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Report on availability and future plans of MTRs and consistency with JHR roadmap

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List of abbreviations

ASTM	American Society for Testing and Materials
BR2	Belgian Reactor 2
CEA	Commissariat à l'énergie atomique et aux énergies alternatives
CSA	Coordination and Support Action
CVR	Centrum výzkumu Řež
EC	European Commission
EFPD	Effective Full Power Day
EJP	European Joint Programme
EU	European Union
HFR	High Flux Reactor
JHOP	Jules Horowitz Operation Plan
JHR	Jules Horowitz Reactor
LVDT	Linear Variable Differential Transformer
LWR	Light Water Reactor
MTR	Materials Test Reactor
MOX	Mixed oxide fuel
NRG	Nuclear Research & consultancy Group
OECD	Organisation for Economic Co-operation and Development
ORIENT NM	Organisation of the European Research Community on Nuclear Materials
PWR	Pressurized Water Reactor
PSF	Pool Side Facility
R&D	Research & Development
SRA	Strategic Research Agenda
VVER	Vodo-Vodyanoi Enyergeticheskiy Reaktor (Water-Water Power Reactor)
WP	Work-package

Summary

As part of ORIENT-NM, Task 4.5 Interaction of the EJP with infrastructure and facilities, this report examines the current status of Material Test Reactors (MTR) in Europe, their current and future planning, and consistency with the JHR Roadmap.

MTRs have a critical role in Europe for the fuel and material testing capability to support the existing nuclear fleet as well as to enable R&D for the future nuclear developments, particularly SMRs. Material and fuel testing has been recognized as one of the potential bottlenecks to bring the new reactor designs to market in the near future.

Potential availability of MTRs in Europe is decreasing due to the high cost and lack of funding for new projects. The overall number of MTRs in Europe and the world has significantly decreased in the last decades. It is expected that the JHR under construction at Cadarache, France, will at least partially resolve these issues once it is completed and operational. Unfortunately, there is a significant delay of the JHR project, already leaving a gap between the operation period of the OSIRIS reactor, which was the most recent European MTR to have been closed in December 2015, and JHR. Based on the latest JHR project agenda, its commissioning is now also coming rather close to the expected closure period of other MTRs in Europe. By the current JHR roadmap, full fleet of test capacity and research programmes will be available from 2034 onwards with first criticality currently foreseen in the beginning of the 30’s.

Introduction

This report is part of the deliverables for Task 4.5 Interaction of the EJP with infrastructure and facilities.

Strategic Research Agenda (SRA) on nuclear materials needs to take into consideration the current and future availability of the proposed facilities and infrastructures. The most important facilities are those that allow exposure of the materials to the conditions expected in nuclear reactors of current and future generations, together with those that allow their subsequent analysis and characterisation. In this respect, MTRs, hot cells and loops or other equipment to study the compatibility of materials with specific fluids, together with a wide variety of mechanical property testing and microstructural characterisation facilities are the obvious target.

Availability and future plans of MTRs

MTRs have a critical role in Europe for the fuel and material testing to support the existing nuclear fleet as well as R&D for the future nuclear developments, particularly SMRs. Their potential availability is decreasing due to the ageing of the existing ones, and limited funding for the future ones. Availability for testing of materials is also, in some cases, competing with the production of medical isotopes, thereby further reducing their experimental capacity.

It is expected that the JHR will at least partially resolve these issues once it is built and operational. Given the current situation it is of particular importance the consistency of the SRA on nuclear materials with the JHR roadmap.

Current European MTRs, their capabilities and plans for the future, are listed below in alphabetical order.

Belgian Reactor 2 (BR2), Belgium

The BR2 material test reactor is located at SCK CEN’s technical and operations site in Mol, Belgium. With its nominal power of 125MW and unique adaptable core configuration, the BR2 is one of the most powerful and flexible material test reactors in the world. Since its start-up in 1962 the reactor has operated on highly enriched metallic uranium fuel with pressurised water as coolant. The core moderation is done by a combination of light water and metallic Beryllium. This combination of materials and the unique geometrical design provides a compact core with high neutronic performance combined with maximal manoeuvrability and accessibility.

The neutronic performance of the light water cooled, beryllium moderated core offers a wide range of neutron fluxes for experiments:

- At regular operating power (55 to 65 $MW_{thermal}$), the total flux in the central core region reaches 10^{15} n/cm²s. This flux can be highly thermalized in the central flux trap, yielding thermal flux levels of 10^{15} n/cm²s, while at the peripheral reflector channels, flux levels go down to 7×10^{13} n/cm²s.
- Fast neutron flux irradiation positions are available in the central cavity of fuel elements or irradiation channels surrounded by fuel elements. The fast flux (E>1 MeV) with standard fuel elements ranges from 3×10^{14} down to 5×10^{12} n/cm²s.

