STUDY ON SEVERE ACCIDENTS AND COUNTERMEASURES FOR VVER-1000 REACTORS USING THE INTEGRAL CODE ASTEC

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ABSTRACT

The research field focussing on the investigations and the analyses of severe accidents is an important part of the nuclear safety. To maintain the safety barriers as long as possible and to retain the radioactivity within the airtight premises or the containment, to avoid or mitigate the consequences of such events and to assess the risk, thorough studies are needed. On the one side, it is the aim of the severe accident research to understand the complex phenomena during the in- and ex-vessel phase, involving reactor-physics, thermal-hydraulics, physico-chemical and mechanical processes. On the other side the investigations strive for effective severe accident management measures.

This paper is focused on the possibilities for accident management measures in case of severe accidents. The reactor pressure vessel is the last barrier to keep the molten materials inside the reactor, and thus to prevent higher loads to the containment. To assess the behaviour of a nuclear power plant during transient or accident conditions, computer codes are widely used, which have to be validated against experiments or benchmarked against other codes. The analyses performed with the integral code ASTEC cover two accident sequences which could lead to a severe accident: a small break loss of coolant accident and a station blackout. The results have shown that in case of unavailability of major active safety systems the reactor pressure vessel would ultimately fail. The discussed issues concern the main phenomena during the early and late in-vessel phase of the accident, the time to core heat-up, the hydrogen production, the mass of corium in the reactor pressure vessel lower plenum and the failure of the reactor pressure vessel. Additionally, possible operator's actions and countermeasures in the preventive or mitigative domain are addressed. The presented investigations contribute to the validation of the European integral severe accidents code ASTEC for VVER-1000 type of reactors.

1. INTRODUCTION

It is the main objective of the nuclear safety to maintain the barriers for the retention of radioactivity in order to protect the workers and the public from the hazards of nuclear radiation. The safety functions required to achieve this fundamental protection goal are implemented by operational and safety systems. All these systems are elements in a staggered safety concept, known as the defence-in-depth concept, which is meant to optimize the interaction of the manifold systems with respect to their effect on the safety of the nuclear power plant, i.e. the ultimate goal to maintain the effectiveness of the physical barriers. In case that during an accident the borders of this safety concept are violated (e.g. due to multiple failures of safety systems or due to external hazards) a progression into a severe accident, characterized by a degradation of the reactor core, is possible.

Usually, the safety case has to be demonstrated for design basis accidents (DBA) and for beyond design basis accidents (BDBA) /IAEA: 23/, /IAEA: NS-R-1/. The design basis accidents represent a class of postulated accidents to which the safety systems and functions of the nuclear power plant had been designed in order to control the event and to exclude the release of harmful amounts of radioactivity. For beyond design basis accidents with unavailability of major safety systems, accident management measures can be applied to terminate the progression of the accident into a severe one or to mitigate the consequences of the severe accident. Two accident management measures are discussed in detail: depressurization of the reactor coolant circuit and subsequent feeding of the primary side by use of passive systems.

For the current studies the ASTEC code, which is jointly developed by IRSN and GRS since several years, was applied for the investigation of the whole accident progression till failure of the reactor pressure vessel. With the help of the ASTEC code simulations the course of the events without application of accident management measures has been investigated for a loss of coolant accident and a station blackout accident (Part 2). As an example of possible accident management measures and strategies, the effect of the primary bleed and feed strategy has been analysed for the station blackout case (Part 3). This investigation has been performed within the framework of the SARNET Project WP4 RAB under the EURATOM 6th framework programme.

2. ACCIDENT MANAGEMENT SCENARIOS WITHOUT APPLICATION OF AACCIDENT MANAGEMENT MEASURES

2.1. Small Break Loss of Coolant Accident (SBLOCA)

The ASTEC v1.2 version of the code has been applied. The activated modules are CESAR for the thermal-hydraulic behaviour in the circuits, DIVA for the core degradation and CPA for the containment. In order to achieve proper initial conditions for the transient calculation steady-state iteration has been applied. After 500 s a stable stationary state has been obtained. The imposed reactor nominal parameters are controlled during steady-state by regulation procedures for pressurizer water mass, primary and secondary mass flow, as well as system pressure and temperatures.

A 60 mm leak in the cold leg of the pressurizer loop next to the reactor pressure vessel was simulated. All active emergency core cooling systems and the hydro-accumulators are assumed to fail. Failure is assumed also of the secondary side heat removal (BRU-K valves, BRU-A valves, secondary cooldown procedure). This calculation has been performed for a total “problem time” of 8.3 hours (about 30000 s) therefore it is focused only on the “in-vessel phase” of the accident (up to vessel failure).

