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GASFLOW-MPI: A new 3-D parallel all-speed CFD code for turbulent dispersion and combustion simulations Part II: First analysis of the hydrogen explosion in Fukushima Daiichi Unit 1

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ABSTRACT

The core-melt in Fukushima-Daiichi Unit 1 represents a new class of severe accidents in which combustible gas from core degradation leaked from the containment into the surrounding air-filled reactor building, formed there a highly reactive gas mixture, and exploded 25 h after begin of the station black-out. Since TMI-2 hydrogen safety research and management has focussed on processes and counter-measures *inside* the containment but the reactor building remained unprotected against hydrogen threats. The code GASFLOW-MPI is currently under development to simulate hydrogen behaviors, including distribution and combustion, for scenarios with containment leakage.

This paper describes a first analysis of the hydrogen explosion in Unit 1. It investigates gas dispersion in the reactor building, assuming a leak at the drywell head flange, shows the evolution of a stratified, inhomogeneous H_2–O_2–N_2–steam mixture in the refueling bay, simulates the combustion of the reactive gas mixture, and predicts pressure loads to walls and internal structures of the reactor building. The blast wave propagated essentially sideways, which explains why all side walls were blown out and the ceiling just collapsed onto the floor of the refueling bay. The blast wave propagation into the free environment was also simulated. The over-pressure amplitudes are sufficiently high to cause damage to adjacent buildings and to injure people. The hitherto existing presumption that the blowout panel of Unit 2 was removed by the Unit 1 explosion can be confirmed which likely prevented a hydrogen explosion in the Unit 2.

GASFLOW-MPI provides validated models for the integral simulations of leakage related core-melt scenarios. Furthermore, the code contains extensively tested submodels for catalytic recombiners, igniters and burst foils, which allow design of new hydrogen risk mitigation systems for currently unprotected spaces in reactor buildings.

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Introduction and objectives

About 1 h after the earthquake on March 11, 2011, loss of all onsite AC power and DC battery power occurred at the Fukushima Daiichi nuclear power plant, starting an unprecedented multiple-unit station-black-out (SBO) accident. In Unit 1 a devastating hydrogen explosion happened 23.8 h later, which had grave consequences for the total course of the accident: it caused massive building damage, dispersed highly radioactive debris, opened new release paths for radioactive fission products, injured personal, destroyed provisional power and water supply systems, interrupted practically all accident management activities on the site, and caused evacuation of urgently needed personal. Due to increased radiation levels after the explosion in Unit 1, the evacuation zone was enlarged from 10 to 20 km radius, which forced the evacuation of additional 78.000 inhabitants. The hydrogen explosion is regarded as the "game-changing event" in the whole accident sequence [1].

How could such a hydrogen explosion happen, despite of the fact that the USNRC had implemented a hydrogen rule for Mark I reactors, that required installation of inerting systems in the containment? The basic reason is that mixing of inert accident atmosphere from the containment (N_2, H_2, H_2O) with inert air (N_2, O_2) in the reactor building can result in highly combustible mixtures in the reactor building (N_2, O_2, H_2, H_2O). Prerequisite for the formation of such mixtures are leakage paths from the pressurized containment into the surrounding air-filled reactor building (*Note*: Although there is no direct proof for CO generation in Unit 1, the presence of CO is practically certain due to the very well known MCCI phenomena and results of MELCOR simulations [2]. In the present first analysis of the Unit 1 explosion CO-effects are neglected. The term "hydrogen" is used in the following text as synonym for combustible H_2–CO mixtures).

The hydrogen explosion in Unit 1 (and those in Unit 3 and 4) have shown in a striking way that established hydrogen mitigation measures in the containment, like containment inertization, venting systems, catalytic recombiners and igniters, are not able to prevent energetic combustible gas reactions and their grave consequences, if the containment develops leakages during the accident progression. For Unit 1 two different leak mechanisms have been proposed: a) thermally overloaded elastomer sealings in hatches, as well as in electric cable penetrations [3], and b) a leak at the drywell head flange due to excessive over-pressure in the drywell [2].

From what has been known from containment research during the last three decades, the development of containment leakages in Unit 1 under severe accident loads is not surprising.

The response of German BWR and PWR reactor plants to core-melt accidents has been investigated by extensive studies of the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS). The main results for the study of BWR-72 design are [4]:

- weakest part of the steel-lined concrete containment is the elastomer seal in the closure head of the steel containment; degradation and leakage is expected above 150–170 °C,
- a core melt would penetrate the concrete containment with high probability, e.g. through cable ducts, providing path ways for discharge of N_2, H_2, CO, and H_2O into the reactor building,
- the resulting H_2–CO–steam–air mixtures in the reactor building would be able to support fast combustion modes after an unintended ignition, could cause serious damage to the building, and open new pathways for radioactive release.

This prognosis for BWRs from 1998 agrees perfectly with the events in Fukushima Daiichi Unit 1. Also for the German KONVOI PWR, which has a large spherical air-filled steel containment, similar predictions were made:

- accident scenarios with extended molten-core-concrete-interaction (MCCI) lead to inert H_2, CO, N_2 and H_2O mixtures in the containment [5],
- PSA-studies show different potential release paths for fission products, and therefore also for H_2–CO containing atmosphere from the containment into adjacent air-filled rooms [6],
- leakage of the spherical steel shell is expected for 1/3 of all scenarios.

Unpublished studies of German utilities lead to similar results. The general conclusion for German plants was therefore that core-melt accidents would create combustible H_2–CO–air mixtures outside of the primary containment with significant probability. This is not a result of inadequate engineering. Containments of the second reactor generation were designed for a large break of the primary circuit, and not for the long lasting, high-pressure and high-temperature loads developing in extended core-melt accidents.

The comprehensive US research on containment structure integrity performed at Sandia Natl. Laboratories was summarized in Ref. [7]. The strength under static pressure was evaluated for Mark I containments in terms of S-shaped fragility curves which give the failure probability as function of internal pressures. For most of the Mark I containments the pressures for 10% failure probability are between 1.4 and 1.7 times the design pressure. The static pressures for 90% failure probability are between 1.8 and 2.5 times the design pressure. Since the typical design pressure for Mark I containments is 3.8 (±0.3) bar, the 90% failure probability would be reached between about 6.8 and 9.5 bar. The MELCOR predicted containment pressure during the MCCI gas generation phase was 7.5 bar [2], which indicates that the Unit 1 containment was probably close to its ultimate failure pressure. The dominant failure locations identified in Ref. [7] include a leak at the drywell head, leak or rupture of the wetwell, or rupture of the drywell wall. Containment leakages were identified as an important failure mechanism and they are quite likely for Unit 1, after core relocation to the lower RPV plenum had occurred (after 9 h).

The EU- funded SARNET Project completed in 2014 was an international network which defined joint research priorities with respect to core-melt accidents in nuclear reactors of the second and third generation [8]. The SARNET project was a direct European response to the multi-unit SBO accident in Fukushima Daiichi. Among the topics with high research priorities were identified: "risk of combustions in the reactor building where

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