Title: Effect of Reactor Radiation on the Thermal Conductivity of TREAT fuel

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Abstract

The Transient Reactor Test Facility (TREAT) at the Idaho National Laboratory is resuming operations after more than 20 years in latency in order to produce high-neutron-flux transients for investigating transient-induced behavior of reactor fuels and their interactions with other materials and structures. A parallel program is ongoing to develop a replacement core in which the fuel, historically containing highly-enriched uranium (HEU), is replaced by low-enriched uranium (LEU). Both the HEU and prospective LEU fuels are in the form of UO_2 particles dispersed in a graphite matrix, but the LEU fuel will contain a much higher volume of UO_2 particles, which may create a larger area of interphase boundaries between the particles and the graphite. This may lead to a higher volume fraction of graphite exposed to the fission fragments escaping from the UO_2 particles, and thus may induce a higher volume of fission-fragment damage on the fuel graphite. In this work, we analyzed the reactor-radiation induced thermal conductivity degradation of graphite-based dispersion fuel. A semi-empirical method to model the relative thermal conductivity with reactor radiation was proposed and validated based on the available experimental data. Prediction of thermal conductivity degradation of LEU TREAT fuel during a long-term operation was performed, with a focus on the effect of UO_2 particle size on fission-fragment damage. The proposed method can be further adjusted to evaluate the degradation of other properties of graphite-based dispersion fuel.

Keywords: Thermal conductivity, TREAT, uranium oxide, fission fragments, radiation damage, low-enriched uranium (LEU), highly-enriched uranium (HEU)

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1. Introduction

The Fukushima Daiichi nuclear accident [1] in 2011 stimulated materials development efforts to improve the current fuel design for Light Water Reactors (LWRs). From fuel element to fuel cladding, new concepts in advanced nuclear fuel development have been proposed, studied, and tested. The new fuel systems are expected to have better resistance to damage, melting, and chemical reaction during accident scenarios. To test the performance and accident tolerance of the advanced fuel, it is advantageous to investigate their response to off-normal or accident transients, in addition to the standard in-core fuel performance testing available at High Flux Isotope Reactor at Oak Ridge National Laboratory and Advanced Test Reactor at Idaho National Laboratory. The Transient Reactor Test Facility (TREAT) at the Idaho National Laboratory is now being prepared to restart its operation and will perform transient testing to assist advanced fuel development and safety evaluations [2]. TREAT was operated from 1959 to 1994 [3], during which it was extensively used to generate thousands of reactor transients for various programs [4]. The reactor has not operated since 1994.

The original fuel used at TREAT was graphite-based dispersion fuel with highly-enriched uranium (HEU, 93.1% enriched uranium) [5]. For nonproliferation purposes, a fuel conversion program, supported by U.S. Department of Energy, National Nuclear Security Administration (NNSA), Office of Material Management and Minimization (NA-23) Reactor Conversion Program, is underway to develop a replacement core in which the UO_2 contains only low-enriched uranium (LEU). The LEU core design concepts have a much lower carbon/uranium atomic ratio than the HEU core design, to accommodate the much lower uranium enrichment. To achieve performance similar to the HEU core, the prospective LEU core will thus contain a much higher density of UO_2 particles. If the particle sizes in the LEU fuel are controlled to be smaller

than those of HEU fuel, the number density of particles in the LEU fuel can be significantly higher than in the HEU fuel. The LEU fuel will then contain a larger area of particle-graphite interphase boundaries. As a result, a higher volume fraction of graphite may be exposed to the fission fragments that escape from the UO_2 particles, which thus may induce a higher volume of fission-fragment damage on the graphite matrix of the fuel. To support LEU fuel design, the present study investigates the effect of reactor radiations on the thermal conductivity of graphite-based dispersion fuel. Both neutron- and fission-radiation induced degradations of the fuel are modeled and validated based on the available experimental data. Finally, in order to provide a guideline for fuel design of the LEU TREAT core, the effect of UO_2 particle size on the radiation damage in TREAT fuel is evaluated.

