

BASIS FOR CALCULATING BORON DILUTION SCENARIOS IN PWR BY 3D NEUTRON KINETICS

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Abstract

The origin of the performed study was the analysis of 20 cm² small break LOCA in the lower plenum in a four-loop PWR nuclear reactor by Relap5 code stand-alone (0DNK) in which boron dilution was observed in more than one loop seal. In order to have a more precise result of the boron dilution NK feedback effect, the original nodalization was refined axially in the core area to couple with PARCS v.2.7 code (3DNK). The neutron macroscopic XSec database was generated by the lattice transport code HELIOS.

Before using the new model to predict boron dilution transients, a necessary activity is the qualification of the model (the boron feedback calculated by the Neutronic Cross Sections) against boron changes, so a group of sensitivity calculations injecting more or less borated water in the cold leg were performed either with Relap5 code stand-alone (0DNK) and with Relap5 coupled with PARCS v.2.7 (3DNK) code in order to analyze the reactor power response to the boron injection and the differences using a 0DNK or a coupled 3DNK nodalization.

To complete the study a benchmark calculation was performed considering a 20 cm² break in the lower plenum, in which the reactor trip by control rods has been disabled and boron injection was simulated in the cold leg. This calculation utilized the Relap5 code stand-alone (0DNK) and the Relap5 coupled with PARCS v.2.7 (3DNK) code, in order to see the differences using a 0DNK or a coupled 3DNK model.

Non negligible differences have been found in all cases in the comparison of 0DNK and coupled 3DNK results analyzed, in relation to the core power. These results challenge the evaluation of the uncertainties in case of coupled thermalhydraulic-3DNK calculations. A comprehensive evaluation of the relevant uncertainties of the 3D NK TH coupled calculations is needed.

Introduction

Reactivity accidents can occur originated by internal boron dilution in the primary system (PS) of a nuclear pressurized water reactor type (PWR or VVER) [1]. The problem is caused by boron dilution following vaporization and condensation of the primary system coolant in case of decrease of primary system mass inventory, for example during a small-break loss of coolant accident. This

may include boiling in the core with condensation of steam in the steam generators. When the liquid level in the reactor vessel decreases below the hot leg elevation, steam begins to flow to the steam generators and condenses there. This steam carries no boron and thus boron concentration in the cold leg loop seals begins to decrease. If for some reason this water plug with low boron concentration begins to flow towards the core and enters it without any major mixing with the borated coolant, the result is a positive reactivity insertion. The potential for this scenario is more relevant for the reactor core at beginning of life (BOL), when boron added to the coolant has its maximum effect and therefore any dilution is significant.

When boron dilution analyses are performed several aspects could be distinguished like: formation of the diluted boron plug; transport of the diluted boron plug; mixing of the diluted boron plug; de-boration (for example from a break in primary system) and boration (for example associated with ECCS actuation); reactivity feedback, necessarily associated with a three-dimensional performance of the neutron flux and coolant distribution in the vessel and in the core region. Some of the aspects are mandatory associated with the use of specific type of tools for the analysis, e.g. Computational Fluid Dynamics (CFD) codes to analyse the mixing of the diluted boron plug [2] and 3D Neutron Kinetic (3DNK) codes for the reactivity feedback [1], [3].

The formation and transport of the diluted boron plug, boron dilution, boration and de-boration and reactivity feedback (0DNK and 3DNK) aspects are all associated with the use of system thermalhydraulic tools and thermalhydraulics - NK coupled tools.

The origin of the performed study was a precedent analysis of 20 cm² small break LOCA in the lower plenum in a four-loop PWR nuclear reactor by Relap5 code stand-alone (0DNK) in which boron dilution was observed in more than one loop seal [4].

In order to have a more precise result of the boron dilution neutron kinetic feedback effect, the original Relap5 model nodalization was refined axially in the core area (from 12 to 22 axial nodes) and run with Relap5 (version a.i) coupled to PARCS v.2.7 3DNK code. In this model the boron neutron kinetic feedback is given by the 3DNK code coupled.

The activity, aim of the paper presented, is the necessary qualification of the new coupled 3DNK model (i.e. the boron feedback calculated by the Neutronic Cross Sections) against boron changes. For this purpose sensitivity calculations injecting more or less borated water in the cold leg were performed either with Relap5 (version a.i) code stand-alone (0DNK) and with Relap5 (version a.i) coupled with PARCS v.2.7 (3DNK) code in order to analyze the reactor power (or energy) response to the boron changes and to analyze the differences using a 0DNK or a coupled 3DNK nodalization.

To complete the study, a benchmark transient calculation was performed considering a small break of 20 cm² in the lower plenum, run with Relap5 (version a.i) code stand-alone (0DNK) and with Relap5 (version a.i) coupled with PARCS v.2.7 (3DNK).

The transient selected was not a standard small break LOCA, but a transient in which the SCRAM due to control rods was disabled and a boron injection from the beginning of the transient was simulated in the cold leg in order to observe the power behavior and the differences using a 0DNK or a coupled 3DNK model in a case of boron transient occurring simultaneously with LOCA.

All these activities are also important from the uncertainty point of view in case of coupled thermalhydraulic-3DNK calculations.

1. The Thermalhydraulic and Neutron Kinetic models

1.1 The Thermalhydraulic model for RELAP5 code

The model is based in an original full qualified Relap5/mod3.2.2 gamma input-deck used in past activities performed at the University of Pisa [1], [6], [7], [8]. It represents a four-loop PWR reactor type of 3765 MWth (Figure 1).

In the original model for small break LOCA calculations the number of nodes per each steam generator U-Tube bundle was increased and three parallel U-Tubes per each steam generator (not shown in Figure 1) were modelled because it was found to be of large interest for the understanding of natural circulation phenomena [8], which is also of importance in the analysis of boron dilution events.

Eight downcomer stacks end up in a single stack of nodes into the lower plenum. This implies the prediction of boron full mixing in the lower plenum.

The original input-deck was adapted in the part of the core from 12 axial nodes to the 22 nodes axial sub-division of the neutronic core [5] (Figure 2) in order to keep the consistency with the neutron cross-section (XSec) axial subdivision. The pressure drop coefficients and crossflow junctions were adjusted consistent with this new nodalization. The core is surrounded by a reflector peripheral part including also a top and bottom parts, which is all nodalized.

The heat structures were also adapted in the core region to the new configuration for the stand-alone steady-state calculation. The heat structures include the following zones: peripheral reflector, core central part, corresponding to 113 fuel assemblies (FA) (including the hot bundle in the central part with the “conservative” and the “realistic” representative rods) and four external zones corresponding each one to 20 FA (Figure 3). This subdivision was performed based in the original model in precedent activity [4] and it was developed like that due to the emergency core cooling system disposition in the concerned NPP.

The level of detail of the thermalhydraulic core radial nodalization is coarse (only six TH channels) compared with present capabilities of TH tools. The adopted detail derives from the consideration of the specific configuration of the concerned reactor where hot leg emergency system injection plays an important role connected with 3D phenomena in upper plenum.

The model considers the ECCS which comprises:

- Eight accumulators (ACC), two in each loop, one of them injecting into the cold leg and one of them injecting in the hot leg.
- Four HPIS, injecting into the hot legs.
- Four LPIS, injecting into the cold legs and hot legs.

