Neutronics computations in support of the CABRI core safety analysis

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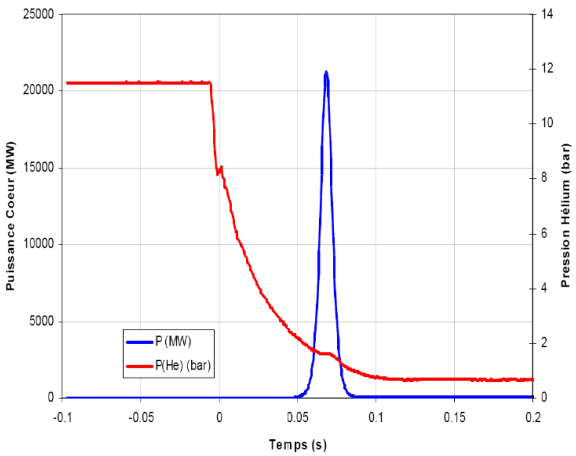
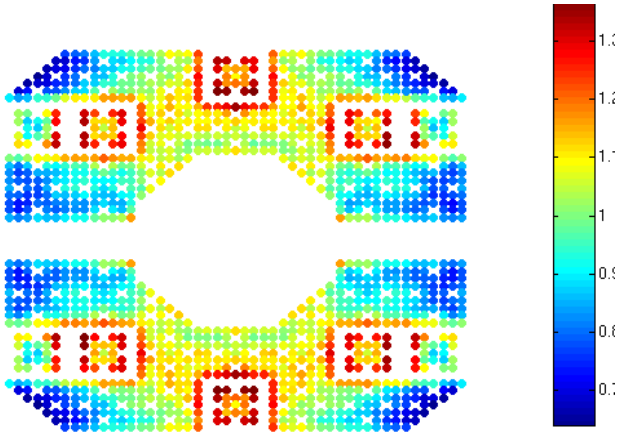
Abstract:

CABRI is a pool-type reactor, with a core made up of 1488 stainless steel clad fuel rods with a 6% 235U enrichment. A 30 cm diameter test loop is installed in the core centre to receive an experimental device. This loop has its own cooling system. The old test facility, featuring a sodium cooled device, was shut down in 2003. Recently, a pressurized water loop has been installed, in order to be more representative of thermal hydraulics effects occurring during a fast power increase in a PWR. The project is funded by the French "Institut de Radioproctection et de Sûreté Nucléaire" (IRSN) through the CABRI International Programme (CIP) framework.
To take into account the evolution of the installation and because the last major commissioning computational characterization of the core was made 30 years ago, it was important to re-evaluate the neutronics parameters and to improve the interpretation of experimental results before its upcoming starting. Several neutronics computations have also been realized in the perspective of core commissioning and will thus be considered as an example in a context where several new reactors will start in the coming years (RES, JHR...).
CABRI was designed to realize power transients from 0.1 to 20000 MW in order to test a fuel pin in the experimental device under conditions representative of PWR rod ejection.
In this context, kinetics and feedback parameters must be determined with a high accuracy. They are evaluated with 3D Monte Carlo reference calculations (MCNP [1] and TRIPOLI4 [2]). Then, they are used in the DULCINEE [3] code, which includes a thermal and point kinetics model to simulate the dynamic effects in CABRI transients.
This paper presents
• the computational methods used to determine the fundamental neutronics parameters
• the qualification of the model by comparison with experimental results
• an application using the methodology for safety studies.



Power and pressure evolution during the transient



Power shape

I. INTRODUCTION

The CABRI experimental reactor is located at the Cadarache nuclear research center, southern France. It is operated by CEA and devoted to IRSN safety programmes. It has been successfully operated during the last 30 years, enlightening the knowledge of FBR and LWR fuel behaviour during RIA and LOCA transients in the frame of IPSN and now IRSN programmes devoted to reactor safety. This operation was interrupted in 2003 to allow for a whole facility renewal programme. The main goal of this reconstruction project is to meet thermal hydraulics parameters identical to LWR standard and downgraded conditions, in particular for the need of the CABRI International Programme (CIP) carried out by IRSN under the OECD umbrella. For this, the sodium cooled experimental loop is now being replaced by a pressurized water loop.

