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Original Article

Reactivity feedback effect on loss of flow accident in *PWR*

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ABSTRACT

In this work, the reactor kinetics capability is used to compute the design safety parameters in a PWR due to complete loss of coolant flow during protected and unprotected accidents. A thermal-hydraulic code coupled with a point reactor kinetic model are used for these calculations; where kinetics parameters have been developed from the neutronic SRAC code to provide inputs to RELAP5-3D code to calculate parameters related to safety and guarantee that they meet the regulatory requirements. In RELAP5-3D the reactivity feedback is computed by both separable and tabular models. The results show the importance of the reactivity feedback on calculating the power which is the key parameter that controls the clad and fuel temperatures to maintain them below their melting point and therefore prevent core melt. In addition, extending modeling capability from separable to tabular model has nonremarkable influence on calculated safety parameters.

1. Introduction

The fundamental objective of nuclear safety is to ensure that under normal and credible abnormal conditions nuclear facilities are operated in an acceptably safe manner. The plant needs to be designed so that to assure the transfer of energy generated in the fuel to the coolant while maintaining fuel and clad temperature limits even under the most severe anticipated or design basis transient conditions. To maintain fuel rod integrity and prevent the release of fission products, the fuel and clad must be prevented from overheating [1].

Although pressurized water reactors are designed with safety features such as the negative reactivity feedback effect, various unusual events such as a reactivity initiated accident (RIA), a loss-of-flow accident (LOFA), or a loss-of-coolant accident (LOCA) can happen, and this can cause a power excursion in the reactor core even though the heat transport system remains in perfect operating order. This paper emphasises LOFA simulation with the aim of investigating reactor safety.

A LOFA occurs in a reactor due to many causes, such as loss of off-site power, pump failure, heat exchanger blockage, pipe blockage, valve closure, etc. The danger of a LOFA is that it could lessen fuel integrity due to overheating that arises from a low heat transfer coefficient in the reactor core. As a LOFA is classified by regulatory bodies as a design-based accident [2].

The loss of coolant flow is classified into three types in accordance with its severity: partial loss of coolant flow, complete loss-of-coolant flow, and pump locked rotor accident [3]. These accidents are considered to be American Nuclear Society (ANS) condition II, III, IV accident, respectively. Of the three types, the complete loss-of-coolant flow CLOFA is only considered.

The CLOFA is one of the important conditions of reactor coolant system flow down in the design basis accidents. In this condition all the reactor coolant pumps stop due to loss of power, and it leads to the loss of forced core flow, thereby causing a rapid increase in reactor coolant temperature, and there is a possibility that minimum DNBR (departure from nucleate boiling ratio) limits are exceeded, resulting in partial failure of the fuel rods [4], where the DNBR is defined as the ratio of the critical heat flux to the average heat flux [5].

Major parameters such as power, fuel and clad temperatures, and the DNBR are evaluated to investigate whether these parameters exceed their design limits and the reactor is sufficiently safe [6].

It is customary to consider systematically the CLOFA transient with and without a scram event, called, protected and unprotected or self-limited transients, respectively. However, most of the research reported so far is in the context of protected transients.

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Unprotected transient CLOFA analysis, in contrast, has received very limited attention or is just beginning to attract attention, therefore this paper mainly focuses in unprotected CLOFA analysis. Initially, CLOFA during an unprotected condition is studied to determine the time margin limit for scram activation imposed by clad melting temperature and confirm dynamic reactivity feedback models under complete loss of flow condition. Additionally, CLOFA under protected conditions is simulated because in real case the reactor scram signal appears when reactor coolant mass flow rate are reduced below certain level and the control rods start to drop into the core [7] and [8].

The reactivity feedback effect is very important in calculating the power and therefore all other parameters related to safety [9]. The neutronic code SRAC is used to calculate reactivities as function of moderator density and moderator and fuel temperatures [10]. While the thermal-hydraulic RELAP5-3D code is used to calculate power, temperatures, DNBR, etc. [11].

There are two options for the computation of the reactor power in the RELAP5-3D code [11]. The first option is the point reactor kinetics model (either separable or tabular models are used for reactivity feedback), while the second option is a multi-dimensional neutron kinetics model (nodal expansion method (NEM). The decay heat model developed as part of the point reactor kinetics model has been modified to compute the decay power for both the point reactor kinetics model and for the multi-dimensional neutron kinetics model.

In the present studies, the point reactor kinetics model is only used for calculating the power for PWR core.

Unprotected CLOFA is considered as a pioneer study, consequently comparison is only made between the results of protected CLOFA scenario. The results for the PWR calculated by RELAP5-3D code were compared with those of the TRACE codes presented in Ref. [12], taking into account the reactivity feedback effect. The results show that there is good agreement between RELAP5-3D and TRACE results.

The paper is organized as follow: In section II, the reactivity feedback approximations calculated by either separable and tabular models are explained. The results of both protected and unprotected CLOFA scenarios with and without reactivity feedback effect are shown in section III. Finally, the conclusion is summarized in Section IV.

2. Theory

RELAP5-3D is a best estimate system code suitable for the analysis of all transients and postulated accidents in Light Water Reactor (LWR) systems, including reactor point kinetics [9].

The point reactor kinetics model in the RELAP5-3D code is the simplest model that can be used to compute the transient behavior of the neutron fission power in a nuclear reactor. The power is computed using the space-independent or point kinetics approximation which assumes that power can be separated into space and time functions. This approximation is adequate for cases in which the space distribution remains nearly constant.

The point kinetics equations are:

$$\frac{dn(t)}{dt} = \frac{[\rho(t) - \beta]}{\Lambda} n(t) + \sum_{i=1}^{N_d} \lambda_i C_i(t) + S \tag{1}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i f_i}{\Lambda} n(t) - \lambda_i C_i(t) \tag{2}$$

where

n is the neutron density or power
Λ is the prompt neutron generation time
ρ is the reactivity
β is the effective delayed neutron fraction
β_i is the effective delayed neutron precursor yield of group i
C_i is the delayed neutron precursor concentration in group i
λ_i is the delay constant of group i
S is the source rate density
f_i is the fraction of delayed neutron of group $i = \beta_i/\beta$

The RELAP5-3D begins with initial values such as initial power and coolant velocity. In the next step according to the initial value of power, the initial temperatures (fuel, coolant, clad) were calculated. Knowing the initial temperatures is very essential for the next step that named reactivity feedback. In the point kinetic module the power in each step was calculated according to the previous reactivity step. In the final step, temperatures are calculated in hot and average channels. Hot channel temperatures are calculated considering peaking factors. The flowchart is completed when the average temperatures were used for calculation of the reactivity feedback for the next steps [13].

The reactivity feedback in Eq. (1) is calculated by using either separable or tabular models.

