THE PRELIMINARY PERFORMANCE ANALYSIS OF THE TRANSMUTATION FUEL FOR HYPER

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Abstract

A U-TRU-(40-60)Zr metallic fuel is being considered as the transmutation fuel for HYPER in Korea. U-TRU-15Zr is also considered as the reference fuel for a critical system. The MACSIS-H for an metallic fuel is being developed as a steady-state performance computer code. The fuel temperature calculation schemes were developed and implemented into the MACSIS-H code to analyze the fuel rod performance. A constituent migration subroutine has been made and installed into the MACSIS-H code to simulate the constituent redistribution. The He production rates calculated by other code were inserted into the swelling/FGR routine of the code. The burnup limits and CDF (cumulative damage fraction) were analyzed by the MACSIS-H code. There are lots of uncertainties in the modelling, so some experimental tests are needed for clarifying the uncertainties of the fuel modeling.

Introduction

Metallic fuel is being considered as the transmutation fuel for the HYPER (Hybrid Power Extraction Reactor) in Korea[1]. U-TRU-Zr is being considered for the alloy fuel slug, and the cladding material is HT9. The performance analysis of the fuel rod design is essential to assure an adequate fuel performance and integrity under irradiation conditions. In this paper, parametric study has been performed by the MACSIS-H (Metal fuel performance Analysis Code for Simulating the In-reactor behaviour under Steady-state conditions-HYPER)[1]. This code will be used for simulating the operational limits of the metal fuel under steady state conditions.

Present studies represent the parametric results and the capability for efficiently predicting the performance parameters as a function of the burnup. There were a few fuel characteristics relating to the TRU material, so the material data of U-Pu-Zr was used for those of the U-TRU-Zr. The radial fuel constituent migration subroutine has been made and installed into the MACSIS-H code to simulate the constituent redistribution. The He production rates calculated by other code are inserted into the swelling/FGR routine of the code. The thermal creep strain limits were analyzed according to the plenum-to-fuel ratio by MACSIS-H. A statistical failure analysis was performed to analyze the rupture behaviour of the metallic fuel pins. The failure probabilities were estimated in terms of CDF, based upon the known Weibull statistical failure model.

Code description

MACSIS-H is a metallic fuel performance computer code which calculates thermal performance characteristics and dimensional changes of the fuel rods in a fast neutron environment. MACSIS-H code calculates the temperature distribution, mechanical deformations, fission gas release, and the constituent migrations of the fuel elements during irradiation. It assumes that the fuel slug is an infinitely long rod concentric with an infinitely long cladding.

The key phenomena which are significant in controlling the metallic fuel pin behavior and reliability are known to be a large fission gas release and a fuel swelling in the early irradiation stage, the fuel constituent migration and the bonding material behavior and the fuel/cladding eutectic reaction. [2]

Main structures of the code consist of the temperature profile calculation routine, the swelling/FGR calculation routine, the chemical/metallurgical behavior calculation routine and the deformation calculation routine.

Design parameter

The alloy fuel consists of a U-TRU-Zr metallic alloy slug and a liquid metal thermal bonding in HT9 steel, much like the Experimental Breeder Reactor II (EBR-ll) fuel or the fuel developed for the Integral Fast Reactor (IFR) concept. A fission gas plenum is located above the fuel slug. The key design parameters for the alloy fuel are shown in Table 1.

Table 1. **Key Design Parameter**

	Sub-critical (HYPER)	Critical
Fuel Slug Contents (wt%)	8U-34.3Pu-4.2Am-1.6Cm-1.7Np-0.2RE-50Zr	65.5U-18Pu-0.5Am-0.4Cm-0.5Np-0.5RE-14.6Zr
^{241}Am Content (wt%)	2.66	0.1
Fuel Slug Diameter (mm)	5.63	6.63
Smeared Density (%)	75	75
Pin Outer Diameter (mm)	7.7	8.8
Cladding Thickness (mm)	0.6	0.57
Fuel Slug Length (mm)	1,500	1,240
Peak Linear Power (kW/m)	28.5	28.9
Coolant Outlet Temperature (°C)	490	540
Cladding Material	HT9	HT9

Fuel temperature prediction

Thermal conductivity and power-to-melt

The precise prediction of the fuel temperature distribution is one of the most important factors in a fuel performance code, because the fuel temperature affects almost all of the fuel element behaviors. It is known that the key factors which may affect the temperature distribution of a metallic fuel are porosity formation, bonding material infiltration, heat generation rate and the fuel constituent migration.

