On the degradation progression of a BWR control blade under high-temperature steam-starved conditions

Anton PSHENICHNIKOV*, Yuji NAGAE* and Masaki KURATA*

*Japan Atomic Energy Agency (JAEA), Collaborative Laboratories for Advanced Decommissioning Science (CLADS), 790-1 Otsuka, Motooka, Tomioka, Futaba District, Fukushima, 979-1151, Japan
E-mail: pshenichnikov.anton@jaea.go.jp

Received: 14 October 2019; Revised: 23 December 2019; Accepted: 30 March 2020

Abstract
High-temperature control blade degradation tests simulating a beginning phase of a severe accident in BWRs has been comprehensively performed in Japan Atomic Energy Agency (JAEA). In the latest test, a mock-up of BWR bundle material has been investigated under postulated Fukushima Dai-Ichi (1F) unit 2 accident conditions in a complex heating transient scenario including a phase of lack of available steam. The progress in control blade degradation was monitored with help of an in situ video and the detailed analysis of the solidified metallic melt, so-called “metallic debris”, was carried out by conventional SEM EDS method. These results indicated that the composition of the metallic debris at different elevations has been significantly changed due to the redistribution and relocation of steel elements under the influence of B and C, sometimes accompanied by a formation of high-melting-point layers. The results of this paper significantly contribute to the physical understanding of control blade degradation phenomenology during beginning phase of a core degradation for a special case of steam-starved conditions at 1F unit 2.

Keywords: Boiling water reactor, Severe accident, Fukushima Dai-Ichi decommissioning, Control blade degradation, Boron carbide, Fuel debris

1. Introduction

Much of previous work was devoted to understanding the fuel/core degradation under severe accident conditions of Pressurized Water Reactors (PWRs). It has been shown that under prototypic accident conditions a formation of two melt pools in the core region can be possible (Coryell et al., 1994). Relocation of molten materials starts firstly by metallic melt with lower melting temperature. The liquid metallic melt proceeds downwards leaving material with higher melting point (in most cases oxides) on the top of the bundle. Then, under much higher temperatures, oxides can create an oxidic melt pool, which also propagates downwards. In general, many features of accident progression in PWRs are similar to those of BWRs. However, differences exist in the distribution of fuel rods and coolant paths within a fuel element, which might affect the relocation and blockage processes, timing of the accident progression and heat-up rates (OECD/NEA, 2000). That is why studies on BWRs’ accident progression have been additionally carried out, for example, at Sandia National laboratory (SNL) (Gauntt and Gasser, 1990), Kernforschungszentrum Karlsruhe (KfK) (Hagen et al., 1994a and 1994b) and Forschungszentrum Karlsruhe (FZK) (Hagen et al., 1997), (Sepold et al., 2009). After Fukushima Dai-Ichi (1F) accident (TEPCO, 2012), one of the most important problems became the accident progression under a steam-starved conditions, which might occur with a high probability in the accident sequence of 1F unit 2 (Attachment 2-14 of The 5th Progress Report, TEPCO, 2017). The degradation of a control blade could largely affect the sequence. Japan Atomic Energy Agency (JAEA) actively contributed to decrease of the knowledge gaps in understanding of control blade degradation behavior of BWRs (Kurata et al., 2014), (Shibata et al., 2016), (Pshenichnikov et al., 2019a). Still it is difficult to predict the progression of melt at each unit of 1F and characteristics of debris and their final distribution in damaged reactors (IRID and IAE, 2018), probably, because of unique accident conditions at 1F.

The understanding of the 1F unit 2 behavior involved many challenges as a less studied case of a BWR accident in
a case of a steam-starved environment of a depressurized reactor (Attachment 2-14 of The 5th Progress Report, TEPCO, 2017). As it is well understood by many authors (Hofmann et al., 1990), (Nagase et al., 1997), (Steinbrück, 2010) that the first step interaction must be a eutectic liquefaction of stainless steel (SS) with B\textsubscript{4}C. This interaction could create large amounts of metallic melt, mainly consisted of SS. Then, in the second step, the metallic melt attaches the surface of Zircaloy channel box. Obviously, the significance on the second step interaction much depends on the thickness of the preliminary formed oxide layer on the Zircaloy surface. Under steam-starved conditions the protective properties of the layer are not fully guaranteed because oxygen atoms diffuse further into the region of available Zr metal. This process depletes the oxide layer in oxygen (external supply from the surface is limited, but concentration gradient drives the diffusion further, so the source of the oxygen becomes the oxide layer itself), which transforms the zirconium oxide back into α-Zr(O) firstly near the surface (Stuckert et al., 2002). This means that the lack of pre-oxidation can promote severe melting by involving large volume of Zircaloy into the eutectic melt of SS and B\textsubscript{4}C.

