Behaviors of High-burnup LWR Fuels with Improved Materials under Design-basis Accident Conditions

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ABSTRACT

Fuels for light water reactors (LWRs) which consist of improved cladding materials and pellets have been developed by utilities and fuel vendors to acquire better fuel performance even in the high burnup region and also raise the safety level of current nuclear power plants to a higher one. In order to evaluate adequacy of the present regulatory criteria in Japan and safety margins regarding the fuel with improved materials, Japan Atomic Energy Agency (JAEA) has conducted ALPS-II program sponsored by Nuclear Regulation Authority (NRA), Japan. In this program, the tests simulating a reactivity-initiated accident (RIA) and a loss-of-coolant accident (LOCA) have been performed on the high burnup advanced fuels irradiated in commercial PWR or BWR in Europe. This paper presents recent results obtained in this program with respect to RIA; and main results of LOCA experiments, which have been obtained in the ALPS-II program, are summarized.

1. Introduction

Japan Atomic Energy Agency (JAEA) performs the Advanced Light Water Reactor (LWR) Fuel Performance and Safety Research (ALPS) program to better understand the behavior of high burnup fuels under accident conditions and to evaluate the adequacy of the present regulatory criteria in Japan for design-basis accidents (DBAs) by providing a database for regulatory judgment. The program was launched in the Japanese fiscal year (JFY) 2002 and extensive examinations including reactivity-initiated accident (RIA) studies and loss-of-coolant accident (LOCA) studies have been performed by using some unique facilities such as Nuclear Safety Research reactor (NSRR) and Reactor Fuel Examination Facility (RFEF) in JAEA, Tokai.

In the first phase of the ALPS program (ALPS-I) which had been conducted from JFYs 2002 to 2007, high burnup UO_2 and MOX fuels with conventional fuel rod materials were subjected to pulse-irradiation experiments in the NSRR and LOCA-simulated tests in the RFEF. Based on these results, the threshold for fuel failure, data on fuel behavior, etc. under RIA and LOCA have been successfully obtained in the high burnup range from ~45 to 78 GWd/t. Following the ALPS-I program, the second phase of the program (ALPS-II) started in JFY 2008. The main objectives of the ALPS-II program are to obtain technical information about the applicability of the present Japanese regulatory criteria to the fuels with improved fuel materials e.g. advanced cladding materials with high corrosion resistance and/or pellets with lower fission gas release, in terms of DBAs. By using the NSRR and RFEF, JAEA has conducted various DBAs-simulated experiments on fuels with improved fuel materials. Up to now, five RIA-simulated tests, eight LOCA-simulated integral thermal shock tests, post-test examinations related to these tests, etc. have been completed [1-4].

The present paper describes some recent results of post-irradiation examinations in terms of a test fuel after RIA test and summarizes the main results of LOCA experiments which have been obtained in the ALPS-II program.

2. Test fuel for ALPS-II program

In the ALPS-II program, UO_2 and MOX fuels with improved fuel rod materials, which had been irradiated to 49 − 91 GWd/t (local burnup) in six European commercial reactors, are subjected to tests simulating DBAs. The cladding materials of these fuels are M-MDA^{TM}, low-tin ZIRLO^{TM}, M5^{TM}, Zircaloy-2/LK3, and the pellet materials are undoped UO_2, UO_2 with additives and MOX.

The test fuels were gathered at a site in Europe and successfully transported from Europe to JAEA, Tokai in Japan at the beginning of 2011.

Some information about the advanced fuels for the ALPS-II program is presented in Tab 1.

Reactor type	Fuel type	Nuclear power plant	Cladding material	Burnup (GWd/t)
PWR	17x17 UO_2	Vandellos	Low-tin ZIRLO^{TM}	80
			M-MDA^{TM}	73-81
		Gravelines	M5^{TM}*	84-87
	15x15 UO_2	Ringhals	M5^{TM}	68
	17x17 MOX	Chinon	M5^{TM}	64
BWR	10x10 UO_2	Leibstadt	Zry-2/LK3	73-91
	10x10 Doped-UO_2		Zry-2	49
		Oskarshamn	Zry-2	63

(*M5 is a trademark or a registered trademark of Framatome or its affiliates, in USA or other countries.)

