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Approach for the adaptations of a nuclear reactor model towards more flexibility in a context of high insertion of renewable energies

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Abstract. The massive penetration of renewable energy sources (RES) that are variable and not "dispatchable", may weaken the power system supply-demand balance. Nuclear power plants (NPP) contribute in part to this daily and seasonal balance thanks to the "load-following" mode in France for example, but there are still limits to their use. These limits prevent a nuclear power modulation as efficient and quickly as the conventional thermal power plants. The need in terms of power ramps for nuclear in a constrained power system has been quantified in previous studies. Nuclear may compensate for the removal of thermal power plants, in order to fulfill energetic strategies of CO_2 reduction. The possibility that nuclear reactors can achieve power ramps of significant values (>5%Pn/min) is put forward and could make possible to replace the services currently provided by thermal power plants. The objective of the study is then to use these power system requirements as the main input parameter for the modelling of a current simplified nuclear reactor capable of responding to frequency control within a specific hypothesis framework. In this paper, a French 1300 MW pressurized water reactor is modelled. Parametric studies are carried out in order to reveal technical and technological constraints when increasing electric power ramp. The study explores ways of design, which may influence reactor flexibility, such as the neutron parameter, Doppler coefficient, or the thermohydraulic parameter, delay in the primary loop.

1 Introduction

In an electric power system, a balance between electricity production and consumption must be ensured at all times. Therefore, the large-scale penetration of variable Renewable Energy Sources (RES) such as wind and photovoltaic generation sources, which are not "dispatchable" in the electric mix, represents a challenge for the power system in the near future. Moreover, energetic strategies plan to reduce CO_2 emissions thanks to thermal power plants reduction in favour of the increasing penetration of renewable energy sources. The compensation of the production/consumption fluctuation is currently carried out by using these dispatchable generation units such as conventional fossil, hydraulic or nuclear plants. But low-carbon flexible levers are still necessary, in order to meet the CO_2 reduction targets. The development of flexible solutions such as power modulation of generation sources to accommodate power demand, storage or even load shedding are low carbon ways of adapting electricity production to consumption.

In order to consider a flexibility solution for nuclear power plant (NPP), the electric power ramp determined by the grid constraints is taken as input and then used in the nuclear design part. This parameter is required to guarantee power system stability [1]. This being said, this paper suggests using this criterion as an input data for the design. Above all, this paper aims at introducing an innovative approach to integrate flexibility as a dimensioning criterion in the design. The specific French case is studied in the paper, which means that only pressurized water reactor (PWR) are taken into account in the following.

The paper first presents the current nuclear resources to ensure grid stability. This section also recalls details related to frequency control and the performance of nuclear power in terms of power modulation. Following this state of the

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art, the description of the simplified reactor model is given with the associated validation as a baseline tool to introduce flexibility in the nuclear design. The current model is then constrained by larger power variations to highlight the potential issues that nuclear power may have under these operating conditions. Finally, a sensitivity study is performed to deduce the main parameters impacting the flexibility of the model, however these parameters may impact at the same time the reactor safety.

2 State of the art on current nuclear capacities

2.1 Nuclear capacities

The French nuclear fleet participates in primary, secondary and tertiary frequency control, which are part of the ancillary services provided by the generation units to the grid. In fact, according to the TSO (transmission system operator) requirements, any generating unit with a capacity of more than 120 MW_e must be able to participate in frequency control (except in exceptional cases such as a fault or maintenance return). A nuclear generating unit providing load-following or frequency control service must be able to operate stably at any power level between its rated power and a defined minimum power level. It must also be capable of increasing or decreasing power at a defined rate between any power levels within that range [2]. A NPP participating in the primary control reserves ±2.5% of its nominal electric power to meet the needs of the network, and must be able to respond in 30 s. For load-following, time scale is much longer until 30 minutes and tertiary control reserve is larger than primary reserve.

Most current plant technologies are designed to perform power operations in the range of 50–100% of rated power, and can perform power ramps as fast as 5%Pn/min [3]. For French nuclear power plants, the general operating rules govern power ramps; this electric power ramp value of 5%Pn/min is taken as the maximum authorized value of all French nuclear reactors for both grid following and load following. Some limits prevent nuclear power for power modulation. Material, thermo-mechanical, environmental and operating constraints, etc., are restrictive from a safety point of view because they impose limits on power variations in the core, on restart times or operating time at reduced power or on the progress in the cycle. However, in order to fulfill the grid requirements, NPP may use regulations for both cases; grid-following and load-following.

In a PWR reactor, there are two means of controlling the power of the core, namely the insertion (or removal) of control rods in the core, and the injection of soluble boron in the primary circuit. They are mobilized to follow a temperature program according to the power requested from the turbine, while regulating the reactivity and the spatial distribution of power in the core. These regulations are used in particular when the reactor operates in "priority turbine" mode, i.e. they are used to adjust the core power according to the power demanded by the turbine. Such mode is fully adapted to system services.

The control rods are a very effective way to adjust the thermal power produced by the core. These control rods are made of neutron absorber rods and are inserted into the core from above. There are two types of control rods, black and grey, which are divided into two groups according to their function within the core [4].

– The power compensation groups (GCP) regulate the core thermal power in an open loop. The displacement of these control rods is normally given by a calibration curve, allowing to anticipate the effects of a variation of electrical power on the core. However, this regulation is only activated if the electrical power deviation exceeds 38 MW_e on a 1300 MW_e PWR (i.e. 2.8%Pn) in the case of frequency primary control, which is extremely rare. In fact, this threshold makes it possible to avoid stressing the power rods during frequency control, and also limits the mechanical wear of the control mechanisms. It also allows not to excessively degrade the thermodynamic efficiency of the reactor.
– The temperature control groups (GRT) unit acts on the water moderator temperature and therefore on the inlet temperature and outlet temperature of the core in closed loop. The deviation of the moderator temperature must not exceed ±0.8 °C in relation to a set point temperature defined according to the electrical power. The temperature control, among other things, has been added to avoid thermal expansion effects that would prevent industrial and robust use of the reactors for grid and load monitoring.

The control rods are used for rapid power changes. Both temperature and power control will automatically activate if they exceed a threshold. Two control modes exist in French PWR; A and G. Reactors using both modes A and G can participate in frequency monitoring. However, only the engines with G control mode are able to participate in load following. GCP control is only available in G control mode as opposed to A mode which uses temperature control exclusively.

In parallel, the control of the turbine-generator set allows the power produced by the turbine to be adjusted instantaneously following a disturbance on the grid side. To do this, the regulation controls the opening or closing of the valves directly linked to the secondary flow and therefore to the power produced by the turbine. In other words, as soon as the electrical power evolves, the opening rate of the! turbine inlet valve varies accordingly, then the core accommodates these variations either in free dynamics or thanks to its regulation system described above, if the thresholds on the average temperature and then the electrical power deviation are exceeded. The quantity of steam sent to the turbine conditions its rotation speed and therefore the power it supplies to the network. This loop is permanently active in priority turbine mode.

