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Validation of ASTECV2.1 based on the QUENCH-08 experiment

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HIGHLIGHTS

- ASTECV2.1 can reproduce QUENCH-08 experimental trends e.g. hydrogen generation.
- Radial temperature gradient and heat transfer through argon gap are underestimated.
- Mesh sizes lower than 55 mm needed to capture the strong axial temperature gradient.
- Minor variations of external electrical resistance strongly affect bundle heat-up.
- Modelling of a bypass and inclusion of currents partially overcome discrepancies.

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ABSTRACT

The Fukushima accidents have shown that further improvements of Severe Accident Management Guidelines (SAMGs) are still necessary. Hence, the enhancement of severe accident codes and their validation based on integral experiments is pursued worldwide. In particular, the capabilities of the European integral severe accident ASTECV2.1 code are being extended within the CESAM project through the improvement of physical models, code numerics and an extensive code validation.

Among the different strategies encompassed in the plant SAMGs, one of the most important ones to prevent core damage is the injection of water into the overheated core (reflooding). However, under certain conditions, reflooding may trigger a sharp hydrogen generation that may jeopardize the containment. Within this work, ASTECV2.1 models describing the early in-vessel phase of the severe accident and its termination by core reflooding are validated against data from the QUENCH test facility. The QUENCH-08, involving the injection of 15 g/s (about 0.6 g/s/rod) of saturated steam at a bundle temperature of 2073 K, has been selected for this comparison.

Results show that ASTECV2.1 is able to reproduce the experimental temperatures and oxide thicknesses at representative bundle locations. The predicted total hydrogen generation (76 g) is similar to the experimental one (84 g). In addition, the choices of an axial mesh size lower than 55 mm and of an external electrical resistance of a 7 mΩ/rod have been justified with parametric analyses. Finally, new modelling options are introduced to overcome some discrepancies present in the reference case and to cancel the effect of the external electrical resistance in the QUENCH model. Results show that the predicted hydrogen generation during quenching (38 g) is closer to the experimental one (37 g) than the reference case, giving a better reproduction of the radial temperature gradient within the hottest zone.

1. Introduction

In the Nuclear Power Plants (NPPs) operated today in Europe, the Defence-in-Depth concept is applied to ensure that public and environment are protected from a harmful release of radioactive material. Within this safety concept, measures to

Abbreviations: FRS, Fuel Rod Simulator; LWR, Light Water Reactor; NPP, Nuclear Power Plant; SAM, Severe Accident Management; SAMGs, Severe Accident Management Guidelines; Zry, Zircaloy.

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control severe accidents by means of Severe Accident Management (SAM) are an essential component.

Although significant progress has been made on the development of SAM Guidelines (SAMGs) after the severe accidents of Three Mile Island Unit 2 (1979) and Chernobyl (1986) (EPRI, 1993; European Commission, 2000), the Fukushima accidents (2011) have shown that further improvements of those are necessary (NEA-OECD, 2013). This requires an extensive number of deterministic analyses performed with severe accident codes (Braun et al., 2014), whose validation is pursued worldwide through dedicated research programs (Van Dorsselaere et al., 2015).

The integral severe accident ASTECV2.1 code (Chatelard et al., 2016), jointly developed by the "*Institut de Radioprotection et de Sûreté Nucléaire*" (*IRSN*) and "*Gesellschaft für Anlagen und Reaktorsicherheit*" (GRS), is being extended for the analysis of severe accident sequences in different Light Water Reactors (LWRs) operating in Europe. This task is being performed within the EU CESAM project (2013–2017) (GRS, 2016) through improvements in the physical models and code numerics, code validation and plant application, considering the lessons learned from Fukushima (NEA-OECD, 2013) and the current status of knowledge on SA phenomena (Klein-Heßling et al., 2014).

Among the different SAM strategies to prevent the failure of the safety barriers, one of the most important ones is the injection of water into the reactor (core reflooding). However, under certain conditions, the generated steam would enhance the oxidation of the remaining Zircaloy cladding and trigger a sharp hydrogen generation (Hering et al., 2015; Schanz et al., 1992; Steinbrück et al., 2010), which could jeopardize the containment at an early stage of the accident. In particular, the hydrogen generation derived from core reflooding at an early in-vessel phase of the accident has been extensively investigated at the QUENCH test facility of the Karlsruhe Institute of Technology (KIT) (Steinbrück et al., 2010).

Within this work, ASTECV2.1 capabilities to describe the phenomena occurring during the early in-vessel phase of a severe accident will be assessed using data from the QUENCH test facility. The QUENCH-08 experiment (Stuckert et al., 2005), devoted to the fast cooling of a prototypical PWR bundle with saturated steam from the bottom, has been selected for the extension of the ASTECV2.1 validation matrix on QUENCH.