Tab 1: List of advanced fuels for ALPS-II program

3. RIA study
3.1 Outline of the RIA test conducted at the NSRR

The NSRR is a modified TRIGA (Training, Research, Isotopes, General Atomics) annular core pulse reactor which can be used for simulating a rapid power pulse anticipated in a RIA. The details of the NSRR and NSRR experiments were described elsewhere [5]. As a typical power history during the pulse irradiation, the full width at half maximum of power pulse and peak power is about 4 ms and 23 GW at the maximum reactivity insertion of $4.6, respectively.

In the case of tests on the fuel irradiated in commercial reactors, a short test rod is prepared from the irradiated fuel rod at the RFEF in JAEA. Detailed examinations, e.g. visual and dimensional inspections, axial gamma scanning, X-ray radiography, etc., are carried out for the test rod before the pulse irradiation test. After the test rod has been transported from the RFEF to the NSRR, the test rod and test capsule are instrumented and assembled in hot cells in the NSRR. In the case of RIA tests on irradiated fuels at the NSRR, a test capsule with double walls which is made of stainless steel is used in order to ensure air tightness for preventing the release of radioactive materials from the capsule during the test.

At the center of the reactor core of the NSRR, there is a large cavity (220 mm in diameter) with the loading tube; and after the test capsule is loaded into the reactor core through the loading tube, the reactor is operated under the desired test condition. After a pulse irradiation test, the test capsule is dismantled; only the inner capsule with the test rod is transported to the RFEF and the test rod is retrieved from the inner capsule in the RFEF hot cells. The retrieved test rod is subjected to detailed post-test-examinations which are similar to those conducted before the pulse irradiation test.

3.2 Recent results of post-RIA test examinations for test fuels with improved material

JAEA has conducted RIA tests on fuels with improved materials such as cladding materials with high corrosion resistance.

In this paper, some results of post-test examinations which were newly obtained on the test fuel rod of test VA-8 are described. In terms of test VA-8, the test fuel rod was segmented from a PWR fuel rod irradiated in Vandellos-2 Nuclear power plant in Spain. Here, the test fuel rod for test VA-8 had M-MDA^{TM} cladding with a final fabrication heat treatment under a recrystallized (RX) condition, and had pellets with a conventional grain size. The average burnup of the test segment was 78 GWd/t. The test rod with a 49.5 mm fuel stack length was prepared by cutting the segmented fuel rod in the RFEF in JAEA. The test rod was subjected to a pulse irradiation under a high-temperature and high-pressure condition (HTHP condition) in the NSRR with a peak fuel enthalpy of ~500 J/g and a pulse width of 4.4 ms, respectively. Test VA-8 resulted in fuel failure at an enthalpy of 174 J/g. A visual appearance of the test fuel rod after test VA-8 is shown in Fig. 1, and pellets were scarcely released from the inside of the rod.

When the effects of coolant temperature on the fuel failure at RIA are discussed, the comparison of experimental results between tests VA-8 and -6 is considered to be useful, because the test fuel rod of test VA-6 had the same materials as that of test VA-8: both test fuel rods were prepared from the same segmented fuel rod and test VA-6 was conducted under a coolant condition at room temperature. Detailed specifications of test rods for tests VA-6 and -8, NSRR test conditions and some test results are summarized in Tab. 2.

The fuel enthalpies at failure are plotted as a function of fuel burnup in Fig. 2 [3], with the current Japanese PCMI failure criteria defined up to 75 GWd/t in 1998. It is found that the PCMI failure limits obtained from tests VA-6 and 8 tend to be lower than those of the test fuels which have M-MDA^{TM} cladding with a final fabrication heat treatment under a stress-relieved (SR) condition (tests VA-5 and -7). This trend is considered to be due to the effect of the hydride precipitated along the radial direction in the cladding in connection with a final fabrication heat treatment under RX condition, in addition to the difference in the amount of hydrogen absorbed during base irradiation between these cladding tubes [2].

