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Original Article

Heat pipe type double-containment vessel as passive safety system for small modular reactors

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ABSTRACT

This paper proposes a new concept of a passive safety system for small modular reactors as a heat pipe type double-containment vessel. The proposed system configures an additional vessel outside the existing one and functions as a heat pipe that utilizes phase change heat transfer to reduce the requirement of a large pool design. The double-containment vessel features a design consisting solely of the evaporation and condensation sections for heat pipe function with compactness. The ultimate heat sink only serves as a condenser part for the double-containment vessel, therefore significantly reducing the volume of stored fluid. The study numerically evaluates several postulated accidental cases to compare the performance of the double-containment vessel using the thermal hydraulic system code, MARS-KS. Even for serious conditions such as multiple failure accidents without safety systems operation, the double-containment can secure extra golden time to core damage by more than 2,090 sec compared to the original design. This study highlights the feasibility and potential of the heat pipe-based double-containment vessel as a passive safety system, enhancing the safety and reliability of small modular reactors. This kind of design can provide the advantages of safety and economics in the construction of small modular reactors.

1. Introduction

Small Modular Reactors (SMRs), which have an electrical output of less than 300 MWe, are designed with high safety features compared to traditional large nuclear power plants (NPPs) [1]. The safety of SMRs is a critical consideration in their design and deployment. Passive principles are usually adopted for SMR safety-related designs to secure and enhance safety and reliability. Passive safety technologies operate by the force of nature, such as gravity, and have the advantage of inherently eliminating mechanical defects. In particular, the natural circulation cooling system in integrated pressurized water reactors (iPWRs), like NuScale [2] and CAREM [3], facilitates the elimination of the large piping system and large break-loss of coolant accident (LB-LOCA) scenarios. However, SMR designs tend to require excessive auxiliary systems to preserve high reactor safety; this reflects a conservative design from a safety perspective. While diversity and redundancy of the passive system of SMR ensure high safety, these may lead to degrading the economic feasibility of small modular NPPs due to construction costs [4]. Some of the codes of account can be determined as reduction factors to overcome the loss of economics of scale [5]. This trend may contribute to delaying the international development of SMRs and hindering the growth of the nuclear industry and technology as low-carbon energy sources.

One possible option to be considered to improve the design compactness and simplification is incorporating a phase-change heat transfer mechanism. From this perspective, the heat pipe and thermosyphon are noteworthy, using the two-phase natural circulation principle [6], as illustrated in Fig. 1. For example, the heat pipe is a highly efficient heat transfer device that operates on the principles of evaporation and condensation. It consists of a sealed envelope containing a working fluid. At the heat source (evaporator), the fluid absorbs heat and becomes vapor. This vapor then travels to the cooler end of the pipe (condenser), where it releases the heat and condenses back into a liquid. The condensed fluid returns to the evaporator, often aided by gravity or a wick structure, completing the cycle. This process allows for rapid and efficient heat transfer over relatively long distances with minimal temperature difference, making heat pipes particularly advantageous in compact reactor designs like SMRs [7]. Currently, the heat pipe and thermosyphon are being proposed for nuclear power applications, enhancing design compactness with the strength of their passive operation, high thermal efficiency, and heat transfer performance. Based on

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Nomenclature

Q	power [W]
C_K	fitting constant [−]
l	length [m]
A	hydraulic area [m^2]
h	enthalpy [J/kg]
D	hydraulic diameter [m]
g	gravitational acceleration [m/s^2]
K	fitting constant [−]
Bo	Bond number [−]

Greek symbols

ρ	density [kg/m^3]
σ	surface tension [N/m]

Subscripts

v	vapor
l	liquid
lv	vaporization

Abbreviations

BDBA	Beyond design basis accident
CCFL	Countercurrent flow limitation
CFD	Computational fluid dynamic
CNV	Containment vessel
CRDM	Control rod driving mechanism
CWL	Collapsed water level
DCA	Design certification application
D-CNV	Double-containment vessel
DHRS	Decay heat removal system
ECCS	Emergency core cooling system
iPWR	Integrated pressurized water reactor
LB-LOCA	Large break-loss of coolant accident
LOCA	Loss of coolant accident
LWR	Light water-cooled reactor
NPP	Nuclear power plant
MARS-KS	Multi-dimensional analysis of reactor safety
MASLWR	Multi-application small light water reactor
MPHP	Multi-pod heat pipe
MSIV	Main steam isolation valve
PCCS	Passive containment cooling system
PCT	Peak cladding temperature
PINCs	Passive in-core cooling system
PWR	Pressurized water reactor
PZR	Pressurizer
RCS	Reactor cooling system
RPV	Reactor pressurized vessel
RRV	Reactor recirculation valve
RSV	Reactor safety valve
RVV	Reactor vent valve
SB-LOCA	Small break-loss of coolant accident
SBO	Stationary black-out
SCRAM	Safety control rod axe man
SG	Steam generator
SMART	System-integrated modular advanced reactor
SMR	Small modular reactor
UHS	Ultimate heat sink
WWER	Water-cooled and water-moderated energy reactor

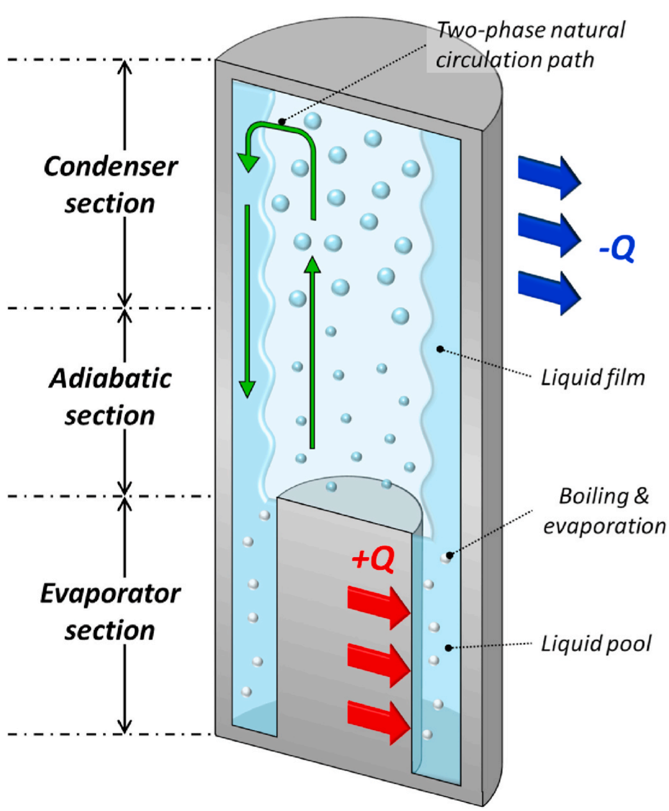

Fig. 1. Fundamental principle of heat pipe or thermosyphon concept with two-phase natural circulation heat transfer for the proposed design.

high reliability, reactor safety can be guaranteed, and economic enhancement can be expected, with these designs being more compact than traditional reactor systems [8]. The heat pipe and thermosyphon are being developed and evaluated for their applicability in the nuclear field.

Several previous studies have been conducted to investigate the functionality and feasibility of applying heat pipes or thermosyphon systems for nuclear system design applications. Table 1 summarizes previous research in which the heat pipe or thermosyphon concept was proposed as a passive heat transfer system. Kim and Bang proposed a Passive IN-Core cooling system (PINCs) that takes the hybrid form by integrating the heat pipe into the existing control rod [9]. A series of experiments were conducted to analyze the thermal performance of the proposed PINCs. Although the applicability of PINCs to SMRs was not demonstrated, it is expected to enhance the safety and reliability of NPPs. Kim et al. applied PINCs to the upper part of the control rod driving mechanism (CRDM) of SMRs as an additional passive decay heat removal system and analyzed and evaluated the performance of PINCs in SMR design under the stationary black-out (SBO) transient accidents [10]. PINCs showed excellent performance as an additional decay heat removal system, and the applicability of the heat pipe concept to SMRs was evaluated. Jeong et al. analyzed the performance of PINCs in operating pressurized water reactor conditions with computational fluid dynamics (CFD) code [11]. The analysis results reveal that the application of PINCs can secure additional golden time before core damage under SBO conditions. As a passive containment cooling system (PCCS) for APR-1400 design, Nam et al. proposed multi-pod heat pipes (MPHPs) located at the top of the containment to discharge the decay heat generated from the core and evaluated heat removal performance with their heat transfer mechanism [12]. The design prevents issues such as countercurrent flow and fluid entrainment, and the heat removal capacity increases with the length of the heat pipes. Although the study recognizes the impact of non-condensable gases on heat transfer,

Table 1
Summary of proposed heat pipe-based passive safety systems.

Type	Length (mm)	D_{in}(mm)	D_{out}(mm)	Heat load (W)	Pressure (kPa)	Reference
Annulus Concentric	1000	22	25.4	25-1600	20	[9]
Cylindrical	>5000	29	30	<6000	70-1000	[12]
Loop	6000	140	150	<271000	N/A	[13]
Cylindrical	N/A	In-core: 1.6–6.4 Out-core: 12.7–25.4		N/A	N/A	[14]

quantitative evaluation was not conducted for this effect. Mochizuki et al. proposed the loop-type heat pipe-based core cooling system, which can passively operate by gravity [13]. The designed system consists of cylindrical evaporators and a naturally cooled finned condenser, which can reduce the reactor temperature from 282 °C to below 250 °C within 7 hours. The heat pipe system operates without relying on active power supplies, enhancing its reliability under accident conditions. Additionally, an initial water charge affects thermal performance and addresses the Leidenfrost phenomenon, offering safe and dependable cooling solutions for nuclear reactors. Dunckel proposed an in-vessel and ex-vessel emergency heat removal system consisting of multiple heat pipes to prevent pressure vessel hot spots during the loss of coolant accident (LOCA) [14]. The proposed heat pipe emergency heat removal system by Dunckel showed passive safety through depressurization performance; however, the multiple heat pipe configurations located inside and outside the core can increase design complexity due to redundancy. Sviridenko proposed a heat pipe heat exchanger as an emergency system to be used in the event of a long-term power loss in the water-cooled, water-moderated energy reactor (WWER) [15]. However, it still has design complexity due to multiple heat pipe configurations, which can act as a factor that undermines the strengths of SMRs.

Most previous research related to heat pipes or thermosyphons has been developed and proposed to further enhance the passive safety of nuclear reactors. The reactor safety can be guaranteed in the established design framework, since SMRs already have a sufficient safety margin with conservatism. However, conservative design may limit the practicality of SMRs leading to over-budget situations and project delays. Addressing economic feasibility remains a challenge while SMRs are designed with safety as a top priority. The heat pipe-based passive safety system is needed to secure safety and reliability while overcoming the economic-related challenges in SMR design. In this context, the motivations of this study are classified into two main objectives.

1) The proposal of the conceptual heat pipe or thermosyphon system as a means of achieving design compactness for practicality and economics, overcoming conservatism.
2) Ensuring core integrity in the proposed SMR design with a heat pipe or thermosyphon system under postulated accidental conditions, as is done in existing designs.

Based on the above objectives, this study aims to offer a potential solution to the economics of current SMR designs and concerns related to their construction feasibility.

2. Concept of heat pipe type double-containment vessel on SMR

2.1. Overview of safety characteristics in light water-cooled SMR

The light water-cooled SMRs feature an integrated reactor pressure vessel (RPV) design encompassing the reactor cooling system (RCS), steam generators (SGs), and pressurizer (PZR) to inherently eliminate LB-LOCA scenarios [2]. A distinguishing feature of these SMR designs is the compactness of the stainless-steel containment vessel (CNV) arranged outside of the integrated RPV to minimize the risk of radioactive material being released into the atmosphere in the event of postulated accidents. The CNV can be submerged in a large water pool, serving as the ultimate heat sink (UHS) for both the emergency core cooling system (ECCS) [16] and the passive decay heat removal system (DHRS) [17]. The ECCS consists of several mechanical valves including reactor vent valves (RVVs) and reactor recirculation valves (RRVs) to compensate for the coolant inventory in the RPV and to prevent the core and nuclear fuel exposure to the atmosphere [16]. Under accidental conditions, the coolant in the RPV boils off or evaporates due to core decay heat and exits through the valves. The vented coolant condenses at the top of the CNV, transferring heat to the heat sink upon contact. The condensed coolant descends, accumulating at the CNV bottom, thereby increasing the water level and facilitating coolant recirculation to maintain adequate coolant coverage over the nuclear core. The decay heat from the core is passively removed by the two-phase flow natural circulation within the DHRS following the safety control rod axe man (SCRAM). The heat exchangers of DHRS consist of several pipes with long lengths to increase the contact surface area with the heat sink [17]. The DHRS can compensate for the loss of natural circulation driving force due to feed water supply interruption. Therefore, given the geometric characteristics of existing designs with passive safety systems, it is reasonable to require a large pool and sufficient water to adequately remove or transfer the decay heat released from the core.

The passive auxiliary system is configured based on the geometric characteristics of a deep and wide water pool capable of cooling the DHRS and the entire CNV [18]. The decay heat generated from the reactor core is ultimately removed to the final heat sink by conduction and natural convection heat transfer mechanisms. A large amount of water comprising the final heat sink has a remarkable heat removal capacity, ensuring the integrity of the reactor core for over 30 days without intervention from operators after SCRAM [19]. Even if all the water in the pool evaporates due to decay heat, abnormal conditions can be avoided by air cooling during long-term transients.

2.2. Design purpose of passive heat pipe type double-containment vessel

The effective passive heat transfer process of heat pipes or thermosyphons can be used as an attractive method in the nuclear field. It has been studied as a concept to passively remove decay heat from the core in an emergency state, serving as an additional passive safety system in PWRs [6]. As described in the previous section, the heat pipe structure can passively transport heat based on internal phase change phenomena, generating a passive two-phase natural circulation path with a simple structure and operating principle. In other words, it has a remarkable heat transfer coefficient and has the advantage of simplifying the reactor design, operating principle, and logic of the decay heat removal system, since phase change heat transfer, such as evaporation/condensation of the working fluid, occurs inside the heat pipe. The passive heat removal performance with the heat pipe concept has been verified through previous research on PINCs [11,20,21]. The proven heat removal performance of the heat pipe can be applied as a solution to the main goal of this study. As a means of improving the economic efficiency and construction feasibility of SMRs, which is the main objective of this study, structural improvement is conceptually proposed to significantly reduce the requirement for the large water pool and the volume of stored water. Fig. 2 illustrates the conceptual SMR design with a heat pipe-based

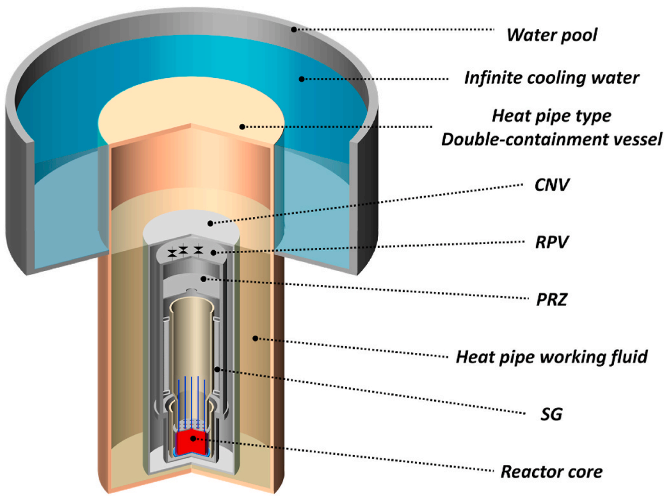

Fig. 2. Conceptual SMR design with heat pipe vessel and modified pool system.

vessel system for the improvement of the large pool structure.

The decay heat removal mechanism utilizes multiple piping and long lengths to ensure a high thermal contact surface area and sufficient natural circulation driving force during accidental conditions [17]. This requires a large volume of stored water to accommodate the decay heat from the core, potentially leading to poor plant economics [4]. To resolve the challenges regarding SMR practicality and feasibility, the ultimate goal of this comprehensive framework is to improve the pool structure by leveraging the use of passive and effective heat transfer devices, such as heat pipes or thermosyphons depicted in Figs. 1 and 2. Moreover, the proposed passive safety system, aimed at reducing the volume of stored water, must ensure the decay heat removal performance to a reasonable level for the reactor integrity. The thermal performance of the proposed heat pipe system is evaluated using the thermal-hydraulic system code under several postulated accidental scenarios in the subsequent sections. The schematic for the heat pipe type double-containment vessel (D-CNV) system conceptually proposed in this study is shown in Fig. 3. This heat pipe-based passive safety system is arranged outside the CNV and can enhance heat transfer performance through the internal phase change phenomenon occurring between the high-temperature CNV and the low-temperature heat sink. Detailed information, including dimensions and configuration of the heat pipe type D-CNV, is described in the subsequent sections.

2.3. Phase change heat transfer in heat pipe structure

The proposed passive safety system adapted a heat pipe concept designed to transfer decay heat from the core to the heat sink. It can provide an additional heat transfer path in the axial direction by utilizing two-phase natural circulation within an enclosed tube. The equivalent thermal resistance model in transients is shown in Fig. 4. In contrast to conventional cylindrical heat pipes, this innovative design features a concentric annular flow path with a long length to cover the reactor vessel, furthermore, heat is transferred from the inner wall to the outer wall. According to previous studies, there are effects caused by the variation in the flow path during the vapor traveling to the condenser [9], therefore, the geometry of the D-CNV is designed with a constant vapor diameter to exclude the complexity of the flow variation effect. Previous research described that incorporating a wick structure is beneficial for ensuring a stable temperature distribution during startup [22], as it fully utilizes not only boiling but also evaporation in the vapor generation mechanism [23]. Considering that D-CNV operates only vertically, we selected a wickless heat pipe, specifically a thermosyphon, to achieve a vapor flow diameter and reduce flow resistance to a minimum.

As the accident condition progresses without proper mitigation after the reactor trip, the entire CNV is heated by decay heat from the reactor core. Hence, from the perspective of the heat pipe, the water level of the working fluid or filled liquid inside the D-CNV is required to sufficiently cover the evaporator. The filling amount were set to be 48.56 m^3 to prevent decay heat accumulation with the actuation of the heat pipe type D-CNV. Although it requires a large amount of working fluid, unlike commercial heat pipe applications, the radial diameter of the D-CNV is relatively small compared to the axial length. Considering the structural characteristics covering the active core height, a relatively small amount of working fluid can be utilized to generate the two-phase heat transfer path. Table 2 summarizes the design information of the proposed heat pipe type D-CNV as a passive safety system.

