FUEL PERFORMANCE EVALUATION OF MINI-PLATE IRRADIATION TEST OF U-7MO DISPERSION FUEL FOR KJRR

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ABSTRACT

National consensus and need for developing successive and sustainable research reactor for a RI production have been matured since the last several years. KAERI had proposed to construct a new research reactor in KiJang province, near Kori PWR complex. As one part of the fuel qualification program for KiJang research reactor (KJRR) in Korea, three irradiation tests with mini-plates of U-7wt%Mo / Al dispersion fuel is scheduled to be performed in HANARO reactor. These mini-plate irradiation tests will provide basic irradiation data including the fuel behavior depending upon fuel burnup.
In this work, a detailed fuel performance analysis of mini-plate such as fuel temperature profile, fuel swelling and corrosion layer growth of the cladding is presented to confirm acceptable fuel behavior in pile. U-7Mo dispersion fuel for KJRR adopt Al-5%Si matrix to minimize the growth of the interaction layer between the U-Mo fuel particle and the matrix. The uranium enrichment of the fuel is about 19.75% and uranium density in the dispersion fuel meat is 8 gU/cc. The evaluation results show that fuel temperature and swelling is predicted to satisfy the fuel design criteria and limits, thus mini-plate irradiation tests in HANARO reactor is expected to be performed with safety and reliability.

1. Introduction

In 2012, KAERI launched a project to construct a new research reactor in KiJang district, near Kori PWR complex. U-7wt%Mo / Al-5wt%Si dispersion fuel with 8.0gU/cm3 is selected to achieve more efficiency and higher performance than U3Si2 fuel. As part of the fuel qualification program for the KiJang research reactor (KJRR), three times of irradiation tests with mini-plates are scheduled to be performed at the High-flux Advanced Neutron Application Reactor (HANARO). These **HA**NARO **M**ini-**P**late Irradiation tests (HAMP-1, 2, and 3) will provide basic irradiation data including the fuel behavior depending on the fuel burnup, i.e. low burnup (about 35-45 %U235 depletion), fuel assembly average discharge burnup (about 65 %U235), and local maximum discharge burnup (about 85 %U235) for KJRR.

For three series of mini-plates irradiation, the fuel performance evaluations of fuel temperature profile, fuel swelling and cladding oxidation are performed to confirm acceptable fuel behavior in pile.

2. Materials properties and irradiation conditions

2.1 Characteristics of U-7Mo mini-plate and irradiation capsule

U-7Mo dispersion fuel for KJRR adopt aluminum-5 wt.% silicon matrix to minimize the growth of the interaction layer (IL) between the U-Mo fuel particle and the matrix [1-3]. The fuel meat consists of an Al-5wt.%Si alloy matrix dispersion of uranium 7 wt.% molybdenum (U-7Mo) metallic alloy with a U235 enrichment of 19.75%. Aluminum alloy 6061 is used as the fuel cladding.

The mini-plates for HAMP-1 and HAMP-2 have a nominal dimension of fuel meat length of 70 mm, fuel meat width of 25 mm, fuel meat thickness of 0.51 mm, fuel plate length of 130 mm, fuel plate width of 35 mm, and fuel plate thickness of 1.27 mm.

The irradiation capsule for HAMP-1 and HAMP-2 consists of 2 clusters and each of top and bottom cluster can contain 4 mini-plates. A cross-section view at bottom cluster of an irradiation capsule is shown in Figure 1 [4].

