Annals of Nuclear Energy 169 (2022) 108969

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Numerical assessment for entry condition of severe accident management guidelines in a Swedish nuclear power plant

Nan Zhao *, Weimin Ma, Sevostian Bechta

KTH Royal Institute of Technology, Roslagstullsbacken 21, 106 91 Stockholm, Sweden

ARTICLE INFO

Article history:
Received 20 August 2021
Received in revised form 17 December 2021
Accepted 3 January 2022
Available online 11 January 2022

Keywords:
Severe accident
Severe accident management guidelines
Entry condition
Core exit temperature
MELCOR simulation

ABSTRACT

The entry conditions of severe accident management guidelines (SAMGs) in pressurized water reactors (PWRs) rely on the indication of core exit temperature (CET). Yet, the setpoints for the CET may be different from plant to plant. Most Westinghouse PWR designs adopt the setpoint of CET at 650℃ as the entry condition of the SAMGs, since this setpoint is an effective indicator of core damage in a wide spectrum of accident sequences. Motivated by the interest in the verification & validation of SAMS after the Fukushima accidents, the present study is conducted numerically to verify the effectiveness of the CET setpoint for the transition from emergency operation procedures (EOPs) to SAMGs in a Swedish nuclear power plant. For this purpose, six representative severe accident sequences covering the main contributors to the core damage frequency (CDF) are analyzed using the MELCOR code. Moreover, different CET readings and alternative entry conditions are also investigated. The simulation results show that the average CET = 650 °C is the effective setpoint as the entry condition of SAMGs, i.e., given this setpoint the transition from EOPs to SAMGs will take place slightly before the occurrence of core degradation, which secures the intended mitigation of SAMGs while keeping EOPs active as long as possible. On the other hand, it is too conservative if the maximum CET = 650 °C is used as the setpoint of entry condition of SAMGs, i.e., it will result in an excessive realization of SAMGs over EOPs. The coolant temperature in the primary circuits, the water level in the RPV and the hydrogen concentration in the containment can also be applied as reference indications of core damage states in the accident management.

1. Introduction

Severe accident management guidelines (SAMGs) have been introduced in the pressurized water reactors (PWRs) since 1990 s for completing and improving the defense-in-depth concept, in line with the recognition that emergency operation procedures (EOPs) cannot cover the severe core damage sequences as the TMI-2 accident (Sehgal, 2012; Sehgal, 2012; Hill et al., 2016). The SAMGs are integrated with the EOPs by utilizing all available off-site and on-site components and resources, focusing on the mitigation of accident consequences in the scenarios with core degradation, e.g., fission products release, hydrogen risk in containment, etc. Compared with the preventive EOPs, the mitigative SAMGs provide the higher flexibility of operation to adopt the remaining components due to the uncertainties under severe accident conditions (IAEA, 2009). The transition from EOPs to SAMG is usually required when the core damage occurs, i.e., the preventive EOPs are considered to save the core and avoid fission products release before the core damage, while the mitigative SAMGs are applied to mitigate the accident consequences like radioactive fission products release or in-containment hydrogen risks after the core damage. The core exit temperature (CET) was proposed as the indicator to monitor inadequate core cooling and core damage in PWRs (Gabor et al., 2012; Hermsmeyer et al., 2014; NEA, 2010) since the peak cladding temperature is unmeasurable in the PWRs. Therefore, the CET is employed as the indication for SAMG entry criteria.

In PWRs, the thermocouples located above the top plate of core are the instrumentation which can be used to obtain the CET, and the resistance temperature detectors (RTDs) in the primary loops are usually taken as the backup measurements of CET in case of the thermocouple failure. The variation of the CET depends on the accident scenarios, and the reliability of readings is limited by the measurement range of the temperature sensors (NEA, 2010). Therefore, the availability of the instrumentation and the setpoint become essential to affect the transition from EOPs to SAMGs (Suh et al., 2006). To avoid the misdiagnosis and reduce the complexity of operation under severe accident, the transition criteria should be consolidated to fit all potential scenarios.

* Corresponding author.
E-mail address: nanzhao@kth.se (N. Zhao).

https://doi.org/10.1016/j.anucene.2022.108969
0306-4549/

An earlier EOP-SAMG transition (i.e., a lower setpoint of CET) may shorten the preventive period of EOPs since there is no fallbacks for the entrance into SAMGs, while a later transition (i.e., a higher CET setpoint) may reduce the mitigative outcomes of SAM measures because they are implemented under more severe conditions of core degradation. Therefore, either early or late entering to SAMGs is not preferable, and there should be a right point in time for the intervention of SAMG.

The selection of SAMG entry criteria may be different in specific reactor designs (Hermsmeyer et al., 2014). In Westinghouse PWRs, the CET of 650℃ is used as the setpoint entering the SAMGs. The Korean PWRs such as OPR1000 (Park and Hong, 2011) and APR1400 (Yun et al., 2015) also apply the CET setpoint of 650℃ as the entry condition of SAMGs. In contrast, the CET setpoint entering SAMGs in VVER plant is lower (450℃–550℃), and some EDF PWRs have a higher CET setpoint as the entry condition of SAMG (Hermsmeyer et al., 2014; NEA, 2010). There are other approaches to direct the EOP-SAMG transition, such as in the Combustion Engineering PWRs, the entry criteria rely on the superheat at core exit, which is a parametric function of RCS pressure and CET (NEA, 2010).

Thus, an essential question is raised for the implementation of the SAMGs of a specific nuclear reactor, which is whether or not the selected entry criteria of SAMGs are proper to direct the EOP-SAMG transition. The present study is concerned with the verification and validation of the SAMGs of a Swedish nuclear power plant, which is a typical Westinghouse 3-loop PWR design with the entry condition of SAMGs at 650℃ CET. Since the plant has experienced several power uprates from its original capacity (in accompany with replacement of components), the CET may be different from the original perception under severe accident conditions, and consequently delay or advance the transition from EOPs to SAMGs in some cases when using the same setpoint of 650℃. Moreover, the PWR has 16 thermocouples installed upon the top plate of the core, which means there are 16 in-situ indicators of CETs. This brings in another question: which CET (maximum, average, minimum or alternatives) should be associated with the setpoint. Therefore, it is necessary to perform an investigation on the reliability of the transitional setpoint from EOPs to SAMGs in different severe accident scenarios. The objective is to verify that the setpoint of the CET chosen is the effective entry condition of SAMGs, i.e., the selection of CET and its setpoint can provide the right point in time for the transition from EOPs to SAMGs during hypothetical severe accident scenarios of the uprated Westinghouse PWR.

To verify the entry condition of SAMGs, six severe accident scenarios induced by three representative initiating events (Travis et al., 1990) are selected to be simulated by the MELCOR code, including loss of coolant accident (LOCA) with concurrent loss of safety injections, total loss of feedwater (TLOFW), and station black-out (SBO). The LOCA accident is accounted as the dominant contributor to the core damage frequency in PWRs. With the hypothetical concurrent loss of AC power, the safety injection cannot be operated immediately to make up the water level in the reactor pressure vessel (RPV), leading to a fast core degradation and RPV failure. The TLOFW accident is initiated by a failure of feed-water pipeline, which directly leads to the loss of coolant in the secondary sides of steam generators (SGs). Subsequently, depressurization of the reactor coolant system (RCS) is implemented once the SG water level is low, prepared for primary side bleed & feed (PBF) actions. The SBO accident is also one of the major contributors to the core damage frequency, which may involve high-pressure core degradation in the RPV. The accident sequences from the three initiating events occupy most of core damage frequency in the PWR.

