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Core melting in the first PHEBUS FP experiment

Livolant, M.[1], Holtbecker, H.F.[2], Adroguer, B.[3], Reppetto, G.[3], Jamond, C.[3]
1) IPSN/CEA, CE-Fontenay-aux-Roses, France
2) European Commission - Joint Research Centre, Ispra, Italy
3) IPSN/CEA, CE-Cadarache, France

Abstract : The aim of the international PHEBUS FP Programme is to investigate in an in-pile facility various phenomena governing degradation of fuel as well as release, transportation and deposition of fission products in LWR severe accident conditions. This paper gives a summary on the current analysis of the core degradation aspect of the first FPT0 test using ICARE2. Results show a core degradation far beyond any other test that has been performed in the past. Weaknesses in current models to predict severe degradation are identified regarding the late phase aspects such as loss of rod-like geometry and molten pool formation.

1. Introduction

The PHEBUS FP program [1] is a light water reactor severe accident research program, initiated by the 'Institut de Protection et de Sûreté Nucléaire' (IPSN) of the 'Commissariat à l'Energie Atomique' (CEA) and the Joint Research Centre of the European Commission (EC). The program is also supported by NRC (USA), NUPEC and JAERI (Japan), COG (Canada) and KAERI (Corea). All European Union research organisations participate to the interpretation of the results.

The aim of the experimental program is to study during a PWR severe accident the core degradation and the release of the fission products (FP) and their behaviour in the reactor circuit and the containment vessel. This program consists in six in-pile tests [2].

The specific objective of the core aspect of the first test FPT0 was to investigate FP release from trace irradiated PWR fuel under severe accident conditions characterised by significant fuel rod degradation and melting under low pressure and oxidising conditions. Objectives were fulfilled and FPT0 shows a degradation progression far beyond any other global experiment that has been performed up to now [3] : PBF SFD, PHEBUS SFD, CORA, FLHT, LOFT-FP-2.

This paper describes preliminary results and the current status of the analysis of core degradation using the mechanistic ICARE2 code which is under development by IPSN/CEA. A first validation of ICARE2 has been done on the previous PHEBUS SFD Programme devoted to study the early phase of core degradation up to 2800 K [4].

2. Description of the PHEBUS FP facility and of the test device

2.1. PHEBUS FP facility

It consists mainly of a driver core, a FP circuit and a pressurised water loop [5] :
- the driver core (rods of 0.8 m fissile length) supplies the test bundle with nuclear power.
- the FP circuit allowing to reproduce the thermal hydraulics conditions in case of severe fuel damage accident. The main components are : a test bundle device, the FP circuit with a U tube (steam generator simulator) and a containment tank.
-the pressurised water loop is used as a cooling circuit of the bundle during the irradiation phase and of the external pressure tube during the degradation phase.

2.2. Bundle

The test device located in a cell on the vertical axis of the reactor (**Fig 1**) consists in a bundle of 20 fresh fuel rods (10.5 kg of UO_2) 1 meter long with zircaloy claddings and a central absorber rod (80% Silver-15% Indium-5% Cadmium) with stainless steel cladding housed in a zircaloy guide tube. The in-pile device involves from the bottom to the top : the foot valve (allowing to shift from irradiation phase to degradation and FP release conditions), the bundle and the vertical line (beginning of FP circuit). All elevations are referenced to the bottom of the fissile length in the bundle (**Fig.1**).