As the reactor is cooled by pressurized (1.2 MPa) water, the allowable heat flux on the fuel surface, exposed to the nominal primary flow, is 470 W/cm² for the driver fuel, up to 600 W/cm² in experimental-set ups cooled by the primary water. The fuel elements are tubular, with 6 concentric tubes, each made of 3 circular formed fuel plates. In the centre of the fuel elements, there is sufficient space for an irradiation device. The fuelled zone is 762mm long, the reactivity control of the load occurs through the addition of burnable poisons in the fuel meat and the vertical motion of the shim/control rods. The driver fuel elements are reloaded typically for 5 or 6 cycles, accumulating up to 60% of average burn-up.

The position and number of control rods and fuel elements are not fixed by design and therefor adaptable to the requirements of all experiments in a reactor cycle. For a typical configuration, as shown in Figure 1, between 30 and 35 driver fuel elements are loaded, together with 6 control/shim rods. A regulating rod and eventually a safety rod are added. Such configuration can typically be operated 21 to 28 days at a reactor power between 55 and 70MW. The standard type of fuel element used is the six plate element (with F1 type of irradiation position in its centre). Upon request of experimenters, 5 plate elements can be made available (F2). Historically, also other types of elements have been used and can be refabricated for dedicated experiments.

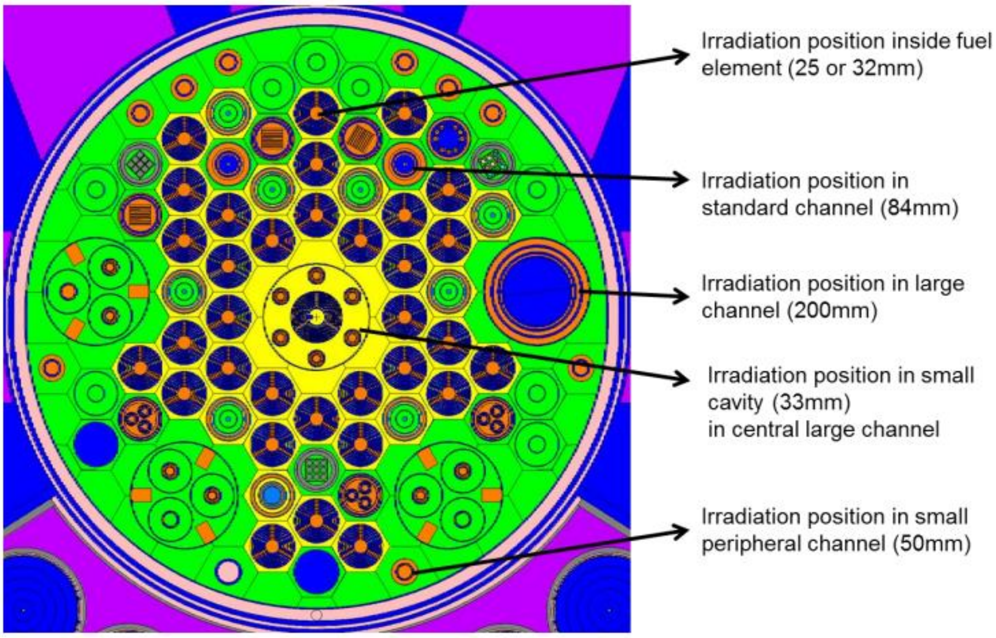


Figure 1 Cross section of BR2 reactor at mid plane with indication of irradiation channel types.

Fuel irradiation

Pressurized water capsule (PWC-CD) for fuel pin irradiation

The pressurized water capsule for fuel irradiation is an instrumented capsule that can be used for base irradiation of fuel pins up to 1m long, with on line power monitoring and control of the cladding temperature by setting the water pressure in the capsule. The device can also be used for transient testing, either by loading a mobile absorber in the vicinity (multiple transients with small amplitude) or by varying the overall reactor power (large single transients). The setup of the device is such that fuel pin failure can be tolerated. Eventually, a fuel pin with limited instrumentation can also be loaded in the device.

MTR plate irradiation

Material test reactor fuel plates can be irradiated in the primary water of the BR2 reactor in different ways. The most straightforward for flat plate irradiations is the use of the so-called FUTURE basket. Up to 4 flat fuel plates can be loaded in this basket (in its current design), replacing a standard fuel element of the reactor (standard channel S). Fuel plate failure can be tolerated up to the contamination limit of the primary water. The environment of the basket is adapted in order to achieve the desired power level in the basket. The basket allows the loading of activation dosimeters. Four devices (1 for 2 plates and 3 for 4 plates) are currently available, but others can be constructed relatively quickly to accommodate plate geometries or experimental requirements.

