FRAGEMA Modelling of PWR Fuel Rod Behaviour during Transients

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1 INTRODUCTION

The fuel rod of the PWRs is subjected to complex physical, mechanical and chemical phenomena. FRAGEMA pays specific attention to these in order to gain a better knowledge of them to deduce some models which, when integrated into computer codes, allow a prediction of the fuel rod behaviour during their in reactor irradiation.

Amongst these phenomena, Pellet Cladding Interaction (PCI) and SCC are examined particularly closely as these have been identified as the major causes of damage to the fuel rods and of any decrease in the reactor performances arising from a reduction in their manoeuvring capacity when they operate in load follow or frequency control mode, leading to possible cladding failure during incidental power transients. In order to estimate this risk, FRAGEMA has, on the one hand undertaken some experimental programs and, on the other developed a set of models for the fuel rod behaviour during transients.

These models use :

- structural mechanics, in order to calculate the stress and strains to which the cladding is subjected during the power transients,
- the SCC, in order to estimate the Zircaloy damage caused by the presence of corrosive fission products,
- the mechanics, in order to evaluate the average lifespan of a fuel rod when subjected to severe power transients in the reactor.

2 THE MECHANICAL MODELLING INGREDIENTS

2.1 Initial condition before power transients :

The prediction of the initial conditions, especially the fuel rod geometry and chemical inventory of its inner atmosphere due to the time based power evolution, involves mainly the following models : thermal calculation, fission gas release, uranium oxide densification and swelling, cladding creep down, external corrosion. An experimental data base has been carried out to study

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these phenomena, this data base includes the results of measurements obtained from the examinations of more than 150 fuel rods irradiated in commercial or experimental reactors and whose maximum burnup is greater than 60 MWd/kgU [1].

2.2 Transient stress modelling :

The stress during power transients are computed with the finite element code SYSTUS, with a 2D axisymetric geometry. The model takes into account the pellet hour glassing shape at the interface betwen the pellets. The evolution of the stresses depends on the initial geometry especially on the pellet to cladding gap and also of the power level at which the fuel rod was operated. Figure 1 shows the stress evolution, calculated by this method, versus the LHGR, in the cladding inner surface, in front of the interpellet gap. Some improvements are made to give a more realistic idea of the clad behaviour. Amongst these improvements, a 3D model makes it possible to take into account the axial and radial cracks in the pellet. In addition the introduction of an elastic-plastic law of behaviour for the uranium dioxide, and a visco-plastic one for the zircaloy cladding make it possible to minimise the computed stresses. The creep law, especially developed for zircaloy tubes and valuable at high stress level, which is applied to the visco-plastic modelling is formulated as follows :

$$\varepsilon = A\ (T)\ {}^{n}\ t^{m}\ \varphi^{p} \quad (1)$$

where ε is the strain, A (T) a temperature dependant function, T the temperature, σ the stress, t the time and φ the neutron flux, m and p are experimental constants, n a function of σ.

2.3 Experimental qualification of the model :

An experimental data base more especially oriented towards the fuel rod behaviour studied during transient regimes has been established and includes the results :

- of power ramping experiments performed on more than 230 rods of various characteristics, in the framework of R & D programs, in cooperation with CEA and EDF [2], in international programs, or coming from open literature, the main characteristics of this data base concerning PCI are indicated on Table I.

Table 1.Fragema PCI Data Base main characteristics.

Type of Design	Test Result	Burnup ($MWd\ kgU^{-1}$)	Irradiation LHGR (kWm^{-1})	Power Ramps Peak LHGR (kWm^{-1})	Hold time (min)
FRAGEMA	NF*	19 - 58	8 - 35	32 - 51	1 - 1654
	F**	21 - 41	13 - 21	45 - 47	20 - 30
Other	NF	17 - 4	8 - 27	31 - 67	0,43-4200
PWR	F	8 - 45	4 - 27	28 - 61	0,57-4320

* NF : non failed
** F : failed

- of analytical experiments carried out in experimental reactors on especially instrumented rodlets CANSAR [3] HATAC, CONTACT, FLOG [4]. The strains and stresses calculated by the modelling have been compared to the measured ones from the in pile experiments performed on short rods instrumented with strain gauges and thermocouples. The Figure 2 allows a comparison between estimated and measured values during a power ramp test of the CANSAR experiment. The qualification of the model including the improved law of behaviour is in progress using the FLOG experiments.

