Session I

PCI TESTING AND RESULTS

Chairman

M. ICHIKAWA

Secretary

N. KJAER-PEDERSEN

POWER RAMPING TEST IN THE JMTR FOR PCI STUDY OF WATER REACTOR FUEL

H. NAKATA, M. KANBARA, M. ICHIKAWA
Fuel Reliability Laboratory 1,
Japan Atomic Energy Research Institute,
Tokai-Mura, Ibaraki-ken, Japan

Abstract

Power ramping test is essential for PCI study of water reactor fuel. Boiling water capsules have been used for the tests in the JMTR. Heat generation of fuel rod in the capsule can be channed by the He-3 power control facility during reactor operation.

Four specially designed fuel rods have been ramped to about 41-43 kW/m; the two of them have small gap filled with iodine, the other two are equipped with centerline temperature thermocouple. Fuel rod elongation detector is equipped to each capsule. For the fuel rods with small gap, unique contraction followed by ordinary fuel relaxation behaviour was observed right after the fast ramping. None of them failed.

Future programme includes a series of tests of fuel rods irradiated in the high-pressure water loop at the JMTR and a verification test of remedy fuel which allows daily-load-following operation of BWRs.

1. Introduction

LWR fuel rod irradiation programme has been conducted in the Japan Materials Testing Reactor (JMTR) since 1977 under the national reactor safety study programme.

The JMTR was designed to provide suitable facilities for conducting nuclear irradiation experiments necessary for the research and development of power reactor in Japan [1]. The irradiation facilities being used for the LWR fuel programme are the two high-pressure water loops (OWL-1, OWL-2) and He-3 power-controlled boiling water capsules.

A series of fuel centerline temperature measurements was carried out in the loop (OWL-1) to study various parameters of fuel rods such as gap size (small, normal), filling gas (He, Xe) and cladding (standard, barrier) (2). In the other loop (OWL-2), a bundle of fuel rods has been irradiated for the future power ramping tests.

The boiling water capsule facility was installed in 1981 for a power ramping test of LWR fuel. The four specially designed fuel rods were already ramped in this facility. The two of them (R-2 and R-3) have small pellet-cladding gap filled with iodine and the other two (R-4 and R-5) are equipped with centerline temperature thermocouple.

This paper describes the power ramping facility and power ramping behaviours of these four fuel rods. Furthermore, the future test programme is dealt with.

2. Power ramping facility

2.1 JMTR

The JMTR is owned and operated by Japan Atomic Energy Research Institute (JAERI), and is located in the Oarai Establishment, one of three establishments of JAERI. The construction work of the reactor began in June 1965. The reactor attained the first criticality in March 1968 with power operation of 30 MW for irradiation experiment commencing in May 1970. The reactor power was increased to 50 MW in November 1971 with a minor change of reactor core configuration.

The reactor is cooled and moderated by light water, and reflected by beryllium. The core, 1560 mm in diameter and 750 mm in active height, is divided into a fuel region and a reflector region. The fuel region is a 7 x 5 array (540 mm x 386 mm) containing 22 U-Al alloy type fuel elements, 5 control rods with fuel follower each, and 8 experimental positions. These positions are used mainly for testing structual materials which require irradiation of fast neutron flux. The reflector region, because of presence of relatively constant and high thermal neutron flux during operation, is used mainly as a space for irradiation of fuel materials. The two water loops and the boiling water capsule have been installed in the reflector region. Figure 1 shows a reactor core configuration.

The reactor vessel containing the core is a stainless steel tank of 3 m in diameter and 9.5 m in height as shown in Fig. 2. The top head flanged to the shell has openings for access to the core and many nozzles for experiments. The bottom head provides the holes for through-loops as well as for the control rods. The vessel is situated in the reactor pool of 6 m in diameter and 13.7 m in depth. Connecting the reactor pool to the hot laboratory adjacent to the reactor building, there is a canal of 3 m in width and 6 m in depth.

The hot laboratory of the JMTR has been in operation since 1971. The laboratory has a capability of performing a wide variety of works such as dismantling of irradiated capsule and loop assembly, and conducting of post-irradiation examination of fuels and post-irradiation test of structual materials. There are three kinds of cells in the laboratory. First, the concrete cells chiefly for dismantling of capsule and examination of fuel. Second, the lead cells for material testing. A scanning electron micro-analyzer (SEM) with shielded beam tube has been recently installed in the laboratory. Third, the iron cells, which was completed in June 1982, have been equipped with the PCI-SCC test machine for LWR fuel cladding. The hot laboratory can accept materials irradiated not only in the JMTR but also in other reactors.

Design characteristics of the reactor are listed in Table 1.

