ANALYSIS OF THE THERMAL BEHAVIOR OF DUPIC FUEL

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ABSTRACT

In order to evaluate the thermal behavior of DUPIC fuel, the irradiation test of simulated DUPIC fuel was accomplished at the HANARO research reactor, and the post-irradiation analysis of irradiated pellets was performed at IMEF in KAERI. To find out the temperature distribution of DUPIC fuel pellets, the FEMAXI-IV code that can analyze the performance of a fuel rod was selected and the thermal conductivity model suitable for characteristics of DUPIC fuel was proposed. Considering HANARO operation conditions, the heat transfer routine was also revised. The temperature profile calculated with FEMAXI-IV was compared with the result of post-irradiation analysis. This comparative study shows that the estimate of FEMAXI-IV agreed moderately well with the result of post-irradiation examination.

1. INTRODUCTION.

The DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors) fuel cycle is an alternative fuel cycle technology for reusing spent PWR fuel via direct refabrication. Such utilization of spent PWR fuel, instead of direct disposal, can save more than 30% of natural uranium requirement and reduce overall spent fuel arisings by 2/3 per unit electricity generation compared with PWR and CANDU once-through fuel cycles. The DUPIC fuel cycle bears excellent proliferation-resistance features because sensitive materials in spent PWR fuel are not separated in the direct refabrication process that is performed in a heavily shielded area. Thus, this feature endows remarkable resistance to proliferation[1].

In order to evaluate the in-pile behavior of DUPIC fuel, the irradiation test of simulated DUPIC fuel was carried out at HANARO in KAERI(Korea Atomic Energy Research Institute) from August 4, to October 4, 1999. The objective of the irradiation test was to obtain the basic data of the performance of DUPIC pellets and to suggest the DUPIC pellet design for the fuel fabrication [2].

FEMAXI-IV is a computer code for the analysis of thermal and mechanical behavior of light water reactor fuel rods during steady-state and transient conditions. In this paper, this code was selected to find out the temperature distribution of DUPIC pellets under irradiation and a thermal conductivity model suitable for characteristics of DUPIC fuel was proposed. The heat transfer routine considering HANARO operation conditions was also revised. The estimate of FEMAXI-IV was compared with the results of post-irradiation examination.

2. EXPERIMENTAL

2.1 DUPIC irradiation rig

One group of pellets for the irradiation test were fabricated by using SEU(Slightly Enriched Uranium), whose enrichment was 1.47% U-235. The other groups of pellets of 2.19% enrichment were fabricate by using SEU, which has been mixed with simulated DUPIC powder. The simulated DUPIC powder contains the surrogate fission products corresponding 35,000 MWd/tU discharge burnup.

The DUPIC irradiation rig can accommodate 3 mini-elements, and each contains 5 pellets. The material for cladding and end-caps is the 316L stainless steel with wall thickness of 0.66mm. The length of mini-elements is 159.82mm, and the irradiation rig was designed for the coolant of HANARO reactor to be directly passed through the surroundings of mini-elements. The specification of a DUPIC mini-element is shown in Figure 1. The DUPIC mini-elements is situated at the central part of the irradiation rig, and arranged to be located axially as the center of the mini-element coincides with 12.5cm just above the reactor core center when it was load into the OR4 hole during the irradiation test. Design specifications of the DUPIC pellets were shown in Figure 2.

2.2 Irradiation test in HANARO

The simulated DUPIC fuel pellets were irradiated in the OR4 hole following the HANARO operation procedure, which is operated for 3 days and halted for 4 days under normal conditions. The temperature and pressure of the coolant at HANARO is 313 °K and 0.4 MPa, respectively. HANARO had been operated at 15MW power for 3 weeks and at 20MW power for the rest of the irradiation period until discharge. Linear heat rate value that corresponds to the control rod locations in HANARO was estimated with HANAFMS (HANARO Fuel Management System), and used as basic reference data in temperature profile calculations. The estimated linear heat rates range from is 20.73 to 65.32KW/m. The discharge burnup of simulated DUPIC fuel was evaluated as 1,770MWd/tU.

