The Carbide Version of the Integral Fuel Rod Code URANUS

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Abstract:

The integral fuel rod code URANUS (see Lassmann /1/), developed for the structural analysis of light water- and fast breeder-reactor rods, has been extended to include a carbide version for use with carbide fuel-fast breeder rods.

All the relevant material properties for UC and (U,Pu)C were taken into account: Thermal expansion, thermal conductivity, specific heat, Young's modulus, Poisson's ratio, yield stress, tensile strength, breaking elongation, specific weight, creep properties, swelling and fission gas release. With the standard routines of URANUS for calculating temperatures, geometry in hot state, stresses, strains, fracture mechanics and plasticity, density, pressure, heat transition in the gap, power form factor and burn-up, and with the axial coupling of individual sections, a detailed description of the integral rod is hence available for precalculation and interpretation of carbide rod experiments.

Material data was extracted from published literature and after careful comparison was assessed and correlated. The models for solid body and gas bubble swelling and for fission gas release are newly developed and calibrated to accessible data. The swelling-model is dependent on temperature and burn-up and takes both solid body and gas bubble swelling into account. The model differentiates between unrestrained and restrained swelling due to contact/gap cases; porosity changes and hot pressing are taken into account.

The model for fission gas release is written in terms of the local volume power, temperature and burn-up. The inital saturation of the fuel matrix with fission-gas atoms is also taken into account and hence the time of initial fission gas release determined due to breakaway.

This newly developed version of URANUS was used to run calculations of experiments with both simple and complicated power histories, and to high burn-up values; in doing so, good consistency to the experimental results was obtained. The newly developed models proved good although some more details have still to be worked out.

However, a sufficient and fast running URANUS-version is now available to carry out calculations for experiment-analysis and design of advanced fuel, allowing detailed thermal and mechanical investigations.

1. Introduction

Due to the many parameters which have to be taken into consideration, a sufficient thermal and mechanical structural analysis of fuel rods of a nuclear reactor can only be carried out using fast running fuel element performance codes applied with capable large-scale computers. At present, besides some special programmes to investigate local effects, some 20 extensive fuel rod codes are internationally known. To them belongs the integral fuel rod code URANUS, recently described in several publications (e.g. Lassmann /1/).

As a first step to describe advanced fuel concepts with URANUS, a carbide-version was established. The particular combination of high density of metal with very good thermal conductivity allows the increase of specific volume power without fear of central fuel melting; nevertheless, higher breeding rates, higher burn-up and a more compact core are other advantages which promise better results in using advanced fuel in future breeder reactors. On the other hand, some problems still render advanced fuel more difficult (high swelling velocities, pellet-clad mechanical interaction). Compared to oxide fuel, experimental data for carbide fuel is rather scarce; transient behaviour is nearly unknown and well recommendated material laws to describe swelling, densification, restructuring and fission gas behaviour are still missing. Due to these not conclusive known material properties, the results of the first application of URANUS to carbide fuel experiments - see section 4 - mainly shows the capability of the URANUS-system rather than prove the accuracy of all detailed data brought in.

2. Structural analysis of fuel rods with URANUS

The URANUS-code is an advanced fuel rod design code with a variable multizone concept. In contrast to a 2-d or 3-d finite element solution, which needs extensive computer storage and which therefore can only describe small parts of a fuel rod, URANUS is based on the superposition of one dimensional radial and axial models. As a result of this concept, a very detailed description of the integral rod is achieved, considering the geometric structure, mechanical state, temperature distribution and all needed physical effects due to irradiation. Geometric nonlinearity (large displacements) are involved, the loads are considered as steady state or transient. The crack structure and the axial friction forces are calculated by means of an axial loop. A very detailed description was given by Lassmann /1/ recently. The URANUS code is applicable to every fuel rod type (LWR, LMFBR, HWR), and allows steady state, transient or accident condition investigations; main application is done with pre- and postcalculations of experimental rods.

