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Irradiation Behavior Analyses of Metallic Fuel Pins for Sodium-cooled Fast Reactors

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A metallic fuel concept, which consists of U-Pu(TRU)-Zr metallic fuel slug and oxide dispersion strengthened ferritic steel (ODS) cladding, is considered to be an alternative for sodium-cooled fast reactor (SFR) cycle system. The capability of the U-Pu(TRU)-Zr metallic fuel with ODS cladding under a high burnup condition was evaluated by a simplified calculation grogram developed in JAEA.

The fuel temperature profiles, gap width profiles, and radial temperature distribution at EOL were evaluated. Those results show that the fuel pin has enough safety margin to fuel melting under the irradiation. Also, the profiles of plenum gas pressure and the cladding deformation after irradiation were evaluated. Those results show that the fuel pin has enough plenum volume to avoid considerable cladding deformations by plenum gas pressure.

In case of 0.4 % Am bearing fuel, calculation result shows that fuel centerline temperature becomes high, but increase from U-Pu-Zr fuel is insignificant.

It is deemed from the obtained results that the metallic U-Pu-Zr fuel pin having the specifications and irradiation conditions used in this investigation would be irradiated moderately up to approximately 140 GWd/t with well integrity.

Keywords　Fast Reactor, Metallic Fuel, Fuel Performance, ODS, Calculation Code

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ナトリウム冷却高速炉のための金属燃料ピンの照射挙動解析

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ナトリウム高速冷却炉(SFR)の開発では、燃料の副概念として、U-Pu(TRU)-Z を成分とする金属燃料スラグと酸化物分散強化型フェライト鋼（ODS）による燃料ピンが検討されている。この燃料ピンの高燃焼度への適用性について、JAEA で開発した簡易プログラムを用いて評価した。

燃料温度履歴、ギャップ幅履歴、照射末期の径方向温度分布を評価した結果、燃料ピンは照射中に燃料溶融に対して十分な裕度を持つことが示された。また、プレナムガス圧履歴と照射後の被覆管の変形プロファイルの評価結果から、燃料ピンのガスプレナム体積を十分に確保しているため、ガス圧による被覆管の変形が生じないことも示された。

0.4wt%の Am 添加燃料では、燃料中心温度の上昇が計算されたが、U-Pu-Zr 燃料と比較してこの温度上昇分は僅かであった。

これらの結果から、本研究で検討した燃料仕様と照射条件において、U-Pu-Zr 金属燃料ピンは約 140GWd/t まで健全に照射できると考えられる。

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1. Introduction

In feasibility studies on a commercialized fast reactor cycle system in Japan, a sodium cooled reactor core design has been investigated.[1)2)] A fuel pin concept consisting U-Pu-Zr metallic fuel slug and oxide dispersion strengthened ferritic steel (ODS) cladding has been considered to be an alternative concept to the conventional concept consisting of MOX fuel and ODS cladding for the sodium cooled reactor core.[3)]

In the present work, to investigate the serviceability of the U-Pu-Zr metallic fuel pin under a high burnup condition, some calculations of irradiation performances of the fuel pin were conducted by a simplified calculation program developed in JAEA.

The irradiation behavior models and fuel properties for the analytical code were selected based on the information of metallic fuel characteristics including fuel properties and irradiation behavior obtained from open literatures and collaborative research activities with the Central Research Institute of Electric Power Industry.

2. Outline of calculation program

A simplified calculation program for U-Pu(TRU)-Zr metallic fuel performance analysis has been developed. This program is an R-Z system and models the thermal behaviors of a fuel pin during irradiation using 10 axial nodes, each having 26 radial nodes, 20 of which are for the fuel region and 6 for the cladding region. Mass transports in the direction are not taken into consideration, except for FP gases released into the plenum space and fuel-cladding gap. The program is limited to analyses of fuel pins having a smear density not over 75%T.D. Table 1 shows the evaluated behaviors. Some conservative and simplified models as follows were incorporated into the program;

1) For the FP gas release, the fractional release rate under irradiation was taken as the constant value of 90 %,

2) For the fuel and cladding mechanical analyses, the fuel-cladding contact pressure under irradiation was taken as the constant value of zero, because it was reported that no considerable contacts between fuel and cladding were obtained in the case of fuels having a smear density of less or equal to 75%TD.[4)] Only the stress-strain analysis of cladding due to the plenum gas pressure were conducted,

3) For the fuel restructuring and fuel constituents migrations, they were not taken into account,

4) For the penetration of bonding sodium into cavities or porosities in fuel, it was not taken into account.

The finite difference analysis procedure is applied to the thermal analysis, and the stress-strain analysis procedure based on the generalized plane strain is applied to the mechanical analysis of cladding. Figures 1 and 2 show the geometrical model and flow chart of the program, respectively.

3. Calculation conditions

Table 2 shows fuel specifications and irradiation conditions for this investigation. A fuel pin having a metallic U-Pu-Zr slug with the ODS cladding was taken for this investigation. The bonding material filling the fuel-cladding gap was sodium. The level of bonding sodium was up to the top of the fuel column. The irradiation time was taken as 2205 days (3 cycles). The maximum neutron fluence was taken as $5.50 \times 10^{23} n \cdot cm^{-2}$, then the maximum local burnup was evaluated to be as 140 GWd/t. The coolant inlet temperature was taken as 668K. Calculations were conducted at the following 5 axial positions; X/L = 0.9, 0.7, 0.5, 0.3, and 0.1. Axial distribution conditions at BOL and EOL of LHR and cladding midwall temperature are shown in Fig. 3 and Fig. 4, respectively. Profile conditions of LHR and cladding midwall temperature at each axial position of the calculations are shown in Fig. 5 and Fig. 6, respectively.

These conditions are based on the current results of feasibility studies on a commercialized fast reactor cycle system in Japan.[3)]

4. Results and discussions

From Fig. 7 to Fig. 11, the evaluated fuel temperature profiles at each axial position are shown. It is easily seen that fuel temperature increased with burnup in the early stage of each cycle and slightly decreased with burnup in the later stage of each cycle. The evaluated gap width between fuel and cladding at each axial position are shown in Fig.12. It is easily seen that the gap between fuel and cladding were plugged in the early stage of the irradiation and the gap conductance did not affect the fuel temperature. Then, the obtained temperature profiles were attributed to the profile conditions of LHR. From Fig. 13 to Fig. 17, the evaluated radial temperature distributions at EOL of each axial position are shown. It is easily seen that no anomalies were found in the radial temperature distributions. It can be seen from those results that the fuel pin had enough safety margin to fuel melting under the irradiation.

Figure 18 shows the evaluated profiles of pressure on the cladding inner surface due to the plenum gases. Figure 19 shows the evaluated cladding deformation after the irradiation. It is easily seen from the results that the fuel pin had enough plenum volume not to cause considerable cladding deformations by plenum gas pressure.

Therefore, it is concluded that the metallic U-Pu-Zr fuel pin having the specifications and irradiation conditions used in this investigation would be irradiated moderately up to approximately 140GWd/t with well integrity.

Figures 20 and 21 show the fuel centerline temperature of 0.4%Am(in heavy metal) bearing fuel. In this calculation Am contribution was assumed to be the same as that of Pu and fuel thermal conductivity in Am bearing fuel slightly decreases. Therefore, fuel centerline temperature of Am bearing fuel is slightly higher than that of U-Pu-Zr fuel, which is indicated as broken line in Figs. 20 and 21. The difference of these temperatures is limited within one degree C. The contribution of Am inclusion may not be significant when the Am content is less than 1% of heavy metal.