The thermal conductivity of the unirradiated U-TRU-Zr alloy, k_0, can be expressed as the function of the temperature and alloy composition. [3]

$$k_0 = 17.5\left(\frac{1-2.23\mathrm{Wz}}{1+1.61\mathrm{Wz}} - 2.62\mathrm{Wp}\right) + 1.54\times10^{-2}\left(\frac{1+0.061\mathrm{Wz}}{1+1.61\mathrm{Wz}} + 0.9\mathrm{Wp}\right)\mathrm{T} + 9.38\times10^{-6}(1-2.7\mathrm{Wp})\mathrm{T}^2 \qquad (1)$$

where, T is the temperature in Kelvin and Wz is the weight fraction of zirconium.

Thermal conductivity for more than 20wt% of Zr or more than 30wt% TRU is given by [4].

$$k_{A-B-C} = k_{A-B}v_{A-B}^2 + k_{A-C}v_{A-C}^2 + 4v_{A-B}v_{A-C}\frac{k_{A-B}k_{A-C}}{k_{A-B}+k_{A-C}} \qquad (2)$$

The porosities and bonding material infiltration effect should be considered in calculating the thermal conductivity of an irradiated metallic fuel. These effects can be determined by porosity correction factor, P_f, which was derived by Bauer and Holland. [5]

The predicted temperatures of the U-Zr metal fuel by MACSIS-H versus LIFE-M, etc.[3,6~10] are plotted in Figure 1 which is an evaluation of the various temperatures from the cladding mid-wall to fuel centerline. It is apparent that MACSIS-H has a reasonably good capability in predicting the fuel pin rod temperatures.

Figure 2 shows several cases of temperature variation with respect to the linear heat generation of the typical HYPER and a critical system fuel pin. In the design of a metallic fuel pin, fuel temperature limits on fuel melting should be considered from a temperature point of view. Based on the aggregate of the TREAT tests, ANL concluded that a centerline fuel melting, even an extensive melting exceeding 80 % of a given radial cross-section, is not a problem and does not result in pin failures[11]. However, the prevention of centerline fuel melting is regarded as a design limit of the metallic fuel pins for conservatism. It is expected that the solidus temperature of the U-42TRU-50Zr and U-20TRU-14.6Zr metallic fuel are 1090 and 1295^{o}C, respectively. As an irradiation proceeds, the thermal conductivity of the metallic fuel is degraded to approximately 50 % of the initial value at around 1 at% burnup. For a further irradiation, the conductivity is restored by the bonding material infiltration into the fuel slug. In the case of the 50 % degraded thermal conductivity, which is the worst condition in terms of the fuel temperature, the calculated power-to-melt for the U-42TRU-50Zr and U-20TRU-14.6Zr metallic fuel were about 420 and 500 W/cm, respectively.

Figure 1. **Comparison of the U-Zr metal fuel temperatures predicted by MACSIS and LIFE-M**