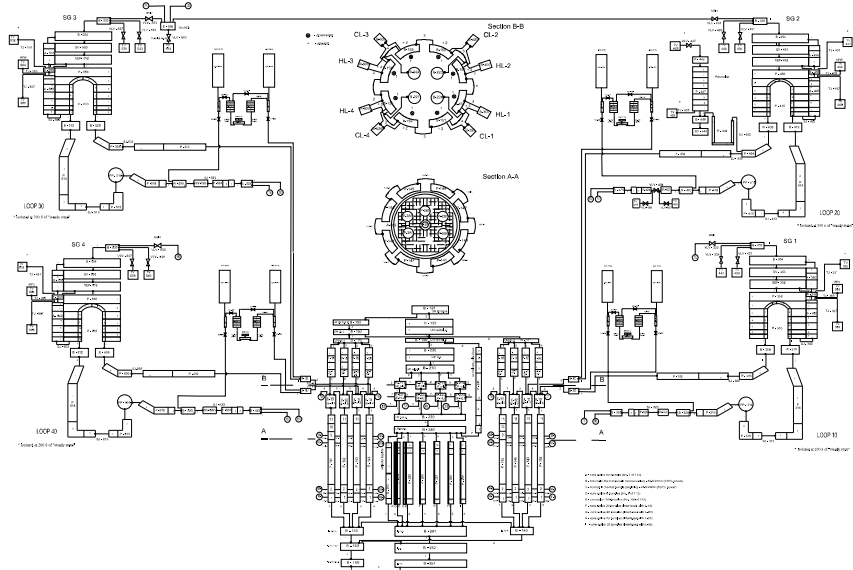


Figure 1 - PWR NPP nodalization overall view.

H accum	OLD CORE	H accum	NEW CORE
4.8270	12	4.8270	22
4.4190	11	4.4190	21
		4.3471	20
4.0290	10	4.1965	19
		4.0208	18
3.6390	9	3.9706	17
3.2490	8	3.7601	16
		3.4494	15
2.8590	7	3.2386	14
		3.0278	13
2.4690	6	2.8170	12
		2.6062	11
2.0790	5	2.3954	10
		2.1846	9
1.6890	4	1.9738	8
		1.7630	7
1.2990	3	1.5522	6
0.9090	2	1.3414	5
		1.0307	4
		0.8202	3
		0.6999	2
0.5190	1	0.5190	1

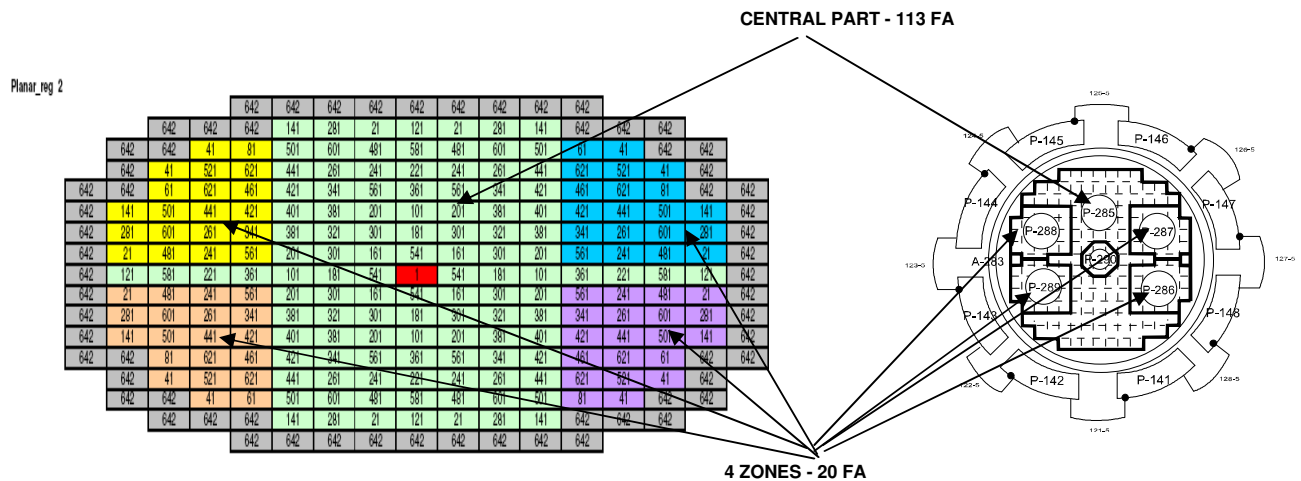
Figure 2 - Original (12 axial nodes) and present (22 axial nodes) axial thermalhydraulic nodalization of the core.

1.2 The Neutronic model for PARCS 3DNK code

The neutronic model for PARCS 3DNK and the neutron macroscopic XSec database were originally prepared for an analysis of LB LOCA in PWR through the same coupled codes [9] and at present the models were adopted for this study.

Radially, the core neutronic model is divided into cartesian nodes, each corresponding to one FA. Each FA is then composed by a 16x16 fuel pin lattice. The core is composed by Uranium-FA and Uranium/Gadolinium-FA types, with an Uranium enrichment up to 4%.

There are a total of 193 FAs (Figure 3) and 61 of them have Control Rod (CR) clusters (i.e. 61 one of them will have rodded and unrodded XSecs). There are 16 CR banks of two different types (Figure 4). The radial reflector is modelled by 64 reflector nodes of the same size as FA nodes. Finally, the neutronic core model utilizes 20 axial nodes in axial direction over the active fuel length plus two reflector nodes (bottom and top reflector).



Input deck for PARCS v.2.7 code was prepared based on the above mentioned configuration, including the 33 FA types compositions for each of the 22 planar regions and the CR bank configuration. The mapping scheme of Figure 3 is linking 5654 neutronic nodes (257 radial nodes by 22 planar regions) with the associated RELAP5 thermalhydraulic and heat structures nodes.

The coupling information is stored in a separate ASCII file (MAPTAB) and it is realized via a PVM protocol. The overall picture of the coupled codes calculation scheme is given in Figure 5.

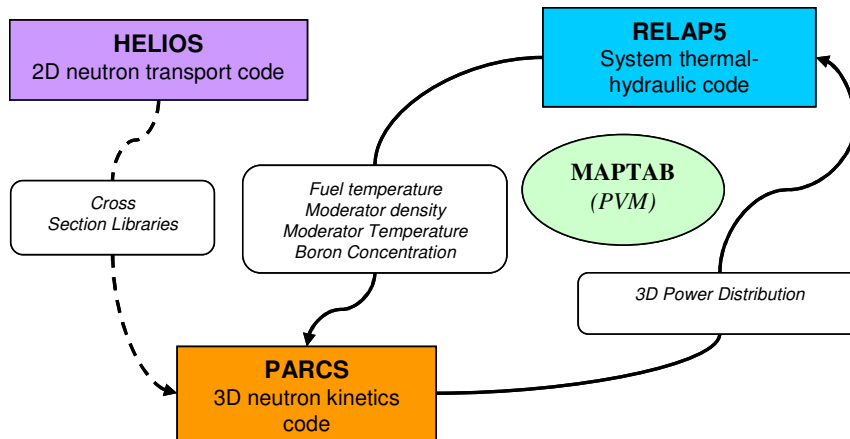


Figure 5 - Coupled codes calculation scheme.

1.3 The Neutron Cross-section database

The neutron macroscopic XSec database were generated by the lattice transport code HELIOS, using RDFMG XSec development methodology. Databases were generated in agreement with a generic PWR core configuration [5] and for a wide range of validity. The calculations presented in this paper refer to a database describing a core at Beginning of Cycle (BOC) condition, with an average burn-up of 18.0 GWd/TonU.

Several HELIOS input decks were developed, modelling different FA types (U and U/Gd FA, rodged and un-rodged) and the reflectors (see Figure 6). An axial core subdivision of 20 layers was used for describing the different burn-up distribution and the partial length of Gd fuel rods. A core 1/4th symmetry was exploited, thus reducing the number of reference FA to 33. Therefore, the neutron cross section database is composed by 663 different compositions or 660 compositions (33x20) for the FA modelling plus 3 compositions for the reflector modelling (radial, top and bottom reflectors).