2. Model development

2.1 Neutron damage

Neutron damage on graphite is one of the two radiation damage sources in UO_2 -graphite dispersion fuel. The neutron damage can be considered homogeneous within the fuel, because neutrons, particularly fast neutrons, have a long travel range within the graphite and thus produce an extensive area of radiation damage. The changes in graphite properties with neutron damage depend on irradiation temperature. Usually, irradiation at low temperatures induces a more significant change in graphite properties; neutron-radiation induced degradation of graphite properties becomes more moderate with increasing irradiation temperature. The neutron radiation damage in graphite can be partially recovered by medium or high temperature annealing [6-8]. Many studies of the neutron irradiation effects on graphite have been conducted since the 1950s [6, 7, 9-18]. In this study, we used the measurements performed by Snead and Burchell [6] for modeling, due to their temperature range being similar to that within the TREAT cores and to their relatively complete data set. Figure 1 shows the normalized thermal

conductivity of neutron-irradiated graphite ($\frac{K_{irr}^n}{K_{unirr}^n}$, where K_{irr}^n is the thermal conductivity of neutron-irradiated graphite, and K_{unirr}^n is the thermal conductivity of unirradiated graphite) at 150° C. The included error bars account for the range of normalized thermal conductivity for several types of graphite and carbon/carbon composites. As seen, the neutron-radiation induced thermal-conductivity degradation is not highly sensitive to the type or quality of the graphite. By assuming damaged and undamaged regions are evenly distributed throughout the graphite, the normalized thermal conductivities were fitted by a natural exponential function. The equation to compute the neutron-radiation degradation of graphite was then built:

$$\frac{K_{irr}^n}{K_{unirr}^n} = 0.88e\left(-\frac{\Phi}{9.76 \times 10^{18}}\right) + 0.12, \quad \text{Eq. 1}$$

where Φ (unit: neutrons/cm²) represents the fast neutron fluence (neutron energy >0.1 MeV) of the graphite. The blue curve in Fig. 1 shows the plot of Eq. 1.

Fig. 1. Neutron irradiation-induced degradation of thermal conductivity; the red data points as a function of fast neutron fluence (Φ) were converted from the damage level in units of dpa [6]. The blue curve shows the best fit to the data (using Eq. 1).

2.2 Fission-fragment damage, and coupling with neutron damage

Fission fragments, unlike neutrons, have a short travel range in graphite. However, they induce a greater level of damage due to their higher kinetic energy. Previous studies on graphite-based dispersion fuel have indicated that the fission-fragment damage depends upon the size of fissionable particles (specifically, the particles containing U_{235}) [8, 19-23]. For a fixed amount of fissile material within the fuel, decreasing the size of the fissile-containing particles increases the number density of particles and thus the total volume of fission-fragment damaged regions near particles. Meanwhile, a larger fraction of fission fragments emerge from fissionable particles when the particle size is reduced. Therefore, unlike neutron radiation damage, which is homogeneous in the graphite matrix regardless of particle size, fission-fragment damage is a function of particle size for a fixed volume or mass density of the fissionable particles in the fuel. Harrison [24] proposed a theoretical method to estimate the effect of particle size on fission-fragment damage of dispersion fuel. His theory can be divided into four steps: (1) computation of the volume fraction of the fission-fragment damaged graphite; (2) computation of the relative exposure in the damaged regions of the graphite; (3) computation of the property changes (e.g., thermal conductivity and electrical resistivity) in the damaged regions of the graphite; and (4) determination of the changes in bulk properties of irradiated graphite, taking into account the results of the three aforementioned computations. This method is overelaborate, and the contribution of some factors (e.g., the “effective exposure” involved in step 2 is negligible in the final results [24]). Additionally, the results of this completely theoretical method did not agree well with the experimental data [23, 24]. With this in mind, we adapted Harrison’s method to specifically evaluate the reactor-radiation induced thermal conductivity degradation of graphite-based dispersion fuel. Detailed computation procedures are as follows:

- i. Determination of the volume fraction of fission-fragment damaged regions in the graphite matrix.

