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ASSESSMENT OF THE MOLTEN POOL INITIAL STATE DURING DEGRADATION OF THE REACTOR CORE WITH HORIZONTAL FUEL ASSEMBLIES ARRANGEMENT

The paper examines the first approximation in the assessment of the corium initial state in the lower plenum during the in-vessel phase late stage of severe accident of a promising small modular reactor of the ESS-SMART project [1]. The features of the presented analysis are related to both the initial supercritical state of the coolant and the horizontal layout of the core in the form of a system of parallel fuel assemblies with a seven-level coolant inlet system [2]. Existing industry computational system tools are limited in representing the original geometric configuration of the reactor flow path, and in particular the reactor core. Moreover, significant difficulties still arise taking into account the pressure decreasing from a supercritical state to a two-phase state, which seriously limits the performance of estimates using the analytical tool only. In this work, the main principles of sequential equivalent approximation are considered, which allow for a notable simplification of the analysis by isolating stages where only one phenomenon is dominant. For example, the evaluation of decompression parameters and impact loads is the first stage, while the immediate reactor core degradation and the formation of the core molten materials pool is the second one. The third stage is a detailed assessment of the reactor vessel degradation. In this work, attention is focused on the second stage. For this, a fast run model was prepared in the MELCOR 1.8.6 integral code with an equivalent approximation in mass fractions of the "components" in the reactor core flow part. In addition, the conceptual design of the lower plenum of the SCW-SMR reactor is similar to that of typical PWR reactors. The results of the design scenarios made it possible to form a general picture of the reactor core degradation progression at the early phase of a severe accident for the selected initial conditions, as well as to evaluate the characteristics of the accumulated melt in the lower plenum at the late phase of in-vessel severe accident stage. The obtained results are an input set for further evaluation of reactor vessel degradation.

Key words: severe accident, supercritical parameters, core degradation, corium, horizontal fuel assembly, ECC-SMART.

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ОЦІНКА ПОЧАТКОВОГО СТАНУ БАСЕЙНУ РОЗПЛАВУ ПРИ ДЕГРАДАЦІЇ АКТИВНОЇ З ГОРИЗОНТАЛЬНИМ РОЗМІЩЕННЯМ ТВЗ

В роботі розглянуто перше наближення в оцінці початкового стану коріуму в нижній камері змішування під час пізньої стадії важкої аварії перспективного малого модульного реактора проекту ECC-SMART [1]. Особливості наведеного аналізу пов'язані як з початковим надкритичним станом теплоносія, так і з горизонтальною компоновкою активної зони у вигляді системи паралельних тепловиділяючих збірок із семирівневою системою заходу теплоносія [2]. Існуючі галузеві розрахункові системні інструменти є обмеженими в представлені оригінальної геометричної конфігурації проточної частини реактора, та зокрема активної зони. До того ж досі виникають суттєві складності із врахуванням декомпресії із надкритичного стану в двофазний, що суттєво обмежує виконання оцінок застосовуючи один аналітичний інструментарій. В даній роботі розглянуті основні принципи послідовного еквівалентного наближення, що дозволяє суттєво спростити аналіз за рахунок виокремлення стадій, в яких домінантним є лише один феномен. Наприклад оцінка параметрів декомпресії і ударних навантажень є першим етапом, безпосередня деградація активної зони та формування басейну розплаву – другий етап. Третій етап – це детальна оцінка деградації корпусу реактора. В даній роботі увага акцентується саме на другому етапі. Для цього була підготовлена швидка розрахункова модель в пакеті MELCOR 1.8.6 з еквівалентним наближенням по масовим долям «компонентів» в проточній частині. Крім того, концептуальна конструкція нижньої камери змішування реактора SCW-SMR подібна до типової на реакторах PWR. Результати розрахункових сценаріїв дозволили сформувати загальну картину перебігу деградації активної зони реактора на ранній фазі важкої аварії для обраних початкових умов, а також оцінити характеристики накопиченого розплаву в нижній камері змішування на пізній фазі внутрішньокорпусної стадії. Отримані результати є вхідним набором для подальшої оцінки деградації корпусу реактора.

Ключові слова: важка аварія, надкритичні параметри, деградація активної зони, коріум, горизонтальні ТВЗ, ECC-SMART.

Introduction

The applications of a coolant with supercritical parameters have significant advantages over other concepts of reactor units of the IV generation [1] – [3]. Firstly, a significant decrease of coolant flow through reactor due to the use of a pseudo-phase transition, in which the coolant temperature practically does not change during the supply of thermal energy [4]. From a design point of view, this means a weighty reduction in the metal capacity of the reactor coolant circulation system primarily due to pumps and heat exchange equipment. Secondly, a significant increase in the average temperature of the coolant (~400 °C – 450 °C) leads to a more efficient production of electricity [5].

