Nuclear Engineering and Design 276 (2014) 191–201

Contents lists available at ScienceDirect

Nuclear Engineering and Design

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

Comparison of CORA & MELCOR core degradation simulation and the MELCOR oxidation model

Jun Wang[a,b], Michael L. Corradini[a,*], Wen Fu[a,c], Troy Haskin[a], Wenxi Tian[b], Yapei Zhang[b], Guanghui Su[b], Suizheng Qiu[b]

[a] College of Engineering, The University of Wisconsin-Madison, Madison, WI 53706, United States
[b] State Key Laboratory of Multiphase Flow in Power Engineering, Xi'an Jiaotong University, Xi'an 710049, China
[c] Institute of Nuclear and New Energy Technology, Tsinghua University, Beijing 100084, China

HIGHLIGHTS

- Oxidation model of MELCOR is analyzed and the improving suggestion is provided.
- MELCOR core degradation calculating results are compared with CORA experiment.
- Flow rate of argon and steam, the generating rate of hydrogen is calculated and compared.
- Temperature spatial variation and temperature history is calculated and presented.

ARTICLE INFO

Article history:
Received 7 February 2014
Received in revised form 18 April 2014
Accepted 30 May 2014

Keywords:
MELCOR
CORA
Core degradation
Hydrogen generating rate
Oxidation model

ABSTRACT

MELCOR is widely used and sufficiently trusted for severe accident analysis. However, the occurrence of Fukushima has increased the focus on severe accident codes and their use. A MELCOR core degradation calculation was conducted at the University of Wisconsin–Madison under the help of Sandia. The calculation results were checked by comparing with a past CORA experiment. MELCOR calculation results included the flow rate of argon and steam, the generation rate of hydrogen. Through this work, the performance of MELCOR COR package was reviewed in detail. This paper compares the hydrogen generation rates predicted by MELCOR to the CORA test data. While agreement is reasonable it could be improved. Additionally, the MELCOR zirconium oxidation model was analyzed.

1. Introduction

Since the first commercial nuclear power plant was built, research efforts have explored phenomena related to severe accidents (Sehgal, 2012). Accurate, multifunctional, and targeted severe accident analysis codes play important roles in ensuring the safety of current and future nuclear power plants (Rubin et al., 1996). With these traits in mind, MELCOR is widely used and sufficiently trusted in severe accident analysis. MELCOR code predictions could be used to qualify the safety margins for beyond design base accident scenarios for operating reactors. In addition, MELCOR could explore possibilities of mitigating postulated accident scenarios before any could lead to a degraded core accident. However, the occurrence of Fukushima increased the focus and use of severe accident codes. The knowledge of in-vessel melt relocation processes is quite important with respect to combustible gas generation, cooling recovery actions (flooding of the core) and reactor pressure vessel (RPV) failure analysis (Hofmann, 1999). As a consequence, MELCOR core degradation analyses are underway in many research groups and recent analyses have been conducted in the University of Wisconsin–Madison using the MELCOR model. The calculation results were checked by comparing with a CORA experiment. In particular, the hydrogen generation rate was examined and the MELCOR oxidation model was analyzed. A separate oxidation analysis model, MYCOAC, was developed and compared.

2. CORA test and MELCOR UW model

2.1. Simple description of the CORA test

The CORA experiment was chosen as OECD-CSNI International Standard problem 31 (ISP-31) (Hagen et al., 1993; Hagen et al.,

* Corresponding author. Tel.: +1 608 263 7451; fax: +1 608 263 7451.
E-mail address: corradini@engr.wisc.edu (M.L. Corradini).

http://dx.doi.org/10.1016/j.nucengdes.2014.05.041
0029-5493/

Nomenclature

Acronym

t	time, (s)
T	temperature, (K)
δ	weight gain or layer thickness, (kg/m^2 or m)
A, B	parabolic rate constants taken from MATPRO
w	oxygen weight gain per unit surface area, (kg/m^2).
ρ	density (kg/m^3)
R	gas constant, R = 8.31 (J/K/mol)
LWR	light water reactor
UW	University of Wisconsin
COR	core
CVH	control volume hydrodynamic
VISIO	a software in office serious
TECPLOT	a software name

1996). The CORA experiment was carried out in an out-of-pile facility at Kernforschungszentrum Karlsruhe (KfK), Federal Republic of Germany, and was part of the international "Severe Fuel Damage" (SFD) program (Fimhaber et al., 1993). The major objectives of this experiment were to investigate the behavior of PWR fuel elements during early core degradation and fast cool-down due to re-flood (Hagen et al., 1993).