At 0 s simulation time a 60 mm SBLOCA occurs in the cold leg. The SCRAM, TURBINE TRIP and MCP OFF signals are actuated by the reactor protection system. The turbine isolation valve closes 10 seconds after the reactor SCRAM signal. As a consequence, the feedwater supply to the steam generators stops, being provided by the turbine driven pumps. Figure 1 to Figure 4 depict the general trend of the accident sequence.

In the beginning of the accident sequence (during the first 250 s), the primary pressure decreases very rapidly due to the loss of coolant through the break. In Figure 1 are depicted the reactor coolant system primary and secondary pressure. With P-prima is noted the primary pressure, with P-secon – the secondary side pressure and UP-plenu stands for the pressure in the upper plenum of the reactor. The leak mass flow leads to a decrease of the pressurizer level as well as the water level in the reactor.

At the beginning of the accident the secondary pressure is immediately increasing and at around 60 s, when the set points for opening of the secondary side safety valves are reached, the steam generator safety valves open and start cycling. From approximately 250 s till 2500 s the primary pressure follows the secondary pressure. The decreasing primary inventory and the increasing void fraction at the reactor outlet lead to deteriorated primary to secondary heat transfer. Due to the increasing void fraction in the steam generators’ primary side (U-tubes) after 2500 s the primary to secondary heat transfer breaks down and the primary pressure decreases below the secondary pressure. From this time the decay heat is only removed via the leak. The steam generators are in hot standby. The continuous loss of coolant leads to the beginning of core heat-up. With the propagation of the dry out, and without injection from the emergency core cooling system, the core starts to heat-up, the fluid and the rod temperatures start rapidly to increase.

The break behaviour can be described as follows. At the beginning of the transient only single phase fluid (water) is released through the break, thus the reactor pressure vessel level starts to decrease. Voiding is observed in the reactor upper plenum and in the primary loops and starts to increase. The void fraction at the break position (in the cold leg) starts also to increase and after that there is a release of two-phase fluid through the break. Later on only single-phase fluid (steam) is released through the break.

From approximately 4000 s the continuous loss of coolant leads to heating-up of the core (Figure 2). After core uncovery the fuel temperature increases due to the low heat transfer to steam. At about 6000 s up to 10000 s the simulation shows an increase of the primary pressure due to corium slumping (Figure 1 and Figure 3). Extended dry-out in the reactor core is observed. The exothermic reaction of steam and Zr-cladding at elevated temperatures results in oxidation, hydrogen release and intensified heat-up. The beginning of the noticeable hydrogen generation (time moment by which > 1 % of hydrogen is released) corresponds to 4900 s. Figure 4 shows the in-vessel hydrogen mass released during the degradation of the core. Fe stands for the hydrogen released by the iron oxidation, B4C for the amount of hydrogen released by the boron carbide oxidation, UO2-Zr for uranium dioxide-zirconium oxidation and Total stands for the total hydrogen release. The total mass of the released hydrogen by the end of the reviewed time interval is 252 kg.

If the accident is unmitigated, the molten material of the degraded core relocates to the lower regions of the core. Figure 3 shows the mass of the relocated corium in the lower plenum of the reactor pressure vessel. In the bottom head of the reactor the molten corium pool starts slowly to ablate the reactor vessel wall.

From the thermal-hydraulic point of view the typical phases of a SBLOCA accident like the depressurization phase, the coupling of primary and secondary pressures and the break down of primary to secondary heat transfer followed by a decrease of the primary pressure below the secondary pressure are predicted very well. The major parts of expected severe accident phenomena in VVER-1000 were modelled in ASTEC v1.2 calculation, namely the core heat-up, the cladding oxidation, the degradation of the core components, the melt relocation and the vessel failure. The lower head vessel failure time predicted by ASTEC is at around 17995 s (~ 5 h) after SCRAM.