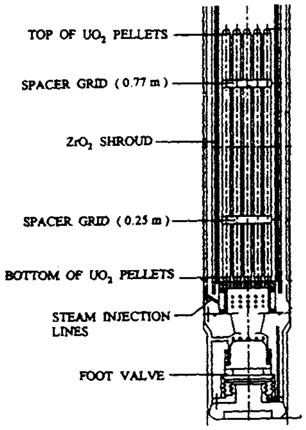


Figure 1 : Axial cross section

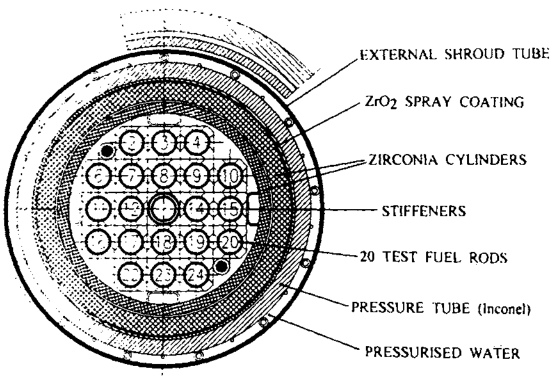


Figure 2 : Radial cross section

The 21 rods are maintained by two zircaloy spacer grids located on either side of the mid plane and four zircaloy stiffeners. The bundle is surrounded by two stabilised ZrO_2 cylinders and a pressure tube of inconel (6 mm) coated on its internal face by a spray of dense zirconia (1mm) (**Fig.2**).These three structures are separated by two gaps in cold conditions.

2.3. Instrumentation and test parameter monitoring

The total power generated in the fuel assembly is deduced from the core power measurement and the coupling factor between the driver core and the test bundle. Thermal hydraulic conditions such as inlet gas flow rate, H2 production and pressure are measured at different locations in the circuit and containment tank [6].

Different kinds of thermocouples at several radial and axial locations allowed temperatures measurements of fuel centrelines, guide tube, stiffeners, fluid (with W/Re TCs), and shroud (with K type TCs). These TCs were developed for the PHEBUS CSD program [7]. Up to 2700 K, the heat up of the fuel rods was measured by bundle TCs. After their failure, the heat up was controlled by shroud TCs.

An On Line optical Aerosol density Monitor (OLAM) is incorporated in the horizontal part to the experimental FP circuit. This instrument measures the attenuation of light across two path lengths during the bundle release phase. Its has been adapted by EG & G, Idaho Falls, from a similar instrument used in PBF test SFD 1-4 [8].

2.3. Gamma-scan, radiography, tomographies, destructive examinations

The purpose of the non-destructive and destructive examinations is to quantify the core damage. First the test train is gamma scanned over the entire length to identify axial profiles of fuel (La140, Nb95, Zr95...) and activation products (Ag110m, Cr51, Co60...). The radiography and tomographies performed fairly well and enabled a rapid and very comprehensive view of the severe degradation of the bundle. Then, after filling the bundle with epoxy, destructive examinations enable cross cuttings for detailed identification of materials, interactions, atomic composition of melts, porosity... This information will be available in the second part of 1995.

3. FPT0 Conditions

Irradiation phase

In order to provide fission products, the bundle was irradiated during 213 h (from November 21st to November 30th 1993 in the PHEBUS reactor) at powers ranging from 230 to 250 kW to provide fission products. After shutdown and before the experimental phase, a transition phase of 36 hours was necessary for Xenon override and preparation of the circuits.

Experimental phase

The transient was performed (December 2nd 1993) in four parts **(Fig 3)**:

- the 1st thermal calibration part enables the thermal response of the bundle to be checked. Comparison with the pre-test calculation enabled the power increase to be adjusted.
- the 2nd part (10000 - 13700 s) was the oxidation phase monitored by step increases of the power and large increase of the mass flow rate (from 0.5 to 3 g/s).
- the 3rd part (13700 - 18138 s) was the heat up phase during which fuel rod reached a maximum temperature of about 3100 K according to an extrapolation from shroud temperatures. This phase was performed by changing the thermal hydraulic conditions and increasing the nuclear power. During the last 70 seconds of this period, different degradation events were observed which led to the shutdown of the reactor at t=18138s. The maximum bundle power reached was about 50 kW.

