CABRI REACTOR SAFETY REASSESSMENT

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The CABRI reactor in the context of future experimental needs

The CABRI reactor is located on the Atomic Energy Commission (CEA) site at Cadarache in Southern France, in a building constructed in 1962.

It is a pool type research reactor with slightly enriched UO2 pins (6% 235-U). It is used for fast power transients and therefore to simulate a reactivity insertion accident (RIA) in a PWR electricity-generating reactor, such as caused by the ejection of a control rod. These tests, requested by the French Safety Authority, are in anticipation of increasing the fuel combustion rates in electricity-generating reactors.

The reactor is made up of (see Figure 1):

- a driver core, with a maximum power of 25 MW in steady-state conditions, cooled by a volume of water (460 m^3) circulating by forced convection,
- an experimental loop, with its own cooling system, one part of which is located at the centre of the driver core and contains the instrumented test device with the fuel pin to be tested.

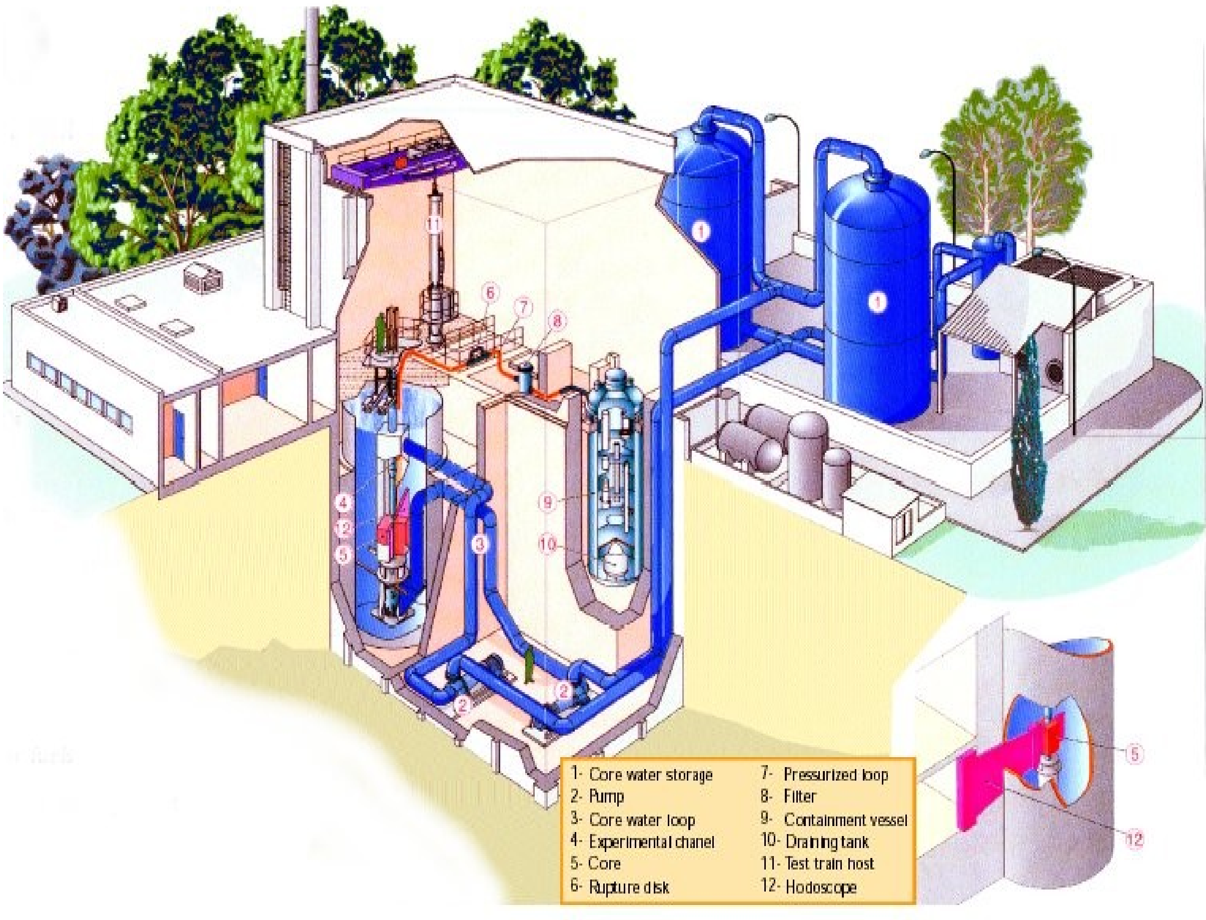

Figure 1: the CABRI reactor and the pressurised water loop

The special feature of CABRI reactor is its reactivity injection system: four driver core assemblies, called “transient rods”, are fitted with empty cylindrical tubes at their periphery; these tubes may be filled with pressurised helium 3 (gas absorbing neutrons) and depressurised by opening the motorised valves, to modify extremely quickly the reactivity and therefore the power of the driver core (power “pulse” corresponding to an injection in the order of $3 at $50/s) and the test pin power by neutronic coupling. The driver core pins are designed to support this injection of reactivity (austenitic steel cladding, large pellet/cladding space).

For the injection of reactivity, initiated from a power level of about 100 kW but which could achieve about 20 GW (see Figure 2), the reactor safety chain is inhibited for 1 s. The power drops, before the control and safety rods fall, under the effect of the neutronic feedback (Doppler effect).

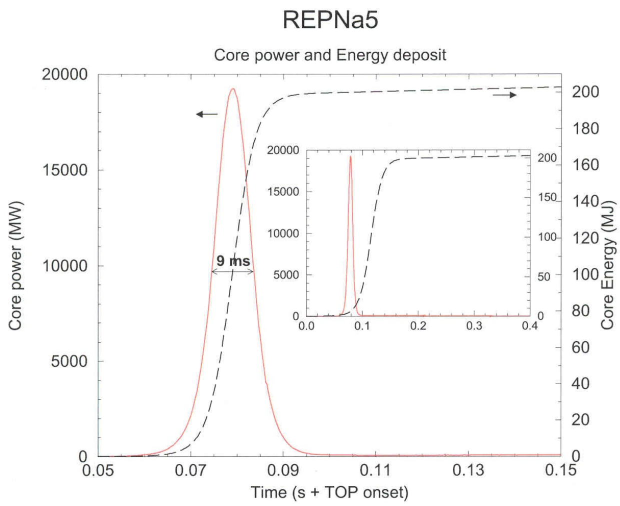

Figure 2: Core power and energy deposit during the fifth experiment with a PWR pin

Experimental programme

