The mechanical behavior and reliability prediction of the HTR graphite component at various temperature and neutron dose ranges

Xiang Fang\textsuperscript{a}, Suyuan Yu\textsuperscript{a}, Haitao Wang\textsuperscript{a,⁎}, Chenfeng Li\textsuperscript{b}

\textsuperscript{a} Institute of Nuclear and New Energy Technology, Tsinghua University, Beijing 100084, China
\textsuperscript{b} Civil and Computational Engineering Centre, College of Engineering, Swansea University, Swansea SA2 8PP, UK

HIGHLIGHTS

• The mechanical behavior of graphite component in HTRs under high temperature and neutron irradiation conditions is simulated.
• The computational process of mechanical analysis is introduced.
• Deformation, stresses and failure probability of the graphite component are obtained and discussed.
• Various temperature and neutron dose ranges are selected in order to investigate the effect of in-core conditions on the results.

ARTICLE INFO

Article history:
Received 22 September 2013
Received in revised form 13 May 2014
Accepted 22 May 2014

ABSTRACT

In a pebble-bed high temperature gas-cooled reactor (HTR), nuclear graphite serves as the main structural material of the side reflectors. The reactor core is made up of a large number of graphite bricks. In the normal operation case of the reactor, the maximum temperature of the helium coolant commonly reaches about 750 °C. After around 30 years' full power operation, the peak value of in-core fast neutron cumulative dose reaches to $1 \times 10^{22}$ n cm$^{-2}$ (EDN). Such high temperature and neutron irradiation strongly impact the behavior of graphite component, causing obvious deformation. The temperature and neutron dose are unevenly distributed inside a graphite brick, resulting in stress concentrations. The deformation and stress concentration can both greatly affect safety and reliability of the graphite component. In addition, most of the graphite properties (such as Young’s modulus and coefficient of thermal expansion) change remarkably under high temperature and neutron irradiations. The irradiation-induced creep also plays a very important role during the whole process, and provides a significant impact on the stress accumulation. In order to simulate the behavior of graphite component under various in-core conditions, all of the above factors must be considered carefully. In this paper, the deformation, stress distribution and failure probability of a side graphite component are studied at various temperature points and neutron dose levels. 400 °C, 500 °C, 600 °C and 750 °C are selected as the core-side temperature of the graphite component, while the range of neutron dose is $0–1 \times 10^{22}$ n cm$^{-2}$ (EDN). It is shown in the numerical analysis that temperature strongly affects the histories of both stresses and failure probabilities as fast neutron dose increases. In addition, the behavior of graphite brick at 750 °C is observed obviously different from that at other temperatures.

© 2014 Elsevier B.V. All rights reserved.

1. Introduction

Graphite is one of the most important structural materials for pebble-bed high temperature gas-cooled reactors (HTRs). The side reflector of the reactor core is made up of a large number of graphite bricks. Therefore, stability of the graphite components is a premise to the operational safety of HTRs (Windes et al., 2011).

The graphite components are direct exposed to high temperature and fast neutron irradiation environment in core when the reactor is on operation. The maximum temperature of the helium coolant commonly reaches about 750 °C, located at the bottom of the core, while the peak value of fast neutron cumulative dose (referred to as neutron dose hereinafter) reaches $1 \times 10^{22}$ n cm$^{-2}$ (EDN), located on the upper 1/3–1/4 part of the core. The resulting thermal strain and irradiation-induced strain may create significant deformation and internal stresses on the graphite brick. Moreover, the creep phenomenon of graphite is obvious on high temperature and neutron dose environment, and acts as a key factor for graphite component’s

http://dx.doi.org/10.1016/j.nucengdes.2014.05.036
0029-5493/© 2014 Elsevier B.V. All rights reserved.
deformation and stress accumulation (Burchell et al., 2010; Davies and Bradford, 2008). Excessive deformation and stress might cause risk of instability or even structural failure issues. In order to ensure the safe and stable operation of the reactor, it is very important to analyze the deformation, the stress state and the failure probability of graphite component at various in-core conditions. Furthermore, high temperature and neutron irradiation engender great changes in graphite parameters, including the Young’s modulus, coefficient of thermal expansion (or CTE for short), creep coefficient, etc. This makes the behavior of graphite component more complex and difficult to predict (Kelly and Burchell, 1994). In order to correctly predict the mechanical behavior of graphite components, all the influencing factors must be taken into account.

