Nuclear Engineering and Design 283 (2015) 8–20

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

A comparison of core degradation phenomena in the CORA, QUENCH, Phébus SFD and Phébus FP experiments

T. Haste[a,*], M. Steinbrück[b], M. Barrachin[a], O. de Luze[a], M. Grosse[b], J. Stuckert[b]

[a] Institut de Radioprotection et de Sûreté Nucléaire, IRSN, BP 3, F-13115 St. Paul-lez-Durance Cedex, France
[b] Karlsruhe Institute of Technology (KIT), Hermann-von-Helmholtz-Platz 1, D-76344 Eggenstein-Leopoldshafen, Germany

HIGHLIGHTS

- The results of the experiments CORA, QUENCH and Phébus SFD/FP are summarised.
- All phenomena expected up to melt movement to the lower head are shown consistently.
- Separate-effect tests performed at KIT and IRSN aid improve their modelling.
- Data from the integral tests help independent validation of new and improved models.
- The improved codes will help reduce uncertainties in safety-critical areas for core degradation.

ARTICLE INFO

Article history:
Received 27 February 2014
Received in revised form 27 June 2014
Accepted 29 June 2014

ABSTRACT

Over the past 20 years, integral fuel bundle experiments performed at IRSN Cadarache, France (Phébus-SFD and Phébus FP – fission heated) and at Karlsruhe Institute of Technology, Germany (CORA and QUENCH – electrically heated), accompanied by separate-effect tests, have provided a wealth of detailed information on core degradation phenomena that occur under severe accident conditions, relevant to such safety issues as in-vessel retention of the core, recovery of the core by water reflood, hydrogen generation and fission product release. These data form an important basis for development and validation of severe accident analysis codes such as ASTEC (IRSN/GRS, EC) and MELCOR (USNRC/SNL, USA) that are used to assess the safety of current and future reactor designs, so helping to reduce the uncertainty associated with such code predictions.

Following the recent end of the Phébus FP project, it is appropriate now to compare the core degradation phenomena observed in these four major experimental series, indicating the main conclusions that have been drawn. This covers subjects such as early phase degradation up to loss of rod-like geometry (all the series), late phase degradation and the link between fission product release and core degradation (Phébus FP), oxidation phenomena (all the series), reflood behaviour (CORA and QUENCH), as well as particular topics such as the effects of control rod material and fuel burn-up on core degradation. It also outlines the separate-effects experiments performed to elucidate specific phenomena such as the impact of chemical reactions involving boron carbide absorber material. Finally, it indicates the remaining topics for which further investigation is still required and/or is under way.

Abbreviations: ACM, advanced cladding materials; AIC or Ag/In/Cd, silver–indium–cadmium; BWR, boiling water reactor; CEA, Commissariat à L'Energie Atomique; EC, European Commission; FP, fission product; GRS, Gesellschaft für Anlagen- und Reaktorsicherheit; IET, Institute for Energy and Transport; IRSN, Institut de Radioprotection et de Sûreté Nucléaire; JRC, Joint Research Centre; KIT, Karlsruhe Institute for Technology; LWR, light water reactor; PWR, pressurised water reactor; SFD, severe fuel damage; SNL, Sandia National Laboratory; TMI-2, Three Mile Island Unit 2; USNRC, United States Nuclear Regulatory Commission; VVER, Vodo-Vodianoï Energuetitcheski Reaktor (PWR of Russian design).

* Corresponding author at: Institut de Radioprotection et de Sûreté Nucléaire (IRSN), PSN-RES/SAG/LETR Bât 702, BP 3, F-13115 St. Paul-lez-Durance Cedex, France. Tel.: +33 4 42 19 95 67; fax: +33 4 42 19 91 67.

E-mail addresses: tim.haste@irsn.fr (T. Haste), martin.steinbrueck@kit.edu (M. Steinbrück), marc.barrachin@irsn.fr (M. Barrachin), olivier.de-luze@irsn.fr (O. de Luze), mirco.grosse@kit.edu (M. Grosse), juri.stuckert@kit.edu (J. Stuckert).

http://dx.doi.org/10.1016/j.nucengdes.2014.06.035
0029-5493/

1. Introduction

Since the TMI-2 accident in 1979 (Broughton et al., 1989) fuel degradation under severe accident conditions has been studied extensively in integral and separate-effects experiments, programmes which continue to this day. This paper compares the main results obtained in four major integral test programmes, the Phébus SFD and FP nuclear-heated experiments performed in-reactor at Cadarache, and the out-of-reactor electrically heated CORA and QUENCH series performed at Karlsruhe Institute of Technology (KIT, formerly KfK, FZK). These experiments are generally of similar scale, with typically 20–25 fuel rods of up to about 1 m heated length enclosed radially by an insulating shroud, usually with control rod material present (Ag/In/Cd (AIC) or B_4C), though a few CORA tests were carried out at roughly twice the radial scale (up to 57 rods). Many of the KIT tests also involved water reflood.

After summarising the experimental programmes, this article concentrates on the main overall conclusions so far drawn concerning the main phenomena observed in these major programmes. Brief notes are provided on supporting separate-effects tests that help to understand the phenomena in more detail and form the basis for model development, with the integral data forming the basis for their assessment. Finally, an indication is given of work still in progress and proposed to help complete understanding of core degradation, essential for reliable estimation of fission product release and potential source terms to the environment in case of containment failure or venting, through codes such as ASTEC (Van Dorsselaere et al., 2009; Chatelard et al., 2013) and MELCOR (Gauntt et al., 2005).

