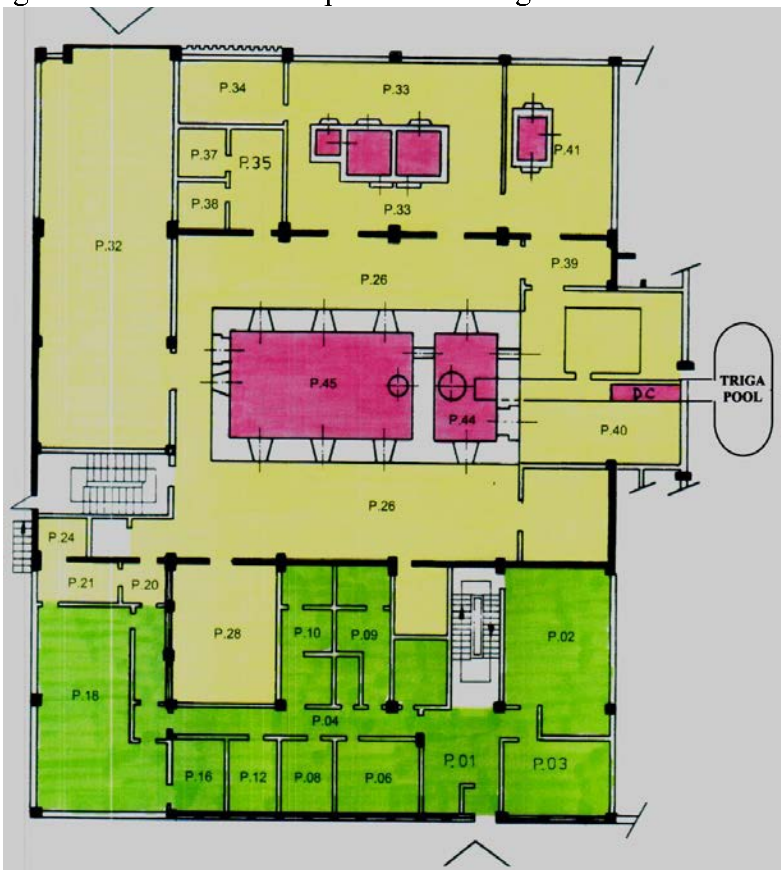


FIG. 8. General arrangements of PIEL.

TABLE 3. HOT CELLS CHARACTERISTICS

Hot cell type	Inside dimensions W×D×H (m)	Shielding wall thickness (mm)	Number of shielding windows	Maximum gamma activity (TBq)
Transfer cell	3.4 × 6 × 7	1100 (Heavy concrete: 3300 daN/m^3)	2	37×10^3
Examination cell	10 × 6 × 7	1100 (Heavy concrete: 3300 daN/m^3)	7	37×10^3
Metallographic cell	2 × 2 × 2	280 (Steel)	1	37
Microscope cell	1 × 1.2 × 0.8	200 (Steel)	1	3.7

Chemistry cell	2 × 2 × 2	280 (Steel)	1	37
Mechanical testing cell	2 × 2× 2	150 (100) (Lead)	2	3.7

The Institute PIEL use a licensed cask forged stainless steel for irradiated fuel bundle transportation from Cernavoda NPP to Institute INR Pitesti, and back.

2. EXISTING AND PROSPECTIVE FACILITIES

2.1. GENERAL DESCRIPTION AND TESTING FACILITIES

The experimental and testing facilities mainly in core irradiation devices procured from importation or designed, built and commissioned by Institute were used for irradiation of nuclear fuel and materials belonging to CANDU6 technology. Those devices are an irradiation loop and several capsules for parametric studies (see Table 4). The testing cover the entire range of steady-state conditions, transient conditions and accident conditions, detailed description is presented in the following section.

TABLE 4. IRRADIATION DEVICES AT THE INR-RR, PITESTI

	Reactor	Irradiation device
Normal conditions	14 MW SSR	– Capsule C1: on-line fission gas sweep analysis – Capsule C2: fuel internal clad pressure evolution and central temperature – Capsule C7: corrosion under irradiation – Capsule C9: load follow-up – Loop A: overpower
Transient conditions	14 MW SSR	Capsule C2: ND loop A ramp tests
Accident conditions	ACPR	Capsule C6: (RIA simulation)
Future plans in terms of accident conditions	14 MW SSR	– Modified capsule C2: for LOCA type accident, using fresh and irradiated fuel – Modified loop A: to simulate LOCA at PWRs
	ACPR	– Design a new irradiation device to allow testing of pre-irradiated fuel in RIA conditions – Designed and build a Pb-Bi capsule for MAXSSIMA project
Material irradiation	14 MW SSR	– Capsule C5: for structural materials – Loop A: pressure tube material

2.2. LOOP FOR TESTING NUCLEAR FUEL AND MATERIALS (LOOP A)

The irradiation facility loop A (see Figs 9 and 10) serves for investigating the behaviour of the CANDU type fuel at the INR-RR. Its main features are as follows:

— Total power 100 kW;
— Water flow rate at the samples 3–7 m^3/h;
— Maximum pressure in the primary circuit 135 bar;
— Maximum water temperature 310°C;

— Useful internal diameter 54 mm.

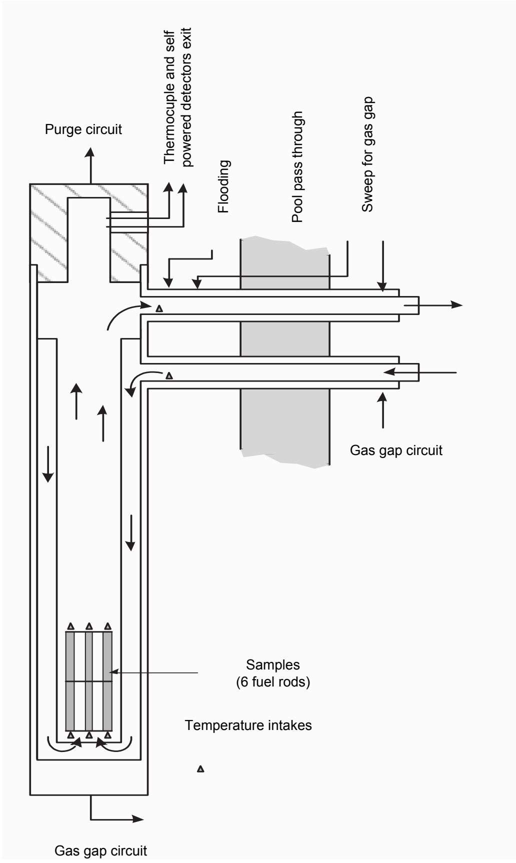


FIG. 9. The in-pile section of loop A.

The in-pile section of the loop consists of two main elements: the 3.80 m long tubular envelope and the sample holder basket fitted at the top with a plug for the passage of instrumentation cables. Except at the top, the in-pile section is double walled and the two tubes, which form the wall, confine a low pressure gas. The out-of-pile section mainly includes a primary circuit to maintain nominal fuel rod cooling conditions (pressure, temperature, flow rate, water quality), a gas circuit to vary heat transfer, and an effluent circuit to collect drainage from the primary circuit and either send it to the effluent tanks after condensation or process it before evacuation. The out-of-pile section is made of AISI304 stainless steel and is located in a tight cylindrical containment shielded with 150 mm lead walls (35 t) to ensure the protection of its surroundings in case of an experimental clad rupture.

Loop A allows the performance of overpower tests as well as ramp tests, for three or six fuel rods simultaneously (see Figs 9 and 10). Table 5 summarizes the history of the experiments performed.

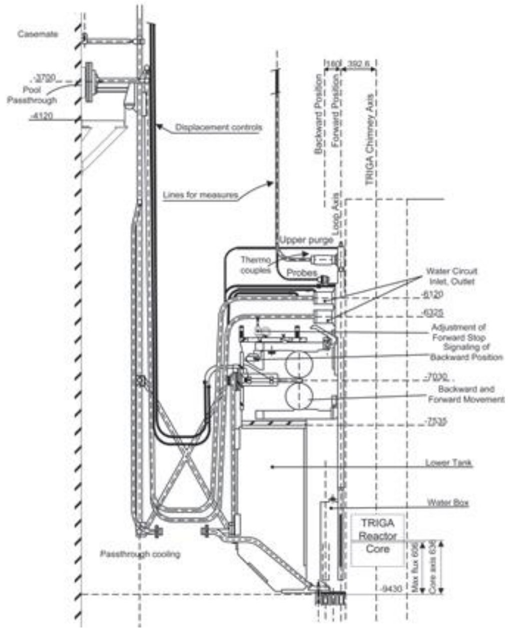


FIG. 10. The in-pile section of Loop A.

The high pressure primary circuit of loop A (see Fig. 11) contains the elements of a power reactor primary circuit at reduced scale, allowing the simulation of thermal hydraulic parameters, a specified forced flow across the in-pile section and the water chemistry established through demineralizing bad resins , water purification by mixed bed ion exchange filters.

TABLE 5. IRRADIATION TESTS PERFORMED IN LOOP A

	Irradiation test	Irradiated nuclear fuel
1.	Overpower test	EC10, EC 11, EC12
2.	Overpower test	EC35, EC36, EC37
3.	Power ramp test	EC15, EC16, EC18, EC19, EC76, EC77
4.	Power ramp multiple test	EC80, EC81, EC82, EC84, EC85, EC86
5.	Power ramp multiple test	ECA01, ECA02, ECA03, ECA04, ECA05, ECA06
6.	High burn – up power ramp multiple test	ECA28, ECA30, ECA34, TR1, TR2, TR3
7.	Cernavoda NPP Zr – 2.5% Nb pressure tube samples irradiation	– Corrosion – 48 samples: 24 in flux, 24 out of flux – Mechanical properties – 24 samples: 12 in flux, 12 out of flux

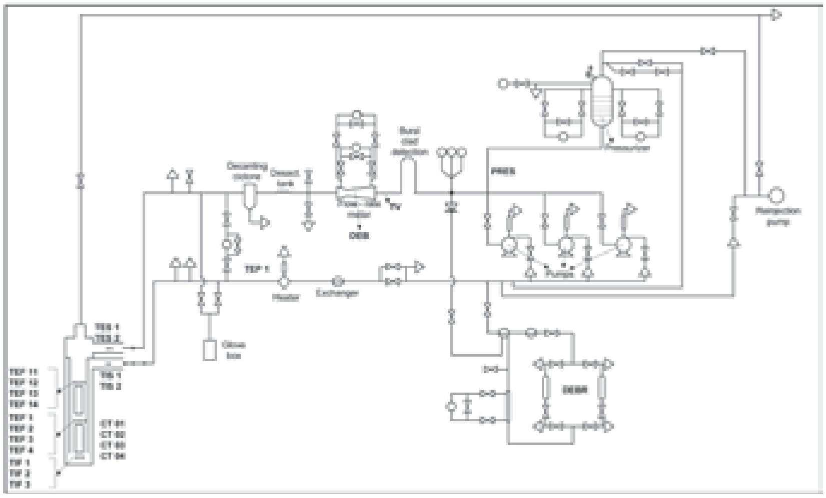


FIG. 11. Loop A flow scheme.

2.3. CAPSULES C1 AND C2

Capsules C1 and C2 are almost similar irradiation devices designed to meet the main parameters, as follows:

- Maximum pressure 150 bar;
- Maximum outer fuel temperature 330°C;
- Cooling water flow rate 300 L/h;
- Purification flow rate 2.5 L/h;
- Useful inner diameter 29.5 mm;
- Maximum fuel element diameter 15 mm.

Capsules C1 and C2 allow the performance of instrumented fuel rod irradiation in order to:

- Measure the central temperature of one fuel rod in correlation with several ranges of power and burn up (see Figs 12 and 13);
- Measure the internal fission gas pressure evolution versus varying burn up and power originating from power ramps;
- Measure the composition of fission gas release during irradiation (see Table 6);
- Determine the elongation of one fuel rod during irradiation.

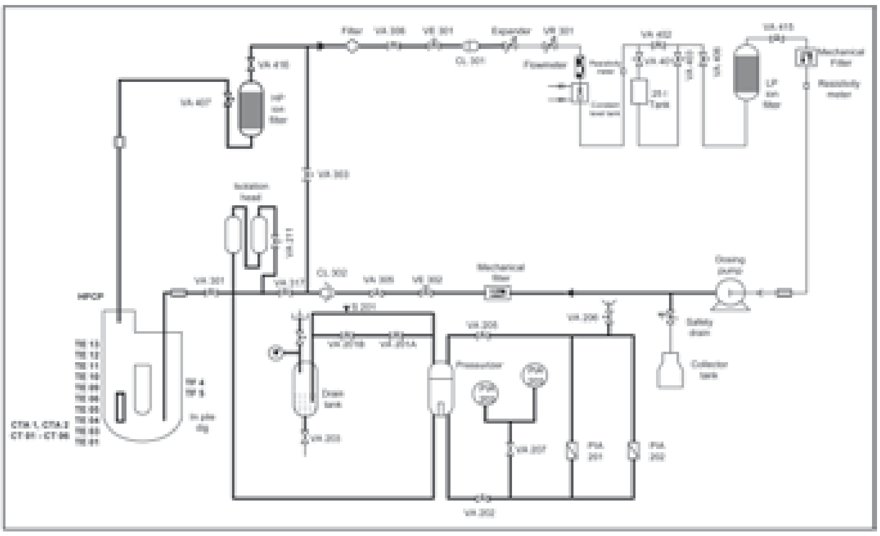


FIG. 12. Capsule C1/C2 flow scheme.

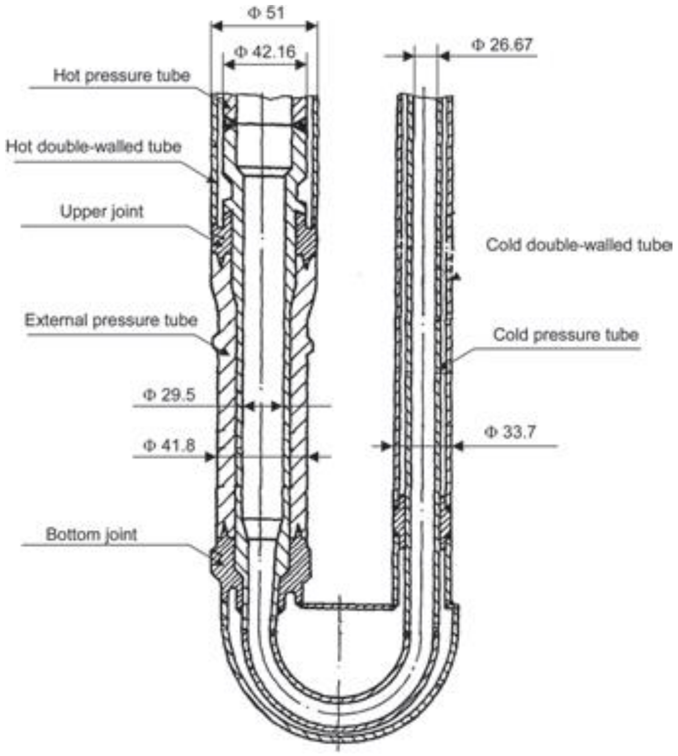


FIG. 13. Capsule C1/C2 in-pile section.

Each type of experiment performed in the capsules is under natural convection cooling inside the in-pile section. The heat generated by one experimental fuel rod, i.e. 35 kW, is dissipated to the pool through the external pressure tube. The out-of-pile section is shielded using lead bricks together with remote control valves. For each type of test listed in Table 6, dedicated instrumentation of the fuel element and the in-pile part was installed to collect the test data during irradiation.

2.4. CAPSULE C5

Capsule C5 was designed and used for the irradiation of steel samples and zirconium alloys in helium atmosphere at temperatures between 200 and 300°C (see Fig. 14). Table 6 contains the experiments performed using C5. The C5 in-pile section was designed to irradiate several types of samples of cladding as well as pressure tubes. The specified test temperature is obtained by gamma heating and/or heating by an electrical furnace.

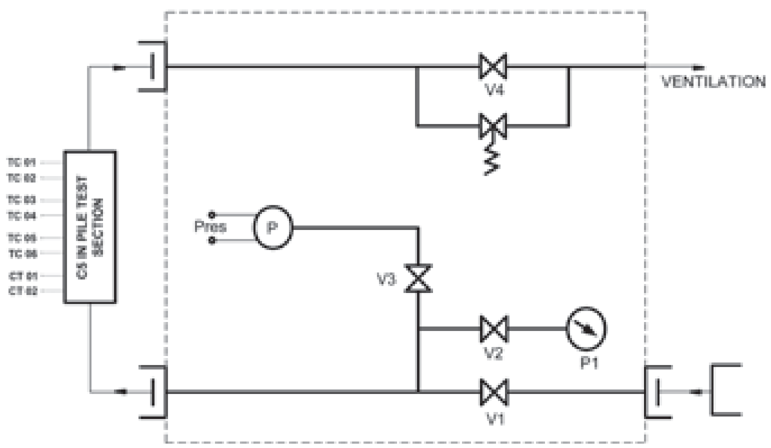


FIG. 14. Capsule C5 flow scheme.

The present capsule C5 is intended to be used for irradiation of samples of special materials for ITER project. Considering the future requirements for qualifications of candidate materials for new generation of reactors a new irradiation in core section will be designed and build to ensure an inert atmosphere and 600°C on samples.

TABLE 6. TESTS PERFORMED IN CAPSULES C1, C2 AND C5

Capsule	Experiment	Irradiation period
C1	Fission gas composition for CANDU type fuel element	20.03.84–15.08.86
C1	Densification tests for CANDU type fuel element	15.06.87–18.09.87
C1	Power ramping tests	13.10.87–19.07.89
C2	Dimensional measurement	10.81–10.84
C2	Fission gas pressure	26.03.84–21.09.94
C2	Power ramp	03.12.85–30.05.87
C2	Irradiation conditions calibrations	26.02.92–26.06.92
C2	Residual deformation of cladding determination	12.12.96–05.12.01
C2	Central temperature measurement in fuel	13.05.98–08.10.98
C2	Fission gases release effects on temperature	15.05.00–07.11.00
C5	C5 SS 403 M irradiation (irradiation time: 1983 h)	18.06.85–04.10.85
C5	C5 SS 403 M irradiation (irradiation time: 1121 h)	30.05.86–19.08.86
C5	C5 Zircaloy-4 tube irradiation (irradiation time: 6603 h)	22.12.87–28.11.88
C5	C5 Zircaloy-4 tube irradiation (irradiation time: 6660 h)	23.07.92–21.07.94
C5	C5 Zr-2.5% Nb alloy irradiation in inactive atmosphere (helium) (irradiation time: 11400 h)	11.10.95–10.12.01

2.5. CAPSULE C9

Capsule C9 was designed and operated to simulate the behaviour of one fuel rod in load follow-up mode under power plant conditions. Figure 15 shows the flow diagram with its specific features to allow continuous power variation while the pressure is kept constant inside the device. Figure 16 adds the main mechanical parts: the single walled pressure tube in a calorimeter tube and the mechanical displacement system.

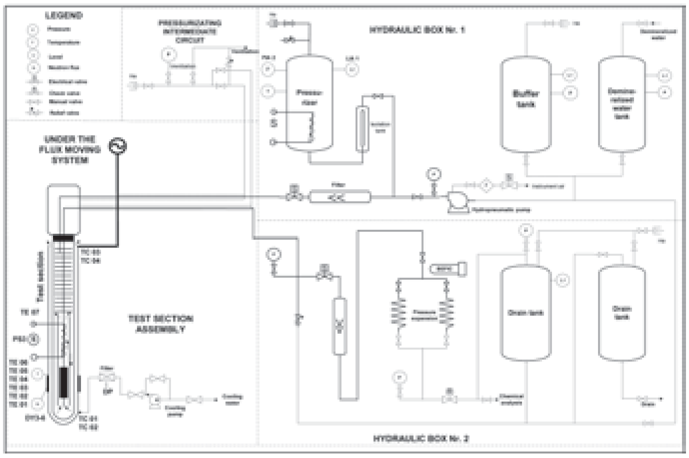


FIG. 15. Capsule C9 flow scheme.

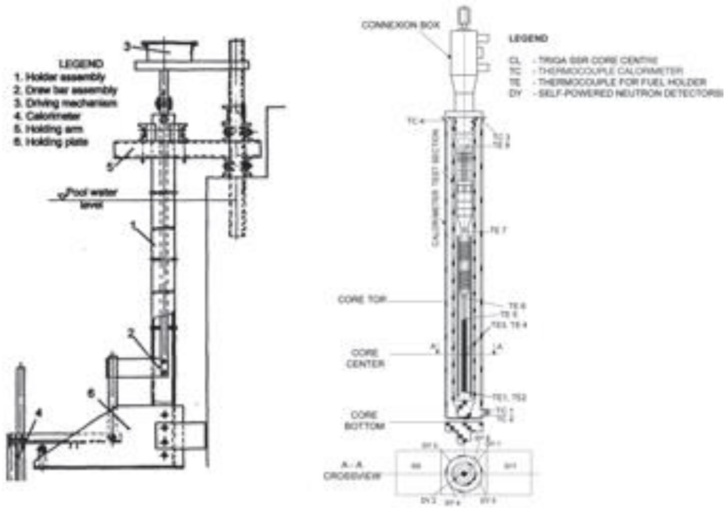


FIG. 16. Mechanical structure and instrumentation of capsule C9.

2.5. CAPSULE C6

Capsule C6 was designed to simulate reactivity insertion accidents (RIA), inserting it into the ACPR with a pulse amplitude of up to 20 000 MW (see Fig. 17). Based on the specific requirements of the fuel manufacturer, a safety research programme on CANDU type power reactor fuel was run between 1984 and 1989. At the beginning of 1997, the experiments were resumed and four more samples have been tested.

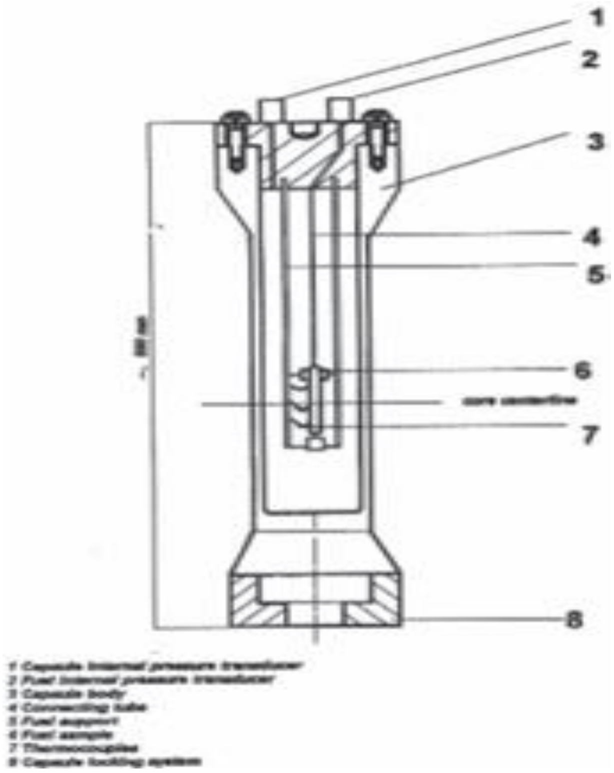


FIG. 17. Mechanical structure of capsule C6.

Another irradiation devices for RIA test using a special design capsule with Pb-Bi eutectic for transient testing of MYRRHA fuel for determination of the pin failure threshold, in the framework of MAXSSIMA project is in the design phase and preliminary licensing and will be in operation at the end of 2014.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

The cask was built following the IAEA recommendations concerning spent/irradiated fuel transportation in the framework of IAEA project *RER 9076 Enhancement of Nuclear Fuel and Materials Safety in Nuclear Power Plant.* The cask was built by licensed nuclear equipment manufacturer in Romania and tested by Institute owner. The B(U)F cask is currently licensed by Romanian Nuclear Authority CNCAN. The cask is transported with a dedicated equipped truck licensed also for this type of transportation following IAEA Regulations endorsed by CNCAN.

3.2. HOT CELLS, PIE FACILITIES

3.2.1. Technical capacity

- Available hot cell facility;
- Available non-destructive PIE techniques:
  - Visual examination and photography;
  - Dimensional measurements;
  - Axial gamma scanning and tomography;
  - Eddy current defect testing;
  - Oxide layer thickness measurement by eddy current;
- Available destructive PIE techniques:
  - Fission gas analysis-internal pressure and free volume of fuel;
  - Optical microscopy — hydrogen combined in the clad and other samples;
  - Scanning electron microscopy (SEM);
  - Burn-up by mass spectrometry;
  - Mechanical testing;
- Sealed sources production.

3.2.2. Available hot cell facility

(1) Heavy concrete hot cells;

FIG. 18. Heavy concrete hot cell operating area.

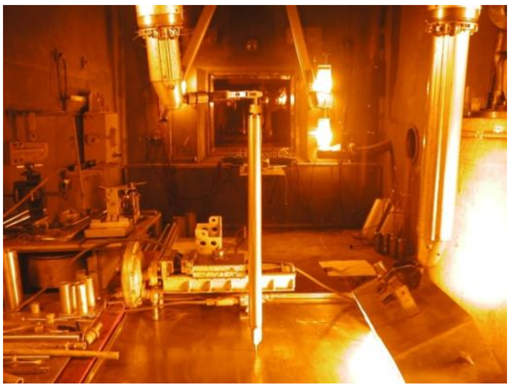


FIG. 19. Working station inside heavy concrete hot cell.

The alpha-gamma *transfer cell* receives loops and capsules from TRIGA reactor for fuel element extraction or loading. It is also used for ^{192}Ir and ^{60}Co sealed sources production. The CANDU pressure tube samples for mechanical testing are also prepared in this hot cell. Remote welding TIG is also installed in this cell.

The alpha-gamma *examination cell* is used for non-destructive examinations as well as for fuel element puncture test with fission gas analysis and for fuel cutting to remove the samples for metallographic examination, burn-up and mechanical testing.

(2) Steel hot cells;

FIG. 20. Steel hot cell operating area.

FIG. 21. Metallographic cell and microscope cell.

— The alpha-gamma metallographic cell is used for sample preparation: grinding, polishing, chemical and electrochemical etching;
— The beta-gamma microscope cell is located adjacent to the metallographic cell and is interconnected to it by a door to transfer the samples on the optical microscope stage;
— The alpha-gamma chemistry cell is under preparation for ^{99}Mo production.

(3) Lead hot cell

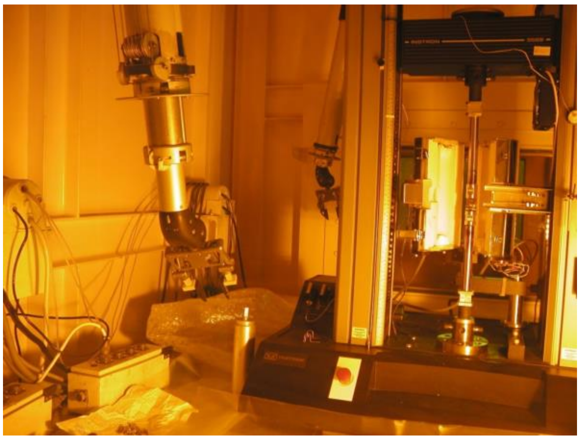


FIG. 22. Working station No.1 of lead hot cell.

FIG. 23. Working station No.2 inside lead hot cell.

The beta-gamma *lead hot cell* has two opposed working stations: No. 1 and No. 2. The working station No. 1 is used for mechanical testing. The working station No. 2 will be used for scanning electron microscopy (SEM). A TESCAN MIRA scanning electron microscope is under installation inside the hot cell.

FIG. 24. Leica TELATOM 4, remote controlled inverted wide field metallographic microscope for radioactive materials.

3.2.3. Available destructive PIE techniques

(a) Optical microscopy;
  (1) Basic parameters:
    — Resolution: 0.3 µm for working distance of 1.6 m;
    — Magnification range: from 8x to 2500x;
    — Motorized sextuple nosepiece;
    — Plan EPI objectives: 2.5x/0.075; 5x/0.10; 10x/0.20 IK; 20x/0.40 IK; 50x/0.80 IK; 100x/0.95 IK;
    — Plan Apo objective: 1x/0.03;
    — Binocular viewing tube with a pair of widefield eyepieces 10x;
    — Motorized magnification changer: 0.8x-1x-1.25x-2.5x;
  (2) Examination techniques:
    — Brightfield;
    — Oblique illumination;
    — Polarized light with +/– 90° rotatable polarizer and analyser;
    — NOMARSKI differential interference contrast (DIC).
  (3) Applications:
    — Grain size and distribution;
    — Phase identification;
    — Porosity measurements;
    — Corrosion susceptibility (oxide layer thickness);
    — Features and orientation of hydride precipitates in the cladding;
    — Integrity evaluation of welds, brazes etc.

(b) Scanning electron microscopy (SEM);
  (1) Basic parameters:
    — Resolution in high vacuum mode with SE detector: 1.2 nm at 30 kV; 1.5 nm at 15 kV; 2.5 nm at 3 kV;

— Resolution in low vacuum mode with SE detector: 1.5 nm at 30 kV; 3.0 nm at 3 kV;
— Resolution in high/low vacuum mode with BSE detector: 2.0 nm at 30 kV;
— Chamber working vacuum in high vacuum mode: < 1×10^{-3} Pa;
— Chamber working vacuum in low vacuum mode: 7–150 Pa;
— Magnification: 4x to 1 000 000x;
— Accelerating voltage: 200 V to 30 kV in 10 V steps;
— Electron gun: high brightness Schottky emitter;
— Probe current: 2 pA to 100 nA;
— SE and retractable BSE detector.

(2) Applications:
— Topographic analysis: the surface features of a material, its texture;
— Morphologic analysis: the shape and size of the particles making up the material;
— Identification and evaluation of microstructure on sample surface, fission gas porosity at grain boundary, fission product precipitates, zirconium oxide layer structure;
— Failure analysis.

FIG. 25. TESCAN MIRA II LMU CS Schottky field emission variable pressure scanning electron microscope

FIG. 26. INSTRON 5569 50 kN tensile testing machine.

(c) Mechanical testing;
(1) Basic parameters:
— 50 kN load cells;
— Displacement rates: .005–500 mm/min;
— Automated data analysis capabilities;
— ASTM standard available;
— Equipped by a furnace which ensure test conditions from 20°C to 1000°C under air atmosphere;

(2) Applications;
— Strain-stress diagram and load extension;
— Yield strength (offset method at 0.2%);

— Elastic limit;
— Ultimate tensile strength of the sample;
— Threshold stress intensity factor (KIH) and delayed hydride cracking (DHC) velocity in Zr-2.5% Nb CANDU pressure tube samples.

Tensile tests on rings, tubes, plane and C-shape samples removed from CANDU fuel cladding, CANDU reactor pressure tube and other component parts of nuclear reactor can be performed.

(d) Axial gamma scanning and tomography;
  (1) Examination equipment:
    — Fuel element positioning machine equipped by step-by-step motors for longitudinal and transversal movement and rotation of fuel element in front of collimator;
    — Collimator with a variable aperture slit set in the hot cell shielding wall;
    — PGT IGC 12 coaxial intrinsic Ge detector (volume: 59 cm^3; resolution: 1.81 keV/1.33 MeV);
    — Multichannel analyser (4096 channel memory).
  (2) Applications:
    — Axial distribution of fission products (FPs) activity in the fuel column: gross gamma FPs activity profile and specific gamma isotope activity profile such as ^{137}Cs, ^{134}Cs, ^{95}Zr-Nb, ^{103}Ru-Rh, ^{106}Ru-Rh, ^{140}Ba-La, ^{144}Ce-Pr;
    — Flux peaking and loading homogeneity;
    — Radial distribution and tomography of volatile and non-volatile FPs in the fuel element cross section;
    — Migration of volatile FPs (^{137}Cs) inside the fuel element;
    — Fuel column length measurement;
    — Assessment of the gap between pellets;
    — Burn-up.

(e) Fission gas analysis;
  (1) Examination equipment:
    — Installation for punching of fuel element and measurement of released fission gas;
    — Quadrupole-mass spectrometer gas analyser installed at the outside of the hot cell;
  (2) Applications:
    — Fission gas pressure;
    — Fission gas volume;
    — Inner void volume of fuel element;
    — Fission gas composition and concentration

The pressure and volume of the fission gas inside the fuel element are measured using a standard expansion volume of 148.5 cm^3 and the isothermal gas transformation law.

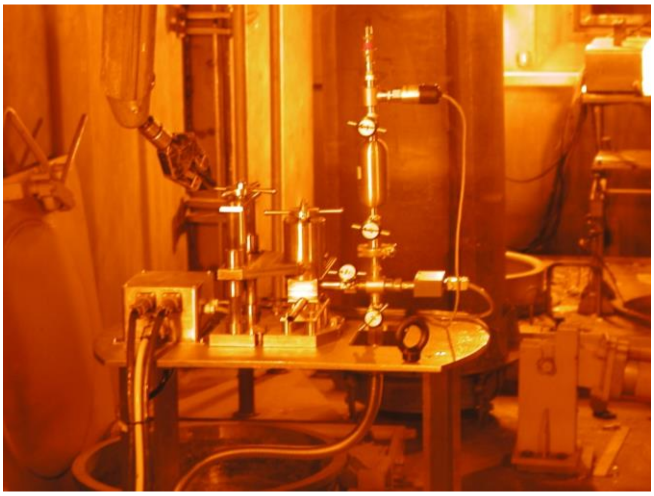


FIG. 27. Installation for punching of fuel element and measurement of released fission gas.

(f) FINNIGAN MAT 261 mass spectrometer

(1) Basic parameters:

- Mass range: 5–350 atomic mass unit (amu) at an accelerating voltage of 10 kV;
- Working resolution: 500 at 10% valley definition;
- Abundance sensitivity: <2 ppm at 237 amu (due to 238 amu peak);
- Collector slit width: 0.6 mm;
- Transmission coefficient: >0.20;
- Air pressure achieved in the analyser: $<10^{-6}$ mbar;
- Cross contamination figure: $<6 \times 10^{-9}$;
- Possibility for loading 13 samples on filaments welded on detachable mounts;
- Ion detection system with Faraday collector; for very low signals it is available a secondary electron multiplier (SEM).

(2) Applications — burn-up of nuclear fuel by ^{235}U depletion method. The accuracy of measurement is ± 3%.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEM

The Institute infrastructure organized in several departments cover the main activities for research, design, construction, operation and analysis of experiments including licensing documentation for reactor and testing facilities authorization, able to sustain capabilities and capacities of TRIGA research reactors towards deployment of innovative nuclear energy systems and technologies. The presentation of departments follows:

- Department of reactor physics and nuclear safety has the capacity to perform neutron and thermal hydraulic analysis and design of requested core configuration including irradiation devices for nuclear fuel testing, material testing and radioisotopes;
- Department of nuclear fuel and material research has the capacity to conceive experiments and testing samples for experimental fuel and materials specimens to be irradiated and ensure support for data processing and analysis for behaviour evaluations during irradiation and post irradiation examination in regards of irradiation and corrosion effects;

— Department of out of pile testing operating loops and rigs and some experimental devices in preliminary stage before design and construction of equipments to be used in reactors;
— Department of nuclear electronics which design and build the instrumentation and control system for irradiation devices;
— Department of design of nuclear equipments including in core components of irradiation devices licensed by Pressure Vessel Authority, design of radioactive materials and handling means as cask/containers for shipment of irradiated fuel;
— Department for manufacture of nuclear equipments licensed by Pressure Vessel Authority has a long experience of fabrication of equipments for nuclear power plant, for research reactor and components for irradiation loops and capsule used for irradiation in TRIGA research reactor. This department is specialized in work concerning fabrication of components from high nickel alloys, zirconium, aluminium and special qualified welding;
— Department of TRIGA research reactor ensures operation and utilization of reactor of irradiation loops, irradiation capsules, irradiation devices for materials and for radio isotopes. This department perform the in core instrumentation of experiments with thermocouples, self powered neutron detectors, pressure transducers and neutron flux monitors for experimental fuel rods and materials specimens using dedicated technologies for welding, brazing, joint diffusion;
— Department of post irradiation examinations (PIEL) ensure operation of hot cells and associated instruments for post irradiation examination of experimental irradiated fuel rods, of material specimens and in the latest time examination of spent fuel from Cernavoda NPP. This department ensures also radio isotopes production, transpiration of sources and transportation of irradiated nuclear fuel;
— Department for radioactive waste and conditioning of waste generated by reactor operation, by experiments/irradiation devices and post irradiation activities as well transportation of conditioned waste to national repository;
— Department for radiation protection and environmental monitoring licensed by Romanian Nuclear Regulatory Authority (CNCAN).

All above described activities are subject of an integrated quality management system dedicated to accomplishment of research goals, nuclear safety and security, protection of people and environment.

4. RECENT ACHIEVEMENT

Examples of research and development studies performed by Institute during the last year could be found on: www.nuclear.ro/programecercetare_en.htm.

Recent publication during annual Institute conference could be found on: www.nuclear.ro/arhiva_evenimente_stiintifice and www.jrnd-nuclear.ro.

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Profile 19

BIGR

RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA

Fast Pulse Graphite Reactor (BIGR) is a fast pulse reactor with ceramic (UO_2+C) core. The reactor is categorized as aperiodic self-quenching pulse reactor and is the most powerful fast pulse reactor.

Works on reaching first criticality in the reactor were conducted from February till December 1976. Reactor was commissioned in March 1977.

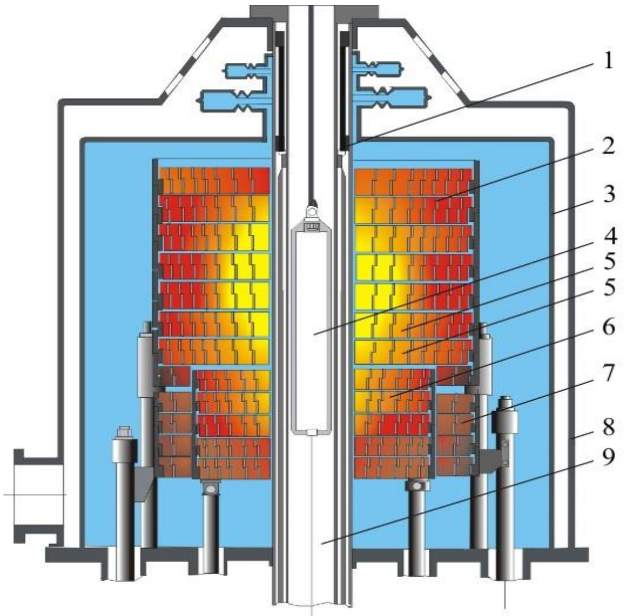
Designed life was not specified. Estimated core material life is not less than 3000 pulses with maximum power release.

The reactor was not reconstructed. Automatic physical characteristic measurement system was modernized. Control and safety system is being modernized. Control room and a view of BIGR core are presented in Fig. 1.

Lifetime reactor operational characteristics by September 2013 are 636 start-ups in the static mode and 1123 pulses on prompt and delayed neutrons.

FIG. 1 (a). BIGR control room.

FIG. 1 (b). View of BIGR core.



1 – pulse block;
2 – core unmovable part;
3 – core vessel;
4 – central channel loading cask;
5 – core fuel rings;
6 – rough control block;
7 – precise control block;
8 – air-cooled shroud;
9 – centre line chamber.

FIG. 2. BIGR reactor section.

The BIGR reactor longitudinal cross-section is presented in Fig. 2. Reactor core is made in shape of hollow cylinder (with the diameter of 76 cm and height of 67 cm) and consists of free-hanging coaxial unfixed rings. Each ring rests upon the neighbouring one or upon the external case by circumferential ledge made at ring half-height. The core is divided in three units, namely: core unmovable part, rough control block and precise control block. Steel tube is used as pulse block. The core is enclosed in a leak tight vessel filled with helium.

TABLE 1. BIGR MAIN PERFORMANCE

Static mode:	steady-state operation
Maximum power	0.5 MW
Energy-release	up to 500 MJ
Quasi-pulse mode:	delayed neutrons power excursion
Ultimate energy-release	280 MJ
Minimal quasi-pulse half-width	0.5 s
Power in a quasi-pulse peak	up to 1.5 GW
Pulse mode:	prompt neutrons power excursion
Ultimate energy-release	280 MJ
Minimal pulse half-width	2 ms
Power in pulse peak	up to 75 GW

Pulse generation frequency is one pulse a day.

Mixture of compacted and sintered UO_2 (enrichment in ^{235}U is 90%) and graphite is used as core fuel. Moderator-to-fuel nuclear ratio is equal to 18.

Cumulative burn-up of ^{235}U is about 3.2 g. Since total amount of ^{235}U in the core is about 440 kg this burn-up is negligibly small.

Reactor core is cooled after the pulse and during operation at steady-state power by either natural convection or forced one due to air vent through air-cooled shroud and central channel.

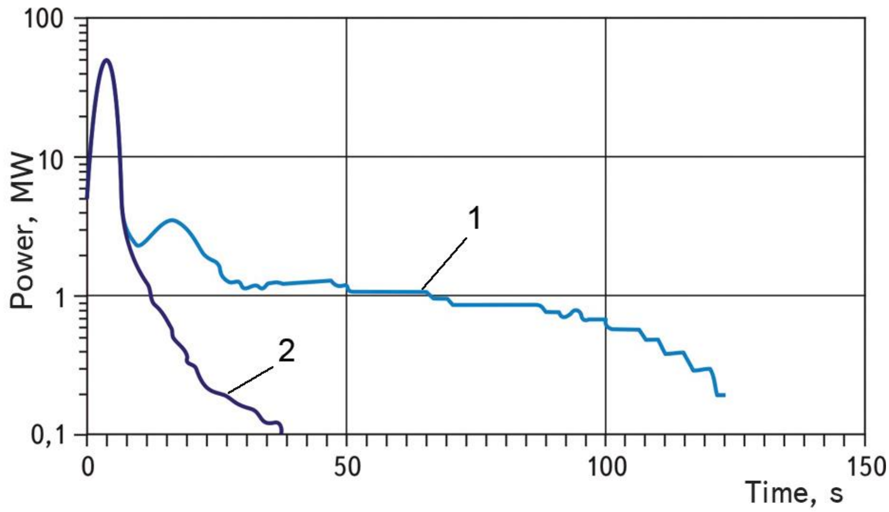
Samples can be housed within the 55 cm-high central channel loading cask, Ø 10 cm, during radiation resistance tests. Large-sized samples can be remotely delivered to the core on specially designed trolleys from two opposite sides.

The facility is equipped with additional equipment which makes it possible to vary n-γ-radiations in a greater range, namely: different n-γ-converter modifications, large-sized reflectors made of steel, graphite, polyethylene and beryllium.

The following modes of fission pulse generation are being used at the reactor:

- Generation of fission pulses on prompt neutrons in flyby mode — pulse rod flies through the core centre;
- Generation of fission pulses on prompt neutrons when pulse rod stops in the centre of the core;
- Generation of fission pulses on prompt neutrons in case of initiation from power or powerful source of delayed neutrons;
- Quasi-pulse mode — pulses are generated on delayed neutrons.

Pulses on delayed neutrons (quasi-pulses) can be of various shapes including rectangular one. Fission quasi-pulses have triangular shape at initial reactor runaway period within the range from dozens of seconds to hundreds of milliseconds. Fission quasi-pulses at the same periods but with power control have trapezoidal shape. Quasi-pulses have a shape close to rectangular when reactor reactivity is near to critical



1 – pulse when pulse block stops in the centre of the core;
2 – pulse when pulse block flies through the core centre.

FIG. 3. Shape of fission pulse on prompt neutrons in BIGR reactor.

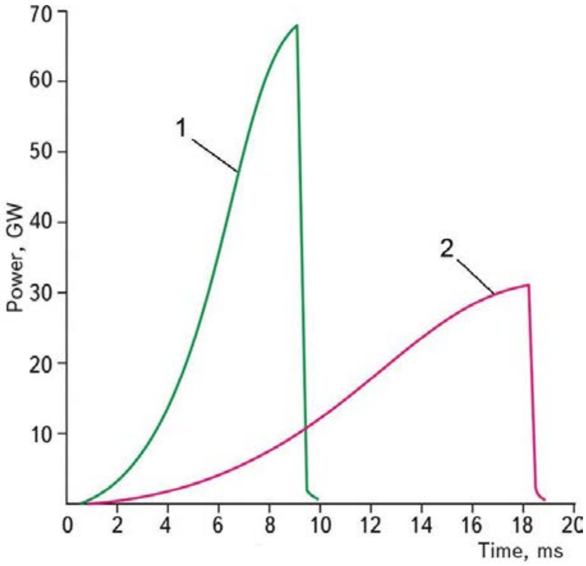


FIG. 4 (a). BIGR quasi-pulse shape with runaway period 1.3 s (1) and 3.1 s (2).

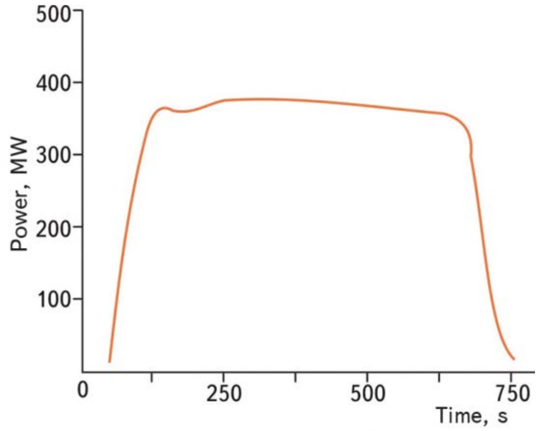


FIG. 4 (b). BIGR quasi-pulse shape with runaway period 14.7 ms.

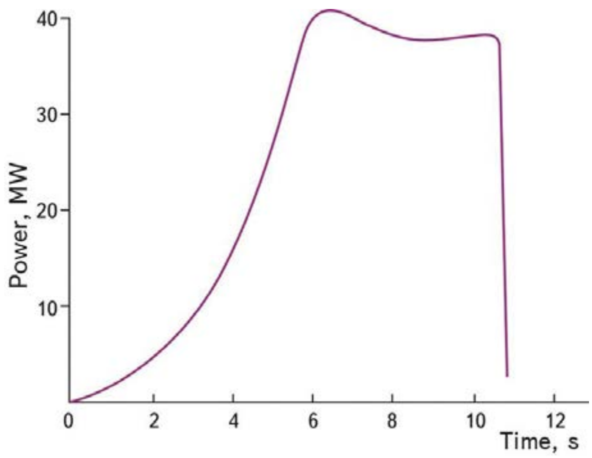


FIG. 4 (c). BIGR quasi-pulse shape with runaway period 1.3 s and with power control.

TABLE 2. BIGR IRRADIATION CHARACTERISTICS AT ENERGY RELEASE OF 280 MJ

BIGR irradiation characteristics	Value
Total neutron fluence in the central channel of the core	1×10^{16} n·cm^{-2}
Neutron fluence (E > 0.1 MeV) in the central channel of the core	8.5×10^{15} n·cm^{-2}
Maximal accompanying γ-radiation dose in the central channel of the core	1.9×10^{4} Gy
Total neutron fluence on the core surface,	1×10^{15} n·cm^{-2}
Neutron fluence (E >0.1 MeV) on the core surface	0.8×10^{15} n·cm^{-2}
Maximal accompanying γ-radiation dose on the core surface	2.2×10^{3} Gy

2. EXPERIMENTAL FACILITIES

2.1. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS RIA

On BIGR reactor there was developed a complex of equipment, meant for experimental studies of behaviour of power reactor fuel elements under conditions, simulating reactivity-initiated accidents on nuclear power reactors.

The developed complex allows tests of experimental fuel elements, containing both fresh (un-irradiated) fuel, as well as burnt-out (irradiated) fuel.

The test aims are:

- Determining of threshold values of specific energy release in the fuel, corresponding to different levels of fuel elements damageability;
- Study of fuel elements disruption mechanisms;
- Study of interaction between melt fuel and coolant.

When performing tests there are used irradiating devices, meant for conversion of reactor leakage neutron spectrum — reflector-moderator blocks (RMB).

Integral moderated neutron fluence in the irradiating cavities of existing modifications of reflector-moderator blocks can comprise up to $2 \cdot \times 10^{15}$ $n \cdot cm^{-2}$.

From the standpoint of geometry, RMB represents a right-angle prism with characteristic linear sizes up to 1 m. RMB may comprise blocks made of graphite, beryllium, polyethylene, steel and other materials in different combinations depending on test conditions. In RMB area facing the reactor there is placed a vertical cavity, meant for irradiated objects placement.

Between the reactor core and RMB there is mounted a shield, comprising boron, enriched in isotope ^{10}B and cadmium. The shield is meant for absorption of moderated neutrons reflected in the direction of the core.

Fission density increase in objects containing FM is achieved through leakage neutron spectrum softening and through the fact that RMB performs a function of neutron reflector.
Fig. 5 shows an example of RMB arrangement near the reactor side surface. RMB is delivered to the reactor by special transfer mechanisms.

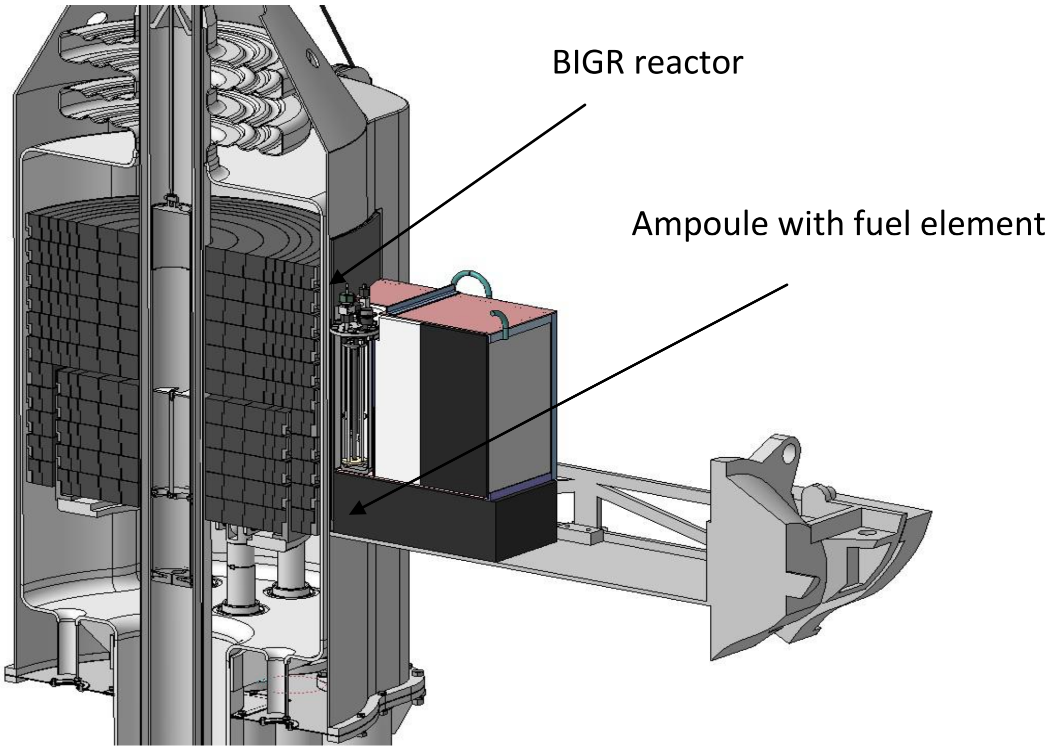


FIG. 5. Arrangement of RBM with an irradiating ampule near the side surface of BIGR reactor.

The RMB structure is selected on the basis of results of neutron-physical calculations and experience of similar devices' employment in experiments on pulsed reactors. Structurally, RMB represents a steel case, serving for placement of beryllium and graphite blocks. Beryllium and graphite RMB sections are picked up for separate elements. Fig. 6 presents an image of RMB general view (case elements are partially hidden).

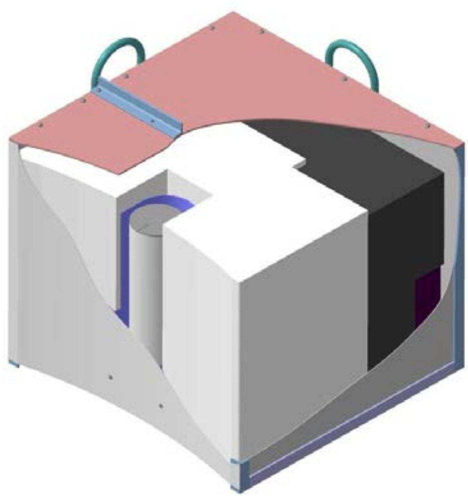


FIG. 6 . Image of RMB general view.

Experimental fuel elements are tested in special instrumented capsules (ampoules) ensuring safety of experiments' performance (protection from radioactive contaminations).

There have been developed experimental ampoules of different types, meant for tests of fresh and burnt-out fuel elements under different conditions.

Tests can be performed at the following parameters of immobile medium in ampoules:
- Water at pressure from 0.1 MPa up to 17 MPa;
- Air at atmospheric pressure.

Ampoules meant for experiments with fresh fuel, provide measuring of temperatures of fuel element shell and ampoule elements as well as medium pressure in the ampoule.

To ensure safety, when performing tests of burnt-out fuel, ampoules with double hermetic container are applied. In this case, no temperature and pressure measurements are made inside the hermetic ampoule.

There were developed the following promising schemes for creating a methodology of testing and experimental capabilities:
(1) Performance of fuel element ampoule tests in the central experimental channel;
(2) Development of irradiating devices for ampoule tests of fuel elements with fresh and burnt-out fuel of VVER reactors under pressures 16 MPa and elevated initial coolant temperature;
(3) Modernization of irradiation complex on the basis of BIGR reactor by its equipping with a transportable multiplying assembly, what allows testing of both fuel elements with low concentration ^{235}U (involving high burn-up fuel), and fuel assemblies, containing several experimental fuel elements and having a large (up to 1 m) length of fuel part.

The modernization project foresees development of loops for tests of fuel elements in different coolants at their parameters, maximally corresponding to real parameters in power reactors.

Supplementary process equipment placed on the reactor site (in the reactor building):
- Spot laser welder applied for equipping of experimental fuel elements and irradiating ampoules with thermocouples;
- Gas filling bench meant for generating gaseous medium pressure up to 17 MPa in the volume of irradiating ampoules.

2.2. COMPUTER CODES AND RESEARCH TECHNIQUES

The computer codes designed in RFNC-VNIIEF for numerical analyses of experiments are:
- The neutron-physical code CMK (an analogue of MCNP);
- The 1D thermo mechanical code CAFR;
- The 3D thermo mechanical code LEGAK-DK.

The computer codes are required to:
- Analyse experimental results;
- Optimize the experimental facility;
- Perform numerical simulations of experiments to design experiments with required parameters (energy deposition, cladding temperature, etc.);
- Find temperature field in the object under study and peak fuel enthalpy.

Application of codes allows to reduce the number of experiments and provide a wide range of additional information about the phenomena occurring during experiments.

The CMK code is based on the C-007 method and:

— Is a Monte-Carlo code designed to solve coupled neutron and gamma transport problems in 3D using both spectral and multigroup constants;
— Can be used for safety analyses, evaluations of critical parameters (K_{eff} and neutron multiplication coefficient), nuclear safety and reactor analyses.

Its main purpose is to solve spatial and temporal distributions of energy release in experimental fuel rods.

The 1D thermo mechanical code CAFR:
— CAFR is designed for numerical support of research reactor experiments with fuel rod specimens of various types and also as a fuel rod solver to be used within thermal hydraulics codes. It is included as a fuel rod solver in the RATVEL thermal hydraulics code (see Ref. [3]). It can be also used separately;
— It is used to analyse experimental results measured for various fuel rod types in a number of research reactors BIGR (RFNC-VNIIEF), IGR (Kazakhstan), NSRR (Japan);
— CAFR provides adequate modelling of all key phenomena occurring in fuel rods, of thermo mechanical, physical and chemical properties of fuels and cladding throughout the operation period, of heat transfer and interactions with coolant;
— CAFR models phenomena occurring in fuel rods and core of power reactors during operational regimes and also in RIA and LOCA type accidents;
— Is used to model the performance of other parts of reactors: RBMK graphite stack and channels, pipelines, etc.

The main difference between CAFR and other exiting fuel analysis rod codes is that it employs numerical simulations of continuum mechanics including strength, ductility and heat conduction. The equations of state designed into CAFR can handle thermal expansion and phase transformations of materials. It can treat radial and axial heat conduction, gas flow between gap and plenum, cladding material oxidation, etc.

A state-of-the-art 3D parallel computation thermo mechanical code LEGAK-DK is being developed now that will include the main capabilities available in CAFR and will enable simulations of fuel rods in a variety of shapes.

It is used for numerical support of BIGR reactor experiments with fuel rods of complex configuration containing metallic fuels.

3. MAIN AREAS OF STUDIES

— Radiation-resistance tests;
— Study of fuel sample behaviour in reactivity initiated accident (RIA).

Experiments on generating ultra-cold neutrons and studying their properties were conducted at BIGR reactor in cooperation with JINR specialists (Dubna).

FIG. 5. WWER fuel samples after their irradiation in BIGR reactor.

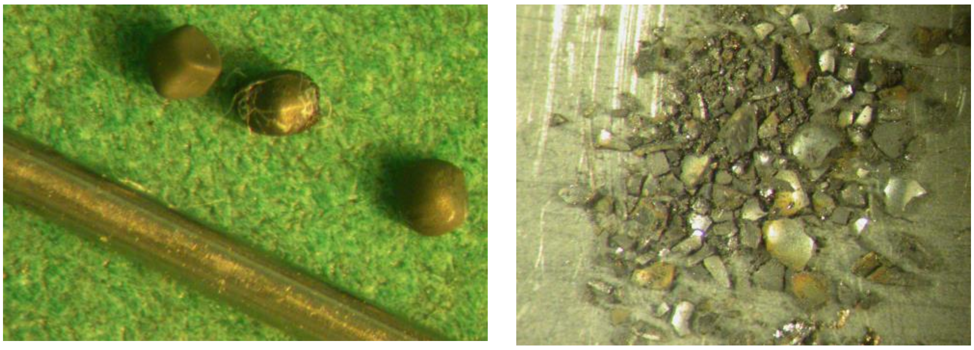


FIG. 6. Coated particle fuel before and after its irradiation in BIGR reactor.

4. ENGINEERING AND RESEARCH INFRASTRUCTURE

Russian Federal Nuclear Centre VNIIEF possesses a powerful calculation, experimental and production base and represents a system of closely collaborating institutes: theoretical and mathematical physics, physics of explosion, nuclear-radiation physics, laser-physics research, high energy densities. It involves also construction departments, production-and-technical and engineering complexes.