CABRI reactor was originally devoted to the study of fast breeder reactors (FBR) fuel pin behaviour. So, the coolant contained in the experimental loop was sodium. Some sixty tests were performed between 1978 and 1997. In addition, twelve tests have already been performed on PWR fuel pins in the sodium-cooled test loop between November 1993 and November 2002.

A new pressurised water loop installed in place of the sodium-cooled loop will be used to re-create the thermohydraulic conditions existing in a PWR reactor (P = 155 bar, T = 350°C). An international programme of tests scheduled for 2008 (about sixty tests over twenty years) will involve UO2 and MOX pins with high burn-up. They will be used in particular to:

- improve knowledge of the interactions between fuel, cladding and coolant,
- estimate the safety criteria margins (mass enthalpy thresholds for cladding failure and fuel dispersion) applied to fuels with high burn-up,
- study fuel relocation, zirconium (a major component in cladding composition) oxidation by steam during reflooding, damage to the cladding from thermal shocks.

The international test programme will gather designers, utilities, safety authorities and assessment agencies.

Pressurised water loop

The test loop (see Figures 3 and 4), representative of a pressurised water reactor, will include:

- one section installed in the centre of the CABRI reactor core, manufactured in Zircaloy to aim a relative neutronic transparency, that will include:
    - o the test device, that is specific to each experiment, housing the experimental pin,
    - o the pressurised water containment (PW containment) which houses the test device and contains the radioactive products (first barrier towards the test fuel),
    - o the safety tube, providing thermal isolation between the PW containment and the reactor pool water (second barrier towards the test fuel);
- one out-of-pile section in stainless steel, comprising:
    - o a "containment vessel" with all the systems required for the experiments (systems which are first barrier towards the test fuel) and which will act as the second barrier towards the test fuel,
    - o double walling for the connecting piping between the PW containment and the loop coolant system in the containment vessel,
    - o a filter, catching virtually all the particles from any rupture of the test rod,
    - o a collecting tank for the loop gaseous effluents, located outside the "containment vessel".

The experimental loop and related systems, that are currently being manufactured, will all be located in the reactor building. This will therefore act as a third barrier.

