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TRACE ANALYSIS OF PHENIX CORE RESPONSE TO AN INCREASE OF THE CORE INLET SODIUM TEMPERATURE

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Abstract

This work presents the analysis, using the TRACE code, of the Phénix core response to an inlet sodium temperature increase. The considered experiment was performed in the frame of the Phénix End-Of-Life (EOL) test program of the CEA, prior to the final shutdown of the reactor. It corresponds to a transient following a 40°C increase of the core inlet temperature, which leads to a power decrease of 60%. This work focuses on the first phase of the transient, prior to the reactor scram and pump trip.

First, the thermal-hydraulic TRACE model of the core developed for the present analysis is described. The kinetic parameters and feedback coefficients for the point kinetic model were first derived from a 3D static neutronic ERANOS model developed in a former study. The calculated kinetic parameters were then optimized, before use, on the basis of the experimental reactivity in order to minimize the error on the power calculation. The different reactivity feedbacks taken into account include various expansion mechanisms that have been specifically implemented in TRACE for analysis of fast-neutron spectrum systems. The point kinetic model has been used to study the sensitivity of the core response to the different feedback effects.

The comparison of the calculated results with the experimental data reveals the need to accurately calculate the reactivity feedback coefficients. This is because the reactor response is very sensitive to small reactivity changes. This study has enabled us to study the sensitivity of the power change to the different reactivity feedbacks and define the most important parameters. As such, it furthers the validation of the FAST code system, which is being used to gain a more in-depth understanding of SFR core behavior during accidental transients.

Introduction

Fast breeder reactors are being developed for a sustainable energy supply and closure of the fuel cycle. Among the different candidates considered by the Generation IV International Forum (GIF), the Sodium-cooled Fast Reactor (SFR) currently appears to be the most mature technology and the best candidate for mid-term implementation of fast-neutron spectrum technology.

The FAST project at PSI focuses on fuel cycle and safety analysis for comparison of the different proposed, advanced fast reactor concepts. The calculational tool used for this purpose is the FAST code system [1], which is assembled from different individual codes. The present study concerns an analysis of Phénix core behavior following an increase in the core inlet temperature, using the thermal-hydraulic code TRACE. The considered experiment was performed within the frame of Phénix End-Of-Life (EOL) tests and is part of the so-called Natural Convection (NC) test. In particular, it corresponds to the first 8 minutes of the transient, before the reactor scram. A detailed description of the considered experiment is given is Section 1. The reader should refer to [2] for a full description of the NC test.

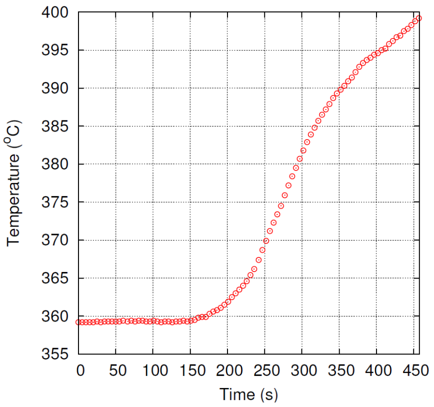
For the current analysis, a point kinetic model of the Phénix core has been developed and is presented in Sections 2 to 4. The reactivity feedbacks specific to fast-neutron systems – mainly relative to the various expansion mechanisms – have been implemented in the TRACE code to allow a realistic modeling of the core behavior. The sensitivity of the core power and reactivity to the different feedbacks has been studied and is presented in Section 5. Finally, the conclusions from this analysis are summarized in the last section.

1. Description of the considered transient

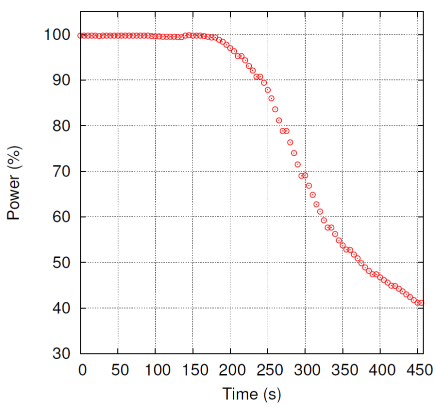
Phénix is a French pool-type, sodium-cooled, industrial prototype fast breeder reactor, originally rated at 580 MWth (260 MWe). After more than 35 years of successful operation, the Phénix reactor was permanently shut-down in 2009. Prior to the final shut-down, a series of EOL tests were performed in order to better understand the core behavior under special operating conditions and acquire test data for the qualification of calculational tools.