2.1. Separable feedback model

The separable model defines reactivity as:

$$\begin{aligned} \textit{Total reactivity} = {} & \textit{initial reactivity} - \textit{bias reactivity} \\ & + \textit{reactivity from tables} \\ & + \textit{reactivity from control variables} \\ & + \sum_{i=1}^{n_\rho} \left[W_{\rho i} \cdot R_\rho(\rho_i(t)) + a_{wi} \cdot T_{wi}(t) \right] \\ & + \sum_{i=1}^{n_F} [W_{Fi} \cdot R_F(T_{Fi}(t)) + a_{Fi} \cdot T_{Fi}(t)] \end{aligned}$$

$$\begin{aligned} r(t) = {} & r_0 - r_B + \sum_{i=1}^{n_s} r_{si}(t) + \sum_{i=1}^{n_c} V_{ci}(t) + \sum_{i=1}^{n_\rho} \Big[W_{\rho i} \cdot R_\rho(\rho_i(t)) \\ & + a_{wi} \cdot T_{wi}(t) \Big] + \sum_{i=1}^{n_F} [W_{Fi} \cdot R_F(T_{Fi}(t)) + a_{Fi} \cdot {}_{Fi}(t)] \end{aligned}$$

where

$r_0$is the initial reactivity
$r_B$is the bias reactivity
$r_{si}$are obtained from input tables defining n_s reactivity (or scram) curves as a function of time
$V_{ci}$are n_c control variables that can be user-defined as reactivity contributions
$R_\rho$is a table defining reactivity as a function of the current moderator density of fluid $\rho_i(t)$ in the hydrodynamic volume i (density reactivity table)
$W_{\rho i}$is the density volume weighting factor for volume i
$T_{Wi}(t)$is the spatial density averaged moderator fluid temperature of volume i
$a_{Wi}$is the volume fluid temperature coefficient (not including density changes) for volume i
$n_\rho$is the number of hydrodynamic volumes in the reactor core
$R_F$is a table defining reactivity as a function of the heat structure volume average fuel temperature $T_{Fi}(t)$ in heat structure i (Doppler reactivity table)
$W_{Fi}$is the fuel temperature heat structure weighting factor for heat structure i

$a_{Fi}$is the heat structure fuel temperature coefficient for heat structure i
$n_F$is the number of heat structures in the reactor core

The separable option uses two tables, one defining reactivity as a function of moderator density $R_\rho(\rho_i)$ (it requires N_{ρ_i} data points) and the other defining reactivity as a function of average fuel temperatures $R_F(T_{Fi})$ (it requires $N_{T_{Fi}}$ data points). Data for the separable option can be obtained from reactor operating data, reactor physics calculations, or a combination of the two, where the number of data points is the summation of N_{ρ_i} and $N_{T_{Fi}}$.

The model assumes nonlinear feedback effects from moderator density and fuel temperature changes and linear feedback from moderator and fuel temperature changes. It is called the separable model because each effect is assumed to be independent of the other effects (a change in one of the any parameters does not affect the others), and the total reactivity is the sum of the individual effects [15]. Boron feedback is not provided, therefore the separable model can be used if boron changes are quite small.

2.2. Tabular feedback model

The separable model assuming that no interactions among the different feedback mechanisms is an approximation. The tabular feedback model computes reactivity from multi-dimensional table lookup and linear interpolation. The tabular model overcomes the objections of the separable model since all feedback mechanisms can be nonlinear and interactions among the mechanisms are included (e.g., the dependence of the moderator density feedback as function of the moderator fluid temperature may be modeled). The penalty for the expanded modeling capability is greatly increased by input data requirements.

The tabular model using the standard variables defines reactivity as:

$$r(t) = r_0 - r_B + \sum_{i=1}^{n_s} r_{si}(t) + \sum_{i=1}^{n_c} V_{ci}(t) + R\left(\overline{\rho}(t), \overline{T}_W(t), \overline{T}_F(t), \overline{\rho}_b(t)\right)$$

$$\overline{\rho}(t) = \sum_{i=1}^{n_\rho} W_{\rho i}\rho_i(t)$$

$$\overline{T}_W(t) = \sum_{i=1}^{n_\rho} W_{\rho i}T_{Wi}(t)$$

$$\overline{T}_F(t) = \sum_{i=1}^{n_F} W_{Fi}T_{Fi}(t)$$

$$\overline{\rho}_b(t) = \sum_{i=1}^{n_\rho} W_{\rho i}\rho_{bi}(t)$$

where ρ_b is the spatial boron density. The reactivity function R is defined by a table input by the user, it is evaluated by a direct extension of the one-dimensional table lookup and linear interpolation scheme to multiple (three- or four-dimensions) dimensions.

The four-dimensional table lookup and interpolation option (TABLE4) computes reactivity as a function of moderator densities, moderator temperature, fuel temperature, and boron concentration. The three-dimensional option (TABLE3) does not include boron concentration. In our work, TABLE3 is used (neglecting boron effect).

Using N_{ρ_i}, $N_{T_{Fi}}$, and $N_{T_{Wi}}$ as the number of values in the three sets of independent variables, the number of data points for a three-dimensional table is the product of them $N_{\rho_i}{\cdot}N_{T_{Fi}}{\cdot}N_{T_{Wi}}$. Using only four values for each independent variable, a three-dimensional table requires 64 data points.

SRAC code calculations provide kinetics parameter inputs to the RELAP5-3D point-kinetics model. SRAC code system is multi-groups (107 energy-groups) neutronics calculation code based on the collision probability method [10].

In Separable model, the SRAC neutronic code is used to calculate the reactivity for all moderator densities (733 ~ 73.3 kg/m3) = 10

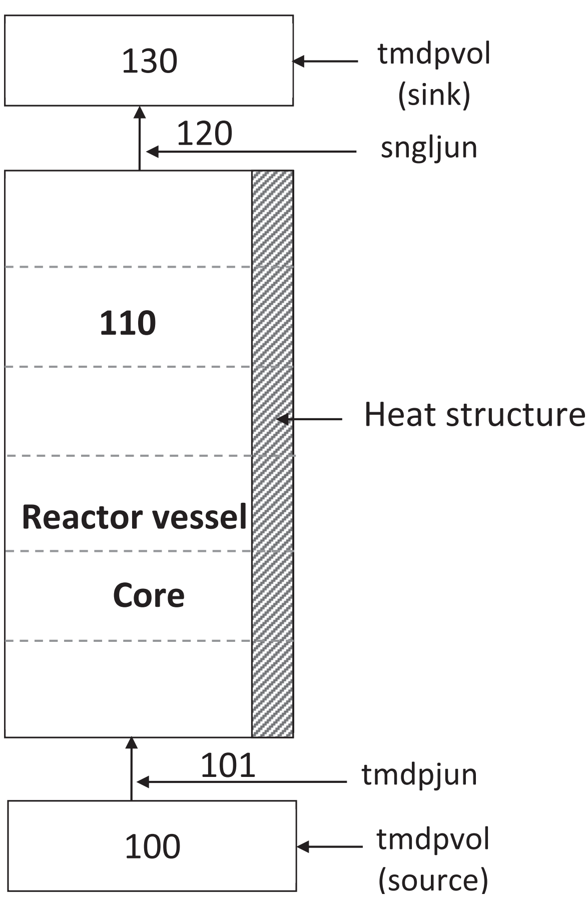

Fig. 1. Simplified PWR nodalization scheme.

