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### ► **To cite this version:**

B Fontaine, L Martin, G Prulhiere, R Eschbach, J.-L Portier, et al.. Recent analyses of PHENIX End of Life Tests and perspectives. IAEA International Conference FR13, Mar 2013, Paris, France. <hal-04807194>

**HAL Id: hal-04807194**

**<https://hal.science/hal-04807194v1>**

Submitted on 27 Nov 2024

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## Recent analyses of PHENIX End of Life Tests and perspectives

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*Presented by B. FONTAINE*

**Abstract.** This paper presents the recent analyses of Phénix End of Life tests performed with the most advanced modelling developments, and a global assessment of the campaign. The following items are specifically touched upon: core decay heat removal measurement and associated uncertainties, natural convection in reactor vessel using coupled system and CFD models, and neutronic/mechanical behaviour of the core during flowering test.

Finally, it appears that the Phénix decay heat measurement test can be added to the validation basis of DARWIN code associated to JEFF3.1.1. The discrepancies with experimental data are weak (under +/- 8%) and globally covered by uncertainties. Concerning the primary natural convection test, the use of a CFD/system coupled calculation allows to model very accurately the sodium temperatures during the transient. The behaviour of the core during static flowering test can be well estimated by using HARMONIE code for mechanical aspects and TRIPOLI for neutronics. A new test is scheduled in Phénix to assess the core mechanics code in case of dynamic load.

### INTRODUCTION

A campaign of “End-of-Life” tests was jointly performed by CEA, EDF and AREVA in 2009 at the PHENIX French sodium fast reactor prototype, before final shut-down of the plant. The tests covered a wide range of domains: core physics, thermal-hydraulics, mechanics, fuel and some investigations related to the negative reactivity transient events that occurred in 1989 and 1990 and are not yet fully explained.

A first set of interpretation work was carried out in the past years and was published [1]. This paper will present some deeper analyses performed recently associated to further modelling developments, and a global assessment of the campaign. Some international cooperation studies have been going on regarding some tests.

The following items will be more specifically touched upon: core decay heat removal measurement and associated uncertainties, neutron physics test interpretations and feedback on codes, the primary system thermal-hydraulics, using coupled system and CFD models and the mechanical behaviour of the core.

A further test, planned in 2013, before the core unloading, will be finally described. The test will aim at assessing the mechanical dynamic behaviour of the core.

## DECAY HEAT MEASUREMENT TEST

Decay Heat (DH) is an important concern to the nuclear industry. If not extracted, the heat produced by the decay of non stable fission products and heavy nuclides can lead to the meltdown of the fuel. Consequently, as decay heat has a deep impact on the design of the emergency cooling system, the cooling time necessary before maintenance in a reactor, the design of waste transport casks and storage facilities, a specific test has been performed in Phénix reactor in 2008 to measure it. This measurement will be useful to validate the codes used to design the future sodium fast reactors.

### Test description

A complete description of the test is available in [2]. Few days before the test, the reactor was operated under nominal conditions. Then, a fast shutdown was carried out. One and a half hour after the shutdown, the reactor was considered to be in isothermal conditions and behaved as a calorimeter. The decay heat is derived from sodium temperature evolution thanks to an energy balance equation :

$$DH = I \frac{d\theta}{dt} - P_{sources} + P_{losses}$$

Where DH is the estimated decay heat, I the reactor vessel inertia,  $\theta$  the measured temperature of the sodium,  $P_{sources}$  and  $P_{losses}$  the sources and the losses of power in the reactor block.

### Decay heat calculations

In Phénix, the decay heat is generated by several sources : fuel and radial blanket assemblies of the core, spent fuel located in the internal vessel storage, neutron shielding, sodium and vessel structures. Actually, 80% of decay heat is produced by the core. To evaluate the contribution of each source to the uncertainties, a deterministic code, namely CYRUS, has been developed to propagate nuclear data uncertainties on nuclide inventory and decay heat calculations, and applied to Phénix experiment.

It appears that the uncertainty on total decay heat calculated with DARWIN and JEFF-3.1.1 ranges between 2% and 4.5% (see figure 1), depending on the integration or not of the contributions of shielding and structures. This result shows the necessity to calculate accurately neutron flux in those peripheral areas and to have reliable nuclear data for activation reactions (for example  $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$  and  $^{62}\text{Ni}(n,\gamma)^{63}\text{Ni}$ ).