Material irradiation

BAMI capsules

The BAMI capsules are un-instrumented capsules that can be loaded in irradiation positions inside fuel elements (F1) or in standard channels (S). Up to 8 capsules of diameter 15mm (F1) or 25mm (S) are loaded in one irradiation position. The capsules can be loaded in the primary water flow (entire cycle irradiation) or in a thimble tube device (flexible irradiation time). The capsules can be open to the water (irradiation temperature <100°C) or can be gas filled, in which case the irradiation temperature is determined by the irradiation position, the mass of the samples, the composition of the gas (typically He) and the spacing between the samples and the cold wall of the capsule. The BAMI capsules offer the lowest cost and lead time for irradiating structural material samples.

ROBIN

The ROBIN device is loaded in a thimble tube, inserted in a standard channel (S) (flexible irradiation time). The specimens are encapsulated in closed needles (9 needles of diameter 11mm); the irradiation temperature is determined by the design of the needles and is controlled by adjusting the water flow in the thimble. In order to avoid boiling, the positioning of the experiment is limited to relatively low flux positions (thermal and fast flux 0.7 and 0.3×10^{14} n/cm²s, respectively). The temperature in the samples is monitored by adding an instrumented dummy capsule with identical design as the specimen needles.

LIBERTY

The LIBERTY device is also loaded inside a thimble tube in a standard channel (S). The main difference with the ROBIN device is that there can only be 5 sample containing needles, but the needles are larger in diameter (16mm inside) and can be equipped with active temperature control by integrated electrical heating and temperature measurement. In this way, specimens can be preheated before the start of irradiation. The fluxes in LIBERTY are similar to the ones in ROBIN.

MISTRAL

The MISTRAL device is inserted in a 5 plate fuel element (F2) and offers active temperature control in a boiling water environment. The MISTRAL device is designed to irradiate a large number (87) of miniature specimens (5mm diameter or $3 \times 4 \text{mm}^2$ cross section and length of 27mm) in stable temperature conditions (160°C-350°C) with medium to high fast flux level (up to 2.5×10^{14} $\text{n/cm}^2\text{s}$, E>1MeV). The rig can be reloaded, so lead times for experiments are relatively limited and depend largely on the rig's availability. Of the 87 specimens, 26 are located in the zone having over 90% of the maximum flux in the rig. The irradiation temperature is monitored by measurement inside dummy specimens and the irradiation temperature is fixed by setting the saturation pressure in the rig and sustaining boiling by electrical heating if the nuclear heating is insufficient to maintain boiling (during start up and shut down of the reactor).

HTHF

For irradiating materials at maximum fast flux (2.8×10^{14} $\text{n/cm}^2\text{s}$, E>1 MeV) in a standard fuel element (F1) and controlled temperature up to 1000°C, a gas filled capsule (diameter 21 mm) with active temperature control was designed. This capsule is constructed of graphite, allowing high temperature stability and heat evacuation under the highest fluxes available in the BR2 reactor. The design is adjusted according to the experimental needs (specimen number and geometry, temperature range) and the capsules are single use. However, capsule cost and experiment lead time are controlled by the generic design and the reuse of the out of pile control equipment. The availability of several driver fuel elements with comparable neutronic conditions allows for the simultaneous irradiation of HTHF devices, for example to compare different materials or generate data at different irradiation temperatures.

The High Flux Reactor (HFR), The Netherlands

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 and the world's largest producer of medical isotopes. The HFR is owned by 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.

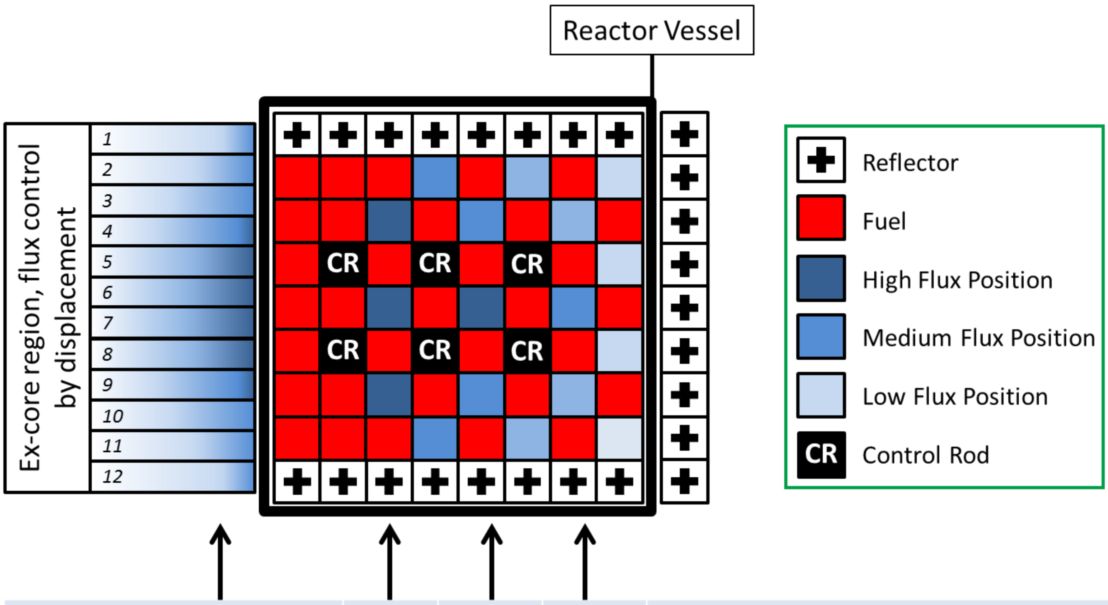
Together with the hot cells of NRG at the Petten site, the HFR has provided for over five decades, an integral and full complement of irradiation and post-irradiation examination services as required by current and future R&D within the field of nuclear energy and healthcare for industry and research organisations. The HFR has a longstanding tradition of multilateral collaboration in combination with 'frontier breaking' irradiation programmes HFR within the domains of transmutation, HTR, graphite research and Molten Salt Reactors.