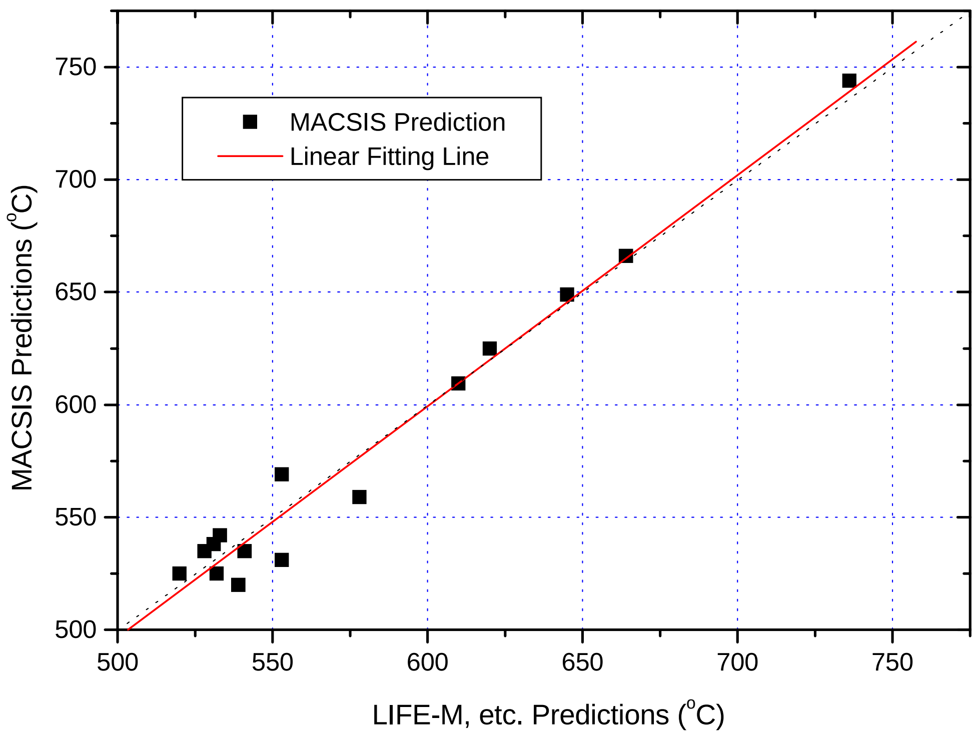
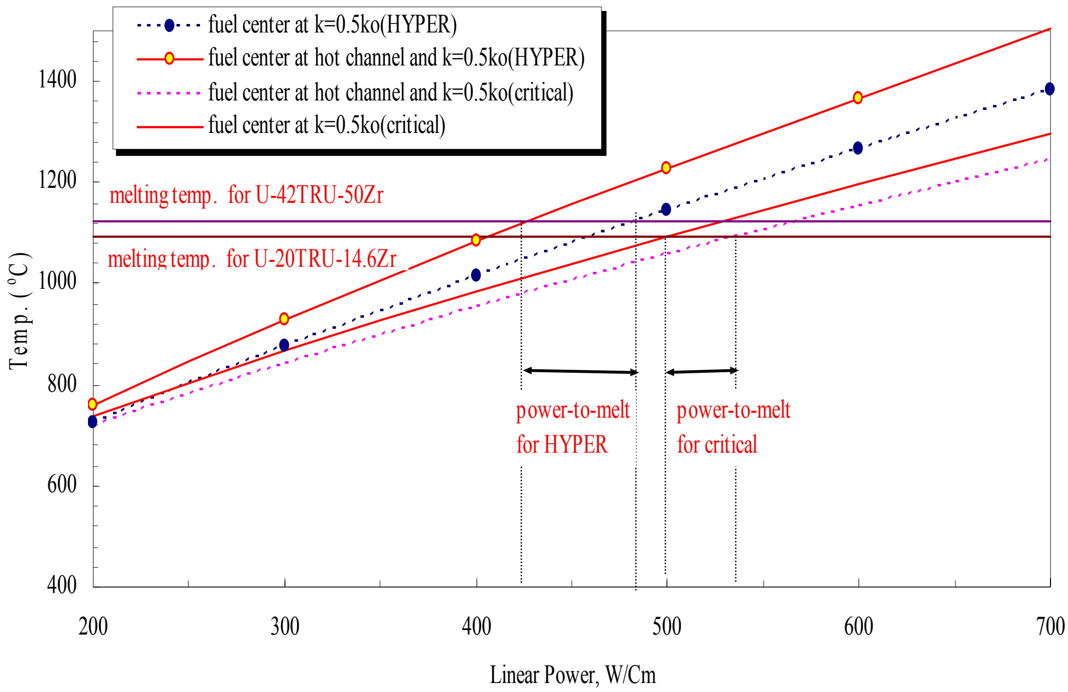


Figure 2. **The operating limits of the linear power rate for a metallic fuel pin**



Fuel constituent migration

The radial fuel constituent migration related to the formation of three distinct phasal zones is a general phenomenon in irradiated U-Pu-Zr and U-Zr alloys.[12] This phenomenon may affect the in-reactor performance of the metallic fuel rods, such as the melting temperature, thermal conductivity, power generation rate, phase boundaries and the eutectic temperature of the fuel slug. Thus, a constituent migration modeling is essential to develop a metallic fuel performance code. The constituent migration model adopted in MACSIS-H was based on the Ishida's model [13] and Hofman's theory. [14]

Ishida's model is a ternary diffusion model that considers a quasi-binary U-Zr phase diagram with constant plutonium contents for the U-Pu-Zr system, but the Zr weight is limited to about 20wt%. So the quasi-binary U-Zr phase diagram has been reconstructed by several polynomial equations only for the fuel slug of the critical system. The diffusion coefficients used in this model are based on the Hofman's model.[14] The fuel constituent migration model for more than 40wt% Zr will be developed in the future.

Figure 3 shows the calculated radial profile of Zr and the measured Zr concentration for the U-19Pu-10Zr. The dashed line shows the initial Zr composition. As shown in Figure 3, a significant amount of Zr is depleted in the middle zone.

Figure 4 shows the simulated Zr redistribution for U-20TRU-14.6Zr fuel. The main reason for Zr addition to the metallic fuel is to increase the melting point of the fuel and to enhance the chemical compatibility between fuel and cladding. In this regard, the fuel constituent migration may affect the integrity of the fuel pin. At around 650 °C of fuel temperature, the model predicts the Zr fraction in fuel centre reaches 0.5, and there will be no centreline Zr depletion expected in all range of temperature. The model also predicts that the sharp Zr depletion occurs when the fuel surface temperature is approached the upper limit of (ζ+γ) phase boundary.