The considered transient is part of the Natural Convection test, which originally aimed at studying the establishment of natural convection in the core and qualification of thermal-hydraulic codes. The acquired data have used for setting up an international benchmark exercise and studied at PSI in the frame of a Coordinated Research Project (CRP), initiated by the IAEA Technical Working Group on Fast Reactors (TWG-FR). The complete test is presented in [2] and will not be detailed here. This work focuses on the first 8 minutes of the transient, prior to the reactor scram and pump trip.

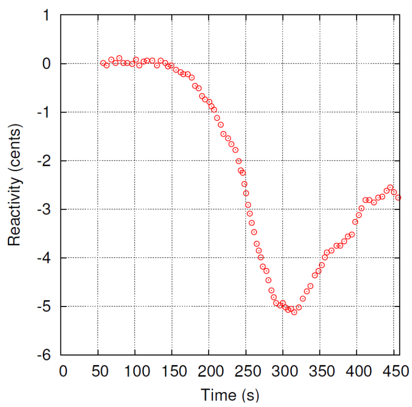
From a reduced power state (120 MWth), the test was initiated by the dryout of the two operating steam generators in the tertiary circuit. The deficiency in cooling led to an increase of 40°C of the core inlet temperature. The different reactivity feedbacks induced a decrease of the power down to 40%. The reactor was manually shutdown after 458 s, when the difference between the primary and secondary temperatures in the intermediate heat exchangers (IHXs) decreased to 15°C. The three primary pumps were simultaneously tripped (t = 466 s), and natural convection began to be established in the core. The main experimental data of interest here, viz. the inlet core temperature, power and reactivity evolution, are presented in Fig. 1. The establishment of natural convection and associated analysis will be addressed in a separate publication.



(a) Core inlet temperature



(b) Reactor power



(c) Reactivity

Figure 1. Relevant experimental data characterizing the considered transient.

2. Description of the Phénix core and corresponding model

2.1 Phénix core

The Phénix core consists of 110 hexagonal mixed-oxide [(U-Pu)O_2] fuel assemblies, ranging from 23% to 28% plutonium content. The inner zone is composed of 54 sub-assemblies (SAs), which contain less Pu than the 56 SAs in the outer zone in order to flatten the power profile within the core. The active core is surrounded by a fertile breeding zone which consists of 86 fertile SAs, initially composed of UO_2. The blanket is surrounded by 212 stainless steel SAs, especially designed for neutron shielding. Figure 2b presents a map of the different core regions. Each core sub-assembly is enclosed in a hexagonal steel tube (wrapper), fixed at the bottom to the diagrid. There is one emergency shutdown control assembly in the center of the core and six control assemblies arranged in the inner core. The absorber rods are, held from the top by control-rod (CR) drives fixed to the slab of the reactor block (see Fig. 2a).

Table 1 summarizes the main characteristics of the Phénix sub-assemblies.