The HFR has 17 in-core and 12 poolside irradiation positions (Figure 2). The HFR uses low-enriched uranium U_3Si_2 fuel and operates at a constant power of 45 MW and uses light water both for cooling and moderation. The current cycle length is 31 operation days with nine cycles per year, for a total of ~279 full power days/year. 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 current and advanced reactor technologies. 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 31 mm) irradiation positions. Maximum flux values are in the range $2.6*10^{14}$ $n.cm^{-2}s^{-1}$ (thermal flux) and $1.8*10^{14}$ $n.cm^{-2}s^{-1}$ (fast flux, E>1 MeV); typical damage and fuel power values are 6 dpa/year and 200-400 W/cm (Figure 2). The flux profile of the HFR is predictable and stable, showing a small upward shift each cycle that can be compensated by vertically translating irradiation devices. Together with gas gap conductivity control, this feature provides strong control over sample temperatures.

Basic irradiation tests are performed either in irradiation facilities with semi-stagnant gas or in closed capsules/ampules. Advanced irradiations are operated with advanced gas systems that enable accurate control and analysis of the sweep gases. In the past, rodlets have been irradiated in stagnant sodium. Maximum irradiation temperatures in the 1000-1200°C range can be achieved. All irradiation facilities are instrumented with up to 24 thermocouples (48 for larger capsules). Irradiation facilities with bellows that apply a fixed load on the samples under irradiation, are used to determine creep under irradiation conditions. LVDT-based sensors are used to measure dimension changes and pressure in-pile.

In addition to the in-core positions the HFR has a pool-side facility. PSF experiments can be moved to and from the reactor core using a trolley system. This allows for reliable power control and power cycling, without significantly affecting other operations in the HFR core.



0-3, location dependent	5-8	3-5	1-3	Material DPA rate (DPA/year)
0-700, displacement controlled	n/a	500-700	300-400	Linear heat rate (W/cm fresh LWR fuel)

Figure 2 Schematic view of the HFR core, with indications of damage build-up for materials and linear heat rate for fresh LWR fuel. The pool-side facility (to the left) allows for power control and power cycling operations.

LVR-15, Czech Republic

Irradiation facility

The LVR-15 reactor is a multi-purpose research reactor. It provides a high-density neutron flux, allowing for a long-term representative material research of the Gen II, III and IV reactor materials as well as the potential material for fusion reactors. Thanks to variable configuration it is possible to simultaneously conduct several experiments at different positions within the reactor core and beyond it including horizontal neutron beam. Horizontal channels and pneumatic rabbit system are used for neutron scattering experiments and activation analysis for nuclear analytical investigations and for fundamental nuclear physics studies. The common experimental in-reactor equipment (e.g. vertical irradiation channels, “Chouca” irradiation rig or others) is used for series of long term irradiation experiments with the possibility to change the irradiation temperature, but in many cases the irradiation rigs are developed and manufactured based on specific requirements for specific experiment and user needs. Loop technology is used in CVR to investigate the behaviour in the environment corresponding to the reactor environment for the investigations of the material – coolant interaction, coolant thermohydraulic studies, impact of impurities on coolant chemistry and other purposes, covering wide range of reactor systems.

LVR-15 is a light water tank-type research reactor placed in a stainless-steel vessel under a shielding cover. It has forced cooling, IRT-4M fuel and an operational power level up to 10 MWt. Reactor operations run in cycles, typically 6-7 per year. Usually the cycle lasts for 30 days, followed by an outage lasting for approximately 20 days for maintenance and fuel reloading. Demineralised water is used as a moderator and coolant. A reflector is composed of a water, or beryllium block, depending on the operation configuration.

A cross section of LVR—15 reactor is shown in Figure 3 and its typical core in Figure 4.