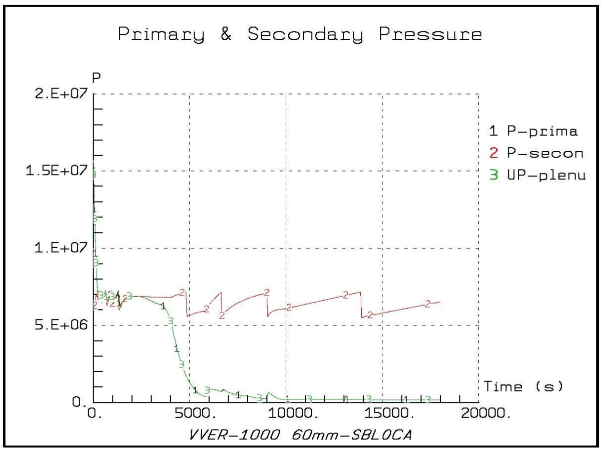

Figure 1: ASTEC v1.2, SBLOCA 60mm, Primary and secondary pressure

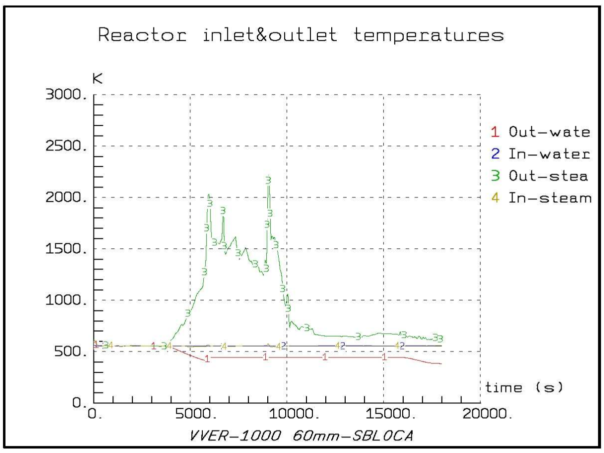

Figure 2: ASTEC v1.2, SBLOCA 60mm, Reactor inlet and outlet temperature

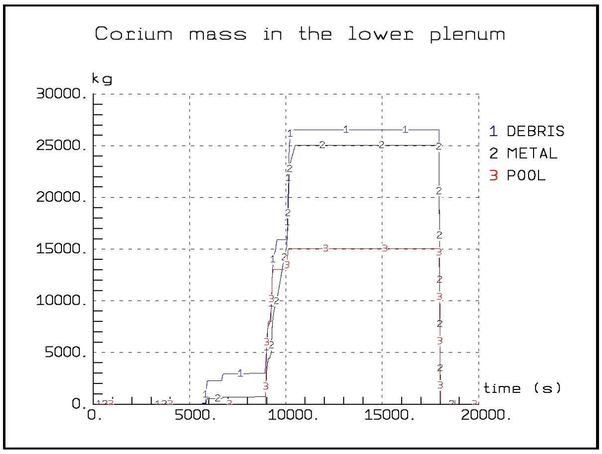

Figure 3: ASTEC v1.2, SBLOCA 60mm, Corium mass in the lower plenum

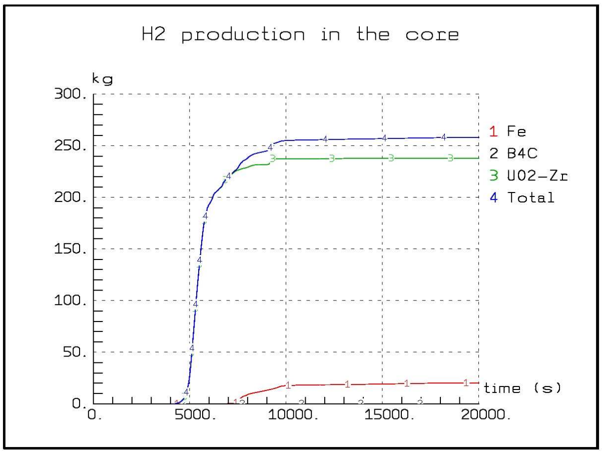

Figure 4: ASTEC v1.2, SBLOCA 60mm, Hydrogen production

2.2 Station Blackout (SBO)

The ASTEC v1.3.0 calculation has been performed assuming an SBO event (loss of the offsite electric power concurrent with a turbine trip and unavailability of the emergency AC power system), leading to unavailability of all major active safety systems /TUS 2010/. The analysis has been restricted to the "in-vessel phase" with a total "problem time" of 8.3 hours (about 30000 s). The simulation has started at nominal operating conditions. The BRU-A valves, the pressurizer relief and safety valves and the passive safety injection system (hydro-accumulators) were assumed available. The active safety injection systems were assumed unavailable, too.

The main parameter trends are depicted on Figure 5 to Figure 8. After the reactor SCRAM and main coolant pumps coast down, the gravitational forces are dominating the flow and the distribution of coolant inside the primary system. The primary side pressure is decreasing for a short time as a result of the reactor SCRAM and the decreasing reactor power.