2.2 Boiling water capsule

The boiling water capsule has been recently developed for a power ramping test of fuel rod under a LWR condition.

A fuel rod to be tested is placed in capsule filled with water pressurized at 7.25 MPa as shown in Fig. 3. A nuclear heat produced in the fuel rod is dissipated through the pressurized water and capsule pressure tube, and is finally removed by the reactor cooling water. A surface temperature of the fuel rod is almost constant over a wide range of linear heat rate because of a subcooled boiling taking place at the surface (see Fig. 4). The fresh demineralized water is continuously supplied to the capsule at a very small flow rate so as to maintain the water quality. The draining water from the capsule is monitored for fission products to detect fuel rod failure. Figure 5 shows a schematic configuration of the capsule. Some self-powered neutron detectors (SPND) and an LVDT type fuel rod elongation detector are equipped in the capsule for fuel rod power determination and elongation measurement.

The capsule is loaded into a "gas screen" of the He-3 power control facility, which has been installed in the reflector region of the reactor core. The "gas screen" has an annular gap between two concentric tubes filled with He-3. Power ramping of a fuel rod is performed by decreasing a He-3 gas pressure in the screen. For a 2.8 % enrichment BWR size fuel rod, the maximum ramped power is about 50 kW/m. Figure 6 shows a schematic configuration of the facility.

3. Power ramping tests of the small gap rods filled with iodine (R-2 and R-3)

3.1 Preparation of the fuel rods

The fuel rods are intended to fail due to PCI during power ramping test. Main features of the fuel rods are of small gap, flat-end pellet and iodine filled in the gap as corrosive agent as in an out-of-pile SCC experiment.

The fuel pellets were prepared by an ordinary sintering technique, and are resistant against densification. A length to diameter ratio is about 1.5. Both ends are flatted so that a strong PCMI is expected. The outer diameter of the pellet was ground to have a diametral pellet-cladding gap of 80 μm. The Zry-2 claddings were stress-relieved at 788 K after rolling. The nominal outer diameter and thickness are 12.26 mm and 0.86 mm, respectively. A quartz ampoule containing pure iodine of 1 gr. was inserted in the gas plenum of each rod and was crashed with a steel ball after welding. Finally, the rods were heated up to 573 K to vaporize iodine and then the fuel stack portion was cooled down faster than the plenum portion so that the pellet-cladding gap was filled with iodine. Pure helium gas was filled at 0.1 MPa before welding.

A schematic drawing and design characteristics of the fuel rods are given in Fig. 7 and Table 2, respectively.

3.2 Irradiation

The fuel rods were pre-irradiated at a low linear heat rate (approximately 20 kW/m) for 263 hours (R-2) or 467 hours (R-3) and then power ramped to high linear heat rates (43.3 kW/m for R-2 and 41.4 kW/m for R-3) in two minutes. The high power levels were kept for about 6 hours (R-2) or 12 hours (R-3), but no indication of fuel failure was found in the radioactivity of the capsule water or in the fuel rod elongation behaviour. A water pressure in the capsules was maintained within 7.25 ± 0.2 MPa during the irradiations.

Figure 8 shows the power histories of the fuel rods and the trends of elongation detector signal during the irradiations. A linear heat rate was determined with a SPND signal calibrated by the centerline temperature measurements (see 4.). The maximum error of power determination has been estimated to be 1.4 kW/m. The elongation signals have been corrected for a capsule thermal expansion. Figure 9 shows the change of elongation with time during the power ramping. A large and sudden contraction followed by a ordinary fuel relaxation was observed in each rod right after the power ramping. Figure 10 shows the relationship between a elongation and linear heat rate during the power ramping.

3.3 Post-irradiation examination

The scheduled non-destructive examinations on these fuel rods have been carried out except eddy-current test (EC). Neither defect nor perforation was found in the fuel claddings during the visual inspection and the leak test. The X-ray radiography shows that fuel stacks are somewhat longer after they were irradiated than before (0.68 mm for R-2 and 1.02 mm for R-3). The fuel claddings, however, do not show distinct residual elongation (<0.1 mm). Figure 11 shows the results of axial total-gamma scanning. Dips in the gamma ray intensity are clearly visible at pellet interfaces of the fuel rod R-3, which was irradiated for longer time and was cooled for shorter time than R-2. Some dips coincide with the pellet-pellet gaps observed in the X-ray radiography.

Figure 12 shows the results of profilometries in three diametrical directions. Ridges are observed at most pellet-interfaces in both fuel rods.

