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Investigation of SAM measures during selected MBLOCA sequences along with Station Blackout in a generic Konvoi PWR using ASTECV2.0

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ABSTRACT

The Fukushima accidents have shown that further improvement of Severe Accident Management Guidelines (SAMGs) is necessary for the current fleet of Light Water Reactors. The elaboration of SAMGs requires a broad database of deterministic analyses performed with state-of-the art simulation tools. Within this work, the ASTECV2.0 integral severe accident code is used to study the efficiency of core reflooding (as a SAM measure) during postulated Medium Break LOCA (MBLOCA) scenarios in a German Konvoi PWR.

In a first step, the progression of selected MBLOCA sequences without SAM measures has been analysed. The sequences postulate a break in the cold leg of the pressurizer loop and the total loss of AC power at a given stage of the accident. Results show the existence of a 40 min grace time up to the detection of a Core Exit Temperature (CET) of 650 °C providing that the AC power has been maintained at least 1 h after SCRAM.

In a second step, an extensive analysis on core reflooding has been carried out. The sequences assume that the plant remains in Station Blackout (SBO) and that reflooding occurs at different times with different mobile pumps. The simulations yield the following results:

- Reflooding mass flow rates above 60 kg/s have to be supplied as soon as the CET exceeds 650 °C in order to prevent core melting.
- Reflooding mass flow rates ranging from 25–40 kg/s at CET = 650 °C mitigate the accident without major core damage depending on when the plant enters in SBO.
- Reflooding mass flow rates lower than 10 kg/s cannot prevent RPV failure.

The performed investigations elucidate the ASTECV2.0 capabilities to describe the in-vessel phase of a severe accident in a German Konvoi PWR and to assess the performance of core reflooding for slightly degraded cores. Moreover, they form the basis of future analysis on sequences with a higher contribution to the overall risk of such nuclear plant.

1. Introduction

The Defence-in-Depth is the key principle for all Nuclear Power Plants (NPPs) operated worldwide. This principle establishes the necessity of deploying safety levels containing diverse provisions aiming at maintaining the integrity of the cladding, vessel and containment, thereby avoiding any harmful release of radioactive material to the public. Despite the events occurred at Fukushima overwhelmed the provisions allotted to each safety level, the fundamental concepts of the Defence-in-Depth are still valid. However, it has been agreed that further improvements at each safety level must be made (NEA-OECD, 2013).

In this connection, the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMUB) asked the German Reactor Safety Commission (RSK) to carry out a comprehensive safety review of all operating NPPs. This study attested a high level of robustness to all German NPPs (BMUB, 2011) and formed the basis for later RSK recommendations aiming at improving existing safety margins (BMUB, 2016, 2014, 2012). The focus

Abbreviations: ACCUs, accumulators; AM, Accident Management; CET, Core Exit Temperature; ECCS, Emergency Core Cooling System; FOM, Figure of Merit; HPIS, High Pressure Injection System; LP, lower plenum; LPIS, Low Pressure Injection System; MBLOCA, Medium Break LOCA; MCP, Main Coolant Pump; NPP, Nuclear Power Plant; RCPS, Reactor Control Protection System; RCS, Reactor Cooling System; RMFR, reflooding mass flow rate; RPV, Reactor Pressure Vessel; SAM, Severe Accident Management; SAMGs, Severe Accident Management Guidelines; SBO, Station Blackout; SG, Steam Generator.

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was especially put on the improvement of Accident Management (AM) measures, particularly during situations involving a Station Blackout (SBO).

In Germany, AM measures to prevent core degradation are stipulated in the "*Notfallhandbuch*" (Emergency Operating Procedures (EOPs)) (KTA, 2009) while mitigative measures are described in the "*Handbuch für Mitigative Notfallmβnahmen*" (Severe Accident Management Guidelines (SAMGs)) (Braun et al., 2014). The technical basis for the development and optimization of such measures are provided by deterministic analyses using state-of-the-art (Klein-Heßling et al., 2014) validated severe accident codes. Consequently, the improvement, validation and application of integral severe accident codes is pursued worldwide (European Commission, 2017; Van Dorsselaere et al., 2015).

The ASTEC code (Chatelard et al., 2014), jointly developed by *Institut de Radioprotection et de Sûreté Nucléaire* (*IRSN*) and *Gesellschaft für Anlagen und Reaktorsicherheit (GRS)*, is able to simulate the complete evolution of a severe accident sequence i.e. from the initiating event till the release of radioactive material from the containment. In the frame of the EU CESAM project (GRS, 2017), ASTEC is being extensively validated, applied and improved to support the development of SAMGs for current European NPPs and of the new ASTECV2.1 version (Chatelard et al., 2016), characterized by enhanced numerical robustness and improved physical models.