CORA-13 was chosen in this paper as it was one of the MELCOR validation cases and the MELCOR input was provided by Sandia National Lab. In the CORA experiments, the decay heat was simulated by electrically heated rods. Great emphasis was given to the fact that the test bundles contain all materials used in light-water reactor fuel elements in order to investigate the different materials interactions. 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 (Fimhaber et al., 1993). The CORA test facility simplified flow diagram is shown in Fig. 1.

The CORA test bundle consisted of 16 heater rods, 7 unheated rods representing typical pressurized water reactor (PWR) fuel elements, and 2 absorber rods. The detail parameters are shown in Table 1.

The experiment consisted of three phases: a preheat phase, a transient heat-up phase, and a cool-down phase. The preheat phase lasted from 0 to 3000s. During this time, the rods were heated at a low electric power input of 0.65 kW in preparation for the protracted heating phase during which the actual test is performed. During the transient heat-up phase, which lasted from 3000 to 4870s, the heater rod power was increased linearly in time from 6 to 27 kW. Of course, the electrical heating produced an increase of the fuel rod temperature. By 4000s, fuel cladding temperatures were beginning to exceed 1273 K whereupon measurable hydrogen production was detected. At this point, oxidation energy became increasingly important as it accounted for nearly 50% of the total heat input during the experiment. Shortly after 4200s, cladding temperatures in the upper regions of the bundle were observed to increase very rapidly, exceeding the melting point of both the thermocouples in use as well as the zircaloy cladding. The final phase was initiated at 4870s when the bundle was quenched by means of a water-filled quench cylinder that rose directly into the test bundle. Finally, the rods were cooled for 180 s (Humphries et al., 2008).

Table 1
Design characteristics of bundle CORA-13.

Bundle type		PWR
Bundle size		25 Rods
Number of heated rods		16
Number of unheated rods		7
pitch		14.3 mm
Rod outside diameter		10.75
Cladding material		Zircaloy-4
Cladding thickness		0.725 mm
Rod length	Heated rods	1960 mm
	Elevation	−489 to 1471 mm
	Unheated rods	1672 mm
	Elevation	−201 to 1471 mm
Heated length		1000 mm
Heater material		Tungsten(W)
Heater diameter		6 mm
Fuel pellets of heated rods		UO_2 annular pellet
Fuel pellets of unheated rods		UO_2 full pellets
U-235 enrichment		0.20%
Pellet outer diameter		9.1 mm
Absorb rob materials		80Ag, 15In, 5Cd (wt%)

2.2. MELCOR UW model

The development of the MELCOR UW model was based on the MELCOR Sandia model with the help of Sandia National Lab (Humphries et al., 2008). MELCOR model has separated CORA system control volumes and CORA core cells nodalizations. Even if making it simplification, it is still too long in this article. As a result, the author tied to explain all the information into one figure to make them simpler, and presents the relationship between control volumes and core cells better. This is MELCOR UW model, as presented in Fig. 2. This innovation makes this MELCOR CORA model much easier to understand as a whole system. Same as the MELCOR Sandia model (Humphries et al., 2008) provided by Sandia National Lab, the MELCOR UW model for the fuel rod section splits the test bundle into four radial rings. Referring to Fig. 2, the first ring includes a central unheated rod. The second ring has four heated rods, while the third ring has two absorber rods and six unheated rods. The last ring contains twelve heated rods.

3. MELCOR calculation results and verification by test CORA

3.1. Gas distribution

Argon and steam distributions were the original boundary conditions in the CORA-13 experiment. They were also stated directly in MELCOR Sandia input. This chapter just confirms that these flow rates were input correctly (not calculated).

3.1.1. Argon distribution

During the whole process of the CORA-13 test, there was a steady flow of 8 g/s preheated argon to the bundle (Hagen et al., 1993). Argon is a kind of chemical stability gas. Under the effect of argon, air is definitely got rid of CORA experiment faculty. Meanwhile, the facility could be pre-heated by argon gas and steam condensation could be avoided. On the other hand, to keep the video-scope windows clear, a total flow of 1 g/s argon was directed to the front of the windows of the video-scopes (Hagen et al., 1993). In consideration of the importance of argon gas, the MELCOR calculation result of argon flow rate was selected and compared with CORA data in Fig. 3. As shown in the following figure, the argon of the MELCOR calculation equals to the sum of argon in the CORA experiment. In other words, the MELCOR calculation results accurately reflect the real situation of argon in the CORA.

3.1.2. Steam distribution

Zirconium alloys are widely used in nuclear industries (Steinbruck et al., 2010). However, at the basis temperature of 1273 K, there appears to be an interaction between zircaloy and steam (Shi et al., 2013). The oxidation reaction of the zirconium