Table 1
PWR steady-state operating conditions.

Parameter	Value
Power (MW)	3411
Pressure (Pa)	1.551E+7
Initial flow rate (kg/sec)	1.863E+4
Coolant inlet temperature (K)	565
Coolant outlet temperature (K)	599
Number of fuel assemplies	193
Assemply type	17*17
Core diameter (m)	3.37
Fuel radius (m)	4.1265E-3
Clad radius (m)	4.7435E-3
Coolant radius (m)	7.1186E-3
Fuel length (m)	3.66
Clad Type	Zirconium
Fuel Type	UO_2
Fuel enrichment (wt%)	4.9

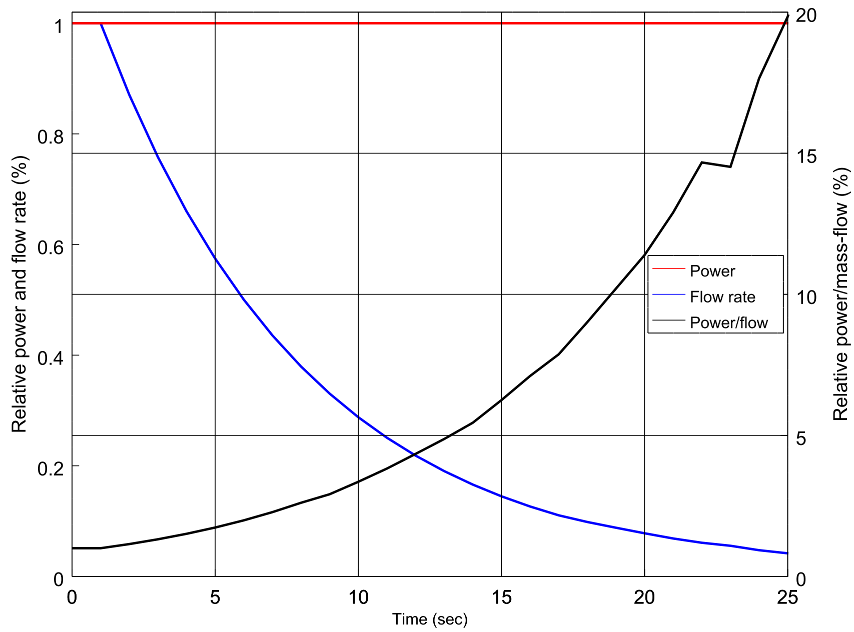

Fig. 2. Relative power and flow rate for unprotected core without reactivity feedback effect.

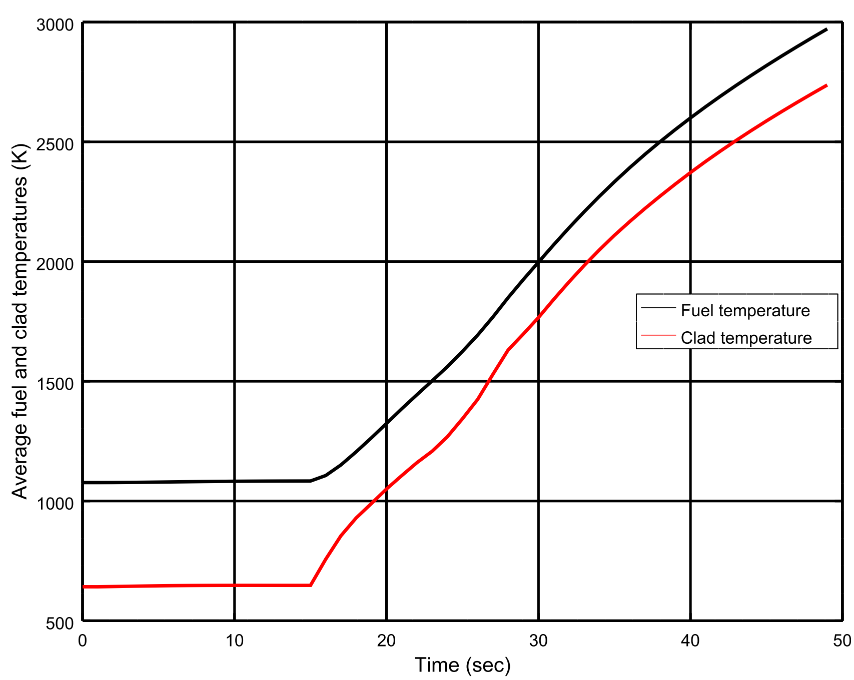

Fig. 3. Fuel and clad maximum temperatures for unprotected core without reactivity feedback effect.

runs and for fuel temperatures (600 ~ 1400 K) = 5 runs. Therefore the total number of runs = 10 + 5 = 15 only.

In Tabular model: for a three-dimensional table (TABLE3), the SRAC code is used to calculate the reactivity for: Moderator densities (733 ~ 73.3 kg/m3) = 10 runs, fuel temperatures (600 ~ 1400K) = 5 runs, and moderator temperatures (560 ~ 620K) = 9 runs. Therefore the total number of runs = 10*5*9 = 450.

Thus, the extension of modeling capability (from Separable to Tabular model) leads to a serious increase in the input data requirements. These reactivities are necessary as input for RELAP5-3D in order to calculate power and temperatures, etc.

3. Numerical results

The descriptions herein are based on the standard Westinghouse four-loop PWR, but generally applicable to smaller units as well (three-loop and two-loop plants), where the configurations for two- and three-loop plants differ only in the size of the components.

