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The efficiency of sequential accident management measures for a German PWR under prolonged SBO conditions

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ABSTRACT

In this paper, the results of ATHLET-CD simulations of an SBO accident for a German Siemens KWU type PWR are reported. The developed model is used in a series of calculations to evaluate SBO coping time provided by a set of countermeasures relevant to the defense-in-depth Level 4. The analysed accident management measures cover a sequence of the bleed and feed procedures, starting/ending with secondary/primary side depressurization followed by the feeding of SGs in the passive (AMM-1) or active (AMM-2) mode and coolant injection from hydro-accumulators (HA) to the primary system (AMM-3).

A sequential implementation of the first two measures with almost equal efficiency (AMM-1 and AMM-2) delays the core degradation onset (CDO) by 21.5 hours compared to the case without AMMs, extending SBO coping time to 24 hours. This time window can be further extended (more than twice) through sequential feeding of a single SG from the four emergency feedwater tanks of the plant. The third measure (AMM-3) is significantly inferior to AMM-1 and/or AMM-2 in contributing to the coping time, since it delays CDO by less than 1 hour.

Keywords: Accident management measures, Station Black Out, PWR, ATHLET-CD

Abbreviations

AC	Alternating Current
AMM	Accident Management Measure
CDO	Core Degradation Onset
CL	Cold Leg
DID	Defense-in-Depth
ECCS	Emergency Core Cooling System
EFT	Emergency Feedwater Tank
EOC	End of Cycle
FWS	Feedwater System
FWT	Feedwater Tank
HA	Hydro-Accumulator
HL	Hot Leg
LOOP	Loss of Offsite Power
LPIS	Low-Pressure Injection System
L_{PRZ}	Pressurizer coolant level
L_{RPV}	Coolant level in reactor pressure vessel
MCP	Main Coolant Pump
MSH	Main Steam Header
NKV	Notkühlvorbereitungssignal / Emergency core cooling preparation signal
NPP	Nuclear Power Plant
p.a.	per annum
ΔP_{CONT}	Containment pressure excess above atmospheric
P_{FWT}	Pressure in feedwater tank (absolute)

$P_{P\ (GAUGE)}$	Primary pressure (gauge)
PWR	Pressurized Water Reactor
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
PRZ	Pressurizer
PSD	Primary Side Depressurization
SBLOCA	Small-Break Loss-Of-Coolant-Accident
SBO	Station Blackout
SCRAM	Reactor shutdown (Safety Cut Rope Axe Man)
SG	Steam Generator
SSD	Secondary Side Depressurization
$T_{CORE-OUT}$	Core outlet temperature

1. Introduction

Following the 2011 accident in Fukushima (Japan), the attention of the international reactor safety community was drawn even more to the risks of severe core damage (Prošek and Cizelj, 2013; Cho et al., 2017; Sanchez-Espinoza et al., 2017; Wilhelm et al., 2018; etc.) than already done as part of the analyses for standard licensing. Such risks are minimized through an implemented safety concept of defense-in-depth (DID), also called “multilevel defense concept” (INSAG 10, 1996; SSR-2/1, 2016). DID levels 1 and 2 refer to normal plant operation, while the levels 3 and 4 correspond to design and beyond-design accident conditions, respectively (ILK, 2008). For operating German NPPs, the sublevel 4b of level 4 is the last one at which severe core damage could be avoided through preventive measures. In the extremely rare case of a failure of all preventive measures, mitigative measures must be initialized to reduce the release of radioactive material (DID sublevel 4c).

Among accident initiating events in German Siemens KWU type PWRs[1], the loss of electrical power supply for house load fed from the external grid (LOOP) or the main generator has the third highest frequency of occurrence (0.025 p.a.)[2], leading to a damage state[3] with a probability of $5.45 \cdot 10^{5}$ p.a. (GRS-184, 2002). Failures of steam generator feedwater supplies (both operational and emergency) contribute about 95% of this value ($5.2 \cdot 10^{-5}$ p.a.)[4]. Such beyond-design accident conditions correspond to the SBO scenario with a high primary pressure, when the time available for prevention of the core damage is limited by 2 to 3 hours (GRS-184, 2002; Kozmenkov et al., 2017; Gómez-García-Toraño et al., 2018).

The SBO accident can be handled, if at least one of thefour emergency feedwater redundancies of the Siemens KWU type PWR is restored before a collapsed coolant level in the primary system drops to the elevation of the core outlet. The recovered supply of emergency feedwater to SGs results in resuming the primary heat removal (via natural circulation of the primary coolant), which in turn leads to the primary pressure reduction and prevention of the further primary level decrease. In case the emergency feedwater supply of SGs remains unavailable within the first hour of SBO progression, special accident management measures (AMMs) have to be undertaken to depressurize and cool down the primary system, extending the available time slot for prevention of core damage. The bleed and feed measures, relevant to all Siemens-KWU plants (Sonnenkalb and Mertins, 2012), have been described for the KONVOI plant by (Roth-Seefrid, 1994). The efficiency of following AMMs (and their combinations) implemented in operating German PWRs are considered in this paper, with the focus on the extra time they provide to regain control over the plant under beyond-design conditions of SBO:

AMM-1: Secondary bleed by main steam safety valve and passive feed by feedwater system;

AMM-2: Secondary bleed by main steam safety valve and active feed by a mobile pump;

AMM-3: Primary bleed by pressurizer valves and passive feed by hydro-accumulators (HAs).

The efficiency of the first measure (AMM-1) has been earlier analysed for the Siemens KWU type PWR plant in the case of "failure of the main heat sink with total loss of feedwater supply" (Herbold, 1995). The application of the second measure (AMM-2) to the same plant type has been studied in detail by the authors of (Gómez-García-Toraño et al., 2018), but without preceding implementation of AMM-1. The procedure of the primary bleed and feed (AMM-3) at the Siemens KWU type PWR has been modelled in a series of ASTEC (Gómez-García-Toraño et al., 2017, 2018) and ATHLET-CD (Wilhelm et al., 2018; Jobst et al., 2018) simulations for SBO and SBLOCA scenarios. This paper considers a sequential implementation of all the above AMMs at a PWR plant of Siemens design.

