Irradiation-Induced Stress Build-Up on Graphite in VHTR

Ji-Ho Kang*, Nam-Il Tak
Korea Atomic Energy Research Institute, Daedeok-Daero 989-111, Yuseong-Gu, Daejeon, Korea
**Corresponding author: jhkang@kaeri.re.kr*

1. Introduction

A Very High Temperature Reactor (VHTR) is a nuclear reactor that uses helium as a coolant with the outlet temperature at around 950℃. A VHTR is inherently safe and highly efficient especially for hydrogen production. The prismatic block type VHTR is considered as a design candidate for the Nuclear Hydrogen Development and Demonstration (NHDD) project[1].

The reactor core is constituted with an assembly of nuclear grade graphite blocks due to high temperature up to 1200 °C in normal operation. In the active core region, a graphite fuel block plays a role of containment of TRISO nuclear fuels as well as formation of coolant channels. Because the fuel block bears nuclear fuels, the graphite material shows noticeable volumetric contraction from the neutron flux and the additional stress build-up is expected due to the difference of volumetric contraction in a neighboring area caused by temperature gradient.

Literature survey showed that structural integrity of the graphite fuel blocks was attained[2]; however detailed irradiation-induced stress analysis for the fuel blocks has not been done yet. In this study, irradiation-induced mechanical material properties and volumetric contraction of nuclear grade graphite, IG-110[3] were introduced. An equivalent thermal expansion coefficient was proposed to apply additional strain from irradiation-induced volumetric contraction. The equivalent thermal expansion coefficient was implemented into a commercial FE program, Abaqus[4] and full 3-D stress analysis was performed on a fuel block under various neutron fluence until the design life time of the block.

2. Methods and Results

2.1 Irradiation-Induced Graphite Material Properties

The irradiated mechanical material properties and volumetric change of IG-110 was shown in the Fig. 1 through Fig. 3. In the reference[3], the change the irradiated material properties were showed in functions of neutron fluence. However, in the FE analysis, the material properties should be listed in functions of temperature. In that reason, the material properties in the figures are shown in functions of temperature. The considered neutron fluence levels were 0, 0.5, 1, 2, and 4×10^{25} n/m^2. Up to 1×10^{25} n/m^2, the material property change is relatively rapid and the fine divisions were chosen in the range. Contrarily coarser divisions were chosen in the range beyond the fluence level of 1×10^{25} n/m^2 where the rate of change lessens. For the volumetric changes of the material are almost linear functions of neutron fluence and the size of the intervals in the divisions does not affect the analysis. In Fig. 3, the equivalent coefficients of thermal expansion were shown instead of volumetric changes of the material.

The irradiation-induced swelling was simulated by equivalent thermal expansion because the commercial FE program, Abaqus cannot implement the irradiation-induced swelling. For example, if we assume that the region of consideration be exposed in neutron fluence of 1×10^{25} n/m^2 and the local temperature be 800 °C, then the resultant volumetric change would be -0.07% and this results in additional compressive strain corresponding to cube root of 0.07% in all direction. This additional strain can be expressed in the form of equivalent coefficient of thermal expansion as shown in the equation (1).

$$\epsilon_{total} = \epsilon_{load} + \alpha(T - T_0) + \alpha_{equiv}(T - T_0) \qquad (1)$$

where, α_{equiv} is the coefficient of thermal expansion equivalent to the irradiation-induced swelling. If we assume that the volumetric change should be an isotropic phenomenon, then the equivalent coefficient of thermal expansion can be expressed in function of volumetric change.

$$\frac{\Delta V}{V} = \frac{(L+\Delta L)^3 - L^3}{L^3} \approx 3\frac{\Delta L}{L}\ , \quad where\ \frac{\Delta V}{V} \approx 0 \qquad (2)$$

$$\alpha_{equiv} = \frac{\Delta L/L}{T - T_0} = \frac{\Delta V/V}{3(T - T_0)} \qquad (3)$$

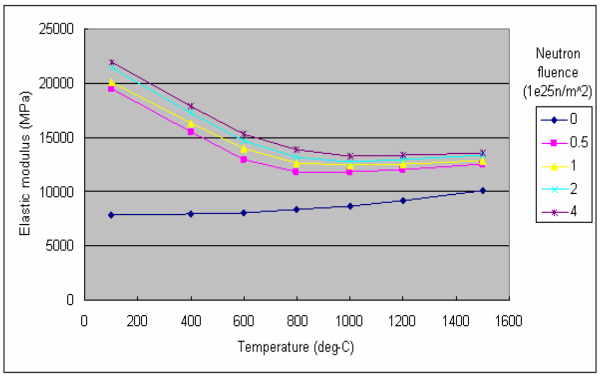

Fig. 1. Irradiated elastic modulus of IG-110.

