INVESTIGATION ON ACCIDENT MANAGEMENT MEASURES FOR VVER-1000 REACTORS

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ABSTRACT

A consequence of a total loss of AC power supply (station blackout) leading to unavailability of major active safety systems which could not perform their safety functions is that the safety criteria ensuring a secure operation of the nuclear power plant would be violated and a consequent core heat-up with possible core degradation would occur.

Currently, a study which examines the thermal-hydraulic behaviour of the plant during the early phase of the scenario is being performed. This paper focuses on the possibilities for delay or mitigation of the accident sequence to progress into a severe one by applying accident management measures (AMM). The strategy "Primary circuit depressurization" as a basic strategy, which is realized in the management of severe accidents is being investigated. By reducing the load over the vessel under severe accident conditions, prerequisites for maintaining the integrity of the primary circuit are being created. The time-margins for operators' intervention as key issues are being also assessed.

The task is accomplished by applying the GRS (Gesellschaft für Anlagen- und Reaktorsicherheit mbH) thermal-hydraulic system code ATHLET. In addition, a comparative analysis of the accident progression for a station blackout event for both a reference German PWR and a reference VVER-1000, taking into account the plant specifics, is being performed.

I Introduction

This paper presents results of an accident analysis for a hypothetical station blackout scenario in a reference VVER-1000 reactor. The initiating event with complete loss of AC power, so called station blackout (SBO), belongs to the typical beyond design basis accidents (BDBA) for which the time of plant survivability without severe fuel damage depends solely on built-in safety.

The occurrence of the specific thermal-hydraulic phenomena appearing during such an event is being investigated with the GRS thermal-hydraulic system code ATHLET. The SBO scenario is characterised by complete unavailability of all active safety systems, except the battery supplied steam dump to atmosphere (BRU-A) valves. This event represents a classical

high pressure accident scenario, and if no safety systems are activated or any accident management measures are applied, it could lead to failure of the reactor pressure vessel (RPV) under high pressure, which from its side could cause challenges to the containment. Early controlled depressurization of the primary circuit applied as a mitigative measure could prevent failure of the RPV under high pressure. Moreover, the ejection of the melt under high pressure (HPME) often accompanied by phenomena known as direct containment heating (DCH) represents a serious threat to the containment integrity.
Taking into consideration the specifics of the VVER-1000 reactors, a comparative analysis with a reference German PWR reactor has been additionally realized.
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II ATHLET Modelling and Initial Conditions

Reference VVER-1000

For the present investigations a reference VVER-1000 type of reactor has been taken into consideration. It is a pressurized water reactor (PWR) with a thermal power of 3000 MW and gross electric output of 1000 MW. The unit has four circulation loops, each including a main circulation pump and a horizontal steam generator. The pressurizer is connected to one of the main circulation loops.

Reference German PWR

A reference German PWR with a thermal power of 3765 MW (1300 MW electric output) has been used for the comparison with the reference VVER-1000. The unit has four circulation loops, each including a main circulation pump, a vertical steam generator as well as a pressurizer, connected to one of the main circulation loops (loop No2).

For both reactor types in the analysed cases all major components of the primary and the secondary side of the reactor coolant system, the necessary reactor protection and safety injection systems are included. On the primary side, the reactor core, the reactor pressure vessel, main circulation pumps, main circulation pipes, the pressurizer and the relief and safety valves have been modelled. On the secondary side, special attention has been paid to the modelling of the horizontal steam generators of the VVER-1000.

The ATHLET input deck for the VVER-1000 reactor was elaborated at GRS in co-operation with Kurchatov-Institute and OKB Gidropress. Requirements for a realistic simulation of the sophisticated performance of safety and control valves in VVER-1000 plants, which were not previously considered in ATHLET, led to some source code modifications. This specially developed version (ATHLET 1.2d) has consequently been used for this analysis. A generic ATHLET input file for the reference German PWR has been used for the investigation of several accident scenarios at FZD. An improved version of the input file with special attention to the accident management strategies has been applied for the investigations presented in this paper.

The nodalization schemes of the primary circuit and the steam generators for both reactor types are depicted in Figure 1 and Figure 2, respectively. They illustrate typical differences between the nuclear power plants, like horizontal steam generators and the connection of the passive safety injection systems (hydroaccumulators) to the primary circuit. In the VVER-1000 reactor the accumulators are directly connected to the RPV (two into the upper plenum and two into the downcomer with 50m³ borated water each), in the German PWR the accumulators inject water to the hot and cold legs (8x50m³).

