IMPROVING THE WIMS-DIREN-RELAP METHODOLOGY TO MODEL LOCA TRANSIENTS IN A CANDU REACTOR

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Abstract

The nuclear safety analyses require modelling different kinds of transients in the reactor cores. Deterministic safety analysis (frequently referred to as accident analysis) is an important tool for confirming the adequacy and efficiency of the safety systems to cope with transients in accident conditions. As far, the most demanding application is the large loss of coolant accident (LOCA), because of the relatively fast kinetics and the spatial effects associated with flux tilts and shut-off rod or liquid absorber insertion. The goal of this paper is to present improvements brought to the WIMS-DIREN-RELAP methodology in order to take into account for thermal hydraulic (TH) parameters and neutronic properties variation during a Rupture of Inlet Header (RIH) LOCA in a generic CANDU reactor core. Since 2010, several studies performed in RATEN ICN have been accomplished by coupling WIMS, DIREN and RELAP programs in order to perform neutronic and thermal hydraulic coupled calculations. A method based on the use of interpolated nuclear cross sections was proposed, but in spite of initial fair results, after intensive tests some gaps were identified. The paper discusses the modifications in the cross section table generation algorithm with respect to the TH parameters variation. The duration of the transient modelling was also extended long enough after the Shut Down System no. 1 (SDS1) rods insertion to take into account for Emergency Core Cooling (ECCS) intervention. The paper results consist in: relative neutronic power, channel and bundle powers, fuel and clad temperatures variations with respect to the time.

Keywords: *CANDU, WIMS, DIREN, RELAP, Coupling, LOCA*

Introduction

Romania owns two 700 MWe output CANDU 6 units in operation at Cernavoda Nuclear Power Plant (NPP). The contribution of Unit 1 and 2 to the total national electricity amount is about 20%, both nuclear units being usually operated at full capacity. On the other side, the operational experience along with many studies performed in the last 50 years shows that CANDU reactors have been being operated in good safety conditions, worldwide [1-3]. No serious incidents or accidents have been reported by now. Nevertheless, performing and improving safety analyses is a current task for all CANDU reactors owners.

RATEN ICN Pitesti has a specific department in charge of performing safety analyses for research reactors, actual operating CANDU type reactors, and innovative reactors. Since 2010, several studies performed in RATEN ICN have been carried out by coupling the computer programs WIMS [4], DIREN [5-7] and RELAP [8], respectively, in order to perform neutronic and thermal hydraulic (TH) coupled calculations for CANDU reactors.

A method based on the use of interpolated nuclear cross sections was proposed in [9], but in spite of a few initial fair results, after intensive tests some gaps have recently been identified.

The goal of the paper is to present improvements brought to the WIMS-DIREN-RELAP methodology in order to cover the identified gaps. Modifications in the cross section table generation algorithm with respect to the TH parameters variation are presented and applied to model 15% and 30% Rupture Inlet Header (RIH) LOCA transients in a generic CANDU reactor core. The duration of the transient modelling was also extended long enough after the Shut Down System no. 1 (SDS1) rods insertion to take into account for Emergency Core Cooling (ECCS) intervention. The paper analyses the evolution of the main neutronic and TH parameters with respect to the time.

Methodology Outlines

In normal operation conditions of a CANDU reactor, the TH parameters vary in a straight domain and there is no significant influence on the neutronic properties of the elementary fuel lattice. On the contrary, in Design Basis Accident (DBA) conditions when large amounts of coolant are supposed to escape from Primary Heat Transport System (PHTS), TH parameters register large variations, which may affect the lattice cross sections, accordingly. As a result, the cross sections have to be recalculated with respect to the significantly larger domain of TH parameters' variation.

The TH parameters considered in an early approach were fuel temperature (T_f), coolant density (d_c) and coolant temperature (T_c), and a local parameter method was developed [10, 11]. There were two approaches targeting to consider explicitly the neutronic properties of every fuel bundle in the reactor core. The first one supposes performing a lattice calculation for every fuel bundle at each time step of the considered transient, thus taking into account for the heterogeneity of the CANDU core. This method is very costly in terms of computing time, but the results obtained are fair. The second method supposes using of interpolated values for neutronic properties, previously generated in cross section tables at a finite and convenient number of points, for the above mentioned TH parameters. This latest method will be presented in the following. The intervals in which TH parameters variation is supposed to take place are as follows:

- fuel temperature: 600 to 1300 ^oC,
- coolant density: 0.2 to 0.9 g/cm^3,
- coolant temperature: 260 to 333 ^oC.

