MAXIMUM POWER FOR A SELF-CONTROL LOW POWER REACTOR

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ABSTRACT

Low Power Research Reactors are attractive tools to start with nuclear activities as they can be used for educational purposes and to develop quality skills in the nuclear field. A relevant aspect to be considered in the design of these reactors is safety. According to IAEA's guidelines, key factors guaranteeing safety include (i) negative overall reactivity feedback coefficients through all operating stages and conditions, (ii) inherent safety features as well as passive systems for decay heat removal and (iii) self-actuated safety systems without any required operator action for a given time after a postulated initiating event.
Reactivity insertion is an initiating event to be considered in the Safety Analysis. When such event takes place, the power generated in the core goes up leading to an increase in the fuel and in the moderator temperatures. Reactors having a strong negative feedback coefficient, however, will respond by reducing the power generated in the core, even at power levels close to nominal conditions, without the need of conventional electromechanical devices or any operator action.
While the values of the feedback coefficients are important in the reactor control, the power at normal operation may also condition the possibility of the reactor to reach a stable power and remain in a steady state. The present study analyses the case of a Self-Controlled Low Power Reactor having negative feedback coefficients and being cooled by the natural convection regime.
The analysis is performed using RELAP5, a best-estimate code which considers the point kinetics model and values of reactivity feedback coefficients to estimate the fission power and its response to the changes taking place in the coolant and in the fuel temperatures. The reactor is modelled using RELAP5 code and a transient simulating an uncontrolled withdrawal of a control rod is analysed considering two different operational states.
For the nominal fission power guaranteeing the self-control of the reactor, the margins to the thermal-hydraulic critical phenomena are calculated and an uncertainty analysis is performed to account for the deviations from the best-estimate input values considered in the model. A sensitivity analysis is also performed to quantify the influence of the uncertainty on the different input parameters on the uncertainty in the reactor behaviour during the transient.

1. Introduction

Low power reactors constitute a useful tool for student and human resources training purposes and they have been used by different countries as a starting point for nuclear energy development. Low power reactors have certain design characteristics enhancing safety, which include:

- Excess reactivity lower than one dollar and large shutdown margins
- Negative feedback coefficients
- A natural convection cooling regime with a large reactor pool acting as the heat sink

The study presented in this paper analyses the behaviour of a low power reactor operating at a nominal fission power of 30 kW when a reactivity insertion accident (RIA) takes place. As reactivity is inserted into the system, the total reactor power is expected to increase thus resulting in an increase in the fuel and in the moderator temperatures. The negative feedback coefficients, however, lead to a reduction of the power generated in the core, thus reaching a stable operation which may guarantee reactor safety without the need of any active safety device or any external action.

The reactor being studied is cooled by a natural circulation cooling regime, meaning that the flow through the core is the result of a balance between the friction and the buoyant forces. The buoyant force depends on the changes taking place in the coolant density as it flows through the circulation loop. For a single-phase flow, these changes in density are attributed to the variations in the coolant temperature which depends, at the same time, on the flow through the core and the power generated. In case of an accident, this power varies with the coolant density and the fuel temperature due to the effect of the feedback coefficients. The interaction described strongly influences the behaviour of the reactor during the transient. Having an accurate calculation model therefore becomes a relevant issue for the Safety Analysis.

In the present work, a RIA is analysed by using the thermal-hydraulic system code RELAP5. The influence of different parameters affecting the steady state is first analysed with the aim of adopting a suitable nodalization. The transient is later evaluated considering a best-estimate calculation approach for a nominal reactor power equal to 30 kW. Deviations and uncertainties in relevant input parameters should also be considered during the analysis. Consequently, uncertainty bands defining maximum and minimum values achieved for the relevant safety figures of merit (design criteria) are also calculated. A sensitivity analysis is performed to determine the influence of the uncertainty in the different input parameters on the uncertainty over these figures of merit. Finally, the transient is evaluated for the reactor during start-up at cold conditions (1 Watt and T_{pool} = 20 °C), "**cold start-up**".