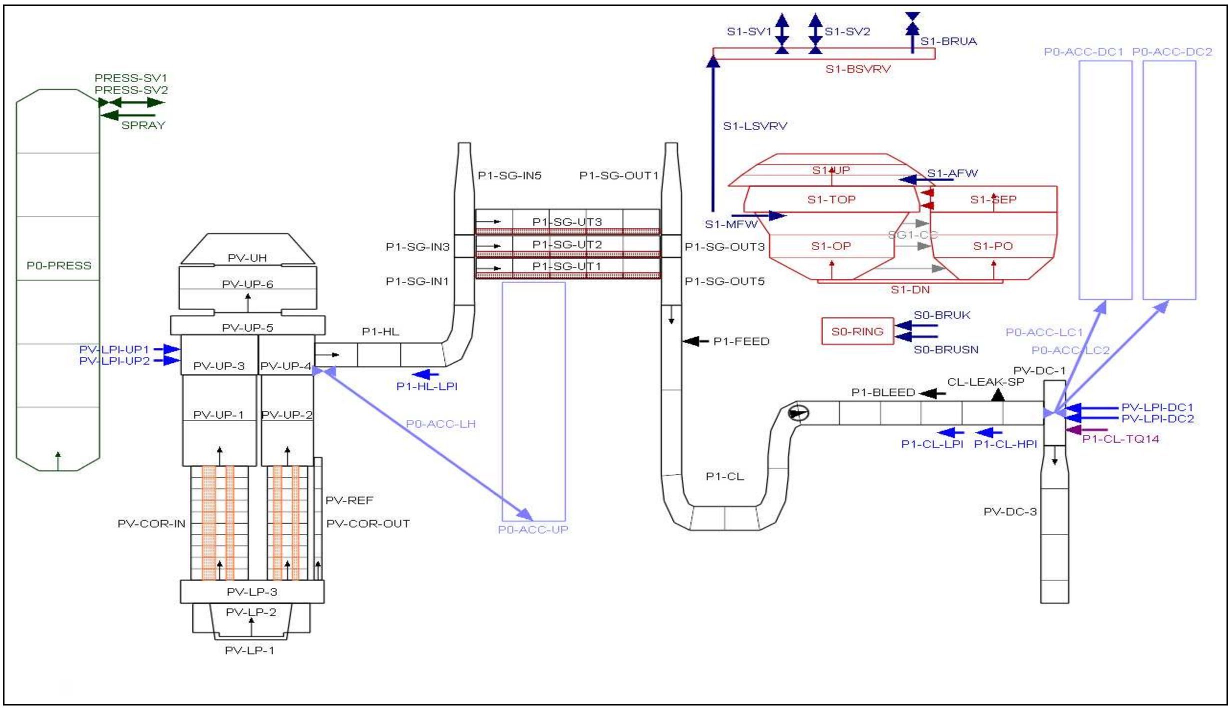

Figure 1: Reference VVER-1000/ Nodalization scheme

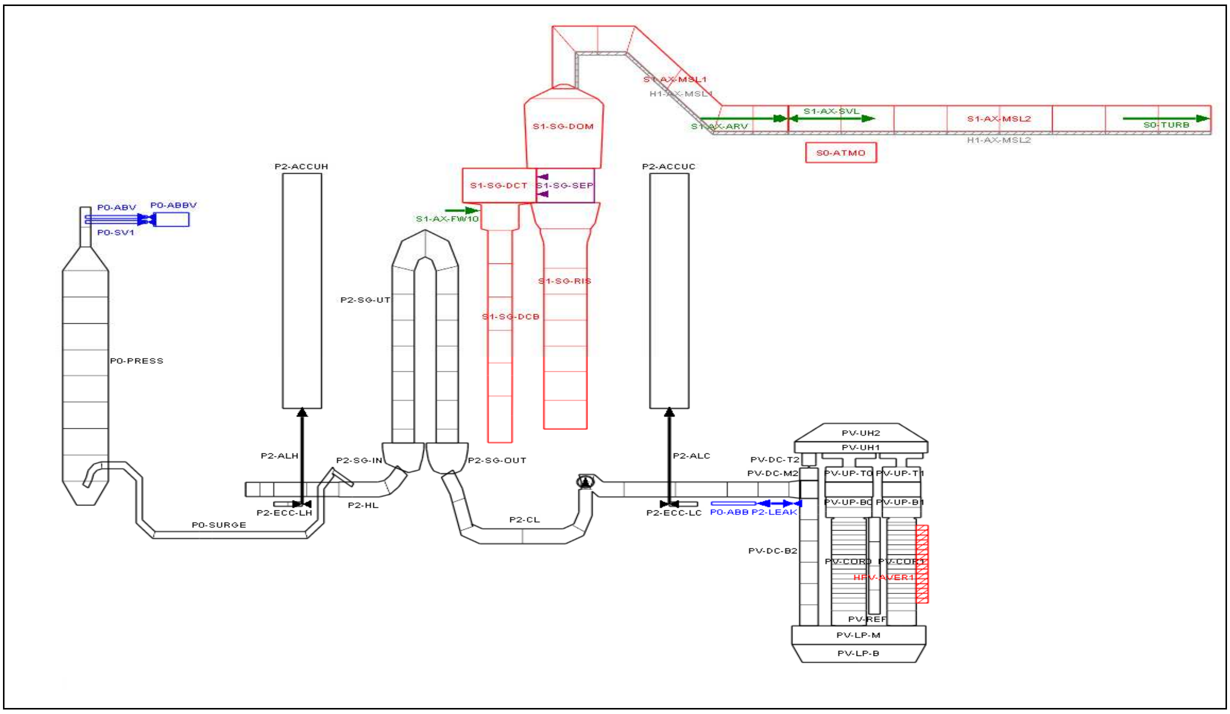

Figure 2: Reference German PWR-1300/ Nodalization scheme

Before performing a transient simulation, for both input decks a steady-state calculation for adjusting the boundary conditions necessary for the analyses of the discussed accident sequences has been performed. In that way the initial plant conditions as reactor power, SG-power, temperatures, water mass and mass flow rates are achieved. The steady-state calculation has been performed for 500 s. In Table 1 the conditions at the end of the steady state simulations have been summarized for both reactors.