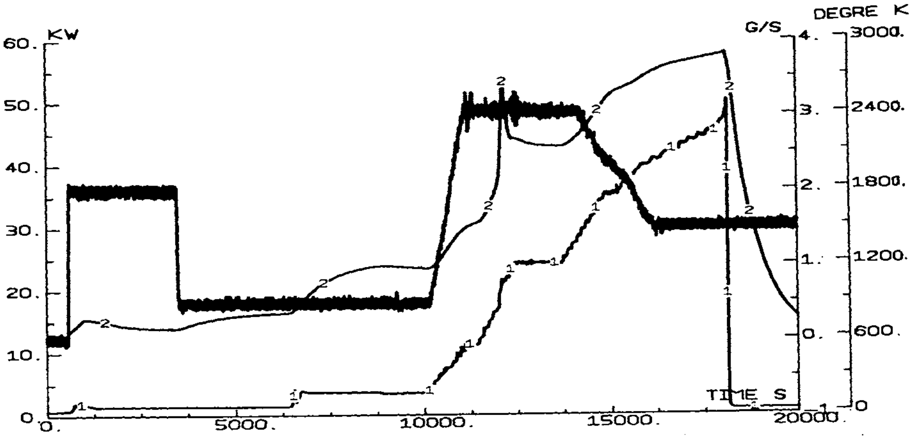


Figure 3 : Bundle power, mass flow rate scenario and calculated fuel temperature (0.7 m)

- the 4th part (duration : 40 mn) was the cooling phase. The steam flow rate was maintained to transport FPs and aerosols to the outlet circuit.

4. Measured Bundle Degradation Behaviour

The analysis of measurements in particular the two qualitative signals from the OLAM (**Fig. 4**) led the main degradation events to be identified as follows :

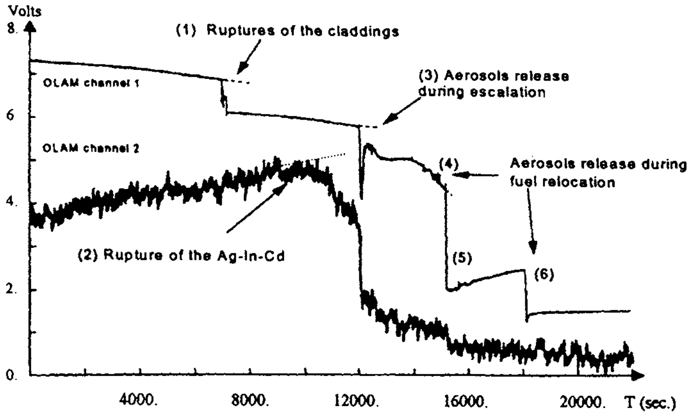


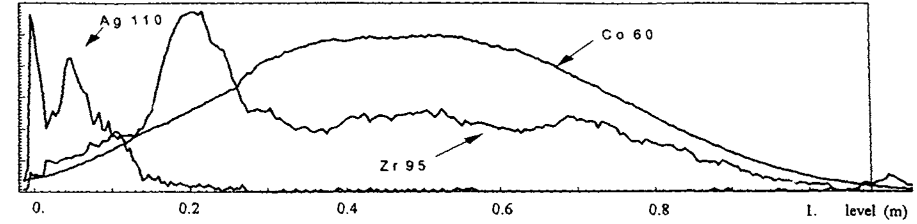
Figure 4 : OLAM signals
Channel 1 : Signal through pipe diameter
Channel 2 : Signal through partial diameter

Event 1: Clad failure of pressurised rods during the calibration phase at ~ 1050 K
Event 2: Control rod failure around 1500 K followed by an increase of aerosol and Indium release (gamma spectrometers in the FP line) when melting of steel is reached (~ 1700 K)
Event 3: Clad oxidation escalation at ~ 12000 s, Zr melting, large aerosol production
Event 4: Relocation of fuel rod materials to lower elevations after 14500 s
Event 5: Large aerosol production at 15200 s probably due to a large flow area blockage
Event 6: Final fuel motion just before the reactor shut down.

The gamma-scanning results are shown in the **Fig. 5**. The Cobalt-60 coming from the activation of the Inconel tube shows the irradiation axial flux profile. The axial distribution of the Zr-95 (fuel material indicator) and the Ag 110 m (activation of control rod silver) show material relocations by comparison with Co 60. In particular control rod material is mainly accumulated under 0. 1 m.