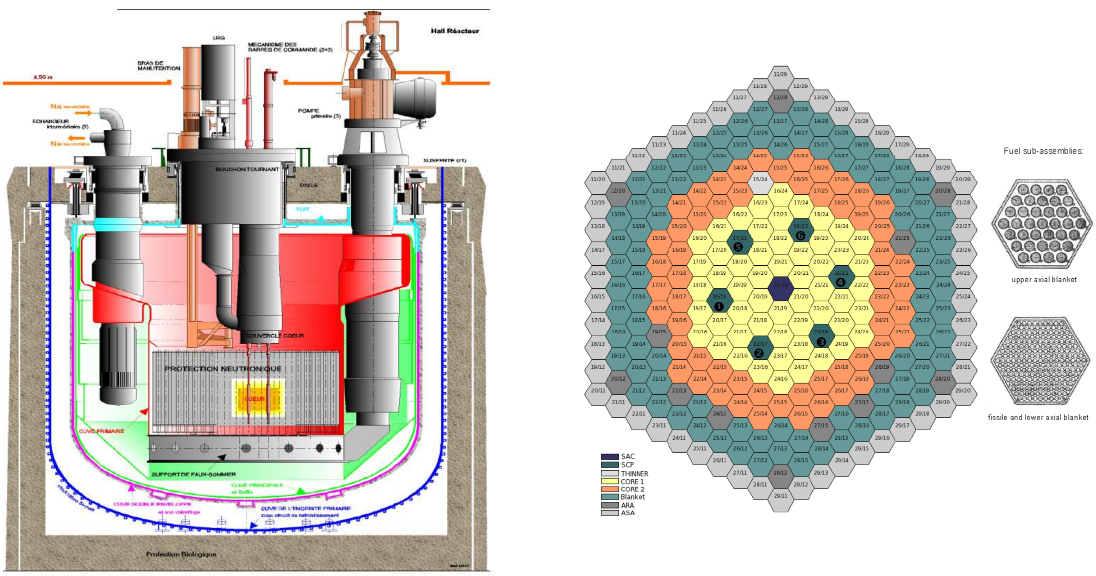
2.2 Phénix primary circuit

The core SAs are fixed to the diagrid, which ensures a distribution of the sodium coolant to the different core regions in relation to the thermal power of the different SAs. The diagrid is connected to the main vessel via a conical shell.

Figure 2a presents the Phénix reactor block. The main vessel is supported by the slab that forms the upper part of the reactor block. Inside the main vessel, the sodium coolant is separated into two pools. The hot sodium (560°C) is contained in the hot pool and flows to the cold pool through the intermediate heat exchangers (IHXs). The cold sodium (400°C) is pumped into the core by the primary pumps.

Table 1. Characteristics of Phénix core sub-assemblies [4]

		Fuel	Axial upper blanket	Axial lower blanket	Radial blanket
Number of pins	-	217	37	217	61
Lattice pitch	cm	12.72	12.72	12.72	12.72
Pellet external diameter	cm	0.542	1.295	0.55	1.215
Clad external diameter	cm	0.655	1.425	0.655	1.34
Spacer wire diameter	cm	0.115	0.386	0.115	0.108
Pins pitch	cm	0.777	1.820	0.777	1.457
Height	cm	85.0	26.2	33.7	164.9



(a) Phénix reactor block and primary circuit (b) Radial cut of the core

Figure 2. Phénix reactor primary circuit and core

2.3 TRACE model of the Phénix core

In the scope of the present analysis, the modeling of Phénix has been limited to the core region. A schematic of the developed TRACE model is shown in Fig. 3a. The model consists of four parallel channels representing the different core regions, i.e. inner core, outer core, fertile core and reflector region plus all bypasses. The correct flow distribution was achieved thanks to singular friction losses in the diagrid. A heat structure (HTSTR) was linked to the pipe representing the diagrid to account for the radial expansion. Â single-node structure with high exchange area was used in order to ensure that its temperature follows that of the inlet core during transient.

The heated regions were coupled to heat structures. According to CEA recommendations, 90% of the fuel pins were modeled as linked fuel (with closed gap), the remaining 10% being considered as free fuel with open gap. This corresponds to different gas-gap conductance, viz.

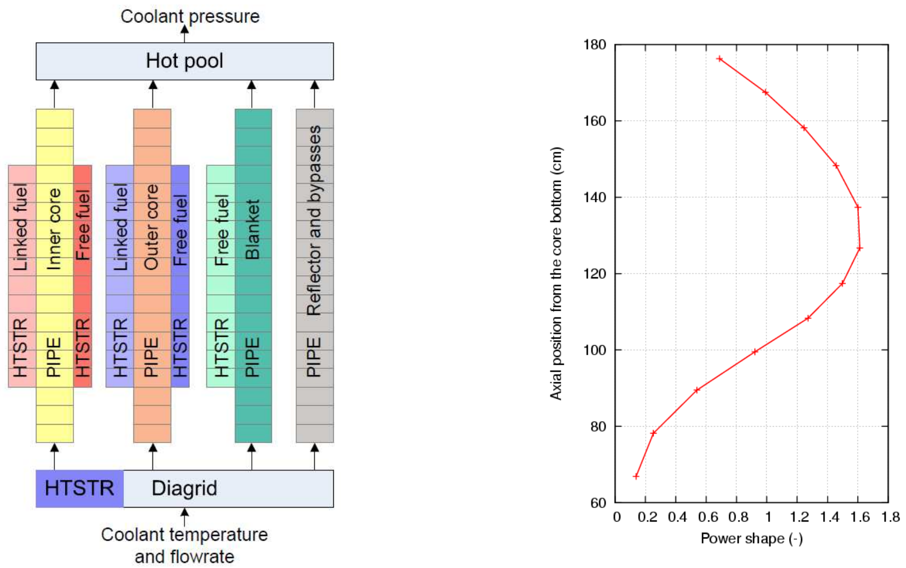
5000 and 1500 W/m^{2}K for fuel with closed and open gap, respectively. A gap conductance of 2500 W/m^{2}K was used in the blanket pins. The axial power distribution was computed from a 3D static neutronic ERANOS model of the core, developed in the frame of a previous study [5], and is shown in Fig. 3b. The radial power distribution is indicated in Table 2.