2. Safety relevant parameters

The present work aims at determining the changes in the reactor power after a RIA and verifying that the thermal-hydraulic design criteria are guaranteed. These design criteria include:

Burn-Out Ratio **(BOR):** the ratio between the heat flux leading to the Burn-Out (BO) phenomenon (q''_{BO}), and the maximum heat flux (q''_{max}) in the hot channel, i.e., $BOR = \frac{q''_{BO}}{q''_{\max}}$

Boiling Power Ratio **(BPR):** the ratio between the boiling power (BP) and the maximum power (P_{max}) in the hot channel, i.e., $BPR = \frac{BP}{P_{\max}}$

A BOR and a BPR equal to 1.3 and 2.0 are adopted for both ratios for the transient and for the steady-state respectively. The minimum between these two ratios is considered in the analysis.

3. Analysis

As it has been established, the present work aims at evaluating the behaviour of a self-controlled low power reactor as a RIA takes place. For the adopted nodalization, the study is divided into three parts:

- A best-estimate plus uncertainty (BEPU) analysis to determine the upper and lower bounds for the safety-relevant parameters, with the reactor operating at a nominal power of 30 kW.
- A sensitivity analysis to determine the effect of the uncertainty in input parameters on the uncertainty in the results.
- An evaluation of the transient for the reactor operating at a "cold start-up".

The BEPU analysis is performed considering the input error propagation methodology. Uncertainties on potentially relevant parameters attributed to deviations in calculations and / or in measurements are adopted based on engineering judgement and experience. A normal distribution is considered for each parameter, resulting in a distribution of responses characterizing the reactor behaviour during the transient.
The number of calculations performed has been determined by considering the Wilks' formula. For a one-sided tolerance limit, a total of 59 runs are required to guarantee a 95% confidence level that the maximum or minimum code result will not be exceeded with a 95% probability of the corresponding output distribution (1).
The sensitivity analysis performed is based on the determination of the partial correlation coefficient. This coefficient quantifies the relationship between two variables while keeping the rest of them constant. A linear relationship between the variables involved is considered as the most relevant hypothesis.

3.1 Calculation model

Fig 1 illustrates the natural circulation cooling loop. The coolant flows through the core removing the energy released during the fission process. The reactor pool acts as a heat sink and its temperature is kept between 20 °C and 40 °C by the Pool Cooling Circuit. The flow through the core is induced by the buoyant forces originated by the changes in the coolant density as it moves through the circuit.

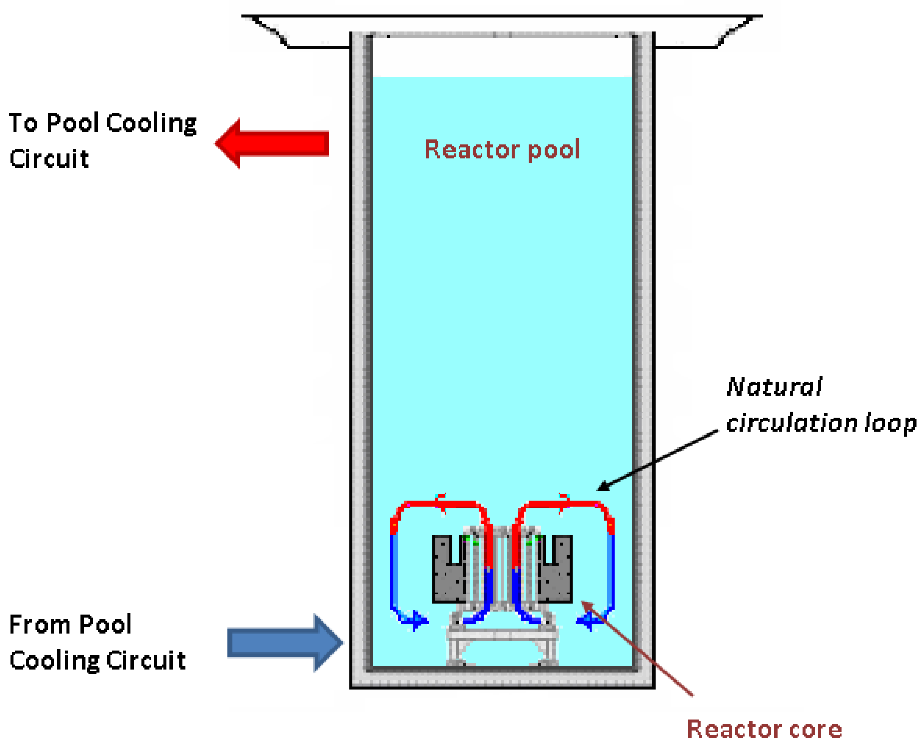

Fig 1: Cooling circuit

The cooling circuit is modelled in RELAP5 and the nodalization adopted to perform the analysis is illustrated in Fig 2.