Later, in the third step, the mutual diffusion of elements could result in a stabilization of some new phases other than oxides (B\textsubscript{4}C could form borides and carbides by reacting with SS elements (mostly Fe and Cr) and/or Zr alloy). The steam-starved condition might influence the stabilization of new phases. Enough preliminary oxidation could probably suppress severe melting and restrict the interactions only the one, namely between SS and B\textsubscript{4}C.

This paper aims at understanding of the control blade degradation sequence (relocation of metallic melt and formation of debris) in BWRs under the steam-starved conditions. In particular, it is important to know changes in the phase composition of interacting materials at different temperatures, because it results in variations in their mechanical and chemical properties and thus can change an accident progression at the beginning phase of an accident.

2. Experimental

2.1 LEISAN facility

The LEISAN facility (Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors) (Pshenichnikov et al., 2019b) has been developed in the research building of Collaborative Laboratories for Advanced Decommissioning Science (CLADS) in JAEA. It is a high-temperature furnace for investigation of non-radioactive materials behavior in steam or Ar environment. Customized assemblies up to 1.2 meter in length can be tested. The large-scale facility consists of:

- furnace, with an internal volume 1.3×0.6×0.5 m\textsuperscript{3};
- gas supply system;
- steam generator, with a range of flow: 0 – 50 g H\textsubscript{2}O/min;
- furnace outer surface cooling system;
- furnace control unit;
- electric power unit;
- resistive graphite heaters, to increase temperature rapidly and to create a necessary temperature gradient;
- water-cooled exhaust line, with alumina tube inside for catching aerosols,
- windows, for in situ process observation
- quadrupole mass-spectrometer, for exhaust gas analysis.

One of the biggest advantages of the facility is the possibility of obtaining a video history of degradation process in situ from three viewpoints: top view and two side views (assembly hottest top part and in the upper middle). It allowed us to analyze the shape degradation, melting and material relocation in details and precisely control the time of observed phenomena.

2.2 Test assembly

A representative part of a BWR fuel assembly (Fig. 1) contained 20 fuel claddings divided in two groups. Each group was encased into a Zircaloy-4 canister, which simulates a channel box. Between two canisters, a part of a control blade was placed, which was consisted of sim-control rods with 3.5 mm inner tube diameter, a U-curved SS-sheath and a central supporting tie rod (SS). All technical control-bade-to-channel-box or cladding-to-channel-box gaps were prototypic for the BWR fuel assembly in Japan. The cladding-to-cladding gap was guaranteed by the original BWR fuel spacer. The claddings sealed from both sides in an inert environment under atmospheric pressure. Uranium dioxide pellets were substituted by alumina pellets. Twelve thermocouples of the W-Re type (W5Re/W26Re) were attached to the sample. Four of these twelve were attached to the control blade directly. Temperature history was recorded by a data

Fig. 1 Sample for CLADS-MADE-01 test a) a mock-up of BWR fuel, b) a horizontal cross section of prototypical fuel elements, consisting of four fuel assemblies and one control blade, c) sample top view.

logger GL840-M (GRAPHTEC, Japan) with a 1 s of sampling interval.

A prototypic thickness of the oxide layer is in the range of 20 to 50 μm on the Zr alloy materials of the fuel assemblies after several campaigns of operation (IAEA, 1993). That is why in present study we included a pre-oxidation phase to 45 μm thick oxide scale during 3 hours at around 1138 K in hot steam. The lower part of the assembly, being colder, was less oxidized. It is usually the same during normal reactor operation conditions, that the both ends the fuel are less oxidized because of complex influence of oxidative factors (IAEA, 1993), (Hoffmann et al., 2005). That is why it was an advantage of the experiment, that the axial gradient of oxide layer has been created much like in the real situation.