Table 2. Comparison of TRACE calculated results (Calc.) with the experimental data (Exp.) at 120 MWth

	Power	Mass flow (kg/s)		Temperature (°C)	
	(MW)	Exp.	Calc.	Exp.	Calc.
Inner core	61.7	1055	554	447	446
Outer core	48.8		503	-	438
Blanket	8.2	149	148	-	399
Reflector	1.1	80	79	-	359

Appropriate boundary conditions were defined at the core inlet and outlet in order to reproduce the correct core state. The mass flow rate and temperature evolution were specified at the core inlet with the FILL component. The nominal core outlet pressure was fixed via the BREAK component (1.6 bar at the bottom of the hot pool).

Local friction losses have been used to reproduce the correct core gagging and obtain a realistic flow rate distribution in the different regions of the core. The core pressure drop was successfully reproduced at nominal state (which corresponds to a power of 360 MWth with an inlet flow rate of 1200 kg/s). Table 2 shows that the flow rates and temperatures calculated by TRACE at reduced power (120 MWth) satisfactorily reproduce the experimental data. This state corresponds to the reactor operating conditions prior to the NC test.



(a) Schematic of TRACE model (b) Axial core power distribution

Figure 3. TRACE model of Phénix core

3. Optimization of the kinetics parameters

The kinetic parameters, viz. the delayed neutron fractions β_i and corresponding decay constants λ_i for each group of precursor i, were first calculated from an ERANOS model of the Phénix core developed in the frame of a previous study [5]. The computed data have then been optimized such that, when specifying the experimental reactivity, the calculated power accurately reproduces the experimental values. Figure 4 shows the optimized kinetic parameters together with CEA specifications and the values calculated from the ERANOS model. It can be seen that the adjusted set of parameters corresponds to the CEA specification within 10% (-10% for β_i and +10% for λ_i) and give an effective fraction of delayed neutron β_{eff} of 292 pcm compared to 325 pcm specified by CEA. In order to minimize the uncertainties due to the kinetic parameters, the optimized set has been used for the sensitivity study on the different feedback coefficient presently included. A neutron lifetime of 0.38 µs was used throughout the analysis.