Table 1: Steady state parameters of the reference PWR VVER-1000 plant and the reference German PWR plant

Parameters	Value Reference VVER-1000	Value Reference German PWR
Core power, [MW_{th}]	3000.00	3765.00
Primary pressure, [MPa]	15.70	15.80
Average coolant temperature at reactor outlet, [°C]	320.15	324.20
Maximum coolant temperature at reactor inlet, [°C]	290.00	291.70
Mass flow rate through one loop, [kg/s]	4400.00	5025.75
Pressure in SG, [MPa]	6.27	6.51
Steam mass flow rate through SG, [kg/s]	408.00	515.00
SG total water mass for one SG, [kg]	47000.00	39000.00

III General Description of the SBO Transient

The initiating event "Loss of the offsite electric power system concurrent with a turbine trip and unavailability of the emergency AC power system" (station blackout), belonging to the typical beyond design basis accidents, results in reactor shut down, loss of feed water and trip of all reactor coolant pumps. With closing of the turbine stop valves, the heat generated on the primary side can not be transferred through steam dump to the condenser and consequently the pressure in the secondary circuit is increasing. After RCPs coast down the coolant flow rate through the reactor decreases and in the early phase the decay heat removal takes place in natural circulation mode on the primary side and with help of steam dump to atmosphere (relief / safety valves) on the secondary side. Continuous evaporation of the secondary side leads to steam generators' depletion and break down of steam generator power with following heating up of the core at high pressure. Increase of primary pressure leads to continuous mass loss through the pressurizer relief and safety valves. Due to the decreasing RPV level and vaporization on the primary side the natural circulation breaks down.

III.1 Initial Conditions and Availability of Systems (VVER-1000 and PWR-1300)

- NPP at normal operating conditions (100 % reactor power)
- SG pressure regulation is available (VVER-1000: BRU-A, PWR-1300: secondary pressure regulation and cool-down)

- For VVER-1000: BRU-A stops at 7200 s (batteries depletion)
- Pressurizer relief and safety valves are available
- Active ECCS (HPIS, LPIS) are not available
- Passive ECCS (accumulators) are available

III.2 Additional Assumptions

- Conservative approach for decay heat curve (DIN + 2 sigma)
- Criterion for start of PSD (core outlet temperature):
    - VVER-1000: $T_{core\text{-}out}$ > 650 °C (additional simulation with $T_{core\text{-}out}$ > 350 °C)
    - PWR-1300: $T_{core\text{-}out}$ > 400 °C

IV Main Results

IV.1 Comparative Analysis Reference VVER-1000/ Reference German PWR-1300

The primary and secondary pressures for the discussed scenario are given in Figure 3. Shortly after plant blackout the secondary pressure increases to the set point pressure thresholds of the SDA (Steam Dump to Atmosphere) for VVER-1000 case, and after opening of the BRU-A valve this causes continuous decreasing of water inventory on the secondary side of SGs. During the first seconds of the accident, the primary pressure is dropping due to decreasing of the core decay heat. The BRU-A valve is working and it maintains the secondary pressure at 6.67 MPa. Likewise, for the reference German PWR the pressure regulation for the secondary side is activated and shortly after that the partly cool-down procedure for the secondary side is actuated leading to a cool-down rate of 100 K/h at 7.5 MPa.

Due to the heat transfer from primary to secondary side the secondary water inventory is continuously evaporating, decreasing the liquid level on the secondary side (SG), see Figure 4. With decreasing levels in the SGs, the SG power is also decreasing and after reaching its minimum the primary pressure starts to increase. Later on the primary pressure reaches the threshold for opening the PRZ relief valve and around 1900 s for the GPWR-1300 and around 3500 s for VVER-1000 there is a blow-down through the valve. At 2300 s for the GPWR-1300 and at 4900 s for VVER-1000 the minimum SG level is reached, due to the larger water inventory on the secondary side in VVERs. After SG's depletion the heat transfer from primary to secondary side breaks down and the pressurizer relief valve opens and closes much faster, see Figure 3. With the actuation of the pressurizer relief valve the pressurizer level is increasing and when the level reaches the position of the relief valve, a transition from two-phase to single-phase water flow through the valve can be observed.

Due to the increasing loss of primary coolant through PRZ relief valves the primary mass inventory is decreasing and an RPV cover bubble. When the water level in reactor vessel drops below hot nozzles elevation, the natural circulation in the primary system is interrupted.

As there is no water supply to the primary circuit the core starts to heat-up (Figure 5). The reduced primary system mass leads to core uncovery and its consequent dryout.