The specificity of the supercritical (SC) state of the coolant, is immensely decreasing of moderation properties of the medium, that requires a partial separation of the flow in the fuel assemblies (FA), which actually, the formation of a parallel channels system [6]. To improve the conditions for neutron leakage beyond the conditional limits of FAs, it is advisable to use so-called fuel assembly's boxes, which also contributes to better neutron thermalization. Despite the accumulated experience of PWR and BWR reactors and the inherent pragmatism of nuclear energy industry, the vertical arrangement of fuel elements is not the best option. This is primarily because of the heat transfer deterioration phenomena [7] (experimentally confirmed during the upward and downward of the coolant flow) and experimentally confirmed instability of parallel channels system of the super critical parameters during the incremental increasing of thermal power [8]. That is why the ESS-SMART [2] reactor concept proposes a horizontal arrangement of FAs with a rather significant complication of the reactor flow part.

From the perspective of justifying the neutronic reliability of such a solution, no significant problems arise [9] and by optimizing the FA cross section it is possible to obtain compromise parameters for initial enrichment, energy release uniformity and fuel burnup depth [10]. The system of passages in the reactor core, as well as partial mixing of the coolant, helps to align the flow and, according to the designers, helps to reduce the progression factors of critical thermophysical phenomena.

Such structural solutions lead to a significant complication of the reactor flow part, which becomes critical in terms of dynamic loads. For example, for VVER-1000 reactors, the initial decompression dynamics at maximum design basis accident [11] does not cause to the FA and in-vessel internals damage. Similar assessments (reactor cooling system cold pipe guillotine rupture) for ECC-SMART [12] show the presence of critical loads in the first 0.1 second of decompression on FA boxes, which is likely to lead to the reactor core geometry failure (the load on the in-vessel internal elements exceeds 1000 MPa [12]). From the point of view of the water-water reactors development, this is a non-typical situation where the LB LOCA for the next generation is more crucial than for the previous one.

Before distinguishing the critical components of the concept, it is logical to assess the integrity margin of the reactor vessel from the point of view of thermal degradation during a severe accident [SA]. Unusual for this type of evaluations are two key factors: the supercritical initial state of the coolant, and the horizontal arrangement of fuel assemblies. The difficulty here lies in the application of specialized codes such as MELCOR, ATHLET-CD [13] or RELAP5/SCDAP. Each of them has its own characteristics, which are either related to the possibility of predicting decompression from a supercritical state, or have geometric limitations (the vertical arrangement of the FA only). Given the existing variety of SA codes, the features of the spatial representation of the in-vessel internals and reactor core with a horizontal FA arrangement are taken into account only on the MAAP based code [14], developed for Canadian heavy water reactors CANDU.

Based on the description from Sandia National Laboratories [15], certain capabilities for representing horizontal fuel channels are also provided by the MELCOR code family. However, an analysis of user functions and the documentation of more modern code versions has shown that such capabilities are not implemented.

Thus, the assessment of the in-vessel phase severe accident for supercritical water reactor requires unconventional approaches in order to use existing tools and the experience of their application.

The aim and objectives of the study

The main objective of this work is to evaluate the state parameters of corium in the reactor lower plenum (thermal and material states) of a prospective IV gen supercritical coolant small modular reactor. Considering the features of the ECC-SMART concept (SC, horizontal arrangement of fuel assemblies), applying well-known specialized tools like MELCOR code in the literal sense is incorrect due to geometric and physical limitations. Therefore, some simplifying principles have been formulated in the work, which allow for preliminary assessments to be made. In general, the main objective is to assess the integrity of the reactor vessel depending on the external cooling conditions, which will be addressed in further studies.

1. The principle of sequential equivalent approximation

The assessment of the reactor core degradation parameters during SA, which is performed in a horizontal approximation, is quite atypical, as most nuclear reactor core designs have vertical arrangement. Since specialized codes have a rather rigid abstraction of processes both in terms of geometric configuration and empirical information, their application is quite limited, for example, compared to universal computational fluid dynamics codes.

Also, the presence of an initial supercritical state is a feature, which is distributed in a certain way throughout the volume of the core, since the inlet temperature is subcritical (280 °C), and outlet is supercritical (~500 °C). The reference pressure in this case is 25 MPa. The analysis of the phenomenology of direct impact on the reactor core degradation under supercritical parameters and the horizontal arrangement of fuel rods showed that there are no fundamental differences compared to a subcritical reactor. This allows to some extent the simplification of the analysis by introducing sequential steps, the results of each of which serve as the input set (initial state for the next one).

Thus, the basic scheme of the analysis (the main goal of which is to simplify and harmonize approaches with different types of codes) is shown in Fig. 1.

