Analysis of the feedback coefficients of the Superphenix start-up core with APOLLO3

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Abstract

The development of Sodium-cooled Fast Reactors (SFRs) is promoted by the Generation IV International Forum (GIF) since its design and operation is competitive against other technologies and projects. In France, considerable experience has been acquired with three experimental facilities: Rapsodie, Phénix and Superphénix. The Superphénix was a large-size SFR that remains as a unique source of data for these cores. Particularly, during the start-up a set of tests were performed to check the safety criteria of the core in which the feedback coefficients of the core were assessed. These feedback coefficients are the *k* (1°C variation at the inlet of the core temperature), *g* (1°C variation in the sodium temperature elevation through the core) and *h* (1% variation of the nominal power). In this paper, the evaluation of these coefficients is performed using two neutronic platforms: ERANOS and the new APOLLO3. By using these two codes, the elementary feedback coefficients are calculated and allow evaluating the global feedback coefficients with a simplified model. The results show the same trend than measurements for the *k* and *h* coefficients, while for the *g* coefficient a significant discrepancy is observed at 80% nominal power. The use of a thermal-hydraulic system code such as CATHARE-3 (eventually complemented by CFD calculations) is envisaged to better understand the sources of this bias, which might be due to the simplistic thermal hydraulic models unable to tackle some physical effects of the core operation.

KEYWORDS: Superphénix, feedback coefficients, APOLLO3.

Introduction

The Generation IV International Forum (GIF) has selected several advanced reactor concepts to be developed in the research and industry roadmap of nuclear power, being one of these the Sodium-cooled Fast Reactor (SFR). A considerable amount of operation experience has been accumulated for SFRs, for instance, the French experimental program provides rich feedback from three reactors: Rapsodie, Phénix and Superphénix. The latter of these three facilities was a large-size SFR with a thermal power of 3000 MWth that operated between 1985 and 1997 and is currently in decommissioning process. During the Superphénix start-up, a set of tests were performed to check the core's behavior, which currently are a unique source of SFR data.

The SFRs design must be performed in such manner that during incidents or transients no major affectation is seen in the core, avoiding severe accidents from its proper design. However, transients in SFRs present a particular performance with interrelated feedback coefficients, which hardens the predictability of the core behavior under such situations. For this reason, the whole core behavior is decomposed in elementary feedback coefficients that separately characterize the core performance in the most fundamental responses for the SFR design.

Given the importance and complexity of transients in SFRs, the production of the elementary feedback coefficients is of major importance for its analysis. Its evaluation for the Superphénix core at different power conditions is assessed with the novel neutronic platform APOLLO3. The calculated elementary feedback coefficients with APOLLO3 are then compared with those obtained with former codes, such as ERANOS, which would enable to verify the code's consistency results.

The Superphénix core

The Superphénix was a large-size SFR that operated between 1986 and 1997 in the south of France. The Superphénix core had a nominal power of 3000 MW_{th} with a pool-type vessel design; it was the successor of Rapsodie and Phénix with a commercial design to eventually deploy a SFR power fleet in France.

The Superphénix was constituted by 358 fuel subassemblies divided in two cores, inner (190 sub-assemblies) and outer (168 sub-assemblies) cores and a total of 222 fertile subassemblies were radially surrounding these assemblies. The main control rod system (SCP, Système de Commande Principale) was constituted by 21 subassemblies divided in two curtains, the first composed by 6 subassemblies and located in the inner core, the second composed by 15 subassemblies located in the interface between the inner core and the outer core. These SCP sub-assemblies contained 31 absorber pins of B_4C enriched in ^{10}B at 90%. Additionally, to limit the reactivity excess at the start-up 18 dummy assemblies were loaded next to the SCP assemblies [1].

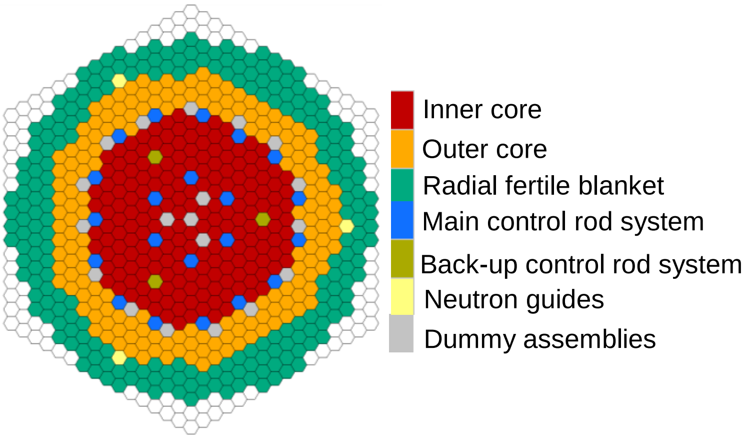

Figure 1 : Superphénix core layout.