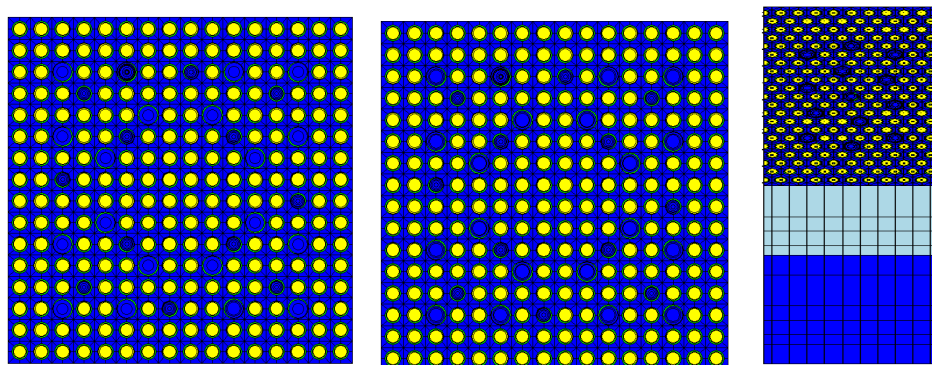


Figure 6 - Unrodded U/Gd FA, Rodded U/Gd FA, and Reflector Modelling.

In order to guarantee a wide range of validity, XSec databases were parameterized according to 5 different independent parameters or Fuel and Moderator temperatures, Moderator density, Boron concentration and Xenon number density. Thus Xsec variation as function of each one of these parameters was calculated, using different number of reference points (see Table 1).

Independent Parameters	Range	Number of Reference Points
T fuel	400 – 1400 – 1800 – 2200 – 2800 K	5
T mod	330 – 450 – 525 – 600 K	4
Rho moderator	10 – 300 – 550 – 650 – 750 – 1000 kg/m ³	6
Cb mod	0 – 2200 ppm	2
Xenon Number density	0. – 1.5E+15 – 3.0E+15 atoms/cm ³	3

Table 1 - Range of independent variables for XSec parameterization.

Therefore, the Xsec database has a 5-dimensional space described by 720 points (5x4x6x2x3). A 5 dimensional linear interpolation routine (LINT5D) performs the automatic interpolation of the XSec database during coupled codes calculations, sending the most suitable value to the PARCS code.

2. Performed Cases

2.1 Sensitivity calculations performed in PWR increasing or decreasing boron in the Reactor Coolant System

All sensitivity calculations were preformed with Relap5 (version a.i) code stand-alone (0DNK) and with Relap5 (version a.i) coupled with PARCS v.2.7 (3DNK) code. After a 300s and 350s period calculation respectively of steady-state in which reactor parameters became stable, a tank injecting water in the cold leg of loop #4 was simulated at different massflows and different boron concentrations.

The steady-state initial boron concentration in the reactor coolant system (RCS) was imposed to 1480 ppm, which is a typical DBA beginning of cycle value in this kind of reactor type. The injection of water with more or less boron concentration from this reference value implies respectively an

increase or decrease (dilution) of boron concentration in the RCS and a decrease (shutdown) or increase (excursion) in the reactor power.

Table 2 summarizes the calculations performed showing boundary and initial conditions (BIC) for cases injecting water with boron concentration higher than 1480 ppm. Table 3 summarizes all the calculations for cases injecting water with boron concentration lower than 1480 ppm.

Case N°	Injecting boron conc. (ppm)	Injection Massflow (kg/s)	CASE ID LABELS	Code
1	1600	500	500-1600-0DNK and 500-1600-3DNK	All calculations were preformed with Relap5 code stand-alone (0DNK) and with Relap5 coupled with PARCS v.2.7 (3DNK) code
2	2000	500	500-2000	“
3	2200	500	500-2200	“
4	1500	1000	1000-1500	“
5	1600	1000	1000-1600	“
6	1700	1000	1000-1700	“
7	1800	1000	1000-1800	“
8	2000	1000	1000-2000	“
9	2200	1000	1000-2200	“
10	1600	2000	2000-1600	“
11	2000	2000	2000-2000	“
12	2200	2000	2000-2200	“
13	1600	5000	5000-1600	“
14	2000	5000	5000-2000	“
15	2200	5000	5000-2200	“

Table 2 - Sensitivity calculations performed in PWR increasing boron concentration in RCS.

Case N°	Injecting boron conc.(ppm)	Injection Massflow (kg/s)	CASE ID LABELS	Code
16	500	500	500-500-0DNK and 500-500-3DNK	All calculations were preformed with Relap5 code stand-alone (0DNK) and with Relap5 coupled with PARCS v.2.7 (3DNK) code
17	1000	500	500-1000	“
18	1300	500	500-1300	“
19	0	1000	1000-0	“
20	500	1000	1000-500	“
21	800	1000	1000-800	“
22	1000	1000	1000-1000	“
23	1200	1000	1000-1200	“

24	1300	1000	1000-1300	“
25	1400	1000	1000-1400	“
26	500	2000	2000-500	“
27	1000	2000	2000-1000	“
28	1300	2000	2000-1300	“
29	1200	5000	5000-1200	“
30	1300	5000	5000-1300	“

Table 3 - Sensitivity calculations performed in PWR decreasing boron concentration in RCS.

2.2 Benchmarking small break LOCA calculations performed in PWR

A benchmark transient calculation was performed considering a small break of 20 cm² in the lower plenum, run with Relap5 (version a.i) code stand-alone (0DNK) and with Relap5 (version a.i) coupled with PARCS v.2.7 (3DNK) code.

As boundary conditions the SCRAM due to control rods was disabled, in order to enhance possible differences 0DNK-3DNK, and a tank injecting water in the cold leg of loop #4 at 1000kg/s and 2000 ppm of boron concentration was simulated in order to observe the power behavior and the differences using a 0DNK or a coupled 3DNK model in a case of boron transient occurring simultaneously with LOCA. The steady-state initial boron concentration in the RCS was imposed to 1480 ppm. Table 4 summarizes the calculations performed showing BIC.

Case ID	BIC	Injecting boron conc.(ppm)	Injection Massflow (kg/s)	Code
SBLOCA-0DNK	8 accumulators available, 2 LPIS and 2 HPIS available in loop #1 and loop#2	2000	1000	Relap5 code stand-alone (0DNK)
SBLOCA-3DNK	8 accumulators available, 2 LPIS and 2 HPIS available in loop #1 and loop#2	2000	1000	Relap5 coupled with PARCS v.2.7 (3DNK)

Table 4 - Benchmarking small break LOCA calculations performed in PWR.

3. Results of the calculations

3.1 Results of sensitivity calculations increasing or decreasing boron in the RCS

The results of the sensitivities are presented firstly comparing the calculations performed with Relap5 code stand-alone (0DNK) with the calculations performed with Relap5 coupled with PARCS v.2.7 (3DNK). The reactor power evolution just after the injection of water more or less borated and its integral, the energy increase/decrease referred to zero initial value (cntrlvar 531) just after the injection are the key parameters that have been selected for the comparison. The energy increase/decrease represents the reactor response due to the boron changes and gives an indication of the qualification of

the model (i.e. the boron feedback calculated by the Neutronic Cross Sections response against boron changes).

Figure 7 shows, as example, these two parameters in the case of 1000kg/s massflow injection in which 0ppm of boron concentration is injected, so water without boron, for both 0DNK and coupled 3DNK calculations (cases 19 in Table 6).

