PRESSURE TUBE OXIDATION HEAT ADDITION IN AN ACCIDENT IN A CANDU TYPE NUCLEAR POWER PLANT

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În această lucrare se prezintă studiul evoluției parametrilor în cazul unui accident de pierdere a agentului de răcire pe un canal cald cu combustibil caracteristic centralelor PHWR. Analiza este făcută în scopul evidențierii aportului de căldură al oxidării structurilor adiacente fascicolelor de combustibil în momentul atingerii temperaturilor de oxidare a aliajelor de zirconiu din care acestea sunt fabricate, aport suplimentar față de cel al oxidării creioanelor de combustibil din zona activă. Modelarea oxidării acestor structuri cu codul SCDAP/RELAP5 este posibilă de la introducerea modelului de incintă de curgere (shroud) ce ia în considerare oxidarea.

This paper presents the evolution of parameters in the case of a loss of coolant accident in a hot channel loaded with typical PHWR fuel. The analysis is aimed to underline the additional heat generated by the oxidation of structures adjacent to the fuel bundles at the high temperature of the zirconium alloys in the core, heat that is added to the one generated by the oxidation of the cladding material in the core. Modeling the oxidation of these structures with the SCDAP/RELAP5 code is made possible by the shroud model that considers their oxidation.

Keywords: pressure tube, heat, oxidation, hydrogen, flow shroud

Abbreviations:
PHWR – **P**ressurized **H**eavy **W**ater **R**eactor
CANDU – **Can**adian **D**euterium **U**ranium
AECL – **A**tomic **E**nergy of **C**anada **L**imited
LOCA – **Lo**ss of **C**oolant **A**ccident

1. Introduction

RELAP5 is a system code destined to the analysis of transients in light water reactors, initially developed by the U.S. Nuclear Regulatory Commission, at the Idaho National Laboratory, aimed for technical support in nuclear regulations, licensing analysis, operating manuals evaluation and as a basis for nuclear power

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plant analysis. The hydrodynamic model of RELAP5 is a transient mono-dimensional one, with two fluids in a biphasic mixture of water and steam that may contain non-condensable components in the gaseous phase and a soluble component in the liquid phase.

The SCDAP/RELAP5 code was developed for best-estimate modeling of light water cooled reactor transients during severe accidents. The code models the reactor cooling system behavior coupled with the core behavior and with the release of fission products release during a severe accident. This is the result of the unification of the RELAP5 code used in thermal-hydraulic analysis with the SCDAP code used in the core behavior modeling during a severe accident. The result is a flexible tool due to his general approach on modeling that allows the simulation of specific systems and is used in the study of a large margin of transients for different power plants, research reactors and experimental facilities.

The code can be adapted to CANDU specific systems due to the heavy water properties library and the capacity to model flow thru horizontal channels present in a CANDU core.

A large variety of studies has been made by Romanian researchers on CANDU accidents using SCDAP/RELAP5, starting from RELAP5 capabilities to study CANDU type reactors [1]. Thermal-hydraulic analysis have been made including reactor inlet and outlet headers [2][3]. The results of these studies are compared with the results of the CATHENA code [4] used for design basis accident studies by the CANDU manufacturer AECL, as well as a feeder positioning analysis in the case of a header brake [5].

The task of modifing the SCDAP models for PHWR severe accidens simulation is a continuous one. The final aim is to create a SCDAP/RELAP5 version that is capable to analize complete accident sequences for this tipe of reactor.

The existing codes that study severe accident in CANDU are: MAAP4-CANDU[6], a modular analysis code, and ISAAC, a integrated CANDU accident analysis code [7]. These codes approach for the phenomenology within or outside the reactor core is a simple one in order to sustain rapid sequencing calculations for probabilistic safety analysis studies and the core degradation is treated on design criteria.

SCDAP/RELAP5 is more detailed and complex due to the dedication to the inner core sequences of accidents. The fuel rod components of SCDAP that define the fuel bundles undergo degradations as cylindrical elements, while MAAP4-CANDU and ISAAC represent the bundle as concentric circle geometry.

The final aim of the program development is the creation of a code that recreates the main phenomenon needed in a accurate and coherent prediction of transients and that, in the same time, is simple and cheap enough to be purchased for parametric or sensitivity studies.

2. Shroud component in SCDAP/RELAP5

The most important accident management measure to terminate a severe accident transient in a Light Water Reactor (LWR) is the injection of water to cool the uncovered degraded core. In the PHWR-CANDU type reactors, the emergency core cooling system injection, in the case of a late detection of an emptied channel has the same effects. Analysis of the TMI-2 accident [8] and results of various integral in-pile and out-of-pile experiments (CORA [9], LOFT [10], PHEBUS [11], PBF [12]) have shown that before the water succeeds in cooling the fuel pins there could be an enhanced oxidation of the zircaloy cladding and other core components that in turn causes a sharp increase in temperature, hydrogen production and fission product release.

