Annals of Nuclear Energy 83 (2015) 274–282

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

Technical note

Comparative study of the hydrogen generation during short term station blackout (STSBO) in a BWR

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ARTICLE INFO

Article history:
Received 15 December 2014
Received in revised form 21 April 2015
Accepted 22 April 2015

Keywords:
Several accident NPP
BWR
Short-term SBO
MELCOR
SCDAP/RELAP5

ABSTRACT

The aim of this work is the comparative study of hydrogen generation and the associated parameters in a simulated severe accident of a short-term station blackout (STSBO) in a typical BWR-5 with Mark-II containment. MELCOR (v.1.8.6) and SCDAP/RELAP5 (Mod.3.4) codes were used to understand the main phenomena in the STSBO event through the results comparison obtained from simulations with these codes. Due that the simulation scope of SCDAP/RELAP5 is limited to failure of the vessel pressure boundary, the comparison was focused on in-vessel severe accident phenomena; with a special interest in the vessel pressure, boil of cooling, core temperature, and hydrogen generation. The results show that at the beginning of the scenario, both codes present similar thermal-hydraulic behavior for pressure and boil off of cooling, but during the relocation, the pressure and boil off, present differences in timing and order of magnitude. Both codes predict in similar time the beginning of melting material drop to the lower head. As far as the hydrogen production rate, SCDAP/RELAP5 predicts 15.8% lower production than MELCOR.

1. Introduction

To design a nuclear power plant is very important to make the safety evaluation in order to prove safety features of systems and installed equipment to prevent any accident. Although the analysis of Design Base Accidents (DBA) are considered as reference, DBA have differences with the modeling of severe accident related to the fact that the modeled system itself is not well defined due that molten core material have constant changes of geometrical factors, composition and into its properties.

Typical codes for severe accident analysis in nuclear reactor cores are:

- MELCOR: Methods for Estimation of Leakages and Consequences of Releases (Gauntt et al., 2000),
- MAAP: Modular Accident Analysis Program (MAAP4, 1994), and
- SCDAP/RELAP5: Severe Core Damage Analysis Package (Siefken et al., 2001a,b).

These codes are widely used for the integral analysis of core melt accident progression and the resultant lower head response expected at the reactor lower plenum. However, they have been developed from different approaches and for different purposes. MELCOR was originally intended to be a probabilistic risk assessment tool. Latter versions of MELCOR contain significant modifications, including the addition of a large number of physics models. The initial objective for the MAAP code was to predict severe accidents, using simple models based on first principles. MAAP has shown to produce credible results for several severe accident scenarios despite relatively coarse spatial mesh and run times two or three orders of magnitude shorter than those of MELCOR and SCDAP/RELAP5. SCDAP/RELAP5 began as a best estimate code with physics-based models. While the primary concern of the MELCOR and MAAP4 codes is to provide valuable information for severe accident management as well as to support probabilistic safety assessment (PSA), the SCDAP/RELAP5 is often used to examine the core and lower head phenomena in detail.

1.1. MELCOR

Several works have been performed on different issues with the MELCOR code, some of these works which are remarkable are listed below:

Carbajo (1994) performed the analysis to assess the effect of a variety of design parameters and operational procedures on a station blackout severe accident at the Peach Bottom Atomic Power Station; he found that the optimum steam cooling of the core is accomplished when the automatic depressurization system is

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http://dx.doi.org/10.1016/j.anucene.2015.04.027
0306-4549/

actuated when the core water level is at one-third of the active core height, delaying vessel failure by minutes and containment failure by hours.

Haste et al. (2006) evaluated the MELCOR code independently, using empirical data consistent with the recommendations of the Committee on the Safety of Nuclear Installations of the Organization for Economic Co-operation and Development (OECD/CSNI) validation matrix for core degradation codes. The results were compared with observed and deduced data for the major accident signatures and rough estimates for exposure based on off-site monitoring. These authors found that MELCOR gave adequate treatment of the thermal-hydraulics, core heat up and oxidation while the core was mainly intact.

Ahn et al. (2006) performed a sensitivity analysis on the evolution of severe accidents that can be expected during a Loss of Coolant Accident (LOCA) at an Advanced Power Reactor 1400-MW (APR1400). Their study focused on the impact of a direct vessel injection into the reactor down-comer in mitigation of a severe core degradation; the impact of different break locations and sizes; as well as on the timings of the key thermal-hydraulic responses of the containment spray systems, the severe degradation of the core, and evolution of the core materials.

Martín-Fuertes et al. (2007) analyzed several aspects related to the source term in the Phebus FPT1 experiment, which contemplate circuit thermal-hydraulics, fission product and structural material release, vapors and aerosol retention in the circuit and containment, vertical line aerosol deposition and steam generator aerosol deposition. The detailed calculations concerning aerosol deposition in the steam generator tube were obtained by means of an in-house code application, named COCOA, as well as with CFD. These authors found that total fission product release is coherent with experimental results for volatile species. However, the non-volatiles and structural material release are under predicted.