In general in all cases the 0DNK calculations predict a higher power and higher energy release in absolute values in the longer term compared to coupled 3DNK calculations at the same conditions of injection massflow and boron concentration.

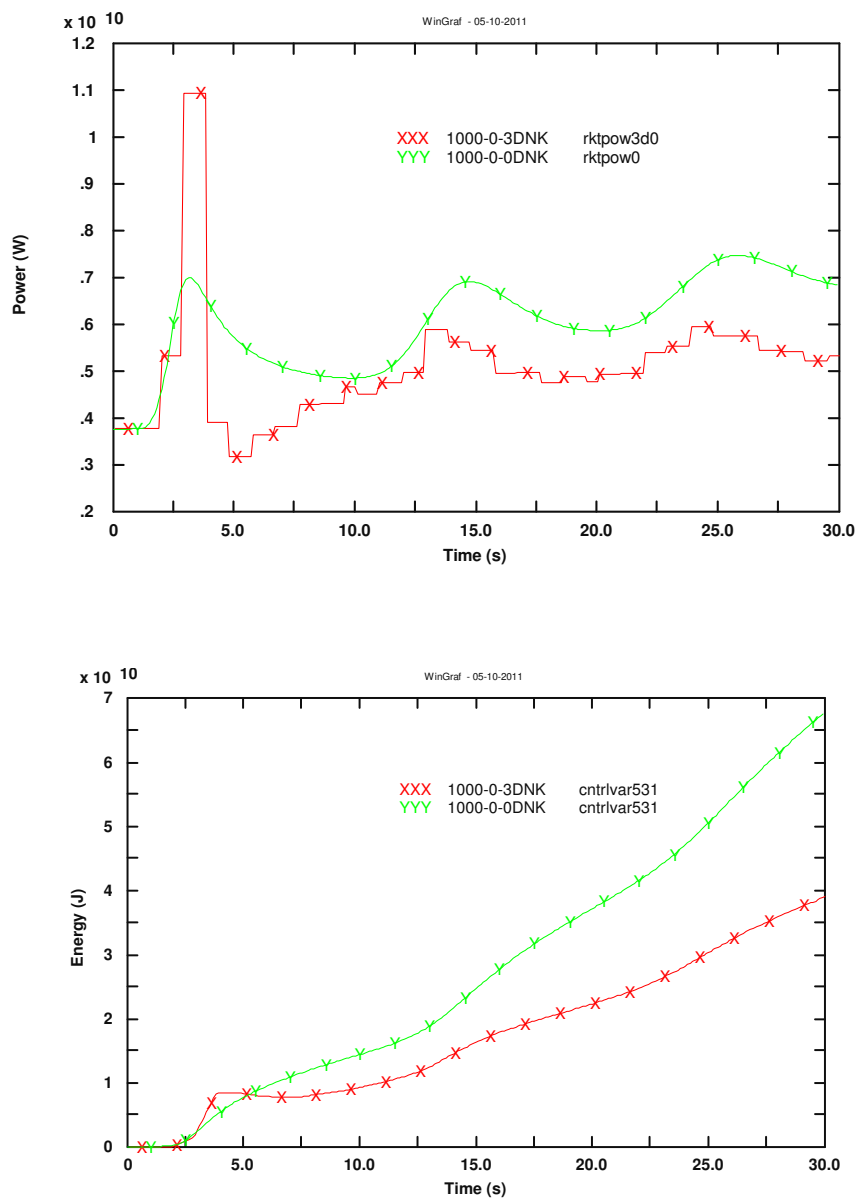


Figure 7 - Reactor power and its integral value (energy, referred to zero origin) versus time in cases 0DNK and coupled 3DNK injecting 0ppm at 1000kg/s.

The 3D radial relative power for coupled 3DNK case is shown in Figure 8 for the first 3.8s of transient showing soft radial changes in the power.

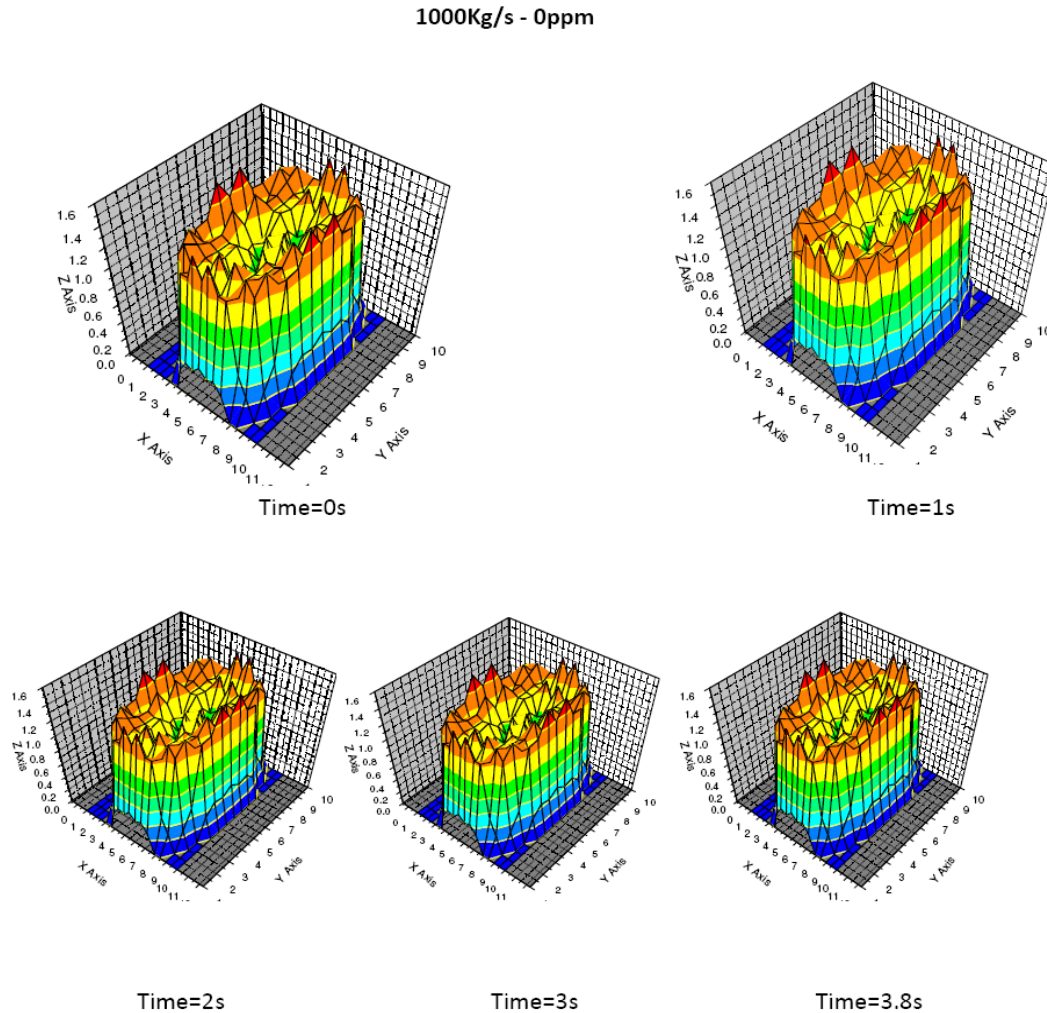


Figure 8 - 3D radial relative power for coupled 3DNK case injecting 0ppm at 1000kg/s.

The axial power profile for coupled 3DNK case in radial position (15,8) is also shown for the first 3.8s of transient and compared with the imposed axial power profile in 0DNK case for the central hot assembly (Figure 9).