Generally, the cross-sectional area for vapor flow is smaller compared to the cylindrical type, which can make concerns for counter flow more pronounced. This hypothesis is similar to the countercurrent

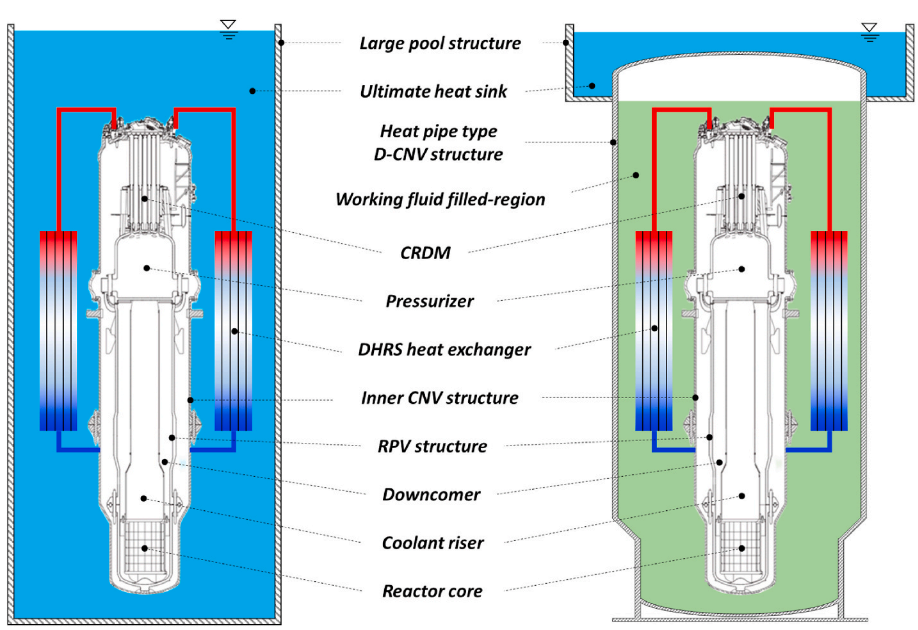

Fig. 3. Schematic of the submerged SMR design (left) [17], and heat pipe type D-CNV as passive decay heat removal system (right).

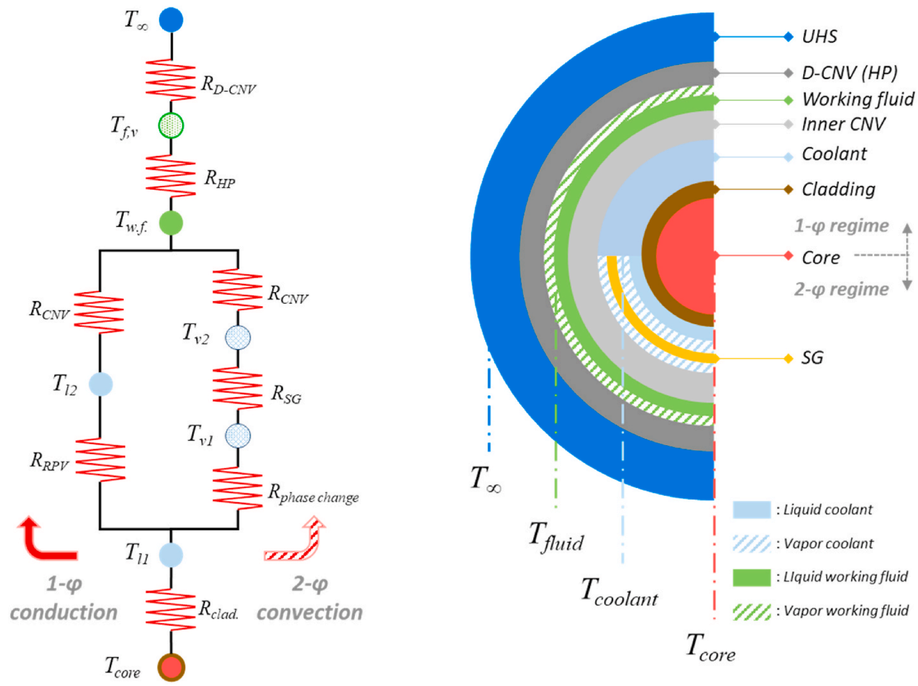

Fig. 4. Simple equivalent thermal resistance model of SMR with proposed heat pipe-based system in transients.

Table 2
Detailed information of the proposed D-CNV system with heat pipe principle.

Index	Parameter	Unit	Value
			Heat pipe type double-containment vessel
Envelop	Length	mm	27390
	Outer/Inner diameter (Core section)	mm	5000/4500 (3930/3430)
	Wall thickness	mm	82.55
	Material	–	Stainless-steel 304
Working Fluid	Filling amount	m^3	48.56
	Material	–	H_2O
	Pressure	kPa	400–772

flow limitation (CCFL) or flow limitation model that cannot be adequately predicted for heat pipe and thermosyphon systems. Kim et al. [22] observed the flow pattern of a concentric annular heat pipe and described that the rapid coalescence of generated bubbles due to a narrow flow area leads to an increase in shear forces between the liquid and vapor phases. Especially, in high heat flux operations, the suggested design can induce a higher velocity of vapor. This in turn results in considerable interfacial shear where the condensed liquid faces challenges in returning to the evaporator. This results in a bottleneck situation at the evaporator's upper section which can induce the dry-out at the evaporator leading to fluctuations in wall temperatures. Consequently, it is vital to highlight the flooding limit for a design approach for the D-CNV system.

The conventional approach for evaluating the flooding limit of a water thermosyphon typically involves the use of several semi-empirical correlations. These methods encompass Wallis' correlation [24] and Kutateladze's correlation [25], which have been employed to predict the flooding effect in two-phase flow for the CCFL phenomena. To predict the flow limitation in a two-phase flow pattern, the MARS-KS code incorporated the CCFL model to calculate the phenomena of flooding in the reactor cooling system [26]. Addressing the limitations of the Wallis and Kutateladze correlations, Tien and Chung [27] combined these approaches and accounted for the influence of surface tension and diameter. Additionally, Faghri et al. [28] extended their analysis method, which is based on wave-induced phenomena, to predict the flooding limit in concentric annular thermosyphons. Hence, this study utilized the CCFL model option based on the semi-empirical correlation embedded in the MARS-KS code. With this option, the heat pipe function can be activated under specific accident conditions, and the unique flooding phenomena inside the heat pipe-based D-CNV system can be simulated with the reactor input model.

$$Q_{\text{Tien and chung}} = C_K{}^2 A_v h_{lv} \left[\rho_l{}^{-1/4} + \rho_v{}^{-1/4}\right]^{-2} [g\sigma(\rho_l - \rho_v)]^{1/4} \tag{1}$$

$$C_K \equiv \sqrt{3.2}\tanh\left(0.5Bo^{1/4}\right) \tag{2}$$

$$Q_{\text{Faghri}} = K h_{lv} A [g\sigma(\rho_l - \rho_v)]^{1/4} \left[\rho_v{}^{-1/4} + \rho_l{}^{-1/4}\right]^{-2} \tag{3}$$

$$K \equiv \left(\frac{\rho_l}{\rho_v}\right)^{0.14} \tanh^2 Bo^{\frac{1}{4}}, \text{ where } Bo = D\left(\frac{g(\rho_l - \rho_v)}{\sigma}\right)^{1/2} \tag{4}$$

The variables are defined as follows: Q is the flooding thermal limit of the heat pipe, ρ_l is the density of the liquid phase, ρ_v is the density of the vapor phase, h_{lv} is the enthalpy of vaporization, g is the gravitational acceleration constant, σ is the surface tension, A is the cross-sectional area of the pipe, and Bo is the Bond number. The parameters C_k and K are the fitting constants for the flooding limit model. Based on Equations (1)–(4), the operation limit of the proposed heat pipe type D-CNV will be supposed to be evaluated with a comparison of calculation values with empirical correlations and the MARS-KS code.

3. Analytical methodology

3.1. MARS-KS code

The one-dimensional thermal-hydraulic safety analysis code, known as MARS-KS (Multi-dimensional Analysis of Reactor Safety), was developed by the Korea Atomic Energy Research Institute [29]. The MARS-KS code is integrated with the RELAP-5/MOD3 and a sub-channel analysis code, namely the COBRA-TF. In this study, the MARS-KS code was employed to analyze the transient behavior of the SMR, both with and without the application of a heat pipe type D-CNV system. The

MARS-KS code employs a comprehensive two-fluid and three-field model, which comprises six-field constituent equations governing continuity, momentum, and energy conservation for two-phase flow regimes. It is worth noting that the MARS-KS code is extensively utilized for regulatory evaluations in the Republic of Korea regarding NPP safety. The predictive capabilities of the MARS-KS code for steady-state and transient thermal-hydraulic phenomena, particularly in the context of light water reactors (LWRs) and LWR-based SMRs, have been rigorously validated and verified through a series of benchmarking studies. These studies involve scaled-down integral effect test facilities, including the ATLAS [30–32], the SMART-ITL/VISTA-ITL [33–35], and the OSU MASLWR test facilities [36]. Subsequent to the validation of the input model for SMR design in the target transient, further computational investigations were conducted based on the established input model applied with heat pipe type D-CNV. These subsequent calculations aim to elucidate the integrated effects of the heat pipe type D-CNV system when applied to the SMR design.

3.2. Reference SMR model

The reference reactor is the NuScale SMR design developed by NuScale Power LLC. The MARS-KS input model used in a previous study [10] was referenced and modified to perform additional modeling on the proposed concept. The MARS-KS input model of the original design was first established by Kim et al. [10]. This model was employed to calculate major parameters for the geometric and thermal-hydraulic information [2,16–18]. According to previous research, representative velocity boundary conditions only include time-dependent junctions employing the mass flow rate to provide feedwater to two steam generators [10]. In other words, the established input data for the MARS-KS code simulation was properly modeled with the representative natural circulation behavior in the RCS. Validation and verification for the MARS-KS input model were conducted with a reasonable agreement with the steady-state calculation values. The established model was analyzed in SBO accident analysis, and an additional passive decay heat removal system, namely PINCs, was modified by applying the heat pipe concept to CRDM [10]. However, additional validation and verification were needed for specific transient conditions focuses on since the model was used in the SBO accident analysis calculation. The nodalization of the original input model used for steady-state and SBO accident calculations is shown in Fig. 5.

3.3. Application of heat pipe type D-CNV on SMR

As a passive safety system applying the heat pipe concept described in Sections 2.2 and 2.3, a heat pipe type D-CNV was modeled with a reference SMR. The modified input model was established by applying the heat pipe type D-CNV to the MARS-KS input model. The modified MARS-KS input model is shown in Fig. 6.