2.3 CLADS-MADE-01 test scenario

The Mock-up Assembly DEgradation test (CLADS-MADE-01) was aimed at improving our knowledge in BWR severe accidents, especially at the initial accident phase – a control blade degradation phase, accompanied by melting of low-temperature eutectics of stainless steel with B$_4$C. The scheme of scenario with temperature and a steam flow rate is in the Fig. 2.

Fig. 2 CLADS-MADE-01 test scenario.
After the preliminary oxidation during approximately 3 hours at 1138 K under steam flow of 5 g/min the test sample was cooled down to 873 K. The carrier gas (Ar) flow rate was constant 20 L/min throughout the test. Two independent groups of heaters created an axial gradient in the test sample. Accident sequence started by transient heating from \( \approx 873 \) K with approximately 0.4 K/s near the upper heaters and 0.35 K/s near lower heaters. The heating rate was chosen based on report data (OECD, 2000) and plant data for unit 2 (TEPCO, 2017). Steam was stepwise increased from 5 to 20, 30, 40 g/min to maintain the level enough for normal oxidation during transient heating based on the minimum oxygen necessary for each elevation of the test bundle calculated by oxidation kinetics (Leistikow et al., 1978) taking into account given temperature and preliminary oxidation history. After 55 minutes, temperature was stabilized at around 1750 K (the sample hottest point) with axial gradient 420 K/m and steam was switched off to create steam-starved conditions for 45 minutes. Natural cooling by heat losses in Ar finished the test. This scenario approaches the sequence that most likely happened in the same way in the reactor core of 1F unit 2 during boil-out of the coolant and subsequent prolonged full absence of coolant in the core region (TEPCO, 2017).

2.4 Post-test investigations

2.4.1 Cross-sections choice

![Six cross-sections of the CLADS-MADE-01 sample made at the corresponding distance (in mm) from the bottom.](image)

Fig. 3 Six cross-sections of the CLADS-MADE-01 sample made at the corresponding distance (in mm) from the bottom.

In all, six cross-sections have been prepared. Three the most interesting of them has been chosen for post-test analysis (Fig. 3). The upper part of the degraded control blade claddings 1090 mm (UP), the blockage melt at 425 mm (BLM) and a cross-section from a relatively cold region at 84 mm (CR).

2.4.2 Scanning electron microscopy

Scanning electron microscopy (SEM) has been done using JEOL JSM-7800F Schottky Field Emission Scanning Electron Microscope. The images were acquired using a lower electron detector (LED). It enabled acquisition of high-quality high-resolution images with a 3-dimensional appearance and possibility to observe material contrast. The working distance was 10 mm; the acceleration voltage was 15 kV. The elemental analysis was performed by energy-dispersive x-ray spectrometry (EDS) system attached to SEM with the same electron beam parameters.

3. Results

3.1 Understanding of the control blade degradation (UP zone)

The hottest upper part of the mock-up assembly (named as UP zone) after the test consisted of materials that survived by 1750 K. Zr parts were significantly oxidized. Control blade structures underwent severe melting. For example, stainless steel sheath fully disappeared in the top part. Control rods remnants bended, fused with each other, but survived in the top part at 1750 K. The initial tube geometry was lost, the material softened and according to in situ video, only leaning against the oxidized channel box did not allow an immediate slumping of the control blade downwards. However, during the transient stage structure became stable again, probably solidified, because of further diffusion of B and C into surrounding materials (see also in Pshenichnikov et al., 2019c).

In the Fig. 4, degraded B\(_4\)C granules trapped in the metallic melt can be observed. According to EDS analysis the degraded granules contained B and C (Fig.4c, d). However, the characteristic energies of B and C were overlapping on the EDS-spectrum, which made it difficult to conclude on a precise chemical composition in the region. Additional investigations by another method would help to confirm precise B and/or C content of dissolved granules’ traces.
The enrichment of melt with B resulted in formation of Fe-rich phase which was observed relatively homogeneously in the cross-section, whereas Cr-rich phase was observed right in front of the granules creating a kind of a new shielding layer in front of them (Fig. 4b). The enlarged region of reacted cross-section with dark and bright appearance of control blade melt phases is represented in the Fig. 5a. The detailed EDS-analysis (Fig. 5b) revealed that green zones contained more Fe and red zones more Cr. The green zone contained more metallic elements than red one (Table 1). At the same time the highest concentration of B was always detected in red (Cr-rich) zones, whereas green (Fe-rich) zones were having less B content (Table 1). The content of C was relatively the same in both zones. It proves the selective affinity of B to Cr (in absence of Zr).

