ORIENT

ORIENT- NM

Organisation of the European Research Community on Nuclear Materials

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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
CEP	Co-funded European Partnership
CSA	Coordination and Support Action
CVR	Centrum výzkumu Řež
EC	European Commission
EFPD	Effective Full Power Day
EJP	European Joint Programme
ESN	European Stakeholder Network
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
MOX	Mixed oxide fuel
MPa	MegaPascal
MTR	Materials Test Reactor
MW	MegaWatt
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

MTRs have a critical role in Europe for the fuel and material testing to support existing nuclear fleet and 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 the market in the near future.

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 JHR Roadmap.

Potential availability of the MTRs in Europe is decreasing due to the high cost and lack of funding for the new projects. It is expected that the JHR will resolve these issues once when it is built and operational, but that project is suffering from serious delays. By the current JHR roadmap, full fleet of test capacity and research programmes will be available only from 2034. This is why it is important to look at the situation concerning presently operating and other planned facilities.

Introduction

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

Any Strategic Research Agenda (SRA) on nuclear materials needs to take into consideration the current and future availability of relevant 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 generation, 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.

Compared to the draft version, this revision adds information about the MARIA reactor in Poland and the PALLAS reactor in the Netherlands. The chapter about the JHR roadmap has been updated with the latest available information from JHOP2040.

Availability and future plans of MTRs

MTRs have a critical role in Europe for the fuel and material testing to support existing nuclear fleet and 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 future ones. The availability for testing of materials is also, in some cases, competing with the production of medical isotopes, thus further reducing the capacity of the existing facilities.

It is expected that the JHR will resolve these issues once it is built and operational. However, at the moment the JHR is suffering from significant delay. Nonetheless, it is important that the SRA of the partnership on nuclear materials is consistent with the JHR roadmap, and looks at alternative facilities while the JHR is not available. For this reason, the current European MTRs, their capabilities and plans for the future, are described below, listed in alphabetical order.

BR2, Mol, Belgium

The BR2 material test reactor is located at SCK CEN's operational 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, i.e., offering 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 762 mm 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 therefore 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 in the six plate element (with F1 type of irradiation position in its centre). Upon request of experimenters, 5 plate elements can be regularly made available (F2). Historically, also other types of elements have been used and can be refabricated for dedicated experiments.

