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Post-test analysis of the ROSA/LSTF and PKL counterpart test

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HIGHLIGHTS

- TRACE modelization for PKL and ROSA/LSTF installations.
- Secondary-side depressurization as accident management action.
- CET vs PCT relation.
- Analysis of differences in the vessel models.

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ABSTRACT

Experimental facilities are scaled models of commercial nuclear power plants, and are of great importance to improve nuclear power plants safety. Thus, the results obtained in the experiments undertaken in such facilities are essential to develop and improve the models implemented in the thermal-hydraulic codes, which are used in safety analysis. The experiments and inter-comparisons of the simulated results are usually performed in the frame of international programmes in which different groups of several countries simulate the behaviour of the plant under the accidental conditions established, using different codes and models. The results obtained are compared and studied to improve the knowledge on codes performance and nuclear safety.

Thus, the Nuclear Energy Agency (NEA), in the nuclear safety work area, auspices several programmes which involve experiments in different experimental facilities. Among the experiments proposed in NEA programmes, one on them consisted of performing a counterpart test between ROSA/LSTF and PKL facilities, with the main objective of determining the effectiveness of late accident management actions in a small break loss of coolant accident (SBLOCA). This study was proposed as a result of the conclusion obtained by the NEA Working Group on the Analysis and Management of Accidents, which analyzed different installations and observed differences in the measurements of core exit temperature (CET) and maximum peak cladding temperature (PCT). In particular, the transient consists of a small break loss of coolant accident (SBLOCA) in a hot leg with additional failure of safety systems but with accident management measures (AM), consisting of a fast secondary-side depressurization, activated by the CET. The paper presents the results obtained in the simulations for both installations using TRACE, observing, in general, a good agreement with the experiments. However, ROSA/LSTF calculations underestimated the maximum PCT value, what might be explained by the higher core level predicted in the simulation compared with the experiment. In PKL calculations, PCT maximum value is slightly higher than in the experiment, and the core level predicted is lower. In the comparison of the evolution of both installations a different timing in the transient events is observed, due to the difference in the pressure vessel design. Thus, when PKL vessel is modified with some of the ROSA/LSTF features, the evolution of the new PKL model behaviour is closer the one observed in ROSA/LSTF calculations.

1. Introduction

Experimental facilities are of great importance in nuclear safety to improve the knowledge on commercial nuclear power plants behaviour under normal and accidental situations. Thus, it is possible to evaluate the evolution of the main safety variables under an

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Nomenclature

ACC	accumulators
AFW	auxiliary feedwater
AM	accident management
CET	core exit temperature
ECCS	emergency core cooling system
HPIS	high pressure injection system
LPIS	low pressure injection system
MSIV	main steam isolation valves
NEA	Nuclear Energy Agency
NV	normalized value referred to steady state
PCT	peak cladding temperature
PKL	Primarkreislauf Versuchsanlage
PV	pressure vessel
PWR	pressurized water reactor
ROSA/LSTF	Rig-of-Safety Assessment Large Scale Test Facility
RV	relief valves
SBLOCA	small break loss-of-coolant accident
SG	steam generator

accidental situation and identify generic issues that may affect the safety of nuclear power plants. Such facilities are scaled models of commercial plants, so the lessons learnt from the analysis of the safety variables evolution during the accidental sequence should serve to extend the knowledge on the behaviour of their nuclear power plant of reference, in order to improve nuclear power plants safety.

Moreover, the results obtained in the experiments undertaken in such facilities are essential to develop and improve the models implemented in the thermal-hydraulic codes. Thus, the data collected in the experiments are necessary in the assessment of the capabilities of thermal-hydraulic codes to reproduce the different physical phenomena that may take place inside the reactor in accidental situations. Such simulations are performed using best estimate thermal-hydraulic codes, as RELAP-5, TRAC, CATHARE, ATHLET or TRACE (Belaid and ZerkakVihavainen, 2010; Carlos et al., 2008; Freixa and Manera, 2010). Among these codes, RELAP-5 and TRAC have traditionally been used to reproduce transients of pressurized water reactors (PWR) and boiling water reactors, respectively. Nowadays, TRACE code (TRAC/RELAP Advanced Computational Engine) is being developed to make use of the most favourable characteristics of RELAP-5 and TRAC codes to simulate both, PWR and BWR, technologies.

