Failure and its mechanism of LWR and research reactor fuels

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Abstract: A failure of LWR fuel under a normal operating condition up to the burn-up of 20MWd/kgU was PCI-SCC. A failure of LWR fuel under RIA occurred at 260cal/g・fuel for the fresh fuel and 118cal/g・fuel for pre-irradiated PWR fuel to 42MWd/kgU. The mechanism was the melt-brittle for the former and strong PCI combined with transient FGR for the latter. The silicide fuel failed by the through-plate cracking, where the tensile thermal stress governed by temperature drop (>94deg.C) and time to quench (<0.13s) is the principal mechanism.

Keyword: LWR fuel; silicide fuel; PCI failure; transient FGR; melt-brittle failure; through-plate cracking; RIA

1 Introduction

In Japan, the gross electricity produced by 54 light water reactors (LWRs) was about 287TWh in 2005 (35% to the total) and the economic scale of the electricity produced by nuclear was 42,682 million dollars (M$) at the demand end. The nuclear energy contributed to the welfare of 127million Japanese people now [1]. The high cost performance of LWR was principally attributed to the safety usage of LWR fuel rods, which was attained from a long term research and development (R&D), carried out in the Japan Atomic Energy Research Institute (JAERI, now JAEA).

In this report, the author describes the new findings obtained from the study. They are addressed to the success of on power monitoring of PCI failure, to finding the PCI threshold by the computer code, to establishing the PCI failure threshold for Japanese LWR fuels by the LWR loop. Second, under the reactivity initiated accident (RIA), they are addressed to the success of finding the failure of the pre-irradiated Japanese PWR fuel, to establishing the preliminary failure threshold (mechanism is not yet), and to finding the mechanism of the transient fission gas release (FGR). Lastly, the author's work addressed to find the failure threshold and its mechanism of the silicide fuels for research reactors.

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2 Experiments

In this chapter, the author describes a type of the research reactor and a kind of tests performed. Lastly, the author introduces the fuel modeling code developed to clarify the fuel failure mechanism.

2.1 HBWR

One of the typical fuel failure occurred in the LWR was a failure initiated from a pellet-cladding interaction induced stress corrosion cracking (hereinafter denoted as the PCI failure). To study this, the author carried out the in-core experiment at the Halden Boiling Water Reactor (HBWR), located in Halden, Norway. This test reactor was run by the IFE (Institute for Energy technology). JAERI as a representative of Japan joined to the Halden Reactor Project since 1967 for the nuclear fuel irradiation test. The outline of the HBWR is described elsewhere [2].

As shown in Table 1**,** the reactor had a unique feature having the coolant (D_2O) temperature of 240deg.C and the coolant pressure of 3.4MPa (hereinafter denoted as the HBWR condition).

Table 1 Summary of the irradiations in HBWR

Test reactor	HBWR		
Irradiation condition	HBWR	BWR loop	PWR loop
Fuel type	8x8 BWR,	8x8RJ BWR	17x17 PWR
Fuel (enrichment)	UO2 (10wt%)	UO2 (10wt%)	UO2 (10wt%)
Cladding	Zircaloy-2	Zircaloy-2	Zircaloy-4
Method	Power increase	Power ramping	
Purpose	To know the PCI failure threshold		
Base irradiation rig	IFA-515	IFA-523	IFA-524
Burn-up(MWd/kgU)	18	0, 6, 15, 20	0, 5, 10, 20
Ramp rig	IFA-508	IFA-520	IFA-525
Test specimens	8	10	10

Note. 8x8RJ BWR means 8x8 type remedy BWR fuel developed in Japan.

2.1.1 Experiment under HBWR condition

For the PCI failure, the typical 8x8 type BWR fuel (rod 32) as shown in Table 2 was designed by JAERI and fabricated by the Nuclear Fuel Industries Ltd., in Japan. The test specimen was base irradiated in the instrumented fuel assembly IFA-515 and transported to the IFA-508 to measure the rod diameter during the power increase (0.001kW/ms), where a rod linear heat rating was increased from 40kW/m to 48.7kW/m at the burn-up of 18.7MWd/kgU.

Table 2 Summary of the fuel rod physical parameters used in power increase experiment in HBWR

Fuel	UO2
Density （g/cc）	10.44（95.3%TD）
Pellet O.D.（mm）	11.31
Pellet Length （mm）	15.19
Enrichment （weight %）	10.5
Grain size diameter （micron）	11
Additive	none
Dishing （both ends）	0.4mm deep and 7mm wide
Cladding	Zircaloy-2
Cladding O.D.（mm）	12.20
Wall thickness （mm）	0.395
Heat treatment	Fully annealed, autoclaved both sides
Assembly	
Diametral gap （mm）	0.100
Total column length（mm）	420
Enriched column length （mm）	390
Filler gas （MPa）	0.1 pure He
Attained peak burn-up（MWd/kgU）	18.7

2.1.2 Experiment under simulated LWR condition

PCI failure was the typical result of power ramp test but not reported by the manufacturers until the first report by H. Mogard in 1971[3]. Since the first observation of the PCI failure, the Halden project started the OVERPOWER RAMP project, where more than 60 LWR fuel rods with different burn-ups were power ramped. The most prominent finding from the project was that PCI failure threshold should decrease from 60kW/m to 50kW/m with increasing burn-up from 10MWd/kgU to 20MWd/kgU. Because a test fuel rod irradiated under the HBWR condition had a relatively low coolant temperature of 240deg.C (320deg.C for LWR) and a relatively low coolant pressure of 3.4MPa (7MPa for BWR, 15MPa for PWR). It was also revealed that PCI failure threshold observed under HBWR condition might be different from that of LWR condition. This was mainly attributed to the HBWR condition, in which the irradiation damages given to the fuel were less than those of the fuel irradiated in LWR.