5. Conclusion

Some calculations of irradiation performances of the fuel pin were conducted by a simplified calculation program developed in JAEA to investigate the serviceability of a fuel pin, consisting of metallic U-Pu-Zr fuel slug and ODS cladding, under a high burnup condition.

The fuel temperature profiles, gap width profiles, and radial temperature distribution at EOL were evaluated. Those results show that the fuel pin has enough safety margin to fuel melting under the irradiation. Also, the profiles of plenum gas pressure and the cladding deformation after irradiation were evaluated. Those results show that the fuel pin has enough plenum volume not to cause considerable cladding deformations by plenum gas pressure.

In case of 0.4%Am bearing fuel, calculation result shows that fuel centerline temperature becomes high, but increase from U-Pu-Zr fuel is insignificant.

It is deemed from the obtained results that the metallic U-Pu-Zr fuel pin having the specifications and irradiation conditions used in this investigation would be irradiated moderately up to approximately 140GWd/t with well integrity.

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Table 1 Behaviors evaluated by the calculation program

Evaluated fuel behaviors
Temperature distribution
Thermal expansion
Fission gas release
Swelling
Evaluated cladding behaviors
Temperature distribution
Thermal expansion
Void swelling
Creep deformation due to plenum gas pressure
Cladding corrosion due to FPs
Cladding liquid phase penetration
Creep damage

Table 2 Designed fuel specifications and irradiation conditions

Item		Unit	Value
Fuel	Type		Slug
	Outer diameter	mm	6.496
	Density	%TD	100
	Pu cont.(including MA)	wt.%	11.47
	Zr cont.	wt.%	6.0
Fuel colum length		mm	750
Plenum	upper	mm	1350
Cladding	Material		ODS
	Inner diameter	mm	7.5
	Outer diameter	mm	8.5
	Thickness	mm	0.5
Bonding	Material		Sodium
	Filling level	mm	up to fuel column
Irradiation duration		day	2205 (1cycle : 735)
Max. LHR		W/cm	347
Max. Cladding midwall temperature		K	878
Max. Neutron fluence(>0.1MeV)		n/cm^2	5.50E23
Max. Burnup (local position)		GWD/t	140
Coolant	Material		Sodium
	Inlet temperature	K	668