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[1] Also known as KONVOI or - the former evolution – as Pre-KONVOI

[2] After the loss of main feedwater without loss of main heat sink (0.12 p.a.) and the loss of main heat sink without loss of main feedwater (0.038 p.a.)

[3] Failure of the operational and safety related measures on the first three DID levels

[4] To the author's knowledge, the corresponding data for Pre-KONVOI plants also from other utilities have not been published. With a high confidence, the referred frequencies are representative for both KONVOI and Pre-KONVOI plants, owing to their similar designs. In more recent probabilistic safety analyses of the PreussenElektra KONVOI and Pre-KONVOI fleet these per annum-values are even smaller since more reactor operating years of the German PWR fleet were considered without any further incident.

The presented results were obtained in a research collaboration between HZDR and PreussenElektra on the analysis of severe accidents for German PWRs.

2. Accident scenario and plant model description

The generic model of a Siemens KWU type PWR was developed at HZDR to analyze the plant behavior for different accident scenarios, including the scenario of an SBO considered in this paper. The layout of power supply for a typical reactor of this type (not only in Germany) is shown in Figure 1. In comparison to the worldwide NPP fleet, the Siemens KWU type PWR contains not only one quartet of emergency diesels - one for each redundancy - as usual with a 10 kV-busbar or the so-called D1-busbar. Moreover, it contains also a 2nd quartet of emergency diesel engines as a special design feature of the Siemens KWU type PWR, the so-called safeguard emergency diesel engines which operate with lower voltage level and which are even housed in the bunkered emergency feed building (Kosowski and Seidl, 2018).

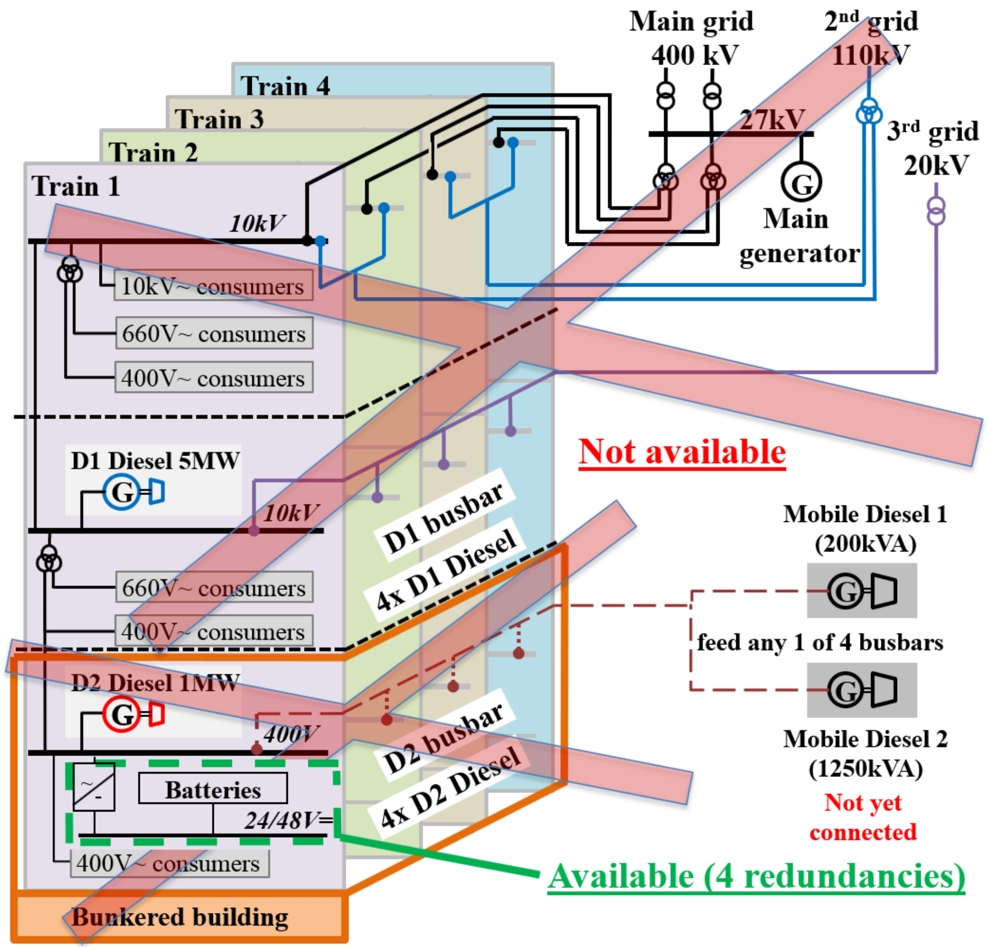

Fig. 1: Layout of power supply of a Siemens KWU type PWR
and postulated losses (Kosowski and Seidl, 2018))

Considering an SBO from international point of view means usually a loss of offsite power (LOOP) coinciding with the unavailability of all four emergency diesel engines (from Siemens KWU-point of view this would be only the loss of the 1st quartet). To remain in the concept of an SBO in a Siemens KWU type PWR as well, an unavailability of all emergency diesel engines - thus all eight and therefore so to say an extended[5] SBO - is postulated in this investigation. The considered scenario is described by the following boundary conditions:

- Before SBO initiation, the plant operates under nominal conditions (100%-power) at the end of cycle (EOC);
- An initiating event is the loss of offsite power;
- Both the first quartet of emergency diesel power supply (10 kV-busbar or D1-busbar) and the second quartet of bunkered safeguard emergency diesel engines (400 V-busbar or D2-busbar) are not available (complete loss of AC-emergency power supply);
- The 3rd grid connection (Fig. 1) are not credited and the mobile diesel generators (alternative power supply for the D2-busbar, which is kept prepared for immediate start) are not available;
- DC emergency power supply is provided by batteries (short term period), i. e. for the reactor protection system (RPS);
- All active safety systems of the plant regarding emergency core cooling are unavailable, while the passive safety injection with HAs remains available;
- Operational and emergency feedwater supply to the SGs is completely lost;
- Pressurizer (PRZ) relief and safety valves are available for the primary side overpressure protection as well as for the primary side depressurization procedure on demand;
- For the SG the safety valves are available for the secondary side overpressure protection, instead of the controlled discharge valves the safety valves also undertake the task for the secondary side depressurization procedure (for reasons of simplifying the model);
- A mobile pump[6] is available to feed the SG after its depressurization.