VNIIEF provides turnkey production: starting form development and ending with manufacturing.

According to normative documents in force, VNIIEF secures all requirements on nuclear materials treatment (storage, account and control, transportation, employment in studies and tests).

At VNIIEF there exists all required equipment for study of irradiated nuclear fuel. However, usually all fuel investigations are conducted in Public Corporation ‘SSC RIAR’, with which VNIIEF is unified by a single collaboration.

5. ACHIEVED RESULTS, EXAMPLES OF RESEARCH AND DEVELOPMENT WORKS PERFORMED IN THE LAST TEN YEARS

BIGR reactor specific feature is a capability for generating two types of pulses with half-width $\tau_{1/2} \sim 2–10$ ms and $\tau_{1/2} > 0.4$ s. This allows:

- Comparison of behaviour of WWER fuel and that of PWR and BWR fuel. In the most complete fuel test programme that have been carried out in Japan on NSRR reactor for more than 20 years with PWR and BWR fuel, there are used reactor pulses with half-width $\tau_{1/2} \sim 3–15$ ms;
- Emergency analyses with most probable duration (half-width) of a pulse of energy release $\tau_{1/2} \sim 1–3$ s.

The level of energy action:

- Fresh nuclear fuel with enrichment 4.4% — ~2 kJ/g·uranium;
- Burnt-out nuclear fuel (burn-up ~60 MW·day/kg·uranium at initial enrichment 4.4%) – 1 kJ/g·uranium.

On the irradiation complex the tests have been carried out and are performed at present in the conditions, simulating reactivity-initiated accident:

- Experimental fuel elements with fresh and burnt-out fuel of WWER type reactors;
- Samples of micro-fuel elements with kernels made of uranium dioxide and coated by four layers;
- Experimental fuel elements with metallic uranium-zirconium fuel.

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Profile 20

IR-8

RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA OF RR

The IR-8 pool-type reactor is the result of upgrades at the IRT reactor which had been commissioned in 1957. IR-8 reached first criticality on 12 August 1981, and was brought to first power on 30 October 1981. IR-8 is used for studies in the fields of solid-state physics, nanotechnologies and nanomaterials, radiation materials science, nuclear physics, radiation chemistry, radiobiology, for tests of fuel composition specimens and production of various medical radioisotopes.

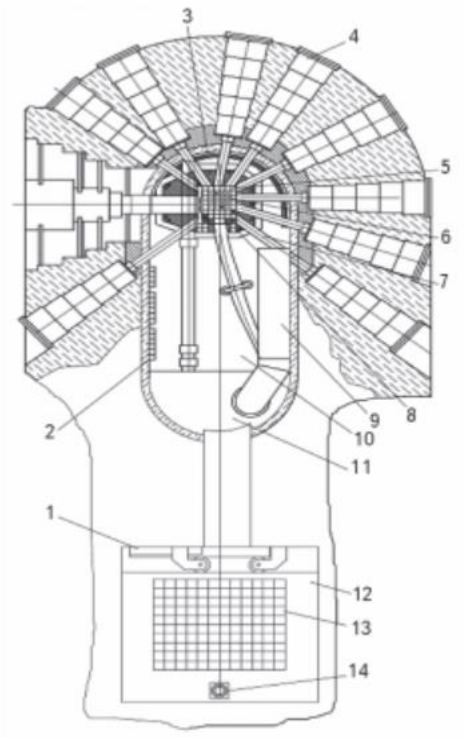
The configuration of the reactor core ensures high neutron flux parameters both in the core and in the reflector. The IR-8 core is immersed in a water pool to provide biological shielding, to ensure safety of maintenance and handling operations as well as of the reactor facility itself, and to reduce accidental radiation release. The main technical characteristics of IR-8 are presented in Table 1.

TABLE 1.THE MAIN TECHNICAL CHARACTERISTICS OF IR-8

Characteristics	Value
Thermal power	8 MW
Number of driver fuel assemblies	16
Fuel	UO_2, 90% enrichment in ^{235}U
Burn-up of discharged fuel	52%
Height of core	580 mm
Core volume	47.4 L
Maximal thermal neutron flux:	
– thermal in the core	1.5×10^{14} n·cm^{-2}·s^{-1}
– fast (E > 3 MeV) in the core	$5.7 \cdot \times 10^{13}$ n·cm^{-2}·s^{-1}
– thermal in a replaceable beryllium block	3.0×10^{14} n·cm^{-2}·s^{-1}
– fast (E > 3 MeV) in a replaceable beryllium block	1.8×10^{13} n·cm^{-2}·s^{-1}
– thermal at the inlet of the horizontal channel	1.1×10^{14} n·cm^{-2}·s^{-1}
– thermal at the outlet of the horizontal	1.8×10^{10} n·cm^{-2}·s^{-1}
Number of neutron beams	12
Number of vertical channels	up to 42
Average annual power days	120

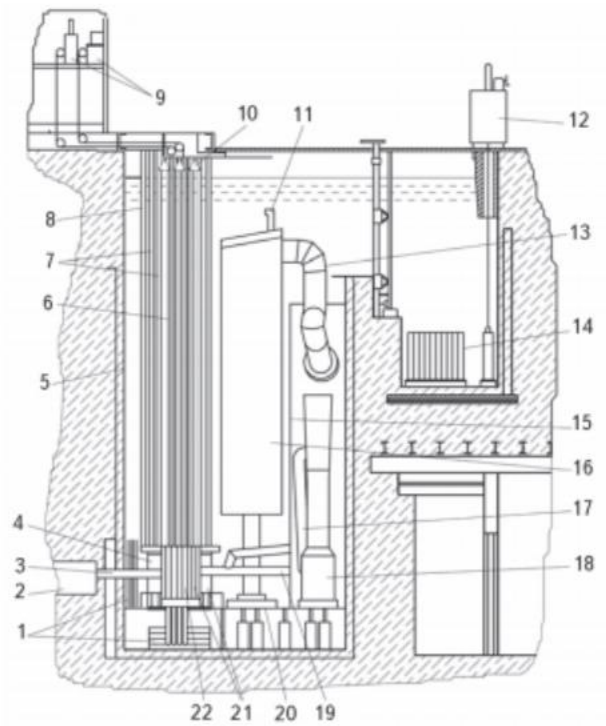
An IRT-3M assembly consists of tube-type fuel elements of square cross-section. The middle fuel layer is uranium dioxide in an aluminium matrix with the thickness of 0.4 mm and the height of 580 mm. There are 13 control rods: 2 for emergency protection (scram), 10 for shimming, and 1 for automatic control. The absorber of the control rods is boron carbide pellets in a sheath of stainless steel. The height of the absorber part of the rods is 600 mm.

Fuel assemblies and replaceable beryllium blocks are mounted on a support grid 90 mm in thickness. In Figs 1, 2 and 3 cross-sections of the IR-8 and core map are shown.



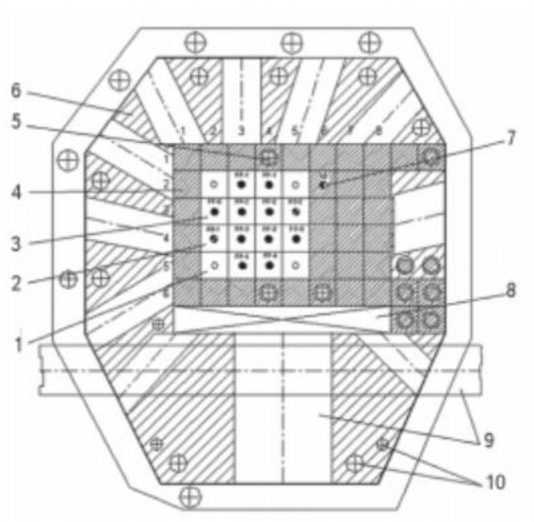
1 – lock gate;
2 – interim SNF storage;
3 – steel shield;
4 – neutron beam;
5 – replaceable beryllium block;
6 – fuel assembly;
7 – HEC;
8 – stationary beryllium block;
9 – retention tank;
10 – reactor pool;
11 – retention tank;
12 – storage pool;
13 – SNF storage;
14 – container receptacle for unloading spent fuel from the pool.

FIG. 1. Cross-section of the IR-8.



1 – steel shields;
2 – HEC shutter;
3 – HEC;
4 – reactor vessel;
5 – reactor tank;
6 – CPS rod channels;
7 – VEC;
8 – CPS channels;
9 – CPS rod drives;
10 – sprinkler;
11 – air vent;
12 – transport cask;
13 – suction pipeline;
14 – SNF storage;
15 – vertical partition;
16 – retention tank;
17 – pressure pipeline; 1
8 – ejector;
19 – ultra cold channel;
20 – dividing bottom;
21 – beryllium reflector;
22 – fuel assembly

FIG. 2. Longitudinal section of IR-8.



1 – fuel assembly with a displacer;
2 – fuel assembly with an EP rod;
3 – fuel assembly with a shim rod;
4 – replaceable beryllium reflector block;
5 – replaceable beryllium block with a vertical experimental channel;
6 – stationary beryllium reflector;
7 – automatic control rod;
8 – lead shield;
9 – horizontal experimental channels;
10 – vertical experimental channels

FIG. 3. Core map.

The reactor's heat transfer system has a two-circuit arrangement. The primary circuit consisting of reactor coolant pumps, heat exchangers, ion exchange filters, valves and pipelines between these components, removes heat from the reactor core and carries it to the recirculating water circuit. The primary water runs through fuel assemblies from the top down.

In 2011, the IR-8 reactor approached the end of its specified 30-year service life, some of its systems grew old, and for its operation to continue, it became necessary to repair or replace:

- Heat exchangers, primary coolant pumps and cooling tower;
- Instrumentation, control, safety systems;
- Radiation monitoring system;
- Electric power supply system.

Work is under way at present to replace the heat exchangers, pumps and instrumentation in the primary circuit.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES

The reactor has 12 horizontal beam channels with equipment installed in them for neutron study of condensed media, and vertical channels (up to 42) where radioactive isotopes can be produced and fuel and structural materials can be irradiated with online measurements. Other facilities of IR-8 include:

- Complex of neutronic hardware for studies of the structure, phase transitions, heterogeneity of and flaws in materials, including various research devices mounted on horizontal channels of the IR-8 reactor with the average thermal neutron flux at channel ends equal to $\approx 5 \times 10^9$ n·cm^{-2}·s^{-1}, such as:
    - five-circle neutron diffractometer MOND;
    - triaxial crystal spectrometer ATOS;
    - polycrystal multidetector circular diffractometer DISK;
    - triaxial perfect crystal-based spectrometer STOIK;
- Complex of hardware for neutron and gamma introscopy of high-tech materials and products (turbine blades, welded joints, fuel assemblies, etc.), including additional devices installed in monochromatic neutron beam traps of the above facilities. Gamma-ray chambers and Imaging Plates are used as neutron and gamma-ray detectors.

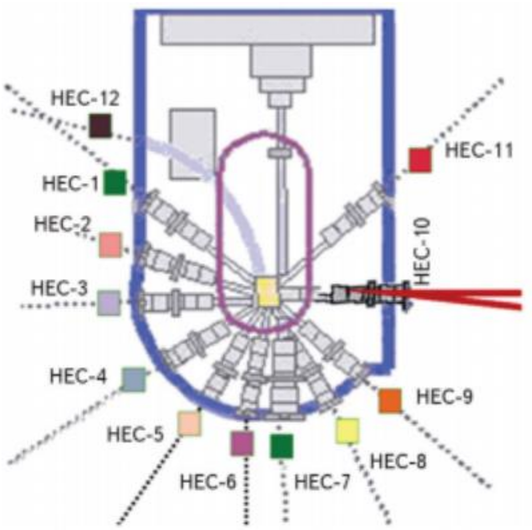
FIG. 4(a). MOND diffractometer.

FIG. 4(ab ATOS spectrometer.

FIG. 4(c). DISK diffractometer.

3. MAIN AREAS OF STUDIES AND ACTIVITIES

— Examinations of materials with polarized and ultra cold neutrons;
— Conversion electron spectroscopy;
— Element analysis and medical biological research;
— Investigations of materials by inelastic and elastic neutron scattering;
— Examinations of fuel elements by neutron and gamma introscopy;
— Development of methods and study of new materials in a wide range of pressures and temperatures;
— Studies of the structure, properties and flaws of non-irradiated and slightly activated reactor materials.



HEC-1 – neutron microscopy;
HEC-2 – nuclear spectroscopy;
HEC-3 – fission physics;
HEC-4 – single crystals (MOND);
HEC-5 – excitation spectra (ATOS);
HEC-6 – high pressures (DISK);
HEC-7 – capillary optics;
HEC-8 – neutron radiography;
HEC-9 – small-angle scattering (STOIK);
HEC-10 – inelastic scattering;
HEC-11, HEC-12 – ultra cold neutrons.

FIG. 5. Experimental hall layout.

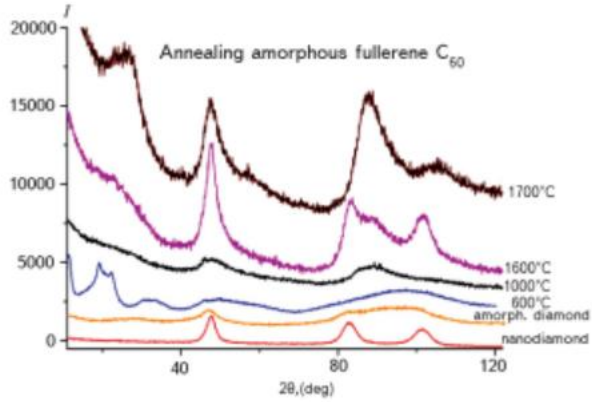


FIG. 6. Polymorphic transition in amorphous fullerenes.

The IR-8 reactor is used in research activities in cooperation with Belarus, Latvia, Armenia and other countries.

4. RECENT EXPERIMENTAL INVESTIGATIONS

- Phase transitions were discovered in nano size fullerites C_{60}, C_{70} and their mixtures at high temperatures;
- Experiments were carried out to study the structure of and manufacturing flaws in mono- and polycrystalline materials of turbine blades, welded joints, micro- and macro-fuel compositions, composite and monocrystalline superconducting materials.

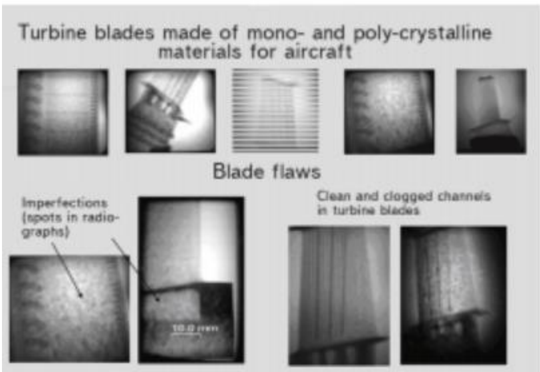


FIG. 7(a). Flaws in aerospace products.

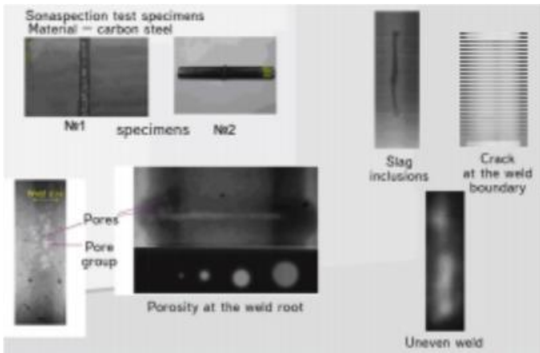


FIG. 7(b). Neutron inspection of flaws in welds.

Profile 21

IVV-2M

RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA OF THE IVV-2M

The multi-purpose nuclear research reactor IVV-2M is the heterogeneous water-cooled water-moderated reactor of the pool-type. It was constructed in 1966. The physical start-up of the IVV-2M reactor was on the 23 April 1966 and a power start-up was on the 18 October 1966. At present time the rated power of the reactor is 15 MW.

The main characteristics of the nuclear research reactor IVV-2M are provided in Table 1.

TABLE 1. MAIN CHARACTERISTICS OF THE NUCLEAR RESEARCH REACTOR IVV-2M

Reactor heat power	15 MW
Maximal neutron flux density: – thermal – fast (E > 0.1 MeV)	 5×10^{14} n·cm^{-2}·s^{-1} 2×10^{14} n·cm^{-2}·s^{-1}
Core height	500 mm
Effective core diameter	500 mm
Primary circuit coolant	Water
Secondary circuit coolant	Water
Primary coolant flow rate	to 1200 m^3/h
Secondary coolant flow rate	to 1200 m^3/h
Coolant temperature: – at the core inlet – at the core outlet	 ≤ 40°C ≤ 65°C
Number of stationary experimental channels: – horizontal – vertical	 10 6
Experimental devices in the core (loop, ampule): to Ø 60 mm Ø 120 mm Ø 130 mm Ø 190 mm Ø 400 mm	 to 40 2 1 1 1
Reactor cycle	300 to 500 h

The IVV-2M is used for a solution of the tasks as follows:

- Investigations of an operational capability of structure elements of new-type reactors;
- In-pile testing of the core materials of high temperature gas-cooled reactors in the anticipated operation (rated, transient and emergency) conditions;
- A justification of the structure and life time of the instrumentation of the thermo-ionic energy conversion;
- A research on an in-pile radiation effect on an instrumentation, sensors and semiconducting materials and different equipment;

— An experimental and calculation — experimental substantiation of nuclear, radiation and ecological safety and reliability of nuclear installations;
— Fundamental studies and experimental works in the field of nuclear power;
— A production of radioactive isotopes for technical and medical applications.

The nuclear research reactor IVV-2M is used for fundamental studies in solid physics, studies of materials' magnetic structure and a magnetic interactions origin.

From 1975 to 1988 the reactor was modernized through the installation of a corrosion-resistant steel vessel, steel-zirconium joints of the tubes of the horizontal experimental channels and additional upper biological shielding; and the replacement of a support grid of the core and reflector, a reactor vessel shell and an intake pipeline of the core, and the installation of a fuel cladding failure detection system; a transfer to tubular fuel assemblies of the type IVV-2M, the replacement of a heat-exchanger and a primary circuit pump, and the control and protection system.

The modernization permitted to improve neutron-physical characteristics of the core, to increase a rated power to 15 MW and extend the IVV-2M operation life- time to the 2025. The fuel assemblies employed in the core are of the type IVV-2M. The core layout varies depending on specified task for a reactor cycle and is based on the standard or newly developed core layouts. A core loading option is shown in Fig. 1.

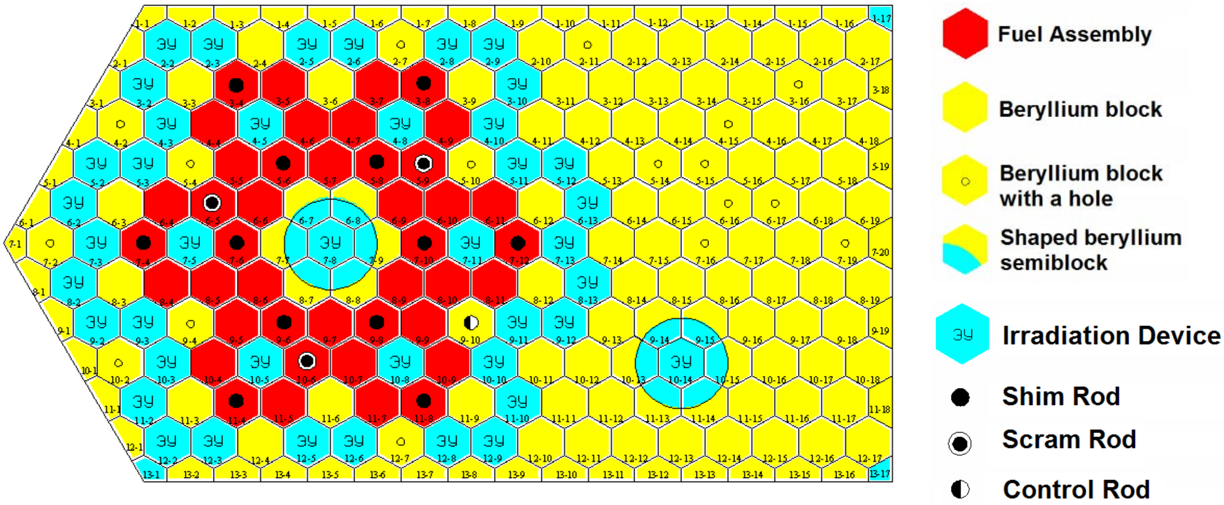


FIG. 1. IVV-2M reactor core with an experimental device in the central cell.

The fuel element assemblies consist of five tubular hexagonal fuel elements that are coaxially placed between hexagonal shroud tubes. The structural material of the fuel assemblies and claddings is an aluminium alloy. Each fuel element is a three-layer hexagonal tube that consists of meat, internal and external coatings, upper and lower end-caps.

In the operating loading of the core a fuel assemblies quantity can vary from 30 to 42 through the combination of an addition of fresh fuel assemblies and a rearrangement of the fuel assemblies that reached different burn-up values.

The primary coolant circulates through the fuel assemblies from top to bottom (downward) by a circulation pump installed in the reactor pool. Primary circuit heat is removed through the heat-exchanger, which is installed directly in the reactor vessel. The heat-exchanger is cooled with service water.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RESEARCH REACTOR IVV-2M

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TEST FACILITIES

The nuclear research reactor IVV-2M is equipped with ten horizontal experimental channels, including six radial and four tangential ones, and six vertical experimental channels. The horizontal channels are equipped with neutron diffractometers and spectrometers to investigate polycrystal and monocrystal materials.

The IVV-2M is equipped with test-beds and installations to study and test fuel, fuel element mock-ups, structural materials and other components of the cores of reactor facilities of different application.

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE (fuel, structural materials, coolant technologies: lead, gas) AT STEADY-STATE CONDITIONS

Low-pressure loop is intended to test structural and fuel materials in an environment of a water coolant of a special composition at a coolant temperature of 60–100°C and a water rate of not more than 4 m^3/h. Controlled parameters during the tests are: a flow rate, a temperature and a coolant composition.

Lead-coolant loop (LCL) is intended to solve the tasks in designing reactors with a heavy liquid-metal coolant (lead). The LCL simulates the operation conditions of a reactor facility with a lead coolant and permits to investigate a number of processes that are important for its safety operation:

- A corrosion of structural materials under in-pile and out-of pile conditions at different temperatures (420–540°C);
- An activation of a coolant and a transfer of radionuclides and corrosion products in the primary coolant circuit;
- A radionuclide release into a reactor gas-space and a precipitation onto structure components of the reactor core;
- Processes of a radionuclide release through a lead mirror, a transfer and a precipitation of these products into a gas cavity.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS

Not available at the reactor IVV-2M.

2.4. FACILITIES FOR INVESTIGATIONS OF CORROSION OF REACTOR MATERIALS

The facility for corrosion testing of materials in a flow of superheated steam permits to perform tests under the conditions simulating the emergency mode conditions of operation of reactor facility structural components at a temperature to 1000°C and a steam flow rate to 100 m/s. The facility's corrosion chamber is installed in the reactor core. Water is fed from a deaerator through a steam generator, where the steam is preliminary heated with an electric heater, and then it is delivered to a working chamber. Superheating of steam is provided by a radiation heater. The controlled parameters during the tests are: a temperature, a composition and rate of environment.

2.5. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENTS AT A WIDE RANGE OF TEMPERATURES AND DOSE RATES, etc.

The experimental devices available for the irradiation of structural material specimens in the core attain and maintain temperature within 60–1500°C. When the pre-set damage doses are reached the specimens are sent to hot cells for post-irradiation mechanical tests and investigations.

Test parameters are as follows:

- Exposure time to 7000 hours per year;
- Neutron flux density to 2×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$ ($E > 0.1$ MeV);
- Irradiation environment: vacuum, gas (nitrogen, helium), dilute gas or gas mixtures.

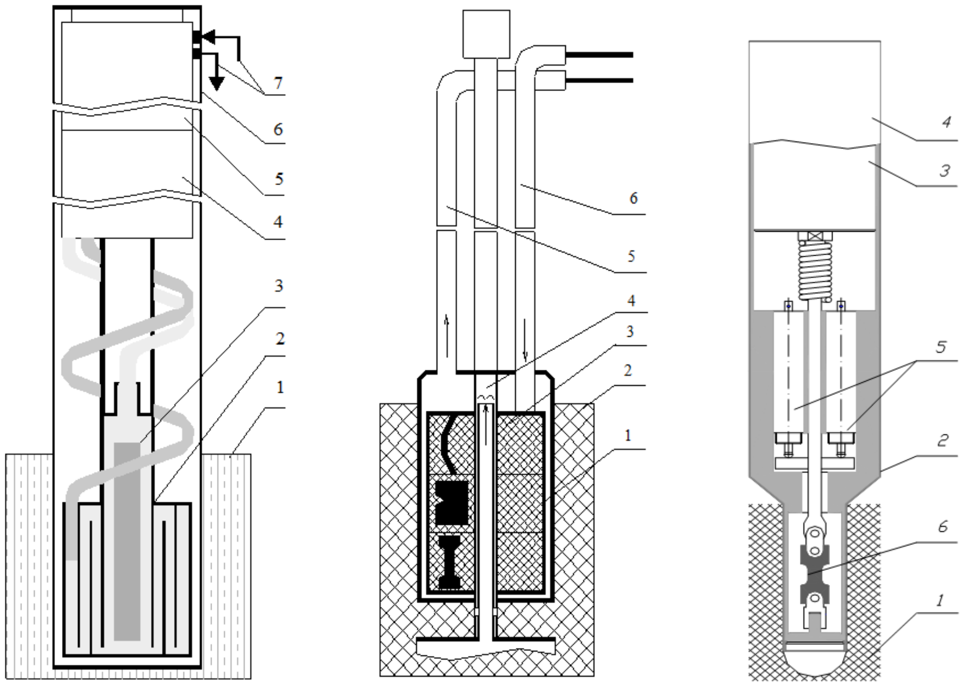
2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (swelling, gas releases, creep, etc.)

The experimental devices complex for the short-term and long-term mechanical tests of the 'URAL' series permits to carry out short-term and long-term mechanical tests under uniaxial loading of specimens exposed to in-pile irradiation. The complex comprises the following devices for:

- A determination of short-term mechanical properties;
- Creep tests;
- Durability tests.

The experimental devices are placed in a vertical channel so that test specimens are in the core.

A loading mechanism ensures a pre-set strain rate while determining short-term mechanical properties or a desired strain level by using a sequentially imbedded elastic element (a spring) in the creep tests and durability tests.



(a) (b) (c)

(a) The facility for corrosion testing materials in a superheated steam flow:
1 – the IVV-2M core;
2 – a superheater of steam;
3 – a container with specimens;
4 – a travel mechanism;
5 – a biological shielding;
6 – a casing;
7 – an inlet and outlet of steam.

(b) The facility for capsule/ampule testing materials:
1 – an irradiation device;
2 – the core;
3 – stake sockets with specimens;
4 – a central cooler;
5 and 6 – pipelines of feed and removal of a process gas.

(c) The experimental devices complex for short-term and long-term mechanical tests:
1 – the core;
2 – an irradiation device;
3 – a loading mechanism;
4 – a travel mechanism;
5 – detectors of linear travels;
6 – a specimen.

FIG. 2. Irradiation devices.

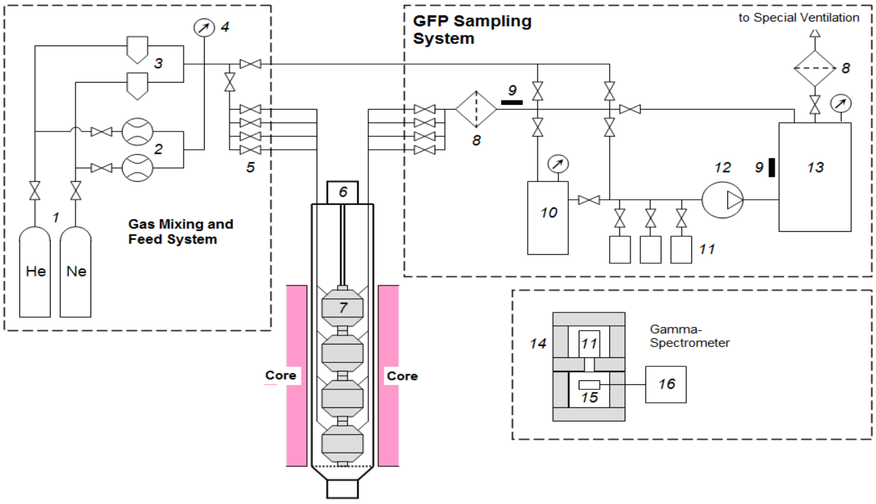
The test-bed 'RISK' provides in-pile tests of fuel and dummies of fuel elements of WWER, BN, RBMK reactors, pebble-bed fuel elements/fuel compacts and micro-fuel of HTGR at temperatures from 400 to 2100°C in inert gas environment (helium, neon, helium-neon mixtures of pre-set composition with excess pressure from 0.1 to 1.5 bar) and gamma-spectrometric measurements of GFP release from fuel in a periodic sampling regime and on-line mode (with a gas rate from 1 to 10 cm^3/s through a Marinelli glass).

The irradiation devices of 'ASU-18' series are intended for testing fuel pellets and fuel element dummies at a fuel temperature from 500 to 2100°C and a specific power density in fuel to 1000 W/cm^3 at GFP (*R/B*) release from 10^{-4} to 10^{-1}. The properties under study are a control of gas swelling of fuel; a determination of diffusion coefficient of GFP in fuel.

The irradiation devices of 'MT' series are intended for testing coated particles (CP) and fuel compacts of HTGR at a fuel temperature from 500 to 1800°C and a specific power density in fuel to 4500 W/cm^3 at GFP (*R/B*)-release from 10^{-4} to 10^{-1}. The properties under study are a control of a leak tightness of CP coatings; a monitoring of uranium contamination of CP coatings; a control of a fraction of damaged CP; a determination of an effective diffusion coefficient of GFP in fuel kernel and coatings of CP.

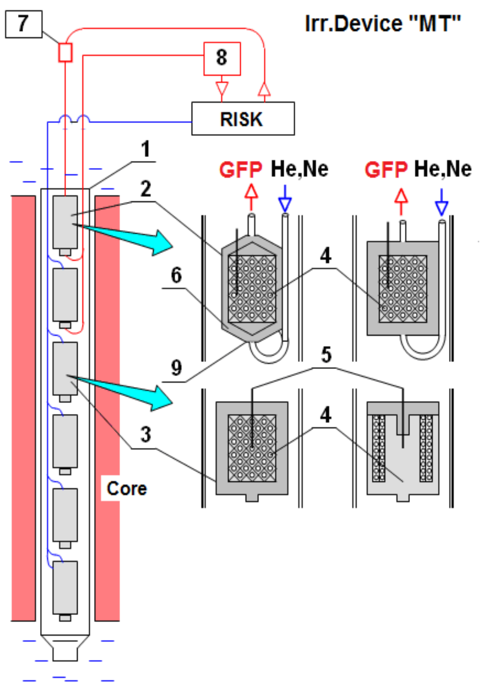
The irradiation devices of 'VOSTOK' series are intended for testing full-scale pebble-bed fuel elements and HTGR matrix-graphite specimens at a fuel/specimens temperature from 500 to

1800°C and a specific power density in fuel to 4500 W/cm^3 at GFP (R/B) release from 5×10^{-9} to 10^{-1}. The properties under study are the control of a leak tightness of pebble-bed fuel; the monitoring of uranium contamination of matrix graphite; the control of a fraction of damaged CP.



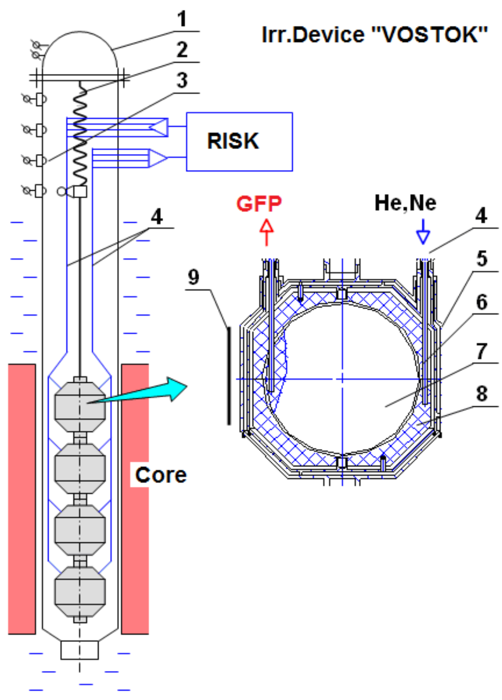
1 – cylinders with helium, neon;
2 – flowmeters;
3 – gas feeding metering tanks;
4 – pressure detector;
5 – gas manifold;
6 – mechanism of capsule travel along the core height;
7 – capsule with specimens;
8 – iodine filter;
9 – dosemeter;
10 – accumulator tank;
11 – samplers;
12 – fore-vacuum pump;
13 – GFP cooling tank;
14 – lead shielding;
15 – Canberra gamma-detector;
16 – computer.

FIG. 3. Block-diagram of 'RISK'.



1 – body;
2 – ampule;
3 – capsule;
4 – specimen;
5 – thermocouple;
6 – adsorber;
7 – loading hopper;
8 – sampler;
9 – mesh separator.

FIG. 4(a). Irradiation device 'MT'.



1 – electric drive;
2 – helical gas pressure regulator;
3 – limit switch;
4 – gas paths;
5 – capsule;
6 – thermocouple;
7 – pebble-bed fuel element;
8 – graphite thermal insulation;
9 – thermal-neutron detector.

FIG. 4(b). Irradiation device 'VOSTOK'.

The test-bed 'PURS' provides in-pile tests of thermionic electro-generating channels (EGCh) for two-mode nuclear power installations of space application. The test-bed is rigged with modern gas-vacuum equipment and a loading-diagnostic device that sustains electric parameters of the EGCh in a wide range of power, the measuring system (that records to 150 different parameters of EGCh with a frequency of one time per second) an on-line gamma-spectrometry of GFP release from fuel.

The test-bed 'PURS-RUGK' provides in-pile tests of full-size fuel elements for space nuclear power engines of a megawatt class in inert environment (helium, neon, helium-neon mixtures of a preset composition under pressure). The test-bed is rigged with a modern gas-vacuum and measuring systems, that maintain the preset temperature of fuel coatings and record to 80 different parameters with a frequency of one time per second.

The test-bed 'RITM-FM' is intended for functional, in-pile tests of tritium breeding mock-ups of breeding blanket elements of thermonuclear reactors. In the in-pile tests this test-bed provides mesurements of tritium permeability through structural materials and on-line recording of a tritium composition and speed of production.

The test-bed comprises the systems and components as follows:

- Experimental channel device;
- Gas-vacuum system, that support the following functions:
  - Preparation, feed and parameter maintenance of a gas -carrier;
  - Circulation of a gas-carrier with a help of the metal bellows pump MB-41;
  - Clean-up of a gas-carrier from tritium;
  - Tritium recording system based on Tritium Monitor Kit 7500-MKIT-001-C produced by Tyne Engineering Inc.

TABLE 2. TEST PARAMETERS

Model	1	2
Mass of a breeder material (Li_4SiO_4)	67.2 g	54.5 g
Typical size	pebble 1–1.5 mm	pellet Ø – 11 mm, height – 10 mm
Test temperature	220–550°C	340–850°C
Lithium burn-up	0.91%	1.31%
Tritium production	8.9 Ci/g	11.4 Ci/g

FIG. 5. RITM-FM test facility.

2.7. OTHER FACILITIES

Not available.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTICS

The JSC ‘INM’ has got the following capacities towards development of innovative nuclear energy systems and technologies with regards to experimental nuclear materials logistics:

- Storage facilities for fresh and irradiated nuclear materials;
- Transportation infrastructure to transport radioactive materials;
- Transport containers for nuclear materials, radioactive substances and radioactive waste;
- Governmental licenses granting authorization to perform related specified activities, and to use specialized transport means and package designs.

3.2. HOT CELLS, POST-IRRADIATION EXAMINATION FACILITIES

The complex of equipment to perform post-irradiation examination of irradiated materials in hot cells and shielding boxes includes:

- NDT installations and instrumentation: a precision measurement of geometrical sizes, visual examination, photography, video-recording, defectoscopy, gamma-scanning;
- Machines for a fragmentation of items into sections, making specimens of different configurations for all kinds of post-irradiation examinations;
- Machines for mechanical testing to determine characteristics of strength, plasticity at temperatures to 1800°C, crack resistance, durability, low-cycle fatigue, impact elasticity;
- Automatic system to analyse micro- and macro-hardness of materials;
- The measuring complex ‘Zvuk’ to determine absolute values of elastic characteristics: Young’s module (E), shear module (G) and Poisson’s ratio (μ) of isotropic materials;
- Quarts dilatometer;
- Remote analytical balances for gravimetric and hydrostatic weighing;
- Equipment for preparing specimens for microstructural investigations;
- Metallographic microscopes of different application with a multifunctional system of pattern analysis;
- Transmission and scanning electron microscopes;
- X-ray diffractometers;
- Quadrupole mass-spectrometer for the isotopic analysis of gas environments and solid specimens (metals and insulators), a depth analysis and a surface cartography;
- Gamma- and alpha-spectrometers;
- Facilities to control a gas environment composition and a gas content in inorganic solid materials;
- Electrochemical complex for corrosion and electrochemical examinations of steels and zirconium alloys;
- Complex of in-chamber installations for post-irradiation examination of HTGR fuel that comprises;
- Facility for a zone electrolytic disintegration of HTGR pebble-bed fuel elements and fuel compacts;
- Facility for selective etching of protective claddings of micro fuel elements;

- Facility for a determination of a fraction of defective coated particles by the IMGA method;
- Facility for a determination of burn-up of HTGR pebble-bed fuel.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

The structure of the JSC ‘INM’ includes the experimental-and-design division, that develops and manufactures experimental facilities and rigs to perform in-pile tests of fuel and structural materials with maintenance of the test parameters: temperature, load, strain rate, environment composition and pressure as prescribed by a customer.

There is a wide experience gained by the personnel in the development, manufacturing, adjustment and operation of experimental in-chamber installations and irradiation devices for in-pile tests of structural materials and nuclear fuel.

4. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS

4.1. STRUCTURAL MATERIALS OF FAST NEUTRON REACTORS

The regularities of the evolution of a structure and properties of structural materials of fast neutron reactors were obtained, they were used for:
- Prediction of maximum life-time of safety operation;
- Optimization of a composition and modes of a fuel-coating production;
- Development of new steels of austenite and ferrite classes;
- Determination of remaining safe operation life-time of fuel elements with different options of coatings.

The influence of vacancy porosity on mechanical properties of austenitic steels was determined. The model was developed to calculate a strength deterioration induced by the vacancy porosity from the characteristics of radiation porosity.

The regularities of a radiation parameters influence on radiation swelling and porosity characteristics of fuel coatings made of austenitic steels were found.

The causes of the onset of a steady-state swelling phase were determined. The apparatus was developed for a determination of the initial moment of a steady-state swelling phase. The steady-state swelling rates in steel ChS68 at different temperatures of neutron irradiation were calculated and proved by experiments.

The regularities of the evolution of austenitic steels microstructure exposed to high dose neutron irradiation in a wide range of temperatures were determined.

The regularities of influence of high dose neutron irradiation on physic-mechanical properties were found. The model was developed to calculate a variation of characteristics of elasticity and electric resistance induced by radiation swelling.

The model of a mechanical stress influence on austenitic steel swelling was developed. It was verified by the experimental results obtained from the examination of specimens of materials-science assemblies after their irradiation in the reactor BN-600. The mechanism of pore nucleation of a critical size in austenitic steels under neutron irradiation was revealed. A notion of incubation period of swelling under neutron irradiation was formulated correctly and ways to calculate a swelling incubation dose were found.

4.2. INVESTIGATIONS OF DIOXIDE URANIUM FUEL AND MIXED URANIUM-PLUTONIUM FUEL

The evolution of a structure and composition of uranium and uranium-plutonium fuels and their interaction with fuel cladding in operation in the reactor BN-600 are described. The investigation results obtained were used for:

- Prediction of behaviour of dioxide and mixed U-Pu fuel;
- Determination of remaining operation life-time of fuel elements with different variants of the dioxide fuel and cladding.

4.3. DEVELOPMENT OF MATERIALS FOR FAST NEUTRON REACTORS

The results obtained during many years of studies of fuel elements after their operation life were used to develop materials for fast neutron reactors.

4.4. PHYSICS OF RADIATION DAMAGE OF METALS

4.4.1. Experimental methods of investigation of displacement cascades

Methods of experimental investigations of displacement cascades regions and vacancy clusters were developed and integrated into practice.

4.4.2. Evolution of radiation defects in metals

Evolution of radiation defects in steels at different irradiation temperatures was classified. Characteristics of intrinsic radiation point defects and their complexes formed in pure metals and alloys were obtained.

4.5. RESEARCH REACTOR FUEL AND STRUCTURAL MATERIALS

In-pipe tests parameters influence on a porosity formation, structure changes and defect formation in fuel elements with dispersion uranium-molybdenum fuel, necessary to predict their performance capability, was studied. Data on corrosion resistance of aluminium alloys in the operation conditions of research reactor fuel claddings were obtained. They permit to make a choice of most corrosion and erosion resistant alloys and predict life-time of their operation.

4.6. RBMK REACTOR MATERIALS

These works were performed to substantiate and increase life-time safe operation of RBMK fuel channels.

4.7. IN-PILE TESTS, PRE-IRRADIATION AND POST-IRRADIATION INVESTIGATIONS OF FUEL AND FUEL ELEMENTS OF HTGR

Fuel quality was estimated for reactors HTR-10 (China) and PBMR (SAR).

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Profile 22

MIR.M1

RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA

The research reactor MIR.M1 is a heterogeneous loop- and channel-type reactor operated with a thermal neutron spectrum and immersed in a pool of water. It is designed for:

- Loop tests of dummies of fuel rods and assemblies from nuclear reactors of different types and purposes;
- Tests of structural materials and items of nuclear reactors of different types and purposes;
- Accumulation of radionuclide preparations.

The main research base of the reactor includes loop facilities with in-pile experimental channels having autonomous process circuits with various coolants.

FIG. 1. Main building of reactor MIR.M1.

The research reactor achieved its first criticality on 24 December 1966, and was commissioned on 11 August 1967 when startup and adjustment of its systems and experimental rigs was conducted. The experimental research activities in the reactor started in 1968.

In 1975, the reactor was reconstructed and since then it has been named MIR.M1. The reconstruction covered changes of the loop channels arrangement in the core, and now each channel is surrounded by 6 operating FAs and 4–5 control rods. It made all the core loop cells equally worth; the conditions for individual control of test modes in each channel were improved.

Continuous improvements and extension of the reactor operation make it possible to utilize the reactor for research purposes for approximately 15 more years.

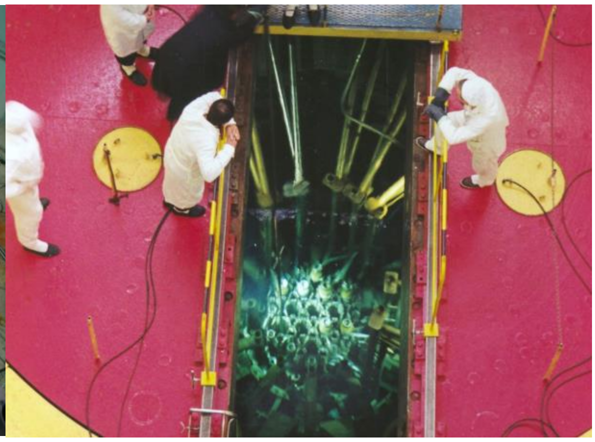


FIG. 2 (a). Central hall of the reactor.

FIG. 2 (a). Reactor pool.

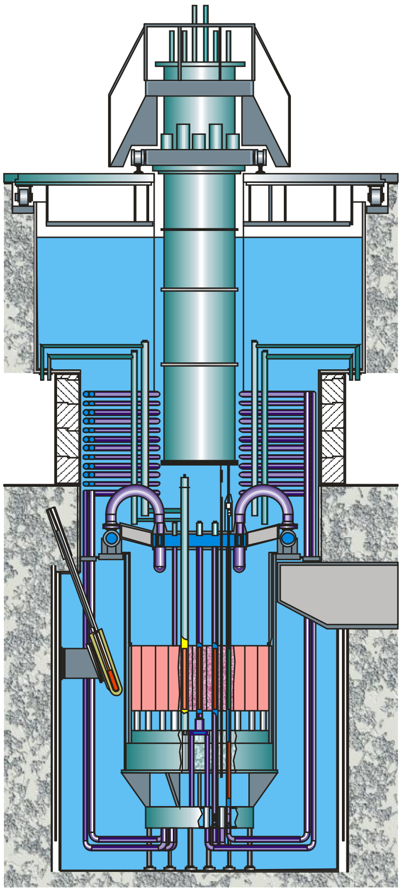


FIG. 3 (a). MIR.M1 reactor layout.

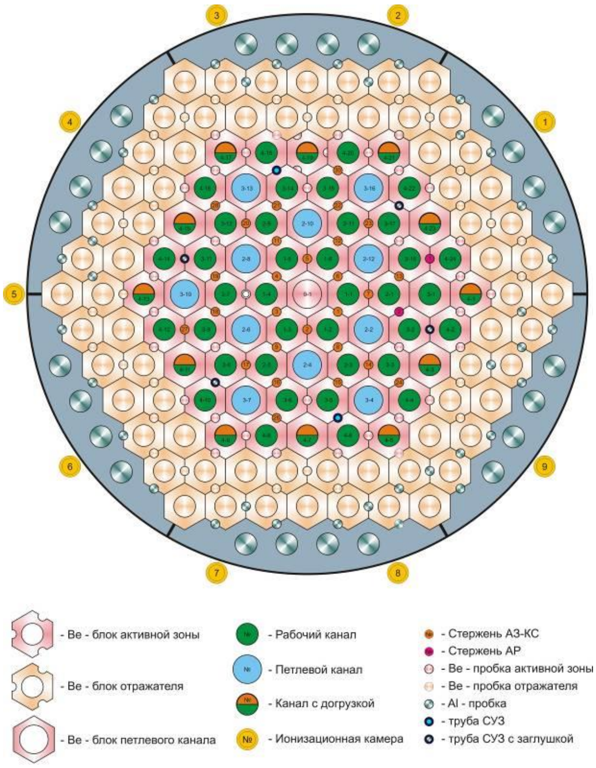


FIG. 3 (b). MIR.M1 reactor core arrangement.

TABLE 1. KEY TECHNICAL SPECIFICATIONS

Maximal thermal power	100 MW
Loop channel diameter	≤148
Number of loop channels, max.	11
Thermal neutron flux density in the experimental channel	$\leq 5 \cdot 10^{14} \cdot n \cdot cm^{-2} \cdot s^{-1}$
Volumetric heat rate in the core	0.85 MW/l
Coolant:	Water
– pressure at the reactor core inlet	1.25 MPa
– temperature at the reactor inlet	≤70°C
– temperature at the reactor outlet	≤98°C
Fuel cycle duration	up to 40 days
Operating time at power,	~ 240 d/year
Scheduled operation time	2015
Planned operation time	More than 2030

The reactor MIR.M1 is a nuclear research facility with heat removal from the reactor to the cooling tower using a double-circuit scheme with five lines in the primary circuit and two lines in the cooling circuit.

The primary circuit incorporates the main circulation pumps, heat exchangers, pressure compensators, and valves and pipelines connecting this equipment. It is designed for heat removal from the core and its transfer to the recirculation water supply circuit, as well as for retaining of the active medium. Circulation of the primary coolant through FAs is downwards.

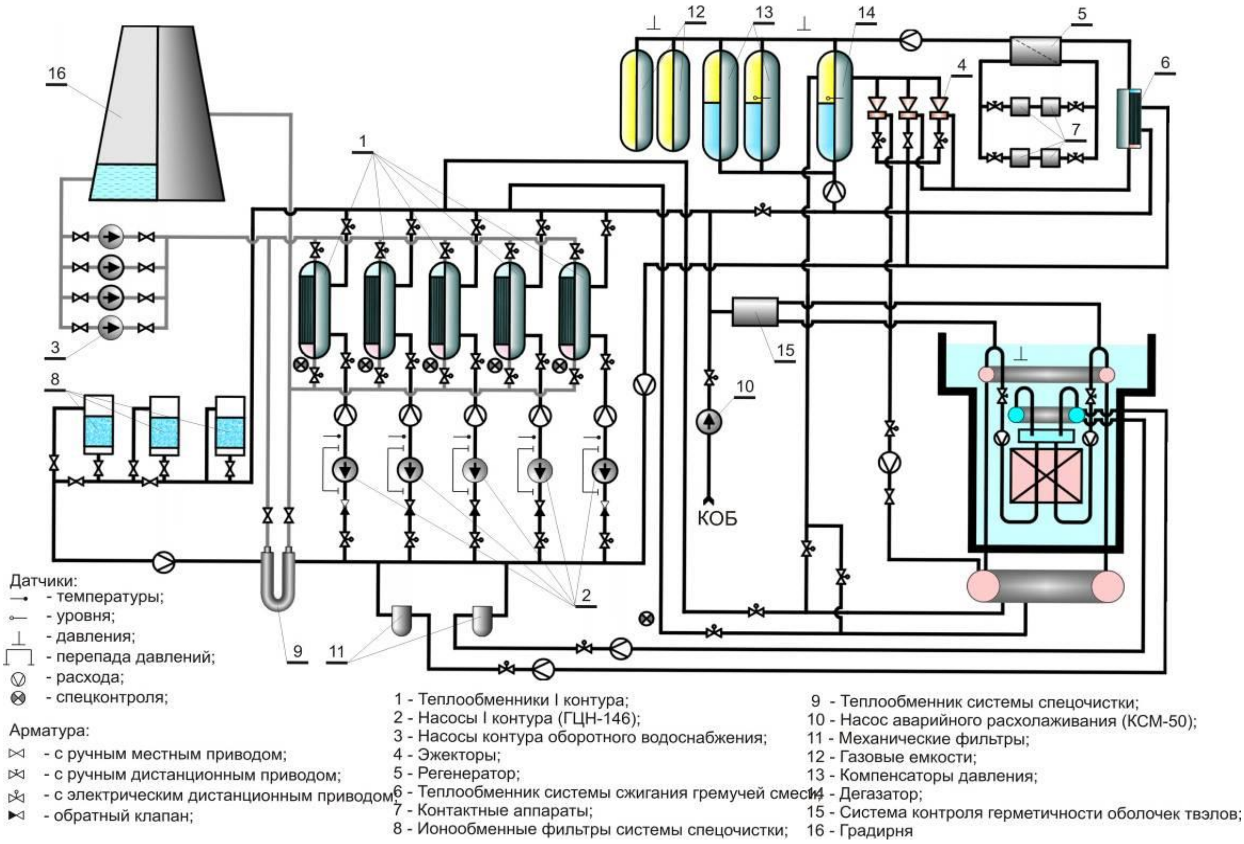


FIG. 4. Principle diagram of the primary circuit in reactor MIR.M1.

To create biological shielding, provide safety during maintenance and reloading operations under water, enhance safety of the reactor and limit radioactive releases in accidents, the reactor core is immersed in a pool of water that is a part of the reactor pool cooling circuit. The pool cooling circuit is designed for removal of radiation heat in Be blocks, control rod actuators (except for movable FAs with absorbers) and structural elements of the reactor.

The coolant of the pool cooling circuit circulates downwards through the gaps (1.5 mm) between Be blocks of the core stacking, control rod pipes and process gaps in the reactor structures.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL CAPACITIES AND FACILITIES

2.1. LOOP FACILITIES

Reactor MIR.M1 is designed for tests of experimental fuel rods and structural materials of nuclear facilities of different purposes (transport, power) operating under various loads in various environments (gas, water, liquid metals and organic compounds). The key feature of the reactor is availability of 11 experimental loop channels in the core. The channels are connected to autonomous loop facilities with various coolants to perform tests under various thermal-hydraulic conditions.

Nowadays the reactor operates seven loop facilities each of which is connected to 1–2 loop channels where experimental rigs with fuel rods and FA dummies are accommodated.

2.1.1. Water cooled loop facility PV-1

It is designed for in-reactor tests of fuel rods and structural materials of water-cooled nuclear reactors. The facility is connected to one or two in-reactor experimental loop channels.

TABLE 2. MAXIMAL PARAMETERS (PV-1)

Thermal capacity of the facility	2000 kW
Thermal capacity of one channel	1000 kW
Pressure in the primary circuit	17 MPa
Coolant temperature at the channel inlet	300°C
Coolant temperature at the channel outlet	350°C
Coolant flow rate through the channel	16 t/h

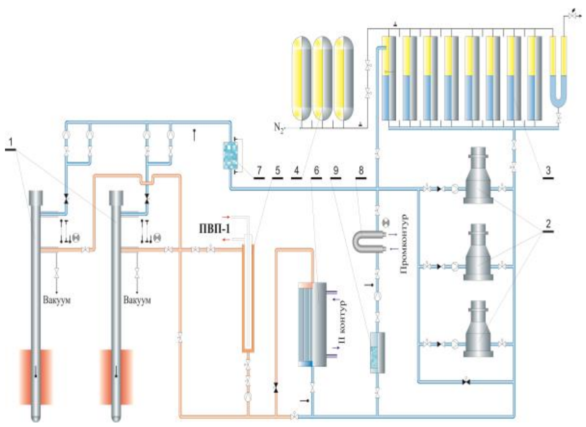


FIG. 5 (a). PV-1 loop facility circuit.

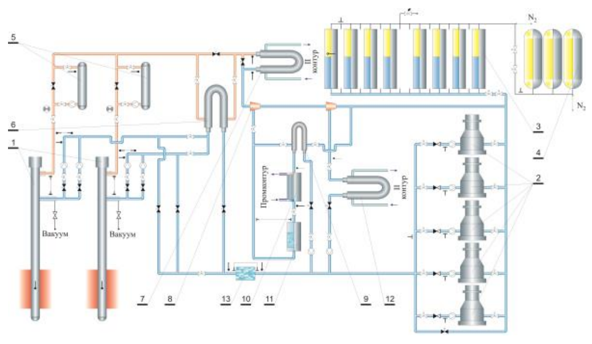


FIG. 5 (b). PV-2 loop facility circuit.

2.1.2. Water cooled loop facility PV-2

It is designed for in-reactor tests of fuel rods and structural materials of water-cooled nuclear reactors. The facility is connected to one or two in-reactor experimental loop channels.

TABLE 3. MAXIMAL PARAMETERS (PV-2)

Thermal capacity of the facility	2500 kW
Thermal capacity of one channel	1500 kW
Pressure in the primary circuit	18 MPa
Coolant temperature at the channel inlet	310°C
Coolant temperature at the channel outlet	350°C
Coolant flow rate through the channel	16 t/h

2.1.3. Water — boiling water cooled loop facility PVK-1

It is designed for in-reactor tests of fuel rods and structural materials of water-cooled nuclear reactors both under coolant boiling and without it. The facility is connected to one or two in-reactor experimental loop channels.

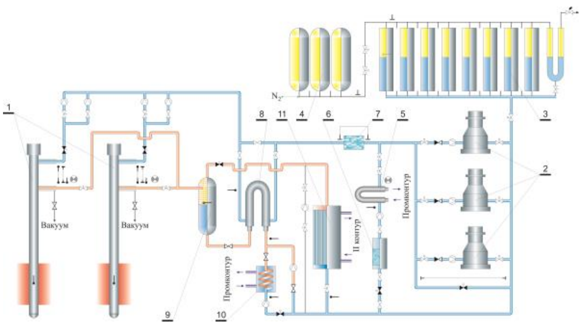


FIG. 6 (a). PVK-1 loop facility circuit.

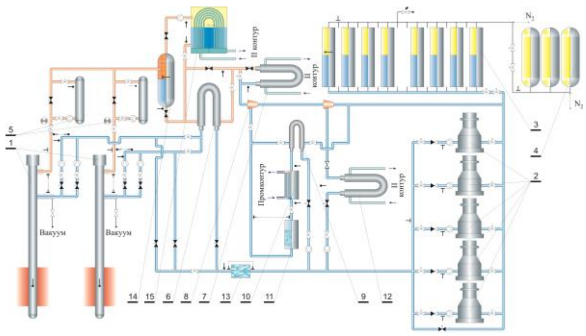


FIG. 6 (b). PVK-2 loop facility circuit.

TABLE 4. MAXIMAL PARAMETERS (PVK-1)

Thermal capacity of the facility	2000 kW
Thermal capacity of one channel	1000 kW
Pressure in the primary circuit	17 MPa
Coolant temperature at the channel inlet	300°C
Coolant temperature at the channel outlet	350°C
Coolant flow rate through the channel	16 t/h;
Steam weight fraction in the channel	40%

2.1.4. Water — boiling water cooled loop facility PVK-2

It is designed for in-reactor tests of fuel rods and structural materials of water-cooled nuclear reactors both under coolant boiling and without it. The facility is connected to one or two in-reactor experimental loop channels.

TABLE 5. MAXIMAL PARAMETERS (PVK-2)

Thermal capacity of the facility	2500 kW
Thermal capacity of one channel	1500 kW
Pressure in the primary circuit	18 MPa
Coolant temperature at the channel inlet	350°C
Coolant temperature at the channel outlet	355°C
Coolant flow rate through the channel	16 t/h;
Steam weight fraction in the channel	40%

2.1.5. Steam cooled loop facility PVP-1

It is designed for lifetime tests of experimental fuel rods, fuel assemblies and structural materials in the core loop channel. In the primary circuit of the coolant circulation, slightly overheated steam is generated in the heat exchanger, and this steam cools FAs of the loop channel and transfers heat to the condenser. The facility is connected to one experimental loop channel.

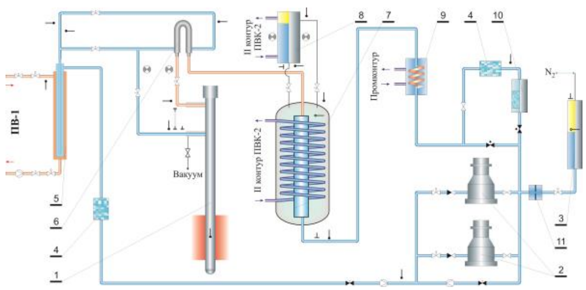


FIG. 7. PVP-1 loop facility circuit.

