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Permeability of observed three dimensional fracture networks in spent fuel pins

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ABSTRACT

The three-dimensional fracture network within a spent fuel pin is characterised using sequential grinding, and its permeability is numerically estimated. Advanced Gas-cooled Reactors (AGRs) produce spent fuel pins consisting of an outer steel cladding enclosing ceramic uranium dioxide (UO_2) pellets. During irradiation, fuel pellets may undergo fracturing due to thermal, densification and swelling effects. Fracture patterns are usually observed on the surface of the pellet or through a cross section or longitudinal plane along the pellet. In this work, the three-dimensional fracture pattern within the pellet is characterised using an optical microscope. The pellet is progressively ground and polished, providing sequential cross sections, which together yield a three-dimensional discrete fracture pattern. Multiple large fractures grow to connect the cladding to the internal region of the pellet. Multiple surface fractures are observed that do not penetrate into the matrix of the pellet. The porosity of the UO_2 and apertures of the fractures are estimated by sampling microscopic images. Darcy flow is numerically solved using the finite element method, computing flow through the matrix and fractures simultaneously. The equivalent tensorial permeability of the system is estimated for various approximate fracture apertures. The fracture network raises the permeability of the pellet by an order of magnitude.

1. Introduction

The UK's Advanced Gas-cooled Reactors (AGRs) are graphite moderated, CO_2 cooled reactors operating at pressures of about 4 MPa and gas temperatures in the range 300–675 °C [1]. In an AGR reactor, eight fuel elements are stacked vertically in a stringer; each element contains 36 fuel pins, arranged in three concentric circles, shown in Fig. 1. Elements are numbered 1 to 8, beginning at the bottom of the stringer and working upwards. The operating conditions experienced by fuel pins in different elements varies considerably, with average fuel cladding temperatures ranging between around 350 °C in element 1 fuel pins to a maximum of 800–850 °C in element 7.

Fuel pins contain approximately sixty-four UO_2 fuel pellets stacked vertically inside a ribbed austenitic stainless steel cladding, as illustrated in Fig. 1B [2]. Ribs are machined onto the outer cladding surface to improve heat transfer [3]. In service and during subsequent storage and transport, the cladding provides the primary containment for radioactivity generated within the fuel during irradiation.

During irradiation, the ceramic UO_2 fuel pellets experience high thermal gradients, initially causing densification then swelling due to the production of fission products. Degassing also occurs, altering the porosity of the UO_2. These processes lead to the formation of fractures in the fuel [4]. In fuel elements that experience higher operating temperatures the fuel and cladding form a bond, which leads to additional fracture formation as the fuel cools after discharge from the reactor [5].

After discharge from the reactor, individual fuel elements are separated from each other and stored in station cooling ponds prior to shipment to Sellafield, UK for interim wet storage. If the fuel cladding suffers corrosion, or any damage that penetrates through the entirety of the cladding, pond water can enter the fuel pin and migrate into the fuel matrix. This may lead to corrosion of the fuel and release of radioactivity into the pond water. In the event that fuel is to enter a period of dry storage, or be consigned for

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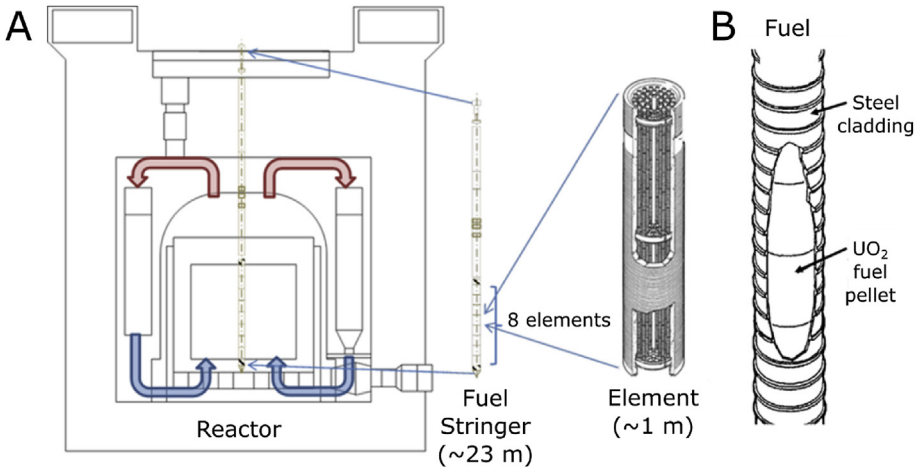