The radiography (**Fig. 6**) and the 52 transmission tomographies gave respectively a general picture and very detailed information about the final state of the degraded bundle. Five main configurations are observed :
1 - lower 0 - 0.15 m zone : almost intact bundle with some relocated materials
2 - just under the lower spacer grid over 0.1 m length, an ingot of frozen materials (**Fig. 7**) composed with a dense peripheral crust and a central part with porosities
3 - over 0. 12 m above the lower grid, only a peripheral crust is remaining
4 - over 0. 38 m under the upper grid, only damaged external rods are remaining
5 - above the upper grid over 0.26 m length, rods with significant UO_2 dissolution (**Fig. 8**)

Figure 5 : Gamma-scanning



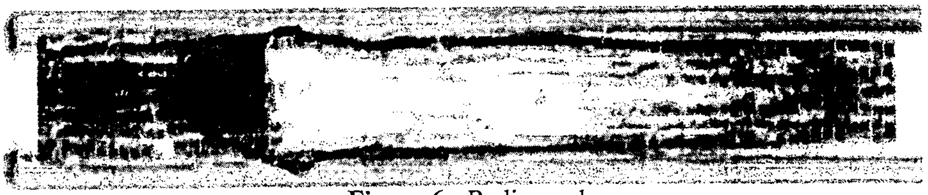


Figure 6 : Radiography

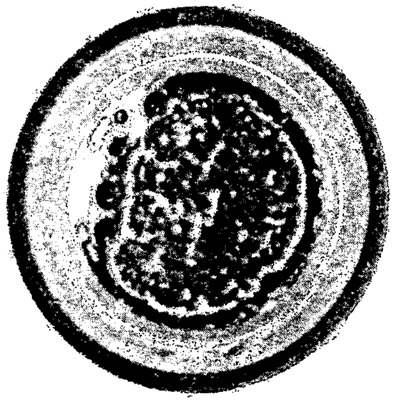
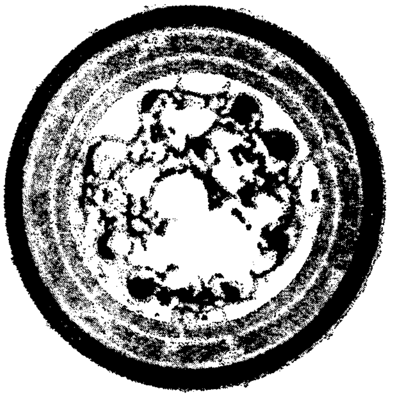



Figure 7 : level 0. 2 m

Figure 8 : level 0. 87 m

All these data show large fuel dissolution in the upper part (which was not expected at so high level in oxidising atmosphere) and large fuel rod relocation with local UO_2-rich pool formation (~ 20 % of UO_2 inventory). This large bundle material accumulation led to a short and large flow area blockage on the lower grid. A detailed analysis of the tomographies show that about 50 % of the fuel inventory was relocated either in solid or molten states. Post irradiation examinations will give more details and should confirm the bundle degradation scenario.

5. Status of interpretation

The interpretation effort is currently carried out within the framework of the PHEBUS FP Scientific Analysis Working Group (SAWG). Code degradation results and calculations are shared and discussed between each organisations in a " Bundle Interpretation Circle ". Main codes used are ICARE 2, ATHLET-KESS, SCDAP/R5, MAAP and MELCOR.

In this section the current status of the ICARE2 post-test calculations performed by CEA-IPSN are presented using the V2 Mod1 version. This code was extensively used to prepare the test [9]. A reference post-test calculation (ICARE2 - Ref.) performed with actual boundary conditions and revised material properties is presented. Additional sensitivity studies have been done in particular for identification of the damage progression. The bundle was modelled with 11 axial meshes, and different components representing inner and outer rods, the Ag-In-Cd control rod with its Zr-guide tube, the two spacer-grids and the shroud with the two ZrO_2 insulating layers, the two annular gaps and the external inconel tube **(Fig. 2).**

Thermal and oxidation behaviour

Calculated and measured temperatures are shown on **Fig. 9 to 11** at 0.8, 0.4 and 0.3 m. These figures show fuel and ZrO_2 shroud temperatures at two radial positions : middle and outer surface of the external ZrO_2 insulating layer. The two first calibration plateaux performed before 6000 s are accurately predicted but are not considered in these figures. Comparisons start at the beginning of the 3rd plateau at 6000 s.