In addition, several key safety issues of the facility have been revisited in order to defend a comprehensive safety case before the safety authority. First item in the case is of course the core. The aim of this paper is to present the path leading to a new core operations domain through expertise, mechanical tests and numerical computations. The reconstruction project is funded by IRSN.

CABRI is a pool-type reactor. The 30 cm diameter pressurized water test loop is installed at the core centre to receive an experimental device. A vertical symmetrical channel across the core allows the hodoscope, a neutron camera, to monitor the evolution of the axial fission distribution in the experimental rod during the experiment.

The core is made of 1488 stainless steel clad fuel rods of 80 cm with a 6% ^{235}U enrichment. The reactivity is controlled via 6 assemblies of 23 Hafnium rods. In addition, the key feature of the CABRI core is its reactivity injection system: a device of 96 tubes filled with ^{3}He (strong neutron absorber) which can be depressurized very rapidly in a discharge tank.

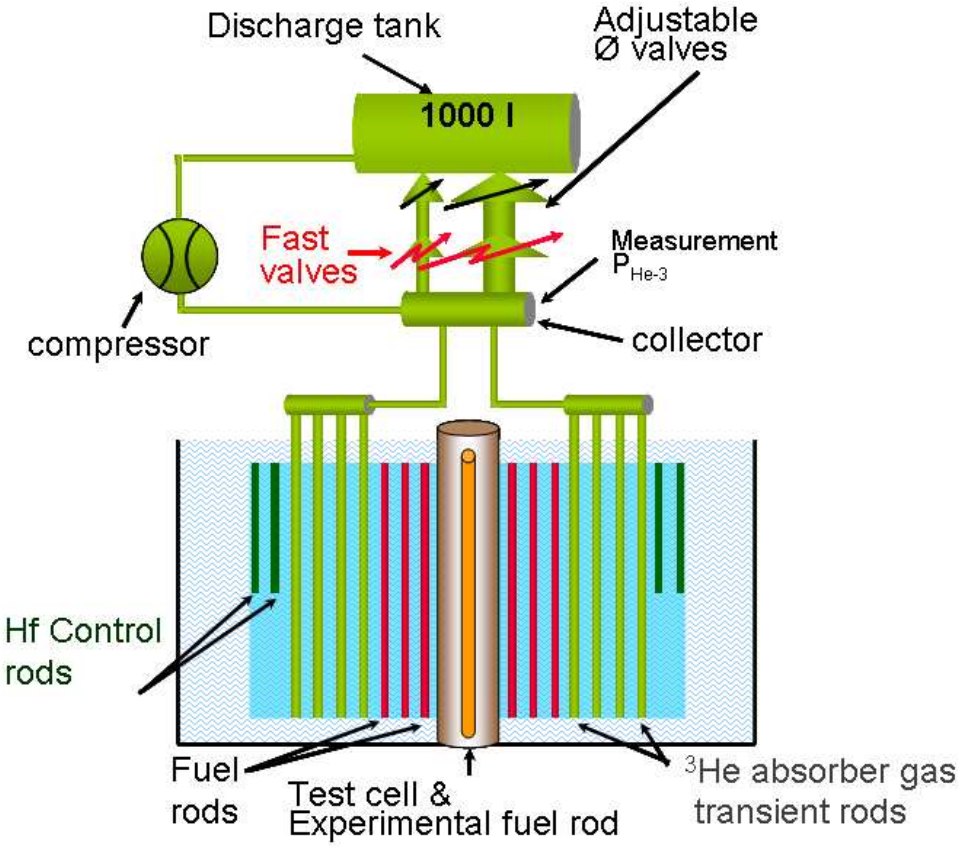


Figure 1: Principle of operation of the CABRI facility.[ref. 3]