Due to the postulated failure of power-dependent safety systems at the plant, the description of the Siemens KWU type PWR model given below is limited to the part relevant to the above-considered SBO scenario. All phenomena in the primary and secondary systems, except those of the core, are modelled with the corresponding modules of ATHLET (Lerchl et al., 2016). The behavior of fuel rods, including their damage, and the process of the core quenching are described by the core module ECORE and the module QUENCHCORE of ATHLET-CD (Austregesilo et al., 2013), respectively. Both codes ATHLET and ATHLET-CD are being developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Germany. The nodalization schemes of the primary side as well as of the secondary side are shown in Figs. 2 and 3.

The reactor coolant system includes the reactor pressure vessel with internals and the 4 primary loops, consisting of the hot and cold leg, SG U-tube bundle and the main coolant pump (MCP). The hydraulic behavior of the MCP is described by a set of homologous curves. Two of the four modelled primary loops are shown

[5] An SBO is already a kind of superlative. The severity of the event cannot be outdone.
[6] The mobile pump will be connected to a flange at the X-building (bunkered emergency feed water building), where the emergency feed water pumps and the D2 diesel are housed in. The mobile pump will pump water through the existing piping of the EFW-System.

in Fig. 2, including the loop with the pressurizer connected to it through the surge line (plant loop # 2). Except the PRZ, the modelled system of the primary side overpressure protection includes one relief and two safety valves (located at the top of the PRZ) through which the primary coolant is discharged to the containment in emergency cases. All three PRZ-valves have different staggered threshold values.

The downcomer in the reactor pressure vessel is subdivided into four parallel channels, each assigned to one cold leg and merged in the lower plenum. In addition, cross connection objects are implemented in order to permit cross-flow between the channels. Cross connections only exist between adjacent channels in the annulus, not between opposite channels (Kosowski and Seidl, 2018).The upper plenum above the core is modelled by two (inner and outer) interconnected parts. The bypass channel between the downcomer and the upper head as well as the bypass between the downcomer sections and the inlets of hot legs are also implemented in the plant model.

Eight hydro-accumulators[7] are attached to the cold and hot legs of the 4 primary loops via injection lines with check valves, preventing a reverse flow from the primary system to HAs. The HAs are pressurized to 26 bar, starting water injection when the primary pressure drops below this value. Possible influence of nitrogen[8] on the heat removal through the SGs and on the primary coolant natural circulation after being injected is taken into account by the plant model.

The reactor core is subdivided into 5 parallel flow channels, including 4 fuel channels and the core bypass channel, all divided into 10 axial nodes. The cross-connections between the adjacent core channels provide a coolant exchange between them. The fuel channels 1, 2 and 3 are connected to the inner area of the upper plenum, while the remaining two core channels are linked to its outer part. Totally, the reactor core contains 45548 fuel rods and 1220 control rods distributed between the 4 fuel channels. The fuel rods inside each channel are modelled by a representative pin with the power averaged over the channel and the active/fuel length of 3.9 m. The 4th channel contains only fuel rods, without any control rods inside. The core power is assumed to be constant before SCRAM and, after SCRAM, it drops to the decay heat level calculated by the ATHLET-CD modules OREST and FIPISO.

The secondary system is represented by two loops (with a single and a triple SG models) to allow modelling of an asymmetric plant behavior. In addition, a detailed model of the feedwater system (FWS) was implemented, containing the feedwater tank (FWT) filled with 510 m^3 of water and pressurized to 10 bar under normal operation conditions, the pipelines with preheaters, the main feedwater pump and the check valves to prevent reverse flow from the SGs to the FWT (Fig. 3). The feedwater is injected to each of the SGs through three separate lines, supplying 10%, 40% and 50% of the total mass flow[9], respectively. The 10%-injection line is connected to the downcomer region, while the other two inject to the SG pre-heating chambers located at the bottom of the riser section and shown in Fig. 3. The main steam lines (MSLs), including the main steam header (MSH), are modelled up to the turbine inlet[10].

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[7] A passive train of ECCS

[8] Initially dissolved in the coolant inside HAs and, after injection to RPV, released into the steam phase

[9] Under nominal operating conditions

[10] MSLs and MSH downstream from the main steam isolation valves are not shown in Fig. 3

To reduce the secondary pressure and initiate a passive feeding of SGs from the FWS under emergency conditions, the steam from the secondary system is released to the atmosphere through one of the modelled safety valves, as mentioned above. Additionally, a mobile pump, injecting water from one of four emergency feedwater tanks (EFT) to the single SG[11], is attached to the secondary system model. The usable water inventory of one EFT is 360 m^3.

The availability of the PRZ and SG safety valves, PRZ relief valve, HAs, and the mobile pump connected to the secondary system by existing flanges outside the feed water building provides the opportunity to delay the transition to severe core damage by implementing the primary and/or secondary bleed and feed procedures. To perform a passive injection from HAs/FWS, the primary/secondary pressure has to be decreased (through the PSD/SSD AMMs) below the prevailing pressure in the HAs/FWT. An active feeding of SGs from EFT by the mobile pump becomes possible after depressurization of the secondary side below the zero flow pump head of the used mobile pump (in this investigation below 17 bar).

As soon as the level in all SGs drops below 4.4 m, the preparations for executing the SSD and the PSD are simultaneously started (Fig. 4). The preparation for SSD takes about 40 minutes, and, for this reason, it is assumed that SSD can not be initiated earlier than 40 min after the SG-level drops below 4.4 m, even if any of the four initiation criteria shown in Fig. 4 is met during this time.

The PSD procedure has to be initiated if SSD implementation does not end with a sustainable core cooling[12]. The criteria to start PSD (Fig. 4) are either a low coolant level in RPV[13] (< MIN3) or a high core outlet temperature (> 400 °C). In the presented analysis a delay of 60 s in opening of all pressurizer valves after meeting the temperature criterion is conservatively assumed.