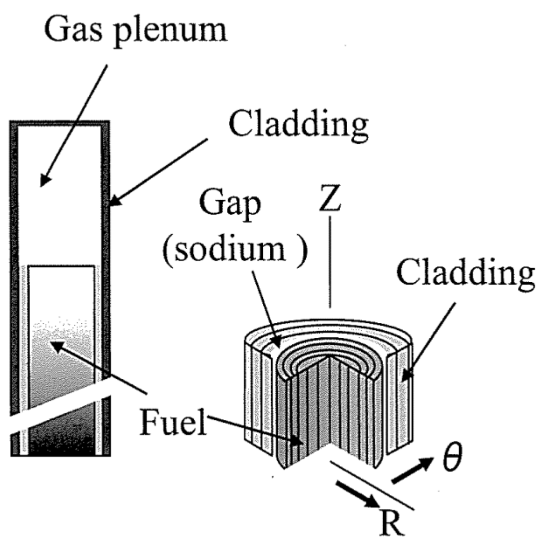

Fig.1 Geometrical model of the calculation program

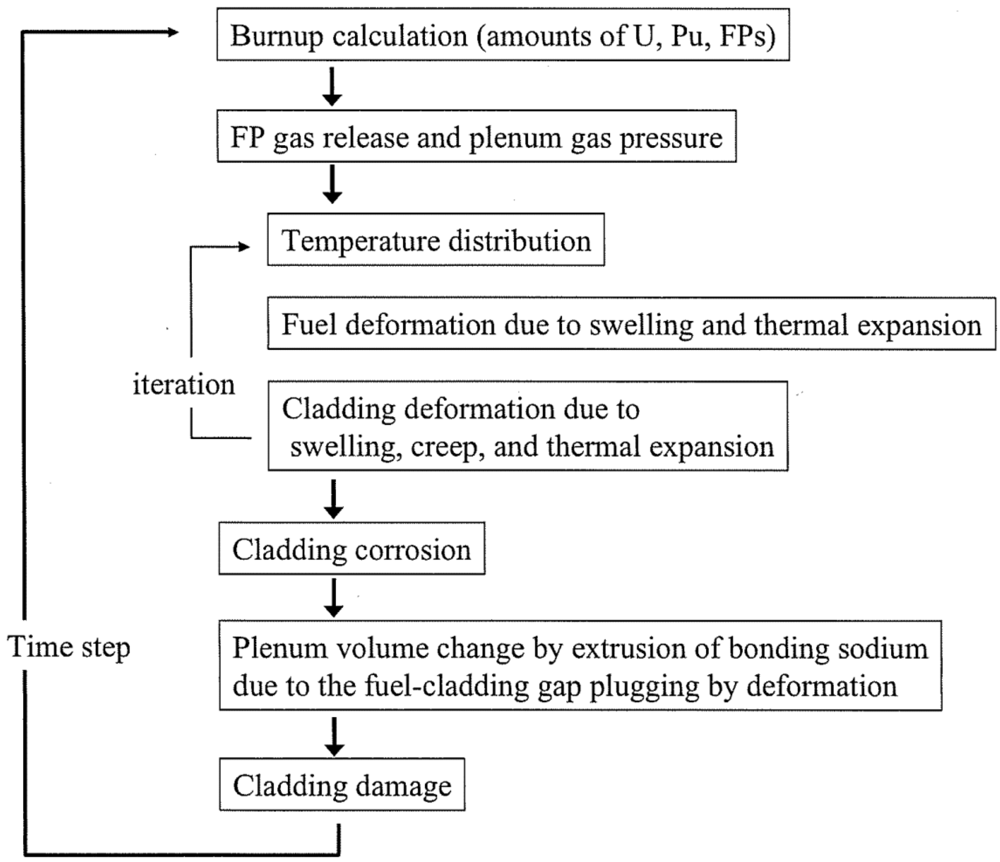

Fig.2 Flow chart of the calculation program

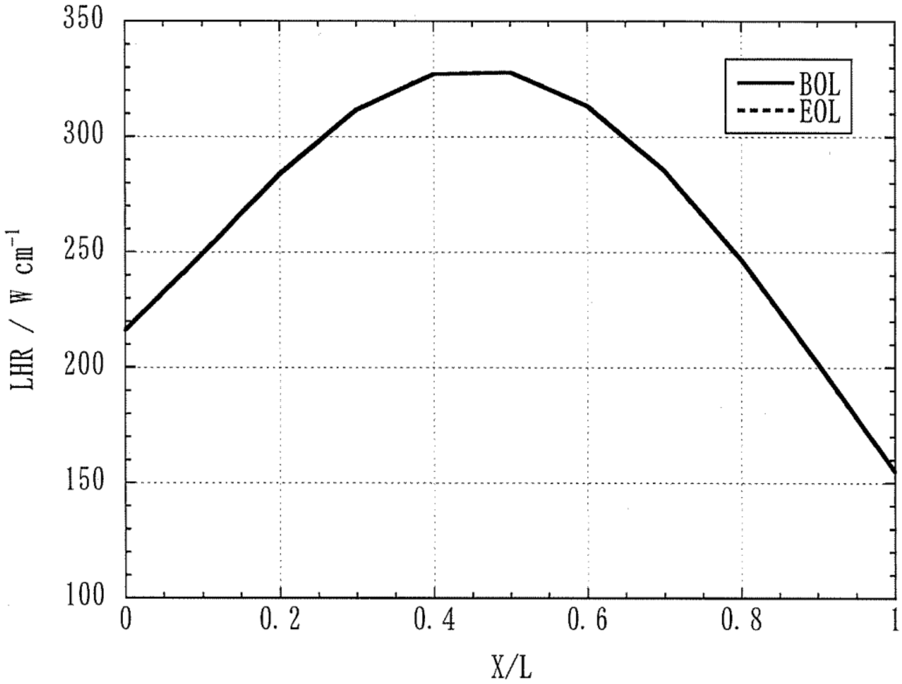

Fig.3 Axial distribution condition of LHR

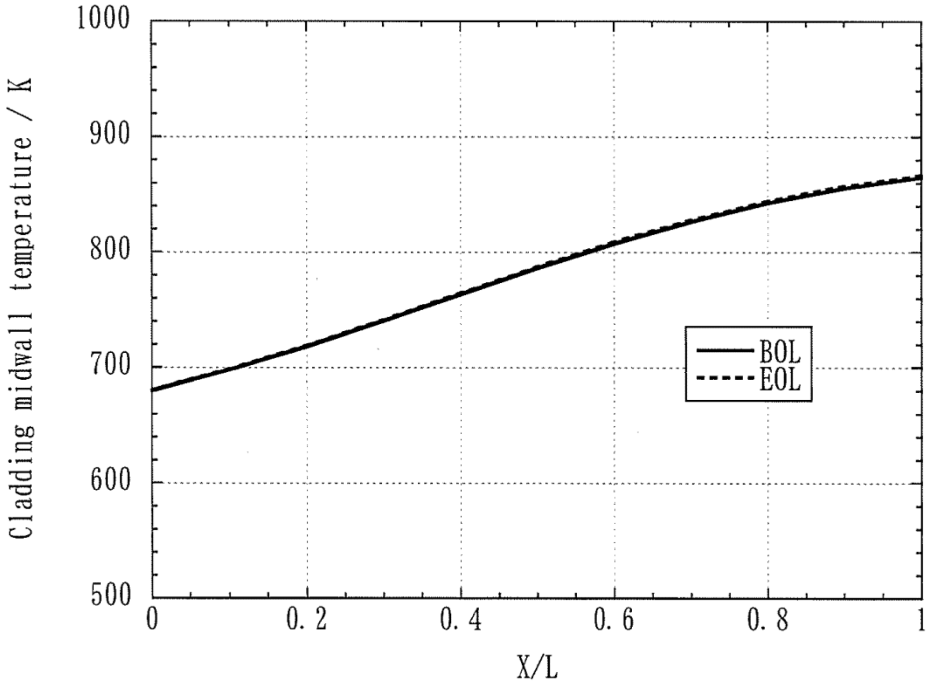

Fig.4 Axial distribution condition of cladding midwall temperature

