EXPERIMENTAL DEVICES IN JULES HOROWITZ REACTOR AND FIRST ORIENTATIONS FOR THE EXPERIMENTAL PROGRAMS

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This paper describes the experimental capacity and the status of the development of the test devices in the Jules Horowitz Reactor (JHR), under construction at the CEA[1]/Cadarache in France.
JHR is a MTR whose design allows performing 20 experiments simultaneously with a large range of neutron fluxes and neutron spectra. The first set of test devices is dedicated to LWR: MADISON, ADELINE and LORELEI for fuel studies respectively under nominal, incidental (power ramp), and accidental (LOCA) conditions; MICA, OCCITANE, for material studies (behaviour under representative thermal conditions, neutron fluxes and possibly under stresses), respectively for SS and/or Zirconium alloys and for pressure vessel steel; and CLOE for IASCC (Irradiation Assisted Stress Corrosion Cracking). Other test devices are under conceptual design.
The paper describes the performances of these test devices, and their status of development. The guideline for the construction of the experimental programmes is also expounded, for fuels (with a priority for LWR, for basis properties and for the behaviour under incidental – accidental situations) and for materials (for claddings, reactor pressure vessel steel, internal and absorbers).

Keywords: Jules Horowitz Reactor, experiment, test device.

[1] Commissariat à l'Energie Atomique et aux Energies Alternatives

1. INTRODUCTION

The Jules Horowitz Material Testing Reactor (JHR) is currently under construction on the CEA Cadarache site. JHR will be operated as an international user's facility for materials and fuel irradiations for the nuclear industry or research institutes [R1] but it is also dedicated to the radio-isotopes production for medical applications [R2].

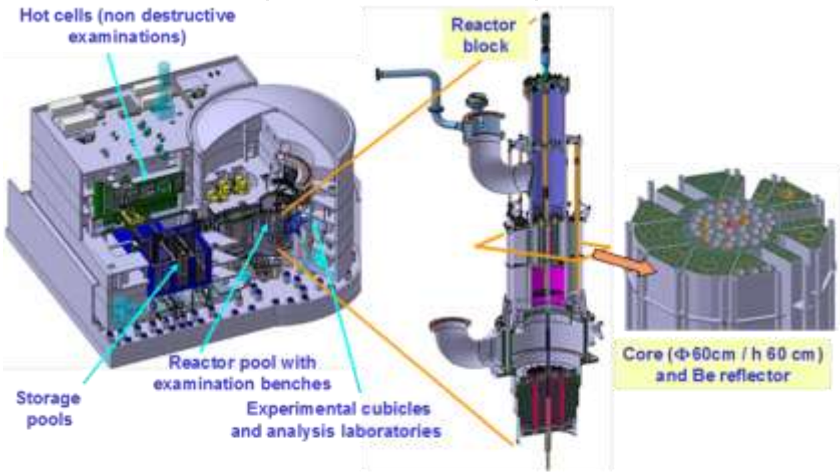
The construction is made within an international framework. Consortium members are presently: CEA, EdF, AREVA (France); JRC (European Commission); CIEMAT (Spain); VTT (Finland); UJV (Czech Republic); Studsvik (Sweden); NNL (UK); DAE (India); IAEC (Israel).

The design of this facility allows a large flexibility in order to comply with a large range of experimental needs, regarding the type of samples (fuels or materials), neutron flux and spectrum, type of coolants and thermal hydraulics conditions (LWR, Gen IV,…), in accordance with the scientific objectives of the programs. This configuration provides an experimental irradiation capacity for the next 60 years, for the nuclear industry. The design of the reactor and the facility allows accounting for the current irradiation needs from the present reactor technologies as well as future needs from advanced or innovative technologies.

1.1. JHR GENERAL DESCRIPTION [R1]

This facility is a 100 MWth pool type reactor with a compact core (60cm fissile length, 70 cm diameter) cooled by a slightly pressurized primary circuit (about 1MPa, in-core water velocity: 15m/s). The core tank is located in a reactor pool of 7 m diameter, 12 m height.

The Nuclear facility comprises a reactor building with all systems dedicated to the reactor and experimental devices; and an auxiliary building dedicated to tasks in support for reactor and experimental devices operation.