The coolant level in the HAs is monitored in the control room. The plant design provides that, firstly, the four HAs attached to the cold legs are blocked via the RPS with the shut-off valves 500 s after two of the following three conditions for the NKV signal generation are met[14]:

- primary pressure < 110 bar (gauge),
- pressurizer water level < 2.28 m,
- containment pressure > 30 mbar (gauge).

Secondly, the injection paths of the four HAs attached to the hot legs are also blocked via the RPS, when the level of the HA drops below 1.65 m, but not earlier than 500s after the previous mentioned NKV signal generation. This disconnection of HAs shall prevent two effects:

[11] SG of the 2nd reactor loop
[12] E.g., due to reactivation of at least one of the emergency core cooling trains
[13] Approximately 0,15 m below the centerline of the hot leg
[14] The level of coolant is not monitored for this group of HAs

- Particularly for small size breaks, the injection of HA could apply pressure to the primary side so that a delay of the injection of the LPIS is provoked.
- The injection of the nitrogen buffer should be avoided.

These shut-off valves are power supplied by the safeguard emergency diesel engines, the 2nd quartet of emergency diesel engines. Thus, the higher rated level of emergency power supply (D2-busbar instead of D1-busbar) shows the importance of these valves in the Siemens KWU design. In fact, regarding the “extended” SBO mentioned previously, the disconnection of the HAs during the course of accident should actually not be triggered by the RPS because of missing power supply. However, the accident sequence is more severe if an early blockage of the injection path or even a complete loss of all four cold leg HAs are assumed. For this reason, the shut-off valves are supposed to be available in spite of missing power supply.