TABLE 6. MAXIMAL PARAMETERS (PVP-1)

FA capacity	100 kW
Steam temperature at the FA inlet	300°C
Steam temperature at the FA outlet	510°C
Fuel rod cladding temperature	655°C
Coolant pressure at the FA outlet	6.5 MPa

2.1.6. Steam cooled loop facility PVP-2

It is designed for the investigation and validation of performance of the core elements from advanced power reactors within a wide range of temperature and pressure, including accident conditions of fuel rod operation.

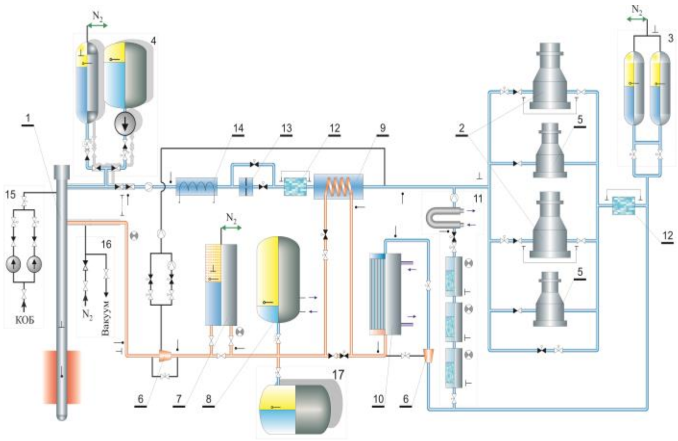


FIG. 8. PVP-2 loop facility circuit.

TABLE 7. MAXIMAL PARAMETERS OF THE FACILITY PRIMARY CIRCUIT

Steam temperature at the outlet of Experimental fuel assembly	1200°C
Coolant temperature at the channel outlet	550°C
Coolant flow rate through the channel	10 t/h
Pressure	20 MPa

2.1.7. Gas cooled loop facility PG-1

It is designed for comprehensive investigations of performance of fuel rods and fuel assemblies from advanced power reactors with gas coolant of the HTGR type. Fuel rods under study are located in the loop channel placed in the reactor core cell. Heat is removed by gas coolant circulating through the primary circuit and transferred to the working medium in heat exchangers of two parallel circuits of the equipment cooling system. The cooling system in turn transfers heat to the intermediate circuit and recirculation water supply circuit.

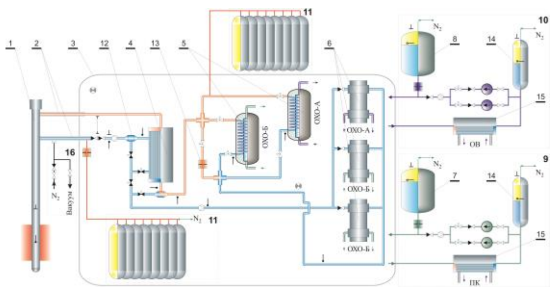


FIG. 9. PG-1 loop facility circuit.

TABLE 8. MAXIMAL PARAMETERS OF THE FACILITY PRIMARY CIRCUIT

Coolant temperature at the channel outlet	600°C
Coolant flow rate through the channel	1.3 kg/s
Pressure	20 MPa

2.2. EXPERIMENTAL TECHNIQUES FOR FUEL TESTING

Types of experimental fuel assemblies:

— Dismountable and instrumented device for testing fuel rods ~1000 mm, containing up to 19 fuel rods;
— Dismountable devices for testing short-size (~ 250 mm) fuel rods, up to four such rigs can be installed one over another in one loop channel;
— Device for combined irradiation of refabricated (~1000 mm) and full-size fuel rods (≤ 3800 mm) of spent NPP fuel;
— Dismountable devices for power cycling and RAMP experiments of instrumented fuel rods by displacement or rotation of the absorbing screens in the experimental channel;
— Instrumented device for testing under LOCA and RIA conditions.

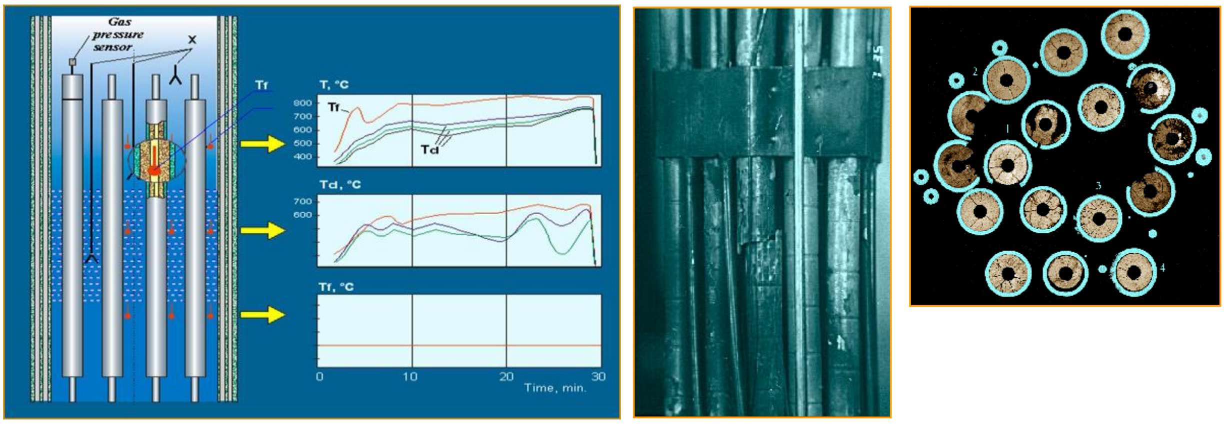


FIG. 10. Technique for simulation of loss of coolant and partial core dry-out accident (LOCA).

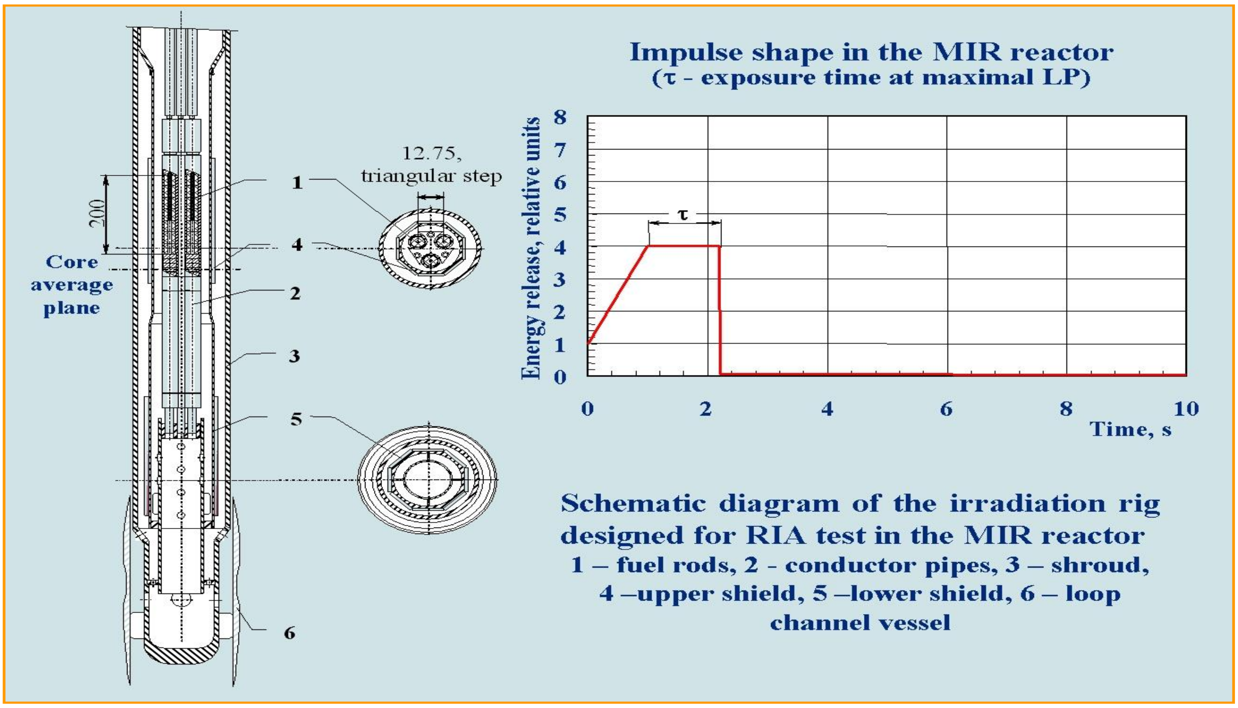


FIG. 11. Technique for simulation of reactivity inserted accident (RIA).

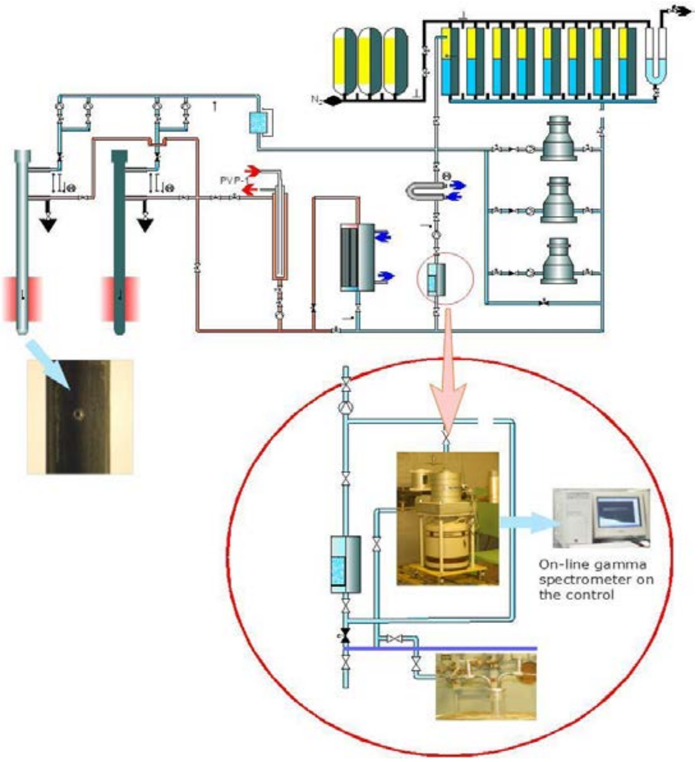


FIG. 12. Technique for investigation of fission products release in the coolant of leaking fuel rods at different burn-up.

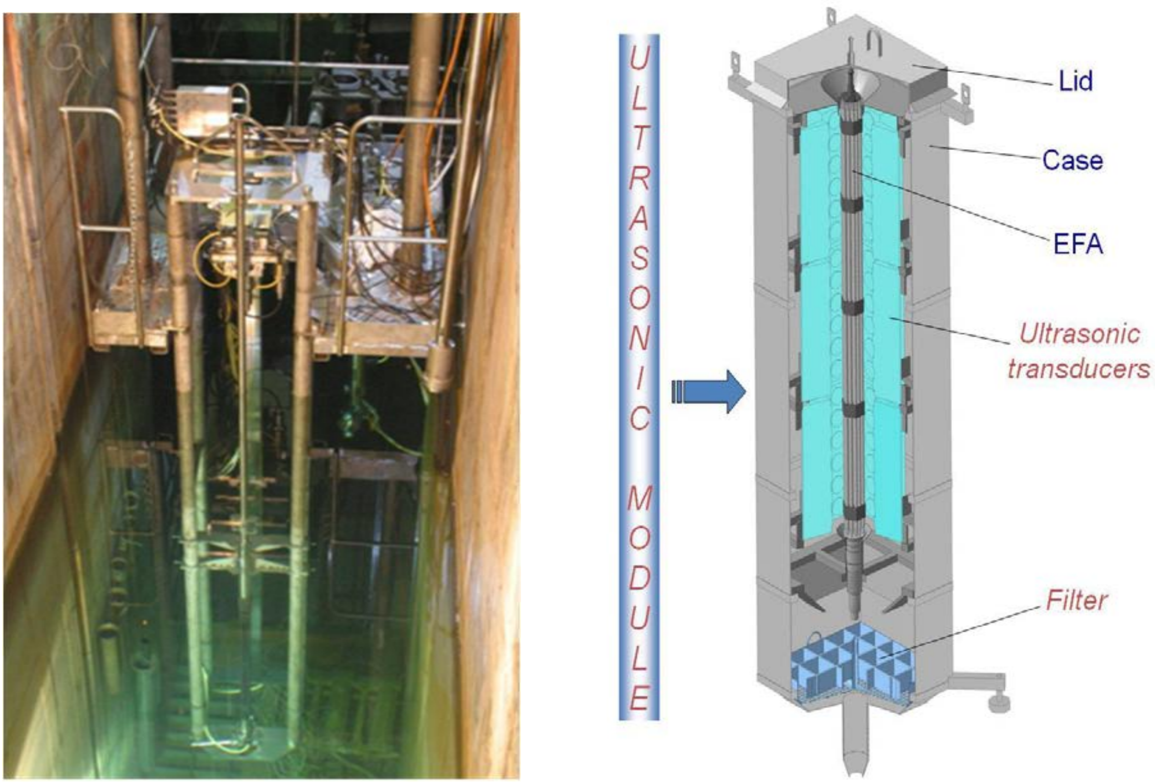


FIG. 13. Technique for irradiated fuel rods inspection and fuel rods ultrasonic cleaning facility in the MIR reactor storage pool.

3. RELATED ENGINEERING INFRASTRUCTURE

MIR.M1 is one of the installations of the largest nuclear centre Research Institute of Atomic Reactors (RIAR) which incorporate:

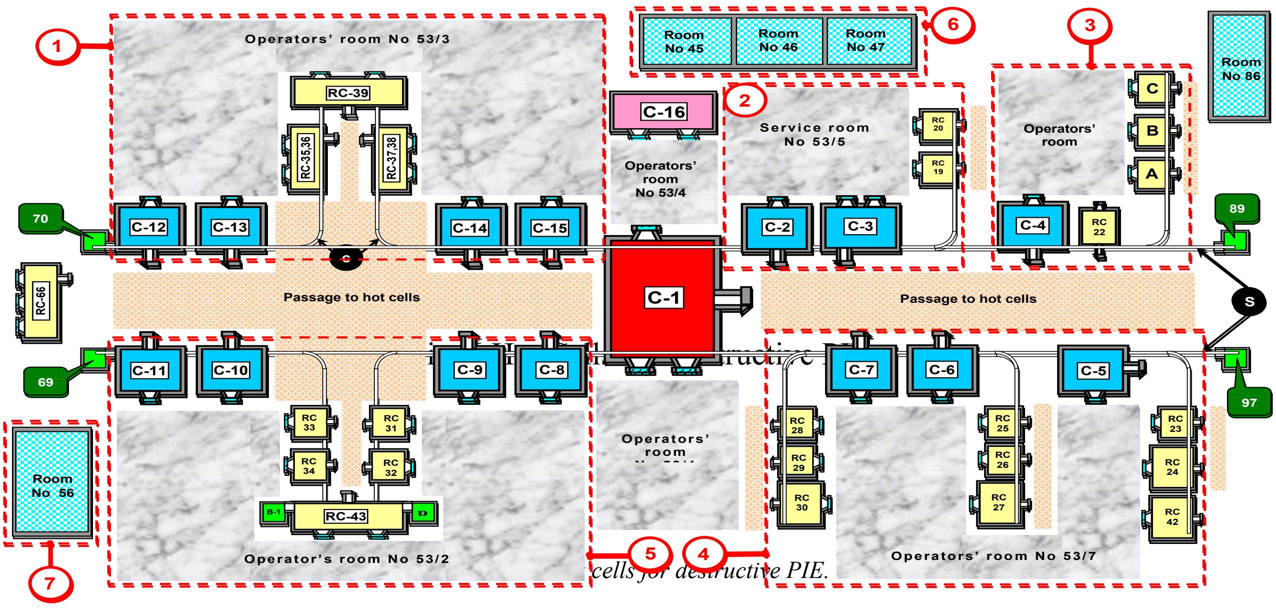
— Research reactor complex consisting of reactors that have laboratories to prepare and perform experimental research;
— The largest in Europe material testing complex comprising hot cells and glove boxes for non-destructive and destructive post-irradiation examinations of core components and wide range of irradiated materials;
— Facilities and technologies for investigation and pilot production of nuclear fuel;
— Radiochemical complex to study and produce trans plutonium elements, radioisotopes and sources for industrial and medical purposes;
— Facilities to dispose radioactive waste and nuclear materials.

RIAR hosts a number hot cells and boxes to perform post-irradiation non-destructive and destructive examinations of core elements and a wide range of irradiated materials. The Material Testing Complex is accommodated in three adjacent buildings housing more than 110 equipment items (devices and facilities) to carry out tests and examinations of irradiated materials and fuel.

There are 7 special large hot cells for investigation of full-size SFAs from NPP, more than 40 hot cells for destructive examinations equipped with up-to-date machines for mechanical testing, examination and analysis of structural and fuel materials.

The non-destructive analysis of full-scale fuel rods and assemblies is possible (measurements of geometric characteristics; visual examinations; gamma scanning; eddy-current and ultrasonic).

The destructive analyses comprise burn-up determination, fission products release, metallographic and ceramographic, micro-hardness; density and porosity, thermal conductivity and electric resistance, X-ray analysis; TEM, SEM, EPMA, AES, SIMS; mechanical testing (tensile, compression, bending, impact etc.).



cells for destructive PIE.

FIG. 15 (a). TEM JEM 2000 FX II.

FIG. 15 (b). Zwick 5113 impact testing machine.

FIG. 15 (c). Instron 1362-DOLI.

FIG. 15 (d). SEM Phillips XL 30 ESEM.

FIG. 15. PIE equipment.

4. RECENT AND CURRENT IRRADIATION PROGRAMMES

The following types of fuel rods and FAs are subject to reactor tests and investigations:
- Modern and advanced designs for power water-cooled reactors;
- For gas-cooled reactors;
- For research reactors.

Intermediate examinations of FAs and fuel rods in hot cells and the reactor pool are provided.

The following tests of refabricated and full-size WWER fuel rods were conducted:
- Long-term tests and tests with simulation of manoeuvring conditions (up to 300 cycles);

— Tests with violation of normal operation conditions and under conditions simulating various accidents (RAMP, LOCA and design-basis RIA experiments;
— Leaky fuel rods under steady state and transient conditions.

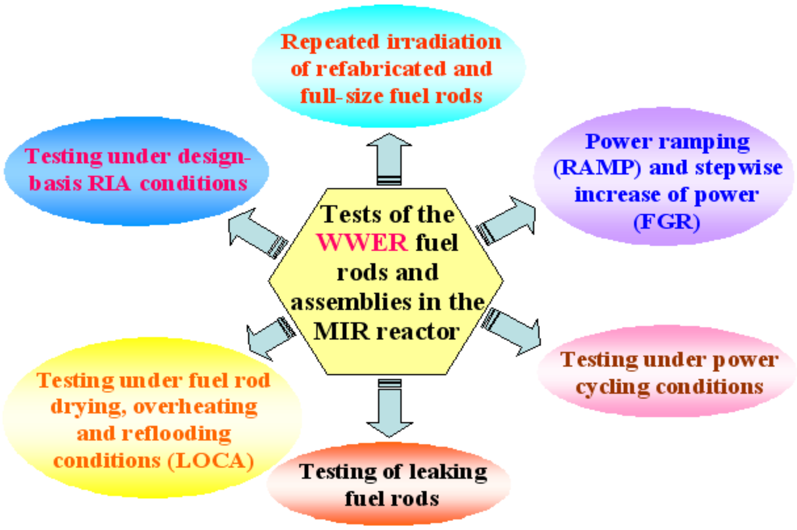


FIG. 15. Diagram of testing and investigation for improving WWER reactor fuel in the MIR reactor.

Profile 23

SM-3

RUSSIAN FEDERATION, DIMITROVGRAD

1. GENERAL INFORMATION AND TECHNICAL DATA

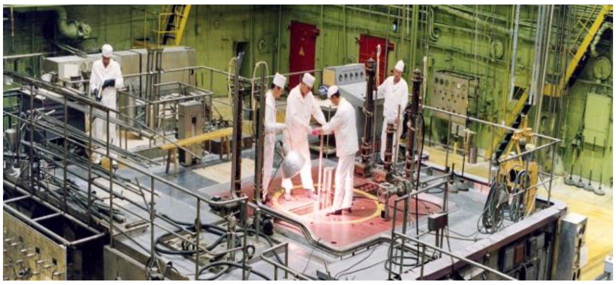


FIG. 1. Views of the reactor building (left) and central hall (right) of SM-3 reactor.

The SM reactor was brought to power for the first time on 15 October 1961, and on 3 November 1962, a neutron flux as high as 2×10^{15} n·cm^{-2}·s^{-1} in the central channel was achieved for the first time in the world when the reactor reached a thermal power of 46 MW. Step by step, as a result of the upgrades and advances of the reactor core, its central part, all circuits and systems, the power was increased to 100 MW, and the neutron flux in the central channel reached 5×10^{15} n·cm^{-2}·s^{-1}.

Throughout the whole lifetime of the reactor, the core, its central part, all circuits and systems have been continuously upgraded and advanced, and these activities are continuing. The SM reactor is a vessel-type high-flux reactor with a trap with intermediate neutron spectrum and cooled by pressurized water. Before its upgrading in 1992, the reactor was known as SM-2, and after its reconstruction it was renamed to SM-3. Reactor SM-3 is a vessel-type high-flux pressurized water-cooled reactor with a hard neutron flux spectrum in the core. A central flux trap has an intermediate neutron spectrum with a high thermal neutron flux and space for 27 irradiation targets with Ø 12 mm.

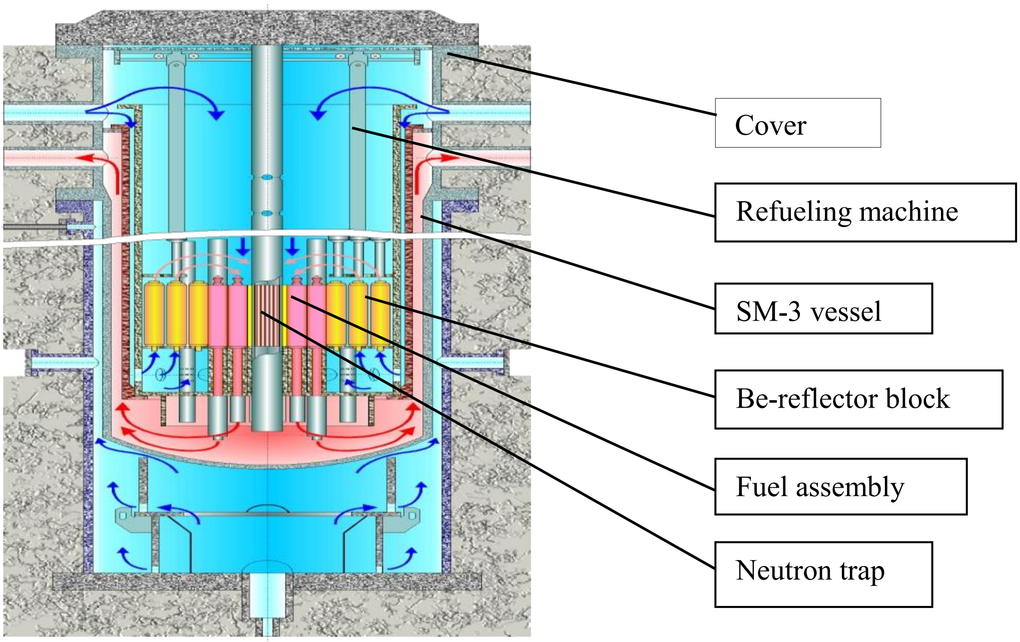
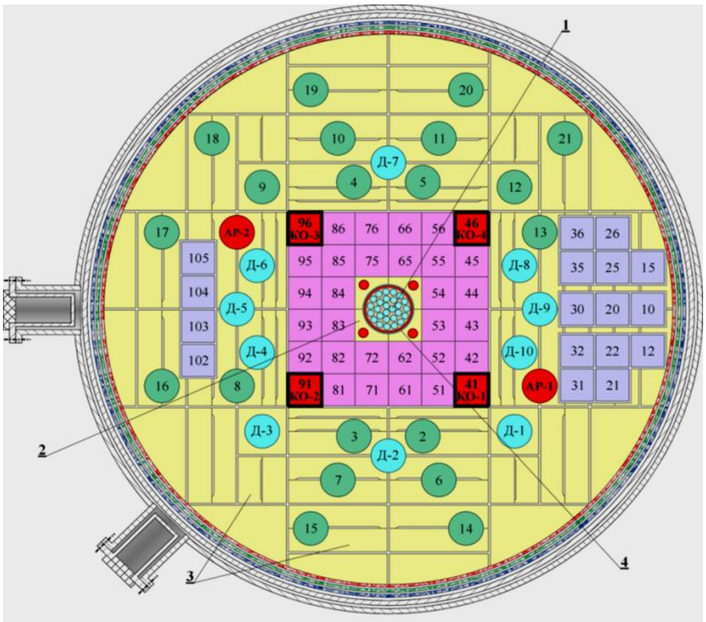


FIG. 2. The SM-3 longitudinal section.

TABLE 1. THE SM-3 MAIN TECHNICAL CHARACTERISTICS

Characteristics	Value
Power	100 MW
Maximal total neutron flux	5.4×10^{15} n·s^{-1}·cm^{-2}
Fuel	UO_2, 90% enrichment
Moderator	Water
Reflector	Beryllium
Coolant of primary circuit:	Water
– core inlet/outlet temperature	< 60 / < 95°C
– flow rate	2400 t/h
– core inlet pressure	4.9 MPa
Operating cycle	
– duration of cycle	10–14 d
Operating time at power	~ 250 d/year
Scheduled operation time	2017
Planned operation time	More than 2030



1 – neutron trap 2 – beryllium liners
3 – beryllium reflector blocks 4 – central compensating element

Cells for irradiation – up to 81;
Trap – up to 27 cells for targets Ø 12 mm;
Core – up to 6 FAs with 4 cells for targets Ø 12 mm, up to 4 FAs with 1 cell for targets Ø 24.5 mm;
Reflector – 30 cells for channels and devices Ø 68 mm, including 20 cells for instrumented devices.

FIG. 3. The SM-3 vessel cross section.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL CAPACITIES AND FACILITIES

2.1. MAIN DATA OF TESTING FACILITIES OF THE SM-3

TABLE 2. THE SM-3 IRRADIATION CELLS CHARACTERISTICS

Number of cells for irradiation	Up to 81	
Trap	Block option: up to 27 cells Ø 12–25 mm; Channel option: channel Ø 50 mm + 18 cells	
Core	Up to 6 and up to 4 FAs with 1 cell for targets Ø 24.5 mm	
Reflector	30 channels (of which 20 cells can be instrumented or supplied with separately coolant), Ø 64 mm	
Irradiation positions:	Neutron flux, $n \cdot cm^{-2} \cdot s^{-1}$	
	total	≥ 0.1 MeV
Trap	$\leq 5.4 \times 10^{15}$	$\leq 1.5 \times 10^{15}$
Core	$\leq 4.3 \times 10^{15}$	$\leq 2.3 \times 10^{15}$
Reflector	$\leq 1.6 \times 10^{15}$	$\leq 5.3 \times 10^{14}$

TABLE 3. THE SM-3 IRRADIATION CELLS CHARACTERISTICS

Design of irradiation rig	Medium	Testing parameters	
		φ, $n \cdot cm^{-2} \cdot s^{-1}$, (E >0.1MeV)	Kt, dpa/year
Loop channel in the reflector	Water (≤350°C, ≤18.5 MPa)	$1.0 \cdot \times 10^{13}$–4×10^{14}	0.1–6.0
Loop channel in the core	Water (≤350°C, ≤18.5 MPa)	1.2–1.5 × 10^{15}	15–18
Ampoule rig in the reflector	Boiling water (≤350°C, ≤17 MPa); heavy liquid metal (≤650°C, ≤1 MPa); supercritical water (≤650°C, ≤23 MPa); gas (He, Ne, N_2) (≤2500°C, ≤23 MPa)	5×10^{12}–5.3×10^{14}	0.1–6.0
Ampoule rig in the core	Boiling water (≤350°C, ≤ 17MPa); heavy liquid metal (≤650°C, ≤1 MPa); supercritical water (≤ 50°C, ≤23 MPa); gas (He, Ne, N_2) (≤2500°C, ≤23 MPa)	1.5×10^{15}–2.3×10^{15}	16–25

TABLE 4. THE SM-3 LOOP FACILITIES CHARACTERISTICS

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

2.2. GENERAL DESCRIPTION OF TESTING FACILITIES OF THE SM-3

In-pile testing devices and facilities:

— Long-term strength and creep tests of steels and alloys under longitudinal tension (facility 'Neutron-8') and internal gas pressure test at 550–800°C;
— In-pile tests of different types of fuel materials at 550–2500°C;
— In-pile investigation of relaxation resistance of structural materials;
— In-pile investigation of creep of nuclear fuel at temperatures 700–1100°C, including pre-irradiated fuel samples to investigate the burn up effect on the creep characteristics;
— In-pile tests of the core material for existing and advanced nuclear facilities at high damage rate of 1–25 dpa/year in the temperature range 100–2500°C and different environments.

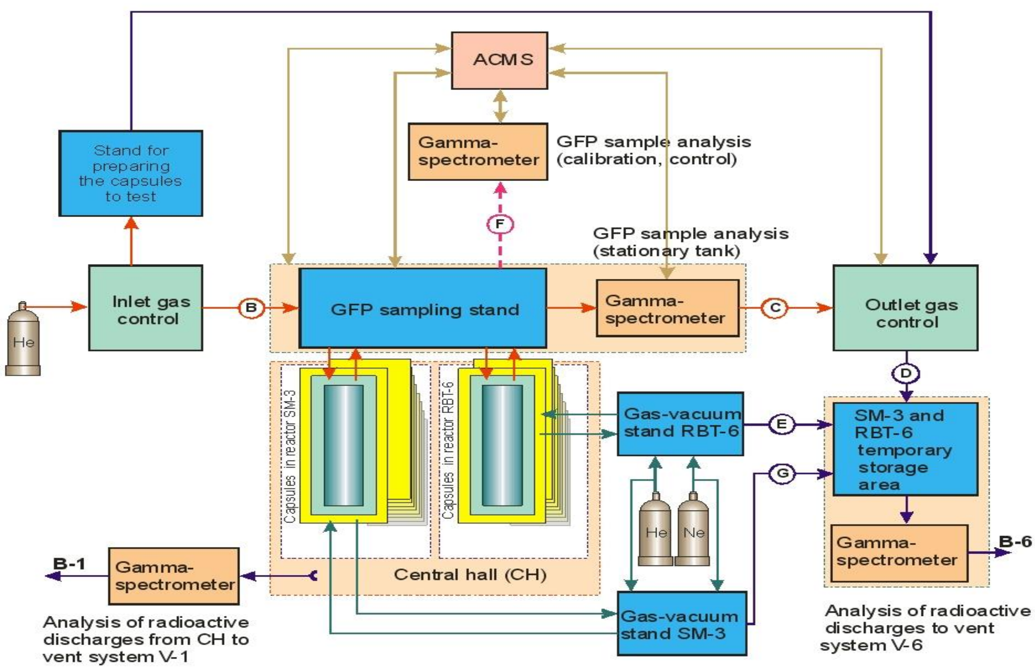


FIG. 4. Complex of test facilities.

Irradiation devices for testing different types of samples with up to 7–17 dpa/year in the near-core reflector cells and in the core cells:

— Tests in a high-temperature loop facility VP-3 in the near-core reflector cells at 250–300°C and 16 MPa with up to 10 dpa/year. The experimental volume available for the samples is Ø 40 mm and 350 mm high. The neutron flux (E $\geq$ 0.1 MeV) is (3–4) $\times 10^{14}$ n·cm^{-2}·s^{-1}. The temperature non-uniformity in the experimental volume is 10–20°C.
— Tests in the boiling water channels in the near-core reflector cell at water temperature 160–300°C and 10 MPa up to 10 dpa/year. Samples should be located in stainless steel containers to prevent corrosion. The experimental volume available for the samples is Ø 60 mm and 350 mm high. The neutron flux (E $\geq$ 0.1 MeV) is (3–4) $\times 10^{14}$ n·cm^{-2}·s^{-1}. The temperature non-uniformity in the experimental volume is 5–10°C.
— Tests in the core cells up to 15–17 dpa/year at $\geq$ 600°C. The experimental volume available for the samples is Ø 20 mm and 350 mm high. The neutron flux (E$\geq$ 0.1 MeV)

is (1.5–2.0) × 10^{14} n·cm^{-2}·s^{-1}. The temperature non-uniformity in the experimental volume is 100–150°C.

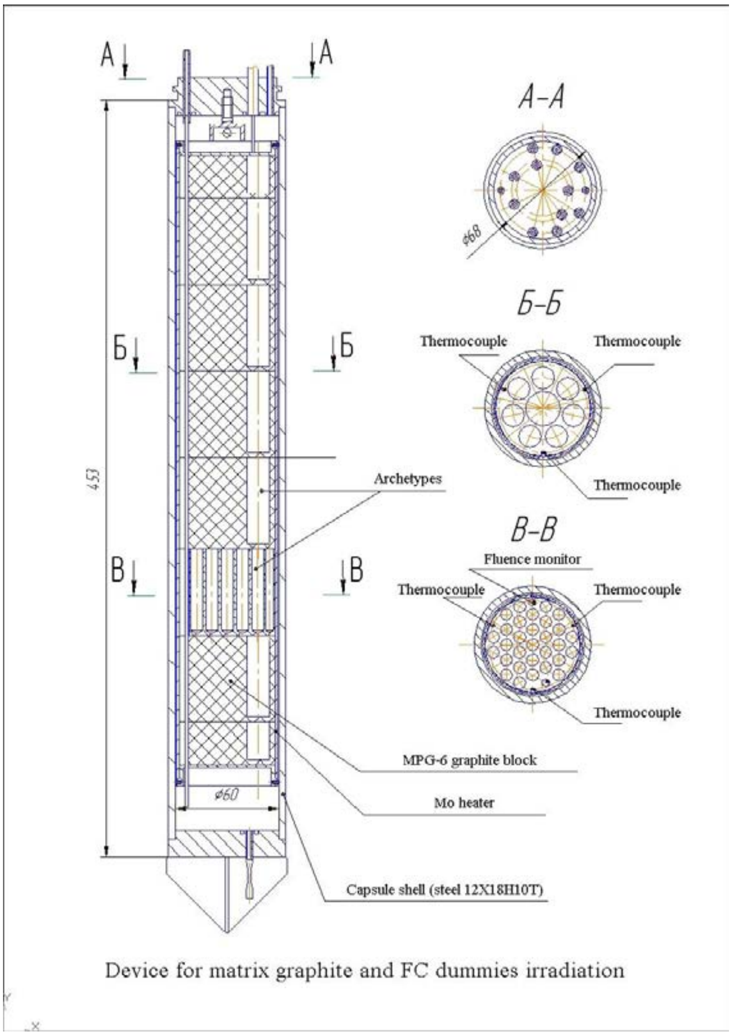


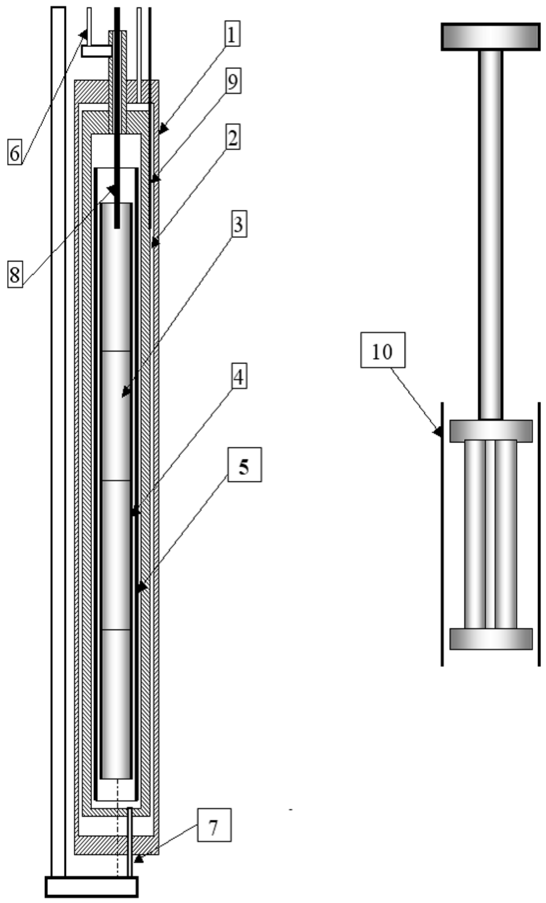
FIG. 5. Irradiation device for graphite testing.

The irradiation device (ID) for graphite testing is shown in the Fig. 5.
The ID core is 350 mm long and consists of seven graphite blocks. The height of the central blocks No. 2–6 is up to 53 mm, and the lateral ones No. 1 and 7 is up to 42 mm.

The blocks contain specimens of matrix graphite and fuel compact dummies (Ø 12.5 mm), specimens of block and reference graphite (Ø 6.0 mm) and graphite cans (Ø 19.0 mm) for special specimens in micro-cells and without can. Irradiation position: first row of the SM-3 reflector.

Neutron flux: φ ~4.0 × 10^{14} n·cm^{-2}·s^{-1} (E >0.1).

Absorbed gamma irradiation rate: steel — 11.5 W/g, molybdenum — 15 W/g, graphite — 9.8 W/g. Irradiation temperature 800–1250°C.



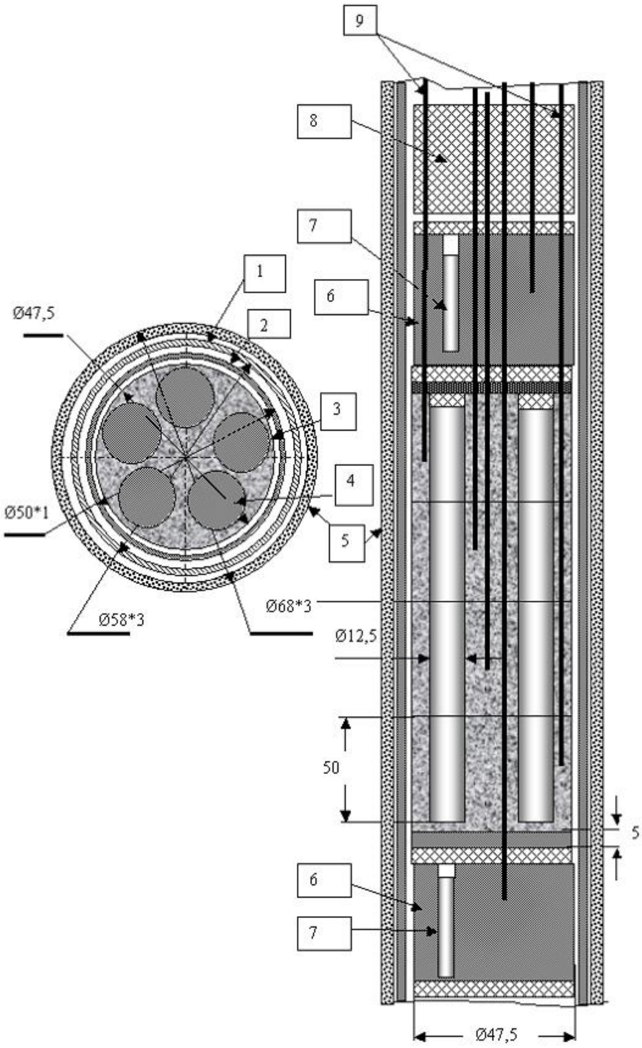
1 – external body	6 – tube
2 – internal body	7 – tube
3 – compact	8 – tungsten-rhenium thermocouple
4 – graphite bushing	9 – chromel-alumel thermocouple
5 – molybdenum screen	10 – hafnium (or boron steel) screen

FIG. 6. Irradiation device for fuel compacts of HTGR testing.

In the Fig. 6 the irradiation device for fuel compacts of HTGR testing is shown. The ampoule consists of external (1) and internal (2) bodies with a gap between them that can be vacuumed and filled with gas supplied from the GVTF. The internal body (2) contains four compacts (3) in graphite bushings (4) and a molybdenum screen (5). Tubes (6, 7) supply and discharge FGP gas-carrier (high-purity helium). The upper compact has a tungsten-rhenium thermocouple (8) 1.5 mm in diameter; the grooves on the external body surface have chromel-alumel thermocouples (9) 1 mm in diameter installed in them. Moreover, thermocouples can be installed in the grooves of the graphite bushings. All the capsules have activation neutron fluence monitors.

Three ampoules (1) can be surrounded by a hafnium (or boron steel) screen (10) and are fixed to the suspension. A screen 0.5–2.0 mm thick maintains the constant fuel burn-up rate. The heat power and thickness of screen are determined by preliminary neutronic analysis.

The specimen temperature is 1150–1400°C.



1 – external body
2 – internal body
3–6 – graphite blocks
7 – micro-capsule
8 – porous graphite block
9– thermocouples

FIG. 7. Irradiation device for HTGR fuel testing.

In the Fig. 7 the irradiation device for HTGR fuel testing is shown. The ID consists of external (1) and internal (2) bodies. The internal body contains 4 graphite blocks (3) (two of them — above the core, and two — below the core) containing fuel compacts in their holes with small gaps (about 100 μm). Two blocks (6) contain micro-capsules (7) with special specimens. In the upper part there is a porous graphite block (8) to absorb solid fission products. From their both ends, the column of blocks with FC and the blocks themselves (6) have radiation heaters and heat-insulating porous graphite screens. The temperature inside the ID can be measured with thermocouples (9) at several points on the section of each block. The temperature of the compacts is 1000–1400°C.

2.3. TECHNIQUES FOR IRRADIATION OF STRUCTURAL MATERIALS IN BOILING WATER FOR SM-3

When studying radiation resistance of materials under irradiation, care must be taken to obtain uniform steady temperature conditions in the operating volume of an experimental unit. A traditional widely used method of heat removal from the samples and levelling the temperature field in a capsule is based on a heat-transfer medium like liquid sodium. Due to high liquid metal thermal conductivity and intensive convection, it is possible to level the temperature field to a certain extent, in particular in reactor channels with low energy release (≤ 3 W/g). If this method is applied in the SM-3 high-flux channels, the non-uniformity of the sample temperature can reach 80°C and more at an average temperature of 300°C, which is totally unacceptable. Therefore, a new technique was suggested to provide a uniform steady temperature field based on boiling water capsules being actually thermosyphons.

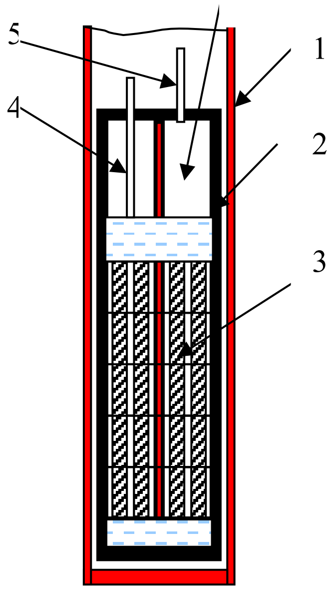
The rig (see Fig. 8) consists of a channel body (1) where a capsule (2) is placed containing samples (3), e.g., vessel steels (size 10 mm × 10 mm × 50 mm) with a definite gap filled with gas (helium, nitrogen) and a tube (5) filled with water to a certain level. The capsule is connected to the stand with a tube (4) in the reactor hall; certain helium pressure above water can be achieved using the capsule. During the reactor operation, boiling on the sample surface occurs (at saturated pressure) with different intensity producing a high temperature field uniformity of ± 5–10°C in the whole experimental volume. A high intensity of the heat removal from the samples during boiling allows an increase of capsule effective loading by several times (compared to the capsules of another type) providing quite low temperatures 180–320°C.

An additional axis heat transfer to the central condensation zone (Fig.8) improves the rig capabilities in terms of heat transfer and effective loading by approximately 50%, which is important for the high-flux reactor channels. The temperature can be controlled during irradiation by changing pressure and gas composition in the gap between the capsule and channel body as well as by changing helium pressure above the water in the capsule. Numerous tests of similar rigs have been carried out in the SM-3 channels at RIAR. The temperature field non-uniformity in the rigs with high energy release due to intensive boiling did not exceed ± 1°C. The samples can be placed in a thin-wall container, e.g., stainless steel, if there is no contact with water. The sample-water temperature difference is low (5–10°C) since the container thin-wall cladding is tightly pressed by external water pressure.

A rig for zirconium alloy testing (see Fig. 9) clearly shows the opportunities of the suggested technique, where the samples can have different configuration and mass at the same temperature. This is important in terms of testing loaded and unloaded samples or studying the neutron flux intensity influence on corrosion and other properties.

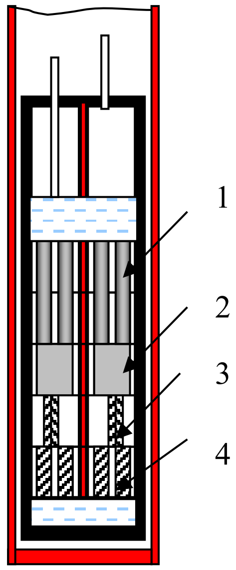
To examine iodine cracking, a technique was developed that allows determining the sample destruction point directly during the irradiation. One of the key factors is high-precision maintenance of the same temperature of all samples. Sealed containers (1) with the samples (2) filled with helium are placed in the rig (see Fig. 10). The sample is previously filled with high-pressure argon (to obtain a stressed sample) with addition of iodine. At the destruction point, argon is mixed with helium, which deteriorates the thermal conductivity of the gap between the sample and the container. A dramatic temperature rise is traced by a thermal block (3) with thermocouples (4). The temperature rise depends on the thermal block material and should exceed the temperature variations caused by other factors.

All the above-listed rigs have been used during the tests under limited temperature conditions. To extend the technique capabilities to higher temperatures, a rig has been developed (see Fig. 11). The rig has no channel body; the heat from the samples (1) is removed using a thermosyphon (2) filled with water through a radiator (3) to reactor water. The capsule cavity with the samples is filled with helium. The samples are separated from the thermosyphon by a gas gap (4) allowing temperature exceeding water temperature in the thermosyphon. The temperature of the samples is determined by the energy release in a channel, gap size (4), factor of the heat removed from the samples directly through the capsule outer wall as well as by gas pressure in the thermosyphon (controlled by the stand). The tests were done at the sample temperature of 450°C; however, the calculations show that the temperature can be raised up to 550–600°C.



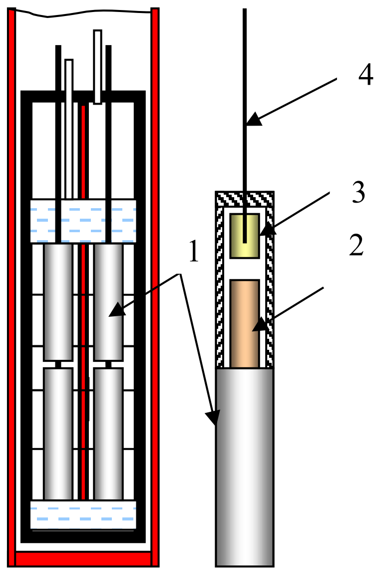
1 – channel body
2 – capsule
3 – sample
4 – water level tube
5 – gas tube

FIG. 8. Capsule to irradiate vessel steel samples.



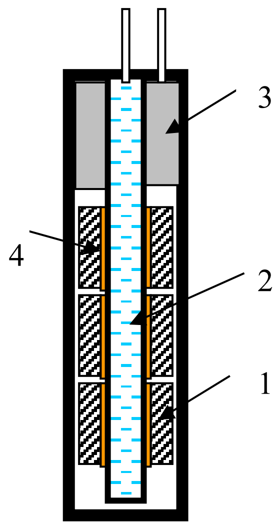
1 – pressurized tubes
2 – corrosion samples
3 – contact pair
4 – impact testing samples

FIG. 9. Capsule to irradiate zirconium alloy samples.



1 – container
2 – sample
3 – thermal block
4 – thermocouples

FIG. 10. Capsule to irradiate pressurized samples.



1 – samples
2 – thermosyphon
3 – radiator
4 – gas gap

Fig. 11. Capsule to irradiate samples at high temperatures.

3. RELATED ENGINEERING INFRASTRUCTURE

SM-3 is one of the installations of the largest nuclear centre Research Institute of Atomic Reactors (RIAR) which incorporate:

— Research reactor complex consisting of reactors that have laboratories to prepare and perform experimental research;
— The largest material testing complex in Europe comprising hot cells and glove boxes for non-destructive and destructive post-irradiation examinations of core components and wide range of irradiated materials;
— Facilities and technologies for investigation and pilot production of nuclear fuel;
— Radiochemical complex to study and produce transplutonics, radioisotopes and sources for industrial and medical purposes;
— Facilities to dispose radwaste and nuclear materials.

RIAR hosts a number of hot cells and boxes to perform post-irradiation non-destructive and destructive examinations of core elements and a wide range of irradiated materials. The material testing complex is accommodated in three adjacent buildings housing more than 110 equipment items (devices and facilities) to carry out tests and examinations of irradiated materials and fuel.

There are seven special large hot cells (see Fig. 12) for investigation of full-size SFAs from NPP, more than 40 hot cells for destructive examinations equipped with up-to-date machines for mechanical testing, examination and analysis of structural and fuel materials.

The non-destructive analysis of full-scale fuel rods and assemblies is possible (measurements of geometric characteristics; visual examinations; gamma scanning; eddy-current and ultrasonic).

The destructive analyses comprise burn-up determination, fission products release, metallography and ceramography, micro-hardness, density and porosity, thermal conductivity and electric resistance, X-ray analysis, TEM, SEM, EPMA, AES, SIMS, mechanical testing (tensile, compression, bending, impact). Several units of experimental equipment are presented in Figs 13(a), 13(b), 13(c) and 13(d).

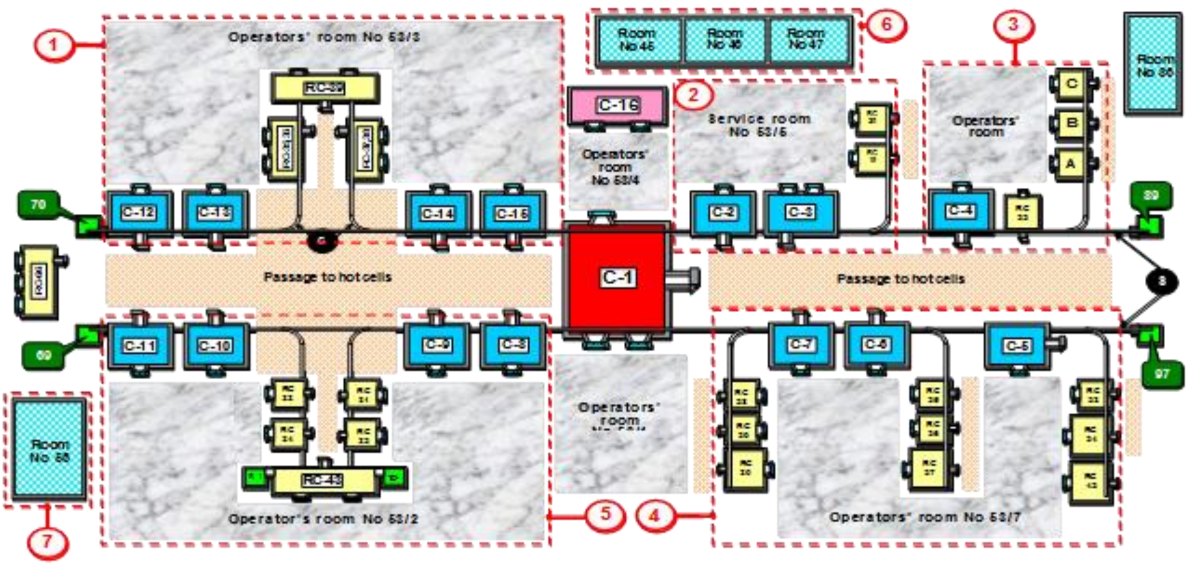


FIG. 12. Hot cells for destructive PIE.

FIG. 13(a). TEM JEM 2000 FX II.

FIG. 13(b). Zwick 5113 impact testing machine.

FIG. 13(c). Instron 1362-DOLI.

FIG. 13(d). SEM Phillips XL 30 ESEM.

4. RECENT AND CURRENT IRRADIATION PROGRAMMES AT SM-3

- Study of creep in E-110 and E-365 zirconium alloys under irradiation at different temperatures;
- Testing of WWER-type vessel steels;
- Study of radiation resistance of materials and brazed joints of the diverter first wall including thermal-cycle testing of ITER diverter target dummies;
- Thermal and thermal radiation tests of electric materials and dummies (electrical steel, micaceous laminate, organosilicate composition, etc.);
- Irradiation of TP-280 graphite samples at high temperatures (670°C) to obtain a constant dose in support of a longer operational life of the RBMK graphite stack;

- Tests of fuel elements with high uranium content are performed in the LF VP-1 channels under the SM reactor core upgrade programme in order to expand its experimental capabilities;
- A multi-stage process is established to accumulate ^{244}Cm, ^{248}Cm and ^{252}Cf (the most expensive metal in the world) heavy isotopes in the neutron trap cells and in two reflector cells nearest to the core. In addition, a number of radionuclides of high specific activity are accumulated: ^{63}Ni, ^{113}Sn, ^{65}Zn, ^{113}Ba, ^{188}W, ^{55}Fe, -59, ^{32}P, ^{33}P, ^{124}Sb, ^{131}I, ^{169}Yb, ^{170}Tm, ^{51}Cr, ^{55}Ca, etc. In addition, the activation of blank sources based on ^{75}Se and ^{192}Ir is performed;
- Precise activation of the medical purpose blank sources based on ^{60}Co is performed in reflector cell 20 in a special instrumented facility ensuring continuous neutron flux monitoring during the irradiation. The annual production amounts to 1000 sources.

Reactor utilization factors are given in Table 2.

Profile 24

BOR-60

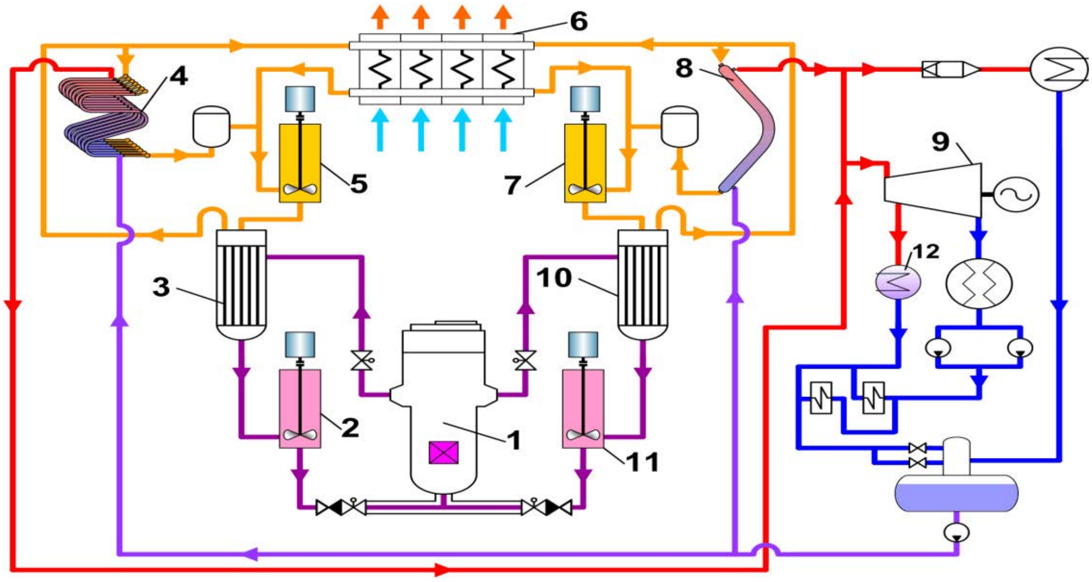
RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA

The BOR-60 is an experimental fast sodium reactor with a three-circuit two-loop scheme of reactor heat removal. The BOR-60 facility has been designed for the experimental justification of advanced fuel, structural and absorbing materials, and lifetime tests of equipment of sodium fast reactors.

FIG. 1. The view of BOR-60 main reactor building (left) and control room (right).

The primary and secondary circuits are cooled by sodium, the third circuit is cooled by water-steam. The water-steam circuit incorporates a turbo-generator and heating unit. The BOR-60 general circuit design and vessel sectional view are presented in Fig. 2.



1	–	reactor vessel	4, 8	–	steam generators
2, 5, 7, 11	–	primary and secondary pumps	6	–	air heat exchanger
3, 10	–	intermediate heat exchangers	9	–	turbine
			12	–	heating unit

FIG. 2. The BOR-60 facility general circuit design.

Basic technical characteristics are given in Table 1.

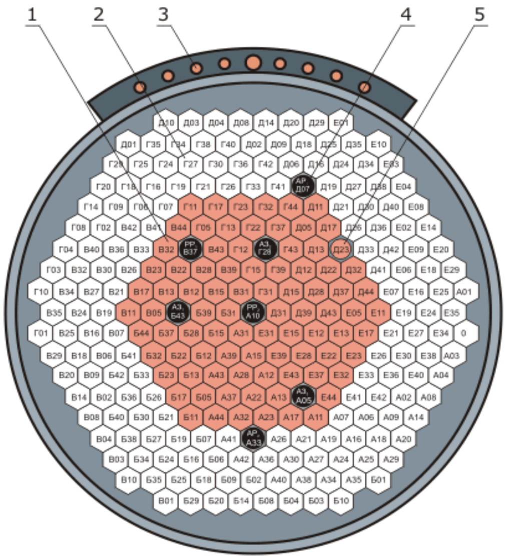
TABLE 1. TECHNICAL SPECIFICATIONS OF THE BOR-60 REACTOR

Characteristics	Value
Thermal power	≤ 60 MW
Primary and secondary circuit coolant	Sodium
Coolant maximal temperature: – at the reactor inlet – at the reactor outlet	 310–340°C 450–515°C
Electric capacity	12 MW
Thermal capacity for commercial heating	24 MW
Core height	450 mm
Maximal neutron flux	≤ 3.7×10^{15} $n{\cdot}cm^{-2}{\cdot}s^{-1}$
Maximal damage dose rate in the steel materials	≤ 25 dpa/year
Time of operation at power per year	220–230 d
Duration of fuel cycle	45–90 d
Number of shut-downs per year	4–5
Scheduled operation time	2015
Planned operation time	2020

2. EXISTING AND PROSPECTIVE EXPERIMENTAL CAPACITIES AND FACILITIES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL POSSIBILITIES

The reactor core (see Fig. 3) can simultaneously accommodate 12–25 irradiation devices with structural and fuel materials depending on the number of driver fuels. The number of experimental fuel assemblies (FA) with advanced fuel compositions in the core and irradiation devices with structural materials in the lateral reflector zone is not restricted. The reactor is also equipped with two horizontal and nine vertical channels located outside the reactor vessel. The vertical channels are 90–230 mm in diameter and are mainly used for irradiation of different types of materials.