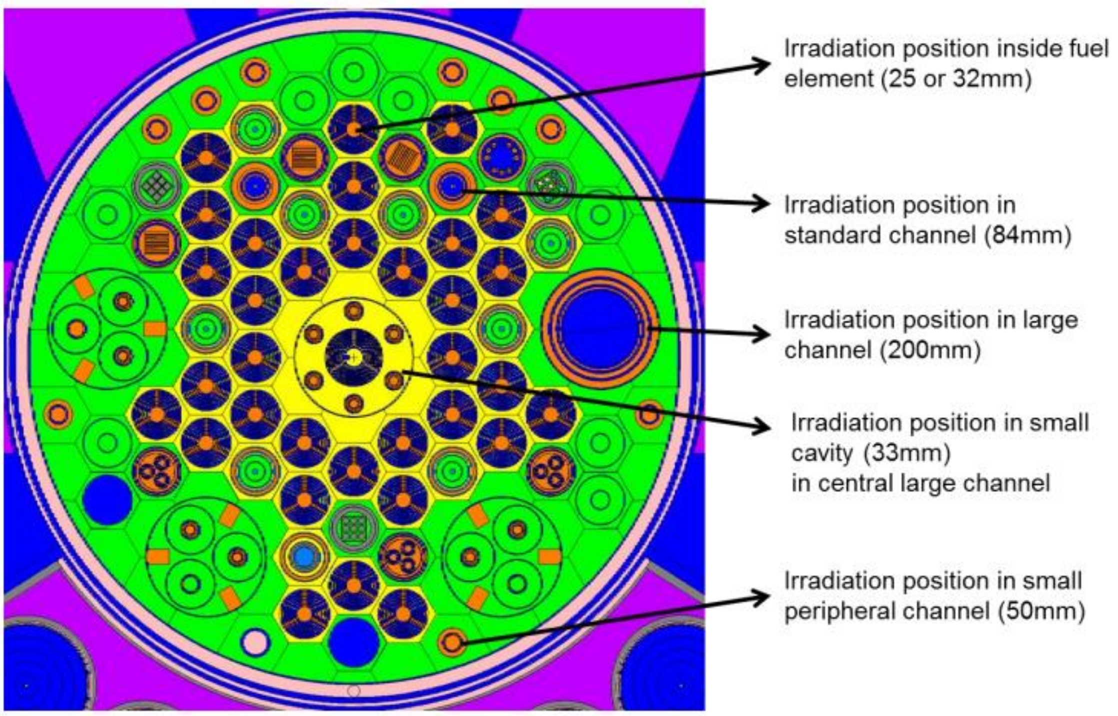


Figure 1. Cross section of BR2 reactor at mid plane with indication of irradiation channel types.

Fuel irradiation

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

The pressurized water capsule for fuel irradiation is instrumented and can be used for base irradiation of fuel pins up to 1 m long, with online 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 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 method 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 quickly to accommodate plate geometries or experimental requirements.

Material irradiation

BAMI capsules

The BAMI capsules are un-instrumented, but can be loaded in irradiation positions inside fuel elements (F1) or in standard channels (S). Up to 8 capsules of diameter 15 mm (F1) or 25 mm (S) can be loaded in one irradiation position, either 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 11 mm); 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 (16 mm 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.

RECALL

The RECALL device is a pressurized water capsule device, loaded in a standard reflector channel (S), with small flow rate. The device allows accurate active temperature control in the range from 250 to 320°C, both before and during irradiation. The device is loaded for the entire reactor cycle and allows 24 standard Charpy V specimens to be irradiated within a homogeneous flux zone (+/-15% axial deviation). The positioning of the device is flexible in order to achieve between 0.05 and 0.15 dpa in steel in one reactor cycle. The device is reusable, offering very short lead times for experiments

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 (5 mm diameter or 3x4mm² cross section and length of 27 mm) in stable temperature conditions (160°C-350°C) with medium to high fast flux level (up to 2.5×10^{14} n/cm²s, E>1MeV). The rig can be reloaded, so lead times for experiments are limited as well as the rig costs. 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} n/cm²s, 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 is 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, Petten, The Netherlands

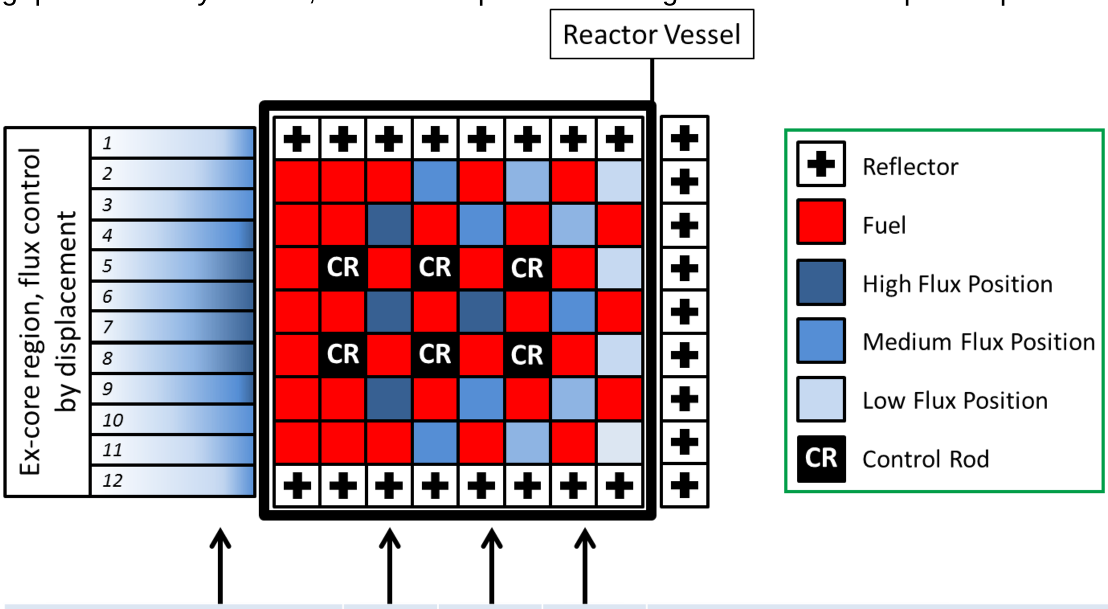
The High Flux Reactor (HFR), located in Petten, in The Netherlands, is one of the most powerful multi-purpose research and test reactors in the world and the world's largest producer of medical isotopes. The HFR is owned by the European Commission (EC). Its operation has been entrusted since 1962 to 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 operations 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 (outside the reactor vessel), is located the Pool Side Facility (PSF).