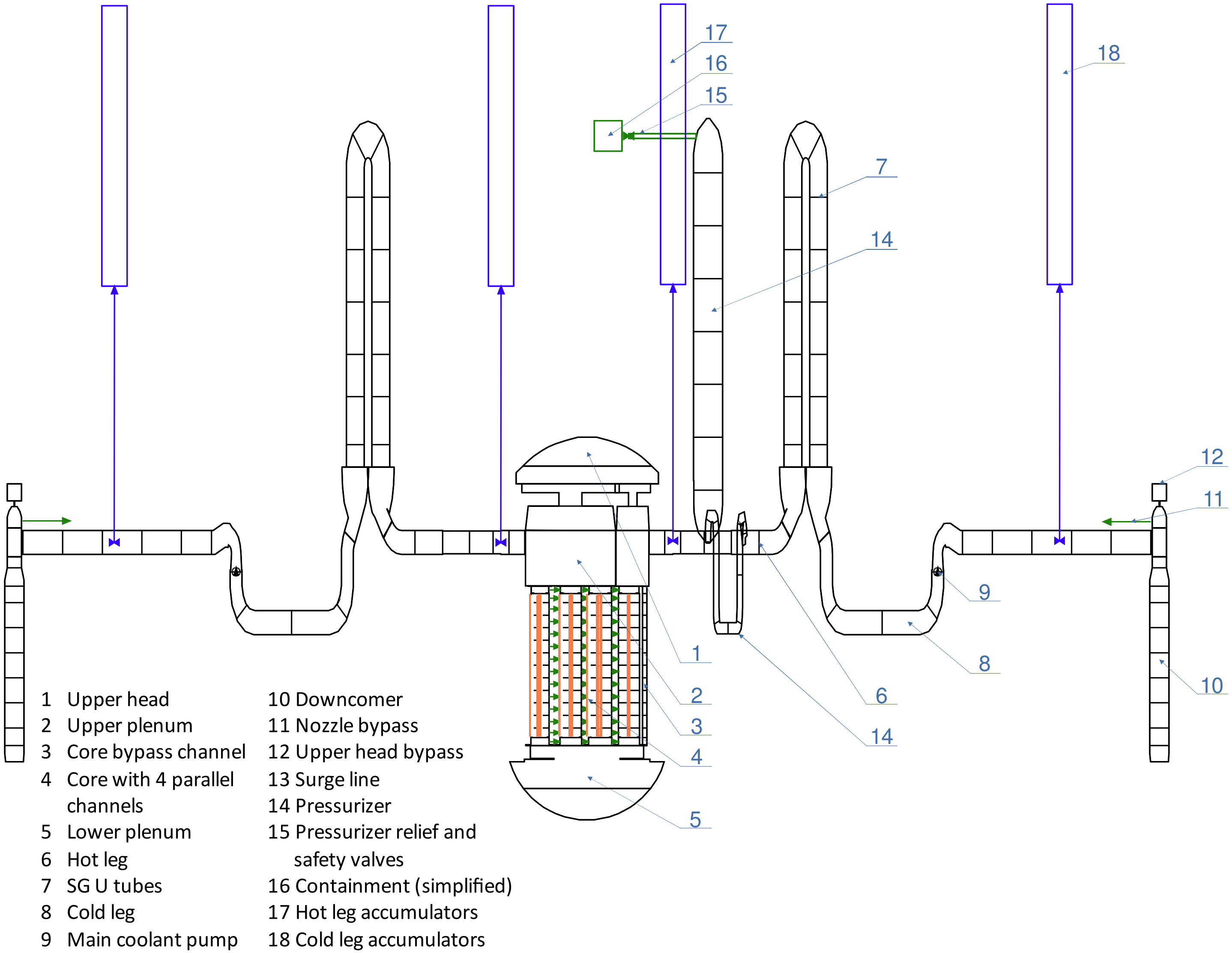

Fig. 2 Primary system model of Siemens KWU type PWR

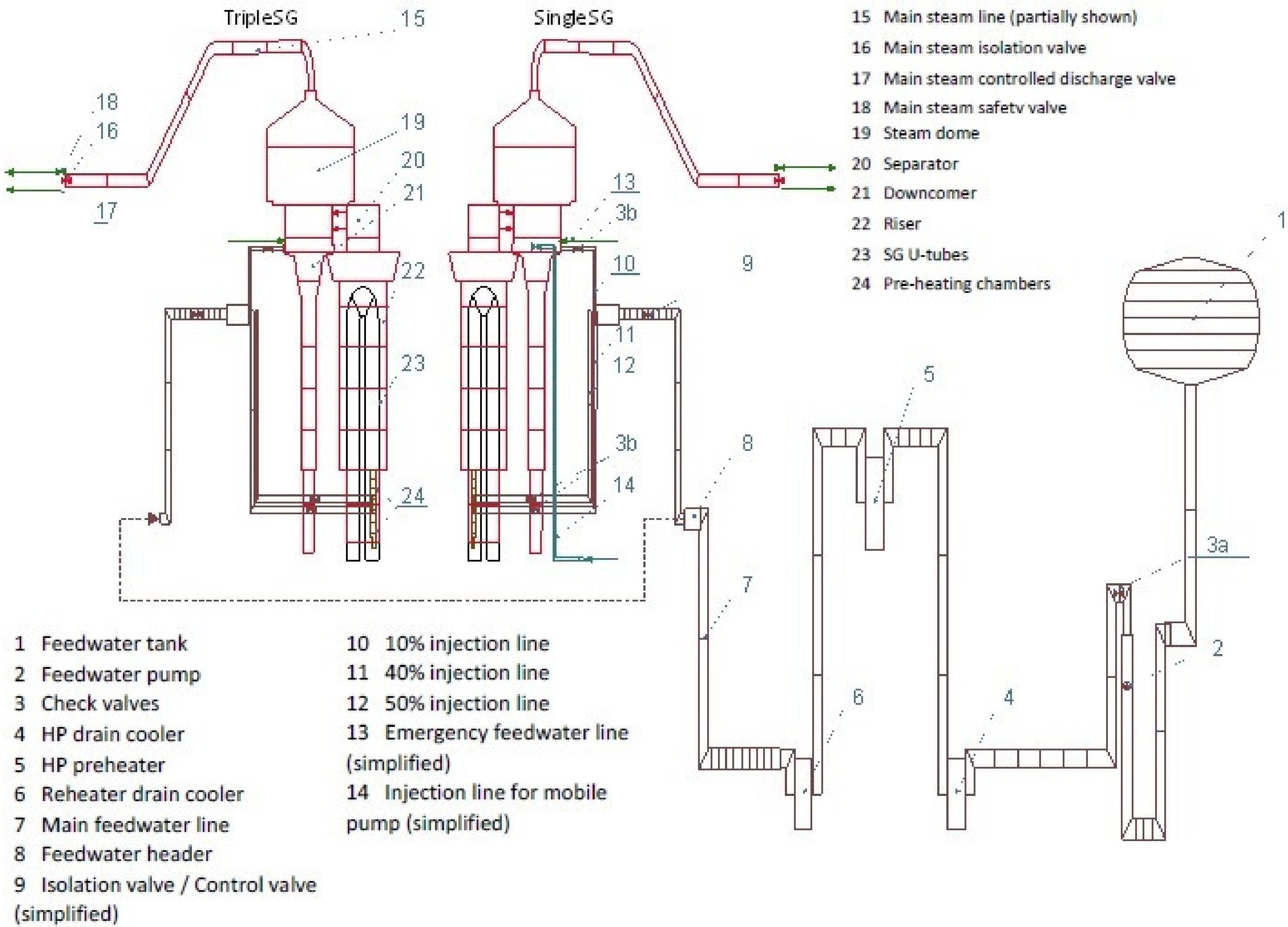

Fig. 3 Secondary system model of Siemens KWU type PWR, in this case of a Pre-Konvoi type