The volume fraction of fission-fragment damaged regions in graphite (defined as V_{irr}^f) is the most important factor in these calculations and can be obtained by [24]:

$$V_{irr}^f = V_{UO_2} \left[\left(1 + \frac{2R_m}{D} \right)^3 - 1 \right], \quad \text{Eq. 2}$$

where V_{UO_2} is the volume fraction of UO_2 in the graphite, D is the average diameter of UO_2 particles, and R_m is the average damage range of fission fragments in graphite. As the damage ranges of fission fragments in graphite were estimated to be $\sim 17 \mu\text{m}$ for light fragments and $\sim 12 \mu\text{m}$ for heavy fragments [22], we herein use $\sim 14 \mu\text{m}$ as the average damage range of the fission fragments. Figure 2 shows the volume fraction of fission-fragment damaged regions in graphite as a function of particle size for fuels with different volume fractions of UO_2 . Note that the V_{UO_2} of 3.9×10^{-4} is the volume fraction of the UO_2 particles in HEU TREAT fuel (detailed assumptions and procedures to attain this value can be found in ref. [19]) and V_{UO_2} of 1.8×10^{-3} is the volume fraction of the UO_2 in a prospective LEU TREAT fuel, assuming an LEU enrichment of 19.75% and the same amount of fissionable material (U_{235}) in both HEU and LEU fuels.

Fig. 2: The volume fraction of fission-fragment damaged regions in graphite as a function of particle size.

- ii. Determination of the amount of fission fragments emerging from UO_2 particles and damaging the graphite matrix.

The ratio of fission fragments escaped from the UO_2 particles to the total number of fission fragments (a ratio defined as F) can be calculated as a function of particle size by [23, 24]:

$$F = \frac{3R_f}{2D} \left(1 - \frac{R_f^2}{3D^2}\right), \quad \text{Eq. 3}$$

where R_f is the travel range of fission fragments in UO_2 particles (its value is $\sim 11.4\mu\text{m}$ [22, 23]). Figure 3 shows the fraction of fission fragments escaped from the UO_2 particles. The computed curve is consistent with the results given in ref. [20]. The small particles with diameters of several μm s contain almost no fission fragments. As the particle size increases, a higher fraction of fission fragments are stopped within the UO_2 particles and a lower fraction escape and damage the graphite. All previous studies aimed to measure fission fragment damage on graphite used ultra fine fissionable particles (with sizes of $< 2\mu\text{m}$ [8, 21-23]) not only because those fine particles produced homogeneous fission-fragment damaged regions within the graphite, but also because almost all the fission fragments emerged from fine particles and dissipated their energy in the graphite.

Fig. 3: The fraction of fission fragments escaped from the UO_2 as a function of particle size.

- iii. Determination of changes in the thermal conductivity of the fission-fragment damaged region within the graphite.

As with neutron-radiation induced thermal conductivity degradation [6, 18], changes in the thermal conductivity of graphite with fission-fragment irradiation are temperature dependent. Hetrick et al. [21] conducted a series of experiments to determine the effect of reactor radiation on the thermal conductivity of uranium-impregnated graphite in the temperature range of ~ 25 to $\sim 340^\circ\text{C}$. The reported values of relative thermal resistivity were converted to normalized thermal conductivity and are presented in Fig. 4. The scattering of the data is due to the wide range of irradiation temperatures. As the fission-fragment damage is always accompanied by fast neutron damage, Eq. 1 is further applied as a component of the equation to fit the normalized thermal conductivity with reactor irradiations ($\frac{K_{irr}^f}{K_{unirr}}$); the best fit of the data is shown in Eq. 4:

$$\frac{K_{irr}^f}{K_{unirr}} = \frac{K_{irr}^n}{K_{unirr}} \times (0.8e^{(-\frac{BU}{1.6 \times 10^{-2}})} + 0.2), \quad \text{Eq. 4}$$

where BU is the burnup, or energy density (unit: KWH/cm^3) dissipated within the fuel.