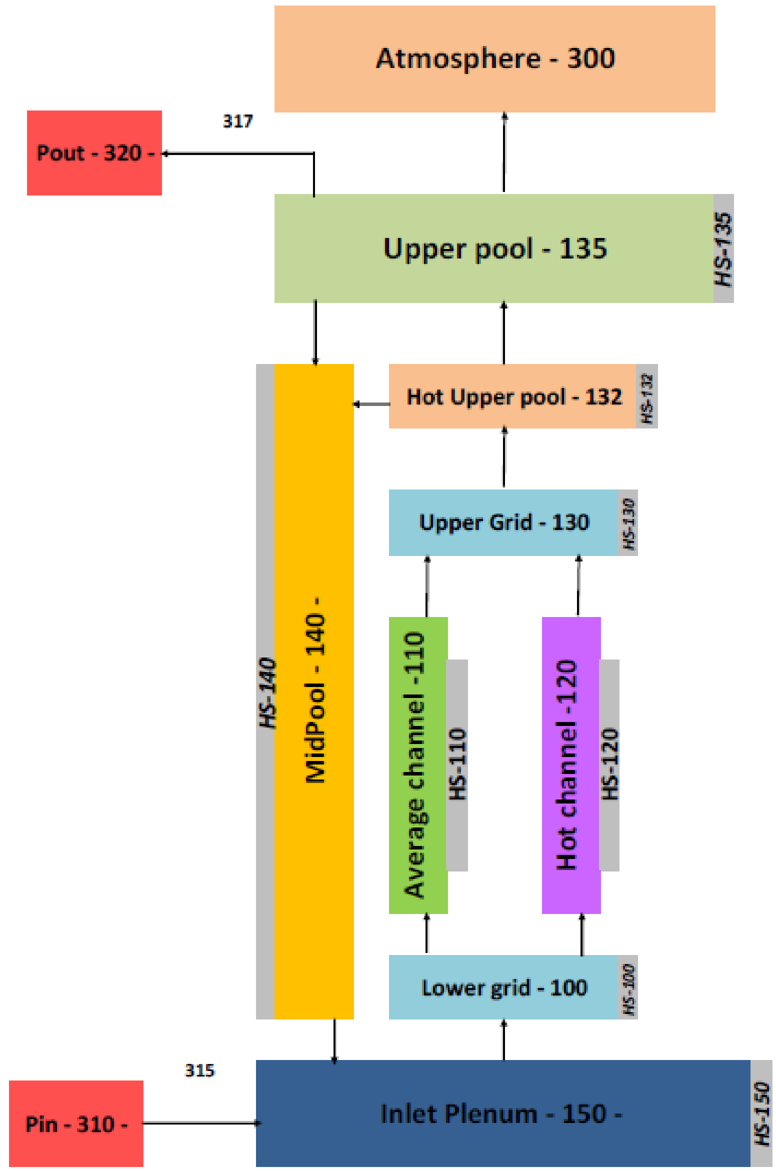

Fig 2: Nodalization in RELAP5

The operating conditions are fixed by the time-dependent volume 300. The pool is divided into two different volumes, a larger one representing the upper pool (135) and a smaller one acting as the heat sink for the coolant leaving the core (132). The reactor core is modelled by two different pipes representing the average (110) and the hot channels (120) used for the thermal-hydraulic analysis. The hydrodynamic component 140 represents the mid-pool and acts as the cold leg in the cooling circuit. The time-dependent volume 310 and the time-dependent junction 315 establish the operating conditions and the flow from the Pool Cooling Circuit. The power generated in the core is calculated by using the point kinetics model. Reactivity feedback coefficients are introduced in the feedback cards to calculate the changes in power as a result of the changes in the coolant density and in the fuel temperature.