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 $2.6{\cdot}10^{14}$ $n.cm^{-2}s^{-1}$ (thermal flux) and $1.8{\cdot}10^{14}$ $n.cm^{-2}s^{-1}$ (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.



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.

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, too. 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, which 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 enables reliable power control and power cycling, without significantly affecting other operations in the HFR core.

LVR-15, Rez, Czech Republic

Irradiation facility

The LVR-15 reactor is a multi-purpose research reactor. It provides a high-density neutron flux, enabling research on materials for Gen II, III and IV reactors, as well as on the potential materials for fusion reactors. Thanks to its 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 the pneumatic rabbit system are used for neutron scattering experiments and activation analysis for nuclear analytical investigations, as well as for fundamental nuclear physics studies. The common experimental in-reactor equipment (e.g., vertical irradiation channels, “Chouca” irradiation rig or others) is used for a 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 of materials in the reactor environment, namely interaction with coolants, coolant thermohydraulic studies, impact of impurities on coolant chemistry, and other effects, covering the needs of a 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.

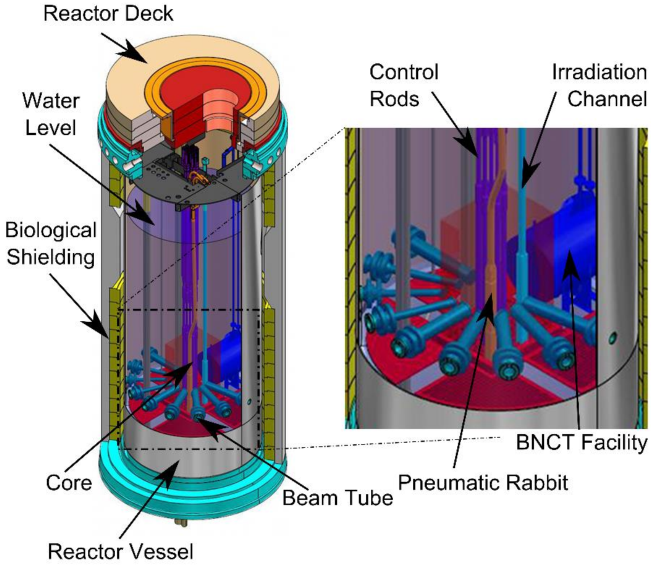


Figure 3. LVR-15 reactor vessel cross-section.