The radial position (15,8) corresponds to the FA with highest power during these first seconds of transient. The FA is located in one of the four thermalhydraulic external zones corresponding to 20 FA, the Relap thermalhydraulic channel 287 (see Figure 9 below, the blue top right external zone).

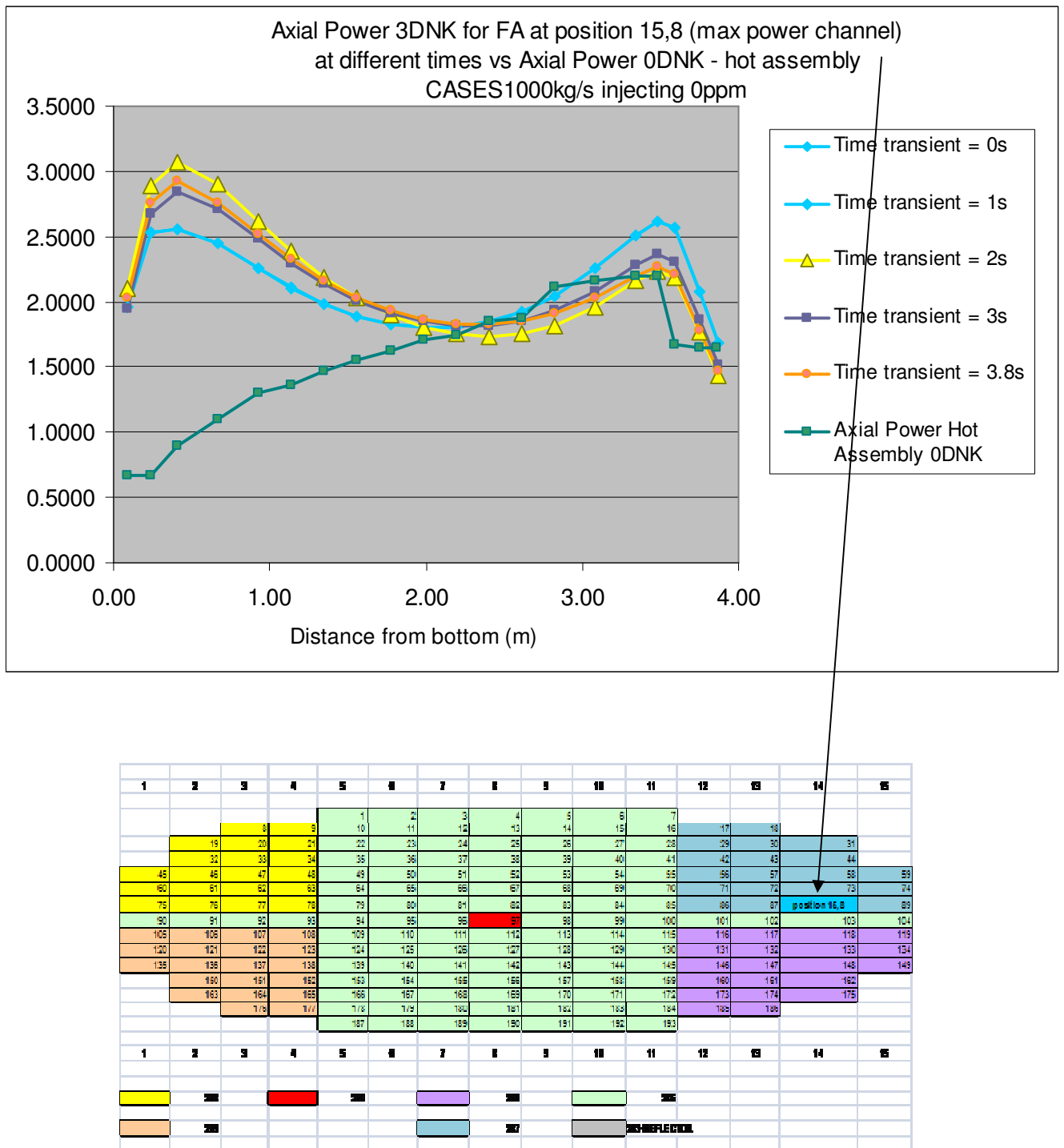


Figure 9 - Axial power profiles in cases 0DNK and coupled 3DNK injecting 0ppm at 1000kg/s.

An increase/decrease of power for coupled 3DNK case in this FA can be observed between 0 and 2s of transient in the lower/upper part of the core, respectively. These changes could be originated by the transient boron distribution inside the core but also due to all other factors that are taken into account in the feedback reactivity coefficients (coolant-moderator void and temperature effects, Doppler...).

Boron concentration in (kg/m³) in the thermalhydraulic channel 287 (where is located the FA) is shown in Figure 10 in this coupled 3DNK case injecting 0ppm at 1000kg/s for the same time intervals till 3.8s at different axial positions. Boron concentration shows a decrease between 0 and 2s in the lower part of the core that could correspond with a higher reactor power, but, as mentioned before, other parameters are influencing core reactivity.

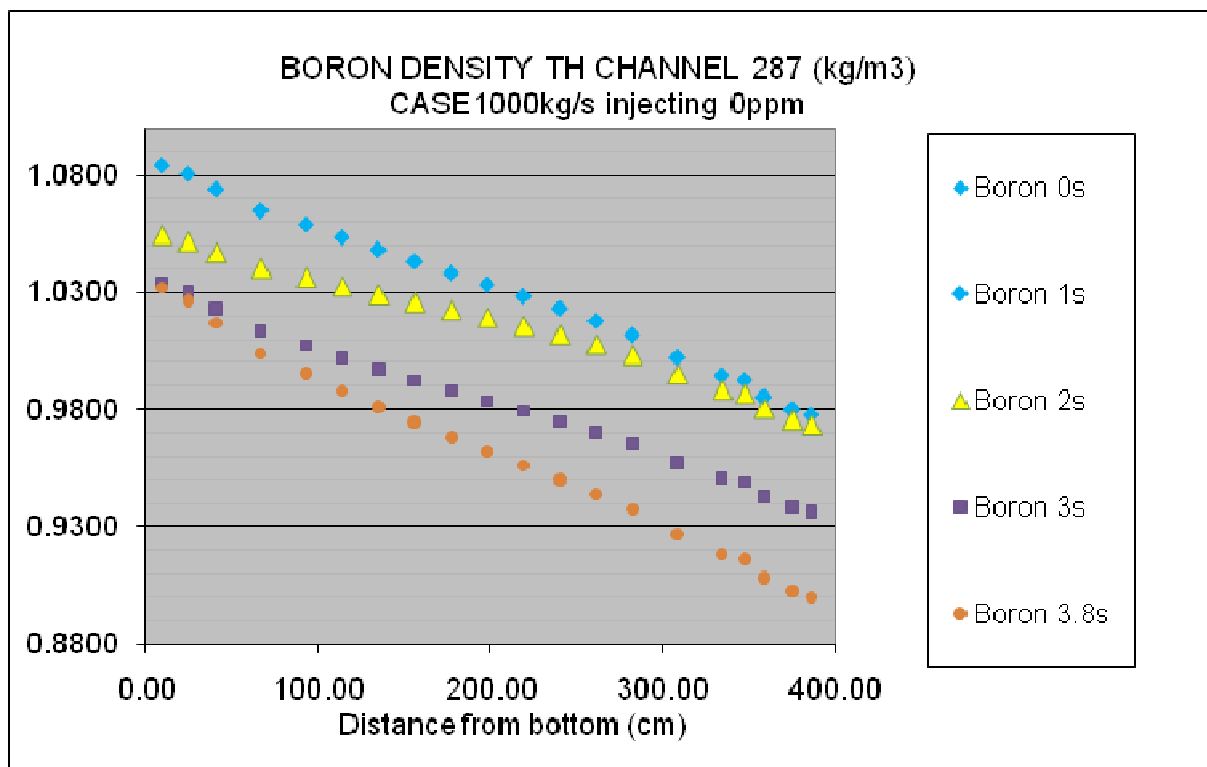


Figure 10 - Boron density (kg/m³) in thermalhydraulic channel 287 at different axial positions for 3DNK coupled case injecting 0ppm at 1000kg/s.