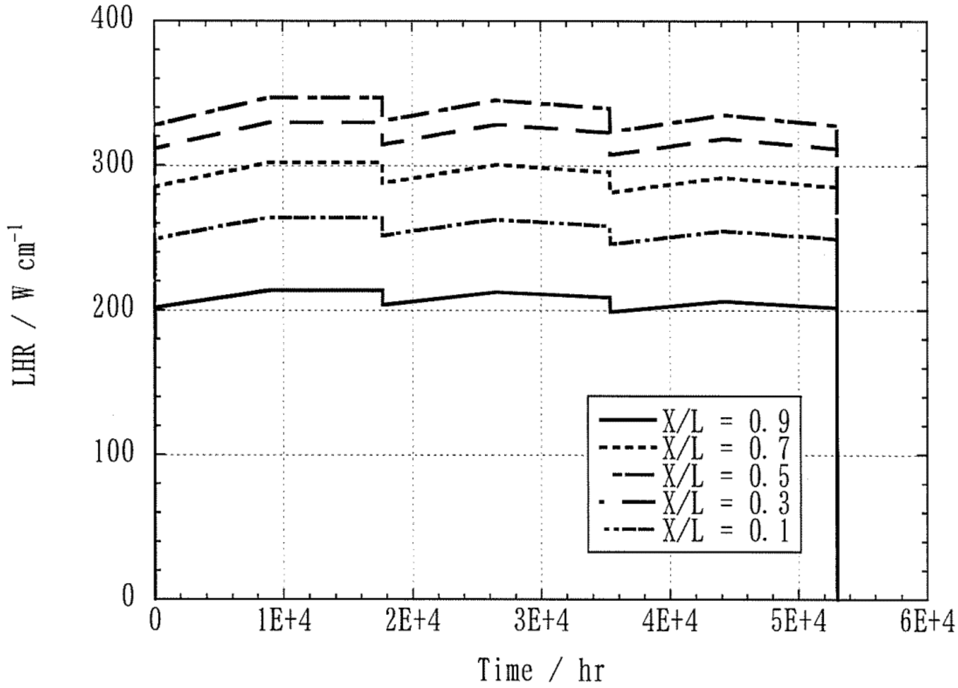

Fig.5 Profile condition of LHR

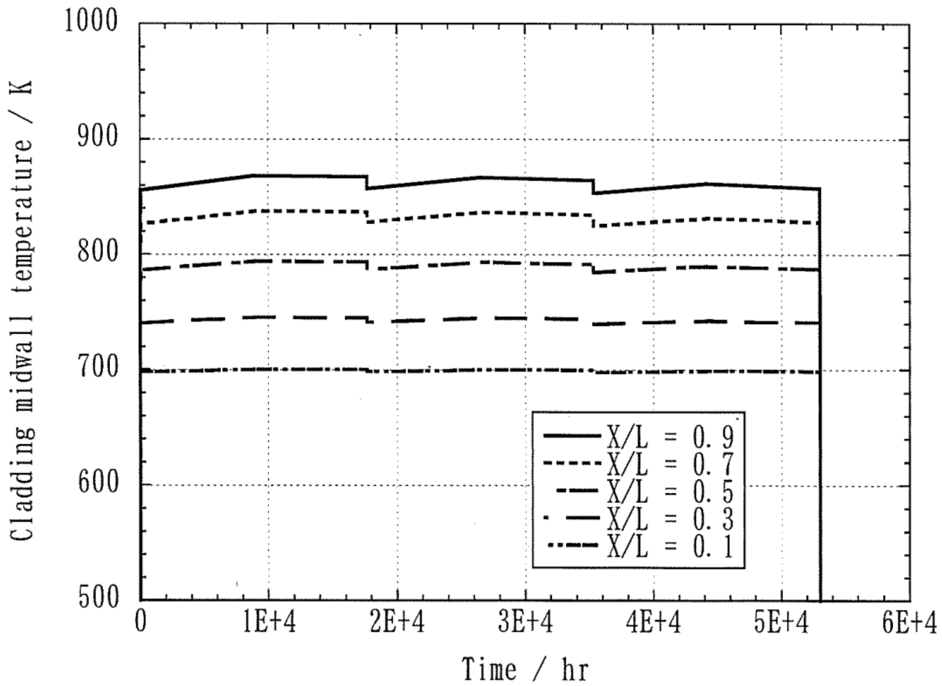

Fig.6 Profile condition of cladding midwall temperature

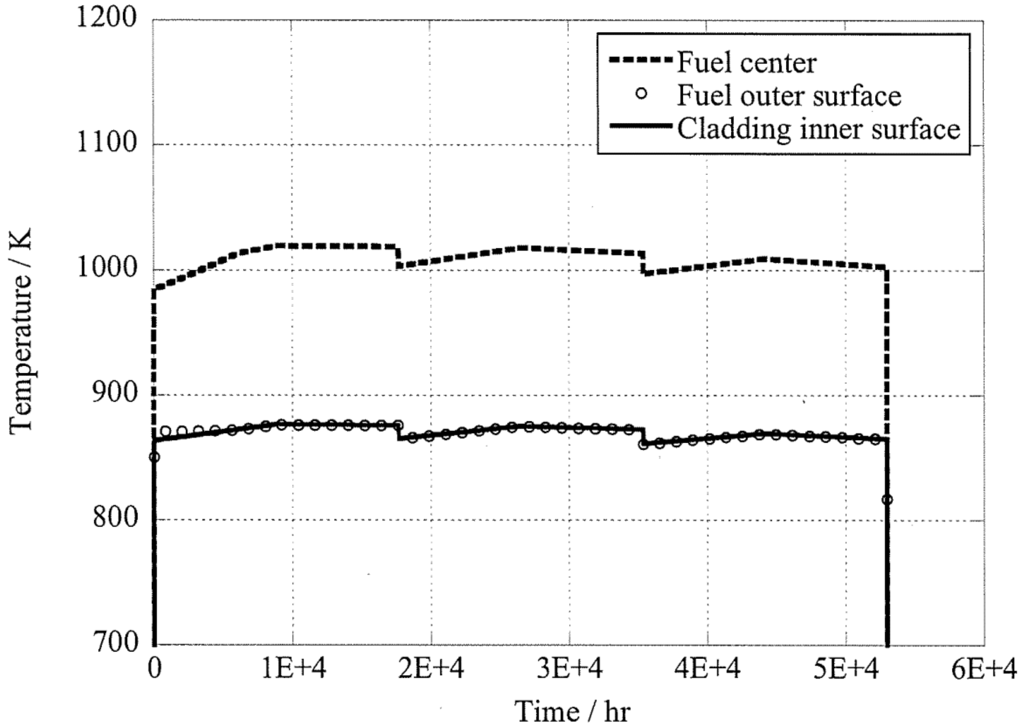

Fig.7 Temperature profiles at X/L=0.9

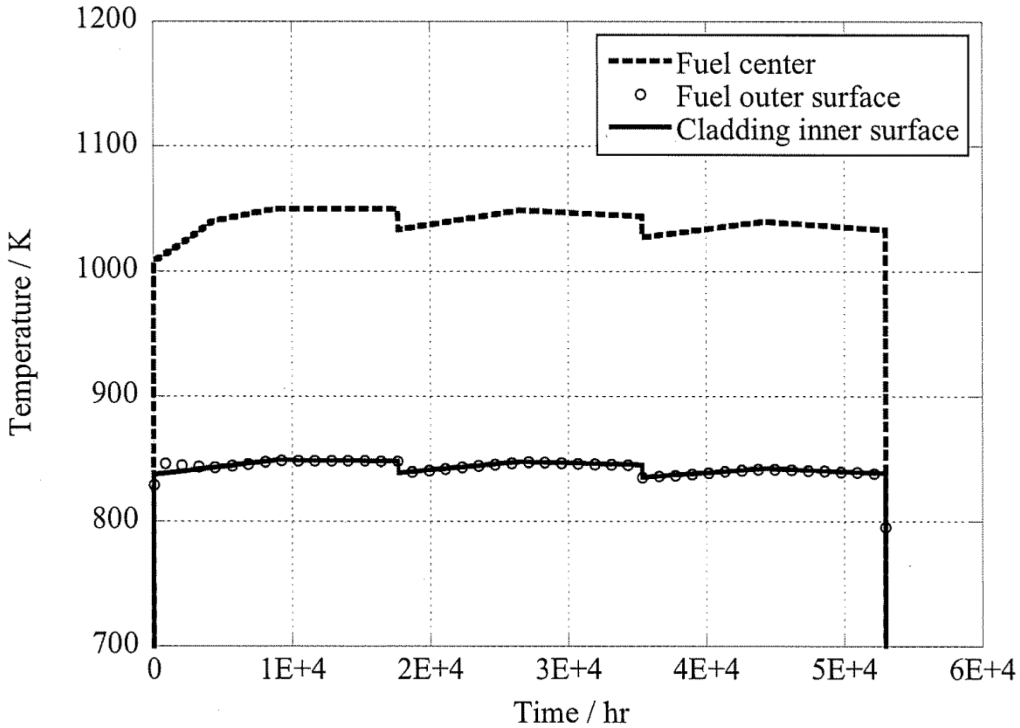

Fig.8 Temperature profiles at X/L=0.7

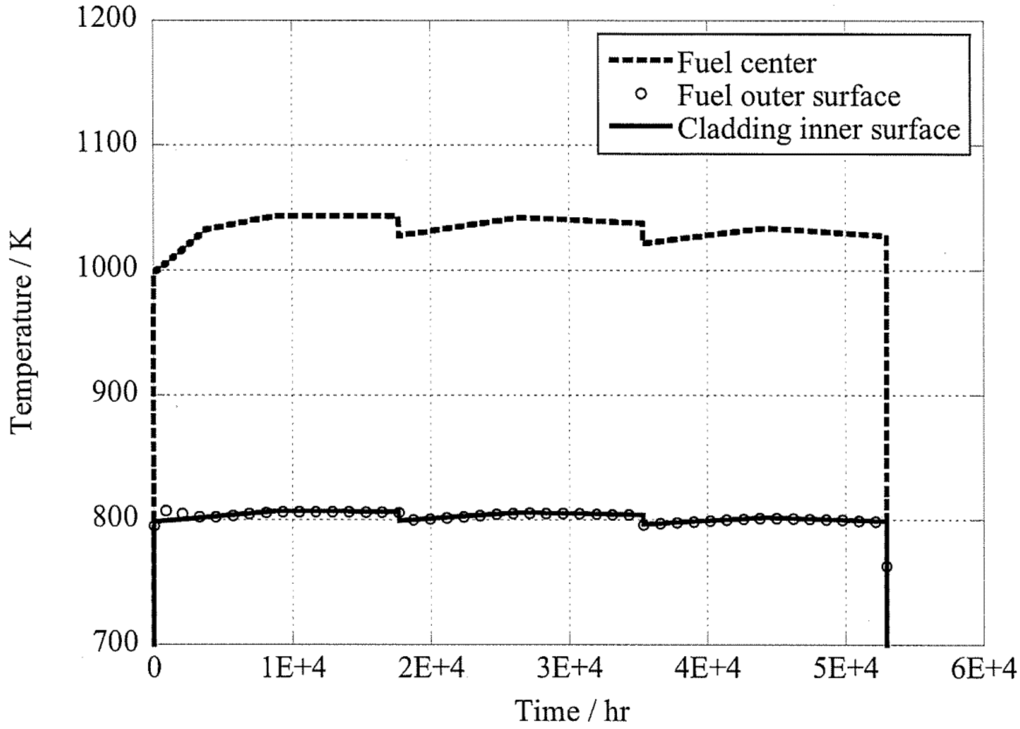