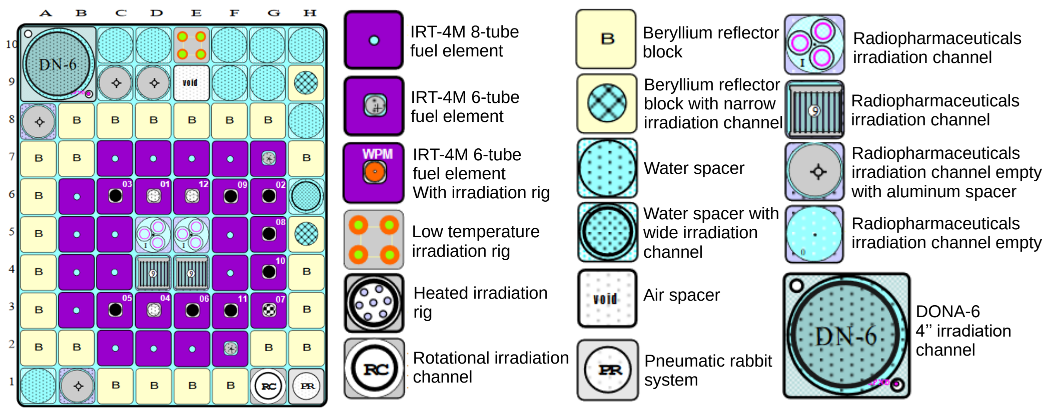


Figure 4. Typical core layouts with in-fuel irradiation rig in position D7 (left) and CHOUCA rig in position D2 (right).

The general operation parameters are:

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

During 2020, CVR extended the LVR-15 operating license, which is now without time limit. Reactor operation until at least 2030 is expected and CVR has plans 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 currently 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 threshold displacement energy (E_d) 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 the shape of thin foils or wires. Basic neutron fluence monitors and reactions of interest are often used:

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

If necessary, additional sets of activation detectors may be used, as well. Certified material or their alloys from Alfa Aesar, USA, are used. Monitor 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 high radiation damage as possible, in the shortest time.

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

MARIA Research Reactor, Poland

The MARIA Research Reactor is located 30 km from Warsaw city centre at the National Centre for Nuclear Research. It is a multi-purpose pool-type reactor with pressurized fuel channels immersed in the reactor pool. It is a high flux reactor of 30 MW nominal power moderated with water and beryllium. Pressurized fuel channels are situated in the open reactor pool in the beryllium matrix enclosed by a lateral reflector made of aluminium canned graphite blocks. Fuel channels are pressurised up to 1.7 MPa at the fuel channel inlet. Three types of fuel are currently used:

- MC-5/485, French production (by Cerca-Areva), with 19.75% enrichment: a dispersion of a uranium silicide alloy powder (U_3Si_2) in Al.
- MR-6/485, Russian production (by Novosibirsk Chemical Concentrates Plant, TVEL), with 19.70% enrichment: a dispersion of UO_2 in Al.
- MR-2/225.7, Russian production (by Novosibirsk Chemical Concentrates Plant, TVEL), with 19.70% enrichment: also a dispersion of UO_2 in Al.

The last one is licensed and tested for the irradiation of material samples in the holder in the fuel central area. In both types, the fuel core is cladded with aluminium. Each fuel element has a form of six (MR) or five (MC) concentric tubes. The thermal and fast neutron flux density may reach $2.5{\cdot}10^{14}$ n/cm²s and $1{\cdot}10^{14}$ n/cm²s, respectively. The reactor has been designed with a high degree of application flexibility. The reactor core has a modular design and its core configuration is fitted to either production or research purposes.

It is anticipated that MARIA will be able to operate until 2050 after several modernizations take place. One of the current challenges is all the preparatory work to license renewal.

The MARIA Reactor core consists of 28 fuel channels and 13 control rods situated in a beryllium matrix. The number of fuel channels may vary from cycle to cycle, while the number of control rods remains constant. The beryllium matrix is enclosed by the graphite blocks forming the reflector. It is also equipped with approx. 26 vertical channels used for irradiation and 8 horizontal channels (7 operational and one still only in conceptual design). The whole core is placed in an aluminium housing, called ‘basket’, located on a table at the bottom of the reactor pool. The top view of the core is shown in Figure 5. The most common operational power of the reactor varies from 18 to 24 MW depending on the cycle needs. In the figure, fuel channels are marked in yellow, rabbit systems in blue, static vertical irradiation channels in green, control rods in orange and red.

The MARIA reactor was designed to operate with several experimental devices in parallel. The most important of them are:

- Vertical irradiation channels for radionuclide production;
- Reactor test rigs for structural materials, biological samples and reactor fuel studies under stationary conditions ;
- Large sample channels for electronic, concrete and another materials irradiation;
- Horizontal experimental channels for neutron beam studies.