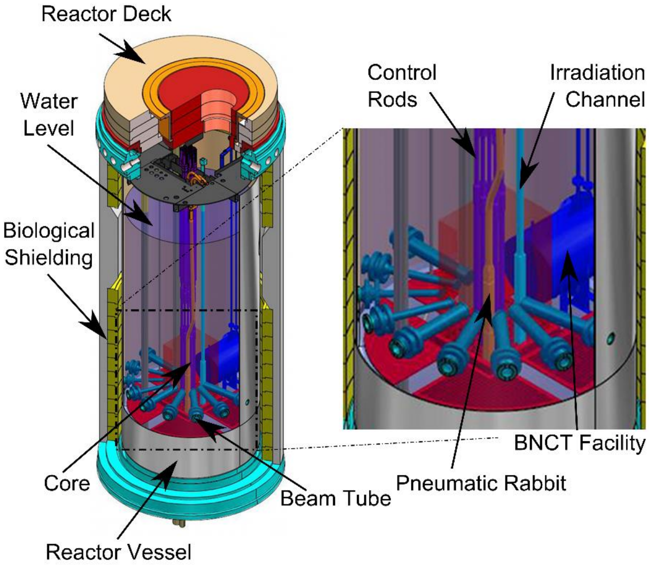


Figure 3 LVR-15 reactor vessel cross-section.

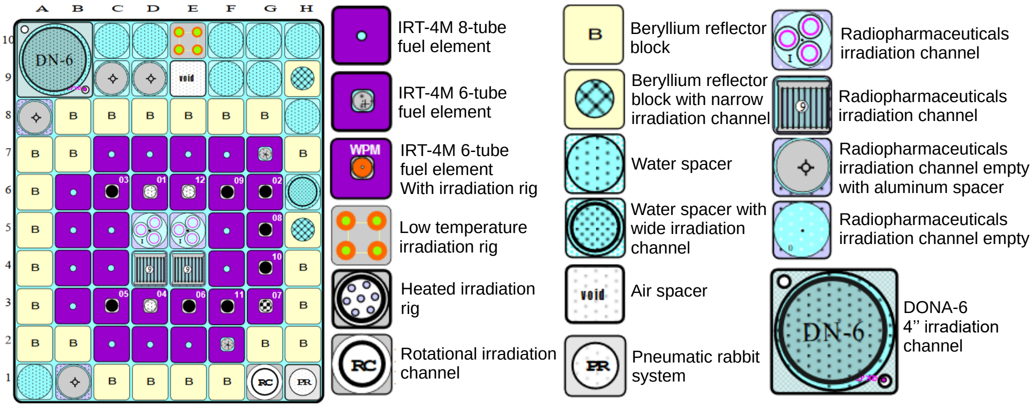


Figure 4 Typical core layouts with in-fuel irradiation rig in position D7 (left) and CHOUCA rig in position D2 (right).

The general operation parameters are:

- Maximal thermal power: 10 MWth;
- Maximal available thermal neutron flux: $2x10^{14}$ $n \times cm^{-2}s^{-1}$;
- Maximal available fast neutron (> MeV) flux: $6x10^{13}$ $n \times cm^{-2}s^{-1}$;
- Pressure: atmospheric;
- Coolant Temperature: max. 56°C.

During 2020, CVR extended LVR-15 operating licenses which is now without time limit. Reactor operation until at least 2030 is expected and CVR has a plan to operate even beyond.

Neutron spectrum and its monitoring

For all calculations, the RSICC CCC-810 package containing MCNP5/MCNPX/MCNP6 Monte Carlo Codes and ENDF/B-VII.0 and ENDF/B-VII.1 nuclear data libraries are being used. Based on these neutron spectra calculations, sample-averaged dpa is also calculated according to the Kinchin-Pease model using the ENDF/B-VII.1 cross-section. The Ed values within the Kinchin Pease model are selected according to the ASTM committee Standards E521.

Irradiation rigs can be equipped with a carrier with activation detectors for neutron flux and fluence determination in shape of thin foils or wires. Following basic neutron fluence monitors and reactions of interest are often used:

Iron	$^{54}\mathrm{Fe}(n,p)^{54}\mathrm{Mn}$
Iron	$^{58}\mathrm{Fe}(n,\gamma)^{59}\mathrm{Fe}$
Nickel	$^{58}\mathrm{Ni}(n,p)^{58}\mathrm{Co}$
Titanium	$^{46}\mathrm{Ti}(n,p)^{46}\mathrm{Sc}$
Niobium	$^{93}\mathrm{Nb}(n,n')^{93m}\mathrm{Nb}$
Cobalt	$^{59}\mathrm{Co}(n,\gamma)^{60}\mathrm{Co}$

If necessary, additional set of activation detectors may be used as well. Certified material or their alloys from Alfa Aesar, USA are be used. Monitors sets design, measurement of their activity as well as fluence determination are performed in accordance with the ASTM Committee E10 Standards (E261, E262, E263, E264, E481, E482, E526, E844, E944, E1006 and E1297).