Fig.9　Temperature profiles at X/L=0.5

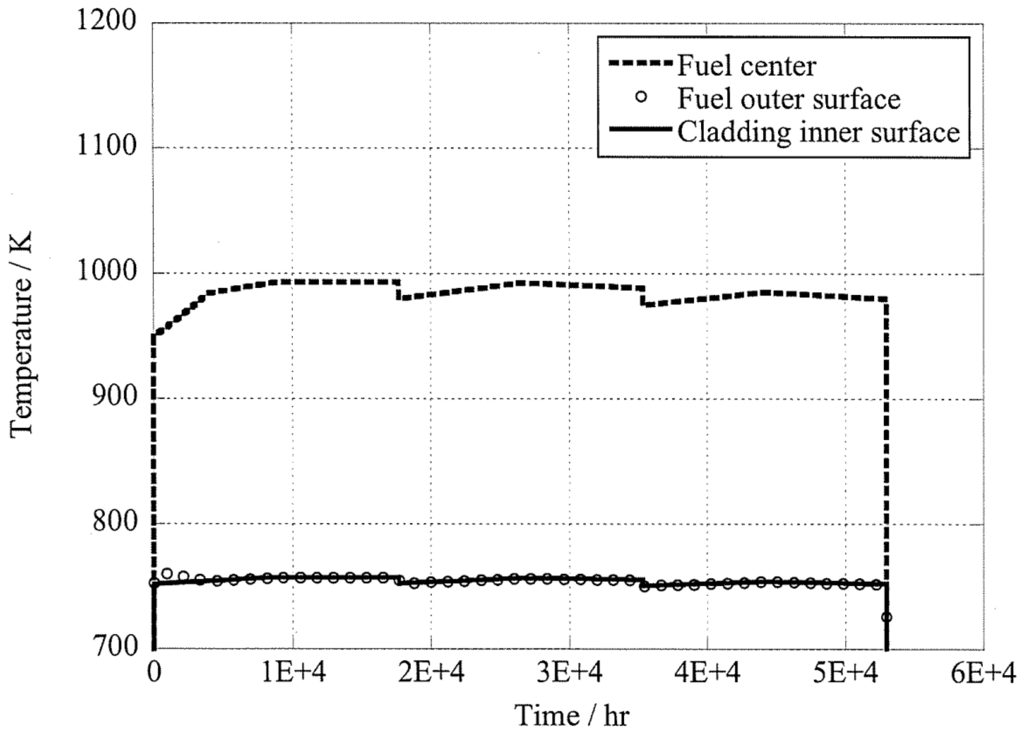

Fig.10　Temperature profiles at X/L=0.3

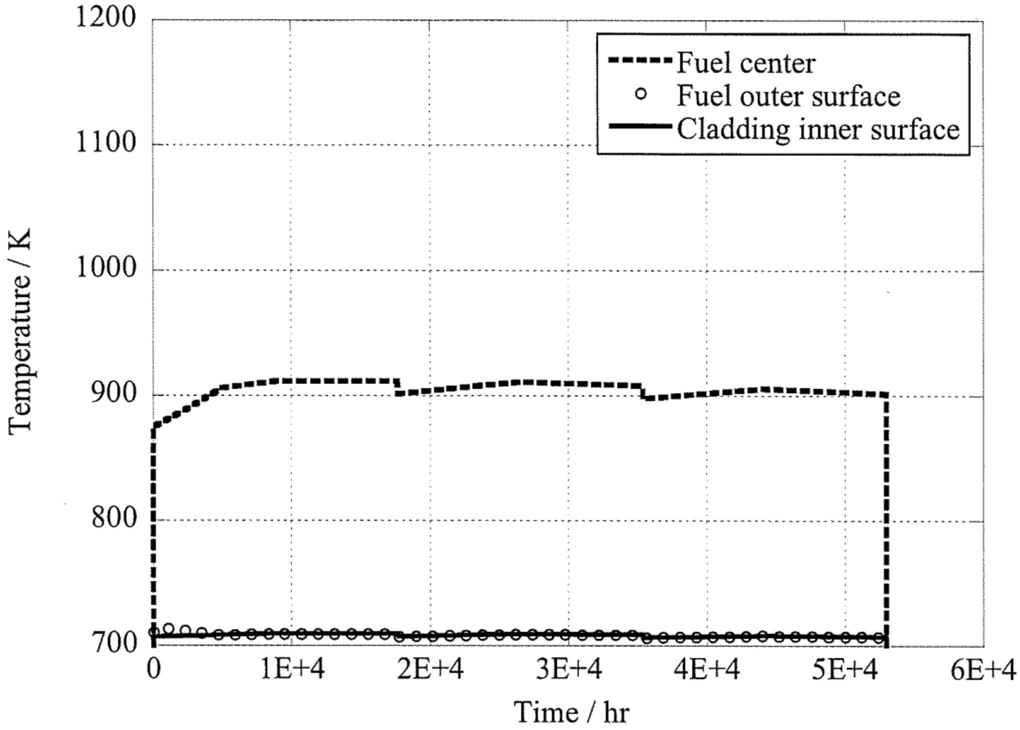

Fig.11 Temperature profiles at X/L=0.1

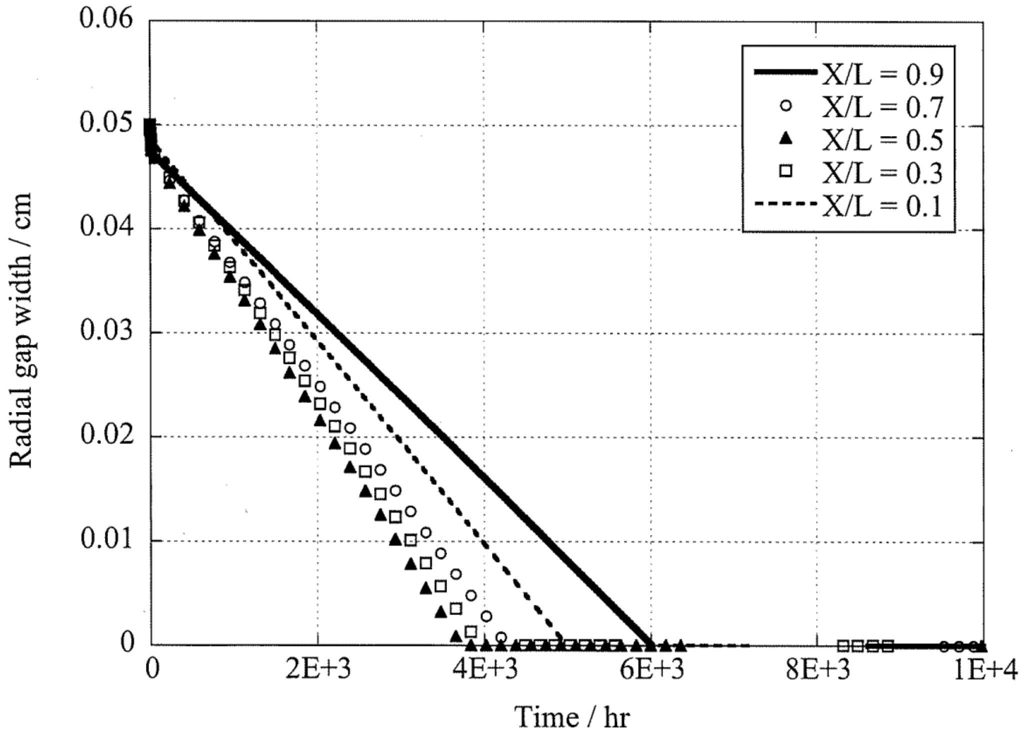

Fig.12 Profiles of gap width up to 1×10^4hr

(From 1E4 hr to the end of irradiation, the gap was found to be plugged at each axial position.)