The reactor building (see Figure 1) is designed to provide the largest experimental capacity possible with the largest flexibility. One half of this building is dedicated to the implementation of devices linked to the in-pile irradiations (for example, water loops). This corresponds to 700 m2 over 3 floors for implementation of experimental cubicles and 490 m2 over 3 floors for instrumentation and control systems.

Figure 1. View of the JHR facility.

The design of the core provides irradiation cavities:

- Located in the core (7 of small diameter – 30mm ; 3 of large diameter – 80mm) which are characterized by a high fast neutron flux (up to $5{,}5.10^{14}$ $n.cm^{-2}.s^{-1}$ above 1MeV) and therefore by a high ageing rate (up to 16 dpa/year)
- Located in the Beryllium reflector zone surrounding the core, (about 20, most of them of 100mm diameter), with a high thermal neutron flux (up to $3{,}5.\ 10^{14}$ $n.cm^{-2}.s^{-1}$). Note that material experiments requiring a low ageing rate (such as the pressure vessel steel) will be installed in the reflector because of the low fast neutron flux which allows an ageing rate of about 0,1dpa/y.
- Four to six water channels (depending on the core configuration) through the reflector are equipped with displacement devices to control accurately the distance to the core and therefore the irradiation flux (for an accurate stable power, for power ramps, or

for power cycling...). The velocity (forward: 5 mm/s) makes possible to reach 700 $W.cm^{-1}.min^{-1}$ even for very high burn-up fuels.
The range of displacement allows getting three orders of magnitude for the neutron flux. Withdrawal velocity of 50mm/s allows going back fast in a safe position for safety purpose.

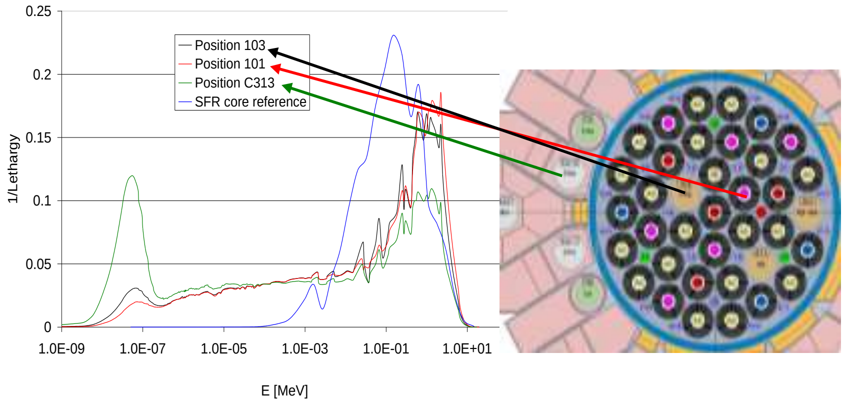


Figure 2. Neutron spectra in the JHR core.

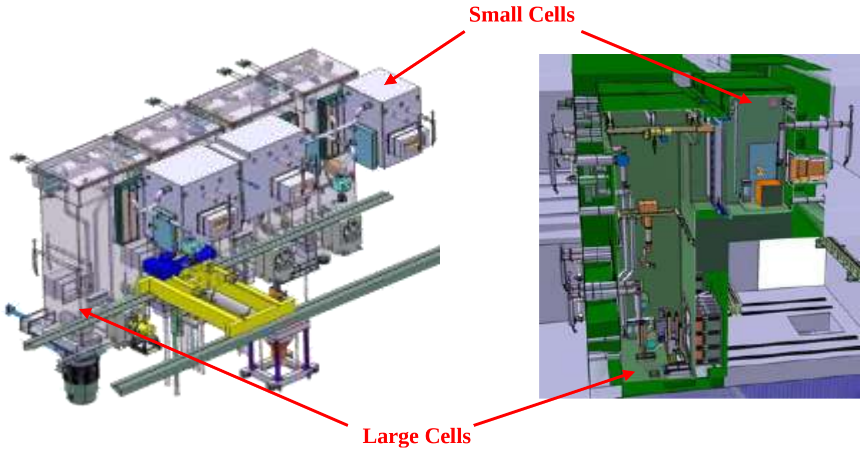
The facility is designed to operate 20 experiments simultaneously.