Two parameters affecting the flow through the core and, consequently, the steady state considered to perform the analysis are the ***friction loss coefficients*** and the ***pool temperature.*** The friction loss coefficients considered are estimated based on CFX calculations for a similar core geometry. In addition, the temperature of the water in the pool varies between 20°C and 40°C depending on the cooling capacity of the Pool Cooling System. The values of these two parameters are varied with the aim of quantifying the influence of these two parameters on the steady-state, in order to define a "reference" steady-state to perform the analysis. A maximum deviation of 10 % in the coolant flow has been calculated being the pool temperature a more influential parameter when compared to the loss coefficient.

3.2 Input data

Tab 1 summarizes the input data and the deviation in each relevant parameter considered to perform the analysis.

Parameter	Best estimate value	Deviation
Initial power (kW)	30	+/- 10%
Pool temperature (°C)	30	1°C
Beta (pcm)	770	
Doppler reactivity coefficient (pcm/°C)	-2.6	+/- 15%
Coolant temperature (and void) reactivity coefficient (pcm/°C)	-14.9	+/- 15%

Tab 1: Relevant input data and deviation in parameters

As it has been established, a total of 59 runs are required to perform the BEPU analysis. The value adopted for each parameter in each run was randomly selected from its normal distribution and each value was randomly combined to generate the 59 inputs.

The starting point considers a 56% insertion of the control rod used to keep the reactor critical. All the other rods are assumed to be withdrawn. The RIA is modelled by considering the extraction of this rod at a maximum velocity of 3 mm/s. Tab 2 shows the positive reactivity introduced by the control rod extraction as a function of time. A deviation of +/ 20% is considered for the uncertainty analysis.

Time (seconds)	Reactivity ($)
0.00	0.00
20.00	0.29
40.00	0.55
60.00	0.75
80.00	0.88
100.00	0.94
120.00	0.94
140.00	0.94
160.00	0.94

Tab 2: Reactivity introduced by the control rod extraction

4. Results

Fig 3 shows the changes in the reactor power during the transient for the 59 and the best estimate (dotted line) cases being analysed.