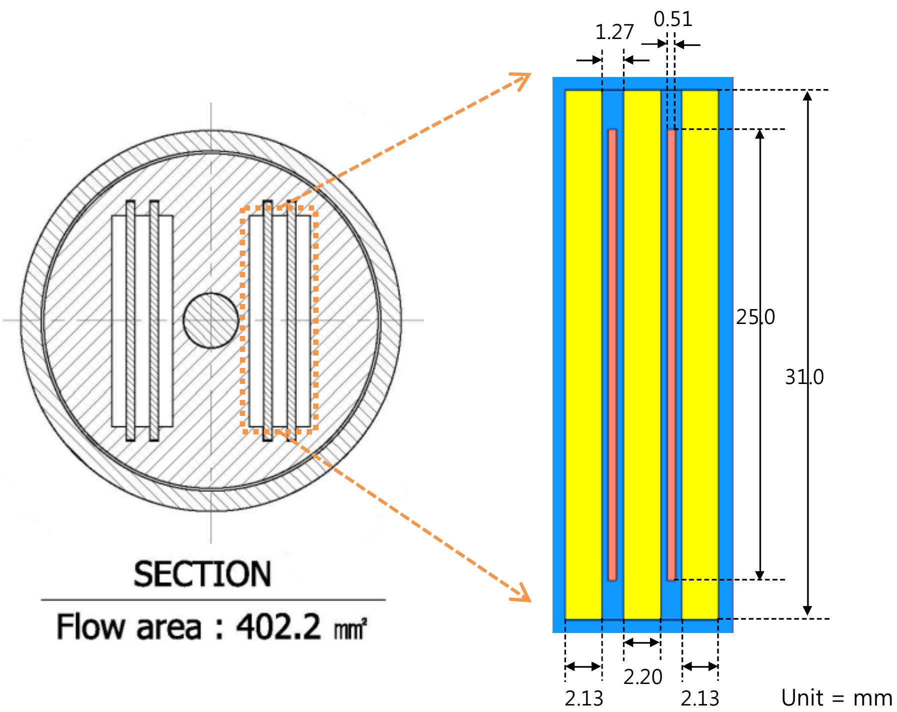


Fig 1. Cross-section of irradiation capsule and mini-plates

The width of a mini-plate of HAMP-3 is the same as HAMP-1 and HAMP-2, which is about half the width of a full-size plate. However, the length of a mini-plate of HAMP-3 is the same as that of a full-size plate for the fuel assembly. Four full length mini-plates will be irradiated in HAMP-3.

Six mini-plates among the total of eight mini-plates to be irradiated in HAMP-1 will consist of fuel meat with a uranium density of 8.0 g-U/cm3. Two other mini-plates will consist of fuel meat with the same uranium density of 6.5 g-U/cm3 as the outer fuel plates of the fuel assembly for equilibrium core of KJRR. However, fuel performance evaluation will be focused on mini-plates of 8.0 g-U/cm3 which show higher heat flux.

2.2 Fuel irradiation test conditions

The irradiation capsules for two sequential HAMP-1 and HAMP-2 test are to be loaded in the irradiation hole (OR-3) with upward coolant flow in the HANARO core. The capsule for HAMP-3 will be loaded in the irradiation hole, which is different from HAMP-2 because the irradiation should run in parallel with HAMP-2. OR-5 is tentatively expected to be used as the irradiation hole for HAMP-3, neutron flux of which is lower than OR-3. Thus, the heat flux of the HAMP-3 in OR-5 would be about 90% compared with that in OR-3. However, fuel performance evaluation for HAMP-3 was performed based on the heat flux of OR-3, conservatively.

The heat flux profiles of mini-plates were obtained from MCNP code calculations [4].

The mini-plate irradiation conditions for HAMP-1, HAMP-2, and HAMP-3 are summarized in Table 1.

Parameter	HAMP-1	HAMP-2	HAMP-3
Fuel meat dimension (mm) (thickness x width x length)	0.51 x 25 x 70	0.51 x 25 x 70	0.51 x 25 x 600
Fuel plate dimension (mm) (thickness x width x length)	1.27 x 35 x 130	1.27 x 35 x 130	1.27 x 35 x 640
Average heat flux at beginning of cycle (BOC), (W/cm^2)	207	207	173
Local maximum heat flux (W/cm^2)	237	237	212
Flow rate of fuel channel (m/s)	11.65	11.65	11.65
Water gap between mini-plates and capsule housing (mm)	2.13~2.20	2.13~2.20	2.13~2.20

Tab 1: mini-plate irradiation conditions

2.3 Thermal conductivity of materials

By multiphase conductivity model [5,6], the thermal conductivity of as-manufactured KJRR fuel meat corresponding to a uranium density of 8.0 g/cm3 is estimated to be 55 W/m-K in the operating temperature. However, as the fuel burn-up increases, the thermal conduction of fuel meat is degraded due to consumption of high conductive Al based matrix as growing the interaction layer by the U-Mo fuel particle and the matrix. About 50% degradation of thermal conductivity of the fuel meat is estimated at burn-up of 85 at% U235 depletion. Consequently, linear decrease of the thermal conductivity for the fuel meat is applied for the fuel performance evaluation.