First of all, a description of the QUENCH test facility and the QUENCH-08 test conduct are presented. Then, the general features of the ASTEC code are briefly described. Afterwards the ASTECV2.1 model for the QUENCH-08 (e.g. geometry and the main modelling parameters) is detailed. Therein, a comparison between the code predictions and the experimental data is performed including the main outcomes of parametric studies on the axial discretization and the external electrical resistance of the bundle. Afterwards, new options for the modelling of the QUENCH facility are investigated and their influence on the bundle behaviour is analysed. Finally, a summary of these investigations and some perspectives for future works are given.

2. The QUENCH experimental program

The QUENCH program (Sepold et al., 2001) arose as a successor of the former CORA program at KIT (Schanz et al., 1992), aiming at deepening the understanding of the processes influencing the hydrogen generation and the core degradation during reflooding. Up to present, 17 tests with severe accident conditions have been carried out (Haste et al., 2015). Generally, reflooding of slightly degraded core at low pressure scenarios (where the system and rod internal pressures are 2 bars) has been usually investigated. This situation usually corresponds to a LOCA scenario, where the fuel rod claddings burst and equalize their pressure to the system pressure. The influence of the following parameters on the hydrogen generation have been studied: degree of cladding preoxidation, temperature at the onset of reflooding, steam starvation, air ingress, water (steam) mass flow rate, effect of absorber material (e.g. boron carbide, silver-indium-cadmium), effect of cladding material (e.g. E110, AREVA M5^{R}, ZIRLO^{TM}) (Steinbrück et al., 2010).

2.1. The QUENCH test facility

The main components of the QUENCH facility and the cross section of the bundle are depicted in Figs. 1 and 2. The superheated steam exiting the steam generator and the argon are injected at the bottom of the test bundle. The argon, the steam and the generated hydrogen exit the bundle at the top. The hydrogen release is analysed by a quadrupole mass spectrometer Balzers GAM300 located at the off-gas pipe of the test facility. The hydrogen mass flow rate is calculated referring the measured hydrogen concentration to the known argon mass flow rate. The reflooding medium, usually water or saturated steam, is injected at the bottom of the test bundle. The two steam flows (saturated and overheated) come from two different pipelines: the pipe with superheated steam is going through the superheater, whereas the one with saturated steam is coming directly from the steam generator.

The test bundle consists of twenty-one fuel rod simulators (FRSs), each of which has an approximate length of 2.5 m. Twenty of these FRSs are electrically heated over a length of 1024 mm through a 6 mm diameter tungsten rod, in contrast to the unheated one placed in the center of the test section. Each heated FRS is composed of a tungsten center rod, which is surrounded by ZrO_2 annular pellets and a Zircaloy-4 (Zry-4) cladding. The tungsten rods are in contact with molybdenum and copper electrodes at the top and at the bottom, both electrodes being connected through cables to the DC electric power supply, and coated with a ZrO_2 fiber. The unheated FRS is composed of ZrO_2 pellets and Zry-4 cladding along its entire length. All FRSs are fixed in their positions by five grid spacers (lowest with Inconel, the rest with Zry-4).

Four corner rods are installed to avoid a flow bypass to the outer positions of the cross section and to obtain a uniform temperature distribution of the rods. Moreover, the corner rod B is removable in order to measure the oxide layer before reflooding. This is possible because of the intensive radiative heat exchange between the heated and the unheated rods, as will be shown later in the comparison. The FRSs and the corner rods are surrounded by a shroud, which is surrounded itself by a tubular cooling jacket. The gap between the shroud and the cooling jacket is filled with a porous ZrO_2 fiber insulation along the heated length of the bundle, and with stagnant argon gas above the heated length (1024–1300 mm). The cooling jacket is argon-cooled along the heated length of the bundle, and water-cooled above the heated length of the bundle. Further details of the facility can be found in Sepold et al. (2001), Stuckert et al. (2005).

2.2. The QUENCH-08 experiment

The test QUENCH-08 (Stuckert et al., 2005) was performed as a reference test for the QUENCH-07 and QUENCH-09 experiments. The aim of the three tests was the investigation of the boron carbide influence on core degradation. In the QUENCH-08 test, no boron carbide was present, in contrast to the other tests QUENCH-07 and -09. The three tests were terminated with the injection of 15 g/s of saturated steam, this representing the dispersed flow region. The QUENCH-08 experiment has been selected due to the lack of this test in the ASTEC validation matrix.

The test phases are illustrated in Fig. 3 and explained as follows: