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Abstract

This paper presents an overview of some recent evolutions in the field of thermal-hydraulics (TH) within nuclear safety studies in France, therefore restricted to Pressurized Water Reactors (PWR) design, from the perspective of the French Technical Support Organization IRSN (Institut de Radioprotection et de Sûreté Nucléaire). These evolutions can concern either the hypothetical accident scenario (recently raised questions, modified rulemaking, new methodology) or the development of recent sources of knowledge (experimental programs, advanced numerical simulation tools).

Introduction

This paper presents an overview of some recent evolutions in the field of thermal-hydraulics (TH) within nuclear safety studies in France, therefore restricted to Pressurized Water Reactors (PWR) design, from the perspective of the French Technical Support Organization IRSN (Institut de Radioprotection et de Sûreté Nucléaire). These evolutions can concern either the hypothetical accident scenario (recently raised questions, modified rulemaking, new methodology) or the development of recent sources of knowledge (experimental programs, advanced numerical simulation tools). As far as TH is concerned, the recent evolution of methods reveals an attempt to more precisely determine the flow structure by taking into account 3D effects. During last decades, lot of development have been made that consider either the extension toward 3D capabilities of system scale or component scales simulation tools or the application of Computational Fluid Dynamics to nuclear safety related issues. We refer here to recent CSNI reports as well as companion papers of this special issue about “Capabilities of 3D thermal-hydraulic codes (3DSYSTH)”, [1,2], and about the CFD activities, [3,4]. As an illustrative example, in France, CFD is now considered in more and more safety studies, [5].

Recent evolutions of understanding of the fuel behaviors at high burnup also lead to better consideration of the burnup impact during accidental conditions [6], and more especially on

- Fuel Fragmentation Relocation and Dispersal (FFRD)
- Embrittlement due to hydriding during Loss Of Coolant Accidents (LOCA)
- Hydrogen assisted PCMI brittle failure during Reactivity Initiated Accidents (RIA)

This leads to needs for updates of safety models and codes. Moreover, those elements led to recent updates of Burnup dependent safety criteria proposed by the USNRC and the regulators of other countries. Their review allows for identifying current trends in TH and potential needs for future research and development programs.

In the first part, the evolution of the French Loss Of Coolant Accident (LOCA) Fuel Safety Criteria demonstration is presented with corresponding methodology assessment and on-going R&D programs. In the second part, the Spent Fuel Pools Loss of Cooling scenario is reviewed according to the lessons learned from Fukushima. In the third part, the question of the post-DNB fuel behavior is raised through different accident scenarios (RIA, LOCA, SGTR). The last fourth part addresses the generic evolutions related to the 3D capabilities of numerical tools (for system scale, component scale and CFD approaches).

1. Loss Of Coolant Accident

Based on the technical assessment of IRSN, the French Advisory Committee for nuclear reactors (GPR) recommended in 2010 in the frame of the French LOCA rulemaking to examine the validity of the safety requirements related to the two modes of fuel degradation during LOCA (ductile mode and brittle mode) in the light of state-of-the-art of the last thirty years. French utility EDF proposed a new methodology to address those points and IRSN made its assessment [7].

A first evolution concerns the reference transients in the safety demonstration, [8,9]: EDF takes now into account the same break sizes limited by pipe whip restraints for both thermal-hydraulics and mechanical analysis. Regarding the thermal-hydraulics analysis, it is important to emphasize that such an evolution of LOCA reference transients leads to focus on a better modeling of the physical phenomena for Intermediate Breaks (IB) LOCA conditions rather than focus only on the 2A break that is now excluded from the safety demonstration (according to justifications concerning pipe break risks). For such IB-LOCA conditions, the level of the core pressure during the transient is higher than for the largest 2A break. At those pressure levels, steam has larger cooling efficiency than at low pressure, but studies revealed that there are still issues regarding the core uncovery phase of the transient, [10]. Most of the past R&D programs regarding reflooding have been devoted to 2A break scenario and constitute the validation database of TH system scale tools for LOCA. Therefore, this reference transients' evolution certainly revealed interest in experimental programs related to IB-LOCA TH's conditions. Moreover, the evolution of the French methodology includes a 3D model for the RPV-TH (including downcomer). Both evolutions require revisiting the database and identifying the pertinent tests conditions for evaluating numerical simulation performances with respect to those aspects, [11]. IB-LOCA conditions and/or 3D flows in undegraded core geometries have been addressed in several past programs (e.g. PERICLES at high pressure, [12], LSTF/ROSA [13], PKL [14], BETHSY [15], ATLAS [16], and for 3D aspects, SCTF and CCTF [17], and PERICLES 2D [18], for the core or DYNAS for the downcomer [19]). Moreover, plant applications revealed higher sensitivity of the transient in IB-LOCA conditions (with respect to LB-LOCA) to some parameters e.g. the flow rate of the safety injection system. For smaller breaks, the pressure decrease is slower and this leads to a delayed injection of water, and a lower driving force for the flow rate. Location of the break can therefore play a different role according to the LB-LOCA scenario. This led the Utility better justifying how conservatisms could cover the induced uncertainty. Moreover, in the frame of the NEPTUNE joint project (French platform on TH gathering CEA, EDF, IRSN and FRAMATOME) a specific METERO-V program has been defined for validation of 3D crossflows, [20].