The neutron-radiation induced thermal conductivity degradation factor (i.e. $\frac{K_{irr}^n}{K_{unirr}}$) and reactor-radiation induced thermal conductivity degradation factor (including both neutron- and fission-fragment radiations), defined as $\frac{K_{irr}^f}{K_{unirr}}$, are plotted as yellow (upper) and red (lower) curves, respectively, in Fig. 4. As seen, the contribution of neutron damage to the total loss of thermal conductivity is small compared to the fission-fragment damage contribution.

Fig. 4: Reactor irradiation-induced degradation of thermal conductivity of uranium-impregnated graphite in the temperature range of ~ 25 to $\sim 340^\circ\text{C}$

- iv. Computation of the effective thermal conductivity of the fuel, taking into account both neutron and fission-fragment radiation damage on graphite.

With the previous computation steps, the effective thermal conductivity (K_{irr}^{eff}) of the bulk fuel as a function of burnup can be calculated by:

$$K_{irr}^{eff} = \frac{V_{irr}^f K_{irr}^f}{F} + (1 - V_{irr}^f) K_{irr}^n \quad \text{Eq. 5}$$

$\frac{V_{irr}^f K_{irr}^f}{F}$ is a term that models the contribution of the fission-fragment damaged regions in the graphite, $(1 - V_{irr}^f) K_{irr}^n$ is the term that models the contribution of the remaining parts of the graphite that were only exposed to neutron radiations.

3. Validation

3.1 Effect of UO_2 particle size

Kernohan [23] reported irradiation tests on graphite samples including four different sized UO_2 particles at the Oak Ridge National Laboratory (ORNL) graphite reactor. The average particle sizes were: (1) $586 \pm 16 \mu\text{m}$, (2) $334 \pm 7 \mu\text{m}$, (3) $94 \pm 0.2 \mu\text{m}$, and (4) $\sim 16\text{-}20 \mu\text{m}$. The particle sizes of samples (1)-(3) were microscopically determined, while the particle size of sample (4) was estimated. Graphite samples without any UO_2 particles, treated as control samples only exposed to neutron damage, were irradiated with the UO_2 particle-containing graphite samples. The irradiation temperature was in the range of 30°C to 88°C . The neutron fluence was reported to be $\sim 1.8 \times 10^{18}$ neutrons/ cm^2 , with a fraction contributed by fast neutrons. The ratio of total neutron flux to fast neutron flux ($\phi_{\text{total}} / \phi_{(En \geq 0.1 \text{ MeV})}$) was estimated to be ~ 2.4 ; this value was used in the present simulation. This estimation was based on the reported neutron fluxes in the ORNL graphite reactor [25]. The energy dissipated by each of the samples (or the burnup) was $1.8 \text{ KWH}/\text{cm}^3$. The normalized thermal conductivities (K_{irr}^{eff}/K_{unirr}) converted from the experimental data are plotted in Fig. 5, showing the effect of particle size on radiation induced thermal conductivity degradation. By using Eq. 5, K_{irr}^{eff}/K_{unirr} was calculated using either measured or computed K_{irr}^n/K_{unirr} . The computed K_{irr}^n/K_{unirr} using Eq. 1 was 0.93, which agrees well with the value of 0.9 measured from the control sample. Both measured and computed values of K_{irr}^n/K_{unirr} were further applied to Eq. 5 to attain K_{irr}^f/K_{unirr} . The results using both values are plotted in Fig. 5 (upper curve is from computed values) and show a reasonable agreement with the measured K_{irr}^{eff}/K_{unirr} of the graphite with different average particle sizes. It is not surprising that there is a better agreement for the simulation using the measured K_{irr}^n/K_{unirr} . However, as experimental data for neutron-irradiated graphite samples is usually unavailable for a specific type of graphite-based dispersion fuel, the method using the

computed K_{irr}^n / K_{unirr} is still useful, especially for fuel that contains UO_2 particles with an average size of $20\ \mu\text{m}$ or smaller.