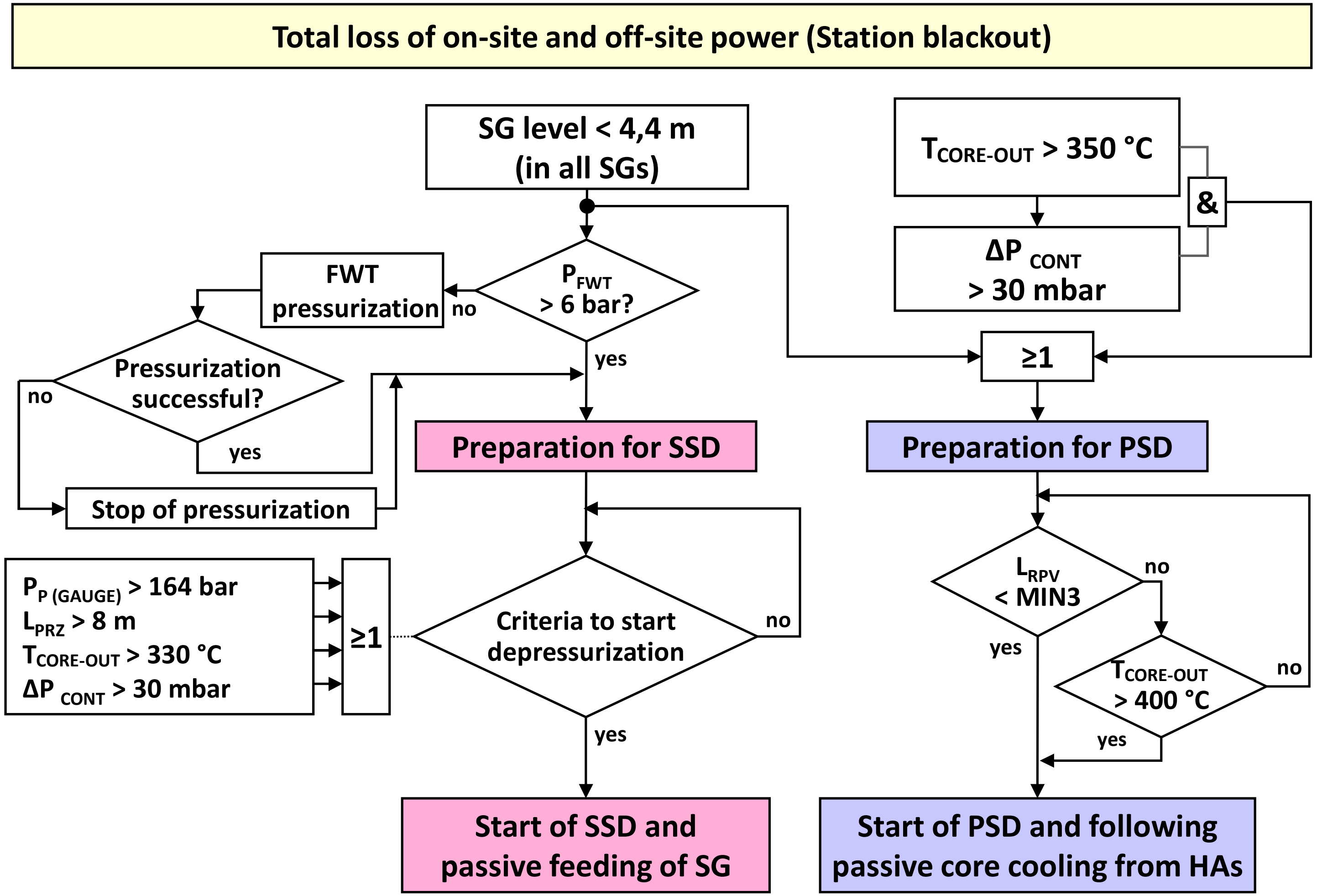

Fig. 4 SSD and PSD procedures for Siemens KWU type PWR under SBO conditions (see also Wilhelm et al., 2018a)

3. Comparison and discussion of results

Prior to the transient calculations, the initial state of the plant has been simulated and compared with the design data related to the reactor operation at full power at EOC. As shown in Table 1, the employed model of a Siemens KWU type PWR reproduces the initial plant parameters with a high accuracy (< 1%).

The following four SBO cases have been simulated, starting from the same initial state but differing in the applied AMMs defined in Chapter 1:

- ***Case 1***: AMM-1
- ***Case 2***: AMM-1 + AMM-2
- ***Case 1a***: AMM-1 + AMM-3
- ***Case 2a***: AMM-1 + AMM-2 + AMM-3

The SBO scenario with AMM-3 only[15] is not considered in this paper. The detailed analysis of this case performed for a generic KONVOI plant model can be found in (Wilhelm et al., 2018). The efficiency of the considered accident management measures is estimated trough the comparison with the results of SBO case calculated without implementation of any AMM, hereafter referred to as Case 0.

Table 1: Plant nominal conditions calculated by ATHLET-CD

Parameter	Units	ATHLET-CD	Deviation from design data, %
Primary system			
Thermal power	MW	3900,0	0,0
Pressure	bar	157,6	-0,3
Mass flow through the core	kg/s	18725	-0,5
Mass flow through 4 loops	kg/s	20025	0,01
RPV inlet temperature	°C	291,7	-0,05
RPV outlet temperature	°C	326,0	0,09
Water level in pressurizer	m	7,46	-0,6
Temperature in pressurizer	°C	345.8	-0,06
Secondary system			
Coolant level in SG	m	SG1: 12,12	-0,7
		SG2: 12,14	-0,5
Steam pressure at SG outlet	bar	68,0	0,15
Feedwater mass flow rate	kg/s	534	-0,4
Feedwater temperature	°C	219,8	-0,09

The loss of offsite power at the plant leads to an MCP trip event. As soon as the speed of at least two of four MCPs falls below 94 % of the initial rated value[16], the SCRAM is triggered followed by a turbine trip. Other possibility of triggering SCRAM – both SCRAM signals take place at almost the same time - is that the voltage of the power supply for the control rods decreases to values below 80%. As a consequence of the SCRAM, the control rods are inserted and the reactor power drops to the decay

[15] This scenario is classified as T1b in the PSA analysis

[16] This event occurs within the first seconds of the accident

heat level. After shutdown of the main coolant pumps, the heat generated in the core is transported to the SGs via the density/gravity driven natural circulation.

After turbine trip the turbine bypass opening remains blocked upon request due to the LOOP, so that the secondary pressure rises from initial 68.0 bars to the setpoint of the SG safety valves (88.3 bars) in less than 2 minutes. The generated steam is discharged through the periodically opening and closing SG safety valves within their opening and closing set-points. In about 30 minutes after this event, because there is no feeding of the SGs, the SG level drops below 4.4 m[17]. Throughout the next 30 minutes it drops even to zero, a complete dryout of the SG (the upper bound of interval I in Fig. 5) takes place due to the unavailability of the emergency feed water supply (high pressure depletion of SGs). In the meanwhile, the check valves in the feedwater lines (Fig. 3) prevent an increase of pressure in the feedwater system.

Due to evaporation procedure and falling SG level and, later on, the complete depletion of SGs, the heat transfer from the primary to the secondary system is decreasing significantly with the increasing exposure of the U-tube bundle and, finally, collapses, when pure steam remains in the SGs, causing the primary pressure to increase up to the setpoint of the pressurizer relief valve.

Shortly afterwards (about 1 h 5 min since the accident onset), the SSD procedure is started by opening the safety valve of the single SG line, as the criterion "primary pressure high" for initiating AMM-1 (Fig. 4) is already met at this time. Since all SGs are interconnected through the main steam header, the secondary side depressurization takes place in all of them. The secondary side depressurization triggers a passive supply of SGs, once the prevailing pressure is reached, with the coolant from the feedwater system, about 2/3 of that amount which was initially stored in the feedwater tank. The process of the passive FWT depletion begins as soon as the secondary pressure drops below 10 bar (the initial pressure level in FWT), approximately 1 h 25 min after the accident initiation, and lasts about 1 hour until the feedwater tank is completely empty (Fig.6).