We conservatively select the value of the thermal conductivities of as-manufactured Al 6061 cladding as 165 W/m-K. To consider the effect of burn-up on the thermal conduction of cladding, 30% degraded thermal conductivity (i.e. 115 W/m-K) of the fuel cladding is used.

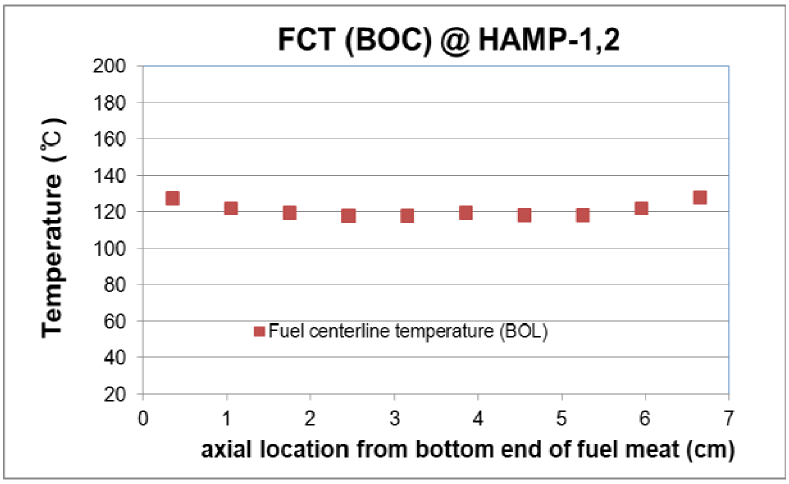
A perfect metallurgical bonding without thermal resistance is assumed between the fuel meat and aluminum cladding. The thermal conductivity value for the oxide layer is 1.85 W/m-K. Conservatively, the maximum fuel temperature is calculated by adding the effect of oxidation layer growth by corrosion of aluminum alloy irradiated in the core.

3. Fuel performance evaluation

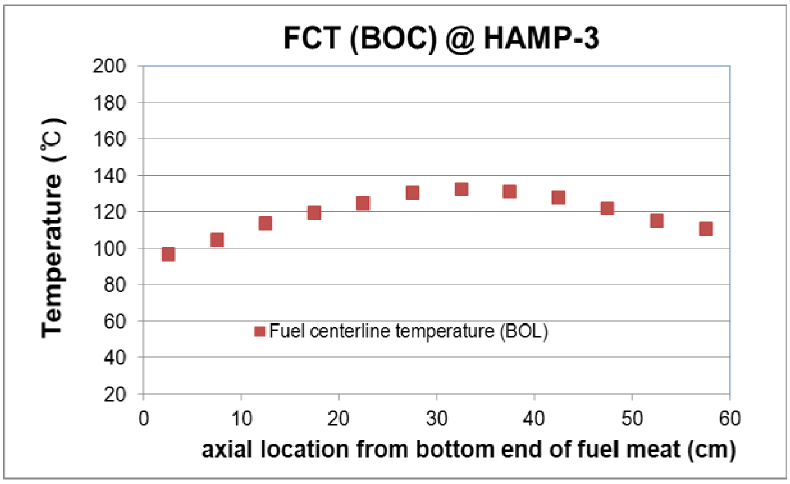
3.1 Peak fuel temperature

The fuel maximum temperature during irradiation is calculated to confirm the suppression of swelling by reaction between U-7Mo particles and an aluminum alloy matrix. With a conservative assumption that the heat flux goes only in the direction of the plate thickness, the temperature gradient is evaluated using a 1-dimensional heat flux equation through the thickness direction. The profiles of centerline temperature of fuel meat are calculated considering the heat flux profile and engineering hot channel factors.