At the lower portion of R-2, there are large ridges at a interval equal to a two-pellet length, while small ridges or secondary ridges are observed between large ridges at the lower portion of R-3. The maximum ridge height is 22 μm for R-2 and 18 μm for R-3. Ridge heights measured in the three direction differ each other at most of ridges. The maximum difference or ovality is 20 μm for R-2 and 15 μm for R-3.

Destructive examination will includes gas analysis, ceramography of fuel pellet and metallography of cladding. Fractography by SEM will be carried out, if an incipient crack is observed by EC or visual inspection of the cladding inner surface.

3.4 Discussion

The fuel rods did not fail, however the elongation behaviours during the irradiation and the results of post-irradiation examinations are very interesting, because of 1) large elongation during the irradiation, 2) sudden contraction right after the power ramping, 3) large ridges along the fuel stacks, and 4) no distinct residual elongation.

The elongation calculated with the FEMAXI code [3] for the power ramping is shown also in Figure 10. Iodine is not taken into account in the FEMAXI calculation. It has not yet been verified whether a discrepancy between measured and calculated elongation is caused by iodine. Destructive examination and further analysis will clear the discrepancy and other incomprehensive behaviours.

4. Fuel centerline temperature measurements (R-4 and R-5)

During reactor start-up, power ramping and reactor scram, the centerline temperatures were measured on fuel rods containing chamfered pellets with diametral pellet-cladding gap of 100 μm (R-4) and 200 μm (R-5). W-Re type thermocouple was inserted into the central hole of fuel pellet from the bottom end plug of the fuel rods. Pure helium gas was filled at 0.1 MPa, and no iodine was introduced. A schematic drawing and design characteristics of the fuel rods are given in Figure 7 and Table 2, respectively.

Figure 13 shows the centerline temperatures measured during the first and second reactor start-up. The temperatures during the second start-up are higher than those during the first start-up. Similar results have been obtained in the OWL-1 centerline temperature measurements, where the FEMAXI prediction agrees better with the second start-up than with the first start-up (see Fig. 14). These repeated observations of temperature increase in the beginning of irradiation should be studied from the standpoint of pellet cracking and relocation mechanisms as well as pellet densification. The results of R-4 and R-5 have been used to calibrate the SPND signal for a power determination togather with the results of OWL-1 experiment and FEMAXI calculation.

Figure 15 shows a response of the centerline temperature of R-4 obtained by reducing the He-3 pressure in the same fashion as that in the power ramping tests mentioned above. This gives a useful information on thermal response of fuel rod during a power ramping test in the JMTR. Figure 15 also shows a transient response of R-4 to a reactor scram. This kind of data is valuable for an evaluation of fuel thermal behaviour under transient condition and will be analyzed using a transient thermal code.

5. Future programme

The power ramping facility is under modification. When completed, a capsule is cooled by water independent of the reactor cooling water and can be loaded into or unloaded from the reactor during reactor operation according to a test programme, and a He-3 pressure in the gas screen can be changed automatically in a programmed mode. In the hot laboratory, the in-cell apparatus has been installed for charging pre-irradiated fuel rod into a capsule.

The future test programme will includes power ramping tests of three kinds of fuel rods; 1) fresh fuels, 2) fuel rods irradiated in the water loop (OWL-2), and 3) remedy fuels.

A fuel rod similar to R-2 or R-3 will be ramped. In order to cause PCI failure in this rod, Xe gas is filled to obtain a higher fuel temperature and longitudinal scratches of approximately 80 μm in depth are produced on the inner surface of cladding to initiate cracking. Two more centerline temperature measurements will be conducted by using two fuel rods, one with circular cladding and the other with oval cladding, the insides of which are filled with Xe, in order to simulate high burn-up fuel.

The fuel rods irradiated in the water loop, OWL-2, up to 7 GWd/t UO_2 under the BWR condition will be ramped in 1984 and 1985. Detail of the power ramping test has not yet been established.

Power ramping tests of some candidate remedy fuels, which allow daily-load-following operation of BWRs, are being planned under the Japanese High Duty Fuel Development Programme (4). Base-irradiation of the segmented fuel elements to be ramped in the JMTR will start in the next spring in a commercial BWR power plant. A further facility modification is necessary to conduct these tests which require a power ramping up to 70 kW/m.

6. Conclusions

1) The power ramping test facility in the JMTR is capable of ramping a BWR size fuel rod of 2.8 % enrichment to 50 kW/m in 2 min. under the BWR condition.

2) Power ramping tests of the fuel rods with small gap filled with iodine have given interesting data on mechanistic study of PCI.

3) The centerline temperature measurements have given a useful information on thermal behaviour of fuel under transient condition.

4) The future programme includes both the ramping tests of fuel rods irradiated in the water loop and the ramping tests of remedy fuels irradiated in a commercial BWR plant.