Fuel rod temperatures : After calculation of the cladding burst as in the test, the correct prediction of the 3rd plateau shows that global heat losses through the shroud are correctly modelled. Energy balance calculations show that most of the nuclear power is dissipated through the shroud : ~ 80 % from the 3rd plateau up to the final heat-up phase except during the strong clad oxidation escalation period observed in the 0.4 - 1 m zone between 12000 and 12400 s. During this early strong heating period (~ 10 K/s) which is correctly calculated, the oxidation power is twice the nuclear power and melting point of non oxidised Zr is reached before total oxidation. Fuel TCs operated up to 2700 K (maximum temperature measured at 0.7 m). The escalation front is correctly calculated to move down rapidly from the hot zone (0.7 - 0.6 m) to 0.4 m at 12400 s. Then the slower progression is also correctly calculated from 0.3 m at 14500 s to 0.2 m at 16000 s. No steam starved condition is calculated. After Zr melting, ICARE2 continues the Zr oxidation with simultaneous UO_2 and ZrO_2 dissolutions.

A correct calculation of the peak temperature was obtained in the 0.55 - 0.75 m zone using a revised clad criteria [10] based on a temperature limit ($T^* = 2700$ K). With this conditions, the U-Zr-O melts created between the two grids remained entrapped in the rods and produced local oxidation heating.

Shroud temperatures : After 14800 s, only shroud TCs survive. Comparisons indicates :
- a correct agreement at 0.8 m **(Fig. 9)** up to 17300 s
- a progressive overestimation between 0.4 and 0.7 m from 14800 s to 15200 s. Then, the large overestimation is maintained up to the end of test **(Fig. 10)**
- a significant underprediction at 0.3 and 0.2 m after 15200 s **(Fig. 11)**.

Sensitivity studies aimed at checking uncertainties on the shroud conductivity and external gap closure could not explain these discrepancies. As discussed latter, this is indicative of the large core material relocation observed in the test after 14800 s near the lower grid which is not calculated.

Hydrogen production : Calculated and measured total H_2 production indicate the same trend ; a rapid production rate corresponding to the early oxidation and a large amount of H_2 generated at 12400 s escalation (~ 70% of the H_2 produced by total oxidation of the Zr inventory of the bundle).

A detailed analysis of H_2 production indicates that during the early oxidation escalation (12000 - 12400 s), a small amount of Zr was melted before total oxidation. The resulting UO_2 liquefaction by molten Zr occurred mainly above the upper grid as confirmed by local tomographies of the bundle. The code-to-data comparison of the H_2 rate derived from the thermal response of the downstream steam generator tube is given on **Fig. 12**. It is observed a correct calculation of both the early H_2 production (oxidation of the hot zone between 12000 - 12400 s and late H_2 production (oxidation of lower levels between 14500 and 16000 s).

Control rod degradation

The control rod failure was detected at 10780 s (event 2 on **Fig. 4**). This event occurred for temperatures of about 1500 K. Later, when stainless steel melting point was reached (~ 1700 K) at 11900 s, large aerosol (event 3 on **Fig. 4**) and Indium release were detected. The first failure is probably due to a local clad-guide tube contact (steel-Zr eutectic) and cannot be calculated by current version of ICARE2. The following control rod degradation with large Ag-In-Cd melt relocation and aerosol production is correctly calculated. The Ag-In-Cd rod degradation progress with the oxidation escalation front. Total disappearance is predicted above 0.15 m at 16000 s. Ag-In-Cd mixtures are predicted to be relocated too far under the bundle. Part of these mixtures remains in FPT0 in the 0.1 m zone. **(Fig. 5)**

Fuel rod degradation

The bundle was more damaged than predicted by ICARE2 and other codes in the pre-calculations. The current version has no specific model able to predict the loss of rod-like geometry and the formation of a molten pool. Current models are limited to predict the formation of U-Zr-O eutectics or pure ZrO_2 or UO_2 melts with flow-down on the external surface of remaining in-place rods. Nevertheless the large flexibility of the code has enabled to analyse and understand the main stages of the bundle degradation by forcing the code to calculate user-imposed degradation configurations. Main results are :

(a) Above the upper spacer-grid and during the early oxidation escalation ICARE2 predicts Zr melting, local UO_2 dissolution (~ 7.5% at 0.9 m) and resulting U-Zr-O melt relocation on the upper grid.

(b) The Zr-rich melt which contains the rod upper plugs, remains trapped on the grid and induces locally large UO_2 dissolution (25 % at 12600 s). This is consistent with upper tomographies which shows significant local UO_2 dissolution **(Fig. 8)**. Grid-trapping by upper grid is necessary to have a correct prediction of local shroud temperatures at 0.8 m **(Fig. 9)**. Oxidation of trapped Zr-rich melts affect the local thermal behaviour of rods and structures.

(c) Between the grids, there is no sufficient UO_2 dissolution to predict large quantities of U-Zr-O melts. These melts are calculated to remain in place and the rod-like geometry is calculated unchanged up to the end to test. This incorrect prediction underlines the main weakness of the code.

Sensitivity calculations showed that if the disappearance of inner rods located in the 0.4 - 0.7 zone is forced by the user at 15000 s, then calculated shroud temperatures are in better agreement with measurements **(Fig. 10).**

Other ICARE2 calculations with a molten pool imposed at 14800 s (inner rods relocated in the 0.25 - 0.32 m zone) showed more correct shroud temperatures at 0.3 m **(Fig.11)**.

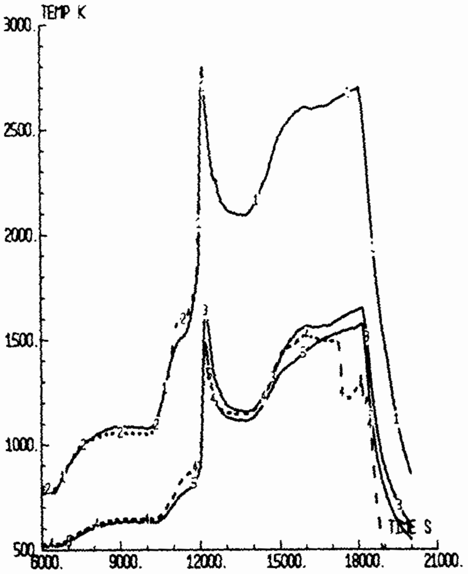


Fig. 9 : Fuel and inner shroud temp. : 0.8 m
Measurements : (2) fuel (4) shroud
ICARE 2-Ref : (1) fuel, (3) shroud
ICARE 2: (5) Shroud (no grid trapping effect)

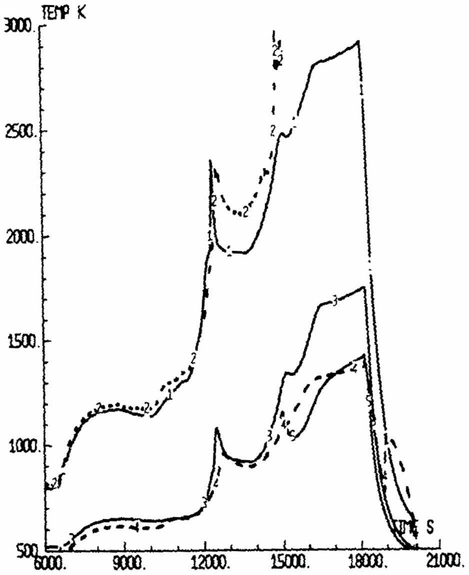


Fig. 10 : Fuel and inner shroud temp. : 0.4 m
Measurements : (2) fuel, (4) shroud
ICARE 2-Ref : (1) fuel, (3) shroud
ICARE 2: (5) Shroud (without inner rods)

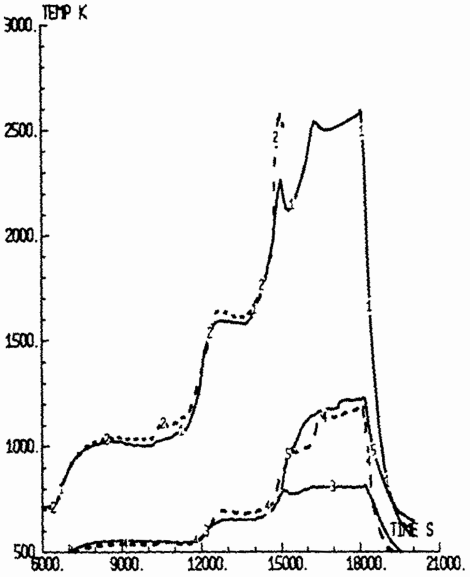


Fig. 11 : Fuel and outer shroud temp. : 0.3 m
Measurements : (2) fuel, (4) shroud
ICARE 2-Ref : (1) fuel, (3) shroud
ICARE 2: (5) Shroud (molten pool case)

