Proceedings of the Fourth International Symposium on ADVANCED NUCLEAR ENERGY RESEARCH
ROLES AND DIRECTION OF MATERIAL SCIENCE IN NUCLEAR TECHNOLOGY

PERFORMANCE OF HIGH BURNED PWR FUEL DURING TRANSIENT

Kazuaki YANAGISAWA and Toshio FUJISHIRO

Department of Fuel Safety Research
Nuclear Safety Research Center
Tokai Research Establishment
Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken, 319-11, Japan
Phone: 0292-82-5955, Fax.: 0292-82-6160

ABSTRACT: In a majority of Japanese light water type commercial power reactors(LWRs), UO_2 pellet sheathed by zircaloy cladding is used. Licensed discharged burn-up of the PWR fuel rod is going to be increased from 39 MWd/kgU to 48 MWd/kgU. This requests the increased reliability of cladding material as a strong barrier against fission product(FP).

A long time usage in the neutron field and in the high temperature coolant will cause the zircaloy hardening and embrittlement. The cladding material is also degraded by waterside corrosion. These degradations are enhanced much by increased burn-up. A increased magnitude of the pellet-cladding mechanical interaction (PCMI) is of importance for increasing the stress of cladding material. In addition, aggressive FPs released from the fuel tends to attack the cladding material to cause stress corrosion cracking (SCC).

At the Nuclear Safety Research Reactor(NSRR) in JAERI, 14x14 PWR type fuel rods preirradiated up to 42 MWd/kgU was prepared for the transient pulse irradiation under the simulated reactivity initiated accident (RIA) conditions. This will cause a prompt increase of the fuel temperature and stress on the highly burned cladding material.

In the present paper, steady-state and transient bahavior observed from the tested PWR fuel rod and calculational results obtained from the computer code FPRETAIN will be described.

Keywords : Performance, PWR fuel, RIA, transient, PCMI, zircaloy cladding, NSRR, FP gas release, SCC, FPRETAIN

1. INTRODUCTION

In LWRs, UO_2 fuel pellet sheathed by zircaloy has been used widely, to date. Recently, a reliability of zircaloy cladding material under the high fluence and the high coolant termperature(320℃) and pressure(16MPa) become more important for the use at higher burn-up stages. The key parameters from the reliability point of view are:

(1) Waterside comosion combined with inside corrosion of the cladding
(2) Zircaloy hardening and embrittlement
(3) PCMI combined with stress corrosion cracking(SCC), so called the PCI-SCC
(4) Irradiation induced growth of fuel rod and assembly

These factors are influential to the fuel integrities not only under steady-state but under transient conditions such as RIA.

In a postulated RIA, the insertion of a large amount of excess reactivity caused by an inadvertent control rod withdrawal or ejection results in a prompt power excursion of the reactor[1]. This reactor power excursion causes a prompt overheating of the fuel rods and may result in fuel rod failure.

In-core experiments have been conducted in the NSRR program to study LWR fuel bahavior under RIA conditions. The Licensing Guideline for RIA Safety in Japan[2] is based on the results obtained from the NSRR program in Japan, and the SPERT [3] and PBF [4] programs in the United States. The Guideline data base is, however, mainly from experiments with unirradiated UO_2 rods. Experiments to extend the data base for the preirradiated LWR fuels have been performed since 1989 in the NSRR program. In the first stage of this experiment series, fuel integrities have been studied with preirradiated PWR type fuel having a burn-up up to 42 MWd/kgU. Attached in-core instrumentation, and pre-pulse and post-pulse irradiation examinations (PIEs) provided transient fuel performance data.

In the present work, the in-core behavior of PWR fuel rod during steady-state operation and transient one is studied by experiment. The experimental data were used for verification of the computer code FPRETAIN developed by JAERI.

2. EXPERIMENT

2.1 Characteristics of the Test Fuel Rod

The test fuel rods used in this study were obtained from the PWR type commercial power reactors. They are denoted as MH and GK fuel rods. The active fuel column was approximately 3.66m in length. As-fabricated characteristics of the fuel rod are shown in Table I. The fuel rod was a 14x14 PWR type having an outer diameter (O.D.) of 10.72mm. Initial enrichment of the fuel was 2.6 w/o for MH rod and 3.4 w/o for GK rod. The cladding material was stress-relieved zircaloy-4. The as-built diametral gap was 0.17 mm. The rod was initially prepressurized to ca.3 MPa with 97%He+3%Air. A longitudinal section of the two fuel rods is shown in Fig. 1. Henceforth, this is called the original test fuel rod and abbreviated OTFR.