An example of both chronologies following a load decrease (primary frequency control and load-following) for a PWR is addressed in the next sub-section in order to visualize nuclear participation and regulations uses for system services.

Table 1. Nomenclature of the study.

GRT regulation	GCP regulation	Turbine-generator control	Type of control in the real life, if possible	Condition	Referred as
0	0	X	Free dynamic		FreeDyn
X	0	X	Primary frequency control or grid following	If moderator temperature deviation exceed ±0.8 °C / ref	With GRT regul
X	X	X	Tertiary frequency control or load-following	If electrical power deviation exceeds 2.8%Pn	With GCP & GRT regul

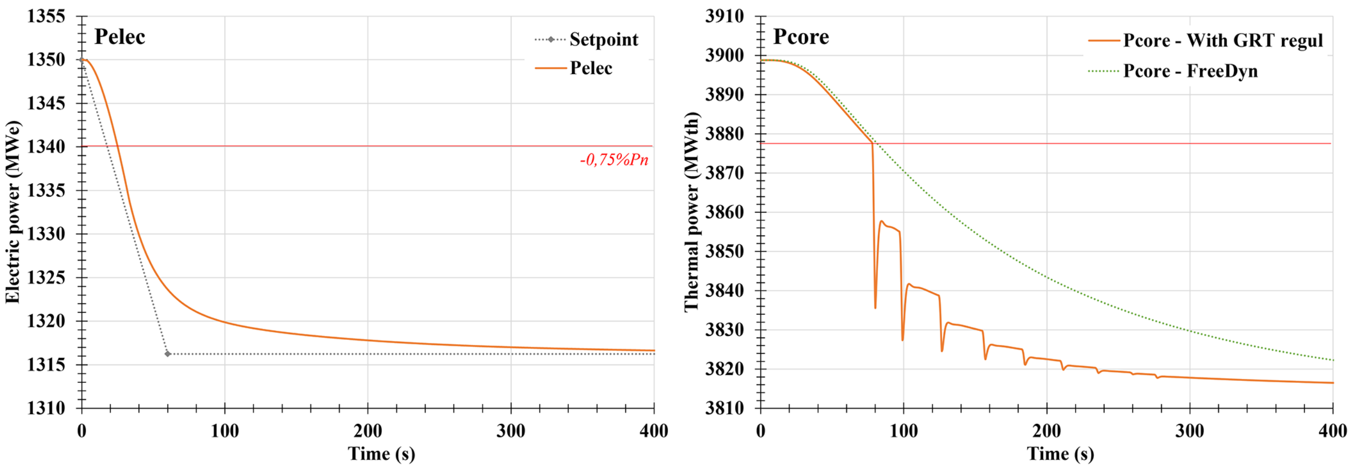


Fig. 1. Electrical and thermal powers during a primary frequency control transient.

2.2 Description of the participation to frequency control of NPP

This section presents transients related to frequency control in order to provide explanations of the regulations involved and the associated activation conditions. All figures in this section were obtained with the C-PWR-1300 academic simulator [5] (more details in Sect. 3.2.). The simulator models a PWR of 1300 MW. The terminology used in the figures is specified in Table 1. It gives the possible configurations of the regulations and their respective activations during controls.

2.2.1 Grid following

A reactor participating in the primary control reserves ±2.5% of its nominal electrical power to meet the grid requirement, and must be able to operate within 30 s. For instance, a typical transient of frequency control is shown in Figure 1. It consists in a small power drop of –5%Pn with a ramp of 5%Pn/min. In practice, the regulation on the average temperature GRT can be activated within the primary frequency control. The behavior of the reactor power is presented in Figure 1 in orange; the electrical power of the reactor on the left side and the corresponding core power response on the right side. We first observe a slight lag between the electrical power set point and the power actually produced; this is due to the inertia of the turbine generator set and the regulation chain controlling the opening of the turbine inlet valve. The response without rod regulation (i.e. free dynamics) is also added to the right figure (green dotted line); we observe that the response time of the core power is longer without regulation. Indeed, after 70 s, the two thermal power curves diverge; the green curve stabilizes more slowly. The regulation of the temperature therefore accelerates the behavior of the core, in addition to the fact that this regulation avoid strong thermal dilatation phenomenon in the core.

During the first 70 s, the two configurations overlap perfectly, i.e. no regulation is activated before 70 s. In addition, the effects of temperature control on the core can be seen with the core power discontinuities. The activation of the GRT control causes strong power gradients in the core and on the fuel, but with a low amplitude. Contrary to the response of the core in free dynamics which is slow and smooth.

As for the average temperatures in the core (inlet, average and outlet), shown in Figure 2, we notice first of all that the variation of the temperatures in the core between

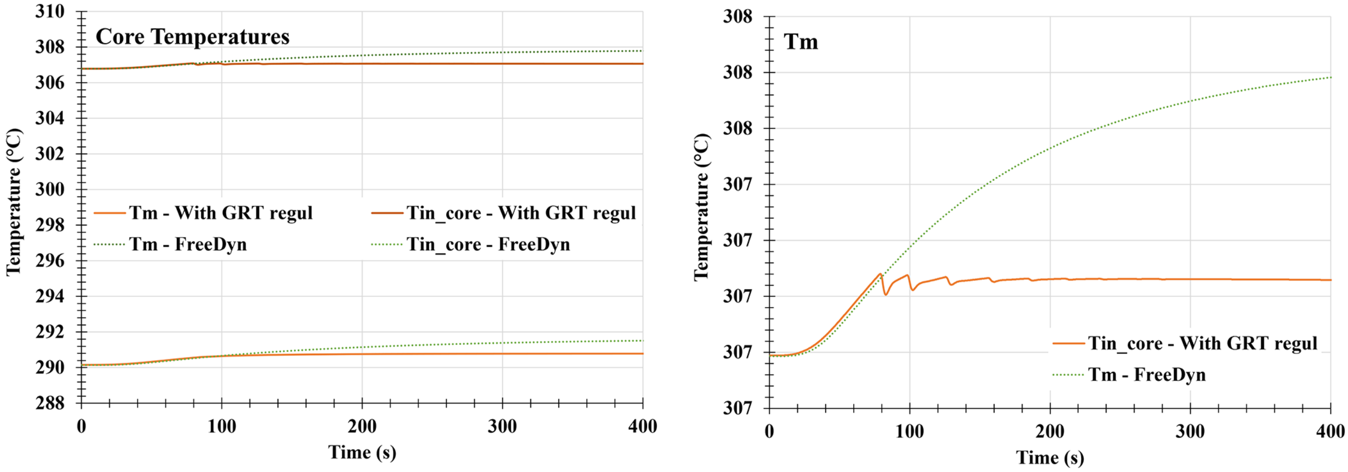


Fig. 2. Temperatures in the core including the average temperature in the core (T_m) and inlet core temperature (Tin_core) during a primary frequency control transient.