A typical reactor cycle is expected to last about 1 month, and CEA targets to operate the reactor at least 10 cycles per year.

The JHR experimental capacity will also include non-destructive examination (NDE) benches: A coupled gamma scanning and X-ray tomography bench located in the reactor pool, a similar bench located in the storage pool of the Nuclear Auxiliary Building. A neutron imaging system bench located in the reactor pool

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

For initial check of experimental samples
For adjustment of experimental protocol or control of parameters after a first irradiation run
For a rapid examination of samples after an irradiation phase
For a delayed detailed examination of samples after the end of an irradiation program



Hot cells (4 large cells (of 12 m height) and 3 adjacent smaller ones) will be used for the loading and un-loading of irradiation rigs from its in-pile containment, for NDE of the samples and for the preparation of the sample transfers to dedicated hot laboratories.

Figure 3. Hot cells in the JHR Auxiliary Building.

1.2. STATUS (CIVIL WORK AND REACTOR COMPONENTS)

Site work started in 2007 when the JHR consortium was set up. The concrete of the first basement has been poured in 2009. The following figure gives the main milestones. The civil work of the reactor building is presently almost completed.

Most of the main components of the reactor are manufactured, some of them are under testing phase, and the installation in the facility will start in 2018. First criticality is expected in 2021.

Figure 4 : main milestones of the civil work

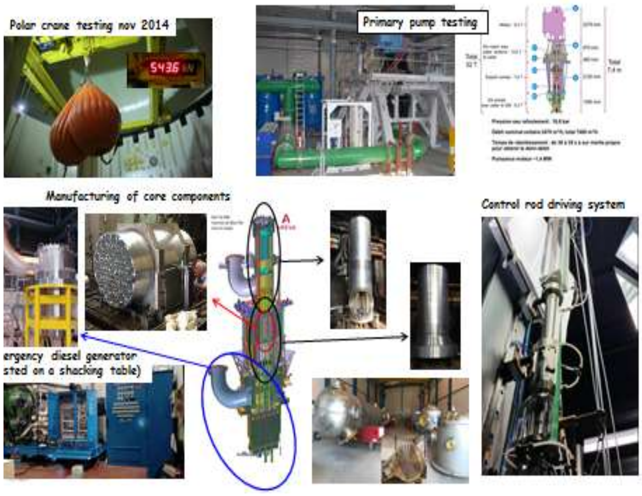


Figure 5: some reactor components

2. DESCRIPTION OF THE FUEL TEST DEVICES [R3] [R4]

2.1. INTRODUCTION

Fuel test devices are presently under development, a specific effort is done for the test devices dedicated to light water reactors (PWR & BWR, including VVER):

Madison dedicated to the investigation under nominal conditions, (evolution of the fuel micro structure, clad corrosion, fission gas releases …)

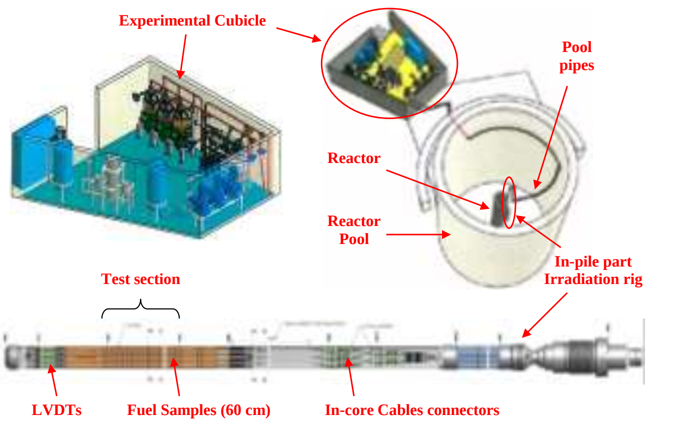
Adeline dedicated to off-normal situations, the reference situations for the design of the loop are the power ramps.

Lorelei dedicated to accidental situations, the reference situation for the design of the loop is the LOCA – large break.

Nevertheless, conceptual design of test devices dedicated to GEN IV is in progress.

