Rod Ejection Accident by the Coupled System Code ATHLET-QUABOX/CUBBOX*

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Abstract

The paper considers a Rod Ejection Accident (REA) which has been calculated by the coupled-code system ATHLET-QUABOX/CUBBOX. For the present study, a MOX/UOX mixed core loading was developed on the basis of a generic PWR. The results are particularly focused on the fuel enthalpy rise which is the main safety criterion for such transient. A parametric REA study has been performed, showing the influence of some important thermal-hydraulic and neutron-physical parameters. Simulations have been performed using realistic or artificially decreased delayed neutron fractions for two different core states (HZP and 30% of the nominal power). Effective fuel rod temperature influence (i.e. Doppler coefficient) has been studied by using different correlations (0.5/0.5 weighting factors or the typical $T_{\text{Doppler}} = 0.7 T_{\text{Surface}} + 0.3 T_{\text{Center}}$) or by changing the fuel gap conductance. It is shown that the maximum enthalpy (and enthalpy increase) does not always appear in the affected fuel assembly but can also appear in the neighboring ones. This result is a direct consequence of the burn up dependence of the enthalpy. The paper also considers the case of local delayed neutron parameters and briefly describes the future REA studies foreseen at GRS such as an investigation of quantitative uncertainty propagation from the nuclear data to the transient behavior.

Key words: Rod Ejection Accident (REA), Reactivity Initiated Accident (RIA), Coupled Codes, Parametric Study

1. Introduction

The Rod Ejection Accident (REA) is the most limiting case among Reactivity Induced Accident (RIA). Due to the fast reactivity insertion leading to prompt criticality and thus to a sharp fuel enthalpy increase in the affected part of the core, REA can cause severe fuel damage. The REA is usually an asymmetric transient where neutron kinetics and the thermal-hydraulics are strongly coupled (through Doppler feedback). 3D coupled models are required to perform such analyses. For the present study, a MOX/UOX mixed core loading was developed on the basis of a generic PWR and modeled with the coupled-code system ATHLET-QUABOX/CUBBOX.

It is known that the transient evolution of the REA is depending on several parameters from which the most important are: the reactivity insertion value (rod worth), neutron kinetics (delayed neutron fraction $\beta$ and lifetime $\lambda$), feedback coefficients (almost exclusively Doppler coefficient) and axial power distribution. A parametric REA study has been performed by modifying the values of some key parameters such as the delayed

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neutron fraction, the initial power level or the heat transfer coefficient of the fuel gas gap.

2. Codes Presentation

ATHLET (Analysis of Thermal Hydraulic of LEaks and Transients) is a TH system code, developed at the GRS. It allows a parallel channel modelisation inside the core vessel and a precise fuel rod model (Lerchl, 2005). QUABOX/CUBBOX is a 3D neutron physics code. It solves the diffusion equation with two groups of energy and up to 6 groups of delayed neutrons (Langenbuch, 1977). Over the year, both codes have been separately extensively validated. The coupled system ATHLET-QUABOX/CUBBOX was developed in the early 90s. Since, it has been validated on a great number of transient and international benchmarks (Langenbuch, 2005).

Maximum fuel enthalpy and especially fuel enthalpy increase are decisive parameters in the study of rod ejection accidents since they are the current safety criteria for REA in most countries. The limits were derived from experimental results on fuel rod exposure to fast power peaks. In this paper, the fuel enthalpy is expressed either in J/g.

The calculation of fuel enthalpy was therefore implemented into QUABOX/CUBBOX core model. This model has the capability to give the enthalpy for each assembly at each time step.

3. Core Description

The core loading is representative of current PWR with 193 assemblies of both Uranium and MOX. The maximum enrichment in UOX fuel is 3.95% and 2.85% in MOX fuel. Fig. 1 shows a representation of the core with the control rods position and the MOX assemblies' position.