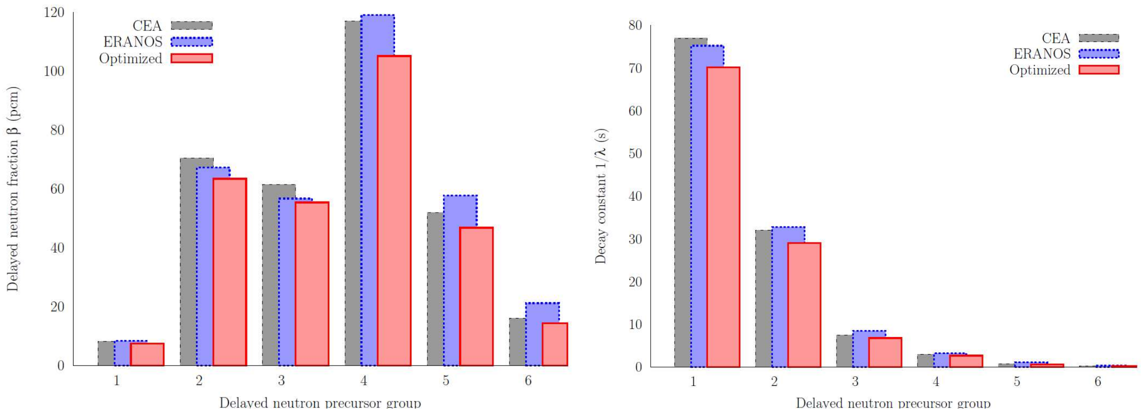


Figure 4. Comparison of different sets of kinetic parameters for the Phénix core

4. Description of the point-reactor kinetic model

Since the TRACE code has been originally developed for the safety analysis of water-cooled reactors with a thermal neutron spectrum, further models have been added at PSI to allow the accurate simulation of fast-neutron spectrum systems. A decomposition of the reactivity as currently computed in TRACE for the present analysis is given by Eq. 1:

$$\begin{aligned}
\Delta\rho(t) &= \Delta\rho_{Doppler}(t) + \Delta\rho_{fuel}(t) + \Delta\rho_{diagrid}(t) + \Delta\rho_{vessel}(t) && (1)\\
\Delta\rho_{Doppler}(t) &= \sum_i K_{D_i} \ln\frac{\overline{T_{f_i}}(t)}{T_{f_i}(0)} \text{ with } i \in \{\text{inner core, outer core, blanket}\} && (2)\\
\Delta\rho_{diagrid}(t) &= K_{rad}\left[T_{in}(t) - T_{in}(0)\right] && (3)\\
\Delta\rho_{fuel}(t) &= K_{axi}\left[\sum_i n_i\left(\overline{T_{c_i}}(t) - T_{c_i}(0)\right) + \sum_j n_j\left(\overline{T_{f_j}}(t) - T_{f_j}(0)\right)\right] \times \frac{1}{n_{rod}} && (4)\\
\text{with} &\quad i \in \text{closed-gap fuel } \{\text{inner core, outer core}\}\\
\text{and} &\quad j \in \text{open-gap fuel } \{\text{inner core, outer core, blanket}\}\\
\Delta\rho_{vessel}(t) &= K_v\left[T_{in}(t-\delta) - T_{in}(0)\right] && (5)
\end{aligned}$$

where ρ is the reactivity and K are the feedback coefficients for the Doppler effect (D), and for diagrid (rad), fuel (axi) and vessel (v) expansion. $\overline{T}$ is the volume-averaged temperature over the fuel length and the sub-scripts f and c refer to the fuel and clad, respectively. n represents the number of rods, and corresponds to 10546 (10937) pins with closed gap and 1172 (1215) with open gap in the inner (outer) core. δ is the time delay characterizing the vessel temperature as function of the core inlet temperature.

It can be seen from Eq. 2 that the Doppler effect has a logarithmic dependency on the fuel temperature, corresponding to the fast-neutron spectrum. For more accurate modeling, a zone-wise Doppler constant has been computed (see values in Table 3).

The sodium density effect has been neglected, since its contribution in the Phénix core is very small (-0.018 pcm/°C). An analysis including this effect showed its very small contribution. Moreover, it is compensated by a positive effect of comparable magnitude, viz. the axial expansion of the hexagonal wrappers (0.013 pcm/°C).

The other effects of interest result from different expansion mechanisms. The most important feedback in the considered transient is the radial expansion of the diagrid, which causes a negative reactivity feedback when there is an increase of the sodium inlet temperature. The expansion of the diagrid, which supports the core, moves the fuel SAs away from one another. The introduction of a relatively higher coolant-to-fuel ratio increases the neutron moderation, thus reducing the reactivity. This effect is mainly due to the increased capture rate of ^{238}U in the resonance region [6].