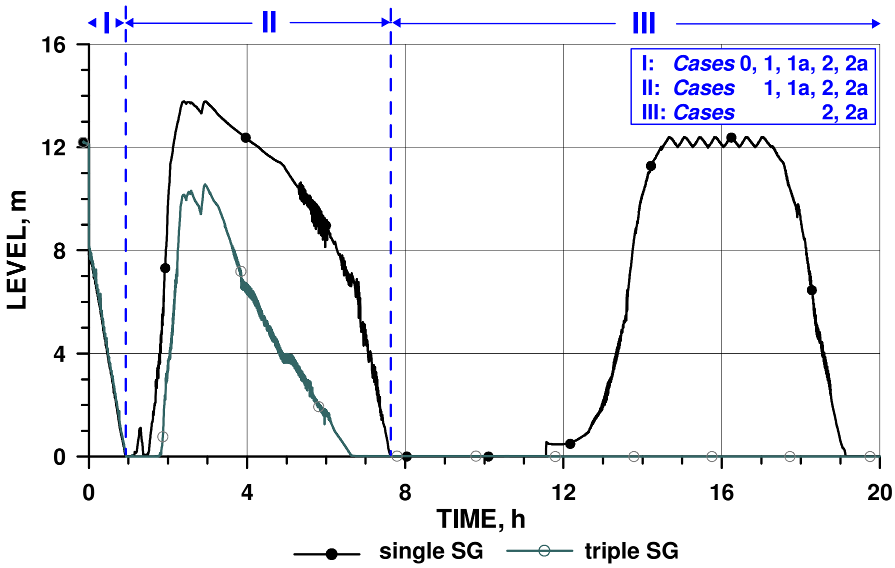

Fig. 5 Coolant level in SGs

[17] The time to start the preparation for SSD

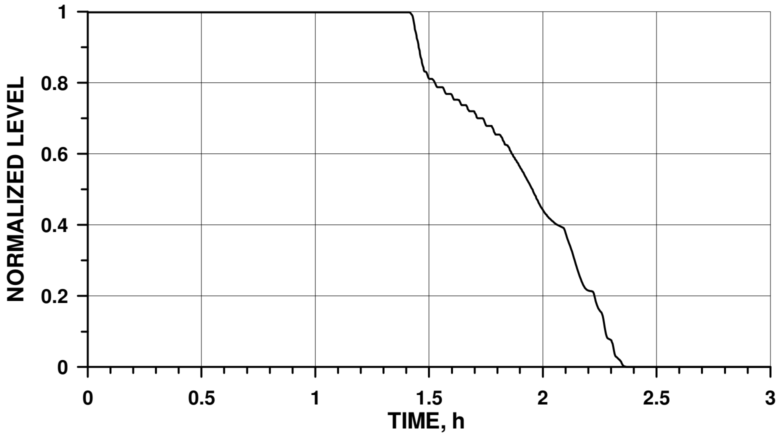

Fig. 6 Normalized coolant level in FWT (for all *Cases* except for the *Case 0*)

The single SG with the opened safety valve is filled up to the level of 13.7 m, while in three other SGs (triple SG in Fig. 3) the maximum level of around 10.5 m is reached (Fig. 5). This result is explained by 0.3 bar pressure difference between the single and triple SGs and, finally, by the fact that the path to discharge coolant from the triple SG (through the main steam header) is longer than that to discharge it from the single one. The increased heat removal from the primary system leads to a cool down of the coolant, which results in an increase of moderator density and, therefore, causes a volume contraction in the reactor coolant system. The outsurge from the pressurizer entails a decrease of pressurizer level whereby volume is released for the upper steam buffer so that primary pressure is reduced. The primary pressure is lowered (Fig. 7) to the level below the prevailing pressure in the HAs, causing a partial (approximately 40%) discharge of those connected to the hot legs. The earlier (500 s after NKV signal generation) disconnection of four HAs from the cold legs (see Chapter 2) prevents their discharge, so that they retain the initial coolant inventory. The inventory from the feedwater system injected to the steam generators maintains the core outlet temperature below 200°C for about 6 hours (Fig. 8). However, after its evaporation, heat removal from the primary system collapses for the second time, once again resulting in the increase of primary side temperature and pressure. Without implementation of further measures (AMM-2 or/and AMM-3) the core continues to heat up until its meltdown starts[18] about 5 hours after the heat removal out of the primary system has been suspended (simulation ***Case 1***).

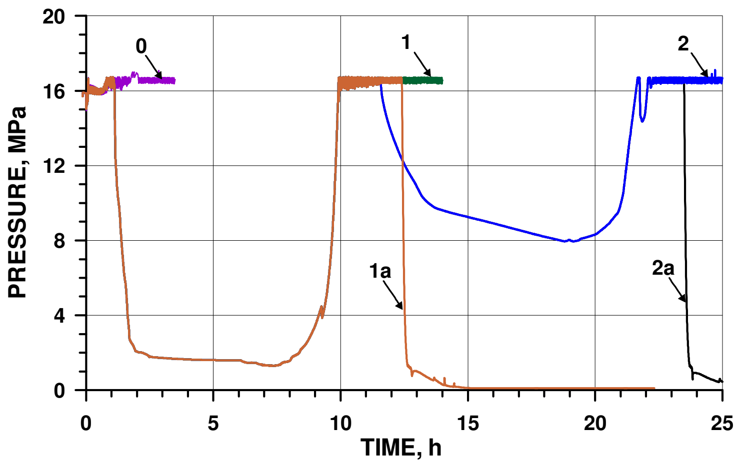

Fig. 7 Primary pressure at the core outlet

[18] Meltdown of Silver-Indium-Cadmium control rods

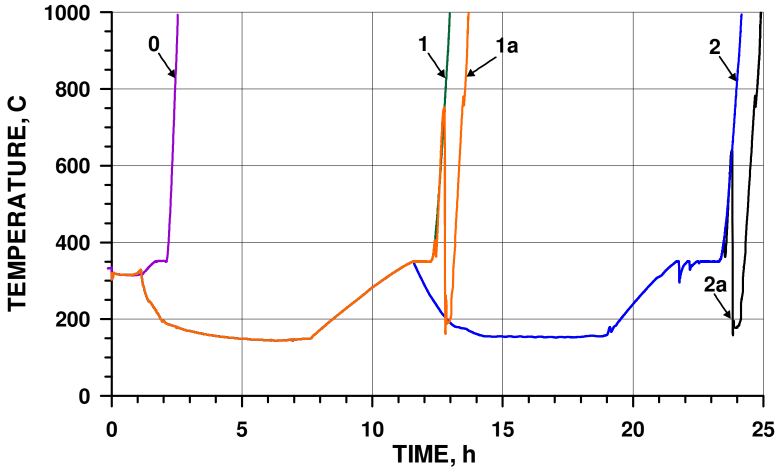

Fig. 8 Core outlet temperature

After the second dry-out of SGs, it takes nearly 4 h for the core outlet temperature to reach 350°C (Fig. 8), representing an artificial setpoint criteria implemented inside the model for starting the mobile pump, which supplies EFT water to the single SG (AMM-2). After 3 hours operating the mobile pump (simulation ***Case 2***), the SG coolant level reaches 12.2 m, which is kept constant at this setpoint for the next 2.5 hours (until the EFT is completely depleted). As a result of AMM-2 implementation, the primary pressure decreases to 80 bar (Fig.7), which is, however, still too high to resume passive injection of about 80 tons of the coolant which remain in the hot leg HAs. Two hours after the depletion of the EFT, the single SG depletes too at about 19 hours (Fig. 5), thus terminating heat removal from the primary system and causing the final (for the simulation ***Case 2***) core heat-up at about 23.5 hours since the accident initiation.