The OTFR was preirradiated to rod average burn-up of 41 MWd/kgU in maximum. The OTFR had an average linear power of 18 kW/m for MH rod and 20 kW/m for GK rod. Coolant pressure was 15.82 MPa and the coolant inlet and outlet temperatures were 289°C and 320°C. The axial flux distribution of the OTFR was smooth and relatively flat except at the rod bottom, top and spacer grid regions. The peaking factor was 1.25.

2.2. Refabrication of the OTFR

Due to the limited height of the NSRR core, 0.38m, the active column length of the OTFR had to be shortened to approximately 0.122m. Henceforth, the shortend test fuel rods are called the segmented test fuel rods, and abbreviated the STFRs. The axial flux distribution of the STFRs was smooth and almost unity. The STFR is schematically shown in Fig. 1. Refabrication process details are described elsewhere[5].

2.3. Pulse Irradiation of the STFR

2.3.1. In-core instrumentation

In-core instrumentations were installed on each STFR. They are cladding axial elongation sensor, fuel stack axial elongation sensor, Pt/Pt-13%Rh thermocouple and strain gauge pressure sensor. As shown in Fig.1, they were attached directly to the STFR.

As shown in Fig. 2, the STFR was located in the center of the irradiation capsule. The midpoint of the active column length of the STFR was positioned at the NSRR core midplane. As a result, the axial power profile of STFR was smooth and flat. All pulse irradiations were conducted in stagnant water at room temperature (~20°C) and at 1 atmosphere of pressure inside the double sealed irradiation capsule.

2.3.2. Pulse irradiation history and energy deposition

A typical pulse power history is shown in Fig. 3. The half power width is about 4.4 ms for a maximum pulse operation of 110 MW•s. This value varies from a minimum of 4.4 ms to about 20 ms depending on the magnitude of the inserted reactivity. The deposited energy E_g (cal/g•fuel) in each test rod was estimated from the integral value of reactor power P(MW•s) measured by two γ-chambers. The power conversion ratio K_g (cal/g-fuel per MW•s), that is, the ratio of fuel rod power to reactor power, was determined through fuel burn-up analysis[6]. The radial and axial power skew was taken into consideration.

3. RESULTS AND DISCUSSION

3.1. Fuel Performance During Preirradiation Stage

3.1.1. Waterside corrosion

Waterside corrosion of the cladding is increased with increasing burn-up. Fig. 4 shows the data obtained from various type of fuels from PWRs[7-14]. The relatively large data scatter may be due to different PWR operating conditions. The OTFR had oxide films with thickness from 14μm to 30μm. These are similar to other corrosion data.

3.1.2. Embrittlement of cladding

It is worthy of mentioning that 10% reduction of the cladding thickness is corresponded roughly to the increased oxide thickness of 60μm. In proportion to the increase of burn-up, amounts of hydrogen pick-up into the cladding are increased resulting in reductions of the ductility of the cladding material. As shown in Fig.5, yield stress of the cladding will increase to the magnitude of 120% of that of initial value. This was due to accumulation of fast neutron and reduced the ductility of cladding material.

3.1.3. Mechanical stress

Burst and tensile stresses of the preirradiated zircaloy as a function of temperature are shown in Fig. 6. Note, that tensile stress of zircaloy at room temperature is about 800MPa while burst stress is the 940MPa. At increased test temperatures the two values are decreased to 650MPa(400°C). It is worthy of mentioning that hoop stress necessary to induce PCI-SCC failure in the zircaloy cladding is about 270MPa for preirradiated LWR fuel rod[15].

3.1.4. Creep-down

Cladding creep-down occurs in PWRs as a function of burn-up. The data obtained recently from domestic PWRs[16-22] and earlier from Westinghouse of the USA[24] are summarized in Fig. 7. The creep-down in PWR fuel rods was < 1%. The creep-down of the OTFR was < 0.8%. There is no significant difference between the OTFR and the others.

