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Validation and application of the system code ATHLET-CD for BWR severe accident analyses

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HIGHLIGHTS

- We present the application of the system code ATHLET-CD code for BWR safety analyses.
- Validation of core in-vessel models is performed based on KIT CORA experiments.
- A SB-LOCA scenario is simulated on a generic German BWR plant up to vessel failure.
- Different core reflooding possibilities are investigated to mitigate the accident consequences.
- ATHLET-CD modelling features reflect the current state of the art of severe accident codes.

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ABSTRACT

This paper is aimed at the validation and application of the system code ATHLET-CD for the simulation of severe accident phenomena in Boiling Water Reactors (BWR). The corresponding models for core degradation behaviour e.g., oxidation, melting and relocation of core structural components are validated against experimental data available from the CORA-16 and -17 bundle tests. Model weaknesses are discussed along with needs for further code improvements. With the validated ATHLET-CD code, calculations are performed to assess the code capabilities for the prediction of in-vessel late phase core behaviour and reflooding of damaged fuel rods. For this purpose, a small break LOCA scenario for a generic German BWR with postulated multiple failures of the safety systems was selected. In the analysis, accident management measures represented by cold water injection into the damaged reactor core are addressed to investigate the efficacy in avoiding or delaying the failure of the reactor pressure vessel. Results show that ATHLET-CD is applicable to the description of BWR plant behaviour with reliable physical models and numerical methods adopted for the description of key in-vessel phenomena.

1. Introduction

In the last years and more recently, following the Fukushima accident, many projects and programs were launched for research on severe accident analysis of nuclear power plants, using different computer codes for simulation. These studies are focused on the importance of accident control management measures to prevent the accident evolution/escalation and to mitigate its consequences (Vitázková and Cazzoli, 2013). In the context of international efforts, the WASA-BOSS (Weiterentwicklung und Anwendung von Severe Accident Codes – Bewertung und Optimierung von Störfallmaßnahmen) project was launched by the German Federal Ministry of Education and Research aimed at investigating severe accident scenarios in Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR). The main goal is to extend the technical basis about the plant behaviour during postulated Fukushima-like accidental sequences using state-of-the-art numerical simulation tools, and to identify modelling issues for further improvements. In this framework, the Karlsruhe Institute of Technology (KIT) has been involved in the assessment of the capabilities of ATHLET-CD (Austregesilo et al., 2014) for the simulation of BWR severe accident scenarios. For this purpose, code models describing core heat-up, oxidation, degradation and melt relocation have been tested and validated on the basis of selected bundle experiments. In particular, among the CORA test series carried out at KIT (Schanz et al., 1992), CORA-16 and -17 experiments have been considered since they provide significant information about the bundle behaviour during temperature escalation up to melting without and with reflooding of the fuel rods, respectively. An input deck has been specifically developed for the CORA test which takes into account the main test section components such as the bundle,

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the canister box and the control rod, as well as the shroud and the surrounding high temperature shield. Calculations of the thermal bundle behaviour and hydrogen production have been compared with the measured values in order to address model weaknesses and other code shortcomings. Based on the CORA validation, ATHLET-CD was applied to perform plant analyses with the aim to investigate the coolability of a partly-damaged core as well as the effectiveness of severe accident (SA) measures to avoid or delay reactor pressure vessel (RPV) failure. In particular, a reference scenario derived from the probabilistic safety analyses performed in Germany (GS, 1993) was selected for the simulation of a small break (SB) loss of coolant accident (LOCA) at the steam line inside the containment. Calculations have been carried out by adopting an input deck developed for a generic BWR by the GRS (Gesellschaft für Anlagen- und Reaktorsicherheit) for the ATHLET code and modified in this work in order to predict the in-vessel phenomena occurring under severe accident conditions. Due to the postulated failure and/or unavailability of the safety systems, the selected transient leads to core temperature escalation, melting and relocation of structural materials in the lower plenum until RPV failure occurs. Under these conditions, the effect of the SA measures such as core reflooding has been investigated by actuating low pressure injection systems at different times during the transient.

The paper basically consists of two major parts. The first one is related to ATHLET-CD validation based on the available experimental dataset which will be presented after a brief overview of the code models connected to core degradation analyses. The second part deals with the BWR plant analysis and gives a general description of the considered plant model, the identification of the reference scenario and the results of the calculations without and with SA-measures. Finally, main conclusions will be drawn.