1 - FA,
2 - reflector assemblies,
3 - vertical channels,
4 - control rod,
5 - instrumented cell (D23).

Core capacity	
Cells total: - driver fuel - control rods - instrumented device	265 85-156 7 1
Non-fuel experiments in the core	12-25
Experiments in the reflector zone	No limit

3

FIG. 3. The BOR-60 core arrangement.

A typical annual diagram of the BOR-60 operation is given in Fig. 4, and radial distribution of the integral neutron fluence, total flux and flux with E > 0.1 MeV in the core is presented in Fig. 5.

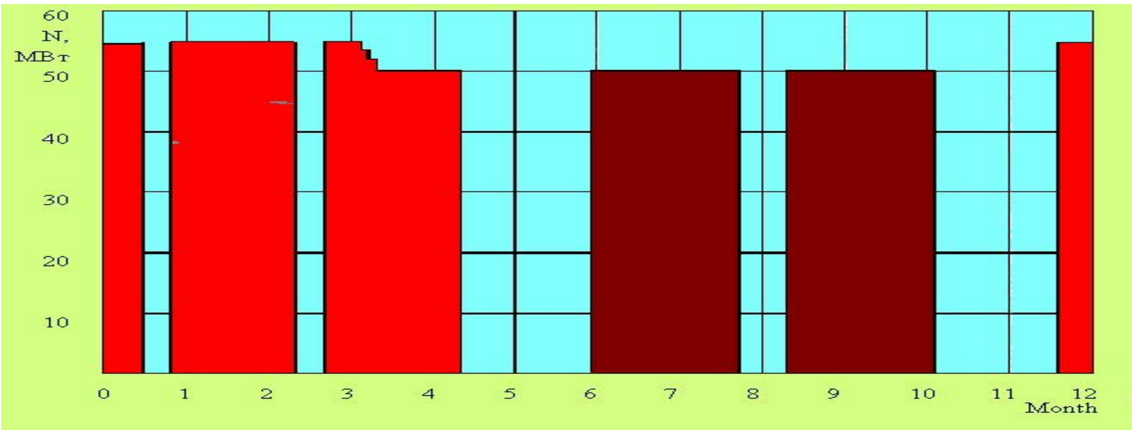
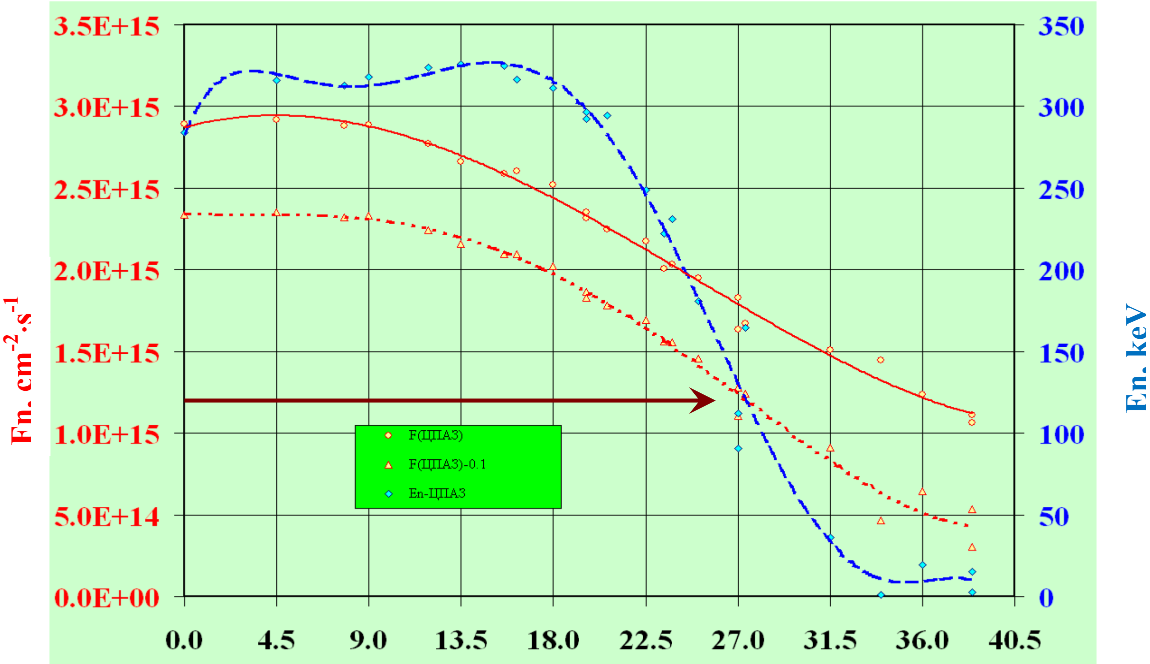


FIG. 4. Typical annual diagram of the BOR-60 reactor operation.



◇ — average neutron energy (En)
○ — integral flux (Fn)
△ — flux with E > 0.1 MeV Fn (0.1)

FIG. 5. Radial distribution of average neutron energy, total flux and flux with E > 0.1 MeV.

2.2. BRIEF DESCRIPTION OF IRRADIATION DEVICES

Irradiation devices of the BOR-60 reactor are designed for the experimental justification of the applicability of advanced fuel, structural and absorbing materials in nuclear power engineering. The damage dose in the steel materials in the irradiation devices installed in the core positions is about 10–25 dpa/year and for irradiation devices installed in the reflector positions about 5–9 dpa/year at maximal power level.

Types of irradiation devices:

— *Instrumented* — they allow obtaining data on irradiation parameters and properties of investigated materials directly in the ionizing radiation field;
— *Semi-instrumented* — they allow obtaining data on the irradiation temperature of investigated material samples using calculation methods, but based on the measured temperature of the coolant at the experimental facility outlet;
— *Non-instrumented* — they provide irradiation of investigated materials without in-process control of parameters.

From the design viewpoint, the irradiation devices can be:

— *Dismountable* — they can be re-used after extraction and replacement of investigated samples;
— *Non-dismountable* — extraction of samples results in a loss of their integrity and impossibility to re-use them.

Instrumented irradiation devices are designed for precision measurements in the radiation fields, and most often irradiation lasts one reactor fuel cycle for verification of design and characteristic of irradiation rigs.

Semi-instrumented and non-instrumented irradiation devices are designed for detection of post-radiation effects during the following out-of-reactor material science examinations. These irradiation devices can accommodate samples of several dimension-types or for several types of tests. The number of samples of one type can be sufficient for using mathematical statistics methods in processing the results of material science examinations (i.e. presentation of a result specifying its confidence probability as minimum). These irradiation devices can be equipped with activation monitors for post-irradiation determination of neutron fluence, as well as with temperature monitors for irradiation temperature measurements. Materials such as silicon carbide (SiC) and alloys of metals with different melting temperatures can be used as temperature monitors.

Dismountable irradiation devices are used for both interrupted and continuous irradiation. In the first case, the data are obtained at intermediate points of irradiation after disassembling of the experimental facility, unloading of some samples and their non-destructive (with further return to the experimental facility) or destructive examinations (i.e. without return for additional irradiation). A scientific result of such irradiation is a so-called dose dependence of a property under study. In case of continuous irradiation, no intermediate data are obtained. The result of such irradiation is a comparison of a property under study in the initial condition and one irradiated condition (mainly up to a lifetime fluence).

The materials under study can include:
- New advanced fuel compositions;
- Advanced absorbing materials for further use in control rods;
- Structural materials (steels, alloys, metals for fast and other reactors, including fusion);
- Non-metal materials: electro-insulating materials, ceramics, etc.

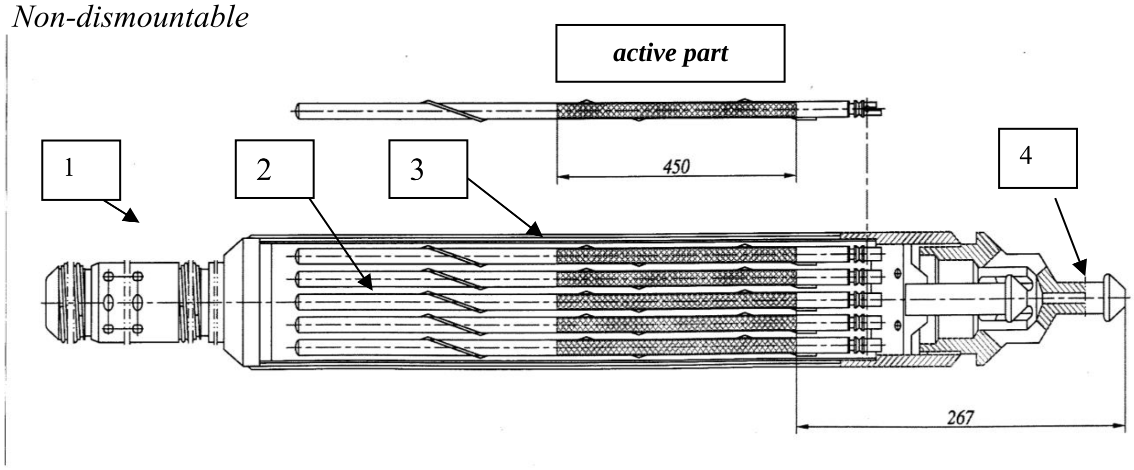
Non-instrumented irradiation devices can be installed in any core or lateral blanket cell. In-vessel loop facilities and instrumented devices are installed into the thermometric channel (cell D-23). The experimental devices where control of the coolant temperature at the outlet of the facility is necessary can be installed in cell D-35 of the lateral blanket and cell A-43 of the core.

The design of all irradiation devices is based on the following principles:
- Maintenance of integrity of the experimental device under normal operation and design-basis accidents;
- Prevention of possible violation of safety limits and/or safe operation conditions during installation to the reactor, operation and withdrawal from the reactor;
- Operation of the experimental facility should not cause heat rate deviations that can result in fuel rods damage;
- Prevention of possible unwanted displacement of the experimental device that can lead to a reactivity change;
- Installation of the experimental device into the reactor and its extraction from it should not cause unwanted displacement of other devices and components of the core.

Despite a variety of types of experimental devices installed in the core and lateral blanket for irradiation, their design has common features. In each case, samples of various materials under study, of various shapes and sizes, grouped in a specific way are placed in a hexagonal jacket (an experimental device case) with a flat-to-flat dimension of 44 mm and overall length of about 1575 mm (see Fig. 6). It has top and bottom adapters and a tail of a standard design. All experimental devices (except instrumented ones) are equipped with a standard transport head to be gripped by a bar of the reloading machine or manual tooling. Fastening of the head to the top adapter of the hexagonal jacket is demountable (bayonet fastening plus additional rodding or installation of a disposable stop ring, providing fixation of the fastening).

Depending on the required sodium flow rate through the experimental device, its tail can provide feeding of the device from a high pressure chamber through openings on the lateral surface, or from a low pressure chamber through a face opening. The flow required rate through the experimental device, which is based on thermal-physical calculation, is provided by a throttling washer selected with a pre-irradiation test of the device in the hydraulic stand.

Non-dismountable



1 – tail, 2 – fuel assembly bundle, 3 – jacket, 4 – head

FIG. 6. General view of irradiation device for fuel testing.

Schematic views of instrumented devices and in-vessel loop facilities are given in Figs 7 and 8. These devices have an overall length of ~ 6500 mm and flat-to-flat dimension of 44 mm in the core zone.

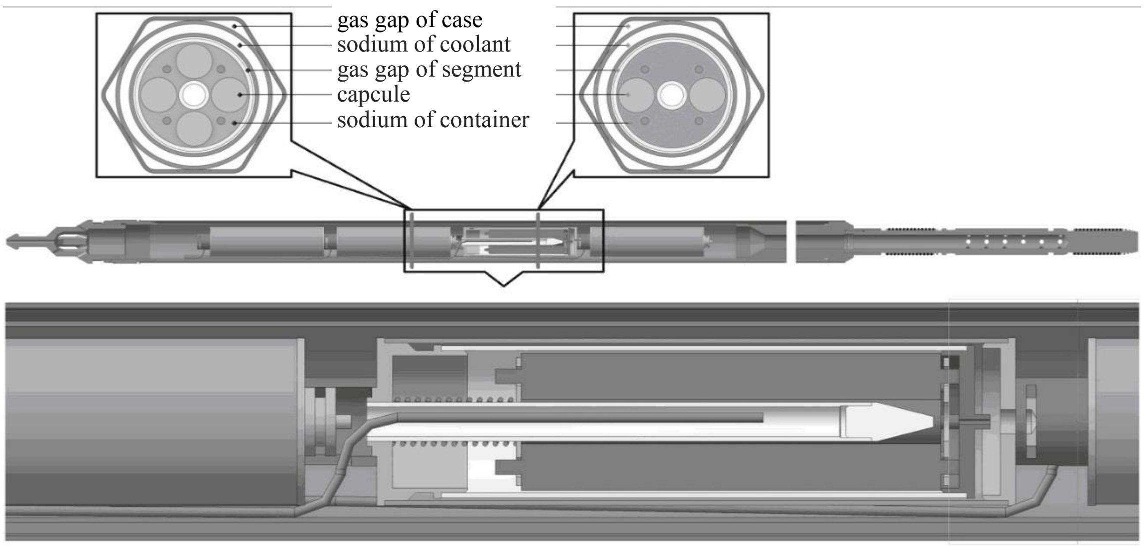


FIG. 7. General view of instrumented irradiation device.

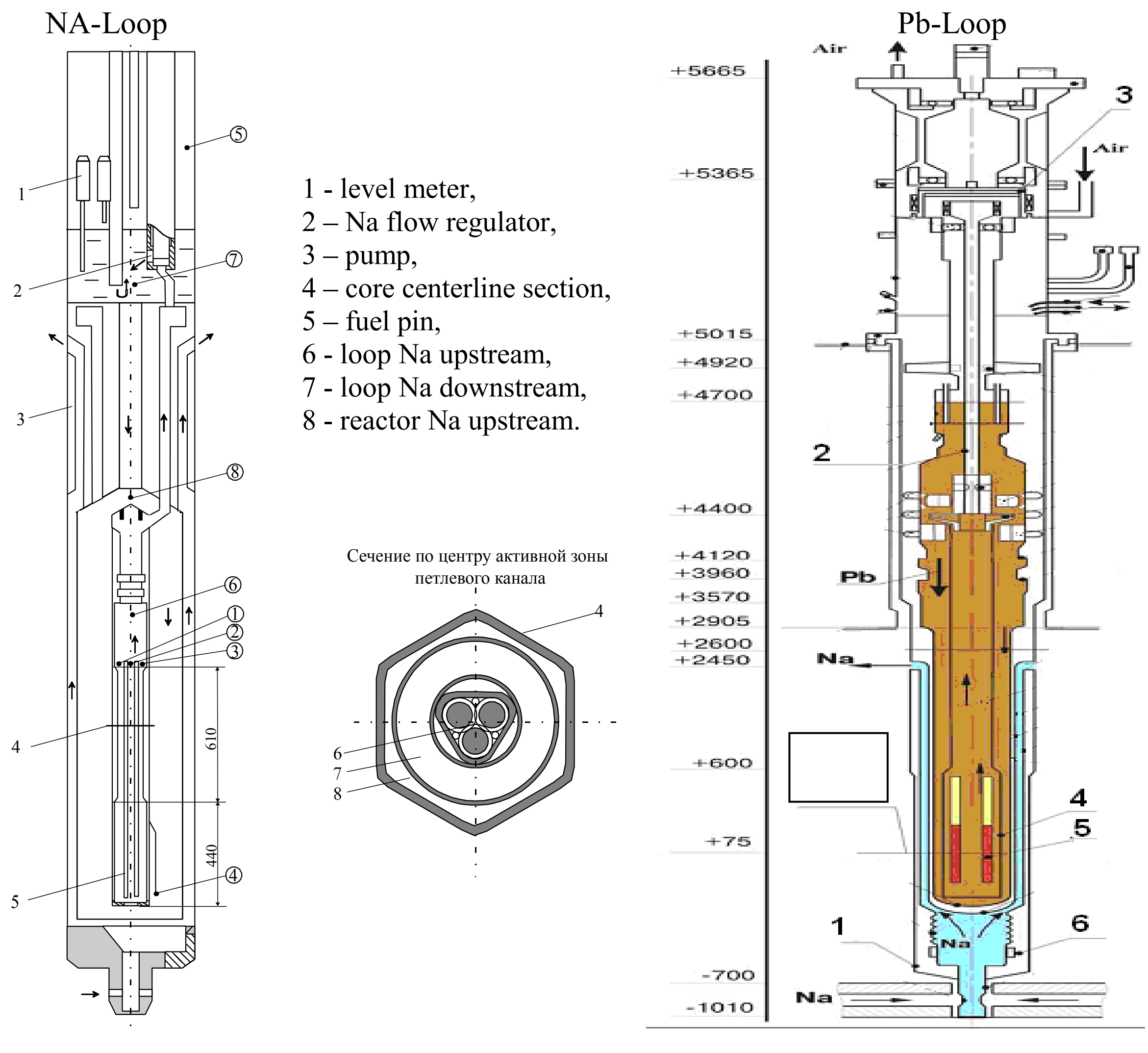


FIG. 8. General view of in-vessel loop facilities.

3. RELATED ENGINEERING INFRASTRUCTURE

BOR-60 is one of the installations of the largest nuclear centre Research Institute of Atomic Reactors (RIAR) which incorporate:

— Research reactor complex consisting of reactors that have laboratories to prepare and perform experimental research;
— The largest material testing complex in Europe comprising hot cells and glove boxes for non-destructive and destructive post-irradiation examinations of core components and wide a range of irradiated materials;
— Facilities and technologies for investigation and pilot production of nuclear fuel;
— Radiochemical complex to study and produce transplutonics, radioisotopes and sources for industrial and medical purposes;
— Facilities to dispose radwaste and nuclear materials.

RIAR hosts a number hot cells and boxes to perform post-irradiation non-destructive and destructive examinations of core elements and a wide range of irradiated materials. The Material Testing Complex is accommodated in three adjacent buildings housing more than

110 equipment items (devices and facilities) to carry out tests and examinations of irradiated materials and fuel.

There are seven special large hot cells for investigation of full-size SFAs from NPP, more than 40 hot cells for destructive examinations equipped with up-to-date machines for mechanical testing, examination and analysis of structural and fuel materials.

The non-destructive analysis of full-scale fuel rods and assemblies is possible (measurements of geometric characteristics; visual examinations; gamma scanning; eddy-current and ultrasonic).

The destructive analyses comprise burn-up determination, fission products release, metallography and ceramography, micro-hardness, density and porosity, thermal conductivity and electric resistance, X-ray analysis, TEM, SEM, EPMA, AES, SIMS, mechanical testing (tensile, compression, bending, impact).

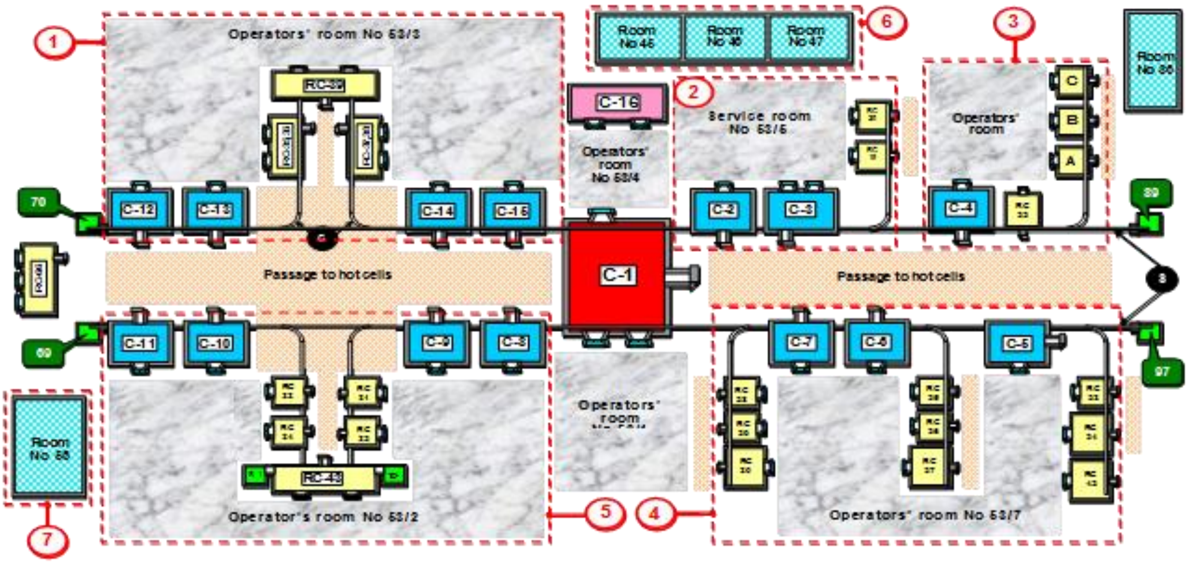


FIG. 9. Hot cells for destructive PIE.

FIG. 10 (a). TEM JEM 2000 FX II.

FIG. 10 (b). Zwick 5113 impact testing machine.

FIG. 10(c.) Test facility Instron 1362-DOLI

FIG. 10 (d). SEM Phillips XL30 ESEM

FIG. 10. PIE equipment.

4. RECENT AND CURRENT IRRADIATION EXPERIMENTS

— Experimental fuel rods of different types fast reactors;
— Steels used for fabrication of in-vessel components for PWR up to 100 dpa;
— Zirconium alloys for PWR cores;
— Vanadium-based alloys in lithium environment for fusion reactors;
— Graphite for the RBMK type reactors;
— Absorbing materials for LMFR;
— Experimental fuel rods with lead coolant for the BREST reactor have completed the first stage of the reactor tests.

TABLE 2.

Material		Type
Fuel	Ceramics	UO_2, UO_2-PuO_2, UC, UN, UN-Pu, UPuCN, $(PuO_2)MgO$, PuN-ZrN
	Metal	U, UPu, UPuZrNb
	Cermet	U-PuO_2, UO_2-U, UN-U
Absorbing	Samples	Ta, Hf, Dy, Sm, Gd, AlB_6, AlB_{12}, EuO_3
	CPS rods	CrB_2, B_4C, Eu_2O, Eu_2O_3+H_2Zr
Structural	Stainless steels	OX18H9, X18H10T, ЭП-450, ЭП-823 03X16H9M2, ЭП-912, ЭИ-847, ЭП-172, ЧС-68, BX-24
	High-nickel alloys	PE-16, X20H45M4Б, ВЦУ
	Fusion reactor materials	V, W, Mo, Nb
	Zirconium alloys	Э-110, Э-635, Э-125
	Graphite	ГРП-2-125, МП6-6, ГР-280, АРВ, IG-11, ПГИ

Profile 25

RBT-6

RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA

The RBT-6 is a thermal neutron pool-type reactor with high stability of irradiation conditions. The reactor was commissioned in 1974; it utilizes the SM reactor spent fuel assemblies as fuel.

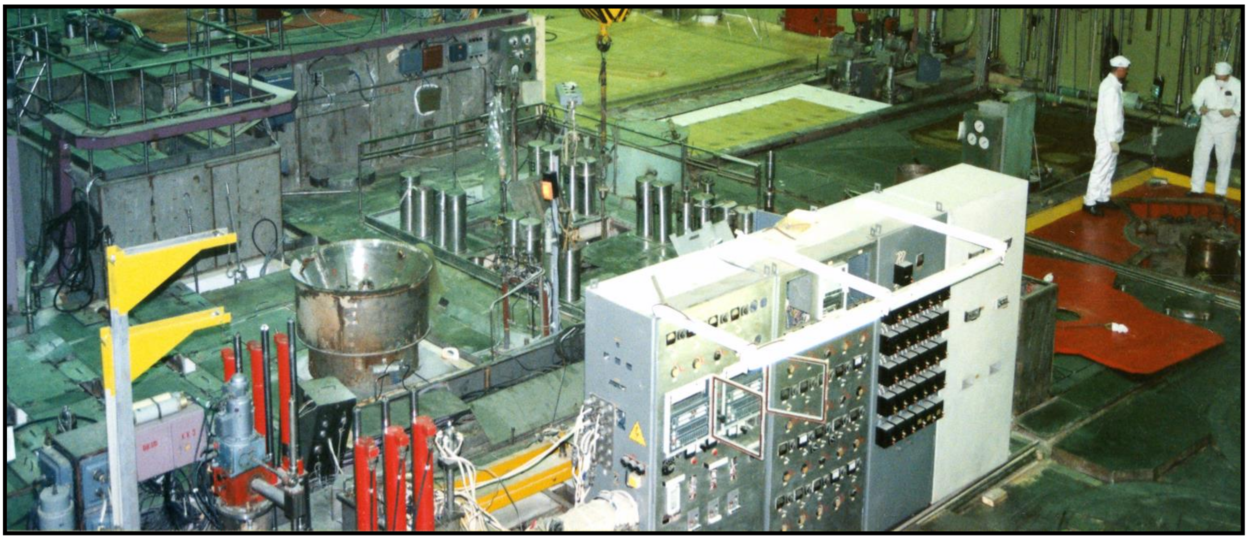


FIG. 1. The RBT-6 reactor central hall.

TABLE 1. BASIC CHARACTERISTICS AND OPERATION PARAMETERS OF RBT-6

Parameter, characteristic	Value
Nominal thermal capacity	6 MW
Fuel	UO_2 (spent FAs of the SM reactor)
Coolant, moderator and reflector	Water
Number in-core cells for experimental channels	8 pcs
Number reflector cells for experimental channels	8 pcs
Number of fuel assemblies	56 pcs
Coolant temperature at the core inlet/outlet	up to 60/75°C
Duration of fuel cycle	40–180 d
Core height	350 mm
Neutron flux: E <0.67 eV E >0.1 MeV	$3.2–22.0 \times 10^{13}$ n·cm^{-2}·s^{-1} $2.0–5.7 \times 10^{13}$ n·cm^{-2}·s^{-1}
Power operation days per year	~ 250 days
Life-time	More than 2020

The reactor is designed for doing experiments to study a change of properties of structural materials during irradiation at a neutron flux of 10^{13}–10^{14} n·cm^{-2}·s^{-1}, accumulation of radionuclide products, radiation doping of silicon. Tests are conducted in the reactor vertical channels located in the core and reflector (see Fig. 2). The total number of channels in the core to irradiate materials with a diameter up to 60 mm is eight. The reflector houses the KORPUS experimental facility for testing of WWER and PWR reactor vessel steels under conditions simulating within a wide range the vessel operation conditions in terms of flux and energy spectrum of neutrons, irradiation temperature, gradients of these parameters, and changes of these parameters during operation. IT also contains the experimental facility for radiation colouring of minerals.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES

The RBT-6 reactor design provides for accommodation of experimental capsule devices in eight core cells. The capsule channel represents a welded structure consisting of two tubes ∅ 75 mm × 3 mm, L = 708.5 mm and ∅ 60 mm × 3 mm, L = 845 mm. For investigation of radiation damage, different types of structural, fuel and absorbing materials can be irradiated in capsule channels.

2.1. TECHNIQUES FOR IN-PILE INVESTIGATION OF STRUCTURAL MATERIALS MECHANICAL PROPERTIES

There are upgraded and implemented techniques for long-term strength and creep of steels and alloys under longitudinal tension (facility ‘Neutron 8’) and internal gas pressure (facility UITO) at 550–800°C. Irradiation rigs were developed and fabricated as well as stands to control temperature and record damage. The effect of irradiation in RBT-6 on the long-term strength of austenitic steels X16H11M3 and X18H9 was studied at 550–600°C in justification of the BN-1200 design.

TABLE 2. PARAMETERS OF ‘NEUTRON’ FACILITY

Samples to be tested	Flat, tubular, cylindrical
Test conditions — load, strain rate and temperature	Static, stepwise
Neutron flux (E > 0.1 MeV)	1–8 × 10^{13} n·cm^{-2}·s^{-1}
Tensile strength	Up to 5000 N (1%)
Damage dose rate	Up to 2 10^{-4} dpa/h
Test temperature	250–900°C (1%)
Elongation measurement range	8 mm (3–5 μm)
Tensile rate range	1–10^{-5} %/h (5%)
Test duration	Up to 2 years
Test environment	Inert gas

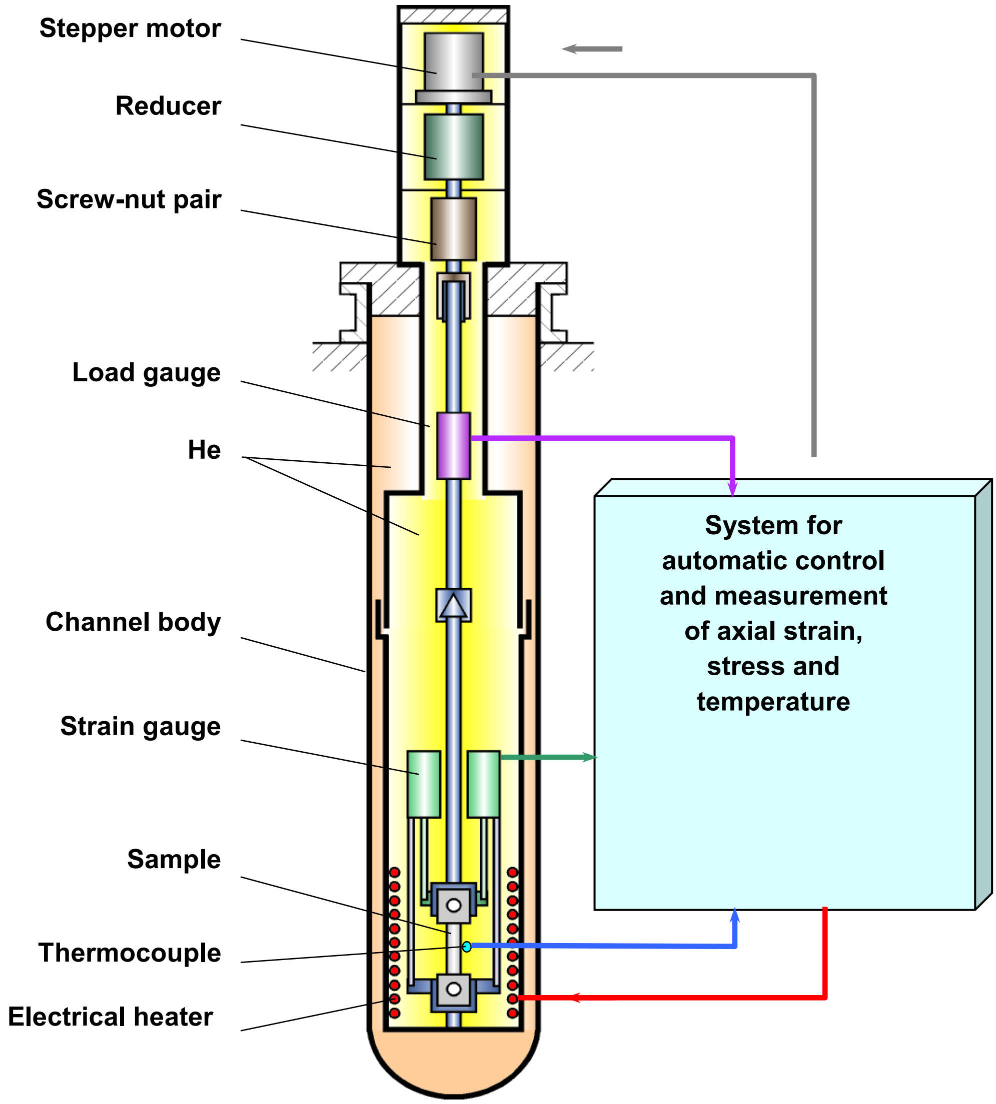


FIG. 2. 'Neutron' facility for tensile tests.

UITO is an in-pile test facility which implements a unique technique to obtain a stress relaxation curve for a bent sample. The stress is defined by the measured force with which a sample affects the supports. The technique is used to test stress and creep of steels and alloys.

The following irradiation rigs were developed and fabricated for in-pile tests of a large amount of flat samples with interim measurements of stress relaxation in the hot cell:

— Case for fix a bend under irradiation; sample is unbent before a periodical measurement of force P to define the stress relaxation;
— Non-dismountable case to test four samples; it has a movable element to periodically measure force P and calculate stress;
— Relaxometer to measure stress relaxation in the hot cell.

TABLE 3. PARAMETERS OF THE UITO FACILITY

Samples to be tested	Tubular, welded, connected to the facility
Number of samples	Up to 30
Neutron flux density (E > 0.1 MeV)	10^{12}–2×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$
Damage dose rate	Up to 3×10^{-3}dpa/h
Stress	- 200–400 MPa
Test temperature	250–600°C (1%)
Size measurement uncertainty	3–7 µm
Test duration	Up to 5 years
Test environment	Water, liquid Na, inter gas

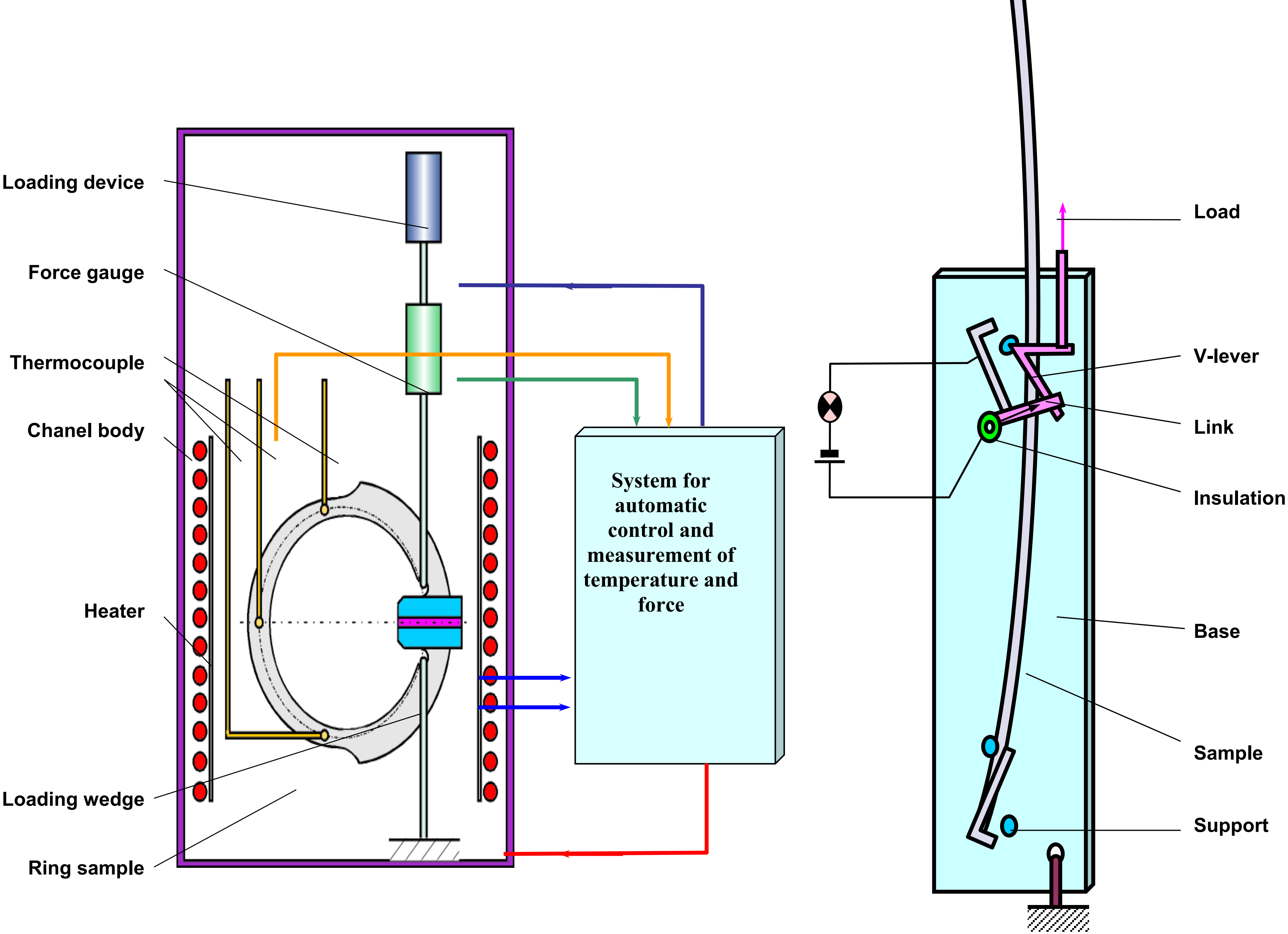


FIG. 3. Facility to test stress relaxation under bend in the ring or flat samples under irradiation.

(1) Technique for relaxation tests of the WWER FA springs under twisting:

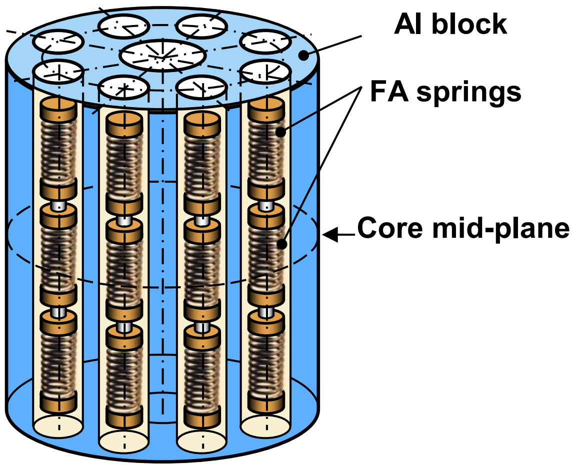


FIG. 4. Block of compressed springs in the IR.

(2) Technique to test a large number of samples under bending with periodical stress relaxation measurements:

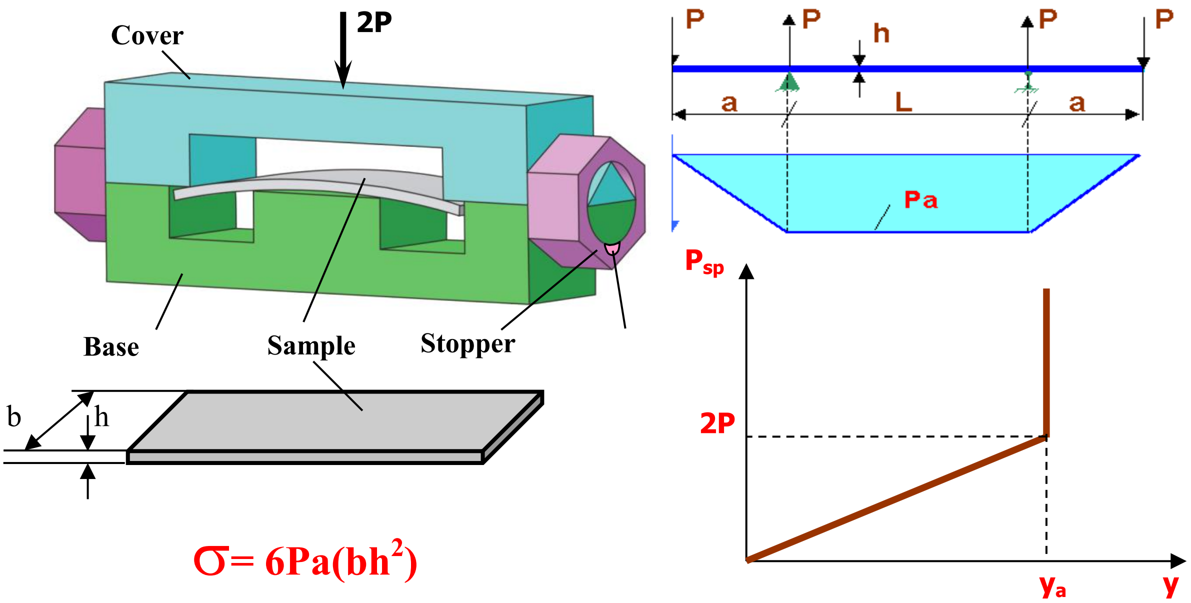


FIG. 5. A device to fix four-point bend y_a and partially unbend a sample before a periodical measurement of force P P to define the stress relaxation.

2.2. TECHNIQUES TO IRRADIATE STRUCTURAL MATERIALS IN RBT-6 BOILING WATER

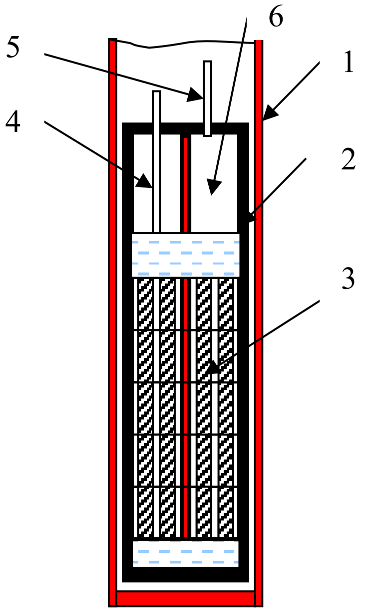
When studying radiation resistance of materials under irradiation in research reactors, there is an issue of maintaining stable temperatures in the irradiation rig. Traditionally, heat is removed from the samples and temperature is levelled in the capsule using a heat transfer medium, e.g. liquid sodium. Due to high thermal conductivity of liquid metal and intensive convection, the temperature can be levelled to some extent, especially in the channels with low heat rate ($q \leq 3$ W/g).

The irradiation rig (see Fig. 6 (a)) consists of a channel body (1), in which a capsule (2) with samples (3) of vessel steel 10 mm × 10 mm × 50 mm in size is inserted. Between the channel body and the capsule, there is a gap filled with gas (helium, nitrogen); there is also a tube (5) filled with water up to a certain level. The capsule is connected to a stand located in the reactor hall to create the required helium pressure above water. Under irradiation, boiling of different intensity on the surface of samples (saturated pressure) provides a high uniformity of the temperature field ± 5–10°C over the whole irradiation volume. High intensity of heat removal allows increasing the capsule effective loading by several times as compared to other types of capsules at quite low temperatures 180–320°C. Additional axial heat transfer to the condensing region increases by about 50% the capacity of the rig in terms of heat transfer and effective loading that is important for high-flux reactor channels. The temperature during irradiation can be controlled by changing pressure and gas composition in the gap between the capsule and channel body as well as by changing the helium pressure above water in the capsule. Multiple tests of such irradiation rigs were carried out at RIAR's RBT-6 reactor.

The capacities of the technique are well demonstrated by a rig to test zirconium alloys (see Fig. 10) where the samples can be of different configuration and mass, the temperature being the same.

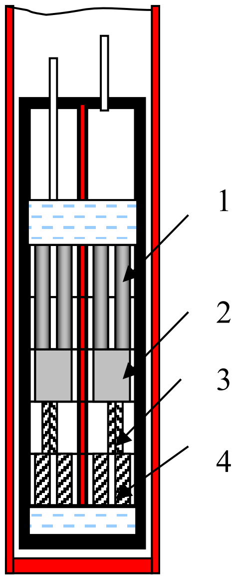
To test iodine cracking, a technique was developed to define a sample fracture moment just under irradiation. One of the key conditions here is to maintain the same temperature for all samples. The rig № 911 contained sealed containers (1) with samples (2) filled with helium. To stress the sample it is preliminary filled with high-pressure argon with iodine addition. At fracture, argon mixes with helium, thus worsening the thermal conductivity in the gap between the sample and container. A sharp temperature increase is recorded by a thermal block (3) equipped with a thermocouple. The value of a temperature jump is determined by the thermal block material and is to exceed the temperature deviations caused by other factors. The rig is easy-to-fabricate and does not require complicated equipment in the reactor hall.

All the above-said rigs were used for tests under a limited temperature mode (water temperature at the saturation line). To enlarge the technique application for higher temperatures, a device was developed shown in Fig. 10. This device does not have a channel body; heat from samples is removed by a thermal siphon (2) filled with water through a pipe to the reactor water. The capsule cavity is filled with helium. A gas gap (4) separates samples from the thermal siphon that allows having the samples temperature increasing the one in the thermal siphon. The samples temperature is determined by heat rate in the channel, gas size (4), portion of heat transferred from the sampled just through the capsule outside wall as well as by gas pressure in the thermal siphon (controlled by the stand). The rig was tested at a samples temperature of 450°C; however, the calculations show that the temperature can be elevated up to 550–600°C. Due to the thermal siphon, the temperature non-uniformity on the samples decreased significantly (till ± 15°C). All the dismountable rigs were tested in the reactor and demonstrated high reliability. They are applied now to test copper alloys for the ITER diverter.



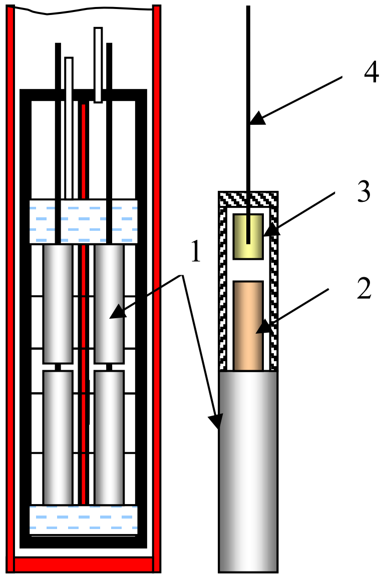
1 – channel body
2 – capsule
3 – sample
4 – water level tube
5 – gas tube

FIG. 6 (a). Capsule to irradiate vessel steel samples.



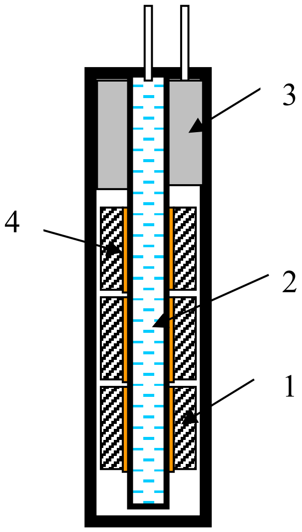
1 – pressurized tubes
2 – corrosion samples
3 – contact pairs
4 – impact toughness

FIG. 6 (b). Capsule to irradiate Zr alloy samples.



1 – container with He
2 – sample with argon
3 – thermal block
4 – thermocouple

FIG. 6 (c). Capsule to irradiate pressurized samples.

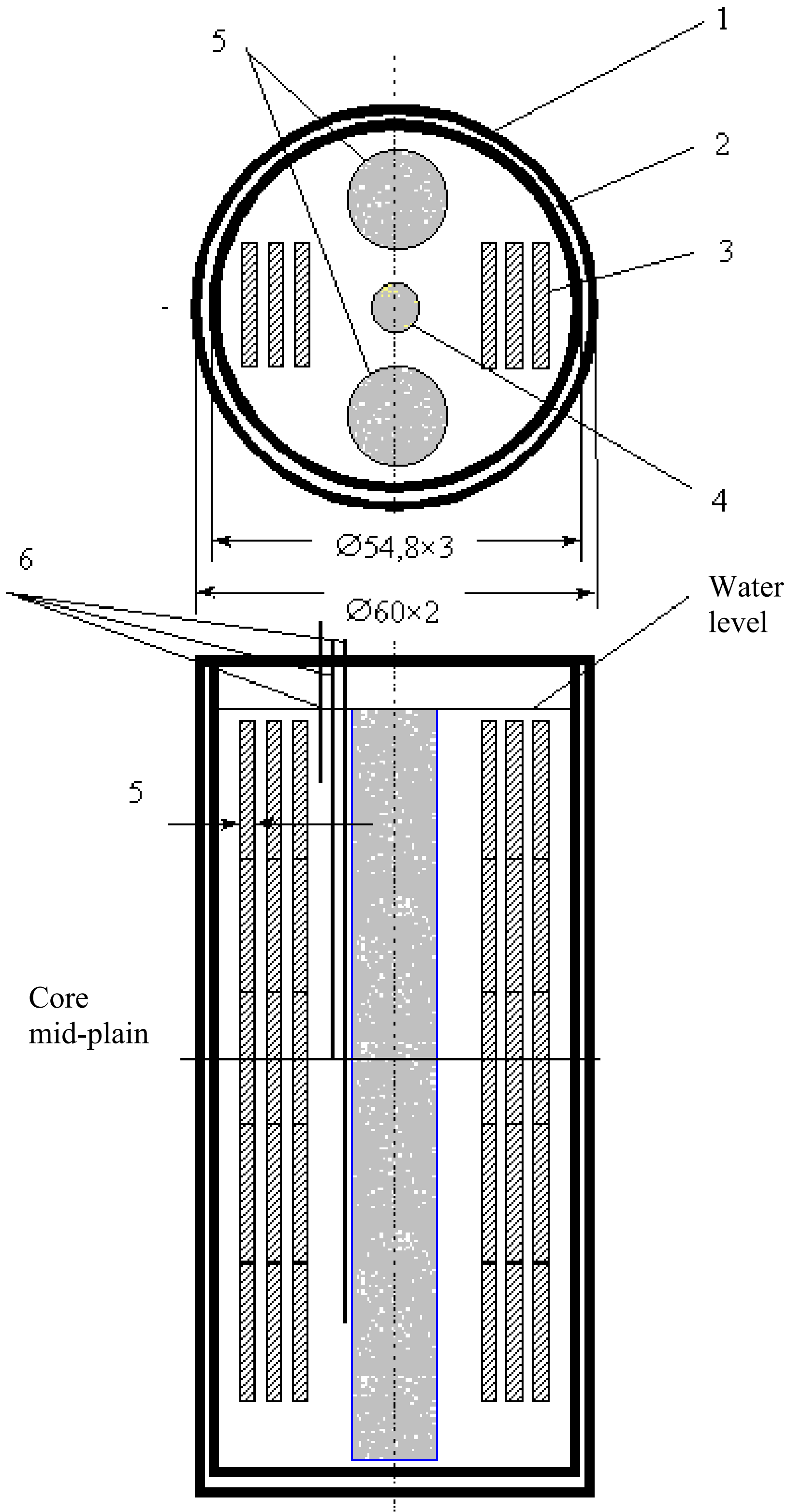


1 – sample
2 – thermal siphon
3 – pipe
4 – gas gap

FIG. 6 (d). Capsule for high-temperature irradiation.

FIG. 6. Technique to test structural material samples in a boiling water channel.

The technique allows testing structural materials in the RBT-6 core at temperatures from 120 to 300°C, the temperature non-uniformity in the experimental volume being ± 15°C.



1 – channel
2 – body
3 – containers with samples
4 – holder
5 – heaters
6 – thermocouple

FIG. 7. Capsule to irradiate copper alloys for ITER diverter.

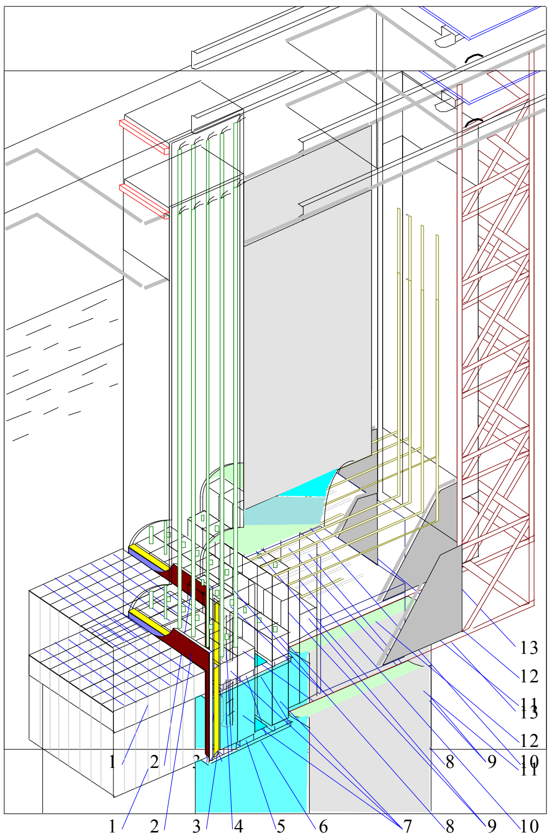
2.3. KORPUS EXPERIMENTAL IRRADIATION FACILITY

The KORPUS facility (see Fig. 8) is designed for vessel material samples irradiation with simulation of operation conditions of the WWER and PWR reactors vessel elements.

It incorporates two systems:
(1) System A located in tank 2 (pool 'A') of the RBT-6 reactor pool;
(2) System B located in tank 1 of the RBT-6 reactor pool.

System B incorporates a baffle, dismountable bed plate and a number of experimental capsules.

System A located in tank 2 of the reactor pool incorporates the following main units:
(1) Manually-driven mobile platform;
(2) Supporting frame;
(3) Type 2 capsules with a grid;
(4) Simulation block of the WWER reactor beyond-the-vessel space.



1 – core	5 – support plate for installation of capsules	9 – Type 2 capsules with samples
2 – baffle	6 – guides along which support plates with capsules move	10 – communication outlets from Type 2 capsules
3 – blanket	7 – type 1 capsules with samples	11 – concrete baffle with a cylindrical hole
4 – samples	8 – communication outlets from Type 1 capsules	12 – simulation block of the beyond-the-vessel space
		13 – manually-driven mobile platform

FIG. 8. KORPUS facility.

According to the experiment condition, blanket assemblies are located in the baffle on the mobile plate that is close to the reactor core. Other plates accommodate shielding capsules and type-1 capsules.

The KORPUS facility has an information and measurement system for automatic maintenance of a set temperature in capsules with accuracy of ± 5°C and for collection and preliminary processing of information from the sensors, as well as for generation of control inputs to switch actuators on and off. All the data from the information and measurement system are displayed on the KORPUS control board.

The RBT-6 gas stand is located in the central hall. It is used for filling capsules with gas, together with gas pressure measurements, during a capsule preparation prior to bringing the reactor to power. When a capsule is filled with gas, the gas stand is cut off from the corresponding communication line. Each communication line (stainless steel tube ∅ 6 mm ×1 mm) has a vacuum pressure gage OBMV-150 with a measurement range of (-1–0–5)kgf/cm^2 that controls inner pressure in a corresponding capsule (except for type-2 capsules) (1kgf=9.80665 N). The KORPUS facility is supplied with electric power from the backup feeder. Voltage 220 V (50 Hz) is provided by the power cabinet and control boards to the capsule electric heaters. The power input is 25 kW.

3. CURRENT PROGRAMMES OF ACTIVITIES AT RBT-6

- Investigation of creep of Zr alloys E-110 and E-365 under irradiation at different temperatures;
- Tests of the WWER and PWR reactor vessel steels;
- Tests of structural materials for the ITER international fusion reactor;
- Thermal and thermal-radiation tests of electro-technical materials and mock-ups (electro-technical steel, micaceous laminate, organosilicate composition, etc.);
- Investigation of long-term strength of materials for BN-1200;
- Investigation of relaxation resistance of the FA-KVADRAT spring materials, Zr alloys;
- Experiments to justify the ^{99}Mo production technology by an activation method using Mo_2C nano-powder containing Mo with natural isotopic composition.

In the coming years, the main research areas of RBT-6 will include:

- Reactor tests of materials in capsule devices at different temperatures and environments and in-reactor investigation of creep and radiation growth;
- Neutron activation analysis and neutron radiography;
- Tests of structural materials for the ITER international fusion reactor;
- Relaxation tests of material samples of springs made of an alloy with a high content of nickel;
- Study into the influence of irradiation on long-term strength of steels;
- Tests of structural materials in the KORPUS facility;
- Expansion of a range of radionuclide products and increase of their output (^{131}Cs, ^{131}Ba, ^{14}C, ^{60}Co, etc.);
- Commercial production of ^{99}Mo;
- Development of radiation technologies on transmutation and change of physical and chemical properties of materials for industrial purposes.

4. RELATED ENGINEERING INFRASTRUCTURE

RBT-6 is one of the installations of the largest nuclear centre Research Institute of Atomic Reactors (RIAR) which incorporate:

— Research reactor complex consisting of reactors that have laboratories to prepare and perform experimental research;
— The largest in Europe material testing complex comprising hot cells and glove boxes for non-destructive and destructive post-irradiation examinations of core components and wide range of irradiated materials;
— Facilities and technologies for investigation and pilot production of nuclear fuel;
— Radiochemical complex to study and produce transplutonics, radioisotopes and sources for industrial and medical purposes;
— Facilities to dispose radwaste and nuclear materials.

RIAR hosts a number hot cells and boxes to perform post-irradiation non-destructive and destructive examinations of core elements and a wide range of irradiated materials. The Material Testing Complex is accommodated in three adjacent buildings housing more than 110 equipment items (devices and facilities) to carry out tests and examinations of irradiated materials and fuel.

There are seven special large hot cells for investigation of full-size SFAs from NPP, more than 40 hot cells for destructive examinations equipped with up-to-date machines for mechanical testing, examination and analysis of structural and fuel materials.

The non-destructive analysis of full-scale fuel rods and assemblies is possible (measurements of geometric characteristics; visual examinations; gamma scanning; eddy-current and ultrasonic).

The destructive analyses comprise burn-up determination, fission products release, metallography and ceramography, micro-hardness, density and porosity, thermal conductivity and electric resistance, X-ray analysis, TEM, SEM, EPMA, AES, SIMS, mechanical testing (tensile, compression, bending, impact).