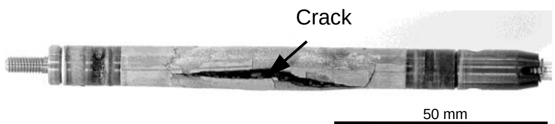


Fig.1. Visual appearance of the test fuel rod after test VA-8

Test ID	VA-6	VA-8
Fuel enrichment (%)	4.9	
Fuel pellet diameter (mm)	8.19	
Cladding inner/outer diameter (mm)	8.36/9.5	
Cladding material	M-MDA^{TM} (RX) †	
Rod burnup (GWd/t)	78	
Averaged cladding oxide thickness (μm)	60	68
Fuel stack length (mm)	117	49.5
Fill gas in the rod (MPa)	0.1 (He)	
Coolant condition	0.1 MPa, stagnant water	6.7 MPa, stagnant water
	~288–290 K	555 K
Max. increase in fuel enthalpy (J/g)	601	496
Fuel enthalpy increase at failure (J/g)	142	174

†Commercial use of M-MDA^{TM} (RX) as cladding is not planned.

Tab 2: NSRR test conditions and results.

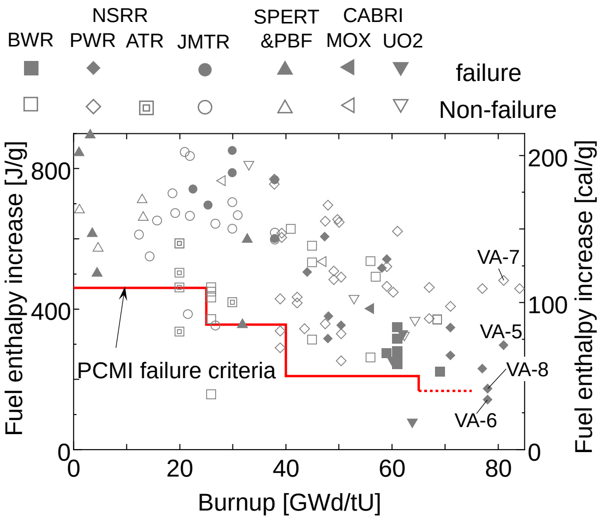
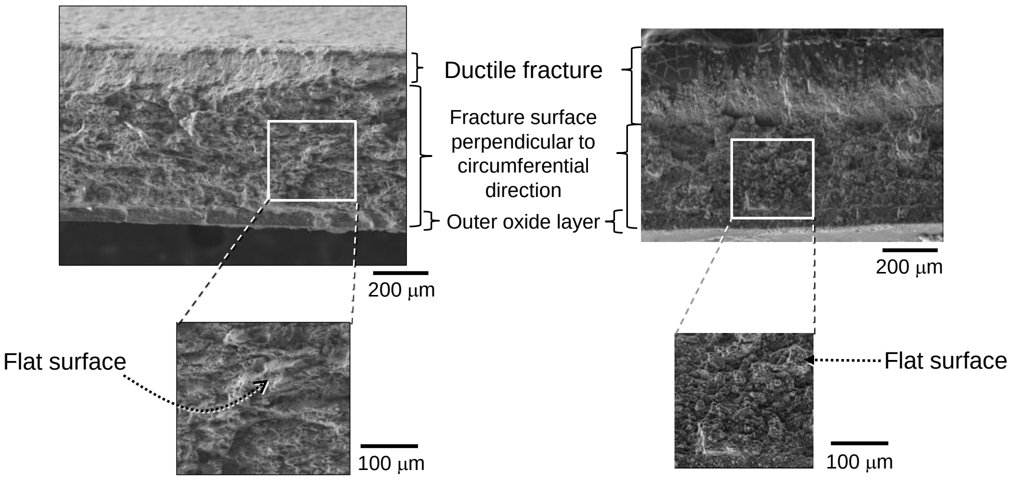


Fig. 2. Fuel enthalpy increases at failure as a function of fuel burnup [3]

In order to investigate the effect of the radially precipitated hydrides on fuel failure under the HTHP condition, crack-surface observations were carried out on the cladding failed in test VA-8 by using a scanning electron microscope (SEM). Examples of the SEM images are shown in Fig. 3. In the figure, SEM images taken on a radial crack surface of the cladding after test VA-6 are also shown for comparison. Similar fractographic morphology was seen on the crack surfaces of both the cladding tubes failed in tests VA-6 and -8: in addition to shallow dimples which indicate ductile fracture, regions with flat and smooth surfaces could be observed between the inner and outer surfaces of the cladding. It is considered that these flat and smooth regions correspond to the fracture surfaces of the hydride precipitates, as well as in the case of tests VA-5 and -7 (M-MDA^{TM}(SR) cladding) [3]. This means that the hydrides precipitated along the radial direction of cladding tube remained in the cladding tube at the fuel failure even under the HTHP coolant condition and affects the failure limit at the RIA test. This may be related to the difference in the fuel failure limit between tests on M-MDA^{TM} (RX) cladding (tests VA-6 and -8) and M-MDA^{TM}(SR) cladding (tests VA-5 and -7) in which hydrides precipitated mainly along the circumferential direction [3].