The temperature and neutron dose are not uniformly distributed on the graphite brick, but very high on the core side and very low on the outer side. This makes the stress and deformation obviously vary at different parts of the brick. Some previous work has shown that high level stress does not accumulate on the areas where the temperature and neutron dose reach their respective peak values, but where the gradients of temperature and neutron dose are large, due to the sharp variation of thermal strain and irradiation-induced strain (Tsang et al., 2010). On any part of the graphite brick, the higher gradient of temperature or neutron dose, the greater the difference of mechanical strain should be variable.

In order to analyze the mechanical behavior of graphite components, two steps of work must be completed (Yu et al., 2012). The first step is stress analysis, in which a three-dimensional constitutive law of graphite is established. A finite element analysis is carried out and both the deformation condition and the stress state can be readily obtained, with graphite properties (Young’s modulus, thermal conductivity, CTE, dimensional change, etc.) and boundary conditions (including temperature distribution and neutron irradiation distribution) being the input data. In this step, establishing a 3D constitutive law and selecting an appropriate creep model are the key points. Researchers have developed various creep models for describing the effects caused by creep. Each creep model constitutes a relationship law between creep and other properties of graphite such as Young’s modulus or dimensional change. Two different creep models, including the UKAEA model and the Kennedy model, are selected for stress analysis in the present paper.

The second step is reliability analysis. The obtained stress distribution is used as input data. The failure probability and service life of graphite component can be gained with some certain deterministic or probabilistic failure models (Davies et al., 2007; Oki and Ishihara, 2004; Yu et al., 2004). Both deterministic and probabilistic models are accepted in the engineering standards and codes, but the probabilistic method is commonly regarded as a more suitable tool to evaluate the mechanical behaviors of brittle materials like graphite (Rose and Tucker, 1982; Burchell, 1996). The Weibull distribution model used in the German HTR code draft KTA–3232 (KTA, 1992) is selected in the present paper.

As the graphite bricks work in an environment with broad ranges of temperature and neutron dose, its internal stress and deformation have very large differences in various serving period. Especially at very high temperature, there is little report on the prediction and analysis results of the graphite brick’s mechanical behavior, the engineering experience is also insufficient (Tsang and Marsden, 2005). In this paper, the deformation, stress distribution and failure probability of graphite components are calculated and predicted at various temperature points and neutron dose levels. 400 °C, 500 °C, 600 °C and 750 °C are selected as the temperature on the core side of brick, while the range of neutron dose is 0–1 × 1022 n cm−2 (EDN). The irradiation material data of nuclear graphite is obtained from completed experiments in history. The results clearly reflect dramatic changes in the behavior of graphite components at various temperatures.

2. Constitutive law and creep models

2.1. The elastic stress–strain relationship

In the FEM calculation, the elastic stress–strain relationship represents the constitutive law. When the reactor is normal running, both thermal strain and irradiation-induced strain will appear on the graphite brick. The constitutive law for operation case can be defined as (Tsang and Marsden, 2008; Wang and Yu, 2008),

\[
\sigma = D(T, \gamma) \cdot \varepsilon^E
\]

\[
\varepsilon = \varepsilon^E + \varepsilon^T + \varepsilon^K + \varepsilon^C
\]

\[
\sigma = \text{stress tensor, } D \text{ is the elastic matrix which is related to temperature } T \text{ and neutron dose } \gamma. \text{ The total strain } \varepsilon \text{ consists of elastic strain } \varepsilon^E, \text{ thermal strain } \varepsilon^T, \text{ irradiation-induced strain } \varepsilon^K \text{ and creep strain } \varepsilon^C. \text{ Eq. (1) can be rewritten into an incremental form as,}
\]

\[
d\varepsilon = D(T, \gamma) \cdot d\varepsilon^E + \varepsilon^E \cdot d\varepsilon^T + \varepsilon^K \cdot d\varepsilon^K + \varepsilon^C \cdot d\varepsilon^C
\]

\[
\varepsilon = \alpha(T, \gamma) \cdot dT
\]

The incremental form of thermal strain \( d\varepsilon^T \) is defined as, \( d\varepsilon^T = \alpha(T, \gamma) \cdot dT \) where \( \sigma \) is the coefficient of thermal expansion (CTE) depending on \( T \) and \( \gamma \). The curves of CTE development ratio \( \alpha/\alpha_0 \) of a certain kind of nuclear graphite at 300 °C, 400 °C, 500 °C and 600 °C are shown in Fig. 1 (Haag, 2000). \( \alpha_0 \) is the initial CTE without irradiation. The CTE increases when neutron dose level is low, but decreases rapidly after reaching the peak value. With the temperature increasing, the peak value reduces, but appears earlier. At the dose range of 5–10 × 1022 n cm−2 (EDN), the values of CTE at various temperature points are close, but the absolute values differ from each other. The CTE data in Fig. 1 is used to calculate \( d\varepsilon^T \) by Eq. (4).