A review of past experimental programs notably revealed a TH issue related to the criterion of "coolability" for French PWRs. This criterion must be satisfied for the safety demonstration for PPC-4 postulated accidents (like LOCA). It corresponds to the following objective for the first barrier of the fuel containment: the geometrical structure of the core must not be damaged to ensure adequate core coolability. During the core uncovering, the clad temperature increases up to values about 750-800°C. From a thermo-mechanical viewpoint, due to the pressure drop in the primary system, fuel rods are subjected to internal pressure higher than the external one. Thus, under the effect of tensile stresses and high temperatures reached during the transient, clads are able to balloon. The clad ballooning phenomenon leads to blocked geometry, which can under a number of conditions reduce the heat exchange surfaces between the fuel rods and the coolant and redistribute coolant flows, jeopardizing core coolability [21,22].

Considering those phenomena in the safety demonstration requires additional models (taking into account fuel deformation impacts on TH) and the corresponding experimental database for validation. The state of the art study revealed that past experimental programs devoted to the reflooding with ballooned fuel rods had a systematic bias: they did not simulate properly the ballooned region thermal inertia associated with the potential fuel relocation within balloons, [23].

The IRSN PERFROI program, [24], addresses both thermal-mechanics and thermal-hydraulics aspects of the LOCA with respect to the renewed French rulemaking. For thermal-mechanics, it focused on the role of H_2 pickup that plays a major role on clad mechanical properties besides the more classical focus on clad oxidation. Clad deformation dynamics due to pressure differential between internal and coolant pressure, impact of neighboring contact on balloon diameter and length are studied in the corresponding ranges of temperature and pressures thanks to dedicated experimental devices. It will allow improving the accuracy of LOCA predictions and validation of the R&D software for coupled thermal-hydraulics and thermal-mechanics DRACCAR, [25,26]. More especially, the COAL experiments are dedicated to the coolability of ballooned regions simulating the effect of fuel relocation through a wide range of TH conditions covering IB-LOCA cases, that clearly impact the reflooding [27].

LOCA is one of the most typical transients for TH in the field of nuclear safety. Despite decades of R&D on the topic from the very first developments of power plants, the subject will still lead part of the future TH developments.

2. Spent Fuel Pools Loss of Cooling accident

Fukushima Daïchi 2011 events certainly draw attention to the safety issues related to SFP accident scenario and extend the interest about transients of loss of cooling. It led to several plant actions in order to improve the reliability of cooling systems. In the frame of the NEA/CSNI activities, the state of the art report produced in 2015, [28], also revealed some lacks of knowledge that were more precisely addressed in the PIRT exercise, [29]. Please refer to the companion paper of this issue for more information, [30].

In France, EDF addressed updated safety demonstration of the scenario of loss of cooling and studied for example the ability to restart the cooling system within a boiling pool. This situation requires evaluating precisely the temperature and potential void fields within the pool under buoyancy driven flow conditions with evaporation at the pool's free surface. Numerical solving of this fully 3D problem is nothing but trivial. For this purpose, EDF developed CFD based studies [31,32], that were assessed by IRSN in the frame of the safety demonstration of French EPR FA3, [33]. EDF had to provide complementary elements to ensure the restart of the cooling system since CFD codes were still insufficiently validated for such flow conditions. In particular, the evaporation at the free surface in such highly turbulent natural convection requires specific validation data that are not available. Moreover, the potential nucleation of gas (dissolved and vapor) bubbles within liquid water at temperature close to the saturation one have to be considered. Meanwhile, IRSN developed the R&D program DENOPI, [34]. It covers TH issues for both loss of coolant and loss of cooling issues. Within this project, MEDEA already provided results concerning potential CCFL (Counter Current Flow Limitation) issue in fuel assemblies within a pool under a spray cooling system, [35]. ASPIC did provide data for the thermal aspects of the problem and evaluate spray cooling efficiency for partially uncovered assemblies, [36]. MIDI facility, [36], is a large basin built for the study of buoyancy driven flow of large pools near boiling conditions. A first test matrix achieved during DENOPI project revealed the facility can determine the partition between the three main vapor regimes (namely 1/Evaporation at the free surface, 2/ Flashing of superheated water in the upper part of the pool, 3/ Nucleate boiling into the assemblies) in competition during loss of cooling conditions. Tests analysis revealed that both flashing and dissolved gases are playing a significant role in the process. Further tests with potential improved instrumentation are required to deeper understand those phenomena. IRSN is currently in discussion with interested partners for the proposal of an OECD joint project based on those facilities to extend the actual database and propose experiments *vs* numerical simulation benchmarks on these topics.