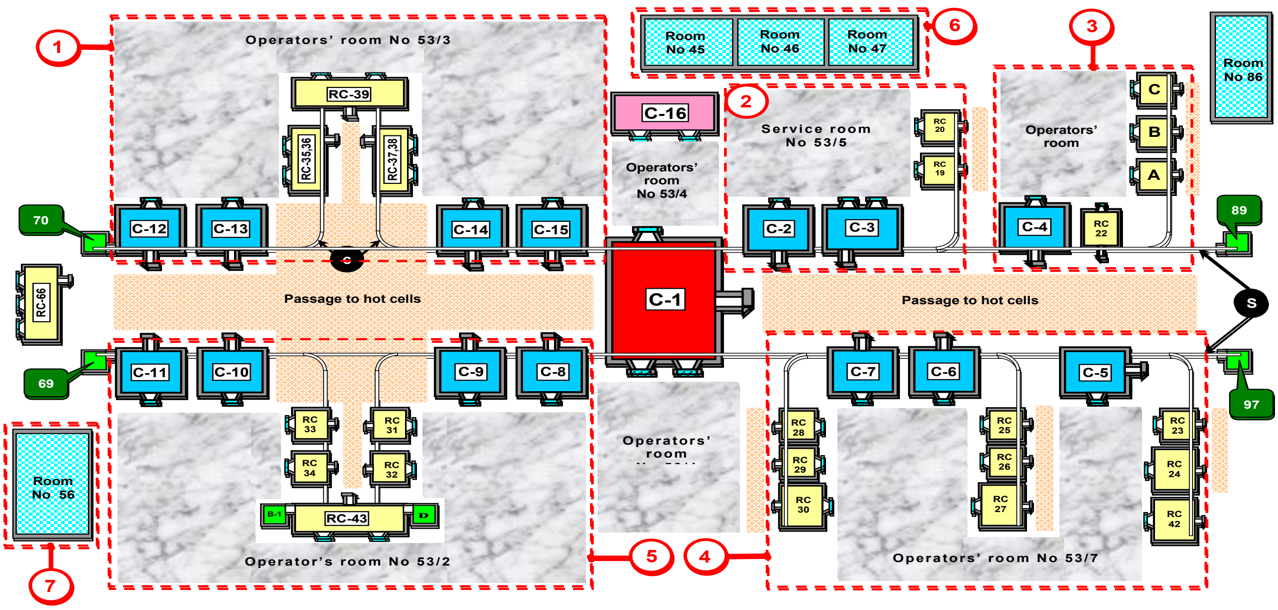


FIG. 9. Hot cells for destructive PIE.

FIG. 10 (a). TEM JEM 2000 FX II.

FIG. 10 (b). Zwick 5113 impact testing machine.

FIG. 10 (c). Instron 1362-DOLI.

FIG. 10 (d). SEM Phillips XL 30 ESEM.

FIG. 10. PIE equipment.

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TRIGA MARK II

SLOVENIA

1. GENERAL INFORMATION

TRIGA Mark II reactor is part of Jožef Stefan Institute. Reactor is placed 10 km away from centre of Ljubljana in Podgorica and is the only Slovenian research reactor.

TRIGA is a pool type reactor. Reactor tank is 6 m high and has a diameter of 2 m. Reactor core is 5 m under water level.

Its maximum thermal power in steady state mode is 250 kW. Reactor is also capable of pulse operation. Maximum power of pulse is approximately 1 MW.

Reactor reached its first criticality in year 1966 and is today almost 50 years old. In 1991 reactor was almost completely reconstructed. Since then the core is loaded only by fuel elements that contain 20% enriched uranium. Uranium is mixed by ZrH (standard TRIGA fuel) which makes reactor inherently safe. To control the power, four control rods are used. As an absorber is used boron carbide (B4C). Three rods have fuel followers, transient rod (used for pulse operation) has air follower.

Reactor operates on average 800 working hours per year and produces on average 125 MW(th) of thermal energy. Currently there are three licensed operators and one operator, trainee. The total number of staff is five.

Nowadays, reactor is mainly used for irradiating different samples (neutron activation analysis and study of radiation hardness), neutron radiography and training of future nuclear power plant (NPP) staff, Slovenian students and IAEA courses participants.

Link to the IAEA research reactor database is cited in [1].

2. EXPERIMENTAL FACILITIES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

Reactor has four beam ports, two tangential and two radial. One tangential penetrates graphite reflector, the second does not penetrate even reactor tank. Beside second tangential channel, all others are placed 10 cm lower than the centre of the core (see Fig. 1).

2.1.1. Radial beam port

The first radial beam port

penetrates only water and ends at the beginning of the reflector (53 cm from the centre of the core). The tube has a diameter of 15 cm. At the end of the tube thermal neutron flux equals 2 $\times 10^{11}$ n·cm^{-2}·s^{-1}. Thermal flux is 10 times higher than epithermal and 20 times higher than fast neutron flux. Neutron flux is inversely proportional to the distance from the centre of the core.

2.1.2. Radial piercing beam port

The second radial beam port penetrates also graphite reflector and ends at the edge of the core (23 cm from the centre of the core). Tube has a diameter of 15 cm. Where the beam port is the closest to the core, thermal neutron flux equals to 2×10^{12} $n \cdot cm^{-2} \cdot s^{-1}$. Epithermal and fast neutron fluxes are approximately two times lower. Once again, neutron flux is inversely proportional to the distance from the centre of the core.

2.1.3. Tangential channel

Tangential channel pierces graphite reflector. At the centre, the distance between edge of the reactor core and edge of the beam tube is only 2.5 cm. Maximum thermal flux is in the middle of the channel and equals 10^{12} $n \cdot cm^{-2} \cdot s^{-1}$. Epithermal flux is there two times lower and fast neutron flux is three times lower. Further we go off centre, lower is the neutron flux.

2.1.4. Tangential channel 2

Second tangential channel does not penetrate the reactor tank. Inner diameter is 15 cm and is positioned 50 cm higher than first tangential channel. Because of the low neutron flux is not well characterized and used for irradiation.

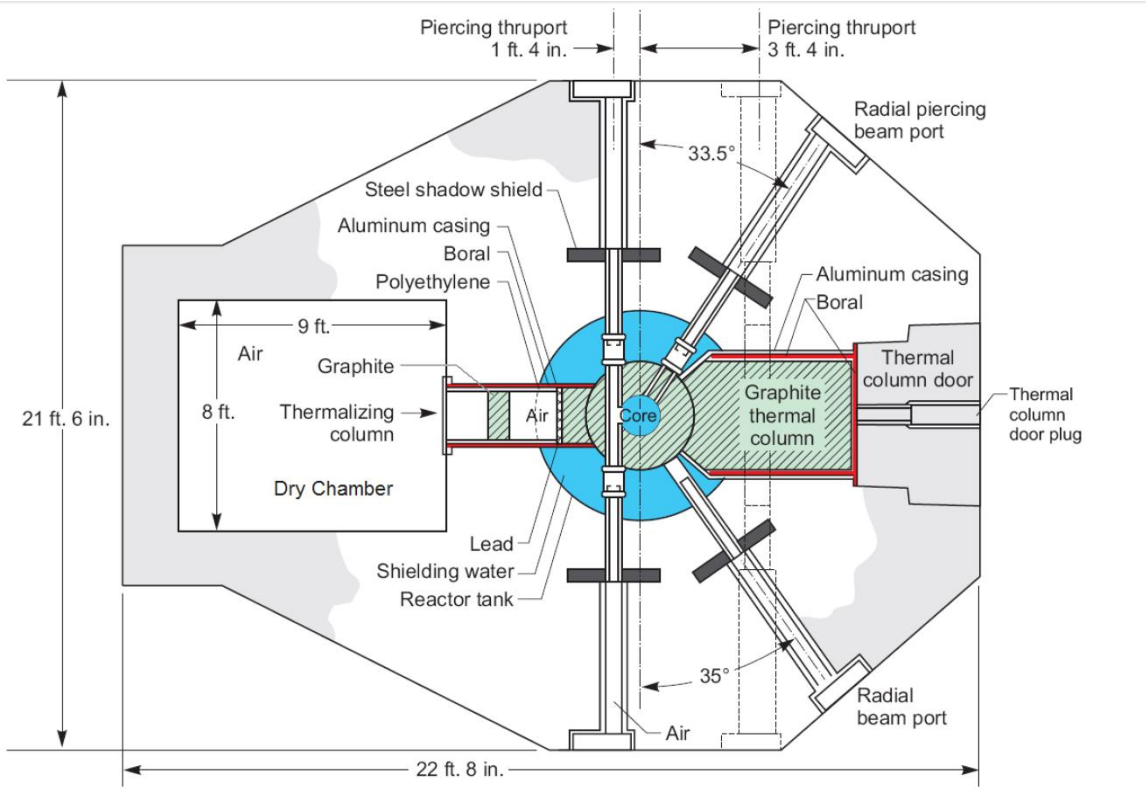


FIG. 1. Top view of the TRIGA reactor.

2.1.5. Thermal column

Beside four beam ports there are two graphite blocks which lead to irradiation facilities. Inside thermal column, samples can be irradiated with a high proportion of thermal neutrons. Samples can be irradiated in a square shape tube that is 10 cm wide and 10 cm high. At the end, the distance from the centre of the core is 92 cm. At that point, thermal flux equals 10^{11} $n \cdot cm^{-2} \cdot s^{-1}$. Epithermal and fast fluxes are approximately two orders of magnitude lower.

2.1.6. Thermalizing column

On the other side of the thermal column, is smaller graphite block, called thermalizing column. It ends with a dry chamber. Dry chamber is used to irradiate large samples. Beam of neutrons is 60 cm wide and 60 cm high. Inside dry cell is optionally placed fission plate made out of 20% enriched uranium. It is used to harden the neutron spectre. Thermal neutron flux equals to 3×10^{10} n·cm^{-2}·s^{-1}. Epithermal flux is six times lower and fast flux is two orders of magnitude lower. A detailed description of external irradiation facilities can be found in [2].

2.1.7. Central channel

In last years, most of the samples were irradiated in vertical aluminium irradiation channels (see Fig. 2). One is located in centre of the reactor core. Samples are inserted into aluminium ampoule with inner diameter of 2.5 cm. Also polyethylene ampoules are used for shorter irradiations (~min). Thermal, epithermal and fast neutron fluxes equal to 5.5×10^{12} n·cm^{-2}·s^{-1}, 6.5×10^{12} n·cm^{-2}·s^{-1} and 7.6×10^{12} n·cm^{-2}·s^{-1} respectively.

2.1.8. Triangular channel

Triangular irradiation channel is located 14 cm away from the centre of the core and is surrounded by the fuel elements. Samples with the diameter of 5 cm can be irradiated inside. They are inserted and withdraw using fishing line. Thermal, epithermal and fast neutron fluxes equal 4.4×10^{12} n·cm^{-2}·s^{-1}, $3.3 \cdot \times 10^{12}$ n·cm^{-2}·s^{-1} and $3.6 \cdot \times 10^{12}$ n·cm^{-2}·s^{-1} respectively.

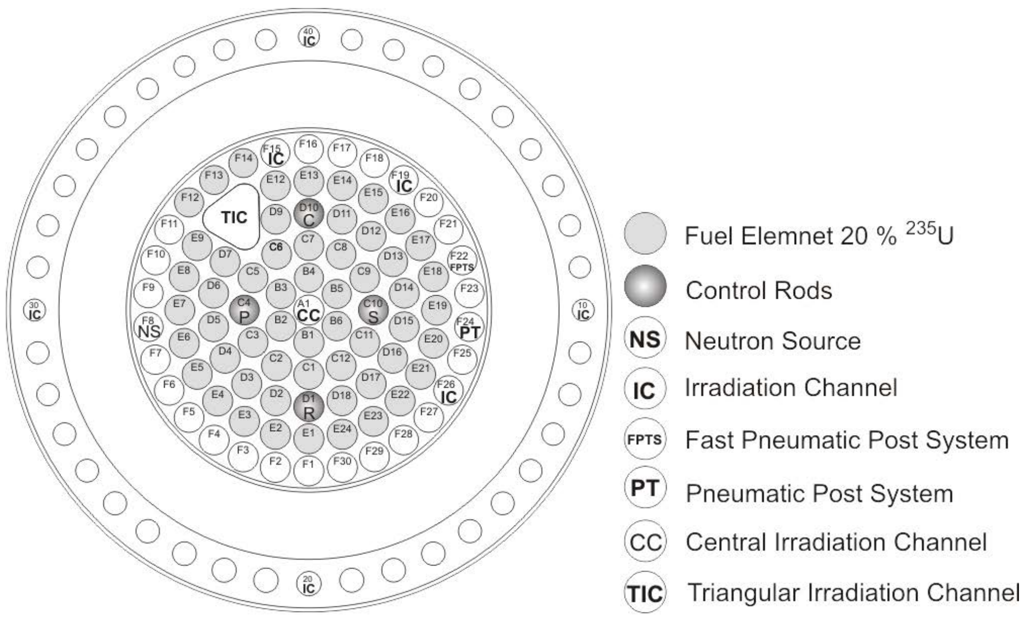


FIG. 2. Irradiation channels inside the reactor core and standard core configuration.

2.1.9. Irradiation channels in outer ring

Inside outer ring of the core are five irradiation channels. Three of them are not equipped with pneumatic post system. They are placed in positions F15, F19 and F26 (see Fig. 2). Samples are irradiated in aluminium ampoules that are 10 cm high and 2.5 cm in diameter. Ampoules are inserted and withdrawn using fishing line and special robot that automatically insert and lift the sample after desired time of irradiation. Thermal, epithermal and fast neutron fluxes equal to 3.5×10^{12} n·cm^{-2}·s^{-1}, 1.7×10^{12} n·cm^{-2}·s^{-1} and 1.7×10^{12} n·cm^{-2}·s^{-1} respectively.

Channels in positions F22 and F24 (see Fig. 2) are equipped with pneumatic transfer system. Samples are inserted in polyethylene ampoules with inner diameter of 1.9 cm and height of 5 cm. They are used only for shorter irradiations (< 3 h at 250 kW). Samples are inserted and withdrawn using pneumatic system. After the irradiation, samples are sent into reactor basement where a gamma spectrometer is located. Samples can be additionally sent to the chemical laboratories in the neighbour building. Thermal, epithermal and fast neutron fluxes equal to 4.0×10^{12} n·cm^{-2}·s^{-1}, 1.4×10^{12} n·cm^{-2}·s^{-1} and 1.3×10^{12} n·cm^{-2}·s^{-1} respectively.

2.1.10. Rotary groove

Around the reactor core there is a graphite reflector. There are 40 irradiation positions and samples are inserted inside aluminium ampoules with inner diameter of 2.5 cm and height of 10 cm. Pneumatic system is used to withdraw the samples and sent them directly inside hot cell facility. Thermal, epithermal and fast neutron fluxes equal 1.5×10^{12} n·cm^{-2}·s^{-1}, 4.3×10^{11} n·cm^{-2}·s^{-1} and 2.6×10^{11} n·cm^{-2}·s^{-1}respectively. A detailed description of the in-core irradiation channels can be found in [3].

2.1.11. DT converter

In near future, it is planned to install a device called DT converter. This is device used to irradiate samples with 14 MeV neutrons. Inside converter is a material that consist deuterium and lithium. When lithium atom absorbs neutron, alpha particle and atom of tritium are formed. If tritium atom is fused by deuterium, 14 MeV neutron is released along with the alpha particle. The whole device will be installed inside thermal column.

2.2. LOOPS FOR TESTING COMPONENTS

Not available.

2.3. INVESTIGATION OF ACCIDENTAL CONDITIONS

Not available.

2.4. CORROSION OF REACTOR MATERIALS

Not available.

2.5. CAPSULE/AMPULE TESTING

Not a routine. Testing can be done by additional preparations.

2.6. FUEL AND STRUCTURAL MATERIALS BEHAVIOUR

Not applicable.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. EXPERIMENTAL MATERIAL LOGISTIC

Experiment dependent.

3.2. PIE FACILITIES

3.2.1. Hot cells

Hot cell facility is placed in a neighbour building. There are two hot cells. The first one is connected with reactor by pneumatic transfer system through which samples that were irradiated inside rotary groove are sent inside hot cell. If the samples are irradiated elsewhere, they can be manually redirected into hot cell. The first hot cell is equipped by master-slave manipulators (see Fig. 3). Both cells have large concrete door through which material can be inserted (e.g. irradiated fuel element inside protective container) or removed (usually the irradiated samples). Second hot cell is used to store radioactive material and to handle radioactive sources.

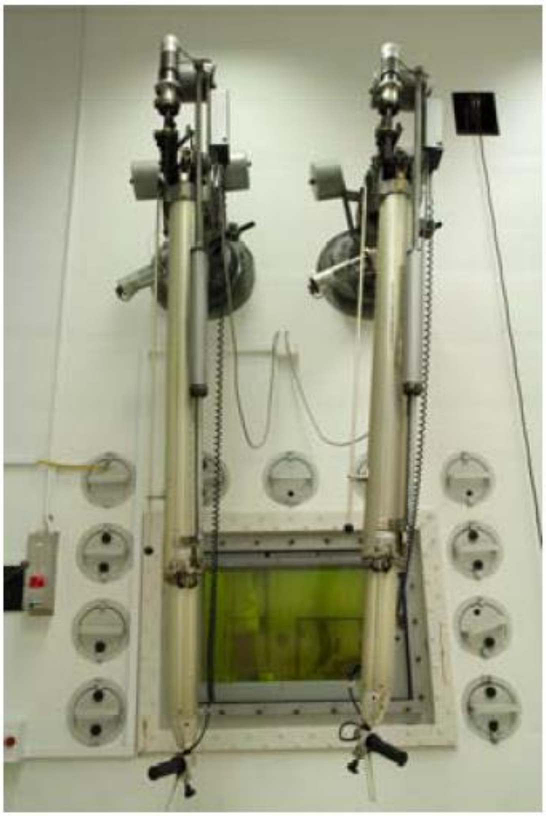


FIG. 3. Master-slave manipulators in hot cell.

3.3. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES

In the past years, we have designed experiments for the students, future nuclear power plant staff and course participants which are performed several times a year.

3.3.1. Void coefficient measurements

With this experiment it is clearly shown what happens when voids appear inside working reactor. Usually this happens due to water boiling. Water in our TRIGA cannot be boiled; therefore a special system was developed.

Aluminium tubes are inserted into reactor core through measuring positions which are located between fuel elements. Up to 20 tubes can be used. System is controlled by laptop (Ethernet connection) where user can open or close each valve individually and set the total air flow through the tubes. During the exercise, reactor is operating at low power (~400 W) and reactivity is observed with digital reactivity meter. Therefore, for each position of aluminium tube, void coefficient can be measured and compared to others.

3.3.2. Flux mapping

Flux mapping is crucial for safe operation; we can avoid overheating the fuel and consequently, we can avoid damaging the fuel elements, xenon oscillations, uneven cooling etc.

At our reactor, we have advanced fission chambers that are 10 mm long and have a diameter of 4 mm. They are inserted in special tubes which can be placed inside measuring positions that are located between fuel elements. A pneumatic system is used to move the fission chamber up and down. The pneumatic system is controlled by laptop where user can set the position of the chamber. User can read the signal of fission chamber and determine flux distribution. During the experiment, reactor operates at low power (~500 W). Uranium-235 and ^{238}U fission chamber can be used, so thermal and fast flux distribution can be measured.

3.3.3. Activation of the primary water

Inside water cooled reactors, water is irradiated by neutrons when it passes the reactor core. This causes higher doses around the primary coolant.

Aluminium tube is used to guide the water from the reactor pool on the platform. Water intake is beneath the core, tube goes than through the core and till the detector. As a detector, HPGe (high purity germanium) or lanthanum bromide (LaBr3) is used. Reactor is operated at different powers (from 1 kW to 250 kW). Gamma spectrum can then be observes using laptop. The main observed nuclides are ^{16}N and ^{19}O. Beside gamma spectrometer also a dose rate metes is used, where it is clearly seen how the activation depends on the reactor power.

3.3.4. Criticality experiment

This is one of the basic experiments in reactor physics. It is done every time after the major core change.

Trainees are provided by the calibrations tables of all control rods. Their task is to make a reactor critical. In steps, control rods are slowly withdrawn. For each step, inserted reactivity is calculates and signal on source range is measured. Reactor is operated by the reactor operator. He follows the instructions given by trainees unless their move would cause unsafe operation. Once the reactor is close to critical, it hard to determine whether it really is. To be totally sure, neutron source is taken out of the core and signal is observed. If the signal is declining, reactor is still subcritical; if the signal is rising, reactor is already supercritical.

3.3.5. Calibration of the control rods

The connection between control rod positions and inserted reactivity is not linear. That is why calibration curve of the control rod is very helpful during the reactor operation.

At this experiment, two different methods are presented for rod calibration. Firstly, we do calibrate one rod using swap method. Reactor power is between 100 and 300 W. Reactivity is measured using digital reactivity meter. Secondly, rod method is used for the same rod. Both methods are compared.

3.3.6. Thermal calibration

Power meters are usually radiation detectors that do not measure thermal power directly. That is why thermal calibration is necessary for reading accurate power of the reactor.

In the first step, electrical heaters are inserted into the core. Their power is 42 kW. Valves inside primary loop are closed. During the heaters operation, temperature is measured at different positions inside the reactor tank. Also the temperature of concrete and air is measured. In the second step, electrical heaters are removed and reactor is used to heat the water. Power is set to the same level as the power of electrical heaters. At the end, both measurements are compared power meters are appropriately calibrated.

Thermal calibration can also be done in a different way. Only reactor is used to heat up the water. Its temperature is carefully measured. Knowing the volume of the tank and speed of heating the water, power of the reactor can be calculated. Although this method is less accurate, it is more appropriate for trainees due to shorter time.

4. RECENT ACHIEVEMENTS

- Collaboration in the AIDA project: http://aida.web.cern.ch/aida/index.html;
- Thermal neutron irradiation testing of NI PXI and cRIO products.
- Equipment developed by National Instrument that will be part of ITER instrumentation was tested at JSI TRIGA reactor;
- Collaboration with CEA. Testing of new types of neutron and gamma detectors, e.g. fission chambers, ionization chambers, SPNDs, SPGDs.

5. REFERENCES

[1] RRDB: http://nucleus.iaea.org/RRDB/RR/HeaderInfo.aspx?RId=333, September 2013.

[2] RADULOVIĆ, V., et. al., Characterization of the Ex-Core Irradiation Facilities of the JSI TRIGA Mark II Reactor, in Proceedings of the 21st International Conference Nuclear Energy for New Europe, Ljubljana (2012).

[3] JERAJ, R., et. al., Monte Carlo Characterization of Irradiation Facilities in the TRIGA Reactor Core, in Proceedings of the International Conference Nuclear Energy in Central Europe, Portorož, Slovenia (2001).

Profile 27

ADVANCED TEST REACTOR (ATR)

UNITED STATES OF AMERICA

1. GENERAL INFORMATION AND TECHNICAL DATA OF RR (link with RRDB)[1]

The advanced test reactor (ATR), located at the Idaho National Laboratory (INL), is one of the most versatile operating research reactors in the world. The ATR has a long history of supporting reactor fuel and material research for the US government and other test sponsors. The INL is owned by the US Department of Energy (DOE) and currently operated by Battelle Energy Alliance (BEA). The ATR is the third generation of test reactors built at the INL with a mission to study the effects of intense neutron and gamma radiation on reactor materials and fuels. The current experiments in the ATR are sponsored by a variety of customers — the US government, foreign governments and companies, and private researcher institutions, and commercial companies that need neutrons. The ATR has several unique features that enable the reactor to perform diverse simultaneous test for multiple test sponsors. The ATR has been operating since 1967, and is expected to continue operating for several more decades.

FIG. 1. Idaho National Laboratory.

The ATR possesses large test volumes in high-flux areas. Designed to generate high radiation exposures in a short period of time, a maximum unperturbed thermal neutron flux of 10^{15} $n \cdot cm^{-2} \cdot s^{-1}$ can be achieved in some locations at the rated thermal power of 250 MW. Since most contemporary experimental objectives generally do not require the limits of its operational capability, the ATR typically operates at much lower power levels (~ 110 MWth) with some sections of the core operating at a higher power than others to accommodate user needs. The ATR is cooled by pressurized (2.5 MPa (360 psig)) water that enters the reactor vessel bottom at an average temperature of 52°C (125°F), flows up outside cylindrical tanks that support and contain the core, passes through concentric thermal shields into the open

[1] ATR National scientific user facility — http://atrnsuf.inl.gov/;
Advanced test reactor — http://www.inl.gov/research/advanced-test-reactor-research/;
Advanced test reactor irradiation facilities and research http://anes.fiu.edu/Pro/s8Fur.pdf.

upper part of the vessel, then flows down through the core to a flow distribution tank below the core. When the reactor is operating at full power, the primary coolant exits the vessel at a temperature of 71°C (160°F). The unique design of the ATR core (see FIG.) and its control elements permit large power variations among its nine flux traps using a combination of fuel management, control cylinders (drums), and neck shim rods. The beryllium control cylinders contain hafnium plates that can be rotated toward and away from the core, and hafnium shim rods, which withdraw vertically, can be individually inserted or withdrawn for minor power adjustments. Within bounds, the power level in each corner lobe of the reactor can be controlled independently to allow for different power and flux levels in the four corner lobes during the same operating cycle.

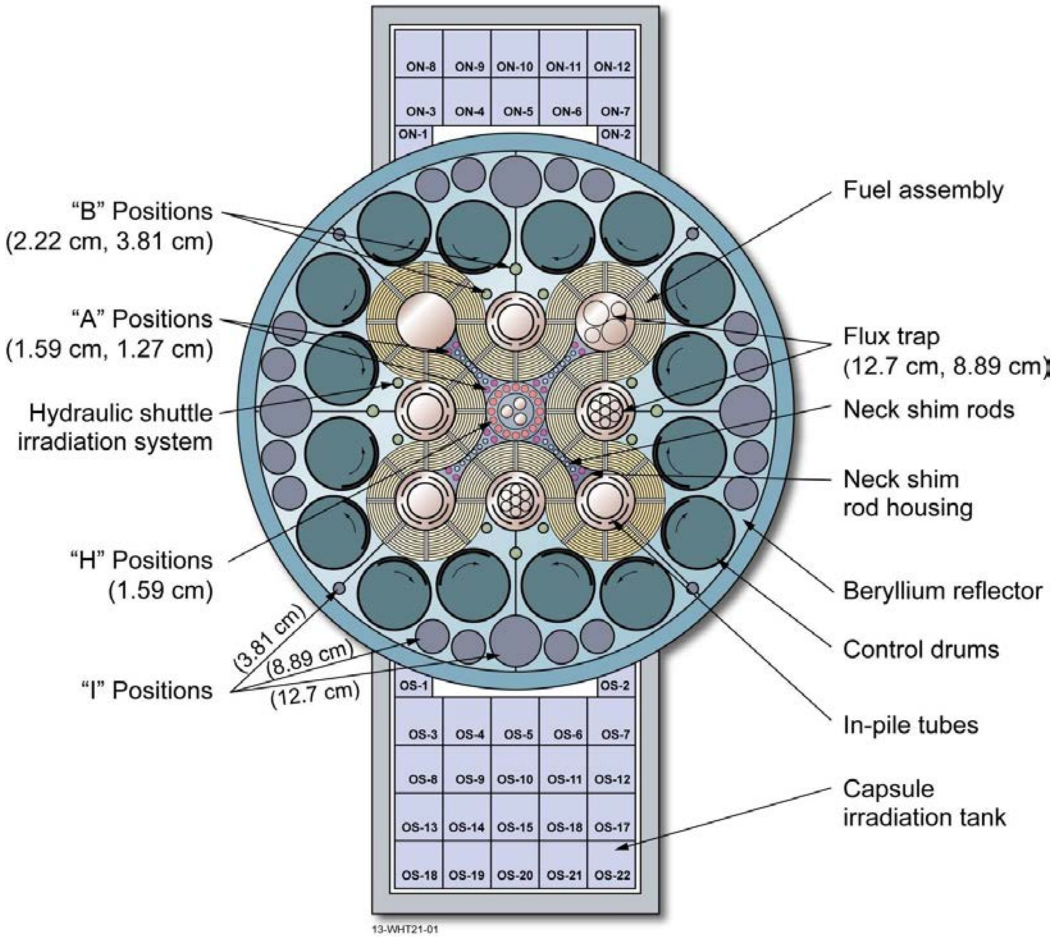


FIG. 2. Core layout of the ATR.

Each of the nine lobes can be fitted with an independent cooling system (in-pile tube) to enable ‘loop’ testing of materials under a wide range of temperature and coolant conditions. Currently, six of the nine lobes are fitted with in-pile tubes.

The reactor is used primarily by the US DOE, but with increasing use by other US government programmes, commercial organizations, and international customers. The ATR was designed to accommodate a wide variety of testing requirements. The key design features are as follows:

- Large test volumes — up to 1.22 m long (at all testing locations) and up to 12.7 cm diameter;
- A total of 77 testing positions;

— High neutron flux — up to 10^{15} $n \cdot cm^{-2} \cdot s^{-1}$ for accelerated irradiations of fuels and materials at full power (the ATR is currently operated at a typical power of 110 MW, with peak thermal flux of 4.4 x 10^{14} $n \cdot cm^{-2} \cdot s^{-1}$);
— Variety of fast/thermal flux ratios (0.1–1.0);
— Constant axial power profile;
— Power tilt capability (four independent lobe powers);
— Individually controlled experiments in different test locations;
— Cycle length of 1 to 55 days;
— No core life limit — core internals are replaced every ten years;
— Power axial locator mechanism to create operational transient conditions in loop experiments.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT ATR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

The unique core design ATR places the reactor fuel closer to the flux trap experiments than is possible in a rectangular grid. The ATR has nine of these high-intensity neutron flux traps and 68 additional irradiation positions inside the reactor core reflector tank, each of which can contain multiple experiments. This is further complemented by two capsule irradiation tanks outside the core with 34 additional low-flux irradiation positions.

The physical dimensions of the available test positions in the ATR range in size from 1.27 cm (0.50 inches) in diameter to 12.7 cm (5.00 inches) in diameter; all test positions are 1.22 m (48 inches) long. Sizes and typical flux levels for positions in flux traps, neck shim housing and reflector are listed in TABLE 1.

Targets are inserted into the ATR inside ‘experiment assemblies’. The components of these assemblies are the targets, the capsule, and the basket. The capsule serves to provide a boundary to contain the target material and isolate it from the reactor primary coolant. The capsule is designed with an internal annulus generally filled with an inert gas such as helium or argon. The basket serves as the housing of the capsule(s) and is designed to mate with the irradiation position in the reactor.

Baskets and capsules are supplied by INL where feasible. Designs for these baskets and capsules have already been qualified within the ATR safety and operational envelope. In the future, users may work with the INL to develop new capsule and basket designs; however, each of these will have to be analysed to ensure that the experiment does not compromise the ATR safety basis.

The following is a summary of the irradiation spaces available in ATR:
— Group A positions are near the cruciform-shaped neck shim rod housing; the inner eight positions (A-1 to A-8) are used mainly for long-term irradiations due to limited accessibility;
— Group B positions are located in the beryllium reflector surrounding the core, inside the rotating control cylinders. The smaller B positions are located close to the fuel elements and, as Table 2 indicates, have considerably higher neutron flux than the larger B positions;
— The I positions are in the periphery of the beryllium reflector, outside the rotating control cylinders, thus the flux is lower than the other ATR positions, but these positions are larger than most others in the ATR;

— The Hydraulic shuttle irradiation system (HSIS) operates in ATR position B-7. This system enables test samples to be inserted into, and removed from the ATR during operations;
— The flux traps are the highest flux positions, and, with the exception of the large I positions, are the largest test locations in the core. These can be used for static capsule, instrumented lead, and loop experiments;
— Group H positions are in the centre flux trap assembly. Positions H-3 and H-11 are used for N-16 monitors and are thus not available for irradiations. The other 14 H positions extend 15.24 cm (6.0 inches) below the active core for a total length of 137.24 cm (54 inches);
— The Outer irradiation tank positions (ON-1 to ON-12 and OS-1 to OS-22) are constructed with three variable diameters to enable simultaneous irradiation tests of differing dimensions.

TABLE 1. APPROXIMATE PEAK VALUES IN IRRADIATION POSITIONS AT 110 MW (th)

Position	Diameter (cm/in.)[a]	Thermal flux ($n \cdot cm^{-2} \cdot s^{-1}$)[b]	Fast flux (E>1 MeV) ($n \cdot cm^{-2} \cdot s^{-1}$)	Typical gamma heating W/g (SS)[c]
Northwest and Northeast flux traps	13.3/5.250	4.4×10^{14}	2.2×10^{14}	
Other flux traps	7.62/3.000[d]	4.4×10^{14}	9.7×10^{13}	
A-Positions				
(A-1 – A-8)	1.59/0.625	1.9×10^{14}	1.7×10^{14}	8.8
(A-9 – A-16)	1.59/0.625	2.0×10^{14}	2.3×10^{14}	
B-Positions				
(B-1 – B-8)	2.22/0.875	2.5×10^{14}	8.1×10^{13}	6.4
(B-9 – B-12)	3.81/1.500	1.1×10^{14}	1.6×10^{13}	5.5
H-Positions (14)	1.59/0.625	1.9×10^{14}	1.7×10^{14}	8.4
I-Positions				
Large (4)	12.7/5.000	1.7×10^{13}	1.3×10^{12}	0.66
Medium (16)	8.26/3.500	3.4×10^{13}	1.3×10^{12}	
Small (4)	3.81/1.500	8.4×10^{13}	3.2×10^{12}	
Outer tank positions				
ON-4	Var[e]	4.3×10^{12}	1.2×10^{11}	0.15
ON-5	Var[e]	3.8×10^{12}	1.1×10^{11}	0.18
ON-9	Var[e]	1.7×10^{12}	3.9×10^{10}	0.07
OS-5	Var[e]	3.5×10^{12}	1.0×10^{11}	0.14
OS-7	Var[e]	3.2×10^{12}	1.1×10^{11}	0.11
OS-10	Var[e]	1.3×10^{12}	3.4×10^{10}	0.05
OS-15	Var[e]	5.5×10^{11}	1.2×10^{10}	0.20
OS-20	Var[e]	2.5×10^{11}	3.5×10^{9}	0.01

[a] Position diameter; capsule diameter must be smaller.

[b] Average speed 2,200 m/s.

[c] Depends on configuration.

[d] East and south flux traps each contain seven guide tubes with inside diameters of 1.76 cm (0.694 in.). The center flux trap holds the ITV, which has three tubes: two with 1.73 cm (0.681 in.) inside diameter and one with 2.24 cm (0.881 in.) inside diameter.

[e] Variable; can be either 2.22, 3.33, or 7.62 cm (0.875, 1.312, or 3.000 in.)

Experimenters are encouraged to design and/or manufacture targets to fit within the capsule designated for the position(s) into which the targets are intended to be inserted. The availability of individual positions depends upon demand.

The simplest experiment performed in the ATR is a *static capsule* experiment. The material to be irradiated is sealed in aluminum, zircaloy, or stainless steel tubing. The sealed tube is placed in a holder that sits in a chosen test position in the ATR. Capsules typically have no instrumentation, but can include flux-monitor wires and temperature melt wires for analysis following the irradiation. For most of these experiments temperature control is achieved by design of the gap size between the test specimen and the outer capsule; the gap is filled with an inert insulating gas jacket. For some experiments the capsules are not sealed and the test specimens are exposed to the ATR primary coolant system. One advantage of the static capsule test configuration is the relative ease of placement and removal into and out of the reactor. This allows for repositioning of capsules within a basket for different operating cycles.

The next level in complexity is an *instrumented lead experiment*. A common application is the temperature-controlled capsule. During a temperature controlled capsule experiment, a conducting (helium) and an insulating (typically neon or possibly argon) gases are mixed to control the thermal conductance across a predetermined gas gap. Thermocouples measure temperature continuously and provide feedback to the gas system that adjusts the mixture to achieve the desired temperature. This type of configuration can also incorporate a fission gas monitor.

The *pressurized water loop experiment* is the most comprehensive type of testing performed. An in-pile tube runs through the reactor core from vessel top to bottom and is attached to its own individual water system. The cooling system includes pumps, coolers, ion exchangers, heaters to control test temperature, pressure, and chemistry. Figure 3 illustrates a typical loop layout. Loop tests can precisely represent conditions in a commercial power reactor. Each loop has an independent control console, independent of the reactor control room. Loops are typically used for testing reactor core components (fuel, control rods, structural materials, coolant technologies: lead, lead-bismuth, sodium, heavy water, molten salt, gas).

The *Hydraulic shuttle irradiation system (HSIS)* operates in ATR position B-7. This system enables test samples to be inserted into, and removed from the ATR during operations. There are a total of 14 capsules in the HSIS 'train' that will be inserted and removed together. Each capsule is 5.715 cm (2.25 inches) long, and 1.5875 cm (0.625 inches) outer diameter, and all the capsules are titanium. It is anticipated that most uses of the shuttle system will be for durations of a few hours to a few days. In a single HSIS train irradiation capsule contents can be the same (as might be the case for isotope production) or different, as might be desired for material irradiations.

2.2. CONDITIONS

(1) Steady state — ATR operates at a steady state of around 110 MWth. This provides fluxes at the values shown in Table 1;
(2) Transient — a powered axial locator mechanism (PALM) attaches to a test assembly and moves it in and out of the core (axially) to create a time-varying flux. The oscillation rate is programmable;
(3) Accident conditions — loop testing can accommodate tests in which fuel specimens are designed to fail.

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS LOCA, LOFT, RIA) AND THE CORROSION OF MATERIALS

Various prototypical and accident conditions can be created inside of in-pile tubes (loops) with specially designed test assemblies and monitoring equipment. These loops can be used in six of the flux trap positions. Each loop has its own temperature, pressure, flow, and chemistry control systems which can sustain conditions that exceed those in pressurized water reactors. Operational transient conditions can be created with the use of the power axial locator mechanism (PALM) which moves a test train in and out of the core at programmable frequencies.

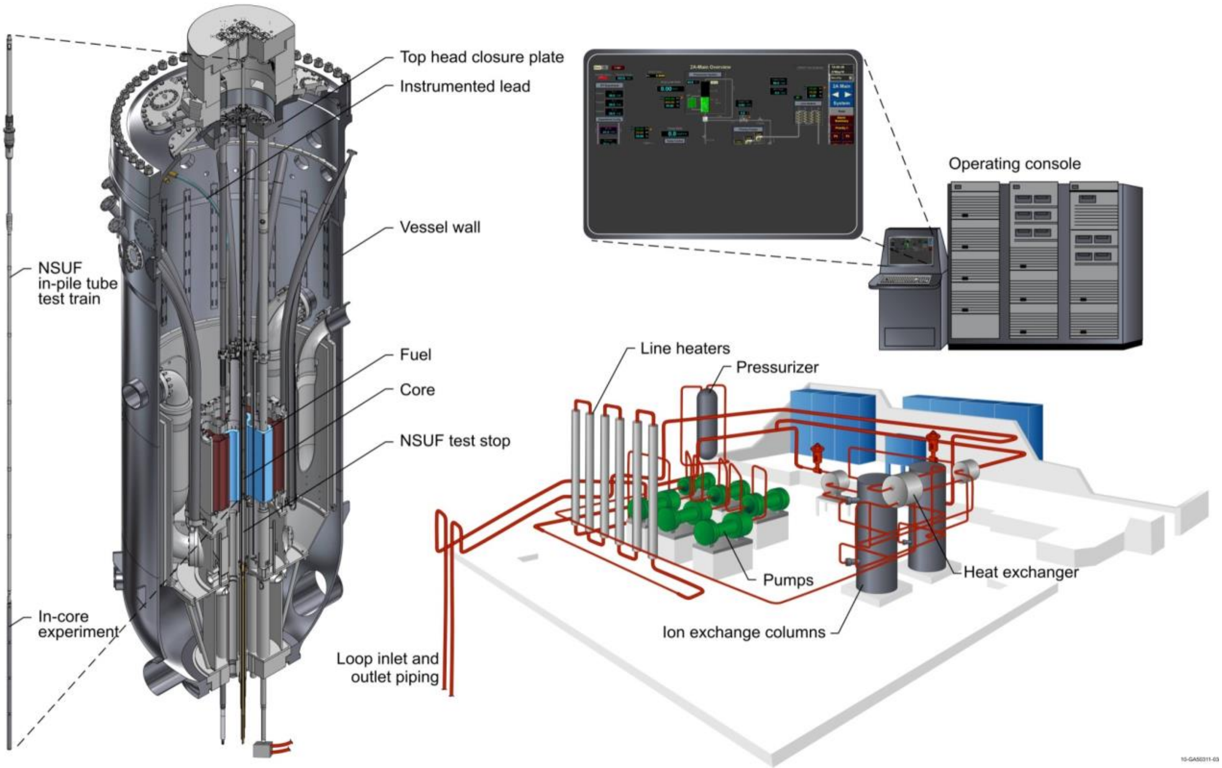


FIG. 3. In-pile tube (loop) in ATR with supporting systems.

2.4. DEVICES FOR CAPSULE/AMPULE TESTS OF MATERIALS IN DIFFERENT ENVIRONMENTS

See next section.

2.5. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR AND CHARACTERISTICS (swelling, gas release, creep, long-term strength, relaxation resistance, etc.)

(1) Irradiation Test Vehicle (ITV);
The ITV (see FIG.4) is one of the sophisticated irradiation capsules designed specifically for ATR. It provides a pressure boundary, a gas jacket, temperature control, and an umbilical tube for instrumented leads or environmental (temperature) control. This system was used for one experiment, and was removed from ATR, however, the

concept was proven successful and similar systems have since been used for experiments.

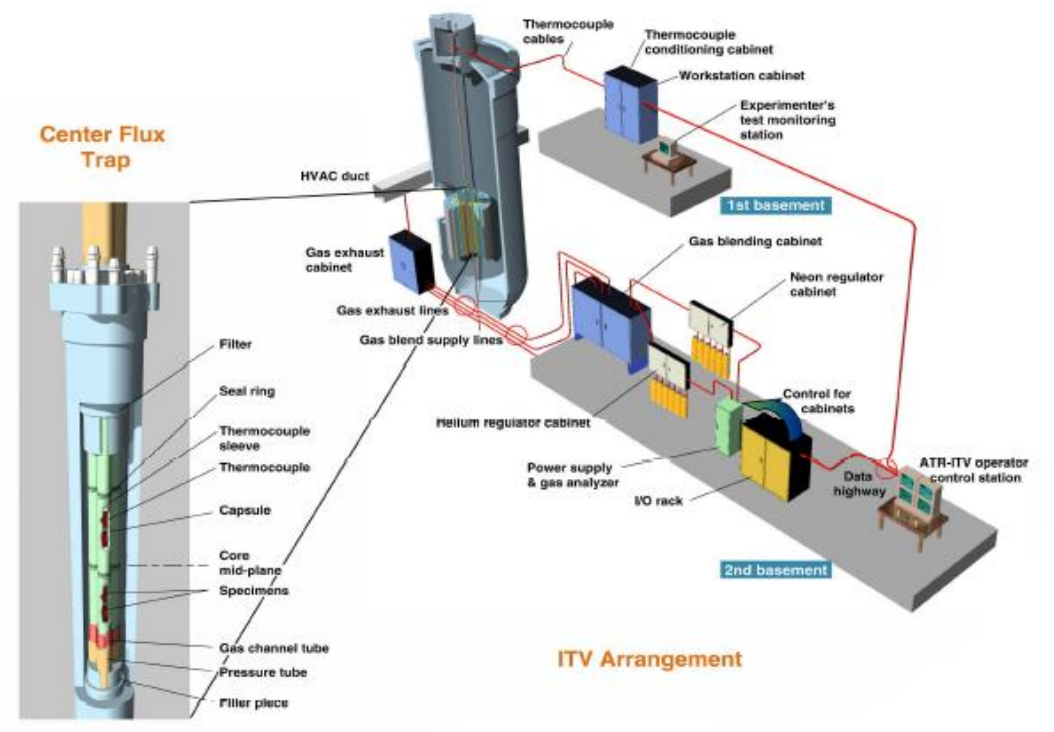
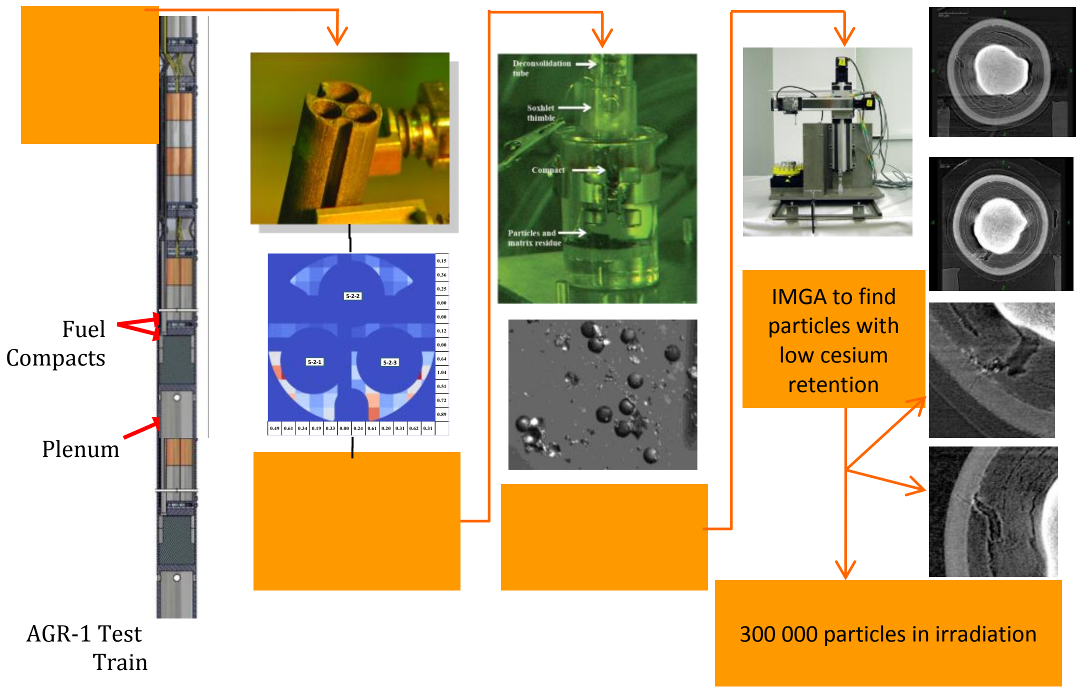


FIG. 4. ITV and online control and data acquisition systems.

(2) Advanced gas reactor fuel test (AGR);

The advanced gas-cooled reactor fuel test assembly (see **Error! Reference source not found.**5) was designed to expose TRISO high temperature reactor fuel particles to irradiation in ATR under a wide range of fluence and temperature conditions while monitoring fission product releases.



(3) Ad

FIG. 5. Gamma scan tomography to identify cesium hot spot and compact location.

The Advanced Graphite Creep test assembly (**Error! Reference source not found.**) is another example of the sophisticated capsule experiment that has been designed for ATR. Several graphite samples are loaded and kept under tightly controlled conditions of temperature, fluence and mechanical stress to obtain data on the response of various grades of graphite to different reactor conditions.

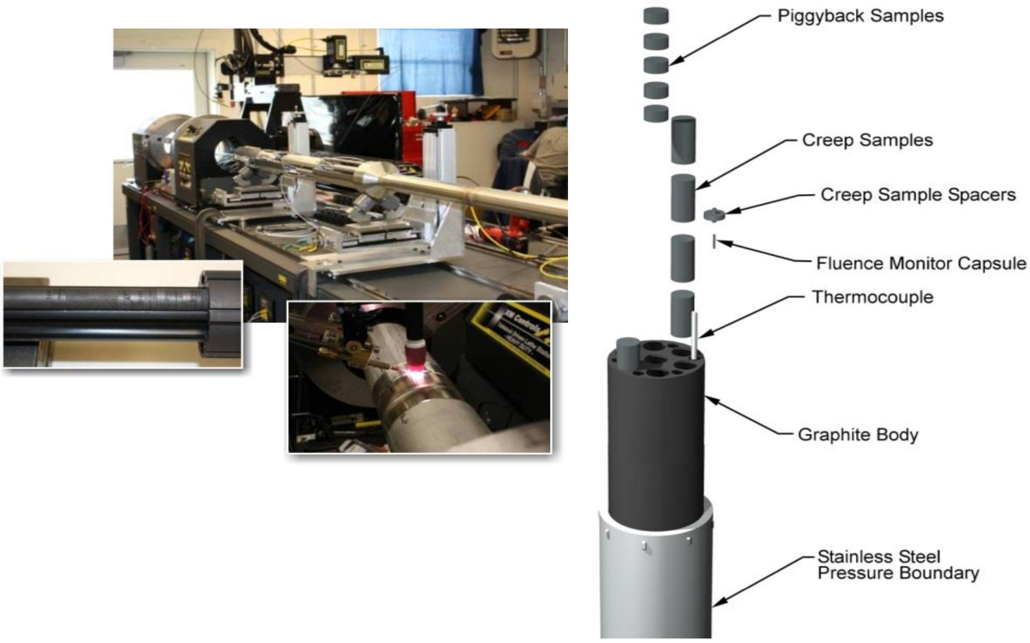


FIG. 6. X-ray tomography and detailed micro-analytic techniques to study nature of defect; deconsolidation to obtain particles from compact.

3. OTHER FACILITIES

Advanced test reactor critical facility (ATRC) — a separate critical (low power) reactor virtually identical to ATR operates to measure the reactivity worth of experiments before insertion into ATR. It can be used for other low power physics experiments and education.

4. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

4.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTICS

The Test train assembly facility (TTAF) is located near the advanced test reactor (ATR) at the Idaho National Laboratory and is used for final assembly of experiments to be installed at the ATR. The TTAF was designed and furnished with the equipment (welding, brazing, machining) necessary to support the assembly of various configurations and design of test train experiments. The TTAF’s staff is assigned to support engineering and perform the experiment assemblies.

Unirradiated targets that contain radioactive constituents cannot be transported over public highways unless packaged in Department of Transportation (DOT) approved containers and appropriately manifested. Irradiated capsules containing target materials are typically transported in a Nuclear Regulatory Commission (NRC) licensed transport cask after cooling.

INL staff can provide support for shipping activities for users that lack qualified shipping personnel.

4.2. HOT CELLS, PIE FACILITIES (radiochemistry facilities, SEM, TEM, X-Ray installations, gamma scanning, neutron beams facilities, etc.)

Post-irradiation examination capabilities are centred at the INL's *Materials and Fuels Complex,* about 30 km from the ATR Complex. The primary facilities here include the Hot fuel examination facility (HFEF), the Analytical laboratory (AL), and the Electronic microscopy laboratory (EML).

The hot fuel examination facility (see FIG. 7) is a large, heavily shielded, alpha-gamma hot cell facility designed for remote examination of highly irradiated fuel and structural materials. Its capabilities include non-destructive (dimensional measurements and neutron radiography) and destructive examination (such as mechanical testing or metallographic/ceramographic characterization). It can accept full-size light water reactor fuel assemblies. HFEF is comprised of two adjacent large, shielded hot cells in a three-story building, as well as a shielded metallographic loading box, an unshielded hot repair area and a waste characterization area. The main cell (argon atmosphere) has 15 workstations, each with a viewing window and a pair of remote manipulators. A decontamination cell (air atmosphere) has six similarly equipped workstations. The cells are equipped with overhead cranes and overhead electromechanical manipulators. Cell exhaust passes through two stages of HEPA filtration. The facility is linked to analytical laboratories and other facilities by pneumatic sample transfer lines.

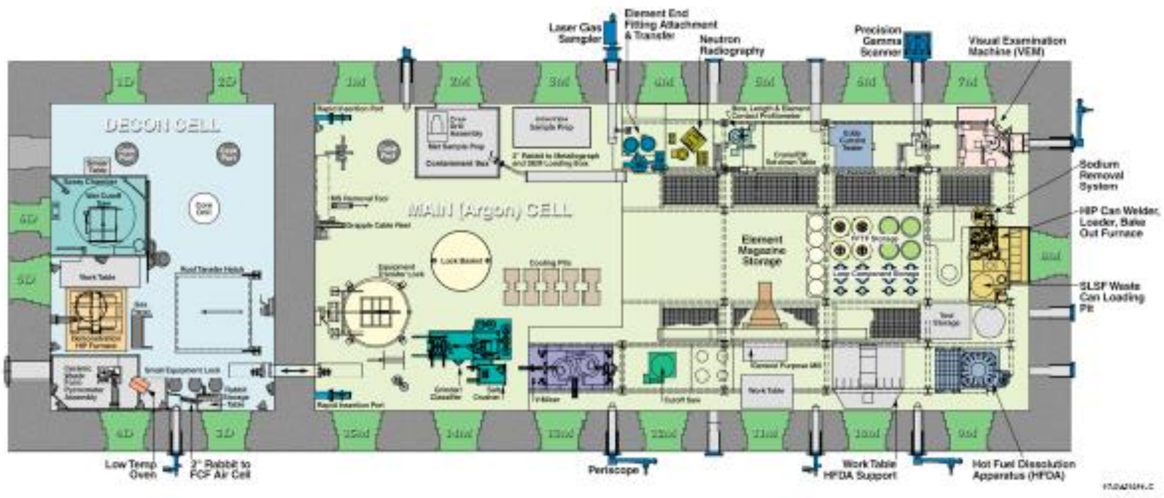


FIG. 7. Layout of the HFEF.

Each main cell work station has removable electrical and lighting feed-through that can be changed to accommodate the mission of the station. The main cell is equipped with two rapid insertion ports for quick transfer of small tools and items into the cell. The decontamination cell contains a spray chamber for decontaminating equipment and non-fissile material using a manipulator-held wand. Material handling takes place via a 750-lb electro-mechanical manipulator, a five-ton crane and six sets of master-slave manipulators. The hot repair area is available for contact maintenance on cell equipment; it can also be used for transfer of equipment and materials to or from the decontamination cell. HFEF also has a 250 kW

Training Research Isotope General Atomics (TRIGA) reactor, for neutron radiography irradiation to examine internal features of fuel elements and assemblies.

TABLE 2. MEASUREMENT AND CHARACTERIZATION EQUIPMENT WITHIN HFEF

Capability	Device
Neutron radiography	250 kW TRIGA reactor
Element/capsule diameter measurements	Element contact profilometer
Element/capsule gas sampling	Gas assay sample and recharge
Element/capsule weight	Element/capsule balance (Mettler)
Element/capsule fission and activation product distribution	Precision gamma scanning
Element/capsule bowing and length	Bow and length machine
Element/capsule visual exam	Visual exam machine
Macro photography	High resolution digital photography
High precision specific gravity measurements	Pycnometer
Metallography	Leitz metallograph
Microhardness	LECO AMH43 automatic hardness tester

The Materials and fuels complex analytical laboratory (AL) is coupled to HFEF via a pneumatic sample transfer system. The AL (see FIG. 8) offers NIST-traceable chemical and isotopic analysis of irradiated fuel and material via wide range of techniques, such as ICP-MS (Inductively Coupled Plasma - Mass Spectrometry), ICP-OES (Inductively Coupled Plasma - Optical Emission Spectrometry), and ICPMS-DRC (Inductively Coupled Plasma - Mass Spectrometry - Dynamic Reaction Cell), and TIMS (Thermal Ionization Mass Spectrometry).

FIG. 8. Hot cells within the Analytical Laboratory.

The *Electron Microscopy Laboratory* is a radiological materials area (RMA), permitting work to be performed with both radioactive and non-radioactive materials. A portion of the laboratory is dedicated to sample preparation, providing the researcher with facilities support, equipment, safety systems, and procedures to prepare samples of diverse materials for analysis. The primary instruments in EML are a JEOL 2010 scanning transmission electron microscope (TEM), a JEOL JSM-7000f scanning electron microscope (SEM), a Zeiss DSM 960a SEM, and a Focused Ion Beam (FIB). The TEM is capable of operating at 200 kV, and is capable of magnifications from 2000 X to 1 500 000 X. It is equipped with an Oxford Instruments energy dispersive X-ray spectrometer that can be used to gather information about the elemental make-up of a sample. Crystallographic information can be obtained by recording the diffraction patterns formed by electrons as they pass through the sample.

The JEOL SEM is a field emission instrument capable of operating at 30 kV, and is capable of magnifications from 15 X to 100 000 X. It is equipped with Oxford Instruments energy dispersive (EDS) and wavelength dispersive X-ray spectrometers (WDS) that can be used to obtain quantitative information about the elemental composition of a sample. It is also equipped with an electron back scatter diffraction detector (EBSD) that can be used to obtain crystallographic information about a sample by recording the diffraction patterns formed by electrons when they tunnel through a sample at glancing angles.

The Zeiss SEM is capable of operating at 30 kV, and is capable of magnifications from 6 X to 50 000 X. It is equipped with Oxford Instruments energy dispersive and wavelength dispersive X-ray spectrometers and an electron back scattered diffraction camera.

This FIB has the ability to analyse the three-dimensional structure and chemistry of materials on a submicron scale. The goal is to characterize irradiated nuclear fuels to detect submicron-level damage, which would make the INL instrument unique in the world. A better understanding of this process has significant potential to improve in-reactor fuels and materials performance.

In addition to the TEM and SEM, EML also has several optical microscopes. Some of these are used to support sample preparation, and others are used for optical characterization of samples. Capabilities for sample preparation include cutting, grinding, and polishing, as well as specialized methods such as ultramicrotomy (cutting ultrathin slices of material with a special machine using a diamond knife), chemical and ion milling to produce thin, electron-transparent samples, etching, and coating. Fume hoods (radiological and non-radiological) and a radiological glove box are available to protect workers and the environment from hazardous materials.

In addition to the hot cell facilities at MFC, additional PIE equipment is available in Idaho Falls (about 75 km from ATR) at the Centre for Advanced Energy Studies (CAES). CAES is a laboratory facility, cooperatively owned and operated by INL and several universities. One of the laboratories in CAES is the Microscopy and Characterization Suite (MaCS). Instruments available for use in MaCS are as follows (these can be used for low level radioactive samples or very small samples):

- Local Electrode Atom Probe (LEAP) — this CAMECA 4000XHR LEAP can create 3-D images of atoms in solids;
- FIB — this FEI Quanta 3DFEG FIB sections materials at micro- and nano- scales for TEM and LEAP microscopy;
- TEM — FEI Technai TF30-FEG STwin STEM for nano-scale material structure images;
- Spark Plasma Sintering — Creates fully dense metals, ceramics and metal-ceramic composites;

— Nanoindenter and Atomic Force Microscope — Hysitron model TI-950 TriboIndenter measures mechanical properties on very small scale samples;
— Automated Hardness Tester — Measures and evaluates the micro-hardness of materials;
— SEM — JEOL JSM-6610LV Images material surfaces at the nano-scale.

4.3. FACILITY UPGRADES

The following equipment will be installed at the Materials and Fuels Complex in the near future to enhance the INL's measurement and characterization capabilities:

— Shielded electron microprobe designed to assess fission product distribution in irradiated fuels, this new instrument performs micro-structural and micro-chemical analysis of fresh and irradiated fuels and waste forms. As a specialized scanning electron microscope, it can also analyse localized micron-scale chemical composition data of irradiated fuels and materials.
— Thermal ionization mass spectrometer 14 replacing an existing instrument that has reached the end of its operational life, this instrument will perform elemental assay and isotopic composition on plutonium, uranium and minor actinides prepared from fresh and irradiated fuels.
— Micro X-ray diffractometer — the purpose of this device, which performs micro-scale phase identification, small-sample powder diffraction and texture determination, is to track the evolution of fuel structure during irradiation.
— Mechanical test equipment and sample preparation equipment funded by Battelle Energy Alliance — these upgrades include new mechanical test and sample preparation equipment in the HFEF hot cells — specifically a mechanical test load frame, power supply and an out-of-cell control console as well as sample cutting and preparation tools.
— TN-FSV cask NRC License Modification — this work comprises modifying the Certificate of Compliance for the TN-FSV transportation cask to include payloads important to the mission of INL fuels research and reactor development. The scope includes fabrication of a new inner-shielded cask insert.

5. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS INCLUDING HUMAN RESOURCES DEVELOPMENT

The INL has several facilities where ATR experiments can be fabricated and/or assembled. At the MFC, there are fuels and materials facilities for fabricating nuclear fuels and non-fissile targets. The Test train assembly facility (TTAF), which is a portion of the ATR Complex hot cell facility building, serves as a location for the mechanical assembly for both test capsules and test trains that will be inserted into the ATR. The TTAF has two 30-ft (9.144 m) long tables to facilitate the assembly of long test train assemblies. The facility contains power equipment including two different sizes of lathe welders, induction brazing equipment, tungsten inert gas (TIG) welders, spot welder, drying ovens, drill presses, vacuum systems, and electro-plating equipment.

The High temperature test laboratory (HTTL) contains specialized equipment to conduct high-temperature testing. HTTL's trained staff evaluates high-temperature material properties and develops custom high-temperature instrumentation for nuclear and non-nuclear applications.

6. RECENT ACHIEVEMENTS, SOME EXAMPLES OF R&D STUDIES PERFORMED DURING THE LAST TEN YEARS (link to the list of publication is recommended)

Graphite Materials testing in ATR for lifetime management of MAGNOX reactors:
http://www.ntis.gov/search/product.aspx?ABBR=DE2001797458;

Advanced gas reactor fuel development and qualification:
https://secure.inl.gov/VHTRRDTR12/pres/Opening/03_Petti_StatusofNGNPAGRFuel.pdf,
https://secure.inl.gov/VHTRRDTR12/presentations.aspx, (fuels breakout sessions);

Advanced graphite creep experiment (Very high temperature reactor R&D):
https://secure.inl.gov/VHTRRDTR12/pres/Graphite/01_Swank_AGCExperiment.pdf,
https://secure.inl.gov/VHTRRDTR12/presentations.aspx, (graphite breakout sessions);

High temperature test laboratory:
https://inlportal.inl.gov/portal/server.pt/community/distinctive_signature__icis/315/httl_accomplishments.

Profile 28

HIGH FLUX ISOTOPE REACTOR (HFIR)[1]

UNITED STATES OF AMERICA

1. GENERAL INFORMATION

TABLE 1. TECHNICAL DATA

Operating power	85 MW
Average cycle length	24 days
Peak power density	2 MW/L
Peak thermal neutron flux	2.3×10^{15} n·cm^{-2}·s^{-1}
Peak fast neutron flux	1.2×10^{15} n·cm^{-2}·s^{-1}
Max DPA per cycle	2 dpa
Total number of irradiation sites (holes)	79 full-length sites (are/can be subdivided axially)

FIG. 1. Aerial view of the HFIR and REDC facilities.

Operating at 85 MW, HFIR is the highest flux reactor-based source of neutrons for condensed matter research in the USA, and it provides one of the highest steady-state neutron fluxes of any research reactor in the world. The fast, thermal and cold neutrons produced by HFIR are used to study physics, chemistry, materials, engineering, and biology. HFIR is currently scheduled to provide 140 days of 100%-power operation per year.

[1] Oak Ridge National Laboratory managed by UT-Battelle for the Department of Energy.

In addition to in-core irradiations for medical, industrial, and isotope production and research on severe neutron damage to materials, HFIR offers several other facilities supporting basic and applied research:

- Neutron activation analysis (NAA) to examine trace elements and identify the composition of materials;
- Gamma irradiation capability that uses spent fuel assemblies and is capable of accommodating high gamma dose experiments;
- Four beam lines with 12 world-class instruments for condensed matter research[2]. To use the neutron scattering capabilities of HFIR, please contact the Neutron Scattering Sciences User Group at neutrons.ornl.gov.

1.1. REACTOR OVERVIEW

HFIR is a beryllium-reflected, light water-cooled and moderated, flux-trap type reactor that uses highly enriched ^{235}U as fuel. Operating at 85 MW, HFIR produces an average thermal neutron flux of 2.3×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$. Additionally, a peak fast flux of 1.2×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$ is available in the flux trap. The image below is a cutaway of the reactor which shows the pressure vessel, its location in the reactor pool, and some of the experiment facilities.

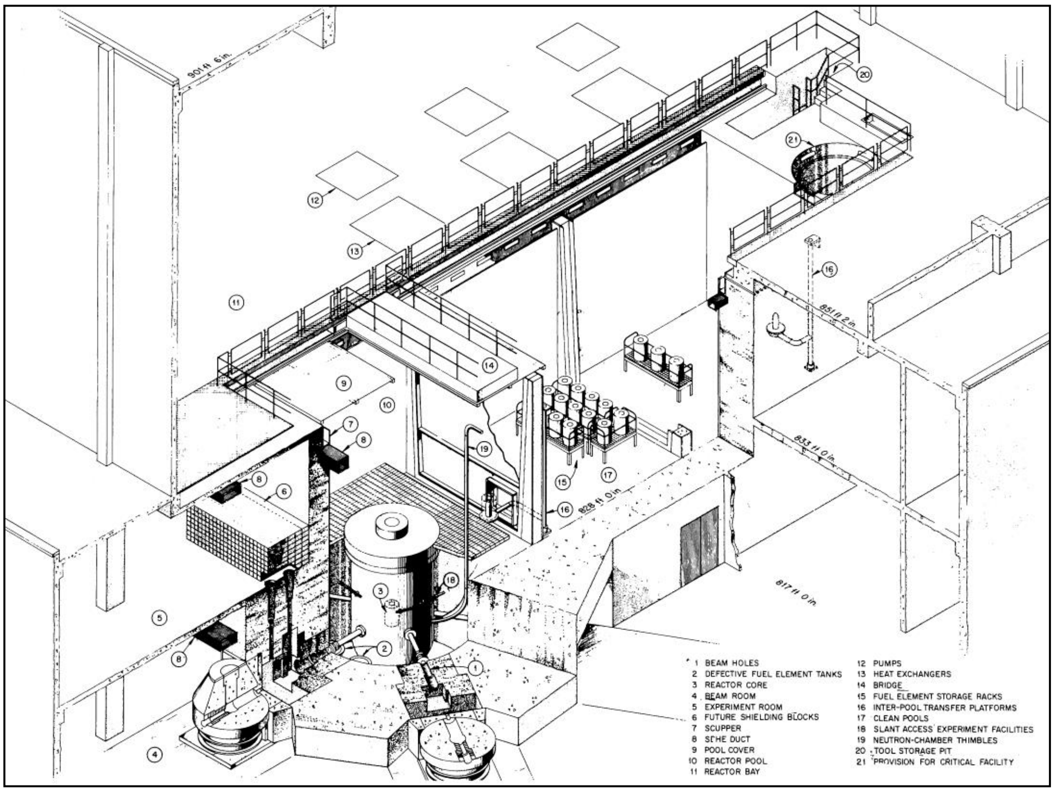


FIG. 2. HFIR building.

The HFIR core consists of four annular regions:

(1) Flux trap region;

(2) Fuel region;

[2] To use the neutron scattering capabilities of HFIR, please contact the Neutron scattering sciences user group at neutrons.ornl.gov.

(3) Control region;and
(4) Reflector region.

The annular fuel cores create a moderating region or 'island' in the flux trap region. This design permits fast neutrons emanating from the fuel to be moderated in the island and thus produces a region of very high thermal-neutron flux at the centre of the island. This reservoir of thermalized neutrons is 'trapped' within the reactor, making it ideal for isotope production. Additionally, the edges of the flux trap experiences high fast neutron flux since it is physical very close to the fuel, providing an excellent environment for materials damage testing. The large thermal neutron flux in the reflector outside the fuel is accessed by beam tubes extending from the sides of the reflector and vertical experiment facilities (VXF) accessed from the top of the reflector. The beam tubes allow neutrons to be streamed into experiments outside the reactor shielding.

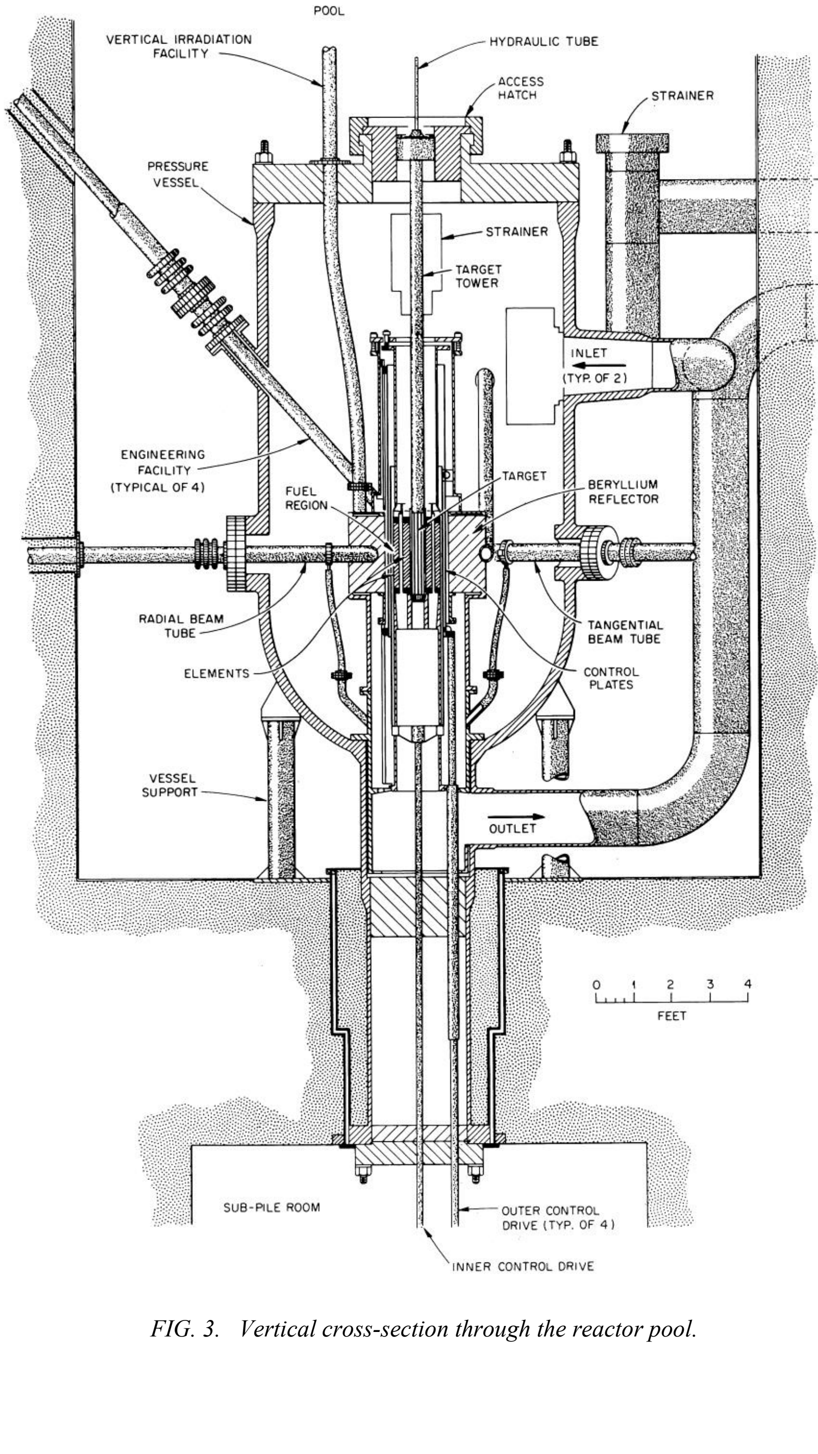


FIG. 3. Vertical cross-section through the reactor pool.

1.2. REACTOR CORE ASSEMBLY

The reactor core assembly is contained in an 8 ft (2.44 m) diameter pressure vessel located in a pool of water. The top of the pressure vessel is 17 ft (5.18 m) below the pool surface, and the reactor horizontal mid-plane is 27.5 ft (8.38 m) below the pool surface. The control plate drive mechanisms are located in a sub-pile room beneath the pressure vessel. These features provide the necessary shielding for working above the reactor core and greatly facilitate access to the pressure vessel, core, and reflector regions.

The reactor core is cylindrical, just over 2 ft (0.76 m) high and 17 (43.18 cm) inches in diameter. A 5 inches (12.70 cm) diameter hole, referred to as the ‘flux trap’, forms the center of the core. The target positions typically contain both materials for damage studies and trans plutonium isotopes in the flux trap. The fuel region is comprised of two concentric fuel elements. The inner element contains 171 fuel plates, and the outer element contains 369 fuel plates. The fuel plates are curved in the shape of an involute, thus providing a constant coolant channel width between each curved plate. The fuel (U_3O_8-Al cermet) is non-uniformly distributed along the arc of the involute to minimize the radial peak-to-average power density ratio. A burnable poison (boron-10) is included in the inner fuel element primarily to flatten the radial flux peak providing a longer cycle for each fuel element. The average core lifetime with typical experiment loading is between 23 and 25 days at 85 MW.

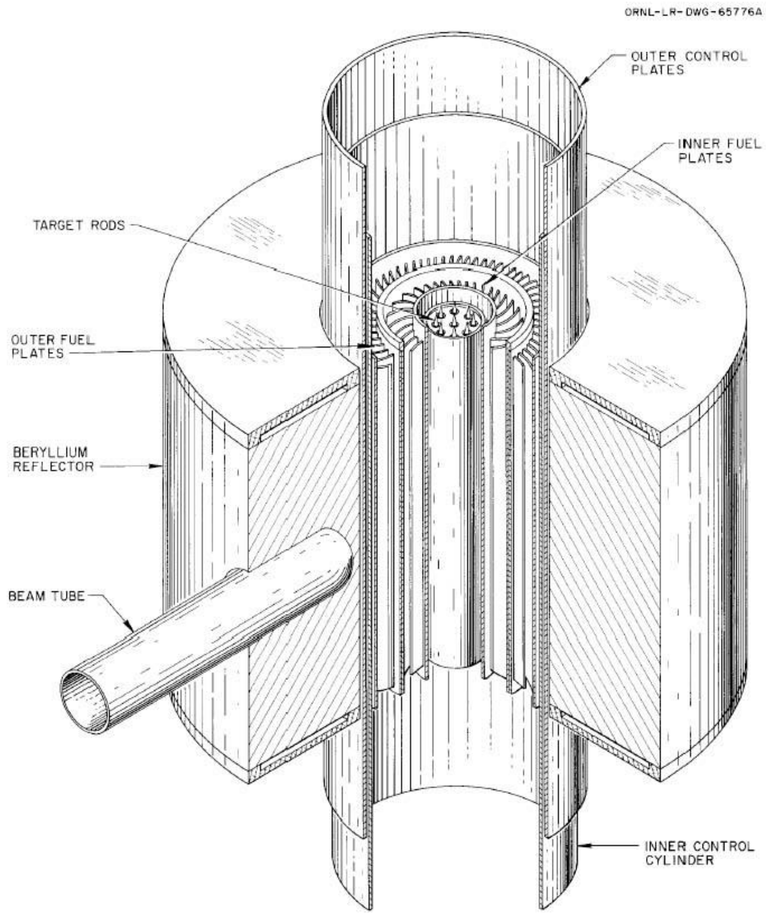


FIG. 4. Basic components of the HFIR core.

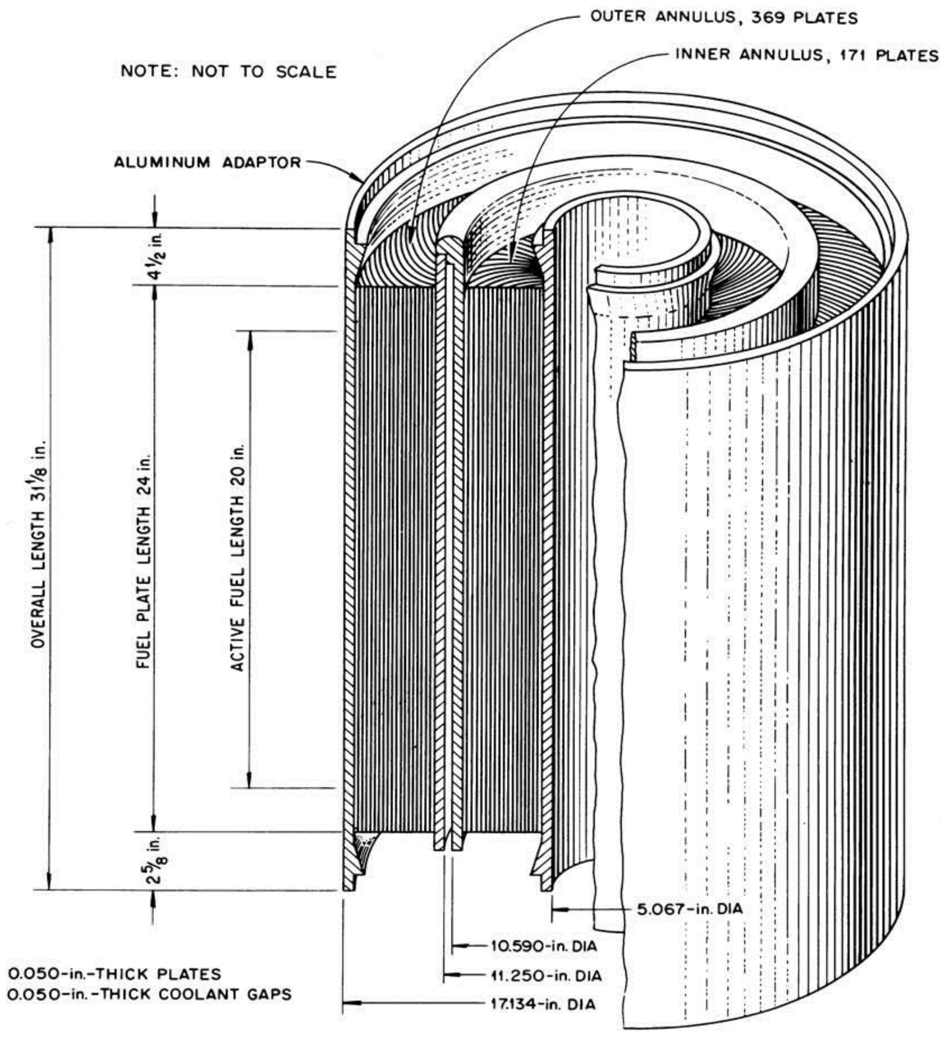


FIG. 4. HFIR fuel element assembly.

The fuel region is surrounded by concentric rings of beryllium reflector totaling approximately 1 ft (30 cm) thick. The beryllium is surrounded by a water reflector of effectively infinite thickness. In the axial direction, the reactor is reflected by water.

The control plates, in the form of two thin, poison-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector. These plates effectively block neutron communication between the fuel and the reflector until they are driven in opposite directions to open a window at the core mid-plane. The inner cylinder is used for shimming and power regulation and performs no fast safety function. The outer control cylinder consists of four separate quadrant plates, each having an independent drive and safety release mechanism. Any single quadrant plate or cylinder is capable of shutting the reactor down.

FIG. 5. HFIR control room.

The cooling system consists of a primary and a secondary system. The primary coolant enters the pressure vessel through two 16 in. (40.64 cm) diameter pipes above the core, passes through the core, and exits through a single 18 in. (45.72 cm) diameter pipe beneath the core. The flow rate is approximately 16 000 gpm (1.01 m³/s), of which approximately 13 000 gpm (0.82 m³/s) flows through the fuel region. The remainder flows through the target, reflector, and control regions. The system is designed to operate at a nominal inlet pressure of 468 psig (3.33×10^6 Pa). Under these conditions the inlet coolant temperature is 120°F (49°C), the corresponding exit temperature is 156°F (69°C), and the pressure drop through the core is approximately 110 psi (7.58×10^5 Pa).

From the reactor, the coolant flow is distributed to three of four identical heat exchanger and circulation pump combinations, each located in a separate cell adjacent to the reactor and storage pools. Each cell also contains a let-down valve that controls the primary coolant pressure. A secondary coolant system removes heat from the primary system and transfers it to the atmosphere by passing water over a four-cell induced-draft cooling tower.

A fuel cycle for the HFIR normally consists of full-power operation at 85 MW for a period of 23 to 25 days followed by an end-of-cycle outage for refuelling, maintenance and upgrades. End-of-cycle refuelling outages vary as required to allow for control plate replacement, calibrations, maintenance, and inspections[3]. Experiment insertion and removal may be accomplished during any end-of-cycle outage. Interruption of a fuel cycle for experiment installation or removal is not permitted to avoid impact on neutron scattering research. Deviations from the schedule are infrequent.

1.3. NEUTRON FLUXES AND GAMMA HEATING RATES

Neutron flux in the HFIR core is relatively constant throughout the 25-day cycle, however some shifts occur at the top and bottom of the fuelled region due to the withdrawal of the

[3] An operational schedule can be found at neutrons.ornl.gov.

control plates throughout the cycle. Peak thermal flux in HFIR in the flux trap has a magnitude of 2.3×10^{15} n·cm^{-2}·s^{-1}. Fast flux in this region reaches 1.2×10^{15} n·cm^{-2}·s^{-1}. Outside of the fuel, in the reflector, fast flux diminishes quickly and most neutrons are thermalized in the first few centimetres of the beryllium reflector. The outer-most positions of the reflector offer 1.0×10^{14} n·cm^{-2}·s^{-1}flux and near 99.5% thermalization.

The neutron flux values herein represent unperturbed values and were obtained from one-dimensional 33-group diffusion theory calculations supplemented by a limited number of two-dimensional few group diffusion theory calculations. The calculated unperturbed fluxes are believed to be accurate to within less than ± 15% of the actual values in regions away from the control plates. The accuracy of calculated values near the control region is more uncertain. This applies especially to thermal-neutron fluxes at the start of the fuel cycle when the control plates are partially inserted.

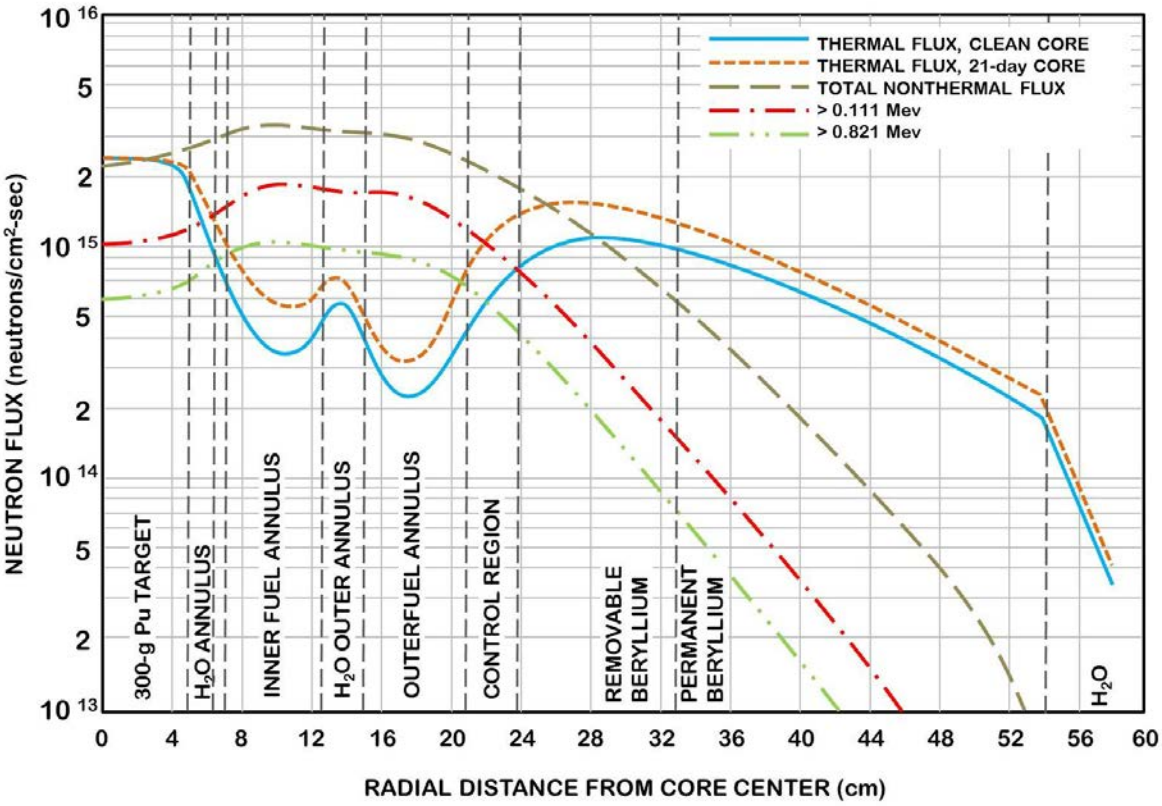
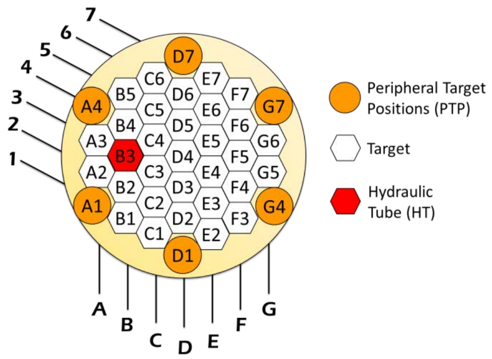


FIG. 6. HFIR neutron flux as a function of distance from the centerline of the core.

With regard to gamma-heating rates, the values provided here were obtained from calculations and may differ from the actual values that exist in the reactor. Specifically, the calculated values are believed to be conservative (i.e., higher than actual values), based on comparison with a limited number of measurements in the flux trap region. Therefore, if safety considerations are paramount, the calculated values given here should be used. Conversely, if the nature of the experiment requires an accurate knowledge of the gamma-heating rates to calculate the temperature of the material being irradiated, then measured values for the particular locations should be obtained and used. Assistance in this area is available. These gamma-heating rates are in units of W/g of aluminium. The results of limited measurements of gamma-heating rates in different materials, performed in the centre of the flux trap, are given at the end of this section. In the absence of other information, these values can serve as a basis for estimating gamma-heating rates in materials other than aluminium applicable to any experiment facility in the reactor.

Additionally, Monte Carlo N-Particle Transport Codes can be used to simulate specific conditions and material experiments in HFIR[4].

TABLE 2. GAMMA HEATING RATES AT VARIOUS POSITIONS IN THE FLUX TRAP



	Vertical centerline Radial distance from reactor (cm)	Gamma heating rate in aluminum (W/gram) Elevation above reactor Midplane (cm)					
		0.00	5.08	10.16	15.24	20.32	25.40
C3, C4, D3, D5, E4, E5	1.689	37.7	34.3	31.7	28.2	23.5	17.0
B3, C2, C5, E5, E6, F5	2.926	38.8	34.7	31.8	28.3	23.5	17.0
B2, B4, D2, D6, F4, F6	3.378	39.3	35.0	32.0	28.4	23.6	17.0
A2, A3, B1, B5, C1, C6, E2, E7, F7, G5, G6	4.465	39.6	35.6	32.5	28.8	24.1	17.0
A1, A4, D1, D7, G4, G7	5.067	40.9	36.0	32.6	29.0	24.1	17.0

2. EXPERIMENT FACILITIES IN THE HFIR CORE

The original mission of HFIR was the production of trans-plutonium isotopes. Fortunately, the original designers included many other experiment facilities. The following sections describe each facility in enough detail to provide guidance for researchers[5].

[4] Visit http://neutrons.ornl.gov/facilities/HFIR/data to download the most current simulation MCNP data.
[5] For additional information and contact information please visit http://neutrons.ornl.gov/facilities/HFIR/status.shtml.

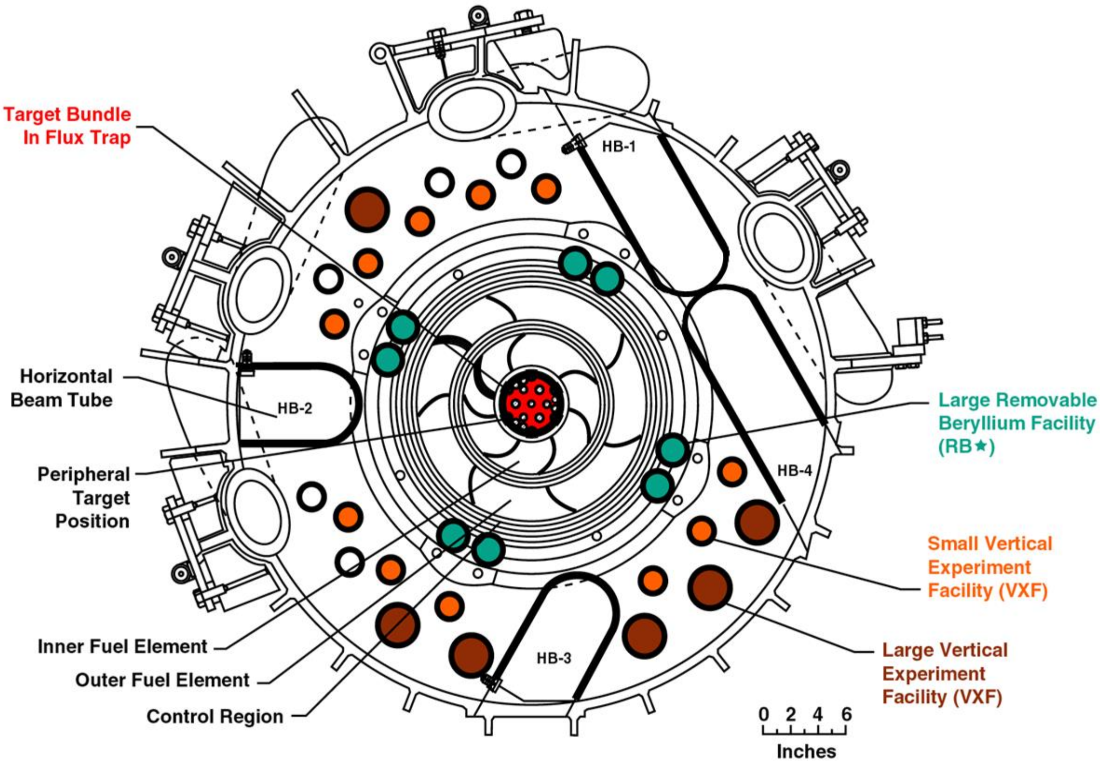


FIG. 7. Experiment positions at HFIR.

TABLE 3. IN-CORE EXPERIMENT FACILITIES

30	*Target positions* in the flux trap (two of these positions can accommodate instrumented experiments)
1	*Hydraulic tube* irradiation facility, located in the very high flux region of the flux trap, which allows for insertion and removal of samples while the reactor is operating
6	*Peripheral target positions* located at the outer edge of the flux trap
21	*Vertical irradiation facilities* of various sizes located throughout the beryllium reflector
2	*Pneumatic tube facilities* in the beryllium reflector, which allow for insertion and removal of samples while the reactor is operating for neutron activation analysis
2	*Slant access facilities*, called ‘engineering facilities’, located on the outer edge of the beryllium reflector

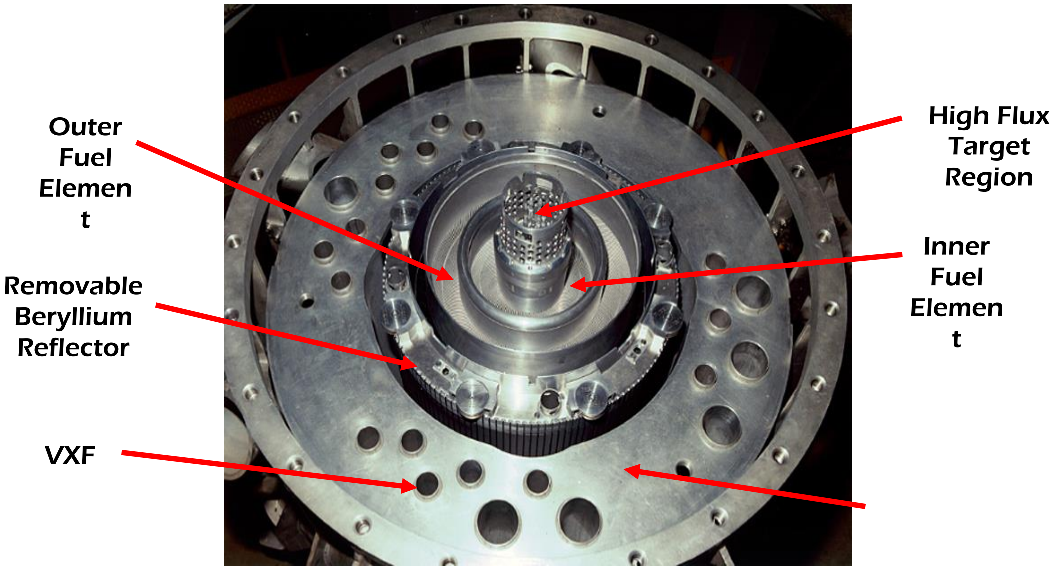


FIG. 8. Photo of the HFIR core.

2.1. FLUX TRAP (TARGET REGION)

The flux trap consists of 31 target positions. These positions were originally designed to be occupied by target rods used for the production of trans-plutonium elements; however, other experiments can be irradiated in any of these positions. We have developed target capsule designs that can be used in numerous areas in the reactor experiment facilities. The use of these types of capsules simplifies fabrication, shipping, and post-irradiation processing, which translates to a cost savings for experimenters.

Target irradiation capsules of each type must be designed such that they can be adequately cooled by the water flow available outside the target-rod shrouds. Excessive neutron poison loads in experiments in target positions are discouraged because of their adverse effects on both trans-plutonium isotope production rates and fuel cycle length. Such experiments require careful coordination to ensure minimal effects on adjacent experiments, fuel cycle length, and neutron scattering beam brightness. Two positions are available for instrumented target experiments: positions E3 and E6.

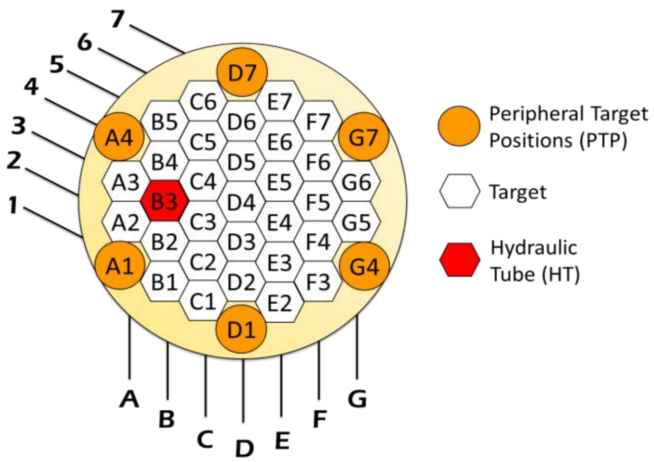


FIG. 9. Target region irradiation positions.

2.2. HYDRAULIC TUBE FACILITY

The HFIR hydraulic tube (HT) facility provides the ability to irradiate materials for durations less than the standard ~23-day HFIR fuel cycle, which is ideal for the production of short half-life medical isotopes that require retrieval on-demand. The system consists of the necessary piping, valves, and instrumentation to shuttle aluminum capsules (called *rabbits*) between the capsule loading station and the reactor.

Normally, the heat flux from neutron and gamma heating at the surface of the rabbit is limited to 74 000 Btu/h-ft² (2.3×10^5 W/m²). Furthermore, the neutron poison content of the facility load is limited such that the reactor cannot be tripped by a significant reactivity change upon insertion or removal of the rabbits.

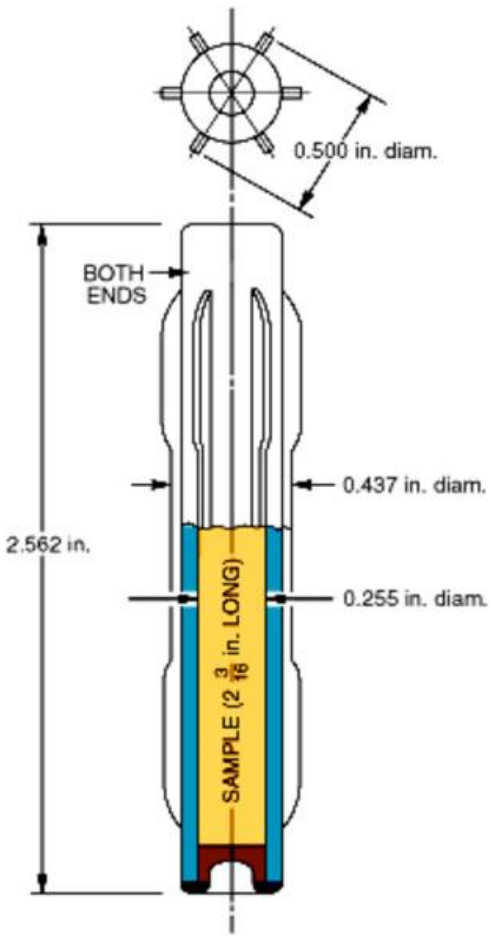


FIG. 11. Typical hydraulic rabbit capsule.

2.3. PERIPHERAL TARGET POSITIONS

HFIR provides six peripheral target positions (PTPs). Fast-neutron fluxes in these positions are the highest available to experiments in the reactor, though there is a steep radial gradient in the thermal-neutron flux at this location.

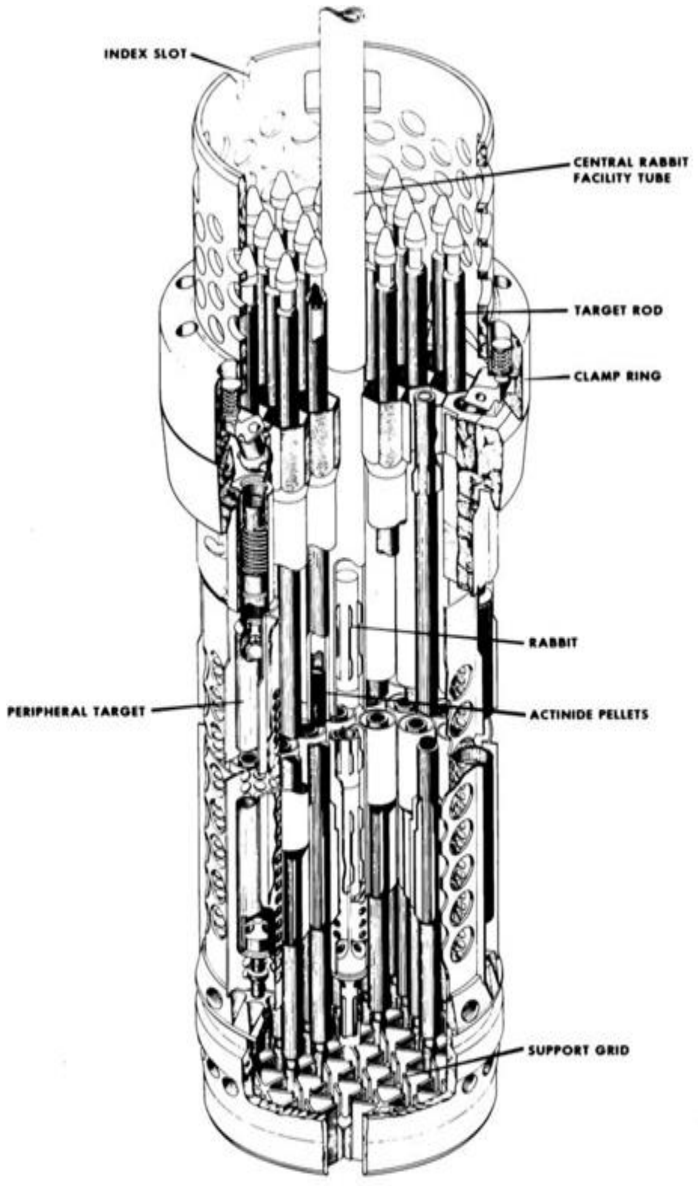


FIG. 10. Target basket in the HFIR flux trap.

Like the target positions, a style of PTP capsule is available to house smaller isotope or materials capsules, which are similar to the hydraulic facility capsules. The use of this type of irradiation capsule simplifies fabrication, shipping, and post-irradiation processing, translating into cost savings for the experimenter.

PTP irradiation capsules of each type must be designed such that they can be adequately cooled by the available coolant flow. Typical experiments contain a neutron poison load equivalent to that associated with 200 g of aluminium and 35 g of stainless steel distributed uniformly over a 20 in. (50.8 cm) length. PTP experiments containing neutron poison loads in excess of this are discouraged because of their adverse effects on isotope production rates, fuel cycle length, and fuel element power distribution.

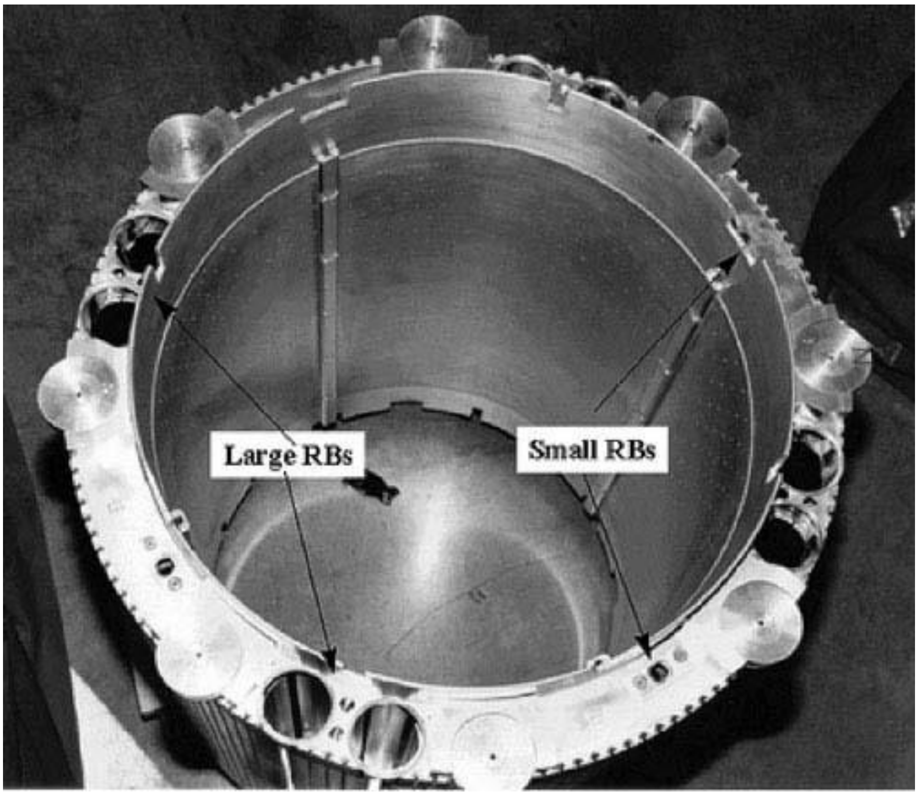


FIG. 12. Removable beryllium experiment locations.

2.4. LARGE REMOVABLE BERYLLIUM FACILITIES

Eight large-diameter irradiation positions are located in the removable beryllium (RB) near the control region and are generally referred to as the RB* positions. These facilities are designed for either instrumented or non-instrumented experiments. The instrumented capsule design can also employ sweep or cooling gases as necessary. Instrument leads and access tubes are accommodated through penetrations in the upper shroud flange and through special penetrations in the pressure vessel hatch.

When not in use, these facilities contain beryllium or aluminium plugs. Because of their close proximity to the fuel, RB* experiments are carefully reviewed with respect to their neutron poison content, which is limited because of its effect on fuel element power distribution and fuel cycle length.

These positions can accommodate shielded and spectral tailored experiments, making them well suited for fusion materials irradiation.

Uses for the RB* facilities have included the production of radioisotopes; High temperature gas-cooled reactor (HTGR) fuel irradiations; and the irradiation of candidate fusion reactor materials. The later type of experiment requires a fast neutron flux. For this application the capsules are placed in a liner containing a thermal neutron poison for spectral-tailoring. These experiments are carefully reviewed with respect to their neutron poison content, and limited to certain positions to minimize their effect on adjacent neutron scattering beam tubes.

2.5. SMALL REMOVABLE BERYLLIUM FACILITIES

Four small-diameter irradiation positions are located in the removable beryllium (RB) near the control region. The small RB positions do not have an aluminium liner like the RB* facilities. When not in use, these positions contain beryllium plugs.

These facilities have been used primarily for the production of radioisotopes. The neutron poison content limits and the available pressure drop requirements for experiments in these facilities are the same as in the RB* facilities previously discussed.

2.6. CONTROL ROD ACCESS PLUG EXPERIMENT FACILITIES

The control rod access plugs (CRAP) consist of four separate beryllium pieces that are typically removed in order to access the control plate mounting hardware. These beryllium pieces currently provide two irradiation holes each; however alternate designs have been used that include additional holes for added capacity. The proximity of these positions to the fuel make them good candidates for materials or isotope irradiations, and at the same time the capacity and location of experiments in these positions have less impact on cycle length than the removable beryllium positions.

2.7. SMALL VERTICAL EXPERIMENT FACILITIES (ISVXV AND OSVXF)

Sixteen irradiation positions located in the permanent reflector are referred to as the small vertical experiment facilities. These are further designated as the Inner small and the Outer small vertical experiment facilities (ISVXF and OSVXF). Each of these facilities has a permanent aluminium liner, located concentric within the core. Normally, non-instrumented experiments are irradiated in these facilities. VXF-7 is dedicated to one of the pneumatic irradiation facilities, which supports the Neutron Activation Analysis Laboratory (NAA) and is unavailable for other use.

A pressure drop of approximately 100 psi (6.89×10^5 Pa) at full system flow is available to provide primary system coolant flow for cooling experiments. When not in use, these facilities may contain a beryllium or aluminium plug or a flow-regulating orifice and no plug.

Large neutron poison loads in these facilities are typically acceptable and are of no particular concern with respect to fuel element power distribution perturbations or effects on fuel cycle length because of their distance from the core. However, experiments are still carefully reviewed with respect to their neutron poison content, which is limited to minimize their effect on adjacent neutron scattering beam tubes.

2.8. LARGE VERTICAL EXPERIMENT FACILITIES (LVXF)

Six irradiation positions located towards the outer edge of the permanent reflector are referred to as the large vertical experiment facilities (LVXF). These facilities are similar in all respects (as to characteristics and capabilities) to the small vertical experiment facilities described in the preceding section except for location and size. When not in use, these facilities contain beryllium or aluminium plugs.

2.9. SLANT ENGINEERING FACILITIES

Provisions have been made for installation of up to two engineering facilities to provide additional positions for experiments. These facilities consist of tubes that are inclined upward 49° from horizontal. The inner ends of the tubes terminate at the outer periphery of the beryllium. The upper ends of the tubes terminate at the outer face of the pool wall in an experiment room one floor above the main beam room. One of the engineering facilities houses the PT-2 pneumatic tube, which is used in the Neutron Activation Analysis (NAA) Laboratory. The other available tube is available to provide access to this highly thermalized region of the core.

2.10. MATERIALS IRRADIATION FACILITY (MIF)

The HFIR materials irradiation facility (MIF) is one of only two such facilities in the USA that can host fully instrumented and monitored experiments in a high neutron flux environment. The MIF facility is located in the experiment room one floor above the reactor core, and provides communication with experiments located in the target region, RB*, and VXF positions.

Interest in monitored and controlled materials irradiations has grown significantly in recent years. This growth is driven in the USA by several programmes including the Light Water Reactor Sustainability (LWRS) programme that is directed towards maintaining and improving the safety record of the existing fleet of reactors, the Advanced Reactor Concepts (ARC) programme that focuses primarily on closing the technology gap that is essential for the next generation of nuclear reactors, the Accident-Tolerant Fuels Initiative designed to develop improved fuels and clad materials, and the Used Fuel Disposition programme that is interested in examining materials characteristics in spent fuel. These are in addition to other US Department of Energy and international Fusion Energy programmes as well as private companies that are working to develop radiation tolerant materials.

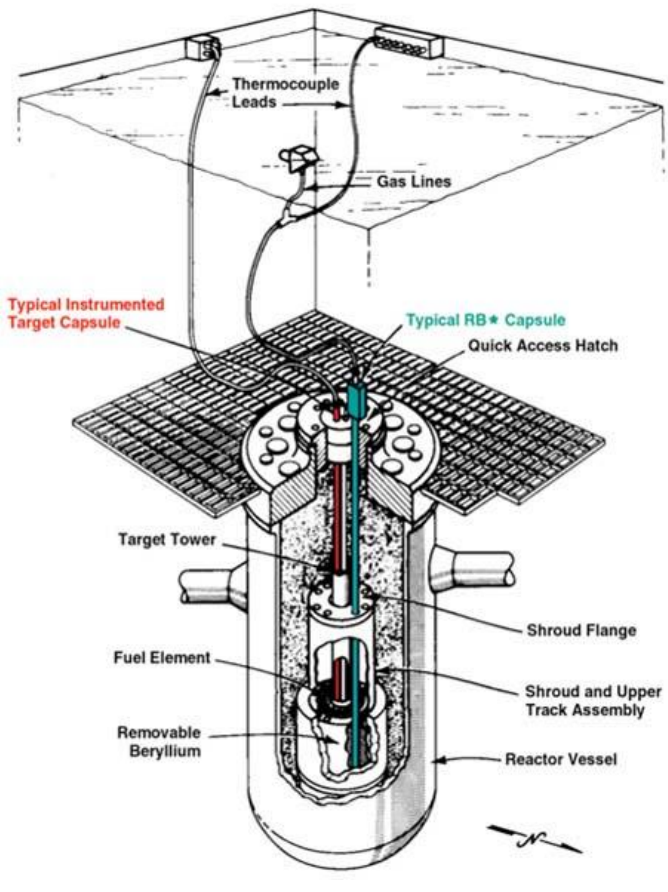


FIG. 12. Diagram of the HFIR materials irradiation facility (MIF).

The MIF offers the unique advantage that the researcher can monitor and acquire in-situ data, and change capsule operating conditions real-time during the irradiation. This enables tight control of desired experimental conditions so that they match the design specifications throughout the irradiation. Meanwhile, the experiment boundary conditions, i.e. temperature and pressure, can be altered during the test enabling significant flexibility on conditions and ability to collect considerably more data from a single experimental capsule. Additionally, this capability can allow measurement of a variety of material properties (conductivity, fission product composition, etc.) that otherwise would only be available upon post irradiation examination where non-equilibrium conditions brought upon by irradiation do not exist.

Additionally, the MIF offers control options; the researchers can introduce transients that might be of interest while the material is being irradiated, and counter reactor-based changes that might occur without control. This level of monitoring and control is not possible with other common material irradiation targets in the reactor (standard targets rods, capsules, etc.).

The MIF design allows for instruments to perform measurements in the off-gas, such as fission product identification to investigate fuel failures and monitoring low-concentration gas constituents by mass ratio to understand hydrogen isotope uptake and release mechanisms in next generation fusion reactor plasma-facing materials. The flexible reconfiguration of MIF allows for the addition of gamma and mass spectrometers for future research. Additionally, the ability to feed deuterium in the sweep gas is beneficial for fusion materials qualification experiments.

3. ORNL POST IRRADIATION EXAMINATION (PIE) FACILITIES

ORNL offers many facilities for post-irradiation examination and testing of materials. Three of these facilities are described below.

3.1. IRRADIATED MATERIAL EXAMINATION AND TESTING (IMET) FACILITY

The Irradiated material examination and testing (IMET) facility was designed and built as a hot cell facility. It is a two-story block and brick structure with a two-story high bay that houses six heavily shielded cells and an array of sixty shielded storage wells. It includes the Specimen Preparation Lab (SPL) with its associated laboratory hood and glove boxes, an operating area, where the control and monitoring instruments supporting the in-cell test equipment are staged, a utility corridor, a hot equipment storage area, and a tank vault room. Tests and examinations are conducted in six examination ‘hot’ cells and/or in a laboratory hood or modified glove boxes in the SPL.

FIG. 13. Cell bank at the irradiated material examination and testing facility (IMET).

In-cell equipment and capabilities:

- Sample sorting and identification;
- Sample machining using a CNC milling machine and diamond saws;
- Furnace annealing;
- Automated welding;
- Ultrasonic cleaning;
- High-temperature, high-vacuum testing;
- Tensile testing with high-vacuum chamber option;
- Impact testing, fatigue and fracture toughness testing of standard and subsize impact specimens;
- Automated micro-hardness testing;
- Profilometry;
- Scanning Electron Microscopy (Philips XL30.

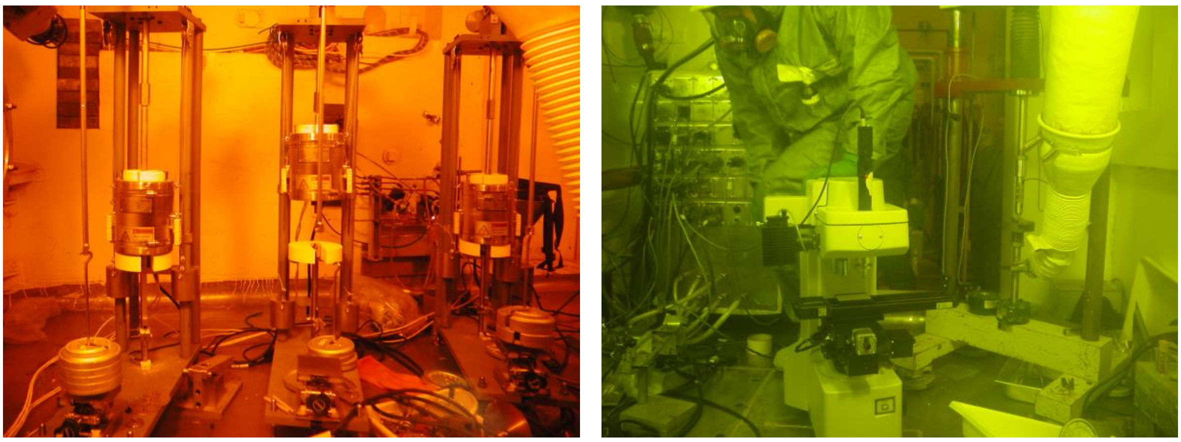


FIG. 14. In-cell creep testing (left) and in-cell hardness testing (right).

3.2. IRRADIATED FUELS EXAMINATION LABORATORY (IFEL)

The Irradiated fuels examination laboratory (IFEL) was designed and constructed to permit the safe handling of increasing levels of radiation in the chemical, physical, and metallurgical examination of nuclear reactor fuel elements and reactor parts. The IFEL is a two-story brick building with a partial basement.

FIG. 15. Irradiated fuels examination laboratory (IFEL).

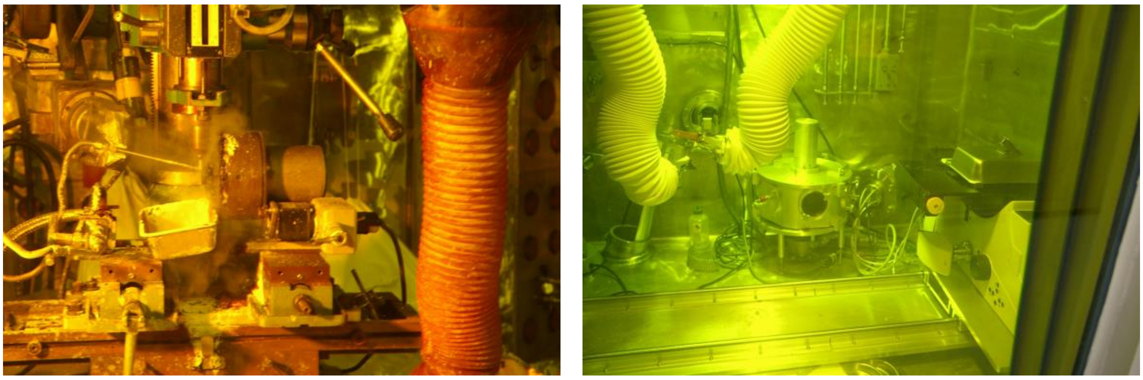


FIG. 16. Fuel targets being disassembled (left) and SEM cubical (right) in IFEL.

Equipment and capabilities:

- Full-length LWR fuel examination;
- Repackaging of spent fuel;
- Metrology, metallography, grinding/polishing, optical and electron microscopy, gamma spectrometry;
- Fission gas sampling and analysis;
- Thermal imaging;
- SEM/microprobe;
- Microsphere gamma analyser for individual fuel particle analysis.

3.3. LOW ACTIVATION MATERIALS DEVELOPMENT AND ANALYSIS LABORATORY (LAMDA)

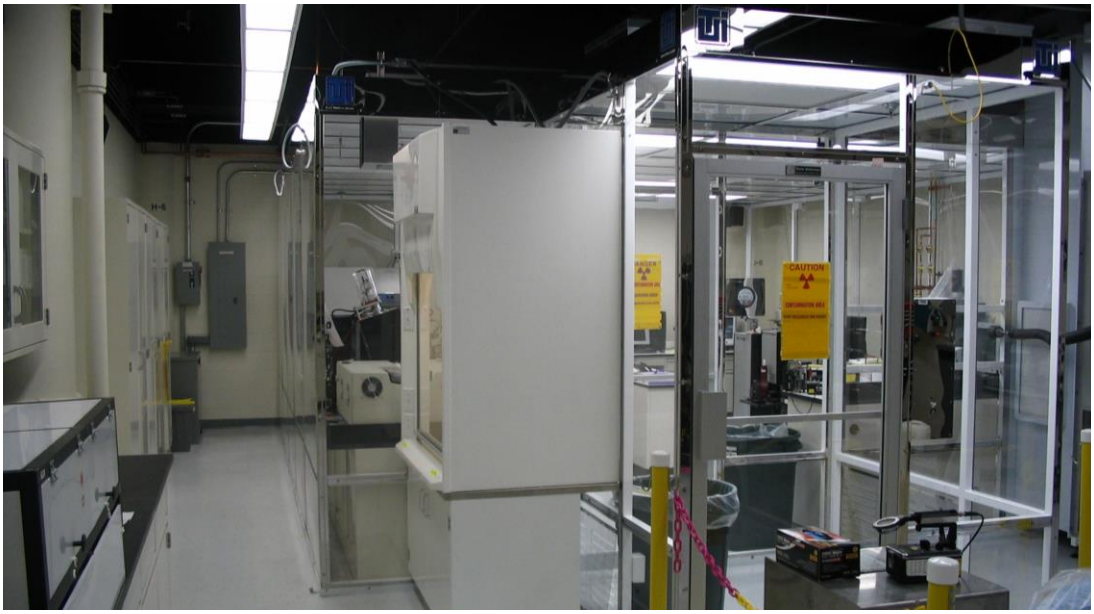


FIG. 17. Low activation materials development and analysis laboratory (LAMDA).