Acknowledgments

The authers wish to thank Dr. Morishima of JAERI for his direction to the programme, Prof. Okubo of Sophia University for his encouragement and guidance to the tests, and all staff in JAERI who are working on the power ramping test programme in the JMTR.

References

(1) H. Nakata, I. Tanaka, Y. Ichihashi, H. Itami, and H. Itoh, "Irradiation Facilities in JMTR", JAERI-M 82-119, September 1982.

(2) H. Ando, K. Matsubara, H. Sakai and H. Kawamura, "A Final Report on the First Test Assembly (77LF-33J) for Fuel Center Temperature Experiment", JAERI-M 83-003, February 1983 (in Japanese).

(3) M. Ichikawa, T. Okubo, Y. Iwano, K. Ito, K. Kashima, H. Saito and T. Nakajima, "FEMAXI-III; an axisymmetric finite element computer code for the analysis of fuel rod performance", IAEA Specialists' Meeting, Blackpool, UK, March 1980.

(4) Y. Mishima, "Japanese Programme on the Development of High Duty Fuel and Related Power Ramping Tests", IAEA Specialists' Meeting, Petten, Netherlands, September 1982.

Table 1 Characteristics of the JMTR

1. Power	50MW thermal		
2. Moderator / Coolant	Light Water (1.5MPa, 323K)		
3. Reflector	Beryllium		
4. Fuel ; Material Enrichment Loading Type	U-Al Alloy 93% 6.6kg of U-235 (in Total) Modified ETR		
5. Control Rod	5 Hf Rods with Fuel Follower each		
6. Max. Neutron Flux ($\times 10^{18}$ $n/m^2 \cdot s$)		Fast (>1MeV)	Thermal
Fuel Region		4	4
Reflector Region		1	4

Table 2 Design Characteristics of the fuel rods

	R-2	R-3	R-4	R-5
1. Pellet				
material	UO_2			
enrichment	2.8 %			
diameter	10.46 mm		10.70 mm	10.60 mm
height	16.0 mm		11.0 mm	
end shape	flat		chamfered	
density	95 % T.D.			
2. Clad				
material	Zircaloy-2			
heat-treatment	stress-relieved		re-crystallized	
outer-diameter	12.26 mm		12.52 mm	
thickness	0.86 mm			
P/C gap	80 μm		100 μm	200 μm
3. Pin				
stack-length	400 mm			
filling gas	helium			
pressure	0.1 MPa			
4. Remarks	1 gr. of nat. iodine is filled.		Diameter of center hole for thermocouple is 2.3 mm.	