Figure 2 (a) and (b) show the axial fuel centerline temperature distribution at the beginning of cycle (BOC) for HAMP-1 (or HAMP-2) and HAMP-3, respectively. The maximum fuel centerline temperature of mini-plate for HAMP-1 (or HAMP-2) is about 128 °C at the top and bottom edge of fuel meat with higher heat flux due to the end effect and moderating water volume. The maximum fuel centerline temperature of mini-plate for HAMP-3 is about 133 °C at the axial position about 2.5 cm upper from the middle plane, although the peak position of neutron flux and heat flux of plate is about 2 cm lower than the middle plane.



(a)



(b)

Fig 2. Fuel centerline temperature at BOC: (a) HAMP-1,2 (b) HAMP-3

For the case of HAMP-3, additional temperature evaluation taking into account the influence of oxide layer formation on the surface of the mini-plate is shown in the Figure 3 as a function of burnup. The maximum peak fuel temperature is about 153 °C at burnup of around 15% U235 depletion. Upto local maximum burnup of about 85%, the fuel temperature shall be kept below the design limit of 200°C for irradiation test.

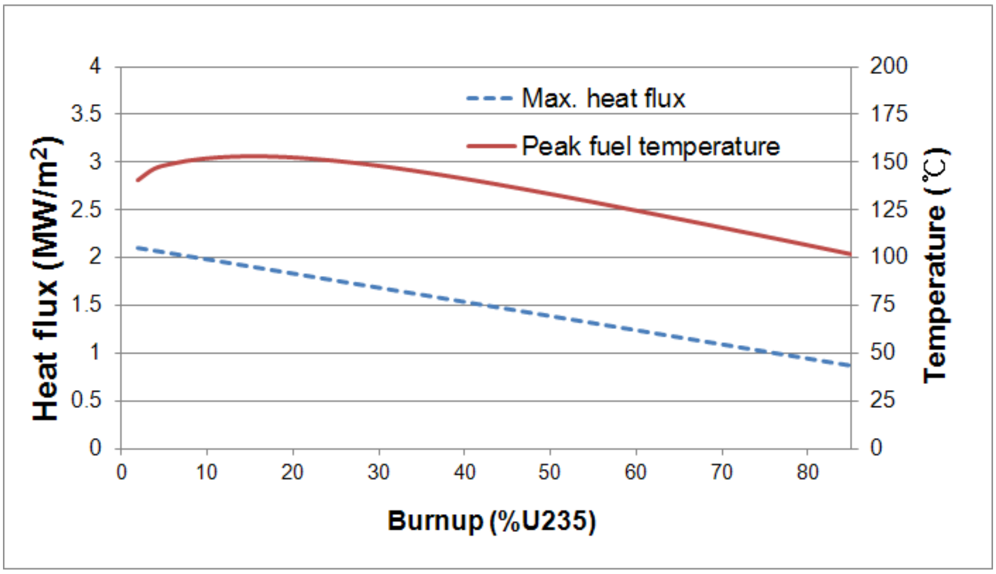


Fig 3. Peak fuel temperature of HAMP-3

3.2 Oxidation

Aluminum alloy cladding experiences oxidation layer growth on the surface during the reactor operation. A prediction of the aluminum oxide thickness of the fuel cladding and maximum temperature difference across the oxide film is needed for a reliable evaluation based on the design criteria and limits, which prohibit the spallation of the oxide film.

The oxide growth model developed by Kim and Hofman, et al. [7] which uses a variable rate-law power in a function of irradiation time, temperature, surface heat flux, water pH, and coolant flow rate, was used for estimating the oxide film thickness.

Figure 4 shows oxide growth of cladding as a function of burnup for HAMP-3. Because the predicted oxide thickness is sensitive to water pH, it is assumed that water pH will be evenly distributed in the range of 5.5 ~ 6.2 during the whole irradiation time.
The oxide thickness of the discharged fuel with maximum burnup of 85% and maximum temperature difference across the oxide film (ΔToxide) are predicted to be about 26 μm and 25.5 ℃, respectively. Therefore, ΔToxide remains lower than 114 ℃, and the oxide spallation and following subsurface corrosion will be precluded.