The PWR nodalization model is shown in Fig. 1, and the steady-

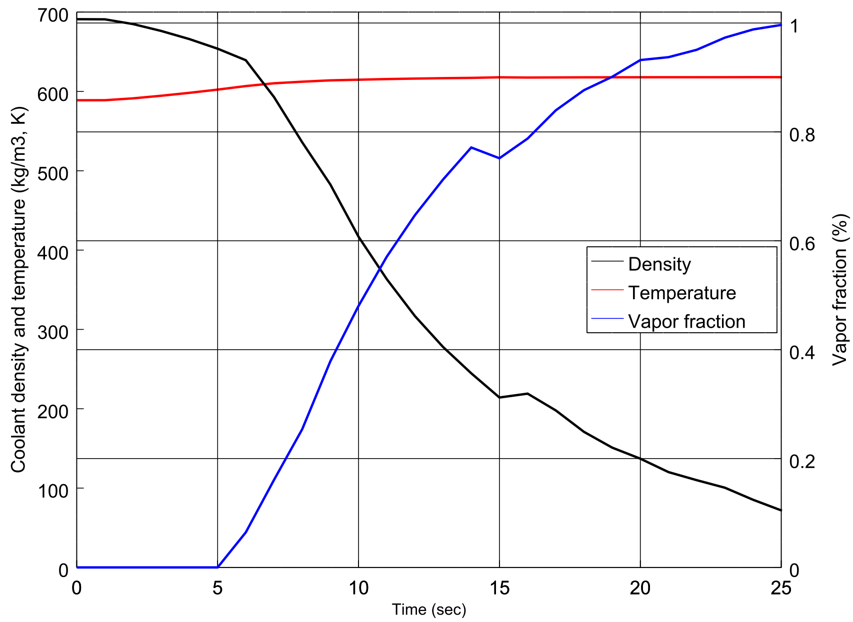

Fig. 4. Coolant density and temperature for unprotected core without reactivity feedback effect.

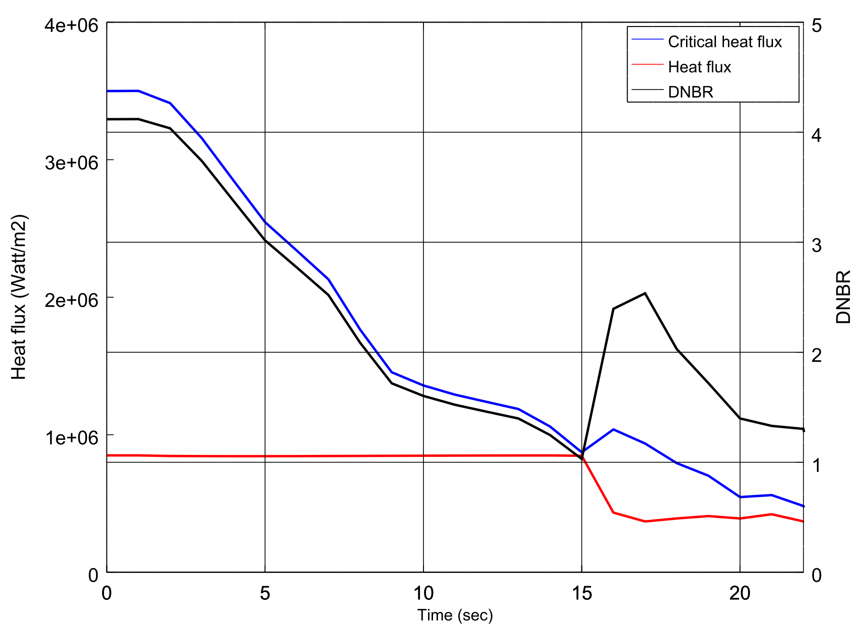

Fig. 5. DNBR and heat flux for unprotected core without reactivity feedback effect.

state operating conditions are summarized in Table 1. In order to investigate the temperature distribution within the fuel, the heat structure is divided into six vertical and also six radial volumes.

In this work, The CLOFA transients with and without a scram (called protected and unprotected transients, respectively) are considered. It is assumed that a reactor trip at time zero initiated the transient with one second time delay with a coast-down time of five seconds, where the flow exponentially decreases to 50% of the rated flow after 5 sec. The flow rate keeps decreasing until 28 sec and it becomes constant after that.

3.1. Complete loss of flow accident for unprotected core

This paper emphasises the unprotected transient with the aim of determining the reactivity insertion limit imposed by clad melting temperature and investigating the dynamic reactivity feedback effect under complete loss of flow condition.

The hypothetical scenario of unprotected loss of flow accident is chosen, where scram signal is assumed not to be released. Furthermore, two cases with and without reactivity feedback contributions are calculated.

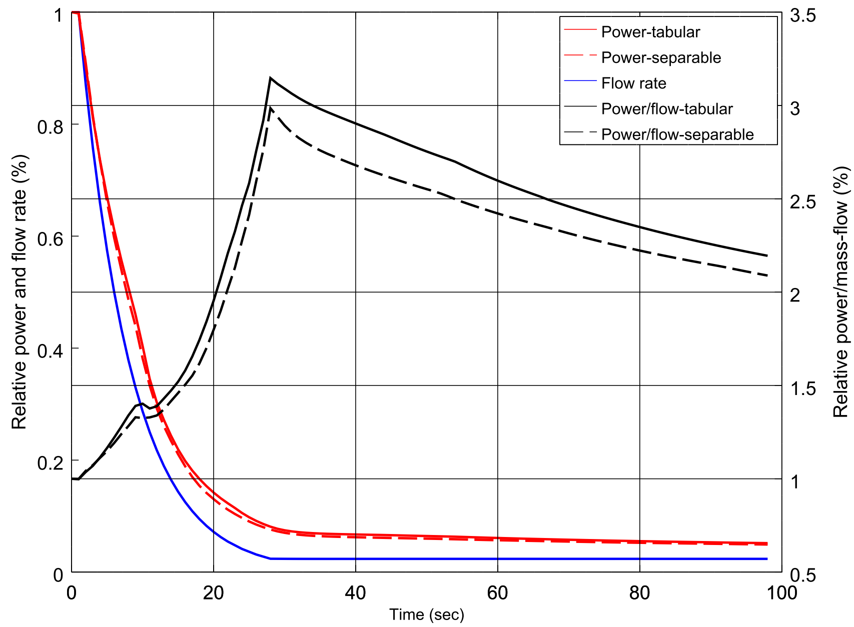

Fig. 6. Relative power and flow rate for unprotected core with reactivity feedback effect.

