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DRACCAR: A multi-physics code for computational analysis of multi-rod ballooning, coolability and fuel relocation during LOCA transients Part one: General modeling description

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ABSTRACT

Computational predictions concerning ballooning of multiple fuel pin bundles during a loss of coolant accident with a final reflooding phase are now more than ever of interest in the framework of light water reactor nuclear safety. To carry out these studies, two difficulties have to be overcome. First, the modeling has to take into account many coupled phenomena such as heat transfer (heat generation, radiation, convection and conduction), hydraulics (multidimensional 2-phase flow, blockage), mechanics (thermal expansion, creep, embrittlement) and chemistry (oxidation, hydriding). Secondly, there are only a few experimental investigations that can help to validate such complex coupled modeling. Over several years, IRSN has developed the 3D computational tool DRACCAR to investigate rod bundle strain during LOCA transients including prediction of the reflooding phase. DRACCAR code is dedicated to study complex configurations such as the deformation and possible contact between neighboring rods and the associated blockage of thermalhydraulic channels in the ballooned zone of the fuel assembly. Modeling efforts have been devoted to the assessment of the coolability of deformed geometries by coupling the thermo-mechanical behavior of the fuel assembly to the thermalhydraulics. The physical modeling available in the current version of DRACCAR V2.3.1 as well as its flexibility are depicted. As a conclusion, some prospects regarding the development of the future version DRACCAR V3 are provided, in particular accounting for the knowledge acquired through IRSN R&D project PERFROI.

1. Introduction

1.1. Background and context

The loss of coolant accident (LOCA) in a light water reactor subsequent to a break or a leakage on the primary circuit were and are still largely investigated in the framework of nuclear research due to their possible consequences regarding nuclear safety. For instance, core uncovery can lead to an increase of the fuel rod temperatures which can rise enough to lead to their deformation and possible damage. In the meantime, the progressive blockage of the core channel can seriously impact the core coolability. Regarding the consequences of such accident, the possible failure of the first confinement barrier would be accompaigned by fission product release and the unsufficient core cooling would initiate a severe accident due to core degradation. So LOCA accidents are particularly investigated to ensure nuclear safety and for instance to design the emergency core cooling system of nuclear power reactors. The performance of this system is assessed by means of an evaluation model and should comply with acceptance criteria dealing with peak cladding temperature (PCT), maximum cladding oxidation, hydrogen generation, coolability of the geometry and long term cooling.

The core management optimization of the French power reactors (increase of flexibility and availability, cost saving) leads the operators to increase progressively the burn-up of fuel and to introduce new types of fuel material. These evolutions involve the development of advanced cladding materials (M5®, Zirlo™) and fuel micro-structures (gadolinium-doped fuel, MOX). In the meantime, efforts to improve simulation of LOCA lead to the development of more precise simulation tools (3D modeling, local models) and the use of calculation methods involving uncertainties to assess the safety demonstration. IRSN conducted an extensive State-of-the-Art-Review relative to fuel behavior under Loss Of Coolant Accident (LOCA) conditions, covering the aspects of clad ballooning and flow blockage, coolability of partially blocked

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assemblies, clad oxidation and clad resistance to quench and post-quench loads (Grandjean, 2005; Grandjean, 2006). Additionally, a review of existing computational tools devoted to the calculation of the fuel behavior under LOCA was performed (Cunningham et al., 2001; Siefken, 1981; Casadei et al., 1984; Haste, 1982; Uchida and Otsubo, 1984). Based on these reviews, IRSN has decided to launch a Research and Development program with the objectives of:

- Increasing the knowledge and reducing the uncertainties on the prediction of the thermo-mechanical behavior of the fuel rods and of the thermalhydraulics within a fuel assembly in LOCA conditions,
- Simulating the LOCA phenomenology taking into account all parameters: fuel management (burnup, linear power, …), fuel assembly type (pellet, clad, grid, …) and accident scenario (large break, intermediate break, …).

The IRSN Research and Development program is composed of two parts, the first one dedicated to the collection and interpretation of new experimental data complementary to the one acquired from past programmes concerning reflooding and fuel rod thermo-mechanical behavior. The second part is dedicated to the development and validation (using experimental material obtained in the first part of the program) of a simulation tool, named DRACCAR and capable of simulating the reactor core during a LOCA. The experimental data interpretation and the use of the simulation tool will allow:

- Performing an analysis of the safety criteria relevance for the current and future fuel management,
- Giving technical advice on revising the LOCA safety criteria,
- Evaluating the safety margin compared with the current safety criteria.