The QUENCH program at Forschungszentrum Karlsruhe (FZK) investigates hydrogen generation, material behavior, and bundle degradation during reflood. Integral bundle experiments are supported by separate-effects tests and code analyses. The program is providing experimental and analytical data for the development of quench and quench-related models and for the validation of code systems.

Following this experimental program the SCROUD component was introduced in the SCDAP/RELAP5 model family, permitting the modeling of the oxidation of the structure adjacent to the fuel bundles in the experiments above mentioned.

A particular aspect of the PHWR reactor core is the division of the coolant flow into channels. This division brings additional structures on the typical PWR/BWR cores, which are subjected to oxidation during significant temperature rise in a severe accident that implies the loss of coolant, due to the extra metal at high temperature available in the accident sequence. The introduction of the SHROUD component in SCDAP/RELAP5 [13] allows the modeling of these structures by the view point of their oxidation. The SHROUD [14] component can model the pressure tube and the Calandria tube in a Loss of Coolant Accident (LOCA), analyzing the added oxidation heat brought by these structures in addition to that of the fuel rods.

3. Metal-water reaction model

Among the mathematical models used by SCDAP/RELAP5, a special place is for the oxidation of the metallic structures and fuel rods in the core, mainly because of his use in the modeling of CANDU type core structures [15].

The reaction of zirconium and steam is treated using the correlation developed by Cathcart [16]. The metal-water reaction model is coupled with the fuel rod deformation model so that if a rod ruptures, the inside of the cladding can react. The metal-water reaction heat source term for the cladding surface mesh point is added into the total heat source term for the heat structure

The chemical equation being modeled is:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + 5.94 \ x \ 10^8 \ J/(kg \ mole) \quad (1)$$

The oxide layer thickness on the cladding's outer surface at time-point n is given by:

$$dr_n = [dr_{n-1}^2 + (K\Delta t)\exp(-\frac{A}{RT})]^{\frac{1}{2}} \quad (2)$$

Where

dr_n = oxide thickness at time-point n (m)

dr^2_{n-1} = oxide thickness at time-point n-1 (m)

K = 2.252 x 10-6 m^2/s

Δt = time step size (t_{n+1} - t_n) (s)

A = 35889 mole/cal

R = 1.987 cal/(K-mole)

T = cladding temperature (K)

The amount of heat added to the cladding's outer surface between time-point n and n-1 is given by multiplying the volume of cladding undergoing reaction by the density of zirconium and the reaction heat release:

$$Q = \rho\pi[(r_0 - dr_{n-1})^2 - (r_0 - dr_n)^2]\frac{H}{W} \quad (3)$$

Where

Q = heat addition per unit length (J/m)

ρ= density of zirconium = 6,500 Kg/m^3

ro = cladding outer radius (m)

H = reaction heat release = 5.94 x 108 J/(kg-mole)

W = molecular weight of zirconium = 91.22 kg/(kg-mole)

Similar equations are used for the cladding's inner surface if cladding rupture occurs.

The total hydrogen mass generated by the metal-water reaction is calculated by multiplying the mass of zirconium reacted by the ratio of the molecular weight of 4 hydrogen atoms to 1 zirconium atom.

4. Pressure tube oxidation modeling

We are presenting below a study of the influence of the oxidation in a LOCA accident in two different cases: with and without the oxidation of the pressure tube.

The modeled transient is the blockage of the hydraulic pathway at the entrance of the core that leads to the stepped drop of flow towards zero, in the last seconds of the transient there is a reestablishment of flow in accordance with the late action of an emergency core cooling system.
The transient is created for the purpose of underlining the importance of the new shroud model in the analysis of the systems in the PHWR core.

The hydraulic circuit configuration is shown in Fig. 1 and represents a single fuel channel in which flows a time dependent quantity of coolant that simulates the loss of cooling in the channel, boiling of the coolant and drying of the channel. The nuclear heat as well as the one produced by oxidation (in the case of the pressure tube oxidation) is extracted by the coolant or the remaining steam still present in the channel.

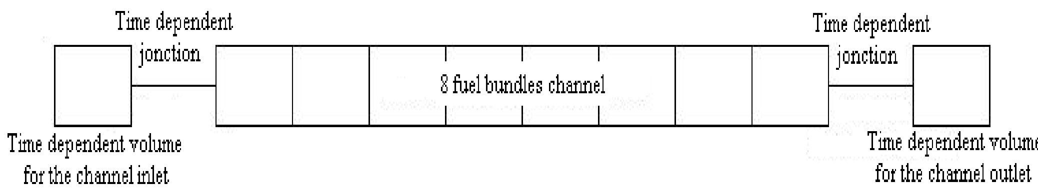

Fig. 1. Horizontal channel representation

For this transient, the flow was varied starting from 1 kg/s initially, dropped suddenly to 0.1 kg/s at the 50th second, then to 0 at the 100th second, and by the end is raised gradually by the 300th second with a 0.0001 kg/s/s (Fig. 2).