After OVERPOWER RAMP project, JAERI decided to install the LWR loop and ramp rig into the HBWR for the ramping of LWR fuels. Such renewed environment was denoted here as the simulated LWR condition. As already shown in Table 1, base irradiated LWR fuel rods were power ramped from 25-30kW/m to 50kW/m at the burn-up up to 20MWd/kgU. The characteristics of those LWR fuel rods were described elsewhere [4]. To study the fuel behavior, the data obtained from in-core instrumentations such as a Pt/Pt-13%Rh thermocouple (T/C) and a diameter gauge were used. Note that T/C was inserted to the hollow fuel pellet (1.8mm O.D.) in the fuel top to measure the fuel centerline temperature directly. Diameter gauge was invented by the Halden Project to measure the change of the O.D. of the fuel rod, where the principal of pick-up coil is used [5].

2.2 NSRR

The experiment related to the reactivity initiated accident (RIA) was carried out at the Nuclear Safety Research Reactor (NSRR) belonged to JAERI. It was built in 1975 to test the fuel accident behavior under RIA. The results were reflected to the Japanese Licensing Guideline for LWRs [6]. RIA test on silicide fuel was done under the request of the Nuclear Safety Commission of Japan prior to the core conversion in the Japan Materials Testing Reactor (JMTR) and the Japan Research Reactor No.3 (JRR-3).

As shown in Table 3, two types of the test specimens were used; one was the LWR fuel and the other the silicide fuel.

Table 3 Summary of the irradiations in NSRR

Test reactor	NSRR		
Irradiation condition	Capsule with stagnant H2O, 0.1MPa pressure		
Fuel type	14x14 PWR	14x14&17x17PWR	Sillicide fuel
Fuel (enrichment)	UO2 (10wt%)	UO2 (<5wt%)	U3Si2 (<20wt%)
Cladding	Zircaloy-4	Zircaloy-4	Al-3wt%Mg
Method	Simulated reactivity initiated accident (RIA)		
Purpose	To know the failure threshold and its mechanism		
Base irradiation	None	Commercial LWR	None
Burn-up(MWd/kgU)	0	42	0
Deposited energy	>240 cal/g･fuel	up to 120cal/g･fuel	up to 164ca/g･fuelplate
Test specimens	>2000	>9	19

2.2.1 LWR fuel

The original NSRR standard fuel shown in the second column of Table 3 had the same dimension used as the 14x14 PWR type fuel, except for a rod internal pressure and the length of active fuel column.

The NSRR standard fuel had no pressurization but the conventional 14x14 PWR was pressurized to 3MPa with a pure helium gas. As shown in Table 4, the NSRR standard fuel had an active fuel column by 0.253m due to the limited height of the NSRR core (0.136m in maximum). All data obtained from the NSRR standard fuel had no accumulated burn-up, that is, the fresh or un-irradiated fuel.

The NSRR was newly licensed around 1989 to grade up the pulse mode from a single to a multi and to install the hot cell to handle the pre-irradiated LWR fuels. Pre-irradiated or burn-up accumulated LWR fuel has to cut one-24th the original length (3.6m). Namely, the original PWR fuel was segmented at the Hot Laboratory to have the active column length by 0.122m. The segmentation, of course, required the new development of technology, for example, the welding of end plug, the refilling of the mixed gas and the welding of T/C at the corroded cladding external surface. After transportation from the Hot Laboratory to NSRR, the test specimen was assembled in the supporting jig with electric cables and loaded into double shielded irradiation capsule. All the irradiation tests were conducted in stagnant water at room temperature at about 20 deg. C and one atmospheric pressure. inside the sealed irradiation capsule.

Table 4 Characteristic of the NSRR standard fuel for RIA test

Fuel	UO2	
Density (g/cc)	10.41	(95%TD)
Pellet O.D. (mm)	9.29	
Pellet Length (mm)	10.0	
Enrichment (weight %)	10	
Additive	none	
Shape	Chamfer	
Cladding	Zircaloy-4	
Cladding O.D. (mm)	10.72	
Wall thickness (mm)	0.62	
Heat treatment	Stress relieved	
Assembly		
Diametral gap (mm)	0.190	
Total column length (mm)	253	
Enriched column length (mm)	135	
Filler gas (MPa)	0.1 pure He	
Attained peak burn-up (MWd/kgU)	0	

2.2.2 Research reactor fuel

The silicide fuel used were designed by JAERI and fabricated by two foreign vendors; CERCA in Romans, France and B&W in Lynchburg Virginia., the U. S. It consists of the fuel core（25×70×0.51mm）sandwiched by Al-3wt%Mg based alloy cladding (35×130×0.38mm), hereinafter abbreviated as “Al cladding”. For the reactor core conversion in the JMTR and JRR-3, the low enriched uranium (LEU<20w/oU-235), hence the 4.8g/cc silicide fuels substituted for the 2.2g/cc aluminide fuels. According to the RERTR (Reduced Enrichment for Research and Test Reactors) Project mainly promoted by the United States and EU, the change from the high enriched fuel (>20w/oU-235) to the low enriched fuel (<20w/oU-235) was recommended at that time. To make the core conversion smoothly, the silicide fuel instead of the aluminide fuel was used worldwide. The in-core instrumentation was T/C (0.2mm diameter), which was directly spot welded to the Al cladding. Assembling and pulse irradiation with an atmospheric capsule was the same as those of NSRR standard fuel.

2.3 Fuel behaviour code

One of main objective of the experiments carried out in HBWR and NSRR was to provide the data for the verification of modeling code known as the FEMAXI-III [4] for the former and FPRETAIN [7] for the latter. Note that FEMAXI-III was the two dimensional, axi-symmetric, finite element code to calculate the fuel centerline temperature, the cladding local hoop stress and the FGR at any burn-up under the steady state condition. This was developed by Ichikawa, M and now opened to the public. FPRETAII had the same thermal and mechanical functions as those of FEMAXI-III. This code was further developed to calculate the fuel transient and accidental conditions such as RIA. The code was developed by the author but not opened to the public yet.

As experimental facts obtained from HBWR, the PCI failure was mainly related to the two parameters; a hoop stress of the cladding inside and an aggressive chemical agent induced by the fission product (FP). They are very hard to quantify because they included many burn-up dependent design factors. Typically, PCI failure threshold is shown by the maximum linear heat rating as a function of burn-up. Well developed fuel behavior code however can plot the

PCI failure by the hoop stress as a function of the fission gas release (FGR). FPRETAIN was developed to predict the transient fuel behavior of the pre-irradiated LWR fuels during the reactivity initiated accident (RIA).