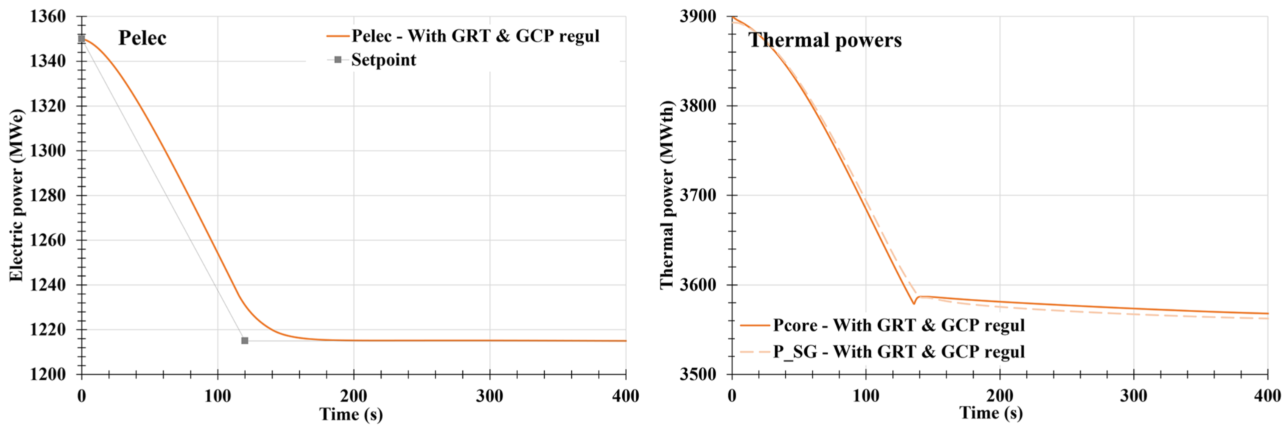


Fig. 3. Electrical and thermal powers (core Pcore and steam generator P_SG) during a load following transient.

the beginning and the end of the transient is small (less than 2 °C). Indeed, in free dynamics, the average core temperature increases by +1 °C due to neutron feedback reactions, whereas it remains almost constant under the effect of the GRT regulation (reference temperature of the GRT regulation varies less than 1 °C between 97.5% and 100%Pn), as shown on the left side of Figure 2. The beginning of the transient is identical in both cases. Moreover, the temperature kinetic is a slow phenomenon according to the figure.

2.2.2 Load following

The nuclear fleet is very much in demand for load following, i.e. to adapt daily consumption. For this purpose, a power program is sent to the reactor participating in the load following. This type of transients is also interesting because the quasi-systematic activation of the regulations is observed and in particular of the GCP regulation which follows a power program. Figure 3 shows the behavior of the PWR in the case of load following where all the rods regulations of the reactor are available. A transient drop in reactor power with a ramp of 5%Pn/min for two minutes (i.e. –10%Pn) is presented.

As before, the electrical power actually produced by the reactor is slightly behind the setpoint, due to the activation of the inertia of the turbine-generator unit and the I&C (instrumentation & control) driving the opening of the turbine inlet valve. On the side of the core response in Figure 3, the power produced by the core follows the electrical power by decreasing in a quasi-instantaneous way thanks to, in particular, the activation of the control rods. The response kinetics is smooth thanks to the GCP rods which regulate the power by anticipation, except after 130 s when GCP rods insertion stops, which causes a slight discontinuity. The temporal evolution of the thermal power of the core results from the competition between two phenomena: the insertion of the rods and the neutron feedback coefficients.

- The purpose of inserting the two sets of control rods in the core is to regulate respectively the average temperature of the water in the core (GRT) and the thermal

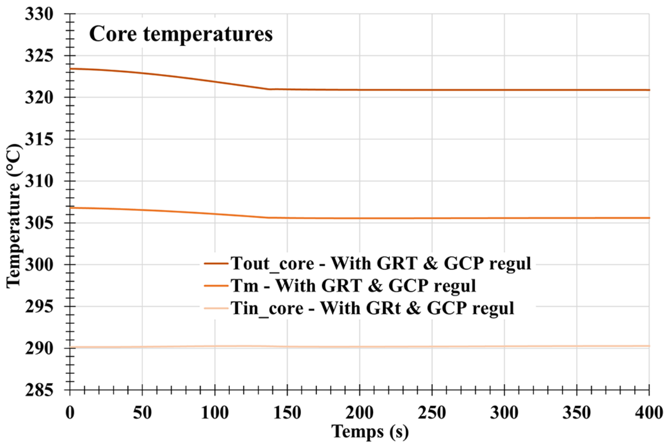


Fig. 4. Temperatures in the core (inlet Tin_core, outlet Tout_core and average Tm) during a load following transient.

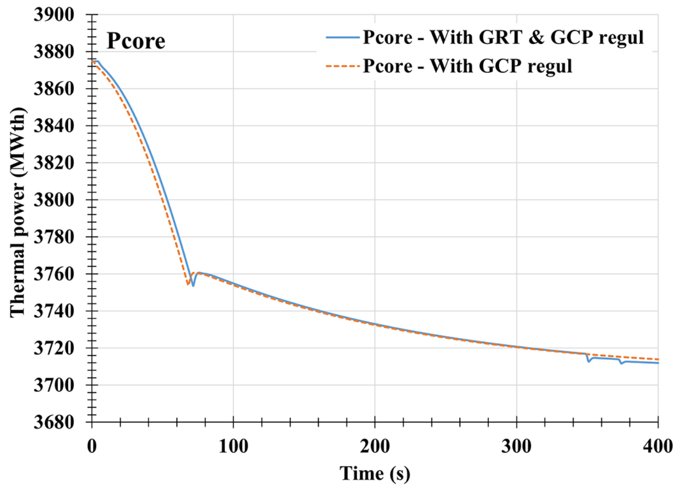


Fig. 5. Thermal core power during load following according to the activated regulation.

power produced by the core (GCP). The core power *Pcore* drops in accordance with the power schedule and then affects the power exchanged at the steam generators *P_SG*. As the power drops, the average core temperature decreases according to Figure 4. The temperature control (GRT) acts on the core inlet temperature, the inlet temperature is kept as constant as possible, the fluctuations are less than 0.2 °C for an inlet temperature of 290 °C. The regulation prevents the average temperature from deviating from the reference value. Finally, the temperature at the core outlet follows the same behavior as the average temperature as the input temperature is constant.