The irradiation-induced strain \( \varepsilon^K \) in Eq. (1) is obtained by experimental data of dimensional change rate. The \( \varepsilon^K \) data of nuclear graphite at 300 °C, 500 °C and 750 °C is shown in Fig. 2 (Haag et al., 1990). The strains on both \( a \) and \( c \) axis of graphite crystal lattice are presented. The graphite shrinks in low neutron dose, and begins to expand rapidly when the neutron dose accumulates to a certain extent. The thermal expansion and irradiation-induced shrinkage
will offset each other in the earlier stage of irradiation. However, as the service time of graphite component is long, the neutron cumulative dose will be very high near the end of service life. Thus the degree of irradiation-induced shrinkage on core side of graphite brick is always greater than thermal expansion, resulting in contraction of the graphite brick. The irradiation-induced shrinkage at 500 °C is the most obvious, and its peak amount is the maximum. At 750 °C, although the amount of shrinkage is smaller, the expansion appears the earliest and the graphite degrades much sooner than at other temperatures.

2.2. The creep models

Creep can reduce the internal stress of graphite component. Most of the internal stress caused by high temperature and irradiation can be released by the action of creep. This means the creep strain $\varepsilon^c$ in Eq. (1) causes great impact to the total strain $\varepsilon$. There are several models which can be used to describe the creep strain, such as the linear visco-elastic model, the UKAEA model and the Kennedy model. Different parameters of graphite are required and used as raw data for various creep models. The entire creep process usually consists of two phases: the primary creep whose creep strain is expressed as $\varepsilon^{PC}$, and the secondary creep whose creep strain is $\varepsilon^{SC}$. $\varepsilon^{SC}$ is commonly considered equal to equivalent elastic strain $\sigma/E_0$, where $E_0$ is the Young’s modulus of virgin graphite. $\varepsilon^{SC}$ is proportional to instant stress $\sigma$ and neutron dose $\gamma$. The incremental form of creep strain can be expressed as [Yao et al., 2007].

$$d\varepsilon^c = d\varepsilon^{PC} + d\varepsilon^{SC} = \frac{d\sigma}{E_0} + k_0 \sigma \cdot d\gamma$$

$k_0$ is the secondary creep coefficient (or creep coefficient for short), which is related to $T$ and $\gamma$. Eq. (5) is also the expression of the linear visco-elastic model. $k_0$ can only be determined by real experiment. Unfortunately, the raw data of $k_0$ at high temperature (especially higher than 500 °C) is very limited. Therefore, the visco-elastic model cannot be applied in creep simulation at high temperature so far.

Nevertheless, the other two creep models are still efficient for creep simulation, and it has been proven that both models are able to substitute the visco-elastic model when lacking the data of $k_0$ (Fang et al., 2012). In the UKAEA model (Kelly and Brocklehurst, 1977), Young’s modulus is used as a factor which takes the place of $k_0$. Eq. (5) can be rewritten as,

$$d\varepsilon^c = d\varepsilon^{PC} + d\varepsilon^{SC} = \frac{d\sigma}{E_0} + \frac{E_p}{E} \cdot k_p \sigma \cdot d\gamma$$

$E_p$ and $k_p$ are Young’s modulus and creep coefficient in a specific region, respectively. Fig. 3 shows the Young’s modulus development ratio $E/E_0$ of the graphite at 300 °C, 400 °C, 500 °C, 600 °C and 750 °C. With the rise of neutron dose, Young’s modulus undergoes three stages: the initial growth, the secondary growth and the rapid decline. The temperature increasing accelerates the speed of evolution, and reduces the amplitude of variation. The factors $E_p$ and $k_p$ are usually selected in dose range between two periods of growth (the region is called “plateau” in some literatures). Their values at neutron dose 1 x 10¹⁷ n cm⁻² (EDN) are selected in the present paper.