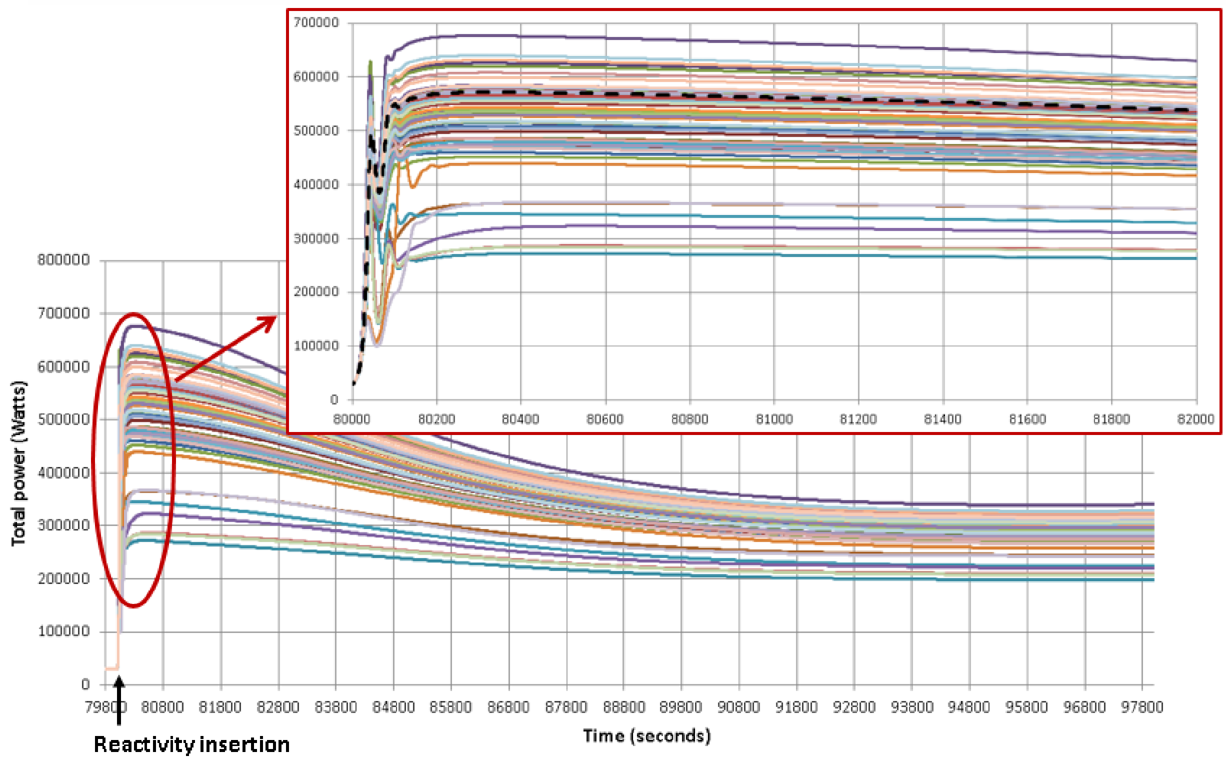

Fig 3: Evolution of reactor power after reactivity insertion

The analysis performed has shown that the BPR is the limiting design criterion. Its evolution during the event is illustrated in Fig 4

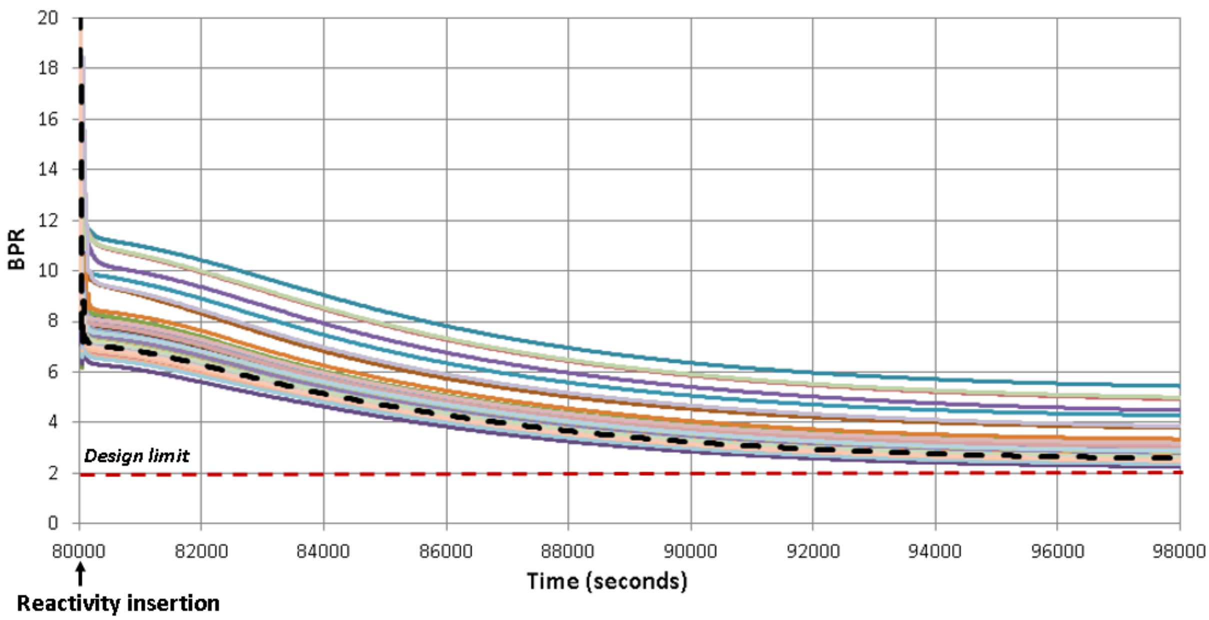

Fig 4: Evolution in BPR during the transient

The results show an increase in reactor power as reactivity is inserted into the core. The feedback coefficients respond to the changes in coolant density and in fuel temperature by reducing the reactor power until it stabilizes into a new steady-state. For the best estimate case, a maximum power equal to 0.56 MW is achieved. For the upper bound the maximum power increases to 0.67 MW achieving a final steady-state power equal to 335 kW. The analysis performed also shows that the BPR satisfies the design criterion during the transient, yet the margin for the new steady state is widely reduced when the lower bound resulting from the BEPU calculation is considered (BPR = 2.3). There is no mention of the BOR criterion as minimum values largely exceeds the margin adopted (BOR=11.0)