Taking into account these aspects, the experimental data and decay heat calculations are finally well consistent since both uncertainty domains are overlapping (see figure 2).

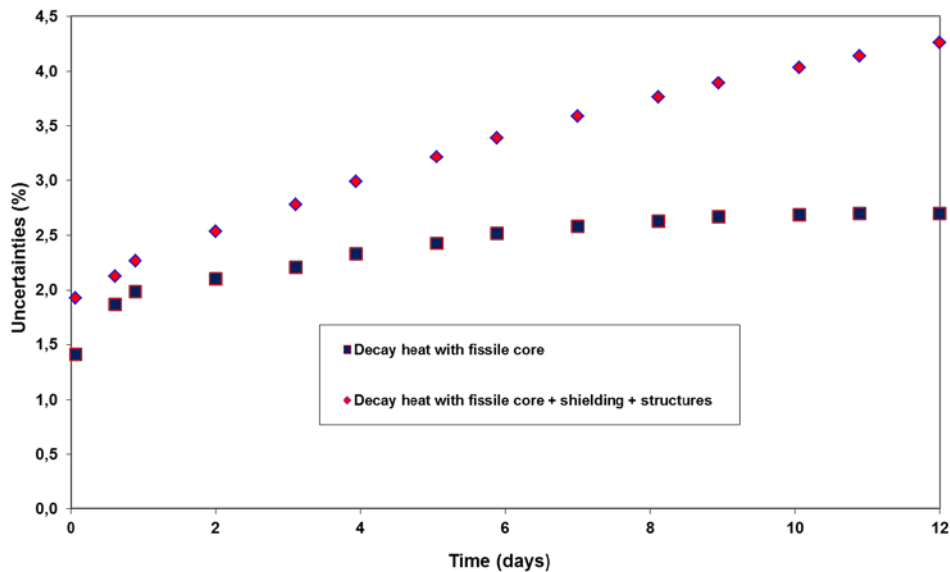


Figure 1 - Decay heat uncertainties calculated with Cyrus

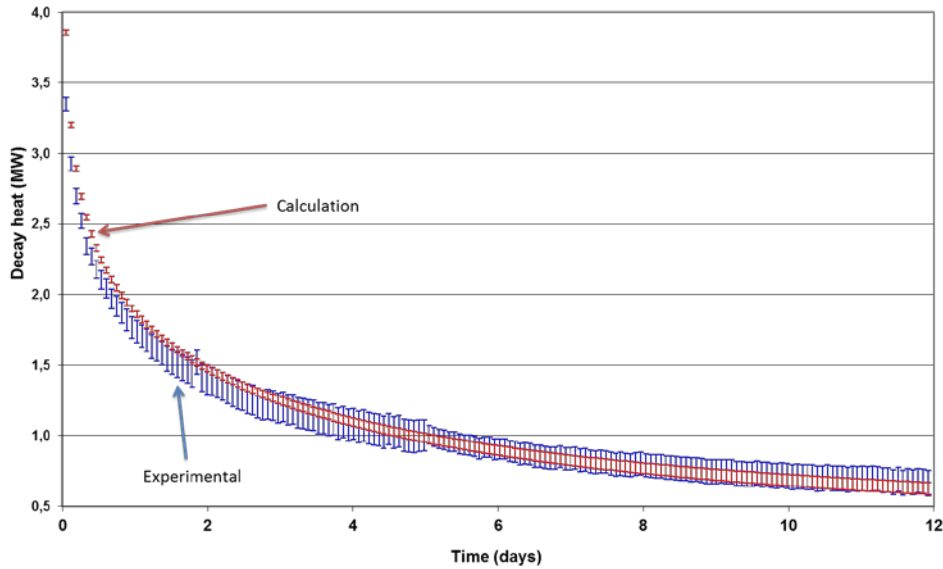


Figure 2 - Comparison of calculated and measured decay heat

As a conclusion, the Phénix decay heat measurement test can be added to the validation basis of DARWIN code associated to JEFF-3.1.1, as a global validation test for SFR applications. The discrepancies with experimental data are weak (under  $\pm 8\%$ ) and globally covered by uncertainties. Nevertheless the experimental uncertainty remains high, a more precise experiment in a controlled neutronic environment (such as experimental reactor) would be necessary to reduce this uncertainty.