For each of the above mentioned intervals, a number of seven equidistant points have been chosen, including the end ones, as shown in Table 1.

Table 1. Points combinations considered for the neutronic properties generation

Point No.	Fuel temperature (^oC)	Coolant density (g/cm^3)	Coolant temperature (^oC)
1	300	0.200	260.0
2	466.6	0.317	272.2
3	633.3	0.433	284.3
4	800.0	0.550	296.5
5	966.6	0.667	308.7
6	1133.3	0.783	320.8
7	1300.0	0.900	333.0

Thus, a total number of 7x7x7=343 sets of lattice neutronic properties were generated by using the WIMS code computations and assembled into a so called *cross section table*. The considered neutronic properties were collapsed into two group energy schema in order to be used by the DIREN – a 3D finite differences diffusion program [7]. A set corresponding to one TH parameters' combination (T_f, d_c, T_c), also depicts neutronic properties with respect to the burnup, being well known that in the CANDU core there are bundles with very different burnups, from fresh fuel values to the discharge burnup.

In the early approach [9] a procedure using a Dynamic Link Library (DLL) was used to run WIMS inside of the DIREN program, this conducting sometimes, and rather randomly, to wrong results, due probably to an incorrect management of the Random Access Memory (RAM). This anomaly was eliminated by adopting a ″classical″ approach, namely: a Visual Basic procedure which calls and waits for WIMS runs, one by one, for every of the before mentioned 343 sets, saving both input and output files for subsequent verifications. In terms of computation time, this latest procedure completion is comparable to that of the original one (DLL based), except that the file storage amount is considerably larger, in exchange for a better results' accuracy. An example of a correct set of neutronic properties is presented in the ″Results″ section along with an incorrect one.

Modelling a LOCA transient in the reactor core asks for solving the time-dependent multigroup diffusion equation. Since the finite differences method is not appropriate to achieve this goal, the so-called improved quasi-static multigroup approximation (IQMA) was developed, as shown in [13-15]. IQMA method was also implemented in the DIREN program [7] and used to model the considered LOCA transient. The technique used in IQMA for solving the time dependent equation is to break the neutron flux into a so-called amplitude function, A(t) - the same over all energy groups (no space dependent), and a space-dependent, slowly time-dependent group spatial form function, Ψ(r,t), as below:

$$\Phi_g(r,t) = A(t) * \Psi_g(r,t) \qquad (1)$$

where $\Phi_g(r,t)$ is the neutron flux in the energy group *g*, *r* is the spatial coordinate and *t* is time.

The amplitude A(t) gives the behaviour of reactor thermal power pulse and also the maximum channel and bundle powers - the most important neutronic safety parameters to be pursued.

The transient duration is divided into a convenient number N of intervals, inside of which is assumed that the TH parameters don't vary, and only at the ends of an interval i (i=1, N) changes in the TH values are considered. According to the chosen N interval, N+1 time moments are defined.

The idea of coupling neutronic and TH calculations is that DIREN and RELAP programs update and change data in the following manner: DIREN calculates the bundle power distribution over the reactor core, submit it to RELAP which, in turn, calculates the actual TH parameters values (coolant density and temperature, and fuel temperature), submits them to DIREN for the next time step calculation of the bundle power distribution, and so on, until all the N intervals are covered. The advantage of such a procedure consists in using of more realistic TH values, obtained by running a TH specialized program (RELAP).

The ″Results″ section presents the main neutronic and TH safety parameters considered in this study, with respect to the time, during 15% and 30% RIH2 LOCA transients, and also a comparison to similar results available in the literature.

Results and discussions

Table 2 presents the main neutronic lattice properties collapsed on two group energy schema with respect to the burnup, for two situations: the actual generation procedure developed in this paper (″correct set″) and the old DLL based procedure (″incorrect set″). The difference, in percent, between the two sets of neutronic properties is given in Table 3.