As previously mentioned, the sensitivity analysis is based on the determination of the partial correlation coefficient. The correlation coefficient quantifying the influence of the uncertainties in the input parameters on the uncertainty in reactor power and in the BPR, and its variation during the transient are illustrated in Fig 5 to Fig 8

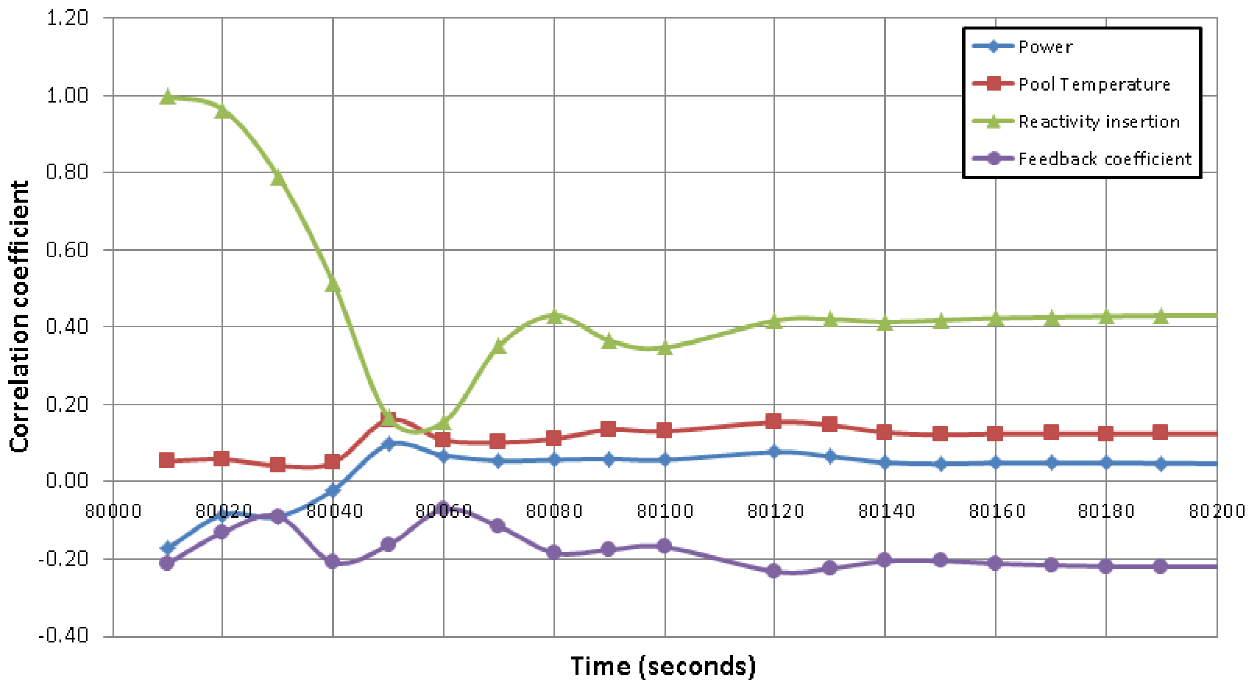

Fig 5: Influence of the uncertainty in input parameters on the uncertainty in **reactor power**