During the start-up a set of tests were performed to check the safety criteria of the core in which the feedback coefficients of the core were assessed [2]. The evaluation of the feedback coefficients serve to determine and characterize the final state of a perturbation at a given power. The main advantage with using these coefficients is that they can be measured in SFRs during the start-up tests as they depend of well known operational parameters: inlet and outlet sodium temperature and the thermal neutronic power [3]. They are defined as follows:

- $k = \delta\rho / \delta Ti$ (pcm/°C) that corresponds to a 1°C variation in the inlet of the core temperature Ti with P and ΔT fixed

- $g = \delta\rho / \delta\Delta T$ (pcm/°C) that corresponds to a 1°C variation in the sodium temperature elevation trough the core ΔT with P and ΔT fixed and

- $h = \delta\rho / \delta P$ (pcm/%nominal power) which corresponds to a 1% variation in the nominal power P with Ti and ΔT fixed.

Experimentally in the Superphénix, the *kgh* coefficients were determined by a three step procedure in which the core power (P), the inlet temperature (T_i) and the core temperature rise (ΔT) were traced.

In Superphénix, the measurement of $d\rho_{CR}$ was directly taken from the S curve of the SCP system, the dT_i was taken from a thermocouple device at the output of the primary circuit pump, the $d\Delta T$ was measured as the difference between the inlet temperature and the average of the measured temperature at the output of each of the core subassemblies and dP was directly measured from the power control interface [1].

The three steps were successively applied after reaching steady state as seen in Figure 2. These perturbations are the following:

- Control rod position (insertion of approximately -60 pcm)
- Secondary flow (≈-10%)
- Primary flow (≈-10%)

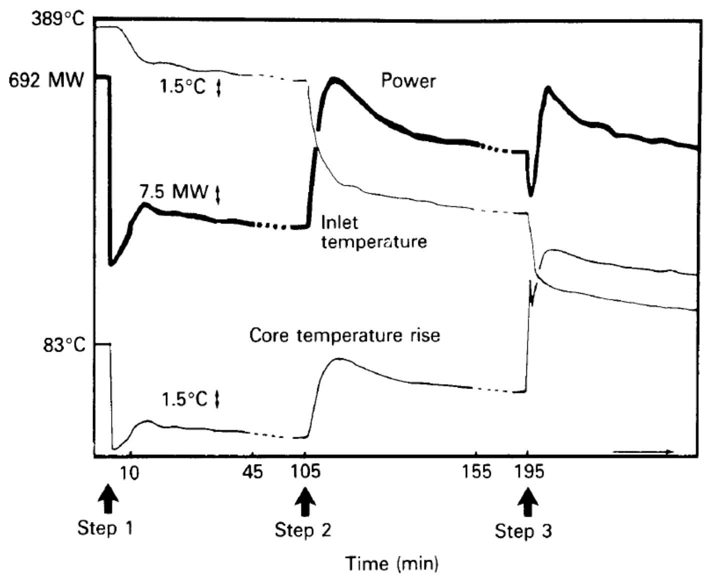

Figure 2 : Measurement of feedback coefficients step by step at 23% [2].

After each step a steady state was reached, and the reactivity change between two steady states is expressed as:

$$k \cdot dT_i + g \cdot d\Delta T + h \cdot dP = \, d\rho_{CR}$$

By measuring $d\rho_{CR}$, dT_i, $d\Delta T$, and dP the three feedback coefficients can be obtained by the resolution of a 3 X 3 system and determine the values of *k*, *g* and *h*.

Methodology

The safety of the core during transients is somehow complex to be determined for the interrelated effects happening in the global behavior of the core. For this reason, the global performance of the core is often decomposed in elementary coefficients to analyze its contribution to the total comportment of the core by the use of a neutronic code.