Table 2. Examples of correct and incorrect sets of neutronic properties

	Correct set			Incorrect set		
Neutronic Property	**No burnup**	**Mid burnup** (3.7 MWd/kgU)	**Exit burnup** (7.4MWd/kgU)	**No burnup**	**Mid burnup** (3.7 MWd/kgU)	**Exit burnup** (7.4MWd/kgU)
STR1 (cm^{-1})	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.57E+00	1.57E+00
SA1 (cm^{-1})	1.64E-03	1.70E-03	1.77E-03	1.64E-03	1.16E-04	1.16E-04
NUSF1 (cm^{-1})	9.32E-04	9.12E-04	8.84E-04	9.32E-04	5.81E-08	5.76E-08
S11 (cm^{-1})	2.31E-01	2.32E-01	2.32E-01	2.31E-01	2.14E-01	2.31E-01
S12 (cm^{-1})	8.27E-03	8.20E-03	8.13E-03	8.27E-03	8.79E-03	8.27E-03
STR2 (cm^{-1})	9.27E-01	9.24E-01	9.23E-01	9.27E-01	9.69E-01	9.69E-01
SA2 (cm^{-1})	3.62E-03	4.04E-03	4.13E-03	3.62E-03	2.56E-04	2.56E-04
NUSF2 (cm^{-1})	4.53E-03	4.80E-03	4.64E-03	4.53E-03	1.74E-06	1.73E-06
S21 (cm^{-1})	7.68E-05	7.76E-05	7.86E-05	7.68E-05	8.75E-06	8.75E-06
S22 (cm^{-1})	3.66E-01	3.67E-01	3.67E-01	3.66E-01	3.53E-01	3.53E-01
H1 (kWs/cm)	1.14E+00	1.09E+00	1.05E+00	1.14E+00	6.72E-05	6.66E-05
H2 (kWs/cm)	6.03E+00	6.04E+00	5.66E+00	6.03E+00	2.04E-03	2.02E-03

The significance of neutronic properties is given below:

STR1, STR2 = transport cross section in rapid (1) and thermal (2) group,
SA1, SA2 = absorption cross section in rapid (1) and thermal (2) group,
NUSF1, NUSF2 = production cross section in rapid (1) and thermal (2) group,
S11, S12, S21, S22 = scattering cross sections,
H1, H2 = rapid (1) and thermal (2) power to flux ratios

Table 3 presents differences between the actual and wrong sets, in percent.

Table 3. Percentage differences between the correct and incorrect sets of neutronic properties (%)

Neutronic Property	**No burnup**	**Mid burnup** 3.7 MWd/kgU	**Exit burnup** 7.4MWd/kg
STR1	0.00	-7.68	-7.69
SA1	0.00	1.3×10^3	1.4×10^3
NUSF1	0.00	1.6×10^6	1.5×10^6
S11	0.00	8.5	0.18
S12	0.00	-6.73	-1.61
STR2	0.00	-4.72	-4.80
SA2	0.00	1.5×10^3	1.5×10^3
NUSF2	0.00	2.7×10^5	2.7×10^5
S21	0.00	787.1	798.3
S22	0.00	4.08	4.14
H1	0.00	1.6×10^6	1.5×10^6
H2	0.00	2.9×10^5	2.8×10^5

It can be easily seen that in the case of no burnup, the old method (DLL based) supplies the same results as the new one. Differences appear as the burnup increases and they become strongly large, especially for production cross sections and power to flux ratios, when percent deviations rise to 10^5-10^6, as Table 3 reveals.

The core calculations also showed that the incorrect cross sections' set leads to results with no physical meaning, the convergence on flux solution being impossible. Instead, the actual cross section table generated using the independent procedure elaborated in the paper supplies fair amplitude values for considered 15% and 30% RIH transients, as presented in Fig. 1.

