Analysis of a rod withdrawal in a PWR core with the neutronic-thermalhydraulic coupled code RELAP/PARCS and RELAP/VALKIN

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Abstract

The Reactor Ejection Accident (REA) belongs to the Reactor Initiated Accidents (RIA) category of accidents and it is part of the licensing basis accident analyses required for pressure water reactors (PWR). The REA at hot zero power (HZP) is characterized by a single rod ejection from a core position with a very low power level. The evolution consists basically of a continuous reactivity insertion. The main feature limiting the consequences of the accident in a PWR is the Doppler Effect. To check the performance of the coupled code RELAP5/PARCS2.5 and RELAP5/VALKIN a REA in Trillo NPP is simulated. These analyses will allow knowing more accurately the PWR real plant phenomenology in the RIA most limiting conditions.

KEYWORDS: 3D Kinetics, RELAP5, PARCS, VALKIN, Reactivity transient, rod ejection, 3D coupled codes

1. Introduction

Nuclear industry and licensing authorities need to be able to rely on the good performance of methods and computer programs used in safety analysis calculations. This is best achieved through validation and benchmarking.

With the implementation of advanced fuel management, margins to safety and licensing limits are frequently reduced. This leads to general development of advanced methods that reduce the level of conservation by implementing kinetics methods that capture spatial effects occurring during reactor transients more accurately. The performance of physical models and numerical methods needs to be established over a realistic range of applications.

The Reactor Ejection Accident (REA) belongs to the Reactor Initiated Accidents (RIA) category of accidents, and it is part of the licensing basis accident analyses required for pressure water reactors (PWR). The REA consist of a rod ejection due to the failure of its drive mechanism. The evolution is driven by a continuous reactivity insertion. The main factor limiting the consequences of the accident is the Doppler Effect.

In this paper it has been analyzed the behavior of a Trillo NPP core configuration in a REA (Rod Ejection Accident) with hot zero power (HZP) conditions at the beginning of cycle (BOC) and at the end of cycle (EOC), using the coupled neutronic-thermalhydraulic codes

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RELAP5/PARCS2.5 [7] and RELAP5/VALKIN [4]. The steady-state results have been compared with the physics code CASMO4-SIMULATE3 [5][6], which provides the cross-sections sets and other neutronic parameters for the full transient by using the SIMTAB Methodology [1].

The transient departs from an initially critical core, being the withdrawal speed of the control rod a typical bounding value. In HZP conditions a power increase occurs while important power distribution changes take place in the core. In this accident, the maximum power is less important than its time integral. If the reactivity insertion rate is low, the heating of the fuel may be sufficient to have Doppler anti-reactivity, balancing the inserted reactivity while the power level is still under the trip level.

These analyses will allow knowing more accurately the PWR real plant phenomenology in the RIA most limiting conditions, in this way the conclusions will be realistic.

2. Description of the Model

The model used for simulating this transient is formed by a very detailed 3D core with boundary conditions.

The reactor core studied is composed of 177 fuel elements, being the number of fuel rods per fuel element equal to 236 with 20 guide tubes. The neutronic nodal discretization consists of 177 x 32 active nodes, considering 20 different fuel elements with 611 neutronic compositions. The cross-sections tables are generated with the SIMTAB methodology from CASMO4-SIMULATE3 code. A sensitivity analysis, using more compositions and comparing the results with CASMO4-SIMULATE3 code, demonstrates that the considered number of neutronic compositions is adequate.

The coupled codes RELAP5/PARCS2.5 and RELAP5/VALKIN are neutronic-thermalhydraulic codes to simulate 3D-geometry complex neutronic phenomena and thermalhydraulic events in multiple 1D-geometries channels. The reactor core has been modeled with 10 thermalhydraulic channels connected with branches (BRANCH) and the by-pass has been modeled as an independent channel (see Fig. 1). A time dependent volume (TMDPVOL) and a time dependent junction (TMDPJUN) simulate the boundary conditions at the entrance and exit of the reactor core as is shown in Fig. 2. Each thermalhydraulic channel representing the core is connected to a heat structure.

Figure 1: Thermalhydraulic channels.