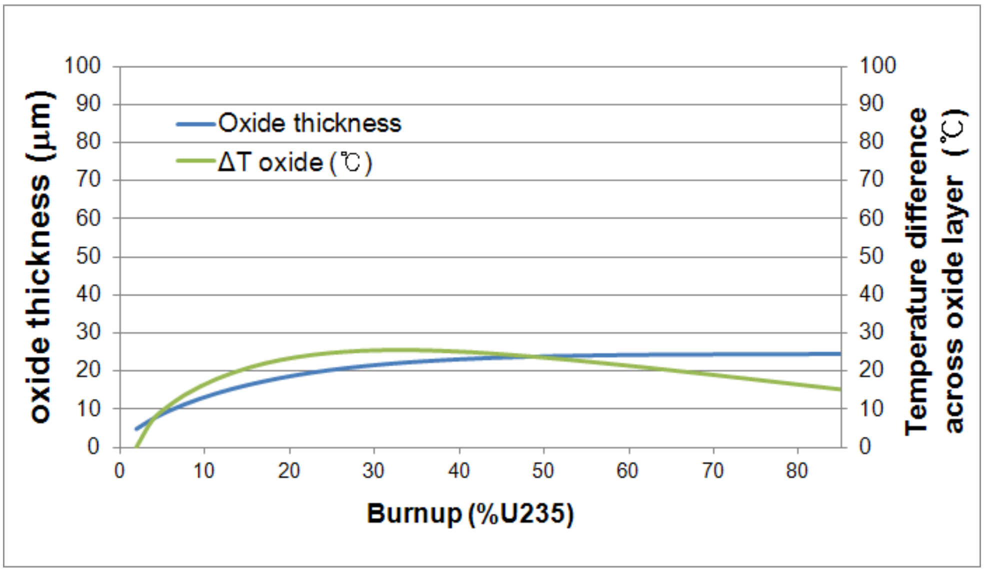


Fig 4. Oxide thickness of cladding as a function of burnup for HAMP-3

3.3 Swelling

The irradiation swelling also needs to be checked for the maintenance of proper channel gap.

The swelling correlation of U-Mo fuel is composed of two parts: solid fission products swelling and gas bubble swelling [8,9].

$(\Delta V/V_0)$ total (%) = $(\Delta V/V_0)_g + (\Delta V/V_0)_s$

The solid fission products swelling is a linear function of burnup (or fission density).
$(\Delta V/V_0)_s$ (%) = $4.0\times10^{-21} f_d$
where f_d is the fission density in fissions/cm^3.

The gas bubble swelling has different rates depending on the fission density.
$(\Delta V/V_0)_g$ (%) = $1.0\times10^{-21} f_d$, for $f_d \leq 3\times10^{21}$ fissions/cm^3
$(\Delta V/V_0)_g$ (%) = $3.0+2.3\times10^{-21} (f_d - 3\times10^{21}) + 0.33\times10^{-42} (f_d - 3\times10^{21})^2$, for $f_d > 3\times10^{21}$ fissions/cm^3

As shown in the Figure 5, the calculated values of fuel meat swelling for HAMP-1, HAMP-2 and HAMP-3 are about 9%, 15% and 20%, respectively. The swelling of a mini-plate occurs almost totally in the thickness direction. Thus, the maximum thickness increase of the fuel meat of HAMP-3 is transferred to the increase of mini-plate thickness by about 12%.

The narrowing of the channel gaps between mini-plates by irradiation swelling for HAMP-1, HAMP-2 and HAMP-3 will be about 2.1%, 3.5% and 4.6% compared to the initial water gap of 2.2 mm.

Conclusively, the substantial swelling caused by interactions between fuel particles and aluminum matrix is prevented by limiting the maximum temperature below 200°C for all normal irradiation conditions. At the maximum fuel temperature of 153 °C for three series of HAMP, the narrowing of coolant channel gaps due to thermal expansion of FPs is less than 0.2% of the initial channel gap. On the other hand, the total narrowing of channel gaps by irradiation swelling and oxidation of dual surfaces of mini-plate is less than 6.9% of the initial channel gap.