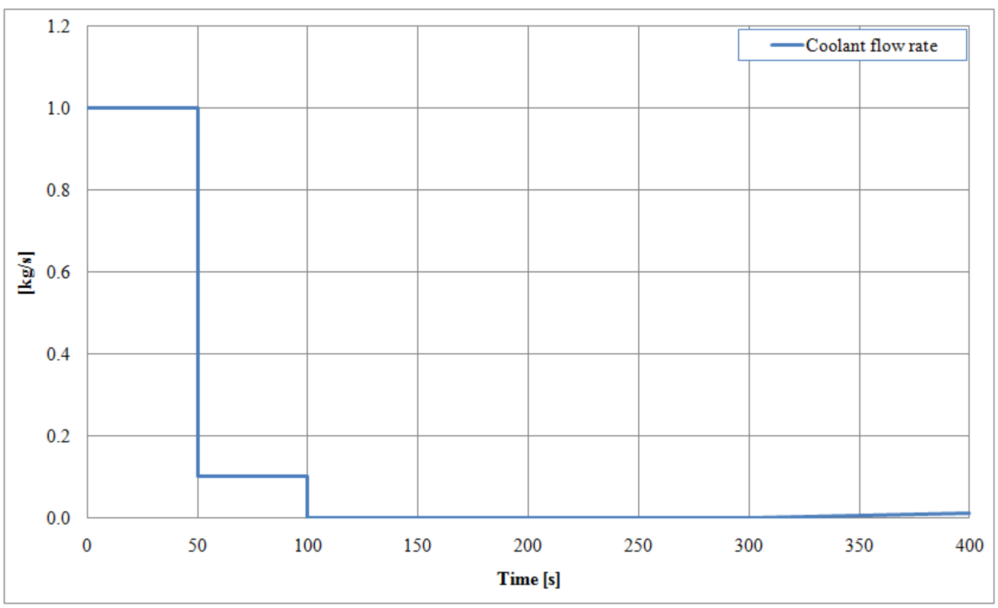

Fig. 2. Flow variation during the transient

5. Transient analysis

In this transient all the heat is extracted by the coolant that flows thru the hydrodynamic volumes, an differences are noted in the cases in which the flow shroud was used, in comparison with the other case, shroud that participates in the oxidation heat generation.

Once the flow is blocked, the coolant boils, the channel is emptied and the fuel bundles are only cooled by the steam still present in the pressure tube. Fig. 3 represents the void variation in the fuel bundle at the entrance in the channel and for the 4th fuel bundle in the middle of the channel. In both cases the void follows the same path overlapping the values.