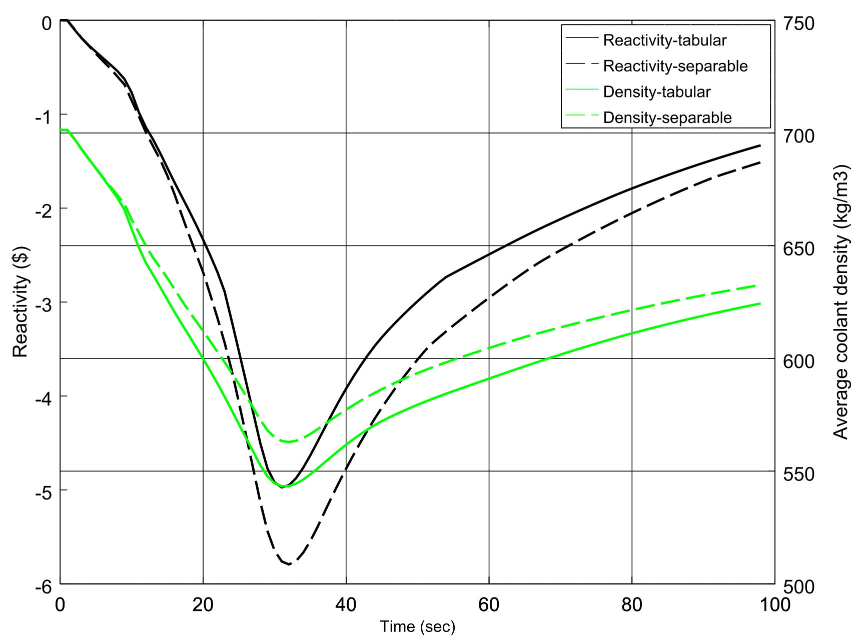

Fig. 7. Reactivity and average coolant density for unprotected core with reactivity feedback effect.

3.1.1. Without reactivity feedback

First let us investigate the power and other parameters behavior if there is no reactivity feedback effect. For that case the power remains constant as the flow rate decreases as shown in Fig. 2.

Fig. 3 shows the average centerline fuel and clad temperatures (where the maximum clad and fuel temperatures are found in zone 4). As the flow rate decreases the fuel and clad temperatures remain almost constant (since the power is assumed to be constant due to neglecting reactivity effect) until 15 sec where there will be a sudden increase in both fuel and clad temperatures. These increases are described as follow:

As the flow rate decrease the coolant temperature increases and density decreases as a vapor bubble forms at the heated surface as shown in Fig. 4. When the temperature of the coolant exceeds the coolant saturation temperature(about 618 K), boiling could be regarded as saturated nucleate boiling. Successively, the maximum heat flux attained when the bubbles become so dense that they form a vapor film over the heated surface, resulting in the heat not being able to pass through the vapor film. Consequently, the heat flux drops appreciably causing rapid increases in fuel and clad temperatures. This maximum heat flux is considered a design limitation, which is referred to as the DNB (departure from nucleate

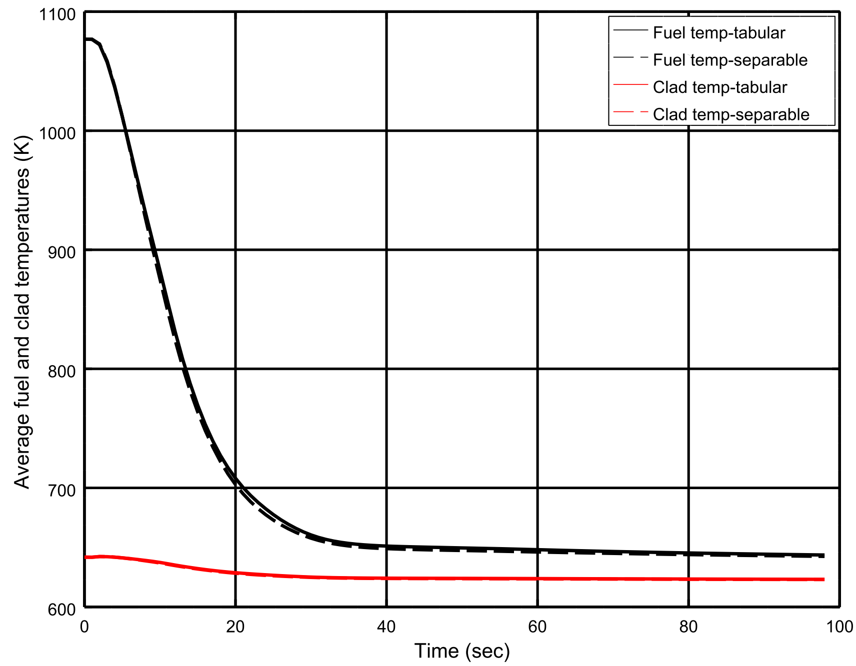

Fig. 8. Fuel and clad maximum temperatures for unprotected core with reactivity feedback effect.

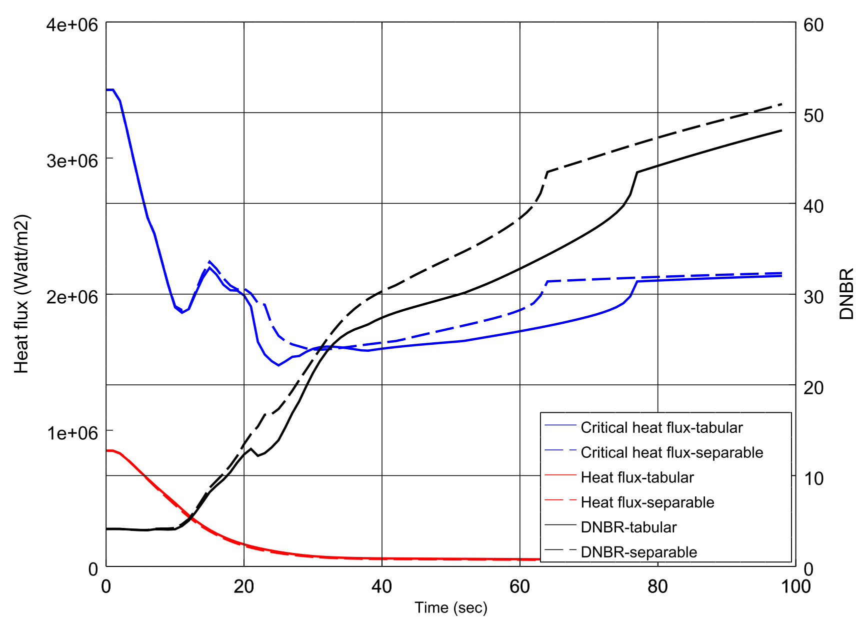

Fig. 9. DNBR and heat flux for unprotected core with reactivity feedback effect.

boiling) as presented in Fig. 5. This condition becomes so severe that damage to the material being heated may result. However, before arriving at these regions, a minimum DNBR value must be imposed as a part of the thermal hydraulic design, this is met by limiting the DNBR to about 1.5, and scram should be activated.

At 25 sec the clad temperature is 1256K while it should not exceed 1255K for normal operation and 1477K (for light-water reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding) for the severe anticipated or design basis transient conditions [12].

From the previous results we can conclude that neglecting the reactivity feedback effect causes overheating of the clad over its melting point and leading to core melt and therefore to severe accident. Hence scram should be activated as the relative flow rate decreased to 70% of the initial flow rate.