2.2. MADISON

This experimental device is designed for experimentation dealing with normal operation of LWR fuel (steady states or transients) therefore, no clad failure is expected. The design of the main experimental part is made by IFE Halden.



This experimental device is made of an in-pile part (holding the fuel samples of 60 cm fissile length) located on a displacement system. This system allows an on-line regulation of fuel linear power of the samples. Thanks to the high thermal neutron flux in the JHR reflector, it is possible to reach high linear power even with high burn-up samples (It is possible to reach 400 W/cm for a burn up of 80 GW.d/t and for a standard UO2 fuel of 4,95%initial enrichment).

Figure 6. View of the MADISON experimental device in the JHR facility with focus on the water loop in experimental cubicle and on the first irradiation rig

The in-pile part is connected to a water loop providing thermal-hydraulics and chemical conditions (representative or specific conditions). The water loop characteristics (implemented in a dedicated cubicle) and the use of SS for the pressure tube of the in-pile part allow reproducing the representative thermal-hydraulics conditions of power reactors (PWR, BWR or WWER technologies) in term of water loop temperature (up to 320°C) and pressure (up to 160 bars).

In order to meet the large range of experimental needs from the nuclear industry, the test section of the in-pile part has a volume as large as possible. This allows loading a large panel of sample holders from high embarking capacity (up to 8 samples) with low instrumentation to low embarking capacity (1 sample), but highly instrumented.

This design allows performing several types of experiments:

- Selection experiments (irradiation of numerous innovative fuel samples),
- Characterisation experiments: (irradiation of few highly instrumented samples with a maximum of instrumentation, particularly on-line physical properties measurement for fuel performance),
- and codes validation or qualification experiments (irradiation of samples under representative normal and off-normal conditions of power plants)

The design of the test device is also done with the objective of being able to embark high performance and sophisticated instrumentations (room in the test device, tight feedthroughs in the head of the test device). At present, the following instrumentation can be easily used in the first MADISON sample holder: Fuel centreline temperature, Clad temperature, Clad elongation, Fuel stack elongation, Fuel plenum pressure, Fission gas release based on acoustic measurement sensor.

2.3. ADELINE

Similarly to the MADISON experimental device, Adeline is dedicated to LWR fuel (PWR including VVER type and BWR fuel). It is made of an in-pile part and an out-of-pile water loop. The in-pile part is located on a displacement system. But the objective of Adeline is to investigate fuel behaviour of a single rod under off-normal situations which can lead to clad failure.

A first version is dedicated to power ramps testing (similar performance to the ISABELLE1 loop in OSIRIS reactor). The design allows reaching high power, the limitation is linked to thermal-hydraulic criteria (700W/cm) or to fuel melting criterion.

High power is reached by using zirconium alloys as material for the test device structure. The consequence is a slightly lower operating condition of the water temperature compared to Power Plants. Representative cladding temperature is obtained by adapting the thermal- hydraulic parameters.