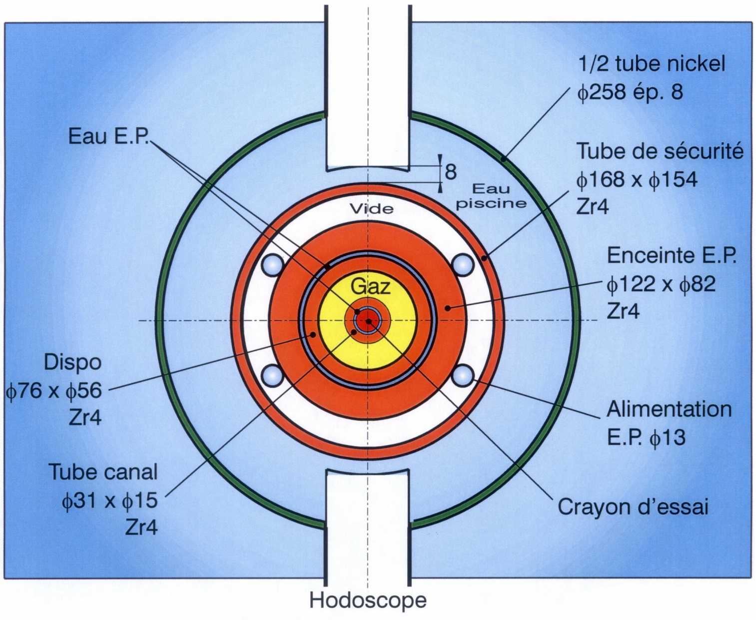

Figure 3: Cross-section of the in-pile section of the experimental loop

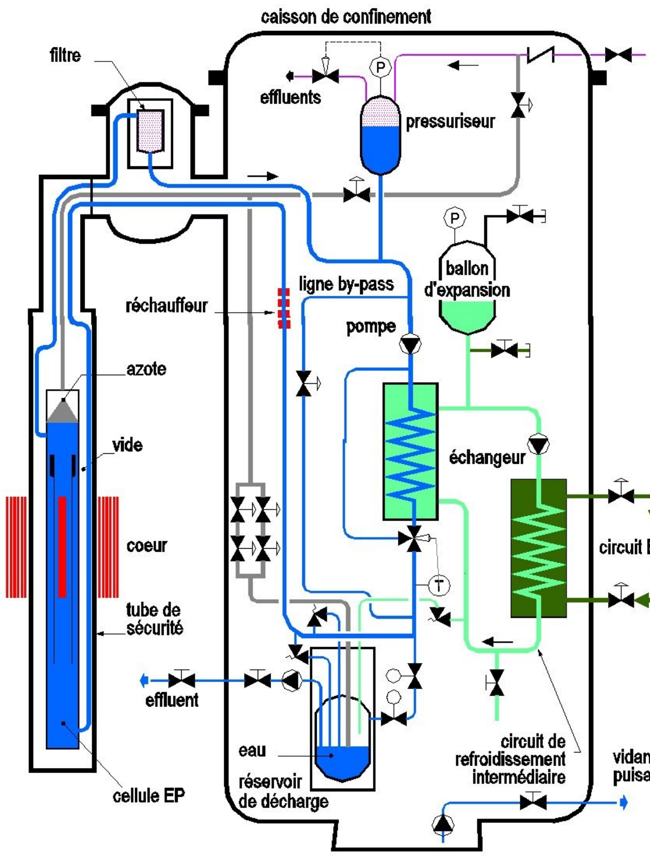

Figure 4: Schematic diagram of the experimental loop

Safety assessment of the CABRI reactor fitted with the experimental loop

Challenges and organisation

Under the request for administrative authorisation to modify the installation, the plant operator submitted a preliminary safety analysis report (PSAR) in February 2002 concerning:

- the design and installation of the pressurised water loop,
- a safety review of the existing installation (including the reactor), comprising:
    - a review of installation compliance with the last approved reference standards, taking into account modifications made during operation; this review should reveal any discrepancies and demonstrate that safety levels have not deteriorated;
    - a safety reassessment, based on:
        - feedback analysis,
        - identification and processing of discrepancies compared with current regulations (acceptance as is, modifications to the installation, renovations, etc.),
        - updated safety studies taking into account the evolution of knowledge (particularly improved calculation tools) and of

analysis methods given the most recent prevailing practices (fundamental safety rules, expert opinions, etc.).