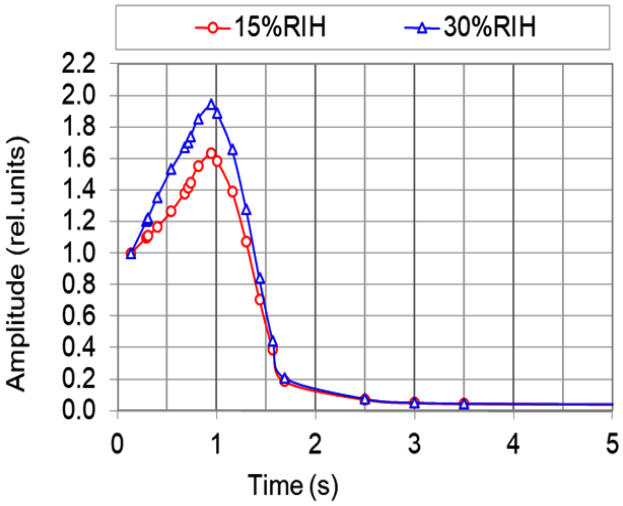


Fig. 1. Flux amplitude supplied by the incorrect cross sections set

In Fig. 2 a comparison between 15% RIH flux amplitude values obtained through different methods are presented, namely:

a) the present paper evaluation through DIREN-RELAP coupled calculations;

b) an older Atomic Energy of Canada Limited (AECL) evaluation using Reactor Fuelling Simulation Program (RFSP) in 1977, [15];

c) another coupled neutronic and thermal hydraulic calculation performed and published in 2008, when ICN Pitesti had an agreement to use the CATHENA (Canadian Algorithm for Thermal hydraulic Network Analysis) program, [16].

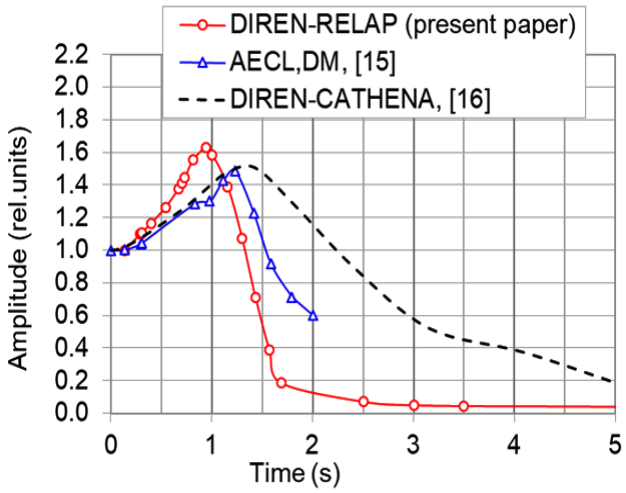


Fig. 2. Comparison between flux amplitudes in the case of 15% RIH transients

Fig. 2 reveals that, in spite of an existing delay in power pulse rising between the actual evaluation (a) and the previous ones (b and c), the three maximum amplitude values obtained are close, the percentage difference being under 8.6%. This observation gives us the confidence to consider our evaluation as being fair.

The channel and bundle power tilts obtained using DIREN-RELAP coupled calculations are illustrated in Figs. 3 and 4, respectively, their evolution tracking rigorously the flux amplitude one.

It can be observed that in about less than 2 seconds the fission power is drastically decreased by the SDS1 intervention. The graphs are limited to 5 seconds because nothing relevant can be seen beyond this period.