Table 2 summarizes the timings of the main events derived for the both reactor types as calculated by ATHLET:

Table 2: Reference VVER-1000 plant and the reference German PWR plant/ Main events

No	Event	Value Reference VVER-1000 Time, s	Value Reference GPWR-1300 Time, s
1	Initiating event – SBO, reactor SCRAM, MCPs are switched off, Turbine stop valves (TSVs) are closed, Feedwater terminated	0.0	0.0
2	Opening of steam-dump to atmosphere valves, BRU- A (VVER-1000) Start of pressure regulation for the secondary side / partly cool-down (PWR-1300)	15.0	10.0 / 25.0
3	Opening of Pressurizer (PRZ) valves	3500.0	1900.0
4	Steam generators dry- out	4900.0	2300.0
5	PRZ valve mass flow transition from two-phase to single-phase water flow	5560.0	2880.0
6	Upper plenum temperature reaches saturation conditions (voiding start-up)	6400.0	3580.0
7	Closing of BRU-A valves	Fully closed at 7200.0	-
8	Break down of natural circulation on primary side	7420.0	4010.0
9	First heating- up of the core	8440.0	4750.0
10	Start of primary circuit depressurization (PSD)	9230.0	5020.0
11	Start of the hydroaccumulators' injection VVER-1000: at primary pressure < 5.9 MPa PWR-1300: at primary pressure < 2.6 MPa	9850.0	5600.0
12	End of the hydroaccumulators' injection (HA empty)	10600.0	8260.0
13	Heating- up of the core after stop of HAs injection	15420.0	10060.0

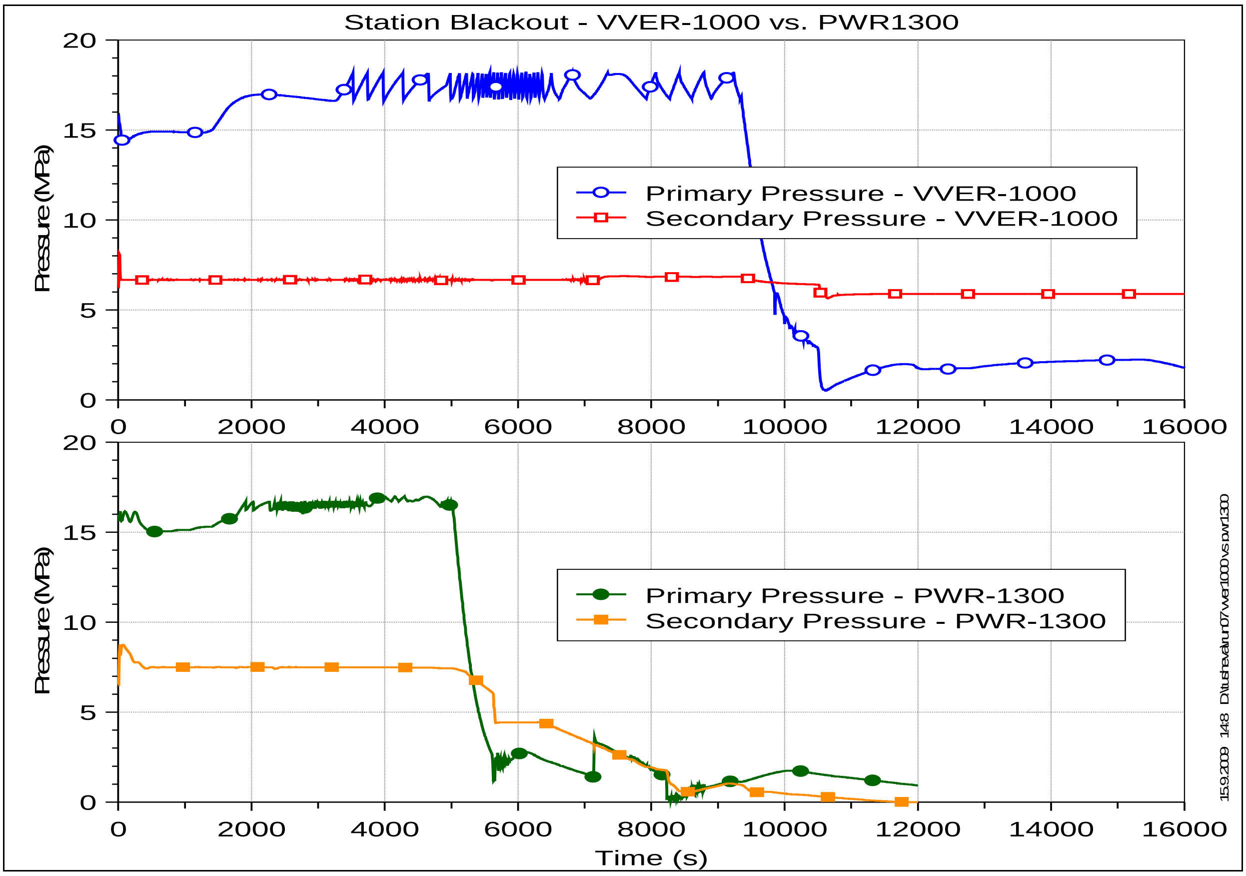

Figure 3: Reference VVER-1000 vs reference GPWR-1300/ Primary and secondary pressures