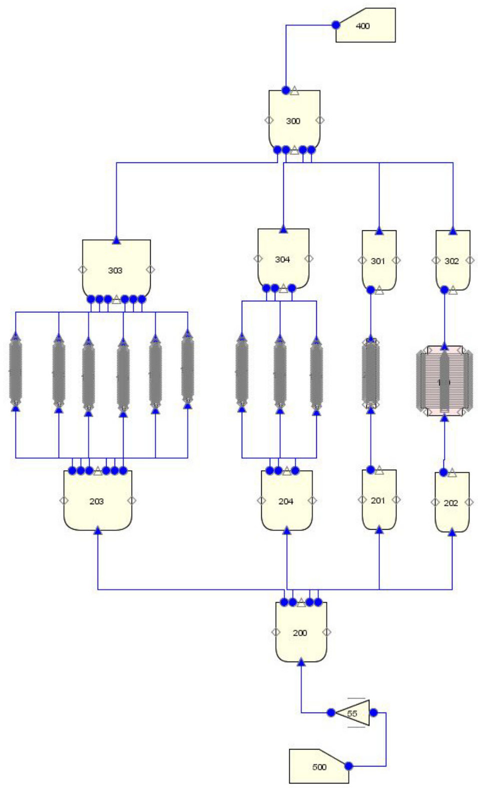
Q	P	O	N	M	L	K	J	I	H	G	F	E	D	C	B	A	
					0	0	0	0	0	0	0						1
			0	0	0	100	100	100	100	100	0	0	0				2
		0	0	100	100	100	100	100	100	100	100	100	0	0			3
	0	0	100	100	100	100	100	100	100	100	100	100	100	0	0		4
	0	100	100	100	100	100	100	100	100	100	100	100	100	100	0		5
0	0	100	100	100	100	100	100	100	100	100	100	100	100	100	0	0	6
0	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	0	7
0	100	100	100	100	100	100	100	100	100	100	100	100	100	100	100	0	8
0	100	100	100	100	100	100	100	100	100	100	100	102	103	104	100	0	9
0	100	100	100	100	100	100	100	100	100	100	100	105	101	106	100	0	10
0	100	100	100	100	100	100	100	100	100	100	100	107	108	109	100	0	11
0	0	100	100	100	100	100	100	100	100	100	100	100	100	100	0	0	12
	0	100	100	100	100	100	100	100	100	100	100	100	100	100	0		13
	0	0	100	100	100	100	100	100	100	100	100	100	100	0	0		14
		0	0	100	100	100	100	100	100	100	100	100	0	0			15
			0	0	0	100	100	100	100	100	0	0	0				16
					0	0	0	0	0	0	0						17

The inlet mass flow through the core is 15605.7 kg/s and it is distributed uniformly among the channels.

The neutronic model uses two prompt neutron groups and six delayed neutron groups, while the boundary condition for the neutron diffusion equation is zero-flux at the outer reflector surface.

Radially, the core is divided in 23 cm x 23 cm cells, each corresponding to one fuel assembly, plus a radial reflector. There are 177 fuel assemblies and 64 reflector assemblies. Axially, the core is divided into 34 layers (32 fuel layers plus top and bottom reflector) with 10.625 cm height each one, with a total active core height of 340 cm.

Figure 2: RELAP5 model.



A previous analysis determined that the control rod with the maximum worth belongs to the bank number 6 and it is located at position D-10 (see Fig. 1 and Fig. 3). The thermalhydraulic channels surrounding the ejected control rod have been modeled as independent channels, while the others have been grouped in a unique channel, because the phenomena of this kind of transients in HZP conditions are localized around the ejected control rod.

Control rods are grouped in 13 banks: initially banks 1, 5 and 6 are totally inserted and the other ones are out of the core. Fig. 3 shows the control rod banks and the ejected rod D-10 is highlighted in red.

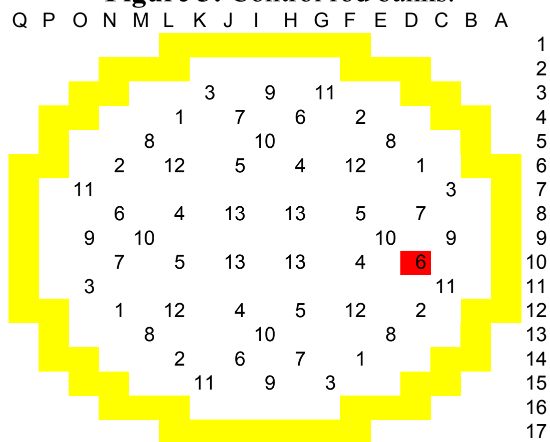
The initial steady state is a HZP where the moderator density is 734 kg/cm^3 and the fuel temperature is 569.55 K.