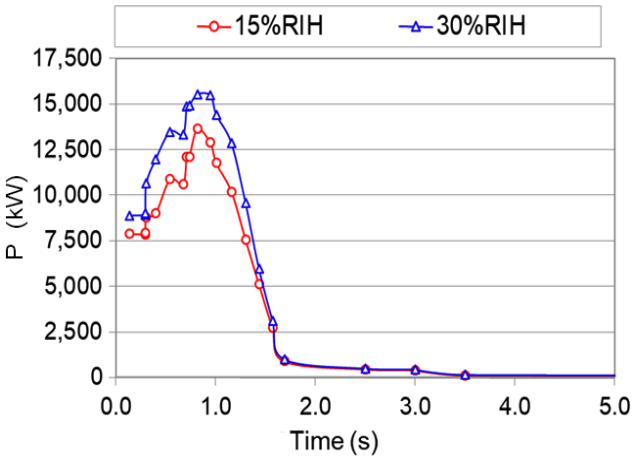


Fig. 3. Channel power evolution

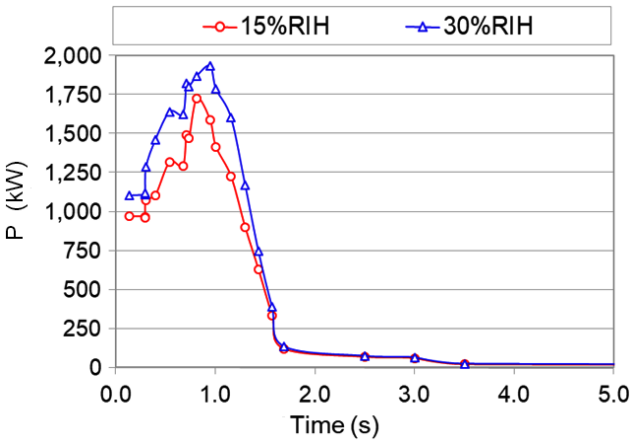


Fig. 4. Bundle power evolution

Another improvement brought to WIMS-DIREN-RELAP methodology was to extend modelled transient duration from 5 seconds in [9] to 70 seconds, in order to catch the ECCS intervention, which has also been modelled.

In the next figures, TH parameters such as mass flow, pressure, clad and fuel temperature, respectively, are represented. A prerequisite condition to start a transient in a CANDU core is to have the core in a fair equilibrium steady state, proven by TH parameters stabilization after enough time. The core behaviour was simulated in steady state conditions for 3000 seconds, and the main TH parameters are shown in Figs. 5 to 8.