Irradiation rigs and loop systems

For material radiation degradation studies the reactor operates standard and non-standard irradiation rigs. Single purpose irradiation rigs are designed, produced and tested in-house according to the needs of the irradiation experiment. Irradiation rigs provide irradiation in an inert gas atmosphere He, He/N_2, He/Ar, online temperature monitoring and temperature control. Fluence evaluation is realized during irradiation trough computational simulations and after the irradiation by the evaluation of passive neutron fluence monitors. Rigs can be designed to fit into reactor positions with the highest available fast neutron fluxes to provide as highest radiation damage as possible in the shortest time.

For investigation of degradation and corrosion behaviour of structural materials' mechanical properties under irradiation and PWR/VVER water chemistry and thermal-hydraulic conditions, RVS-3 in-pile loop is used. CVR has also long-term experiences with in-pile loop systems under Boiling Water Reactor water chemistry and thermo-hydraulic conditions. Currently there are two out-of-pile loops dedicated for material studies under Gen IV. Reactor conditions – Super Critical Water Loop and High Temperature Helium Loop, with a plan for their future introduction into the LVR-15 core.

Consistency of EJP SRA with JHR roadmap

The Jules Horowitz Reactor (JHR), with a power output of approximately 100 megawatts and planned service lifespan of around 50 years, under construction in Cadarache, France, will be the key material test reactor in Europe in the future. JHR will be constructed by an international consortium consisting of European and non-European members.

The JHR is a materials testing reactor, and is designed to be adaptable for a variety of research uses by nuclear utilities, nuclear steam system suppliers, nuclear fuel manufacturers, research organisations and safety authorities. The reactor's versatile modular design allows it to accommodate up to 20 simultaneous experiments. Its instrumentation allows previously unavailable real-time analysis to be performed. Its primary uses will be research into the performance of nuclear fuel at existing reactors, testing of materials used in reactors, testing designs for fuel for future reactors and the production of radioisotopes for use in medicine.

JHR main features

The design of the reactor provides irradiation locations situated inside the reactor core with the highest

ageing rate and irradiation locations situated in the Beryllium reflector zone surrounding the reactor, with the highest thermal flux.

Numerous locations are implemented (up to 20 simultaneous experiments) with a large range of irradiation conditions (Figure 5):

- 7 in-core locations of small diameter for experimental devices up to 33.1 mm diameter (101, 105, 203, 207, 211, 303, 307, 313)
- 3 in-core locations of large diameter (80 mm) for experimental devices up to 86 mm diameter (103, 211, 301)
- 16 fixed reflector locations for experimental devices up to 97 mm diameter (two of them, C311 and C413, will be used for surveillance specimens monitoring irradiation ageing of the JHR vessel material)
- 1 fixed reflector location for an experimental device up to 200 mm diameter (P322)
- 4 displacement devices located in water channels through the Beryllium reflector for experimental devices up to 100 mm diameter (T5, T8, T10, T12)
- 4 additional displacement devices for Moly production (T0 to T3)

During the first years of operation, CEA targets to operate the reactor 180 days per year, 15% at 100 MW and 85% at 70 MW. A mean reactor cycle is expected to last about 34 operating days corresponding to 25 EFPD.

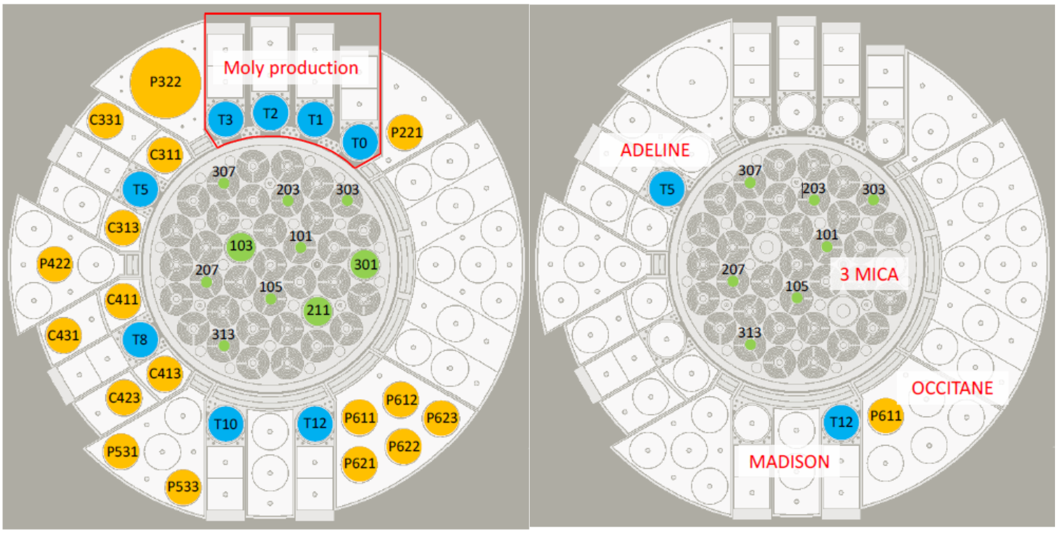


Figure 5 Left: Experimental locations in the JHR core and reflector. Right: Locations of the experimental devices available for the JHR start-up.