The peak cladding temperatures (PCT) and CETs in different core channels are obtained firstly from the MELCOR simulations. The difference between the time at PCT = 1204℃ and the time at CET = 650℃ is considered as the measure (so-called "grace time") for effectiveness of the transition from EOPs to SAMGs since PCT = 1204℃ represents the initiation of cladding failure and core degradation in the accident sequences. The grace times of using maximum, minimum and average CETs are compared, and the CET instrumentation and the alternative transitional condition are discussed by taking the advantage of the detailed outputs from the MELCOR simulations.

2. MELCOR model

2.1. Thermal-hydraulic nodalization

MELCOR is a fully integrated, engineering-level simulation code developed by Sandia National Laboratories under the auspices of the U.S. Nuclear Regulatory Commission (NRC). A broad spectrum of severe accident phenomena is treated by the MELCOR code, including thermal–hydraulic responses in reactor and containment; core degradation and relocation; hydrogen generation and transport; fission products release and transport, molten core-concrete interaction, etc. Code packages, such as CVH, FL, COR, RN and CAV, are employed to simulate the different phenomena in a severe accident sequence (Humphries et al., 2015; Humphries et al., 2015). The code version v.2.2.9541 is applied to perform the simulations on a LENOVO workstation.

In the MELCOR model of the PWR, Fig. 1 shows the nodal scheme of control volumes (CVs) and flow paths (FLs) used to simulate thermal hydraulics in the primary and secondary circuits, while Fig. 1b illustrates the core cells in the RPV, which are defined through concentric radial rings and axial levels, to track the behavior of core materials. Two of the three primary circuits are grouped into the left loop of Fig. 1 which contains a double capacity of single circuit, while the remaining circuit is the right loop which represents the primary circuit with the pressurizer. The secondary circuits are represented by two loops correspondingly. The essential components in the primary loops are RPV, hot/cold legs, steam generators, and reactor coolant pumps (RCP). The pressurizer is modelled by a control volume, connected with the hot leg of the right primary loop though the surge line. The top of pressurizer connects with the pressurizer relief tank through the safety relief valves (SRVs) and three power-operating relief valves (PORVs). The RPV is represented by six CVs in accordance with the geometry: lower plenum, downcomer, core, bypass, upper and

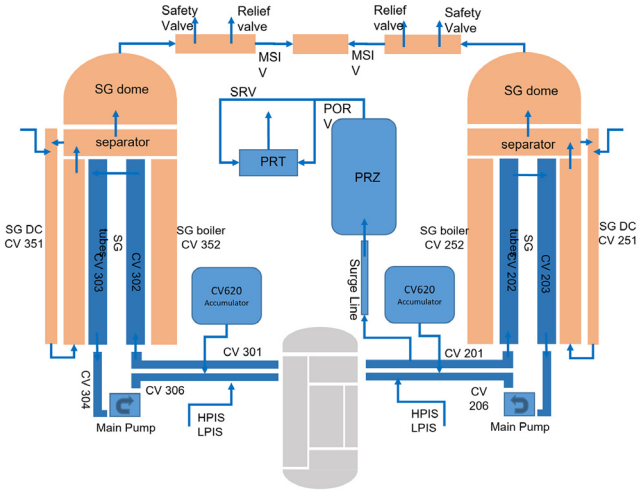

Fig. 1. Thermal-hydraulic nodes of MELCOR model for coolant circuits.

lower plenums. The lower plenum CV covers the coolant space below the core support plate and the core CV covers the entire region of active fuel, so they are interfacing with the COR cells representing the lower plenum and the active fuel, respectively.

The PWR has a large dry containment with a total volume of around 50000 m^3, consisting of a cylindrical building of prestressed concrete with a semispherical dome. In the MELCOR model of the PWR plant, the entire containment is represented by seven CVs as shown in Fig. 2: the reactor cavity, the lower compartment, the annular compartment, the compartment accommodating the two identical SGs, the compartment with the remaining SG and the pressurizer, the refueling pool and the dome.

In addition, the MELCOR model of thermal-hydraulics also includes the safety systems, such as multiple-venturi scrubber system (MVSS), accumulators (ACC), auxiliary feedwater (AFW) system, high pressure safety injection (HPSI) and low head safety injection (LHSI). The MVSS used for containment venting comprises assemblies of small venture scrubbers submerged in the water pool of a storage tank and can be activated either passively by a rupture disc when the containment pressure exceeds 5.14 bar, or by a manual operation through relief valves.

The HPSI system will pump coolant from the refueling water storage tank (RWST) into boron injection tank (BIT) by 3 separate pipelines and drive the water into the cold legs of the primary circuits, with a pumping head up to 17 MPa. The LHSI system consists of 4 centrifugal pumps connecting with the RWST and the containment sump, which are charactered with low injection head and high flow rate up to 110 kg/s. The auxiliary feedwater (AFW) systems are used to provide coolant injection into the secondary sides of steam generators (SGs), by a steam-turbine-driven pump and two electric-motor-driven pumps. The two motor-driven AFW (MDAFW) systems are connected to two SGs, providing a total mass flowrate of 48 kg/s (i.e., 24 kg/s each), while the turbine-driven AFW (MDAFW) system is connected to the remaining SG, with a maximum mass flowrate of 48 kg/s. Both TDAFW and MDAFW systems take water from the condensate water storage tank (CST), which has a total volume of around 1000 m^3.

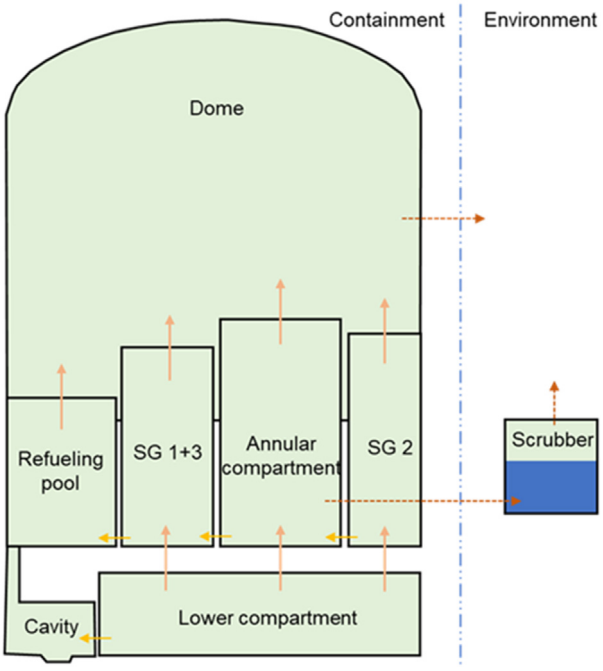

Fig. 2. Thermal-hydraulic nodes of MELCOR for the containment.

The containment spray is considered as two separate spray functions in the dome and the sump, respectively. The spray pumps can provide a stable mass flowrate of 60 kg/s from the RWST to the containment, and from the containment sump under the recirculation mode. The passive safety system such as the accumulators in the primary circuits and the passive catalytic recombiners (PARs) in the containment are also modelled.

2.2. COR nodalization and CET estimations

As shown in Fig. 3, the medium COR nodal scheme in our previous work (Zhao et al., 2021) is used in the present study, which divides the core into 7 rings and 27 levels, and the lower plenum into 10 rings and 16 levels. The MELCOR simulation will not only calculate the temperature of the core materials in each COR cell, but also provide local fluid temperature in the flow channel seen by the cell (Humphries et al., 2015; Humphries et al., 2015). In the present study, the CETs refer to the local fluid temperatures associated with the uppermost COR cells (as illustrated in Fig. 4), and therefore there are seven CETs (one for each COR ring) from the MELCOR modeling.