In the Kennedy model (Kennedy et al., 1980), the raw data of the percentage volume change $\Delta V/V$ is applied instead of $k_0$. Eq. (5) can be rewritten as,

$$d\varepsilon^c = d\varepsilon^{PC} + d\varepsilon^{SC} = \frac{d\sigma}{E_0} + \left[ 1 - \mu \frac{\Delta V/V}{\Delta V/V_m} \right] \cdot k_p \sigma \cdot d\gamma$$

$\mu$ is an empirical constant, $\mu = 0.75$. $\Delta V/V$ can be obtained with the data of $\varepsilon^c$ in Fig. 2 by the following relationship,

$$\frac{\Delta V}{V} = 2\varepsilon^c + \varepsilon^c$$

$(\Delta V/V)_m$ is the maximum volume shrinking of the graphite, corresponding to the lowest point of each curve in Fig. 2.

2.3. The three-dimensional creep strain

For the practical multi-axial case, the tensor expression of creep strain $\varepsilon^c$ must be derived. The incremental form of three-dimensional creep strain tensor can be extended from the uni-axial case in Eq. (5) (Tsang and Marsden, 2005),

$$d\varepsilon^c = d\varepsilon^{PC} + d\varepsilon^{SC} = C_0 \cdot d\sigma + K \sigma \cdot d\gamma$$

In the UKAEA model, the creep strain is expressed as $\varepsilon^c = d\varepsilon^{PC} + d\varepsilon^{SC}$, then the neutron dose dependence of $d\varepsilon^{PC}$ and $d\varepsilon^{SC}$ can be derived by means of Eq. (5). The three-dimensional creep strain tensor can be expressed as:

$$d\varepsilon^c = C_0 \cdot d\sigma + K \sigma \cdot d\gamma$$

where $C_0$ and $K$ are the creep tensor coefficients. The neutron dependence of $C_0$ and $K$ can be calculated by means of Eq. (5).
\( C_0 \) and \( K \) are the unirradiated elastic flexibility matrix and the creep flexibility matrix, respectively.

\[
C_0 = \frac{1}{E_0} \begin{bmatrix}
1 & -\nu^E & -\nu^E \\
-\nu^E & 1 & -\nu^E \\
-\nu^E & -\nu^E & 1 \\
0 & 2(1 + \nu^E) & 2(1 + \nu^E) \\
0 & 2(1 + \nu^E) & 2(1 + \nu^E)
\end{bmatrix}
\]

\( K = k \begin{bmatrix}
1 & -\nu^C & -\nu^C \\
-\nu^C & 1 & -\nu^C \\
-\nu^C & -\nu^C & 1 \\
0 & 2(1 + \nu^C) & 2(1 + \nu^C) \\
0 & 2(1 + \nu^C) & 2(1 + \nu^C)
\end{bmatrix}
\]

\( \nu^E \) and \( \nu^C \) are the elastic and creep strain ratios, respectively. The case when \( \nu^E \) equals to \( \nu^C \) is selected in calculation.

3. The Weibull failure model

The failure probability can reflect the structural integrity and possible service life of graphite component. The obtained stress distribution condition in stress analysis procedure will be used as the initial data to obtain the failure probability of graphite component in the reliability analysis procedure. The two-parameter Weibull model is considered to be a dependable probabilistic model which can evaluate the reliability of graphite components in HTRs. The model is selected from the German HTR code draft KTA-3232 (KTA, 1992). A probability density function \( f(x) \) is defined by two-parameter Weibull distribution,

\[
f(x) = \left( \frac{x}{S_c} \right)^{m-1} \cdot \frac{m}{S_c} \cdot \exp \left[ -\left( \frac{x}{S_c} \right)^{m} \right] \quad x > 0
\]  

The graphite brick is meshed into many integration points in finite element calculation. Eq. (12) provides the probability density function of each integration point, with variable \( x \) standing for the modified stress \( \sigma_{eq} \). \( \sigma_{eq} \) is an equivalent stress for evaluating the service life of graphite component,

\[
\sigma_{eq} = \sqrt{\hat{\sigma}_1^2 + \hat{\sigma}_2^2 + \hat{\sigma}_3^2 - 2\nu^E(\hat{\sigma}_1\hat{\sigma}_2 + \hat{\sigma}_2\hat{\sigma}_3 + \hat{\sigma}_3\hat{\sigma}_1)}
\]

\( \hat{\sigma}_i \) \((i = 1, 2, 3)\) are modified principal stresses which are defined as,

\[
\hat{\sigma}_i = \begin{cases} 
\sigma_i, & \sigma_i \geq 0 \\
\sigma_i/\sigma_T, & \sigma_i < 0
\end{cases}
\]

\( \sigma_T \) and \( \sigma_C \) are the tensile and compressive strength, respectively. In order to simplify the representation, the following \( \sigma_{eq} \) will be abbreviated as \( \sigma \).