II. PHENOMENOLOG+Y

To prepare the experiment, the helium tubes are pressurized (the maximum allowable pressure is 15 bars). The critical state is then obtained by control rod withdrawal to compensate for the ^{3}He antireactivity. It should be mentioned that for different helium pressures, the Monte-Carlo code, TRIPOLI4 [2], is able to predict the critical position with a very fair accuracy as opposed to measurements made in the past program. That means helium reactivity is well known in the CABRI core at this initial state.
The principle of operation for the CABRI facility is that a fast ejection of ^{3}He after opening of the circuit valves translates into the equivalent reactivity injection, then inducing a possibly supercritical power burst.
To start the experiment, the aperture of two adjustable valves (see figure 1) is fitted to the desired ^{3}He depressurization kinetics. For large opening valves and high pressure of helium (10 to 15 bars), the reactivity injection can reach 3-4$ in 10 ms. Consequently, a ~20 GW peak power and a ~1500 degree fuel temperature increase are generated in a few ms. Then, the power decreases mainly due to the Doppler effect and also to a minor delayed feedback. One second later, the reactor is automatically shut down by dropping all the control rods into the core. The full transient can release more than 200 MJ in the whole core.

III. CORE NUMERICAL CHARACTERIZATION

To take into account the evolution of the installation and because the last characterization of the core was made 30 years ago, it was important to re-evaluate the neutronics parameters and to improve the interpretation of experimental results before the reactor startup in 2009. The aim is to be able to predict the energy release for future experiments and especially for accidental situation due to unexpected depressurization (reference accident scenario of the safety report).
Transient consequences are strongly connected to pressure kinetics and feedback parameters. Therefore, they have to be determined with a high accuracy with 3D Monte Carlo reference calculations like MCNP [1] or TRIPOLI4 [2]. These calculations are well adapted because the fuel can be considered as fresh: the very short operation time cumulated over past programs allows for neglecting fuel depletion.

III.a Neutron generation time

The effective (i.e. adjoint weighted) neutron generation time (Λ_{eff}) is defined as:

$$\Lambda_{eff} = \frac{\left\langle \Phi^*, \frac{1}{v}\Phi \right\rangle}{\langle \Phi^*, F\Phi \rangle}$$

where Φ^* is the critical adjoint flux, v the neutron speed, Φ the critical flux and F the total (i.e. prompt and delayed) fission operator of the reference critical configuration.

Verboomen *et al.* [ref. 4] have implemented in the Monte Carlo code MCNP(X) [1], an asymptotically exact method based on perturbation theory for the calculation of Λ_{eff}.
Using the exact perturbation formula for reactivity increments and considering the special case of a poison with a c/v absorption cross section (where c is a constant and v is the neutron speed), Verboomen shows that the derivative of the reactivity increment for this perturbation in c = 0 will be exactly equal to the negative value of the adjoint neutron generation time.
The perturbation was chosen sufficiently small (300 pcm) to be in the range of applicability of the first-order perturbation theory but large enough to keep a good statistical precision (~ 5% at 2σ). For our simulations, the corresponding value of c is around 50 s^{-1}.

We have benchmarked this method against measurements of the MUSE4 experimental program in the MASURCA fast reactor and usual deterministic calculations in the OSIRIS and JHR reactors [5].

	Expected values	MCNP results
MASURCA (MUSE4 experiment)	56 ± 5 µs	53 ± 5 µs
OSIRIS (deterministic calculation)	34.7 ± 2 µs	34.0 ± 1.1 µs
JHR (deterministic calculation)	36.0 ± 2 µs	38.7 ± 1.4 µs

Table 1: Effective neutron generation time (Λ_{eff}) in MASURCA, OSIRIS and JHR reactor

The very good agreement between MCNP results and expected values allows us to trust this methodology. The application to CABRI with the JEFF-3.1 nuclear data library gives:

$$\Lambda_{eff} = 27.7 \pm 2\ \mu s$$

Remark: The effective neutron generation time calculation was also calculated with TRIPOLI4. The result is consistent: $\Lambda_{eff} = 27.2 \pm 1.5$ µs