The transient is started by the ejection of the rod D-10 which is completely extracted in 0.1 s. The values of control rod worth and β_{eff} for BOC and EOC are presented in Tab. 1.

Table 1: Values of β_{eff} and control rod worth.

Case	β_{eff}	Control rod worth (pcm)	Control rod worth ($)
BOC	0.00605	408	0.674
EOC	0.00527	408.8	0.775

Figure 3: Control rod banks.



3. Numerical results

3.1 Steady state results

Initial steady state has been simulated with the RELAP5/PARCS and RELAP5/VALKIN coupled codes. The Tab. 2 shows the obtained results of the k_{eff} and the Fig. 4 and Fig.5 show the axial power profile for the different simulations.

Table 2: Values of k_{eff}.

Case	k_{eff} SIMULATE3	k_{eff} RELAP5-PARCS	deviation (pcm)	k_{eff} RELAP5-VALKIN	deviation (pcm)
BOC	0.982855	0.983032	17.7	0.983029	17.4
EOC	0.982733	0.983570	83.7	0.983180	44.7

Figure 4: Axial power profile at BOC.

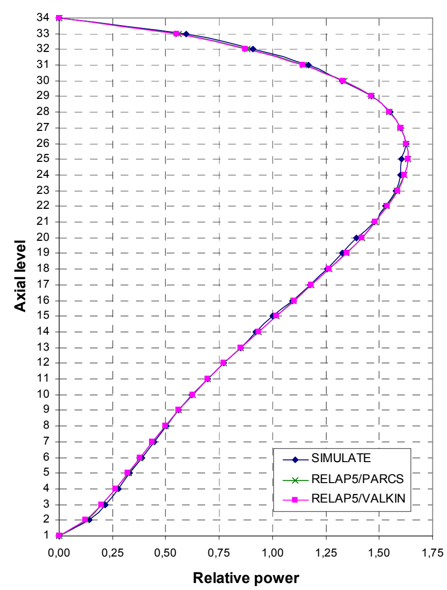
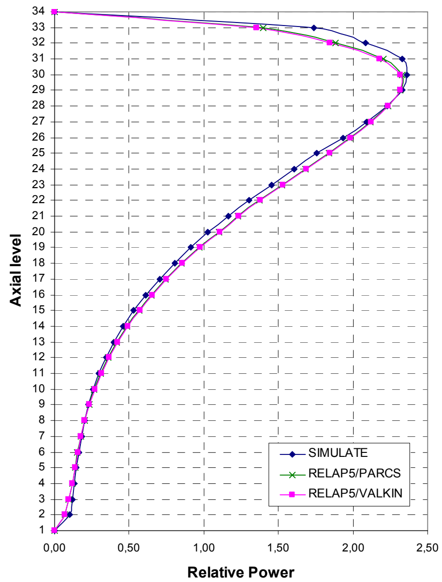


Figure 5: Axial power profile at EOC



3.2 Transient results

In both cases, BOC and EOC, zero-power state was considered as initial state and the control rod group 6 completely inserted, then the control rod D-10 is ejected. The evolution consists of a continuous reactivity insertion. The transient is terminated by the Doppler Effect caused by the increased fuel temperature, but this occurs before it reaches the limit that can be dangerous for the nuclear power plant safety, as it was expected.

Fig. 6 and Fig. 7 show the power evolution during the transient in both analyzed situations BOC and EOC.

Both neutronic modules, PARCS and VALKIN give practically the same transient results. The Fig. 6 to 12 correspond to PARCS calculations.

The Doppler temperature calculated by PARCS and VALKIN codes is found from the fuel temperature at the fuel rod center T_{fc} and the fuel rod surface T_{fs} via the relation:

$$T_f = (1-\alpha)\cdot T_{f_c} + \alpha \cdot T_{f_s}, \qquad (1)$$

where α is taken equal to 0.7.