3 Results and discussion

3.1 HBWR

3.1.1 Power increase

The 8x8 BWR test specimen (rod 32) was base irradiated to 18.7MWd/kgU with an average linear heat rating of 40kW/m. The linear power level was higher than that of a normal BWR fuel (30kW/m) to enhance the rod burn-up. It should be mentioned that the test specimen had a small gap (0.1mm) and thin wall cladding (0.395mm) to give a large PCMI at the power increase test.

Details on PCI failure were monitored by a failure detector (coolant radioactivity) and diameter and axial deformation data. The change in coolant radioactivity was monitored in a continuous manner. The result of monitoring from 40kW/m with the power increase rate of 0.001kW/ms is shown in Fig. 1b. It can be seen from the figure that the first two spurts of the radioactivity occurred within 10 min after reaching maximum power. The second marked spurt occurred after 3h. There is a good correlation between the two spurts and diameter/axial data.

Note that at that time diameter gauge and connected cable was degraded by the action of neutron, gamma heating and the immersed moisture from the coolant. The degradation was to weaken the signals from the core and to increase the noise level significantly. However, diameter profiles at the period could be obtained. The selected times and profiles obtained are shown in Fig.1a.

The profiles were only measured along a single generatrix (0 to 180 deg.). At the time of the first spurts ① and at subsequent periods ② and ③, corresponding profiles gave little change in shape. At the time of the second spurt ④, however, a large irregularity of cladding surface appeared.

The axial deformation of the cladding was monitored for every 15-min interval. The behavior given in the form of axial strain, as a function of power and time, is shown in Fig. 2.

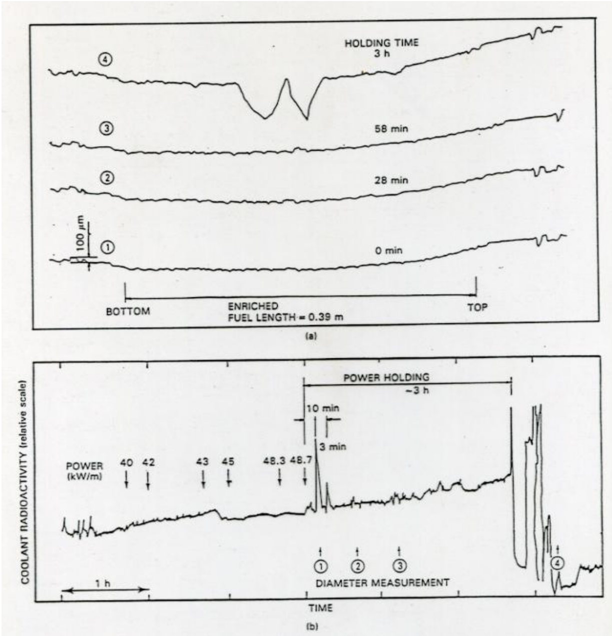

Fig.1 (a) Diameter profiles of failed rod 32 measures at the time, and (b) coolant radioactivity of the failed rod as a function of time, where rod power and time of diameter measurement are indicated.

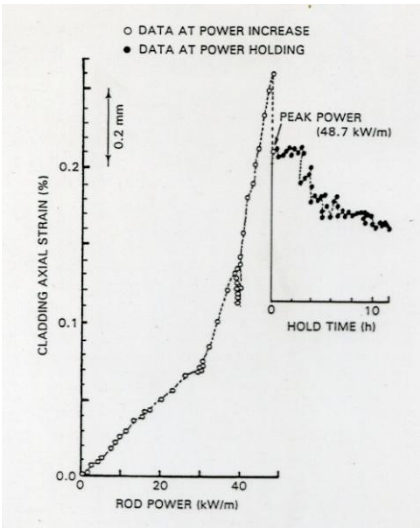

Fig.2 Cladding axial strain of the failed rod as a function of rod averaged power and holding time.

As can be seen from the figure, a sudden drop of axial strain immediately after reaching maximum power occurred. In the subsequent periods, however, a few change occurred. The sudden drop of the cladding strain coincided well with the first spurts of coolant radioactivity. Although a sign of failure in the diameter profile is not observed, the PCI failure (cracking) should be initiated from that time. The marked change of cladding surface coincided well with the second spurt of coolant radioactivity. At that time, significant cracks developed in the rod surface along a path of diameter measurement. Opened cracks released a large amount of FP gas into the coolant.

The failed rod was retained in the reactor a week longer with the power at 40kW/m and then unloaded. After unloading from the reactor, the rod was given visual inspection and post-irradiation examination (PIE). Figure.3 gives the rod outer view taken from the visual inspection and reveals that four apparent cracks existed. The locations are indicated in the photo.

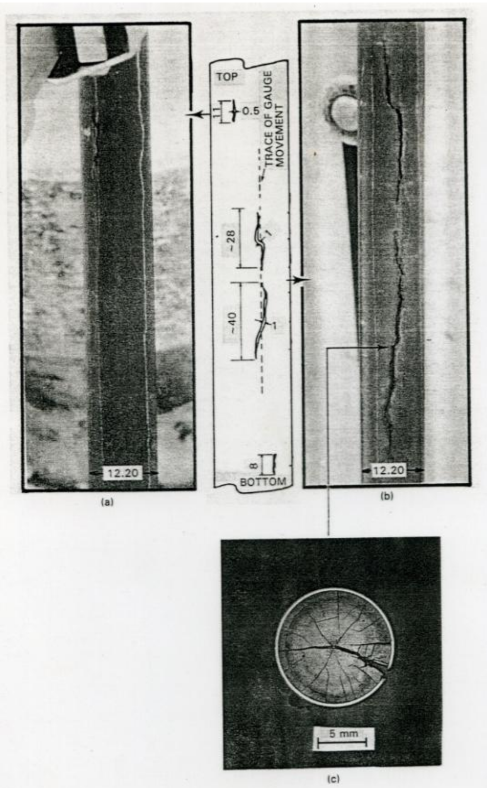

Fig.3 An over view of the failed rod taken immediately after unloading from HBWR core: (a) a crack in the rod top, (b) two marked cracks n the rod middle, and (c) a metallographic cross section taken from (b), Units are in millimeters.