It is obvious that the CETs in the MELCOR modeling is different form the CETs measured by the thermocouples in the reactor. The CETs from the thermocouples are the measurement of local temperatures at the tips of the 16 thermocouples at the core exit, while the CETs in the MELCOR are averaged temperatures of the uppermost cells of the 7 COR rings. Nevertheless, this is the best-estimate representation of CETs from MELCOR simulation, although it may be better predicted by detailed codes (Freixa et al., 2015).

The fission products release during core degradation is modeled by the coupling of COR and RN packages in the MELCOR code. In RN package, there are specific models describing the process of fission product release, aerosol agglomeration and deposition, fission products condensation and evaporation, and the impacts of safety functions on the fission products transportation. Release of radionuclides can occur from the core fuel (with nonradioactive releases from other core structures), from the fuel-cladding gap, and from material in the cavity. The CORSOR-M model with surface-to-volume ratio is applied to calculate the release rates of

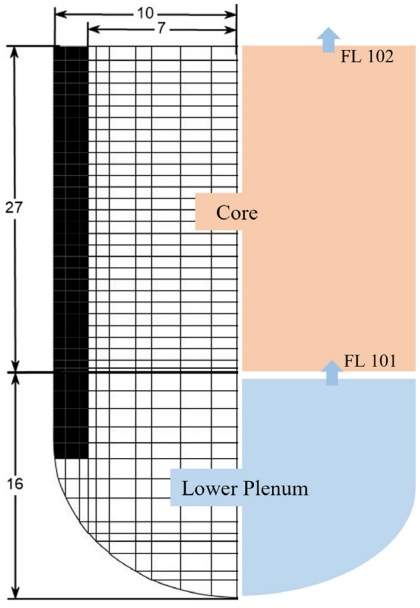

Fig. 3. COR nodalization of MELCOR model.

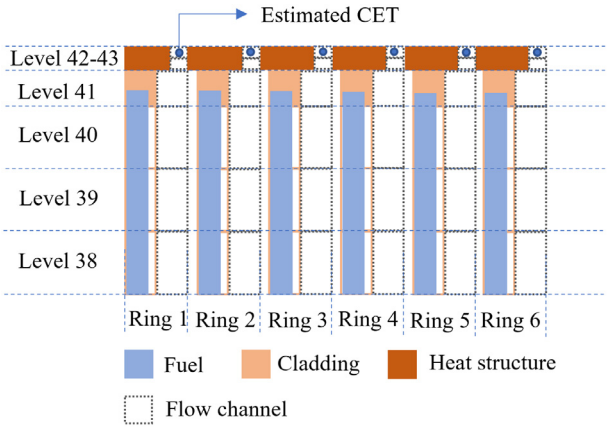

Fig. 4. CETs in the MELCOR simulations.

radioactive radionuclides from core fuel material, which exists in the intact fuel component, in refrozen fuel material on other components, in particulate debris, and in the molten pool components (Humphries et al., 2015; Humphries et al., 2015).

The transportation of fission products is modelled by the coupling of RN package and CVH package, with concerning the processes of aerosol dynamic, fission product evaporation and condensation, and the decay heat distribution, etc. The radionuclides are described as 16 classes by grouping elements that have similar chemical properties. The classes are referred by representative elements (e.g., Xe for noble gases, Cs for alkali metals, I for halogens, Ba for alkaline earths, Te for chalcogens, Mo for early transition metals, etc.). The noble gases are mainly distributed in the atmosphere of CVH volume after its release from fuel. The fission product Cs is treated in the two classes of CsI and CsOH, which are mainly resided in the water pool. The other volatile fission products would be transported in the aerosol released from the surface of heat structure. The release aerosol would be treated as specific fog component in CVH package.

The initial inventory of fission products is estimated by MELCOR through a scaling factor of the ORIGEN calculation results of a Westinghouse PWR at the end of fuel cycle (Ostmeyer, 1985). Upon release from fuel, the total class masses are converted to compound form with a corresponding increase in mass from the added nonradioactive material (e.g., the hydroxide mass in CsOH). The released fission products would be distributed into atmosphere and water pools in CVH packages according to their physical properties. Since the interest of the present study is the transitional point from EOPs to SAMGs, the following discussion is only focused on the initial core damage state near CET = 650 °C and PCT = 1204 °C, where only gap release (of noble gases) occurs.

3. Accident scenarios

As the entry condition of SAMGs, the CET setpoint should be selected in such a way that it reflects the core damage state in a wide spectrum of accident scenarios. To examine the core damage state at the CET of 650 °C, the main contributors to core damage frequency in Westinghouse PWRs (Travis et al., 1990; U.S.NRC, , 1990) are considered here, including loss of coolant accident (LOCA) with concurrent loss of safety injections, total loss of feedwater (TLOFW), and station black-out (SBO). Accordingly, six severe accident scenarios are simulated by the MELCOR code:

- *Scenario SBO*: The accident is initiated by total loss of AC power. With the failure of AC power supply, the reactor scram occurs, followed by trips of the turbine, the reactor coolant pumps (RCPs) and the main feedwater pumps. The coolant recirculation in the primary circuits is terminated with the RCPs trip, and the safety injections are disabled due to loss of AC power. As a result, the core will be uncovered through discharge of SRVs/PORVs and heated up by decay heat, finally leading to core degradation under high pressure. At the same time, the water level in the secondary sides of the SGs will decrease due to boil-off. Other assumptions are as follows: (i) The turbine driven AFW system is activated upon a signal of low SG water level and will be available till the depletion of the CST inventory; (ii) The primary and secondary PORVs can be manually opened before the exhaustion of DC power at 8 h.
- *Scenario LBLOCA*: The accident is initiated by a large break LOCA together with SBO. A double-ended break of the pressurizer surge line (280 mm ID) is assumed, leading to the rapid depressurization in the RCS. The reactor shutdown is triggered following the signal of low RPV pressure, and the RCPs coast-down and the turbine trip occur immediately, and then the main steam lines to the turbine are isolated. The safety injections which need AC power are assumed to be disabled, though the turbine driven AFW system is available. The rapid depressurization will soon trigger the passive accumulator injection, and most of coolant in the RCS will be discharged rapidly.
- *Scenario MBLOCA*: The accident is initiated by a medium break LOCA together with SBO. The break is a 100-mm ID hole located at the lower elbow of the pressurizer surge line. The other conditions are assumed as in Scenario LBLOCA.
- *Scenario SBLOCA*: The accident is initiated by a small break LOCA together with SBO. The break is to simulate by the spurious opening of a PORV at the top of the pressurizer, probably induced by an electrical/mechanical error. The other conditions are assumed as in Scenario LBLOCA.
- *Scenario SBLOCA w/o AFW*: This accident is initiated by the spurious opening of a PORV at the top of the pressurizer, together with SBO. In addition to the loss of safety injections which need AC power, the passive AFW system driven the steam is also postulated to be lost due to the unexpected closure of valves (Carney and Sherman, , 1990). The other conditions are assumed as in Scenario SBLOCA.
- *Scenario TLOFW*: This accident is initiated by the total loss of main feedwater system, together with SBO. The reactor scram is triggered by the signal of low SG water level, followed by the trips of the RCPs and the turbine. All AFW systems are assumed to be unavailable, and thus the SG water level continues reducing without any feed water. To avoid SG tube rupture and containment bypass, the operators must manually open the PORVs of the primary sides to depressurize the RCS when the low SG water level is detected. The primary side will maintain high pressure till the depressurization.