The pool structure is modified to apply the heat pipe concept, and the heat pipe type D-CNV is additionally designed outside the CNV. As described in Section 2.3, the geometric design requirement for the D-CNV is to maintain the same hydraulic diameter of 0.5 m for the evaporation section to minimize changes in the cross-sectional area of the two-phase flow path, which directly affects the heat transfer performance and flooding limit. The CNV is heated as a heat source (evaporator section) due to the core decay heat after the SCRAM under postulated accident conditions. The initial water level was set to cover the active core height corresponding to the heater considering the geometrically long height and relatively small diameter of the CNV, as well as the small hydraulic diameter between the CNV and D-CNV. For the modified design, the height of the large pool as the UHS is significantly reduced from 21 m to 4.78 m. The modified UHS provides a heat sink (condensation section) for the phase change heat transfer phenomenon of the thermosyphon inside the D-CNV. Hence, a passive natural circulation process is generated in which the working fluid

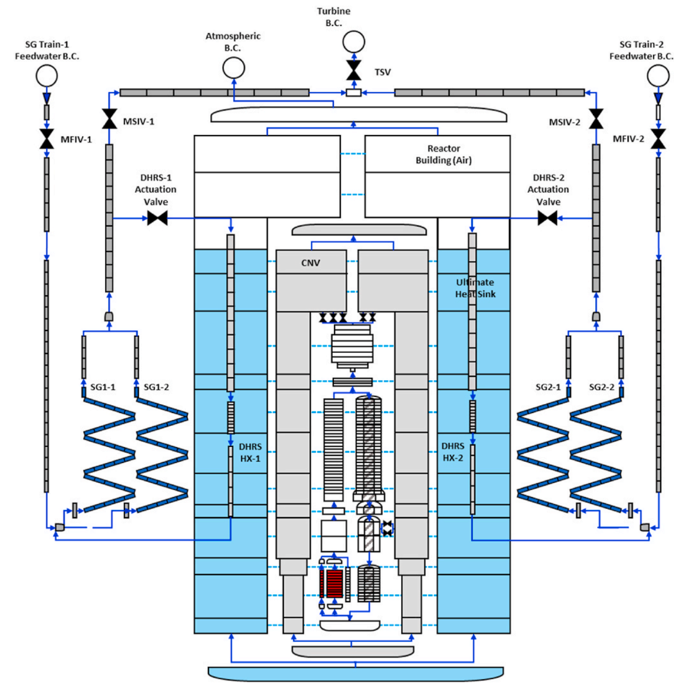

Fig. 5. MARS-KS input model for the reference SMR design [10].

evaporates from the core decay heat, condenses at the top of the D-CNV, and accumulates back to the bottom. These structural characteristics require a height corresponding to the condensation section of the D-CNV cooled by the final heat sink. The water inventory, which had an existing volume of 20,705 m^3, can be reduced by 3,842 m^3 to cool the condenser section of the heat pipe. Fig. 6 shows the MARS-KS input model with the heat pipe type D-CNV and the modified pool design. There were no geometrical and physical modifications to other components except the heat pipe type D-CNV and UHS for the MARS-KS model established in this study. The CCFL model was applied to the two-phase flow pattern of the working fluid inside hydraulic components considering the heat pipe and thermosyphon characteristics of the D-CNV system.

The depiction of the one-dimensional heat pipe and the vessel-type system model is illustrated in Fig. 6. This configuration includes major vessel systems, such as the inner CNV, UHS, and D-CNV, constructed with pipe and branch components featuring single and multi-junction connections [10]. In contrast to conventional loop-type PWR designs characterized by integrated primary systems that exclude the establishment of independent loops, the interconnection between the constitutive pipe components was facilitated through multi-junctions ensuring symmetry in the reactor system. This connectivity is visually represented by dashed lines of distinct colors, cross-connecting the pipe components forming vessel-type systems, as illustrated in Fig. 6. The major systems were interconnected through branch components to establish the upper/lower vessel cavity. The hydraulic volume components essential for calculating the thermal-hydraulic parameters of the internal fluid are enveloped by heat structures. These structures not only account for heat transfer pathways between hydraulic volumes but also incorporate lateral-direction conductive heat transfer. This modeling approach was grounded in the consideration that all vessel systems in the reference SMR design are composed of stainless steel material, motivating the modeling of heat structures surrounding hydraulic volume components of the vessel systems.

3.4. Inadvertent RVV opening scenario in SMR with MARS-KS code

This section deals with the introduction of selected transient scenarios for the SMR design with the heat pipe type D-CNV as a passive

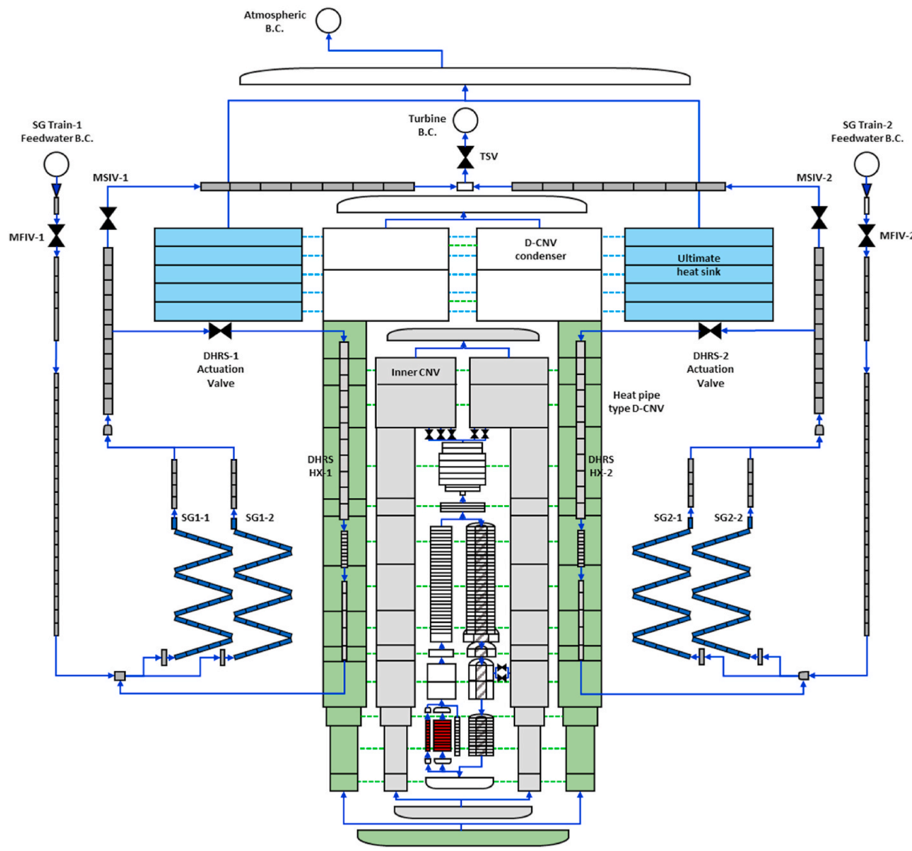

Fig. 6. MARS-KS input model with heat pipe type D-CNV.

safety system with the MARS-KS code. We focus on the small break-loss of coolant accident (SB-LOCA) initiated by inadvertent vent valve opening without DHRS and ECCS actuation. This multiple failure event can seriously affect the reactor systems and even the reactor core [37] since DHRS and ECCS are traditional safety systems in the reference SMR design [16,17]. This failure means that decay heat removal and core water level compensation are not carried out, even though this transient scenario follows a similar sequence to SB-LOCA in general PWRs [38]. Hence, it can be regarded as a beyond-design basis accident (BDBA), resulting in the loss of coolant inventory and uncovering the active fuel rods [37].

According to previous studies [37], the transient is initiated by the inadvertent opening of one of three RVVs located on the top of the RPV, which is supposed to be operated for normal actuation of ECCS. The primary coolant initially present in the high-pressure system, the RPV, leaks into the CNV due to the valve opening. The discharged primary coolant flows out in the form of superheated steam as it is released at high speed into the low-pressure system and increases the collapsed water level (CWL) inside the CNV. When the CWL in the CNV exceeds a certain level (~6.096 m) or that in RPV, an ECCS actuation signal is automatically generated, and the ECCS is supposed to compensate for the coolant inventory along with the opening of two RRVs located above the effective core height [16]. However, as the accident progresses without ECCS actuation, the CWL in the RPV decreases due to the continuous leakage of the primary coolant. Consequently, the reactor core can be uncovered, causing the peak cladding temperature (PCT) to rapidly increase and leading to cladding oxidation behavior and hydrogen generation [37]. This scenario was selected as a reference transient for heat pipe type D-CNV performance analysis, which can effectively transfer the decay heat from the core to the heat sink in a transient. The criteria for selection of cases must be clear, as this transient increases the temperature of the CNV, including the RPV, and the heated CNV can be regarded as the heater or evaporator section of the heat pipe type D-CNV. Simulation cases were established for the presence and absence of traditional safety systems' functions.

The MARS-KS input model of the reference SMR, established in a previous study [10], was verified for inadvertent RVV opening transient accidents prior to modeling the heat pipe type D-CNV system. Several cases for validation and verification of the original reference SMR model are listed in Table 3 to ensure the reliability of the data generated from the MARS-KS calculation. The verification of the MARS-KS input model in the target transient was performed by comparison with the thermal-hydraulic behaviors in the case of functional actuation of traditional safety systems (Case A) and the case of multiple failures (Case B).

Cases C and D were selected to analyze the feasibility and thermal performance of the heat pipe type D-CNV by comparing the results with Cases A and B, along with the application of the D-CNV, respectively. In Case C, the purpose is to demonstrate the feasibility of the proposed concept with two-phase heat transfer alongside the actuation of the traditional safety systems. In Case D, which is the main target scenario of this study, the heat pipe type D-CNV design is intended to analyze the passive heat transfer performance in the event of traditional safety systems failure and to assess the feasibility of D-CNV application. According to previous research [37], the ECCS was emphasized for its exceptional importance in the reference SMR design and its potential significant influence on the temperature behavior of the outer wall of the

Table 3
Summary of transient cases of MARS-KS model for original SMR model.