In this paper, the ERANOS [4] code and the new APOLLO3 [5] code are used to evaluate the elementary feedback coefficients. The route calculation of ERANOS consists in an assembly step calculation to produce effective cross sections with the ECCO code [6] and the TGV/VARIANT core solver (SPn) with the use of the JEFF 3.1.1 library.

The new APOLLO3 code developed by CEA, EDF and FRAMATOME includes the late neutronic improvements and provides great flexibility to the user. In this paper, the APOLLO3 code is used to produce these elementary feedback coefficients with an assembly step calculation with the Two Dimensional Transport (TDT) solver. Cross sections are self-shielded with the Collision Probability Method (CPM) using the Tone method [7], and the multigroup flux solution of Boltzmann neutron transport equation is computed with the Method of Characteristics (MOC) [8]. The use of the JEFF3.1.1 nuclear data is also used.

Core calculations are done with the MINARET [9] core solver and include a simplified core with 2pi/3 reflection and homogeneous control rod description. These elementary coefficients involve the neutronic perturbations on coolant, Doppler Effect, fuel expansion and structure expansion (including hexagonal can, clad, diagrid, and control rod position perturbations). The methodology used for the elementary feedback coefficient determination considers independence between each effect; this means that no interaction between elementary feedback coefficients is supposed. Each effect is defined as follows:

The Doppler Effect is related to the temperature variation of a given state, which affects the thermal motion of the target nuclei in the nuclear fuel. Consequently, the resonances in the isotopes cross sections are modified as temperature changes, thus, the reactivity varies depending on the fuel isotopes temperature. Its evaluation with APOLLO3 includes the production of a set of effective cross section for the fissile and the fertile media at the lattice step with the same nominal state geometry, but isotopes temperature is modified to the desired temperature. On the other hand the effect related to the steel is supposed to be linear as temperature increases. The Doppler constant is calculated as follows, considering a perturbation of 200°C for the Doppler temperature:

$$K_D = \frac{{}^{nom}_{dop}\Delta\rho}{ln\left(\frac{T_{dop}+273}{T_{nom}+273}\right)}$$

The sodium expansion effect is related to the coolant temperature of the core. The sodium density decrease reduces the absorption and scattering neutronic reactions on sodium. This leads to an increase in the core reactivity. However, the increase in the mean free path of the neutrons also leads to an increase in the neutron leakage, which has a negative effect on the reactivity. The first effect is mainly predominant at the core center, where the neutrons have low escape probability, while the negative leakage effect is mainly found at the core periphery. Its evaluation with APOLLO3 includes a 1% sodium density decrease at the lattice calculation in all regions of the core, which are used at the core level calculation. Its calculation is defined as:

$$C_{Na} = \frac{{}^{nom}_{Na}\Delta\rho}{\Delta T_{Na}}$$

The fuel expansion effect is the change of volume of the fuel from a nominal state and it can be separated in axial and radial effects. The axial fuel thermal expansion can have two approaches depending on the fuel situation of the reactor, if it is fresh fuel or with low burnup, it can be considered as free within the clad, which means that it is expanded with its expansion coefficient. However, if the fuel has a burnup higher than a few GWd/t, it sticks to the clad and hence its expansion is linked to the steel expansion properties of the clad. For both approaches, the fuel expansion is represented as a change of volume at constant mass, which perturbs the available number of neutrons for fission events. The radial expansion has no major impact since its change of volume does not replace any other material or component in the core, therefore it can be negligible. In order to avoid the control rod interaction of the SCP with fuel the nominal state is taken to be with the control rods at larger distance from the active core so that the fuel expansion effect is not influenced by the control rods, it is defined as follows, considering a 1% fuel density decrease:

$$C_{Ax-Fuel} = \frac{{}^{nom}_{ax-f}\Delta\rho}{\Delta T_{Fuel}}$$

The structure expansion effect is the change of volume of this element for a temperature variation. This change modifies the density of the structure and obviously perturbs the neutron interaction compared to the nominal state. It can be separated into hexagonal cans and pin cladding having the following behavior:

- Axial clad thermal expansion: the steel concentration decrease (1% density decrease) reduces the neutronic interaction; hence, a positive reactivity effect is seen.
- Radial clad thermal expansion: the steel volume increase replaces sodium volume which has a similar effect to the sodium density diminishment. It is evaluated as follows:

$$C_{rad-Clad} = \frac{C_{Na}}{\Delta T_{str}} * 100 * 0.02 * \left(\frac{1 - F_{TH} - F_{Na}}{F_{Na}}\right)$$

where:

F_{TH} is the hexagonal tube volume fraction in the subassembly

F_{Na} is the sodium volume fraction in the subassembly

- Axial hexagonal can thermal expansion: As in axial clad thermal expansion, the steel concentration diminishes (1% density decrease), reducing the neutronic interaction with this material, therefore a positive reactivity effect is expected.
- Radial hexagonal can thermal expansion: As in radial clad thermal expansion, the steel volume increase replaces sodium volume of the core. It can be evaluated as follows:

$$C_{rad-TH} = \frac{C_{Na}}{\Delta T_{str}} * 100 * 0.02 * \frac{F_{TH}}{F_{Na}}$$

where:

FTH is the hexagonal tube volume fraction in the subassembly
FNa is the sodium volume fraction in the subassembly

The diagrid expansion has a direct impact on reactivity since it modifies the distance between assemblies and therefore the quantity of sodium increases or decreases between assemblies depending on the expansion or contraction of this element. The diagrid thermal expansion considers the variation of 1% of the subassembly pitch in the core. At the lattice step, subassembly geometry is maintained, but an increase of the inter-assembly sodium is done to include the sodium volume increase at the assembly calculation level. At the core step calculation an increase of the subassembly pitch is done. It is calculated from the following expression:

$$C_{Dia} = \frac{{}^{nom}_{dia}\Delta\rho}{\Delta T_{Dia}}$$

Control rod driveline. As the whole systems of the vessel (Vessel, Core, Diagrid and Hot collector) presents a temperature change, the structure and fuel thermal expansion modify the position of the control rods relative to the core which also impact the reactivity. The reactivity change for the control rod position change is directly taken from the S curve of the SCP.

These elementary feedback coefficients were calculated at different power conditions, namely at 100%, 80%, 50% and 33%. This means that a set of the elementary feedback coefficient was produced for each power condition of the core. To determine the operation temperature of the core at different power conditions a fuel behavior analysis was performed with the GERMINAL [3] code to obtain the main hypothesis of the temperature regions of the core. As a preliminar approach, a simplified model does not account the temperature difference between the inner core and the outer core since it is considered as negligible.

Power conditions (in % of Nominal Power Pn)	Core temperature (°C)
100	1200
80	1020
50	770
33	620

Table 1 : Fuel temperature at different power conditions.

The global feedback coefficients are then evaluated with a simplified model, which is function of the calculated elementary feedback coefficients with APOLLO3 or ERANOS, based in Table 2 from reference [2]. Thought the pad effect is shown in Table 2 this effect was never seen in Superphénix, contrariwise to the observed on Phénix reactor.

Feedback coefficient		k (pcm/°C)	g (pcm/°C)	h (pcm/%NP)
Sodium expansion		+	+	
Structure expansion		+	+	
Axial fuel expansion	Free	+	+	+
	Linked	+	+	
Diagrid expansion		+		
Doppler		+	+	+
Control rod driveline position		+	+	
Pad effect			+	

Table 2 : Elementary feedback coefficients influence on global feedback coefficients.

The model to determine the *k* coefficient is simply the sum of the elementary feedback coefficients since they all are assumed to perturb the core as the core's inlet temperature changes. For the *g* coefficient the differential position of the SCP is assumed to be the main factor for its determination, however the other elementary feedback coefficients are also taken into account, but contributing at half its value. The h coefficient is strictly related to the value of the fuel/clad heat exchange coefficient H_{gap}.

A simplified exchange thermal model, considering the beginning of life state of the fuel (in free condition) was used, taking as hypothesis the GERMINAL results on the fuel pin behavior. The fuel/clad coefficient exchange was assumed to be 0.36 (W/cm^2K). Additionally, the hot and cold plenums were supposed to have punctual temperature state, without any temperature profile or stratification.

Results

The results concerning the elementary feedback coefficients with APOLLO3 and ERANOS are presented in this section. As core power changes, some elementary feedback coefficients remain almost constant or their change is minor. For instance, the sodium, diagrid and structure thermal expansion presents very small fluctuations at different power levels. In Table 3, the feedback coefficients at nominal core power are shown with APOLLO3 and ERANOS. Coefficients obtained with ERANOS are consistent with those of APOLLO3, however, larger values are seen in all cases with the ERANOS code. This could occur since the perturbed ERANOS media is not exactly the same compared to the APOLLO3 ones.