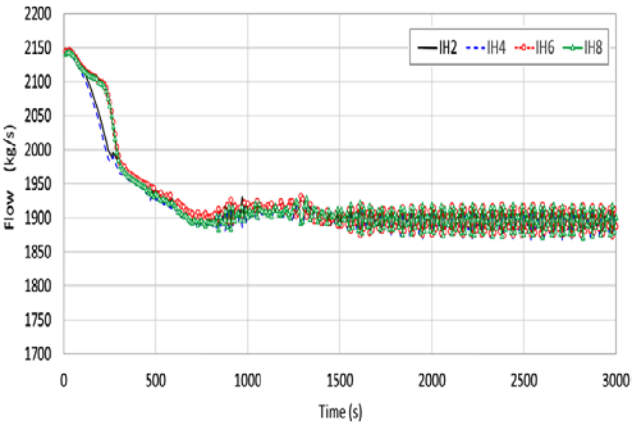


Fig. 5. Inlet Header (IH) mass flow in steady state conditions

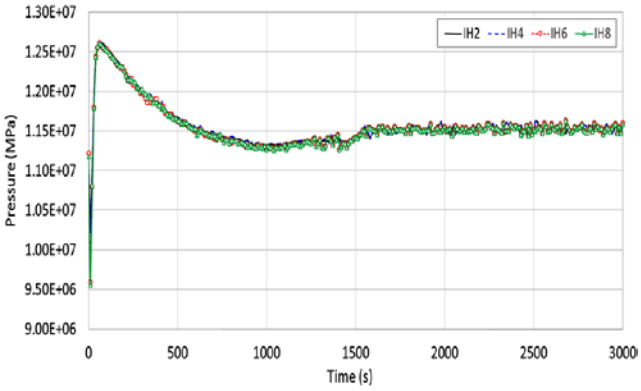


Fig. 6. Inlet Header (IH) pressure in steady state conditions

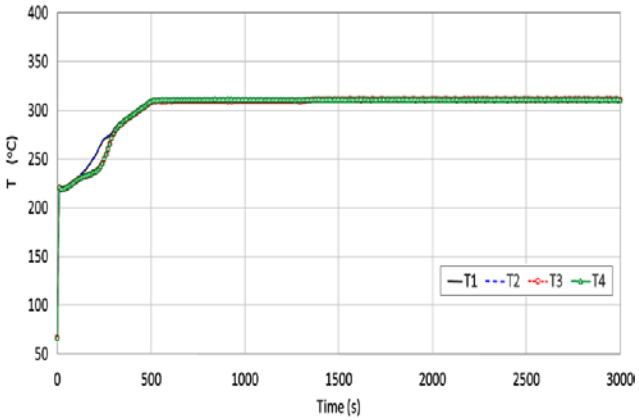


Fig. 7. Sheath temperature in steady state conditions

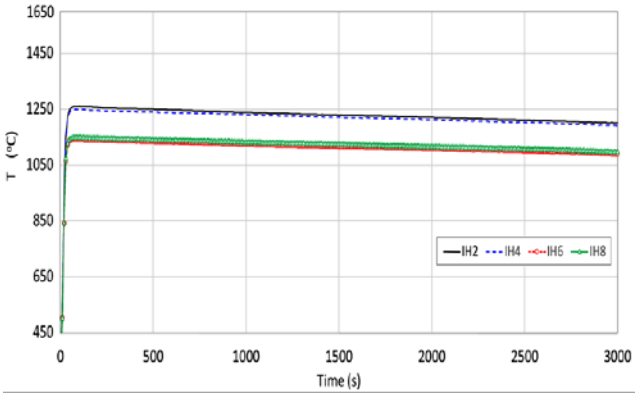


Fig. 8. Fuel temperature in steady state conditions

In Figs. 5-8, IH2, IH4, IH6 and IH8 represent the four inlet headers of the reactor while T1-T4 denote the four passes through the reactor core. A good stabilization of the main TH parameters can be seen in the before mentioned figures, and as result, the proposed transients' simulation can be started.

The TH results corresponding to the 15% LOCA RIH transient are illustrated in Fig. 9-15, given below.

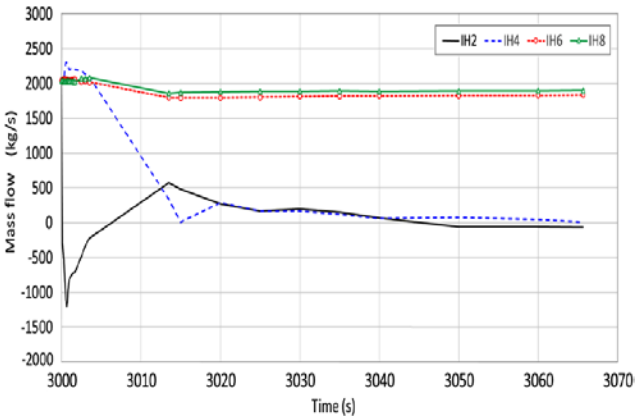


Fig. 9. Inlet header mass flow

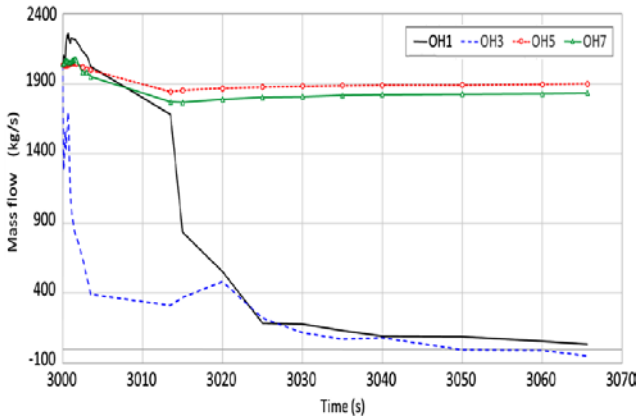


Fig. 10. Outlet header mass flow

Figs 9 and 10 reveal that in the headers of the intact loop (IH6, IH8, OH5 and OH7) the mass flow remains almost unchanged, due to a safety feature of CANDU 6 to isolate the intact loop from broken loop in case of LOCA.

In the broken loop IH2, the reverse flow starts immediately (see black line in Fig. 9) and it holds for about 5 seconds. After that, the IH2and IH4 flows tend to stagnate together for the remaining transient duration. This phenomenon of flow stagnation is also present in the OH1 and OH3 outlet headers, but it starts 20 seconds later, as it can be seen in Fig. 10.

In Figs. 11 and 12 the inlet and outlet header pressures evolution is represented with respect to the time. The pressure behaviour is similar to that of the flow, the loss of flow leading to a rapid depressurization in the broken loop (see the continuous black and blue dotted lines in Figs. 11 and 12). After 30 seconds, the pressure stabilizes to about 2 MPa. Only a slightly decrease of the pressure can be seen in the intact loop, from 11.6 to 9.55 MPa and from 10.13 to 9.0 MPa (see IH6 line in Fig. 11 and OH5 line in Fig. 12, respectively).