3.1.2. *With reactivity feedback*

In this section, both Separable and Tabular models are compared to analyse the reactivity feedback effect. Due to the loss of flow, the moderator temperature increases and density decreases,

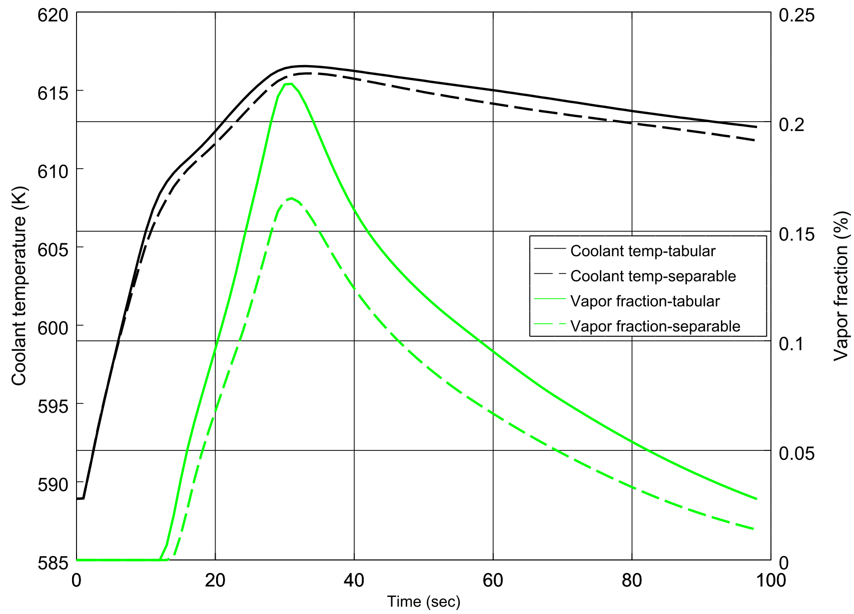

Fig. 10. Coolant temperature and vapor fraction for unprotected core with reactivity feedback effect.

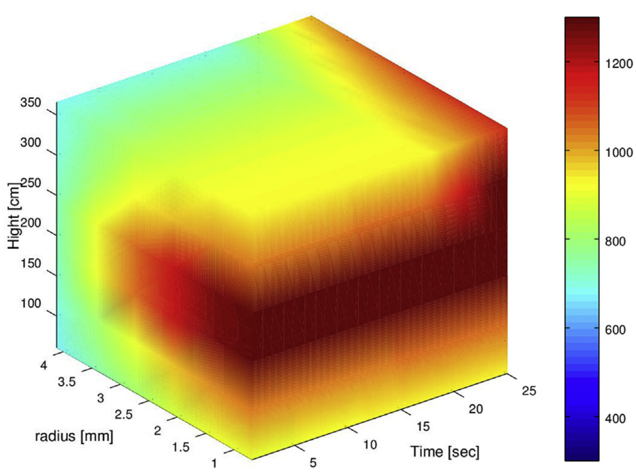

Fig. 11. 3D fuel temperature distribution for unprotected core without reactivity feedback effect.

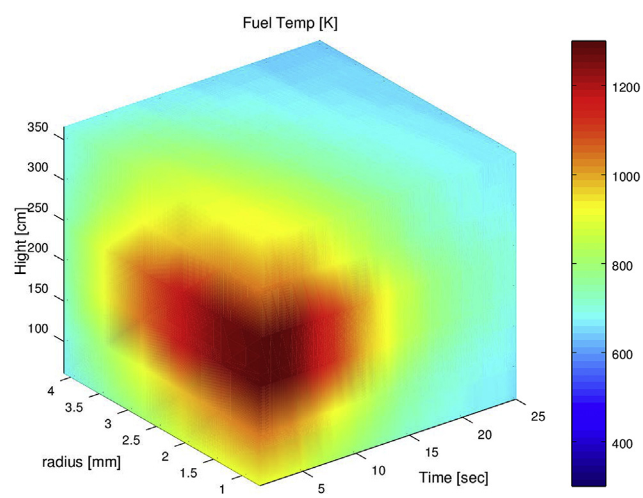

Fig. 12. 3D fuel temperature distribution for unprotected core with reactivity feedback effect.

consequently the number of thermal neutrons decreases and this leads to drop in power, as presented in Fig. 6 for both separable and tabular models. Fig. 7 shows the negative reactivity component (coolant density and temperature reactivities) for the two models.

As the power decreases the fuel and clad temperatures also begin to decrease, as shown in Fig. 8. The reactivity will increase due to Doppler effect, and this is the positive reactivity component (Doppler reactivity).

For separable model the total reactivity is just the summation of the coolant density reactivity and Doppler reactivity (separable model assumes that each effect is independent of the other). In tabular model, the total reactivity is a function of the three reactivities, since all feedback mechanisms are dependent. Therefore the absolute reactivities calculated by the tabular model are smaller than the separable model, although the coolant density is smaller than that calculated by separable model.

The power decreases with the decrease in the flow rate due to negative net reactivity feedback which is dominated by the negative density reactivity feedback as the coolant heat-up (the coolant density reactivity is much greater than Doppler reactivity). After about 28 sec (as the flow rate becomes constant) coolant density and consequently reactivity begin to increase again depending on the power to flow rate ratio as shown in Fig. 6. Before 28 sec, the P/F ratio increases because the drop in flow rate overcomes the decrease in power. At 28 sec, the flow rate becomes constant while there is a little bit of decrease in power, so the P/F ratio decreases.

Fig. 9 describes the DNBR and heat flux behavior, the DNBR increases as a result of lessening in heat flux due to power decreasing. The coolant temperature does not exceed the coolant saturation temperature as presented in Fig. 10 consequently a vapor film will not be formed. The maximum vapor fraction is only about 20% while it reaches about 98% when neglecting the reactivity

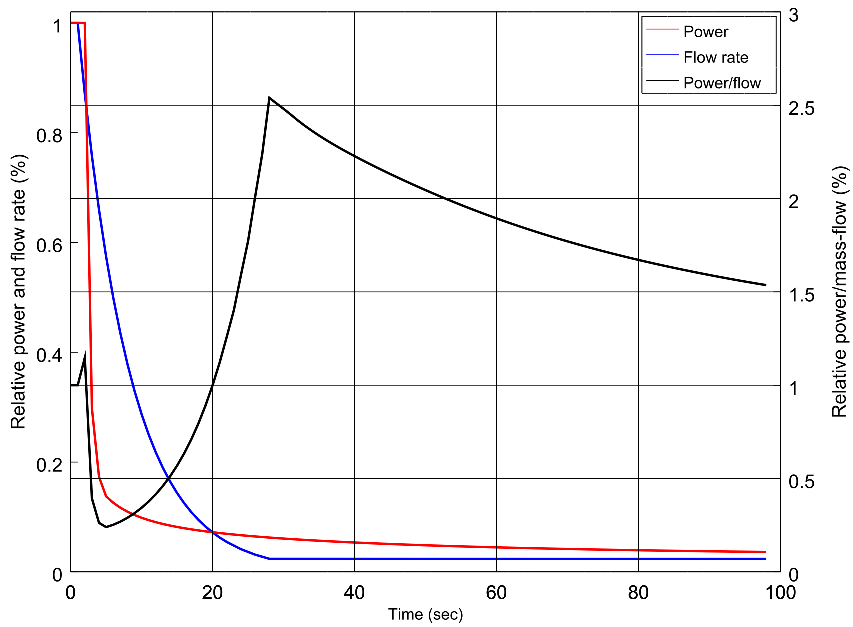

Fig. 13. Relative power and flow rate for protected core without reactivity feedback effect.