At the request of the Safety Authority and based on an analysis report written by IRSN (Insitut de Radioprotection et de Sûreté Nucléaire), the Standing Group for reactors (comprising about 40 specialists from miscellaneous nuclear industry bodies) met on 22 January, 29 January and 13 May 2004 to advise in particular on:

- the design and sizing of the new pressurised water loop equipment,
- the use of Zircaloy for the manufacture of the in-pile section of the water loop,
- the list of incidents and accidents adopted in the safety demonstration and their consequences, particularly radiological,
- the facility safety review and more specifically the upgrading of its seismic behaviour.

Within IRSN, the assessment of the PSAR has been organised by DSR/SEGRE (Reactor Safety Division/Department for the Assessment of Gas-cooled, Fast-neutron and Experimental Reactors) and groups about ten of the Institute's specialist departments. In addition, BCCN, the French Inspectorate of Nuclear Steam Supply Systems, which is part of the French Safety Authority, has contributed to this analysis in terms of the regulations that are applicable to pressurised equipment.

AVN, the technical support agency of the Belgium Safety Authority, has also provided an independent second opinion on some subjects in the design and sizing of the PW loop. The conclusions of this second opinion, which have also been presented to the members of the Standing Group, agree very clearly with those of IRSN.

Analysis context

Whereas the major safety principles remain unchanged, the design and construction codes in nuclear practice, as the regulations, are normally intended to be applied to PWR electricity-generating reactors. The safety analysis methods should be adapted to the special case of each experimental reactor. It is important to take into account not only its potentially positive aspects (low radiological inventory, limited operating time, etc.) but also its own specific risks. For its own safety assessment, IRSN therefore took into account:

- not only a certain number of peculiarities in the CABRI reactor:
    - o the fact that it is not in constant operation (about ten hours for each test, i.e. in total a few days per year considering the maximum of five tests per year),
    - o a relatively limited radiological inventory of the driver core (burn-up about 100 MWj/t),
    - o the absence of staff in immediate proximity during reactor operation or loop pressurisation (control room located 300 m from the reactor),
- but also difficulties inherent to the specific features of the experiments:
    - o power excursions in the driver core with inhibition of the reactor protection system (normal operating condition),
    - o installation of a part of the reactor cooling system outside the reactor building,

- the pressurised fluid casing acting as the first barrier of the PW loop (under normal operation the test could rupture the experimental pin cladding, causing the dispersion of radioactive products through the loop coolant system),
- the fuel/water thermodynamic interaction should the experimental rod cladding rupture, causing a peak of overpressure in the loop coolant system (normal operating condition),
- use of Zircaloy, a fragile material, for the manufacture of the in-pile section of the PW loop (P = 155 bar),
- the inventory of the experimental pin fission products with a high burn-up,
- the management of liquid effluents with high activity due to the experiments.

In addition, the fact that this is the first safety review since the facility was created is an added difficulty given the ageing of the equipment and changes in design and analysis rules.

Lastly, considering the plant operator's wish to operate CABRI reactor for a further twenty years, the Nuclear Safety Authority has requested IRSN to make its safety assessment as if it was a lasting facility, which would involve, in particular, a full review of the behaviour of the facility faced with a seismic hazard like a Maximum Historically Probably Earthquake (MHPE) or a Safe Shutdown Earthquake (SSE) (see in the later paragraph “Seismic Hazard Analysis”).

The main IRSN conclusions

Use of Zircaloy in manufacturing the PW loop

Using zirconium alloys (as Zircaloy) as a cladding material does not comply with present-day industrial practices for pressurised nuclear equipment. In fact, its use is not explicitly provided for in design codes such as the RCC-M or ASME (Section III) codes.

IRSN has therefore examined in detail the aspects relating to the use of Zircaloy to be able to give an opinion on the suitability of the adopted design rules, taking into account the ductility and toughness of this material. This examination reveals principally that it is acceptable for the plant operator to apply the RCC-M sizing rules for ferritic steels, on condition that he makes provision for high quality production, ensuring particularly the absence of nicks. In addition, given the material's low capability for plastic adaptation, it seemed necessary to avoid incursions in the plastic domain by recourse to appropriate sizing rules.

Overpressure produced by fuel/water interaction

The tests performed could, under normal operation, result in the rupture of the test fuel cladding and, therefore, in interaction between fuel and water in the loop coolant system.

The envelope value of the peak in overpressure caused by this interaction is an essential data for sizing the loop. Indeed, the test device is not isolated from the PW loop coolant system ("open" device).