4. Simulation results and discussion

In all the six accident scenarios, the water levels in the RPV will drop below the top of active fuel (TAF), and the fuel starts heating up following the core uncovering. The core degradation is initiated after the cladding temperature exceeds failure criterion. In the present study, three signals are employed to indicate the core damage state: (1) Gap release which represents the occurrence of cladding failure when the peak cladding temperature (PCT) reaches 900 °C; (2) PCT at 1204 °C which represents the starting point for the occurrence of cladding embrittlement and the potential loss of cladding geometry (Kim et al., 2005; Billone et al., 2008; Esmaili et al., 2011); and (3) Initiation of core relocation (core candling) which represents the beginning of significant release of radioactive fission products (e.g. the noble gases) from the core. After the water level in the RPV drops below the bottom of active fuel

(BAF), the entire core is uncovered. Without any SAM measures, the RPV lower head will finally fail due to the attack of the core melt (corium) relocated there. For the above six accident scenarios, Table 1 lists the chronological events of interest in the present study.

Once the core degradation occurs, the transition from EOPs to SAMGs should be made because the actions in EOPs are not designed for the mitigation of severe accident consequences with a significant release of fission products from the fuel. In theory, the entry condition of SAMGs should be an imminent or actual core damage and the failure of all EOPs actions.

Fig. 5 shows the history of the average CET in the duration between TAF uncovered and BAF uncovered in every simulated accident scenario. The average CET is the arithmetic mean value of the 7 CETs in MELCOR simulation. The *x* axis in the figure represents the elapsed time since TAF uncovered. In the scenarios of LBLOCA and MBLOCA, the quick loss of coolant after the initiating events leads to rapid depressurization and water level decrease below the TAF in one minute. The subsequent accumulator injections make up the coolant, resulting in a decrease of the CETs for a short period. The setpoint for the entry condition of SAMGs is reached in 16.8 min and 45.0 min for LBLOCA and MBLOCA, respectively. In the scenarios of SBLOCA w/o AFW and TLOFW, the loss of coolant is slower, so the rapid increase of the core exit temperature is delayed, resulting in that the setpoint of CET at 650 °C is reached in 1.26 h in SBLOCA w/o AFW and 0.99 h in TLOFW. In the scenarios of SBLOCA and SBO, the times reaching the setpoint of CET at 650 °C are significantly longer, thanks to slow loss of coolant due to the small break and the operation of turbine driven AFW which enhances the convective heat transfer in the SG secondary sides. The entry condition of SAMGs is met in 2.84 h and 14.93 h for SBLOCA and SBO, respectively.

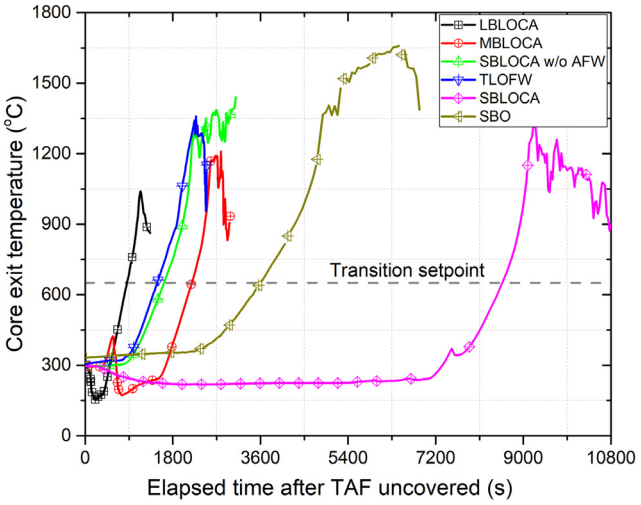

Fig. 5. Averaged core exit temperatures in different accident scenarios.

4.1. Outcomes of different CET setpoints

For the entry condition of SAMGs, what if the setpoint of the average CET is different from 650 °C? For instance, the setpoint of CET is 480 °C for the entry condition of the SAMGs in the Combustion Engineering PWRs (NEA, 2010). Moreover, CET = 760 °C is sometime used as the cladding damage criterion in analysis (Lutz, Editor (1996).), and CET = 1100 °C is the entry condition into SAM guidance of EDF PWRs (NEA, 2010).

In order to answer the above what-if question, the setpoints of 480 °C, 760 °C and 1100 °C are also investigated in the present study, with the objective to verify their candidacy as the setpoint indicating imminent core damage.

Table 2 lists the times reaching these setpoints in the five accident scenarios, i.e., the periods elapsed from reactor scram. In the LBLOCA scenario, the CET of 480 °C is reached in 13.6 min after the TAF is uncovered, and the core damage occurs subsequently in 17 min after reactor scram. It takes around 27 min for CET to increase from 480 °C to 1100 °C. In the SBLOCA and SBO scenarios, it also takes 14 min and 29 min for CET to increase from 480 °C to 1100 °C respectively with the much smaller break sizes. Comparatively, the MBLOCA scenario takes the longest time of 48.8 min for CET to increase from 480 °C to 1100 °C.

Table 2 also lists the maximum temperatures of cladding components in the COR cells at the time when the setpoints are reached during the six accident scenarios, i.e., the peak cladding temperatures (PCT) in the core. In the LBLOCA scenario, the core is uncovered rapidly and uncooled due to large break size, and therefore the PCT is higher than those in other scenarios. In contrast, due to the slow loss of coolant in the SBO scenario, the PCT is lower in SBO than in other scenarios. From this table, it can be found that the PCT reaches1204°C (cladding embrittlement) in any scenario after the average CET is higher than both 480 °C and 650 °C. However, for the CET setpoints of 760 °C and 1100 °C, the gap release (PCT = 900 °C) in all scenarios occurs before the setpoints, and there is no grace time to core damage (PCT = 1204 °C), except for the three scenarios of SBLOCA w/o SFW, SBLOCA and SBO at the setpoint of CET = 760 °C. Especially for CET = 1100 °C, the PCTs have exceeded the melting points of cladding (around 1800 K). As a result, the setpoints of CET = 650 °C is the most appropriate indication for imminent core damage.

According to the timing of events as listed in Table 1, the core relocation begins only a few minutes after the initiation of core damage (cladding embrittlement) in all accident scenarios. Therefore, the grace times in Table 2 indicate the available time windows for preparation of the operator actions in the SAMGs under various entry condition (CET setpoints) of SAMGs and accident scenarios. The negative values of the grace time in Table 2 imply that the best transitional point from EOPs to SAMGs is missed if the

Table 1
Chronological events of severe accident sequences.

Event	Scenario					
	LBLOCA	MBLOCA	SBLOCA w/o AFW	TLOFW	SBLOCA	SBO
Reactor Scram	0 s	0 s	0 s	24 s	0 s	0 s
TAF uncovered	64 s	419 s	2909 s	2103 s	2167 s	50156 s
ACC activated	164 s	960 s	9360 s	4520 s	2210 s	52260 s
Gap release (PCT at 900 °C)	895 s	2617 s	4539 s	3549 s	10142 s	53991 s
Core damage (PCT at 1204 °C)	1024 s	2801 s	4795 s	3759 s	10444 s	54551 s
Core relocation	1156 s	2955 s	4961 s	3912 s	10617 s	54813 s
BAF uncovered	1126 s	3232 s	5713 s	4069 s	16665 s	54563 s
RPV failure	8382 s	10463 s	24152 s	27581 s	19380 s	69484 s

Table 2
Outcomes of different setpoints of the average CET.