### NATURAL CONVECTION TEST

One of the main goal of the Phénix natural convection test (NCT) is to validate thermalhydraulics codes with respect to the establishment of natural circulation in the primary system of a pool type SFR. The present section presents the calculation of the NCT by coupling the 3D computational fluid dynamics (CFD) code TRIO\_U with the best estimate thermal hydraulic system code CATHARE. It is also mentioned that Phénix NCT model has also been simulated at the system scale by CATHARE alone [3][4].

### Methodology

Many SFR transients involve both the system scale, with flows and temperatures varying in different components and circuits, and local 3D effects where fluid behavior can not be modeled by 1D approach. Local simulations accuracy is usually limited by boundary conditions, but it is impossible to model the whole reactor in CFD. On the other hand, system codes do not provide capabilities to account for local extrema or fluctuations. Therefore, it appears necessary to couple a system code with a CFD code.

Because of its high computational cost, CFD-models must be used only where essential. The first phase of the Phénix NCT, a steam generator drying operation, is characterized by a hot-shock occurring at the intermediate heat exchanger (IHX) primary outlet window. During this unprotected phase of the transient, the temperature at the pump inlet is a key physical parameter for the determination of the core inlet temperature and consequently on the core neutronics. Physically, this parameter is determined by the complex 3D flow paths of the cold pool, which are likely to evolve during the transient due to buoyancy forces. After the reactor scram, the second phase begins and natural circulation rapidly takes place in the primary sodium system. It has been shown that the IHX primary inlet temperature is of greatest importance for the determination of the natural circulation mass flow-rate. This is particularly true in situation where the primary/secondary heat transfer-rate is low. The beginning of the second phase of the Phénix NCT is an example of such a physical situation.

Physically, the IHX primary inlet temperature is again determined by 3D complex thermalhydraulic effects occurring in the hot pool. It seems therefore reasonable to prescribe the use of a CFD model for both cold and hot pool. Another argument in favor of CFD-modeling for the hot and cold pool is related to the instrumentation that produces the experimental data used for the validation: some important bias effects, originating from 3D thermal hydraulics phenomena (e.g. thermal stratification, mixing with surrounding sodium ...), can be feared in the natural convection regime. Consequently, for validation purpose, these sensors have to be integrated to CFD-based models able to deal with the above-mentioned local phenomena.

The coupling methodology used is of the domain-overlapping type associated to a defect correction approach. The coupling scheme has been verified by cross-comparing the numerical predictions of a CATHARE reference calculation to a CATHARE/CATHARE coupled calculation. The coupling software architecture relies on a common application programming interface (API) named ICoCo developed by CEA for coupling activities. The methodological aspects related of this work are detailed in [5]. Figure 3 presents the initial state at the beginning of the NCT obtained by the 1D/3D coupled model.

## **Results**

Numerical vs. experimental comparison has been performed against measured quantities consisting of inlet/outlet primary and secondary IHX temperatures, primary pump inlet temperature, core outlet temperature, primary sodium mass flow-rate as well as level measurements in the hot pool, the cold pool and in the main vessel cooling system. For the sake of brevity it is not possible to present all the available results here. Nonetheless, an interested reader can refer to [5] to have a better insight into the predictions of the 1D/3D coupled calculation. It was chosen here to present the evolution of the primary IHX outlet temperature, which illustrates the interest of thermalhydraulic multi-scale coupling for addressing SFR transient analysis.

Obtaining the 120 MWth steady state requires several thousand seconds CFD/system coupled calculation in order to correctly evaluate the cold pool thermal stratification. Stratification originates from both the heating by the hot pool and the cooling by the main vessel cooling system. It occurs at elevations where sodium is poorly mixed by the primary flow, i.e. above the top of the IHX outlet windows. Thermal stratification of the cold pool was found to significantly alter the sodium temperature at pump inlet during phase 1 of the NCT.

During the initial steady state, the primary mass flow-rate, as well as the temperature at the inlet/outlet of the main components provided by the CFD/system coupled calculation were found to be very similar to that of the system stand-alone calculation.