4. Discussions and Conclusions

Figs. 6 to 12 show the transient evolution of the six parameters selected for the analyses: power, Doppler temperature, average coolant outlet temperature, enthalpy rise, fuel centerline temperature and reactivities. It can be seen, that in the EOC scenario a first power peak occurs, followed by a relatively slow power increase, until it reaches a plateau value around 160 times the initial power. In this case the averaged Doppler temperature plateau is 600.56 K. In EOC the power increase is very fast; this fact is coherent with the central rod worth and the β-effective values. Otherwise, in the BOC scenario the power increases slowly reaching a plateau level around 320 times the initial power, higher than the EOC case; in this scenario the Doppler temperature is around 634.07 K, higher than the EOC case. In both cases, the

Doppler feedback is the mechanism which drives the evolution of the parameters. In general, for this real best estimate case, in HZP conditions and for the two scenarios BOC and EOC, the maximum power reached is very lower than the scram trip level. In the same way, the maximum Doppler temperatures are lower than the specifications limit. For Trillo NPP, these analyses show that this event is irrelevant concerning the nuclear safety.

Furthermore, comparing the figures we can see that in the BOC case, the feedback moderator reactivity effect is neglectable in comparison with the Doppler reactivity, otherwise in the EOC case, both feedback reactivities play the same role during the transient.

Figure 6: Power evolution during the transient at BOC.

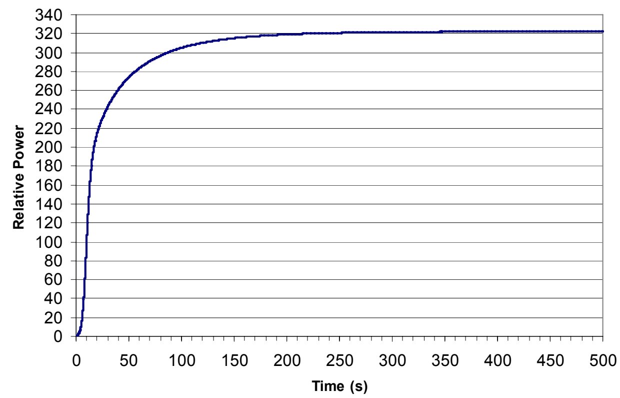


Figure 7: Power evolution during the transient at EOC.

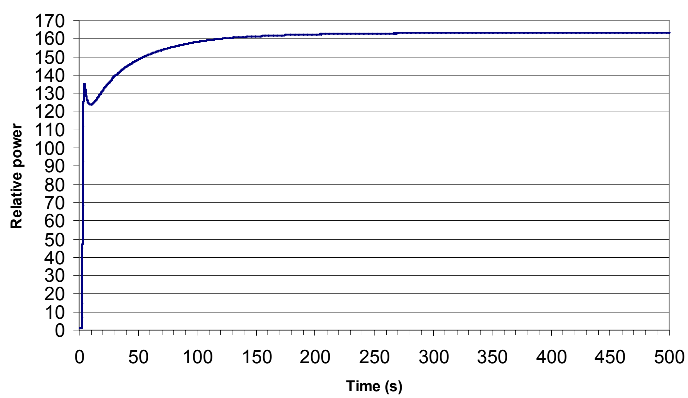


Figure 8: Enthalpy rise.

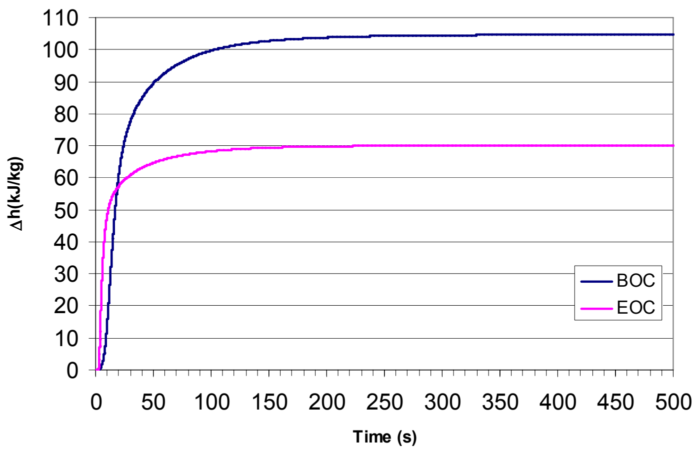


Figure 9: Temperatures at BOC.