The post-irradiation examination of the irradiated DUPIC mini-elements was performed at IMEF (Irradiated Materials Examination Facility) to observe the microstructure changes and the dimensional deformation of the elements.

3. IMPROVEMENT OF THE PERFORMANCE ANALYSIS CODE

3.1 Selection of the code

Many codes for the study of the design and performance analysis of a nuclear fuel rod are currently available, such as ELESTRES[3], FRAPCON[4], EXBURN-1[5], TRANSURANUS[6], COPERNIC[7], ENIGNA, etc. The FEMAXI-IV code [8], which has been chosen for the purpose of the analysis of DUPIC fuel in HANARO, was first developed to analyze in-pile behavior of the light water reactor fuel rod. The motives of selecting the FEMAXI-IV were (1) it includes all properties necessary for the safety analysis estimation for the irradiation test at HANARO and (2) it can describe better the HANARO research reactor environment than other codes do. This code has already acquired the certification and validation through Battelle High Burnup Experiment Program and Halden Reactor Program.

In order to apply the FEMAXI-IV code to the HANARO irradiation test of DUPIC fuel, thermal conductivity routine and heat transfer coefficient routine have been revised.

3.2 Thermal conductivity model of DUPIC fuel

The thermal conductivity of nuclear fuel is the most important factor for the analysis of in-pile thermal behavior. The model of a DUPIC pellet was developed based on the experimental data recently obtained using simulated reference DUPIC fuel, which has the reference composition determined from via the compatibility study of DUPIC fuel on CANDU[9]. The formula proposed for sintered stoichiometric uranium dioxide proposed by Lucuta et al. was also used for considering the burnup dependence[10].

Referring to the expression developed by Harding and Martin (1989) for UO_2[11], the thermal conductivity model of unirradiated fully dense DUPIC fuel pellets can be expressed as follows:

$$K_{D0} = \frac{1}{A + BT} + \frac{C}{T^2}\exp\left(-\frac{D}{T}\right) \tag{1}$$

where K_{D0} = conductivity of unirradiated fully dense DUPIC fuel pellets [W/m-K]
T = temperature [K]
A, B, C and D = constants.

The constants A, B, C and D were obtained through the nonlinear least-squares procedure using the experimental data for simulated reference DUPIC pellets. The Nedler-Mead simplex algorithm was used to minimize the sum of errors [12]. Consequently, we suggested the following equation is suggested as the thermal conductivity model of an unirradiated fully dense DUPIC pellet.

$$K_{D0} = \frac{1}{0.1044 + 2.058 \times 10^{-4} T} + \frac{1.327 \times 10^{10}}{T^2}\exp\left(-\frac{19359}{T}\right) \tag{2}$$

As it has been known for sintered urania, irradiation damage and the progressive buildup of fission products with increasing fuel burnup will also reduce the thermal conductivity of DUPIC fuel pellets. It can be thought that they have the same effects on the conductivities of sintered urania and DUPIC fuel. So, the formula proposed by Lucuta et al. that includes the effects of temperature, radiation, fuel burnup and porosity was adopted to assess the degradation of the thermal conductivity of DUPIC pellets with increasing burnup. Consequently, the burnup-dependent thermal conductivity for irradiated DUPIC pellets was proposed as equation (3):

$$K_D = K_{D0} \cdot f_d \cdot f_p \cdot f_m \cdot f_r \tag{3}$$

where K_{D0} = conductivity of unirradiated, fully dense DUPIC fuel pellets [W/m-K]

f_d = effect of the dissolved fission products

$$= \left[\frac{1.09}{B^{3.265}} + \frac{0.0643}{\sqrt{B}}\sqrt{T}\right] \arctan\left[\frac{1.09}{B^{3.265}} + \frac{0.0643}{\sqrt{B}}\sqrt{T}\right]^{-1}$$

B = burnup in atom% (1 atom% = 9.383 GWd/tU at 200 MeV/fission)

T = temperature [K]

f_p = effect of the precipitated fission products

$$= 1+\frac{0.019B}{3-0.019B}\left[1+\exp\left(-\frac{T-1200}{100}\right)\right]^{-1}$$

f_m = effect of porosity

$$= \frac{1-p}{1+0.5p}$$

p = porosity fraction (as-fabricated plus swelling)

f_r = radiation effect

$$= 1-\frac{0.2}{1+\exp\left(\frac{T-900}{80}\right)}$$

Figure 4 shows the proposed thermal conductivity of freshly prepared DUPIC pellets, K_{D0}, along the temperature variation. As shown in this figure, it is noted that the thermal conductivity of DUPIC fuel is somewhat lower than that of UO_2.