Given the calculations performed and subject to feedback from the next tests, a value of 480 bar in the interaction zone has been adopted at the pre-sizing stage. Nevertheless, the analysis of the special case of MOX experimental pins (presence of plutonium-rich clusters, different coupling than for the UO2 pins, etc.) should be developed further and the first test on MOX fuel will be subjected to authorisation from the Safety Authority.

Consideration of the fuel/water interaction in sizing the PW loop

The overpressure from the fuel/water interaction propagates axially and attenuates inside the test device before moving into the loop coolant system. As this device has an effect on the loop first containment barrier behaviour, IRSN has estimated that it is an item important for safety and that the level of associated design and construction requirements should take account of this dynamic overpressure.

The sizing of the loop coolant system will therefore normally take account of this dynamic overpressure. The plant operator should in particular show that the protection valves on the coolant system do not open during the passage of this wave of overpressure occurring under normal operating conditions.

Failure of the loop barriers

An examination of the various rupture scenarios of the loop coolant system by IRSN reveals principally that:

- the second PW loop barrier should be sized by taking large margins towards overall loading (pressure and temperature) induced by the assumed rupture of the first barrier, so as to satisfy the operating requirements of leaktightness and non-aggression of the driver core and the pool;
- for the local dynamic loadings (jet effects, whipping of pipework, shocks etc.), smaller margins could be used when sizing the safety tube and eliminate its failure nevertheless.

In addition, under the defence in depth, IRSN believes that the plant operator should check that the consequences of simultaneous failure of both PW loop barriers located in the centre of the reactor core remain acceptable.

List of operating conditions

With the help of a structured and innovative approach, in comparison with a large number of experimental reactors, the plant operator has drawn up a list of operating conditions (potential incidents and accidents as a result of equipment failure or human error) classified into four categories depending on the annual frequency of occurrences estimated from the initiating event. These operating conditions concern the reactor, the pressurised water loop and the facility effluents. Each one is analysed specifically for prevention, detection and limitation of the consequences and, if appropriate, a preliminary estimation of its radiological consequences is made.

IRSN examined this list taking the specificities of the facility into account. Without questioning the legitimacy of the plant operator's approach, IRSN has formulated a few requests relating, for example, to:

- consideration of additional, different aggravating circumstances from those considered by the plant operator,
- consideration at the design stage of the hazards from very high-activity effluents,
- installation of redundancies to stop the reactor core coolant system pumps,
- demonstration of the possible evacuation of residual power from the experimental pin should the loop coolant system fail.

Seismic hazard analysis

The behaviour requirements assigned in the event of an earthquake to the various equipment (leaktightness, non-conversion into missile, functionality, etc.) and structures (lack of crumbling, control of overall deformations, etc.) have already been analysed by IRSN in 2002. On this basis, IRSN has investigated the feasibility of the plant operator's suggested reinforcement solutions.

The seismic movements spectra presented by the plant operator correspond to the Maximum Historically Probably Earthquake (MHPE) and the Safe Shutdown Earthquake (SSE, with an intensity higher by one unit than the MHPE on the MSK and EMS scales), adapted to the Cadarache site and approved by the French Safety Authority.

The MHPE is combined in particular with all normal operating configurations for the reactor and the PW loop. Given the short operating times of the CABRI reactor, IRSN believed it acceptable for the SSE to be combined with the pressurised operation of the experimental loop (excluding the extremely short power excursion phase), but assuming reactor shutdown.

IRSN has not identified an element that would preclude the seismic reinforcement of the buildings. Nevertheless, it has underlined that the reinforcement process using carbon fibre fabrics, as planned locally by the plant operator, has yet to be validated. The plant operator should therefore suggest other solutions based on proven reinforcement techniques (concrete, for example).

Lastly, in accordance with IAEA recommendations for new facilities and given the planned additional operating time, the plant operator has undertaken to fit a seismic detection system to the CABRI reactor that will shut the CABRI reactor down automatically in the event of an earthquake.

Conclusion

The analysis of the preliminary safety report on the CABRI reactor fitted with a pressurised water loop represents an important stage in the life of this facility.

Subject to the plant operator taking into account the requests formulated by the Standing Group based on the IRSN analysis and respecting his commitments, the French Safety Authority has responded favourably to the modification of the facility and its safety review.

The intermediate safety analysis report that will be forwarded early in 2006 in response to the requests by the Safety Authority will be analysed before the facility re-enters service, scheduled for 2008.