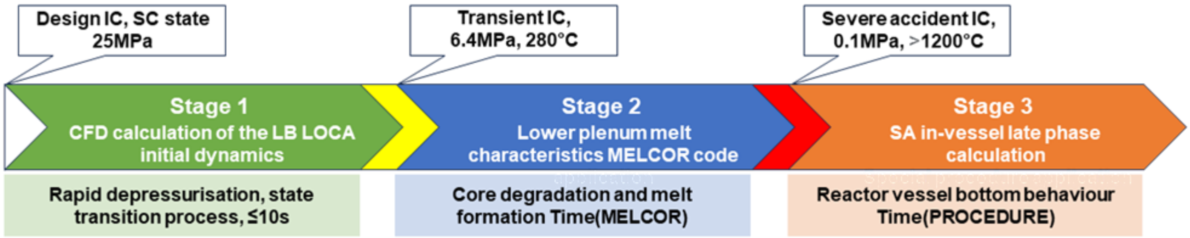


Fig. 1 – Principal scheme for analyzing the SA in-vessel phase
for the reactor with supercritical coolant parameters

In fact, **Stage 1** considers decompression [12] and evaluates impact loads on the structural elements of reactor core, which allows for the formation of an initial state for **Stage 2**, which is already subcritical and can be applied in MELCOR. Moreover, at **Stage 1**, it is possible to determine whether the design geometry is maintained, meaning if the critical failures and potential changes in the flow part configuration occur during the initial dynamics. All evaluations are conducted in a three-dimensional setting, taking into account the features of the flow part [11, 12].

The second stage (**Stage 2**) involves the assessment of the parameters (material and thermal-physical) of the molten pool, that is the input set for **Stage 3**, based on which the final conclusions regarding the "success" of the concept from the perspective of the flow part design will be formulated.

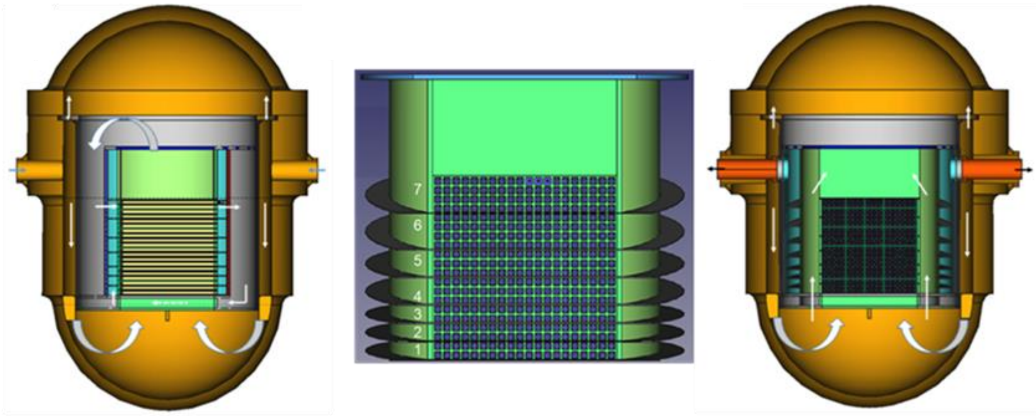
Let's consider in more detail the features of calculating the main parameters of the molten pool in a prospective reactor. It is worth noting that the total operational time before potential reactor vessel failure consists of the characteristic duration at each stage. From the perspective of the integral duration of the transient mode, **Stage 1** can be disregarded, as the characteristic time of the initial dynamics does not exceed 10 seconds, and the stabilization of the parameters observed in the flow section is determined by the critical flow and is close to the saturation parameters at the inlet temperature (if the break of the cold part of the circulation loop is considered) [16].

2. Equivalent horizontal core model

The concept of a promising small modular reactor [2] was proposed by Professors Schulenberg and Otic from the Karlsruhe Institute of Technology. It is based on the concept of a high-efficiency large-power reactor [17]. Essentially, this is a low-power water-water reactor with an original (non-conventional core design) in the form of a system of parallel horizontal heat generation channels with boxed fuel assemblies, which are arranged horizontally and grouped by levels. Thus, seven riser passages are formed, between which the coolant is mixed, primarily to ensure uniform temperature at the entrance of each fuel assembly at the next level (Fig. 2). It is believed that such a configuration of the core allows avoiding two important phenomena: deteriorated heat transfer from the surface of the fuel assembly and unstable operation of the parallel channel system during changes in thermal power. One of the defining features is that the lower plenum (LP) of the SCW-SMR concept is similar to PWR-type reactors with one fundamental difference. There are no support structures there in the ECC-SMART concept, meaning that the specific mass of steel per geometric unit volume of lower plenum is significantly lower, which, in terms of thermal capacity (and thus the action front on the vessel), determines the time reserve component.

The equivalent representation by the MELCOR code of the real internal reactor core modeling area is based on the concept of an axisymmetric cylinder with individual characteristics and balance ratios and predicate conditions for each elementary cell. Conditional sampling into elementary control volumes (cells) occurs by dividing into concentric radial rings and axial levels in height. Thus, from the point of view of the development of a severe accident in the postulated state of the drained core, the assessment is based on an equivalent approximation of the mass distribution.

