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ABSTRACT

The decision to use mixed-oxide (MOX) fuel in PWR's involved re-investigation of a certain number of accidents and notably control rod ejection transients. It has thus been shown that this accident would be no more severe than in the case of all-uranium cores, since the positive effects on the ejected rod worth would counterbalance the negative effects on the delayed neutron fraction. A new approach to the kinetics aspect of the calculation method for this accident is also presented, involving a 3-D kinetic calculation with only a few axial meshes.

INTRODUCTION

In 1985, EDF decided to recycle plutonium derived from fuel reprocessing in the French Pressurized Water Reactors (PWR). The recycling ratio presently adopted is 30% and the first sixteen MOX assemblies have been loaded into St. Laurent B1, which went critical in November 1987 and has been operating at full power since December of that year.

Various accidents the consequences of which could be exacerbated by the presence of plutonium in the reloads, have now to be examined.

This is the case in a control rod ejection accident, where failure of the mechanism pressure housing ejects a control rod to fully withdrawn position. The resulting reactivity insertion causes a violent power transient with severe power peaking in the vicinity of the ejected rod position. The greater the ejected rod worth and the smaller the effective delayed neutron fraction βeff, the higher will be the overpower in this area. The power excursion will be physically limited by Doppler and moderator feedbacks related to the fuel temperature increase, before scram occurs after a few seconds. It must be demonstrated that clad integrity and fuel behaviour in these circumstances remain within the limits defined by the safety authorities.

The use of plutonium decreases both neutron fraction and rod worth. The first of these effects tends to heighten the seriousness of the accident whilst the second mitigates its consequences and it was difficult to determine without further investigation which was the prevailing influence. It was consequently decided that specific studies were necessary and these were performed jointly by EDF and CEA.

CALCULATION METHOD

Presentation

The main codes involved are as follows:

APOLLO (1) : transport code, generating the cross section data libraries used for diffusion calculations.

CRONOS (2) : finite element diffusion code suitable for 1, 2 or 3 D calculations. This code also has a thermal hydraulic model for feedback effects and can solve neutron kinetic problems.

FLICA (3) : code for steady-state and transient thermal hydraulic calculation of 1 or 2-phase fluid circulating in sets of channels coupled or not.

Using 3-D kinetic reference calculations, a simplified method was developed for the calculation of control rod ejection accidents at zero power, considered the most penalizing with regard to uranium fuel management. This simplified calculation method has been compared with the reference method and its conservativeness analyzed and checked (4). It has in fact been widely used for all-uranium cores. It comprises three steps.

First of all, 3-D static calculations are performed before and after control rod ejection to establish the variables characterizing the accident, the ejected rod worth, the hot channel power distribution factor F_Q. These calculations are performed with precise parabolic or cubic finite elements and a fine mesh (about twenty axial meshes and from one to four radial slices per assembly). In addition, static calculations are performed for feedback modelling (see next paragraph).

The next step consists of a 1-D kinetic calculation to determine average core power evolution P(t). The ejected rod worth used in this calculation is that determined in the previous step.

Finally, in the third and last step, the hot channel thermal evolution calculations are performed, assuming the previously calculated F_Q factor to apply throughout the transient and the power release in the hot channel to follow the P(t) law determined during the previous step. We thus obtain $F_Q P(t)$ as the hot spot energy level. This calculation provides clad and fuel temperature data, which has to be compared with the authorized limits.

Feedback modelling

A rod ejection accident involves a reactivity insertion causing severe power peaking in the vicinity of the ejected rod. The power transient is limited by feedback effects and the more the flux distribution is distorted, the stronger will be the feedback effects, because maximum flux increase occurs where the temperature rise is highest. An axial representation of the reactor under these conditions will necessarily include feedback and flux redistribution effects, which depend on the mean temperature corresponding to each step in the transient.

The moderator temperature is conservatively assumed to be constant throughout the accident for core mean power distribution analysis . Only Doppler feedbacks have consequently to be modelled.

Spatial flux distributions under static and dynamic conditions are assumed to be the same, providing the average fuel temperatures remain the same. On the basis of 3-D calculations performed at various power levels, an average cross section library is constituted for each core slice (Figure 1). Reflector leakages are calculated in each axial slice, in order to be able to keep the 3-D calculation axial flux and eigenvalues.

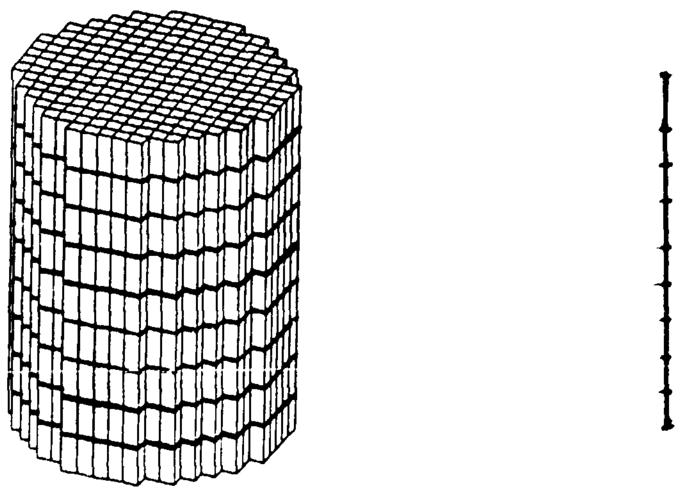

Fig. 1. Constitution of axial cross section libraries by 3-D - 1-D condensing.

In slice k, the mean macroscopic cross section for a given energy group is:

$$\Sigma\alpha_{\cdot}(\overline{Tk}) = \frac{\sum_{i,j,k} \Phi_{ijk}\, \Sigma\alpha_{ijk}\ (T_{ijk})}{\sum_{i,j,k} \Phi_{ijk}}$$

$\overline{Tk}$: mean fuel temperature in slice k
Tijk : temperature in slice k for assembly ij
Φijk : flux integrated on mesh ijk
α : type of cross section (absorption, fission, etc.)
$\Sigma\alpha$ijk : macroscopic cross section in mesh ijk, depending on Tijk

The condensing is performed on the basis of static 3-D calculations performed at different power levels, keeping the moderator temperature identical to that before rod ejection. This gives, for slice k, tabulated results for different power levels, which are themselves dependent on different mean temperatures of the slice Tkp.

During the transient, the cross section $\Sigma_{\alpha k}$ $(T_k\ (t))$ is linearly interpolated between $\Sigma_{\alpha k}(T_{kp-1})$ and $\Sigma_{\alpha k}(T_{kp})$ for $T_{kp-1} < T_k(t) < T_{kp}$. This method gives a conservative value for the reactivity (i.e. the reactivity is overestimated) because the mean temperature depends non-linearly on the Keff.

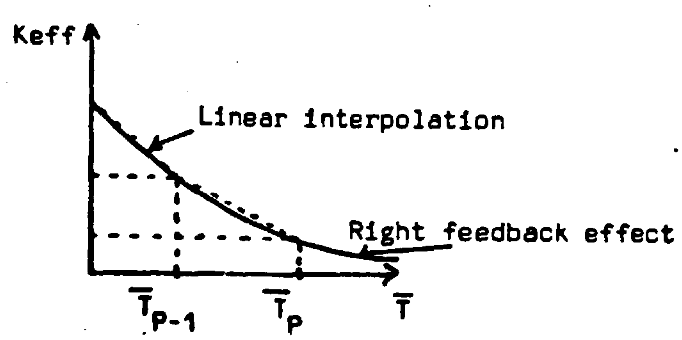

Fig. 2. Example of the conservativeness regarding the Doppler effect.

ANALYSIS OF A ROD EJECTION ACCIDENT IN A 30% MOX CORE

Presentation of the analysis

Rod ejection accident analysis is included in the specific refuelling calculations. The main purpose of this analysis is to determine whether the consequences of this accident would be more severe in the case of a mixed oxide core.

The analysis was consequently carried out on a realistic representative case, consisting of an equilibrium cycle in a 900 MW PWR comprising 30% MOX assemblies and corresponding to an annual 3-batch cycle. Figure 3 shows the MOX assembly (mean plutonium content: 5.3%) and its zoning. The cycle of this assembly is equivalent in length to that of the 3.5% enriched uranium assemblies. The refuelling schedule together with the position of the C and D rod banks is shown in Figure 4.

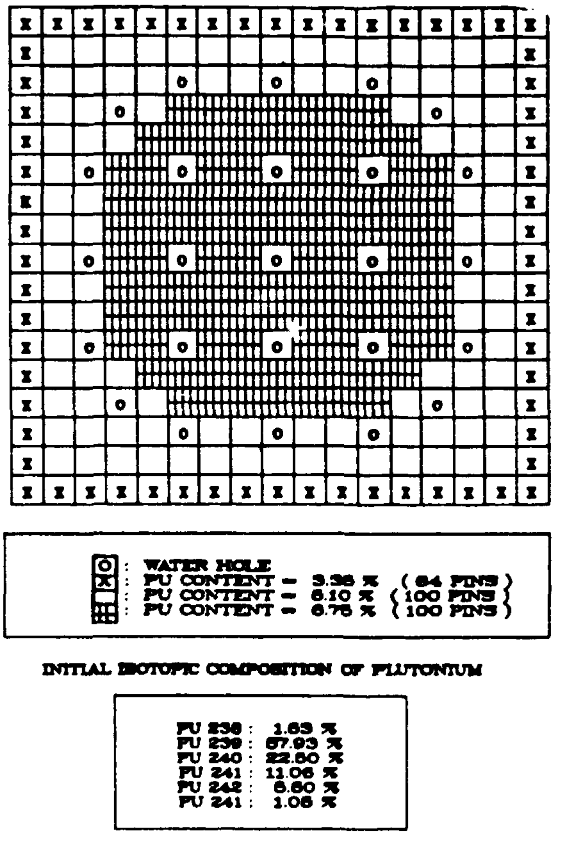

Fig. 3. Assembly zoning and plutonium distribution

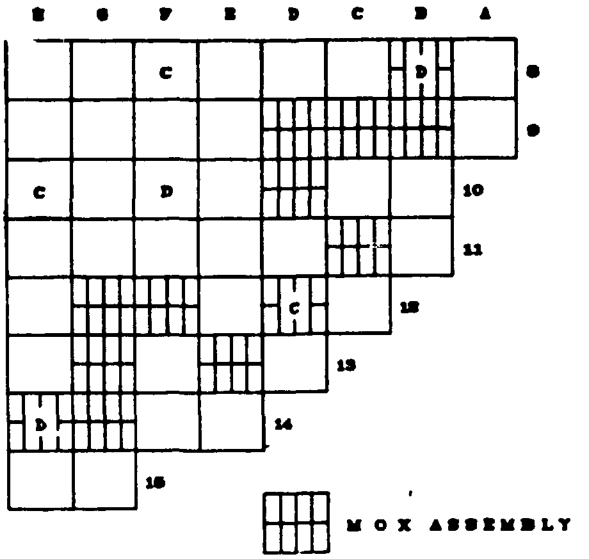

Fig. 4. Loading pattern

Of the four situations which have to be considered in the context of the safety analysis report (beginning and end of cycle at 0 and 100% PN) the most penalizing proved to be the case of end of life at zero power. In this case, the D rod bank is inserted up to 80% of the active length and the C rod bank up to 25%. The ejected rod with highest worth is B8, which belongs to rod bank D and is worth 0.680 percent, which, given the value of the delayed neutron fraction $\beta eff = 0.00480$ corresponds to a rod worth of 1.42 dollars.

If the most severe scenario is the same as for all-uranium cores, it should be noted, on the other hand, that the values of ρ and β, which are in the vicinity of 0.750 percent and 0.00550 for all-uranium cores, are significantly lower for the MOX cores. (The ρ/β ratio is also in the vicinity of 1.4 dollars).

Results of the analysis

This analysis was undertaken on the basis of a simplified method using an axial kinetic calculation, modelling an instantaneous ejected rod worth of 0.680 percent. It was shown that the speed with which the rod was ejected has no effect on the accident sequences, its only consequence being to displace the time origin.

As can be seen on Figure 5, a maximum power peak corresponding to about 8 times the core P_N occurs at t = 0.09s. This is followed by a sharp power drop, due to the Doppler effect, which is included in the calculation via axial cross section libraries, constituted according to the methodology previously described. Scram takes place after 3 seconds (penalizing time lag), at which time the negative reactivity is 3.400 percent for N-2 rods, the rod with the highest worth of the N-1 others being assumed to be jammed in fully withdrawn position. The axial kinetic calculation model corresponds to 4 seconds.

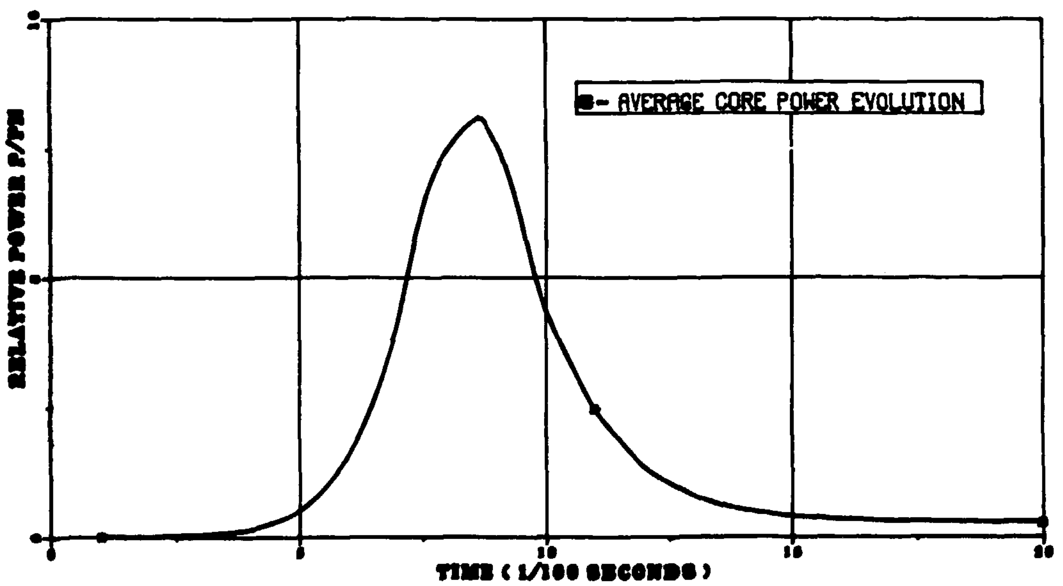

Fig. 5. Nuclear power for 1-D neutron kinetics 0-0.2s

The hot spot factor F_Q = 17.5 was also determined for the MOX fuel assembly B8 from which the rod had been ejected. This value of 17.5 makes allowances for calculation uncertainties. Using this F_Q value and the data obtained on mean core power distribution evolution versus time, we can then perform thermal hydraulic calculations to obtain the clad and fuel temperatures. The results, given in Figure 6, show that clad dryout occurs at about 0.5s and that the clad temperature, which is the limiting parameter for this type of accident, reaches 960°C at about 2.8s, after which it decreases. We are clearly very far removed from the limit value of 1482°C.