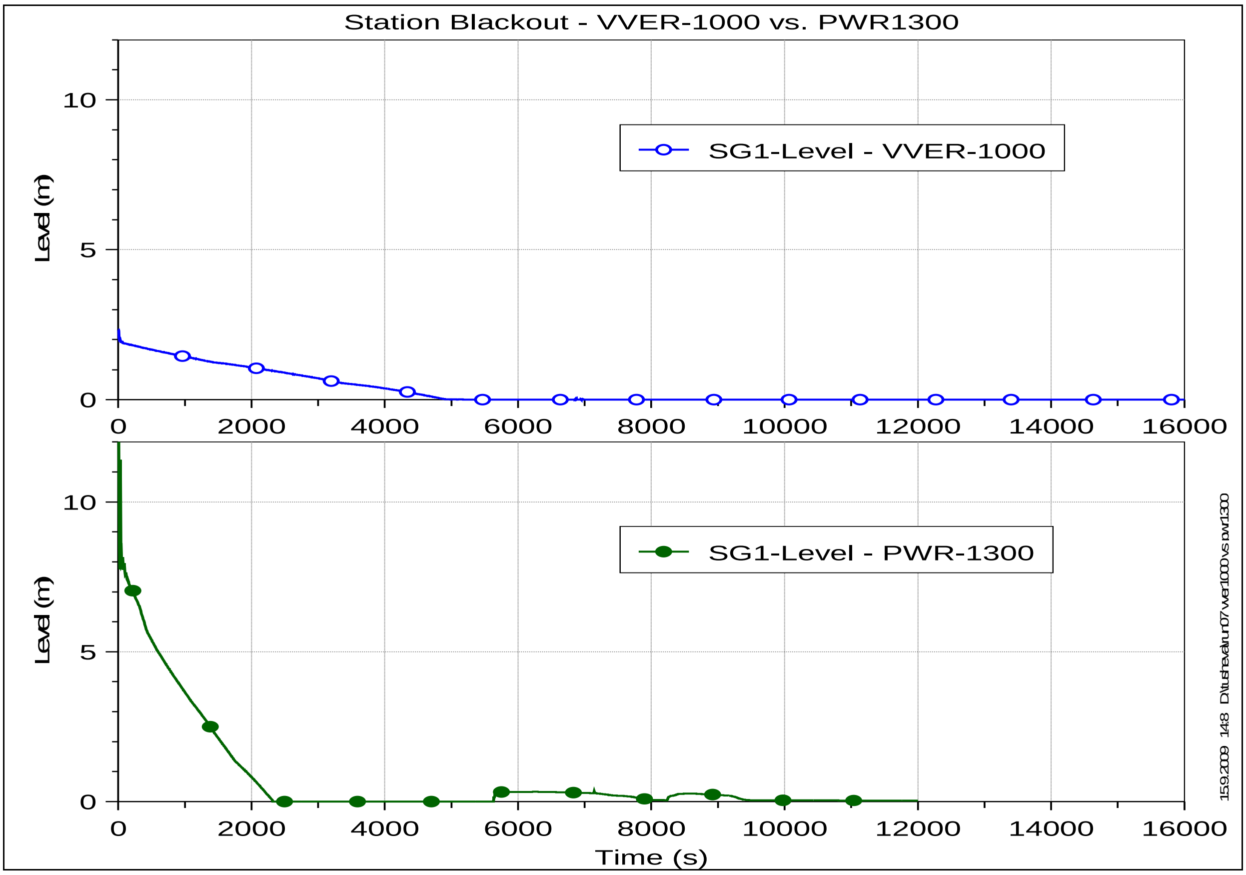

Figure 4: Reference VVER-1000 vs reference GPWR-1300/ Steam generator level

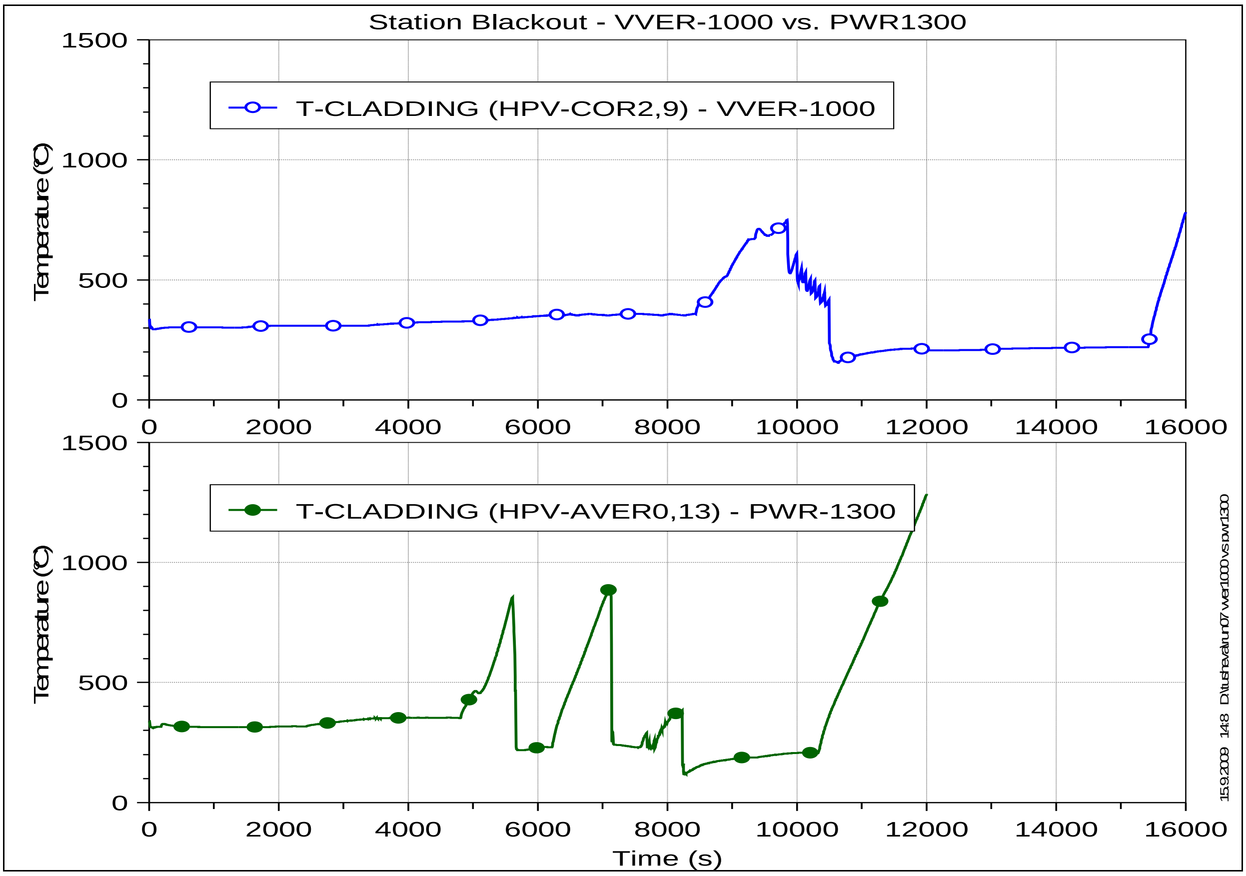

Figure 5: Reference VVER-1000 vs reference GPWR-1300/ Cladding temperature

State of the NPP after ~ 2 h (VVER-1000) resp. ~ 1 h (GPWR-1300)

- Primary pressure ~ 17-18 MPa (p >> set point of accumulator injection, passive ECC)
- Heat removal from reactor core only via pressurizer valves (steam generator power ~ 0)
- Continuous decrease of RPV level and increase of fluid and cladding temperatures

At a temperature $T_{core\text{-}out}$ > 650 °C for VVER-1000 and $T_{core\text{-}out}$ > 400 °C for GPWR-1300, a depressurization of the primary side is initiated manually by the operator. For this AM measure the PRZ safety valves are fully opened, which allows a reduction of the primary pressure below the set-point of the accumulators and in this way an injection of water from the passive ECC systems. The accumulator injection leads to an increasing water level in the RPV and the rise of the cladding temperatures is temporarily stopped (Figure 5). Due to evaporation in the reactor core the decrease of the primary pressure is slowing down and caused by a reversed heat-transfer in the SGs the secondary pressure slightly decreases. In the PWR-1300 simulation after start of the accumulator injection a sharp increase of the primary pressure can be observed. Due to this the accumulator injection has been stopped for a while and the cladding temperatures start to increase. Approx. 2600 seconds later a second injection phase takes place and the increase of the cladding temperatures is stopped again. With the feeding from the accumulators in both simulations the core temperatures are lowered for a limited period of time. When the accumulators are emptied, the cladding temperatures start to increase and without additional measures (recovery of electricity supply and start of active ECC pumps) the cladding temperatures could exceed the safety margin of 1200 °C.