### **Primary IHX outlet temperature**

The transient evolution of the primary IHX outlet temperature, as measured by a sensor located at the bottom of the IHX outlet window is shown in Fig. 4. This sensor is immersed in the sodium stream exiting the IHX in the forced convection regime. The 60 K increase, resulting from the steam generator dry out during phase 1 and during the beginning of phase 2, is very well reproduced by the CFD/system coupled calculation. A 15 K temperature drop is then observed (at  $t = 5490$  s), 24 s after the primary pump trip. Visualization of 3D temperature and velocity fields have revealed that this temperature drop originates from the transition between the forced convection to the natural circulation regime. Indeed, when buoyancy forces become predominant, the sodium stream exits the IHX at the top of the outlet window and the measurements of the sensor experiences a sudden shift from the average outlet temperature of the IHX to the local temperature of the cold pool which happens to be 15 K lower than the former one at this stage. The predictions of the CFD/system coupled calculation are also in very good agreement with the measured values during phases 2 and 3. To illustrate the effectiveness of coupling CFD and system codes in addressing SFR transient analysis, the predictions of the system code alone were also shown in Fig. 4. Three liquid temperatures corresponding to the 0D elements located at the elevation of the sensor are presented (the system code model of the cold pool is azimuthally sectorized into three equal parts).

More generally, the spatially averaged quantities calculated by the system code present smoother evolutions than the experimental values obtained from local measurements. If correct, the temperature field obtained from the CFD/system coupled calculation may eventually be used to define the spatially averaged quantities that could be directly compared to the results of a system code. This could lead to a more accurate validation procedure for the use of system codes to simulate SFR.

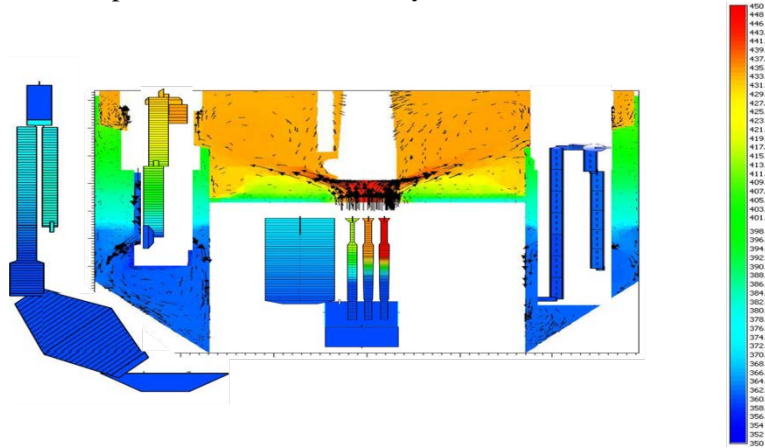


Figure 3 - Initial temperature and velocity fields in the primary circuit

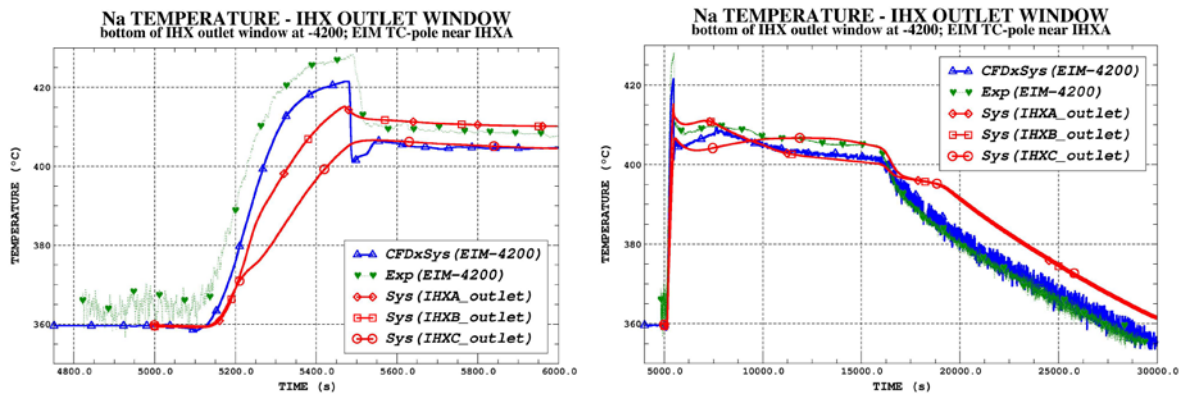


Figure 4 - Short- (left) and long- (right) term evolutions of the IHX primary outlet temperature (transient starts at  $t_0 = 5000$  s).