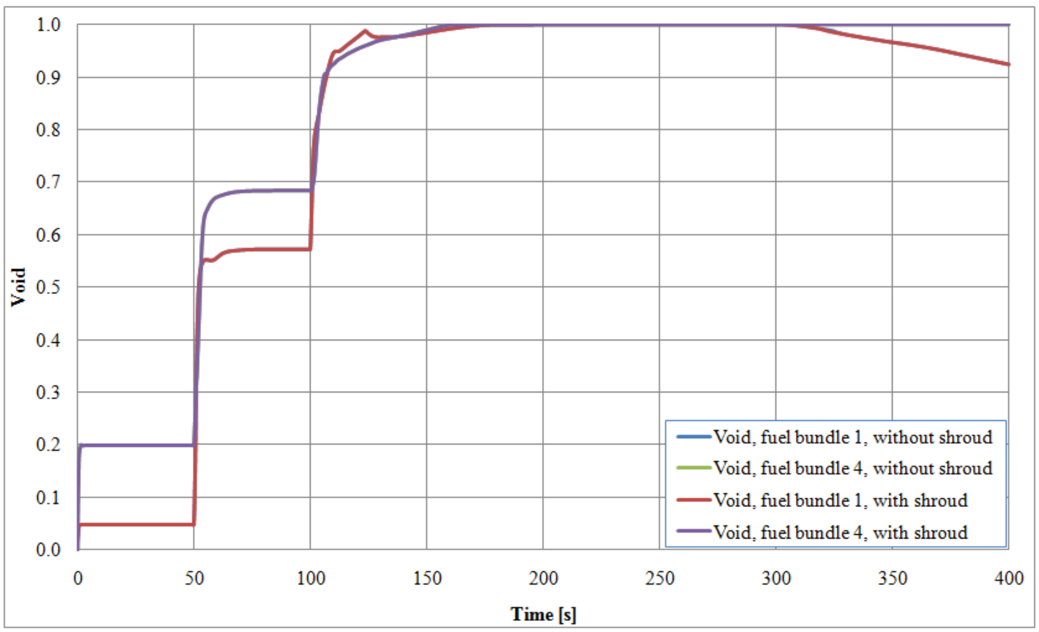

Fig. 3. Void during the transient

The flow shroud accomplishes in this case two things. The first is to increase the quantity of heat produced by oxidation and the quantity of hydrogen. The second one is to reduce the oxidation of the fuel rods, decreasing the oxide layer thickness around them.

Beside the charts for flow rate evolution, void and nuclear heat in the channel, the time scale is reduced to the portion in which the flow is blocked, the studied phenomena linked to the oxidation appear after this moment, and we get a much better observation of the values for the parameters important to our study.

Thus, in Figs. 4 and 5, we noted that in the case of the use of a shroud, the hydrogen quantity generated is larger, and the maximum temperature in the fuel reaches comparable values as those in the case without the flow shroud, with a different evolution, the peak being pushed back toward the introduction of fresh coolant.

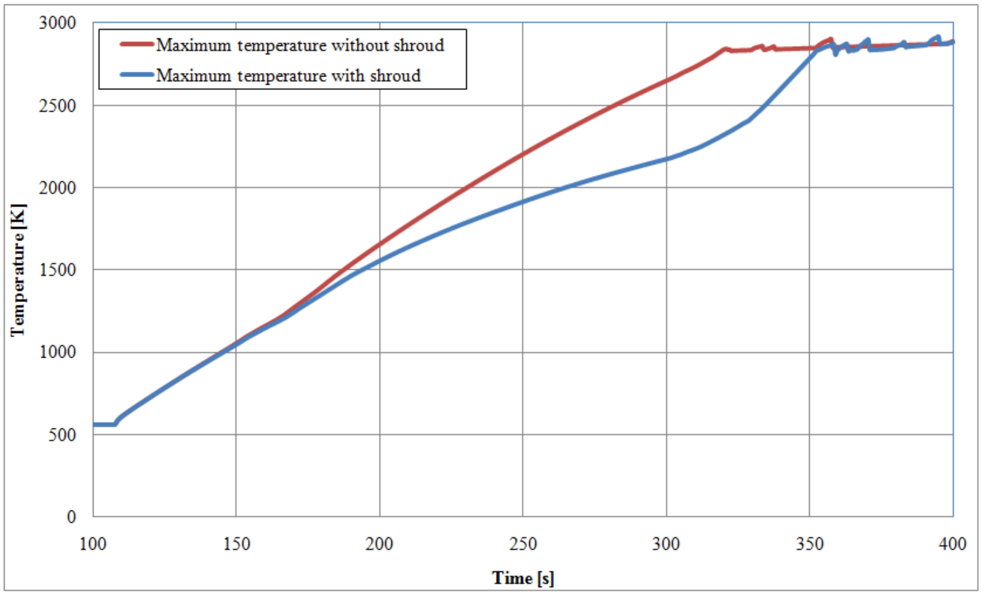

Fig. 4. Maximum temperature

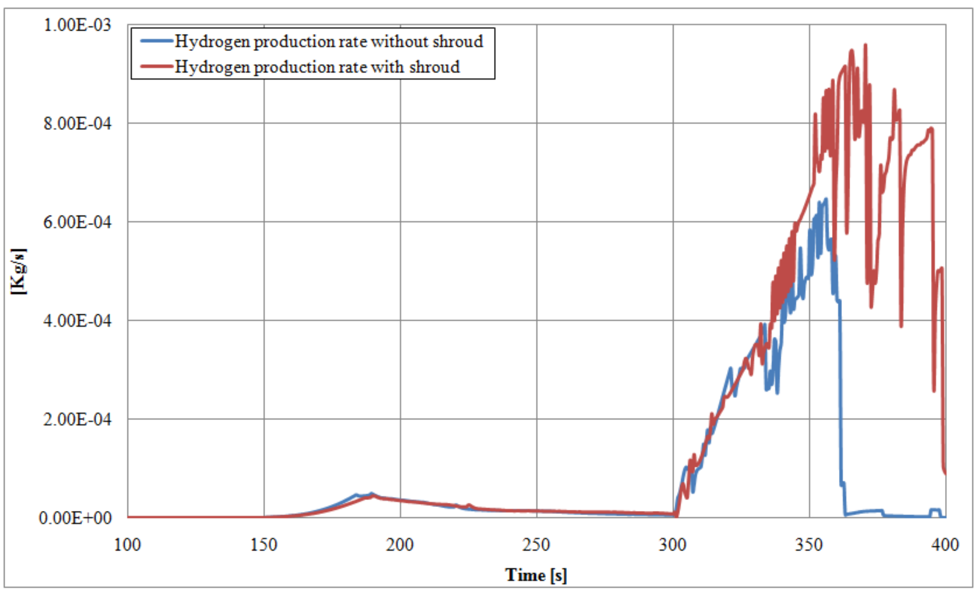

Fig. 5. Hydrogen production

Also, Figs. 6 and 7 show the nuclear heat maintaining the same value in both of the studied cases but, at the same time, a larger quantity of heat produced by oxidation in the case with the use of a shroud.