(a) VA-6(M-MDA^{TM}(RX), RT condition) (b) VA-8(M-MDA^{TM}(RX), HTHP condition)

Fig. 3. Crack surface observation results of the cladding tubes failed in tests VA-6 ((a)) and -8 ((b))

4. LOCA study

In a safety analysis for a postulated LOCA (a design-basis-accident LOCA, a DBA-LOCA), it is estimated that fuel cladding is exposed to steam at high temperatures for several minutes until an emergency core cooling system (ECCS) is activated and the cooling water injected by ECCS quenches the fuel bundles in the reactor core. During this process, it is possible that the fuel cladding is severely oxidized and embrittled. In order to maintain the coolable geometry of the reactor core during and after the DBA-LOCA, the Japanese regulatory criteria for LOCA require that the oxidation of the cladding, calculated by using the oxidation rate equation proposed by Baker and Just [6], shall not exceed 15% of the cladding thickness (ECR: Equivalent Cladding Reacted) in order to prevent massive failure of fuel rods and/or fuel bundles by quenching. The limit is mainly based on the thermal shock resistance (fracture/no-fracture boundary) of unirradiated and oxidized cladding which was experimentally determined under simulated DBA-LOCA conditions. With increasing fuel burnup, effects of corrosion, hydrogen absorption and neutron irradiation on the mechanical properties of cladding become considerable. Considering that the current Japanese regulatory criteria for DBA-LOCA are based mostly on the data obtained by using unirradiated Zry-4 cladding, the thermal shock resistance of the fuel rod with not only high burnup but also advanced cladding material has been a current concern for the safety of LWRs in Japan. From the viewpoint of the safety review for the nuclear power plant in which the fuel with improved materials will be loaded into its reactor core, investigations on the behavior of fuel rods with advanced cladding material under DBA-LOCA conditions is needed.

JAEA has conducted studies on fuel behavior under a DBA-LOCA condition in the ALPS-II program, by carrying out experiments such as integral thermal shock tests, cladding oxidation rate tests and four-point bend tests.

4.1 Results of integral thermal shock test and four-point bend test on the high-burnup cladding with improved material

The outline of the integral thermal shock test is as follows: a segment of about 190 mm in length was cut from the fuel rod, and the fuel pellets inside the segment were mechanically removed by drilling. Afterwards dummy Al_2O_3 pellets were inserted into the cladding specimen in order to simulate the thermal condition of a fuel rod during a DBA-LOCA, and end plugs were welded on both ends of the cladding specimen. The inside of the test rod was pressurized with Ar gas of about 5 MPa at room temperature and sealed. Four R-type thermocouples were spot-welded onto the outer surface of the test rod at different elevations to control and measure the cladding temperature during the test. The test rod was set inside the quartz tube of a test apparatus, and heated up at a rate of 3 to 10 K/s to a desired target oxidation temperature (usually 1473 K (1200 ^{o}C)) by using an infrared image furnace under a steam flow condition. The test rod was ballooned and ruptured during the heat-up stage. After the rupture, both the inner and outer surfaces of the ruptured cladding of the test rod were isothermally oxidized for a predetermined oxidation period at the desired target oxidation temperature. After the isothermal oxidation, the rod was cooled in a steam flow to about 970 K and finally quenched by flooding water from the bottom of the test rod. To achieve a restrained condition at the quench during the DBA-LOCA, which is expected when a constricted condition occurs between fuel cladding and parts in the space or grid in the case of the bundle geometry, both ends of the test rod were fixed just before the cooling stage initiates. The tensile load increases as the rod is cooled and quenched, because cladding shrinkage is restrained. Since it is considered that a fully constrained condition is too severe, the restraint load was controlled not to exceed 540 N in the ALPS-II program: this load was determined based on previous studies [7, 8].