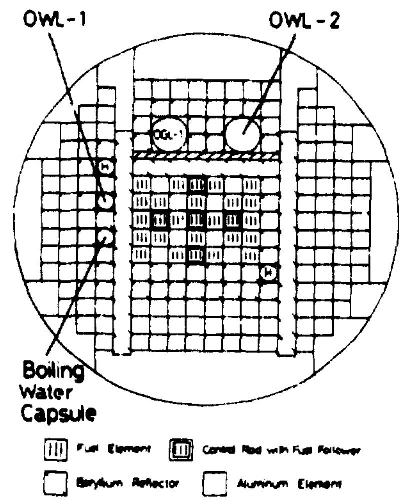

Fig. 1 Irradiation Facilities in the JMTR Core

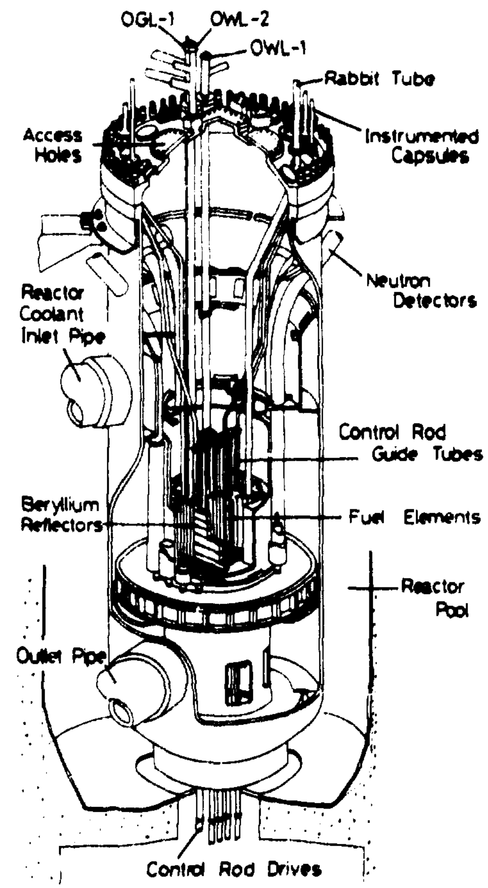

Fig. 2 Pressure Vessel of the JMTR

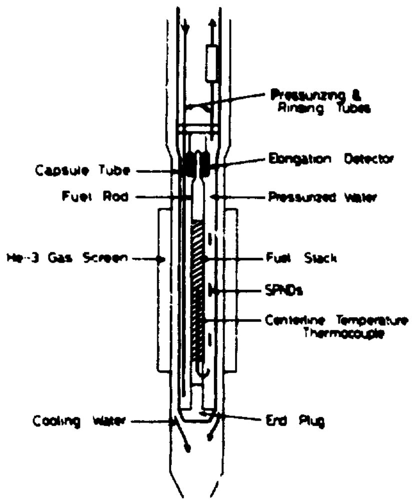

Fig. 3 Fuel Rod in the Boiling Water Capsule

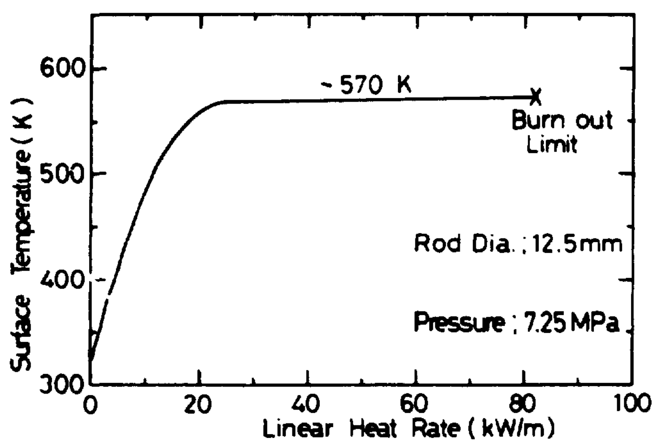

Fig. 4 Surface Temperature V.S. Linear Heat Rate of Fuel Rod in the Boiling Water Capsule