3.3 Heat Transfer Coefficient

The surface temperature of the cladding is obtained from the heat transfer differential equation for the fuel. To obtain this solution, Jens-Lottes empirical equation(BWR) and Dittus-Boilter(PWR) equation can be used. In the HANARO reactor case, the Dittus-Boilter(PWR) model has been chosen, since because boiling of cooling water does not occur.

4. RESULTS AND DISCUSSION

4.1 Temperature distribution

According to the power history shown in Figure 3, the temperature distribution of simulated DUPIC fuel was calculated with FEMAXI-IV.

A photograph of the cross section of an irradiated mini-element is shown in Figure 5, and the microstructures at the central area and near the surface are in Figure 6. Considerable grain growth was observed in the center region of an irradiated DUPIC pellet. While the average as-fabricated grain size was measured to be 4.58 mm, the average grain size in this region after irradiation was 11.1 mm, which is twice as much as that before irradiation. This implies that the temperature of the center region is higher than normal uranium fuel. The average grain size near the surface was measured to be 4.77 mm after irradiation. The temperatures in this region are too low to cause any observable restructuring of the simulated DUPIC fuel.

4.2 Diametral deformation

The maximum ridge height of the DUPIC mini-element during the irradiation period was calculated to be 57.7 mm using FEMAXI-IV. As represented in Figure 7, the profilometry results of the DUPIC mini-element shows that maximum and minimum ridge heights were measured to be 60 mm and 41 mm, respectively. Therefore, it is thought that the estimate for diametral deformation calculated by FEMAXI-IV agrees quite well with the measured values from the post-irradiation examination.

5. CONCLUSION

In order to evaluate the in-pile thermal behavior of DUPIC pellets at the HANARO, the FEMAXI-IV code was selected and revised. The thermal conductivity model of a DUPIC pellet was proposed based on the experimental data obtained using simulated DUPIC fuel. The routine for heat transfer between the coolant and a fuel rod was also modified considering the irradiation rig design and the HANARO operation condition.

The predictions of the performance code in terms of temperature profiles and dimensional changes were compared with the results of the post-irradiation examination. They agreed moderately well considering various kinds of uncertainties contained in the current thermal conductivity model of DUPIC fuel and linear element ratings in the test hole of HANARO.

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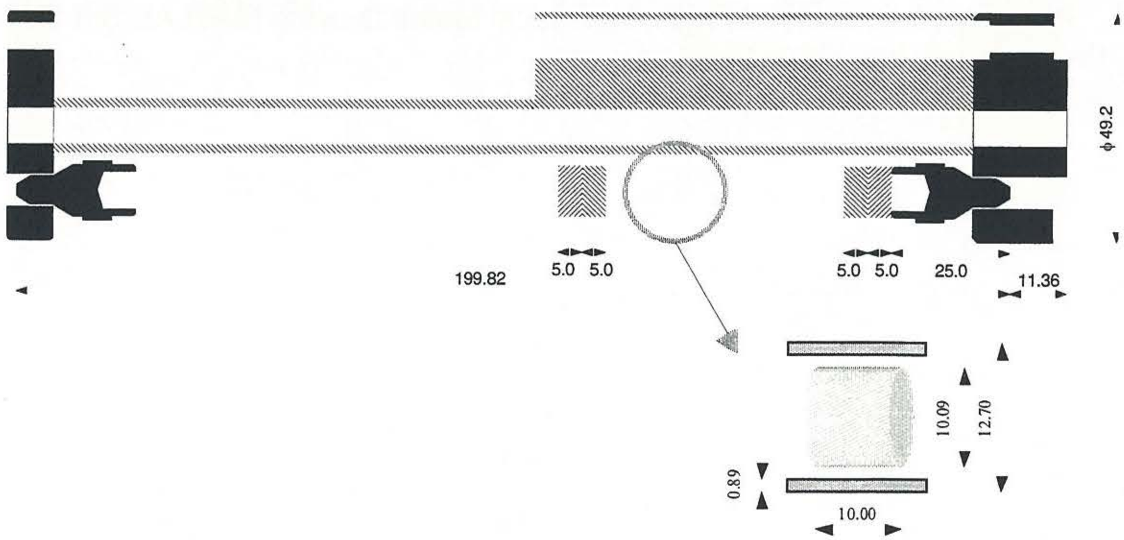