In the ALPS-II program, eight integral thermal shock tests for several improved cladding materials, of which characteristics are summarized in Tab. 3, have been conducted.

Test rod ID	Burnup (GWd/t)	Cladding material	Specimen conditions before the integral thermal shock test		Results of the integral thermal shock test		
			Oxide layer thickness (μm)	The amount of absorbed hydrogen (ppm)	Isothermal oxidation temperature (K)	ECR* (%)	Fracture/ non-fracture
MMDA1	81	M-MDA™	27.0	249	1505	21.6	Non-fracture
MMDA2				227	1475	20.8	Non-fracture
LMFI3	84	M5™	10.3	64	1474	21.5	Non-fracture
LMFI4				63	1473	29.6	Non-fracture
LZIR4	80	Low-tin ZIRLO™	17.9	144	1474	21.0	Non-fracture
LZIR5				123	1475	20.8	Non-fracture
LZRT3	85	Zry-2 (LK3)	38.9	492	1472	20.0** (10.0**)	Non-fracture
LZRT4	73		34.6	395	1474	19.8** (9.9**)	Non-fracture

* Calculated by using the Baker-Just equation with oxidation temperature and time. In this calculation, the metal layer thickness in the cladding before the test and the decrease in cladding-wall thickness due to ballooning during the test were taken into account.
** Since the oxidation condition at the inner surface during these integral thermal shock tests was not the same as that at the outer surface (only in these cases, the oxidation on the inner surface during the integral thermal shock test was not significantly observed), the amount of oxidation is considered to be between these upper and lower values, which correspond to double-sided and single-sided oxidation cases, respectively.

Tab. 3: Characteristics of the specimen subjected to an integral thermal shock test.

Fig. 4 presents a fracture map for the fuel cladding tubes with improved materials, which was obtained in the ALPS-II program and relevant to ECR and oxidation temperature. In this figure, the black and gray symbols show the data obtained in the ALPS-II program and previous tests [9], respectively. In addition, the notations show the specimen IDs which include cladding materials (MMDAs: M-MDA™, LMFIs: M5™, LZIRs: low-tin ZIRLO™, and LZRTs: Zircaloy-2(LK3)), and the amount of oxidation which is shown as ECR was calculated by using the Baker-Just equation [4] with oxidation temperature and time, taking into account the metal layer thickness in the cladding before the test and the decrease in cladding-wall thickness due to the ballooning that occurred during the test. Here, the data points of LZRT3 and 4 correspond to those of the upper and lower ECR values (see the caption of Tab. 3). As seen in this figure, it was confirmed that the fracture limits of the fuel cladding tubes with the improved materials are located above the current ECCS regulatory criteria in Japan in the burnup range up to about 85 GWd/t.

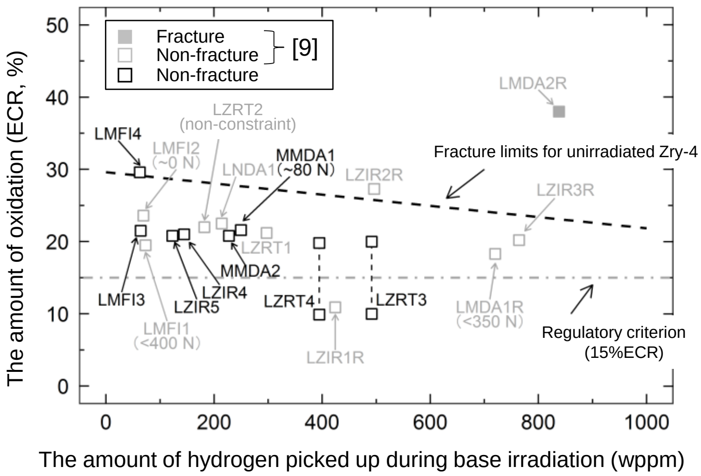
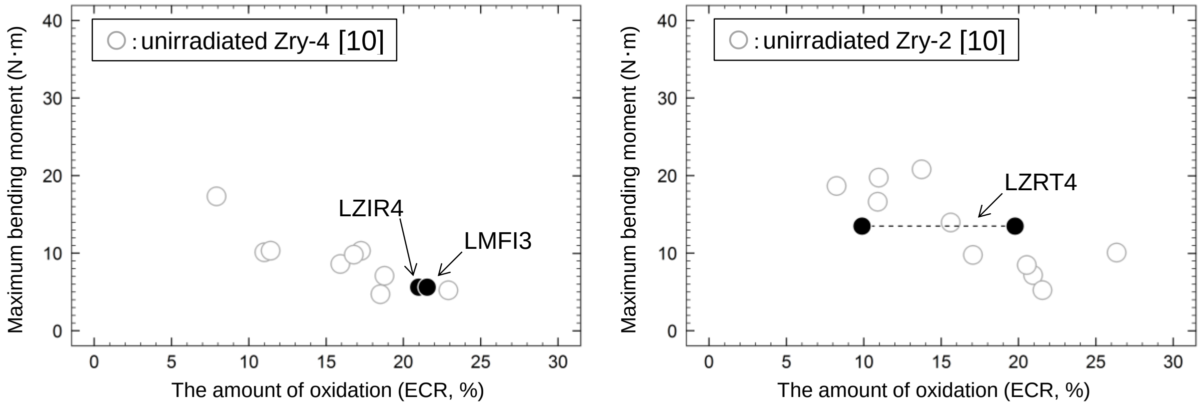