Scenario	*LBLOCA*	*MBLOCA*	*SBLOCA w/o AFW*	*TLOFW*	*SBLOCA*	*SBO*
Time reaching the different setpoints of CET:						
CET 480 °C	820s	2420s	4230s	3300s	9840s	53180s
CET 650 °C	1010s	2700s	4560s	3580s	10230s	53780s
CET 760 °C	1120s	2840s	4750s	3770s	10440s	54140s
CET 1100 °C	2440s	5350s	5080s	4140s	10680s	54920s
Peak cladding temperature (PCT) at the setpoints of CET						
CET 480 °C	775 °C	684 °C	642 °C	645 °C	682 °C	601 °C
CET 650 °C	1159 °C	1015 °C	917 °C	938 °C	972 °C	810 °C
CET 760 °C	1550 °C	1286 °C	1136 °C	1221 °C	1194 °C	962 °C
CET 1100 °C	2228 °C	2129 °C	2124 °C	2180 °C	2089 °C	1774 °C
Grace time prior to core damage (PCT = 1204 °C)						
CET 480 °C	204s	381s	565s	459s	604s	1372s
CET 650 °C	14s	101s	235s	179s	214s	772s
CET 760 °C	−96s	−39s	45s	−11s	4s	412s
CET 1100 °C	−1416s	−2549s	−285s	−381s	−236s	−368s

corresponding setpoints of CET are applied, and a large grace time implies too early transition.

What is the outcome of an SAM action (e.g., safety injection to RCS) if the above setpoints are chosen no matter what core damage state is? The concern is whether the severe accident can be terminated or stabilized even if the SAM action is implemented after severe core damage. To answer this question, below is an investigation on the effect of safety injection (as an SAM action in SAMGs) implemented at different CET setpoints.

Figs. 6–8 show the histories of PCT in the scenarios of LBLOCA, SBLOCA and TLOFW with the safety injection activated at the average CET = 480 °C, 650 °C, 760 °C and 1100 °C, separately. The safety injection is realized by activation of 1 HPSI pump. The simulation results show that in the SBLOCA scenario the safety injection is capable of preventing core damage with the setpoint at CET = 480 °C or 650 °C, i.e., PCT is below 1204 °C. If the CET setpoint is 760 °C or 1100 °C, the PCT will exceed 1204 °C (core damage) even with the safety injection. While the setpoint CET = 650 °C in the TLOFW scenario would avoid a severe core damage (with a short period of PCT slightly over 1204 °C), a lower CET setpoint would be better for the LBLOCA scenario (due to an extended period of PCT well above 1204 °C if the setpoint is 650 °C).

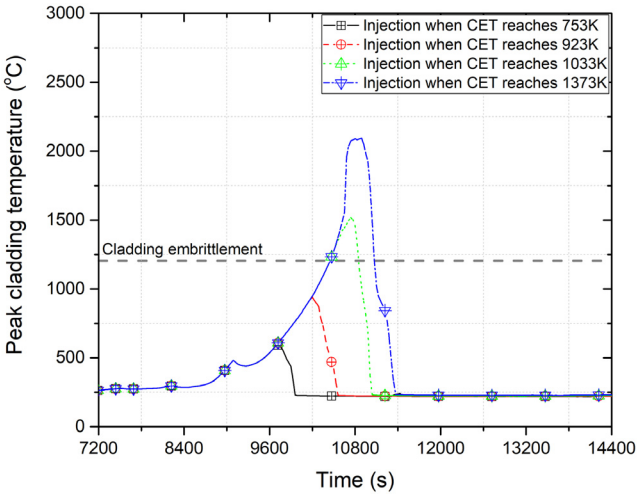

Fig. 7. Peak cladding temperatures in TLOFW scenario with safety injection at various CETs.

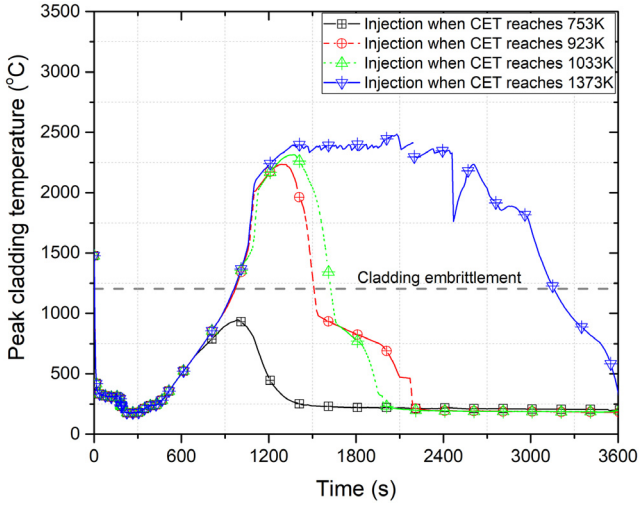

Fig. 6. Peak cladding temperatures in LBLOCA scenario with safety injection at various CETs.

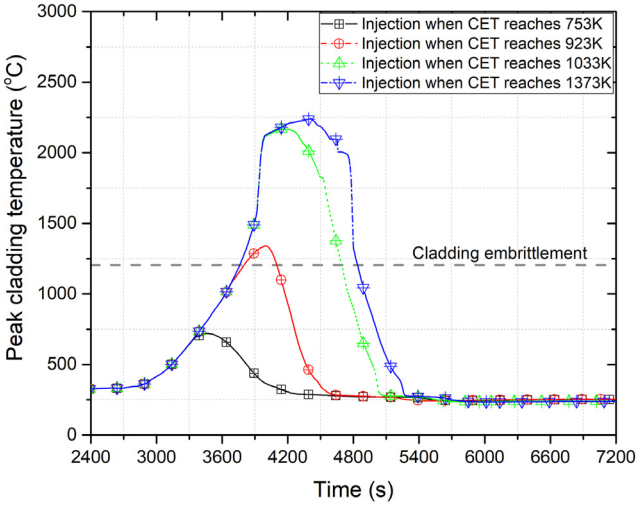

Fig. 8. Peak cladding temperatures in SBLOCA scenario with safety injection at various CETs.

Interestingly, in all above scenarios with the CET setpoints of 480 °C, 650 °C, 760 °C and 1100 °C being the entry condition of the SAMGs, the safety injection as a SAM action simultaneously activated at any of the setpoints is sufficient to cool the reactor, although different extents of core damage are predicted for the higher setpoints of CETs. For the low-pressure accident sequences, e.g., LBLOCA and MBLOCA, the core damage is more significant even with the safety injection at CET = 650 °C.

4.2. Outcomes of different temperature indicators

In the preceding discussion, the average CET is used for the setpoint of SAMG entry condition, which is averaged value of the 7 CETs (cf. Fig. 4) from MELCOR simulation. What would happen if the setpoint is one of the 7 CETs instead of the averaged one? In addition, the RTDs in the primary circuits can also be used as alternative indicators of core damage in case the thermocouples fail (NEA, 2010). The operators need to make decisions based on the remaining indicators in the RCS (Kim, 2014; IAEA, 2015; Rempe et al., 2016), under an extreme condition (e.g. loss of all thermocouples in the RPV).