The reactor construction allows various kinds of irradiation facilities and neutron sources to be installed. The following channels are available:

- channels of 23 mm diameter in beryllium blocks;
- channels of 28 mm diameter in the graphite reflector;
- channels of 38 mm diameter in the graphite reflector;
- channels of 46 mm diameter inside the modified fuel element MR-2;
- bulk material irradiation facility for large-size (up to 85 mm in diameter and 900 mm height) target materials

- channels equipped with a hydraulic transport (‘rabbit’) system located in the beryllium and graphite area of the reactor core.

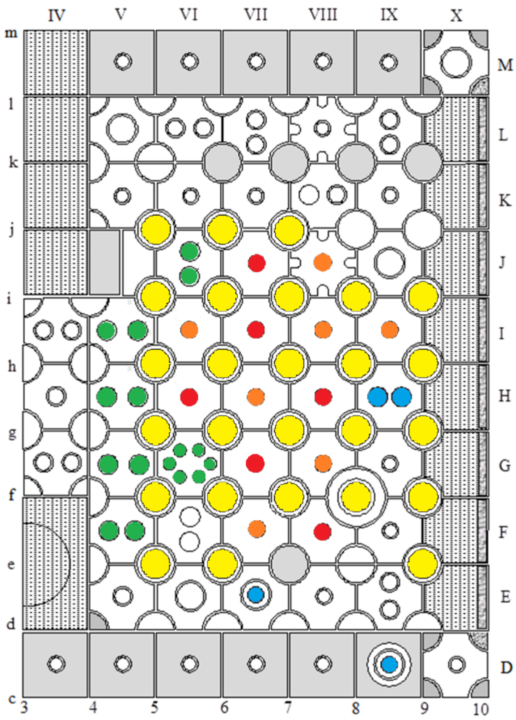


Figure 5. MARIA Research Reactor top view.

The customized fuel channel can be used to irradiate the fission materials, like targets or fuels, under strictly controlled conditions. Such technology has been applied for irradiation of uranium targets for ^{99}Mo production. This activity plays a special role in the utilization of the MARIA reactor.

For very long time MARIA reactor has been known for its isotope production activity, with some capabilities to irradiate material samples in standard irradiation vehicles. In recent years, several thermostatic irradiation rigs were designed and implemented. So far only single use devices were irradiated.

PALLAS, Petten, The Netherlands

The PALLAS-reactor is as of early 2022 in the last stages of Basic Design, after which the Detailed Design will commence, with construction planned to be underway in 2023. The site clearance at Petten has already started.

PALLAS continues the work of the HFR by producing medical isotopes and enables high-quality research. It will produce medical isotopes for diagnosis, therapy and medical research and the reactor is designed to have the largest production capacity in the world. With the arrival of PALLAS, a sufficient supply of medical isotopes will remain available after closure of the HFR, and research and development of new treatments will be given a boost. In addition, the PALLAS reactor will be at the heart of future nuclear energy research programmes on behalf of private and public parties. Important research themes include safety and performance of existing nuclear reactors and the development of Gen IV technologies, such as high temperature gas-cooled and molten salt reactors, continuing the longstanding tradition of multilateral collaboration and ‘frontier breaking’ irradiation programmes of NRG and the HFR.

The PALLAS-reactor is a pool type reactor, uses low-enriched uranium U_3Si_2 fuel arranged in a compact core (4x5 fuel element array), has an operating power of 25 MW and uses light water for core and pool cooling. The core is surrounded, essentially on three sides, by a sealed heavy water filled tank. A beryllium reflector makes up the fourth side, consisting of a 3x5 array of individual beryllium blocks.