The third of the considered sequential accident management measures (AMM-3 or PSD) is used for injection of coolant into the primary system, which still remains in the hot leg HAs after implementation of AMM-1. This measure is initiated following the implementation of AMM-1 (***Case 1a***) or AMM-2 (***Case 2a***), when the core outlet temperature exceeds 400 ^{0}C (Fig. 4). As a result, about 70% of the remaining 80 tons of water is injected to the hot legs before the coolant level in the HAs drops to 1.65 m, triggering their disconnection from the primary loops, as noted in Chapter 2. With such a low amount of injected coolant, AMM-3 delays the core degradation onset by less than 1 hour (approximately 35 minutes)[19], which is significantly inferior to AMM-1 and/or AMM-2 in this aspect. This measure is primarily aimed at using the restored LPIS trains to cool down the core or, in a more severe case, at avoiding RPV failure under high pressure conditions.

The sequences of events related to the considered AMMs are summarized for all simulated cases in Table 2. The secondary bleed and feed procedures AMM-1 and AMM-2 demonstrate similar efficiencies, each providing a 10- to 11-hour time slot for core damage prevention. The sequential implementation of these two measures delays CDO by 21.5 hours compared to the case without AMMs (*Case 0*). The primary bleed and feed procedure is partially activated in parallel with (and as a consequence of) AMM-1 implementation. PSD initialization (AMM-3) finalizes this procedure, extending the above-mentioned time slots for the *Cases 1a* and *2a* by approximately 4.5% and 2.4% respectively.

[19] The early implementation of AMM-3 for an SBO without SSD (T1b scenario) delays SDO by approximately 80 minutes due to complete depletion of all eight HAs in this case (Wilhelm et al., 2018)

Table 2 AMM-related sequence of events and their impact on CDO

<table>
<tr><th rowspan="2">Event</th><th colspan="5">Elapsed time since SBO initiation [h:min]</th></tr>
<tr><th>Case 0</th><th>Case 1</th><th>Case 1a</th><th>Case 2</th><th>Case 2a</th></tr>
<tr><td>Coolant level in SGs < 4.4 m</td><td colspan="5">00:35</td></tr>
<tr><td>Depletion of SGs at high pressure</td><td colspan="5">01:00</td></tr>
<tr><td>Initiation of AMM-1</td><td rowspan="12">–</td><td colspan="4">01:05</td></tr>
<tr><td>Start of FWT depletion</td><td colspan="4">01:25</td></tr>
<tr><td>Disconnection of HAs from CLs</td><td colspan="4">01:30</td></tr>
<tr><td>Start of coolant injection from HAs to HLs</td><td colspan="4">01:40</td></tr>
<tr><td>End of FWT depletion</td><td colspan="4">02:20</td></tr>
<tr><td>Termination of coolant injection from HAs to HLs</td><td colspan="4">07:20</td></tr>
<tr><td>Depletion of SGs after AMM-1</td><td colspan="4">07:40</td></tr>
<tr><td>Initiation of AMM-2</td><td rowspan="5">–</td><td rowspan="3">–</td><td colspan="2">11:30</td></tr>
<tr><td>Complete depletion of EFT</td><td colspan="2">17:20</td></tr>
<tr><td>Depletion of the single SG after AMM-2</td><td colspan="2">19:10</td></tr>
<tr><td>Initiation of AMM-3 (PSD)</td><td>12:20</td><td rowspan="2">–</td><td>23:30</td></tr>
<tr><td>Disconnection of HAs from HLs</td><td>12:50</td><td>23:50</td></tr>
<tr><td>Core degradation onset (CDO)</td><td>02:30</td><td>12:50</td><td>13:25</td><td>24:00</td><td>24:35</td></tr>
</table>

4. Summary and conclusions

The presented work was focused on the capability of German Siemens KWU type PWR to withstand SBO conditions relevant to the DID sublevel 4b - the last one to prevent severe core damage. It was evaluated in terms of "SBO coping time", defined in (IAEA, 2015) as the "time available from loss of all permanently installed AC power sources until onset of core damage". The coping time (available time slot) depends on the efficiencies of the implemented countermeasures (AMM-1, AMM-2 and AMM-3), which were quantified in a series of ATHLET-CD simulations using the developed model of a Siemens KWU type PWR plant.

Each of the two implemented secondary bleed and feed procedures (AMM-1 and AMM-2) provides nearly equal contributions to the coping time, increasing it (compared to the case without AMMs) by 10 to 11 hours, so that the cumulative time window reaches 24 hours. In contrast, the primary bleed and feed countermeasure (AMM-3) is significantly less efficient (in particular due to a limited amount of available coolant in HAs), being able to extend SBO coping time by slightly more than half an hour.

With the coping time as long as 24 hours, the recovery of AC power for the Siemens KWU type PWR plant before reaching CDO becomes more likely. It should be noted that to compare the efficiencies of AMM-1 and AMM-2, only 1 of 4 EFTs has been depleted (by the operating mobile pump) in the simulations. In fact, although there is a strict separation of redundancies, there are – indeed closed in normal stand-by mode - connecting pipes between the EFTs, which can be opened via manual operation to provide the inventory of all four EFTs i.e. to one recovered emergency feed water pump or, in the considered scenario, to the mobile pump. This additional amount of water increases the efficiency of AMM-2 by at least four times, thus extending more than twice the coping time.

The efficiency of AMM-3 could be enhanced if the disconnection of the HAs through the threshold values "level low" or "time passed after NVK signal" has not been taken into account. More injected water would be available to delay even more the CDO. The accident sequence would be less severe than "with disconnection".

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