LAMDA is specifically designed for working with low-activity samples. Low-activity specimens include materials that do not activate significantly under irradiation (such as composites or reactor pressure vessel steels) or higher activity specimens that are either smaller in size or have decayed to lower levels. Samples are routinely received in LAMDA by way of the ORNL hot cell facilities where irradiation capsules are dismantled or higher activity packages are sorted. LAMDA can also receive specimens directly from facilities around the world if the activity is low enough.

Research tools encompass mechanical testing for tensile, compression, bend-bar, and fatigue tests in a variety of environments. Physical properties such as density can be measured on a variety of specimens while thermal and electrical properties are also characterized over a range of materials and temperatures. LAMDA also frequently performs sample polishing, grinding, cutting and other metallographic processes.

Equipment and capabilities:

- *Mechanical properties testing*: a versatile MTS system is available for testing graphite, SiC, and composite materials. This system has a 500 lb ultra-fine-thread frame used for testing in air. Lab view data acquisition provides for custom testing applications. Laser extensometry and pneumatic sample grips are also available.
- *Thermal diffusivity*: three Anter Flash thermal diffusivity systems are available. A four sample stage allows for rapid data acquisition. Data can be taken at temperatures ranging from liquid nitrogen up to 1000°C with solid state and infrared detectors.
- *Dilatometer*: an Anter Workhorse Dilatometer can operate from room temperature up to 1000ºC.
- *Elastic modulus*: Grindosonic system is available for measuring elastic modulus.
- *Calorimetry*: TA Instruments scanning calorimeter is capable of measurements from room temperature up to 1000ºC.

— *Four-point probe electrical resistivity*: the measurement of electrical resistivity allows for non-destructive evaluation of changes in microstructure as well as information on radiation-induced changes on electrical and thermal properties.
— *Density measurement*: density can be accurately measured using density gradient columns for densities up to 3.2 g/cm^3 with accuracy to 1 mg/cm^3 (0.1% error). Immersion density can be used on micro tensile or other specimens larger than three mm diameter with accuracy to < 0.4% error.

Other instrumentation and capabilities
— *Metallography*: slicing and polishing facilities are available for preparing ceramic and metallic samples of various sizes. Disc punches are also used to create transmission electron microscopy specimens.
— *Sample preparation*: a Ficione ion miller with a liquid nitrogen stage is available for TEM preparation. Sample slicing, grinding, and polishing operations are also performed.
— *Hitachi SEM*.
— *Laser profilometer*.

Profile 29

MIT NUCLEAR REACTOR LABORATORY

Massachusetts Institute of Technology Reactor (MITR)

UNITED STATES OF AMERICA

1. MIT NUCLEAR REACTOR LABORATORY (MIT-NRL) MISSION AND GOALS

The MIT Nuclear Reactor Laboratory (MIT-NRL) is an interdepartmental laboratory that conducts interdisciplinary research in radiation and nuclear energy applications. MIT-NRL operates a 6 MW research reactor, the MIT Research Reactor (MITR), which is the second largest university research reactor in the USA. Our mission is to provide faculty and students from MIT and other institutions with both a state-of-the-art neutron source and the infrastructure required to facilitate use of the reactor.

MITR is a preeminent university centre for research and education in the areas of materials and fuel research for advanced nuclear energy systems. The MITR is a partner reactor facility of the ATR National Scientific User Facility (ATR NSUF) for advanced materials and fuel research and development to facilitate collaboration among national labs, universities, and industry.

1.1. EDUCATIONAL ACTIVITIES AND PUBLIC OUTREACH

The MITR has the advantage of being affiliated with the very highly regarded Nuclear Science and Engineering Department (NSED) at MIT. The NSED and other departments use the MITR for both thesis research projects and laboratory exercises. Students are particularly enthusiastic about experimental work on the MITR because it gives them the opportunity to apply their academic learning to challenging engineering and scientific problems. Also, they acquire the skills needed to coordinate projects and are imbued with the 'safety culture' needed for the proper operation of nuclear facilities. Training, research, and educational opportunities at the MITR are offered through:

(1) MIT undergraduate and graduate-level laboratory courses;
(2) Research projects as part of the Undergraduate Research Opportunity Programme (UROP);
(3) Student reactor operator training programme;
(4) Senior and graduate thesis research projects.

The MIT-NRL also maintains a successful public outreach programme. The educational and outreach activities include:

- Tours of the MITR by the general public and high school students (about 1000-1500 tours per year);
- Assistance to students with science fair projects involving photon irradiations.

2. MITR GENERAL DESCRIPTION

The MITR is a tank-type research reactor. It is owned and operated by the Massachusetts Institute of Technology, a non-profit educational institution, and is licensed by the US Nuclear Regulatory Commission. Its current license, issued in November 2012, authorizes steady-state 6 MW operation for 20 years. As shown in Fig. 3, the reactor has two tanks: an inner one for the light water coolant/moderator and an outer one for the heavy water reflector. A graphite reflector surrounds the heavy water tank.

The reactor utilizes flat, plate-type fuel elements. Each rhomboidal fuel element consists of fifteen plates of UAl_x cermet clad with 6061 aluminium alloy. Longitudinal fins on the fuel plates increase the heat transfer area. The core has 27 fuel element positions and is normally configured with 24 fuel elements and 3 positions available for in-core experiments. The close-packed hexagonal core design maximizes the thermal neutron flux in the heavy water reflector region where the re-entrant thimbles of the beam ports are located. The light-water core, heavy-water reflector, and graphite region are all separately cooled; each transfers heat to a secondary coolant that dissipates it to the atmosphere via two cooling towers.

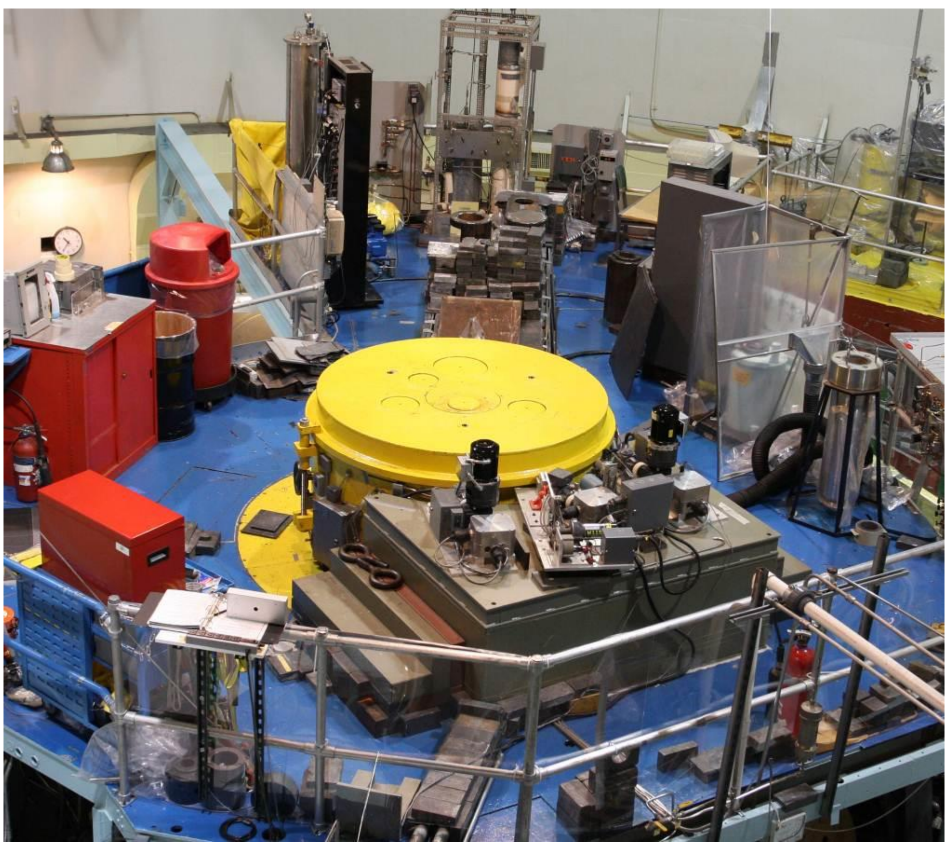


FIG. 1. The MITR reactor top.

The MITR operates at atmospheric pressure. Primary coolant, at a nominal flow rate of 2000 gpm (~125 kg/s) enters the bottom of the core tank through the core shroud, flows upward through the fuel elements and then exits at the outlet piping about 2 m above the top of the core. The primary coolant core inlet temperature is approximately 42°C and outlet temperature is about 50°C. The hexagonal core structure is about 38 cm across with an active fuel length of about 56 cm. The compact core has an average power density of about 84 kW/l, with fast, thermal, and gamma fluxes similar to those of a commercial light water power reactor (LWR).

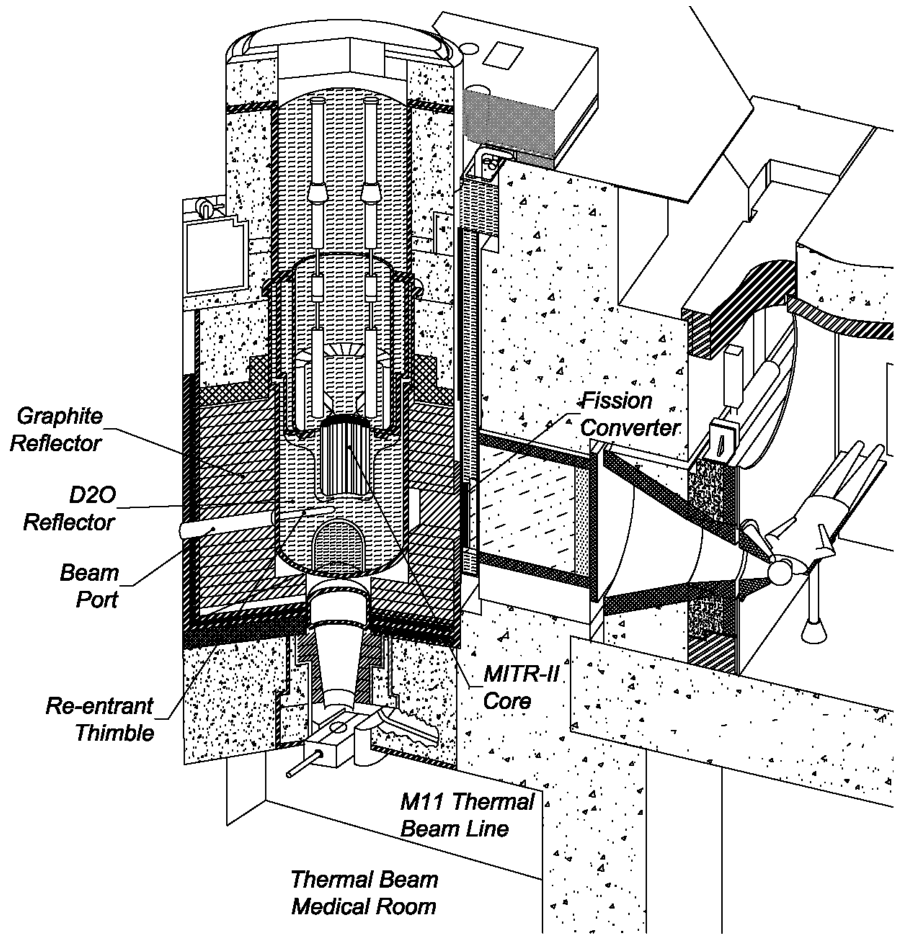


FIG. 2. Cutaway schematic of the MITR.

The reactor design includes a number of passive safety features. The principal ones are negative reactivity temperature coefficients of the fuel and moderator, a negative void coefficient of reactivity, the use of anti-siphon valves to isolate the core tank from the effect of breaks in the coolant piping, a core-tank design that promotes natural circulation, and the presence of a full containment. During a loss-of-primary flow transient, decay heat is removed by natural circulation within the core tank through the natural circulation and anti-siphon valves and no forced cooling is required. The anti-siphon system prevents uncovering of the core in the event of a loss of coolant accident.

The MITR is equipped with a wide variety of sample irradiation facilities, with fast and thermal neutron fluxes up to 1.2×10^{14} $n{\cdot}cm^{-2}{\cdot}s^{-1}$ and 5×10^{13} $n{\cdot}cm^{-2}{\cdot}s^{-1}$, respectively; these facilities are described in the following section. The MITR operates 24 hours a day, seven days per week. A typical fuel cycle lasts up to nine weeks followed by a two-week refuelling and maintenance outage.

3. MITR IRRADIATION POSITIONS AND EXPERIMENTAL FACILITIES

3.1. OVERVIEW OF EXPERIMENTAL FACILITIES

The irradiation positions at the MITR include three in-core positions, radial and vertical beam ports, shielded medical rooms, and pneumatic tubes. In-core irradiation facilities are not permanently installed and can be designed to tailor specific requirements. Table 1 gives flux and sample dimension information for the irradiation positions.

TABLE 1. REPRESENTATIVE DIMENSIONS AND FLUX LEVELS IN MITR IRRADIATION FACILITIES

Facility	Size	Neutron flux ($n{\cdot}cm^{-2}{\cdot}s^{-1}$)
In-core	3 available Max in-core volume ~ 1.8” ID × 24” long (4.572 cm × 60.96 cm)	Thermal: 3.6×10^{13} Fast: up to 1.2×10^{14} (E >0.1 MeV)
Beam ports	Various radial: 4” to 12” ID (10.16 cm to 30.48 cm)	Thermal: $1 \times 10^{10} – 1 \times 10^{13}$ (source)
Vertical irradiation position	2 vertical (3GV) 3” ID × 24” long (7.62 cm × 60.96 cm)	Thermal: $4 \times 10^{12} – 1 \times 10^{13}$
Through ports	One 4” (10.16 cm) port (4TH) One 6” (15.24 cm) port (6TH)	Average thermal: 2.5×10^{12} to 5.5×10^{12}
Pneumatic tubes	One 1” (2.54 cm) ID tube* (1PH1)	Thermal: up to 8×10^{12}
	One 2” (5.08 cm) ID tube* (2PH1)	Thermal: up to 5×10^{13}
Fission converter beam facility (FCB)	Beam aperture ~ 6” (15.24 cm) ID	Epithermal: ~ 5×10^{9}
Thermal beam facility (TNB)	Beam aperture ~ 6” (15.24 cm) ID	Thermal: up to 1×10^{10}

* Dimensions of the sample holders used in the pneumatic irradiation facilities are 1” (2.54 cm) diameter by 3-1/4” (7.62 cm–6.35 mm) length for the 1PH1 location, and 1-3/8” (2.54 cm–1.06 mm) diameter by 6-1/4” (15.24 cm–6.35 mm) length for the 2PH1. Images of these holders are shown in Section 3.3.

3.2. IN-CORE EXPERIMENTAL FACILITIES

The MITR core is shown in Fig. 3. An in-core experimental facility at the MITR is installed by removing a solid dummy fuel element and replacing it with a dummy element that accommodates the desired in-core loop configuration. In some cases the dummy fuel element is integral with the in-core experimental facility structure. Note that these facilities are not permanently installed and are removed from the core tank when they are not in use. This means that a given facility can generally be tailored to meet the requirements of a particular experiment or multiple experiments/specimens.

There are two general types of in-core experimental facilities, illustrated schematically in the 3-D model of the reactor shown in Fig. 4. The first, designated ‘In-core sample assembly’ (ICSA), uses an S-bend tube to give access to the in-core irradiation space from the top of the reactor core tank. This facility is generally used for sample irradiations in an inert gas atmosphere with limited requirements for in-core instrumentation (typically thermocouples only). The ICSA is cooled by the reactor coolant, but sample capsules can be insulated to take advantage of nuclear heating for elevated temperature exposures. Active heating or cooling is also potentially available in an ICSA subject to design and safety review. Figure 5 shows the neutron spectra comparison of an ICSA and a typical PWR core.

The second type of in-core facility encompasses a variety of more complex irradiation rigs that are expressly designed for a specific purpose. Past examples include LWR loops used to study various aspects of coolant chemistry, passively and actively loaded mechanical tests under LWR conditions, corrosion test loops for advanced clad materials, a test of internally and externally cooled annular fuel, an irradiation test for high temperature gas reactor materials at temperatures approximately 1000-1600°C, and irradiation of multiple uranium-zirconium hydride fuel rods at PWR temperatures. These experiments are described in more detail in Section 4 to illustrate the types of in-core facilities that can be used and the design envelope that has been previously approved and demonstrated. In some cases, similar experiments can utilize existing irradiation rigs. The modular design and small size of these facilities, however, makes it possible to design and construct new facilities for specific purposes at moderate cost.

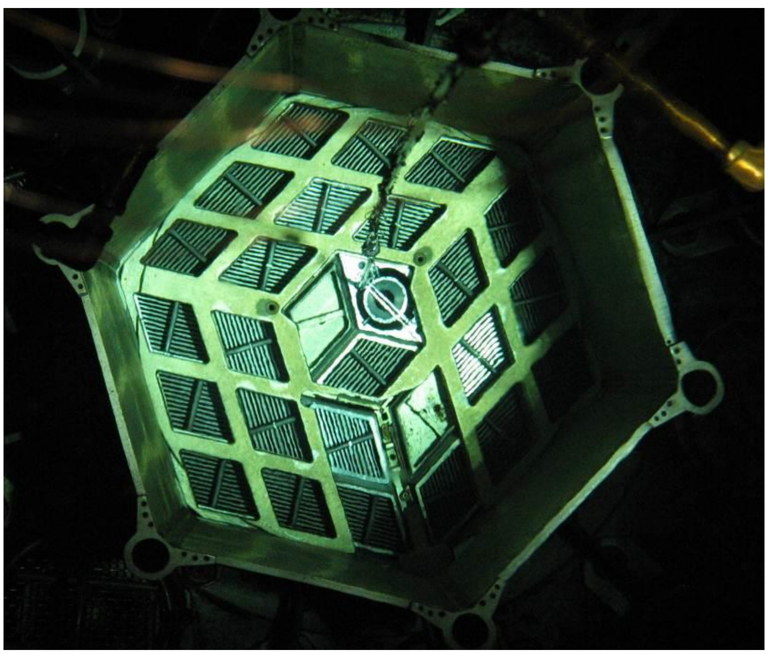


FIG. 3. *Photograph of the MITR core with two solid dummy fuel elements and in-core experimental facility installed.*

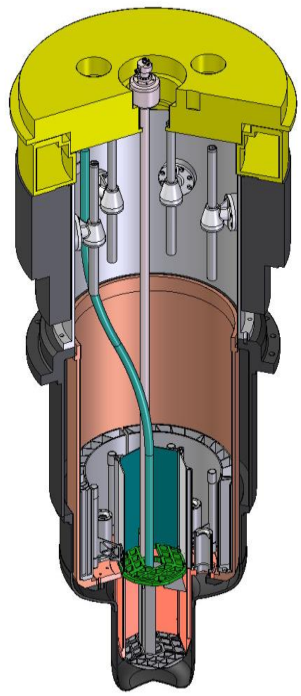


FIG. 4. Cutaway 3-D model of the MITR showing an ICSA (blue) and an in-core water loop installed in the reactor core (fuel elements removed for clarity).

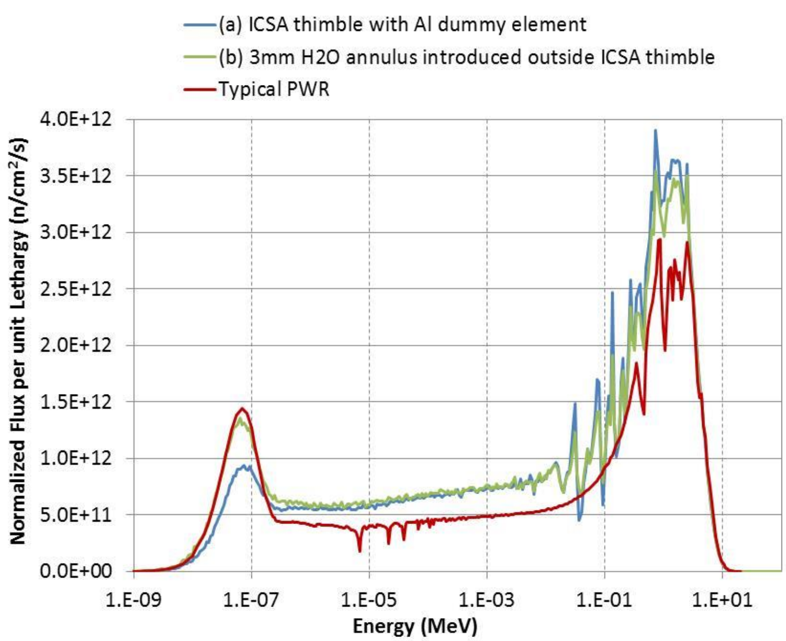


FIG. 5. Comparison of the neutron spectra for:
(a) — an MITR In-Core Sample Assembly (ICSA) with a 0.3 cm water annulus;
(b) — a MITR ICSA with an aluminium annulus; and

(c) — a typical PWR core.

Note: Calculations were performed using MCNP with a three-dimensional MITR model at 6 MW. The MITR in-core neutron spectrum and flux magnitudes are comparable to those of a commercial PWR.

A variety of in-core and support instrumentation and measurement can be provided to support in-core loop irradiation. Thermocouple temperature measurement is the most common in-core measurement. Other in-core instrumentation that has been used includes electrochemical corrosion potential and other electrode measurements and DC potential drop strain and crack growth measurement. A variety of data acquisition equipment is available if experimenters desire to use or test their own in-core instrumentation. A wide variety of parameters are routinely monitored and recorded at out-of-core locations for the in-core facilities. For water loops these include temperature, pressure, flow, dissolved hydrogen and oxygen, and conductivity. Real-time residual gas analysis (mass spectrometry) is also available. Radiochemical and chemical assays can be performed on site on batch-sampled radioactive coolant samples. For chemical assays, ICP-OES, INAA and prompt gamma NAA are available. MITR staff will support use of specialized instrumentation required for a particular experiment, subject to funding and manpower constraints

3.3. PNEUMATIC TUBE TRANSFER SYSTEM

The MITR is equipped with two pneumatically-operated irradiation facilities that allow materials to be exposed to high or intermediate-level neutron fluxes. A 2” pneumatic facility (2PH1) offers a high thermal flux of up to 5×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$ with a significant fast neutron flux (cadmium ratio of about 20). A 1” pneumatic facility (1PH1) offers an intermediate, highly thermal flux of up to 8×10^{12} $n \cdot cm^{-2} \cdot s^{-1}$ (cadmium ratio of about 200). These facilities are normally used to support MIT-NRL’s Neutron Activation Analysis programme and for isotope and radiotracer production.

Materials are transferred through the pneumatic facilities in sample holders known as ‘rabbits’. The 1PH1 system uses a polyethylene rabbit with internal dimensions of 1” diameter by 3-1/4” length; the 2PH1 system uses either polyethylene or titanium rabbits with internal dimensions of 1-3/8” diameter by 6-1/4” length. Titanium rabbits are generally only used for 2PH1 irradiations longer than 10 hours. The 1PH1 pneumatic system is capable of ejecting the sample to a hot cell within the reactor containment or to a laboratory in an adjacent building. This allows materials to be irradiated for short periods (typically from 10 seconds to 10 minutes) and examined within minutes.

4. IN-CORE EXPERIMENTS

This section provide a brief review of the in-core experiments that have been carried out at the MITR to the outline the in-core capabilities that can support fuel, materials, water chemistry, and high temperature instrumentation testing. Although future experiments are not limited to the types of experiments that have been previously performed, these examples should serve to illustrate more fully the nature and scope of in-core research that can be contemplated. Furthermore, experiments that fall in or near the envelope of conditions previously investigated will benefit from the established techniques, safety analyses and, in some cases, existing equipment that can be re-used or adapted.

Many of the in-core experiments at the MITR have been used for studies of commercial light water power reactor (LWR) technology. These include facilities to study: radioactive corrosion product transport in PWRs, specifically the effect of pH and zinc injection; BWR normal and hydrogen water chemistry and techniques to reduce radioactive nitrogen carryover into the steam phase; irradiation assisted stress corrosion cracking (IASCC) using actively loaded constant strain rate and constant load tests and passively loaded crack growth monitoring specimens; corrosion and mechanical property response of candidate ceramic fuel clad materials to LWR coolant and irradiation environments; dependence of shadow corrosion on base and counter materials, gap size and irradiation environment; the feasibility of developing internally and externally cooled annular fuel (IXAF) in order to increase the power density of PWRs while maintaining or improving safety margins, and real-time measurement of the thermal conductivity of liquid metal bonded uranium-zirconium hydride fuel during irradiation.

In addition to the LWR facilities, a high temperature irradiation facility was demonstrated that provides irradiation space for a variety of mechanical and thermal property specimens for post-irradiation examination. These specimens were exposed in-core at temperatures up to 1600ºC. A generic in-core sample assembly (ICSA) capsule design was developed and demonstrated for several types of experiments with temperatures up to 900°C.

4.1. PRESSURIZED LWR LOOPS

4.1.1. Coolant chemistry loops

The PWR coolant chemistry loop (PCCL) and its companion BWR coolant chemistry loop (BCCL) are described in the following sections. They were among the earliest of the advanced in-core experiments to be carried out at the MITR. These loops were operated under prototypic pressure and temperature to enable the simulation of PWR or BWR conditions.

(1) PWR coolant chemistry loop (PCCL);

The PCCL was designed as an approximately 1/3 scale model of a single flow channel in a PWR. Its components included a core section of Zircaloy-4 tubing representative of the flow channel between fuel pins in a PWR core, a steam generator section of Inconel tubing representative of a single steam generator tube, and connecting sections representative of the recirculation piping. PWR coolant was circulated in the loop at representative velocities by a seal-less, magnetic drive pump. In-core heating was provided by an electric resistance heater coupled to the loop tubing by a lead bath that was predominantly liquid at operating temperature. Full operating temperature and pressure for the loop were maintained independent of the reactor power and the loop was designed to run for periods of up to 3 months at constant conditions.

An important feature of the loop was that the bulk of the tubing was replaced for each run. This facilitated the performance of a series of closely controlled experiments where only the pH of the coolant was changed from run-to-run. The initial conditions for each run were virtually identical; there was no need to decontaminate the tubing between runs and the tubing itself was available for analysis after each run. Coolant chemistry was maintained at a constant boric acid and lithium hydroxide level for each run, with feed and bleed to the loop set at a rate proportional to the total loop volume. Inlet dissolved hydrogen levels were controlled and monitored as was the inlet dissolved oxygen content of the coolant. In addition to study pH effects on radioactive corrosion product transport, a series of longer runs were undertaken to investigate the possibility of reducing transition metal transport in PWR coolant using zinc injection.

(2) BWR coolant chemistry loop (BCCL);

The BCCL was also designed as an approximately 1/3 scale model of a single flow channel in a BWR. In this case, the parameters that govern BWR coolant radiolysis and the associated parameters of electrochemical corrosion potential (ECP) and nitrogen chemistry were simulated as closely as possible in the in-core loop. The loop operated at the same temperature and pressure as BWR core with capability for outlet quality up to 15%. Operating in single phase mode the loop could also be used to simulate bypass region conditions.

For the BCCL, makeup and let-down systems were designed to provide heated feed water at a variety of conditions. Feed water gas content (H_2 and O_2) was independently controllable and other dissolved species could be added through a separate injection system. The core exit stream was separated into steam and water phases that were independently assayed for N^{16} content using on-line gamma monitors. ECP was monitored in the steam separator plenum and in the liquid phase let-down line. A cooled sampling system was installed on the loop to allow core exit hydrogen peroxide concentration to be measured. Hydrogen peroxide decomposes rapidly by thermal processes at BWR temperatures in the bulk coolant and on sample line walls. The possibility of close access to the loop during reactor operation and the use of the sample cooler combined to provide a unique capability for accurate hydrogen peroxide measurements.

Several sets of BCCL experiments were carried out. The first were aimed at providing well characterized datasets to be used for benchmarking radiolysis codes. Many codes were semi-empirical in nature and had been optimized to predict parameters measured in operating plants. The availability of a test bed that was better characterized and more controllable than a commercial plant was of significant value in the improvement of the codes. Another set of experiments studied the possibility of achieving the benefits of hydrogen water chemistry (HWC) in a BWR without the concomitant increase in carryover of radioactive nitrogen into the steam line. Alternate ways to produce reducing coolant environments were evaluated, as was the possibility of injecting other species together with hydrogen to achieve reductions.

4.1.2. Fuel cladding test facilities

Two major types of in-core fuel cladding test have been carried out at the MITR. The first is designed to expose cladding samples to prototypical LWR coolant and neutronic environments. This facility is referred to as the Advanced Clad Irradiation (ACI) and has been used to test alumina and silicon carbide-based fiber composites. The facility is essentially the same in both cases and therefore the more recent implementation of the ACI for SiC composite candidate clad materials is described. A separate facility used to study the phenomenon of shadow corrosion under BWR conditions is also described.

(1) Advanced clad irradiation facility (ACI);

The aluminium thimble for the ACI facility is shown installed in the MITR core tank in Fig. 4 of Section 3. Within this thimble there is a titanium autoclave serving as the main pressure boundary, and within the autoclave there is a set of internal parts to position the specimens and direct the water flow. Water flow is downward around the return tube and the flow shrouds that isolate the specimens, and then back up through the spacers and over the specimen surfaces. At the top of the specimen stack the water flow enters the return tube, which is fed through the autoclave pressure boundary at the top head of the autoclave. It is generally possible, as was done in this experiment, to place specimens in a region immediately above the core as well as directly in core. This feature has been used in several experiments to directly study the effect of neutron irradiation on the relevant phenomena, because the in-core and above-core specimens are exposed to otherwise virtually identical coolant temperatures and chemical conditions.

The ACI facility is inserted through a port in the reactor top lid. This system has the advantage that the high temperature and pressure coolant lines and thermocouples connect above the reactor top lid and are accessible for assembly and maintenance. These lines are routed to the circulating pump and heater adjacent to the reactor top. Shielding is required to reduce personnel dose, principally from the ^{16}N activity in the circulating coolant. A photograph showing the autoclave top head and connections, and the coolant lines, prior to installation of insulation and shielding is shown in Fig. 6.

The coolant conditions used for the SiC clad test in the ACI were chosen to represent mid-cycle conditions in a typical PWR. The coolant temperature was about 300 ± 2ºC at the specimen location and was controlled at that point independent of the reactor power level. In this facility, with relatively low in-core mass, the in-core heating is relatively small and the temperature change from specimen inlet to outlet is therefore also small. The coolant contained boric acid and lithium hydroxide. In addition to the parameters above, inlet and outlet coolant conductivity was monitored throughout the run and periodic batch coolant samples were obtained for assay of radiochemical content using an HPGe gamma spectroscopy system.

FIG. 6. Photographs of a sample module during assembly and the fully assembled sample stack before insertion into the autoclave.

(2) Shadow corrosion test facility;

The shadow corrosion test facility uses essentially the same out-of-core system as the ACI but has been operated under BWR coolant conditions with no lithium hydroxide or boric acid addition. Both normal water chemistry (NWC) and HWC~~)~~ are used in the loop. This facility is also similar to the ACI in that it is installed through the reactor top lid into a B-ring position at the reactor and is designed to expose a set of specimens to LWR conditions including a representative in-core irradiation environment.

The samples in this experiment are short segments of clad tubing that are exposed to 'counter' materials with a varying gap along the length of the specimen. Following irradiation, the samples are removed from the rig and the oxide thickness is evaluated using eddy current measurement techniques. By scanning circumferentially and axially, the dependence of oxide thickness on counter material type and gap thickness can be determined.

4.1.3. Irradiation assisted stress corrosion cracking studies

This section describes three facilities that were used to study various aspects of the problem of irradiation assisted stress corrosion cracking. The first of these is a unique facility that provided for slow strain rate testing of un-irradiated or pre-irradiated mechanical property test specimens in-core under BWR conditions. The second is an adaptation of this facility for a long-term constant load test of multiple specimens. The third is a facility that provided for irradiation of passively loaded crack growth sensors and accompanying ECP electrodes. These facilities are not currently available at the MITR, so only brief descriptions are given here to provide the scope of research that can be performed.

(1) Slow strain rate test (SSRT) facility;

The slow strain rate test facility used a commercially available ball-screw type test machine installed on the reactor top lid to actively load a specimen in an in-core autoclave in a B-ring position. The autoclave was operated under BWR conditions using the same out-of-pile system used for the ACI and shadow corrosion tests. Both NWC and HWC conditions were used during testing. The mechanical load on the specimen was applied through a dynamic seal at the reactor top lid and transmitted to the specimen using a pull rod and reaction tube system that extended from the reactor top to the in-core portion of the autoclave. Tests were performed at constant cross-head speed with strain rates sufficient to produce failure of the specimens in a period of 10-20 days. Both thermally sensitized (un-irradiated) specimens and specimens pre-irradiated in an inert gas environment at about 300°C to a dose of about 1 dpa were tested. Specimen change-out was performed in a reactor floor hot cell. Fractured specimens were removed from the hot cell, sectioned to reduce background activity and examined by SEM to assess intergranular fracture percentages.

(2) Constant load test facility;

The constant load test facility used the same test machine, autoclave and other major components as the SSRT facility. In this case, however, the internals were designed to accommodate a train of up to nine specimens over the active core height. All of these specimens were loaded by the test machine to the same load that was maintained for a period of months, again at BWR conditions. The objective of this test was to emphasize the effect of water chemistry on the cracking behaviour of specimens by using a long-duration test at lower loads than was the case in the SSRT. The specimen train was designed to allow continuation of testing after failure of one more specimens without the need to remove the failed specimen. The design also permitted the identification of a failed specimen at the time of failure.

(3) Crack growth sensor test;

This test was an approach to studying IASCC that was particularly aimed at demonstrating the ability of HWC to arrest the progress of a growing crack. To accomplish this, a set of double cantilever beam (DCB) crack growth monitors of a type adapted from those used in BWR core and recirculation piping tests was installed in an in-core autoclave. Electrochemical corrosion potential (ECP) electrodes were placed in the same autoclave to determine the ECP changes associated with coolant chemistry changes and the associated changes in crack growth behaviour. The DCB specimens were pre-cracked and passively loaded with direct current potential drop measurements used to assess crack growth *in situ*. After operation at NWC to establish crack growth in one or more specimens, coolant chemistry was switched to HWC while crack growth rates continued to be monitored. At the end of the irradiation period the in-core portion of the autoclave was stored in the wet storage ring in the reactor core tank to allow for decay of the highly radioactive stainless steel specimens. The specimens were then shipped off-site for post irradiation examination.

4.2. HIGH TEMPERATURE IRRADIATION FACILITY (HTIF)

The HTIF was designed, installed and operated for a period of several months in order to demonstrate the capability for this type of irradiation in the MITR. Such a facility is an important test bed for irradiation testing of materials essential to the development of high temperature gas reactors. The demonstration test achieved temperatures up to 1600°C. A variety of materials relevant to high temperature gas reactor design, including SiC, gas reactor matrix graphite, and non-fuelled coated particles, were irradiated.

As with other in-core experiments the facility is installed in an in-core position after removal of a solid dummy fuel element. In this case, the experiment dummy element is an integral part of the facility. The rig is comprised of a set of tungsten sample holders with graphite-lined sample spaces designed to hold a variety of samples. For the demonstration experiment, mechanical property test specimens of two types and thermal conductivity test specimens were accommodated. Tungsten was chosen because of its high density and good high temperature properties; the large in-core mass provides a large heat input from the nuclear heating. By thermally insulating the tungsten sample holders from the aluminium dummy element using a heat shield and gas gaps, high temperatures were achieved in the facility. The thermal conductivity of helium is approximately ten times that of neon. Thus increasing the helium fraction in the cover gas decreases the temperature while increasing the neon fraction increases the temperature. At the temperatures achieved in the facility, a large part of the heat loss from the sample stack is due to radiative cooling and is not affected by the gas composition.

4.3. IN-CORE FISSILE MATERIALS IRRADIATION EXPERIMENTS

There are a number of constraints on a fissile material irradiation experiment established by the US Nuclear Regulatory Commission definition of a research reactor, by the license amendment authorizing these experiments and by the technical specifications relevant to all in-core experiments at the MITR. These are: the fuel may not be contained in a ‘circulating loop through the core’, the fissile content is limited to 100 g ^{235}U or equivalent, the fissile material must be doubly encapsulated, the reactivity worth of the experiment may not exceed 1.8% ΔK/K (2.3 beta), onset of nucleate boiling (ONB) on surfaces exposed to the MITR primary coolant must be avoided, and provision must be made to sample for fission product gas at least weekly during the irradiation and for redundant automatic over-temperature reactor scrams. Due to these constraints, the focus of such experiments at MITR is to investigate the basic properties of advanced nuclear fuels at lower burn up using small aggregates of fissile material and is intended to complement the full-scale, accelerated burn up fuel evaluation programs at test reactors.

Two fuelled irradiations have been performed at the MITR. The hydride fuel irradiation experiment is given below as an example.

4.3.1. Hydride fuel irradiation

The Hydride fuel irradiation (HYFI) experiment is an irradiation of (U, Zr) H_X metallic fuel pellets in zircaloy cladding in a dedicated in-core facility. The fuel rod design and experiment conditions are meant to approximate the typical operating conditions (geometry, flux, and temperature) of a short length of LWR fuel. This fuel is of interest because of its higher thermal conductivity (reducing fuel temperatures) and favourable neutronics properties (built-in moderator and stronger negative Doppler feedback). The data from the experiment will reveal how the conductivity changes with irradiation.

Of additional interest is the fuel-cladding interaction, and in particular the possible transport of hydrogen out of the fuel and into the cladding, which may cause cladding degradation during irradiation. To attempt to ameliorate this interaction, the gap between the fuel and cladding is filled with lead bismuth eutectic (in place of traditional helium cover gas), which provides an excellent thermal bond and possibly chemical protection to the cladding inner surface. The cladding surfaces are also pre-oxidized to help slow hydrogen diffusion.

Three HYFI fuel rods were assembled into capsules and irradiated in the MIT reactor. The hydride fuel rod is constructed from zircaloy cladding with U (30 wt%)-$ZrH_{1.6}$ fuel pellets. The approximate dimensions are as follows: fuel pellet diameter is 0.950 cm by 1 cm height and 5 pellets are included in one fuel rod. The fuel/cladding gap is about 125 μm and is filled with lead-bismuth eutectic (LBE). Zircaloy cladding inner and outer diameters are 0.975 and 1.116 cm, respectively and the total height of the zircaloy cladding for one mini fuel rod is 7.5 cm. Figure 7 shows cross-sections of the fuel rod, titanium capsule, and loaded dummy element.

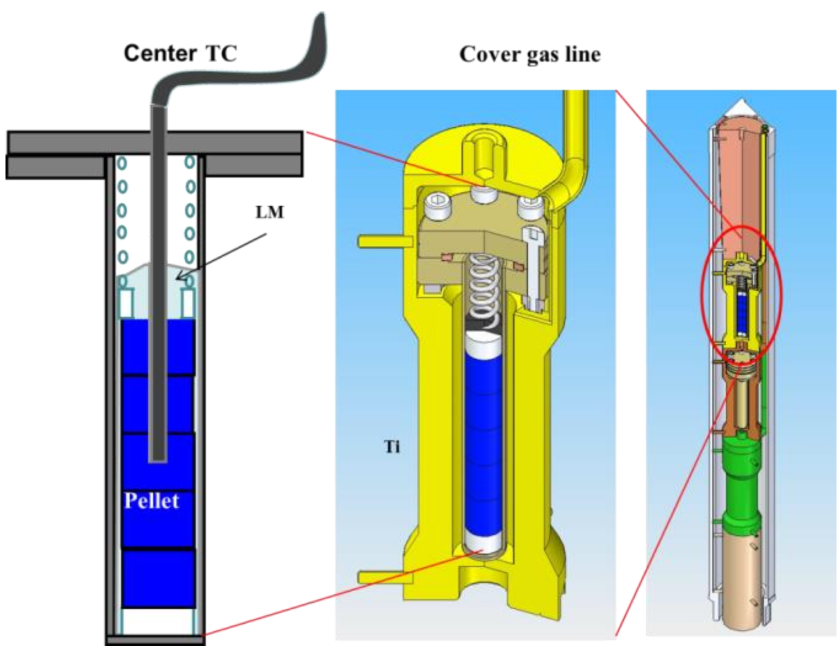


FIG. 7. HYFI fuel rod, capsule, and loaded dummy element cross sections.

4.4. INERT GAS IRRADIATIONS IN THE ICSA

The in-core sample assembly (ICSA) facility provides space for in-core irradiations isolated from the water coolant with an inert cover gas and aluminium thimble. In general, experiments in the ICSA desiring elevated temperatures utilize gamma heating (heat produced by gamma radiation from the reactor core interacting with high-Z materials. Because the heat rejection from samples inside the ICSA thimble occurs across the inert cover gas gap, the conductivity of the gas mixture can be used, along with reactor power, to control sample temperatures. Figure 8 shows an experimental verification of the control of temperature in the ICSA as a function of reactor power and the supplied helium/neon inert gas mixture. Figure 9 is a photograph of a high temperature capsule. Irradiation experiments performed in ICSA capsules includes:

(1) High temperature materials specimens consist of tensile bars, resistivity bars, and thermal diffusivity disks;
(2) High temperature fiber optic sensors;
(3) Static fluoride salt (flibe; 66% LiF-33% BeF_2) irradiation with graphite, SiC composite, metal alloys, and surrogate TRISO particles.

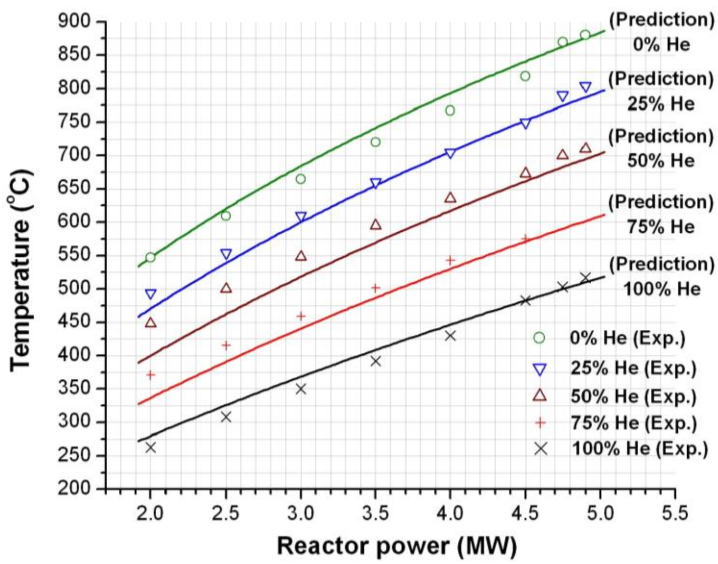


FIG. 8. Experimental verification ICSA with various gas mixtures.

FIG. 9. An High Temperature irradiation capsule.

5. SUPPORT SERVICES, EQUIPMENT AND SPECIALIZED FACILITIES

5.1. ELEMENTAL ANALYSIS FACILITIES

The MIT-NRL is currently equipped for both delayed and prompt gamma neutron activation analysis (NAA). Delayed gamma NAA (or Instrumental NAA) techniques are often used in environmental health studies to evaluate the transport of air pollutants, mineral uptake in the body, aerosol formation, the role of aerosols in acid rain, and geochemical studies. In the past few years, the MIT-NRL's NAA capability has been expanded for toxic trace element analysis in biological samples such as animal and human hair and tissue. The 1PH1 pneumatic tube can be set up so the irradiated sample is transferred automatically to the hot lab in the adjacent building for short irradiations (from a few seconds to a few minutes) for the analysis of short-lived isotopes. The NAA laboratory has three high purity germanium (HPGe) detectors, digital multi-channel analysers, and the Genie-2000 gamma spectrum analysis system.

A prompt gamma neutron activation analysis (PGNAA) facility is also available for trace element analysis. PGNAA was used primarily for the quantification of trace B-10 in blood for BNCT research. The PGNAA facility is normally installed at one of the MITR's horizontal beam ports, 4DH3. The thermal neutron flux at sample location is approximately 1.8×10^7 $n{\cdot}cm^{-2}{\cdot}s^{-1}$. The PGNAA counting facility is equipped with a HPGe detector with a multichannel analyser, and the Genie 2000 Gamma spectrum analysis.

MIT-NRL is also equipped with an inductively coupled plasma - optical emission spectrometer (ICP-OES) that is used to complement the NAA facilities. ICP-OES is a widely used multi-element analysis technique that offers good precision and speed and can be useful for elements that are not suitable for NAA or where interfering activities are present. The ICP-OES can be used for radioactive sample analysis and has been used for chemical analysis of in-pile loop coolant. Note that ICP-OES analysis requires that the sample be in aqueous solution or uniform suspension and that a standard is available with the matrix matched as closely as possible to the unknown.

6. POST-IRRADIATION EXAMINATION (PIE) FACILITIES

6.1. HOT CELLS AND HANDLING FACILITIES

The reactor containment building is equipped with an overhead polar crane with 20-tonne and 3-tonne hooks. These cranes are used for installations and removals of in-core and other experiments. A variety of shielded transfer casks are also available for transfers. There are two hot cells in the reactor hall as pictured in Figs 12 and 13. The larger cell is generally used for handling and disassembly of full-height in-core experiments. This cell is accessible for installation of custom fixturing required for particular experiments. The smaller cell has been used to handle small, high activity components and fuel from in-core experiments. A collimated gamma scan facility can be installed in the small cell. The reactor spent fuel pool is also available for storage, handling and packaging of irradiated experiments. Shipping casks up to the GE2000 can be loaded dry or wet.

FIG. 12. Reactor floor main hot cell.

FIG. 13. Reactor floor hot box.

6.2. HOT SAMPLE PREPARATION FACILITIES

Laboratory space is available within the reactor exclusion area (outside the containment building) with two standard fume hoods and a perchloric acid-capable fume hood for electro-polishing (Struers electropolisher is available.) A controlled-atmosphere 4-port glove box with furnace is also available. A ventilated hot box with manipulators is located in this laboratory and is available for specialized PIE activities requiring more shielding than can be installed in the fume hoods. Standard metallurgical sample preparation (epoxy mounting, sectioning and polishing) can be carried out on activated samples. Photography and macro-photography for irradiated specimens is also available.

6.3. ELECTRON MICROSCOPY FACILITIES

Although there are no electron microscopes within the reactor containment or exclusion area, non-dedicated facilities can be used for hot sample microscopy at MIT. The instruments available in the MIT Department of Materials Science and Engineering Central Facility are described at http://web.mit.edu/cmse/facilities/electron.html. Use of these facilities for irradiated materials is subject to dose limits and approvals from MIT's Radiation Protection Office. In some cases, dedicated sample holders may be required to reduce the probability of contamination of shared equipment.

6.4. CONTACT

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Profile 30

EXAMPLES OF LOW POWER RR

I. TRAINING RESEARCH ISOTOPES GENERAL ATOMIC REACTOR CASACCIA 1 TRIGA RC-1, ITALY

1. GENERAL INFORMATION AND TECHNICAL DATA OF RR

The TRIGA RC-1 nuclear research reactor (Training research isotopes general atomic reactor Casaccia 1) is a source of thermal neutrons.

TRIGA RC-1 (see Fig. 1) was built in 1960 in its first version with 100 kW power as part of the US Atom for Peace initiative. In 1967 its power was upgraded to 1 MW based on the ENEA staff design.

FIG. 1. TRIGA RC-1 reactor.

The TRIGA core consists of an annular structure immersed in water which serves as primary coolant (see Fig. 2). The core is arranged in a honeycomb-like array forming an annulus with seven coaxial cylindrical rings of fuel elements.

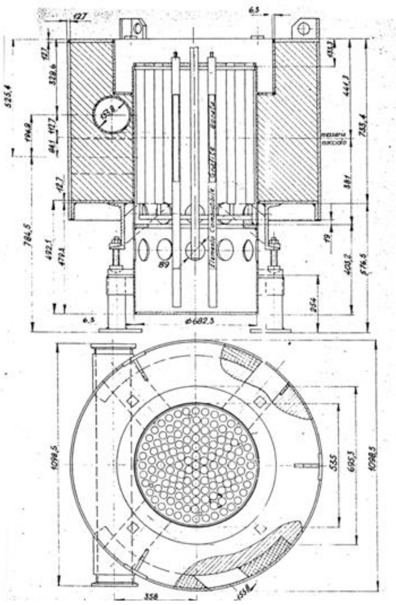

FIG. 2. Schematic view of the reactor assembly.

The reactor and the experimental facilities are surrounded by a concrete shield structure. The core and the reflector assemblies are located at the bottom of an aluminium tank (190.5 cm diameter). The overall height of the tank is about 7 m, therefore the core is shielded by about 6 m of water. The core which is surrounded by the graphite reflector, consists of a lattice of fuel elements, graphite dummy elements, control and regulation rods. There are 127 channels divided in seven concentric rings (from 1 to 36 channels per ring). The channels are loaded with fuel rods, graphite dummies and regulation and control rods. One channel houses an Am-Be source, while two fixed channels, the central one and a peripheral, are available for irradiation or other experiments.

The diameter of the core is about 56.5 cm while the height is 72 cm. Neutron reflection is provided by graphite contained in an aluminium container which is surrounded by 5 cm of lead acting as a thermal shield. An empty aluminium tube (15 cm diameter and 0.6 cm thick) traverses the graphite reflector tangentially to the reactor core for thermal flux irradiations. The core components are contained within a top and bottom aluminium grid plates: the top grid has 126 holes for fuel elements and control rods and a central thimble for high flux irradiations. The reactor core is cooled by natural convention of the water in the reactor pool.

The fuel elements consist of a stainless steel clad (AISI-304, 0.05 cm thick, 7.5 g/cm^3 density) characterized by an external diameter of 3.73 cm and a total height of 72 cm end cap included (see Fig. 3). The fuel is a cylinder (38.1 cm high, 3.63 cm in diameter, 5.8 g/cm^3 of density) of a ternary alloy uranium-zirconium-hydrogen (H-to-Zr atom ratio is from 1.7 to 1; the uranium, enriched to 20% in ^{235}U, makes up 8.5% of the mixture by weight: the total uranium content of a rod is 190.4 g, of which 37.7 g is fissile) with a metallic zirconium rod inside (38.1 cm high, 0.5 cm in diameter, 6.49 g/cm^3 of density). There are two graphite cylinders (8.7 cm high, 3.63 cm in diameter, 2.25 g/cm^3 of density) at the top and bottom of the fuel rod. Externally two end-fittings are present in order to allow the remote movements and the correct locking to the grid.

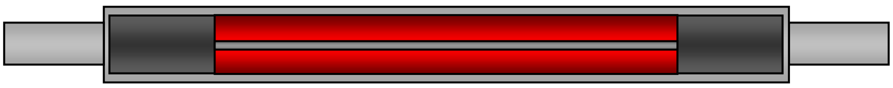

FIG. 3. Fuel rod section.

The regulation rod has the same morphological aspect as the fuel but instead of the mixture of the ternary alloy U-$ZrH_{1.7}$ there is the absorber (graphite with powdered boron carbide). Some control rods are 'fuel followed': the upper section of the rod is graphite; the next 381 mm is the neutron absorber. The follower section consists of 381 mm of U-$ZrH_{1.7}$ fuel and the bottom section of 165 mm of graphite. The graphite dummies are similar to the fuel rod but the central volume is filled by means of graphite.

A typical core loading is shown in Fig. 4. The reactor's main features are:

— maximum power: 1 MW;
— maximum neutron flux: 2.7×10^{13} n·cm^{-2}·s^{-1};
— core cooling by natural convection;
— irradiation facilities:
  - One central channel;
  - Forty positions in rotating rack;
  - One pneumatic transfer system ('Rabbit');
  - One loop for irradiations of liquids;
  - One thermal column;

- One thermalizing column;
- Six horizontal neutron channels;
- Irradiation cavity in the core (3 el. space);
- Irradiation cavity in the thermal column inside the reactor pool.

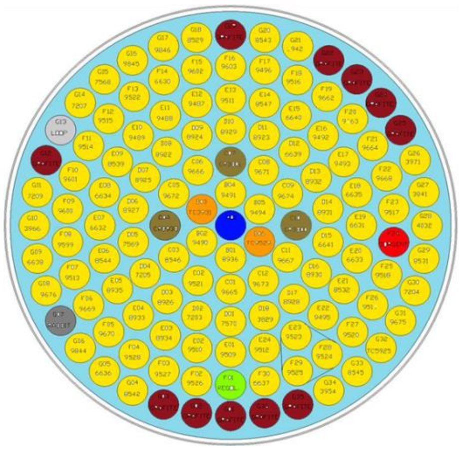

FIG. 4. TRIGA RC-1 core.

In Table 1 are shown the characteristics of the core.

TABLE 1. CHARACTERISTICS OF THE TRIGA CORE

Core	Cylindrical diameter	535 mm
	Height	670 mm
Fuel	Type	Uranium – ZrH alloy (8.5% Wt U)
	Enrichment	20% ^{235}U
	Moderator	H_2O, ZrH
	Coolant	Demineralized water in natural convection
Control rods	Type	n°3 B_4C fuel follower n°1 B_4C regulating rod
Reflector	Cylindrical inner reflector diameter	543 mm
	Outer reflector diameter	1098.5 mm
	Overall height	733.4 mm
	Radial thickness	214 mm
	Material	Graphite

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

A global view of the TRIGA RC-1 experimental facilities is shown in Figs 5 and 6.

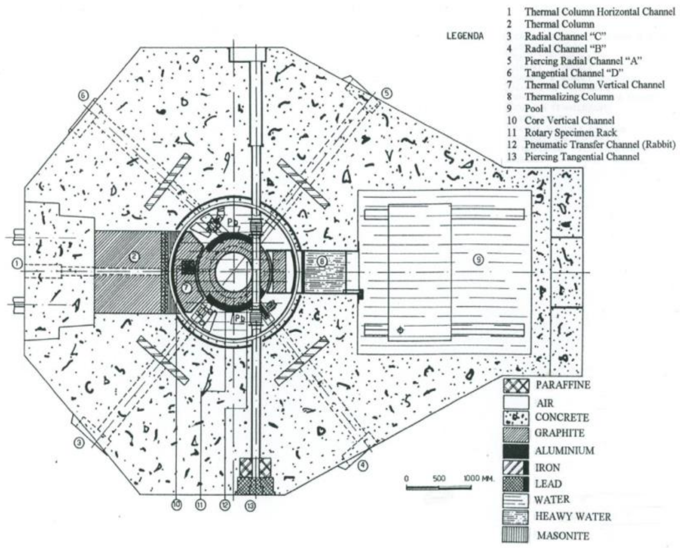

FIG. 5. Irradiations facilities.

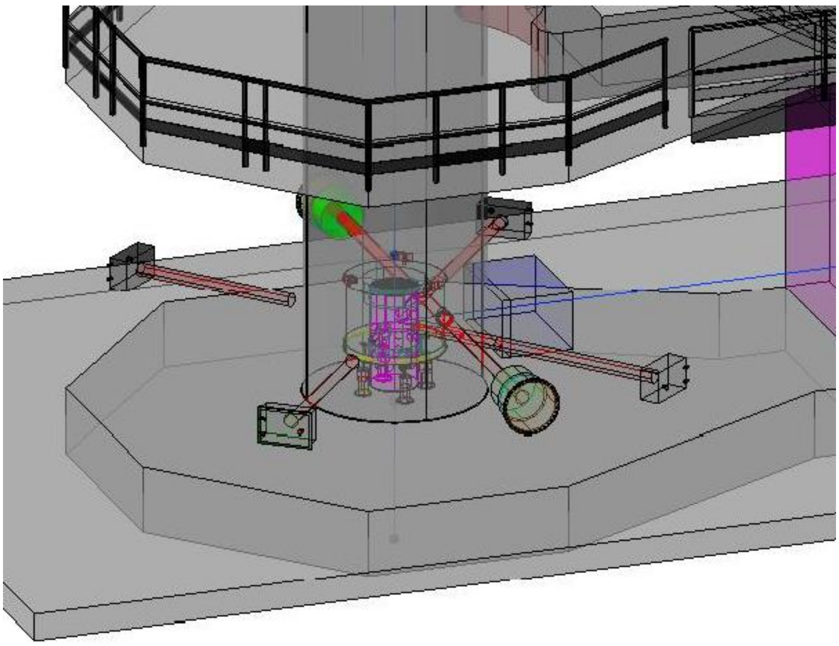

FIG. 6. Radial neutron channels.

2.1.1. Thermal column horizontal channel

The thermal column is crossed by a cylindrical hole (see Fig. 7 in red) which collimates the neutron beam. The channel, just outside the concrete shield, is provided with a mobile pneumatic shutter allowing its opening and closure (see Fig. 8). This device can be driven remotely from the reactor hall. The whole zone in front of and around the channel is shielded by concrete and paraffin blocks to reduce exposition for researcher and workers. The thermal column is suitable for neutron imaging applications.

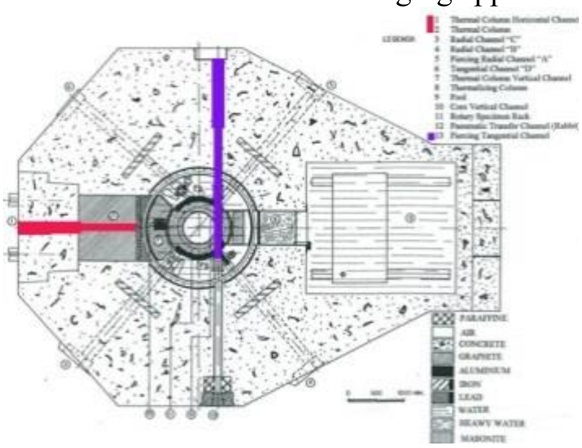

FIG. 7. Thermal column horizontal channel.

FIG. 8. Thermal column shutter.

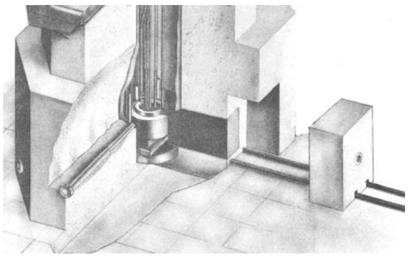

FIG. 9. Thermal column.