The experiments and inter-comparisons of the simulated results are usually performed in the frame of international programmes in which different groups of several countries simulate the behaviour of the plant under the accidental conditions established, before the experiment, what is known as blind tests, and after the experiment, what are the post-test simulations. The results obtained are compared and studied to improve the knowledge on codes performance and nuclear safety. Thus, the Nuclear Energy Agency (NEA), in the nuclear safety work area, auspices several programmes which involve experiments in different experimental facilities (Carlos et al., 2011; Reventos et al., 2008; Gallardo et al., 2012). Among them, one can find the OECD/NEA Rig-of-Safety Assessment (ROSA-2) project and PKL-III project. The first one, was focused on the validation of simulation models and methods for the complex phenomena of high-safety relevance for thermal-hydraulic transients in design basis events (DBE) and beyond-DBE of light water reactors. To achieve this objective, different experiments are performed in the Rig-of-Safety Large Scale Test Facility (ROSA/LSTF) located in Japan, which represents a PWR Westinghouse design. The second programme was mainly focused on investigating safety issues relevant and complex heat transfer mechanisms in current pressurized water reactor PWR plants as well as for new PWR design concepts (Nakamura et al., 2009; Umminger et al., 2011; Jonnet et al., 2013). In this programme, the experiments were undertaken at Primarkreislauf Versuchsanlage facility (PKL) in Germany, which is a scaled model of a Konvoi PWR design.

Among the experiments proposed in those programmes, one of them consisted of performing a counterpart test between both installations. Thus, the experiment is undertaken in both installations to analyze the effect that different technology and scale may introduce in the evolution of the main safety variables under the same accidental situation. In particular, the transient consists of a small break loss of coolant accident (SBLOCA) in a hot leg with additional failure of safety systems but with accident management measures. The counterpart experiment was proposed as a result of the experiments performed by the NEA Working Group on the Analysis and Management of Accidents in different facilities (Toth et al., 2010). In those experiments a significant difference between core exit temperature (CET) and peak cladding temperature (PCT) evolution was observed in the measurements obtained in all the installations. Accident management (AM) measures are triggered by the CET values but the safety variable normally followed in nuclear safety studies is the PCT, so, to determine the effectiveness of the AM proposed, a more detailed study of both variables and the relationship between them was suggested. When a SBLOCA occurs the water inventory and pressure of the reactor coolant system decrease and this leads to empty the reactor pressure vessel, and to core uncovering. Therefore, it is necessary the actuation of the safety systems to inject water in the primary circuit, high pressure injection system (HPIS) in this case, to maintain the core full of water and cooled. The safety system failures postulated in this transient are no HPIS injection and no automatic secondary-side cooldown. This situation leads to core uncovering and the clad temperature increases until core-melt scenario if no action is performed. Therefore, it is necessary to explore the AM measures necessary to prevent this scenario. The AM measures proposed to prevent core melting is a fast secondary-side depressurization, initiated after core uncover to re-establish the steam generators secondary side as heat sink aiming for a fast reduction of the primary pressure, what permits the injection through the accumulators and makes possible the low pressure injection safety (LPIS) activation. These AM measures are activated by the core exit thermocouples measurements so, the use of CET as a valid criterion for the initiation of accident mitigation measures involving emergency operating procedures and/or severe accident management measures has to be assessed, and the efficiency of the accident mitigation measures proposed has to be analyzed (Belaid and ZerkakVihavainen, 2010).

In order to reproduce the same transient in both installations, a conditioning phase is needed to reach in ROSA/LSTF the same working conditions as the ones in PKL. Once this is achieved the break is produced and the physical phenomena occurring in both installations can be compared.

The paper is organized as follows: In Section 2, both facilities are presented and described. In Section 3 the experiment is explained. The TRACE model developed for each facility is exposed in Section 4. In Section 5, the results of the transient simulation are discussed. Finally, Section 7 presents the main conclusions obtained.

2. Experimental facilities

Both facilities are scaled models of different nuclear power plants. Thus, ROSA/LSTF is a scaled model of a Westinghouse and has Tsuruga-2 as reference plant and PKL is a Konvoi type and has