3. Carbide fuel material laws

To establish the carbide version of the URANUS code, material data was extracted from published literature and after carful comparison was assesed and correlated. Compared to oxide fuel, results are rather scarce and often vary widely. Nevertheless, a sufficient description of carbide fuel behaviour is possible using the present data. Within the scope of this short presentation only some typical points and the newly developed models to describe swelling and fission gas behaviour will be discussed; the publication of the unshortened paper is in preparation.

3.1 Mechanical behaviour of carbide fuel

Carbide possesses a cubic face centered lattice of type NaCl. The mainly used Uranium-monocarbide UC with 4.8% carbon does not experience any lattice changes in solid state, the melting point lies around 2365 $^{\circ}C \pm 165$ (Nickerson et al /2/). The higher density of metal is a main advantage of carbide fuel compared to oxide, which leads to more compact cores and better breeding properties. The elastic and plastic behaviour is influenced by five main parameters: temperature, porosity, stoichiometry, grain size and alloy composition. For UC and (U,Pu)C no data are available which take all these parameters into account; most data is known only for room-temperature. However, the main values are known.

Padel et al /3/ describe the elastic modulus as temperature- and porosity dependent in the range 0-27% porosity. Their equations are comparable to those of Matthews /4/, who in addition takes the alloys into account. Matthews' equations were adopted for URANUS:

$$\text{UC} \quad : E\ [\text{MPa}] = 215000\cdot(1-2.30\cdot P)\cdot(1-0.92\cdot10^{-4}(\vartheta-25)) \qquad \vartheta \text{ in } ^{\circ}C, \qquad P<0.3$$

$$\text{(U,Pu)C}: E\ [\text{MPa}] = 202000\cdot(1-1.54\cdot P)\cdot(1-0.92\cdot10^{-4}(\vartheta-25)) \qquad \vartheta \text{ in } ^{\circ}C, \qquad P<0.3$$

Poisson's ratio is given by Padel et al /3/:

$$\nu = 0.288 - 0.286\cdot P \qquad 0.05<P<0.27$$

The fracture behaviour of ceramic fuel can be divided into three different temperature dependent regions: brittle behaviour at low temperature, decrease of breaking resistance with increasing temperature, ductile behaviour and high breaking elongation at high temperature. Steinkopff /5/ gives the following values for breaking strength of sintered carbide fuel (90% th.D.) at 25, 800 and 1000 $^{\circ}C$ as 160, 120 and 69 MPa; above 1200 $^{\circ}C$ plastic deformation will occur. From these and other values a law can be abstracted:

$$\sigma_s\ [\text{MPa}] = 9.81\cdot(13-3\cdot10^{-3}\cdot\vartheta) \qquad \vartheta \text{ in } ^{\circ}C$$

$$\delta_B\ [\ /\ ] = 0, \quad \text{if } \vartheta < 1400\ ^{\circ}C; \qquad \delta_B\ [\ /\ ] = 10^{6}, \text{ if } \vartheta > 1400\ ^{\circ}C$$

The treatment of plasticity and fracture within the URANUS fuel rod mechanics is solved with a special iterative procedure by Lassmann /1/.

3.2 Thermal behaviour of carbide fuel

The metallic bonding with free electrons results in extremely good thermal properties, the very flat temperature gradient allows a higher specific rod volume power. The most interesting parameters which concern the thermal behaviour are temperature, porosity, alloy composition, stoichiometry and burn-up; in general, only temperature and porosity dependence are described in most laws. Fig. 1 shows several values for the heat conduction coefficient of UC as published. For URANUS, the equation by Steiner /6/ was chosen:

$$\lambda\ [\text{W/mm}^{\circ}\text{C}] = \frac{1-P}{1+P}\cdot(0.02 + 1.3\cdot10^{-6}\cdot(\vartheta-500)) \qquad \text{if } \vartheta > 500\ ^{\circ}C$$