– The feedback coefficients of the core are intrinsically linked to the evolution of the temperature of the water and the fuel. These phenomena lead to an increase in the temperature of the water and in the core inlet, because the power extracted from the steam generators decreases, but this is by nature a slower phenomenon than the regulation, which explains why it is not observable.

As a result, both the average water temperature and the core power stabilize after 140 s. This load-following transient allowed to visualize the effects of power regulation (GCP), whose main role is to accelerate the response of the core; the electrical power reacts quickly. It is the power regulation that primarily guides the power decrease.

This section has shown the actual response of a nuclear unit during a frequency transient. The aim of the modelling, presented in the following, is thus to reproduce this kind of behavior. Before presenting the model, consider the role of the temperature regulation in load following.

2.2.3 Sensitivity of the GRT regulation

A load following of –5%Pn is performed with the simulator. According to Figure 5, the core power is almost the same with or without GRT regulation. It seems that the temperature control do not impact the core dynamic when power control is activated. The GRT regulation influence is neglected in the following in a first approach.

In sum, Table 1 gives the possible configurations of the regulations and their respective activations during the controls.

This section showed the role and the conditions of activation of regulations during transients of electrical power variation; next step consists in defining a model dedicated to the simulation of frequency and load control transients.

3 Modelling and validation

3.1 Objectives of the modelling

The idea of this paragraph is to define the specifications for developing a nuclear reactor modelling tool. The aim of this model is to reproduce the behaviour of a reactor in case of grid disturbances and to observe the evolution of physical quantities during the transient considered.

This model is primarily used to observe the behaviour of a reactor and the analysis of variables of interest during power transients imposed by the electrical network. The tool must be able to reproduce transients related to current and prospective primary frequency controls. The network requirements for primary frequency control may evolve in the coming decades; the transients performed will certainly become more constrained. The tool must be able to take this evolution into account. To fulfill that, two main assumptions are made for such transients:

– Electric power ramps may become greater than 5%Pn/min according to [1].
– Larger variation magnitude than 2.5%Pn are considered. For example, a magnitude of 10%Pn is applied.

In the following, the power transients may have higher amplitudes than those classically related to the primary frequency control, i.e. they may have amplitudes of the order of all frequency controls (primary, secondary or tertiary). In other words, a new equivalent primary frequency control is introduced; it includes all three frequency controls, also called “equivalent frequency control”. Technically, this equivalent control remains a

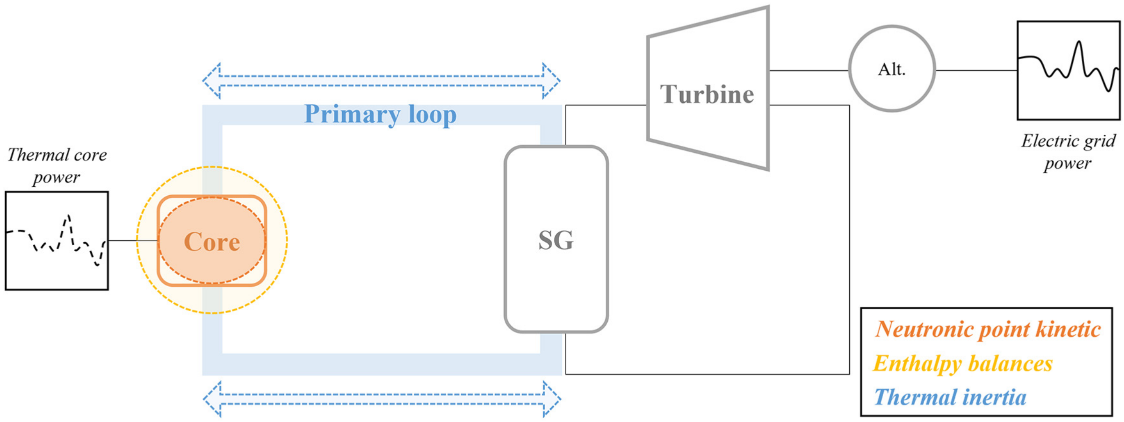


Fig. 6. Diagram of the simplified modelling of a 1300 MW PWR type reactor.

primary frequency control with the associated constraints (continuous response to network variations and repeated occurrences, etc.). Thus, all the characteristics related to the primary frequency control remain the same except for the amplitude of the variation, which can be significantly increased. It is considered that the threshold value 2.8%Pn of the GCP regulation activation is not modified, i.e. for small magnitudes, the regulation does not activate at all.

Secondly, the modelling must be easily modifiable because the aim, in the long run, is to modify design parameters to improve the flexibility of the reactor.

Therefore, the chosen level of modeling must allow the restitution at a sufficient level of the important physical phenomena governing a transient driven by the power demand, while ensuring fast calculations in order to perform parametrical and sensitivity studies. The 1300 MW_e PWR is selected for the modeling and it is implemented in MATLAB. The nuclear reactor is considered in normal operation and close to its nominal operating point at 100%Pn.

3.2 Main assumptions and short description of the modelling

The modeling must link the temporal evolution of the electrical power imposed by the network via the alternator of the power plant, and the power generated by the core of the reactor through the various primary and secondary circuits. To do this, a model of the different stages of energy conversion and heat transfer from the fuel to the grid is suggested. Neutronic and thermal aspects are taken into account in the modeling as well as the neutronic-thermal-hydraulic coupling. Figure 6 shows a schematic diagram of a reactor and the elements taken into account in the selected modeling. This model was inspired by the references [6,7].

For neutronics, only the feedback coefficients due to the Doppler effect and the moderator effect are taken into account because during the time involved, other feedback coefficients such as Boron or Xenon do not act significantly. The calculation of the temporal evolution of this global power is calculated using a point kinetics (0D) model taking into account the feedback coefficients. At first approximation, the use of point kinetics is applicable because spatial effects are negligible as power transients are closed to the nominal point. The reactor is not at the end of the irradiation cycle in order to avoid additional constraints for the power fluctuations due to the end of the irradiation cycle.

The thermal exchanges in the core are assumed to be homogeneous and uniformly distributed in the core during normal operation. This is why the study is restricted to the 0D modeling of a single fuel rod representative of the core. This assumption allows to deduce the temperatures at each stage of heat exchange by applying enthalpy balance. As an example, the temporal evolution of the average water temperature Tm is given in equation (1). The fuel temperature T_{fuel} is obtained in the same way.

$$\frac{dT_m}{dt} = \frac{1}{\rho_w c_{pw} V_{pipe}} \frac{T_{fuel} - T_m}{R} + \frac{Q_{vpipe}}{V_{pipe}} (T_{in_{core}} - T_{out_{core}}). \quad (1)$$

It depends of several thermalhydraulic parameters listed below:
- ϱw et c_{p_w}: density and mass heat capacity of the water;
- V_{pipe}: volume of water around a fuel rod in the core;
- R: Equivalent thermal resistance between fuel and water from the thermal/electrical analogy, this corresponds to the “resistance” of the material (solid or fluid) to the passage of a heat flux;
- $Q_{v_p i_p e}$: volume flow rate of the water in the primary circuit.