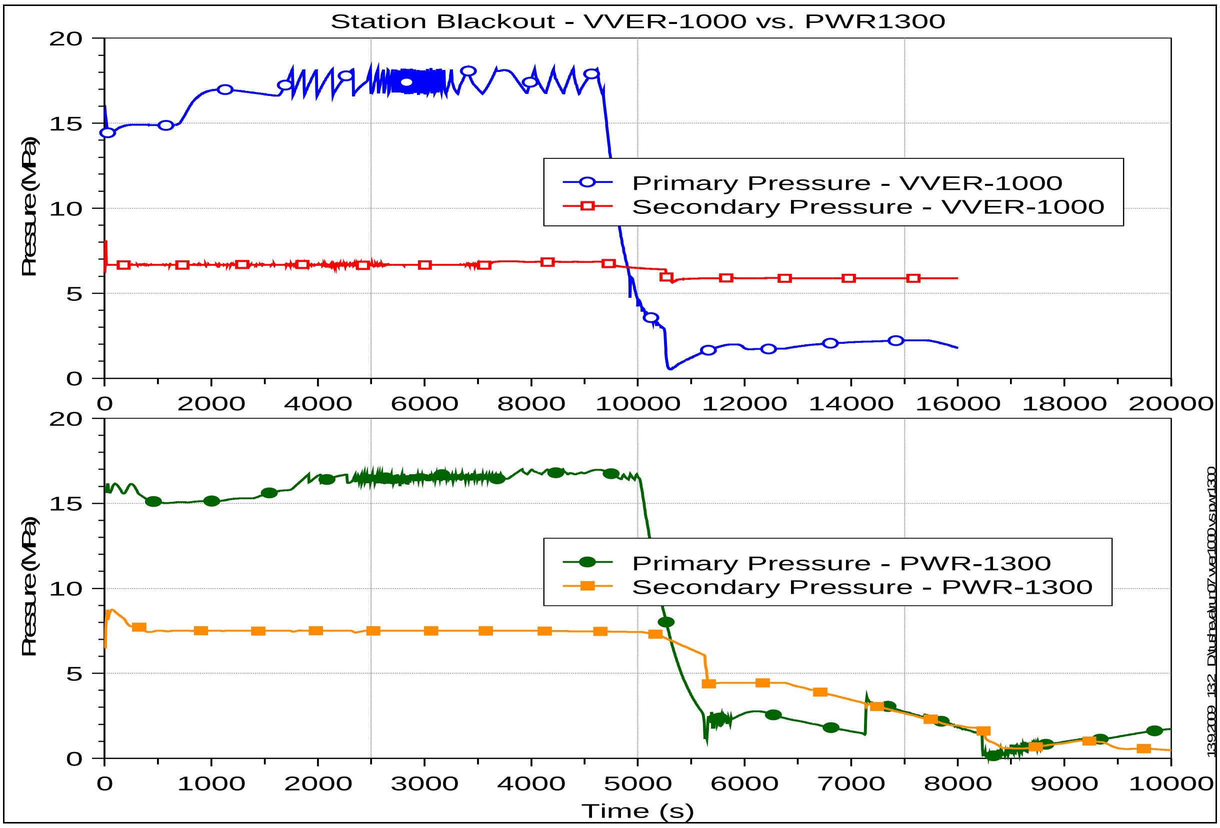

Figure 6: Reference VVER-1000 vs reference GPWR-1300/ SBO - general transient behaviour

The general trend of the transient for both reactor types is very similar; the difference is in the timing of the events. This could be seen in Figure 6 with a changed scaling for the time-axis. The main differences concerning the different timings for characteristic events are from one side the total SG water mass and the reactor power for the both plants (see table 1), and with a factor of 2 higher steam velocities and higher steam mass flow rates for the GPWR-1300; and last but not least the spatial orientation of the steam generators with the differences in the heat transfer characteristic (heat transfer area and SG power depending on SG level). Finally these differences result in an accident progression which is approx. two times faster in the GPWR-1300. As a consequence the operators in the VVER-1000 NPP have more time to prepare accident management measures.

IV.2 Investigation on AMM for VVER-1000: Primary Side Depressurization

Two investigations have been done: firstly, depressurization of the primary side at a temperature criterion $T_{core\text{-}out}$>650 °C and secondly, depressurization of the primary side at a temperature $T_{core\text{-}out}$>350 °C. The main purpose has been to determine the maximum response time for the stuff to recover electricity supply and in such a way to be seen the effectiveness of the AMM on the accident progression. Figures 7-10 represent the main results from this investigation.