3 SCC MODELLING

The SCC rupture process is divided into two steps : the first one corresponding to the micro-crack initiation, the second one to the crack propagation. An initiation model and a propagation model have been carried out and calibrated from tube burst tests with iodine at 350°C, performed on non irradiated tubes some of them having artificial axial inner cracks [5]. The crack growth rate $\mathring{c}$ is formulated as follows :

$$\mathring{c} = f\ (I)\ \exp\ (-\ Q/RT)\ (\alpha_i\ \dot{c}_i + \alpha_p\ \dot{c}_p) \qquad (2)$$

where : f (I) is a function which depends on the corrosive fission product concentration (iodine) and has been determined by the Konashi method [6], Q, the activation energy, T the absolute temperature, R the perfect gas constant, and with ($\alpha_i = 1$, $\alpha_p = 0$) during the crack initiation period, and ($\alpha_i = 0$, $\alpha_p = 1$) during the propagation period. The subscript i refers to initiation whereas p refers to propagation. The crack growth rate during propagation is given by the following equation :

$$\mathring{c}_p = g\ (K_I) \qquad (3)$$

where g is a function of the stress intensity factor K_I given by : $K_I = a\sigma\sqrt{\pi c_p}$,

where a is the influence factor of the crack and depends only on the geometry, the average stress in the cladding outside the crack and c the crack depth.

The crack initiation rate c_i is deduced from burst tests using a method established by Mrs I. Schuster [7]. Figure 3 shows the initiation time compared to the total lifespan versus the applied hoop stress. Some identical tests on irradiated tubes are in progress and will make it possible to quantify and predict the damaging effect of neutrons and recoil fragments. In addition, with a transient fission gas release model we will be able to calculate in the best way the iodine evolution with time.

4 SIMPLIFIED MECHANICAL MODELLING

In order to be able to analyse all the situations which are anticipated in a reactor it has been necessary to develop a simplified model for stress computation, from the 2D method in the previous paragraphs. The local stress depends on the local FEM calculation $\Delta P = P - P_{ref}$, as on the local initial geometry through the value of an equivalent cladding stress P_{eq}, the equation which relates all these parameters is the following :

$$\sigma = \sigma_{eq} + \alpha\ \Delta P \qquad (4)$$

where : α depends on the geometry and the pellet nature, P is the maximum instantaneous LHGR, P_{ref} the reference LHGR which corresponds to a thermomechanical equilibrium between the pellet and the cladding. The reference LHGR P_{ref} is directly related to the cold gap J_F, which is measured in laboratory conditions. Figure 4 shows the cold gap evolution versus burnup for a typical FRAGEMA fuel rod design irradiated in a commercial reactor. The modelling application to the power range tests has made it possible to define a stress rupture threshold, which depends on the fuel burnup (Figure 5).

5 CONCLUSION

The data base related to irradiated fuel, established by FRAGEMA, made an improvement in the fuel rod behaviour modelling under possible irradiation conditions. The use of structural mechanics as well as the use of SCC and fracture mechanics concepts establishes suitable fuel rod models for PCI analysis. These models have been calibrated from in pile experiments. A 3D model as well as the use of visco-plastic law will make significant improvements in the fuel rod thermomechanical modelling during power transients. Some burst tests on irradiated cladding tubes as well as the use of a transient fission gas release model will allow the best possible prediction of cladding damage caused by SCC and the corresponding lifespan of a rod. Some analytical experiments will make it possible to calibrate all these models in the future.

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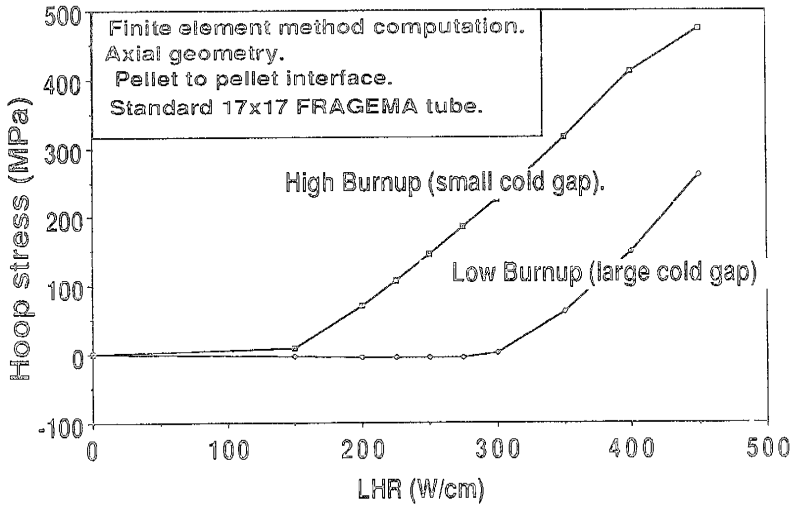

Fig. 1 Typical Evolution of Hoop Stress in FRAGEMA cladding Fuel Rod vs. LHGR at various Burnup.