### CORE MECHANICAL FLOWERING TEST

A core flowering test was carried out by introducing a mechanical pushing device inserted in the code sub-assembly lattice at two different positions: at the center and at a the periphery (figure 5). The effect of core flowering was measured at different temperatures in the range of 180 °C to 350 °C. The purpose of this test was to measure the reactivity effect of such core deformation at subcritical and at critical zero power conditions, and to measure the core mechanical behaviour under an imposed quasi-static flowering.

During the test, the measurements relative to core mechanics were :

- pushing forces necessary to make the core flower,
- radial displacement at the top level of four assemblies located at 3<sup>rd</sup> and 7<sup>th</sup> ring (see fig. 5) The devices used for the measurement were ultrasonic transducers (SONAR system).

A detailed description of the test and instrumentation is available in [6].

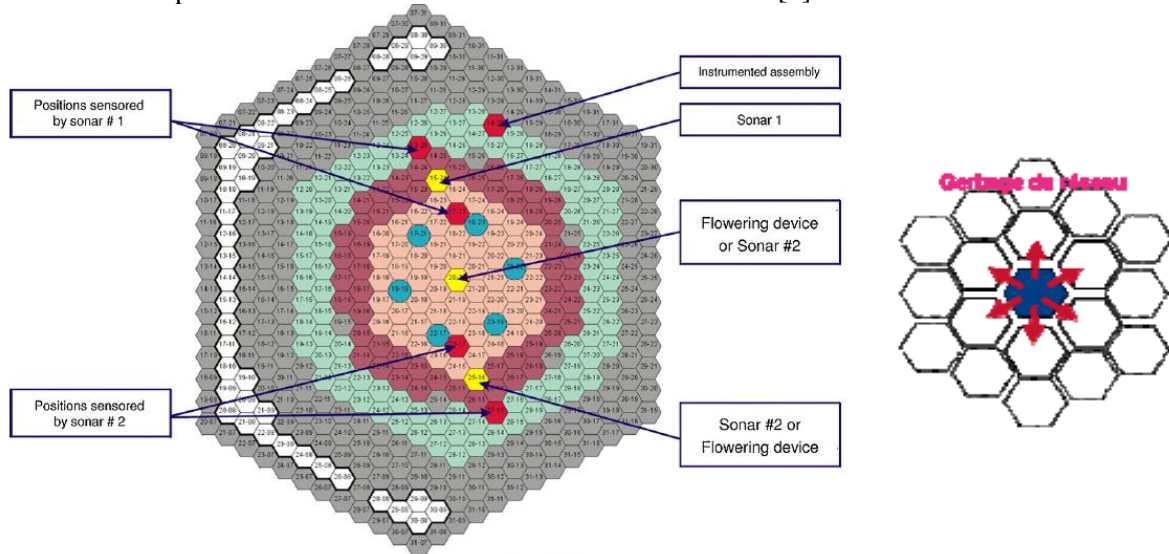


Figure 5 - Phénix core description and principle of core flowering device

### Mechanical aspects

The maximal displacements of the assemblies and the interaction forces between assemblies were estimated for all experimental configurations when the flowering device was partially and completely infolded. These estimations were performed using the static mechanics code, HARMONIE, which is dedicated to the calculations of SFR core deformations.

For this analysis, the mean gaps between assemblies were calculated, taking into account the differential thermal expansion between assembly wrapper tubes and the core support plate. As the material of wrapper tube is EM10 ferritic-martensitic steel with good properties under irradiation, no irradiation swelling and creep was first considered.

The results of these calculations show that HARMONIE was able to estimate the good order of magnitude of the core stiffness. The prediction of individual assembly displacements was more difficult to achieve, due to the local distribution of gaps between assemblies which are very sensitive to manufacturing allowances and small deformations.

To complete the validation of HARMONIE, another specific test was performed in PHENIX after flowering test. It consists in measuring the vertical forces necessary to lift subassemblies during fuel-handling operation. The calculated forces issued from HARMONIE were very similar to those measured.

As the flowering test performed in 2009 was quasi-static, it would be now interesting to investigate experimentally the core behaviour under dynamic load. Indeed, when core is submitted to dynamic movement, hydrodynamic forces due to sodium acceleration occur and may strongly affect the assembly displacements. In order to observe such behaviour in fast reactor cores, a new test is scheduled in PHENIX in 2013. The aim of this test is to make the core flowering dynamically in order to observe assembly displacements of about 10 mm. This test will be performed at shutdown state, as the reactor is now in final shutdown conditions.