The start of operation of the JHR (first criticality scheduled for the beginning of the next decade) will happen in two fleets for the experimental capacity. The first fleet will contain five different testing devices: MADISON for normal operation, ADELINE for off-normal operation, OCCITANE for pressure vessel steel experiments and 3 MICA material test capsules for more conventional material testing. In addition, there will be Non-Destructive evaluation techniques available for different purposes.

MADISON (Fuel device)

- Designed to carry out irradiations of LWR fuel samples during which no clad failures are expected. Consequently, the experimental conditions correspond to normal operation of power reactors (steady states or slow transients that can take place in nuclear power plants).

As an example fuel sample instrumentation for a 4 rods irradiation rig:

- For each fuel rod:
    - 1 temperature measurement: fuel temperature or clad temperature
    - + 1 LVDT measurement: fuel clad elongation or fuel plenum pressure

Example of fuel sample instrumentation for a 2 rods irradiation rig

- First fuel rod
    - Fuel stack elongation
    - + 1 LVDT measurement: fuel clad elongation
- Second fuel rod
    - 1 temperature measurement: fuel temperature or clad temperature
    - + 1 fission product measurement (acoustic sensor)
    - + 1 LVDT measurement: fuel plenum pressure
- The first operational irradiation rig will be designed to host 9.5 mm UO_2 fuel rods with 5% ^{235}U and 60 cm length. Smaller or higher rod diameters (from 7 to 12.5 mm) and higher ^{235}U content (or MOX fuel) would be manageable but require additional studies.
- PWR conditions (155 bar and 320°C for the inlet temperature).

ADELINE (Fuel device)

- Able to hold a single experimental fuel rod from all LWR technologies to reproduce various experimental irradiation scenarios in which clad failure is either a risk or an experimental objective.
- First version of the ADELINE device is mainly dedicated to power ramp testing. In particular, the design is optimized to provide a qualified thermal balance (with a targeted 5-6% precision) and good accuracy on the clad failure instant and, consequently, a good knowledge of the linear power inducing the failure.
- A typical PWR power ramp sequence is made of the following phases:
    - A low power plateau (from 12 hours up to 7 days) with control of clad surface temperature while the sample linear heat rate is controlled between 100 and 200 W/cm, depending on end-user request;
    - A linear power ramp at a continuous rate ranging between 100 and 700 $W.cm^{-1}.min^{-1}$;
    - A high power plateau that may last 24 h (performed if the cladding has not failed during the power transient).
- The first irradiation rig is designed to host 9.5 mm UO_2 fuel rods with 5% ^{235}U and 50 cm length. Smaller or higher rod diameters (from 7 to 12.5 mm), higher length (up to 60 cm) and higher ^{235}U content (or MOX fuel) would be manageable but require additional studies.
- For a reactor power of 100 MW, the limit for the power at maximum flux plane is the thermal-hydraulic limit of 620 W/cm whatever the burn-up (0 to 120 GWd/t).

MICA (Material device)

- Designed to irradiate structural materials in the core of the JHR, within a fuel element. Its typical temperature range is between 280 and of 450°C for the samples. Seven different locations are available for MICA devices: two in the first ring, two in the second ring and three in the third ring of the JHR core.
- The experimental samples are loaded in a sample holder at the centre of the MICA test device. The available space in the MICA is limited by the internal stainless steel tube of diameter 24 mm. The sample holder is immersed in liquid metal (Na-K eutectic) which ensures optimal thermal homogeneity with its high thermal conductivity.
- Nuclear heating in the JHR core transfers energy to the MICA and thus to the experimental load, which is foreseen to be between 15 and 35 kW depending on the reactor power, the location of the MICA in the JHR core and the experimental load in the MICA. Additional energy is provided by heating elements embedded in a sprayed aluminium coating around the internal stainless steel tube.
- Several thermocouples and dosimeters can be integrated to the sample holder to monitor temperature and neutron fluence at different locations.