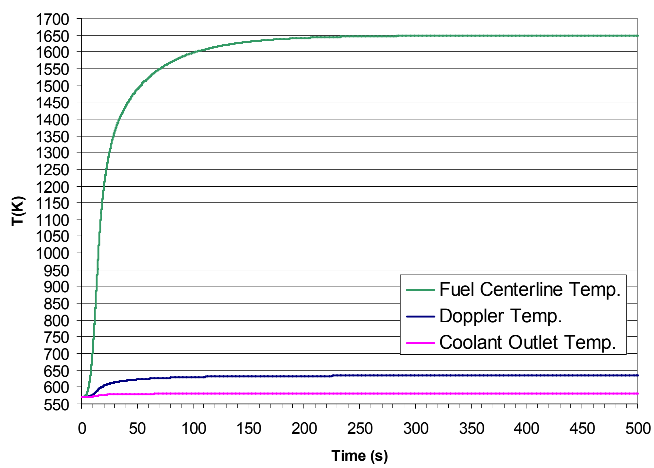


Figure 10: Temperatures at EOC.

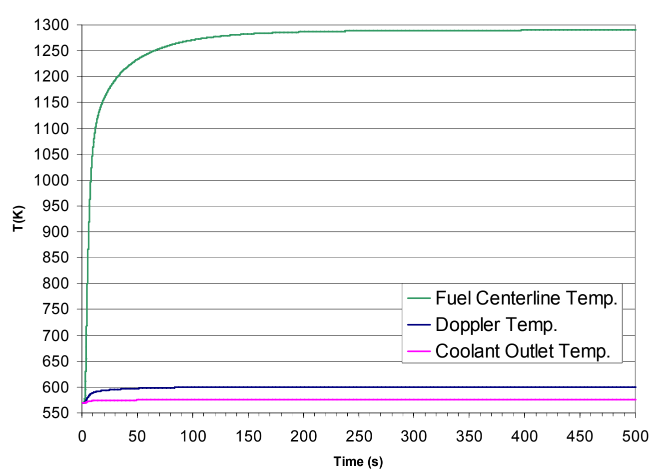


Figure 11: Reactivity at BOC.

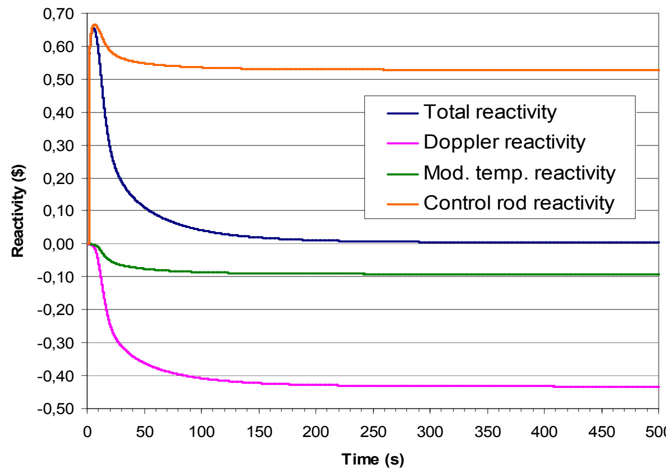
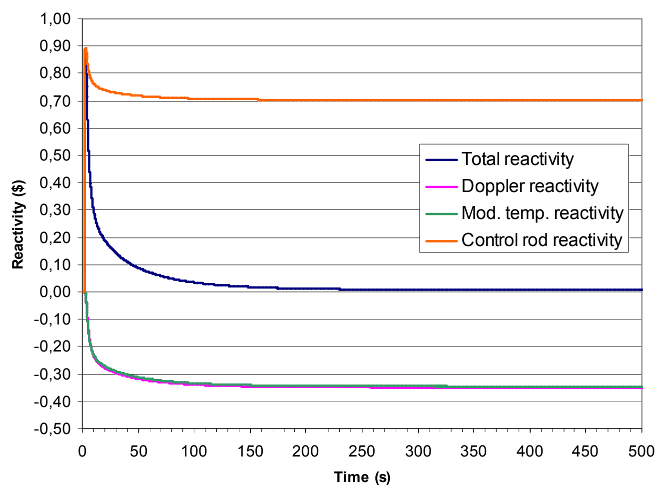


Figure 12: Reactivity at EOC.



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