3.1.5. Fission gas release

Fission gas release(FGR) obtained from the OTFR was compared with the data obtained from other domestic PWR fuels these were preirradiated up to burn-ups of 41 MWd/kgU[16-22]. The result is shown in Fig. 8. The FGR of the OTFR was < 0.5%. The FGR of the OTFR was the same magnitude as that of other domestic and foreign PWRs[25]. The FGR from these fuels are attributed mainly to the recoil/knockout mechanism.

3.2. Transient Fuel Bahavior During Pulse Irradiation

The STFRs had energy depositions up to 112cal/g•fuel. No indication of fuel failure was observed for any of these. Figure 9 shows a typical example of in-core data obtained from the STFR pulsed at 112cal/g•fuel. The discussions on performance of the STFRs during pulse irradiation will be made in the followings.

3.2.1. FGR

As indicated in Fig. 9, the rod internal pressure increased very rapidly. The maximum pressure increase recorded was 1.93 MPa. The increased pressure was returned mostly to the initial level. Hence, the main reason of the pressure increase during transient was due to thermal expansion of the filler gas. The increase of pressure by FGR was significantly diminished by the ballooing of cladding occurred simultaneously.

As shown in Fig. 10, STFR GK-1 caused a net FGR of 12.2%. Code verification was made against this by using the transient computer code FPRETAIN [26]. The net FGR predicted by the code was almost 0%. The calculation was made by Booth type diffusion model. It can be said from this that the most possible mechanism of FGR during transient was not according to diffusion.

From the fuel microstructural study, it was founed that the most possible mechanism was FGR burst from the cracked fuel at fuel periphery. Due to this, the micro hair-cracking model was newly developed and implemented into the code. Obtained result from recalculation is shown in Fig. 11. As clear from the figure, the predicted FGR is in excellent agreement with experiment.

3.2.2. PCMI

To correctly understand PCMI, one should use the radial deformation data [27]. These data, however, could not obtain from the experiment so that cladding and fuel axial elongation data were used instead of them.

Figure 12 showed that during pulse irradiation the cladding axial strain increased as much as 1.1%. This indicated that PCMI followed by cladding thermal expansion occurred. As shown in the figune, the peak cladding strains obtained were 0.43, 0.54 0.99, and 1.1% at energy depositions of 60, 68, 83 and 112 cal/g•fuel, respectively. The 14x14 PWR rod (BU=41 MWd/kgU) had a peak cladding strain of 1.1% but did not fail. Peak power was attained within 0.2s. Then very rapid relaxation of strain occurred within a few seconds. This relaxation was effective to the STFR preventing not only from PCMI failure but also from PCI-SCC failure[28].

Usually, the cladding axial strain in commercial LWR fuel is as high as 0.1%[23]. In the case of power ramping, the observed cladding strain may be as high as 0.4%[27]. For example, a 14x14 type PWR rod in TRANS-RAMP II (BU=30 MWd/kgU) showed an axial strain of 0.08% and then failed[29]. It should be noted the rod was power ramped with a rate of 1.5 kW/ms and held at 60 kW/m for 34s without causing a significant relaxation .

Predicted cladding maximum hoop stress by the computer code FPRETAIN at energy depositions of 112 cal/g•fuel of the tested fuel rod was < 200 MPa. The values were much lower than those required to cause either PCMI failure (ca.900 MPa) or PCI-SCC failure (ca.270 MPa)[26]. Hence, the tested fuel had not enough stress to cause the fuel failure.

3.3. PIE After Pulse Irradiation

Visual inspection revealed no significant damages to the STFRs used. Diameter profile and eddy current data, however, showed that there were apparent ridge formations, hence the PCMI have occurred.

3.3.1. Fuel microstructural study

Fuel microstructures of the STFR examined at pre-pulse and post-pulse conditions(112 cal/g•fuel) are respectively shown in Fig.13(a) and 13(b). From the figure it is clear that there occurred micro hair-crackings at fuel periphery. This might be a strong reason for FGR burst observed.

4. CONCLUSION

A study was made on steady-state and transient fuel performance of a 14X14 PWR fuel rod preirradiated to a rod average burn-up of 41 MWd/kgU. The results obtained are as follows:

(1)A hard PCMI represented by ridging occurred during transient. Axial peak deformation was 1.1% at a deposited energy of 112 cal/g•fuel. However, fuel failure did not occur within this experimental range. Computer code calculation by FPRETAIN revealed that the cladding hoop stress was < 200 MPa and less to cause either PCMI failure or PCI-SCC one.