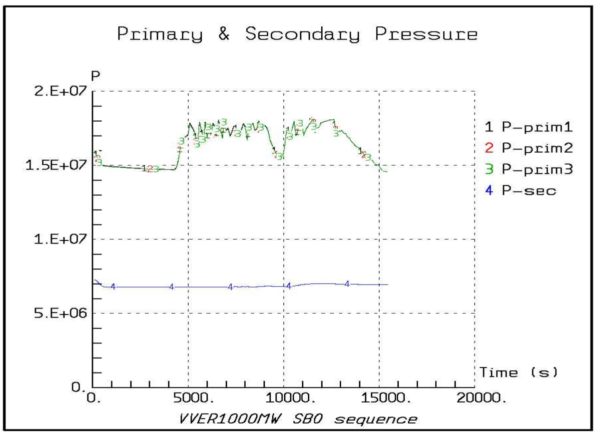

Figure 5: ASTEC SBO, no AMM, Primary and secondary pressure

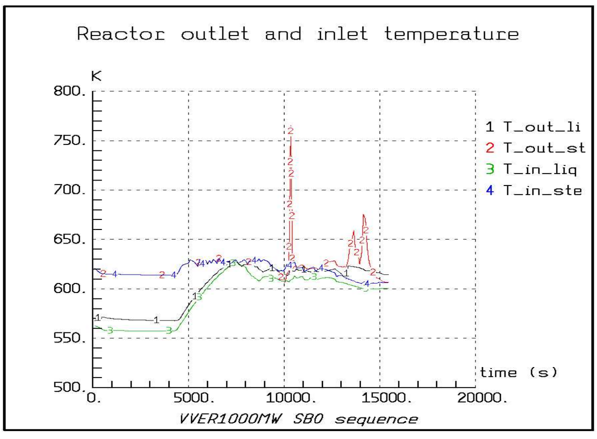

Figure 6: ASTEC SBO, no AMM, Reactor inlet and outlet temperature

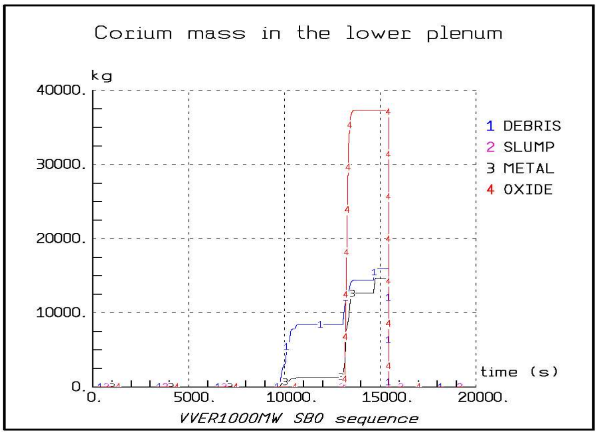

Figure 7: ASTEC SBO, no AMM, Corium mass in the lower plenum

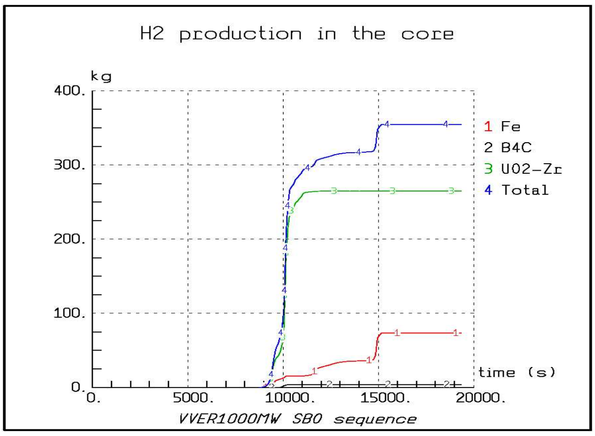

Figure 8: ASTEC SBO, no AMM, Hydrogen production

Due to the isolation of the secondary side, the secondary pressure increases and the decay heat is removed from the primary circuit via the steam generators by opening the BRU-A valves, leading to a continuous decrease of water inventory on the secondary side of the steam generators. During this phase the primary pressure remains nearly constant at around 15 MPa. After the depletion of the steam generators (4310 s), the primary to secondary heat removal breaks down and an instantaneous pressure increase on the primary side occurs (Figure 5). Later on, the core undergoes a high-pressure boil-off. In Figure 5 P-prim stands for the primary pressure and P-sec stands for the secondary pressure. After reaching the set point of the pressurizer relief valve continuous depletion of the primary coolant inventory through the valves uncovers the core and leads to temperature increase in the core (Figure 6). In Figure 6 T_out_liq/T_out_st correspond to the outlet liquid/steam temperatures and T_in_liq/T_in_st are the inlet liquid/steam temperatures, respectively. The fuel cladding temperature is also increasing and at temperatures above 1100 °C the Zirconium in the fuel cladding is oxidized by steam.