Fig. 5: Reactor-radiation induced thermal conductivity degradation of graphite with four different sizes of UO_2 particles, shown with the simulation results.

3.2 Thermal conductivity degradation of HEU TREAT fuel

During the design stage of the TREAT HEU core in the 1950s, two fuel samples were encapsulated and irradiated in the CP-5 reactor at Argonne National Laboratory [5]. The fuel samples finally achieved an integrated flux of 1.27×10^{18} neutrons/cm² at an average temperature of 290°C. The ratio of total neutron flux to fast neutron flux in the CP-5 reactor was reported to be ~ 1.64 [26], and this number was used in the present simulation. Because the volume fraction of UO_2 particles in the original HEU fuel was small (estimated to be about 3.89×10^{-4} by assuming full density of UO_2 (10.97 g/cm³)) and the average UO_2 particle size in the

fuel was relatively large (reported to be about 16-20 μ m in diameter [5]), the fission fragment damaged region within the fuel is very small (see Fig. 1). Hence, the changes in thermal conductivity of the HEU TREAT fuel were mainly attributed to the fast neutron damage. The thermal resistivity of the fuel samples was reported to increase 7% after the reactor irradiation, corresponding to K_{irr}^{eff}/K_{unirr} of about 0.935. The simulated value of K_{irr}^{eff}/K_{unirr} using Eq. 5 is in the range of 0.927-0.935 for the average particle size of 16-20 μ m, which agrees well with the measured value in the experiment.

4. Prediction of TREAT fuel

4.1. Reactor radiation damage during a long-term operation of TREAT

TREAT was operated for about 35 years using the original HEU fuel. The estimated average annual energy production in the TREAT HEU core was 350,000 MJ, corresponding to $\sim 1.1 \times 10^{22}$ fissions per year [27]. The average annual burnup per unit volume and core-average neutron fluence were estimated to be $\sim 2.65 \times 10^{-2}$ KWH/cm³ and $\sim 2.21 \times 10^{18}$ neutrons/cm² (fast neutron fluence: $\sim 4.13 \times 10^{17}$ neutrons/cm²), respectively. To achieve the same test sample total energy depositions possible in the current HEU core, a prospective LEU core may need to operate at higher powers than the HEU core for some sample and experiment vehicle combinations. For this analysis, an increase of 10% more annual energy production in the LEU core was therefore assumed. For the current LEU conceptual design [28], the corresponding annual burnup per unit volume and core-average neutron fluence of the LEU core are thus assumed to be $\sim 2.92 \times 10^{-2}$ KWH/cm³ and $\sim 2.38 \times 10^{18}$ neutrons/cm² (fast neutron fluence: $\sim 4.15 \times 10^{17}$ neutrons/cm²), respectively. Table 1 summarizes the parameters used in the simulation of reactor-radiation induced thermal conductivity degradation of both HEU and LEU TREAT fuels. The changes in thermal conductivity of both types of fuels are then calculated using Eq. 5. The simulation results for the fuels with UO₂ particle size of 20 μ m are shown in Fig. 6. No significant difference was

observed in thermal conductivity degradation between the two types of fuels. The bulk thermal conductivity of both fuels decreased $\sim 50\%$ after usage for 20 years and $\sim 72\%$ after 40 years.

Table 1: Input parameters for the simulation of reactor-radiation induced thermal conductivity degradation

Fuel Types	Volume fraction of the UO_2 in the graphite (V_{UO_2})	Annual burnup (BU , KWH/cm^3)	Fast neutron fluence per year (Φ , neutrons/ cm^2)	UO_2 particle size (D , μm)
HEU	3.9×10^{-4}	2.65×10^{-2}	4.13×10^{17}	3-20
LEU	1.8×10^{-3}	2.92×10^{-2}	4.15×10^{17}	3-20

Fig. 6: Thermal conductivity degradation of HEU and LEU TREAT fuels with UO_2 particle size of $20 \mu\text{m}$.