Stefanova et al. (2009) used MELCOR "input model" for Kozloduy Nuclear Power Plant (NPP) Voda Voda Energo Reactor (WWER-1000) for the research of Passive Autocatalytic Recombiners (PARs) capabilities for hydrogen recombination in case of a station blackout scenario. Two types of WWER-1000 fuel assemblies where considered: the old Modernistic type of fuel assemblies (TVSM) and the recently new installed alternative type of fuel assemblies (TVSA) in Kozloduy NPP. The main conclusion was that for both types of fuel assemblies there were no disturbance of PARs' capabilities and safety criterion of NPP.

Szabó et al. (2012) compared MELCOR and GASFLOW codes results of the thermal hydraulics in a simplified generic Pressurized Water Reactor (PWR) containment; these authors found integral values such as pressure, mass and energy balances were in good agreement, but more realistic averaged and local distributions of steam and hydrogen could be captured in the GASFLOW results. In a more recent study, Szabó et al. (2014) coupled MELCOR and GASFLOW, they tested the coupling by calculating the TH7 experiment in the THAI facility with the coupled code system and with GASFLOW in stand-alone mode for comparison, the results agreed very well with the experimental data. Additionally, they compared the results obtained using the coupling for a postulated LOCA in a generic PWR, and found significant differences in the containment pressures which caused deviating leak flow rates and differences in the hydrogen distributions, this could also affect the steam and hydrogen flow rates through the leak calculated by MELCOR.

Wang et al. (2014) compared the hydrogen (H_2) generation rates predicted to the CORA test data; the authors found reasonable agreement. Longze et al. (2014) analyzed a severe accident at a Chinese Pressurized Reactor 1000-MW (CPR1000) power plant caused by a station blackout (SBO) with failure of the steam generator (SG) safety relief valve (SRV); these authors found the SG-SRV stuck in the open position would greatly accelerate the sequence for a severe accident induced by SBO and their results show that an auxiliary feed water supply can mitigate SBO accidents very well and provide more time for human intervention. Kim et al. (2014) compared MELCOR code versions 1.8.5 and 1.8.6 for Large Break LOCA scenarios for the APR1400 developed in Korea, tested the corium relocation models and lower head penetration models. They found the primary differences in the analysis results of hydrogen generation rates and the lower head failure timing, MELCOR 1.8.6 predicted more hydrogen generation and molten pool model, and the hemispherical lower head geometry give more realistic results.

Bonelli et al. (2014) analyzed a SBO for the Atucha 2 Nuclear Power Plant; they found that during transient, the water in the fuel channels evaporates first while the moderator tank is still partially full. The moderator tank inventory acts as a temporary heat sink for the decay heat, which is evacuated through conduction and radiation heat transfer, delaying core degradation. This feature, together with the large volume of the steel filler pieces in the lower plenum and a high primary system volume to thermal power ratio, derives in a very slow transient in which RPV failure time is four to five times larger than that of other German PWRs.

1.2. RELAP5/SCDAPSIM

Immediately after the accident at Fukushima Daiichi several authors presented an assessment of the possible core/vessel damage states of the Fukushima Daiichi Units 1–3 station blackout, LOCA, among other scenarios using RELAP5/SCDAPSIM (Allison et al., 2012; Espinosa-Paredes et al., 2012; Nuñez-Carrera et al., 2012; Trivedi et al., 2014).

Allison et al. (2012) presented an assessment of the possible core/vessel damage states of the Fukushima Daiichi Units 1–3 station blackout scenarios; the main conclusion is that the responses to the accident once the accident is underway can make a significant difference in the consequences of the accident.

Espinosa-Paredes et al. (2012) performed a loss of coolant accident (LOCA) analysis, which leads to severe fuel damage, and it also analyses the impact of flooding using the high pressure core spray (HPCS) system for the above low pressure scenario.

Nuñez-Carrera et al. (2012) analyzed a hypothetical simulation of a LOCA with simultaneous loss of off-site power, and without injection of cooling water in a Boiling Water Reactor (BWR) lower head during a severe accident; they evaluated the temperature distribution and heat up of the reactor core material that slumps in the lower part of the RPV.

Trivedi et al. (2014) analyzed the influence of water addition in a BWR during a core isolation event; they found that injection of water is impacted by time as well as reactor vessel water level, additionally the cladding tends to collapse onto the fuel at much lower temperature for high pressure sequences.

Historically, the SBO accident sequence has been considered to be loss of off-site power and reactor scram combined with failure of the station diesels to start and load. In this (long-term) accident sequence, water is injected into the reactor vessel by the steam turbine-driven HPCI or RCIC systems as necessary to keep the core covered for as long as the direct current (DC) power for the turbine governor control remains available from the unit batteries (Hodge and Ott, 1990). However, the definition of SBO has been expanded to include the Loss of Injection, called short-term station blackout (STSBO) sequences. In these STSBO sequences, the capability for water injection to the reactor vessel is lost at the inception of the accident sequence. The short-term designation derives from the fact that all of the safety systems immediately become inoperable and the core is uncovered relatively quickly in these sequences, then core damage occurs in the "short term". The early total loss