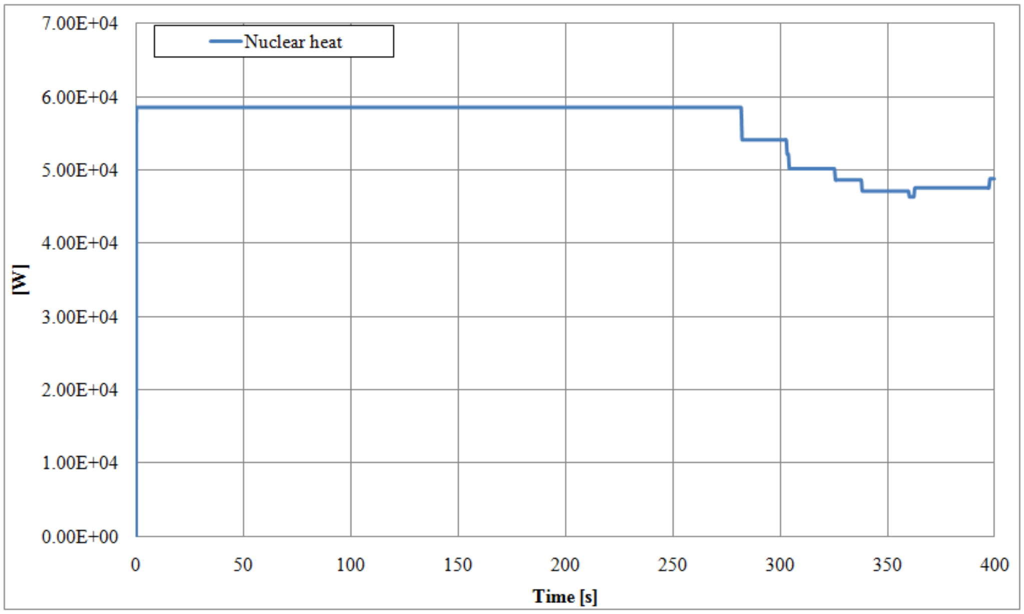

Fig. 6. Nuclear heat

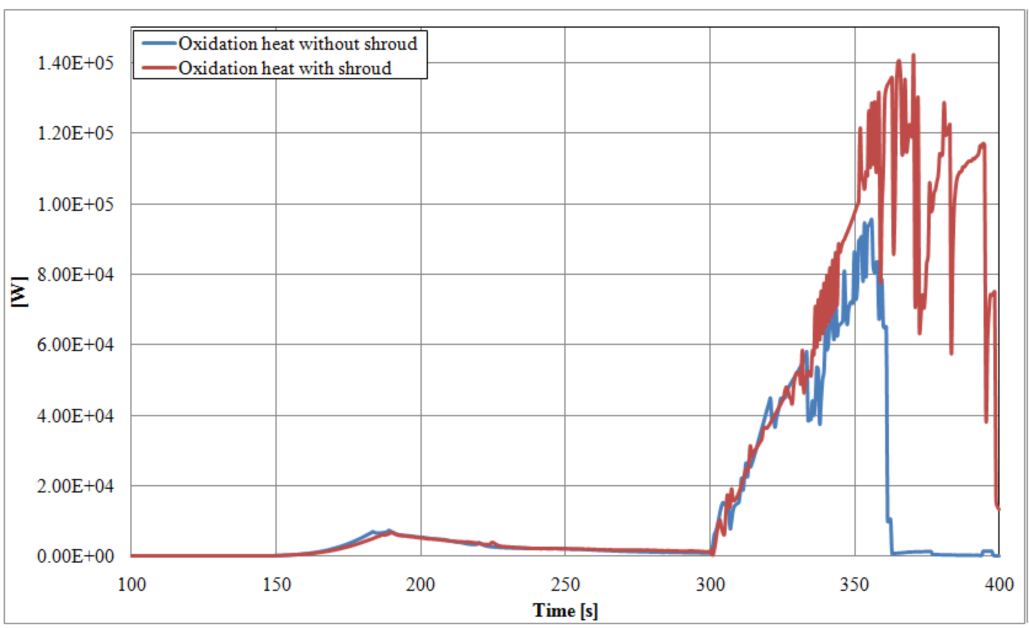

Fig. 7. Oxidation heat

The heat produced by the pressure tube oxidation leads to the increase of the hydrogen quantity, but for the same steam flow, the heat is contained not only in the fuel rods but in the adjacent structures also, relieving the thermal stress of the fuel.