Feedback coefficient	APOLLO3 (pcm/°C)	ERANOS (pcm/°C)
Sodium thermal expansion	0.34	0.45
Diagrid thermal expansion	-0.89	-0.86
Cladding axial thermal expansion	0.037	0.070
Cladding radial thermal expansion	0.069	0.092
Hexagonal can axial thermal expansion	0.013	0.029
Hexagonal can radial thermal expansion	0.012	0.016

Table 3 : Feedback coefficients with APOLLO3 and ERANOS

The axial fuel thermal expansion was supposed to vary as core power changed, besides since no major burnup conditions were reached during these tests the fuel conditions is considered as free and not linked to the cladding. For its evaluation the control rods were supposed to be in parking position so that it does not influence the axial fuel thermal expansion effect. The axial fuel thermal expansion coefficients with APOLLO3 and ERANOS are consistent and share tendency, between each other, however, the obtained results with APOLLO3 are more negative than those obtained with ERANOS, as in past elementary feedback coefficients. It should be stated that different routes have been produced for the effective cross sections; in APOLLO3 the heterogeneous axial leakage model of the MOC has been used and different references suggest this is a significant improvement [11]. This might be the main source of the difference between codes for the axial fuel thermal expansion.

	Core power			
as % of Pn	100%	80%	50%	33%
APOLLO3	-0.156	-0.148	-0.134	-0.129
ERANOS	-0.146	-0.145	-0.123	-0.121

Table 4 : Axial fuel thermal expansion (pcm/°C) as function of core power.

Finally, the Doppler coefficients were evaluated with both ERANOS and APOLLO3 codes at different core power states, results are shown in Figure 3. A good consistency is found between both codes at different power conditions and good agreement is seen between codes for this elementary feedback coefficient.

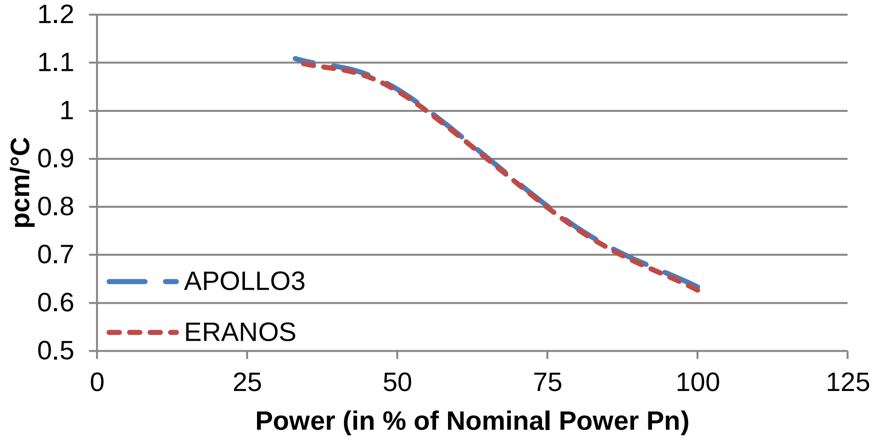

Figure 3 : Doppler coefficient (pcm/°C) at different core power levels.

Once the elementary feedback coefficients were calculated, the global feedback coefficients are evaluated and compared to the experimental results provided in [2]. Figure 4 presents the *k* coefficients as function of core power. The obtained results with APOLLO3 and ERANOS share the same tendency that measurements, however, both calculations underestimate the experimental *k* coefficients with ERANOS evaluations greatly underestimating compared to the APOLLO3 results.