<table>
<thead>
<tr>
<th>Zone</th>
<th>B</th>
<th>C</th>
<th>O</th>
<th>Cr</th>
<th>Fe</th>
<th>Ni</th>
<th>Mo</th>
</tr>
</thead>
<tbody>
<tr>
<td>Green</td>
<td>7.1 (24.0)</td>
<td>6.1 (18.5)</td>
<td>0.3 (0.7)</td>
<td>14.2 (10.0)</td>
<td>62.1 (40.6)</td>
<td>9.8 (6.1)</td>
<td>0.2 (0.1)</td>
</tr>
<tr>
<td>Red</td>
<td>14.8 (40.2)</td>
<td>6.9 (16.8)</td>
<td>0.4 (0.6)</td>
<td>41.3 (23.4)</td>
<td>34.1 (18.0)</td>
<td>0.9 (0.4)</td>
<td>1.7 (0.5)</td>
</tr>
</tbody>
</table>

3.2 Blockage region (BLM)

The melt originated from the eutectic interaction of stainless steel and B$_4$C in a hot region of the bundle proceeded downwards along the control blade at first without contact with channel box. But very soon the amount of melt was enough to fully occupy the space between the channel box and control blade. It happened in the region where the temperature was close to the freezing point of melt 1447 K.
There was a big difference observed in behavior of a preliminarily oxidized channel box (Fig. 6, zone “b”) and another unprotected zone at the same elevation (Fig. 6, zone “e”) (Pshenichnikov et al., 2019d). In the zone “b”, after melt had covered the surface the oxide layer thickness was enough to protect the channel box from a melt-through. The zone “e” was created artificially. It was the place where an alumina insulation of a thermocouple attached the surface of the channel box throughout the test. Oxidation of the place was significantly reduced, giving the possibility to a local development of a eutectic interaction of Fe melt with Zircaloy-4.

The detailed investigation of the melt attack was performed by EDS-analysis (Fig. 7). On the SEM image one can distinguish three layers containing melt elements. A penetration depth correlated to their common diffusion abilities (Cr - lowest, Fe - intermediate, Ni - highest). Surface layer of the oxidized channel box contained elements from the melt 13 µm deep inside (Fig. 7, OxL-1). The layer OxL-2 contained mostly Ni and Fe. The layer OxL-3 stands for ZrO₂ layer. Only alone-standing penetrations of Ni and Fe were detected. The Fig. 7 Zr+Sn overlay showed that Ni and Fe penetrated in the oxide mostly using these Sn-rich points and cracks in oxide layer. Though Fe+Cr+Ni have been found in the oxidized channel box, the Zr atoms could be hardly found in the surrounding melt.

![Fig. 7 Appearance of polished cross section of protective oxide layer that was in contact with melt detected after CLADS-MADE-01 test in blockage (BLM) region; a) SEM-image, b) Fe, Cr, Ni EDS-maps overlay, c) O, Fe, Ni EDS-maps overlay, d) Zr, Sn EDS-maps overlay.](image-url)
Fig. 8 Penetration of an insufficiently oxidized channel box wall by SS/B₄C melt detected after CLADS-MADE-01 test in blockage (BLM) region; a) SEM-image, b) Fe EDS-map, c) Zr EDS-map, d) Cr EDS-map.

In absence (or insufficient thickness) of oxygen-containing layers, a local formation of Fe-Zr eutectic was possible (Fig. 8). Intensive penetration of Fe and Cr into metallic Zircaloy-4 has been observed resulting in a round-shaped front of Fe+Cr+Ni diffusion into Zr metal starting right from the end of existing oxide. It seems from Fig. 8 b, it was difficult for Fe+Cr+Ni elements to penetrate even an α-Zr(O) layer. Thus oxygen-containing surface layers at the temperature of Fe-B eutectic melting remain protective. This example showed that penetration of oxygen-containing layer α-Zr(O) by SS/B₄C eutectic melt under accident conditions was 100 times slower in comparison to an unprotected region. Penetration of Zr oxide at the same conditions can be considered as negligible at the temperatures and time ranges of the beginning phase of an accident.