Figs. 8 and 9 show a delay in the damage progression in the fuel rods, due to a lower temperature reached by them in the late phases of the transient, meaning a later time for reaching the break point considered in the SCDAP model for the fuel. These charts represent only the damage progress for the fuel bundles in the middle of the channel, the damage being much reduced for the fuel bundles at the ends of the channel.

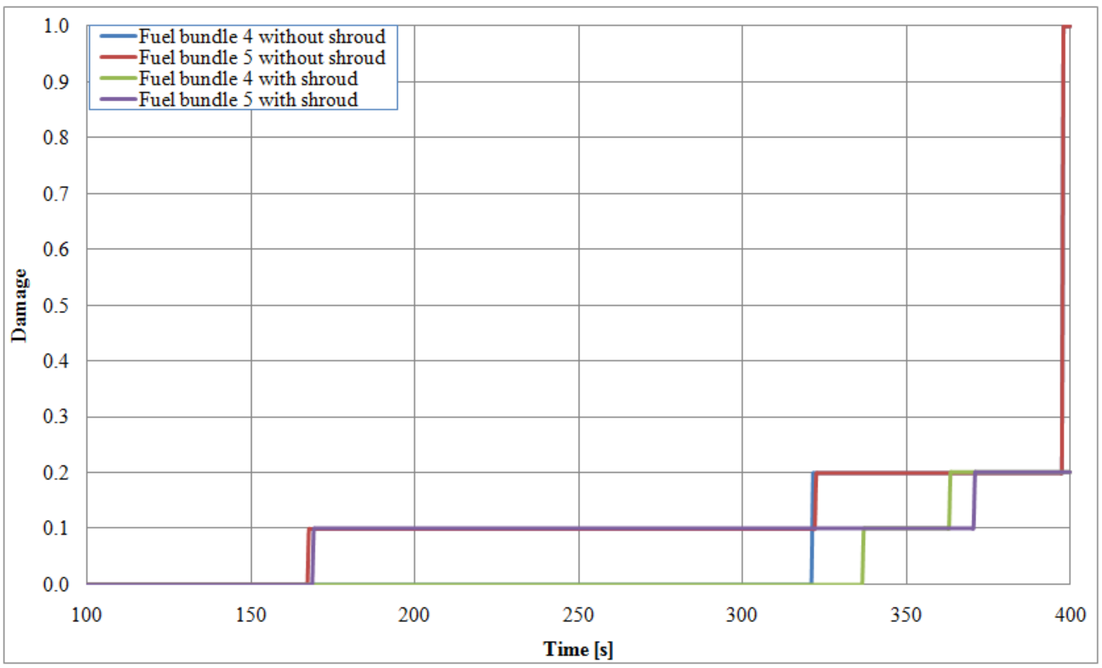

Fig. 8. Fuel damage progression, fuel bundles 4 and 5

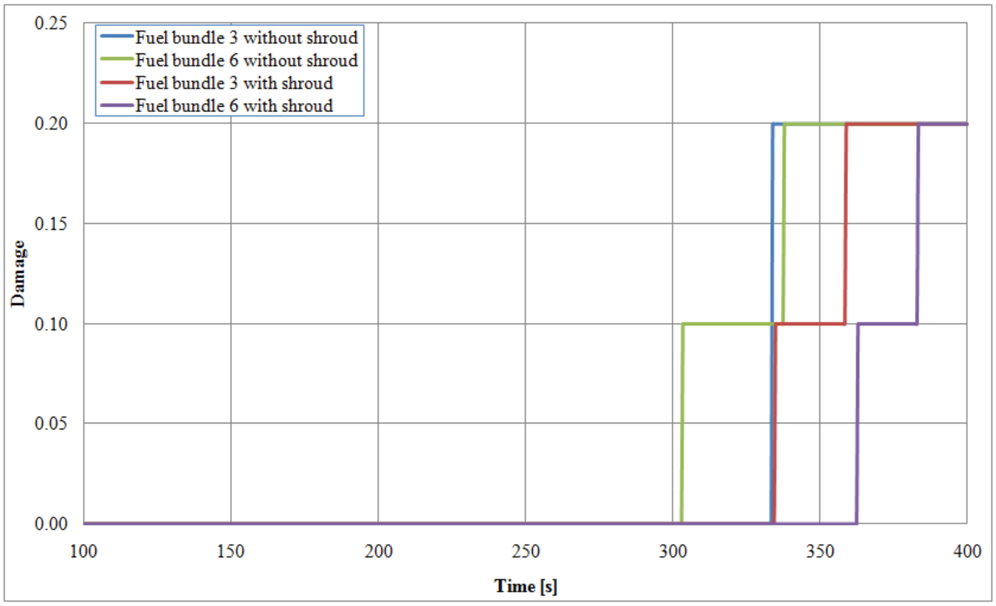

Fig. 9. Fuel damage progression, fuel bundles 3 and 6

Figs. 10 and 11 show a rise for the hydrogen production rate in both cases but notable differences. In the case without shroud, the