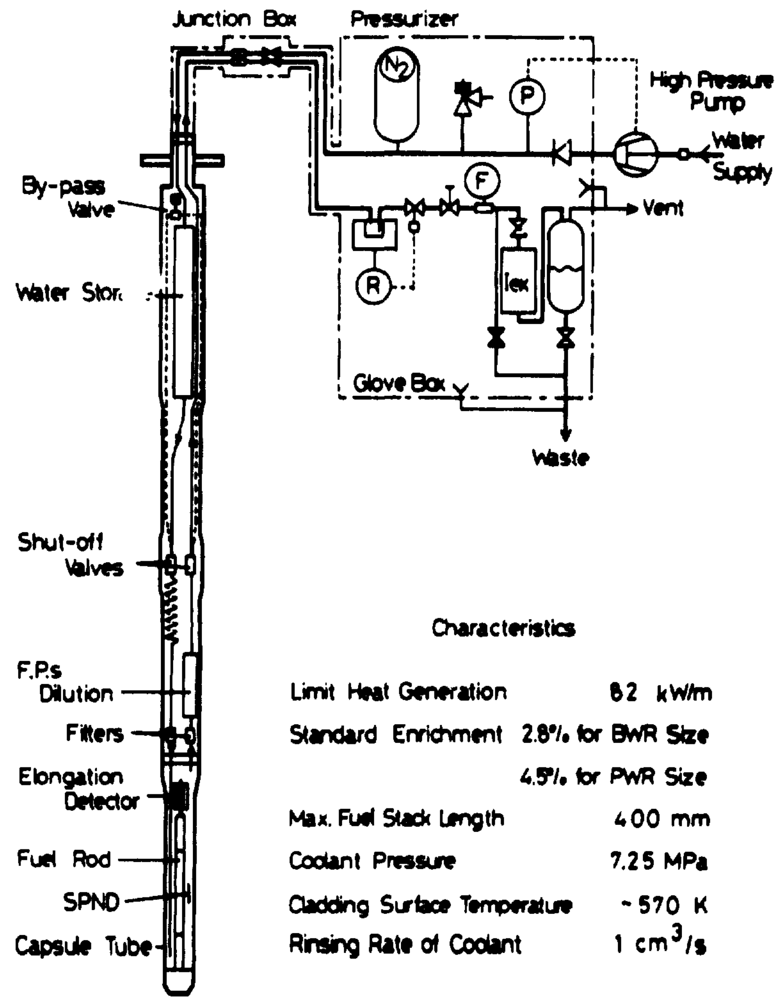

Fig. 5 Schematic Configuration of the Boiling Water Capsule

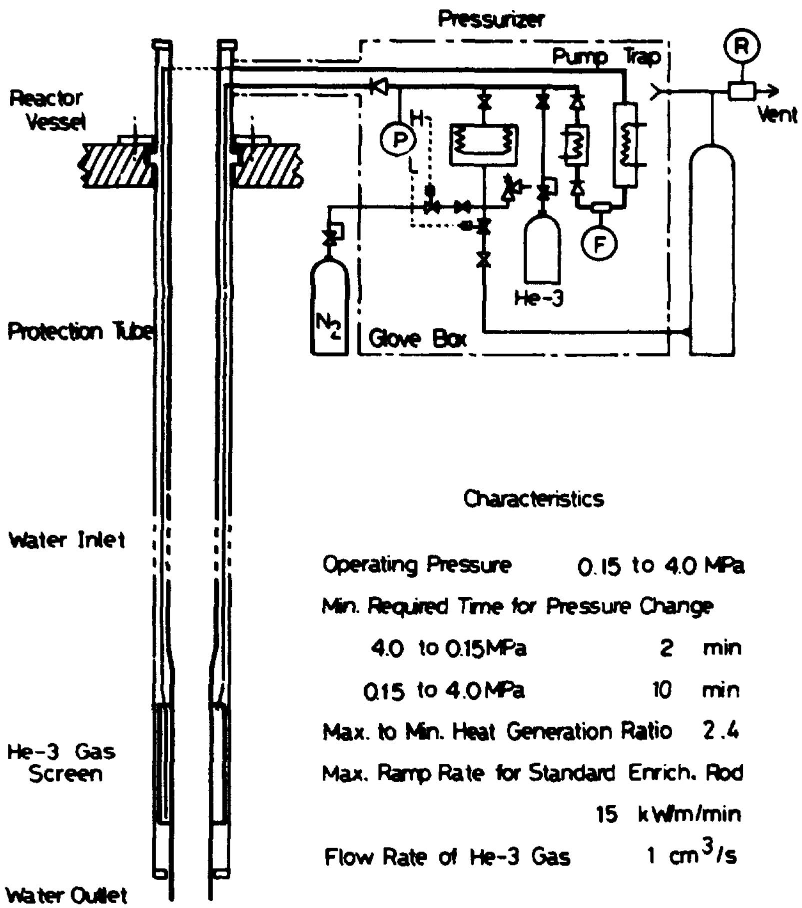

Fig. 6 Schematic Configuration of the He-3 Power Control Facility

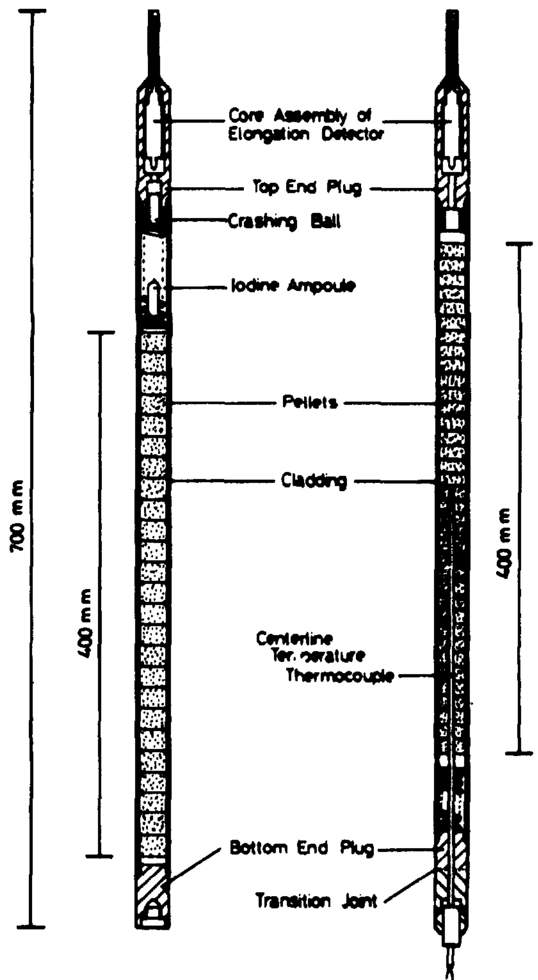

Fig. 7 Schematic Drawing of the Fuel Rod

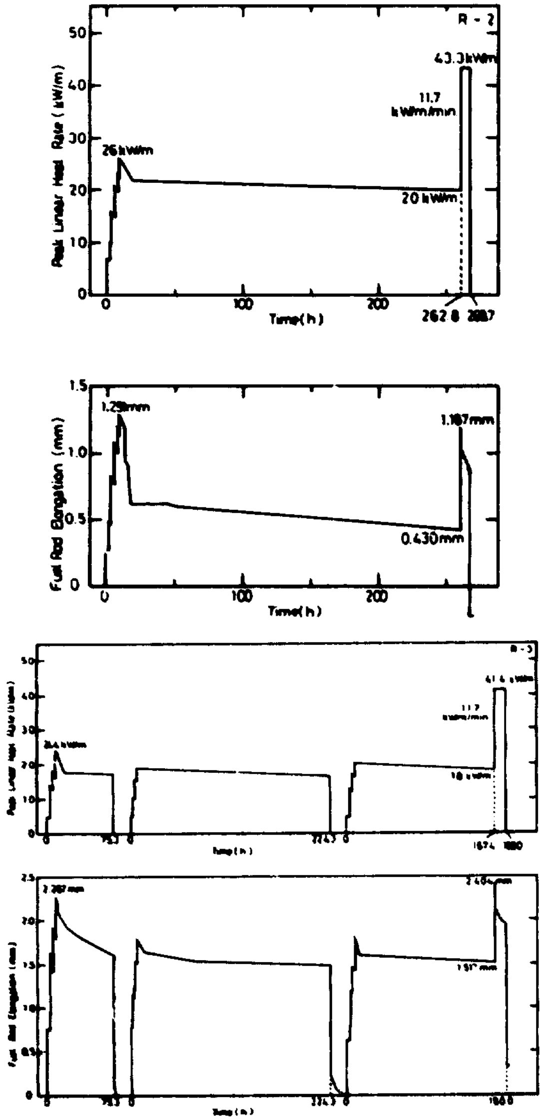