The test will provide a better knowledge of core dynamic behaviour and experimental data for validation of the core mechanical modeling (French CAST3M code).

A specific device for dynamic flowering actuation is being designed on a “pushing apart/release” principle. The displacement of the sub-assemblies will be measured with the ultrasonic system already used for the static test and with a new device based on deformation gauges.

### **Neutronic aspects**

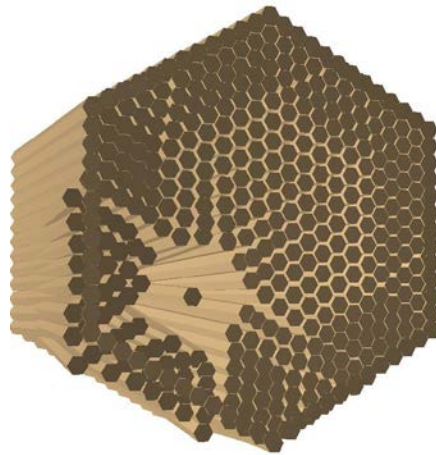
During PHENIX flowering test, the core reactivity was monitored at zero power state and compared afterwards to calculation results.

Each assembly displacement calculated previously with the HARMONIE code at midcore level was transferred to the neutronic TRIPOLI code based on Monte-Carlo method.

In the TRIPOLI model, the distorted position of each subassembly was considered for every flowering states, taking into account two assumptions concerning the axial deflection : pure translation or beam bending. Some examples of core deformations simulated with TRIPOLI and considering the bending assumption are presented in figures 6 and 7. In the translation assumption, the bending of the assemblies is not considered and displacements calculated by the HARMONIE code at midcore level were applied all along the assemblies. This assumption seemed reasonable considering that the maximum bending deflection is lower than  $0.16^\circ$ .

In figure 7 are compared experimental and calculated reactivities for different flowering states with spreading device located at core center and two isothermal temperatures ( $180^\circ\text{C}$  and  $357^\circ\text{C}$ ). One may observe from these results that flowering always leads to negative effects. A small flowering generates an internal arrangement of fuel assemblies in a fissile core but does not still affect fissile core limit. The evolution of reactivity remains slight because the fuel displacements occur in a zone where the adjoint flux is almost uniform. With a larger flowering, the external row of fuel elements begins to move changing the fissile core size. The effect on the reactivity is greater and we can now observe an increase in the curve slopes. The discrepancies between calculations and experimental data remain relatively small, in the range of statistical errors of the TRIPOLI results ( $\sigma = 10$  pcm).

The next step of the analysis will be the use of deterministic code, namely ERANOS/PARIS, to calculate reactivities in order to assess this type of codes for the evaluation of neutronic impacts of SFR core deformations.



*Figure 6 - Example of TRIPOLI core model with non-centred flowering device*

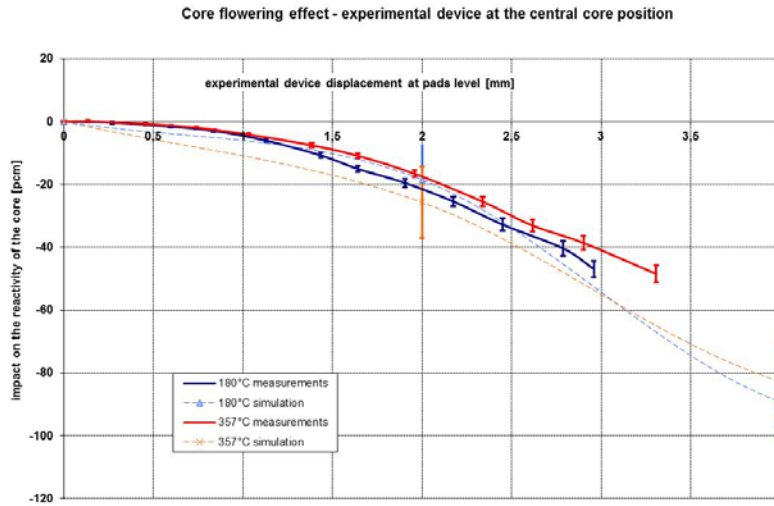


Figure 7 - Calculated and measured reactivities vs. core flowering at 180°C and 357°C

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