FIG. 1 DESIGN SPECIFICATION OF DUPIC MINI-ELEMENT

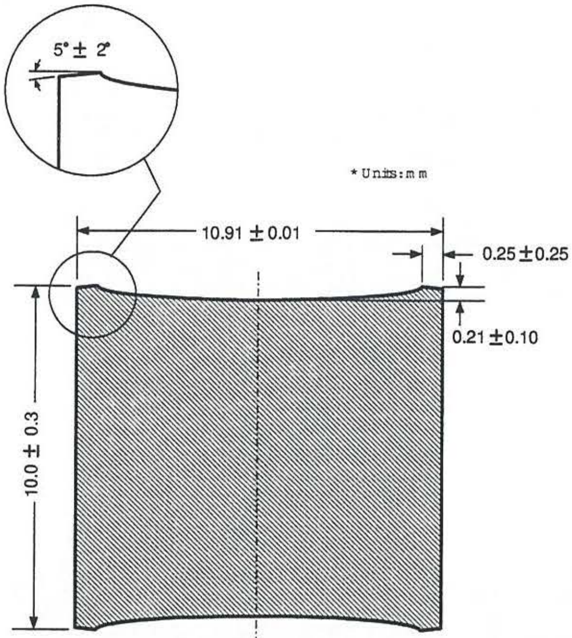


FIG. 2. DESIGN SPECIFICATION OF SIMULATED DUPIC PELLET

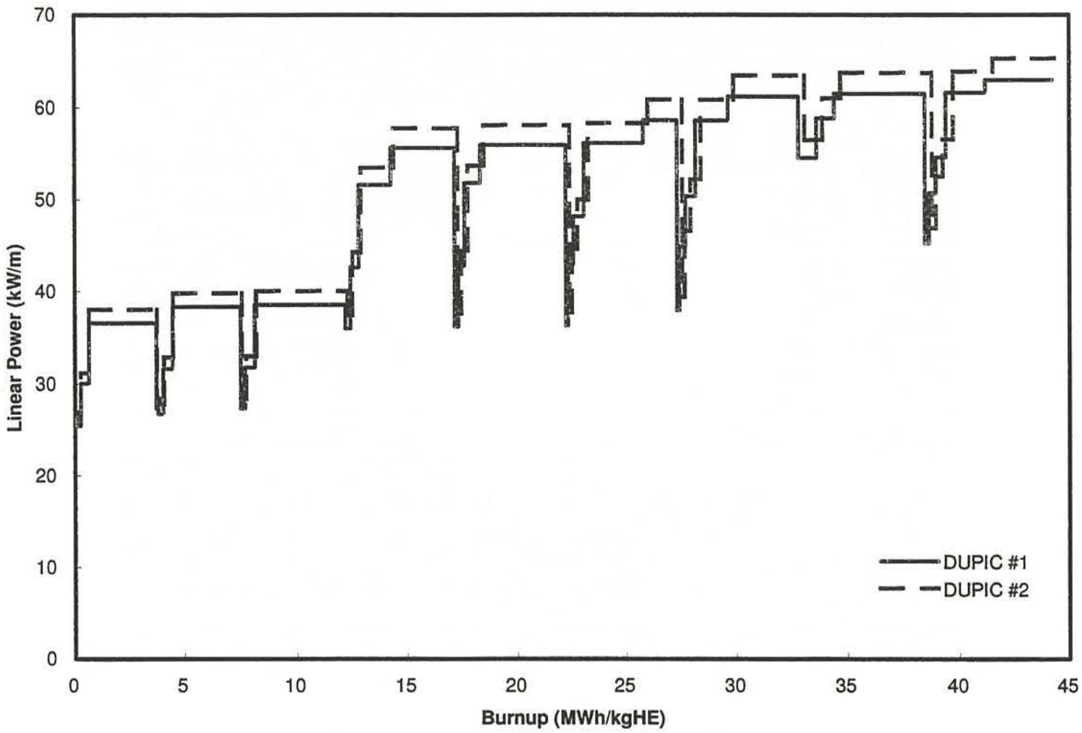


FIG. 3 IRRADIATION HISTORY OF SIMULATED DUPIC FUEL