Figure 3a shows a small crack located in the rod top section, and Fig.3b shows two marked cracks located in the rod middle section. The latter was on the path of diameter measurement. In addition, its location coincided well with information that was observed from the diameter profile of the failed rod (Fig.1a, ④). The cracks observed in the visual inspection stage seem to be more extended than during in-core failure stage due to hydriding. The cross sectional photograph taken from one of the marked cracks is shown in Fig.3c. A very large radial crack starting from the failure position is seen. A more detailed observation showed that many hydridings existed around cracked locations. Not mentioned in detail here but the author detected the propagation of PCI crack by the eddy current on power measurement. Usually the measurement is known to be very difficult because the crack was initiated from the inner surface of the cladding.

As mentioned above, the author succeeded to monitor on-power PCI failure by the diameter profiles, the axial elongation and the coolant activity study. In-core observations coincided with those from PIE.

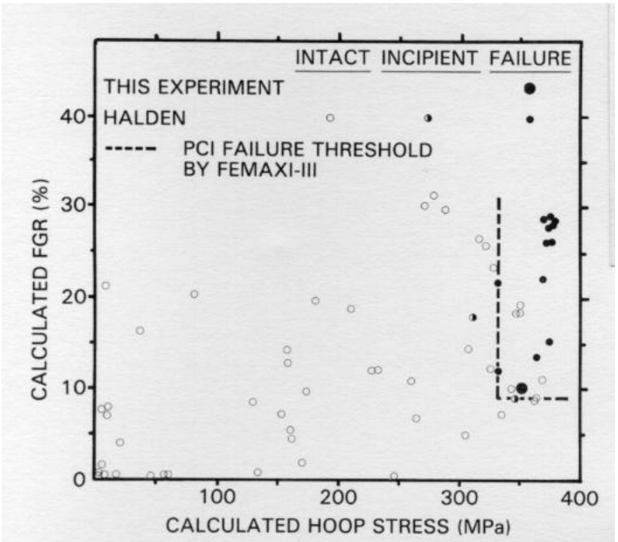

Fig.4 Failure threshold of the HBWR fuel rods (dashed line) and rod 32 used in the present study

Note: THIS EXPERIMENT means the IFA-508 experiment done by the author. Meanwhile HALDEN means that all data were from the Halden OVERPOWER tests carried out by the Halden Project. The author collected the necessary data from them as the input of FEMAXI-III code and plotted the calculated results here.

Obtained in-core data were used for the verification of FEMAXI-III code. In the verification stage, more than 60 power ramped rods under HBWR conditions were also included. PCI failed or no failed rods were plotted together as a function of calculated maximum hoop stress and FGR. The result is shown in Fig. 4.

The dotted line gives estimated PCI failure boundary for HBWR rods, where the failed rod in this experiment was within the boundary. The calculated hoop stress and FGR of the rod 32 are 350MPa and 10%, respectively.

As mentioned here, the hoop stress and FGR to cause the PCI failure are clarified by the code study.

3.1.2 Power ramp

In case of 17x17 type PWR fuel rods, they were base irradiated in the PWR loop at the linear heat rating of 20kW/m and power ramped in the PWR ramp rig to 50kW/m with a step wise power increase mode. The adoption of the step wise power was due to the ease determination of the failure threshold. Also this mode permitted to carry out the diameter measurement at the each step. In case of 8x8 type BWR fuel rods, they were base irradiated in the BWR loop at the linear heat rating of 30kW/m and power ramped in the BWR ramp rig to 50kW/m with a step wise power increase mode.

As shown in Fig.5, the PCI failure threshold for LWR (PWR and BWR) is slightly lower than that of HBWR. Note that a standard PWR fuel at 18MWd/kgU was subjected to power cycling (power range 30 to 45kW/m with period 20-25min) to 620 times without failure. A standard BWR fuel at 15MWd/kgU was also subjected to power cycling (power range 30 to 45kW/m with period 20-25min) to 826 times without failure. It was monitored that the FGR rate during the power cycling was the diffusion and dependent on the square root of the holding time at the power. This fact was also observed in the PWR fuel.

As mentioned here, the PCI failure threshold of the Japanese LWR fuels was clarified together with that of the HBWR fuels.

Resultantly, many Japanese fuel vendors especially for BWR changed the fuel design. Most of all is now used a sponge zirconium co-extruded with the fully annealed zry-2 cladding as the second layer to cease the crack propagation. Due to the results of power ramp tests, 8×8 BWR was revealed to have less resistivity to the PCI failure. The use of 8×8RJ BWR, fabricated by a sponge zirconium co-extruded with the zircaloy-2 as the second layer was recommended. PWR vendors did not use this kind of a barrier fuel.

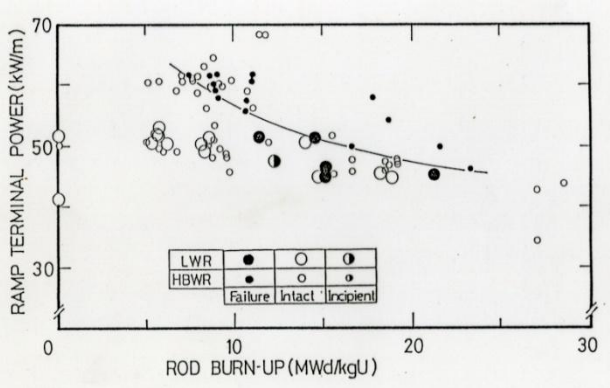

Fig. 5 PCI failure threshold for HBWR rods and LWR ones ramped in the LWR loop installed in the HBWR core.

3.2 NSRR

With respect to NSRR experiments done by the fresh LWR fuels, almost all achievements described in the subsequent section 3.2.1 were obtained from the colleague of the author. The author's contribution to the NSRR was started from the section 3.2.2.