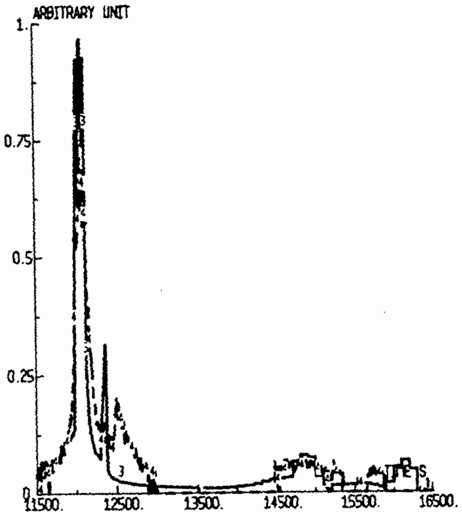


Fig. 12 : H_2 production rate
Measurements : (4)
ICARE2-Ref calculation : (3)

(d) The large flow blockage near the lower grid could induce reduction of the local convective rod-fluid heat transfers. Sensitivity calculations confirmed this effect and enabled correct rod and shroud temperatures to be calculated at 0.2 m.

6. Current understanding of the core degradation and discussion

The current understanding of the degradation progression up to 18068 s is as follows :
(a) 12000 - 14800 s : During and just after the early oxidation period, an apparently quite intact rod-like geometry is remaining (i.e. no fuel relocation). Significant UO2 dissolution occurred mainly above the upper grid.
(b) 14800 - 15200 s : Loss of rod-like geometry of inner ring rods located between the two grids and relocation on the lower grid with formation of a local molten pool.
(c) 15200 - 18068 s : The molten pool is maintained in place, there is no significant fuel relocation.

Between 18068 s and 18138 s, a final fuel motion was observed (event 6 on **Fig. 4**). It induced an unexpected increase of core reactivity and a downward relocation of the molten pool. This final period is under study.

The period (a) is the transition between the early and late phase of core degradation. Its understanding is crutial for modelling purposes. Codes usually do not calculate this transition correctly and FPT0 in this area brings unique data on the loss of geometry of rods with oxidised clads. This period was characterised by a nearly progressive degradation-relocation of inner ring rods and partially lower outer rods with UO_2 - rich melt accumulation on the lower grid. Different processes could explain this loss of geometry particularly : (a) Dissolution and melting of rods by external relocated Zr-rich melts previously refrozen between the two grids, (b) Embrittlement by the thermal chock of the oxidation escalation followed by a rubble bed formation which induces melting and relocation of UO_2, ZrO_2 debris.

7. Preliminary conclusions

The current FPT0 results indicate that large rod degradation occurred far beyond those observed in any previous tests. Major events such as sharp clad oxidation escalation, melting of non oxidised Zr and UO_2 dissolution, large fuel rod relocation and local molten pool formation were observed with significant volatile FP release (above 50 % of the inventory) from the trace-irradiated fuel. The large rod relocation and molten pool formation were unexpected and understanding the progression of these events, in particular the total loss of rod like geometry is our current challenge.

The ICARE2 code is extensively used for the analysis of experimental results. This code validated for the prediction of the early phase of core degradation ($T < 2800$ K, limited degradation) could not calculate the large fuel rod relocation observed on the bundle tomographies. Main deficiencies concern the loss of rod-like geometry and the lack of transition toward a core debris configuration. In these areas, new development efforts are promoted.

Final analysis of FPT0 is waiting the completion of destructive post-irradiation examinations and will integrate all the analysis work of the PHEBUS FP partners. These results should bring valuable information on the core degradation progression up to a complete loss of geometry and on the strong coupling between core degradation, FP release and aerosol production.

Concerning the bundle design for the future test FPT1 (same objectives using irradiated fuel coming from BR3), improvements of instrumentation and modifications of the shroud design are foreseen (in particular using ThO_2 as first ring of the thermal shielding).

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