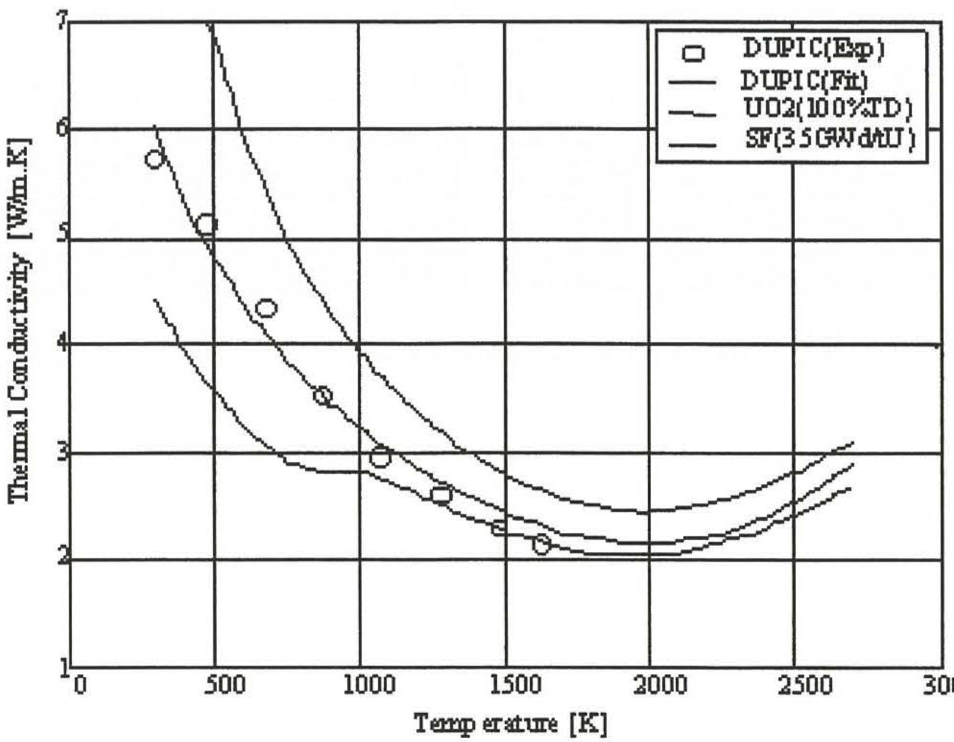


FIG. 4 THERMAL CONDUCTIVITY OF SIMULATED DUPIC FUEL

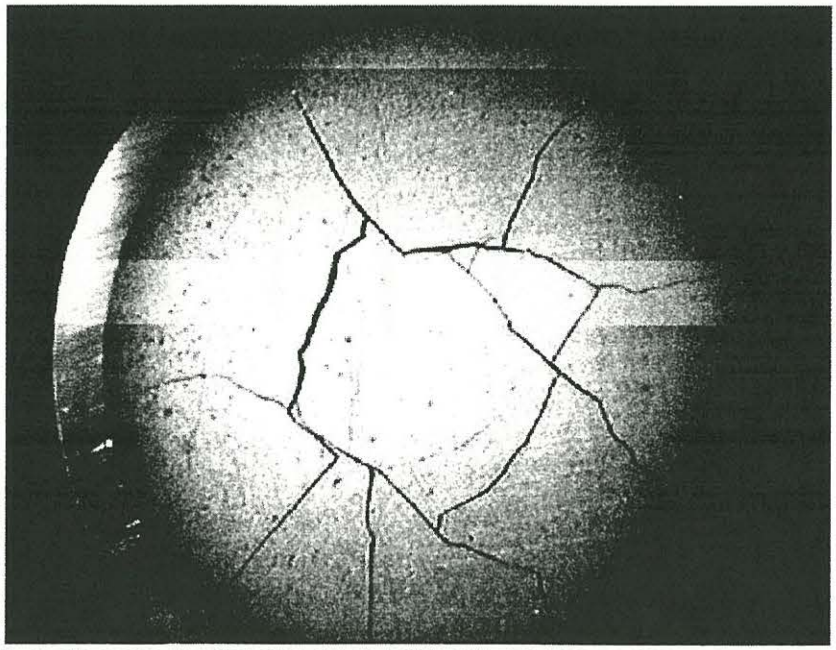


FIG. 5 CROSS-SECTIONAL VIEW OF SIMULATED DUPIC FUEL

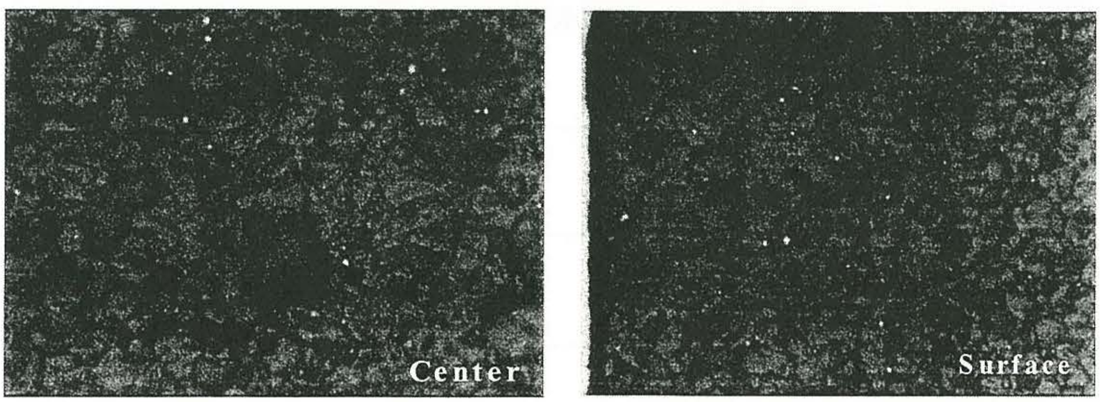