Case	DHRS	ECCS	D-CNV
A	O	O	X
B	X	X	X

CNV due to the compensation of the CWL above the reactor core. Hence, Case E, in which the actuation of not DHRS but ECCS was considered, was additionally selected for the performance evaluation case. Table 4 summarizes all transient cases considered for MARS-KS calculations for the model applying the heat pipe type D-CNV.

4. Results and discussion

4.1. Steady-state analysis result

As a basis of transient analysis for performance evaluation, steady-state analysis was performed using the MARS-KS code for the original and heat pipe type D-CNV applied SMR models described in Sections 3. The steady-state calculation values for the reference SMR design are summarized in Table 5, along with the values contained in the design certification application (DCA). The steady-state calculation values show good agreement with the original design implemented in DCA documents [17], with an error of around 1 %. This justifies further analysis as a standard model for applying heat pipe type D-CNV and postulated transient conditions. The modified model with heat pipe type D-CNV also shows similar consistency to the original model. It is deemed to hardly affect the major thermal-hydraulic parameters inside the reactor system since the D-CNV is arranged outside the existing CNV. The inside of the CNV is almost a vacuum and there is no heat transfer medium, although the heat pipe type D-CNV is filled with working fluid to the height of the CNV, so heat loss is ignored during the normal operation. The original and heat pipe type D-CNV applied models used in this section were utilized for performance analysis in transients in the following sections.

4.2. Inadvertent RVV opening accident of reference SMR (Case A)

First, the MARS-KS simulation was performed on Case A, an inadvertent one RVV opening accident with normal actuation of traditional safety systems to validate and verify the original SMR model. Transient event timing and MARS-KS calculation results for Case A are summarized in Fig. 7, respectively. As the transient starts, the primary coolant flows out of the upper part of the RPV into the CNV in the form of superheated steam, as illustrated in Fig. 7 (b). The pressure of CNV increases and reaches the maximum pressure value due to the merger of the high-pressure system in the RPV (~12.76 MPa), as shown in Fig. 7 (d). Afterward, signals for the reactor trip, the DHRS, and the main steam isolation valves (MSIVs) are generated within 1 sec after the transient. The SCRAM and DHRS actuation operate 2 and 30 sec after each signal generation, respectively. After the reactor is shut down, the DHRS continuously transfers decay heat generated from the core to the UHS and alleviates heat accumulated in the RPV as shown in Fig. 7 (a). The primary coolant leaked due to decay heat is condensed in the upper part of the CNV due to the external UHS, and the condensed coolant accumulates in the bottom of the CNV. As the coolant inventory of the RPV is lost and the CWL of the CNV increases, the ECCS actuation signal is generated. At the same time, the two RRVs arranged beside the core and the two remaining RVVs at the upper part of the RPV operate to compensate for the reduced coolant inventory in the RPV. The core exposal situation can be excluded through the normal operation of ECCS since the two RRVs are located above the active fuel rods. The coolant flow rate through the reactor safety valves (RSVs) is ignored since they passively operate by the pressure difference between the RPV and the CNV [17], as shown in Fig. 7 (c). The DHRS and ECCS models are verified for the original SMR design based on the results in this section. The heat removal capacity of the DHRS was appropriately calculated using the MARS-KS code for comparison with the other cases.

Table 4
Summary of transient cases of MARS-KS model with heat pipe type D-CNV.

Case	DHRS	ECCS	D-CNV
C	O	O	O
D	X	X	O
E	X	O	O

Table 5
Steady-state calculation values for original and D-CNV applied SMR model with MARS-KS code.

Design parameter	Unit	NuScale SMR (DCA)	NuScale SMR (MARS-KS)	SMR with D-CNV	Error
Core power	MWth	160.00	160.00	160.00	0.00 %
Core inlet temperature	K	531.48	532.40	532.41	0.17 %
Core outlet temperature	K	587.04	584.07	584.07	0.01 %
RPV pressure	MPa	12.76	12.76	12.76	0.00 %
CNV pressure	MPa	0.00	0.00	0.00	0.00 %
Feed water inlet temperature	K	422.04	421.92	421.92	0.03 %
Steam operating pressure	MPa	3.45	3.45	3.45	0.00 %
Primary mass flow rate	kg/s	587.15	587.07	587.07	0.01 %
Feed water mass flow rate	kg/s	67.07	68.00	68.00	1.30 %
Core bypass mass flow rate	kg/s	42.86	42.417	42.417	1.04 %

4.3. Inadvertent RVV opening accident without DHRS and ECCS (Case B)

In this section, verification of the inadvertent RVV opening without traditional safety systems transient (Case B) was conducted. The reference data for verification are the results using the RELAP/SCDAPSIM code for the NuScale SMR model [37]. The MARS-KS calculation results compared with reference data for Case B transient are shown in Fig. 8. This scenario exhibits a similar sequence to Case A in the early stages after the transient. However, it shows different thermal-hydraulic behavior over time since DHRS and ECCS are unavailable. The primary coolant inventory in the RPV continuously decreases over the transient time, and it cannot be compensated due to the absence of the ECCS. Therefore, 3 hours after the accident, the coolant inventory is lost up to the effective height of the nuclear fuel, and the PCT increases with the exposure of the active fuel rods. In the previous study [37], when the ECCS is unavailable, the fuel cladding oxidation and core damage (PCT >1,477 K) occur within 4.5 and 4.8 hours after the transient, respectively. This behavior can cause physically problematic situations in the core, the reactor, the containment structure, and even entire NPP systems. The calculation results also emphasize the importance of ECCS as a reactor safety system and these accidents can be completely mitigated through the opening of at least one RRV. The original SMR model was sufficiently simulated with postulated accident tendency excluding the PCT calculation results. Thus, the reliability of the established model is deemed to be obtained for further application of heat pipe type D-CNV. In the case of PCT calculation results, it reasonably decreases in proportion to the saturation temperature of the coolant along with the collapse of the pressure boundary inside the RPV. However, after the nuclear fuel is uncovered in the air, the behavior in response to the sharp temperature increase is conservatively observed. It can constitute an excess of the capabilities of the MARS-KS code since the calculation results after the rapid increase of the PCT show the progress to a severe accident leading to the core damage consequence.

4.4. Application of heat pipe type D-CNV (Case C)

This section deals with the feasibility of the heat pipe type D-CNV

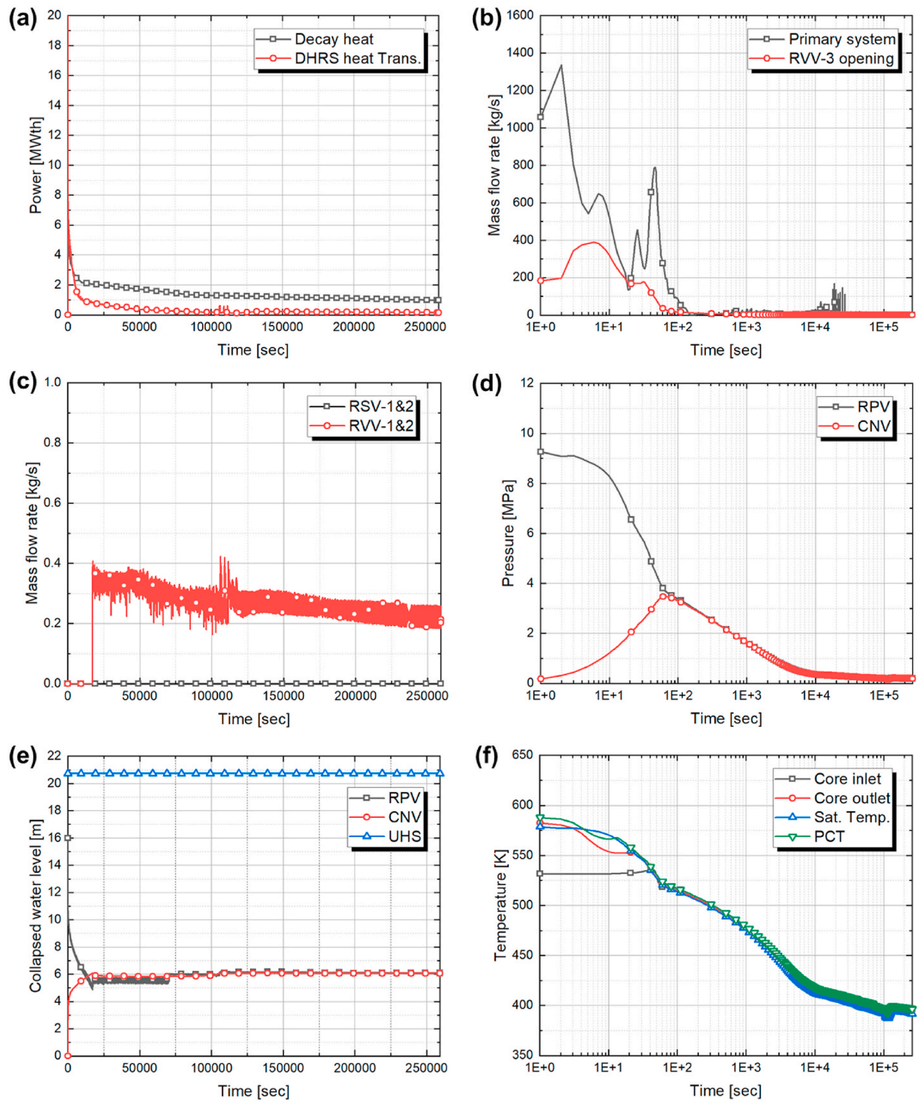

Fig. 7. Transient analysis results due to inadvertent one RVV opening for reference SMR; (a) core decay heat and decay heat removal rate, (b) primary system mass flow rate, (c) safety system mass flow rate, (d) system pressure, (e) collapsed water level, (f) primary system temperature.