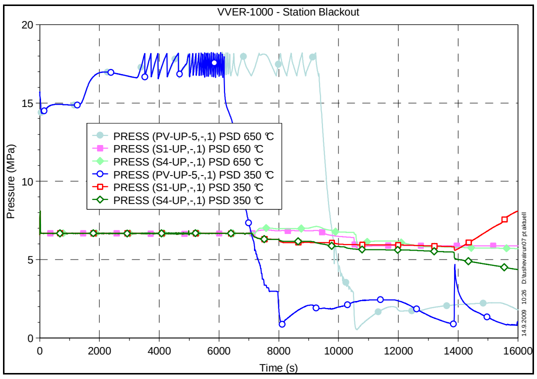

Figure 7: Reference VVER-1000 PSD/ Primary and secondary pressures

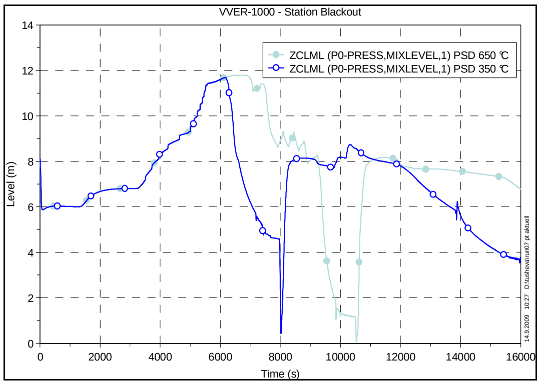

Figure 8: Reference VVER-1000 PSD/ PRZ Level

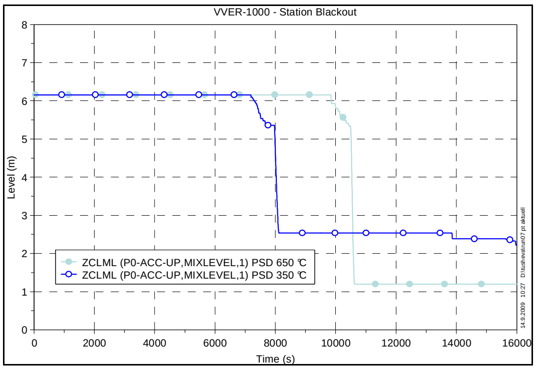

Figure 9: Reference VVER-1000 PSD/ HA level

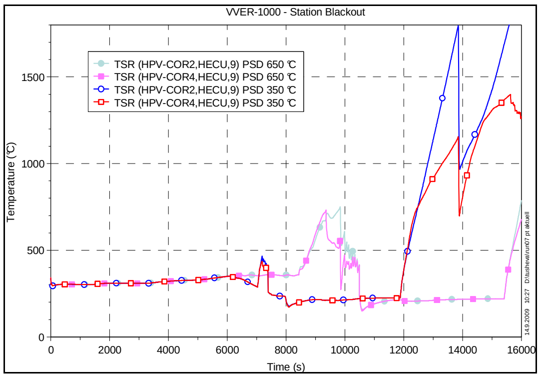

Figure 10: Reference VVER-1000 PSD/ Cladding temperature

Two peaks in the cladding temperature could be observed (Figure 10). When the PSD is actuated at $T_{core\text{-}out}$>350 °C, the first peak is much less pronounced than if the PSD is actuated later in time due to the higher temperature criterion. Independently from the PSD temperature actuation criterion the first peaks (heat-ups) in the cladding temperature which occur during this period are mitigated by the reflood from the hydro-accumulators. Finally, after the HAs depletion the temperature in the core increases again, a sharper increase could be seen in the simulation by actuation of the PSD at lower core outlet temperature. For the case with higher core outlet temperature criterion for PSD actuation, a second increase in the cladding temperature is observed much later in time approx. 15500 s from the beginning of the transient, in comparison to 11950 s for the case with earlier actuation of the PSD. Nevertheless it should be kept in mind that before initiation of the PSD-procedure an early initiation of secondary side depressurization measure which should normally precede the PSD, can effectively minimize the risk of core damage by preventing fuel rods from heating up throughout the transient.

V Summary

The work presented in this paper aims at the investigation of the thermal-hydraulic behaviour of a generic VVER-1000 reactor in case of a station blackout accident. The initial event leads to reactor scram, turbine trip, total loss of feedwater and trip of all main coolant pumps. As a consequence the scenario results in a core heat-up under high pressure conditions. With help of an accident management measure (primary side depressurization) the primary pressure can be reduced, so that the passive ECC systems (accumulators) can inject water into the primary system and the start of an extended core dry-out can be significantly delayed.

Comparative analyses for the SBO accident with the thermal-hydraulic code ATHLET have been done for a reference German PWR and a reference VVER-1000, taking into account the plant specifics. The comparison of the results shows, that the general behaviour with respect to the main events and thermal-hydraulic phenomena is very similar. The differences in the reactor power and in the construction of the steam generators (orientation, mass inventory, steam velocity and steam mass flow rate) are directly responsible for the different timing. A preliminary result of the comparative study is that the operators in the VVER-1000 NPP have more time to prepare accident management measures to prevent or mitigate possible core damage. Further analyses are needed to determine the differences in timing more precisely. Therefore the influence of the SG model (especially the nodalization in vertical direction) and the influence of the accumulator injection (condensation rates, effect on primary pressure) on the course of events have to be investigated in more detail.

An additional simulation for the VVER-1000 reactor with the earliest possible time for starting PSD has been performed. For the simulation the temperature criterion to start the procedure has been modified. With a core outlet temperature of 350 °C the fluid temperature reaches the saturation temperature and steam appears in the RPV. With the modified criterion PSD starts much earlier. Caused by the higher decay heat in the early phase of the transient the primary pressure reduction after initiation of PSD is not so effective than in the first simulation with PSD at a core outlet temperature of 650 °C. Compared to the first simulation the accumulator injection starts earlier and at a higher level of the decay heat. As a consequence the accumulators can not inject the full amount of water and with the changed criterion the cladding temperatures start much earlier to rise and then exceed the safety margin of 1200 °C. As a conclusion from the results of the two simulations it should be mentioned, that a start of the PSD procedure later in time is more effective.

Finally it has to be pointed out, that without any active ECC systems a core dry-out with increasing cladding temperatures can not be avoided at all. With help of the PSD procedure the start of an extended core dry-out can be significantly delayed and the core cooling is ensured for at least two hours, but to prevent severe core damage a recovery of the electricity supply and the start of active ECC systems after depletion of the accumulators is an essential requirement for both nuclear power plants.

NOMENCLATURE

AM	Accident Management
AMM	Accident Management Measures
ATHLET	Analysis of THermal-hydraulics of LEaks and Transients

BDBA	Beyond Design Basis Accidents
ECCS	Emergency Core Cooling System
FZD	Forschungszentrum Dresden-Rossendorf
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit mbH
HA	Hydroaccumulators
HPIS	High Pressure Injection System
HPME	High Pressure Melt Ejection
LPIS	Low Pressure Injection System
NPP	Nuclear Power Plant
PRZ	Pressurizer
PSD	Primary Side Depressurization
PWR	Pressurized Water Reactor
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SDA	Steam Dump to Atmosphere (BRU-A)
SG	Steam Generator
VVER	Water-Water Energetic Reactor

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