Summary

An analytical expression for the steady-state creep of UO₂ is required for use in computer codes which predict fuel dimensional changes during irradiation. Out-of-pile (1-8) and in-pile (9-15) experiments have been reviewed and an analytical model based on theoretical models and experimental data for the steady-state creep of UO₂ has been developed and verified. The effects of temperature, stress, density, grain size, and fission rate on the various measured UO₂ creep rates were evaluated and are reflected in the model, represented by Equation (1):

$$\dot{\varepsilon} = \frac{(A_1-A_2 F) \sigma e^{-Q_1/RT}}{(A_3 + D)G^2} + \frac{A_4 \sigma^{4.5} e^{-Q_2/RT}}{(A_5 + D)} + A_6 \sigma F e^{-Q_3/RT}$$

(1)

where:
- $A_1 = 9.728 \times 10^6$
- $A_2 = 3.24 \times 10^{-12}$
- $A_3 = -87.7$
- $A_4 = 1.376 \times 10^{-4}$
- $A_5 = -90.5$
- $A_6 = 9.24 \times 10^{-28}$
- $Q_1 = 90,000 \text{ cal/mole}$
- $Q_2 = 132,000 \text{ cal/mole}$
- $Q_3 = 5200 \text{ cal/mole}$
- $\dot{\varepsilon} = \text{Creep rate} \quad \frac{\text{in.}}{\text{in.-hr}}$
- $F = \text{Fission rate} \quad (8.4 \times 10^{17} \text{ to } 1.18 \times 10^{20} \text{ fission/m}^3 \cdot \text{sec})$
- $\sigma = \text{Stress} \quad (1000 \text{ to } 16000 \text{ psi})$
- $T = \text{Temperature} \quad (713 \text{ to } 2073 \text{ °K})$
- $D = \text{Density} \quad (92 \text{ to } 98\% \text{ TD})$
The first two terms of this equation reflect high temperature thermal creep mechanisms. The first term represents a viscous creep mechanism with some grain boundary sliding occurring at low stresses. The viscous creep is linearly proportional to stress; inversely proportional to the square of grain size; and enhanced by a factor that is proportional to the fission rate. A dislocation-climb mechanism expressed in the second term of Equation (1), operates at stresses greater than a transition stress that is governed by the grain size \( G \) (5, 16). The dislocation-climb creep rate is proportional to the stress to approximately the 4.5 power. Data are not available for irradiation enhancement of UO\(_2\) creep in the dislocation-climb controlled regime and, therefore, although certain authors (14) have suggested equations incorporating irradiation enhanced dislocation-climb mechanisms, such mechanisms have not been incorporated into this model.

The stress necessary for transition from viscous creep to dislocation creep is assumed to be independent of temperature because of the apparently small difference in measured activation energies for the two mechanisms. The transition stress \( \sigma_{\text{trans}} \) is given by the following expression:

\[
\sigma_{\text{trans}} = 24,000G^{-0.5714} \tag{2}
\]

When the stress is less than \( \sigma_{\text{trans}} \), the actual stress in the fuel is used in the first term of Equation (1); for stresses greater than \( \sigma_{\text{trans}} \), the transition stress is used in the first term of Equation (1). For stresses below \( \sigma_{\text{trans}} \), the contribution to the creep rate from the second term is negligible.

The fission process induces creep at low temperatures for which out-of-pile thermal creep normally does not occur. This fission-induced creep rate is represented by the last term in Equation (1) and is proportional to stress and
fission rate. Early experimenters reported the process to be athermal \(^{14}\), but suggested that the temperature dependency could be masked by material variations. Later, Brucklacher \(^{15}\) reported a slightly temperature dependent process with an activation energy of 5,200 cal/mole. The effect of material properties on the irradiation induced creep of UO\(_2\) has not been determined.

Analytical predictions obtained using Equation (1) are compared in Figure 1 with experimental data selected from compressive creep tests. Good agreement with all the data is obtained except with the data from Poteat \(^{4}\) in the high stress region, data for the 55\(\mu\) grain size material used by Wolfe \(^{5}\), and data from Perrin \(^{13}\) and Solomon et al \(^{12}\) for unirradiated fuel. No explanation is available for these discrepancies.
REFERENCES

1. R. Scott et al., J. Nucl. Mat., 1, 39 (1959)
12. A. A. Solomon and R. H. Gebner, Nucl. Tech., 13, 177 (1972)
Figure 1: Comparison of experimental data for irradiated and unirradiated UO$_2$ with corresponding calculated values obtained from Equation (1).