Eq. 4 details the computation of the reactivity feedback due to the fuel axial expansion. Distinction has been made between the linked and free fuel with closed and open gas-gap, respectively. In the first case, since the fuel is linked to the cladding, the axial expansion is assumed to be driven by the average cladding temperature whereas, for fuel with open gap, it is assumed to be driven by the average fuel temperature. Figure 5 shows the contributions of the different core regions and fuel types during the considered transient. The average cladding temperatures in the different core regions follow the evolution of the increasing inlet temperature, producing a negative reactivity feedback. In the free fuel regions, the reduction of power leads to a decrease of the fuel temperatures, associated with a positive reactivity feedback. Consequently, it can be noticed that, in the fuel regions, the different fuel types (linked or free fuel) bring opposing contributions to the reactivity, which significantly reduces the effect of the fuel expansion in both the inner and outer core. From this point of view, the blanket region contributes the highest reactivity feedback when considering the fuel expansions. Expansion of the cladding and radial expansion of the hexagonal wrappers are negligible (less than 0.01 pcm/°C) and have not been modeled here.

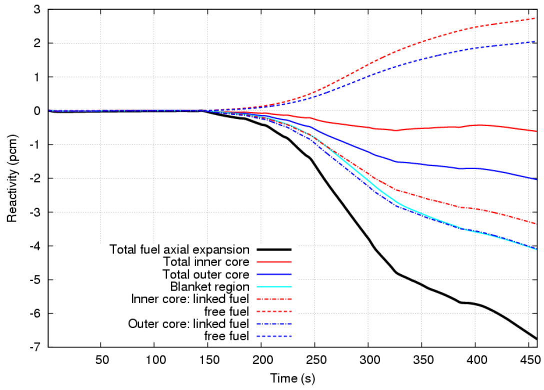


Figure 5. Decomposition of the fuel axial expansion reactivity feedback showing the contributions of the different core regions and fuel types

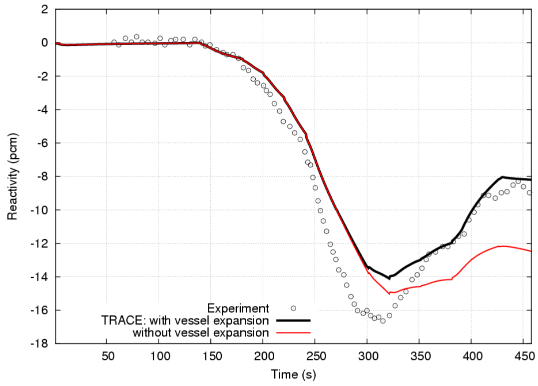


Figure 6. Comparison of the calculated and experimental reactivity with and without the vessel expansion feedback

The last effect taken into account in this study is the vessel expansion. Following the evolution of the core inlet temperature, the vessel is slowly heated up by the increasing temperature in the cold pool. The vessel temperature is assumed to follow the core inlet temperature with a time delay δ [7]. Since the vessel is fixed to the slab at the top of the reactor, and since it indirectly holds the core, an expansion of the vessel will slightly move the core downward. This results in a relative extraction of the control assemblies, which are held by drive rods fixed to the same slab at the top of the reactor. The relative expansion of the core, vessel, and control rods should thus be taken into account accurately in pool-type reactors. In the considered transient, the core outlet temperature remains almost constant (less than 10°C variation). Therefore, in a first approximation, the expansion of the CR drivers has been neglected. Taking into account the vessel expansion reactivity feedback has enabled us to qualitatively reproduce the evolution of the reactivity as measured during the test. To illustrate the importance of this phenomenon, Fig. 6 shows a comparison between the measured and the calculated reactivity, with and without vessel expansion feedbacks. The time delay δ and feedback coefficient K_v have been determined on the basis of the experimental reactivity (δ = 130 s and K_v = 0.5 pcm/°C).