The state-of-the-art report OECD/NEA (2009) (chapter 8) describes the different types of codes which are necessary to achieve LOCA analysis. This includes fuel performance codes and system thermalhydraulic codes. Steady-state fuel performance codes (such as FRAPCON (Geelhood et al., 2015), CYRANO3 (Cayet et al., 1998), COPERNIC (Bonnaud et al., 1997)) can provide the initial state of fuel and cladding after normal operation (burn-up, swelling, fuel-gap thickness, fission gases distribution). From this initial state, single rod transient codes can simulate the response of a single fuel rod in accidental conditions (with a possible coupling to a thermalhydraulic code). Some of the code used are an extension of steady-state fuel performance code (such as FRAPTRAN (Cunningham et al., 2001) which is based on FRAPCON). Regarding the thermo-mechanical behavior, some 3D approaches are used in single rod codes (such as ALCYONE (Marelle et al., 2017)). These approaches are often supported by complementary analysis using finite element methods applied in mechanics (ASTER (Electricité de France, 2011), CAST3M (Le Fichoux, 2011)). Concerning the peak cladding temperature (PCT), the methodology widely generalized to achieve the LOCA safety assessment is to use a system code whose description is limited to the thermo-mechanical behavior of one average weighted fuel rod per core ring. In addition pessimistic conditions and assumptions are cumulated to ensure an over-estimation of the PCT. However such approach cannot deal properly with the presence and the behavior of neighboring rods or with the complex flows induced by the progressive blockage of the fuel channels and alternative modelling are developped such as DRACCAR or BISON (Pastore et al., 2015). The aim of the DRACCAR platform is to realistically describe the 3D thermo-mechanical behavior and reflooding of a fuel assembly including its coolability as well as its embrittlement during a LOCA. Thus, the multi-rod and multi-physics platform DRACCAR proposes an alternative to the use of single rod codes coupled to system thermalhydraulic codes for the evaluation of the peak cladding temperature (PCT), the equivalent cladding reacted (ECR) and the reflooding efficiency (Ricaud et al., 2013). DRACCAR modeling allows to take care of changes in the core geometry, as well as the ability to cool the assemblies in deformed configuration. This analysis is essential for a detailed understanding of circumstances and conditions for the assemblies quenching even in the case of a large flow blockage ratio due to clad ballooning. This implies complex computations involving two-phase flows coupled with heat transfers (decay heat, conduction, convection and radiation), mechanical behavior (creep behavior, thermal expansion, contact, oxidation and fuel relocation).

The flexibility of the DRACCAR code allows a modeling range from a simple fuel rod up to a full core including the surrounding shrouds. In order to achieve this goal, DRACCAR allows the modeling of any kind of core components (pellet, cladding, grids, …), fuel (UO_2, MOX, …), cladding (Zircaloy-4, M5®, Zirlo™, …), core management (burn-up), and types of water-cooled reactor including loops and safety systems.

Moreover, since the Fukushima accident, increased attention has been paid to the vulnerability of the Spent Fuel Pools (SFP). This vulnerability associated with the dramatic radiological consequences of an unmitigated accident is of a major concern for nuclear safety. Those consequences are related to the location of SFPs outside the reactor containment building and only the cladding acts as a confinment barrier. Accidents involving SFPs can be the result of a loss of heat removal system or a loss of coolant due to a breach in the pool structure or in a connected fluid circuit. To improve knowledge on the fuel behavior during these accident situations, IRSN has decided to extend its Research and Development program. Several experiments and specific developments in DRACCAR dedicated to SFP accident studies have been defined (Mutelle et al., 2014).

2. DRACCAR: a coupling platform

2.1. Modular approach

The main objective of DRACCAR is to describe the fuel assemblies' behavior during a LOCA on a reactor or on a spent fuel pool. Nevertheless, during such scenario, it is important in order to estimate well this behavior to:

- calculate the thermalhydraulics in all the reactor circuits in particular in the core or in the spent fuel pool,
- simulate the normal, auxiliary and safety systems (pumps, pressurizer heater, hydro-accumulators, ECCS, safety valves, removal residual heat system) which could be used during the accident,
- calculate the residual fuel power evolution as a function of the fission products.

To achieve this main objective, DRACCAR couples two codes, the first one describes the thermo-mechanical behavior of the fuel assemblies and the second one the thermalhydraulics in these assemblies. The fluid code is also capable of simulating the thermalhydraulics in all the reactor circuits or the thermalhydraulics in a spent fuel pool (to some extent). Several additional modules are available in the DRACCAR platform. The codes and modules of the DRACCAR platform are presented on Fig. 1.

Two different thermalhydraulic codes (CESAR and CATHARE-3) can currently be coupled to the thermo-mechanical code ICARE3D. CESAR and ICARE3D are two codes developed by IRSN. CATHARE-3 is the advanced thermalhydraulic code developed by CEA and financed by the main organisations in the French nuclear sector (Framatome, EDF, CEA and IRSN).

Material and fluid properties used in both codes are gathered together into a material library called Material Data Bank (MDB) and partially based on NUCLEA project (Bakardjeva et al., 2008). In fact, MDB allows the management of any kind of validated property sets and includes a huge data bank composed of:

- thermal, mechanical and chemical properties of all structural