Figure 3. **Calculated and measured radial profile of Zr for U-19Pu-10Zr**

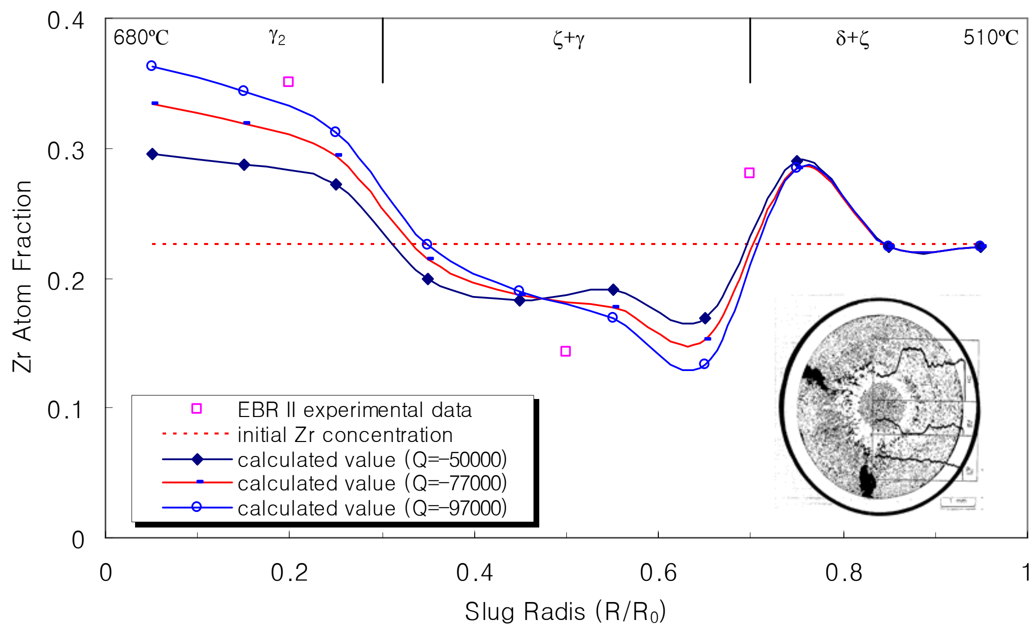
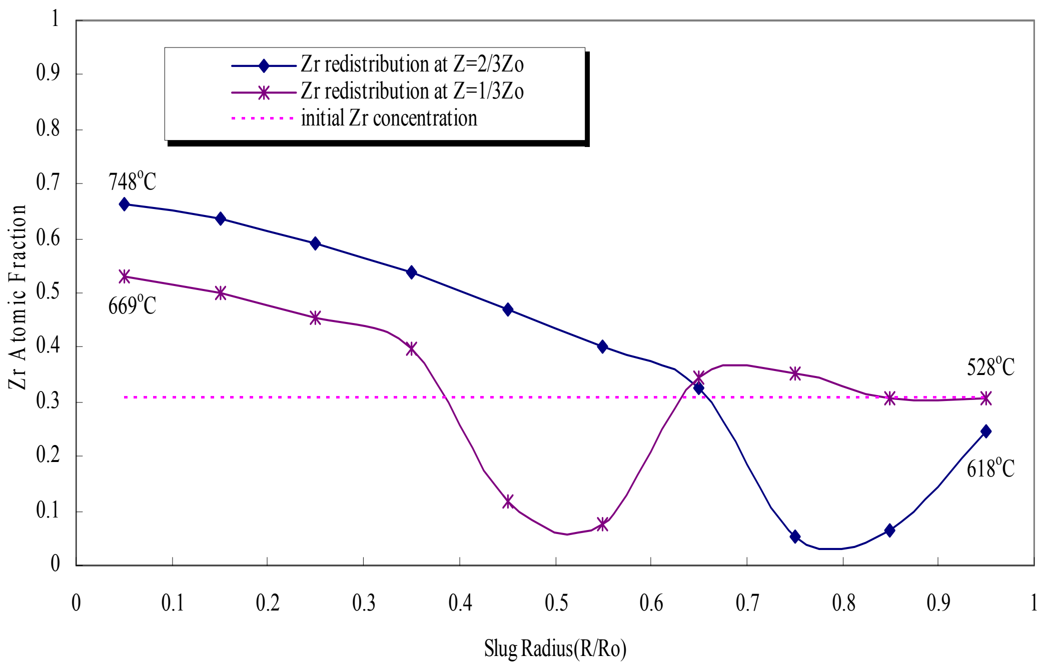


Figure 4. **Calculated radial profile of Zr for U-20TRU-14.6Zr**



Margin to slug centreline melting

Figures 5 and 6 show the radial distribution of the temperature of the slug centre. The centreline peak temperature is considerably lower than the solidus temperature. It can be concluded, therefore, that the metallic fuel has a sufficient margin to the slug melting temperature.