The radial division is chosen to be the maximum possible (9 conditional rings), and to consider the design of the fuel assembly boxes, a BWR-type reactor option is used. Therefore, the internal volume of the SCW-SMR reactor vessel, including the upper and lower parts, reactor core and the peripheral area of the downcomer, is divided into 30 axial segments in height and 9 rings in the radial direction. The first 8 rings are designated for the reactor core region. The core internal side panels, as well as other external steel structures and elements, are defined by boundary heat structures. The nodalization scheme of the volumetric space of the reactor core, included in the modeling area, is presented in (Fig. 3).



a *b* *c*

Fig. 2 – General view of the SCW-SMR concept:
a – the axis of section along the nozzles of the coolant inlet of the subcritical temperature;
b – the reactor core; *c* – along the outlet nozzles

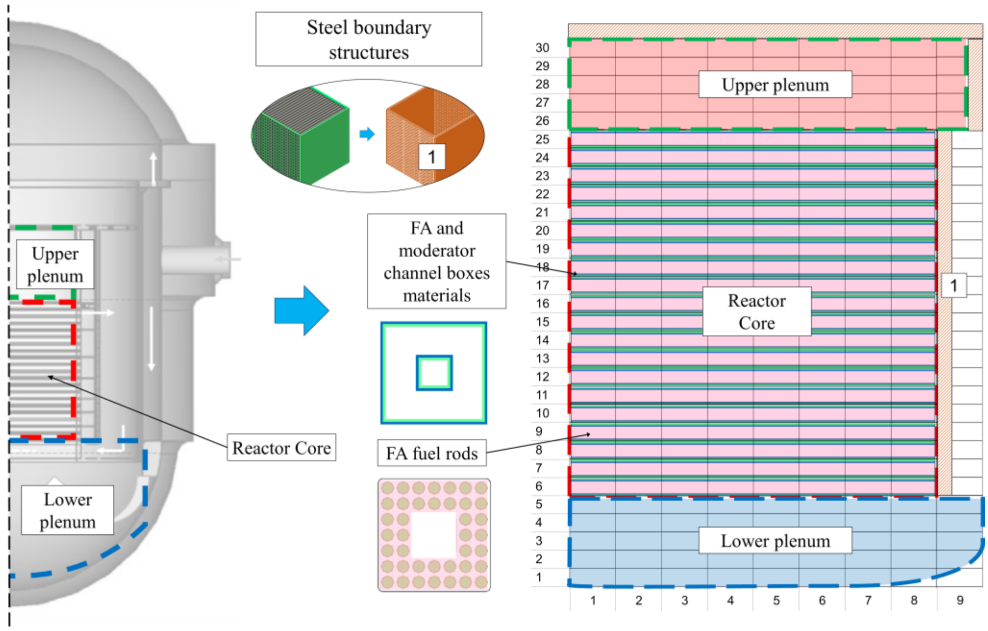


Fig. 3 – SCW SMR core MELCOR nodalization scheme

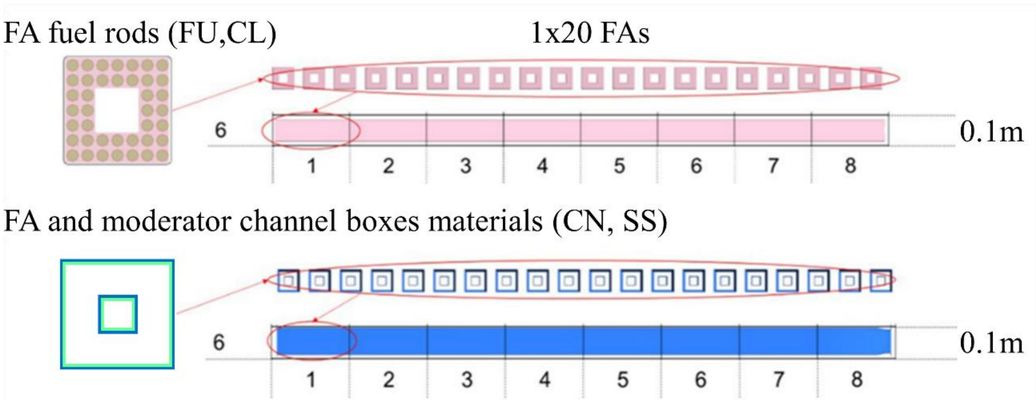


Fig. 4 – General approach to mass distribution of the main structural components

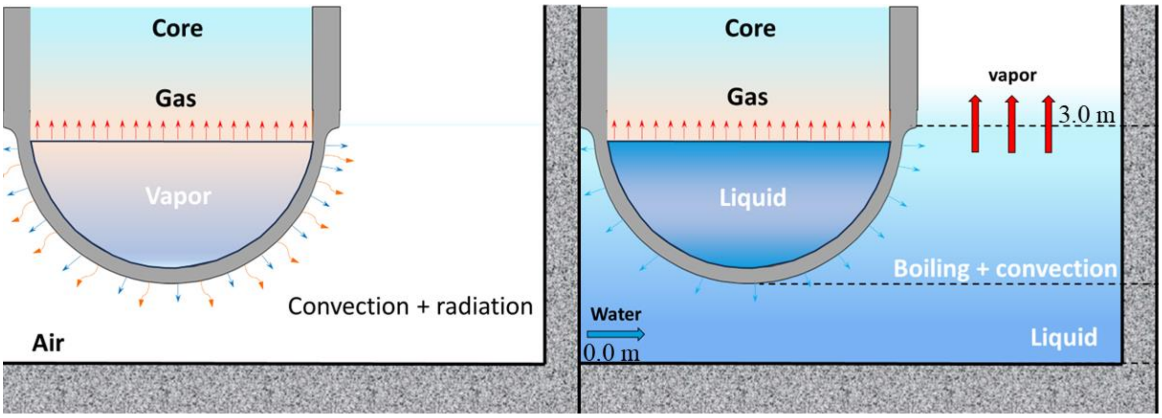


Fig. 5 – Schematization of initial state from the point of view of LP and reactor concrete cavity (gas-gas, liquid-liquid)