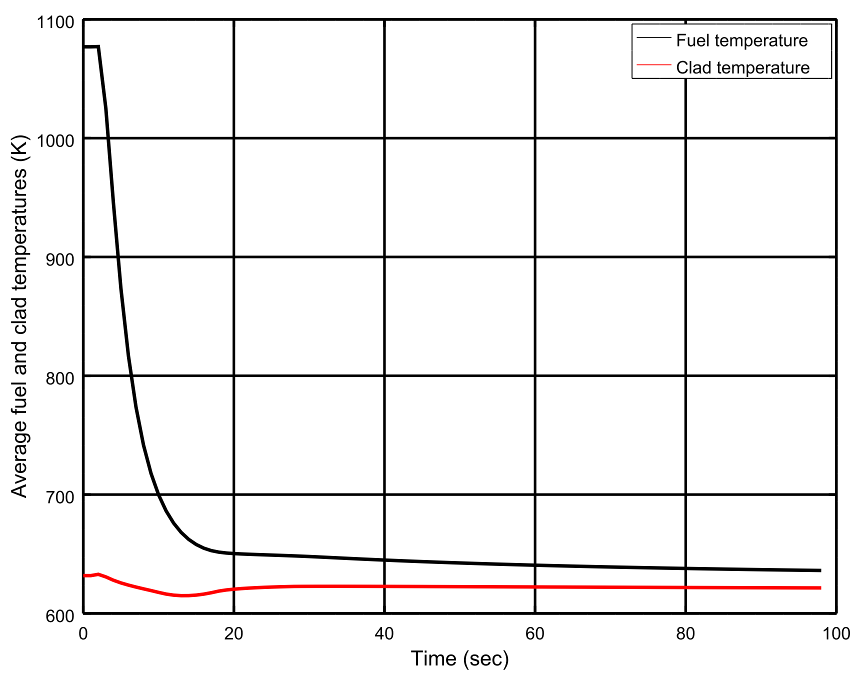

Fig. 14. Fuel and clad maximum temperatures for protected core without reactivity feedback effect.

feedback (as shown in Fig. 4). It can be seen that the DNBR calculated by tabular model is smaller that separable model due to the difference in the coolant temperatures calculated by the two models.

Generally speaking, extending modeling capability from separable to tabular model has nonremarkable influence on calculated safety parameters.

Figs. 11 and 12 depicted the distribution of fuel temperature (3D) with and without reactivity feedback, respectively. The intense red color represents the central region of the rod. The power profile imposed on the system, in the form of a cosine chopped, influences the distribution of the temperatures. Without reactivity feedback, as the flow rate decreases the fuel temperature increases leading to core melt. While the fuel temperature decreases if the reactivity effect is considered (fuel temperatures calculated by tabular and separable models are almost same).

3.2. Complete loss of flow accident for protected core

In real case the reactor scram signal appears as the reactor coolant mass flow rate declined below certain level and consequently the control rods start to drop into the core.

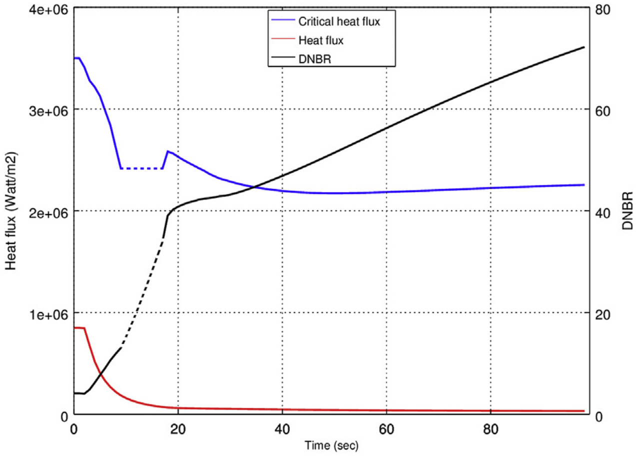

Fig. 15. DNBR and heat flux for protected core without reactivity feedback effect.

3.2.1. *Without reactivity feedback*

In the analysis, the calculations is performed without reactivity feedback effect, where the reactor scram signal is assumed to be activated at 2.5 sec when mass flow rate reduced to 70% of the initial flow rate.

Fig. 13 represents the change of power due to flow rate reduction. The power remains constant until 2.5 sec and it drops as a result of control rod insertion. The fuel and clad temperatures follow the decrease in power, as described in Fig. 14.

Fig. 15 explains the DNBR and heat flux behavior, the DNBR increases due to decrement in heat flux as a result of power drop. The results indicate that the reduction in power following the trip is sufficient to ensure that there is adequate margin to DNBR and the fuel cladding does not fail. However, the critical heat flux is exceeded within a few seconds (7 ~17 sec, dashed line) due to flow instability caused by void formation in the sub-cooled boiling regime, where coolant temperature decreases blow its initial value as shown in Fig. 16.

3.2.2. *With reactivity feedback*

For the purpose of comparison, the results of the three-loop PWR calculated by RELAP5-3D code were compared with those of the TRACE code [14] presented in Ref. [12] with 10 sec flow coast-down.

Fig. 17 compares the relative power and flow rate calculated by RELAP5-3D and RACE code presented in Ref. [12] (denoted by the symbol Ref). Here we only consider the separable model since both models (separable and tabular) give similar results.

The smooth decrease in reactor power before the scram is due to a negative reactivity feedback for coolant density and temperature, and the fuel Doppler coefficients, while the sudden drop at 2.5 sec is mainly due to negative scram reactivity insertion. As can be seen, there is good agreement between RELAP5-3D and reference results.

The fuel and clad temperatures also follow the decrease in power as shown in Fig. 18, however there is a small difference between the fuel temperature calculated by RELAP5-3D and reference results.

4. Conclusions

In the present study, the complete loss of flow accidents were performed including two scenarios with protected and unprotected (no scram) PWR core. The calculations were carried out with and without reactivity kinetics feedback to investigate their impact on the design parameters. The results indicate that both separable and tabular models give similar results, consequently it is recommended to use the separable model for saving computational time.