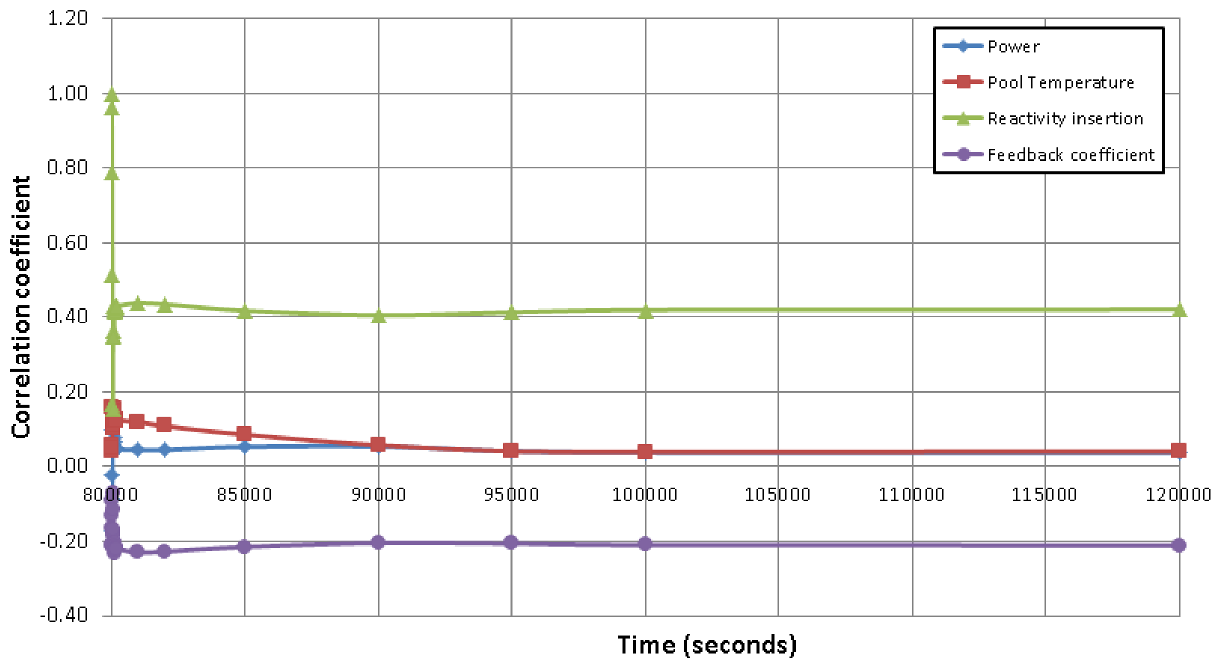

Fig 6: Influence of the uncertainty in input parameters on the uncertainty in **reactor power** – long term –

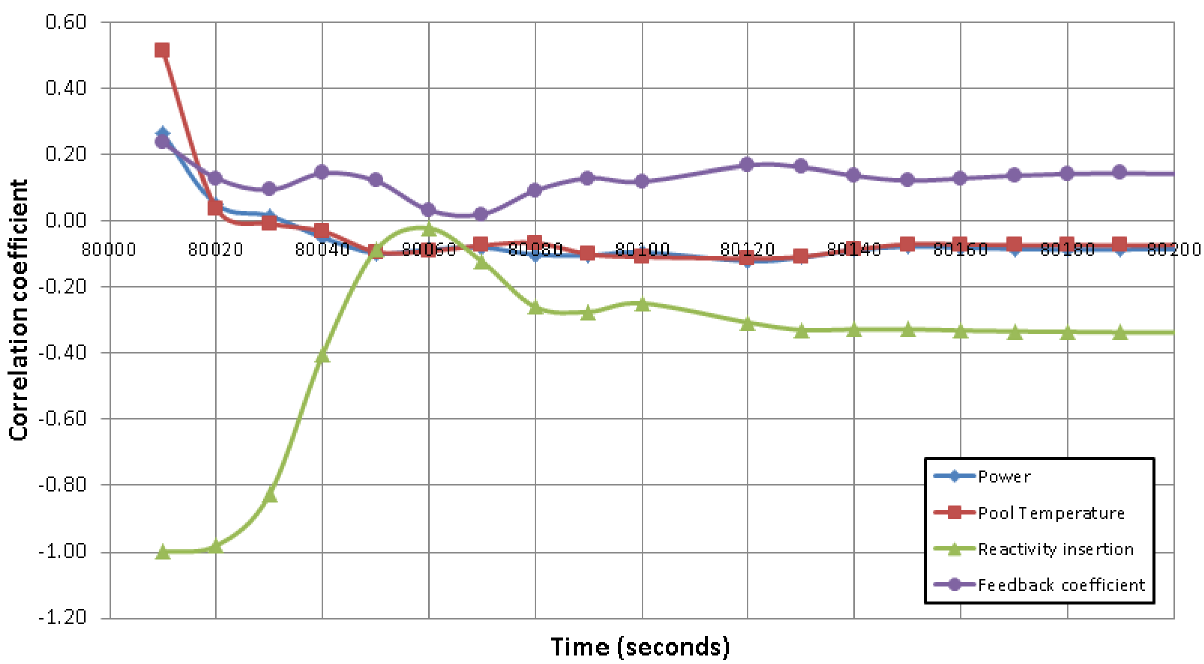

Fig 7: Influence of the uncertainty in input parameters on the uncertainty in **BPR**