Within the CESAM project, the Institute of Neutron Physics and Reactor Technology (INR) at the Karlsruhe Institute of Technology (KIT) is devoted to the extension of the technical basis for the improvement of AM measures in a German Konvoi PWR by means of ASTEC calculations. These studies consider the outcomes of the Probabilistic Safety Analysis for the German Konvoi PWR (GRS, 2002) and the recommendations issued by RSK after Fukushima (BMUB, 2014). Within this work, the effectiveness of core reflooding as a Severe Accident Management (SAM) measure is investigated by means of ASTECV2.0 (rev3). The sequences assume a Medium Break LOCA (MBLOCA) in the cold leg of the pressurizer loop as well as the total loss of AC power at a given stage of the accident.

Despite the limited contribution of Medium Break LOCAs to the Core Damage Frequency in German Konvoi PWRs (GRS, 2002), this sequence is ideal to test ASTECV2.0 capabilities to simulate reflooding, due to the lack of repressurization in the reactor upon water injection. During such sequences a rapid depressurization of the primary side is expected and hence, water injection into the reactor becomes the most imperative SAM measure (Braun et al., 2014). However, since the plant undergoes a Station Blackout, water injection from the active safety systems is no longer possible, which opens the possibility of considering an external water injection by means of portable equipment as an alternative choice.

2. The ASTEC code

The integral severe accident code ASTECV2.0 simulates complete severe accident sequences in water-cooled reactors (Chatelard et al., 2014). Its application range covers source term determination, Probability Safety Assessment and assessment of SAM efficiency. The structure of ASTEC is modular, each module considering a domain of the reactor or a set of physical phenomena. In this work, the CESAR and the ICARE modules have been used.

The CESAR module uses a 5-equation modelling approach to describe one-dimensional two-phase thermal-hydraulics throughout the primary and secondary circuit, including the Reactor Pressure Vessel (RPV) up to the beginning of core degradation (Chatelard et al., 2014). From that point in time, CESAR calculates the thermal hydraulics in the aforementioned domain except in the RPV, which is handled by ICARE. The equations are broken down in 2 energy conservation equations to calculate the temperature of the liquid and the gas; 2 + N mass conservation equations to calculate the mass of liquid, steam and N non-condensable gases and 1 momentum equation to calculate the average velocity of liquid and gas. In addition, CESAR makes use of an algebraic equation to calculate the drift between the liquid and the gaseous phase. Therefore, phenomena such as the counter current flow of water in ascending steam cannot be properly reproduced, which limits the study of water injection into the hot legs of the reactor.

The ICARE module takes over the thermal-hydraulics in the RPV when certain criteria are fulfilled e.g. mass of non-isolated accumulators and temperature at the upper plenum (Chatelard et al., 2014). Similarly to CESAR, ICARE uses a 5-equation modelling approach to describe 1-D thermal-hydraulics in the core region. This spatial resolution is valid during the first stages of degradation, but it is no longer adequate when significant corium blockages have been formed in the core or in the lower plenum. In such case, the coolant flow patterns are mostly 2-D, which requires the calculation of the cross flows between adjacent channels.

The reflooding model of ASTECV2.0 is devoted to bottom to top reflooding assuming that the core geometry is sufficiently intact so that it can be treated with a 1-D approach. The basic idea is to calculate the heat flux downstream of the quench front, first by applying a heat transfer corresponding to a prescribed boiling curve downstream of the quench front; then, by integrating this curve considering that the heat flux at the quench front location is equal to the Critical Heat Flux (Chikhi and Fichot, 2010). The model has been validated up to slightly-degraded core conditions (Chikhi et al., 2012; Chikhi and Fleurot, 2012). For advanced stages of degradation, the quench front would progress through a porous medium (debris particles, corium) and hence, the model is no longer valid.

The previous reasons limit ASTECV2.0 capabilities to evaluate the efficiency of core reflooding for degraded cores (i.e. more than 20 corium tons in the core region for this work). Therefore, simulations involving such situations shall be verified with the new major version ASTECV2.1, characterized by a 2-D treatment of the thermal-hydraulics in the RPV and a new reflooding model for degraded cores (Chatelard et al., 2016).

3. Generic PWR Konvoi plant model with ASTEC V2.0

The German Konvoi PWR plant consists of 4 loops, one of which contains the pressurizer. Each loop consists of a hot leg, a Steam Generator, a cold leg and a Main Coolant Pump (Siemens, 1991). The different domains of the generic plant model of the German Konvoi in ASTECV2.0 are described hereafter. Details about the containment and the rest of modules of the ASTECV2.0 generic model can be found in Nowack et al. (2011).

3.1. Primary and secondary circuits

The four loop PWR is represented by two loops, in which the loop B is connected to the pressurizer and the loop A represents the other three loops. A simplified sketch of the primary and secondary circuits (together with relevant safety systems for this work) is depicted in Fig. 1. The discretization of the loop A is analogue to the one of loop B, except for the pressurizer and the surge line.

The primary coolant exits the core from the upper plenum to the hot legs and to the upper head of the RPV. Once the coolant enters the hot leg, it flows towards the Steam Generator (SG) inlet