(2)The FGR of the tested fuel during steady-state stage was < 0.5%. This might be governed the recoil/knockout mechanism. The net FGR during transient at energy deposition of 112 cal/g•fuel was 12.2%. The net FGR predicted by computer code FPRETAIN with diffusion model was 0%. While the net FGR predicted by micro hair-cracking model was 11%. The latter was in excellent agreement with experiment. The most probable mechanism, therefore, to explain the net FGR observed in the experiment is the FGR burst caused by micro hair-cracking at fuel periphery. This was confirmed by the fuel microstructural study during PIE.

5. ACKNOWLEDGMENT

Appreciation is addressed to the KEPCO, Japan , for providing the preirradiated fuel rod to JAERI. The technical support obtained from the NSRR Operation Section is gratefully acknowledged. Thanks are also due to the engineering staffs of the Hot Laboratoryies, JAERI for their careful PIEs.

6. REFERENCES

[1] D. J. Diamond, C. J. Hsu, and R. Fitzpatrick, NUREG/CR-5368 (1990)
[2] Nuclear Safety Commission, (1984), (in Japanese)
[3] T. Fujishiro, R. J. Johnson, P. E. MacDonald and R. K.McCardel,NUREG/CR-0269, TREE-1237,(1978)
[4] P. E. MacDonald et al., Nuclear Safety 21, 582(1980)
[5] K. Yanagisawa et al., JAERI-M 90-091,(1990),(in Japanese)
[6] T. Suzuki et al., Fall Mtg. At. Energy Soc. Japan, H17 (1990)
[7] F. Garzarolli, H. Stehle, Ref.[16], IAEA-SM-288/24
[8] T. S. Tulenko et al., Ref.e[24], p.49
[9] M. G. Andrews et al., Ref.[20], p.90
[10] M. G. Balfour et al., WCAP-10180, (1982)
[11] T. D. Dyecha et al., IAEA Technical Committee Mtg. on External Cladding Corrosion in Water Power Reactor, Cadarache, France, (1985)
[12] F. G. Reshetnikov et al., ibid
[13] S. Suzuki et al., Ref.[7]
[14] Y. Irisa et al., Ref.[20], p.102
[15] K. Yanagisawa, JAERI-M 83-228 (1984), (in Japanese)
[16] T. Miki, Y. Kondo, IAEA Specialists' Mtg. on Examination of Fuel Assembly for Water Cooled Power Reactors ", Tokyo, (1981), p.79
[17] Y. Mishima, T. Okubo, ibid, p.161
[18] Y. Takada et al., IAEA Spec. Mtg. on PIE and Experience, Tokyo (1984)
[19] S. Suzuki et al., IAEA International Symposium on Improvements in Water Reactor Fuel Technology and Utilization, Stockholm, Sweden, IAEA-SM-288/66 (1986)
[20] Y. Irisa et al., ANS Topl. Mtg. on LWR Fuel Performance, Williamsburg, Virginia, U.S.A.(1988), p.102
[21] Y. Kondo, Proc. 6th Pacific Basin Nucl. Conf. Beijing, China,(1987), p.497
[22] T. Baba et al., ANS Topl. Mtg. on LWR Fuel Performance, Williamsburg, Virgina, U.S.A., (1988), p.142
[23] K. Yanagisawa et al., J. At. Energy. Soc. Japan, Vol.28, no.7, (1986), p.641, (in Japanese)
[24] J. B. Melehan et al., Proc. ANS. Topl. Mtg. on LWR Fuel Performance,Portland, Oregon, U.S.A.,(1979), p.29
[25] S. R. Pati et al., Ref.[20] p.204
[26] K. Yanagisawa, JAERI-M 89-095 (1989), (in Japanese)
[27] K. Yanagisawa, Nucl. Technol. Vol.73, (1986), p.361
[28] K. Yanagisawa et al: ANS/ENS Topl. Mtg. on LWR Fuel Performance, Avignon, France (1991), P850
[29] H. Mogard, D. A. Howl, M. Grounes, Ref.[11], p.232