3.2.1 LWR fuel rod experiments with the fresh fuel

To establish the safety criteria, many RIA experiments (more than 2,000 times) were performed with a NSRR standard fuel, which was essentially the same as the 14x14 PWR fuel. The highlighted finding was that the NSRR standard fuel failed around the deposited energy about 260 cal/g・fuel or the peak fuel enthalpy about 220cal/g・fuel. The estimated failure mechanism was as follows. The fuel after the pulse became very hot and moved promptly toward the cladding inside until the fuel interacted with the cladding. As a result, the cladding was melted partly or thinned the thickness partly. Because the interaction assisted the formation of a eutectic layer, the local point became significantly brittle. The cladding outside at that time was heated up due to the occurrence of departure from the nucleate boiling (DNB). When the fuel was quenched, as shown in Fig.6., the circumferential crack would be propagated at the local point until the fuel rod was broken into a few pieces. When the peak fuel enthalpy exceeded 325cal/ｇ・fuel, the fuel was fragmented into small pieces and reacted directly with the coolant. As for severe case, a generated mechanical energy (a pressure pulse) could destroy the integrity of the reactor components, as really occurred in the SL-1 accident in 1961 [8].

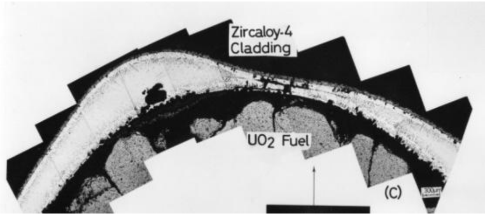

Fig.6 A failure of NSRR standard fuel by the melt-brittle mechanism, where the circumferential crack would occur during the quench and the fuel rod was broken into a few pieces.

3.2.2 LWR fuel rod experiments with the pre-irradiated fuel

With respect to the pre-irradiated LWR fuel, the author would like to focus on the PWR fuel in this report. The PWR fuel was used in the commercial power reactor for the electricity generation about 3 years. Attained burn-up during the reactor usage was about 39-42MWd/kgU. As shown in Fig.7, an original test fuel rod (OTFR) was then transported to the JAERI Hot Laboratory to shorten the size as long as about 0.122m. The author defined it as the segmented

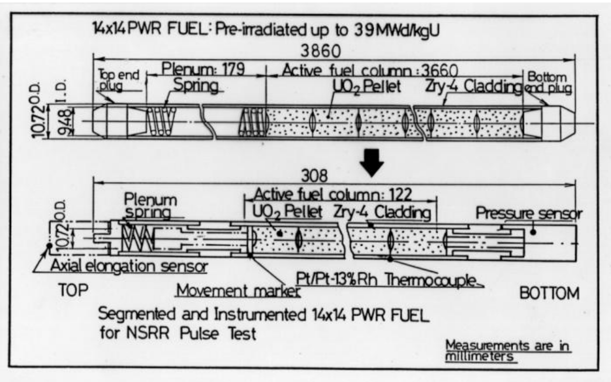

Fig.7 (Top) OTFR; (Bottom) STFR having an active column length by 0.122m.

test fuel rod (STFR). The STFR was transported again to the NSRR to attach the axial elongation sensor, the pressure transducer and the T/C. The main experimental parameter was so called the deposited energy *Eg*(cal/g/fuel), which was estimated from the integral value of reactor power *P*(MW・s) measured by gamma chambers. The conversion ratio *kg* (cal/g・fuel per MW・s), that is, the ratio of the fuel rod power to reactor power was determined through a burn-up analysis.

As mentioned here, the author and colleague succeeded to develop the technology for the fabrication of the STFR.

Up to now, several pulse irradiations were performed by the use of STFR-PWR (STFR in PWR type). During the PIE it was revealed that a strong PCMI would occur. Figure.8 shows an axial gamma scanning and diameter profiles at two generatrices obtained from the pulse irradiated PWR fuels. Residual ridges occurred at the pellet-to-pellet interfaces with a magnitude of about 50 μm, which was about ten times the normal, namely a normal residual ridge height is up to 10 μm for a fresh fuel and 2-3 μm for pre-irradiated fuel.

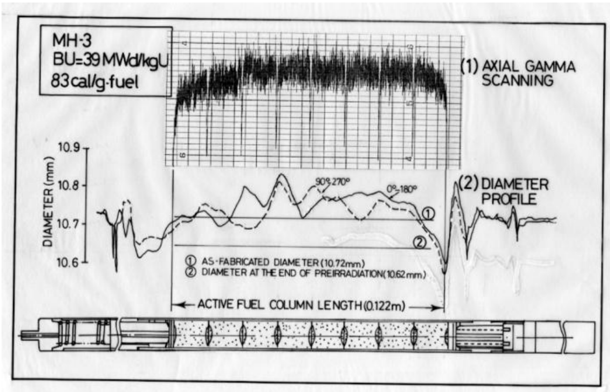

Fig.8 (1) Axial gamma scanning and (2) diameter profiles at two generatrices, observed in the PIE of MH-3 PWR rod pulsed at 83 cal/g・fuel.

The FGR of the OTFR during the use in the commercial power reactor was <1%. During the segmentation the amounts of FGR was purged and refilled 100% helium gas.

As shown in Fig.9, a transient FGR during the pulse was increased with an increase of deposited energy except for the triangle data which had the slow ramp. A maximum was around 12% and seems to be not increased more The transient FGR seemed to be not followed the diffusion mechanism because the time at the elevated temperature was too short. The author considered that the principal mechanism might be the micro hair-crack occurred at the fuel periphery, where much fission product gases were retained. As expected, during PIE, the author revealed the evidence. As shown in Fig. 10, the fuel had lots of micro hair-cracks at the periphery. The author modeled the micro hair-cracking into the computer code FPRETAIN and predicted the transient FGR. As shown in Fig.11, the predicted value coincided well with that of experiment.

As mentioned here, the transient FGR by the formation of micro-hair cracking was the new finding. The assumption from the PIE observation was proven by the theoretical consideration through the computer code FPRETAIN.