2.1.2. Thermal column

It is formed by two parts (see Fig. 9), internal and external to the tank. The internal part constitutes the thermal column vertical channel. The outer part is constituted by an aluminium box of square section 1.2 m × 1.2 m containing lead, graphite and a mobile part in concrete. The outer part is crossed by a cylindrical hole that collimates the neutron beam and constitutes the thermal column horizontal channel.

2.1.3. Radial channels ‘B’ and ‘C’

Consist of an aluminium inner cylindrical tube of internal diameter 152 mm. The radial channel C allows the introduction of experimental samples from the outside to the outer surface of the tank.

2.1.4. Piercing radial channel 'A'

Consists of an aluminium inner cylindrical tube of internal diameter 152 mm. The piercing radial channel allows the introduction of experimental samples from the outside to the outer surface of the reflector.

2.1.5. Tangential channel 'D'

Consists of aluminium inner cylindrical tube of internal diameter 152 mm. The tangential channel D allows the introduction of experimental samples from the outside to the outer surface of the tank.

2.1.6. Thermal column vertical channel

Consists of an aluminium box containing graphite (202 mm × 178 mm) to which it is possible to connect a cylindrical tube collimator having vertical axis (internal diameter 56 mm out of the tank).

2.1.7. Thermalizing column and shielding tank

— The thermalizing column;
  Its vertical section size is 608 mm × 608 mm, and it is divided into two parts: the first one enclosed between the shielding tank and the external wall of the reactor pool, the second one between the internal wall of the reactor pool wall and the core reflector.

  The first part is an aluminium box filled with heavy water and the second one consists of a sealed box of aluminium alloy shaped in such a way as to leave the minimum thickness of water toward the reflector and into the reactor vessel, and it is filled with graphite and air.
— Shielding tank;
  The shielding tank that originally was designed for shielding measurements has the following dimensions: 4.65 m in depth, 2.44 m wide, 2.74 m in length. Impermeability is achieved by applying a layer of epoxy (epoxy white) on the concrete of the bottom and of the walls.

  On the bottom of the tank are located two rails to allow, in the origin, the translation of a trolley for experimental devices. On top of the tank, just over the security wall which surrounds it (90 cm height), there are other two rails allowing the translation of the special equipment for the positioning of a new irradiation device.

  The main physics peculiarity of the thermalizing column, in the shielding tank, is a large, uniform and well thermalized neutron flux.

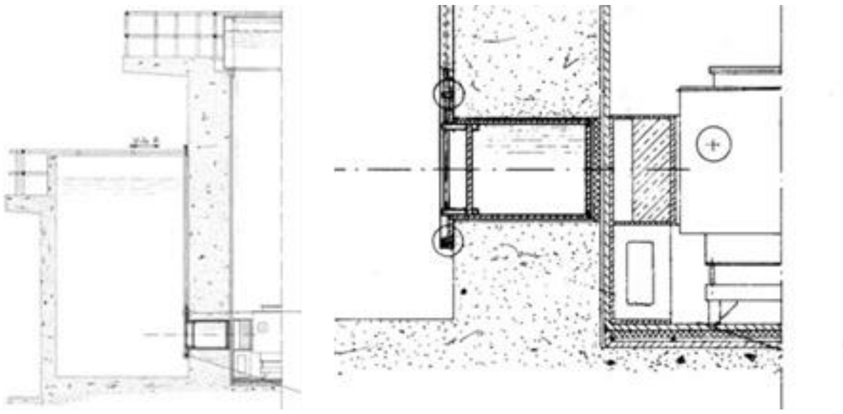

FIG. 10. Sections of the shielding tank and of thermalizing column.

2.1.8. Central thimble

The central thimble, in the center of the reactor core, allows the irradiation of small samples at locations of maximum flux and to extract a collimated beam of neutrons and ϒ-ray. The channel is constituted by an aluminium tube of 7.10 m in length and internal diameter of 34.04 mm which can be filled with air or water.

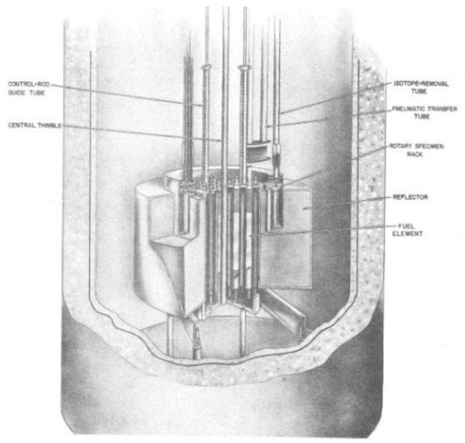

FIG. 11. Rotary specimen rack.

2.1.9. Rotary specimen rack (Lazy Susan)

The rotary specimen rack (see Fig. 11) consists of an aluminium ring mounted on a steel bearing, contains forty aluminium cups evenly spaced. These cups serve as holders for the radioisotope specimen containers. The rotary specimen rack can be rotated manually, automatically (continuous rotation or step). A single removal tube is used for inserting and removing irradiation specimen. The specimen containers, which fit into the cups, are cylinders 188 mm high and 30 mm in diameter.The tops of these containers are approximately at the same level of the top of the core. The specimen is used for the production of isotopes of average half-life in useful quantities.

2.1.10. Pneumatic transfer channel (Rabbit)

A pneumatic transfer tube (located in the outermost ring of the core) is provided for fast insertion and removal of irradiated specimens into the core. The Rabbit allows the production of radioisotopes with extremely short half-lives which are transferred from the reactor to a counting room. Special containers (inner diameter of 14.2 mm, length 100 mm), made of nylon or aluminium, can be accommodated in this facility.

2.1.11. Piercing tangential channel

This channel (see Fig. 12 in purple) crosses all the length (east-west direction) of the biological shield. It is tangential to the inner surface of the reflector, just 113 mm above respect to the core mid-plane. The channel is equipped with a collimator (see Fig. 13) designed also to minimize the gamma and neutron streaming due to structure discontinuities. The collimator filter and geometry are optimized, using also Kobayashi method, in order to maximize the neutron flux and the beam diameter. The substitution of the inner part of the collimator allows to try collimator’s response for different configuration: graphite, bismuth and air. The scope is to obtain an optimization for the n/γ ratio.

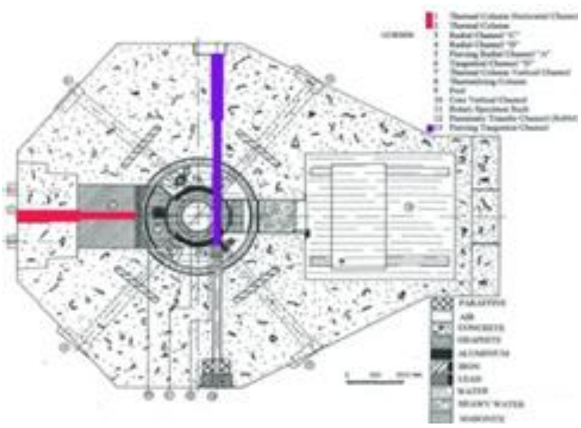

FIG. 12. Piercing tangential channel.

FIG. 13. Piercing tangential channel shutter.

2.1.12. Removable grid cavity

On the upper grid (see Figs 14 and 15) of the core is provided a removable part structured into three contiguous holes arranged in a triangular fashion. Removing three fuel elements, the spacer and the removable grid become available for irradiations of specimens having a diameter up to 60 mm.

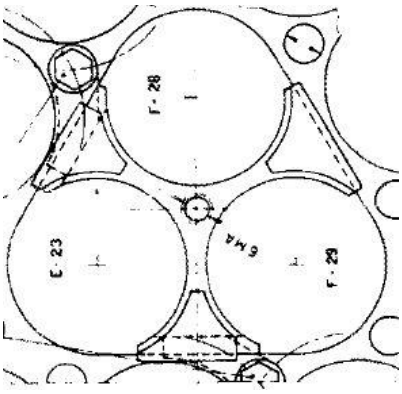

FIG. 14. Removable grid cavity.

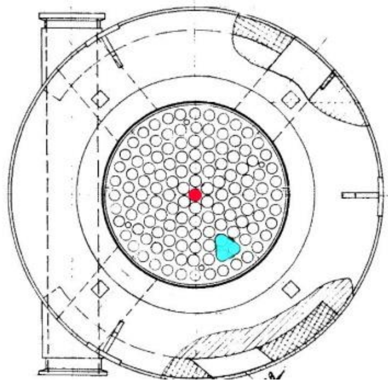

FIG. 15. Core postion of the removable grid cavity.

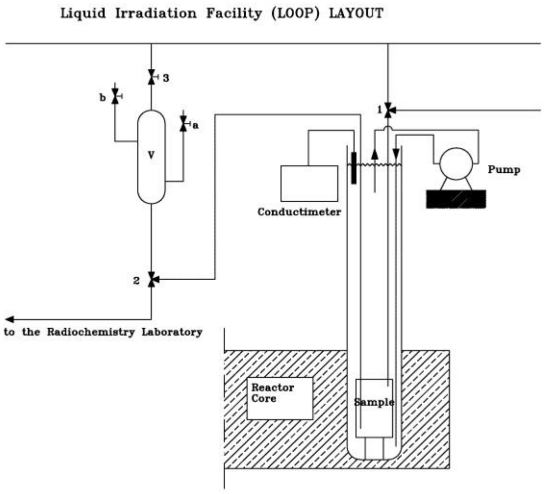

FIG. 16. Layout of the liquid irradiation facility (LOOP).

2.1.13. Irradiation facility for liquid samples (LOOP)

In the peripheral zone of the core (ring G) it is possible to place a special capsule, stainless steel made, in which can flow the liquid solution to be irradiated. The capsule is connected by means of a small stainless steel pipe with a special receiving station placed in the radiochemistry laboratory. A pneumatic system injects and extracts the liquid into and from the capsule. The stainless steel capsule is faced with the centre of the core and it is contained into an aluminium pipe filled with demineralized water which ensures, by means of a forced circulation, the removal of the thermal power produced by the capsule. The water also ensures the shielding from radiations in the axial direction. Figure 16 shows the layout of the circuit. The irradiation of liquids can be done in continuous or in batch mode.

In Table 2 is provided an overview of the main characteristics of the experimental locations.

TABLE 2. TRIGA RC-1 EXPERIMENTAL LOCATIONS MAIN CHARACTERISTICS

Experimental facility	Thermal flux $n \cdot cm^{-2} \cdot s^{-1}$	R_{Cd}*	Shape	Dimensions (mm) (unless differently specified)
A — radial channel	4.8×10^{12}	≈ 2.2	Cylinder	Ø int. = 152

B - radial channel	4.3×10^{10}	≈ 3	Cylinder	Ø int. = 152
C - radial channel	4.3×10^{10}	≈ 3	Cylinder	Ø int. = 152
D - radial channel	5.4×10^{10}	10.4	Cylinder	Ø int. = 152
Tangential channel	1.1×10^{6}	2.22	Cylinder	Ø int. = 152
Piercing tangential channel	1.1×10^{6}	1.24	Cylinder	Ø int. = 180
Thermal column horizontal channel	2.2×10^{6}	3.2	Cylinder	Ø int. = 40
Thermal column vertical channel (with plug of graphite)	1.9×10^{10}	4.3	Square	Side = 100
Thermal column vertical channel (without cap of graphite)	4.2×10^{9}	~ 4	Square	Side = 100
Central thimble	2.68×10^{13}	1.73	Cylinder 'S' shaped	Ø int. = 34.04
Thermalizing column	1.3×10^{8}	>100	Parallelepiped	608 × 608 × 155

TABLE 2 (cont.). TRIGA RC-1 EXPERIMENTAL LOCATIONS MAIN CHARACTERISTICS

Experimental facility	Thermal flux n·cm-2·s-1	RCd*	Shape	Dimensions (mm) (unless differently specified)
Rotary specimen rack	2.0×10^{12}	2.7	Cylinder 'S' shaped	Ø int. = 32
Removable grid cavity	1.25×10^{13}	2.21	Triangular prism	side ≈ 75 height = 650
Rabbit (pneumatic transfer tube)	5.1×10^{12}	2.00	Cylinder	Ø int. = 14 Ø int. tube = 27
Loop for irradiation of liquids	$\approx 5.0 \times 10^{12}$		Cylinder	V ≈ 150 ml

*R_{CD} = Cadmium ratio

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

Not available

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS

Not available

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Not available

2.5. DEVICES FOR AMPULE TESTS OF MATERIALS

2.5.1. Irradiation device at the thermalizing column

It is a plexiglas cylindrical (diameter 170 mm, length 330 mm) waterproof cavity (see Figs 17 and 18) that can be moved in the water and placed in front of the thermalizing column neutron beam, deeply in the shielded tank. The cylinder It is equipped also with a tube allowing the connection of the cavity, by wires or cables, with the external of the pool. It is possible to introduce a wide type of objects: from gold foils to ampule or others containers of sufficient dimension to test irradiation on various materials. It is provided with a positioning system to facilitate operations.

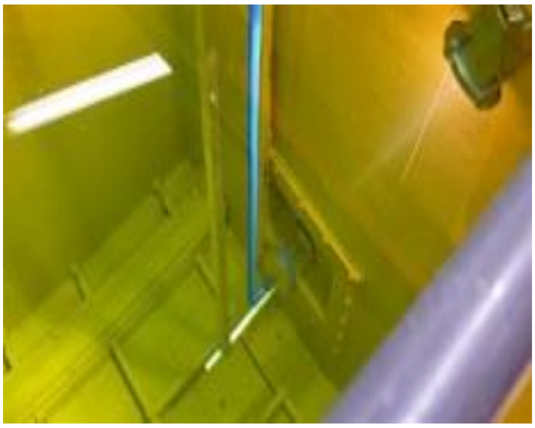

FIG. 17. Cavity position deep in the pool in front of thermalizing column.

FIG. 18. Detail of the cavity.

2.5.2. Neutron radiography and tomography device

Neutron fluxes provided by the thermal column and the piercing tangential channel are utilized to obtain a radiography image of objects and, with a time dependent image acquisition, a tomography reconstruction of such objects. The device is composed by:

- A support for object able to rotate and translate so that the operator can reach the optimal object interaction with neutron flux;
- A neutron converter provided with efficiency parameters;
- A system equipped with an optical system used to focus the light produced;
- A CCD camera connected with an acquisition and analysis system composed by hardware and software;
- A software tool for tomography reconstruction.

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR

Not available

2.7. EXPERIMENTAL FACILITIES FOR APPLIED RESEARCH

2.7.1. Isotopes for medicine

TRIGA reactor has been used for the preparation of isotopes for nuclear medicine. ^{18}F has been produced irradiating $LiNO_3$ in the fast pneumatic system of the reactor, by $^{16}O(t, n)^{18}F$ reaction using tritons from $^{6}Li(n, t)^{4}He$ reaction. The produced ^{18}F has been efficiently purified in short times by an original radiochemical purification technique, and has been used for the synthesis of ^{18}F-FGD medical grade, in a semi-automatic home-made synthesis apparatus. A scale up of the preparation system is currently under development, by using a new device that is able to perform direct high-yield irradiation of natural and enriched Li solutions.

The reactor has also been used for the preparation of the isotope ^{166}Ho, by strating from ^{165}Ho; the obtained ^{166}Ho, an high energy beta emitter, was employed for the synthesis of different therapeutic carriers (labelled polymers, radioactive bio-absorbable micro and nano-spheres), and was used in the experimental therapy of human patients in solid tumors (skin cancer, glioblastoma, liver metastasis), with promising clinical results. An international patent on the technique has been filed.

Actually a new original facility is under installation, devoted to the preparation of isotopes of clinical interest in therapy and diagnostics, (^{131}I, ^{133}Xe, ^{99}Mo).

2.7.2. Isotopes for industry

TRIGA reactor has been used for the characterization and follow up of industrial processes. In one application radioactive ^{198}Au has been prepared by neutron irradiation in central channel reactor facility. The radioactive metal was dissolved in molten aluminium of some large scale industrial electrowinning cells. The follow up of isotopic ^{198}Au dilution allowed a complete study of mass balance, and the optimization on-line of efficiency electrolysis parameters of the industrial cells.

A second application has been developed, in which ^{133}Xe has been prepared by neutron irradiation in fast pneumatic reactor irradiation facility. The radioactive gas was used as a gaseous tracer in the process of coke preparation. After in -field test, the system has been applied in industrial coke oven process, and allowed a precise monitoring of temperature set point for industrial continuous coke preparation. An international patent on the technique has also been filed.

2.7.3. Neutron activation analysis

Neutron activation analysis (NAA) has been widely employed by means of TRIGA reactor since 1963.

In the TRIGA reactor NAA is mainly performed by pile irradiations using either a vertical channel passing through the core centre (central thimble), a second vertical channel characterized by a pneumatic tube to transfer the irradiated samples (Rabbit), a rotating rack with forty holes for samples introduction (Lazy Susan), and a water pool, separated from the reactor core, with a thermalizing thickness of D_2O (Thermalizing column).

Gamma spectrometry measurements are performed by mean of HPGe detectors supplied from Canberra and ORTEC, equipped with adequate instrumentation and software. The laboratory is also equipped with an anti-coincidence measurement system utilizing a NaI 12 inches diameter × 12 inches length annular single crystal Bicron detector characterized by a relevant spectral background reduction. Another useful detector is constituted by a HPGe planar detector with high efficiency in measuring x and γ rays of energy < 100 keV and for XRF counting.

By exploiting the instrumental analysis (INAA) potentials it is possible to study analytically and determine, either by thermal, epithermal, or fast neutrons, or by the radiochemical separation, the majority of the macro constituents, minor constituents, trace, and ultra-trace elements in a very wide set of matrices and materials. The samples that may be analyzed and the applications of the INAA range from alloys to soils; from sediments and suspended matter to the atmospheric particulate matter; from archaeological materials to the materials for the study of the physics of sub atomic particles; from radiotracers to the execution of forensic studies.

2.7.4. Radiological characterization

During the last years the activities carried out at the TRIGA reactor by Nuclear materials characterization laboratory's personnel has been aimed to the radiological characterization of drums containing wastes produced in the routine activities of the plant.

The characterization was carried out using mobile equipment:

- *ISOCS (In situ object counting system)*: radiological characterization system for assaying objects of any shape and nature containing ϒ-emitting radionuclides; the measurement system operates with a Germanium detector, whose response to a series of point sources or distributed in predefined arrays was characterized using Monte Carlo codes;
- *Portable multi-channel*, equipped with neutron probe and gamma radiation detector. This instrument is characterized by high accuracy and speed in response and was used for preliminary inspections of the drums.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

Not available

3.2. SNF AND RADWASTE MANAGEMENT

Not available

3.3. HOT CELLS PIE FACILITIES

Not available

3.4. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS

TRIGA RC-1 facility has a section devoted to design and manufacture experimental devices (mechanicals, hydraulics and electronics). Some examples of this capability are:

- Neutron collimators;
- Channel shutters;
- Irradiation devices;
- Optical bench for neutron imaging;
- 'Ad hoc' hydraulics loops;
- Ancillary systems for experiments;
- Electronic control panels;
- Neutron activation analysis.

4. HUMAN RESOURCES DEVELOPMENT (EDUCATION AND RR TRAINING FACILITIES)

- Training for university students;
- Experiment design (with Monte Carlo and deterministic codes);
- Neutronic characterization of irradiation channels;
- Hands-on educational experiences for university students;

— Integral control rod worth measurement by positive period method;
— Neutron flux measurement;
— Training for reactor operators.

5. SOME RELEVANT EXAMPLES OF R&D EXPERIENCES OCCURRED IN THE LAST TEN YEARS

— Test and utilization of a special instrumentable TRIGA fuel element;
— Multiplication factors measurements by means of D-T tubes;
— Investigation by neutron tomography of a volumetric 3D display for visualization of archaeological samples;
— Neutron imaging tests of archaeological samples;
— Neutron imaging tests of electronic/mechanical equipment;
— Neutron imaging tests of biological samples;
— Environmental (neutron/gamma) tests of electronic components;
— Design and characterization of neutron collimators;
— Tests and development of innovative neutron detectors;
— Irradiations and neutron activation of many kind of solid/liquid samples for several purposes;
— Developing of a digital console for control rooms parameters supervision.

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Note (*): 2002-2010 — The TRADE programme was planned to investigate the static and dynamic behaviour of ADS at power in thermal neutron spectrum. Because problems in financial backing, the programme was interrupted at the end of 2004. In spite of these unlucky circumstances, a huge experimental data bank has been set up during the time period prior to the programme interruption. In 2007 IAEA endorsed this experimental campaign, and an experimental benchmark, named 'pre-TRADE experimental benchmark', was launched in the frame of the Coordinated Research Project 'Analytical and Experimental Benchmark Analyses of Accelerator Driven Systems (ADS)' coordinated by IAEA. The benchmark was focused on the evaluation, via computation, of the correction factors to be applied to the PNS Area-ratio and MSA results for the selected reactivity estimates to take into account the role of the spatial/energy effects on the rough experimental data.

7. CONTACTS

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IAEA-RRDB: TRIGA

II. RESEARCH NUCLEAR REACTOR TAPIRO, ITALY

1. GENERAL INFORMATION AND TECHNICAL DATA OF RR

The TAPIRO nuclear reactor (see Fig. 1), located at the ENEA Casaccia Research Centre near Rome, is a fast neutrons source and its design is based on the Argonne Fast Source Reactor — Idaho Falls. The reactor name comes from the Italian acronym *TA*ratura *PI*la *R*apida Potenza Zer*O* (Fast pile calibration at 0 power). TAPIRO was built in the 1960s and its first criticality was on April 1971. In the frame of an agreement between ENEA and SCK/CEN Mol (Belgium), an extensive neutronic characterization of the TAPIRO source reactor was carried-out (1980-1986). It was found that TAPIRO is able to provide a family of neutron spectra of extremely variable hardness (about pure fission spectrum near the core centre). This remarkable feature makes the TAPIRO most suitable to many metrology applications, also taking into account that a good spherical symmetry of the neutron flux shape was evidenced by the joint ENEA-SCK/CEN experimental campaign.

The TAPIRO reactor can operate at the maximum power of 5 kW, and the neutron flux at the centre of the core at full power is about 4×10^{12} n·cm^{-2}·s^{-1}.

The reactor core is a cylinder made of highly enriched metallic uranium (weight 98.5% U; 1.5% Mo) enclosed in a stainless steel cladding. The core is split in two parts; the first one is fixed and takes up 2/3 of the total volume, whereas the second one is movable. The cylindrical core is surrounded by a cylindrical reflector made of copper, of thickness about 30 cm and height 72 cm. The reflector is housed in a steel sheath surrounded by the biological shield made of borate concrete. There are many irradiation channels crossing the reflector. The main channels on the horizontal plane are: two radial channels, a diametral channel and a tangential channel.

FIG. 1. TAPIRO reactor room.

The experimental equipment is complemented by a thermal-column (see Fig. 2). The purpose of the thermal column is to provide an epithermal neutron flux, allowing at the same time the assembling of large experimental equipment. The thermal column was originally filled with graphite in order to provide thermal neutrons. In any case, different materials can be placed inside the thermal column cavity to obtain specific neutron spectra of interest. The thermal column cavity is delimited by a metal case having 2 m^3 of available volume with 1.6 m of depth.

FIG. 2. Thermal column.

The thermal column shape is a parallelepiped with a straight section of 110 cm × 110 cm radially arranged with the longitudinal axis perpendicular to the diametral channel. The thermal column door, made of concrete with the same characteristics of the biological shield, has a thickness of approximately 60 cm.

The reactor is controlled by several rods made of copper, capable of vertical movement of extraction and insertion inside the cylindrical reflector housings. The reactor control is achieved by varying the neutron leakage level from the reflector, i.e. a control rod extraction provides a negative reactivity introduction whereas a control rod insertion provides a positive reactivity introduction.

The core is cooled by helium circulating in a gap of a few millimeters between core and reflector.

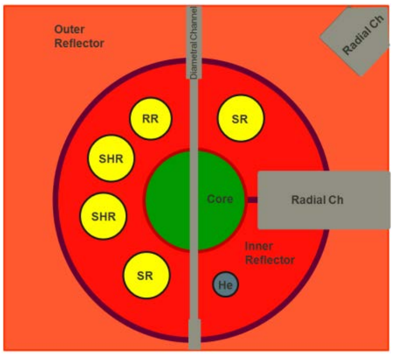
The star-up source consists of Am-Be of 185 GBq (5 Ci), and it is moved by a mechanism allowing its displacement from the peripheral area of the shield to the core along the radial channel named ‘Source channel’ (see Fig. 4).

Some characteristics of the reactor are provided in Table 1.

TABLE 1. CHARACTERISTICS OF THE TAPIRO REACTOR

Core	Cylindrical: diameter about 120 mm Diameter/height ratio about 1	
Fuel	Uranium-molybdenum alloy (weight 98.5% U – 1.5% Mo) Density: 18.5 g cm^{-3} Enrichment: 93.5% U-235	
Cladding	Stainless steel:	thick 0.5 mm
Reflector	Cylindrical inner reflector: Outer reflector: Overall height: Material: Weight:	diameter 348 mm diameter 800 mm 700 mm Copper 2600 kg
Cooling system	Forced He: Heat exchanger + Cooling Inlet core temp:	100 g/s at 7.35 bar 25°C — outlet 35°C
Biological shield	Shape: Thickness: Material: Density:	near spherical 1.75 m high density borate concrete 3.7 kg·dm^{-3}
Irradiation channels	3 channels at the reactor midplane 1 tangential (to the top edge of the core)	
Control rods (see Table 2)	2 shim rods + 2 safety rods +1 regulating rod	

In Figure 3 some characteristics of the control rods are provided.



Control rods — five control rod made of copper with a driving distance of 150 mm

Two shim rods (SHR) — used for coarse control and/or to remove reactivity in relatively large amounts

One regulating rods (RR) — used for fine adjustments and to maintain desired power or temperature

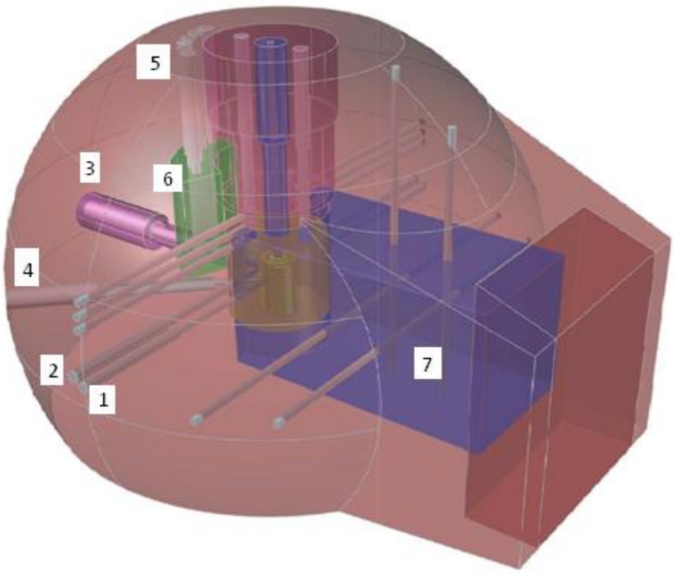
Two safety rods (SR) — provide a means for very fast shutdown in event of unsafe conditions

FIG. 3. TAPIRO control rods.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL FACILITIES AT RR INCLUDING INSTRUMENTATION DEVICES

2.1. GENERAL DESCRIPTION OF EXPERIMENTAL AND TESTING FACILITIES

Figures 4 and 5 show the position of the TAPIRO experimental locations together with their specific description.



1) Diametral channel;
2) Tangential channel;
3) Radial channel 1;
4) Radial channel 2;
5) Detector channels;
6) Paraffin;
7) Thermal column.

FIG. 4. Irradiation facilities.

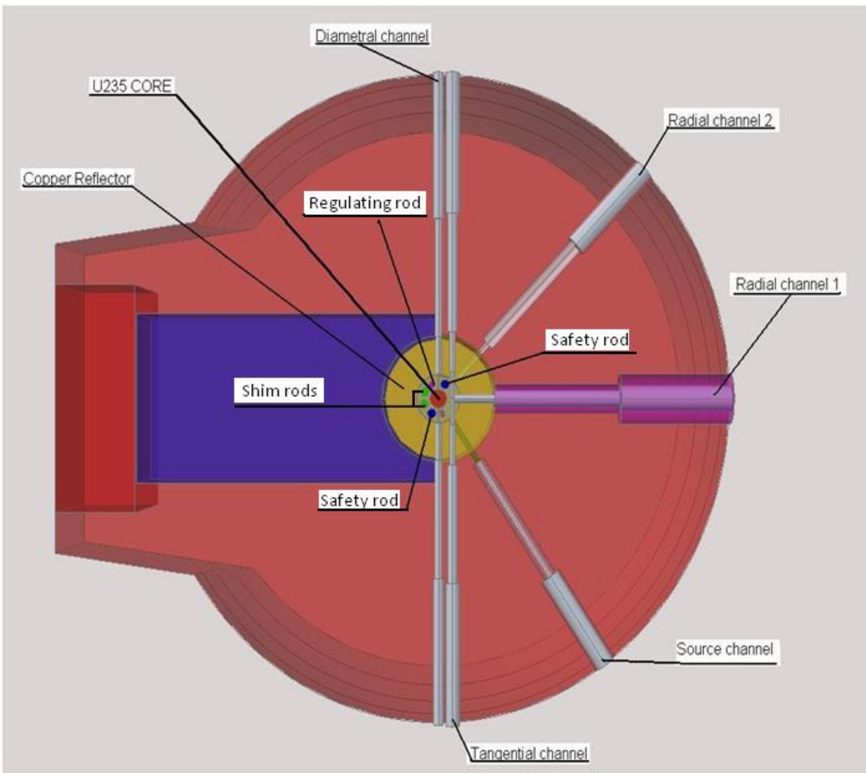


FIG. 5. Horizontal reactor section.

The experimental channels allow the installation of devices or samples at positions with different neutron flux and spectrum. Each channel consists of a metallic cylindrical jacket and a plug for shielding purposes. The channels have a gradually reducing section to lower the gamma streaming effect. Each channel plug is essentially constituted by a casing filled with shielding material for the entire section, and it is provided with a copper extension occupying the area of penetration in the reflector. This extension may be modified for hosting the sample container. The plugs are provided with three holes available for remote control or power cables eventually needed by the experiments.

2.1.1. Irradiation channels

The irradiation channels (see Figs 6 and 7) are:

— Three different channels at the reactor mid-plane and one tangential (to the top edge of the core) channel;
— One mid-plane channel crossing the core, and allowing measures with small samples (internal diameter of the channel in correspondence of the core about 1 cm) in almost pure ^{235}U fission spectrum.

FIG. 6. Grand horizontal channel, radial channel 1.

FIG. 7. Radial 2, tangential, diametral channels.

2.1.2. Thermal column

As mentioned above, a large experimental cavity, labelled thermal column (parallelepiped 110 cm × 110 cm × 160 cm) is present within the shield zone.

In Table 2 is provided an overview of the main characteristics of the experimental locations.

TABLE 2. TAPIRO EXPERIMENTAL LOCATIONS MAIN CHARACTERISTICS

Name	Position	Penetration	Useful diameter
Diametral channel (DC)	Piercing, horizontal, diametral in the core	Inner and outer fixed reflector, core	10 mm in core
Tangential channel	Piercing, horizontal, 50 mm above core mid-plane, parallel to DC, 106 mm from core axis	Inner and outer fixed reflector	30 mm in reflector
Radial channel 1 (RC1)	Radial, horizontal on core mid-plane, at 90° with respect to DC	Inner and outer fixed reflector, up to 93 mm from core axis	56 mm in reflector
Radial channel 2	Radial, horizontal on core mid-plane, at 50° with respect to RC1	Outer fixed reflector, up to 228 mm from core axis	80 mm in reflector
Grand horizontal channel (GHC)	Radial, concentric with RC1	Up to reflector outer surface	400 mm near reflector
Grand vertical channel (GVC)	Above core, on the same axis	Outer fixed reflector, up to 100 mm from upper core base	800-900 mm in reflector
Thermal column	Horizontal	Shield, up to outer reflector	110cm×116cm×160cm
Irradiation cavity	On safety plug upper base	7.4 mm	33 mm

2.2. LOOPS FOR TESTING COMPONENTS OF REACTOR CORE

Not available

2.3. EXPERIMENTAL FACILITIES FOR INVESTIGATION OF ACCIDENTAL CONDITIONS

Not available

2.4. FACILITIES FOR INVESTIGATION OF CORROSION OF REACTOR MATERIALS

Not available

2.5. DEVICES FOR AMPULE TESTS OF MATERIALS

Not available

2.6. DEVICES FOR INVESTIGATION OF FUEL AND STRUCTURAL MATERIALS BEHAVIOUR

Not available

2.7. EXPERIMENTAL FACILITIES FOR APPLIED RESEARCH

The research reactor TAPIRO is a source of fast neutrons with a wide variety of neutron spectra and, therefore, may be used for different areas of application like: validation of calculation codes for generation IV reactors design; fast neutrons damage; benchmark for nuclear data testing, evaluation of fast neutron damage induced on electronic components; qualification of chains of innovative detectors, hands-on experience in nuclear engineering courses.

2.7.1. Medical applications

Experimentation of BNCT (boron neutron capture therapy) for the treatment of certain types of brain tumours by an epithermal neutron beam (energy between 0.4 eV and 10 keV). BNCT experimentation at the TAPIRO reactor has been carried out on anthropomorphic phantoms, cells and animals.

2.7.2. Materials

Characterization of ^{16}N counters devoted to monitoring functions in Eastern Europe power plants. Neutron radiation damage on lead tungsten single crystals, APD's (avalanche photo diodes) and optical flats in the frame of the ECOLE electromagnetic calorimeter design (CERN LHC-Large Hadron Collider Project), muon spectrometer ATLAS (a toroidal LHC apparatuS).

Neutron radiation damage on some devices of the Large Hadron Collider (CERN LHC) as the monitored drift tubes (MDT) of the ATLAS muon spectrometer and the avalanche photodiodes (APD) of the electromagnetic calorimeter CSM.

Neutron radiation influence on airspace electronic components (silicon based). Radiography and sectional radiography in the field of non-destructive analysis techniques.

Neutron irradiation tests for determining the susceptibility of semiconductor devices to degradation in the fast neutron environment (qualification of radiation hard electronic devices):

- Neutron irradiation test of integrated circuits and discrete semiconductors for aerospace applications;
- ATLAS experiment.

2.7.3. Codes validation

Experiences for the validation of codes applied to the analysis of nuclear plants characterized by high degree of heterogeneity, as in the case of HTGR systems (high temperature gas-cooled reactor).

2.7.4. Neutron activation analysis

Gamma spectrometry measurements are performed by mean of HPGe detectors supplied from Canberra and ORTEC, equipped with adequate instrumentation and software. The laboratory is also equipped with an anti-coincidence measurement system utilizing a NaI 12 inches diameter × 12 inches length annular single crystal Bicron detector characterized by a relevant spectral background reduction. Another useful detector is constituted by a HPGe planar detector with high efficiency in measuring X and γ rays of energy < 100 keV and for XRF counting.

By exploiting the instrumental analysis (INAA) potentials it is possible to study analytically and determine, either by thermal, epithermal, or fast neutrons, or by the radiochemical separation, the majority of the macro constituents, minor constituents, trace, and ultra-trace elements in a very wide set of matrices and materials. The samples that may be analysed and the applications of the INAA range from alloys to soils; from sediments and suspended matter to the atmospheric particulate matter; from archaeological materials to the materials for the study of the physics of sub atomic particles; from radiotracers to the execution of forensic studies.

2.7.5. Radiological characterization

During the last years the activities carried out at the TAPIRO reactor by Nuclear materials characterization laboratory's personnel has been aimed to the radiological characterization of drums containing wastes produced in the routine activities of the plant.

The characterization was carried out using mobile equipment:

(1) *ISOCS (In situ object counting system):* radiological characterization system for assaying objects of any shape and nature containing ϒ-emitting radionuclides; the measurement system operates with a germanium detector, whose response to a series of point sources or distributed in predefined arrays was characterized using Monte Carlo codes;

(2) *Portable multi-channel*, equipped with neutron probe and gamma radiation detector. This instrument is characterized by high accuracy and speed in response and was used for preliminary inspections of the drums.

3. RELATED ENGINEERING AND RESEARCH INFRASTRUCTURE

3.1. FRESH AND IRRADIATED EXPERIMENTAL MATERIAL LOGISTIC

Not available

3.2. SNF AND RADWASTE MANAGEMENT

Not available

3.3. HOT CELLS PIE FACILITIES

Not available

3.4. CAPABILITIES TO DESIGN AND MANUFACTURE EXPERIMENTAL DEVICES AND MEASUREMENT SYSTEMS

TAPIRO facility has a section devoted to design and manufacture experimental devices (mechanicals, hydraulics and electronics). Some examples of this capability are:

- Neutron collimators;
- Channel shutters;
- Irradiation devices;
- Optical bench for neutron imaging;
- 'Ad hoc' hydraulics loops;
- Ancillary systems for experiments;
- Electronic control panels;
- Neutron activation analysis.

4. HUMAN RESOURCES DEVELOPMENT (EDUCATION AND RR TRAINING FACILITIES)

- Training for university students;
- Experiment design (with Monte Carlo and deterministic codes);
- Neutronic characterization of irradiation channels;
- Hands-on educational experiences for university students;
- Integral control rod worth measurement by positive period method;
- Neutron flux measurement;
- Training for reactor operators.

5. SOME RELEVANT EXAMPLES OF R&D EXPERIENCES OCCURRED IN THE LAST TEN YEARS ARE LISTED BELOW

— Neutron irradiation test of integrated circuits; discrete semiconductors and electronic boards for:
— Aerospace applications (IMT);
— Devices of the ATLAS experiment at Large Hadron Collider (LHC).
— Self-powered neutron detectors (SPND) were studied to investigate the applicability of such detectors to neutron flux monitoring in ITER-TBM (test blanket module). Three commercial SPND (Rh, V and Co emitters) were located in the tangential channel of TAPIRO reactor at about 20 cm from the core centre.
— Piezo-motor neutron irradiation test (EU Project F4E-GRT-282).

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7. CONTACTS

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III. BFS-1 AND BFS-2 CRITICAL FACILITIES, RUSSIAN FEDERATION

1. GENERAL INFORMATION AND TECHNICAL DATA

The BFS complex of the Institute for Power Engineering (SSC RF-IPPE) includes two critical facilities: BFS-1 and BFS-2, as well as the electron accelerator — M1-30. The BFS is a unique experimental base designed for making studies in the areas of fast reactor physics, safety, and core design optimization. A variety of reactor materials is available at the BFS facilities (highly enriched uranium, plutonium, fertile materials, different absorbers, coolant and structural materials)

1.1. GENERAL INFORMATION AND TECHNICAL DATA OF BFS-1

The BFS-1 facility was built at IPPE for full-scale modelling of fast-reactor cores, blankets, in-vessel shielding, and storage. Horizontal and vertical cuts of the facility are shown in Figs 2 and 3. The core volume can be varied from tens of litres to two cubic meters. The facility was put into operation in June 1961. Since then, core prototypes of many reactor designs have been studied at this facility and many experiments for validation of neutron data have been performed.

The critical assemblies are constructed at the BFS-1 facility from set of vertical stainless steel or/and aluminium tubes with outer diameter of 5.0 cm and 0.1 cm wall thickness. The hexagonal lattice pitch of the tubes is 5.1 cm. The tubes are filled with disks (pellets) of fissile or structural materials in a repeated cell arrangement. Sample geometrical structure of the BFS assembly tubes is shown in Fig. 4.

These tubes are put inside a cylindrical steel vessel with inner diameter of 2002 mm, 2775 mm height, and 22 mm wall thickness. On one side, the cylindrical core vessel is 'cut' by an aluminium wall with a thickness of 10 mm. A graphite 'thermal column' is behind the wall. The distance from the centre of the vessel to this wall is 778 mm. Cadmium lining (1.00 ± 0.06 mm thick) covers the outer surface of the wall, between the aluminium wall and graphite. The graphite thermal column is shown on the left side in Figs 2 and 3.

On the opposite side of the core is a bay for the 'metal column', which is an extension of the reflector designed for studying neutron penetration through thick layers of materials. The 'metal column' is continuous with the central hexagonal grid. The inner width of the bay is 1327 mm. The distance from the centre of the assembly to the far wall of the bay is 1836 mm. Cadmium lining (1.00 mm thick) covers the outer surfaces of the 22-mm-thick steel bay walls.

Inside the curved walls of the vessel, steel segments with a cylindrical outer surface (radius 1000 mm) and flat inner surfaces are fastened. Therefore, the shape of the BFS-1 working volume is an irregular octagonal prism. Cadmium sheets with a thickness of 1.00 mm are placed between the segments and the curved walls of the vessel. Two types of segments are used: two small segments with maximum thickness (arc height) of 29 mm and four big segments with maximum thickness (arc height) of 45 mm. The six segments, which can be seen in Fig. 2, rest on a 95-mm-thick steel lattice plate. The height of segments is 2235 mm.

Large concrete blocks of about 2 g/cm^3 density provide biological shielding around the vessel. The upper layer is made of high-density (approximately 4 g/cm^3) concrete blocks. There is concrete shielding above the thermal column and removable shielding above the 'metal column'. The concrete cover above the 'metal column' was removed during the early stages of the approach to critical, in order to easily move depleted-uranium dioxide tubes from the core to the metal column. Later, it was closed so that the experimentalists could more easily walk around the core. The block masonry forms a cylindrical cavity (206 cm in diameter) for installation of the critical assembly vessel and two rectangular cavities for the graphite and metal columns.

Six SNM-11 type BF_3 impact counters are located near the stainless steel tank containing the critical assembly. The centres of the 15-cm-diameter holes for the counters are about 20 cm from the core vessel. There are also three large cylindrical regions (60 cm diameter), shown in Fig. 2, that are used to store fresh fuel tubes. They are about 80 cm away from the columns of the assembly.

A 9.5-cm-thick steel support and spacing plate, shown in Fig. 3, is located at the bottom of the cylindrical section of the vessel and ‘metal column’. There are 3.5-cm-diameter holes in this grid plate that are drilled in a 5.1-cm hexagonal lattice pitch. Shanks of the tubes forming the assembly are put into these holes (see Fig. 5). Up to 1169 tubes can be installed in the octagonal section of the vessel with 612 additional tubes in the ‘metal column’.

The 14 tubes used as control rods are shown in the central part of Fig. 2. The fourteen holes in the grid plate that are provided for the control rods have a diameter of 5.3 cm. Control-rod tubes have the same diameter as other tubes of the assembly and similar contents, but they have a longer active length and are connected to drives at their bottom. Control-rod drives are shown in the lower part of Fig. 3. The BFS-1 facility has three kinds of control-rod drives: two independent drives of control rods, each connected to one tube; three compensating-rod drives, each connected to two tubes; and three safety-rod drives, each connected to two tubes. When control-rod tubes are in their upper positions, fuel and other contents are at the same levels as those in other tubes. The lower, shutdown position of the tubes is usually 80 cm to 100 cm below the upper level. (End switches installed at the level optimized for each critical assembly determine the lower positions of the control rods.)

1.2. GENERAL INFORMATION AND TECHNICAL DATA OF BFS-2

The BFS-2 facility was designed in the IPPE for the full-size simulation of cores and shielding of large fast reactors with a unit power up to 3000 Mwe (see Fig. 1). It was put into operation in October 1969. In August 1989 it was shut down for a reconstruction of the control room and for changing the electronic equipment of the system of monitoring and protection. It was again put into operation in March 1990.

FIG. 1. BFS-2 facility.

The BFS-2 critical facility is in the same building as the BFS-1.

The walls of the vessel, in which the investigated critical assemblies are constructed, limit a space of an irregular prismatic form with an effective diameter ~ 5 m and a height ~ 3.3 m.

The vessel walls are vertical screens, each of them consisting of two steel sheets, 14 mm and 6 mm thick, with the cadmium layer between them. The cadmium layer is 1 mm thick and its purpose is absorption of neutrons which slowdown in the concrete shielding and come back into the vessel of the critical facility.

Nine thousand four hundred sixty tubes are installed inside of the vessel. They are made of stainless steel (1X18H10T). Their outside diameter is 50 mm and the wall thickness is 1 mm. Tails of tubes are installed in holes of the diagrid plate. The thickness of this steel plate is 100 mm.

The hexagonal lattice pitch of the diagrid is 51 mm.

The space between the tubes can be filled with stainless steel (1X18H10T) sticks of 8 mm diameter and about 3100 mm long. This space can be also used for inserting different small-size detectors for experimental investigations of reaction rate distributions.

2. EXISTING AND PROSPECTIVE EXPERIMENTAL POSSIBILITIES

2.1. EXPERIMENTAL POSSIBILITIES OF BFS-1

An experimental device used for the placement of fission chambers and other experimental detectors is shown in Fig. 3 at the top of the facility, above the assembly.

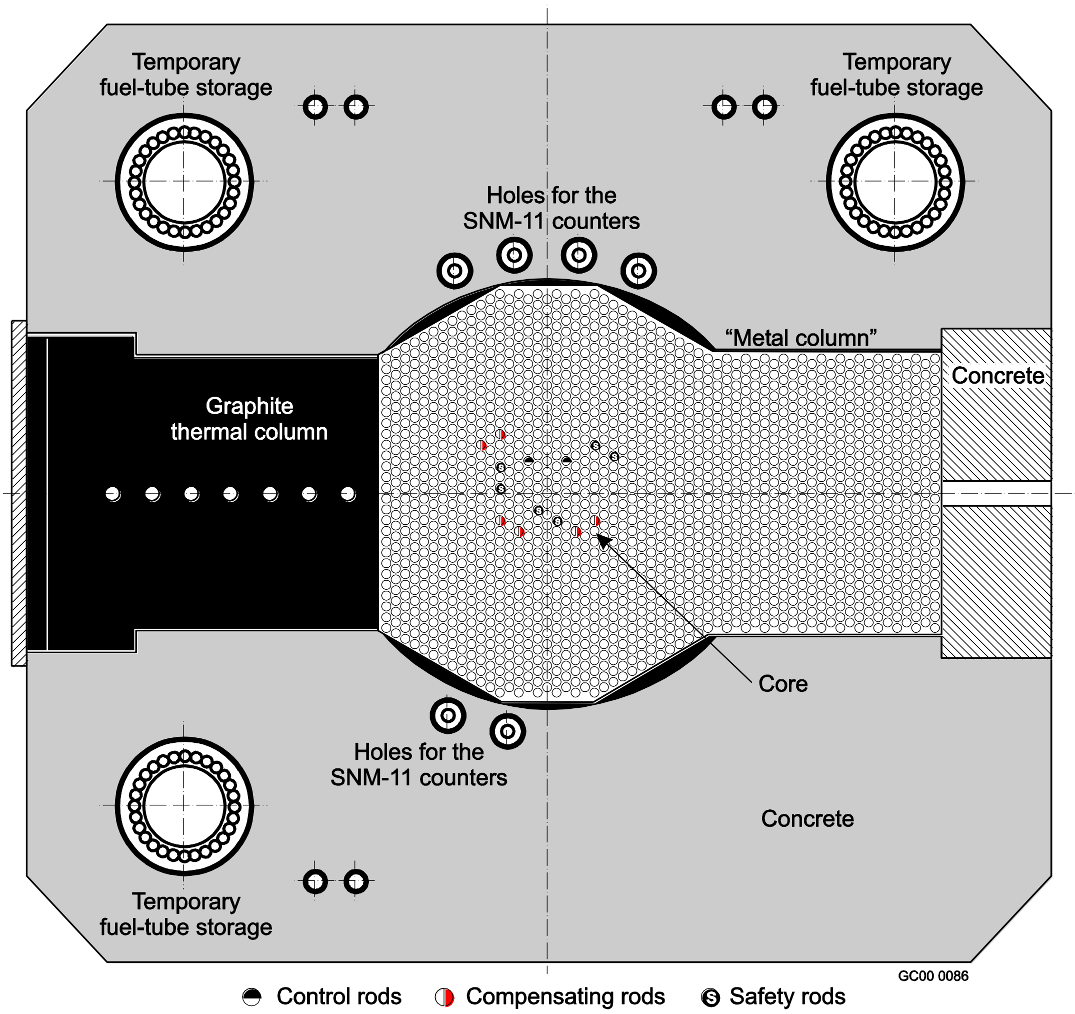

FIG. 2. BFS-1 facility (horizontal cut).

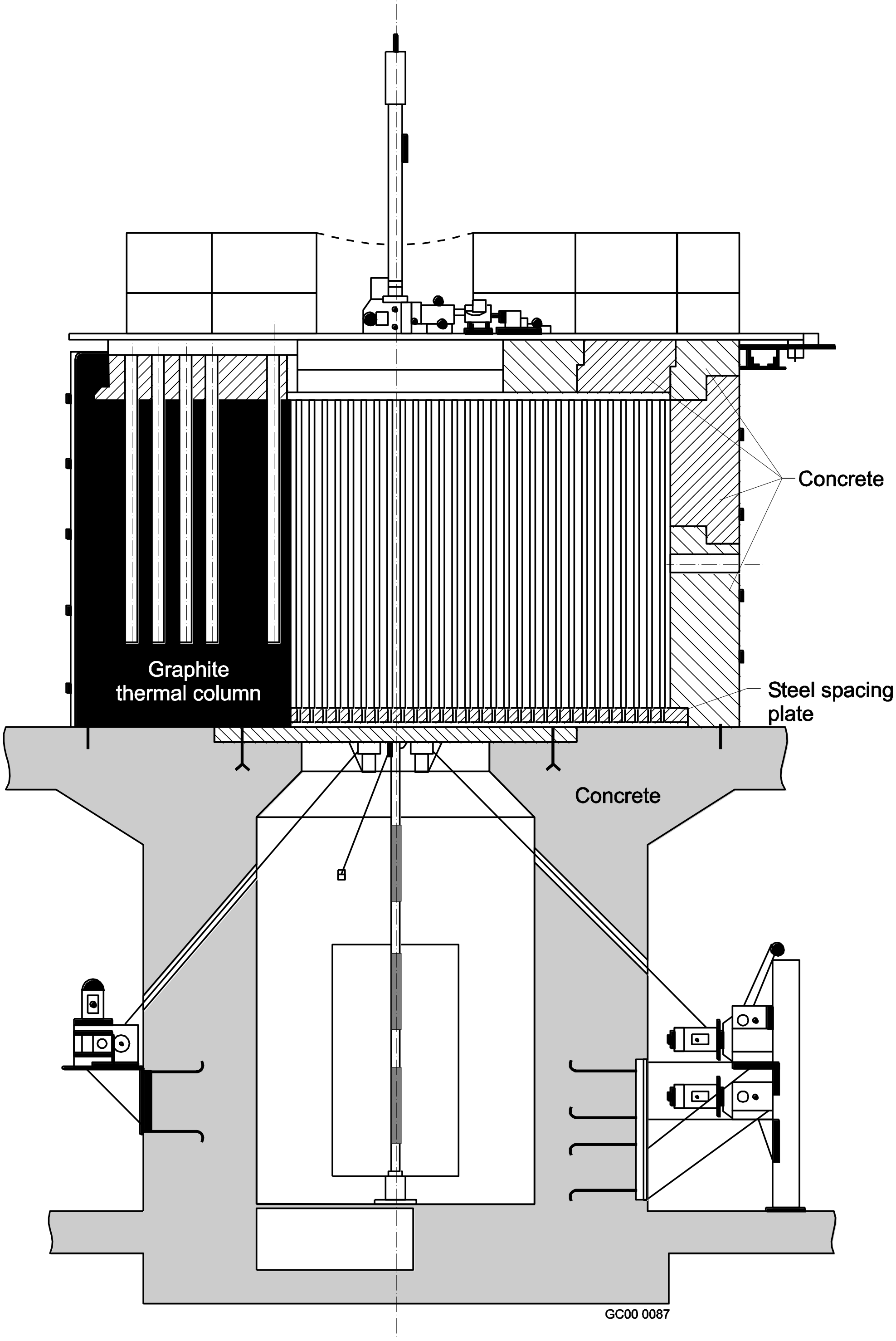

FIG. 3. BFS-1 facility (vertical cut).

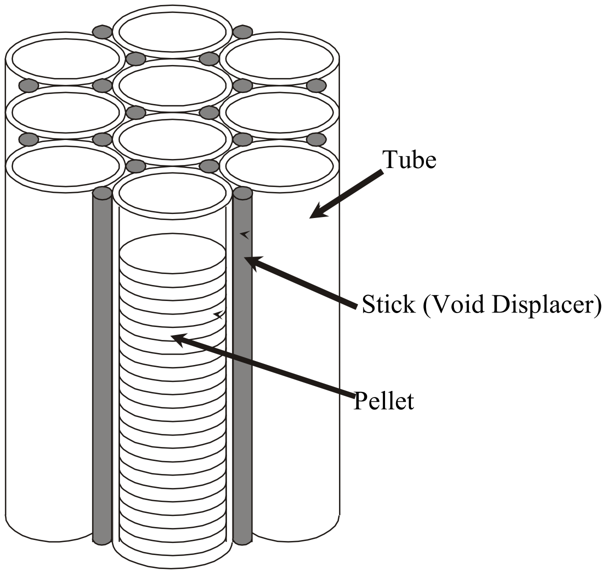

FIG. 4. Sample rod structure of the BFS assemblies.

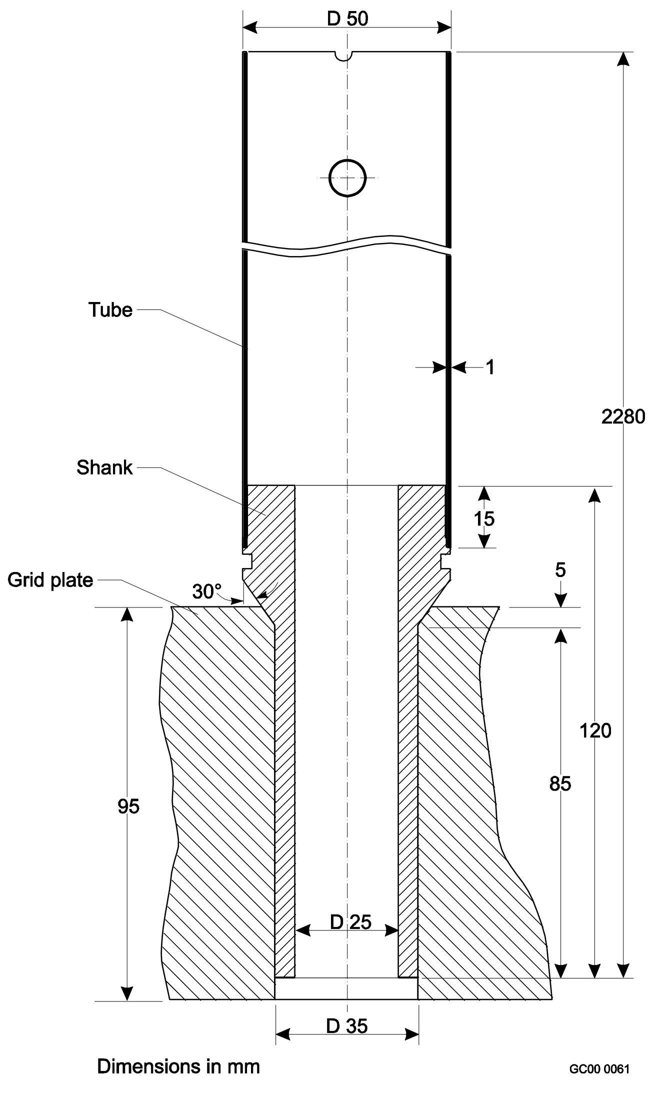

FIG. 5. Tube fixed in grid plate.

Two types of neutron detectors are used for reactor control, namely KNK-59 ionization chambers having gamma compensation, placed in peripheral tubes of the lattice, and SNM-11 boron fluoride counters, placed in special holes in the concrete masonry (see Fig. 2). Five out of the eight KNK-59 chambers are used as reactor control-system detectors (two are used for power-level current channels related to the reactor safety system, two are used for automatic power-control channels, and one for a control-channel and power-doubling-time safety system). A few tube locations in the vicinity of these chambers are filled by polyethylene cylinders of the same diameter. The three remaining KNK-59 chambers are used in the experiments to determine reactivity effects. SNM-11 neutron counters are used for reactor control during the approach to criticality.

2.2. EXPERIMENTAL POSSIBILITIES OF BFS-2

For carrying out calibration measurements requiring the thermal spectrum of neutrons, the critical facility is equipped with the graphite column which is installed closely to the tubes in the vessel. The walls of the thermal column are made of double steel sheets (with thickness 7 mm each) with a 1 mm cadmium sheet between them.

For simulating in-vessel storages, shielding and screens, there is so called metal column in the vessel part opposite to the graphite column. The tube lattice pitch in the metal column is the same as in other parts of the vessel. There are 1330 tubes in this column. Some special cassettes can be installed in the metal column for an extraction of a horizontal neutron beam.

For loading (unloading), changing positions, lifting (lowering) or oscillating fuel and absorber rods in the core as well as for installing detectors in different positions at a power level, the facility is equipped with a coordinate manipulator whose load rating is 100 kg. The grip of the manipulator is guided automatically with necessary accuracy on a pre-selected coordinate for any of 9460 tube positions. Being operated in a measuring regime, the manipulator is controlled by a computer.

At the storages of BFS facilities there are the large amount of fissile materials (metal and dioxide of uranium 36% and 90% enrichments, weapon and reactor grade metal plutonium, about 8 tons), about 280 tons of fertile materials, 120 tons of structural materials (stainless steel, Al, Ni, Nb, Zr, C, B_4C, Al_2O_3) and 200 tons of coolant materials (sodium, lead, lead-bismuth). All the reactor materials are in form of pellets with diameter 47 mm and 10–0.1 mm thickness. Some of the pellets are covered by Al or SS. The same pellets are used for the BFS-1 as well as BFS-2 assemblies.

The possibility of BFS critical facilities permit to investigate not only the models of reactors with fast neutron spectrum, traditional for these installations, but using the pellets of polyethylene as slow-down material, the advanced reactor models LWR type also, where the neutron spectrum is essential thermal.

3. RECENT ACHIEVEMENTS

At the BFS facilities the models of the BOR-60, IBR-2, BN-350, BN-600, BN-800, BN-1200, BREST-300, CEFR(Chinese fast Na cooled), KALIMER (Korean fast Na cooled), SVBR-100, MBIR-100 core designs were investigated and this is besides the plenty of the experiments on testing and of the cross-section libraries (benchmarks).