Fig.4. The fracture map relevant to ECR and oxidation temperature for advanced fuel cladding (Open symbol: non-fractured during the test. Solid symbol: fractured during the test.)

In order to investigate the mechanical strength of a cladding tube against the bending motion applied under *e.g.* a seismic condition after a DBA-LOCA, four-point bend tests have been conducted on several kinds of cladding tubes with improved materials after the integral thermal shock test.

Fig. 5 compares the maximum bending moments obtained in the ALPS-II program with literature data obtained at 410 K for unirradiated Zry-4 and Zry-2 cladding specimens [10]. The ECR values in the figures were calculated using the Baker-Just equation [6] with oxidation temperature and time, and the metal layer thickness in the cladding before the test and the decrease in cladding-wall thickness due to ballooning during the test were taken into account in this calculation. Here, the data points of LZRT4 correspond to those of the upper and lower ECR values (see the caption of Tab. 3). From this figure, it is found that the maximum bending moments of the cladding tubes with improved materials are quite similar to those of unirradiated Zry-4 and Zry-2 cladding specimens. This suggests that the burnup and the difference in the composition of cladding material hardly affects the maximum bending moment of cladding having experienced a LOCA.



(a) PWR cladding (b) BWR cladding

Fig. 5. The relationship between the maximum bending moment and the amount of oxidation for the low-tin ZIRLO™, M5™ ((a) PWR cladding) and Zry-2 (LK3) ((b) BWR cladding) specimens with burnups of ~80GWd/t after the integral thermal shock test. The open symbols show the literature data [10] and the solid symbol shows the data of the specimens obtained in the ALPS-II program

4.2 Test results on the oxidation rate in high-temperature steam

The outline of the cladding oxidation rate test is as follows: a fuel cladding specimen was prepared from a segmented test fuel rod by removing fuel pellets mechanically, and test specimens (~10 mm in length) were prepared from the fuel cladding specimen. One of the test specimens was set on the holder of an apparatus for the cladding oxidation rate test, and the holder with the specimen was inserted into the furnace of the apparatus which had been kept at a desired temperature with a steam flow. After isothermal oxidation of the specimen for a desired period, the holder was quickly withdrawn from the furnace and the specimen was quickly cooled down to room temperature. The weight of the specimen after oxidation was measured, and the rate constant of oxidation was evaluated based on the weight and surface area of the specimen before oxidation.

Tab. 4 summarizes the characteristics of the high-burnup fuel cladding tubes with improved materials. Fig. 7 plots the temperature dependences of the parabolic rate constants for weight gain of the specimens with several kinds of improved cladding materials. The data for unirradiated materials are also shown for comparison. In terms of the oxidation rate, it is found that the specimens obtained from the high-burnup fuels with improved cladding materials have values quite similar to those of unirradiated materials including Zry-4. This tendency suggests that the oxidation rates of these improved cladding materials do not significantly accelerate in the burnup range and temperature level examined. It is also found that the values estimated by using the Baker-Just equation [6] give conservative values in the burnup range up to about 85 GWd/t. These tendencies can be seen in other literature [11].