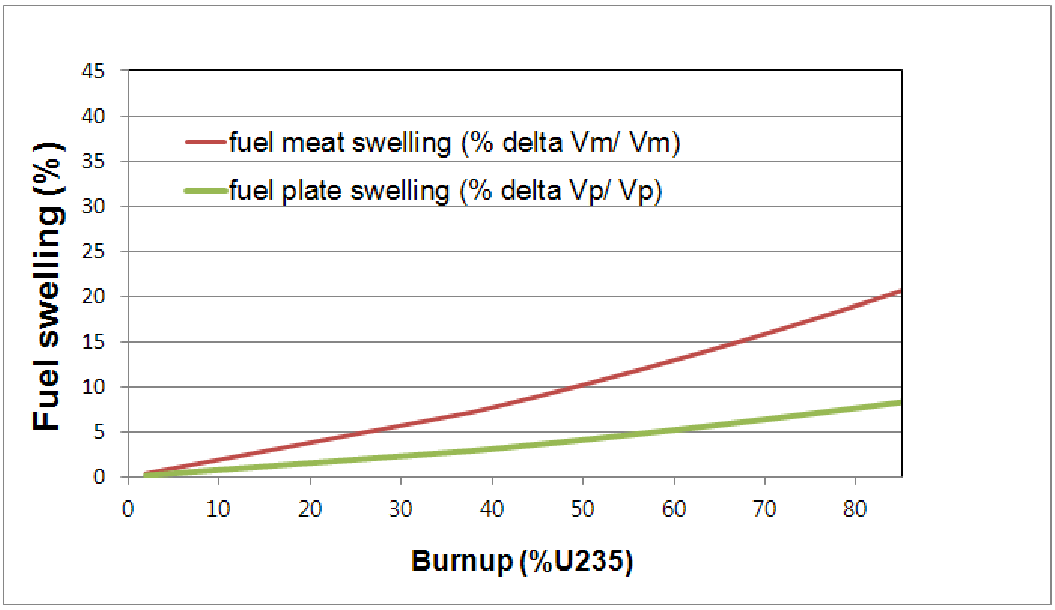


Fig 5. Fuel swelling as a function of burnup

4. Summary

The maximum fuel temperature obtained in the irradiation condition of three separate series of HAMP is around 153°C, which is far below the preset limit of 200°C on the rapid swelling by reaction between fuel particles and the matrix. Thus, substantial swelling will not occur so that the possibility of excessive flow gap narrowing will be precluded.
The calculated temperature difference between the inside and outside of the oxide is a maximum of 25.5 °C and far below the spallation criterion. Thus, the hazardous oxidation effect on the fuel cladding will be precluded.
The fuel plate integrity and fabrication robustness are to be verified through the irradiation test including post-irradiation examination (PIE).

5. References

[1] G. L. Hofman et al., RERTR 2006, Capetown, South African, Oct. 29-Nov. 2, 2006.
[2] G. L. Hofman et al., RRFM 2007, Lyon, France, Mar. 11-15, 2007.
[3] A. Leenaers al., RRFM 2008, Hamburg, Germany, Mar. 2-5, 2008.
[4] Chul-Gyo Seo, Haksung Kim, Daeseong Jo, "Heat flux evaluation of U-Mo mini-plate Irradiation in the HANARO core", Personal communication, Korea Atomic Energy Research Institute.
[5] Hayes, S.L., et al., “Modeling of high-density U-Mo dispersion fuel plate performance”, Proc. Intl Mtg on RERTR, San Carlos de Bariloche, Argentina, 2002, Rep. ANL/RERTR/TM-, Argonne Natl Lab., IL (2002).
[6] Z. Hashin and S. Shtrikman, “A Variational Approach to the Theory of the Effective Magnetic Permeability of Multiphase Materials,” Journal of Applied Physics, Vol. 33 (1962) pp. 3125-3131.
[7] Yeon Soo Kim, G.L. Hofman, A.B. Robinson, J.L. Snelgrove, N. Hanan, “Oxidation of aluminum alloy cladding for research and test reactor fuel”, Journal of Nuclear Materials, Vol. 378, p. 220, 2008.
[8] J. Rest, Y.S. Kim, G.L. Hofman, M.K. Meyer, and S.L. Hayes, “U-Mo Fuels Handbook”, ANL-09/31, 2009
[9] Y.S. Kim, G.L. Hofman, J. Nucl. Mater. vol.419, p.291, 2011.