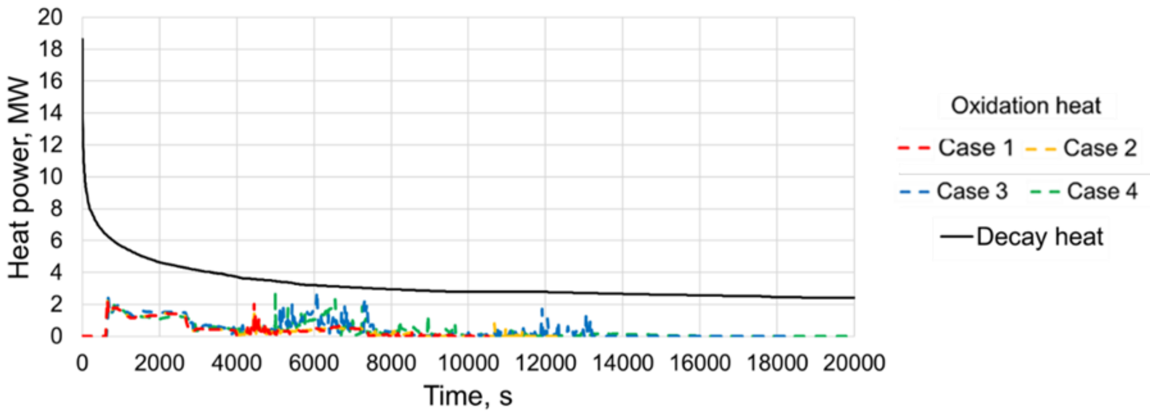


Fig. 6 – Reactor core heat energy sources

The volume of the LP is represented by the first five axial segments. The latter consists of a "fictitious lower support grid" of the reactor core in the part of the large square opening for the coolant to flow into the annular space. The reactor core, made up of a system of parallel horizontal fuel assemblies, is evenly divided into 20 levels (from the 6th to the 25th). The height of one level is taken as 0.1 m for geometric correspondence to one fuel assembly. The volume of the upper plenum is distributed over the last 5 levels. From the center to the periphery, the core is divided in such a way that the mass characteristics of the structural elements of the fuel, the materials of the claddings, and the FA boxes are distributed evenly in the radial rings (Fig. 4).

To take into consideration the retaining capacity of the FA boxes and to limit the movement of the melt in the axial direction, their mass is considered by the structural element of the "SS" type. Also, from an engineering perspective, the space of the square opening on the 5th axial level contains the mass of a "fictitious support grid". This prevents the premature movement of molten materials from the upper axial levels to the lower plenum, although in general, this approach is one of the parametric factors that determines the "early" contact of the melt with the reactor vessel.

The heat release function is initially assumed to be uniform throughout the core, which considers both the core decay heat and energy of oxidation reactions.

3. Parametric analysis of the molten pool formation in the lower plenum

As mentioned above, the initial event of the accident is postulated as the full diameter rupture of the circulation circuit pipeline from the inlet nozzle side, which causes rapid decompression and the emergence of impact loads on the in-vessel internals, including the elements of the reactor core. The stresses that occur in this case [12] can lead to a partial loss of load-bearing capacity of the FA boxes.

The initial thermodynamic state of coolant in the reactor vessel for the analysis of the early internal phase of a severe accident is taken from the results obtained in [12], which are presented in Table 1.

Table 1 – Initial thermodynamic state of coolant

Parameter / Case #	1	2	3	4
Initial decay heat power	18.6 MW			
Reactor coolant pressure	6.4 MPa			
Coolant temperature	280 °C			
Initial LP coolant level	–	–	2.8 m	2.8 m
In-vessel melt retention system water temperature	–	90 °C	–	90 °C
Reactor cavity pressure	0.1 MPa			
Reactor cavity water level	–	3 m	–	3 m

For the purpose of estimating the time reserve for the formation of quasi-stationary parameters of the melt pool, the influence of safety systems (core emergency cooling system) is not considered. The nominal thermal power is 290 MW, and the calculation is performed taking into account a 10 second offset relative to the start of the accident (the offset is determined by **Stage 1**).

The assessment of the melt state, which forms in the reactor core and moves into the LP, is evaluated for the following initial conditions:

Case#1: the LP is "dried out", without flooding the reactor concrete cavity.

Case#2: the LP is "dried out", the reactor concrete cavity is flooded.