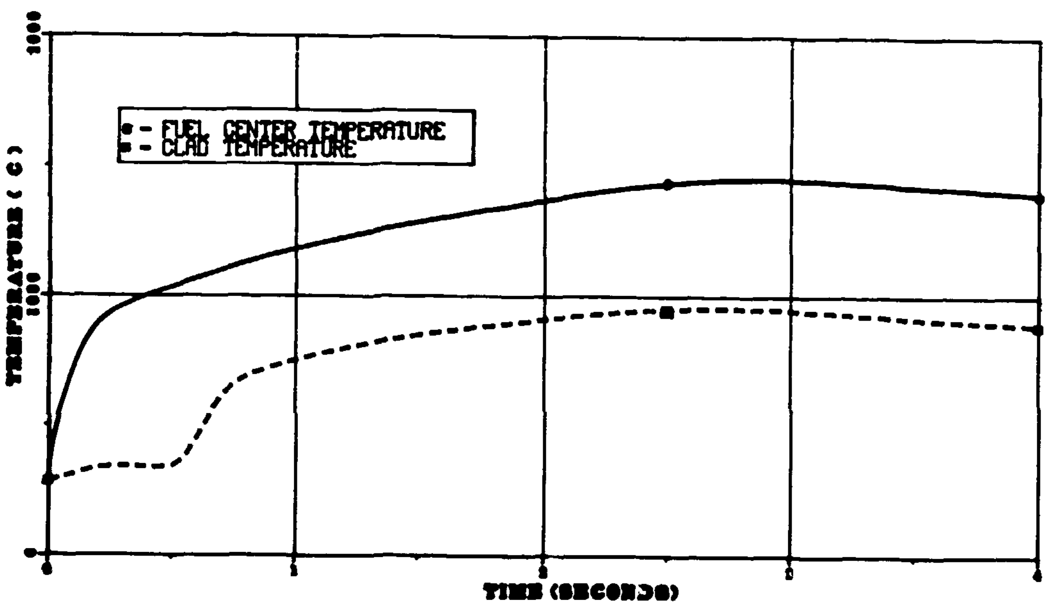

Fig. 6. Hot spot temperature versus time

Comparison with an all-uranium core

The main observation is that the specific properties of plutonium, for the 30% plutonium content adopted, do not significantly affect the sequences or the violence of the accident.

It should also be noted that the main problem encountered concerns the static calculations undertaken to assess the hot spot factor. The code used, CRONOS (2), is based on representation of homogeneous assemblies. The fine structure superposition method was thus used to calculate the hot spot. The fact that the assemblies are of the mixed oxide type then raises a further two-fold problem: first, the uranium-plutonium interface implies systematic use of 4 meshes per assembly and, secondly, the fine structure peak, calculated in an infinite medium is higher because of the zoning and necessitates extremely thorough transport calculations.

Finally, as regards the physical properties of the plutonium rod, the difference in thermal conductivity has practically no incidence on temperature evolution.

ELABORATION OF A SIMPLIFIED METHOD USING 3-D KINETIC CALCULATIONS WITH A SMALL MESH NUMBER

Advantages of the method

The simplified methodology developed uses a 1-D kinetic calculation, which has the drawback of involving a large number of manual stages. Moreover, it is specifically designed for zero power cases and is not easy to transpose to other power levels. It was consequently decided to investigate a new approach, where the kinetic part, corresponding to the second step, of the simplified method would be performed, not with 1-D, but with 3-D calculations, using a coarse mesh. The other steps remain as before.

The main advantages of this new approach reside in the fact that 3-D geometry is kept for the kinetic calculations, with the greater flexibility which this implies, since different cases can be studied simply by modifying geometrical input, and in the compatibility of the method with all power levels, integrating moderator and Doppler feedbacks.

Moreover, with a view to further reducing costs, we opted for a linear finite element system.

Incidence of the axial mesh option

The calculation results presented correspond to the zero power, end of cycle ejection of rod B8, where the reactivity is 1.22 dollars for a βeff of 0.00480, the irradiation distribution differing from that of the previous case.

Several axial slices were investigated, ranging from a coarse 4-slice model of the fuel zone to a precise 14-slice représentation. Results obtained over a period of 0.2s are shown in Figure 7.

The 4-slice axial mesh provides an approximate picture of core power evolution, which it strongly overestimates throughout the transient, reaching a peak of 4.8 P_N 0.13s after ejection of rod B8.

It is interesting to note the consistency between the 8, 12 and 14 fuel slice meshes, both as regards evolution of the transient and the power peak value obtained. The power ratio P/P_N for this series of results is of the order of 3 and also corresponds to 0.13s.