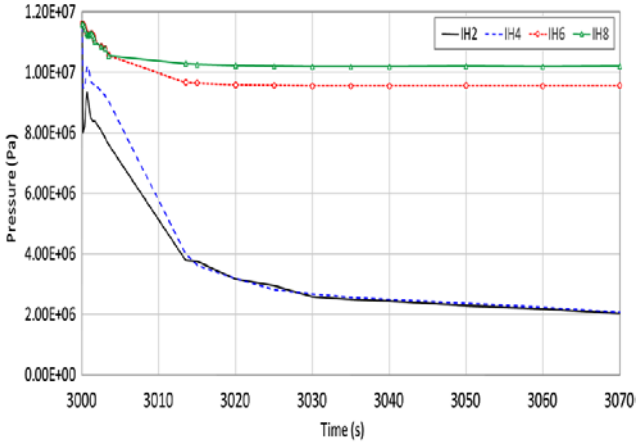


Fig. 11. Inlet header mass flow

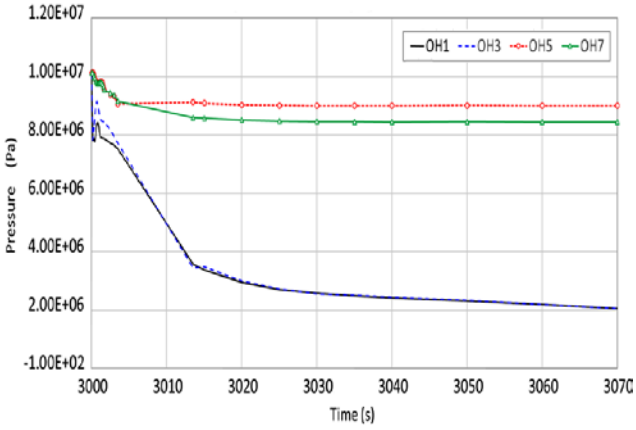


Fig. 12. Outlet header mass flow

As for the 30%RIH transient the TH parameters evolution is similar to that of the 15%RIH one, and only significant safety parameters were represented further on.

In Figs. 13 and 14, the sheath and fuel temperatures in the broken loop are illustrated during the considered 15% and 30% RIH transients.

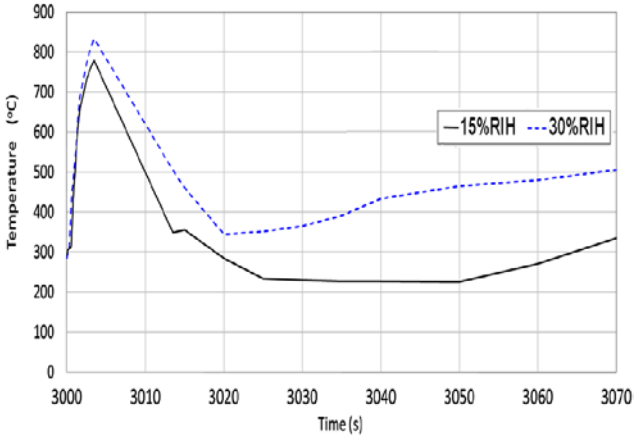


Fig. 13. Sheath temperatures in the broken loop for 15% and 30% RIH transients

Maximum values of 779°C and 833°C have been noticed for the sheath temperature in the case of 15% and 30 RIH transients, respectively (see Fig.13).