III.b Delayed neutron fraction

The effective (i.e. adjoint weighted) delayed neutron fraction (β_{eff}) is defined as [6] :

$$\beta_{eff} = \frac{\langle \Phi^* \chi_d \nu_d \Sigma_f \Phi \rangle}{\langle \Phi^* \chi \nu \Sigma_f \Phi \rangle}$$

where Φ^* is the critical adjoint flux, Φ the critical flux, ν and ν_d the average neutron multiplicity per fission (prompt and delayed), χ and χ_d the neutrons energy spectrum (prompt and delayed), Σ_f the macroscopic cross section of fission.
Its computation hence requires a calculation of both the direct flux and the adjoint function to calculate the weighted production rate.
In Monte Carlo calculations, the physical processes are simulated as realistically as possible. For the development of a Monte Carlo method for the β_{eff} calculation, it is therefore useful to look at the physical interpretation of the adjoint function.
Considering the introduction of a neutron with properties *r* , E , and Ω in a critical system, this neutron will produce other neutrons by inducing fission. These neutrons will then produce fissions and then new neutrons, etc. The number of fissions produced by this way will approach a limit, given by the iterated fission probability. The reference 7 shows that this iterated fission probability is proportional to the adjoint function.
Meulekamp [ref. 8] develops a methodology based on this interpretation to determine β_{eff} in Monte Carlo calculations : for each neutron history initiated in the Monte Carlo run, the code remembers whether it was “prompt” or “delayed” at its creation. During the simulation of the Monte Carlo history of that neutron and all secondary particles, the corresponding number of simulated fissions is recorded. The entire fission chain due to this one neutron should be simulated as one “history,” while counting the number of fissions along the way. At the end of the history, the number of counted fissions is directly proportional to the iterated fission probability, and therefore, proportional to the adjoint function. This is true whatever the status of the initial neutron (prompt or delayed). Using the number of counted fissions and using the knowledge of whether the neutron was prompt or delayed at its creation, one can easily calculate β_{eff}. In fact, many histories will go on indefinitely. This is why in most Monte Carlo codes there is a specific *keff* mode in which the concept of a history is changed. Using the “*k*-code” mode in MCNP, a history starts when a neutron is generated from a fission and finish when the neutron produces a new fission or is removed from the system. Then, the code follows “generations” of neutrons. A major advantage of this history definition is that histories

will stop with a reasonable time. We have benchmarked this method against measurements and usual deterministic calculations in Materials Testing Reactors (non irradiated fuel) [5]. MCNP calculations are performed with the JEFF-3.1 nuclear data library.

	Experimental values	MCNP results (2σ)
Godiva	659 ± 10	673 ± 4
Popsy	276 ± 7	282 ± 2
Mistral 1	788 ± 11.8	791 ± 3
Mistral 2	370 ± 6	365 ± 2
Masurca	331 ± 5	329 ± 3

Table 2: Comparison of experimental and computational values of β_{eff}

	Expected values	MCNP results
OSIRIS (deterministic calculation)	736.8	737.8 ± 1.7
JHR (deterministic calculation)	735	730.2 ± 1.7

Table 3: Effective delayed neutron fraction (β_{eff}) in OSIRIS and JHR reactor

The very good agreement between MCNP results and expected values allows us to trust this methodology. The application to CABRI gives:

$$\beta_{eff} = 758 \pm 2 \; pcm$$

III.c Doppler coefficient

Using the reactivity determined at different fuel temperatures by the TRIPOLI4 Monte-Carlo code, it is easy to calculate the Doppler coefficient.
Usually, this Doppler coefficient, defined in pcm/K, is strongly depending on the temperature (approximately 1/√T evolution). Historically, the CABRI Doppler coefficient was defined in pcm/$K^{0.5}$, in order to use a constant which takes into account the typical temperature dependance.

$$\Delta\rho^{doppler}(t) = A_D.\sqrt{\Delta T}$$

where $\Delta\rho^{doppler}(t)$ is the Doppler Reactivity (pcm), A_D the Doppler Coefficient (pcm.$K^{-0.5}$) and $\Delta T(t)$ the fuel temperature variation during the transient (K).