Case#3: the LP is filled up, without flooding the reactor concrete cavity.

Case#4: the LP is filled up, the reactor concrete cavity is flooded.

The term "dried out" refers to a state where the amount of coolant in the LP is minimal, and the flow part is in steam. An area is considered filled if a larger portion of the coolant is in the liquid phase on the saturation line within the reactor vessel at the moment of initial dynamics (short-term stabilization of state parameters because of critical flow). Such modes are chosen due to the design feature of the SCW-SMR concept, which defines a complicated hydraulic scheme for the flow part of the reactor core. Taking into account the capabilities of MELCOR 1.8.6, this makes the dynamics of degradation and the formation of the molten pool somewhat uncertain. From the perspective of external boundary conditions, it is assumed that the reactor concrete cavity is flooded to a level of 3 meters at the initial moment of the calculation time (**Stage 2**) and is subsequently maintained (Fig. 5).

Additionally, a parametric analysis of the impact of "additional" steel structural elements in the LP (similar to support buckets in PWR) is performed for the calculated options Case#1 and Case#4. Case#1 and Case#4 are selected as those corresponding to the minimum and maximum time during which the transition to quasi-stabilized parameters of the melt pool occurs, with the core being almost completely degraded. So, additional cases are considered where structural elements with an integral mass of 15 and 40 tons are present in the LP, with the distribution being uniform throughout the entire volume. It is also should be noted that materials related to control rods are not taken into account in the reactor core, as their type and design are still under development.

The simultaneous rapid boiling and irreversible loss of coolant in the break leads to a fast drop in the reactor core level and its early drying (all this occurs still at **Stage 1**). The relatively high level of residual heat at the initial stage of the accident process determines the heating and subsequent overheating of the fuel. Against the backdrop of increasing temperatures of the fuel rod cladding, the exothermic oxidation reaction intensifies, accelerating the degradation processes of the core and the onset of its melting. From this point, the set of calculation scenarios can be divided into two subgroups depending on the presence of a liquid phase in the reactor. For the calculation scenarios with the flooded LP, the initial sequence of degradation of the reactor core is similar to the first one, but the presence of water in the liquid phase significantly affects the quantitative and temporal characteristics of the late phase. The contribution to the total energy release of exothermic oxidation reactions increases almost doubles (Fig. 6).

The proportion of reacted zirconium and the proportion of the liquid phase of this metal in the molten pool are increasing. This leads to the generation of a larger amount of hydrogen in the volume of the degraded core and an enthalpy-averaged increase in the temperature of the gas environment. Fig. 7 presents the main results for 4 basic cases, from which it is evident that the presence of a liquid phase of the coolant in the LP, as well as the intensification of heat exchange from the outer surface of the reactor vessel by flooding the concrete cavity, significantly increases the duration of reaching quasi-stabilized parameters of the molten pool. The presented simplified picture of the degradation of the core (the process of melt formation and its movement in the lower core) indicates that in all cases, the formation of a stationary stratification pattern is not observed until the conditions for the failure of the reactor vessel bottom under mechanical load are met (simple one-dimensional estimates). It is worth noting the following here. In general, the methods used in the MELCOR 1.8.6 code to assess reactor vessel failure are somewhat simplified compared to spatial assessments that take into account thermal creep and direct melting of the vessel steel. The "quality" of the correlations for external heat exchange during the boiling of the flooded cavity also plays an important role. Thus, the straight failure of the reactor vessel is not the focus of this work.

The parameters of the quasi-stationary thermal state of the molten materials pool in the lower part of reactor lower plenum, which are the initial conditions for **Stage 3**, are presented in Table 2.