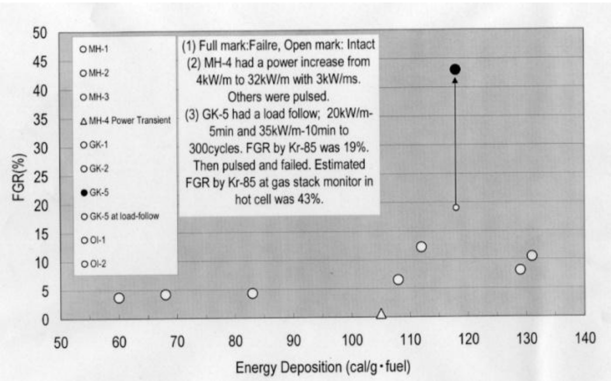

Fig. 9 Transient FGR of STFR-PWR. Open circle is intact and full circle is in failure.

As shown in Fig.9, one STFR GK-5 was in failure. As STFR GK-5 and STFR GK-1 were the sister rod; they were segmented from the same OTFR at the Large Hot Laboratory. As for GK-5, it was loaded into the Pressurized Capsule (POCA) at the JMTR to perform a load follow operation under PWR conditions. The load follow was carried out between the high power level of 35kW/m for 10-min and the low power level of 20kW/m for 5-min time interval. The total cycles were 300. It should be noted that the OTFR had an average linear power of 18kW/m and FGR by 0.42% at the end of life (EOL). During the load follow operation, an FGR occurred additionally to the magnitude of 19%. and caused an increase of the rod internal pressure from 0.28MPa to 1.2MPa. This was the estimation from the non-destructive activity counting by Kr-85 at the fuel plenum [9]. After load follow, GK-5 was inspected at the JMTR Hot Cell and confirmed the non failure. Accumulated burn-up during the load follow was about 0.2MWd/kgU.

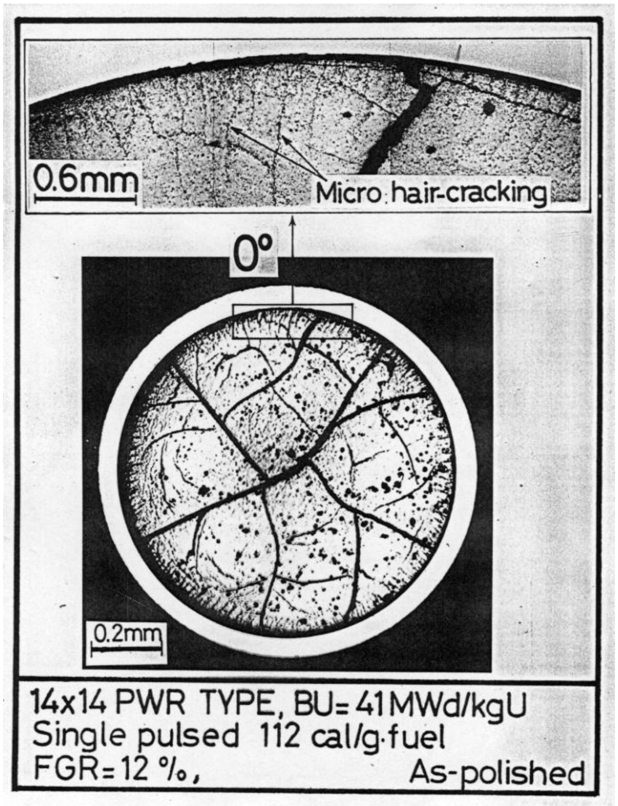

Fig.10 A fuel microstructure to show the occurrence of the transient FGR, which was caused by a fuel micro hair-crack at deposited energy of 112 cal/g・fuel.

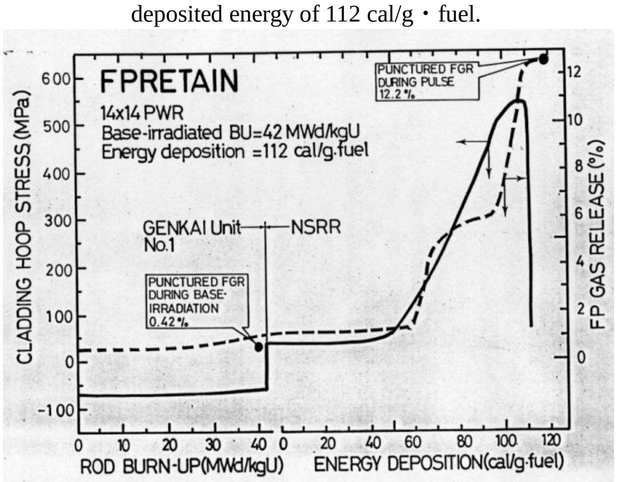

Fig.11 Transient FGR predicted by FPRETAIN and experimental data obtained from the PIE.

Subsequently, the STFR-GK-5 was transported from the JMTR to the NSRR for pulse irradiation. Pulse condition was similar to that of GK-1. As a result, the energy deposition was 118 cal/g・fuel (119 cal/g・fuel for enthalpy) for GK-5 and was 112 cal/g・fuel (104 cal/g・fuel for enthalpy) for GK-1. The peak axial strain observed was 0.85% for GK-5 and 13% for

GK-1. The small magnitude in the former might be due to the failure. As shown in Fig.12, GK-5 was split along one generatrix, where the crack propagated from the end to the top of active column region (0.123m). Estimated FGR by Kr-85 activity counting at the gas stack monitor inside the hot cell was about 43%. The failure mechanism is not revealed yet because it is a significantly difficult to separate the damage occurred in NSRR from that occurred in load follow in JMTR. Accumulated thermal effect such as the fuel cracking during the load follow is also not clarified to date.

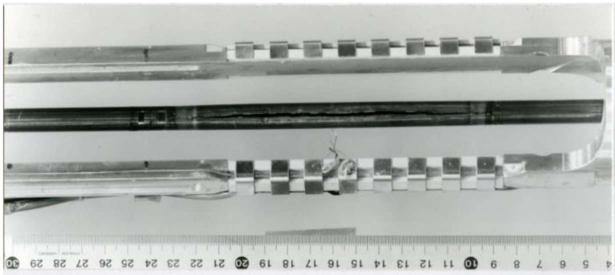

Fig.12 Overview of the STFR-GK-5, failed at the energy deposition of 118cal/g・fuel.