rate increases more slowly once the coolant is reintroduced and goes to zero after approximately 50 seconds after the reflood while in the case with shroud hydrogen is still produced with a larger and larger rate.

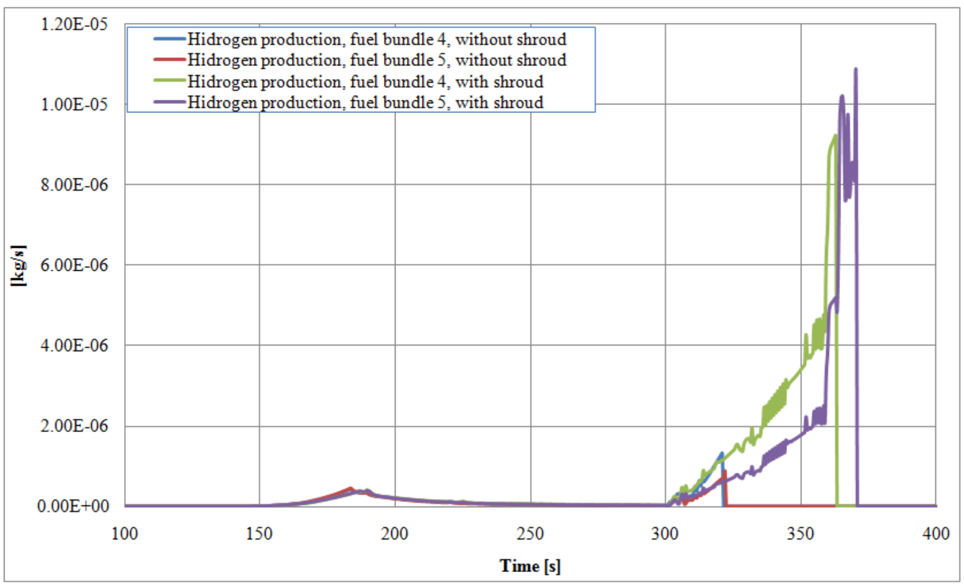

Fig. 10. Hydrogen production rate, fuel bundles 4 and 5

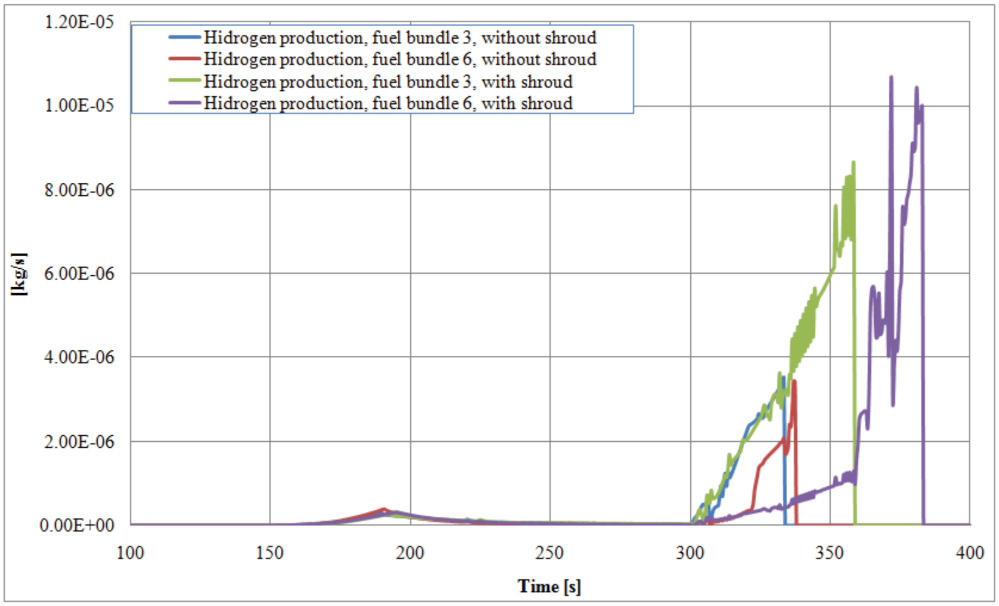

Fig. 11. Hydrogen production rate, fuel bundles 3 and 6

This is probably due to early oxidation of the fuel rods without the oxidation of the pressure tube in the case without shroud that leads to the formation of a thicker oxide layer around the fuel rods. At the reflood moment, the thin layer of oxides in the case with shroud allows the oxidation of the fuel once the coolant is reintroduced.