TRIPOLI4 computations show the weak Doppler coefficient variation for the fuel temperature range 20-1500℃ (see figure).

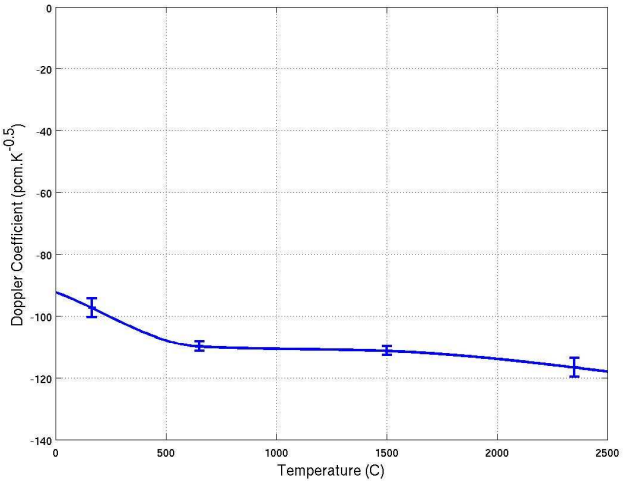


Figure 2: Dependence of Doppler coefficient on fuel temperature

The currently chosen value is 103 pcm.$K^{-0.5}$, what corresponds to the mean range of Doppler coefficient variation. In a short term, a real temperature dependent Doppler coefficient could be implemented in the model.

IV. VALIDATION

The CABRI-REP Na program, initiated in the early 90s, provided many experimental data: several power bursts have been analyzed with different Helium-3 initial pressures and different valve opening rates. This data bank will be used to validate the kinetics model including the key parameters calculated above.

Using experimental power measurements during the bursts, the time-dependent reactivity in the core can be provided by inverse kinetics. The calculated reactivity is the sum of helium reactivity (positive component) and the feedback reactivity (negative component), shown in figure 3 below.

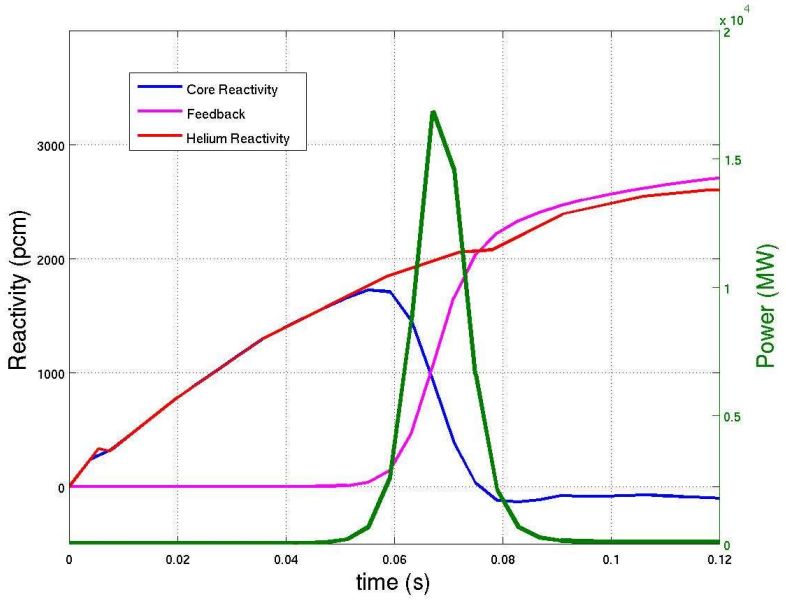


Figure 3: Feedback and helium reactivity during the power pulse

The transient is adiabatic; during the pulse, the energy transfer to clad or coolant can be neglected. Simulations show that the temperature increase only affects the fuel. Therefore the feedbacks at this short time (0.1 second) are mainly driven by the Doppler effect.
The Doppler coefficient and fuel temperature variations are calculated by thermal computations in order to determine the Doppler feedback, which can then be subtracted from the core reactivity to isolate the helium reactivity. The helium reactivity assessment resulting from experimental power burst analysis can then be compared to the expected value, calculated with TRIPOLI4 static computations (see § II.). The static computation has also been validated by a comparison of measured vs. computed critical control rod levels as a function of ^{3}He pressure at steady state. This analysis corroborates the TRIPOLI4 - JEFF-3.1 $\rho(P_{He\text{-}3})$ function.