OCCITANE (Material device)

- 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 and consequently the lifetime of the reactor itself.
- CEA is designing a hosting system named OCCITANE (Out-of-Core Capsule for Irradiation Testing of Ageing by Neutrons), which allows irradiations in an inert gas at least from 230 to 300°C.
- OCCITANE will be located in P611 in the first ring of JHR reflector. The neutron characteristics will be as follows (best-estimate values at maximum flux plane, 100 MW, 27% ^{235}U, core at equilibrium in mid cycle):
    - Fast flux ($E > 1$ MeV): about 8 10^{12} n/cm².s
    - Fast flux ($E > 0.1$ MeV): about 2 10^{13} n/cm².s
    - mdpa/EFPD: 1.0
- Neutron fluxes and dpa should be multiplied by 0.7 for a reactor power of 70 MW.
- The neutron spectrum ratio $Rs = \Phi(E > 0.1 \text{ MeV}) / \Phi(E > 1 \text{ MeV})$ and the nuclear heating in the samples can be adapted with neutron and gamma shields.
- Various kind of samples (creep, tensile, Charpy and microstructure) can be irradiated if their size does not exceed 30 x 62.5 mm². Samples can be stacked on top of each other on to 60 cm height with a damage axial gradient due to the variations of the fast neutron flux. Between each cycle, the sample holder is rotated 180° in the device in order to homogenise damage in the experimental samples.
- The multi-zone furnace controls the required irradiation temperature between 230 and 300°C and compensates the axial thermal gradient due to the above-mentioned axial nuclear gradients.
- The associated instrumentation includes at least thermocouples and dosimeters as close as possible to the samples.

JHR Roadmap

In the second fleet of experimental devices, e.g. a test loop for corrosion studies, a loop for accident simulations as well as additional fuel capsules will be available.

The JHR working groups are currently making the prioritization of the experiments for both fuels and materials.

In parallel with the ORIENT NM project, the Jules Horowitz Operation Plan project (JHOP2040, CSA) is on-going. The goal of this project is to make operation planning of the JHR for the first 15 years. This project will be realized between 2020-2023. In JHOP2040, the main targets are to describe how the Euratom's 6 % access rights can be utilized in the most effective manner for the benefit all EU countries interested in performing nuclear materials or fuels tests in the JHR. More information on the JHR testing capabilities will be given in the deliverables of the JHOP2040 project.

JHR is aiming to have the first reference operation plan available around two years before the start of research programmes (Figure 6). This reference operation plan needs to reviewed yearly but it shall give the main frame according which the loading of the reactor will be done.

At the same time when planning for the EC is done, the JHR community is actively doing the pre-JHR era research for the JHR related topics in e.g. the OECD's Framework for Irradiation Experiments framework. The research projects established there serve the future needs of the JHR and give an opportunity to design and validate some the testing methods also meant for the JHR.

In addition to the existing developments, it is also foreseen that some new technologies/multiplication of the existing ones for the JHR will be needed. Moreover, e.g. the VVER technology user group within JHR should get running in some years.

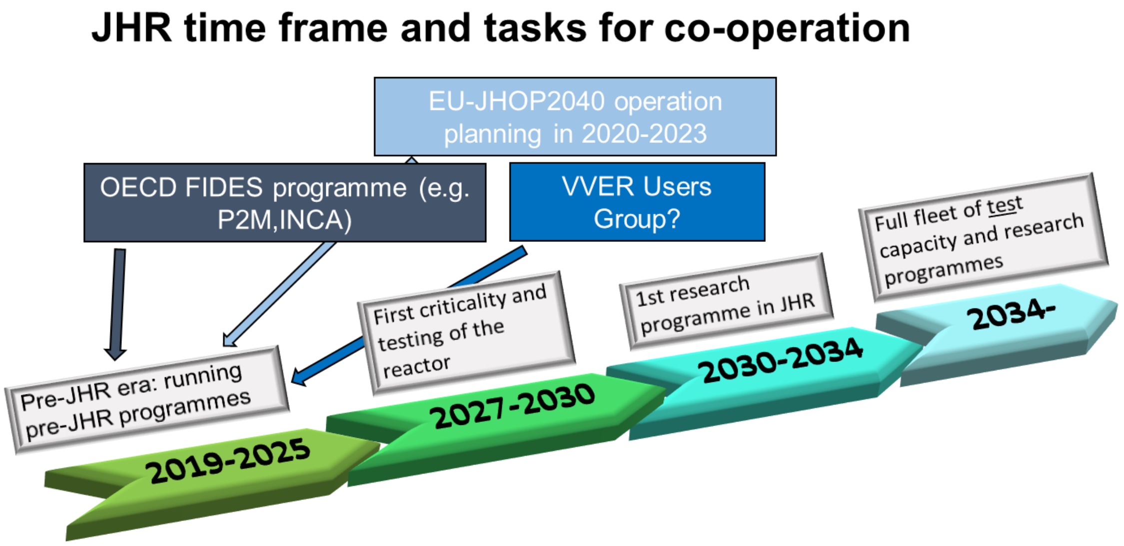


Figure 6 JHR Time frame

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