To analyze all the sensitivities results, the energy increase/decrease referred to zero initial value, cntrlvar 531 has been checked at 5s and at 20s of transient and the relative differences from 0DNK to 3DNK coupled cases have been calculated as:

$$\frac{(value \cdot of \cdot energy \cdot at \cdot time \cdot x \cdot in \cdot 3DNK \cdot coupled \cdot case) - (value \cdot of \cdot energy \cdot at \cdot time \cdot x \cdot in \cdot 0DNK \cdot case)}{value \cdot of \cdot energy \cdot at \cdot time \cdot x \cdot in \cdot 0DNK \cdot case}$$

Where x is 5s or 20s.

Table 5 and Table 6 report these values of comparison between 0DNK and 3DNK coupled cases.

Figure 11 shows these values in a chart (Y axis) depending on the relative boron concentration from 1480 ppm (X axis), i.e. the boron concentration of the RCS in steady-state. Figure 12 shows these values in another way; depending on the value of energy at time 5 or 20s in 0DNK cases (cntrlvar 531) divided by the nominal reactor power (3765MW) per time (5 or 20s) (X axis).

From the figures it can be concluded that non negligible differences can be observed when using a 0DNK model or a thermalhydraulic-3DNK coupled model and that the reason should be further investigated with more detailed analysis.

Case N°	Injecting boron conc. (ppm)	Injection Massflow (kg/s)	CASE ID LABEL	Relative difference from 0DNK to 3DNK coupled at 5s	Relative difference from 0DNK to 3DNK coupled at 20s
1	1600	500	500-1600	1.86	0.19
2	2000	500	500-2000	0.37	-0.25
3	2200	500	500-2200	0.29	-0.28
4	1500	1000	1000-1500	-	0.78
5	1600	1000	1000-1600	0.44	-0.26
6	1700	1000	1000-1700	0.24	-0.31
7	1800	1000	1000-1800	0.18	-0.32
8	2000	1000	1000-2000	0.13	-0.33
9	2200	1000	1000-2200	0.10	-0.33
10	1600	2000	2000-1600	-0.09	-0.46
11	2000	2000	2000-2000	0.01	-0.34
12	2200	2000	2000-2200	0.00	-0.31
13	1600	5000	5000-1600	-0.32	-0.54
14	2000	5000	5000-2000	-0.11	-0.27
15	2200	5000	5000-2200	-0.11	-0.21

Table 5 - Results of sensitivity calculations increasing boron concentration in RCS.

Case N°	Injecting boron conc.(ppm)	Injection Massflow (kg/s)	CASE ID LABEL	Relative difference from 0DNK to 3DNK coupled at 5s	Relative difference from 0DNK to 3DNK coupled at 20s
16	500	500	500-500	-0.10	-0.46
17	1000	500	500-1000	-0.24	-0.52
18	1300	500	500-1300	-0.65	-0.71
19	0	1000	1000-0	0.07	-0.40
20	500	1000	1000-500	-0.03	-0.41

21	800	1000	1000-800	-0.04	-0.41
22 ¹	1000	1000	1000-1000	-0.06	-0.42
23	1200	1000	1000-1200	-0.10	-0.43
24	1300	1000	1000-1300	-0.17	-0.46
25	1400	1000	1000-1400	-0.36	-0.53
26	500	2000	2000-500	-0.14	-0.39
27	1000	2000	2000-1000	-0.01	-0.37
28	1300	2000	2000-1300	0.05	-0.33
29	1200	5000	5000-1200	-0.03	-0.28
30	1300	5000	5000-1300	0.08	-0.24

Table 6 - Results of sensitivity calculations decreasing boron concentration in RCS.

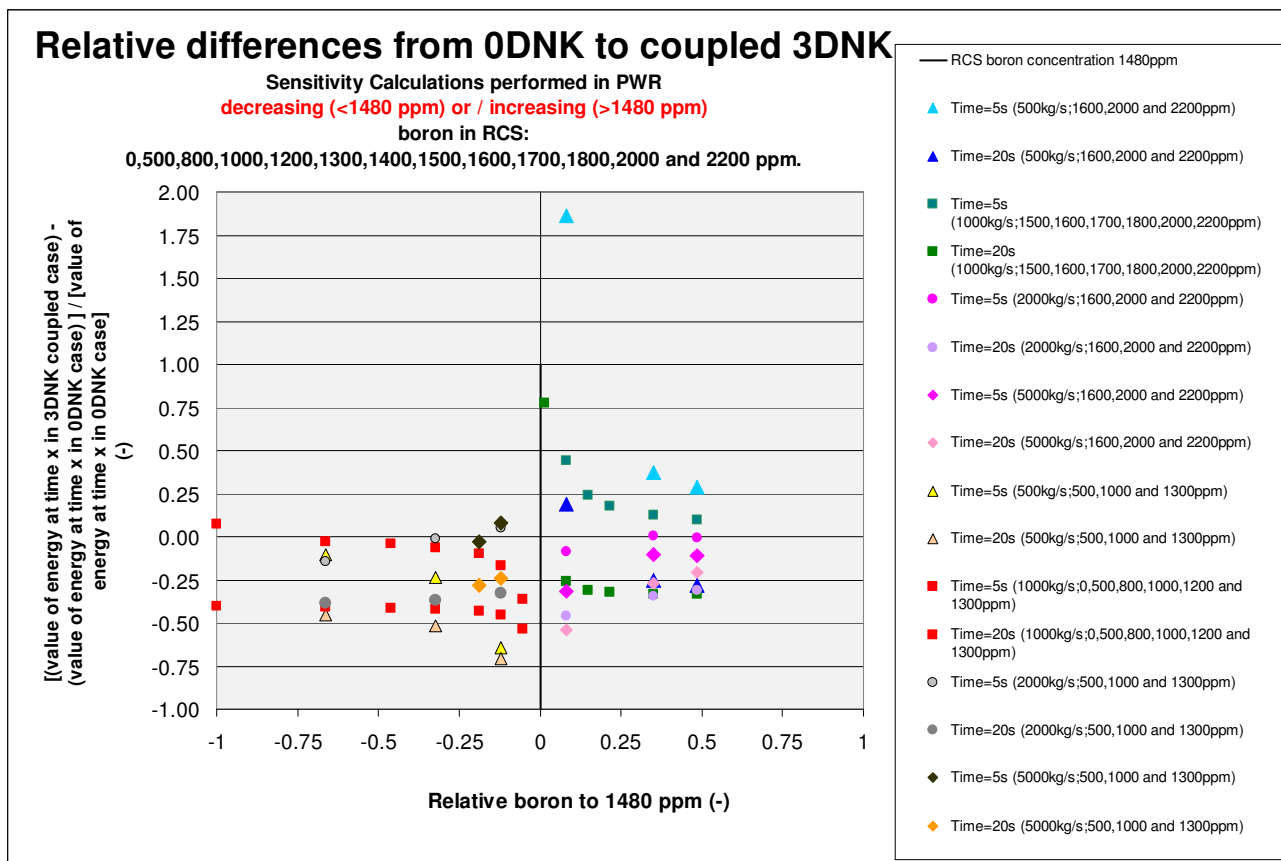


Figure 11 - Relative differences from 0DNK to 3DNK coupled cases in energy increase/decrease at 5s and 20s depending on the relative boron concentration from 1480 ppm.