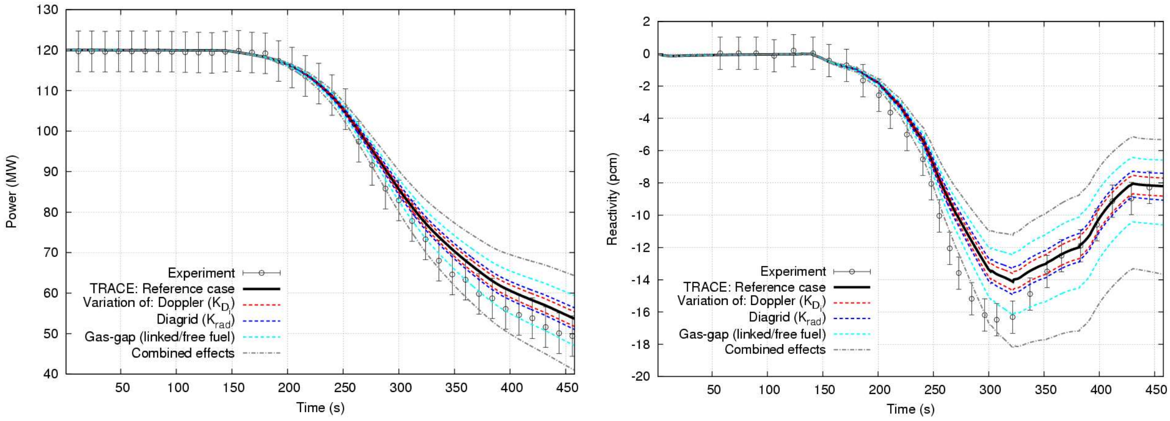
The reference values of the feedback coefficients have been calculated using the ERANOS 3D static neutronic model, simulating different core states at 293 and 1000 K. The computed results are presented in Table 3.

Table 3. Reactivity feedback coefficients of the Phenix core, calculated from ERANOS and used in the TRACE point kinetics model

Doppler constant K_D: inner core	-404.7 pcm	Diagrid radial expansion	-1.11 pcm/°C
outer core	-173.7 pcm	Fuel axial expansion	-0.72 pcm/°C
blanket region	-101.0 pcm		

5. Sensitivity analysis with respect to the different reactivity feedbacks

In order to better understand the core response to the increasing core inlet temperature, a sensitivity analysis has been performed with respect to the different reactivity feedbacks. This was carried out assuming 10% variation on each reactivity coefficient, viz. on K_D, K_{rad}, K_{axi}, K_v as well as on the ratio of linked to free fuel. The reference case was calculated with the values given in the former section. Then, calculations were performed while perturbing each coefficient separately, assuming a variation of +10 or -10%. It should be emphasised that the effects were studied separately. Though this analysis did not use a sampling technique for the propagation of uncertainties, it allowed us to identify the important feedback effects occurring in this transient.



(a) Power (b) Reactivity

Figure 7. Sensitivity of the calculated (a) power and (b) reactivity evolution to variation of different feedback coefficients and comparison with experiment

Figure 7 presents the power and reactivity evolution obtained for a selection of effects. The 'Doppler' case corresponds to a change of 10% of the Doppler constants K_{D_i}, the 'Diagrid' case to that of the radial expansion coefficient K_{rad}. The sensitivity on the fuel burn-up has also been studied assuming a different proportion of linked and free fuel. This is referred to as the 'Gas-gap' case (open or closed), which has been calculated using a different ratio of linked to free fuel: 100/0% and 80/20%, instead of the specified 90/10% ratio. These changes mostly impact the fuel temperature and the corresponding axial expansions, as well as the Doppler effect. The study allowed us to evaluate the reactivity changes for different fuel modeling assumptions (the different gas-gap conductance values, corresponding to an open or closed gap, result in different fuel temperatures). The last case presented in Fig. 7 corresponds to 'Combined effects' and serves to provide an estimate of the extreme boundaries of power and reactivity variation when combining the variations of the different coefficients. The lowest boundary has been calculated using K_D - 10%, K_{rad} + 10%, K_{axi} + 10%, K_v - 10% and 100% of linked fuel. The opposite variations have been used for obtaining the upper boundary, with 80% of linked fuel.

Figure 8 shows the decomposition of the different reactivity feedbacks. The solid lines correspond to the reference calculation. The dashed lines illustrate the variation range of each effect when perturbing the corresponding coefficient, without showing the impact on the other feedbacks. The combined variation of effects, resulting from the variation of all effects together as specified above, is shown on the total reactivity to illustrate the range of variation due to changes of ±10% of the different feedback coefficients. From this figure, it can be seen that the diagrid expansion is the first effect produced in this transient. It is also the most important feedback. It can be noticed that the time dependency of its contribution to the reactivity almost exactly follows that of the core inlet temperature (cf. Fig. 1 (a)). This highlights the importance of an accurate prediction of the inlet temperature during the transient. The second most important feedback is the Doppler effect, and this is mainly an inner core contribution**.**. The fuel axial expansion provides a negative feedback, thanks to the linked fuel – driven by the increasing clad temperature – and the blanket expansion, as detailed in Fig. 5. The decomposition of the fuel axial expansion into the different core regions and its sensitivity to the burn-up (through the variation of the linked/free fuel ratio) has demonstrated the importance of accurately modelling the gas-gap conductance and fuel expansion mechanisms, since the burned fuel (whose expansions are assumed to be driven by the clad temperature) brings an opposite contribution to that of the free fuel (assumed to be driven by the fuel temperature). Finally, the vessel expansion brings a delayed positive contribution opposite to that of the diagrid expansion, due to the relative movement of the core, vessel and control assemblies as discussed in Section 4.