Using experimental results at different pressures, this approach was repeated to evaluate the helium reactivity with both methods separately. For the whole CABRI operating range, the figure 4 below shows the satisfying agreement between the static and dynamic reactivity.

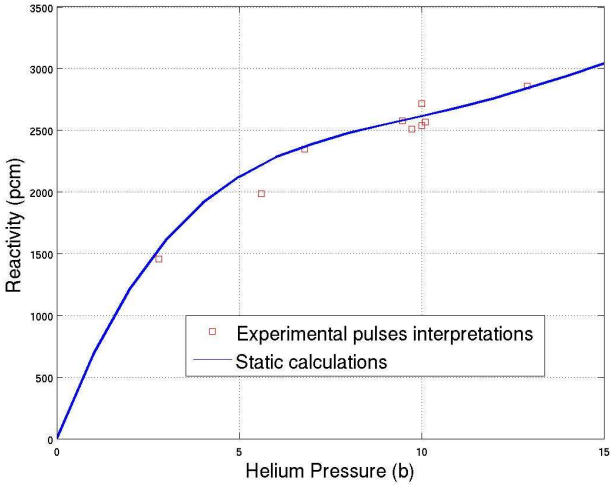


Figure 4: Comparison of the helium reactivity determined by two independent methods

These results validate the model and its set of parameters (Doppler coefficient and kinetic parameters).

V. PREDICTIVITY AND SAFETY ASSESSMENT

CABRI tests are characterized by the initial helium pressure and the valve aperture. An fluid dynamics model is used to calculate the depressurization kinetics and the aforementioned Monte Carlo calculations were applied to determine the helium reactivity at different pressures. Then, the obtained time dependent reactivity can be introduced in the DULCINEE point kinetics model.
The model is then able to predict the power and energy released during experiments. In order to assess the accuracy of predicted energy values, these simulations were performed for tests from the past, i.e. with the previous sodium loop, characterized by large valve apertures and high pressures. The results presented in table 4 show an agreement between measured and predicted energy better than 7%.

Experiment	Measured energy (MJ)	Predicted energy (MJ)	Agreement (%)
REPNA1	209	195	-6.6
REPNA2	204	197	-3.3
REPNA3	212	199	-6.2
REPNA5	213	203	-4.8
SU2REP4	224	227	1.2
S1REP1	172	176	2.8

Table 4: Comparisons of predicted and measured energy

After this validation, the same predictive approach has been used to characterize the energy release during the most severe possible accident. This scenario considers a simultaneous opening of both valves at full aperture when the initial helium pressure is 15 bars (the maximum allowed for the CABRI). The results obtained were increased by 7% corresponding to the maximum error observed during our simulations for this kind of transient. In the end, the calculated energy release is about 403 MJ. It was then introduced in thermo-mechanical computations [9] to attest that the consequences were acceptable to both fuel and cladding. These calculations belong to the safety case, to demonstrate the structural integrity of the core during this reference accident [10].

VI. CONCLUSION

To take into account the evolution of the CABRI facility and because the last core characterization was made 30 years ago, the main neutronics parameters driving a power pulse (prompt neutron lifetime, delayed neutron fraction and Doppler coefficient) were re-evaluated with a high accuracy. Each parameter was calculated by 3D Monte Carlo reference codes and with a validated and new methodology.
Then, all data were used in the DULCINEE code, which includes thermal and point kinetics models, to simulate the dynamics effects in CABRI transients and to predict the energy release during a power pulse. The comparison between experimental results and computational results shows the model is able to provide predictions with an accuracy better than 7%.
This model is consequently used to assess the power and energy release during the reference accident of the CABRI facility. Considering these conservative results, thermal and thermo mechanical computations attest the fuel and cladding integrity during the transient.

ACKNOWLEDGEMENTS
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