Both \( m \) and \( S_c \) in Eq. (12) are parameters which arise from fitting experimental curves. \( m \) is the shape parameter and \( S_c \) is the characteristic strength value. The values of the two parameters of the nuclear graphite are shown in Table 1. Similar to Young’s modulus, \( S_c \) changes with temperature and neutron dose. The variation of \( S_c \) is always expressed approximately as,

\[
\frac{S_c}{S_{c0}} = \left( \frac{E}{E_0} \right)^\beta
\]

\( S_{c0} \) is the unirradiated characteristic strength. \( \beta \) is an empirical constant which equals to 0.5 before Young’s modulus reaches peak value, and equals to 1.0 thereafter.

The failure probability \( P_i \) of an integration point can be derived by Eq. (12),

\[
P_i = \int_0^\sigma f(x)dx = 1 - \exp \left[ -\left( \frac{\sigma}{S_c} \right)^m \right]
\]

The survival probability can be designated as,

\[
1 - P_i = \exp \left[ -\left( \frac{\sigma}{S_c} \right)^m \right]
\]

The graphite brick is the sum of all the integration points, the survival probability of the entire graphite brick can be deduced as,

\[
P = \exp \left\{ -\sum \left[ \left( \frac{\sigma}{S_c} \right)^m \times \frac{V_i}{V} \right] \right\}
\]

4. FEM calculation and boundary conditions

A three-dimensional finite element code INET-GRA3D which is compiled by INET is applied to actualize the FEM calculation. The objective is to predict graphite component’s deformation, stresses and reliability at various temperature and neutron dose levels, especially at high temperature. A 1/4 symmetric model of a candidate design of graphite brick is established and meshed. The finite element model is shown in Fig. 4. The left side of brick is referred to the reactor core. Two channels are drilled on the brick, the inner one for the control rod and the other for the cold gas coolant. The marked nodes A–E are the representative points for subsequent analysis in Section 5.

The subjected load for graphite component comes from two aspects: high temperature and neutron irradiation. Thus the

---

![Fig. 4. 1/4 model of a candidate design of graphite brick and the meshed grid.](image-url)
boundary condition includes the temperature distribution and neutron dose rate distribution on the whole graphite brick. Fig. 5 shows the temperature distribution at the end of service life (EOL). The core side temperature is 500 °C in the figure as a demonstration. The coolant temperature gradually increases from top to bottom of the core, with temperature range of about 300–750 °C. In the simulation, the temperature of graphite brick’s cold coolant side is fixed to 300 °C, while the temperature of core side is sequentially set to 400 °C, 500 °C, 600 °C and 750 °C.

The fast neutron dose in core is also uneven. The position with largest neutron dose rate is located on the upper 1/3–1/4 part of the core. On the position, the neutron dose on the core side of graphite brick grows linearly during lifetime and reaches $1 \times 10^{22}$ n cm$^{-2}$ (EDN) at EOL. Along the radial axis of the reactor core, the neutron dose decreases rapidly, assumed following an exponential law, $\gamma = \gamma_0 e^{-0.1x}$. $\gamma_0$ is the neutron dose rate on the core side. $x$ is the radial dimension of the graphite brick. The distribution of neutron dose rate versus radius of reactor core is shown in Fig. 6.

Fig. 5. Temperature distribution at EOL (unit: °C).

Fig. 6. Neutron dose rate versus radius of reactor core (in the largest neutron dose rate position).

Fig. 7. Displacement-Y at EOL (corresponding core side temperature: 500 °C).
It should be noted that the peak values of temperature and neutron dose do not appear in the same core position. In the maximum neutron dose position, core side temperature of graphite brick is about 400 °C, while in the high temperature zone, the neutron dose is much lower than the peak value $1 \times 10^{22} \text{n cm}^{-2}$ (EDN) at EOL.