3.3 Furnace bottom region (CR)

Below the blockage, all structures have retained their initial geometries (Fig. 9a). No melting of materials happened in this area. Black layer of Zr oxide covered claddings and channel boxes. In fact, the temperature in CR never exceeded 1373 K. However, molten material was detected here as well. Along the control blade and channel boxes there were single solidified rivulets detected.

Fig. 9 a) Appearance of the slightly oxidized part of the bundle between 0 mm and 290 mm from the bottom of the bundle; b) melt accumulates at the sample supporting plate relocated directly from blockage and c) solid debris with molten edges relocated directly from the top part of the control blade.
At the bottom of the furnace (Fig. 9b), there was a significant part of metallic material with the same composition as blockage melt relocated by free fall. Free fall mechanism was confirmed by the in situ video and because the final location was not connected to the control blade. Also on the top of the relocated melt massive previously non-molten debris of the control blade (Fig. 9c) have been found. They originated from the top part of the fuel assembly, and relocated only when mechanical stability was lost due to melting of the structures beneath them.

4. Discussion

The relocation of debris from damaged reactor pressure vessel to containment is very complicated process involving understanding of materials interaction at all stages because some eutectic reactions can decrease the failure temperature of components and cause localized melting. This melting might change the progression of the accident. Generally, the results of the present study are in accordance with all previous studies performed in EG&G Idaho, Inc., SNL, KfK and JAEE. The new CLADS-MADE-01 test complemented them providing new information on effects accompanying melting and oxidation of a BWR assembly with a control blade under specific conditions of steam starvation, which had not been sufficiently studied before. In particular, melt relocation features: onset, progress, candling, blockage formation and its subsequent remelting were directly observed in situ using video cameras.

Melting of the test assembly started as expected from a Fe-B eutectic formation: at \( T \approx 1463 \) K large control blade part was involved into melting and relocated fast downwards from the hot region, leaving behind only high-melting point remnants. In the hottest zone up to max 1750 K (UP zone) there were severely oxidized Zr parts, \((\text{Cr,Fe})_x(\text{B,C})_y\) mixed carboboride phases encapsulating partially intact \( \text{B}_4\text{C} \) granules. Maximum temperature of the test did not allow oxide and carboboride melting. Encapsulation of \( \text{B}_4\text{C} \) by absorber melt and steam-starved conditions did not allow significant \( \text{B}_4\text{C} \) oxidation. Thus, most of \( \text{B} \) and \( \text{C} \) would be involved into melting and stabilization of boride and carbide phases. The main reason why remnants did not melt in the hot region of the bundle because of selectiveness of \( \text{B} \) interaction with Cr. Thermodynamically it is logical because of higher bonding energy of Cr-B compound than Fe-B. Cr-B compounds when formed, should solidify first, because of high melting point of all Cr-B compounds in comparison to Fe-B compounds. In such way, Cr-rich compounds became solid, while Fe-rich phase could be still liquid and relocate to lower elevations or penetrate deeper between granules by a capillary force (supported by the Fig.4 and 5). This liquefaction-solidification mechanism played the most important role in stabilization of degrading control blade remnants in the hottest elevation, where they supposed to be melted. Same effect of Cr-rich layer formation would happen everywhere in case of temperature for liquefaction of Fe material exceeded in contact with \( \text{B}_4\text{C} \) granules. Taking into account, that the control rod structures contained intact \( \text{B}_4\text{C} \) granules that was reported in (Pshenichnikov et al. 2019b), it seems that the direct reaction of \( \text{B}_4\text{C} \) with hot steam at the early phase of an accident in Japanese BWRs is negligible.

The BLM region consisted of metallic melt formed between channel boxes in the temperature range around \( 1447 \) – \( 1520 \) K, while high-temperature materials like oxides and B-rich phases of Fe and Cr remained in the upper part, waiting for the temperature increase. Thus, observed phenomenon is similar to the one observed in OECD LOFT Experiment LP-FP-2 performed by EG&G Idaho, Inc. (Coryell et. al., 1994). There were two melt pools observed during the core degradation test – oxidic and metallic one. The metallic melt will propagate downwards always in front of oxidic melt because of differences in their melting temperatures until they both meet some obstacle, for example, RPV lower head, where mixing of these two pools becomes inevitable. Though their test has been performed with different absorber materials and PWR geometries, plus Zr was involved in relocation, it allowed general understanding of the fact that in most cases the isotherms and material interactions are important when describing core relocation.