The amount of energy that is released during the steam-oxidation reaction is comparable to the residual power. A layer of ZrO_2 is forming on the external cladding surface. Strong hydrogen production is observed (Figure 8). Because of the significant decay heat and the high pressure in the primary side of the reactor coolant system, and due to the unavailability of emergency coolant injection, the necessary core cooling is not provided. At approximately 8750 s the core starts to heat up and subsequently the maximum allowed cladding temperature threshold of 1200 °C is exceeded. Without any additional measures the accident is turning into a severe accident. The reactor pressure vessel failure is at 15452 s from the beginning of the transient simulation. Figure 7 depicts the relocated corium mass in the lower plenum.

3. ACCIDENT MANAGEMENT MEASURES

3.1. General Overview

During severe accident sequences it is of prime importance to depressurize the primary circuit in order to allow injection from the passive and/or active safety injection systems and to avoid reactor pressure vessel failure at high pressure that could cause direct containment heating and subsequent challenge to the containment structure. Accident management procedures are to be applied both in the preventive and in the mitigative domain of an accident sequence. The main objectives are to preserve the integrity of the primary circuit, the pressurized premises and to create conditions for cooling down of the core or of the debris after core damage. The time factor until core damage is a key issue when assessing accident management procedures.

To make available the water delivery to the core by means of active or passive safety systems, or by external sources, in both the preventive and mitgative domain two accident management measures can be applied to reduce the pressure in the reactor coolant system and to start subsequent feeding of the primary and/or secondary side: primary side depressurization (PSD) and secondary side depressurization (SSD). In LOCA sequences, secondary cooldown procedures can be activated to reduce the pressure on both the secondary and the primary side. In case of unavailability of the cooldown procedure, depressurization by manual opening of relief or safety valves can be applied. Especially in SBLOCA sequences with limited decay heat removal via the steam generators and the leak, and unavailability of the high pressure injection system, the depressurization of the reactor coolant circuit could help to delay a possible core heat-up by earlier injection from the hydro-accumulators and/or the low pressure injection system.

Overview on the basic severe accident management actions is given in /SARNET 2006/, /SARNET 2008/, /IRSN 2007/. Cooling a degradated core into the primary circuit would help in the process of stopping the core degradation and retention of the degraded core materials inside the reactor pressure vessel. The procedure is water delivery to the core as soon as possible. Another strategy is flooding of the reactor pressure vessel compartments. The strategy of in-vessel retention by ex-vessel cooling is one of the adopted strategies for mitigation of a severe accident /KYM 1997/, /BEC 2008/. For the containment management of the combustible gases is applied. This strategy is applied to reduce the H2 and CO inventory in the containment. Management of the containment temperature and pressure is performed for keeping the containment integrity. This strategy is realized by the automatic or manual usage of the containment sprays. Management of the radioactivity releases is an additional measure. The purpose is reduction of the containment pressure.

3.2. An example of the application of accident management measures

For the SBO accident the effect of the depressurization of the primary circuit as an accident management measure is studied in an additional simulation. The accident scenario is an SBO, as the general trend of the sequence without application of accident management measures is described above in Part 2.2. The primary side depressurization is applied when the core outlet temperature reaches 650°C /TUS 2008/, /TUS 2010/. The measure is realized by fully opening of the pressurizer relief and safety valves. The pressurizer valves are activated intentionally to reduce the primary pressure. After their fully opening, the primary pressure drops rapidly below the hydro-accumulators' pressure allowing passive feeding to the primary circuit.