5. Numerical results and discussion

5.1. The deformation of graphite brick

The graphite component apparently deforms when exposed to high temperature and neutron irradiation working conditions. At the area at the reactor core side, the temperature and neutron dose level are higher, resulting in more significant deformation. Figs. 7 and 8 show the displacements at EOL at 500 °C on Y-direction and Z-direction, respectively. The results are calculated with Kennedy model. 500 °C is the theoretical maximum temperature of graphite components located where the neutron dose rate is largest (at about $1/3$–$1/4$ the total core height to the top of core). The dilated frame marks the original shape of graphite brick. The deformation is 15 times enlarged, in order to be displayed more clearly. There is an obvious shrinkage at core side of graphite brick, but only a little unconspicuous expansion at cold coolant side. This is due to the neutron dose level at core side is very high, causing the irradiation-induced shrinkage much greater than thermal expansion. At the cold coolant side, the neutron dose rate becomes extremely low and almost negligible. Thus only a slight thermal expansion can be observed. Fig. 9 shows displacement distributions on line BE. The maximum shrinkage appears at core side. The peak values reach 3.03 mm and 2.11 mm on Y-axis and Z-axis, respectively. It is noteworthy that the dimension of expansion reaches near 0.3 mm at cold coolant side. Therefore a gap between two neighboring graphite bricks should be designed in order to accommodate the amount of expansion.

The displacements at 750 °C on Y and Z-direction are shown in Figs. 10 and 11 with the displayed deformation enlarged by 25 times. The deformation of graphite brick is very different from that at 500 °C. The maximum displacement position is no longer located at core side, but moves outward to node C. It can be seen from Fig. 2 that the irradiation-induced shrinkage at 750 °C is not so large than at 500 °C, and even converts to swell when neutron dose reaches nearly $1 \times 10^{22} \text{n cm}^{-2}$ (EDN). Because of the neutron dose on node C is lower than on node B, the irradiation-induced deformation on node C must be more obvious. The displacement distributions on line BE at 750 °C are shown in Fig. 12. The peak values of shrinkage only reach 1.15 mm and 0.65 mm on Y- and Z-directions, much smaller than at 500 °C.

5.2. The stress distribution and history

Fig. 13 shows the modified equivalent stress distributions at EOL. The temperature on core side is set 500 °C as an example. The stress concentrates on two areas. Nodes A and D are selected as the representative point for each area. Comparing Fig. 13 to Fig. 7, it is clear that the stress accumulates at neither large local deformation area nor small local deformation area, but between the two regions, where the gradients of both temperature and neutron dose
are maximum. This demonstrates that the key factor which causes stress concentration is neither high temperature nor fast neutron irradiation, but the uneven distribution of temperature and neutron dose.

The stress development history versus neutron dose level from 0 to $1 \times 10^{22}$ n cm$^{-2}$ (EDN) at node A is shown in Figs. 14 and 15, calculated with UKAEA model and Kennedy model, respectively. There is a negligible rise and fall in the first half of service life [before neutron dose reaches $5 \times 10^{21}$ n cm$^{-2}$ (EDN)], followed by a rapid increase in the second half. The stress level at 400°C is the lowest. Yet the stress level at 500°C is greater than at 600°C. This indicates that the stress level does not increase monotonically with increasing temperature, but turns to a decreasing function of temperature in the vicinity of 500°C. This conclusion is consistent with Fig. 2 which reveals that the irradiation-induced shrinkage at 500°C is the maximum in the neutron dose range selected. At 750°C, the stress level is much higher than at other temperature points. The peak value is close to (calculated with UKAEA model) or exceed (calculated with Kennedy model) 20 MPa. Such a high stress level may greatly increase the risk of the graphite component's failure. In addition, it is obviously that the results obtained with Kennedy model is larger than the results obtained with UKAEA model. This conclusion has been discussed in some previous studies (Fang et al., 2012). More conservative results can be gained with Kennedy model which is more suitable for real crucial engineering design.

The stress development history at node D is shown in Figs. 16 and 17, also calculated with UKAEA model and Kennedy model, respectively. At 400°C, 500°C and 600°C, the stress rises simultaneously and rapidly in the whole service life except near EOL. The stress level at 400°C is lower, and highest at 500°C. The maximum value of stress is located at neutron dose $8 \times 10^{21}$ n cm$^{-2}$

Fig. 10. Displacement-Y at EOL (corresponding core side temperature: 750°C).

Fig. 11. Displacement-Z at EOL (corresponding core side temperature: 750°C).
(EDN), and reaches to about 10 MPa. The entire historical process of stress development is significantly accelerated at 750 °C, yet the value is far less than at other temperature points. The stress rises to peak value about 5 MPa in the first half of service life, and drops down to no more than 1 MPa at EOL. This means that the stress concentration area around node D once appeared but eventually disappeared at 750 °C.