¹ In this case a variation of about $\pm 4\%$ and $\pm 8\%$ in the values of energy (cntrlvar 531) at 5s and 20s respectively can be expected if the 0DNK doppler coefficients imposed vary of about $\pm 10\%$.

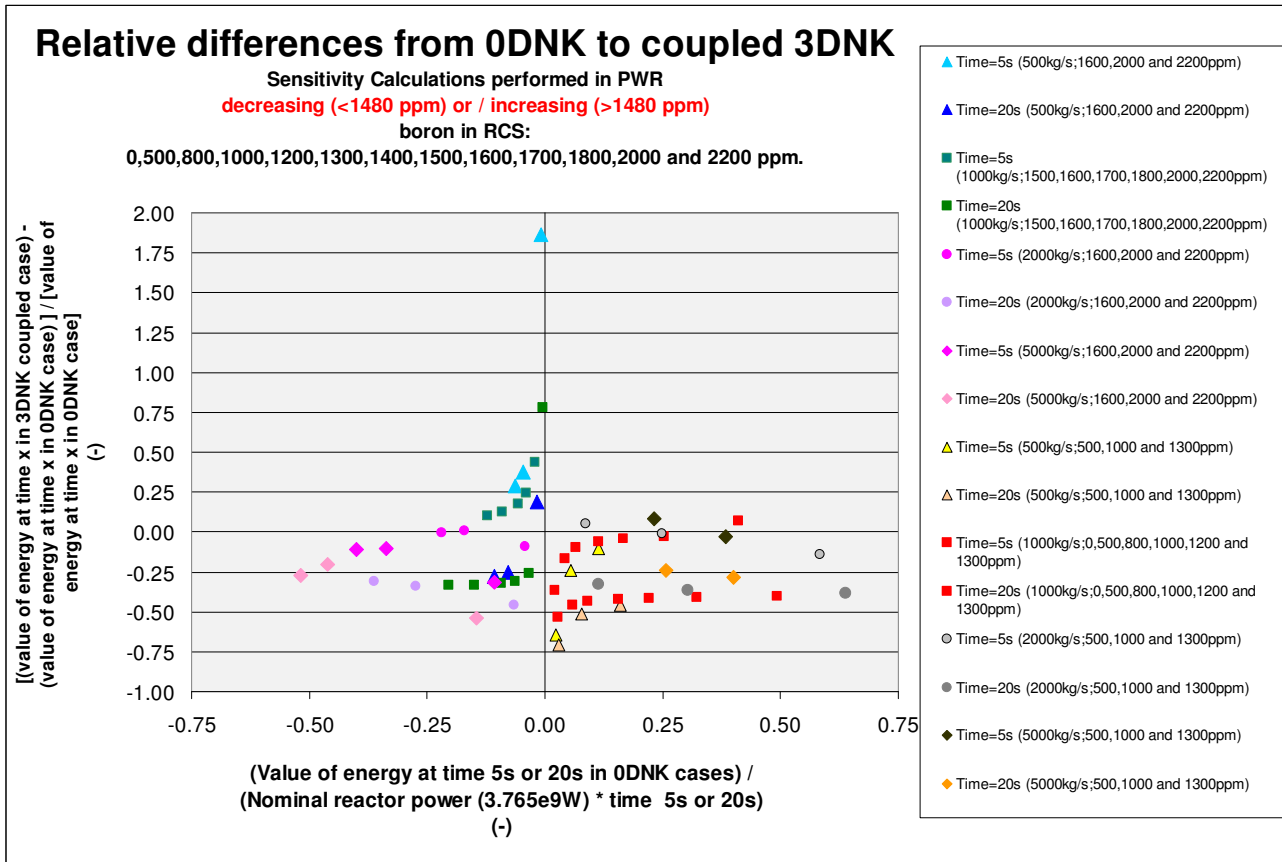


Figure 12 - Relative differences from 0DNK to 3DNK coupled cases in energy increase/decrease at 5s and 20s depending on the value of energy at time 5 or 20s in 0DNK cases / nominal reactor power * time (5 or 20s).

3.2 Results of benchmarking small break LOCA calculations

The benchmark transient calculation of a small break of 20 cm² in the lower plenum was performed to complete the comparison study. The transient selected is not a standard small break LOCA, but a transient in which the SCRAM due to control rods was disabled. Signals related to SCRAM actuation (ECCS actuation, SGs isolation and afterwards cooldown...) were kept.

A tank injecting water in the cold leg of loop #4 at 1000kg/s and 2000 ppm of boron concentration was simulated from the beginning of the transient in order to observe mainly the power behavior in a case of boron transient occurring simultaneously with LOCA when using the Relap5 (version a.i) code stand-alone (0DNK) or Relap5 (version a.i) coupled with PARCS v.2.7 (3DNK). The steady-state initial boron concentration in the RCS was imposed to 1480 ppm.

The calculations, after steady-state period, were run till 60s since by that time different results in both calculations were observed and mainly non negligible differences were observed in the core

power. 8 accumulators were available, and 2 LPIS and 2 HPIS available in loop #1 and loop#2 for both models, however did not actuated during this 60s of transient calculated.

Figure 13 shows primary system pressure behavior different in both cases. Higher pressure is predicted in 0DNK case due to higher reactor power (Figure 14). Good agreement is observed between secondary system pressures.

Figure 15 shows quite similar boron density in the lower plenum referred to zero value at the beginning of the transient in both cases. As mentioned in previous chapter, the influence on the neutron kinetics is due to the transient boron distribution inside the core but also due to all other parameters that are taken into account in the feedback reactivity coefficients.

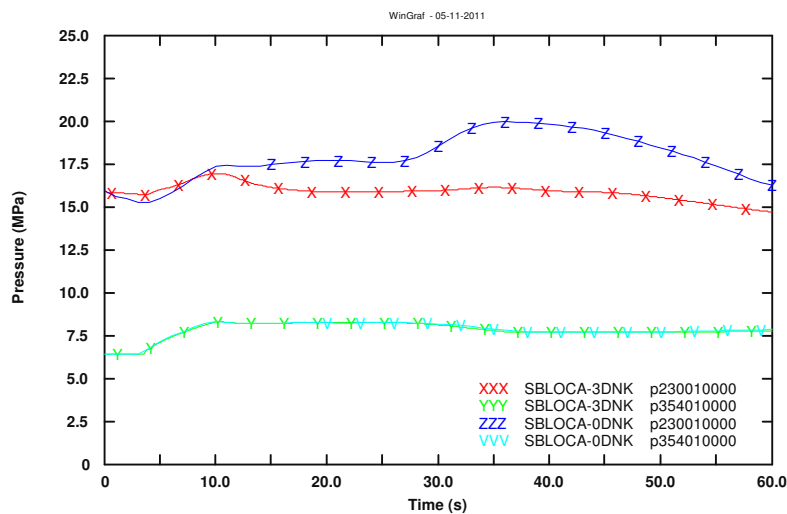


Figure 13 - Primary and Secondary System Pressure in coupled 3DNK and 0DNK cases.