Figure 5 also shows the temperature distribution for the case of the fuel constituent migration. The Zr fraction in the fuel centre is increased by the fuel constituent migration. This indicates that the fuel centreline melting will be retarded as the redistribution occurs during off-normal conditions. But the Zr depletion at the fuel surface may affect deleterious effect on fuel pin integrity. This phenomenon has not been confirmed yet experimentally, but the fuel designers might be careful on this effect for setting up their operating temperature limit.

Figure 5. **Calculated radial profile of Zr for U-20TRU-14.6Zr**

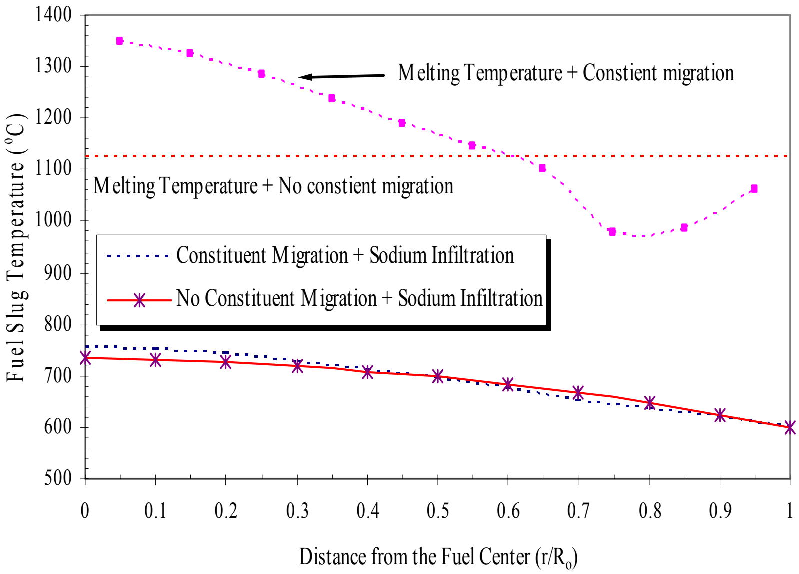
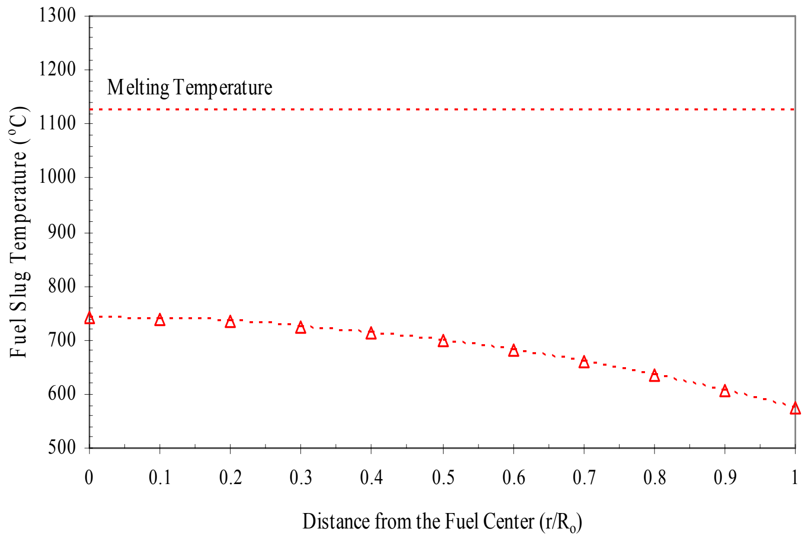


Figure 6. **Calculated radial profile of Zr for U-20TRU-50Zr**



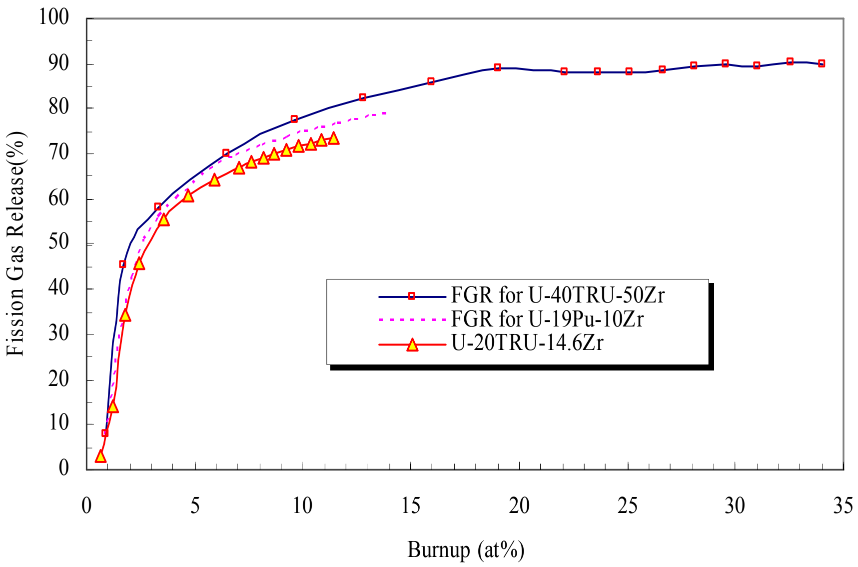
Fission gas release and the He release rate insertion