This phenomenon determines (as shown in Fig. 12) the production of a bigger and bigger hydrogen quantity in the case with shroud, while in the case without it, the integrated hydrogen production (calculated by a control variable introduced by the user) stagnates, indicating the reduction to zero of the hydrogen production rate mentioned above.

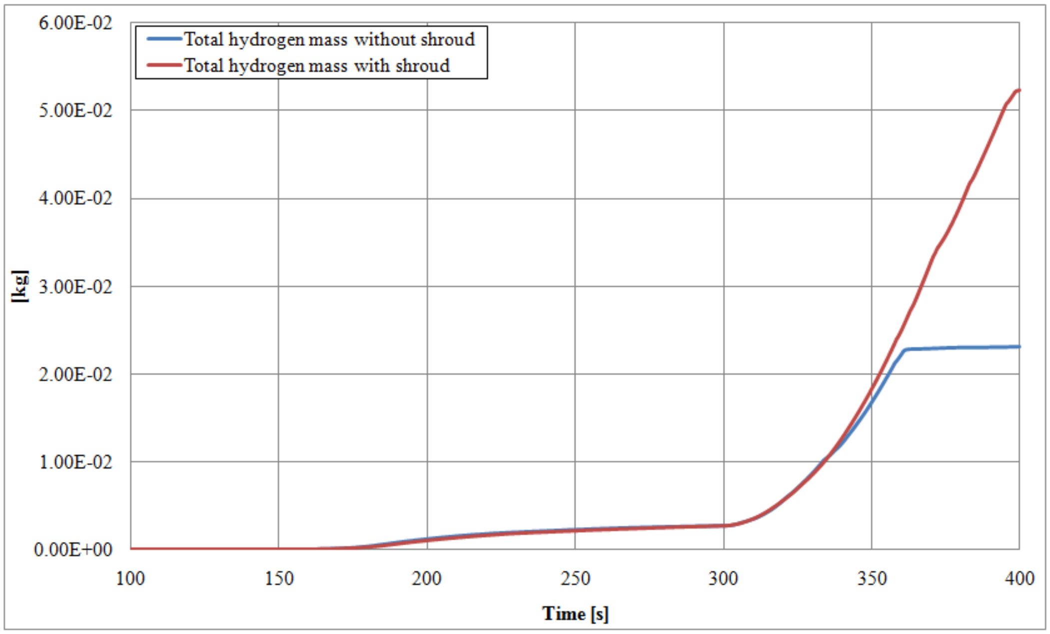

Fig. 12. Hydrogen quantity

6. Conclusions

For the PHWR reactor accident analysis, the introduction of the shroud component represents an important step in the evolution of the simulating capabilities of the systems in this type of nuclear power plants for the SCDAP/RELAP5 code.

The shroud module allows modeling of flow shrouds from the core channels in CANDU thru the modeling of the oxidation of the pressure tube and Calandria tube in the case of accidents that involve loss of coolant.

Pressure tube and Calandria tube modeling, especially the capability to model the oxidation of these structures in the core leads to the better estimation of the heat produced in the core during a LOCA accident and furthermore, the maximum temperature reached in the fuel rods during these kind of transients.

Modeling the oxidation of adjacent structures for the fuel has the effect of reducing the heat stored in the fuel rods, reduces the steam quantity (the main “culprit” for the oxidation) that comes in direct contact with the fuel rods in the case of flow restriction or interruption for the coolant in a flow channel, at the same time reducing the oxide layer thickness on the fuel rods in the case of an LOCA type accident.

During the transient, the most affected fuel bundles are those situated in the middle of the channel being subjected to the highest heat flow in normal operation or in a transient. For this reason, we presented only the parameters of interest in these fuel bundles.

There is more heat produced by the oxidation of the core materials, but at the same time the damage to the fuel rods appears later, cladding oxidation being delayed by the consumption of a large quantity of steam by the oxidation of the pressure tubes.

The total quantity of hydrogen is larger (corresponding to the bigger heat production due to oxidation) in the case of the modeling of the pressure tube an it`s oxidation.

There is a delay in the damage appearance moment similar to the delay in the reaching of the maximum temperatures, the damage occurring after a temperature threshold is reached.

The effect of introducing this model into the input for the CANDU models is a more realistic estimation for the variation of the main parameters in the CANDU fuel channel.

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