Figure 6. Preliminary schematic of the PALLAS-reactor core and reflector in cross sectionshows the layout of the reactor core and the reflector cross section. The Be-reflector will be particularly suited for solid and liquid fuels testing, as well as materials testing for reactor components usually subject to thermal neutron spectra. Maximum flux values in the Be-reflector are in the range $3{\cdot}10^{14}$ n.cm^{-2}s^{-1} (thermal flux) and $3{\cdot}10^{13}$ n.cm^{-2}s^{-1} (fast flux, E>0.6 MeV). The target cycle length of the reactor is somewhat over 40 days, with between 7 and 8 cycles per year, for a total of 300 FPD/year.

For radioisotope production, there are positions available in fixed channels in the heavy water reflector tank. There is also the possibility to replace up to 2 fuel elements in the core with irradiation rigs/capsules for production or research (fast flux approx. $2{\cdot}10^{14}$ n.cm^{-2}s^{-1}). These positions cannot however have any external connections, but can be used for ‘drop-in’ experiments for high dpa irradiations. For most intents and purposes, research irradiations will thus be performed within the beryllium reflector region. Here, 15 different positions can accommodate an irradiation device, made by removing/replacing a single Be-block. It is currently expected that the number of irradiation devices operated in parallel in the Be-grid will be limited to four. PALLAS plans to have a variety of dedicated irradiation devices similar to those used in the HFR, including gas supply lines for temperature and general instrumentation and control, as standard. Irradiation tests are planned to be performed either with semi-stagnant gas or in closed capsules/ampules. Maximum irradiation temperatures in the 1000-1200^{o}C range are expected to be achievable. Irradiation devices will have an effective height of 60 cm and an inner useable diameter of up to about 60 mm (which can be sub-divided into smaller testing channels). The reactor design also allows the removal of several Be blocks, opening up the grid to accommodate significantly larger irradiation devices. Other provisions within the design also allow the expansion of the facilities and capabilities during the life of the reactor, including the addition of a sweep gas system similar to that of the HFR for online analysis of gaseous fission products and the more extensive upgrade of building a high temperature, high pressure water loop with chemistry control. It may also be possible to include cabling for heaters and/or to enable vertical movement of samples (for axial flux gradient change following), as well as a high pressure gas system for operating bellows (for applying loads to samples). However, these capabilities will only be confirmed during Detailed Design.

In addition to the fixed irradiation positions, a Horizontal Displacement System is being developed, which will allow irradiation devices to be moved forward-backwards within the Be-reflector region. This will enable reliable and accurate power transient testing and power cycling.

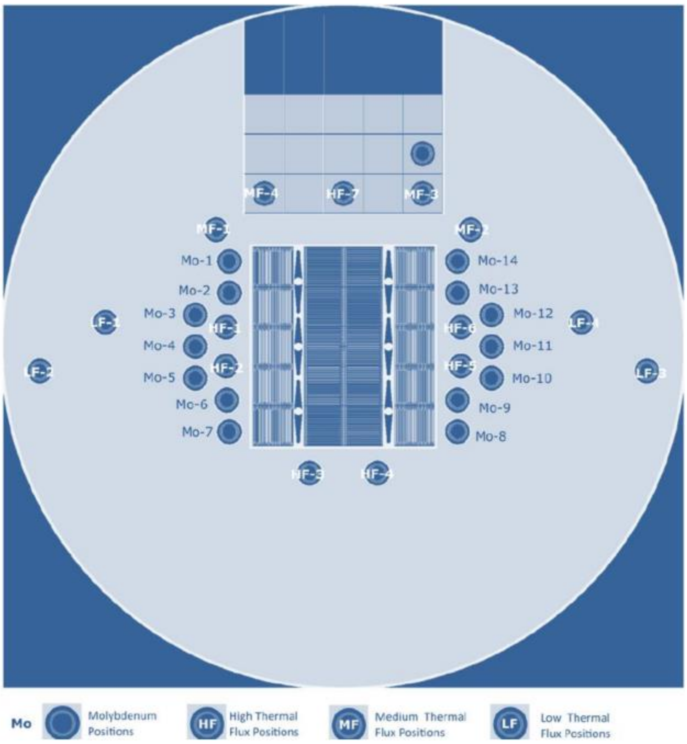


Figure 6. Preliminary schematic of the PALLAS-reactor core and reflector in cross section

Consistency of the CEP SRA with the 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 inside the reactor core with the highest ageing rate and 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 7):

- 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.