Thus, the coupling of kinetics and thermal is possible. Moreover, the components of the primary circuit such as the pressurizer and the primary pumps are not taken into account in the model; the primary flow rate is fixed by the constant rotation speed of the pumps, and the pressure is imposed at 155 bar. These assumptions are possible because it occurs during normal operation of frequency control where the regulation of the pressure is effective and the mass flow rate is constant. An equivalent loop of the primary circuit instead of four loops on a 1300 MW PWR is then considered. Finally, a time delay models the delay in the primary circuit on the hot and cold pipe. For instance,

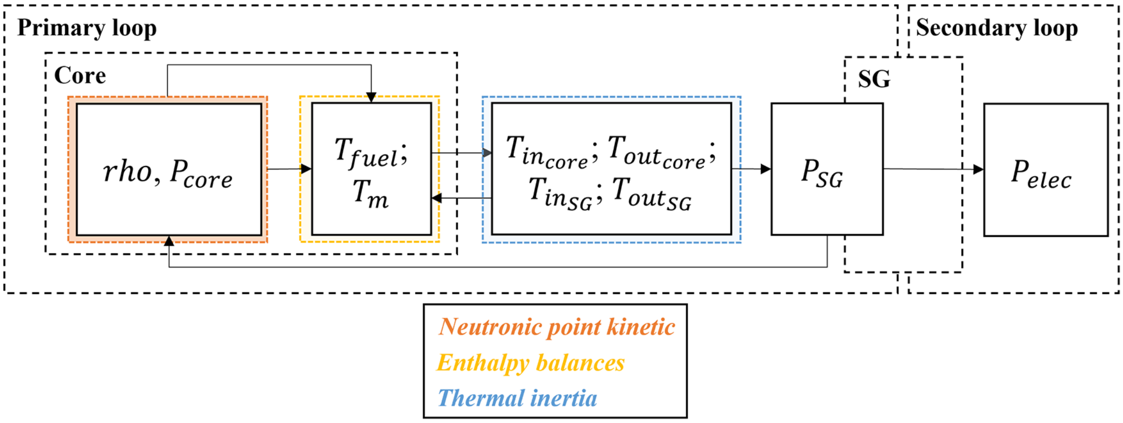


Fig. 7. Schematic of the quantities for the global modelling of the reactor.

Table 2. Role of the equivalent frequency control considered in this paper and associated regulation.

	Temperature regulation	Power regulation	Turbine-generator control
Free dynamic	0	0	X
Equivalent frequency control	0	X	X

the evolution of the inlet core temperature is described in equation (2).

$$\frac{dT_{in_{core}}}{dt} + \frac{T_{in_{core}}}{\tau} = \frac{T_{out_{SG}}}{\tau} \tag{2}$$

with τ, the time delay in the primary circuit.

All the quantities needed at each stage of the modelling are presented in Figure 7.

Finally, as a first approach, a simplified equivalent control is modelled to accelerate the response dynamics of the reactor in case of an electrical disturbance. Based on the sensitivity study in Section 2.2.3, not considering temperature control seems to be a relevant assumption for a preliminary approach. As shown in Figure 6, the regulation depends on the deviation between the power extracted from the steam generator and the thermal power of the core, and then adjusts this deviation according to a proportional coefficient. The activation or non-activation of the control is specified where appropriate.

Table 2 describes the role of the equivalent frequency control, compared to the free dynamic (no rods regulation at all).

3.3 Validity of the model

This section consists in validating the model behavior (referred as MATLAB) in grid-following and load-following modes of conventional reactor, presented in Section 2.2. The section also defines the validity domain of the model with regard to the regulation of the equivalent control. The validation was performed with the PWR-C-1300, named C-1300 in the following simulator which is an academic simulator developed by Corys [5]. The initial operating point is of 100%Pn of its nominal power, i.e. about 3900 MW_{th} and 1350 MW_e.

3.3.1 Load-following transient

Figure 8 shows the comparison of the electrical and thermal powers obtained for a classic load following transient, i.e. a power drop of 10%Pn for 2 min.

The corresponding core power output is similar to the reference case; the final core power drop difference is almost identical between the two cases (discrepancy is <1%). Moreover, the core power ramps are also very close. In a first approach, the maximum core thermal power gradient allows to quantify this slope. The gradient is equal to 4.7% Pn/min for the simulator against 5%Pn/min with MATLAB. The difference between the two cases is small.

It is also shown that the model correctly accounts for the behaviour of the reactor in case of load following. The regulation used in MATLAB is obviously simplified, but it allows the evolution of the core power to be calculated correctly.

For a transient of 10%Pn, it confirms that the model is correct. The following section discusses the behaviour of the model at low power magnitudes, i.e. for traditional primary frequency control.

3.3.2 Grid-following transient

As a reminder, only GRT regulation is active for this type of conventional transient during current operations. According to Figure 9, the behavior of the core power seems satisfactory only at the beginning of the transient (until 80 s). This start of the transient may correspond to free dynamics, as shown on the green curve of the figure

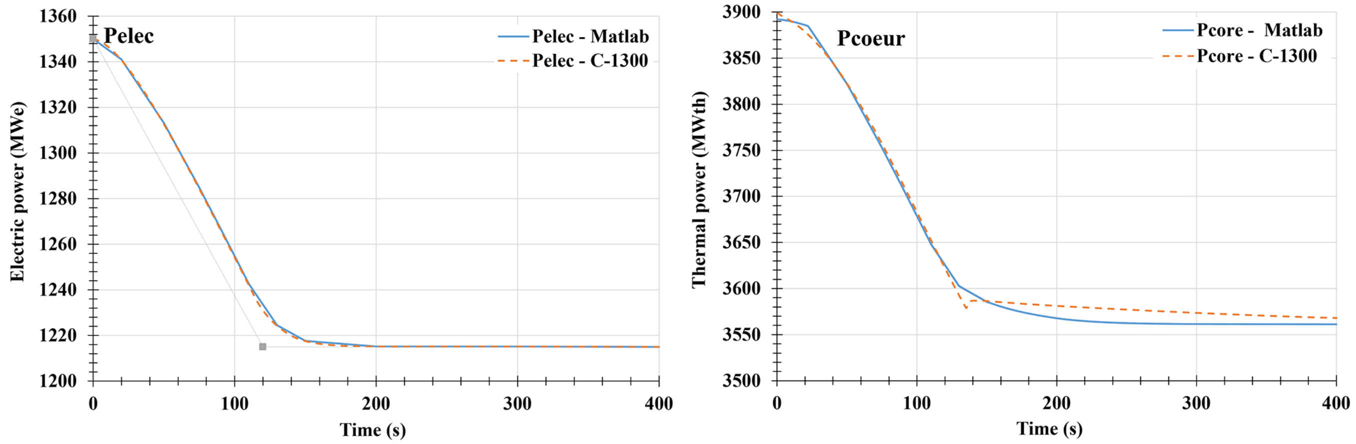


Fig. 8. Electrical and thermal power for the Matlab load-following validation case.