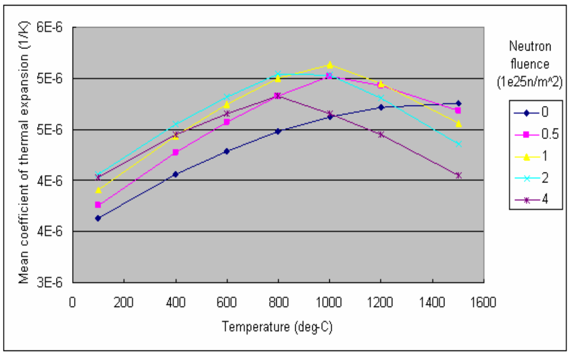

Fig. 2. Irradiated mean CTE of IG-110.

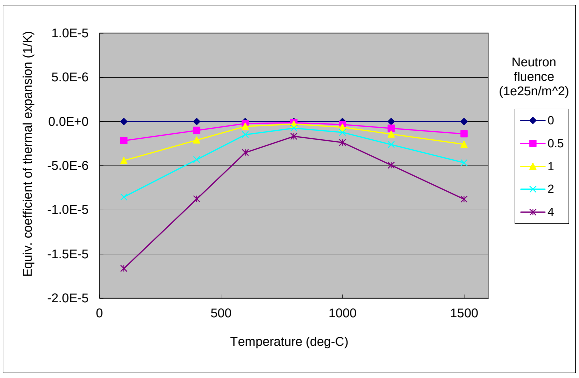

Fig. 3. Irradiated equivalent CTE of IG-110.

2.2 Stress Analysis Results

In the stress analysis, the boundary and loading conditions were identical to the previous study[2]. The top-most fuel block in the fuel assembly was chosen because the block showed maximum stress level in unirradiated environment. Only the graphite block except fuel compacts was considered because the irradiated material properties of fuel compacts were not available and the fuel compacts showed relatively low stress level in the previous study. The stress contours of all neutron fluence cases were shown in the Fig. 4. As shown in the figure, the fuel block suffers severe dimensional contraction when it approaches to its design life (neturon fluence level of 4×10^{25} n/m^2). Also, the differences in irradiation-induced contraction due to the temperature gradient showed considerable stress build-up. The peak values of maximum and minimum principal stresses in the fuel block are shown in Fig. 5. The graph shows that the gradual build-up of stresses as neutron fluence accumulates and the largest tensile stress was 19.93 MPa which is 79% of tensile strength, 25.3 MPa, at the end of design life time of the block.

3. Conclusions

In this study, stress build-up in a fuel block of VHTR was investigated in the irradiation environment investigated. Irradiation-induced material property changes of a nuclear grade graphite, IG-110, was introduced. An equivalent coefficient of thermal expansion to implement irradiation-induced volumetric contraction of the graphite into the commercial FE program was proposed. Stress analysis of a fuel block was done under various neutron fluence levels. The analysis result showed that fuel block suffers severe volumetric contraction and the largest stress is 78% of the strength at the end of design life of the block.

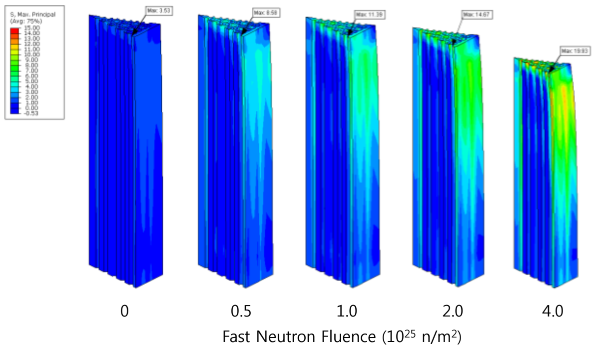

Fig. 4. Deformation and maximum principal stress under various neutron fluence levels.

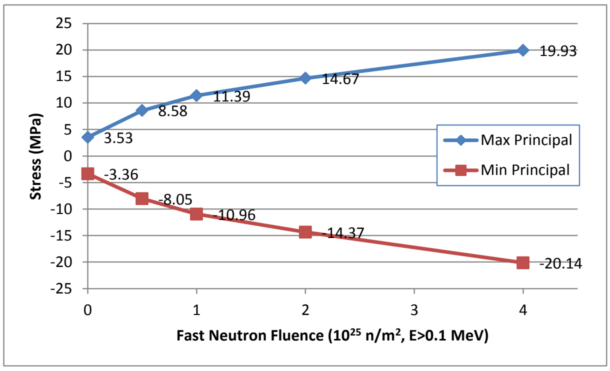

Fig. 5. Maximum and minimum principal stress in fuel block.

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