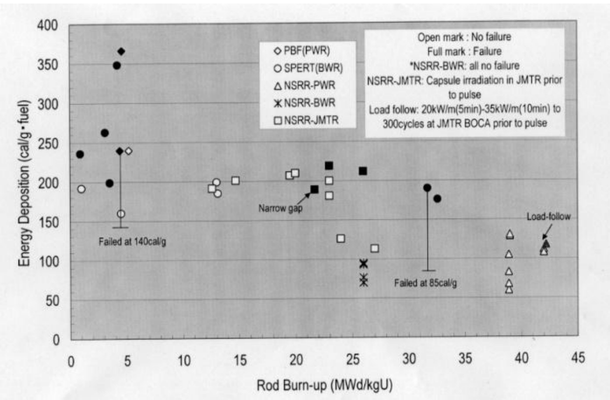

Fig.13 Energy deposition of the STFR-PWR as a function of rod averaged burn-up.

Preliminary failure threshold of the STFR-PWR at the present stage is given by Fig. 13. Note that data points from the STFR-BWR, the NSRR standard fuel irradiated by the capsule in the JMTR [10], and the U. S. [11, 12] were included as the references. Failure threshold of PBF and STFR-PWR seems to be decreased with increasing burn-up. This implies that the magnitude of PCMI as well as the transient FGR (that is also the indication of rod internal pressure) at the pulse irradiation might be the key factor.

As mentioned here, the author found the split type failure of the pre-irradiated Japanese PWR fuel. Though the mechanism is not clarified yet, the preliminary failure threshold was made.

3.2.3 Silicide fuel

One of the important failure mechanism for the silicide fuel was the blister occurred at the temperature >400-500deg.C. Besides the blister, a new type failure revealed by the author will be reported in the followings.

(1) Failure threshold

In almost research reactors in Japan now used the 4.8g/cc silicide fuel. The failure threshold for the silicide was studied in the NSRR. In the series of test denoted as Ex-508, a total of 17 tests were conducted, in which 5 data points were in failure as typically shown in Fig.14.

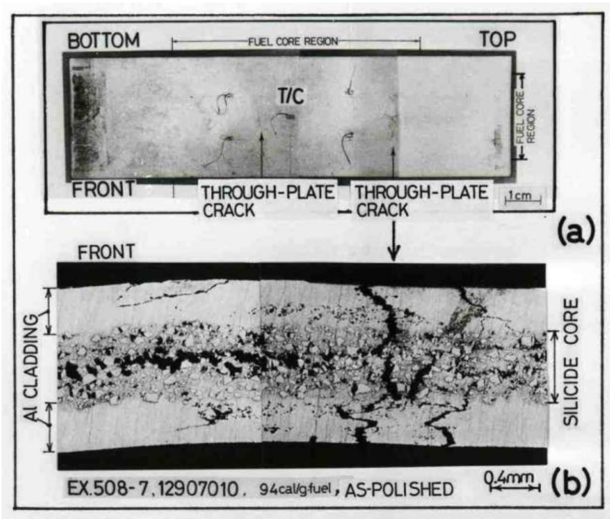

Fig.14 (Top) overview of the tested fuel plate at 94cal/g・fuel plate (B&W), where the PCST was 237deg.C. Two through-plate cracks occurred locally. (Bottom) as-polished longitudinal section cut from the location of the through-plate crack at the plate top.

It was clear that the failure took the form of through-plate cracking, occurred at the peak cladding surface temperature (PCST) above the DNB (182±15deg.C) point. As shown in Fig.15, the 4.8g/cc silicide fuels had the failure points with the energy deposition >94 cal/g・fuel. Of course, if PCST exceeds 640deg.C, the silicide shall fail by the melt mechanism.

The author focused on the two parameters here because they seemed to be related to the failure mechanism.

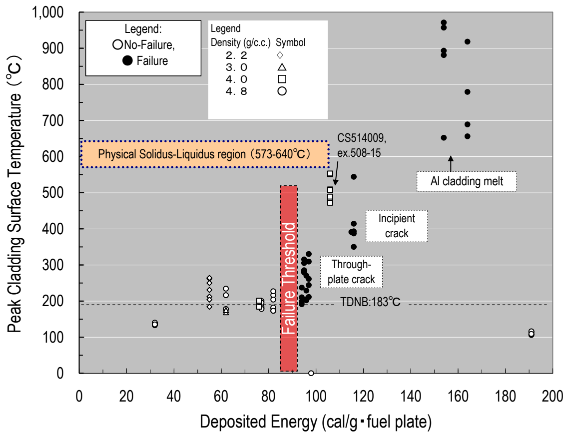

Fig.15 PCST as a function of deposited energy.

(2) The role of DNB
The onset of DNB, namely the initiation of film boiling, might be an important parameter affecting the occurrence of the through-plate cracking, occurred in the silicide fuel during the quench. After the DNB, a fuel plate heated rapidly under a relatively poor heat transfer from the cladding to the coolant. The cladding was partly covered with a vapor, where vaporized hydrogen could interact with the fabricated elements (Al, Mg) existed in the cladding. If some of the hydrogen precipitated into the cladding matrix, the cladding should be hardened to enhance the propagation of through-plate cracking. This DNB would occur at about 50 cal/g･fuel plate or temperature at 182±15deg.C

(3) The role of temperature drop
The temperature drop should occur during the fuel quench. The author considered that the temperature drop ΔT or simply ΔT here was one of the important factors related to the fuel failure mechanism, because the thermal stress arisen during the quench is given by α × E × ΔT, where α is the thermal expansion coefficient and E is the Young’s modulus. In which the value ΔT is defined by PCST-Tp (quench temperature). As shown in Fig. 16, ΔT is revealed to have a linear relationship to the PCST because Tp is almost constant, 112±12deg. C. This is one of the main reasons why the author used PCST as the index of the failure threshold (Fig.15). However, PCST is not sufficient to consider the failure mechanism of the silicide fuel because not only ΔT but also tq (time to quench) will take the important role for the initiation of thermal tensile stress.