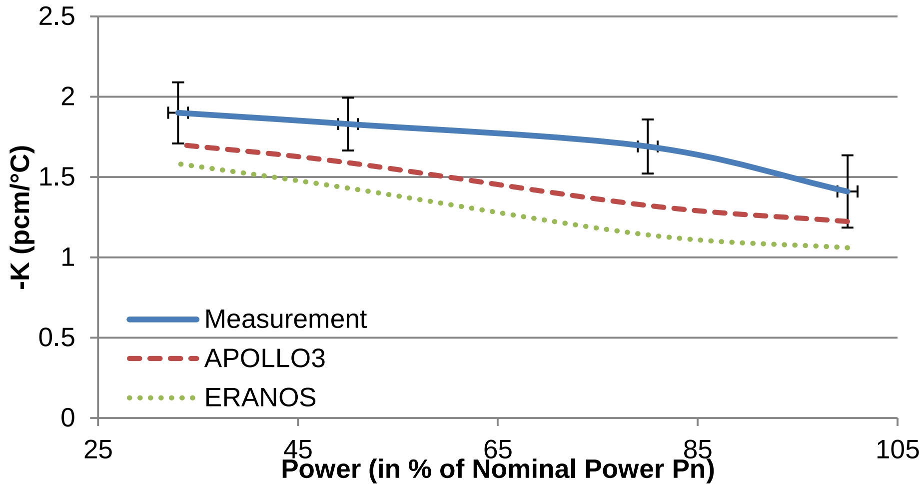

Figure 4 : K coefficient at different core powers.

The *g* coefficient, measured and evaluated, is shown in Figure 5. A discrepancy is seen between measurements and calculations, a different trend of g coefficient is obtained with evaluations. The measurements show an increase of *g* as power rises, however at nominal power it drastically decreases. Contrariwise the evaluations with ERANOS and APOLLO3 show a decrease as the power rises.

The main feedback coefficient involved in the g coefficient is the control rod driveline position and in this case as core power raised the control rods were extracted which suggests that the control rod driveline effect should decrease as power increases, however in this case we observe the opposite in the measurement since at 80% a rise of *g* is observed. A hypothesis of why this discrepancy is seen might be that the steady state was not fully reached at the time of the measurement, the temperature fluctuation was still unstable, and as a consequence the differential thermal expansion of control rods was still ongoing.

Although this discrepancy was seen for the *g* coefficient, the same inconsistency was seen in reference [2], in which the presented evaluations share the same tendency with the ERANOS and APOLLO3 results by decreasing the *g* coefficient as core power rises.

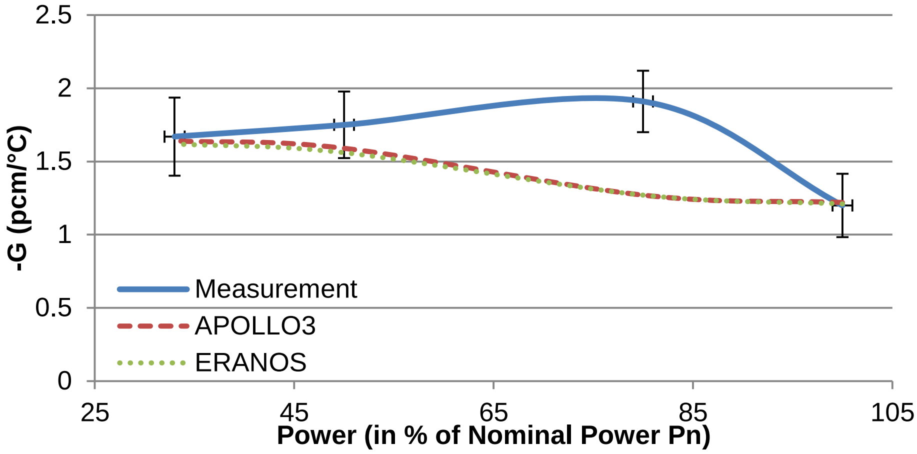

Figure 5 : G coefficient at different core powers.

Finally, the measurement and evaluations of the *h* coefficient are shown in Figure 6. As for the *k* coefficient, the same trend was seen between measurements and evaluations; however, a better agreement is observed between evaluations and measurements for this coefficient. One should be aware that the *h* coefficient is strongly related to the Doppler and axial fuel thermal expansion, for this reason a very good consistency is seen between codes. However, an increasing inconsistency is seen as power rises with both codes, overestimating slightly the *h* coefficient at 80% and 100%.