Fig. 8 Power-histories and Elongations of the Fuel Rods (R-2 and R-3)

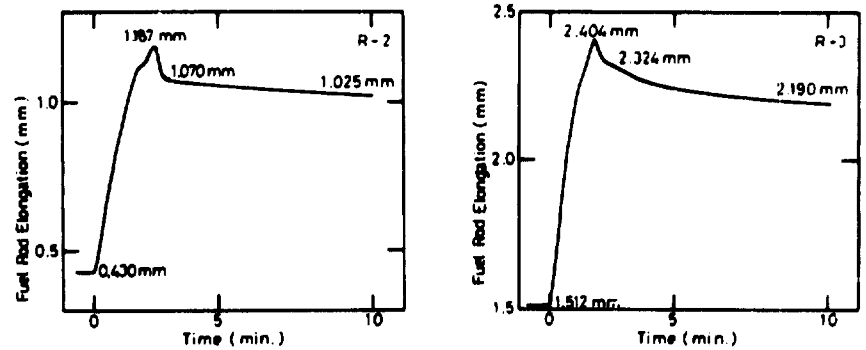

Fig. 9 Elongation Behaviours of the Fuel Rods (R-2 and R-3) during the Power Ramping

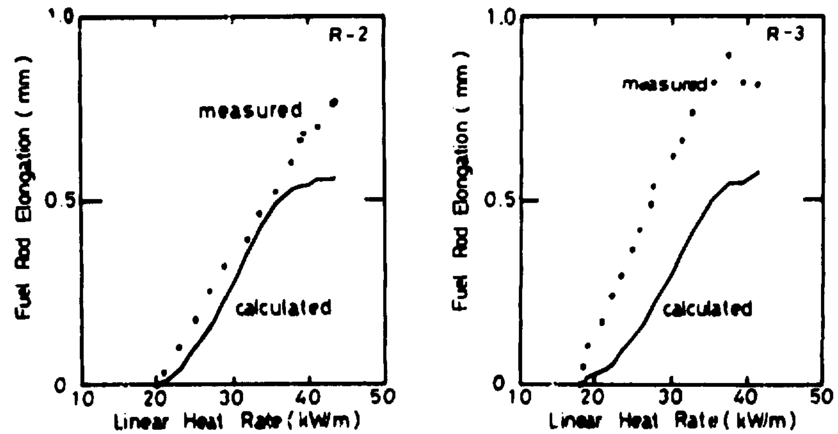

Fig. 10 Elongation V.S. Linear Heat Rate of the Fuel Rods (R-2 and R-3) during the Power Ramping