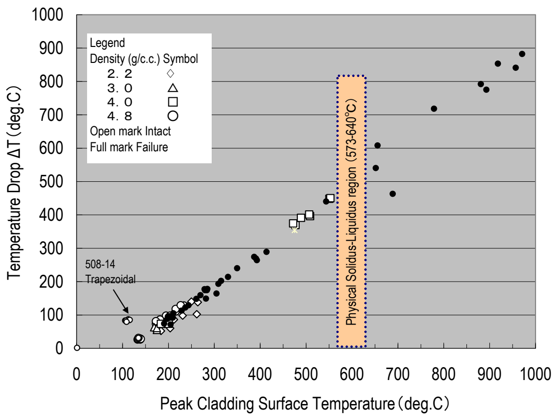

Fig.16 Temperature drop ΔT as a function of PCST.

(4) Failure mechanism
The author considered that the failure mechanism was attributed to an uneven ΔT, which caused an ill-balanced quench front during the quench of the fuel plate. To explain this, three data from Ex. 508-6 (96cal/g, failure), Ex. 508-7 (94cal/g, failure) and Ex. 508-2 (77cal/g, no failure) were prepared. PCST and ΔT at each T/C location are plotted together as shown in Fig.17. The data curve of the PCST of any fuels resembled to that of ΔT, but not constant. Here, the author focused on ΔT (triangle data) from the failed fuel. In-core data showed that the Al-3wtMg cladding at T/C#1 was heated from 22deg. C to 270 deg. C through the DNB (174deg. C for this case), and quenched to 111deg. C. As a result, ΔT was 159deg. C and tq was 0.055s. PIE revealed that the incipient crack occurred around T/C#1-T/C#2 and the through-plate crack occurred around T/C#3-T/C#4 (see the picture at the top-left side of Fig.17).

With respect to ΔT (rectangular data) from the failed fuel, Al-3wt%Mg cladding at T/C#4 was heated from 24deg. C to 237 deg. C through the DNB (179deg. C for this case), and quenched to 115deg. C. As a result, ΔT was 122deg. C and tq was 0.052s. PIE revealed that the incipient cracks occurred around T/C #1-T/C #2 and through-plate cracks occurred around T/C #3-T/C#4 (see the picture at the top right side of Fig.17).

With respect to ΔT(circle data) from the intact fuel, a magnitude of the ΔT was 85±11deg.C (n=5) and tq was 0.06±0.01s. PIE revealed no abnormality at all.

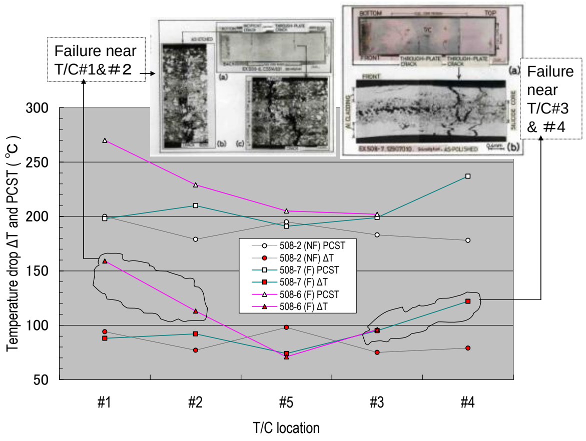

Fig.17 Temperature drop ΔT at each T/C location.

Obtained values for ΔT and tq, except for the fuel melting cases were plotted in Fig.18. This plot implies that there exists a failure threshold at ΔT 94deg.C and tq 0.13s. If a silicide fuel had a high ΔT (>94deg. C) together with a low tq (<0.13s), the risk of the fuel failure should be high and vice versa.

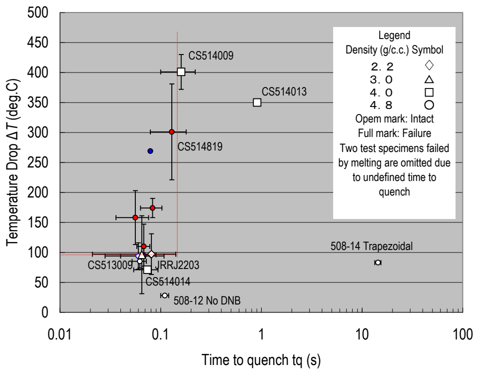

Fig.18 Temperature drop ΔT as a function of the time to quench.

As mentioned here, the author's work is addressed to finding the failure threshold and its mechanism of the silicide fuels for research reactors.

4 Conclusions

(1) The on-power monitoring of PCI failure in HBWR was succeeded, where time dependent data related to the diameter profiles, the axial elongation and FGR release to the coolant were simultaneously logged. Practically, the 8x8 BWR fuel with 18.7MWd/kgU was slow ramped from 40kW/m with the rate of 0.001kW/ms and failed at the power of 48.7kW/m. Meanwhile, the maximum hoop stress estimated was 350MPa and FGR was 10% according to the FEMAXI-III code.

(2) After commissioning the LWR loop I HBWR, Japanese LWR fuels were power ramped at the burn-up up to 20MWd/kgU. It was revealed that the tested fuels failed by the PCI mechanism, where the failure threshold is lower than that of the typical HBWR rods.

(3) It was well known that NSRR fresh standard fuel failed at the energy deposition of 260cal/g・fuel, where the melt-brittle of the zircaloy cladding was the principal mechanism. In contrast with this, the author focused on the pre-irradiated 14x14 PWR fuel up to 42MWd/kgU and found that the fuel failure occurred at 118cal/g・fuel. The effect of burn-up on the failure threshold is significant. The strong PCMI (the residual ridge height 50 μ m) and a significant transient FGR (48%) should be the cause of the fuel split in one generatrix. The mechanism is under consideration.

(4) The new finding is also addressed to the research reactor fuel. The RIA test on the fresh 4.8g/cc silicide fuel was carried out and reveled the occurrence of through-plate cracking at PCST<640deg.C, whereas the cladding was melt at PCST>640deg.C. The former should occur at temperature above DNB (182 ± 15deg.C). The principal mechanism for the through-plate cracking was thought to be the thermal stress, arisen from the temperature drop ΔT>94deg.C and the time to quench<0.13s.

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