2. Overview of ATHLET-CD models for core degradation

The system code ATHLET (Analysis of Thermal–Hydraulics of Leaks and Transient) is designed to describe the reactor coolant system thermal–hydraulic response during normal and off-normal operating conditions. ATHLET is being developed by GRS in collaboration with the Institute für Kernenergetik und Energiesysteme (IKE) of the University of Stuttgart. The code has a highly modular structure in order to include a large spectrum of models and to offer a flexible basis for further development. In this frame, the simulation of in-vessel processes is carried out by different coupled modules which constitute the code version ATHLET-CD (Core Degradation). These modules allow the user to reproduce the core damage progression (ECORE module), the debris bed behaviour (MEWA), the lower plenum behaviour (AIDA or LHEAD) fission products and aerosol behaviour (FIPREM) during severe accidents, to calculate the source term for containment analyses, and to evaluate accident management measures (Austregesilo et al., 2014). A detailed description of the mentioned modules and models are reported in (Trambauer et al., 2001). Most of the validation work has been carried out by GRS in collaboration with several research institutions for PWR conditions on the basis of different experimental databases such as TMI-2, Phébus and several QUENCH tests (Firnhaber et al., 1993; OECD/NEA, 1995; Trambauer et al., 2001, 2009; Drath, 2007; Repetto et al., 2007; Georgiev and Stuckert, 2012). Compared to the studies performed on PWR, BWR validation is restricted to few investigations mainly carried out on the CORA test series (Steinrötter et al., 2000; Hollands et al., 2007; Hoffmann et al., 2013).

The core damage progression described by the ECORE module takes into account the mechanical fuel rod behaviour, oxidation of zirconium and boron carbide, melting of metallic and ceramic components, freezing, re-melting and refreezing, formation and dissolution of blockages. The mechanical rod model consists in a simplified one-dimensional approach aimed at the calculation of the fuel and cladding deformation as a consequence of both thermal expansion and creep. Cladding creep velocity is computed by means of specific correlations which take into account α and β phase transition of Zircaloy cladding, as well as oxygen and hydrogen effects in order to estimate the amount of clad ballooning and the time-to-failure which is established by different criteria available in the code.

As far as the oxidation of core components is concerned, there are different models for the description of zirconium oxidation under steam and air environment as well as boron carbide (B_4C) related processes. The zirconium steam oxidation rate of cladding and canister box (in BWR) is governed by the parabolic law derived from the solution of the diffusion equation where the reaction rate x_p is defined by the well-known Arrhenius formulation:

$$x_p = Ae^{-B/RT}g \quad (1)$$

where R is the gas constant, T is the cladding temperature, g is a reduction factor to consider steam starvation, and A and B are parameters depending on the adopted correlation. In particular, there are 3 different correlations available for oxidation in ATHLET-CD which are combinations of the following well-known correlations according to the different temperature range: (1) Cathcart et al. (1977) $1273\,K < T < 1800\,K$ – Prater and Courtright (1987) $T > 1800\,K$; (2) Cathcart et al. (1977) $1273\,K < T < 1800\,K$ – Urbanic and Heidrick (1978) $T > 1800\,K$; (3) Leistikow et al. (1978) ($1173\,K < T < 1800\,K$) – Prater and Courtright (1987) $T > 1800\,K$. The oxidation model can take into account also the breakaway effect. In particular, when the calculated oxide layer exceeds a user-defined critical oxide thickness, and if the temperature is below 1400 K, the transition from parabolic to linear kinetics occurs with consequent acceleration of the oxidation process. Besides, the inner side cladding oxidation behaviour is also considered, though in a simplified way.

As far as the boron carbide related phenomena are concerned, the code treats the eutectic material interaction between stainless steel and B_4C by considering that the dissolution of the control rod elements starts at lower temperature (1500 K) without distinguishing between the two materials behaviours. The eutectic reaction kinetics and the B_4C oxidation in steam are only available for PWR configurations and, therefore, are not considered in the present calculations.

Modelling of metallic melt takes into account the following phenomena: melting of metallic Zr cladding; dissolution of UO_2 pellet by molten zirconium; cladding failure as a function of temperature and oxide layer thickness; candling and oxidation of melt rivulet composed of molten Zr and dissolved UO_2; freezing of melt rivulet and crust and blockage formation; crust oxidation; re-melting of crust and restart of candling. The ceramic melt modelling includes the melting of remaining UO_2 and oxidized ZrO_2 cladding, as well as candling, freezing and re-melting phenomena without oxidation.

When a damaged core at high temperature is subjected to reflood or refill at low pressure, steep axial temperature gradients and high local heat fluxes may occur along the fuel rods and internals in the vicinity of the quench front, which require the axial heat conduction to be considered. For this reason, a quench front model is available in ATHLET-CD which calculates the position and the movement both of the bottom and the top quench front by applying the Semeria and Martinet (1966) correlation.

Due to the different geometrical configuration, BWR modelling requires specific treatment in ATHLET-CD in order to properly calculate the thermal–hydraulics coupling between bundle and