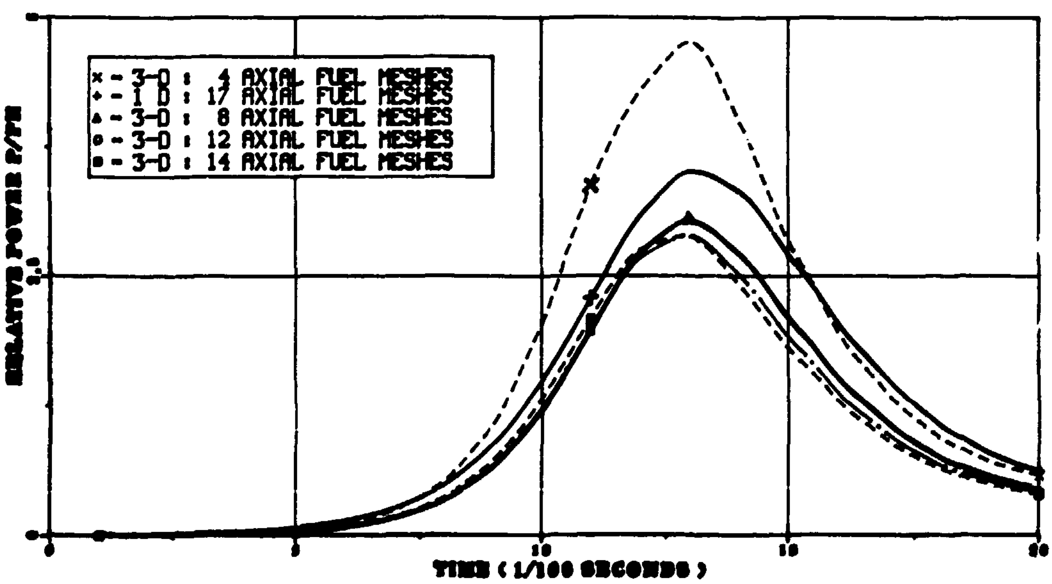

Fig. 7. Average core power evolution (reactivity = 0.580 percent)
3-D and 1-D calculation results

This figure is a perfect illustration of the conservativeness associated with use of a coarse mesh. The reason for this result if that there is a lower Doppler feedback with an axial power distribution which is underestimated owing to the small number of meshes. The temperature of the hottest mesh will consequently be below the real temperature.

Comparison between the conservatism of simplified methods with axial kinetics and with 3-D kinetics and few meshes.

Figure 7 shows power evolutions calculated with the 4, 8 and 14 slice 3-D kinetic model and those obtained with the 1-D kinetic model. The 1-D model gives a maximum power peak of 3.5 P_N at 0.13s, which is an intermediate result between those obtained with the 4 and 8 slice 3-D kinetic model.

Another important element to be considered in these comparisons is the computer time requirements. For information, we give below a few examples of calculation time requirements on an IBM 3090-200 computer. They correspond to the modelling of 3 seconds of the rod ejection transient.

Type of calculation	Calculation Times	
	Static calculations (step 1)	Kinetic calculations (step 2)
3-D 4 slices	20	10
3-D 8 slices	20	20
3-D 14 slices	20	36
1-D 17 slices	40	1

Table I: calculation time (in minutes)

The static calculations necessary for 1-D type calculations correspond to a longer static calculation time because, apart from the ρ and F_Q calculations, additional calculations have to be performed to model the Doppler feedback.

Considering the figures obtained, the 8 slice 3-D method is found to display a conservativeness comparable to that obtained with the 1-D method, for an equivalent calculation cost, although the longer engineer time requirements for preparation of the 1-D calculations should not be overlooked. Finally, 3-D modelling of moderator feedbacks is perfectly straightforward, which is an added advantage for analysis of rod ejection at intermediate power levels.

Conclusion

Analysis of a rod ejection accident on a representative PWR core comprising 30% MOX assemblies showed that this accident had no consequences beyond the limits defined by the safety authorities, since the clad temperature did not even reach 1000°C.

It is thus inferred that the specific characteristics of plutonium, which result in lower delayed neutron fractions and rod worths, do not significantly alter the sequences of the transient nor the violence of the accident.

As regards the physical properties of the plutonium rod, the difference in thermal conductivity induces only slight modifications.

As for the methodology developed to analyze this accident in all-uranium cores, its transposition to mixed oxide cores raised no particular problems. However, it should be noted that it is a little more complicated to obtain the hot spot factor F_Q, because the homogeneous calculation has to be performed with a minimum of 4 slices for each assembly to cover representation of the uranium - plutonium interfaces, and that the fine structure assessment, which is much greater, also requires a high degree of accuracy.

Furthermore, 3-D calculations with only a few meshes greatly enhance computation flexibility, both at the preparation stages and as regards transposition for other power levels. This would appear to imply that it should be possible to define margins with respect to the conventional methods described in the safety analysis report, which would enable us to use parametric studies to get a better understanding of the most penalizing case for given core management and reactor control systems.

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