To investigate the outcomes of the alternatives, six other temperature indicators (in addition to the average CET) in the MELCOR model are considered in the present study, which are as follows:

- Maximum CET among the seven CETs (cf. Fig. 4);
- Minimum CET among the seven CETs (cf. Fig. 4);
- Temperature of the hot leg in the left loop (cf. Fig. 1);
- Temperature of the cold leg in the left loop (cf. Fig. 1);
- Temperature of the hot leg in the right loop (cf. Fig. 1); and
- Temperature of the cold leg in the right loop (cf. Fig. 1).

Table 3 shows the times for the six temperature indicators to reach 650 °C, and their comparisons with the average CET (the reference case). It can be seen that five of the six temperature indicators are all taking longer times to reach 650 °C than the average CET does, and those of the cold legs are the longest. Moreover, the differences among CETs depend on the accident scenarios: they are quite significant for low-pressure scenarios such as LBLOCA (see Fig. 9), while the differences are marginal for high-pressure scenarios such as SBO (see Fig. 10). In other words, the CETs of accident scenarios with high RCS pressure are almost identical (at least until 650 °C), so the setpoint 650 °C is effective to all CET indicators (all CETs reach 650 °C at around 3500 s in Fig. 10). But this is not the case for low-pressure scenarios where only the average CET seems the best as the setpoint, since the maximum CET and the minimum CET reaches 650 °C respectively several minutes earlier and much later than the average CET does.

Table 3 also shows that for all accident scenarios, the setpoint of 650 °C is no longer effective if one wants to use the readings of the RTDs in the hot and cold legs, since the core damage occurs much earlier than the RTDs reach the setpoint (also see Table 4). This means that new setpoints of lower temperatures should be applied when the RTDs are employed as the indicators of core damage and entry into SAMGs. Table 4 lists the readings of different indicators at the timing of entry condition reached (CET = 650 °C) and core damage occurred (PCT = 1204 °C). It can be concluded that the hot leg temperature is more reliable than the minimum CET in the low-pressure scenarios (CET of outermost ring in Fig. 9), but in the high-pressure scenarios (see Fig. 10), the minimum CET would be earlier to reach set point than the hot leg temperatures.

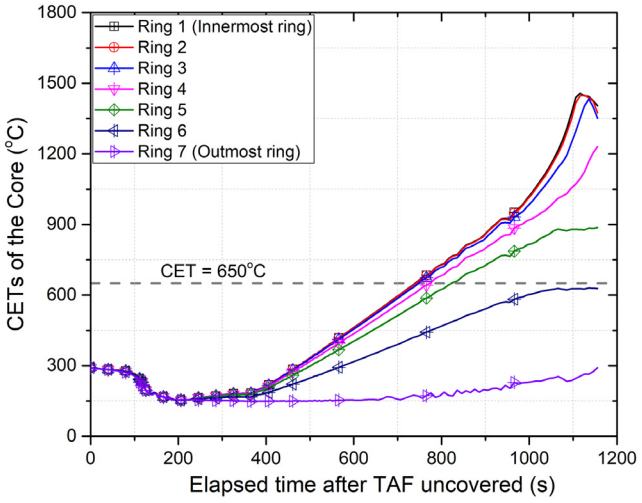

Fig. 9. CETs in LBLOCA scenario.

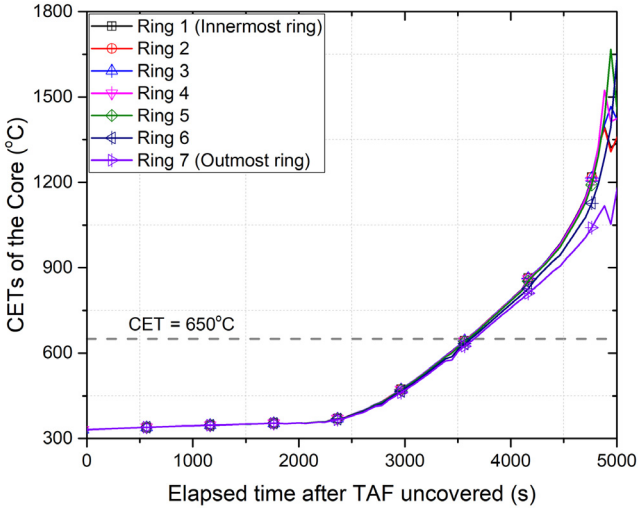

Fig. 10. CETs in SBO scenario.

Table 3
Times of different temperature indicators reaching 650 °C in the MELCOR simulations.

Indicator	*Scenario*					
	LBLOCA	*MBLOCA*	*SBLOCA w/o AFW*	*TLOFW*	*SBLOCA*	*SBO*
Maximum CET	810s	2480s	4470s	3510s	9960s	53755s
Average CET (Reference)	1010s	2700s	4560s	3580s	10230s	53780s
Minimum CET	2430s	5390s	4670s	3670s	10830s	53840s
Hot leg temperature (Left loop)	2640s	5350s	7390s	4800s	12270s	57020s
Cold leg temperature (Left loop)	5590s	N/A*	N/A	24860s	N/A	69720s
Hot leg temperature (Right loop)	3020s	5400s	6990s	4680s	12270s	57020s
Cold leg temperature (Right loop)	5590s	N/A	19466s	24860s	N/A	65520s

* The temperature 650 °C is never reached in the simulations.

Table 4
Temperatures of different indicators at the average CET = 650 °C and PCT = 1204 °C.

Indicator \ Scenario	LBLOCA	MBLOCA	SBLOCA w/o AFW	TLOFW	SBLOCA	SBO
Temperatures of different indicators at the average CET=650°C						
Maximum CET	925 °C	851 °C	704 °C	698 °C	807 °C	662 °C
Minimum CET	216 °C	377 °C	595 °C	599 °C	444 °C	641 °C
Hot leg temperature (Left loop)	502 °C	431 °C	303 °C	350 °C	462 °C	360 °C
Cold leg temperature (Left loop)	194 °C	212 °C	309 °C	317 °C	223 °C	357 °C
Hot leg temperature (Right loop)	499 °C	260 °C	390 °C	303 °C	440 °C	354 °C
Cold leg temperature (Right loop)	232 °C	215 °C	312 °C	311 °C	222 °C	357 °C
Temperatures of different indicators at PCT=1204°C						
Maximum CET	951 °C	958 °C	873 °C	678 °C	1016 °C	957 °C
Minimum CET	229 °C	422 °C	720 °C	582 °C	508 °C	889 °C
Hot leg temperature (Left loop)	499 °C	435 °C	303 °C	349 °C	227 °C	362 °C
Cold leg temperature (Left loop)	189 °C	205 °C	311 °C	317 °C	220 °C	360 °C
Hot leg temperature (Right loop)	444 °C	262 °C	391 °C	304 °C	492 °C	356 °C
Cold leg temperature (Right loop)	226 °C	202 °C	315 °C	311 °C	219 °C	359 °C

4.3. Other indicators of SAMG entry

In addition to the CETs given by the thermocouples at the core exit and the coolant temperatures given by the RTDs in the primary circuits, indications of other available instruments can be adopted by the operators to determine the core damage states during accident management. For instance, the water level measured in the RPV is usually an indication of the core uncovery and core damage state (Esmaili et al., 2011; IAEA, 2015; Murata et al., 2016). From the MELCOR simulations, one can plot the variation of the RPV water level with increasing core exit temperature, as illustrated in Fig. 11. The height of the active fuel is more than 70% uncovered when the CET reaches 650 °C. It implies that 70% uncovering of the core can be employed as an indicator of core damage. Caution must be paid to the reliability of instrumentation when the collapsed RPV water level is taken as the reference information, as the lessons learnt from TMI-II accident.