It was confirmed that in a loss of flow the reactor power decreases due to the negative reactivity feedback effect even if a reactor scram did not occur. Moreover, neglecting the reactivity feedback effect for the unprotected scenario causes overheating of

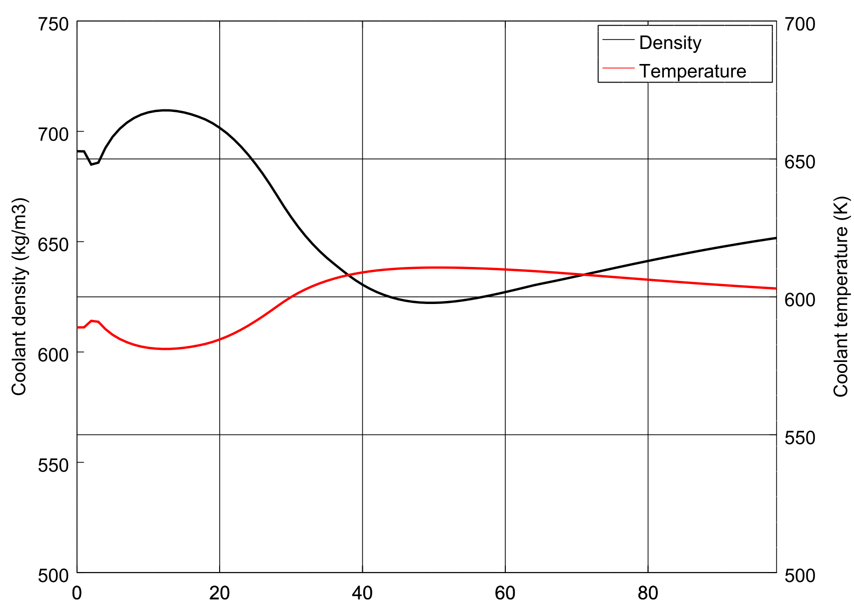

Fig. 16. Coolant density and temperature for protected core without reactivity feedback effect.

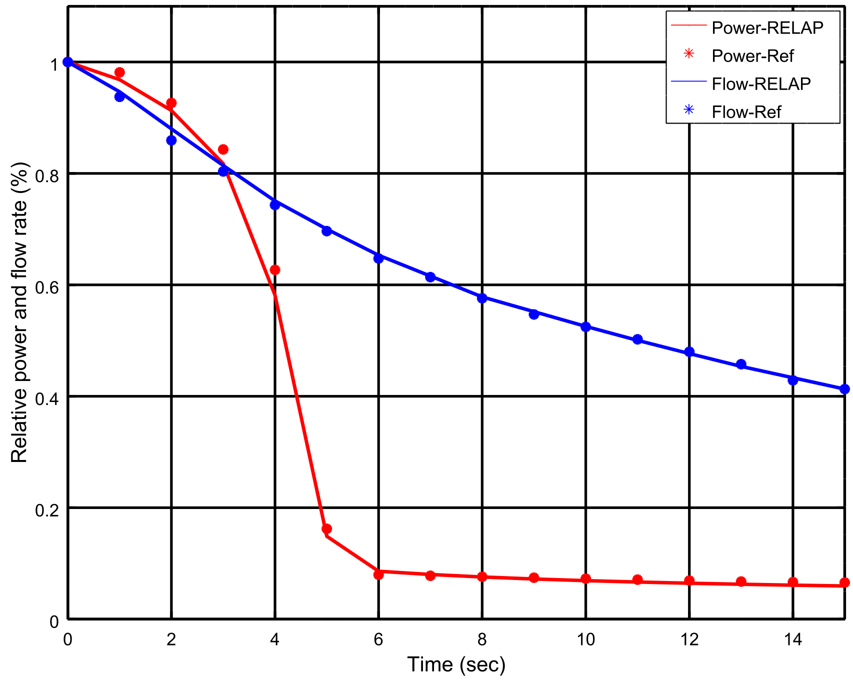

Fig. 17. Relative power and flow rate for protected core with reactivity feedback effect.

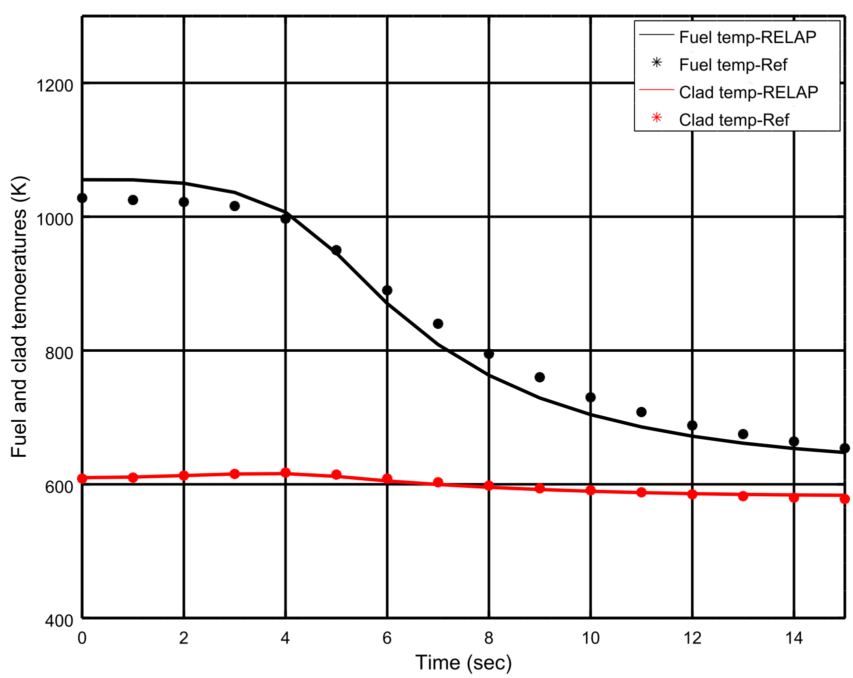

Fig. 18. Fuel and clad maximum temperatures for protected core with reactivity feedback effect.

the clad and therefore the minimum DNBR limits are exceeded resulting in core melting. The results obtained with the RELAP5-3D code agree well with the reference results for protected scenario taking into account reactivity feedback.

The present study is in fact an initial attempt, and the authors are very interested in conducting more analyses, including the nodal expansion method to accurately drive the reactivity coefficients.

5. Future work

In our future work, the advanced nodal expansion method (NEM) will be investigated, this method is widely used in nuclear industry due to its accuracy and short execution time. In addition, the impact of simulation uncertainty on the calculation results such as core integrity will also be clarified under accident conditions. Uncertainty is included in the design margin, which is a part of safety margin. Thus, its quantification is important for nuclear safety.

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