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Severe accident management measures for a generic German PWR. Part I: Station blackout

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This paper focuses on analysis of severe accident management measures for a generic German PWR of type Konvoi. A nuclear power plant model based on the severe accident code ATHLET-CD was developed in order to assess the code applicability for simulation of accident scenarios with core degradation. It was applied for investigation of two main groups of accident scenarios: station blackout and small-break loss-of-coolant accident.

Part I of series of two papers analyses the plant response in case of hypothetical station blackout severe accident. Assessment of accident management measures in the preventive and in the mitigative domain is performed, where a focus is given on the combination of primary pressure reduction and injection by portable equipment directly into the reactor circuit. Key timings for operator actions are deduced. Both positive and negative effects of the investigated accident management measures are discussed. The results from the station blackout simulations showed that the time until core degradation can be delayed by application of primary side depressurization and usage of mobile pump as accident management measures. Depending on the time of injection significant reduction of the hydrogen and fission products releases can be obtained. In case that early water injection is possible, severe core damage might be prevented.

1. Introduction

Research related to accident management measures (AMMs) and important criteria for the accident management procedures, especially in the preventive domain for pressurized water reactors (PWRs) is documented in (Roth-Seefrid et al., 1994; Muellner et al., 2007; Andreeva et al., 2008; Cherubini et al., 2008; Tusheva et al., 2012, 2014b). Accident management strategies and measures to prevent or to mitigate an ongoing accident received substantial reflection after the Fukushima accident (2011), which evoked extensive European nuclear power plant (NPPs) checks, stress tests, focussing on the assessment of the risk and safety of the plant and the plant capabilities to cope with severe accidents (ENSREG, 2011, 2012).

For German NPPs, the implementations based on the stress tests are included in the (BMU, 2011) report, where additional consideration is given to natural hazards, loss of power, loss of water supply, effects of the accident on neighbouring unit etc. Within the AMMs domain core cooling and identification of safety margins are included. Considering management of an accident the focus is put primarily on the prevention of core damage and only in case of unsuccessful preventive measures – on mitigation of core damage. Specifically for the case of loss of power supply, where "no need for measures to further increase the robustness were identified", nevertheless, additional measures to "increase the robustness of the power supply" for primary bleed and feed as AMMs, as well as installation of mobile equipment for water delivery to the secondary side are considered (BMU, 2011).

In parallel, research projects focussing specifically on analysis of core degradation scenarios, with various spectrum of initiating events, and the impact of applied AMMs supported by computer code simulations were initiated. One of those projects was the 2013–2016 joint research project WASA-BOSS (Weiterentwicklung und Anwendung von Severe Accident Codes – Bewertung und Optimierung von Störfallmaßnahmen/Further Development and Application of Severe Accident Codes – Assessment and Optimization of Accident Management Measures) of the German Federal Ministry of Education and Research (Beck et al., 2014; Tusheva et al., 2015b; Jobst et al., 2017). Depending on the reactor type, severe accident analysis and investigation of AMMs for PWRs (Trometer et al., 2014; Tusheva et al., 2015b; Wilhelm et al., 2016) and boiling

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Nomenclature

AC	Alternating current
ACCU(s)	Accumulator(s), passive safety system
(S)AMM(s)	(Severe) Accident Management Measure(s)
ATHLET(-CD)	Analyse der Thermohydraulik von Lecks und Transienten/Analysis of thermal-hydraulics of leak and transients (with core degradation)
ASTOR	Approximated structural time of rupture
BWR	Boiling water reactor
CET	Core exit temperature
CL	Cold leg
ECC(S)	Emergency core cooling (system)
EOC	End of cycle
FPs	Fission products
GCSM	General control simulation module within ATHLET
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH
HL	Hot leg
HPIS	High pressure injection system
HZDR	Helmholtz-Zentrum Dresden - Rossendorf e.V.
IKE	Institut für Kernenergetik und Energiesysteme der Universität Stuttgart
LOCA	Loss-of-coolant accident
LPIS	Low pressure injection system
MCP	Main coolant pump
MPI	Mobile pump injection
NPP	Nuclear power plant
PRZ	Pressurizer
PSD	Primary side depressurization
PWR	Pressurized water reactor
RPV	Reactor pressure vessel
SCRAM	Safety Control Rods Activator Mechanism
SBLOCA	Small-break LOCA
SBO	Station blackout
SG	Steam generator
SSD	Secondary side depressurization
WASA-BOSS	Weiterentwicklung und Anwendung von Severe Accident Codes – Bewertung und Optimierung von Störfallmaßnahmen

water reactors (BWRs) (Di Marcello et al., 2016) were performed. Within the frames of the EU CESAM project the impact of severe accident management measures (SAMMs) for German PWRs was also investigated (Gremme and Koch, 2015; Gómez-García-Toraño, 2017; Gómez-García-Toraño et al., 2017; Sanchez-Espinoza et al., 2017).