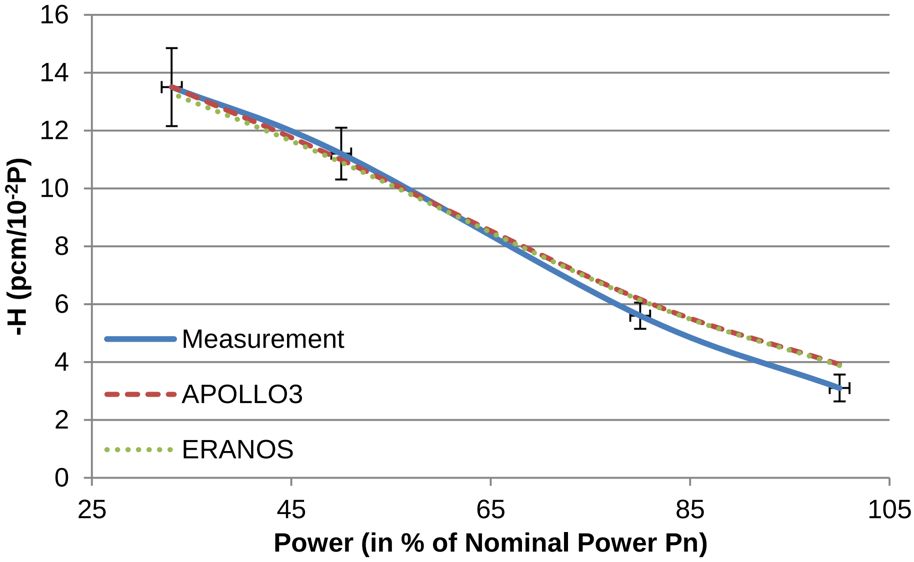

Figure 6 : H coefficient at different core powers.

Despite the fact that the *kgh* coefficients were successfully evaluated, the use of a thermal-hydraulic/neutronic code could largely improve these evaluations. The reproduction of some physical effects such as temperature stratification of the hot plenum of the core could be performed, contributing to a better interpretation of these tests. The measurements concerning this temperature stratification should be recovered or approximated from integral tests performed. This has been done before, as an example, in Phénix the natural convection in the reactor vessel was considerably improved by the use of CFD models and neutronic/mechanical behavior of the core [12] [13]. Additionally, the research of the thermal-hydraulic behavior of ASTRID [14] has been also validated with TrioCFD, CATHARE3, in which promising results were observed. The use of these calculation schemes for the Superphénix could contribute to the validation of these codes and to understand the strange behavior of the measurements in these Superphénix tests.

Conclusion

One of the main objectives of the SFRs of the GEN IV is to improve the safety to avoid severe accidents of the core. To characterize the safety of each SFR design, the global feedback coefficients *kgh* are used to determine the static states at which the reactor may be operated. For instance, in the Superphénix plant these tests were performed at the commissioning phase, during the start-up program of the reactor. These experiments are an invaluable source of data since they provide a set of tests that can be assessed and analytic hypothesis can be confirmed or refuted, which is a major benefit for the SFR research.

Still, the SFRs present a somehow particular complex performance with interrelated effects, reason for which the global effects are often decomposed in elementary feedback coefficients to characterize the contribution of each parameter of the core and its impact in the global design. In this paper, the elementary feedback coefficients were produced with ERANOS and APOLLO3 neutronic codes and the production of the *kgh* coefficient was done from these elementary coefficients.

The production of the elementary feedback coefficients was successfully done with both codes. Discrepancies were observed between codes with higher coefficient values for the ERANOS code but very good agreement between each other for the Doppler and the axial fuel thermal expansion.

The evaluation of the *kgh* coefficients was performed using elementary feedback coefficients produced by ERANOS and APOLLO3. The *kgh* were evaluated and consistent trends were observed between codes and against measurements for *k* and *h* coefficients; however a 2 σ difference between measurements and evaluations is still present at least for the *k* coefficient. Besides, a large inconsistency is seen for *g* coefficient for the measurement at 80% of the nominal Power and it needs to be clarified through a more detailed analysis. The use of a simplified thermal hydraulic model might be neglecting some effects; for instance, the temperature stratification at the hot plenum is totally ignored at this point. Also a particular assessment of the *g* measurement has to be done to determine its validity or to elaborate another hypothesis to explain its behavior as core power rises.

To deepen the analysis of the global feedback coefficients, the use of a thermo-hydraulic code such as CATHARE3 has to be used to consider thermo-hydraulic and neutronic interaction during the core operation for the determination of the *kgh* coefficients. Particularly, the inlet temperature change should be carefully analyzed to assess the measurement consistency.

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