Table I SUMMARY OF AS-FABRICATED CHARACTERISTICS OF THE OTFR

Fuel Type	MH	GK
<1> Fuel	Sintered UO_2	Sintered UO_2
Density (g/c.c.)	10.41 (95%TD)	10.41 (95%TD)
Pellet O.D.(mm)	9.29	9.29
Pellet length (mm)	15.2	15.2
Enrichment (w/o of ^{235}U)	2.6	3.4
<2> Cladding	Zircaloy-4	Zircaloy-4
Cladding O.D.(mm)	10.72	10.72
Wall thinkness (mm)	0.62	0.62
Heat treatment	Stress relieved	Stress relieved
<3> Assembly	14×14 PWR type	14×14 PWR type
Diametral gap (mm)	0.190	0.190
Active column length (m)	3.6	3.6
Filler gas (MPa)	Helium ca. 3.0	Helium ca. 3.0
Gas composition	97% He + 3% Air	97% He + 3% Air
Plenum volume (mℓ)	12.6	12.6

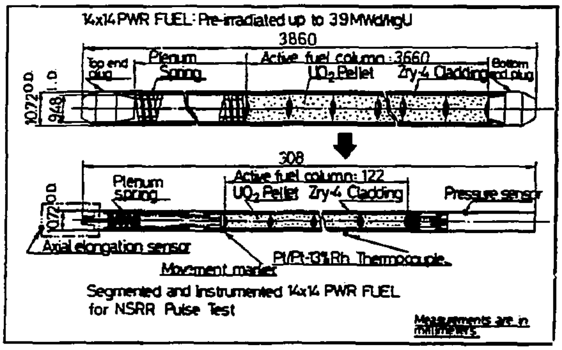

Fig. 1 CROSS SECTION OF OTFR (TOP) AND STFR(BOTTOM)

Table II IN-CORE PERFORMANCE OF THE PREIRRADIATED OTFR AND PULSE-IRRADIATED STFR

I	OTFR characteristics at the end of preirradiation stage		
	Fuel type	MH	GK
(1)	Attained burn-up (MWd/kgU)	38.9 (average)	42.1 (average)
(2)	Rod internal pressure at 0℃ (MPa)	4.66	4.34
	Gas composition (volumetric)	99.3%He + 0.1%Kr + 0.6%Xe	98.4%He + 0.1%Kr + 1.5%Xe
(3)	Plenum volume (mℓ)	15.0(c.f.as-fabricated:12.6)	16.9
(4)	Released amounts of		
	Kr (mℓ)	0.36	1.0
	Xe (mℓ)	3.79	10.8
(5)	FGR (%)	0.17 of Kr + Xe	0.42 of Kr + Xe
(6)	Average grain size (μm)	6.4	9.0

II	STFR characteristics before and after pulsing at NSRR				
	Items / Fuel rod	MH-1	MH-2	MH-3	GK-1
II-1	Before pulse irradiation				
(1)	Active column length (m)	0.122	0.121	0.121	0.122
(2)	Fill gas pressure at 0℃ (MPa)	4.23	4.23	4.63	4.22
(3)	Plenum volume (mℓ)	1.5	1.5	1.5	1.5
II-2	After pulse irradiation				
(1)	Deposited energy (cal/g-fuel)	60	68	83	112
(2)	Pressure increase during pulse (MPa)	1.36	1.31	–	1.93
(3)	Peak cladding surface temperature (℃)	95	82	184	308
	Coolant peak temperature (℃)	18	6	16	32
(4)	Failure(F)/No failure (NF)	NF	NF	NF	NF

Note : (–) Not measured

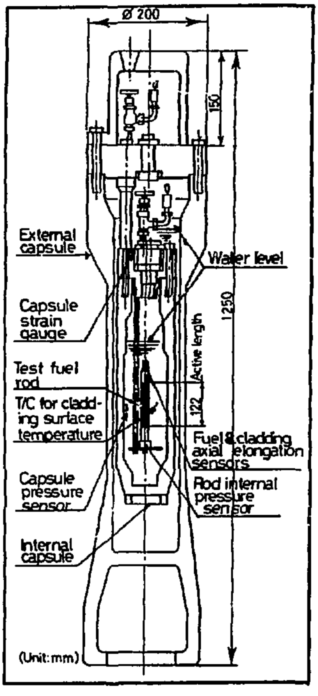

Fig. 2 IRRADIATION DOUBLE CAPSULE FOR PREIRRADIATED STFR

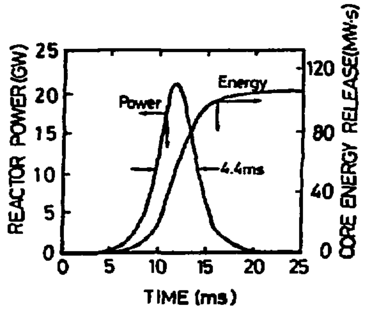

Fig. 3 TRANSIENT REACTOR POWER AND CORE ENERGY RELEASE ATTAINED IN PULSING OPERATION OF NSRR WITH $4.67 REACTIVITY INSERTION