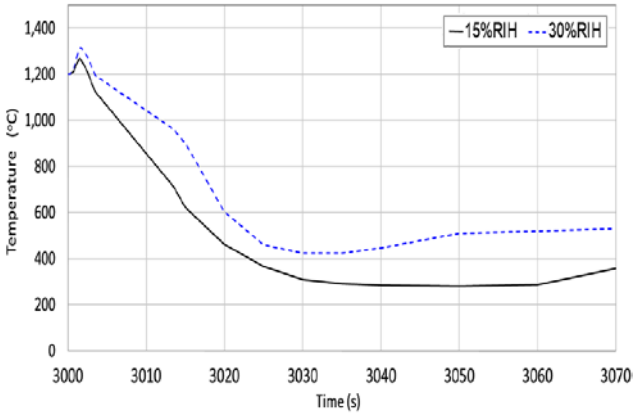


Fig. 14. Fuel temperatures in the broken loop for 15% and 30% RIH transients

The peak values appear after 3.5 seconds from the initiating event start. The fuel heats more rapidly than the sheath does. After only 1.68 seconds from the transient start, the fuel centre temperatures reach their maximum values of 1267°C and 1315°C for the two transients, respectively, and then they decrease rapidly for about 30 seconds. After this moment, a values' stabilization for 20 seconds can be seen, followed by a very slight tendency to further increase, this tendency being more evident for the 30%RIH transient.

Taking into account that the uranium dioxide fuel melts at 2800°C, we conclude that the fuel failure is not expected to happen during the analysed RIH LOCA transients.

On the other side, a comparison with other evaluations [17, 18], reproduced in Figs. 15 and 16, shows that our results are encouraging, the sheath and fuel temperature shapes and values being similar, in spite of existing different times to reach the maximum values.

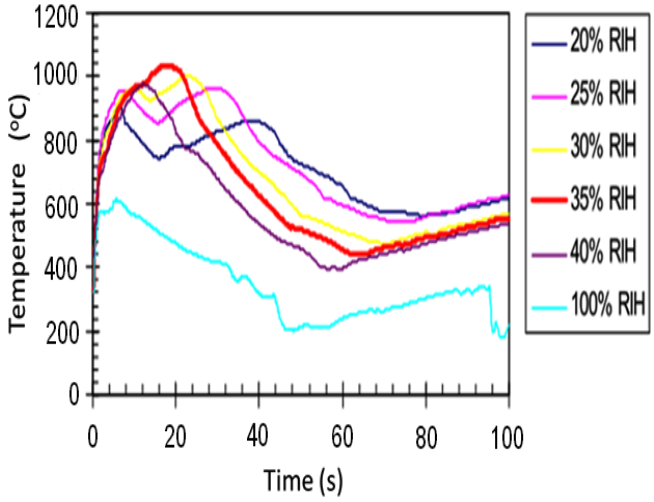


Fig. 15. Sheath temperature in some LOCA scenarios, CATHENA calculations [17]

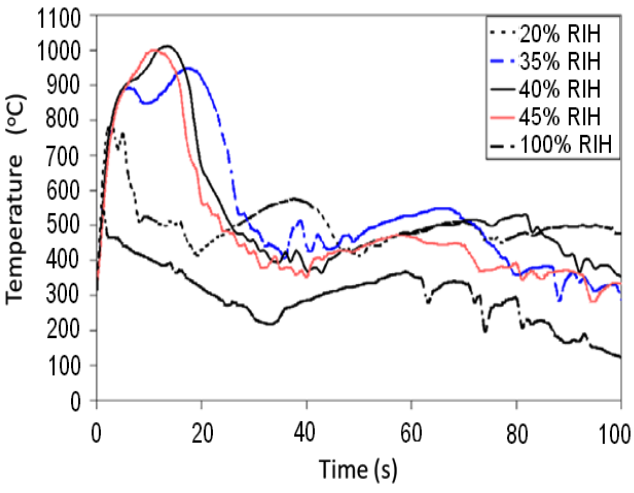


Fig. 16. Sheath temperature in some LOCA scenarios, RELAP calculations [18]

Conclusions

An improved method using interpolated neutronic cross sections was elaborated in order to achieve lattice neutronic properties necessary to model LOCA transients supposed to happen in a CANDU type reactor.

The cross sections table generated for the expected variation domain of thermal hydraulic parameters led to plausible neutronic results, the flux amplitude, the channel and bundle powers being comparable to those reported in previous studies.

Despite that the thermal hydraulic parameters of interest for nuclear safety (sheath and fuel temperatures) reached their maximum values more rapidly than it was reported in similar studies, they showed similar behaviour and comparable peak values.

The main benefit of this work consists in offering a rapid and reliable procedure to model usual LOCA transients in CANDU reactors.

The future work will be focused on improvements of the accuracy of the safety parameters' tilts prediction and a finer discretization of the transients' duration.

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