Cladding material	Burnup (GWd/t)	Oxide layer thickness (μm)	The amount of absorbed hydrogen in the specimen before the oxidation test (ppm)
M-MDA™	81	27.6	292
M5™	84	10.3	65
Low-tin ZIRLO™	80	17.9	99
Zry-2 (LK3)	85	38.9	488

Tab. 4: Characteristics of the irradiated fuel cladding with improved materials subjected to the oxidation tests

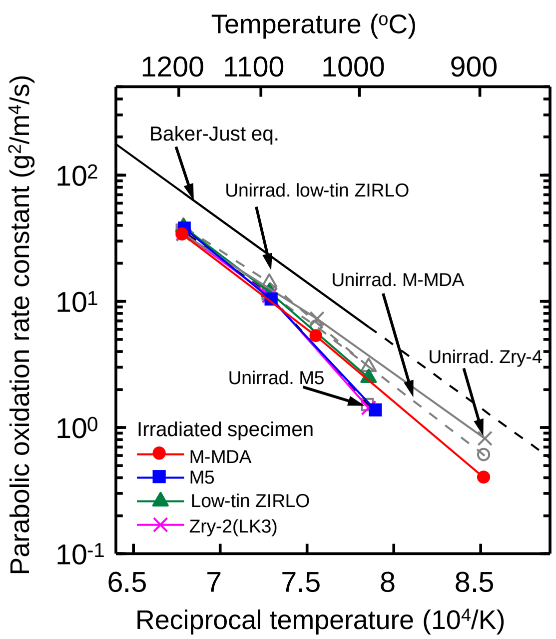


Fig. 7. Temperature dependence of the parabolic rate constants for weight gain of the cladding specimens with advanced materials

5. Summary

Fuels for light water reactors (LWRs) which consist of improved materials such as cladding with high corrosion resistance and pellets with lower fission gas release have been developed by utilities and fuel vendors to improve the fuel performance even in the high burnup region and also raise the safety level of current nuclear power plants to a higher one. In order to obtain technical information about the applicability of the present Japanese regulatory criteria to the fuels with improved fuel materials e.g. cladding materials with high corrosion resistance and/or pellets with lower fission gas release, in terms of DBAs, JAEA has conducted a research program called ALPS-II program, which has been sponsored by the Nuclear Regulation Authority (NRA), Japan.

In this program, tests simulating a RIA and a LOCA have been performed on high burnup fuels with improved materials irradiated in commercial reactors in Europe. The failure limits of the high-burnup advanced fuels under RIA conditions have been obtained by pulse irradiation tests at the NSRR in JAEA. Post-test examinations are being performed on the fuel rods after pulse irradiation. In terms of the simulated LOCA test, integral thermal shock tests, high-temperature oxidation tests and four-point bend tests have been performed at the RFEF in JAEA, and the fracture limits under LOCA and post-LOCA conditions, oxidation rates in high-temperature steam, etc. of the high-burnup fuel cladding with improved materials have been investigated.

In this paper, some recent results of post-test examinations were introduced in terms of fuel rods with M-MDA^{TM} (RX) cladding after the pulse irradiation tests. Information was obtained about the effect of coolant temperature on the precipitation condition of hydrides in the M-MDA^{TM} (RX) fuel cladding and the cause of the difference in the failure limits of the fuel rods with the M-MDA^{TM} (RX) and M-MDA^{TM} (SR) cladding during the pulse irradiation tests. In terms of the simulated LOCA tests on high-burnup PWR and BWR fuel cladding with improved materials, the main results of the integral thermal shock tests, high temperature oxidation tests and four-point bend tests, which have been obtained in the ALPS-II program, are summarized as follows: regarding the integral thermal shock tests on the cladding tubes examined, no fractures were observed. In the results of four-point bend tests, a decrease in the maximum bending moment was not clearly observed. This implies that the fracture boundaries of these materials do not change compared with conventional materials even in the high-burnup region. The high-temperature oxidation rates of these cladding tubes with improved materials were similar to those of unirradiated materials and Zry-4 and lower than the values estimated by the Baker-Just equation, and this suggests that the oxidation in high-temperature steam does not significantly accelerate in the high-burnup region.

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