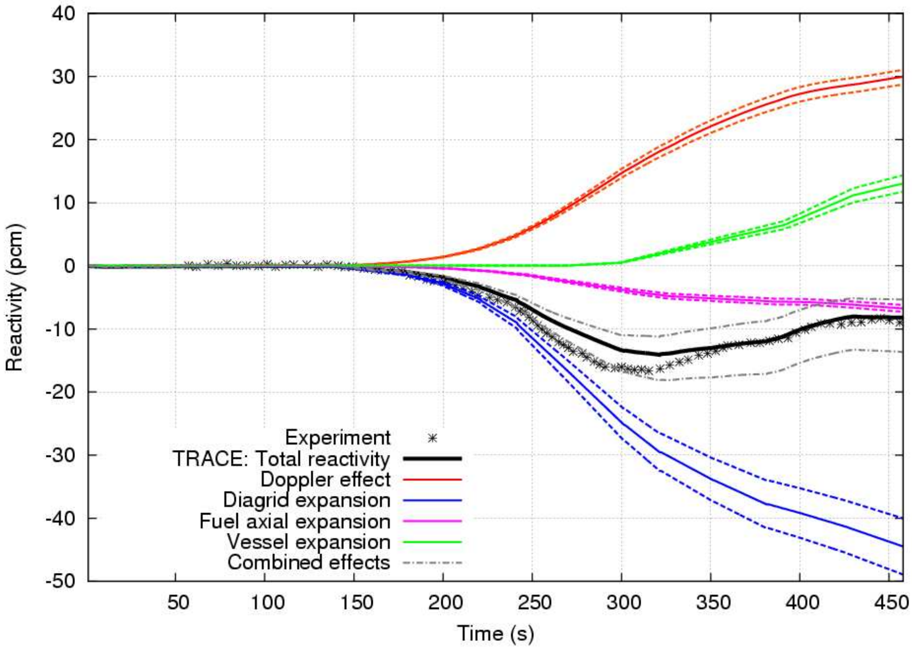


Figure 8. Decomposition of the reactivity and illustration of the sensitivity of the different feedback mechanisms to a variation of ±10% on the feedback coefficients

6. Conclusions

The increase of 40°C of the sodium core inlet temperature at the beginning of the Natural Convection test leads to a reduction by 60% of the power in the Phénix reactor. The presented point kinetic model for the prototype fast reactor, developed in TRACE within the FAST project, has enabled us to better understand the core behavior during the considered transient. It has been shown that the diagrid radial expansion provides the first and most important feedback effect in this transient. The increase of the diagrid temperature moves the core SAs away from one another, thus producing a negative reactivity feedback. The resultant power decrease causes a decrease of the fuel temperature, which yields a positive Doppler reactivity feedback. The opposite evolutions of the fuel (decreasing) and clad (increasing) temperatures has shown the need for an accurate simulation of the fuel axial expansion, since burned fuel could bring an opposite contribution to that of fresh fuel – due to closed or open gas-gap, respectively.

The sensitivity study conducted has enabled us to define the most important effects in the considered transient, viz. the diagrid and fuel axial expansions and the Doppler effect. These have highlighted the need to accurately predict the core inlet temperature, gap conductance and fuel expansion mechanisms.

Finally, the experimental power and reactivity evolution could be satisfactorily reproduced and the core behavior better understood. As such, the present analysis represents a step towards the in-depth understanding of sodium-cooled fast reactor (SFR) behavior under transient conditions.

7. Acknowledgments

The authors are grateful to the CEA for sharing the experimental data of the NC test.

8. References

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