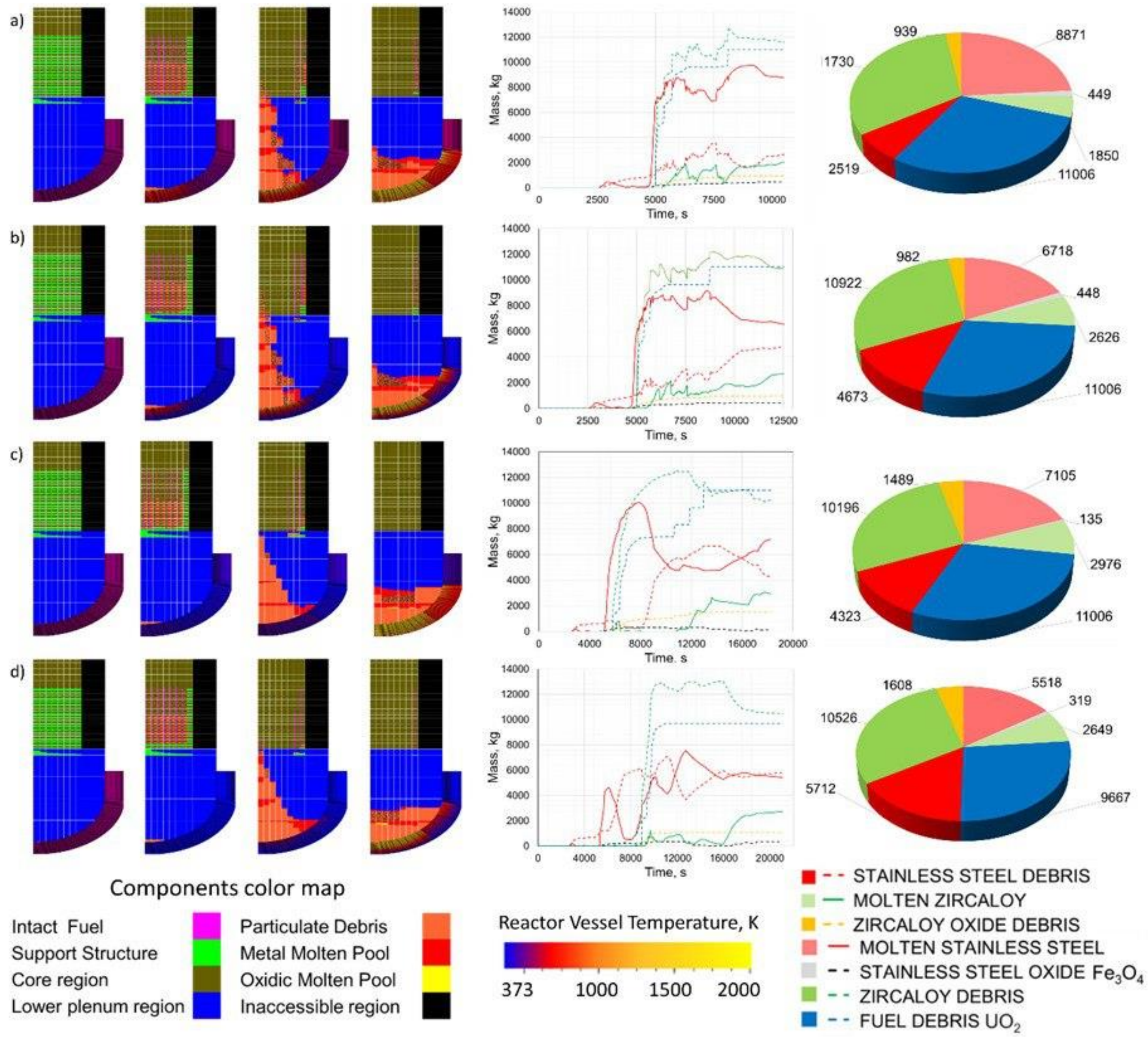


Fig. 7 – Main results of the molten materials pool formation: *a* – Case#1; *b* – Case#2; *c* – Case#3; *d* – Case#4

Table 2 – The parameters of the quasi-stationary thermal state of the molten materials pool in the lower part of reactor lower plenum

Parameter	Case#1	Case#2	Case#3	Case#4
Reference time (for Stage 3 input data)	10000 s	12000 s	18000 s	20000 s
Total material mass in lower plenum	37363 kg	37375 kg	37230 kg	35459 kg
Oxidic melt fraction temperature	2400 K	2400 K	2100 K	2500 K
Metal melt fraction temperature	2100 K	2250 K	2100 K	2140 K
Ex-vessel heat transfer coefficient	10 W/(m^2·K)	~20 kW/(m^2·K)	10 (m^2·K)	~20 kW/(m^2·K)
Reacted Zirconium fraction	6.5 %	6.8 %	10.4 %	11.3 %
Degraded core gas temperature	1200 K	1200 K	1700 K	1700 K
Reactor cavity temperature	726 K	373 K	738 K	373 K
Reactor vessel failure time[1]	10523 s	12527 s	18258 s	21083 s
Reactor vessel failure delay	–	+0.5 h	+2.1 h	+3.0 h
[1)] by mechanical loading (1D mechanical model calculates the thermal and plastic strain)				