Fission gas retention and release

The production rate of the fission gas products is proportional to the power density, and a predictive capability for the behaviour of the fission gas in nuclear fuel during normal and off-normal conditions is essential to any rational estimate of the fuel element integrity. The first step of the fission gas release is the movement of fission gas created in the fuel matrix to the fuel grain boundary. The second step of the fission gas release is the movement of the fission gas on the grain boundary through the fission gas tunnels to the fuel rod gas plenum.

Figure 7 shows the percentage gas releases according to the burnup variation, predicted by the semi-theoretical models in the MACSIS-H code. According to the experimental results, it appears that the fission gas release largely increases at around 1 to 2 at% burnup and the maximum fission gas release at a burnup of 13at% was 74 % for the U-19Pu-10Zr. The predictions by MACSIS-H with the semi-theoretical models agree comparatively well with the experimental results from ANL, according to the burnup increase.

Figure 7. **Fission Gas Release as function of burnup**



He release rate insertion

The estimated helium production rates from ^{241}Am was 50 ml He per gram of transmuted ^{241}Am.[16] As shown in Table 1, the initial loaded ^{241}Am contents by the heavy metal contents of the HYPER and the critical system were 2.66 and 0.1 wt%, respectively. So the initial loaded ^{241}Am weight calculated by the fuel dimensions were 9.1 and 0.5g. It is assumed that the time required to achieve 50% transmutation of ^{241}Am is 2.5 years.[16] The He generation rates of the U-42TRU-50Zr and U-20TRU-14.6Zr metallic fuel were assumed to be about 40 and 9.36ml/165day.

In the MACSIS-H, the He generation rates were inserted into the FGR (Fission Gas Release) analysis module, and then the volume of the fission gas generated was calculated including the He generation rates.

Burnup limits analysis

Figure 8 shows the cladding thermal creep strain comparison with the He effects as a function of the plenum-to-fuel ratio for the fuel of HYPER. The cladding strain of a low-smeared density pin can be accounted for by the plenum pressure stress alone.[2] It was estimated that the swelling and the irradiation creep of the HT9 were very small. The effects on the thermal creep strain with different plenum sizes were analyzed by the MACSIS-H code.

A shorter plenum length results in a larger thermal creep strain. The values of the thermal creep strain without the He effects were about 0.14 and 0.09% at 25at% for the 1.5 and 1.75 plenum-to-fuel ratios, respectively. But the values of total strain with He effects were about 0.4 and 0.19% at 25at% for the 1.5 and 1.75 plenum-to-fuel ratios, respectively. These results indicate that the He effects will be a very important factor for the higher burnup, even if the He weight were very small.

According to preliminary burnup limits, the thermal creep strain limit is 1%[17]. So the burnup limits of the alloy fuel for HYPER were 29 and 33at% for the 1.5 and 1.75 plenum-to-fuel ratio, respectively.

Even though a lot of models such as the eutectic melting and the FCMI by the solid fission products have not been established, it is expected that 1.5 and 1.75 times of the plenum-to-fuel ratios are conservative for satisfying the discharge burnup goal of about 25% by the thermal creep strain limit.

Figure 9 shows the cladding thermal creep strain comparison with the He effects as a function of the plenum-to-fuel ratio for the fuel of the critical system. The values of the thermal creep strain without the He effects were about 2.27 and 0.64% at 9.8at% for the 1.5 and 1.75 plenum-to-fuel ratios, respectively. But the values of the total strain with the He effects were about 4.8 and 1.2% at 9.8at% for the 1.5 and 1.75 plenum-to-fuel ratios, respectively.

These results indicate that the HT9 cladding is not conservative for satisfying the discharge burnup goal, because of the high coolant outlet temperature. It is estimated that the replacement of the cladding material with higher thermal creep resistance may be needed for the critical system.