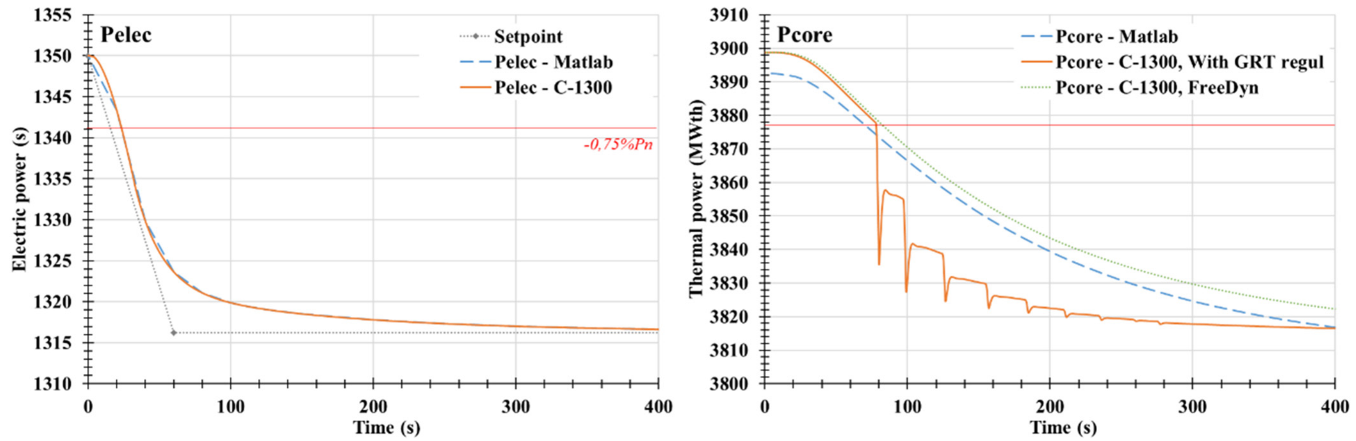


Fig. 9. Comparison of electrical and thermal power for the grid-following validation case.

below. It means that between about 0.75%Pn and 2.5%Pn of electric power, the model is not correct. However, in free dynamic, we see that time response and dynamic of the model is satisfying. A slight deviation of less than 1% remains but it is not significant. The stationary losses refers to the primary pumps power.

At the temperature level, the model is correct, as shown in Figure 10. The difference between MATLAB and the simulator is less than 1 °C for both cases in free dynamic.

In summary, the model described gives satisfying results over the following power ranges:

– 0 to 0.75%Pn of electric power without the use of equivalent regulation;
– For transients whose amplitude of variation is larger than 2.5%Pn of the electrical power; in this case the use of equivalent control is required.

Transients out of the validation range can be further investigated with the addition of a control system. As a first approach, the validity domain is sufficient to study the impact of constrained transients on a current reactor.

4 Testing of the model in stressed conditions

4.1 Main impacted factors

A non-exhaustive list of relevant factors to characterize the reactor's capabilities during constrained transients of electric power, is presented in Table 3. Then, the extremal values of these indicators are selected and are given in absolute.

The next step of this work is to study the MATLAB validated model for power ramps higher than 5%Pn/min in order to observe the behaviour of the model under constraints. The factor values obtained with MATLAB for the case of 5%Pn/min with equivalent frequency regulation are taken as reference in the following. They are supposed to partially represent some of the main current capabilities of a reactor in terms of safety, performance and control. In the following, figures will always refer to MATLAB transients only. The results of the study are compared to the values obtained for the reference case of 5%Pn/min.

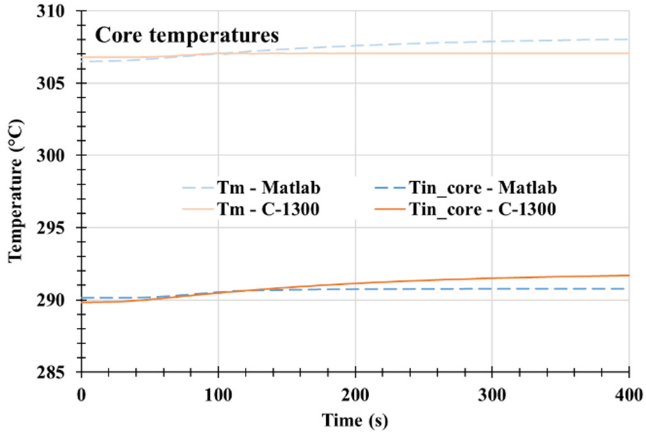


Fig. 10. Comparison of core temperatures for the grid-following validation case.

Table 3. Summary of selected factors.

Factors	Unit	Factor's definition	Impacted domain
$\frac{\Delta}{\Delta t}P_{core}$	MW/s	Core power gradient averaged over 1 sec	Core performance
τ_{core}	s	Response time at 95% of core power i.e. time after which the core power has reached 95% of its final value	Core performance
$\frac{\Delta}{\Delta t}T_{fuel}$	°C/s	Fuel temperature gradient averaged over 1 sec	Safety of the core
$\frac{\Delta}{\Delta t}T_{inSG}$	°C/s	Core inlet temperature gradient averaged over 1 sec	Safety of the steam generator
$\frac{\Delta}{\Delta t}T_{in_{core}}$	°C/s	Steam generator inlet temperature gradient averaged over 1 sec	Control rod use
$\Delta\rho_{ext}$	pcm	Maximum amplitude of reactivity inserted or withdrawn by the power control	Control rod use and associated safety