Due to the unconformity of data in the case of (U,Pu)C no general law could be found. The equations by Steiner /6/ and Caligara /7/ are used optionally:

$$\lambda\ [\text{W/mm}^{\circ}\text{C}] = \frac{1-P}{1+P}\cdot(0.0126 + 4.1\cdot10^{-6}\cdot(\vartheta-500)) \qquad \text{if } \vartheta > 500\ ^{\circ}C$$

$$\lambda\ [\text{W/cm}^{\circ}\text{C}] = (1-P)^{2/3}\cdot(-0.0178 + 0.594\cdot10^{-4}\cdot\vartheta - 0.235\cdot10^{-7}\cdot\vartheta^2) \qquad \text{if } 700 < \vartheta < 1200\ ^{\circ}C$$

Rather typical for the thermal elongation of stoichiometric carbide fuel is the temperature dependence of the gradient of the curves. The material laws by Steiner et al /8/ and Caligara /7/ are very similar:

$$\alpha\,[\frac{1}{^{o}C}] = 1.007\cdot 10^{-5}+1.17\cdot 10^{-9}\cdot\vartheta \qquad \vartheta \text{ in } ^{o}C$$

3.3 Irradiation behaviour

Creep: Thermal creep of UC and (U,Pu)C sets in with temperatures above 1000 ^{o}C and then clearly predominates over irradiation-induced creep. Due to the possible high fission rates and the flat temperature profiles in carbide fuel however, irradiation-induced creep gives a certain contribution to bringing down the mechanical stresses. Steiner /6/ and Matthews /4/ show:

$$\dot{\varepsilon}^{ind}_{cr}\,[h^{-1}] = 3.6\cdot 10^{-22}\cdot F\cdot\sigma_v \qquad F \text{ in fiss/cm}^3\text{s}, \qquad \sigma_v \text{ in MPa}$$

The dependence of the creep rate on stress and temperature is used in the most creep laws and generally leads to increasing values of $\dot{\varepsilon}^{th}_{cr}$ with increasing stresses and temperature. Steiner /6/ gives the following law

$$\dot{\varepsilon}^{th}_{cr}\,[h^{-1}] = 1.49\cdot 10^{10}\cdot\sigma_v^{2.44}\cdot\exp(-63200/T) \qquad T \text{ in Kelvin}, \qquad \sigma_v \text{ in MPa}$$

Freund's /9/ equation is nearly the same; in both cases, the stress exponent is above 2, which agrees with a statement of Matthews /4/.

Swelling: During irradiation, reactor fuel increases its volume due to pore growth and fission product storage. Contrary to oxide fuel, the swelling velocity of carbides is higher than the creep velocity, consequently, a high stress level and a "hard" pellet/clad mechanical interaction results. Nickerson et al /2/ explain this swelling behaviour by the low fission gas release of carbides and with the somewhat different restructuring behaviour compared to oxide fuel; both in consequence of the flat temperature gradient. However, swelling of high density carbides differs from swelling of low density fuel, because low density fuel is able to swell into the pores. The swelling rates increase with temperature due to the higher mobility of gas bubbles and lattice vacancies, where fission gas atoms are trapped. Above 900 - 1050 ^{o}C gas swelling starts (Steiner /6, 8/); due to higher amounts of gas atoms in the matrix with increasing burn-up, this temperature bar decreases (Nickerson et al /2/). This effect of sudden gas swelling after a certain burn-up and above a definite temperature is called breakaway-swelling.

To establish a swelling model for carbide fuel, based on first principals, is rather difficult. Blank et al /10/ describe a possible way by dividing the swelling in four additive parts due to solid fission products, little gas bubbles, pores and cracks; combined with a subdivision of the fuel in four structural regions with well definite physical properties. For use with URANUS, a more simple, empirical model was developed, based on experimental results. The model is dependent on temperature and burn-up and takes both solid body and gas bubble swelling into account. The model differentiates between unrestrained and restrained swelling due to contact/gap cases; porosity changes and hot pressing are taken into account. Solid body swelling is independent of temperature, proportional to burn-up, and cannot be compensated. The volume increase per percent burn-up lies in the range of 0.426% up to 0.9% for swelling with pellet-clad interaction (recent values; older ones give 1.3% - 1.6%).