system under postulated accident conditions. The transient analysis results of Case C are described in Fig. 9. With the temperature increase of the CNV structure, the early-filled working fluid in the D-CNV system is boiled off increasing the internal pressure. Steam ascends to the condenser section in contact with the final heat sink. Enthalpy transfers to the final heat sink, condensing the steam to the liquid. This typical thermosyphon function was successfully observed in the heat pipe type D-CNV system. The heat transfer rate of the D-CNV system gradually increases with the phase change phenomenon and shows a higher value than the DHRS within 1 hour, as shown in Fig. 9 (a). In the modified design, the main distinction between the D-CNV system and the DHRS is the coolant cooling method and principle, which directly removes high decay heat generated in the early stages after reactor shutdown. After 330 sec, the total passive heat transfer rate with DHRS and D-CNV exceeds the decay heat from the core and stabilizes thermal-hydraulic parameters in the RPV. Even after the system pressures of the RPV and CNV are equalized, the coolant inventory continues to leak out due to the decay heat. An increment in the CWL inside the CNV subsequently generates an ECCS actuation signal. After 3 hours into the accident, two RRVs and two remaining RVVs are activated to compensate for coolant and to prevent the core from being exposed to the atmosphere as shown in Fig. 9 (c). By 72 hours into the transient, thermal-hydraulic parameters in the RPV are stabilized with the decay heat transfer from the core and the continuous reduction of the internal pressure boundary due to the actuation of DHRS, D-CNV, and ECCS. The passive decay heat transfer rate of the heat pipe type D-CNV demonstrates sufficient capability after the reactor shutdown based on the transient calculation results in Case C.

4.5. Application of D-CNV without DHRS and ECCS (Case D)

The safety analysis results of Case D are the focus of this study and serve as the basis for performance evaluation of the heat pipe type D-CNV through comparison with reference data for transients. The transient calculation results of Case D are shown in Fig. 10. Within approximately 1.5 hours, the passive decay heat removal performance of the D-CNV exceeded decay heat as shown in Fig. 10 (a). The D-CNV shows appropriate heat removal performance over time similar to the results described in Section 4.4. It is deemed to demonstrate sufficient feasibility as a passive decay heat removal system. However, neither the DHRS nor the heat pipe type D-CNV provided a solution to the continuous leakage of coolant from the inadvertently opened valve. Fig. 10 (e) shows a decrease in the CWL in the RPV due to the continuous outflow of primary coolant evaporated by decay heat. At the same time, an increase

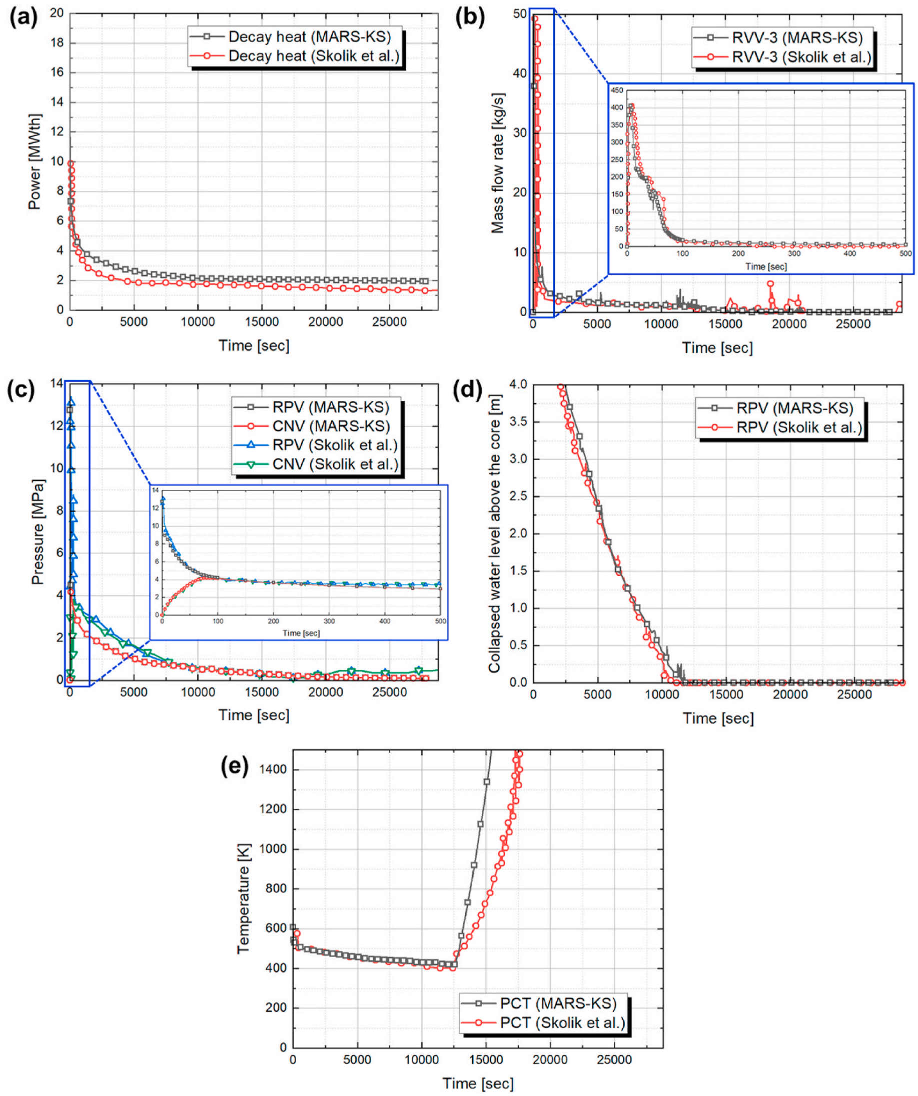

Fig. 8. Transient analysis results due to inadvertent one RVV opening without DHRS and ECCS; (a) core decay heat, (b) break mass flow rate, (c) system pressure, (d) collapsed water level above the core, (e) peak cladding temperature.

in the CWL in the CNV due to the coolant condensed from the D-CNV being accumulated in the bottom of the CNV. The ECCS is unavailable and thus the CWL above the core gradually decreases although the CWL in the RPV has a higher value than that in the CNV. The reactor core is exposed after approximately 17,480 sec, and the PCT exceeds 1,477 K causing the core damage as shown in Fig. 10 (f). These results again emphasize the importance of the ECCS, the significant safety system of the reference SMR design described in previous research [37]. With the results of Case D transient calculation, the feasibility of heat pipe type D-CNV can be confirmed based on the passive decay heat removal performance.

4.6. Application of D-CNV without DHRS (Case E)

An additional case was established, and the analysis was performed to evaluate the feasibility of the heat pipe type D-CNV as an alternative to the existing DHRS based on the results of Section 4.5. In this case, the transient is also inadvertent one RVV opening where ECCS normally operates while DHRS is unavailable with the application of heat pipe type D-CNV. Compared to the results of Case C with DHRS actuation, the saturated pressure equalized between the RPV and the CNV was relatively high owing to the absence of an initial decay heat removal performance of existing DHRS. The system pressure integrated with the RPV and the CNV due to the RVV opening shows a more delayed decompression process than the results from Case C. This effect can increase the outflow of coolant released from the RPV causing the CNV water level to increase quickly. Therefore, in Fig. 11 (e), it is illustrated that the relatively faster operation time of ECCS is approximately 5,316 sec. The D-CNV heat removal performance is consistently higher than decay heat despite the slow decompression process due to the absence of DHRS. In this section, the feasibility of the D-CNV as the passive decay heat removal system was additionally evaluated based on the calculated decay heat removal rate. In the modified model with the heat pipe type D-CNV, integrated system pressure due to the inadvertently opening showed a relatively high value with the relatively low initial decay heat removal rate of the D-CNV. However, based on consistent decay heat removal performance by two-phase natural circulation heat transfer inside the D-CNV, a sufficient accident mitigation process is numerically observed even without the existing DHRS.

4.7. Performance evaluation of heat pipe type D-CNV

The decay heat removal rate of the heat pipe type D-CNV is relatively lower than existing UHS since the heat transfer surface area of the D-CNV is low due to only being served the condenser part as described in Figs. 12 and 13. However, the decay heat removal performance of the D-