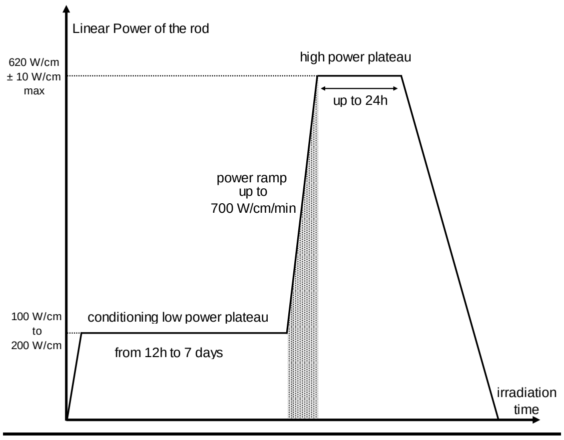


Figure 7. Standard power ramp test profile

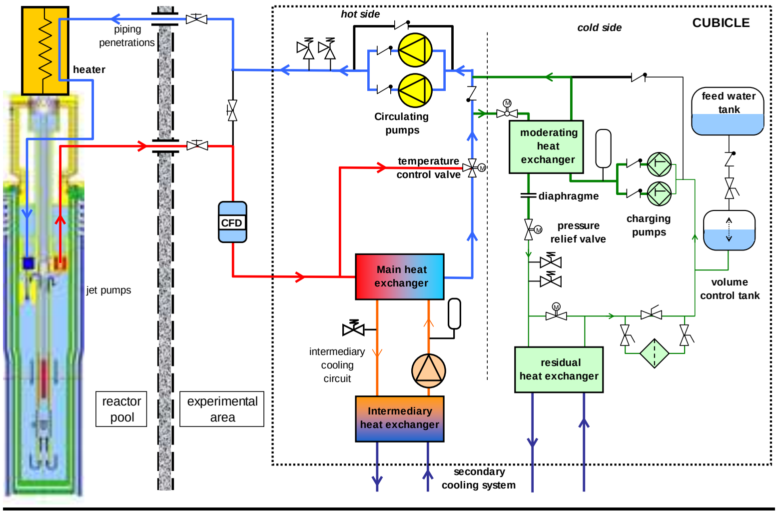


Figure 8. Schematic diagram of the Adeline loop

The in-pile part is instrumented with sensors dedicated to the local thermal-hydraulics conditions (pressure, flowrate temperature). It is equipped with flow amplifiers (jet pump system) which are supplied via an injection system located in the cubicle (high pressure circulating pumps). The amplification factor is around 4 to 5, resulting in a fuel cooling flow rate in the test channel of about 200 g/s for a flow rate in the supplying line of about 50g/s. It makes possible to use pipes of small diameter which are flexible enough to allow large displacements in the pool.

A low pressure and low temperature system is installed in parallel of a part of the main loop. It mainly includes the water treatment and the control of the loop chemical conditions.

The reference configuration will provide an accurate thermal balance, a quantitative measurement of the gamma activity of the water (to quantify the radiological source term released) in case of rod failure and an on-line quantitative clad elongation measurement during power transients. This configuration makes possible to load and unload the experimental rod via a “rod transfer station” without pipes disconnection, and it is expected to be able to perform 3 ramps during one reactor cycle.

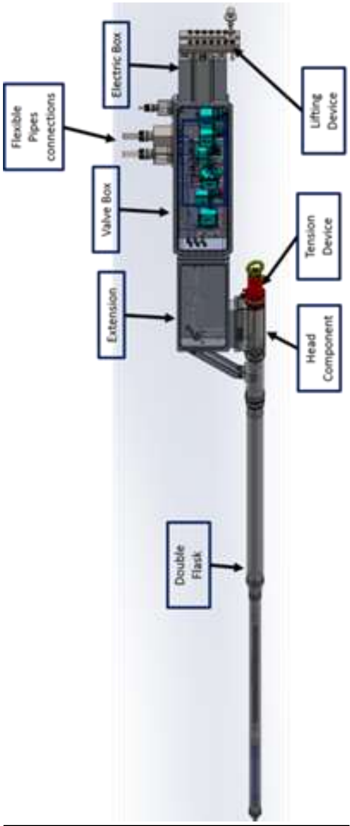
A “highly instrumented configuration” is also taken into account in the design (with fuel and clad temperature measurement, fission gas release measurement, plenum pressure sensor…) but the handling requires a transfer in the hot cells and only one test is possible per reactor cycle. This configuration could be used to investigate: Rod internal over-pressurization (“lift-off”), Rod internal free volumes gas sweeping, Power to melt approach margin mastering. 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

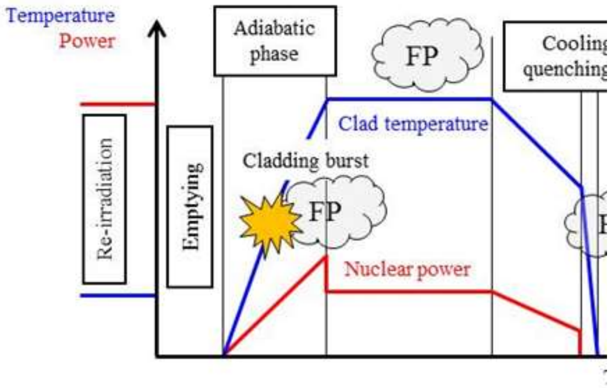
2.4. LORELEI

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 during the experiment does not reproduce all the phases of a LOCA-type power reactor sequence, but the thermal-mechanical conditions of the clad (clad temperature, clad over-pressure, steam environment) will be representative.