In the case of low density fuel in combination with PCMI the lower values as reported by Freund et al /9/ seem to be correct:

$$\frac{\Delta V}{V}\left[\text{Vol \% / 10 } \frac{\text{MWd}}{\text{Kg}}\right] = 0.4667$$

Swelling of high density fuel without contact to the clad can be described as follows:

$$(\text{U,Pu})\text{C}: \frac{\Delta V}{V}\left[\text{Vol \% / 10 } \frac{\text{MWd}}{\text{Kg}}\right] = 1.3838$$

$$\text{UC} \quad : \frac{\Delta V}{V}\left[\text{Vol \% / 10 } \frac{\text{MWd}}{\text{Kg}}\right] = 2.178$$

Above 900 °C for mixed carbides and above 700 °C for carbide fuel respectively, breakaway swelling which can be described by a parabolic function is assumed. The functions are limited to be constant above 4.56 Vol.% for UC and 3.4 Vol.% for (U,Pu)C corresponding to the present knowledge from experiments (see fig 2).

Fission gas release: Due to the flatter temperature gradient in carbide fuel compared to oxides, fission gas behaviour of carbides is quite different. A great deal of produced fission gas is stored in the fuel and causes swelling and changes in porosity. From existing experimental results a set of principal statements for carbide fuel can be assembled: low and medium density fuels release more fission gas than high density fuel; unstoichiometric fuel contains more voids than stoichiometric fuel and hence can store more gas; impurities hinder gas diffusion and therefore bubble formation and also bubble motion.

Most models to describe the kinetics of fission gas are formulated microscopically (consideration of individual bubbles) and are consequently expensive. For this reason and because of the small amount of inpile data which exists for calibration of the model, a simple empirical model was used for the initial development of the carbide version of URANUS. It is based on 32 literature sources dealing with post-irradiation examinations; most results are from tests with He-bonded, low to medium density, mixed carbide rods. The influence of the porosity on the release properties is in general only qualitatively described; accordingly low density rods tend to have higher fission gas release than high density rods. This is because the gas bubbles can collect at the grain boundaries more easily. For the temperature dependence of the fission gas release, three typical regions were defined: with central temperature below 1000 °C, practically no fission gas release occurs; from 1000 °C to 2070 °C the release rate is described as a linear function; and above 2070 °C it follows a steady saturation function.

According to Nickerson et al /2/ there is a certain temperature for every level of burn-up at which bubble forming and hence gas swelling and gas release begins. By consideration of the temperature-dependence and of the temperature/ burn-up dependent start of fission gas release, the relationship between burn-up and fission gas release could be obtained through correlation with the data points. Fig. 3 shows as a result the obtained functions in comparison to the experimental results.

4. Results of URANUS calculations

In this section several typical results are presentend which where obtainend with the carbide version of URANUS. Because of the small number of evaluated experiments, these results should not however be considered as complete verification of this version of the code.

Calculations corresponding to the Mol-11 (KfK) and GOCAR-4 (Euratom) experiments have been run through. The Mol-11 experiments were carried out in four test groups in the thermal flux of BR-2 reactor in Mol/Belgium; the maximum burn-ups reached 35 to 70 GWd/to with linear rod powers between 900 and 1300 W/cm (see Freund et al /9/). A comparison of calculated central-temperature for Mol-11/K2-2 shows good agreement with the results of SATURN code, if one assumes only low fuel densification from 83% th.D. to 88% th.D. This assumption is corroborated by comparison of the cold measured cladding tube radii at the end of irradiation with URANUS results (see fig 4). Good consistency is also achieved in calculating the fission gas pressure build-up, as fig. 5 shows.