Figure 8. **Thermal creep strain according to plenum-to-fuel ratio for the HYPER fuel**

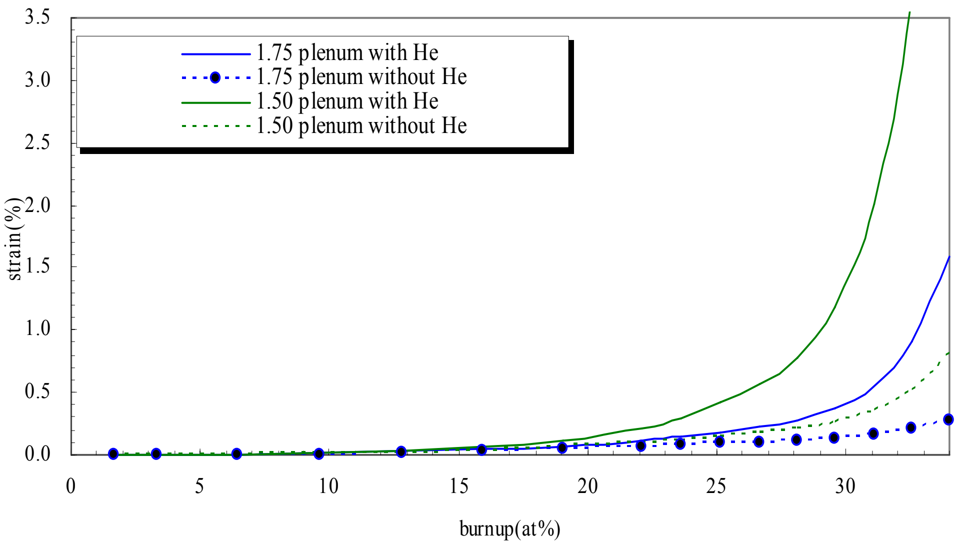
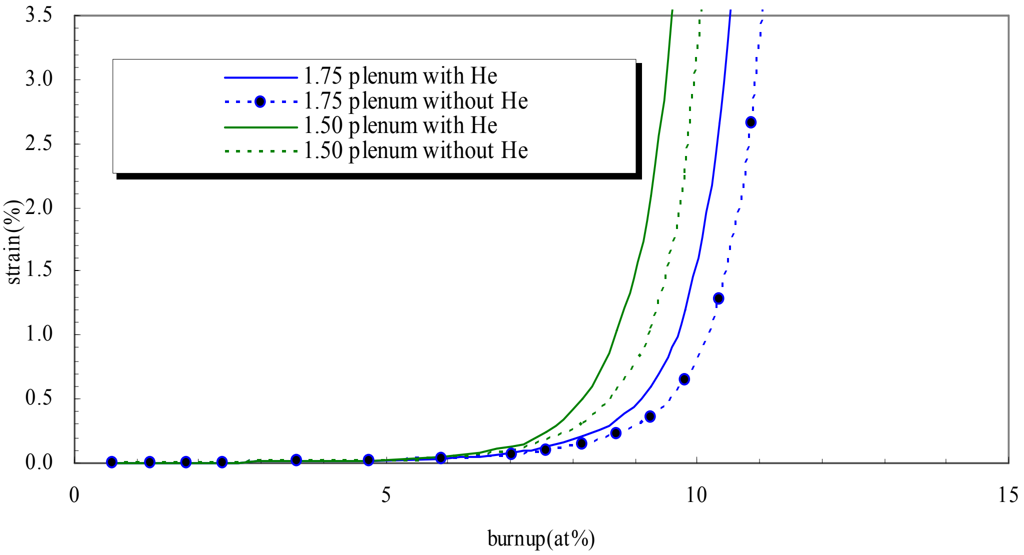


Figure 9. **Thermal creep strain according to plenum-to-fuel ratio for the critical system**



Cumulative damage fraction analysis

To prevent the fuel pin failure, generally two kinds of specific design limits have been used in the fast reactor fuel design. One of them is the strain limit approach. However, this approach has a deficiency in that the rupture strain is strongly dependent on the temperature and strain rate. The other approach is the cumulative damage fraction (CDF) method, which utilizes the linear life fraction rule assuming that the damage accumulates linearly. The CDF is determined from the time-to-rupture correlation as a function of the temperature and stress.

For the evaluation of the component reliability, the Weibull statistical model [18] has been widely used in the area of nuclear components as well as other industries. The Weibull analysis is appropriate for describing the fuel pin breach because it is derived from the situation where the most severe flaw dominates the failure.

For the steady-state conditions, the probabilistic CDF were estimated as follows;

- The cumulative damage fractions of the X447 fuel pins were evaluated from the time-to-rupture correlation by the MACSIS-H code calculations;
- The failure distribution functions of the metallic pins were derived by the Weibull analysis with the application of CDF and burnup;
- Fuel pin performances were estimated under the HYPER and the critical system conditions.