This equipment consists in an integrated water capsule which can be operated as a thermo-siphon able to cool and re-irradiate a single fuel sample, and to produce a short half-life Fission Product inventory.

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 an homogeneous temperature azimuthal distribution and acts as a dynamic thermal insulation in order to get representative adiabatic conditions (initial heat-up rate will be about 7 to 10°C/s maximum due to technological limitation).





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.

Figure 9. Lorelei test device

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 hydriding), 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 improve representativity of 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).

2.5. TOWARDS GEN IV TEST DEVICES

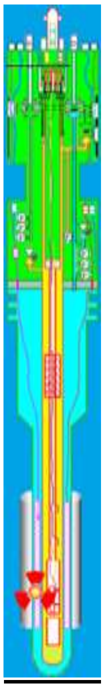
The test devices dedicated to GEN IV fuels are under conceptual design. Several technologies (gas, sodium,…) are taken into account. SFR experimentations will take benefit of the technological developments done in the framework of the “Calipso” test device development (see next section): the “calypso technology” could be used to test fuel under nominal conditions in experimental locations within the core and to test incidental / accidental conditions in the reflector by using the displacement system to simulate abnormal power transients and a change of flow of the electro-magnetic pump to simulate loss of flow transients.

3. DESCRIPTION OF THE MATERIAL TEST DEVICES [R3] [R4]

3.1. INTRODUCTION

Similarly to the fuel, CEA made the decision to develop first new irradiation devices in support of LWR type reactors, taking benefit of the know-how of the previous reactors (particularly OSIRIS). This section describes the test devices under developments and those under conceptual design.

3.2. MICA



The MICA device (Material Irradiation Capsule) is derived from the usual CHOUCA test device widely used in SILOE and OSIRIS reactor). The samples are located in NaK (up to 450°C) or gas (up to 1000°C). The operation principle is that the experimental sample temperature results from the equilibrium between the gamma heating and the heat losses through double wall containment. Heat losses are controlled by adapting the composition of the gas located between the two walls. For a more accurate control of the temperature, several electrical heaters are installed on the inner wall. These test devices are mainly foreseen for in core irradiations were fast flux can lead up to 10 to 12 dpa per year corresponding to the maximum gamma heating being possible to equilibrate.

Figure 10. MICA test device

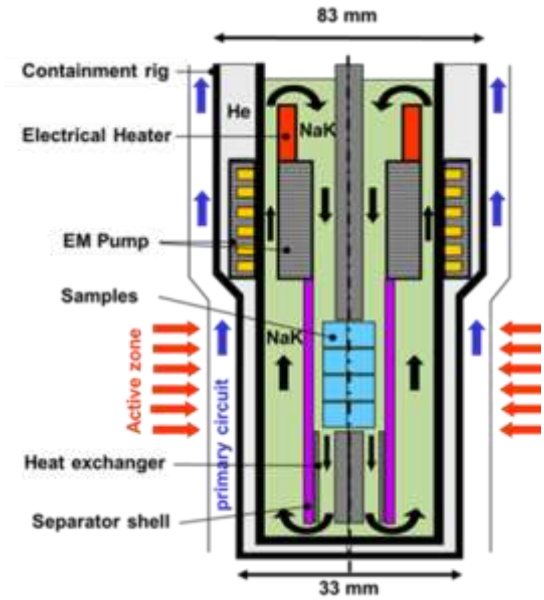
A particular focus has been given on the handling procedure by improving the gas circuits control components (valves, pressure sensors, connection) and the current electrical connections (instrumentation and electrical heating) located in the head of the test device. In addition, effort has been made to improve the thermal behaviour of the device (temperature accuracy and gradients mastering).

This test device can be used to perform sample irradiation for tensile, Charpy tests, CT specimen, or to study irradiation creep and stress release phenomena.