During the severe accidents, the hydrogen masses produced by cladding oxidation are released into the containment through the breaks or PORVs. Therefore, the hydrogen sensors in the containment can be adopted to examine the occurrence of cladding oxidation in the core region (Hoffheins et al., 1997; Xiao et al., 2020). The PWR has hydrogen sensors installed at different elevations in the containment, referring to three compartments (lower compartment, SG1 + 3 compartment and dome in Fig. 2) (Gustavsson and Moller, 2002; Aronsson, 2012). Fig. 12 depicts the hydrogen concentrations of the three compartments and the total hydrogen mass generated from cladding oxidation of the LBLOCA scenario. The hydrogen starts production when the CET reaches 650 °C, and then releasing from the RCS into the containment. The hydrogen concentrations in the lower compartment, the

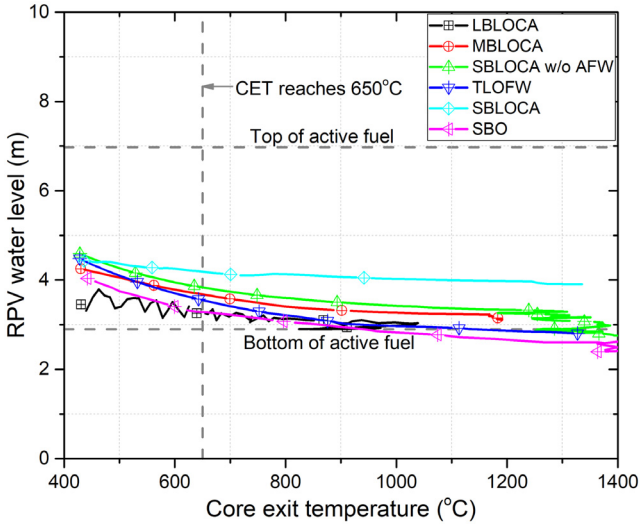

Fig. 11. RPV water level vs. Core exit temperature.

SG1 + 3 compartment and the dome reach the detectable range of hydrogen sensors (over 0.05% in general) at around 1590 s, 1450 s and 1320 s, separately. These times are all later than the time of CET reaching 650 °C, though just several minutes longer.

For high-pressure accident scenarios without breaks (e.g., the SBO in Fig. 13), the hydrogen generation from the core is initiated later than the CET reaches 650 °C. Then, the hydrogen is discharged into the containment through the PORVs. The hydrogen concentration in the containment is detected by the hydrogen sensors at

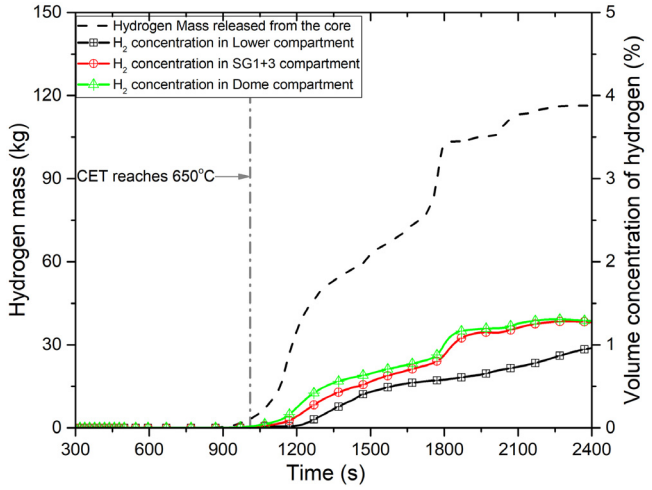

Fig. 12. Hydrogen concentrations in the containment (LBLOCA).

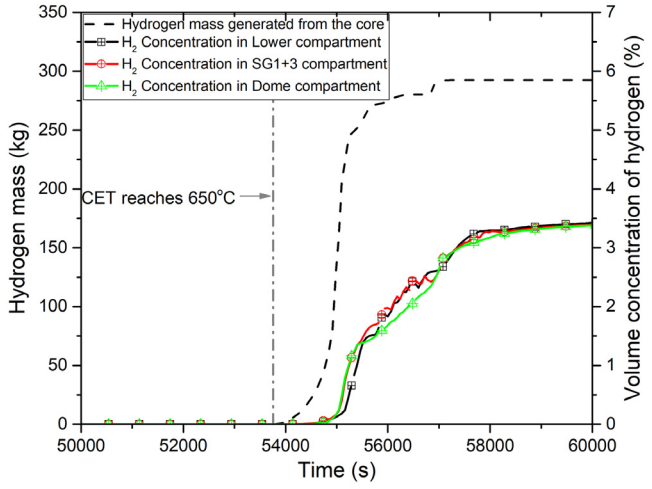

Fig. 13. Hydrogen concentrations in the containment (SBO).

around 55160 s after the scram, which is 23 min later than the timing of CET reaches 650 °C. Therefore, the indications of the hydrogen sensors in the containment could be too late for the entry to SAMGs. Besides, the passive hydrogen mitigation equipment (PARs) may impact the hydrogen distribution in the containment, and therefore affect the reliability of hydrogen sensors.

5. Conclusions

The effectiveness of the setpoint CET = 650 °C as the entry condition of SAMGs is investigated through the MELCOR simulations of six representative accident scenarios in a Swedish PWR plant. The interest of the study is to verify the successful transition from EOPs to SAMGs by the setpoint CET = 650 °C, and the outcomes of using different setpoints and temperature sensors, i.e., the thermocouples at the core exit and the RTDs in the hot and cold legs. In addition, the other parameters like the RPV water level and hydrogen concentrations in the containment are also considered as possible indicators of the entry condition in case all temperature sensors fail. Based on the MELCOR simulation results, the following remarks can be concluded:

- The core damage (PCT = 1204 °C) occurs after the average CET reaches 650 °C - the grace time between the occurrence of CET = 650 °C to the occurrence of PCT = 1204 °C is 12.8 min for the SBO sequence, and 10 ∼ 240 s for the other sequences. The average CET of 650 °C is therefore an effective setpoint as the entry condition of SAMGs.
- Other possible setpoints of the average CET, such as 480 °C or 760 °C, give rise to an inappropriate point in time which is either too early or too late to enter the SAMGs, if the PCT = 1204 °C is considered as the criterion of core damage.
- If the maximum CET from the MELCOR simulation is employed as the indicator of the entry condition of SAMGs, it may earn several minutes of grace time for SAM actions but may be too early to enter SAMGs (i.e. shortening the coverage of EOPs). On the other hand, opposite outcomes can be obtained if the minimum CET is adopted as the indicator of the entry condition of SAMGs.
- The temperatures (corresponding to the readings of the RTDs) in the hot and cold legs can be employed in the entry condition of SAMGs, but their setpoints should be lower than 650 °C in order to be effective indicators of core damage.
- The water level in the RPV can be considered in the entry condition of SAMGs, and for all investigated accident sequences 70% core uncovery seems a reasonable setpoint corresponding to the average CET = 650 °C, but caution should be paid to the reliability of instrumentation.
- The readings of the hydrogen sensors in the containment can also be considered in the entry condition of SAMGs if all other signals are lost. However, it should be kept in mind that the hydrogen concentrations cannot be detected until several minutes to 23 min (depending on accident sequences) after the average CET reaches 650 °C. Moreover, the hydrogen measurement may be impaired by the PARs.

