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Spatial variability in the coefficient of thermal expansion induces pre-service stresses in computer models of virgin Gilsocarbon bricks

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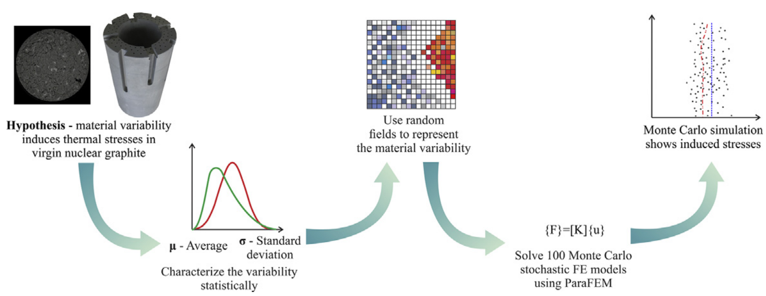
HIGHLIGHTS

- Open source software has been modified to include random variability in CTE and Young's modulus.
- The new software closely agrees with analytical solutions and commercial software.
- Spatial variations in CTE and Young's modulus produce stresses that do not occur with mean values.
- Material variability may induce pre-service stress in virgin graphite.

GRAPHICAL ABSTRACT



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ABSTRACT

In this paper, the authors test the hypothesis that tiny spatial variations in material properties may lead to significant pre-service stresses in virgin graphite bricks. To do this, they have customised ParaFEM, an open source parallel finite element package, adding support for stochastic thermo-mechanical analysis using the Monte Carlo Simulation method. For an Advanced Gas-cooled Reactor brick, three heating cases have been examined: a uniform temperature change; a uniform temperature gradient applied through the thickness of the brick and a simulated temperature profile from an operating reactor. Results are compared for mean and stochastic properties. These show that, for the proof-of-concept analyses carried out, the pre-service von Mises stress is around twenty times higher when spatial variability of material properties is introduced. The paper demonstrates that thermal gradients coupled with material incompatibilities may be important in the generation of stress in nuclear graphite reactor bricks. Tiny spatial variations in coefficient of thermal expansion (CTE) and Young's modulus can lead to the presence of thermal stresses in bricks that are free to expand.

1. Introduction

1.1. Objectives

The principal objective of this paper is to report on work carried out to test the hypothesis that tiny spatial variations in material properties may lead to significant pre-service stresses in virgin

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graphite bricks. Unrestrained temperature expansion with uniform mean material properties will not produce substantial stresses. However, unrestrained temperature expansion with more realistic random material properties could produce stresses that are currently ignored in thermo-mechanical engineering simulations. The authors also report on modifications made to open source parallel finite element software, adding functionality to generate spatially random fields for coefficient of thermal expansion (CTE) and Young's modulus. As the use of open source software is not common in nuclear engineering, we also describe in detail how the software was tested.

1.2. Background

Most of the UK nuclear reactors use graphite as a moderator. The main function of the graphite components is to moderate the nuclear reactions and to provide channels for fuel cooling and control rods entry [1]. The UK currently deploys three types of reactor: Advanced Gas-cooled Reactors (AGRs), Magnox Reactors and Pressurised Water Reactors (PWR). The fleet of 14 AGRs and the remaining Magnox reactor still in operation are graphite moderated.

During the lifetime of a graphite moderated reactor it is essential to ensure the integrity of the graphite core. Distortions, cracking and weight loss in the graphite components can potentially prevent normal reactor operations such as the loading and unloading of fuel or control rods. Dimensional changes and cracking may also impede the correct circulation of coolant. Damage to the graphite core could reduce the lifetime of a nuclear reactor. The graphite moderator bricks cannot be replaced [2], making the graphite core a critical component of a nuclear power station [3]. It is important to note that graphite component cracking is not a safety issue in itself as long as the core can still perform its function under all operating conditions. That is, the fuel can be adequately cooled, the reactor safely shut down, and the fuel safely removed.