5.3. The failure probability prediction

The graphite components in HTRs belong to the first Structural Reliability Class (SRC-I) in KTA–3232 (KTA, 1992), with the admissible failure probability value of $10^{-4}$. Figs. 18 and 19 show the failure probabilities of graphite brick corresponding to stress results calculated with UKAEA model and Kennedy model. At 400 °C, 500 °C and 600 °C, the failure probability maintains a stable growth in the lifetime until neutron dose reaches $7 \times 10^{21} \text{ n cm}^{-2}$ (EDN), and slightly decreases thereafter. The failure probability at 400 °C is lower, and higher at 500 °C. This well agrees with stress calculation results in the previous section, as the failure probability can be seen as the sum of the effects of stresses on all nodes to the reliability of the entire graphite component.

The development process of failure probability at 750 °C is apparently different. The failure probability at this temperature is the lowest in the early irradiation stage, but begins to grow rapidly after neutron dose reaches $6 \times 10^{21} \text{ n cm}^{-2}$ (EDN). The results calculated with Kennedy model exceed $10^{-4}$ at neutron dose $9.7 \times 10^{21} \text{ n cm}^{-2}$ (EDN). At 750 °C, both the end of irradiation-induced shrinkage and the start of expansion happen obviously earlier than at other temperature points, which cause a rapidly rising of the failure probability near EOL. Later in the service life, the reliability of graphite component is significantly lower under very high working temperature.
6. Conclusions

The graphite component in HTRs is subjected to high temperature and fast neutron irradiation constantly when the reactor is in operation. High temperature and irradiation cause the deformation of graphite, and their non-uniform distribution creates high internal stress accumulation. In addition, high temperature and irradiation make most of the graphite properties (such as Young’s modulus and coefficient of thermal expansion) change remarkably. In this paper, the behaviors (including deformation, stress distribution and failure probability) of graphite brick are simulated and predicted at several representative in-core temperature levels (400 °C, 500 °C, 600 °C and 750 °C) and neutron dose range 0 to $1 \times 10^{22} \text{n cm}^{-2}$ (EDN). The finite element method is commonly used to study the behaviors of graphite component and predict the structural integrity. A three-dimensional constitutive law is built up, including thermal strain, irradiation-induced strain and creep strain. By means of the implicit algorithm, more stable and reliable results can be obtained. The UKAEA model and Kennedy model are selected as the creep models, while the Weibull model is selected as the reliability model.

The calculation results show that the behavior of graphite component at 400 °C, 500 °C and 600 °C are in good agreement, with the same trend and similar amount of stress and failure probability.
curves. The stress value and failure probability at 400 °C are the lowest. Yet as the irradiation-induced shrinkage at 500 °C is maximum in the selected dose range, the results at 500 °C are always higher than at 600 °C. The deformation, the stress accumulation history and the failure probability development at 750 °C are all different obviously. This is due to the nature of graphite changes violently at such high temperature, especially the change of irradiation-induced deformation. The results in this paper show that the graphite component should be serving in admissible temperature and neutron dose range. If the temperature or neutron dose exceeds the allowable level, the reliability of graphite component would reduce significantly.

It should be emphasized that the effective in-core distributions of temperature and neutron dose are completely different. The place with the highest neutron dose is located on the upper 1/3–1/4 part of the core. The in-core temperature of this location is about 500 °C and the neutron dose is 0–1 × 10^{22} n\text{cm}^{-2} (EDN) from the beginning to the end of service life. The place with highest temperature is located on the bottom of the core, with high temperature (about 700 °C) but low neutron dose [only 0–1.5 × 10^{21} n\text{cm}^{-2} (EDN)]. The entire calculation results, with temperatures 400–500 °C, 500–600 °C and 750 °C and neutron dose range 0–1 × 10^{22} n\text{cm}^{-2} (EDN), are presented in this paper. In the engineering design, suitable parameter ranges must be selected in order to get accurate and credible prediction results.

Acknowledgements

Financial supports for the project from the Tsinghua University Initiative Scientific Research Program (No. 20131089216 and No. 20111080959) and the IAEA’s Coordinated Research Project. The authors also thank the Royal Academy of Engineering, for its support through the Research Exchanges with China and India Award.

References