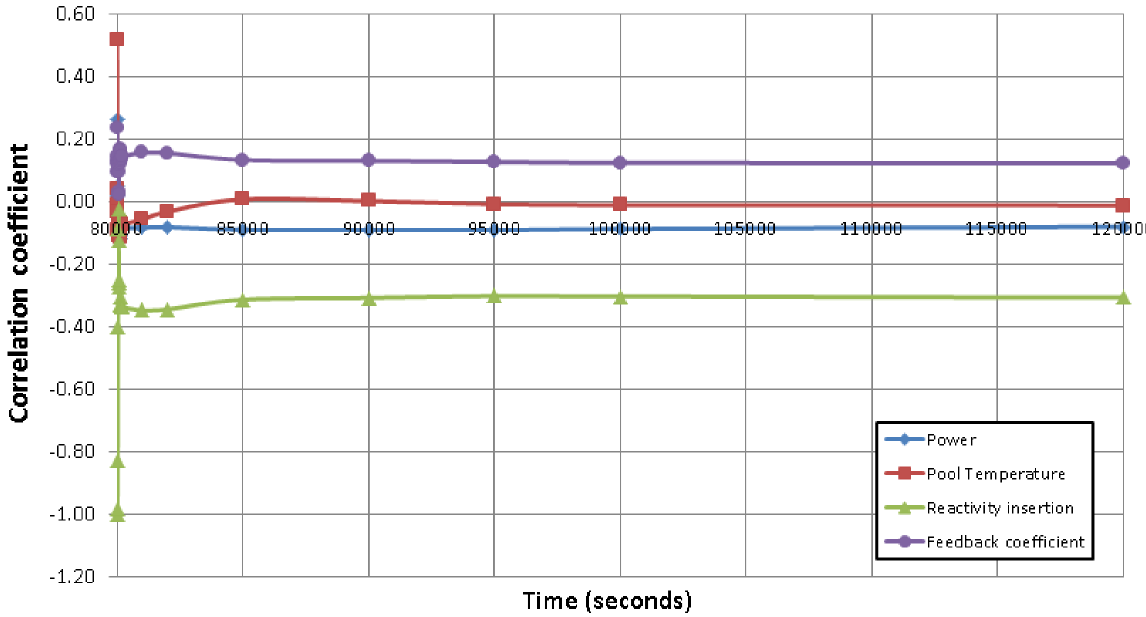

Fig 8: Influence of the uncertainty in input parameters on the uncertainty in **BPR** – long term –

The value for the correlation coefficient changes throughout the transient. Yet, the uncertainty in reactivity insertion is the one having the strongest influence on the results.

Finally, the reactor behaviour after a RIA during the “cold start-up” is analysed, when an uncertainty of +20% in the reactivity insertion is considered. A power peak of 5.9 MW is reached for the upper bound uncertainty while 1.4 MW is the result for best-estimate calculations. The minimum BPR for the best-estimate calculation is satisfied as it results equal to 1.3, however, for the conservative uncertainty of +20%, the BPR reaches a value of 1.1, for the new steady-state in the hot channel, exceeding the design limit.

5. Conclusion

In the present work, the behaviour of a self-controlled low power reactor after a RIA has been studied with the aim of determining whether its inherent safety features are capable of guaranteeing a safe operation. The reactor behaviour was modelled in the thermal-hydraulic code RELAP5 considering the point kinetics model for the determination of the power in the core during the transient.

Uncertainties in input parameters were considered to perform a BEPU analysis and calculate the changes in the reactor power during the transient and to verify that the thermal-hydraulic design criteria are fulfilled. The influence of the uncertainty in the input parameters on the uncertainty in the results was also determined by performing a sensitivity analysis, with results showing that the uncertainty in the reactivity insertion is the most influential one.

Based on this analysis, the RIA was also assessed for the reactor operating at a cold start-up. For the conservative case when the maximum uncertainty (+20%) in reactivity insertion is considered, the BPR design criterion cannot be satisfied for the hot channel.

6. References

1. Glaeser Horst. GRS Method for Uncertainty and Sensitivity Evaluation of Code. Results and Applications