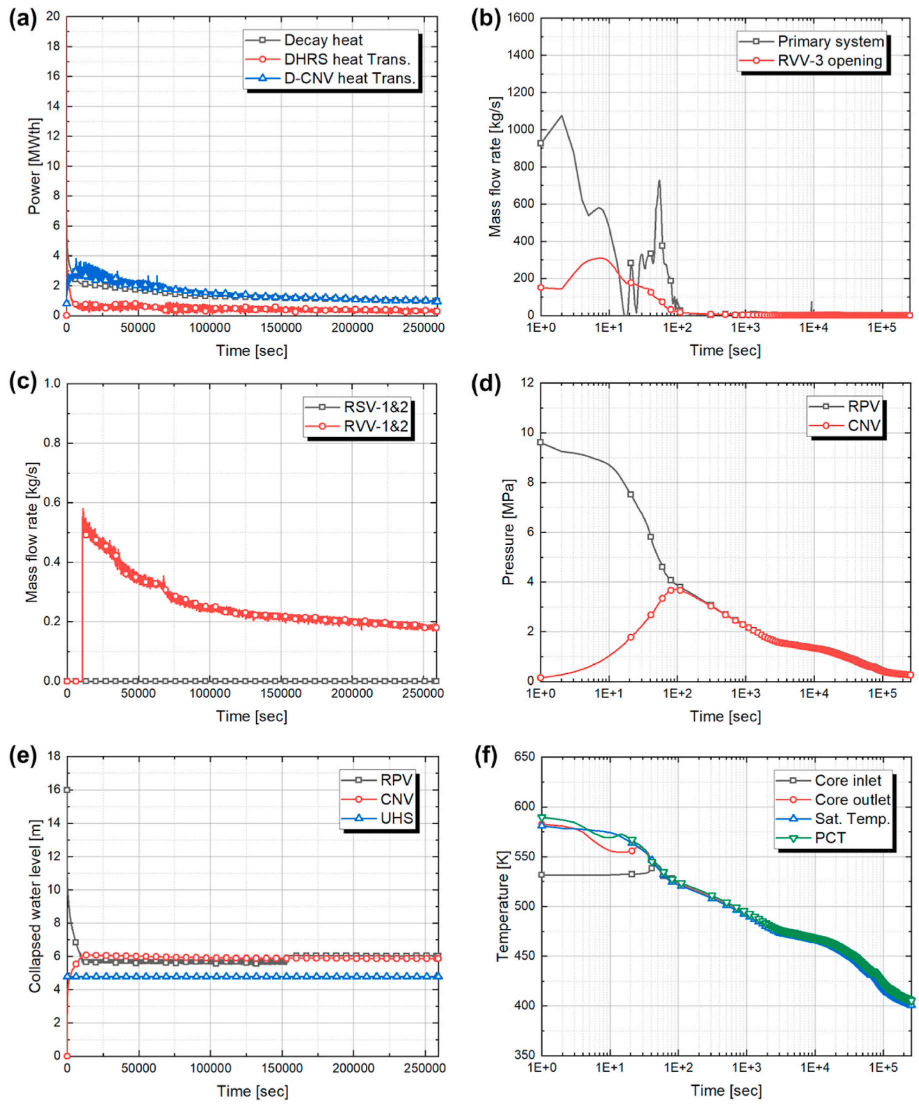

Fig. 9. Transient analysis results for the application of heat pipe type D-CNV; (a) core decay heat and decay heat removal rate, (b) primary system mass flow rate, (c) safety system mass flow rate, (d) system pressure, (e) collapse water level, (f) primary system temperature.

CNV gradually increases and is higher than that of the original design with consistent and effective two-phase decay heat transfer with the actuation of the D-CNV system nearly 6 hours after the transient starts. The notable difference in the decay heat removal performance of the D-CNV is not observed whether the DHRS actuates or not, as shown in Figs. 12 and 14. Moreover, the performances of D-CNV and existing UHS show similar values with transient time after nearly 33 hours. The comparison of simulation results in the accidental transient with safety systems failure is depicted in Fig. 13. The decay heat removal rate of the D-CNV is initially lower than that of existing UHS, however, it surpasses nearly the after 4 hours the transient even with the failure of the traditional safety systems similar to the results from Fig. 12. What is remarkable is the PCT behavior in Fig. 13 (b) and the time when the reactor core is damaged or is exposed to the atmosphere since whether the reactor core is damaged or not specifies the classification of accident level. The simulation results of the main event timing and sequence in all transient cases are summarized in Table 6. When the reference design is faced with the inadvertent RVV opening without safety systems (Case B), the reactor core is damaged with a possibility of a transition to a severe accident at nearly 15,390 sec as shown in Table 6. In the case of the proposed design with the D-CNV, the thermal enthalpy of the primary system is transferred to the inner CNV as well as the D-CNV with the additional supply of bypass line in the transient conditions. This behavior can lead to a decrease in the flow rate through the opened RVV and a decrease in the leakage rate of the primary coolant inventory. Consequently, in Case D, the PCT reached 1,477 K leading to the core damage situation at nearly 17,480 sec expanding the golden time (~2,090 sec) and indicating the high possibility of mitigating the progress of reactor accidents due to the decay heat transfer to the UHS with the phase change natural circulation. Based on this kind of behavior, it can be interpreted that the proposed D-CNV as a passive safety system is a more reasonable design considering the anticipated advantages as well as sufficient decay heat removal performance.

Fig. 15 illustrates the comparison of the simulation results of the proposed SMR design with the heat pipe type D-CNV according to the actuation of DHRS or ECCS. The decay heat removal performance of D-CNV is irrelevant to the actuation of ECCS, which compensates for the inventory or collapsed water level in the primary system. The D-CNV is dependent on or systemically coupled with the ECCS, which can prevent the reactor core from being exposed to the atmosphere since the heat removal capability is higher with transient times. Hence, as confirmed in a previous study [37], the ECCS of the reference SMR design has a high importance compared to the other safety systems although the D-CNV system is additionally applied to it.

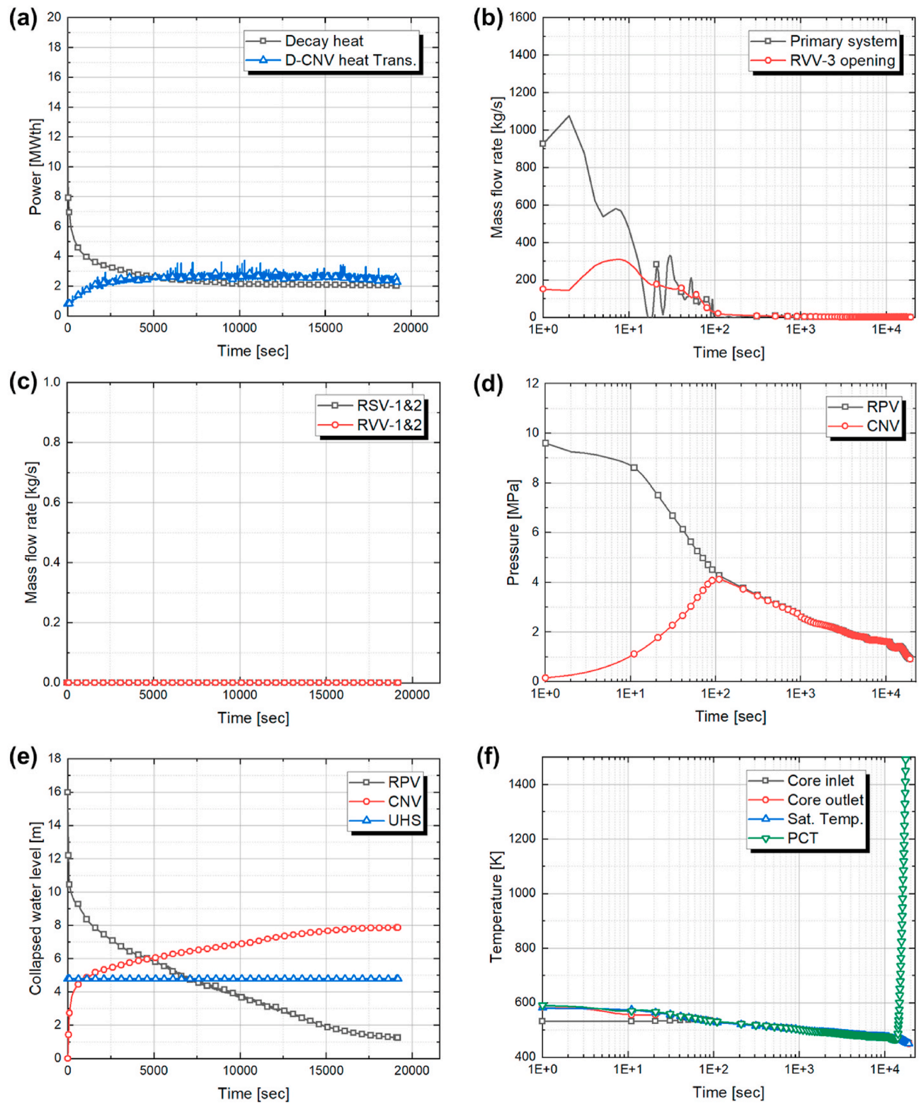

Fig. 10. Transient analysis results for the application of heat pipe type D-CNV without DHRS and ECCS; (a) core decay heat and decay heat removal rate, (b) primary system mass flow rate, (c) safety system mass flow rate, (d) system pressure, (e) collapse water level, (f) primary system temperature.

4.8. Flooding limit of heat pipe type D-CNV

The investigation into the maximum heat removal capabilities of the heat pipe type D-CNV stemming from the proposed design involved a meticulous assessment, with a focus on the theoretical operation limit of a two-phase thermosyphon. In this context, the primary impediment to the heat removal capacity of thermosyphon is the flooding limit. The evaluation of the flooding limit correlation for the D-CNV was conducted featuring the annulus evaporator geometry using Equations (1)–(4) described in Section 2.2. This assessment followed the framework delineated in Equation (1), as carried out by Kim and Bang [9]. Furthermore, the assessment of the maximum heat load for the thermosyphon integrated into the heat pipe type D-CNV design was predicated upon established correlations by previous research [27,28]. These correlations are extensively employed for powerful prediction of the flooding limit of wickless heat pipes.

The maximum heat removal capacity of heat pipe type D-CNV was depicted in Fig. 16. The overall calculation results were observed with lower values than that from empirical correlations [27,28]. The bubbles inside the heat pipe by the phase change can rapidly be merged resulting in the consideration of the flooding limit since the suggested D-CNV is modeled with the shape of the concentric annular heat pipe. However, the geometrical characteristics of the heat pipe type D-CNV were designed relatively high with meter scale of the height and diameter unlikely commercial heat pipe structures. Hence, the design margin can be ensured with the actuation of heat pipe type D-CNV as the passive decay heat removal system with the large value of the vapor cross-section.

5. Conclusion

In this study, the feasibility of a heat pipe type D-CNV was investigated as a passive safety system to improve the challenges of light water-cooled SMR design from the perspective of economics and reactor safety. The conceptually proposed D-CNV system has the advantage of passive heat transfer based on the phase change phenomena with compact geometry. The effectiveness of heat pipe type D-CNV facilitates the reduction of a large amount of fluid in a large pool structure which serves as the heat sink. Furthermore, the heat pipe D-CNV system passively actuates along with the thermal-hydraulic behavior of CNV integrated with the RPV to enhance the reliability of SMR safety.