When in-service, graphite is subjected to fast neutron irradiation and oxidation. These phenomena cause a progressive degeneration of the graphite material. Neutron irradiation affects graphite in several ways; it changes the material properties of graphite and causes change in dimensions. In addition to this, neutron irradiation promotes ‘irradiation creep’ that relieves internal stresses. Graphite used in nuclear reactors can experience two oxidation mechanisms: thermal oxidation and radiolytic oxidation. Thermal oxidation is negligible at normal AGR and Magnox operating temperatures. Radiolytic oxidation is present during the normal operation of an AGR and Magnox reactor and causes considerable weight loss if it is not controlled properly.

Several techniques are used to predict the health of a reactor including direct inspections, measurements taken during the operation of a reactor and computational modelling. Periodic inspections track the ageing effects on graphite caused by nuclear irradiation. A device explores the surface of the graphite core to measure the dimensions and ovality of the channels formed by the graphite bricks and a camera records the condition at the bore. Graphite samples are extracted from the graphite components during fuelling operations and ageing is quantified experimentally, providing insight about the condition of the reactor [2]. Computer modelling is used to predict the structural integrity of the graphite bricks by estimating stresses generated by elastic, thermal, creep and irradiation strains. Modelling is verified by comparing predictions with data from inspections.

1.3. Computer modelling

In the literature, computer modelling focuses on the development and implementation of constitutive equations that capture the full complexity of the physical processes acting on the graphite. For example, Tsang and Marsden [4] [5] have developed Abaqus material user subroutines for Magnox and AGR. In the case of High Temperature Gas-cooled Reactors (HTGR) Mohanty et al. [6] have developed a constitutive model for stress analysis of prismatic graphite bricks. Yu et al. present a microstructure-based model that predicts the lifetime of an HTGR [7].

To date, computer modelling has given important insights into the behaviour of nuclear graphite. However, as is typical in engineering simulation, the output of the models does not match well enough with observations to allow accurate predictions. Operators and regulators cannot yet use simulation alone to assess the maximum period of time a particular reactor can remain in service under safe operating conditions.

One route to improved simulation is to increase the complexity of the constitutive models and increase the temporal and spatial resolution of the finite element meshes. In this paper, the authors instead focus on potential improvements from a different perspective: introducing realism into the definition of material properties in the finite element model.

It is common practice to use a mean value of a material property in an engineering simulation. For example, in a mesh of a graphite brick, all the elements that represent graphite are often given the mean Young's modulus, the mean Poisson's ratio and the mean CTE. These values change as the analysis proceeds when the effects of temperature, irradiation and oxidation are taken into account. The starting point for the analysis is perfect uniformity in the material and any temperature change will lead to expansion without the generation of significant stresses. In contrast, tiny spatial variations in material properties (that exist in graphite bricks before they are brought into service) can induce thermo-elastic strains during a simple temperature change. These pre-service stresses may be an important, but overlooked starting point for simulations that seek to predict how the brick will respond to in-service conditions during the lifetime of the reactor. The most appropriate initial conditions for a simulation may not be uniform material and zero stress during a temperature change.

1.4. Spatial variation in material properties

Spatial variations in material properties, documented in pre-service virgin nuclear graphite [8–14], are related to the properties of the main components used (filler, binder and flour), their arrangement and the method of manufacture. Several types of nuclear graphite have been created for the different needs, stages and generations of nuclear reactor. Pile Grade A (PGA), an anisotropic graphite, was manufactured for the first generation of British reactors, the Magnox reactors. PGA filler particles were obtained from the cracking process in the petroleum industry. The filler particles in PGA have needle shape forms that become preferentially aligned during the extrusion stage of the manufacturing process. This gives the material a direction dependent microstructure and consequentially direction dependent properties. Gilsocarbon, a type of graphite that is quasi-isotropic, is used in AGRs. PGA and Gilsocarbon binders are randomly orientated and thus do not have any preferential alignment [9].

The material properties of the graphite components of AGRs are recorded in heat certificates. A heat certificate is an official document issued by the manufacturer which is linked to a particular batch of components. These records are one of the most important sources of information for the unirradiated material properties of the Gilsocarbon actually used in the construction of the AGRs. A statistical analysis that compared the variation of material properties with the information present in the heat certificates was