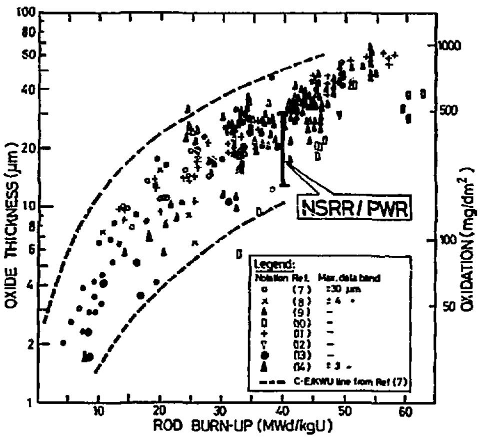

Fig. 4 OXIDE THICKNESS OF PWR FUELS AS A FUNCTION OF ROD AVERAGE BURN-UP

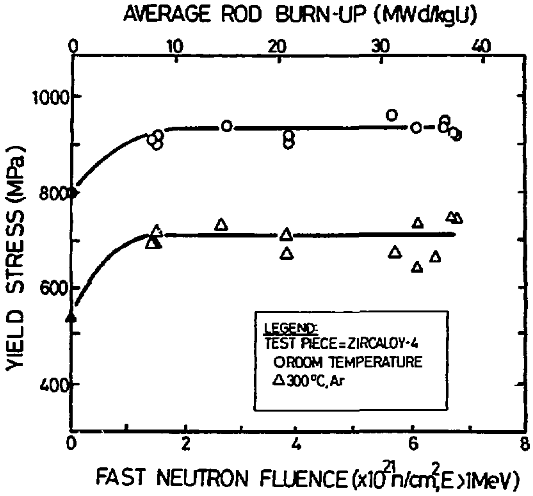

Fig. 5 YIELD STRESS OF ZIRCALOY-4 CLADDING AS A FUNCTION OF FAST NEUTRON FLUENCE AND BURN-UP

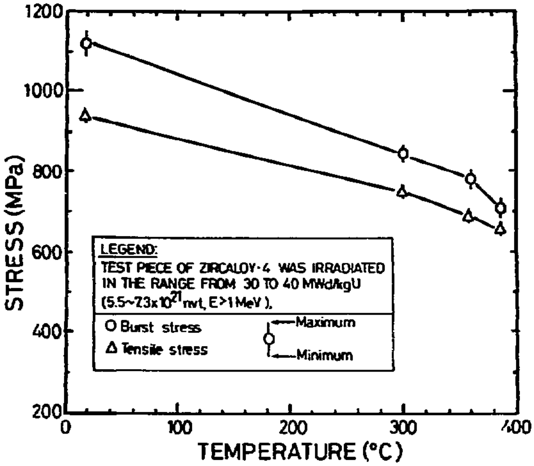

Fig. 6 BURST AND TENSILE STRESSES OF ZIRCALOY-4 CLADDING AS A FUNCTION OF TEMPERATURE

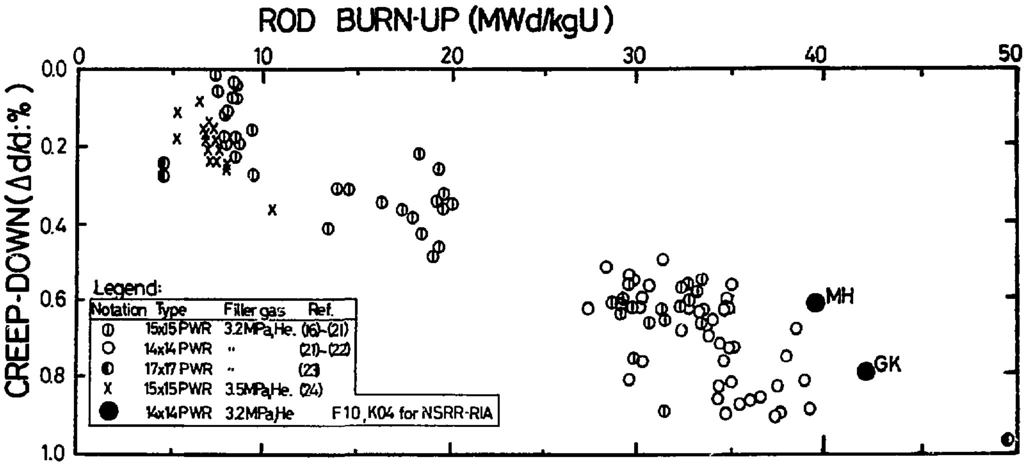