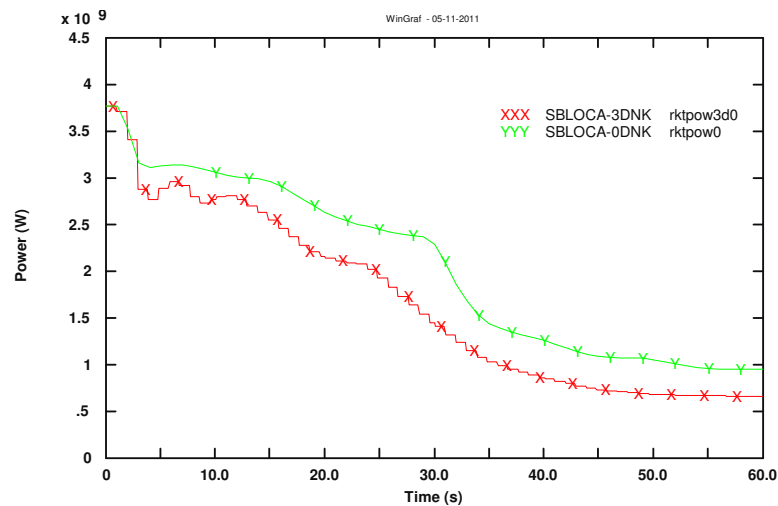


Figure 14 - Reactor power in coupled 3DNK and 0DNK cases.

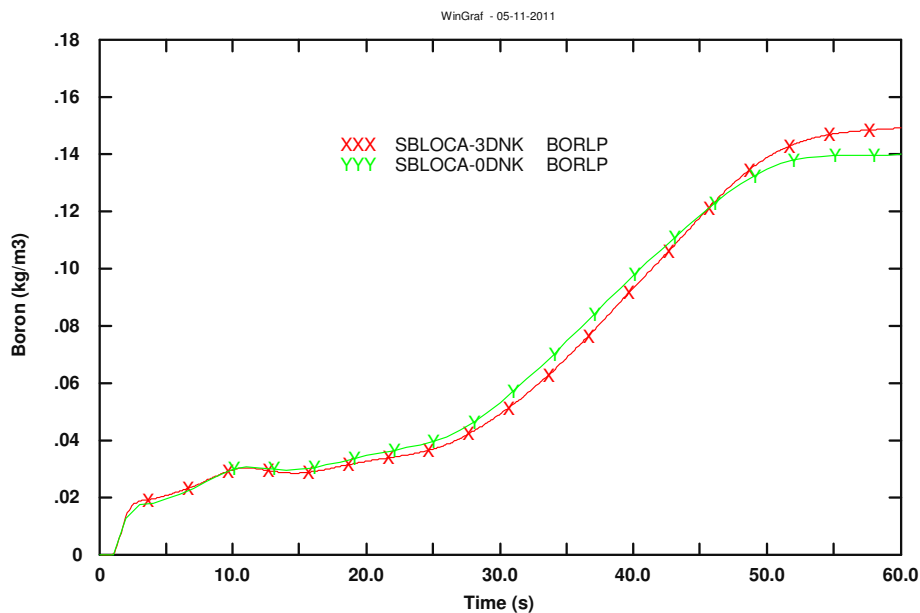


Figure 15 - Boron density in lower plenum in coupled 3DNK and 0DNK cases.

4. Conclusions

The paper presents a qualification activity necessary before using coupled 3DNK models to predict boron dilution transients due to reflux condensation after small break LOCAs.

In order to qualify the coupled 3DNK model (Relap5 coupled with PARCS v.2.7 (3DNK) code) of four-loops PWR against boron changes, sensitivity calculations injecting more or less borated water in the cold leg were performed and compared with the same calculations run with Relap5 code stand-alone (0DNK) in order to analyze the reactor power response to the boron injection and the differences using a 0DNK or a coupled 3DNK nodalization.

Finally a benchmark calculation was performed, considering a 20 cm² break in the lower plenum with Relap5 code stand-alone (0DNK) and with Relap5 coupled with PARCS v.2.7 (3DNK) code, in order to see the differences using a 0DNK or a coupled 3DNK model in case of boron transient occurring simultaneously with LOCA. Boundary conditions considered SCRAM due to control rods disabled and boron injection simulated in the cold leg.

Non negligible differences have been found in the comparison of 0DNK and thermalhydraulic-3DNK coupled results in all cases analyzed in relation to the core power. The reasons should be further investigated with more detailed analysis.

As this work demonstrated, in order to fully exploit the advantages of the 3D NK TH coupled code technology, a comprehensive evaluation of the relevant uncertainties of the calculations is needed. The OECD/NEA UAM (Uncertainty in Analysis and Modeling for LWR) benchmark is an ongoing

project with such scope. In particular, during Phase III, beginning by the next year, coupled code calculations and uncertainty analyses are envisaged [10].

5. References

- [1] F. D'Auria, G.M. Galassi, W. Giannotti, D. Araneo, M. Cherubini, A. Del Nevo, "THE BORON ISSUE IN PWR AND VVER-1000", OECD/NEA/CSNI PKL PROJECT, PKL Analytical Workshop - University of Pisa, Italy, October 11-12, 2005.
- [2] AREVA, FZR, UNIPI, Gidropress, "TACIS PROJECT R2.02/02 - Development of safety analysis capabilities for VVER-1000 transients involving spatial variations of coolant properties (temperature or boron concentration) at core inlet - TASK 4 REPORT", November 2006. Restricted.
- [3] Angelo Lo Nigro, Antonino Spadoni, Francesco D'Auria (Mechanical, Nuclear and Production Engineering Department, University of Pisa), Ana Maria Sanchez Hernandez (Chemical and Nuclear Engineering Department, Polytechnic University of Valencia) "3D Neutron Kinetics to Address the Boron Issue and Analysis of B&W PWR Scenarios", PKL Analytical Workshop, Pisa, October 12, 2005.
- [4] Patricia Pla, Regina Galetti, Francesco D'Auria, Carlo Parisi, Walter Giannotti, Alessandro Del Nevo, Nikolaus Muellner, Marco Cherubini, Giorgio Galassi, Francesc Reventós, "Addressing Boron Dilution Scenario Through RELAP5/3.3 Analysis of PWR SB LOCA" - Proceedings of the 17th International Conference on Nuclear Engineering, ICONE17, Brussels, Belgium, July 12-16, 2009.
- [5] C. Parisi, "PWR Core Neutronic Data and Modelling" University of Pisa Report, 2008. Restricted.
- [6] F. D'Auria, G. M. Galassi, "Best Estimate Analysis and Uncertainty evaluation of Angra-2 plant LBLOCA DBA", University of Pisa Report, DIMNP - NT 433(01), Pisa (I)-rev.1, July 2001, CNEN Contract TERMO No.012-2001, PROCESSO No.1661/2000. Restricted.
- [7] F. D'Auria, G. M. Galassi, "Best-Estimate Analysis of Angra-2 plant ATWS Event Category", University of Pisa Report, DIMNP - NT 526(04), Pisa (I)-rev.4, July 2004, CNEN Contract TERMO No.-2001, PROCESSO No.0548/2002. Restricted.
- [8] F. D'Auria, G. M. Galassi, "DESIGN OF PKL BORON-DILUTION TRANSIENTS", OECD/NEA - SETH PROJECT 2004, DIMNP NT 529(04), University of Pisa, Italy, July 2004.
- [9] Patricia Pla, Carlo Parisi, Francesco D'Auria, Giorgio Galassi, Regina Galetti, Kostadin Ivanov, "Analysis of LB LOCA in PWR through TH - 3D NK Coupled Code" - Thirteen International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-13), Kanazawa, Japan. September 27 - October 2, 2009.
- [10] <http://www.oecd-nea.org/science/egrs/ltb/UAM/>.