4.2. Effect of UO_2 particle size

As shown in the earlier analyses, the fission-fragment radiation damage is highly sensitive to UO_2 particle size. To determine the effect of particle size, thermal conductivity degradations of both HEU and LEU TREAT fuels with different sized particles were simulated. Figures 7 and 8 show the simulation results of HEU and LEU fuels, respectively. Due to the greater number of

UO₂ particles in the LEU fuel (and so also the higher fraction of fission-fragment damaged regions), the thermal conductivity degradation is more sensitive in the LEU fuel than in the HEU fuel. During a long-term operation, similar levels of thermal conductivity loss were found in HEU fuel with particle size 5-20µm. For the HEU fuel with 3 µm sized particles, the thermal conductivity degradation is much more significant: the bulk thermal conductivity decreases ~40% after usage for 2 years, while the fuel conductivity with particle sizes of 5 µm or larger decreases less than ~15% (Fig. 7). Here, we defined a “critical particle size” to represent the minimum particle size of the fuel to maintain a similar level of bulk thermal conductivity degradation; in other words, vast decreases in thermal conductivity can be expected for fuel with particle sizes smaller than the critical particle size. The critical particle size for the HEU fuel is considered to be 5 µm, while this number is doubled (i.e., 10 µm) for the LEU fuel (for the assumptions made in this study), as shown in Fig. 8. The effect of particle size on bulk thermal conductivity degradation and the determination of critical particle size can be better presented when we look at snapshots in different reactor operation time periods (Figs. 9 and 10). The HEU fuel shows a better tolerance for smaller sized UO₂ particles than the LEU fuel, in terms of maintaining the thermal conductivity. Note that the simulated value (in the 40th year) of the LEU fuel with 3µm particles has reached a saturated value of $\frac{K_{irr}^{eff}}{K_{unirr}} \approx 0.05$, which corresponds to the minimum relative thermal conductivity measured in the experiments [21].

Fig. 7: Thermal conductivity degradation of HEU TREAT fuels with different sized UO_2 particles.

Fig. 8: Thermal conductivity degradation of LEU TREAT fuels with different sized UO_2 particles.

Fig. 9: Snapshots of thermal conductivity degradation of HEU TREAT fuels with different sized UO_2 particles after 1, 3, 10 and 40 years.

Fig. 10: Snapshots of thermal conductivity degradation of LEU TREAT fuels with different sized UO_2 particles at 1, 3, 10 and 40 years.

4. Discussion

As indicated in Fig. 6, if the UO_2 particle sizes in the prospective LEU TREAT fuel were similar to those in the HEU fuel (evaluated to be about 16-20 μm [5]), the thermal conductivities of both types of fuel would be expected to experience almost the same level of radiation-induced degradation during a long-term operation. The insignificant difference stems from the small volume fractions of fission-fragment damaged regions (V_{irr}^f) in both types of fuels. In the case of the 20 μm UO_2 particles (see Fig. 1), the values of V_{irr}^f are 0.5% in HEU fuel and 2.3% in LEU fuel, both of which are too small to impact the bulk properties. For the graphite-based dispersion fuel with such large sized particles, the major contribution to the change in bulk thermal conductivity is neutron radiation damage, which can be partially recovered by annealing at elevated temperatures. As the HEU TREAT fuel has been used for 40 years without any major reported issues related to the thermal conductivity degradation of the fuel, the radiation induced thermal conductivity degradation should not be a concern for the prospective LEU fuel that contains similarly sized UO_2 particles.

Smaller UO_2 particles are beneficial to the TREAT fuel in many ways. First of all, most of the fission fragments escape from sufficiently small particles (see Fig. 3) and dissipate their energy in the graphite, and thus the graphite temperature can respond quickly to the reactor power increase. This means the “time lag” [20], defined as the displacement in time between the time-temperature curve for the UO_2 particle, and the time-temperature curve for its surrounding moderator, is minimized for fuel that contains small UO_2 particles. The temperature of UO_2 particles would not be a concern, because (1) the heat produced per particle is small and can be released easily from the particles [19], and (2) the energy carried by the fission fragments directly dissipates into the graphite. However, as demonstrated in this study of the effect of