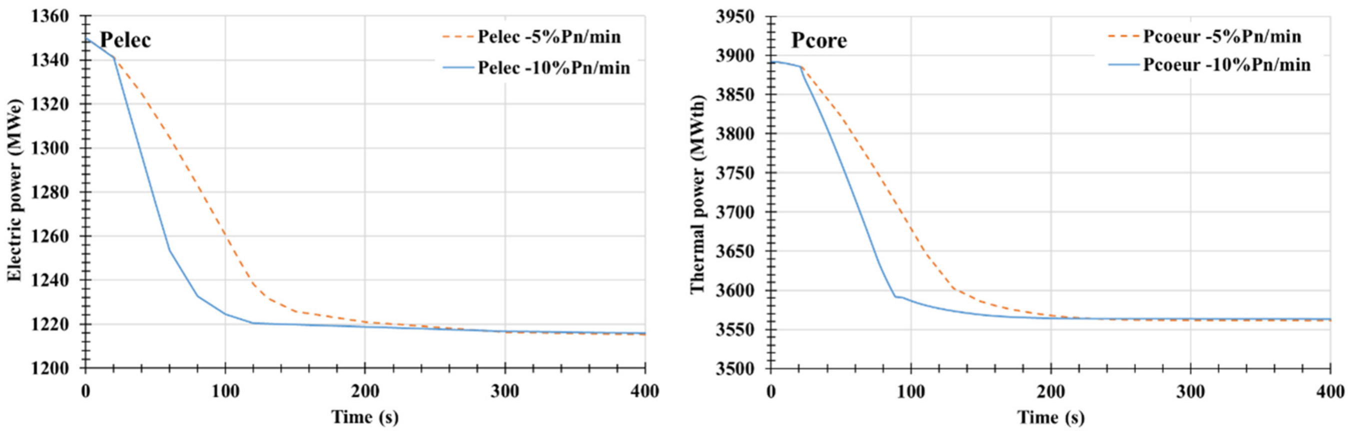


Fig. 11. Electrical and thermal power for a constrained transient at –10%Pn/min.

4.2 Model behavior

A power ramp of –10%Pn/min is applied at a –10%Pn decrease. As explained previously, this kind of transients may correspond to a potential evolution of the nuclear requirements with regard to the primary frequency control. This new transient is directly compared to the reference case (current maximum ramp of –5%Pn/min), which includes with the equivalent frequency control.

Table 4. Comparison of indicators for two constrained transients with a magnitude of 10%Pn.

Factors	Unit	Reference case–5%Pn/min	Stressed ramp–10%Pn/min	Stressed ramp –15%Pn/min
$\frac{\Delta}{\Delta t}P_{core}$	MW/s	3.25 (5.01%Pn/min)	5.41 (−8.34%Pn/min)	6.39 (−12.86%Pn/min)
τ_{core}	s	166	113	99
$\frac{\Delta}{\Delta t}T_{fuel}$	°C/s	0.41	0.64	0.80
$\frac{\Delta}{\Delta t}T_{in_{SG}}$	°C/s	0.01	0.02	0.04
$\frac{\Delta}{\Delta t}T_{in_{core}}$	°C/s	0.03	0.07	0.10
$\Delta\rho_{ext}$	pcm	76	69*	61*

* The reactivity balance between the different cases is slightly different because the balance of the temperatures in the core are modified as a function of the transient.

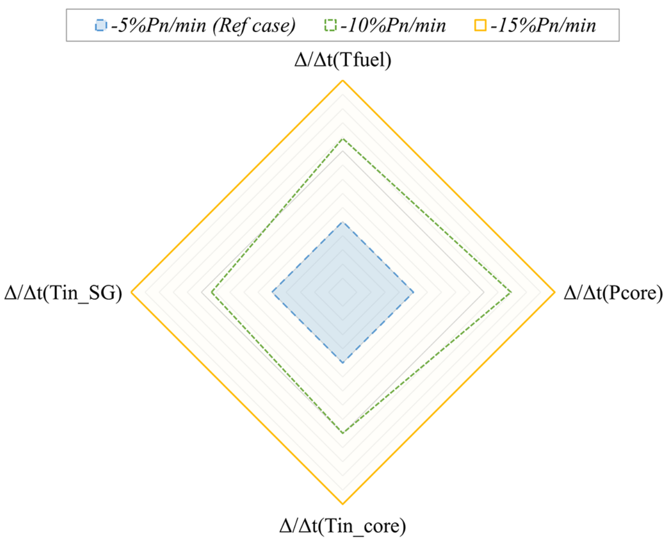


Fig. 12. Radar for two constrained transients with a magnitude of 10%Pn.

Figure 11 shows the core power trends for the two configurations. Table 4 completes the figures by giving the maximal values of all factors between the reference case and the case with −10%Pn/min power ramp.

The figure shows that the larger the electrical ramp, the faster the core power responds. In other words, core time responses are faster, based to the table below. But, according to Table 4, increasing the power ramp degrades the safety indicators in the same time, such as the fuel temperature gradient. A third case with higher ramp of 15%Pn/min is added to the table.

It can also be seen that the thermal constraints are affected as well by the increase in the ramp; gradients are increasing. Moreover, it can be noted that the use of regulation is increased when the power ramp is low.

From the four gradient factors, a normalization on each with respect to the maximum obtained on the different transients considered is performed. Time response as well as external reactivity are not included in the normalization process. Figure 12 shows a radar diagram taking into account the normalised factors of the three above configurations. This radar allows the previous results to be visualised in a different way, and facilitates the comparison between each transient. The reference case at 5%Pn/min is shown in dotted blue in the middle of the radar. The 15%Pn/min case is on the outside of the radar as the quantities are the most degraded. The last case at 10% Pn/min is between the other two. The extreme relative deviations for each quantity are added to the figure in relative terms.

In summary, the safety quantities are exceeded when the electrical power ramp is constrained and *a fortiori*, the use of regulation may be lower.

The following section attempts to recover as much as possible the factors values of the reference case (obtained for the ramp of −5%Pn/min). To do this, adaptations from the reactor design are made in order to take into account the new flexibility requirements.