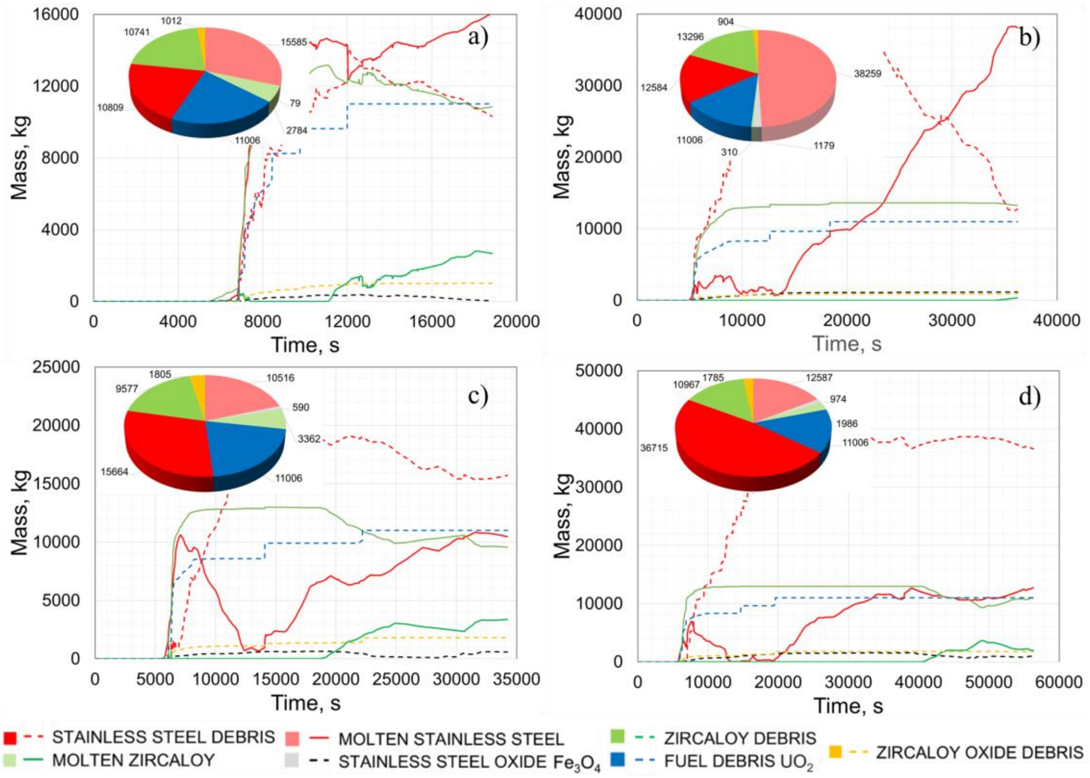


Fig. 8 – Main results of the molten materials pool formation during parametric changes in the homogenized mass of steel in the LP: *a*, *b* – Case#1; *c*, *d* – Case#4

Fig. 8 shows the results of the parametric assessment of the corium state with an increase in the average portion of structural steel elements in the LP, whose integral mass is 15 and 40 tons, for bounding cases #1 and #4 respectively. It is expected that the average time for the formation of the corium pool (also the preliminary estimate of the containment failure time) increases due to the need to dissipate some energy for heating and melting the additional amount of steel. Generally, in the case of a "dry" LP, the decay heat is mostly used for melting steel, while for a "wet" configuration, a significant postponement of the reactor failure time is observed.

The obtained results allow to formulate a series of parametric initial states for a detailed examination of the late phase of a severe accident, with an emphasis on analyzing the features of reactor vessel degradation depending on external cooling conditions. (**Stage 3**). Based on the combination of all conditional assessment stages, it will be possible in the future to draw conclusions regarding the applicability of one or another severe accident management concept (the use of passive external flooding systems or the use of core catchers).

Conclusions

The analysis of severe accidents for prospective SCW-SMR reactors is in general similar to that of subcritical PWR/BWR reactors. At the same time, it is important to understand some features related to the influence of supercritical coolant state parameters on the reactor core degradation progression. The greatest difficulties in this case arise precisely from the perspective of the initial decompression dynamics, taking into account the transition to a two-phase (water-steam) state. Not all "classical" severe accident codes are capable of taking into account the initial supercritical state of the coolant, as well as the geometric arrangement of the fuel assembly (horizontal placement). All this requires certain approaches that allow for the consideration of the specifics of the prospective reactor core using the available specialized tools.

This work examines exact such method, which is based on the concept of sequential equivalent steps characterized by the most dominant processes occurring in transient. In general, three stages are proposed, the first (**Stage 1**) that is independent of the others. In fact, **Stage 1** only considers decompression and evaluates the impact loads on the structural elements of the reactor core, which allows for the formation of the initial condition for **Stage 2**, that is already subcritical and on which this work is focused. The second stage (**Stage 2**) involves the assessment of the parameters (material and thermal-physical) of the molten materials pool, that serves as the input set for **Stage 3**, based on which the final conclusions regarding the "success" of the concept in terms of the flow part design will be formulated.

A fast run model of the SCW-SMR ECC-SMART reactor core was prepared. The results of the computational scenarios allowed to take a general picture of the degradation of the reactor core an early stage under various initial conditions, as well as the assessment of the characteristics of the accumulated melt in the lower plenum at a late stage. Evaluation of the main parameters of the molten pool were carried out in terms of both material and thermal-physical states, including for different amounts of steel (for example, in the form of support elements) in the lower plenum of the reactor, which turned out to be one of the important factors affecting the further degradation of the reactor vessel. Significant limits of the minimum operational time reserve have been obtained, ranging from 3 to 20 hours depending on the conditions of external cooling and the initial state of the LP. In addition, the more structural elements there are in the lower part of the reactor, the better it is for the initial stage of the internal phase, which must be taken into account at the pre-conceptual design.

This work is a preliminary stage for the parametric assessment of reactor vessel failure due to wall melting and steel property degradation, depending on external cooling conditions and the initial configuration of the corium in the lower plenum. This will provide the first clarifying data for the stage of direct development of the project of a small modular reactor with supercritical parameters, in terms of resistance to hypothetical severe accidents.

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