Figure 9 shows the primary pressure behaviour for the time till failure of the reactor pressure vessel. The blue curve describes the case without application of accident management measures. The red curve describes the case with application of accident management measures i.e. primary side depressurization. The two bars below the pressure curves visualize the temperature evolution and indicate the time margins for core heat up and initiation of the accident management measure as well as the vessel failure time. The heating up of the core is calculated approximately after 2 hours and 26 minutes. The two accident paths show vessel failure under high pressure after 4 hours and 18 minutes (blue curve, the case without accident management measures) and vessel failure under low pressure after 5 hours and 48 minutes (red curve, the case with primary side depressurization). With application of the accident management measure ASTEC predicts a prolongation of the vessel failure time by approximately 90 minutes. Slowing the core damage would allow more time for systems to be recovered to mitigate or terminate the accident. This would give more time and different possibilities for operator interventions.

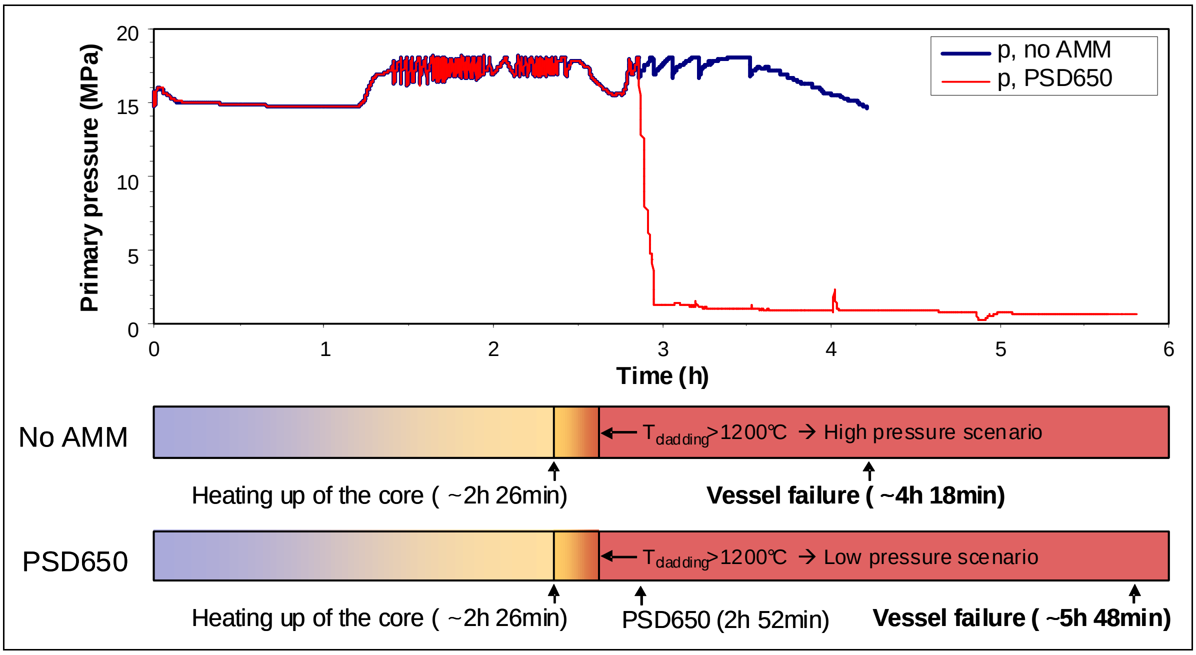

Figure 9: ASTEC SBO, without and with AMM, assessment of the time margins till vessel failure

4. SUMMARY OF THE RESULTS

The work presented in this paper has focused on analysis of the in-vessel phase of two hypothetical severe accident sequences in a nuclear power plant with a VVER-1000/V-320 reactor with the integral computer code ASTEC. Study on an SBLOCA and an SBO scenario without accident management measures has been realized. The code results have shown that in case of unavailability of active safety systems and without additional measures in both scenarios the reactor pressure vessel would fail. By application of primary side depressurization as an accident management measure during the SBO scenario, the reactor pressure vessel failure can be significantly delayed, but without restoration of power supply it cannot be fully avoided.

The results demonstrate the applicability of the integral code ASTEC for the simulation of entire severe accident sequences, covering all main phenomena including the application of AMMs.

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NOMENCLATURE

AMM	Accident Management Measure
ASTEC	Accident Source Term Evaluation Code
B(DBA)	(Beyond) Design Basis Accident
BRU-A	Steam-Dump to Atmosphere
BRU-K	Steam-Dump to Condenser
MCP	Main Coolant Pump
(SB)LOCA	(Small Break) Loss of Coolant Accident
SBO	Station Blackout
PSD	Primary Side Depressurization
SSD	Secondary Side Depressurization
RAB	Reactor Application and Benchmarking
SARNET	Severe Accidents Research Network
WP	Work Package

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