Fig. 7 CREEP-DOWN OF THE PWR FUELS AS A FUNCTION OF ROD AVERAGE BURN-UP

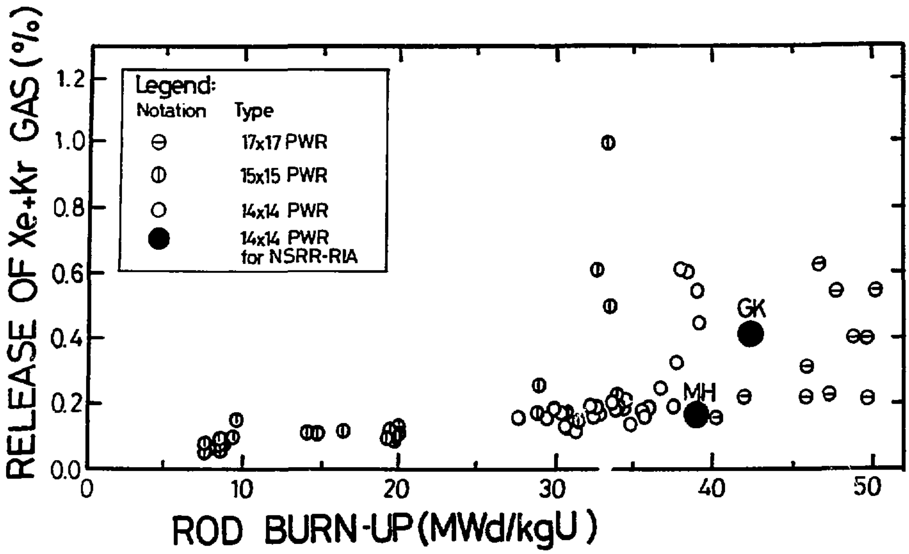

Fig. 8 FGR OF THE PWR FUELS AS A FUNCTION OF ROD AVERAGE BURN-UP

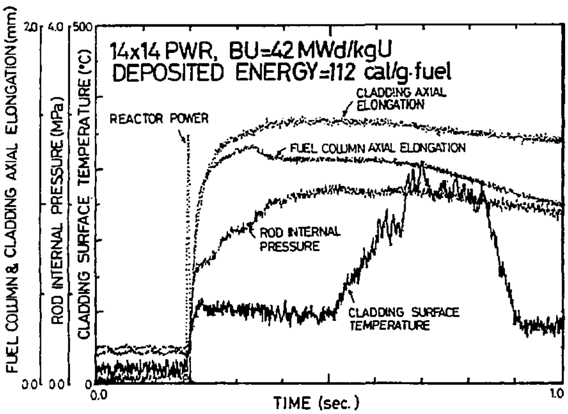

Fig. 9 IN-CORE BEHAVIOR OF PULSE IRRADIATED STFR: ATTAINED BURN-UP=41 MWd/kgU, DEPOSITED ENERGY=112 cal/g·fuel

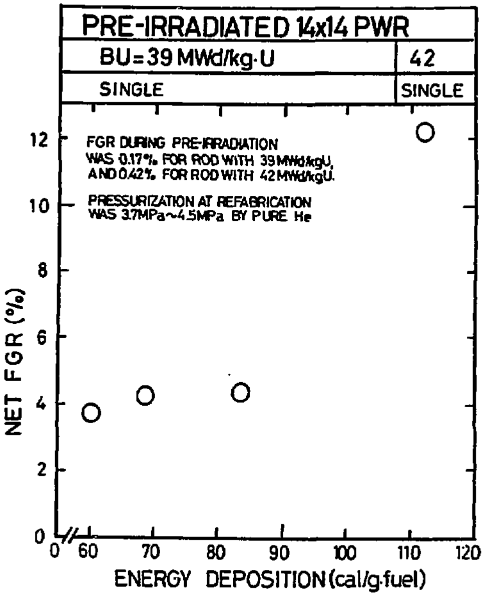

Fig. 10 NET FGR OF THE PWR FUEL AS A FUNCTION OF ENERGY DEPOSITION

