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Assessment of CINEMA code with LOFT LP-FP2 experiment

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1. Introduction

Early-phase of core degradation belongs to the in-vessel phenomena of severe accident and several experiments have been utilized to validate the system code. Present study shows the assessment of CINEMA code using LOFT LP-FP2 experiment and is focused on the reactor thermal-hydraulics and in-vessel core degradation including heat-up and oxidation.

2. Methods and Results

2.1 LOFT LP-FP2 experiment

LOFT (Loss of Fluid Test) facility in Idaho National Laboratory is designed to simulate typical 4-loop PWR reactor with reactor pressure vessel, primary/secondary system and ECCS (Emergency Core Cooling System) and has widely produced the thermal-hydraulic database based on OECD project and USNRC safety program [1]. In particular, LP-FP2 experiment is focused on the fission product release and transportation under severe core damage condition like TMI-2 accident [2]. Center fuel module (CFM) of reactor core is insulated by thermal shroud (ZrO2) to simulate heat-up, oxidation, degradation and relocation of core under high temperature. LP-FP2 experiment does not have any broken loop part and it is substituted by three break lines: ILCL (Intact Loop Colg Leg), LPIS (Low Pressure Injection System) and PORV (Power-Operated Relief Valve). Each break line has their own event time for opening or closing of valve to control the pressure of primary loop. Coolant inventory during blowdown stage is discharged to blowdown suppression tank to simulate containment building.

2.2 CINEMA code

CINEMA code is the integral code to simulate severe accident by coupling reactor thermal-hydraulics (SPACE), in-vessel phenomena (COMPASS), ex-vessel phenomena (SACAP) and source term (SIRIUS) [3]. CINEMA code in the present study is the SPACE coupled with COMPASS. Input dataset of CINEMA code described in the present study is based on the model developed by SPACE [4] and MELCOR [5]. CINEMA input dataset for the LOFT facility consisted on six parts: (i) intact loop, (ii) reactor vessel, (iii) pressurizer, (iv) secondary system, (v) ECCS, and (vi) blowdown suppression tank. Core is simulated with center fuel module (CFM) and periphery fuel module (PFM) as two rings and convection/radiation/conduction between rings is negligible because of thermal shroud. Axial node of CFM is divided by twelve nodes identical to MELCOR. Clad material (Zr) of thermal shroud is also considered and total mass of zircaloy in CFM is 40.2 kg, which is nearly identical to the design value [2]. Blowdown suppression tank is treated by pressure boundary condition using TFBC (Temporal Face Boundary Condition).

2.3 CINEMA Result

Table I: Initial condition by steady state stage

	CINEMA	MELCOR	EXPERIMENT
Core power [MW]	26.8	26.8	26.8
Primary pressure [MPa]	14.98	14.96	14.98±0.1
HL/UP temperature [K]	572.52	567.4	571.6±0.8
CL/LP temperature [K]	562.30	556.7	559.9±0.8
Loop mass flow rate [kg/s]	472.84	477.1	475.0±2.5
Core bypass mass flow rate [kg/s]	62.82	65.3	63.0
SG pressure [Pa]	6.38	6.49	6.38±0.08
SG level [m]	2.91	2.79	3.12±.0.06

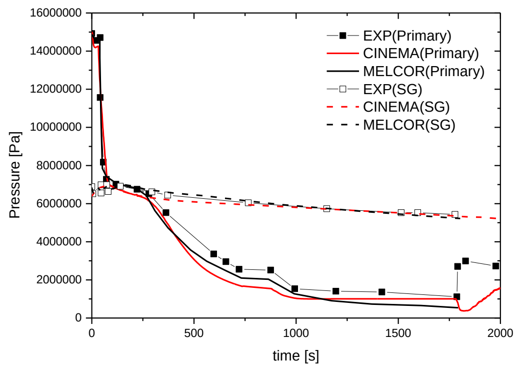

Fig. 1. Pressure of pressurizer and steam generator

Results of steady-state calculation (from – 500s to 0 s) is well matched to experimental or MELCOR result (Table I). Reactor is scrammed when time is 0 sec, and pressure of primary loop is decreased when ILCL break initiated (32.9 s). Control of break lines (LPIS and PORV) is required to an additional decrease of the primary pressure by blow-down during experiment. Loss of coolant inventory in primary loop leads to decrease of coolant water level in the core. When the core is dry-out, single-phase steam convection results in lack of heat removal from core and temperature of fuel and clad is gradually increased. ECCS is activated after the temperature of CFM thermal shroud is reached the high-temperature trip condition (1766 sec) and core is quenched by 1795 sec, which indicates the reflooding in the core.

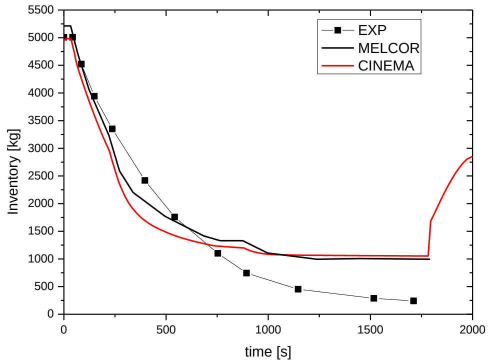

Fig. 2. Coolant inventory of primary loop

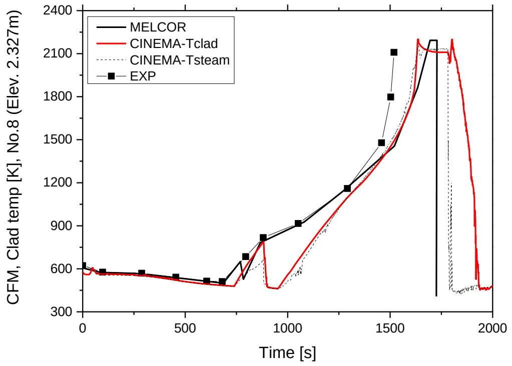

Fig. 3. Cladding temperature of CFM

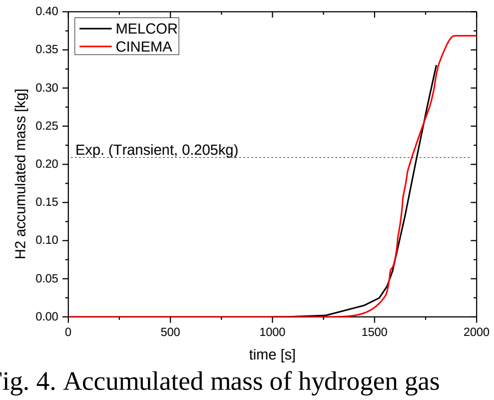

Fig. 4. Accumulated mass of hydrogen gas

Reactor thermal-hydraulics (primary pressure and coolant inventory) and core heat-up and oxidation by CINEMA code are appropriate compared to the MELCOR or experimental data. CINEMA code also well predicts hydrogen mass generated by the zircaloy-steam oxidation.

3. Conclusions

Present study shows the calculated results of CINEMA code and the compared results with LOFT-LP FP2 experiment and MELCOR. From the reactor thermal-hydraulics to in-vessel hydrogen mass, CINEMA code shows the capability to evaluate the early-phase of core degradation during severe accident. LOFT LP-FP2 experiment belongs to case for lower pressure and steam-rich environment during core degradation and oxidation. Additional investigation of core degradation under steam-starved environment is required in the future. Also, LP-FP2 experiment provides the release of fission product and elimination of radioactive aerosol in primary loop system and relevant verification using CINEMA code will be conducted.

ACKNOWLEDGEMENT

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