On the basis of our previous work, focused on eastern type PWRs – VVERs (Tusheva et al., 2012, 2014b, 2015a), we have extended the analyses and the current paper is focused on western types PWRs, considering generic German PWR of type Konvoi. In this paper we discuss the elaboration of the model (the input deck), application of the model for scenarios involving core degradation, assessment of the capabilities of the applied code and assessment and optimization of AMMs. Timings for operators' intervention are also deduced.

Severe accidents analyses of two accident categories were carried out: station blackout (SBO) and small-break loss-of-coolant accident (SBLOCA). The entire transient from the initiating event until the reactor pressure vessel (RPV) failure, considering main severe accident phenomena is simulated with the severe accident code ATHLET-CD: start of core heat-up, cladding failure, release of fission products (FPs, in-core), fuel and absorber material melting, oxidation and release of hydrogen, molten material relocation in the core and to the lower plenum, as well as failure of the RPV. The model was applied for the analysis of preventive and mitigative AMMs for SBO and SBLOCA transients.

Part I of series of two papers is focused on the analysis of SBO severe accident scenario. Detailed analysis of the accident progression without and with application of AMMs and assessment of the effectiveness of the applied measures is performed with a focus on the plant response, assessment of AMMs, effectiveness of the applied AMMs and assessment of grace periods. Investigated AMMs are primary bleed and feed (primary side depressurization (PSD)) and injection by mobile equipment, cold and hot side injection of borated water, hypothetically to the primary side. The mobile equipment is considered as an independent from the plant safety systems possibility for injection. As our analysis is solely based on analytical research to investigate the influence of late water injection on the accident progression in a way to deduce conclusions on assessment of time margins and success of the measure, at that stage of the analysis no consideration was taken about a practical implementation and connection at the NPP. Variation of the initiation criteria for PSD and injection were also analysed. The time evolution of the accident and relevant margins are assessed. The assessment of the time spans was supported by an uncertainty and sensitivity analysis for the early phase of SBO with application of PSD prior core melt (Kozmenkov et al., 2017). The results of the SBLOCA 50 cm^2 analysis are reported in Part II of the paper (Jobst et al., 2018). In addition, the performed numerical analyses contribute to the ATHLET-CD code quantification for plant analysis and application of the code within the accident management domain.

Part I of the paper includes a short overview on the applied ATHLET-CD modules (Section 2), description of the plant model (Section 3), scenario description and assumptions for the SBO simulations (Section 4), discussion of the SBO results (Section 5) and the main findings are summarized in Section 6.

2. The ATHLET-CD code and applied modules for the simulations

The analyses were performed with the severe accident computer code ATHLET-CD (Core Degradation), Mod 3.0 Cycle A (Austregesilo et al., 2013). It is developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Germany, in collaboration with the University of Stuttgart. The structure of the code is modular, as (Trambauer et al., 2001, 2009; Bals et al., 2012; Lerchl et al., 2012; Austregesilo et al., 2013):

- The thermal-hydraulics is simulated by the ATHLET (Analysis of THermal-hydraulics of LEaks and Transients) code;
- The GCSM module (General Control Simulation Module) models the signals for the NPP control and activation of safety systems;
- Reactor core and the core degradation processes are modelled by the ECORE module, including heating up of the core, cladding oxidation and degradation of the reactor core;
- The core quenching processes are modelled by the QUENCH-CORE module;
- The release of FPs is calculated by the FIPREM module (rate equations model). ORNL data (Lorenz and Osborne, 1995) are used as a basis;
- The initial FPs and actinides inventories are simulated by the OREST module, based on the configuration of the core, enrichment of the fuel, burnup, power history. The output data from the OREST module are used as input for the FIPISO module;