5 Sensitivity studies of the model

5.1 Studied parameters

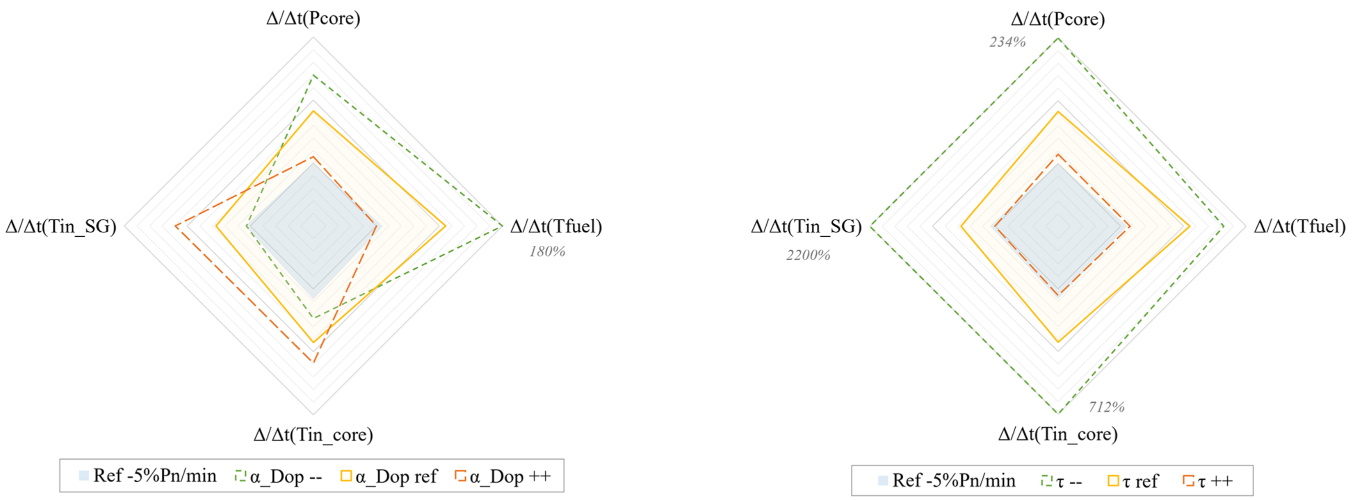
Six parameters are considered for the sensitivity study in order to modify the core design and observe the impact on the behavior following a grid disturbance. Only two parameters out of six (in bold underlined), are selected and presented in the paper, as the latter have the most notable influences:
- The **neutronic parameter Doppler coefficient** α_{Dop} is taken into account in order to modify reactivity balance;
- The neutronic parameter moderator coefficient α_{T_m};
- The **thermohydraulic parameter time response of the entire primary loop** τ is chosen in order to quantify thermal inertia in the loop;
- The thermalhydraulic parameter time response of the hot pipe;
- The thermalhydraulic parameter time response of the cold pipe;
- The insertion speed of the equivalent rod regulation.

Both selected parameters are modified in a range of variation chosen arbitrary and given in Table 5. The two types of parameter may directly affect the design of the reactor itself, i.e. the geometry of the core (size, number of assemblies, choice of fuel) or the format of the primary loop.

The following section consists of simulating an example of a constrained transient, namely a ramp of −15%Pn/min, in order to get closer to the flexibility criterion, with the aim of exploring design levers that act on the flexibility of the reactor.

Table 5. Sensitivity study parameters and modification range.

	τ (s)	α_{Dop} (pcm/°C)
–	0.1	–0.83 (÷3)
Ref	8	–2.5
++	100	–7.5 (×3)



α_{Dop} (pcm/°C)	--	Ref	++
$\Delta\rho_{ext}$ (pcm)	29	61	145
τ_{core} (s)	60	99	162

τ (s)	--	Ref	++
$\Delta\rho_{ext}$ (pcm)	0	61	111
τ_{core} (s)	54	99	149

Fig. 13. Radars for the sensitivity study according to both studied parameters.

5.2 Main results

This section details the impact of both parameters on the behaviour of the model. For this purpose, each of the parameters is modified within its range of variation, as shown in Table 5. Figure 13 summarizes the radar diagrams obtained for both parameters. Each radar diagram is completed by a table which lists the values of the core response time as well as the use of regulation.

The outer ring of the radars indicates the maxima achieved for each of the quantities for the entire sensitivity study (with all parameters non described in the study), while the inner ring indicates the minima. On each of the diagrams, the blue zone represents, for information, the factor's values for the reference case (ramp of 5%Pn/min). At a first approach, this blue zone is almost identical to the inner ring. This means that the factors are close to the minimum in the reference case. The yellow area refers to the values obtained when the model is unchanged and the ramp is 15%Pn/min.

In general, increasing the electrical power ramp degrades the various factors as we move away from the reference case in blue, even though some configurations allow us to return to acceptable values of the factors.

- Doppler coefficient impact: the Doppler effect primarily impacts the response dynamics of the core and fuel. When the Doppler coefficient is high (orange curve), the core response time is increased and the gradient on the core power is de facto low. The high Doppler requires a significant rod intervention i.e. strong external anti-reactivity input in addition to the neutron feedback coefficient of the moderator effect to compensate the Doppler effect. This is even the configuration that puts the greatest strain on the regulation. At the same time, an increase in water temperature variations is observed in the case of a strong Doppler coefficient.
- Primary loop inertia impact: the thermal inertia affects the water temperature gradients, although there is a significant indirect impact on the core. The majority of the extreme values in all simulations are obtained with this parameter. The almost zero delay results in a very strong gradient on the fuel temperature and a fortiori on the power produced by the core, as well as strong temperature variations on the primary circuit water. However, this requires no power regulation. In contrast, a strong thermal buffer reduces the stress on the core but implies a very slow core response. The stress on the control rods is very high in case of a high delay in the whole primary circuit. Apart from the increased use of power regulation, the config-

uration with the maximal thermal delay, for the water temperature gradients, is very similar to the reference case obtained for a light ramp of 5%Pn/min.

In summary, a low Doppler coefficient allows to return close to the acceptable fuel gradient. The neutron parameters have a direct influence on the fuel behavior. In addition, longer response time requires increased use of control rods but stresses on the fuel are lowered. This sensitivity study allowed us to observe the relative influence of both parameters on the model and the various factors of interest. These parameters affect the performance of the reactor, the safety of the core and the use of the control system.

6 Conclusion

In conclusion, the challenge of nuclear flexibility in an integrated power system can be addressed at different scales. In this paper, it has been chosen to focus on amplitude and time variations equivalent to the primary frequency control (see Sect. 3.1). Indeed, for other time scales and amplitudes, different phenomena would have to be considered otherwise.

This paper presented an innovative approach in order to introduce flexibility into the design process model. For this purpose, a model of a current 1300 MW PWR nuclear reactor is proposed. The input data of the electrical power ramp is used. After the validation of the tool during transients related to grid and load following, this modelling allowed to highlight phenomena undergone by the reactor during constrained transients.

Furthermore, it was shown that by modifying several design parameters (for example Doppler coefficient or time delay in the primary loop), the triptych performance, safety and control was reordered. This indicates that it is possible to put forward design parameters capable of improving flexibility. In the perspective, it would be interesting to associate the modification of a neutronic, thermal or regulation parameter with a technological solution.

Even if the model chosen in this tool is a 1300 MW PWR, it can be easily modified to other design of cores or even other solid fuel reactor types. This MATLAB tool could even be directly coupled to a grid simulation software (co-simulation), in order to directly connect the fields of power system and nuclear design, which have so far been unrelated.

Conflict of interests

The authors declare that they have no competing interests to report.

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