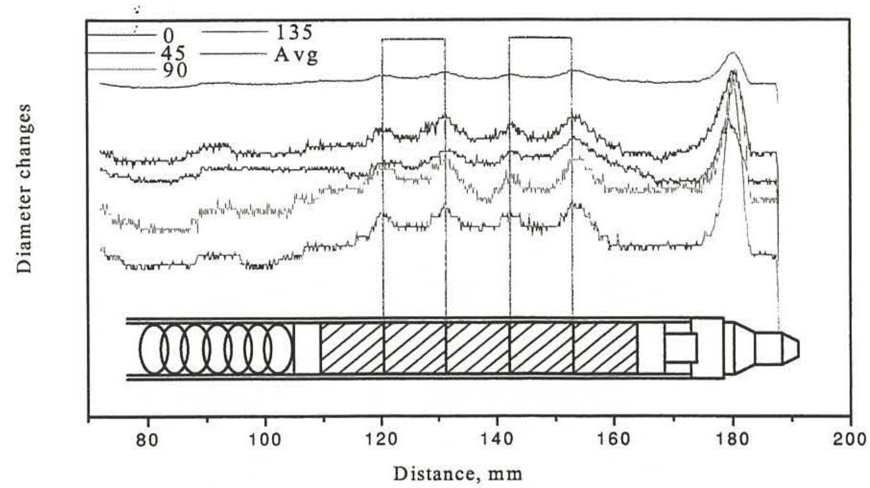


FIG. 6. MICROSTRUCTURES OF SIMULATED DUPIC FUEL AFTER IRRADIATION



Cladding diameter changes along the distance of irradiated DUPIC SIMFUEL-02.

FIG. 7 DIAMETRAL DEFORMATION OF A DUPIC MINI-ELEMENT