EFFECT OF IRRADIATION CREEP STRAIN ON THE COEFFICIENT OF THERMAL EXPANSION OF AGL IM1-24 AND UCAR GCMB NUCLEAR GRAPHITE MODERATOR BRICKS

David A J Carter, British Energy, Barnett Way, Barnwood, Gloucester, GL4 3RS, United Kingdom

Introduction

British Energy Ltd. (BE) operates all seven twin Advanced Gas-Cooled Reactor (AGR) power stations in the UK, in addition to one Pressurised Water Reactor (PWR) power station at Sizewell. In the AGRs neutrons from the heat producing process of fuel fission are moderated by graphite, and both graphite and fuel are cooled by pressurised carbon dioxide. The moderation process together with corrosion by the coolant alters the material properties of the graphite, which in turn influences the functionality of the moderator core. As a consequence it is important to BE to understand the behaviour of irradiated graphite.

Background

The AGR graphite moderator is formed from hollow cylindrical blocks (1m × 0.5m) laid on a square lattice to form ~350 vertical channels, in a stack which is ~12 layers high, has an overall cylindrical shape and is attached to supporting steelwork. The cylindrical blocks are spiggoted together vertically, and connected by keys horizontally (Fig. 1). These interconnections ensure that the geometry of the vertical fuel and control rod channels is maintained, while allowing local deformation of individual blocks and global distortion of the stack within the constraining steelwork.

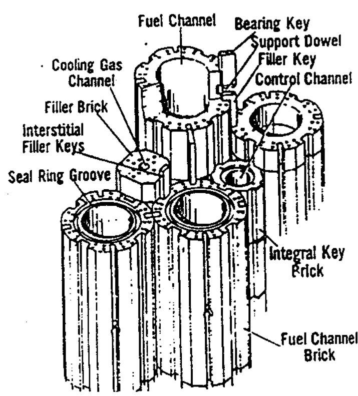

Fig. 1 Typical AGR moderator block geometry and keying arrangement

Fuel is located discretely within the vertical channels, giving rise to periodic spatial non-uniformities in the moderation and corrosion processes through the moderator volume, and in particular in the individual graphite blocks. These variations give rise to associated radial gradients in the material properties within the blocks, which are manifest as a combination of internal stresses, and deformations. Changes to block geometry caused by damage combine with these deformations to influence the possible displacements of the core. Consequently, the irradiated material properties affect the potential geometry of the fuel and control rod channels, and therefore the functionality of the moderator.

Model of Interaction of CTE and Irradiation Creep

When subjected to irradiation the coefficient of thermal expansion (CTE) of Anglo Great Lakes IM1-24 and Union Carbide GCMB nuclear graphite as used in the moderator falls from around 5×10^{-6}°C^{-1} for an unirradiated sample to around 2×10^{-6}°C^{-1} for a fast neutron dose of 150×10^{20} neutrons/cm^2 EDND[1] (See Fig. 2). The reduction in CTE is dependent on the temperature at which the graphite is irradiated, such that graphite irradiated at 500°C would have a greater reduction in CTE than a sample irradiated at 350°C, for the same neutron dose.

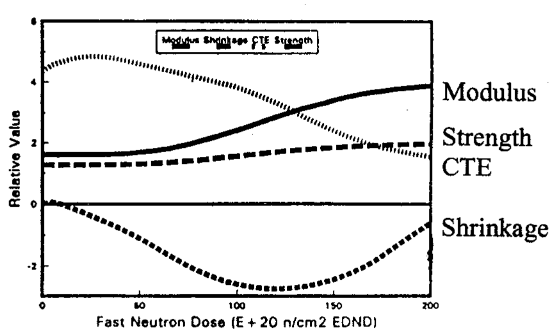

Fig. 2 AGR moderator material property changes with irradiation

1 Equivalent DIDO Nickel Dose. The damage to graphite caused by fast neutrons is a function of the energy spectrum of the incident neutrons. To correct for this doses are converted into an "Equivalent DIDO Nickel Dose", which refers to the particular neutron energy spectrum present at the DIDO Materials Testing Reactor at Harwell, Oxfordshire, UK.