The difference of 100 times slower penetration of oxidized layer is remarkable and can be the evidence, that channel box can only be molten-through during the beginning of a severe accident if oxide layer and \( \alpha\text{-Zr(O)} \) both were self-dissolved in available metallic Zr similar to that described in (Stuckert et al., 2002) or did not exist (Pshenichnikov et al., 2019d). It also proves the possibility, that in case of a thin oxide layer an amount of Zr involved into the Fe-Zr eutectic melt can become quite large. The amount of available Fe in this case is the limiting factor. Anyway, preliminarily oxidized channel boxes would effectively resist melt-through. If the cannal boxes do not allow melt-through, originated metallic melt would probably not create one big metallic pool. Instead of that melt will individually propagate along each control blade channel until it reaches the bottom. However, in some special case under reducing conditions and oxidation below threshold value a channel box melt-through can happen in a BWR.
involving large volume of Zr into melting, thus creating a metallic pool same as for PWRs. The thickness of the oxide layer enough to prevent the melt-through is still not clear. The threshold value that prevents penetration of channel box by ferrous melt is now under investigation. This individual melt relocation is one of the main differences between BWRs and PWRs scenarios at the beginning phase of the accident. However, it has never been explained, that it works only in the case if the melt pool is metallic. The case of oxidic melt will be one big melt pool formation for both PWR and BWR in any case because of high temperature.

Having reached the lower RPV head, metallic melt solidifies and waits for the next temperature increase that could come together with the second wave of the oxidic melt pool. As long as there is a gap between the oxidic and metallic melt pools because of temperature gradient the metallic melt should become liquid before they mix. Thus being at the bottom of RPV, heated up by oxidic melt it attacks welding joints of control rod guide tube nozzles, where first melt come-out of the RPV should happen. First melt may have low concentration or even not contain Zr and U until the oxidic pool is going to be involved into the common pool at the bottom of RPV. In this case, metallic pool can be pushed due to density difference to the upper surface of common pool and would start side attack of the RPV wall due to “heat flow focusing” effect. The lower head is going to be finally filled with hot melt and is going to breach releasing molten material. These effects are highly-likely to happen during the accident at the unit 2 of 1F.

5. Conclusions

As a result of the study the following conclusions can be made:

- A BWR control blade degradation test included a phase of steam starvation using prototypic materials and geometries with preliminarily oxidized test bundle has been performed.
- In situ video captured during the test allowed detailed investigation of a control blade degradation sequence and formation of debris.
- Post-test investigations suggest that preliminary oxidation history should be taken into account during analysis of a severe accident in BWRs, because a change of an accident sequence in a less oxidized core can move an accident progression towards a PWR scenario in the sense of a common metallic pool formation.
- If channel box survives, Zr is not involved in the metallic part of melt, promoting \((\text{Fe}, \text{Cr})_x(\text{B}, \text{C})_y\) phases in the blockage. The precise composition of these phases should be additionally investigated. But even now it is clear that it is an additional challenge for decommissioning because of their extreme hardness.
- A consideration of two molten pools – metallic and oxidic, and investigation of their propagation and contact with lower RPV head can uncover the possible mechanism of RPV damage in 1F reactors.

Acknowledgements

A part of this study was performed in the framework of “Advanced Multi-Scale Modeling and Experimental Tests on Fuel Degradation in Severe Accident Conditions” supported by Ministry of Economy, Trade and Industry, Japan. Help of Dr. H. Pham and Dr. S. Yamazaki from JAEA/CLADS during the test is gratefully acknowledged.

References

Coryell, E.W., Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2 (1994), Report No. NUREG/CR-6160, ID, USA.
Gauntt, R. O., Gasser, R. D., Results of the DF-4 BWR Control Blade-Channel Box Test (1990), Report No. SAND-2716C, NM, USA.,
Hofmann, P., Markiewicz, M. E., Spino, J. L., Reaction Behavior of B\textsubscript{4}C Absorber Material with Stainless Steel and


International Research Institute for Nuclear Decommissioning (IRID) and the Institute of Applied Energy (IAE), Core Status Evaluation Project for Decommissioning of the FDNPS (2018), Tokyo, Japan.