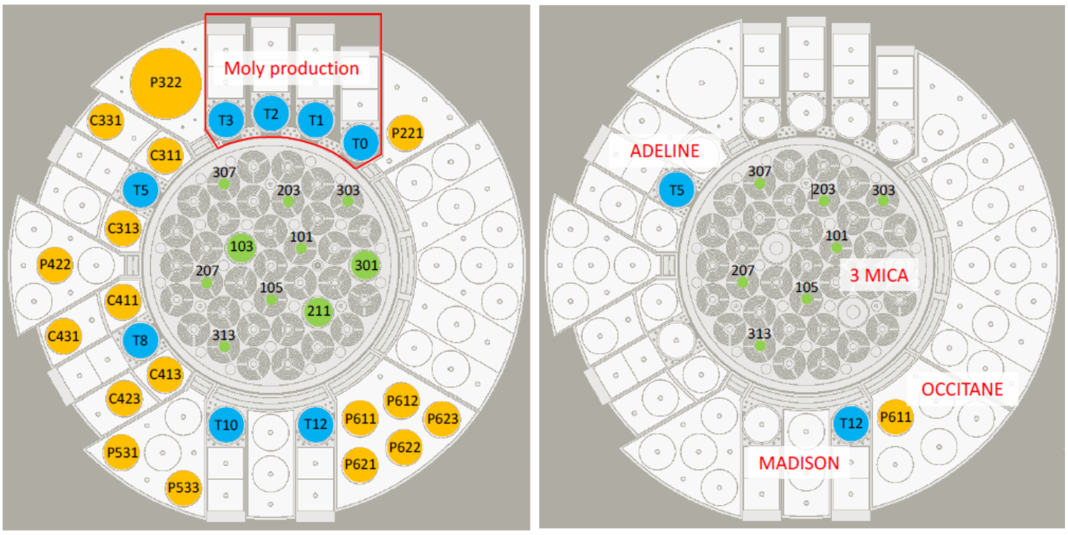


Figure 7. Left: Experimental locations in the JHR core and reflector. Right: Locations of the experimental devices available for the JHR start-up.

The start of operation of the JHR (scheduled at 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, Non-Destructive Evaluation techniques will be 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 $W \cdot cm^{-1} \cdot min^{-1}$;
    - 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 = Φ(E > 0.1 MeV) / Φ (E > 1 MeV) and the nuclear heating in the samples can be adapted with neutron and gamma shields.
- Various kinds 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 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.

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.

JHR Roadmap

The JHR working groups are currently prioritizing the experiments for both fuels and materials. With the goal of planning the operation of the JHR for the first 15 years, the Jules Horowitz Operation Plan project (JHOP2040, CSA) is underway, in parallel with the ORIENT NM project. The main target of JHOP2040 is 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. The JHOP2040 project will propose to Euratom the establishment of the JHR European

Stakeholder Network (ESN). The goal of this network is to collect the ideas and interests from different Euratom members states on how to use Euratom's access rights. The ESN will start collecting the ideas four years ahead of each reference operation plan (the 4-year operation plan in the JHR). In that work the connection also to the nuclear material partnership will be very important.

JHR is aiming to have the first reference operation plan available around two years before the start of research programmes (Figure 9). This reference operation plan needs to reviewed yearly, but it shall give the main frame according to which the loading of the reactor will be done. The key outcome from the JHOP2040 in relation to the forthcoming partnership on nuclear materials is that there are plenty of topics raised in the JHOP2040 project that could be of interest for the nuclear materials community as well. The challenge, however, is that the operation of JHR will not start until well into the 2030s, while the partnership on nuclear materials willstart running long before that. In any case, as the JHOP2040 project and the JHR Working Groups have made the mapping of the future needs, these results could be usable for the whole nuclear materials community. There are obvious needs for different fuel cladding material and reactor pressure vessel steel studies, for example. An example of the material roadmap for the JHR purposes is shown in the Figure 8 below. So, in the JHOP2040 project the different material research topics have been mapped according to technologies and materials and, as a final result, a matrix describing thoroughly the interdependencies have been obtained.