Fig. 1. Schematic of the UO_2 fuel pellet and cladding in the context of AGR reactors. A: AGR fuel element, in the context of the reactor. B: AGR fuel pin. After [2].

geological disposal, it is important that water from inside failed fuel pins is removed to prevent corrosion or pressurisation of the dry storage/disposal container [6]. Therefore, it is essential that the permeability of the fuel matrix is understood as this will affect both saturation and drying of the fuel. Of these, the drying behaviour is particularly important given the potentially long time periods over which outgassing of water vapour can continue [7].

1.1. Fuel assembly and structure

AGR fuel pins consist of a ribbed stainless steel cladding tube that provides the primary containment for the fuel and the fission and activation products generated during irradiation. The ends of the fuel tube are enclosed by end caps that are seam welded to provide a continuous and robust containment boundary and are separated from the fuel by insulating aluminium oxide pellets that reduce the operating temperature of the end caps in service. The overall length of a fuel pin is approximately 1 m.

AGR fuel pellets are fabricated from sintered UO_2 with a theoretical density greater than 97.2% [8]. The annular (hollow) fuel pellets reduce the fuel centreline temperature, and the central void space provides a reservoir for containment of fission gases released from the fuel during irradiation. The fuel pellets are approximately 15 mm long with an external diameter of 14.5 mm and an internal diameter of 6.35 mm [1]. One in six fuel pellets have a groove in the centre of the pellet, known as an anti-stacking groove. The fuel cladding is compressed into the groove during manufacture to ensure that the cladding strain is evenly distributed along the fuel pin during irradiation.

1.2. Fracturing of fuel pins

Understanding of fuel pellet fracturing has been driven by the need to understand pellet clad interaction failures. These fractures occur where the fuel and cladding have bonded, and cladding opposite a fuel crack, which is not bonded to fuel, experiences enhanced strain as the fuel crack opens during irradiation [9]. Important aspects of fuel pin behaviour that affect the fracturing are:

- Densification of the fuel during initial phase of irradiation [1].
- Fuel swelling due to fission gas production [8].
- Clad creep down to contact the fuel [1].
- Clad-fuel bonding [1].
- Differential thermal expansion due to high thermal gradients in the fuel pellets.

The general condition of AGR fuel pellets after discharge is shown in cross- and axial-sections in Fig. 2. In Fig. 2B, there are gaps between the fuel pellets. Although these are a result of manufacturing and therefore within expectations, they have a similar dimension to some of the fuel fractures and may therefore have a similar impact on drying of the fuel. In the element 1 fuel, there is evidence of a gap between the fuel pellet and cladding, with larger voids around the grooved pellets. Shrinkage cracks may provide a preferential flow path similar to the pellet-clad gap in lower element fuels and both may limit the volume of fuel matrix that is remote from a connected flow path. Both shrinkage fractures and pellet-clad gaps may therefore increase the volume of fuel that is connected to a flow path. In both cases, the fuel matrix displays a range of fractures that have large variations in orientation and width. It is therefore concluded that 3D information on the fracture network is required to adequately characterise the fuel fracturing.

1.3. Study goals

This work details a fracture network study undertaken using an element 2 fuel pellet. To create a three-dimensional model of the fracture network in the pellet, the end of the pellet is imaged, ground and polished in five stages. Subsequently, the fractures are traced and interpreted in three dimensions. A numerical approach is used to simulate the flow of coolant fluid through the pellet [10] [11]. The equivalent permeability of the network is quantified, summarising the flow properties of the whole pellet, taking into account flow in the fractures and in the UO_2 matrix. This work provides an important understanding of AGR pellet permeability, as higher or lower values could influence the disposal procedures necessary for handling spent fuel.

2. Methodology

2.1. AGR fuel imaging

The present fracture network study has been undertaken using an element 2 fuel pellet that had been irradiated for approximately six years in the 1980s to a burn-up of 20.5 $GWd.teU^{-1}$, with a cladding temperature in the range 450–520 °C. The sample was selected from a position near the centre of the zone where the fuel