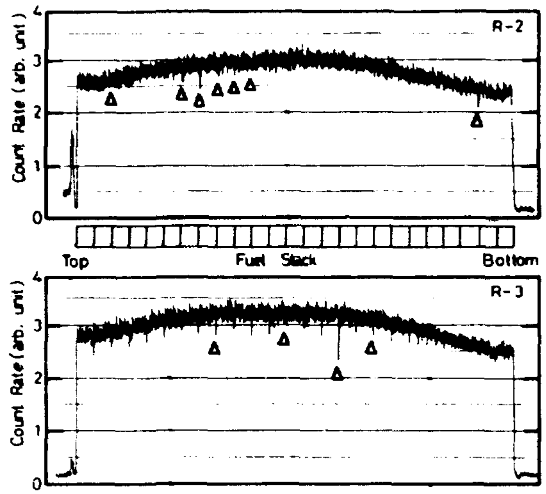

Fig. 11 Axial Total Gamma Distributions of the Fuel Rods (R-2 and R-3) (Δ indicates pellet-pellet gap observed in the X-ray Radiography)

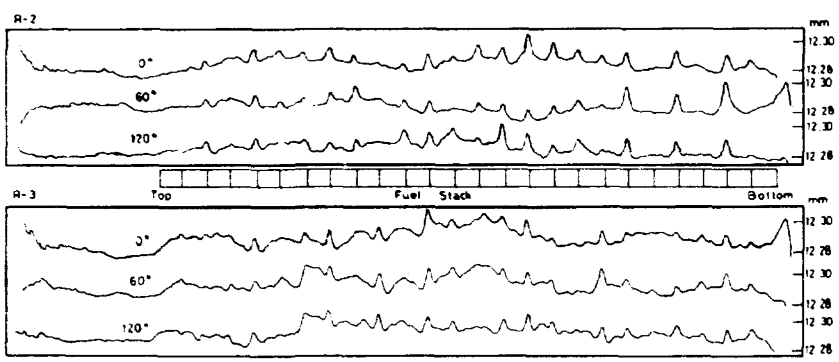

Fig. 12 Results of Profilometries of the Fuel Rods (R-2 and R-3)

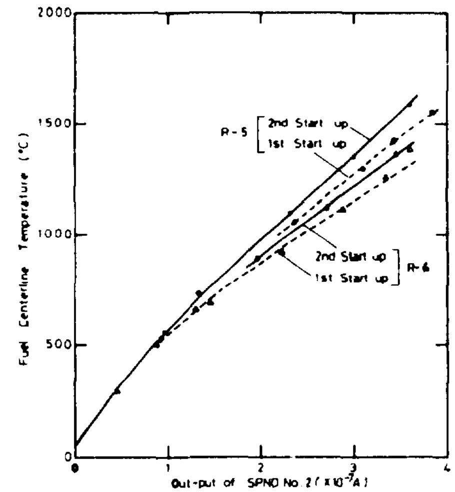

Fig. 13 Centerline Temperatures of the Fuel Rods (R-2 and R-3) during the 1st and 2nd Start-up

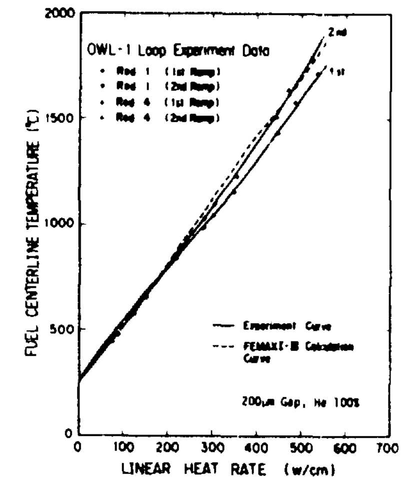

Fig. 14 Centerline Temperature Measurements in OWL-1

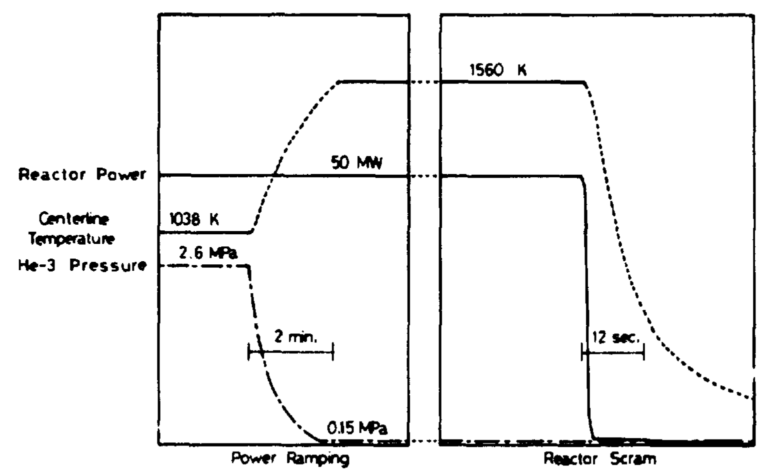

Fig. 15 Transient Responses of Centerline Temperature of the Fuel Rod R-4 to Power Ramping and Reactor Scram