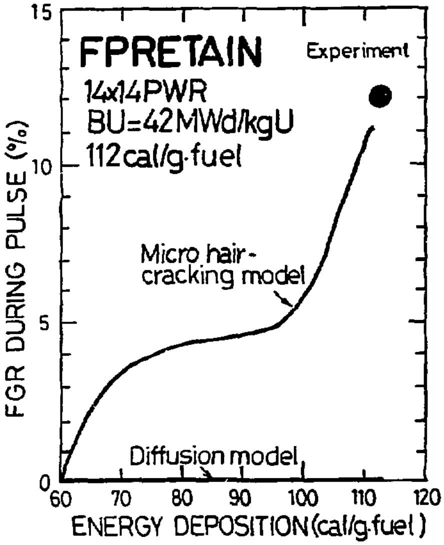

Fig. 11 CALCULATED FGR OF THE GK-1 FUEL ROD BY COMPUTER CODE FPRETAIN AT DEPOSITED ENERGY OF 112 cal/g·fuel

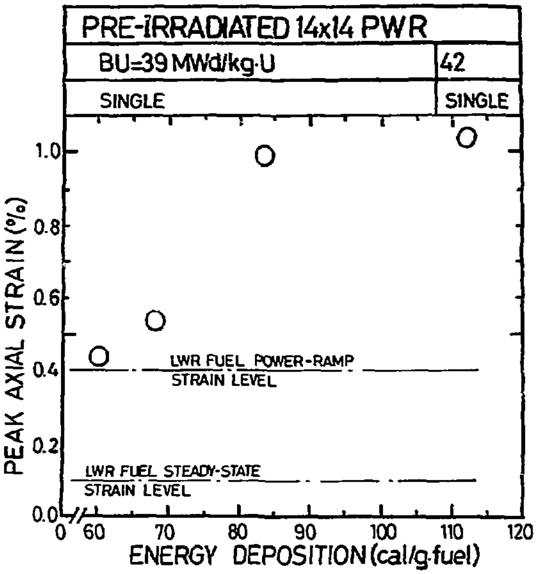

Fig. 12 PEAK AXIAL STRAIN OF THE PWR FUEL AS A FUNCTION OF ENERGY DEPOSITION

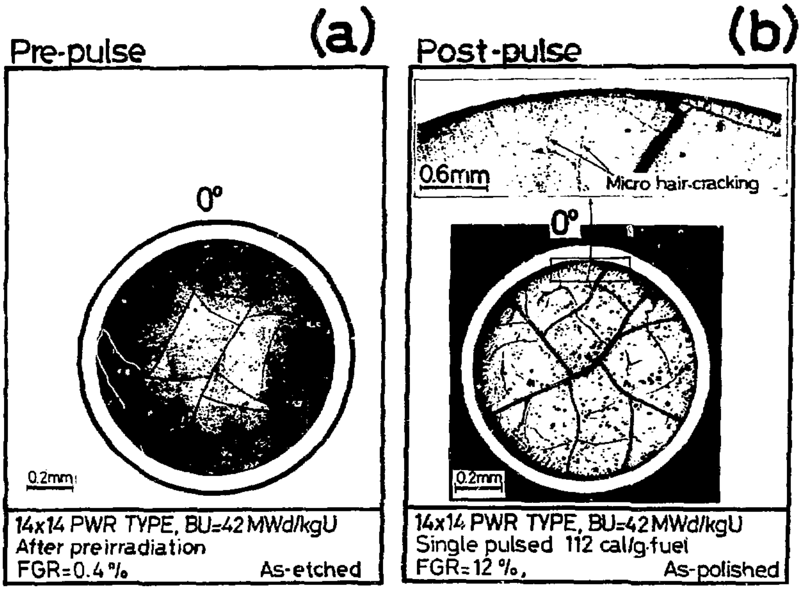

Fig. 13 FUEL MICROSTRUCTURE OBTAINED (a) FROM PRE-PULSE CONDITION AND (b) FROM POST-PULSE CONDITION, WHERE DEPOSITED ENERGY WAS 112 cal/g·fuel