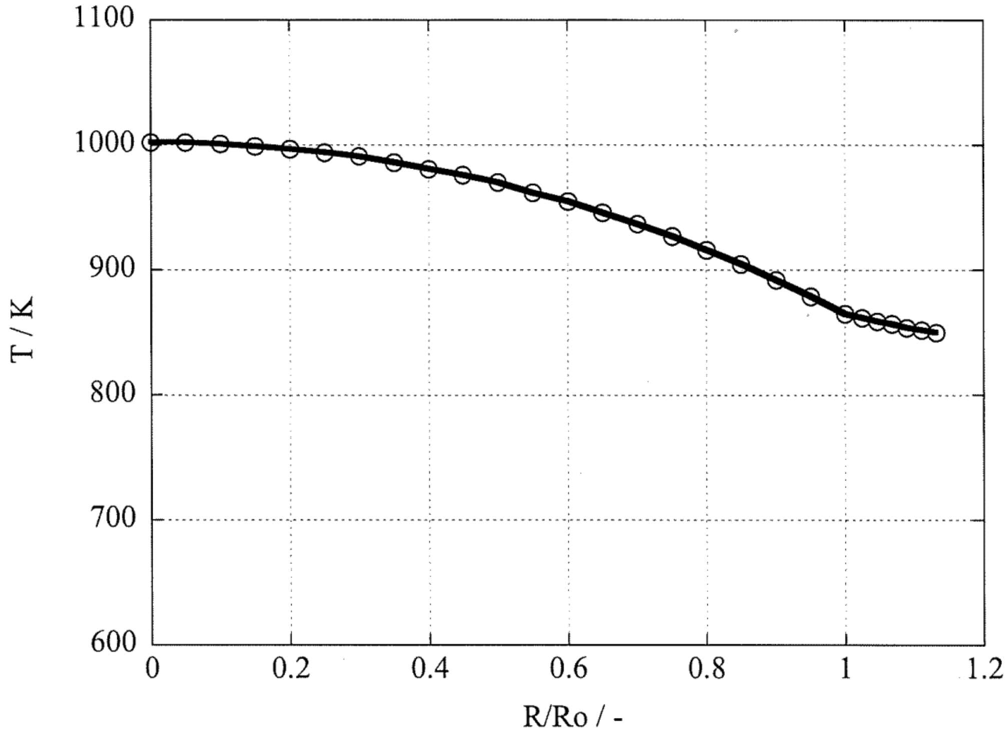

Fig.13 Radial distribution of fuel temperature at X/L=0.9

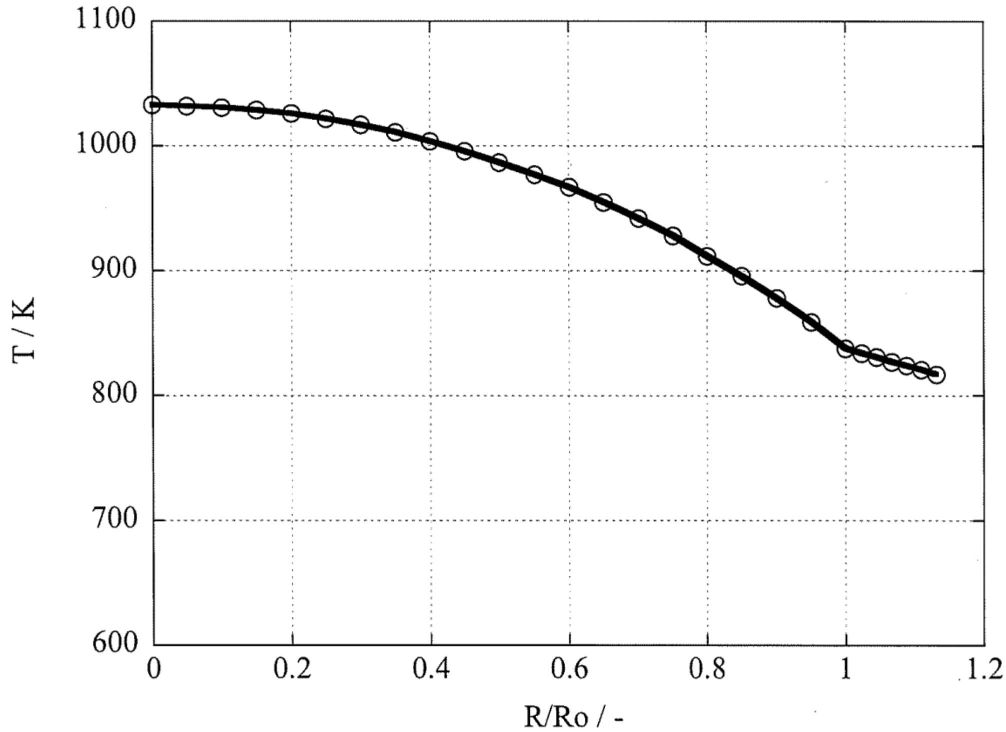

Fig.14 Radial distribution of fuel temperature at X/L=0.7

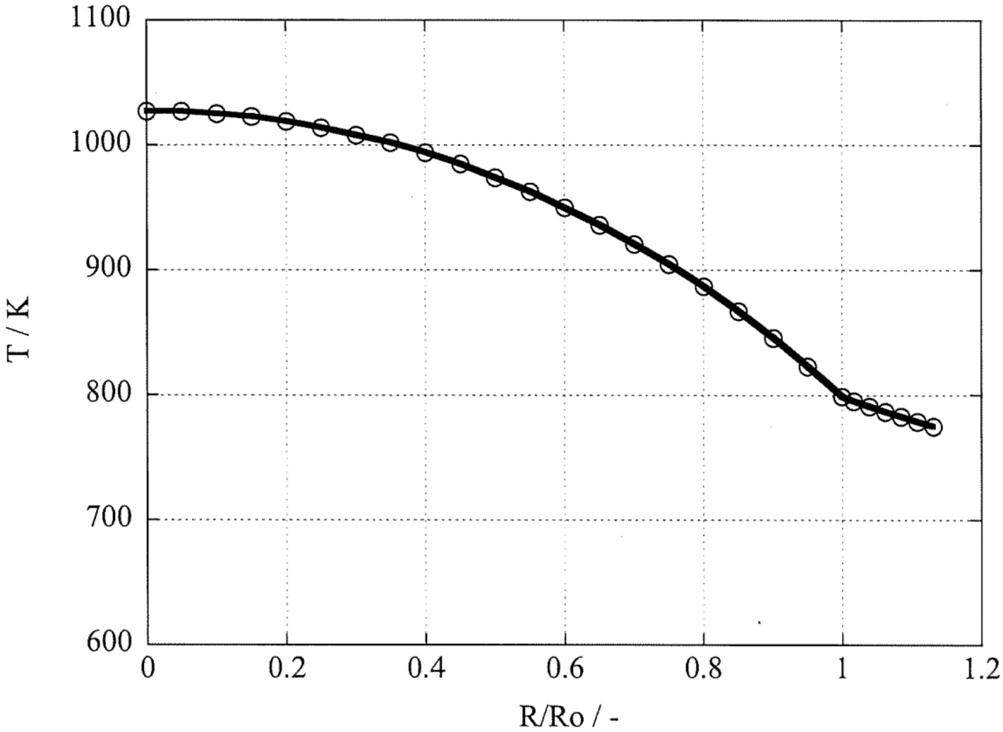

Fig.15 Radial distribution of fuel temperature at X/L=0.5