particle size, the effective or bulk thermal conductivity of the TREAT fuel can be substantially degraded during long-term reactor operation when small sized UO_2 particles are used. Therefore, the UO_2 particle size dependence of thermal conductivity degradation is contradictory to the size dependence of the “time lag”: the smaller the particle, the higher reactor radiation damages and loss in thermal conductivity. It has been shown that if the prospective LEU TREAT fuel contains the same sized UO_2 particles as those in the original HEU TREAT fuel, the thermal conductivity degradation of the LEU fuel would not produce a significant change in reactor performance or safety. It would be still acceptable even if the average particle size in the LEU fuel is half of that in the HEU fuel. Recent pre-conceptual development of an extruded LEU fuel demonstrated an advanced manufacturing technique that produces a homogeneous distribution of ultra fine UO_2 [29]. In order to minimize agglomeration, UO_2 can be ball milled for many hours prior to blending with the graphite powders. The extruded fuel contained milled UO_2 particles a few μm or smaller in size, similar to the size of the particles impregnated by chemical methods into the graphite [8, 21-23]. If LEU fuels with ultra fine UO_2 particles were used, the reactor-radiation damage on the graphite would be significant with a dramatic decrease in thermal conductivity in the first few years. In this regard, the loss in thermal conductivity of the fuel should be considered a factor when performing large-scale thermal analysis of the reactor core. In previous thermal analyses, however, the conductivity change of the fuel has generally been computed to only impact the cool down rate, and not peak temperature, of the fuel. Additionally, the conductivity only has appreciable effect on this rate for the design basis accident scenario in which there is no cooling air flow (with negligible impact on normal operation with flow).

As TREAT is intended to produce transients to support accident tolerant materials development, the reactor core is expected to be exposed to relatively high temperatures frequently during operation. Both neutron and fission-radiation damage on graphite at high temperatures has been

observed to be moderate compared to the severe damage at low temperatures [8, 22]. Moreover, annealing at reactor operating temperatures can recover partial neutron damage of graphite [5, 6]. As the proposed method was developed based on data obtained from low-temperature irradiation experiments, the simulated thermal conductivity degradation of TREAT fuel is conservative and can be considered a lower boundary or safety margin for the thermal conductivity of the irradiated fuel.

Finally, the proposed method can be further adjusted to evaluate the degradation of other properties of graphite-based dispersion fuel. Besides thermal conductivity, many properties of graphite change with radiation damage, for example, the elastic modulus and electrical conductivity (or resistivity). However, more experimental data will be needed to determine the impact of fission-fragment damage on those properties, so that their correlations (like Eq. 5 for thermal conductivity) to the fuel with various amounts and particle sizes of fissionable materials can be built following the same steps given in section 2.

5. Conclusions

In this study, the thermal conductivity degradation of graphite-based dispersion fuel was analyzed. A semi-empirical method to model the relative thermal conductivity with reactor radiations was proposed, validated, and applied to the prediction of LEU TREAT fuel. The following conclusions were reached:

- (1) Within the range of fission fragments escaped from the fuel particle, the fission-fragment radiation damage is much more severe than the neutron radiation damage.
- (2) The level of reactor radiation damage in the graphite-based dispersion fuel is particle size dependent because the fission-fragment damage volume is controlled by the particle size when the volume/mass density of the fissionable particles in the fuel is fixed.

- (3) If the UO_2 particle size in the prospective LEU TREAT fuel were similar to that in the HEU fuel (i.e., 16-20 μm), the thermal conductivities of both types of fuels are expected to experience almost the same level of radiation-induced degradation during long-term operation.
- (4) Based on the assumptions made in this analysis, the critical particle sizes, which represent the minimum particle size of the fuel to maintain a similar level of bulk thermal conductivity degradation, are 5 μm and 10 μm for HEU and LEU TREAT fuel, respectively.
- (5) With sufficient experimental data to build the models, the proposed method can be further adjusted to evaluate the degradation of other properties of graphite-based dispersion fuel.

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