2. Experimental summaries

2.1. CORA

The CORA program (Schanz et al., 1992; Hofmann et al., 1997), Table 1, investigated out-of-pile the integral material behaviour of PWR (11 tests), BWR (6 tests) and VVER (2 tests) bundles up to about 2700 K with oxidised cladding outer surfaces (also one BWR dry core test with negligible oxidation) and at low system pressure of 0.22 MPa (one PWR test also at 1 MPa). One PWR test used high rod internal pressure to give ballooning of the clad. The decay heat was simulated by electrical heating. Great emphasis was given to the fact that the test bundles contained all materials used in LWR fuel elements, to investigate the different material interactions, Fig. 1. Fuel pellets, cladding, grid spacers, absorber rods and the pertinent guide tubes were typical to those of commercial LWRs with respect to their compositions and radial dimensions. The PWR-typical bundle consisted of 16 heated, 7 unheated and two absorber rods. The (Ag80%, In15%, Cd5%) absorber material was sheathed in stainless steel and this rod was surrounded by a Zry-guide tube. The BWR bundle simulated the arrangement of the B_4C absorber-cross placed between two bundles each with 6 heated and 3 unheated rods. VVER-1000 aspects were simulated by use of a 19-rod bundle with a hexagonal arrangement of 13 heated, one B_4C absorber and 5 unheated rods. Three phases can be recognised for each test: (1) gas preheat phase in argon to 700 K, 0–3000 s; (2) transient phase in steam from 700 K to a maximum of 2700 K depending on the test, 3000–4900 s; (3) water quench from the bottom (3 tests) or slow cooling phase in argon, to room temperature. The results of the integral CORA tests may be summarised as follows:

- The CORA data allowed definition of 3 temperature regimes in which large quantities of liquid phases form, which cause extended fuel rod bundle damage and accelerate damage progression:
  - 1500–1700 K: localised core damage;
  - 2100–2300 K: extended core damage;
  - 2900–3150 K: total core destruction;
- A temperature escalation due to the Zr-steam reaction starts in the upper, i.e. hotter bundle half at about 1400 K (axial elevation typically 850 mm up a 1000 mm tungsten heated length, as in CORA-13 (Firnhaber et al., 1993)) and propagates from there downwards and upwards. The maximum heat-up rates and maximum temperatures measured were approx. 20 K/s and 2300 K, respectively;
- The cladding integrity can be lost far below the melting point of Zircaloy by eutectic interactions with stainless steel of absorber claddings or absorber materials themselves, resulting in formation of liquid phases at temperatures as low as 1550 K;
- Significant molten UO_2 relocation can begin at the Zircaloy melting (solidus) temperature of about 2025 K, ~1000 K below the melting point of UO_2;
  - The low-temperature early fuel relocation is important for the increased release of volatile fission products and the redistribution of decay heat sources in damaged cores;
- The CORA quench tests demonstrated that reflood initiated at temperatures close to beginning of melt formation (due to eutectic interactions or achievement of melting point of zirconium alloy) does not result necessarily in an immediate decrease of the bundle temperature;

Table 1
CORA test matrix.

Test	Number of fuel rods/control rods	Fluid	Pressure (MPa) system/rod	Test termination	Date
2	25/0	Ar, Steam	0.2/0.7	Slow cooling in Ar	August 06, 1987
3	25/0	Ar, Steam	0.2/0.5	Slow cooling in Ar	December 03, 1987
5	24/1 Ag/In/Cd	Ar, Steam	0.2/0.4	Slow cooling in Ar	February 26, 1988
12	23/2 Ag/In/Cd	Ar, Steam	0.2/0.3	Quench by water	June 09, 1988
16	18/B_4C blade	Ar, Steam	0.2/0.6	Slow cooling in Ar	November 24, 1988
15	23/2 Ag/In/Cd	Ar, Steam	0.2/6.0	Slow cooling in Ar	March 02, 1989
17	18/B_4C blade	Ar, Steam	0.2/0.5	Quench by water	June 29, 1989
9	23/2 Ag/In/Cd	Ar, Steam	1.0/0.2	Slow cooling in Ar	November 09, 1989
7	52/5 Ag/In/Cd	Ar, Steam	0.2/0.5	Slow cooling in Ar	February 22, 1990
18	48/B_4C blade	Ar, Steam	0.2/0.6	Slow cooling in Ar	June 21, 1990
13	23/2 Ag/In/Cd	Ar, Steam	0.2/0.4	Quench by water	November 15, 1990
29	23/2 Ag/In/Cd	Ar, Steam	0.2/0.4	Slow cooling in Ar	April 11, 1991
31	18/B_4C blade	Ar, Steam	0.2/0.4	Slow cooling in Ar	July 25, 1991
30	23/2 Ag/In/Cd	Ar, Steam	0.2/0.4	Slow cooling in Ar	October 30, 1991
28	18/B_4C blade	Ar, Steam	0.2/0.4	Slow cooling in Ar	February 25, 1992
10	23/2 Ag/In/Cd	Ar, Steam	0.2/0.4	Slow cooling in Ar	July 16, 1992
33	18/B_4C blade	Ar, Steam (trace)	0.2/0.5	Slow cooling in Ar	October 01, 1992
W1	19 VVER	Ar, Steam	0.2/0.35	Slow cooling in Ar	February18, 1993
W2	18 VVER/1 B_4C rod	Ar, Steam	0.2/0.3	Slow cooling in Ar	April 21, 1993