As an example, the “MELODIE” sample holder (qualified in a MICA test device) has been developed to measure the bi-axial creep of a cladding sample submitted to a controlled bi-axial loading. It has been successfully tested in Osiris in 2015. An axial creep rate of 4µm/d was measured under 60MPa in the irradiation conditions of OSIRIS.

3.3. CALIPSO

The objective of the CALIPSO device (in-Core Advanced Loop for Irradiation in Potassium SOdium) is similar to the MICA one but the main difference is to meet the original need of a low temperature axial gradient (a maximum of 8 °C) all along the sample holding, in liquid metal coolant (NaK), in the operating range of the LWR needs (300 to 450°C) and with the highest ageing rate (16dpa/y) it means the highest gamma heating. The design is based on an embedded thermal-hydraulic loop, including a heater, an electromagnetic pump and a heat 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. A prototype has been successfully tested in a dedicated facility (SOPRANO). The performance of this test device (pumping, heat exchanger) makes possible an adaptation to other applications, particularly fuel irradiation within the framework of GEN IV).



Schematic diagram of the Calipso test device

Prototype of the electromagnetic pump

Figure 11 – CALIPSO

3.4. OCCITANE

Figure 12 – IRMA test device (OSIRIS)

In the field of pressure vessel steels of NPPs, irradiations are carried out to justify the safety of this 2nd 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 to 300°C. It will be implemented in the JHR reflector and reach damage rate about 100mdpa/year (E > 1MeV). The associated instrumentation consists in thermocouples and dosimeters, but a “fast neutron flux fission chamber” is envisaged to better control the irradiation condition and better determine the irradiation duration. OCCITANE is based on IRMA device of OSIRIS. The design studies consist mainly in decreasing thermal gradient in the sample area and in integrating the capsule in the JHR environment.

3.5. CLOE

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 is designing a LWR corrosion loop, which will be located in the JHR reflector close to the JHR vessel. 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.

3.6. PROSPERI/O

The JHR vessel (aluminum based alloy) is strongly irradiated during operation. Damaging phenomena are due to fast neutron flux (dpa) and due to thermal neutron flux (transmutation). Two test devices dedicated to the surveillance of the vessel will be installed in the core, close to the vessel (maximizing the fast flux damages) and in the reflector, close to the vessel (maximizing the thermal flux damages). This design of material irradiation test devices, operating at low temperature, can be used for other applications.

3.7. TOWARD GEN IV

Right now, CEA anticipates the fourth generation research program; a first analysis of the SFR or GFR irradiation needs for materials and fuels has been performed. Accordingly, some feasibility studies have begun in order to prepare future irradiations in JHR. As examples, CEA studies the adaptation of the calypso test device for operation at 600°C (SFR), and the adaptation of the MICA test device for operation at 1000°C (GFR). Conceptual design of an in-core high temperature device with a large capacity has been also studied.

3.8. AND ABOUT FUSION NEEDS

After analyzing the needs linked to the fusion technology, three conceptual designs (FUSERO test devices) are under preparation:

- Thermo-Mechanical Fatigue testing: study of components submitted to both mechanical strain and thermal strain (from the breeding blanket to divertor tiles).
- Ceramic testing (for diagnostic windows); samples bi-axially loaded, analysis of optical properties and sub-critical crack growth.
- Cryogenic testing for the study of electrical and structural properties of superconducting magnet materials.

4. START UP TEST DEVICES

The objectives of the start-up test devices are the verification of the performances of the facility, the verification of the safety parameters but also the accurate determination of the irradiation parameters in the experimental cavities. This last point is specifically a challenge because JHR has to be as accurate as the present MTRs which started to operate at least 40 years ago…

The preparation of these test devices led to optimize the strategy for the start-up tests (timing of the tests, accuracy, power level, reactor configuration…) and to analyze the needs in terms of instrumentation

- Neutron flux and gamma heating mapping, neutron spectra
- fissile power, reactivity measurement, in core void effect evaluation
- Thermal-Hydraulics (flowrate in experimental cavities, in core, in reflector ; core under free convection, ...)
- Devices dedicated to the thermal mapping of the reactor structural materials

5. FIRST ORIENTATIONS OF THE EXPERIMENTAL PROGRAMS [R5] [R6]

A specific organization has been set up within JHR consortium framework in order to:

Identify the open issues in the field of nuclear fuel and material development and qualification,

Define criteria to elaborate 'ranking grids" about fuels/materials types, reactor systems,

Define experimental objectives and initiate the future R&D programs.