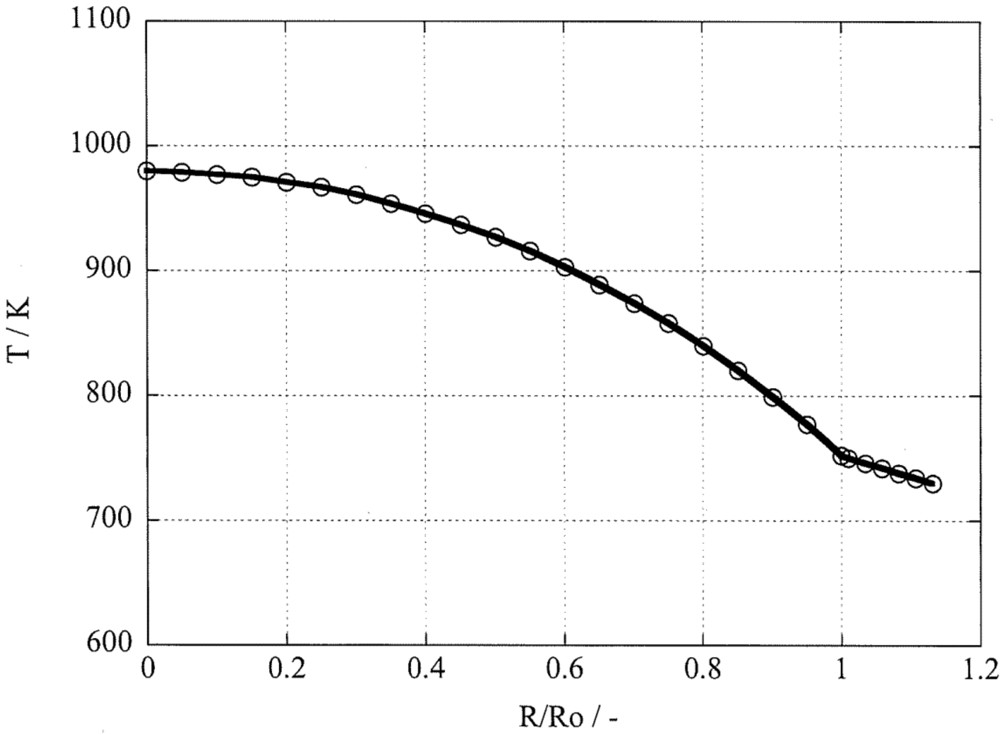

Fig.16 Radial distribution of fuel temperature at X/L=0.3

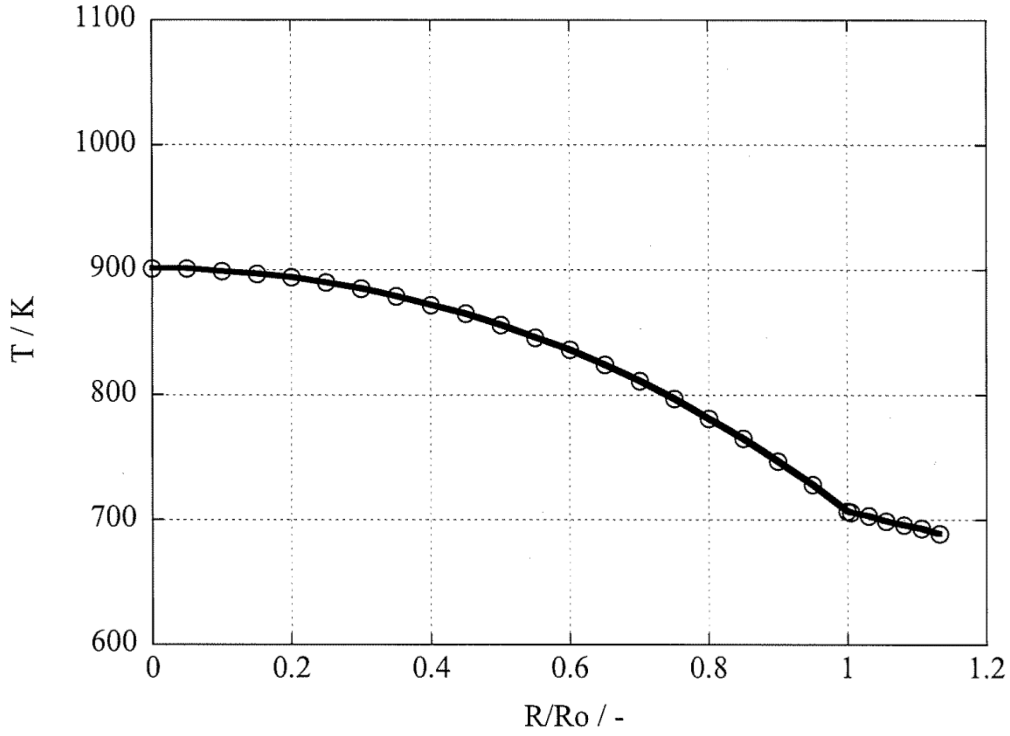

Fig.17 Radial distribution of fuel temperature at X/L=0.1

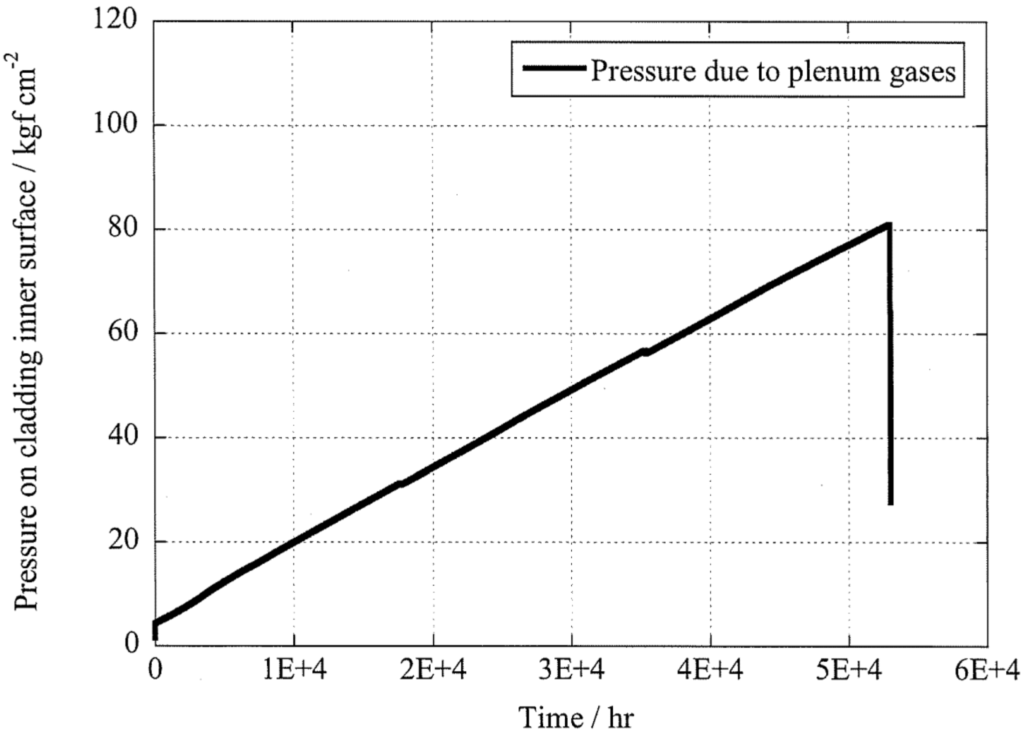

Fig.18 Profile of pressure on the cladding inner surface