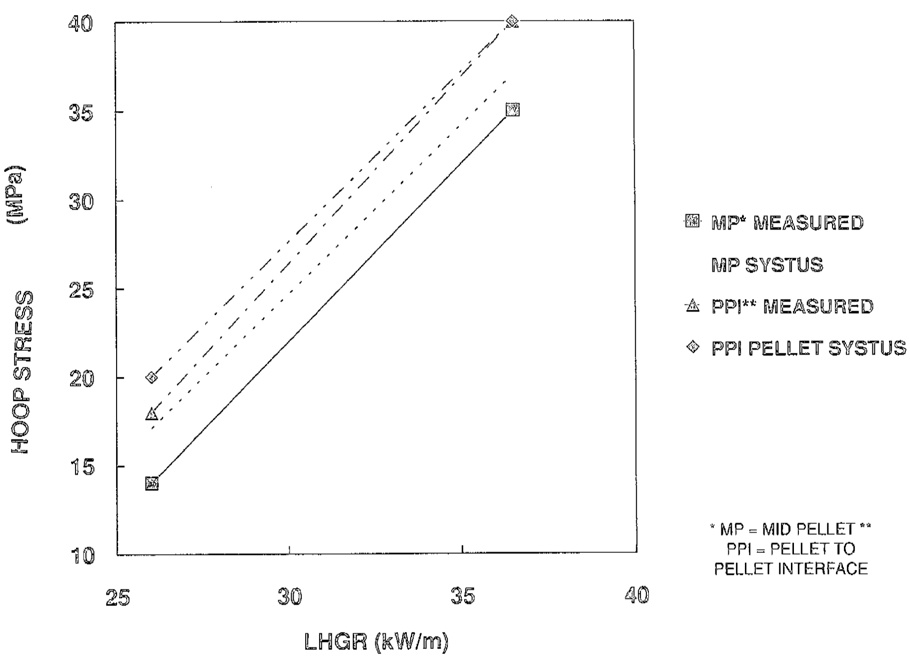

Fig. 2 CANSAR Experiments. Comparaison of the Computed and Measured Hoop Stress.

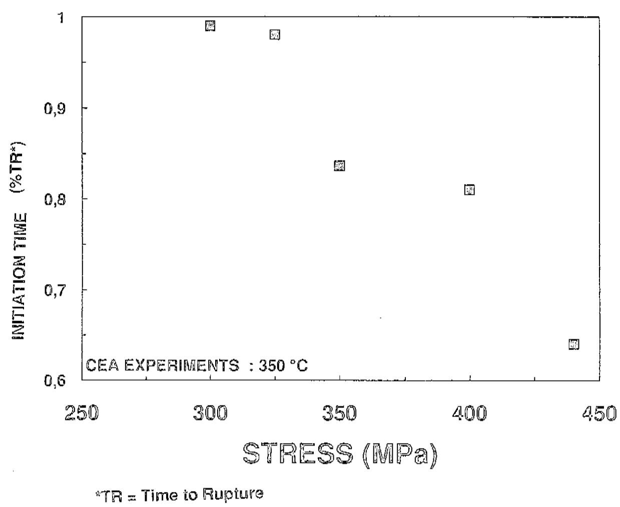

Fig. 3 FRAGEMA SCC Modelling. Evaluation of Initiation Span vs. Hoop Stress.

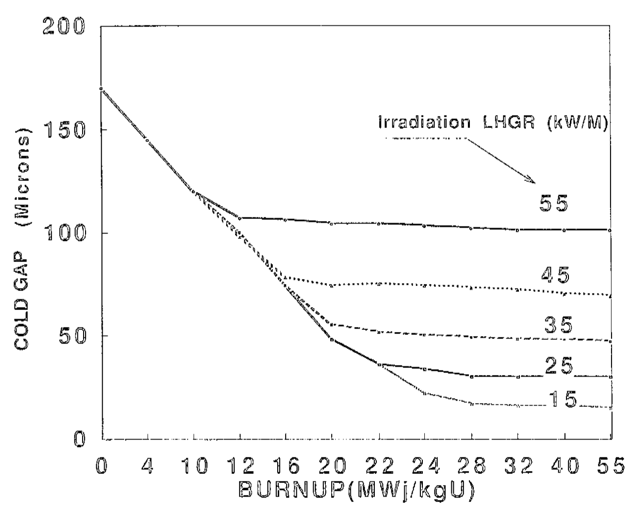

Fig. 4 Typical Evolution of Cold Gap vs. Burnup for the Standard FRAGEMA Rods at Various LHGR.