Generally speaking, accent is given on the mastering and follow-up of the irradiation conditions in terms of local neutron flux and spectrum, temperature homogeneity and thermal gradients, with the possibility to un/re/load irradiated materials in the JHR experimental devices.

The priority is given on Gen II-III power systems, the needs are related to:

- Integral LWR fuel element performance under nominal operation (fuel rod, cladding, fuel assembly)
    - for various new fuel (UO2 with additives, burnable absorbers, …) and clad candidates, including Enhanced Accident Tolerant Fuels,
    - for various chemical environments,
    - and various power conditions (power cycling, load following and Extended Reduced Power Operation).

  With a specific need concerning failed fuel behavior in normal operation.
  For the assemblies, special attention will be given on behavior about guide tubes and grids: grid-spring interaction, SCC, guide tubes axial creep, effect of rod bowing,
  For the control rods and burnable absorbers: thermal-mechanical and geometrical evolutions, neutron absorber consumption.

- LWR fuel testing in incidental conditions: power ramps (crack initiation and propagation, clad integrity thresholds, pellet-cladding chemical interaction, FP release and radiological source-term), power to melt, lift-off phenomenon.

- LWR fuel behaviour in accidental conditions such as LOCA-type conditions (clad ballooning and hydriding, fuel fragmentation, pulverization and ejection, grid effects, radioactive source term, bundle geometry after quenching…),

But needs dedicated to fuel for Gen IV power systems are also established: thermal-mechanical fuel properties fission product effects (FP retention, "joint oxide-gaine" (JOG) formation, fissile-fertile interaction), and integral fuel performance for SFR, GFR, ADS, (high burn-up conditions, FP chemical behavior, behavior in transients…), without forgiving minor actinide transmutation, and new fuel concept.

About specific needs related to materials, it can be mentioned:

- For the cladding:
    - The mechanical behaviour (creep test) with on-line biaxial control (stress and strain),
    - The effect of environment on mechanical behaviour (various LWR environments)
    - The effect of dose accumulation on cladding (microstructure, hardening, embrittlement, creep for Gen II/III, and microstructure, swelling, embrittlement, creep for Gen IV).
- For the Reactor Pressure Vessel (RPV): Dose accumulation (Gen II/III, Gen IV) with the Irradiation effect on the microstructure and mechanical properties.
- For Internals (Gen II/III, Gen IV): dose accumulation and environment effect (LWR) on mechanical behaviour.
- And for Absorbers: Dose accumulation (Gen II/III, Gen IV), effect of irradiation on the microstructure.

This analysis led to a "Position Paper" and a first proposal for a JHR International Joint program, dedicated to:

- Fuel behaviour (improvement of the understanding of the phenomena involved in power transients and having an impact on the clad loading: quantification of the fission gas release effect and impact on pellet-clad interaction during a power transient consisting in successive power steps).
- And material behaviour (effect of neutron spectrum on Stainless Steel behaviour): dose-damage relationship quantified by tensile testing and microstructure characterizations. Effect of ratio "epithermal + fast" neutron flux / "fast" neutron flux (Rs 2 - 5) on mechanical properties and on microstructure of Stainless Steel.

6. CONCLUSION

The Jules Horowitz Reactor has been designed with improved nuclear performances compared to previous MTR, and particularly to OSIRIS, while taking advantage from a strong feedback about its operation. This feedback is also of primary importance for the design and manufacturing of the experimental hosting systems and for the Non Destructive Examination benches. Nevertheless, CEA is maintaining significant R&D programs to improve the performances of the experiments (especially by putting innovation on instrumentation, with a specific effort for on-line measurements and on NDE technics). With the help of improved modelling, efforts of development are in progress on test devices designed to match the requirements and to provide around experimental samples accurate, mastered and reproducible physical and chemical conditions.

The elements of future experimental programs are under preparation, mainly oriented on GEN II & III reactors. The first international program is built in order to gather the scientific community on JHR experimental objectives before the reactor start-up.

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