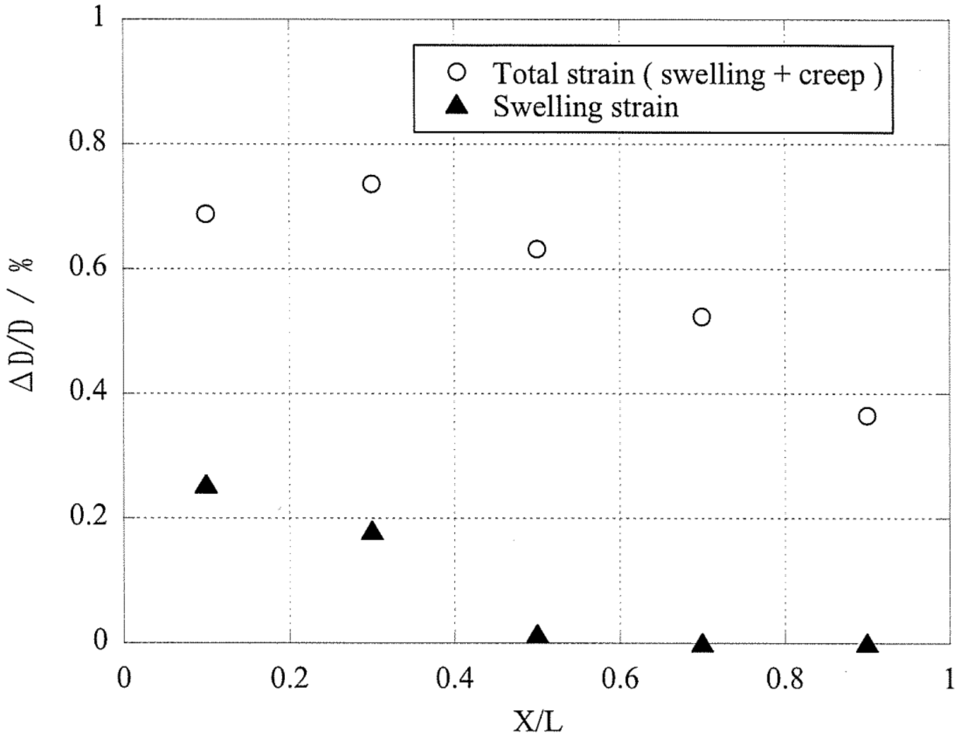

Fig.19 Cladding deformation after irradiation

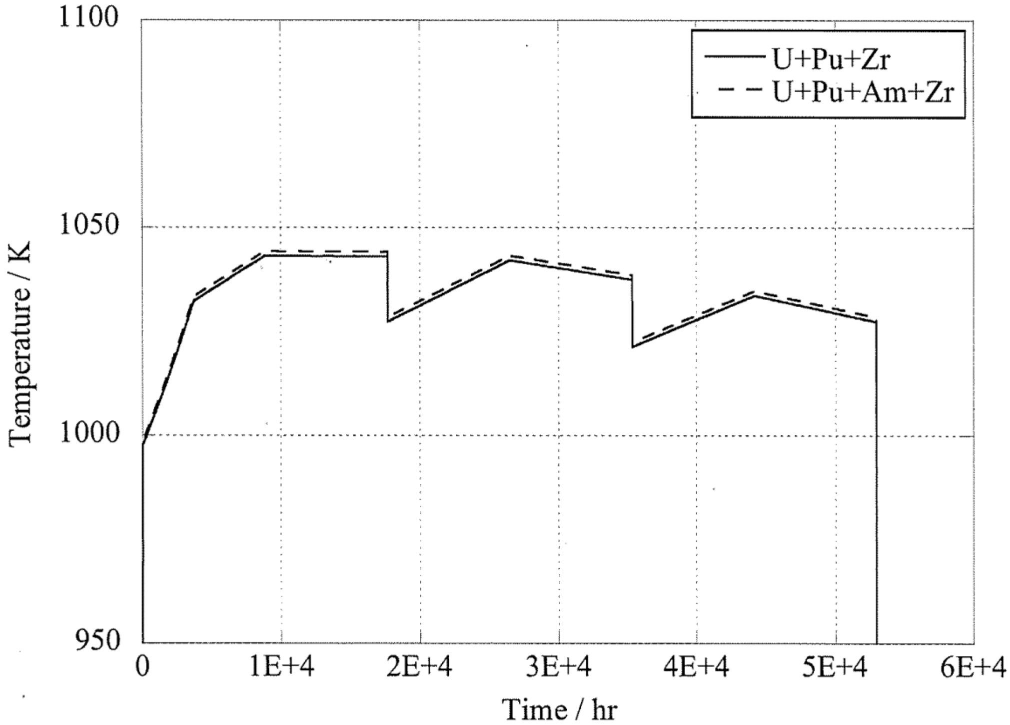

Fig.20 History of fuel center temperature at X/L=0.5

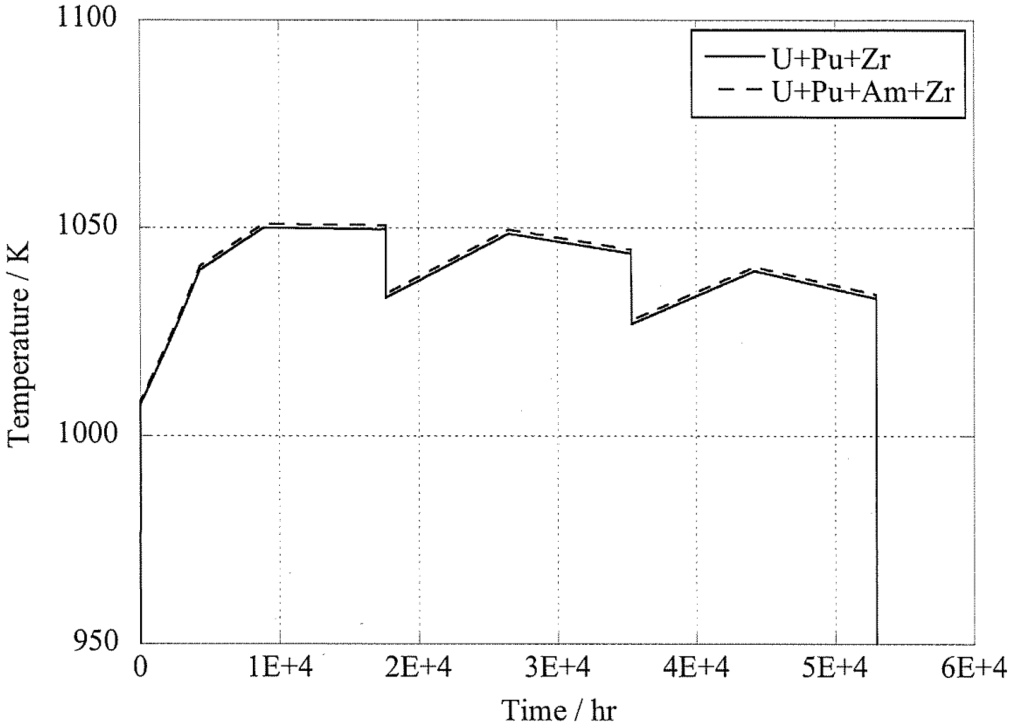

Fig.21 History of fuel center temperature at X/L=0.7