It is worth pointing out that during the last decade, SFP became a renewed subject of interest and led to focus on complex buoyancy driven flows. But the topic of buoyancy driven flow of large pools as being heat exchangers is not restricted to Loss of Cooling accidental scenarios in SFP. Many under development passive systems (either for Small Modular Reactors projects but also for next generation of Light Water Reactors) have large basins that under accidental conditions may encounter similar flow conditions. For such high Rayleigh number conditions and close to nucleation or nucleate boiling conditions, the capability of numerical simulation tools must be assessed [37–39].

3. Post-DNB fuel behavior

Reactivity Initiated Accidents (RIA), and more especially Rod Ejection Accident scenario for PWR (REA), is a PPC-4[1] scenario for which the fuel's cladding is particularly stressed. It has some similarities with the rod drop accident (RDA) for Boiling Water Reactors (BWR). In this scenario, a large power pulse initiated by reactivity transient leads to a very rapid heating of the fuel. The first phase of the transient is a purely thermo-mechanical phenomenology leading to a Pellet-Clad Mechanical Interaction. It is followed by a potential large temperature increase of the cladding. This could lead to film boiling over the cladding: the so-called post-DNB (Departure from Nucleate Boiling) phase. Clad failure risk evaluation must cover both phases of the transient. For the post-DNB phase, the coupling between thermal-mechanics and thermal-hydraulics is strong: at high temperatures, the cladding yield stress strongly decreases, but meanwhile, large deformation cladding induced by internal pressurization thermally insulates cladding from hot fuel. Recent CSNI/WGFS fuel code benchmark showed a huge scattering of clad behavior as soon as DNB events occur, [40]. It led to recommendations in the recently updated version of the OECD RIA state of the art report [41]. The experimental database for RIA is still very limited as soon as realistic TH conditions for a PWR REA are concerned. This concerns both the prediction of the occurrence of the boiling crisis (transient CHF) and the post-DNB clad-to-coolant heat transfer. The on-going OECD CABRI International Program led by IRSN allows for simulating RIA in those conditions, [42]. It will provide new insights of these transient boiling conditions at convective high pressure and high temperature conditions.

In the field of research reactors with plate fuels for material testing, the BORAX scenario is currently considered. This reactivity-initiated accident could lead to massive melting of the core and subsequent vapor explosion. Determination of the thermal energy for this explosion was in the past estimated using conservative assumptions, [43]. CEA recently updated the safety demonstration method by considering a best-estimate simulation of the transient for the JHR project, [44]. In such a BORAX transient, determination of the void related reactivity feedback could play a role in mitigating the violence of the explosion. In those transient boiling conditions, DNB could occur. Validation of the coolant vaporization is clearly a challenge since *(i)* the deposited energy depends on the void creation and *(ii)* melting of the fuel elements depends on the clad-to-coolant heat transfer, both void creation and heat transfer varying in order of magnitude according to the boiling regime. A coupled (neutronics and thermal hydraulics) solving method is used to

[1] Plan Condition Category (PCC): PCC-4 corresponds to Category 4 Operating Conditions in the french set of events considered for the safety demonstration ; Category 4 accidents are hypothetical accidents nevertheless taken into account for safety purposes.

estimate the melting of the fuel elements. Validation of the methodology is based on the past SPERT-IV in-pile program, [45,46], and on a dedicated R&D program on transient boiling, e.g. [47].

EDF recently modified its safety demonstration regarding LOCA and Steam Generator Tube Rupture (SGTR) scenarios, [48]. This modification concerns the determination of the risk of failure of the fuel rods that experience DNB during the very first times of the transient. The demonstration now studies more precisely the potential consequences of the DNB in those specific conditions. During the assessment process, IRSN and EDF considered the review of past experimental programs. The Power Cooling Mismatch experiments program, [49], is of particular interest regarding the fuel behavior under high pressure high temperature post-DNB events and revealed very specific phenomenology like post-DNB induced bowing of the rods. This bowing could lead to potential consequences in the next phases of the considered transients. Such bowing is initiated by localized and transient azimuthal temperature gradients along the rod, [50]: prediction of its occurrence in an accidental scenario is a challenge that intimately couples local thermal-hydraulics and thermal-mechanics predictions. Complementary experiments could be required to validate the conditions of bowing occurrence for well characterized thermal conditions representative of post-DNB.

Therefore, it appears that post-DNB and fast transient boiling phenomenology are still challenges for some nuclear safety related issues. Many past experimental programs are very valuable for the analysis, but corresponding data can be insufficient for validation of best-estimate methodologies. Therefore, this topic can be identified for future TH R&D activities.

5. Conclusions

The not exhaustive but illustrative set of examples of the previous sections clearly illustrates that thermal-hydraulics for nuclear safety studies in water cooled reactors is still a very active field. Evolutions of either safety demonstration methods or scientific new knowledge provide renewed challenges. It also reveals that ongoing R&D considers a more extended validation range for coupling TH with neutronics and thermal-mechanics or for 3D TH computations. This requires revised PIRT and potentially a new experimental database.

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