The CTE is also separately affected by irradiation creep strain, such that tensile creep strain reduces CTE, and compressive creep strain increases the CTE (Fig. 3). This is particularly relevant as moderator graphite is subject to irradiation shrinkage, which is relieved by creep.

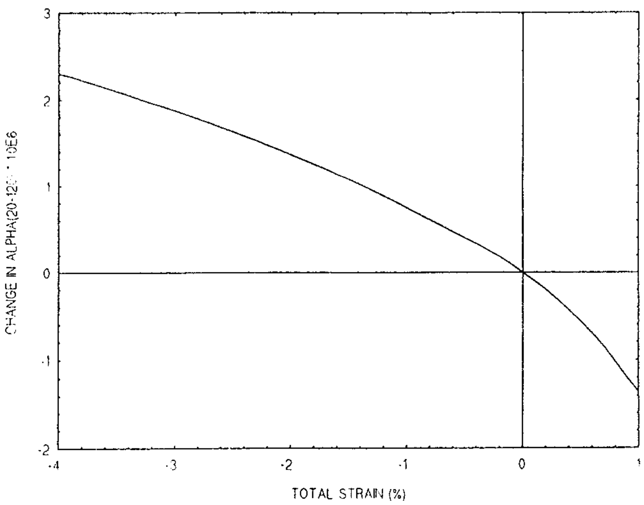

Fig. 3 Change in α_{20-120} v Total Strain

As noted above the moderator graphite in Advanced Gas-cooled Reactors is in the form of hollow cylindrical bricks, with fuel stringers running down the centres of vertical columns of bricks. This has the effect that the bore of the brick experiences both a higher fast neutron dose (by a factor of 2), and a higher irradiation temperature (450°C c.f. 370°C) than the outer surface.

The greater neutron dose at the channel wall gives rise to greater initial shrinkage together with earlier shrinkage turnaround. Such shrinkage is relieved by irradiation creep, and is associated with an initial tension to compression stress gradient along a brick radius. This stress gradient reverses on shrinkage turnaround, with the associated creep strains lagging substantially behind the stress behaviour (Fig. 4). CTE reduction gives rise to thermal stresses which have a tensile to compressive stress gradient on overall cooling of the block (Fig. 4).

Moderator graphite irradiation shrinkage rate is related to CTE behaviour up to shrinkage turnaround, with lower shrinkage (i.e. less negative strain) rates being associated with higher CTE.

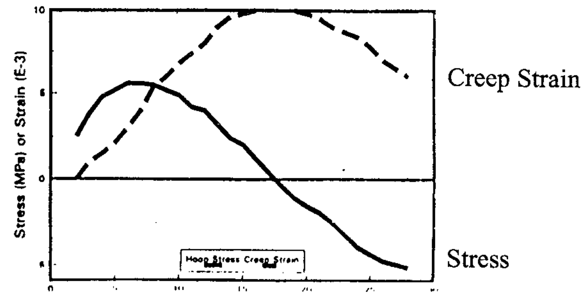

Fig. 4 AGR moderator block internal stresses and creep strain history

As shown in Fig. 4, creep strains in moderator bricks are out of phase and lag behind shrinkage stresses, having a maximum value at stress reversal. In the first instance, the residual tensile creep strains at the moderator block bores reduce the CTE, and exacerbate the block thermal stresses post-turnaround. However, the associated increase in shrinkage rates increases initial shrinkage stresses and delays and offsets subsequent reversed stresses.

Validation

A materials model of how graphite materials properties evolve with increasing dose and fluctuating temperature was developed. The model is time integrated i.e. over a small increment of neutron dose, the changes in CTE, the elastic strains and stresses, the creep strains and stresses, and the CTE for the next increment are calculated. The cycle then repeats by adding another small increment of neutron dose, along with any temperature changes that may occur.

The materials properties calculated using the above model feed into finite element stress analyses of the graphite to predict stresses in the bricks up to the end of reactor life. This model of the interaction of creep strain and CTE is currently being validated by comparison of predicted stresses with those measured in bricks in the reactor and material properties with those measured in small samples withdrawn from the reactor.

Acknowledgement

I would like to express my thanks to the British Nuclear Energy Society for funding my attendance at the Carbon '99 Conference. I also wish to express my gratitude to Dr. Neil McLachlan of British Energy for his technical assistance in preparing this paper.