For the transient conditions, the probabilistic CDF were estimated as follows;

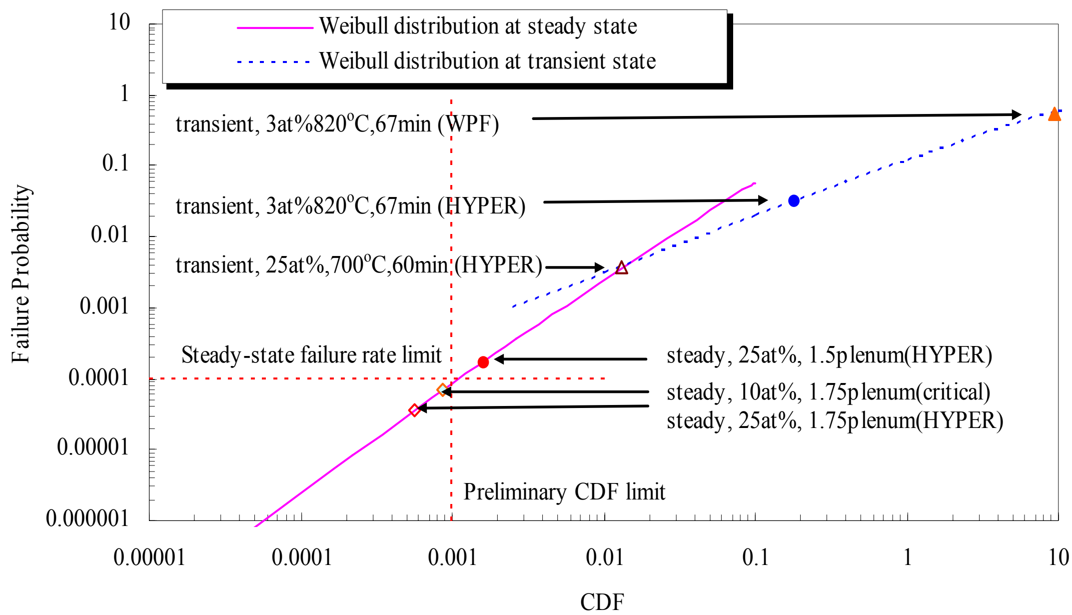
- The cladding performances during transient conditions were evaluated using the result of the WPF (Whole Pin Furnace) tests;
- Cumulative damages about the WPF test fuel pin were estimated by the transient test data and the failure correlation;
- Distribution of the CDF of the WPF data was determined by the Weibull probabilistic analysis;
- Fuel pin performance was estimated under the same conditions as the WPF tests.

Generally the limit on the fuel pin failure rate of the fast reactor core is less than 0.01%, so Figure 10 also shows the CDF limit of 0.001 was reasonable.

The calculated CDF at 25at% of the peak burnup for the fuel pin of HYPER during the steady-state were 1.6×10^{-3} and 5.7×10^{-4} for the 1.5 and 1.75 plenum-to-fuel ratios, respectively. The fuel pin failure rates were 0.017 and 0.003%, respectively. It was estimated that 1.75 times of the plenum-to-fuel ratio is conservative for satisfying the discharge burnup goal of about 25% by the CDF limit. .

Failure probability of the HYPER fuel pin during transient condition were lower than that of the WPF pin, because of a higher plenum-fuel volume ratio and lower cladding inner radius vs. thickness ratio. If there were more transient data points on the fuel pin, a more accurate probabilistic CDF analysis would be performed.

Figure 10. **Cumulative damage fraction of the alloy fuel**



Conclusions

The metallic fuel is being considered as the transmutation fuel for the HYPER in Korea. The MACSIS-H has been developed for the design of the metallic fuel in HYPER, and a parametric study was performed using the MACSIS-H code.

It was estimated that the metallic fuel has a sufficient margin to the slug melting temperature. The constituent migration analysis was performed by inserting the quasi-binary U-Zr model into the MACSIS-H code. The results show that a significant amount of Zr was depleted in the middle zone. So it is expected that the centreline melting temperature of the U-20TRU-14.6Zr fuel will be increased, but the eutectic temperature at the fuel surface will be decreased.

In order to evaluate the He effects by ^{241}Am transmutation, the He generation rates were inserted into the code. It was estimated that the He effect will be a very important factor.

It was expected that the fuel pin of HYPER was conservative for satisfying the discharge burnup goal of about 25% by the thermal creep strain limit of 1%. But the fuel pin of the critical system was not conservative for satisfying the discharge burnup goal, because of the high coolant outlet temperature. It was estimated that the replacement of the cladding material with higher thermal creep resistance may be needed.

It was estimated that 1.75 times of the plenum-to-fuel ratio is conservative for satisfying the discharge burnup goal of about 25% by CDF the limit. Failure probability of the HYPER fuel pin during transient condition were lower than that of the WPF pin, because of a higher plenum-fuel volume ratio and lower cladding inner radius vs. thickness ratio.

There are lots of uncertainties in the modelling such as the material data, the eutectic melting, and the FCMI etc., so some experimental tests are needed for clarifying the uncertainties of the fuel modeling.

Acknowledgments

This work has been carried out under the national nuclear long-term R&D program which is supported by the Korean MOST(Ministry Of Science and Technology).

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