ATHLET thermal-hydraulic core model contains 194 parallel channels (THC). Each THC is assigned to one fuel assembly in order to calculate the thermal-hydraulic feedbacks (i.e. fluid density, fuel temperature and boron concentration) precisely. One additional channel models the reflector/core bypass. The model used is a so-called ‘open core’ model where the primary circuit outside of the core is not simulated. Therefore, the boundary conditions (e.g. outlet pressure, inlet temperature and massflow) have to be defined in the input deck.
4. Transient Description

In all studied cases, the rod was ejected after 2s of 0-transient and within 0.1s which is a typical rod velocity for such transients. The accident sequence of events is depending on the following parameters:
- Neutron kinetic parameters: delayed neutron fraction $\beta$ and lifetime $\lambda$
- Reactivity feedback coefficients: in this case the main feedback parameter is the fuel temperature (Doppler effect)
- The axial power distribution, which is depending on the burnup distribution, the Xenon distribution and the rods position

The accident simulation can be either realistic or conservative. The latter case is achieved by changing values of key parameters in the penalizing direction. For example, one can reduce the delayed neutron fraction, increase the worth of the ejected rod, decrease the Doppler coefficient or use penalizing axial power distribution.

The ejected rods are selected taking into account, not only the rod worth but also the power peaking factor after rod ejection. These parameters are depending on the actual core loading and the control rods position. The values can be obtained by QUABOX-CUBBOX steady state calculations. Stationary results for several control rods are presented in Table 1.

The most penalizing control rod in position (C7) was selected from Tab. 1. Control rod (C7) withdrawal results in a strong increase of $K_{eff}$ to 1.00486 (rod worth = 434pcm) and of power peaking factor to 8.55.

Table 1. Control rod worth and associated power peaking factor

<table>
<thead>
<tr>
<th>Rod position</th>
<th>BU at rod position (GWD/t)</th>
<th>Cycles of neighboring assemblies (0=fresh)</th>
<th>$K_{eff}$</th>
<th>Peaking factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>ARO</td>
<td>12.4</td>
<td>0/1/3/0</td>
<td>1.00000</td>
<td>3.00</td>
</tr>
<tr>
<td>A3</td>
<td>0.0</td>
<td>0/1/1/2</td>
<td>1.00139</td>
<td>3.47</td>
</tr>
<tr>
<td>A5</td>
<td>26.6</td>
<td>2/1/1/0</td>
<td>1.00470</td>
<td>7.92</td>
</tr>
<tr>
<td>D4</td>
<td>0.0</td>
<td>0/0/4/0</td>
<td>1.00432</td>
<td>2.71</td>
</tr>
<tr>
<td>C7</td>
<td>0.0</td>
<td>1/2/0/1</td>
<td>1.00486</td>
<td>8.55</td>
</tr>
</tbody>
</table>

The ejection of rod (C7) has been simulated for two initial states (Hot Zero Power (HZP) and 30% power) and two delayed neutron fractions (one realistic value (= 520 pcm) and one lowered by 20% (= 416 pcm)). This gives the following combinations:
- HZP, $\beta_{eff}$ = 520 pcm
- HZP, $\beta_{eff}$ = 416 pcm
- 30% Power, $\beta_{eff}$ = 520 pcm
- 30% Power, $\beta_{eff}$ = 416 pcm

5. Influence of the Initial Power Level

For the HZP, $\beta_{eff}$ = 520 pcm case, core power, neutron balance and maximum fuel enthalpy are presented in Fig. 2 to Fig. 4 and from Fig. 5 to Fig. 7 for the 30% power, $\beta_{eff}$ =
520 pcm case.

Figure 2. Rod ejection in 100ms, HZP, $\beta_{\text{eff}} = 520$ pcm, Core power

Figure 3. Rod ejection in 100ms, HZP, $\beta_{\text{eff}} = 520$ pcm, Neutron balance

Figure 4. Rod ejection in 100ms, HZP, $\beta_{\text{eff}} = 520$ pcm, Maximum fuel enthalpy
Figure 5. Rod ejection in 100ms, 30 %Power, $\beta_{eff} = 520$ pcm, Core power

Figure 6. Rod ejection in 100ms, 30 %Power, $\beta_{eff} = 520$ pcm, neutron balance

Figure 7. Rod ejection in 100ms, 30 %Power, $\beta_{eff} = 520$ pcm, Maximum fuel enthalpy

One can see that the effect of fuel enthalpy appears much faster in the 30% power case than in the HZP case. This is due to the low initial power level at HZP which induces no thermal-hydraulic response. Because of the delay in the fuel enthalpy response (and thus the fuel temperature), the equilibrium state is reached later in the HZP case. In addition to the power time history, the axial power distribution in THC #99 (ejected rod) and in neighboring channels were studied. For the HZP initial state, the power profile has a peak in the upper core region. The 30% power initial axial distribution is symmetrical.
During the brief period of control ejection (0.1s) no substantial modification of the axial power profiles can develop. At the end of the transient (t=26s), the axial power profiles remain qualitatively the same as during steady state, although the power level is higher.

For the HZP case, the maximum fuel enthalpy and maximum enthalpy rise in function of the local burn-up is shown in Fig. 8 and Fig. 9. Both variables are plotted as ordinate while the abscissa values correspond to the sequential numbers of the fuel in the reactor core.

The maximum fuel enthalpy is reached in fuel assembly (B7). The value for this position is slightly higher than in fuel assembly (C7), where the control rod is ejected. The same observation can be made for the maximum fuel enthalpy rise. This result indicates that the maximum enthalpy does not automatically occur in the fuel element with the ejected rod but can occur in an adjacent fuel assembly.

Fig. 10 also shows the dependence of enthalpy rise on burnup. The fuel elements form three burnup groups corresponding to the operating time of the fuel (Fresh fuel, 2nd cycle, etc). The figure shows that the enthalpy rise decreases uniformly with the burn-up.

For the 30% power case, the initial fuel enthalpy is higher than at HZP which is expected since the fuel temperature is higher. Higher values for the enthalpy rise can also be observed in Fig. 11. The justification for this behavior results from one basic property of the
Doppler coefficient. Finnemann (1992) gives Eq. (1) as an approximate formula for the fuel temperature dependence of the Doppler reactivity coefficient.

\[
(1) \text{Doppler coefficient} (T) = Cst \times (\sqrt{T} - \sqrt{T_0}) \text{ with } T \text{ and } T_0 \text{ in } [\text{K}]
\]

It follows that the effect of delta T is smaller at higher temperatures than at low fuel temperature i.e. in 30% load conditions, the fuel temperature is higher and therefore a larger fuel temperature increase is required to compensate the positive reactivity introduced by the rod ejection.

![Figure 10. Rod ejection in 100 ms, HZP, $\beta_{\text{eff}} = 520$ pcm, Fuel enthalpy rise as function of burnup](image)

![Figure 11. Rod ejection in 100 ms, 30% Power, $\beta_{\text{eff}} = 520$ pcm, $\Delta$Fuel enthalpy (T0 – Tend) and burnup](image)

6. Influence of the Delayed Neutron Fraction

The use of a lower $\beta_{\text{eff}}$ increases the relative reactivity insertion and thus results in a significantly faster and higher power peak. In the present case, it leads to prompt criticality ($416 < 434$pcm) whereas the reactivity insertion stays below the realistic $\beta_{\text{eff}}$ value.

As Fig. 12 shows, for the HZP case, the prompt critical transient has a completely different course of event. In the 30% power case however, the differences are only quantitative (Fig. 14).

At the end of the transient, a new equilibrium state is reached in which the power level is higher than the initial level. By definition, at equilibrium the reactivity is back to 0. The positive reactivity introduced by the rod ejection must be compensated by negative feedbacks, here almost exclusively fuel temperature coefficient. Since the absolute
reactivity insertion is the same, the power level converges toward the same value and so does the mean fuel temperature/fuel enthalpy (Fig. 13 and Fig. 15).

Figure 12. Rod ejection in 100 ms, HZP, Core power, Comparison of two delayed neutron fractions

Figure 13. Rod ejection in 100 ms, HZP, Maximum enthalpy, Comparison of two delayed neutron fractions

Figure 14. Rod ejection in 100 ms, 30% Power, Core power, Comparison of two delayed neutron fractions
7. Influence of the Fuel Rod Model

The exact calculation of the temperature distribution in fuel rod depends on:
- The heat transfer conditions at the cladding surface
- The heat transfer value and the specific heat of the cladding material and of the fuel pellet
- The heat transfer conditions at the gas gap, or in some cases in the contact zone between pellet and cladding.

The fuel rod burnup has an influence on all those parameters by inducing cladding oxidation, structure changes and cracking of the pellet as well as by changing the composition of the gas gap or by the formation of direct contact between pellet and cladding. The described effects can either be calculated or given as input in the available ATHLET fuel rod model.

ATHLET gives ring wise fuel temperature (the number of ring is given as an input parameter). In this study 4 rings are considered. The neutron kinetic model uses only one so-called effective “Doppler temperature” to account for the fuel temperature effect. Recent studies have shown the strong influence of the “Doppler temperature” definition on the REA course of event (Grandi, 2010). Usually this “Doppler temperature” is taken as a weighting of the fuel pellet surface and the central fuel temperature (with T in K)

\[ T_{\text{Doppler}} = 0.7 \times T_{\text{surface}} + 0.3 \times T_{\text{Center}} \]  

A modification of the weighting factors for example 0.5/0.5 leads in the HZP case to a postponement and a change in the maximum peak power, but has little influence on the maximum fuel element enthalpy.

As an example of a variation in the determination of the fuel rod temperature variation of the heat transfer coefficient is seen at the gas gap. The default value of 1.0x10+4 is reduced by a factor of 10. In the HZP case, the variation does not affect the initial state. There are however significantly higher values for the internal fuel enthalpy and especially for the enthalpy rise.

In the 30% power case, there is already a significant increase in the average fuel temperature of about 150 K and therefore a higher fuel enthalpy in the initial state. The average fuel temperature for PWRs at nominal power is in the range of 820 K to 920 K. A comparison of the time functions for the fuel enthalpy is shown in Fig. 16. The peak fuel enthalpy reaches a higher value. First, because the initial fuel temperature level is higher and then because the fuel temperature feedback decreases as it has already been discussed.
Accordingly, the peak power reached is also higher (Fig. 17).

![Figure 16. Rod ejection in 100 ms, 30% Power, Core power, Comparison of two gas gap HTC](image1)

![Figure 17. Rod ejection in 100 ms, 30% Power, Maximum enthalpy, Comparison of two gas gap HTC](image2)

8. Local Delayed Neutron Parameters

To study the dependence of local values of delayed neutron parameters, the rod ejection transient benchmark in the core configuration of PWR UO2/MOX is investigated (Kozlowski, 2006). The core configuration is shown in Fig. 18. It contains UO2 and MOX fuel assemblies with a wide range of different burnups and different enrichments for UO2 and MOX assemblies. In (Klein et al, 2011) a two-group cross section library is produced by the NEWT from SCALE - 6. This library contains $\beta$ and $\lambda$ values for all fuel assemblies with a full dependence on TH parameter and burnup (which is the main parameter influencing $\beta$ and $\lambda$). To be able to treat this local information, the modified version of system code QUABOX-CUBBOX is prepared.

In the ATHLET core model for the thermal hydraulics part, every assembly is treated by a representative channel with an axial power distribution as calculated by QUABOX-CUBBOX. The thermo-physical data for UO2 and MOX are taken from the benchmark specification. All channels are connected to a lower plenum and upper plenum volume. There is no cross flow assumed between assembly channels.

According to the benchmark specification, the rod at the position E5 (see Fig. 18) is assumed to be fully ejected in 0.1 seconds after which no reactor scram is considered. The
control rod ejection is to be performed from HZP, All Control Banks In, All Shutdown Banks Out, critical boron concentration with the highest worth rod at such condition. During the entire calculation the boron concentration and the position of the other control rods are assumed to be constant. On Fig. 19 the comparison of the total core power for the cases with global and local delayed neutron parameters is shown. In the case of the local values of delayed neutron parameters, the peak comes earlier and has lower maximum.

9. Conclusion

The results from the coupled calculations of the control rod ejection in PWR can be summarized as follows:

- The rod worth has a determining influence. First, because it can lead to prompt criticality, and then because this positive reactivity must be compensated by negative reactivity feedback, i.e. an increase in the fuel temperature.

- The neutron kinetic parameters such as $\beta_{\text{eff}}$ affect the time course of the power peak, but not significantly the maximum fuel enthalpy and enthalpy rise.

- The fuel assemblies in the vicinity of the control rod position influence the reactivity insertion but also the radial power redistribution caused by the ejection. The power distribution in the final state of the transient is in agreement with the steady-state calculation.

- The initial fuel temperature level has a strong influence on the Doppler feedback coefficient. The Doppler feedback value decreases when the fuel temperature increases. As
a result, to compensate the same reactivity insertion, in HZP conditions (i.e. with low temperatures) the fuel temperature increase is lower than at 30% initial power (i.e. higher initial temperature).

- The calculations show that the maximum fuel enthalpy or the maximum fuel enthalpy rise can occur not only in the affected fuel element but also in its vicinity.
- The fuel rod model parameters that determine fuel temperature have a strong influence on the fuel enthalpy and enthalpy rise.
- Local values of delayed neutron parameters influence the maximum value and time of the total core power peak.

The next step of this project will be a quantitative uncertainty analysis, propagating the uncertainty from the nuclear data to the final transient results. For this purpose, the XSUSA tool, developed in GRS will be used.

References

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