The GOCAR-4 experiment was irradiated in the thermal flux of Siloe reactor (Grenoble). 93% th.D. mixed carbide fuel was used with a canning of Niob 1% Zr. The conditions were nearly steady state (650 W/cm), interrupted by 9 reactor shut downs; a burn-up of about 3.5% was achieved. Fig. 6 shows the calculated values for fuel and canning temperature in comparison to the fluctuation of the central thermocouple measurement. More than 100 power changes have been considered for the calculation. Fig. 7 shows the tangential stress distributions at the start and end of cycle 1, and at the end of cycle 2. The disturbance of the thermo-elastic stress field around the edge of the pellets caused by cracking can be clearly recognised, as can the stress reduction with time due to creep.At the end of irradiation a permanent cold cladding-tube expansion (radial) of 8 µm is obtained in comparison to 9 µm in the experiment. This shows that the models for the crack-, creep- and swelling mechanics of the fuel work very satisfactorily.

To summarise, it can be stated that the results obtained with the newly developed carbide version are also in application to particular experiments satisfactory. The newly developed models have proved good, but further detailed work is still necessary. Hence, a detailed and easily to handle version of URANUS is available for carbide irradiation experiments and interpretation calculations, which gives a full thermal and mechanical analysis. Transient calculations are in preparation.

5. Acknowledgement

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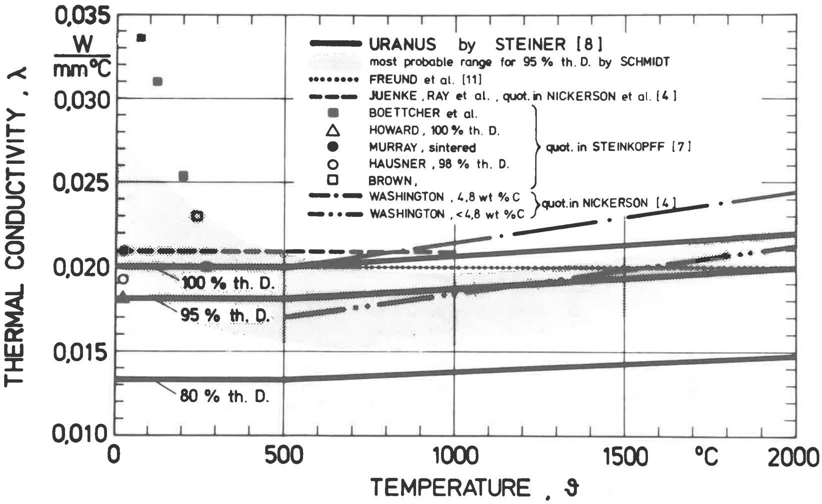

Fig. 1. Thermal conductivity λ of UC - several authors quoted.

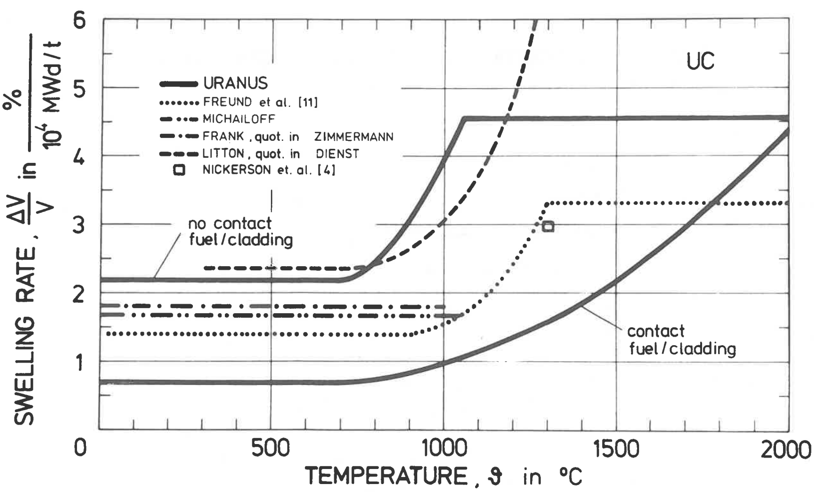

Fig. 2. Swelling rate of UC against temperature - URANUS-model compared to experimental results.

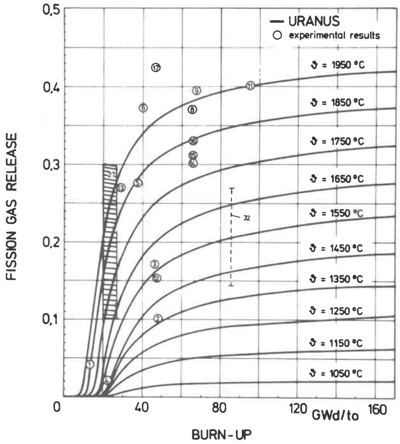

Fig. 3. Fission gas release as a function of burn-up

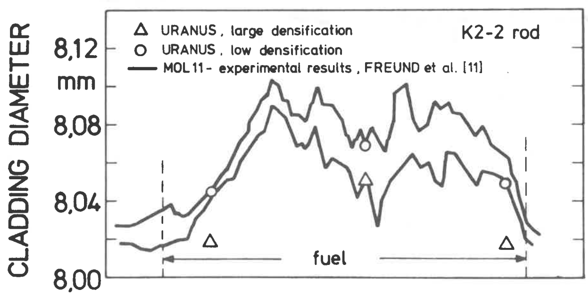

Fig. 4. Cladding diameter of sample K2-2 .

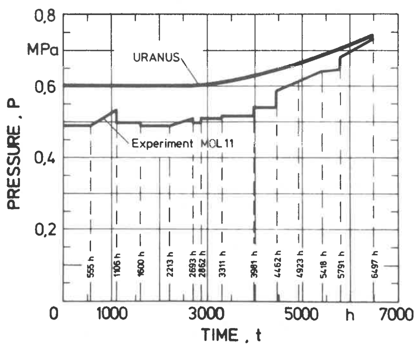

Fig. 5. Gas pressure increase in sample K2-2 .

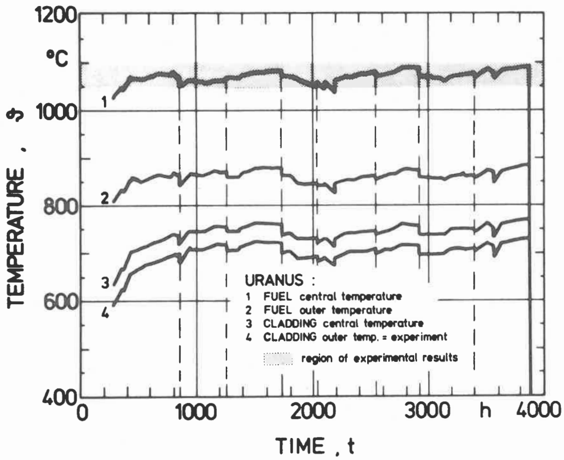

Fig. 6. Temperature profile versus time URANUS-calculation compared to experimental results .

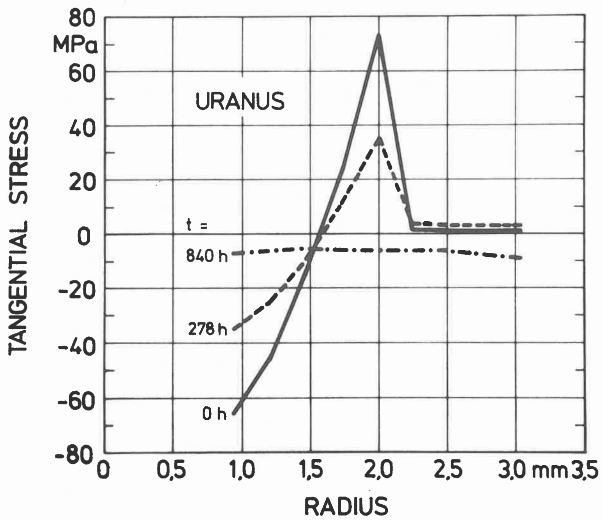

Fig. 7. Radial distribution of tangential stress .