Family	#	Topic	Type of materials	Reactor system of interest											JHR experimental Device	GLOBAL PROGRAM RANKING:
				PWR	BWR	WWER	SFR	LFR	ADS	HTR	SMR	CANDU	Fusion	Spent fuel storage (SP)		
RPV	1	**Embrittlement: effect of neutron dose**	low alloy steels, including MnMoNi (e.g. SA-533, Grade B, Class 1 SA-508, Class 2, 16MND5), MnMo (e.g. SA-302, Grade B), CrMoV (e.g. 15Kh2MFA base metal, Sv-10KhMFT weld metal) and NiCrMo (e.g. 15Kh2NMFA). '18-8' stainless steels (e.g. SA-336) **(base metal, welds, HAZ)**	x	x	x	x	x	x	x	x				**OCCITANE**	3
	2	**Embrittlement: effect of neutron flux**		x	x	x	x	x	x	x	x				**OCCITANE**	3
	3	**Embrittlement: Effect of neutron spectrum**		x	x	x	x	x	x	x	x				**OCCITANE**	2.5
	4	**Modelling: irradiation damage accumulation on materials properties**	Fe-low alloys	x	x	x	x	x	x	x	x				**OCCITANE**	2

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Figure 8 Example of the material study matrix for the JHR

At the same time when the 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 second Framework for Irradiation Experiments (FIDES-II). 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. It is foreseen that these kinds of pre-JHR era studies could be of interest also for this partnership and the synergies between the different projects should be investigated further, taking into account also the possible co-operation between Euratom and OECD/NEA. 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. Several of these new technologies have also wider interest in the nuclear materials research. For more detailed description of the JHR roadmap planning, please visit the project website https://www.jhop2040-h2020.eu/.

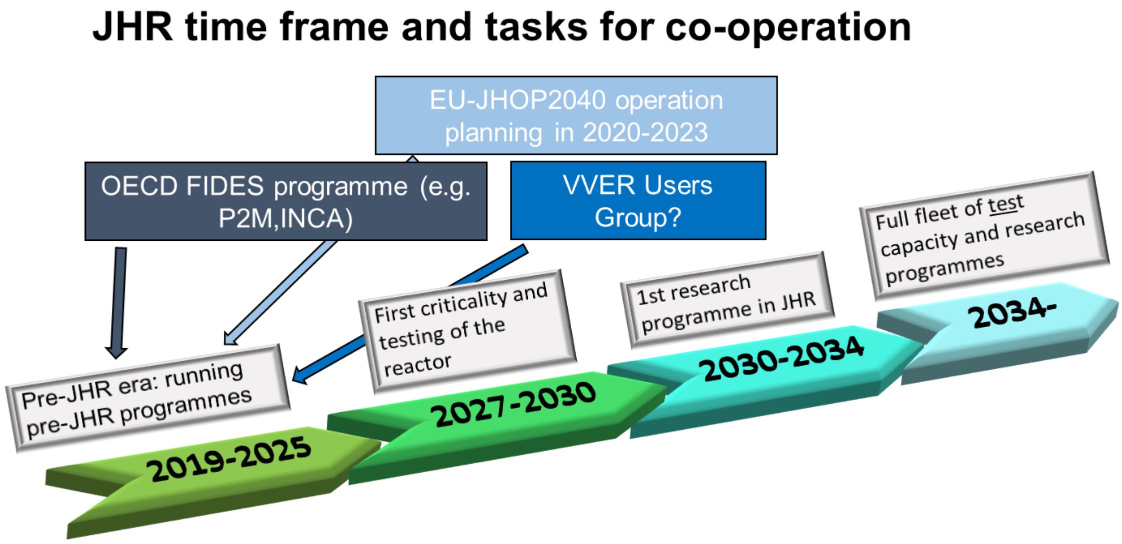

Figure 9 JHR Time frame

ORIENT NM

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