In summary, the present study implies that the setpoint of the average CET = 650 °C is an effective entry condition of SAMGs (i.e. performing transition from EOPs to SAMGs at the right point in time), given the representative accident sequences as the main contributors to the core damage frequency (CDF) of the reactor chosen. The transition from EOPs to SAMGs may be too early if applying the minimum entry condition that at least one reading of thermocouple at the core exit exceeds 650 °C. The readings of the RTDs in the primary circuits, the water level in the RPV and the hydrogen sensors in the containment can also be applied as reference indications of core damage states in the accident management.

CRediT authorship contribution statement

Nan Zhao: Data curation, Writing – original draft, Visualization, Investigation, Software, Validation, Conceptualization, Methodology. **Weimin Ma:** Funding Acquisition, Methodology, Project administration, Resources, Software, Writing – review & editing. **Sevostian Bechta:** Supervision.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgements

The MELCOR code is developed and maintained by Sandia National Laboratory for U.S. NRC. This study was made possible by the support of Swedish Radiation Safety Authority (SSM). Mr. Patrick Isaksson is the project manager at SSM. The authors are also

grateful to the support of the scholarship awarded by the China Scholarship Council (CSC).

References

Aronsson, D., 2012. Detektorsystem och källtermer i kärnkraftverk (in English: Dectector system and source terms in nuclear power plants). CPD kurs Halmstad, Ringhals AB.

Billone, M., Yan, Y., Burtseva, T., Daum, R., 2008. "*Cladding Embrittlement During Postulated Loss-of-Coolant Accidents*", N.R.C. (NRC), Editor Nuclear Engineering Division, Argonne National. Laboratory.

Carney, F.S., Sherman, A., 1990. Three Mile Island Revisited". Ann. Nucl. Energy 17 (7), 393–398.

Esmaili, H., Helton, D., Marksberry, D., Sherry, R., Appignani, P., Dube, D., Tobin, M., Buell, R., Koonce, T., Schroeder, J., "*Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom*", *NUREG-1953* Idaho National Laboratory, U.S. Nuclear Regulatory Commission (2011).

Freixa, J., Martinez-Quiroga, V., Zerkak, O., Reventos,, 2015. F."Modelling guidelines for Core Exit Temperature simulations with system codes". Nucl. Eng. Des. 286, 116–129.

Gabor, J.R., Luxat, D.L., Godknecht, M.P., Henry, R.E., "*Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate HighLevel Actions and Their Effects.*", *CA: 2012. 1025295* EPRI, Palo Alto (2012).

Gustavsson, V., Moller, E., 2002. "*A Study on Hydrogen Deflagration for Selected Severe Accident Sequences in Ringhals 3*", *NKS-46/ISBN 87-7893-099-5*. Swed Power AB, Sweden.

Hermsmeyer, S., Iglesias, R., Herranz, L.E., "*Review of current Severe Accident Management (SAM) approaches for Nuclear Power Plants in Europe*", Joint Research Centre, Institute for Energy and Transport, Luxembourg: Publications Office of the European Union (2014).

Hill, L., Stringfellow, J., Young, D., Lyter, J., Pierson, S., "*Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents*", *NEI 14-01 [Revision 1]* Nuclear Energy Institute, Nuclear Regulatory Commission (2016).

Hoffheins, B.S., McKnight, T.E., Lauf, R.J., Smith, R.R., James, R.E., 1997. "Evaluation of a hydrogen sensor for nuclear reactor containment monitoring", CONF-970649–1 Oak Ridge National Laboratory. Oak Ridge National Lab, TN (United States).

Humphries, L.L., Cole, R.K., Louie, D.L., Figueroa, V.G., Young, M. F., "*MELCOR Computer Code Manuals Vol. 1: Primer and Users' Guide*", *NM 87185-0748*, H. Esmaili, Editor Sandia National Laboratories, Albuquerque (2015).

Humphries, L.L., Cole, R.K., Louie, D.L., Figueroa, V.G., Young, M. F., "*MELCOR Computer Code Manuals Vol. 2: Reference Manual*", *NM 87185-0748*, H. Esmaili, Editor Sandia National Laboratories, Albuquerque (2015).

IAEA, 2009. Severe Accident Management Programmes for Nuclear Power Plants - Safety Guide. No. NS-G-2.15 International Atomic Energy Agency.

IAEA, "*Accident Monitoring Systems for Nuclear Power Plants*", International Atomic Energy Egency, Vienna, Austria. (2015).

Kim, J.H., Lee, M.H., Choi, B.K., Jeong, Y.H., 2005. Embrittlement behavior of zircaloy-4 cladding during oxidation and water quench". Nucl. Eng. Des. 235 (2005), 67–75.

Kim, M.C.,"Insights on Accident Information and System Operations during Fukushima Events". *Science and Technology of Nuclear Installations,* Volume 2014, Article ID 123240, 12 pages (2014).

Lutz, R.J., "*Westinghouse Owners group Core Damage Assessment Guidance*", R. WCAP-14696-A, Editor (1996).

Murata, A., Isoda, K., Ikeuchi, T., Matsui, T., Shiraishi, F., Oba,, 2016. M."Classification method of severe accident condition for the development of severe accident instrumentation and monitoring system in nuclear power plant". J. Nucl. Sci. Technol. 53 (6), 870–877.

NEA, "*Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor*", *NEA/CSNI/R(2010)9, JT03293388* Organisation for Economic Co-operation and Development (2010).

Ostmeyer, R.M., 1985. An Approach to Treating Radionuclide Decay Heating for Use in the MELCOR Code System. U.S.NRC, Sandia National Laboratories.

Rempe, J.L., Knudson, D.L., Lutz, R.J., "*Scoping Study Investigating PWR Instrumentation during a Severe Accident Scenario*", *INL/EXT-15-35940* Idaho National Laboratory (2016).

Park, R.J., Hong, S.W., 2011. Effect of SAMG entry condition on operator action time for severe accident mitigation. Nucl. Eng. Des. 241 (2011), 1807–1812.

Sehgal, B.R., 2012. Nuclear Safety in Light Water Reactors: Severe Accident Phenomenology. Elsevier.

Sehgal, B.R., 2012. Light Water Reactor Safety - Chapter7: safety principles. Academic Press, Nyköping.

Suh, N., Cho, Y., Park, J., Ryu,Y., "Assessment of Core Damage Status using Core Exit Thermocouple". *Transactions of the Korean Nuclear Society Spring Meeting.* Chuncheon, South Korea, May 25-26 2006 (2006).

Travis, R., Taylor, J., Fresco, T., Chung, J., "*Generic Risk Insights for Westinghouse and Combustion Engineering Pressurized Water Reactors*", *NUREG/CR-5637* BrooKhaven National I,aboratory, NY (1990).

U.S.NRC, "Severe Accident Risks : An Assessment of Five U.S. Commercial Nuclear Power Plants", NUREG-1150 (1990).

Xiao, Z., Zhang, L.M., Lia, H., Xua, D., Jin, C.G., Peng,, 2020. H.Q."Design study of hydrogen monitoring equipment for severe accident in containment of nuclear power plant". Energy Rep. 6 (2020), 205–214.

Yun, J., Kim, T., Kim,, 2015. J."Verification of SAMG entry condition for APR1400". Ann. Nucl. Energy 75 (2015), 404–412.

Zhao, N., Chen, Y.L., Ma, W.M., Bechta, S., Isaksson,, 2021. P."A nodal sensitivity study of MELCOR simulation for severe accidents in a pressurized water reactor". Ann. Nucl. Energy 160, (1) 108373.