In this context, several postulated accidental conditions were chosen to evaluate the performance of the heat pipe type D-CNV. An input model showed a good agreement with the reference design. The

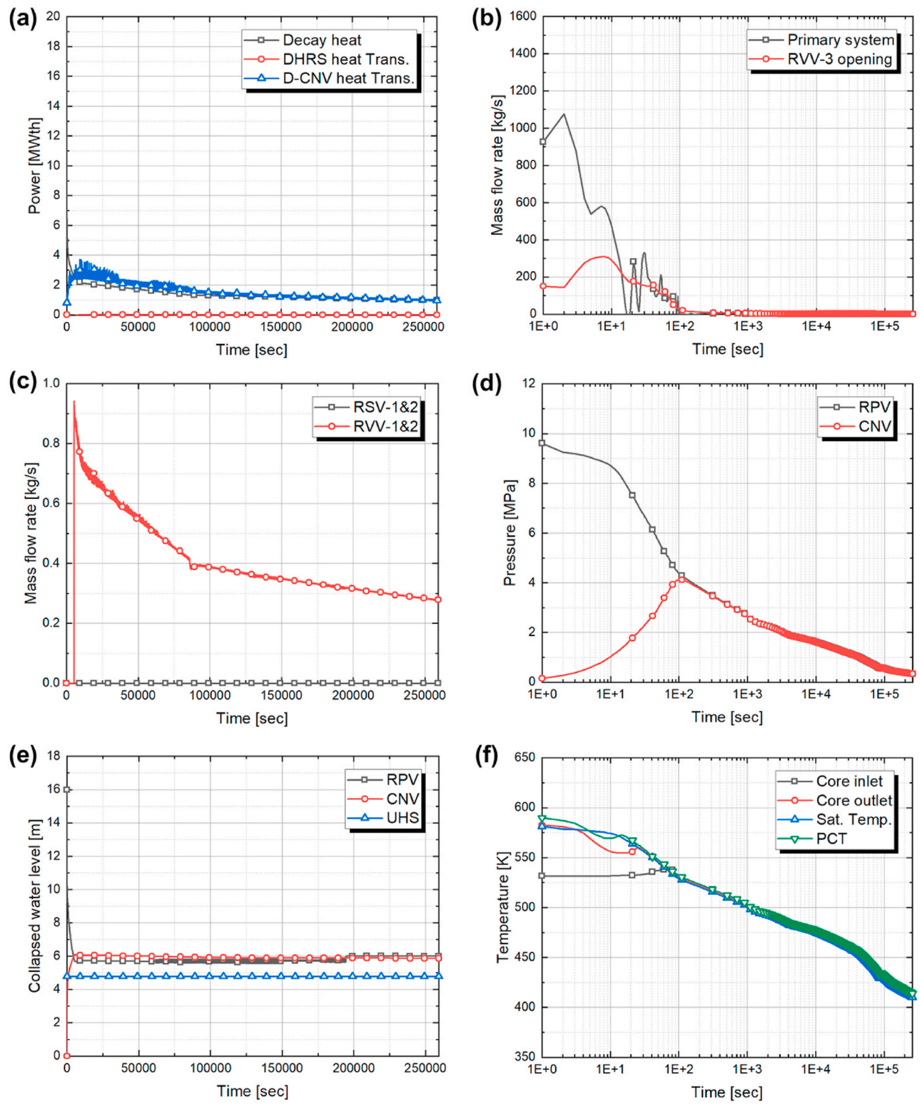

Fig. 11. Transient analysis results for the application of heat pipe type D-CNV with only ECCS actuation; (a) core decay heat and decay heat removal rate, (b) primary system mass flow rate, (c) safety system mass flow rate, (d) system pressure, (e) collapse water level, (f) primary system temperature.

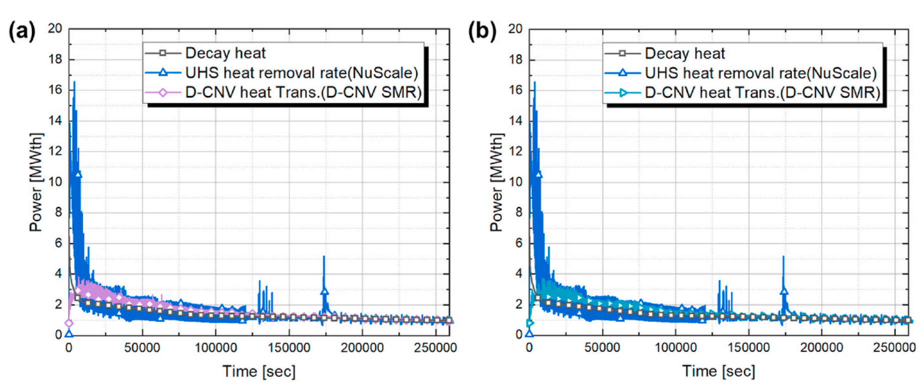

Fig. 12. Comparison of the decay heat removal performance with existing UHS and proposed D-CNV (a) in Case A and C, and (b) in Case A and E.

modified input model applied to the heat pipe type D-CNV was simulated to validate the decay heat removal performance using the MARS-KS code for postulated transient scenarios in SMRs. The selected transient cases encompassed the inadvertent opening of one RVV including the multiple failures of traditional safety systems, such as DHRS and ECCS.

The application of the heat pipe type D-CNV is expected to enhance the practicality of SMR construction based on the improvement of the large pool structure, leading to reduced construction costs and improved economics. The D-CNV system can remove decay heat by cooling only the upper part of the vessel while overcoming the arrangement limitations of the heat sink from the heat pipe characteristics based on two-

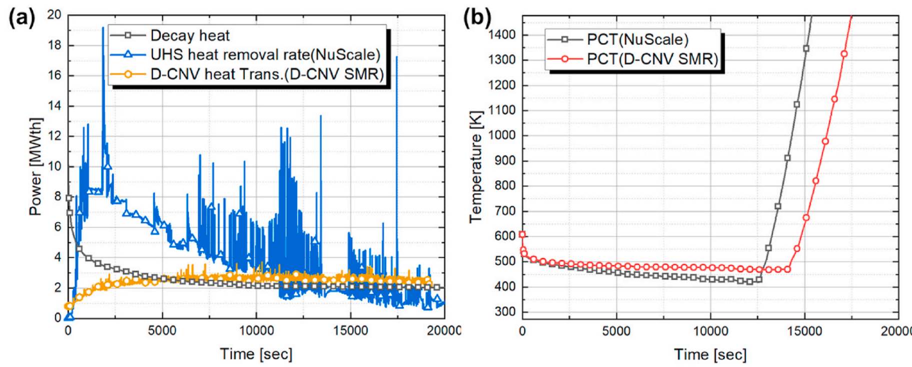

Fig. 13. Comparison of the calculation results for core damage scenarios; (a) decay heat removal performance, and (b) peak cladding temperature.

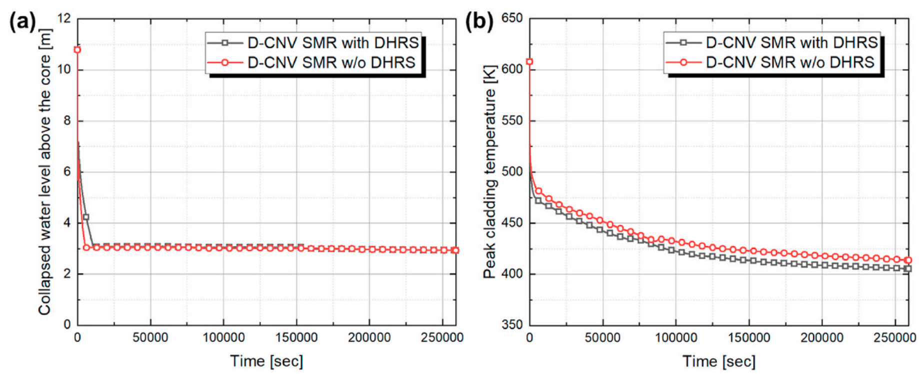

Fig. 14. Comparison results according to the DHRS actuation in proposed SMR design; (a) CWL above the core, and (b) PCT.

Table 6
Summary of main event sequences in all transient cases with MARS-KS code calculation.

Event (sec)	Case A	Case B	Case C	Case D	Case E
RVV opening	0	0	0	0	0
High CNV pressure	0.332	0.327	0.385	0.385	0.385
Low PRZ pressure	0.404	0.404	0.487	0.487	0.487
Reactor trip	2.332	2.329	2.391	2.391	2.391
DHRS actuation	30.332	–	30.391	–	–
Max. CNV pressure	65.020 (3.48 MPa)	82.010 (4.19 MPa)	94.020 (3.70 MPa)	100.000 (4.12 MPa)	100.000 (4.12 MPa)
ECCS actuation	17233	–	10849	–	5316
Core damage (PCT>1,477K)	–	15380	–	17480	–
Transient end time	259200	28827	259200	20170	259200

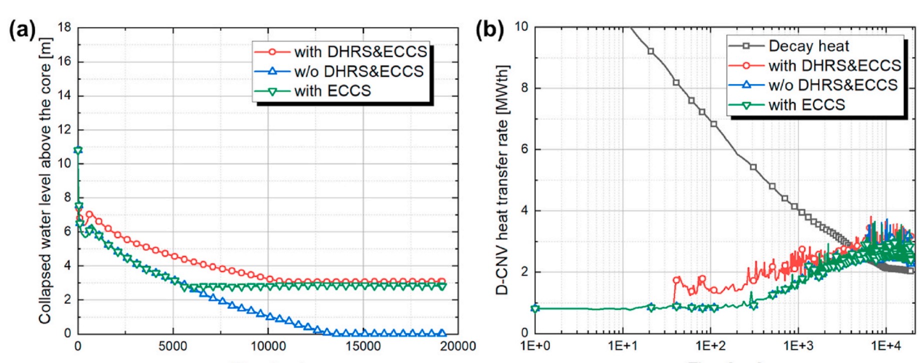

Fig. 15. Comparison results according to the actuation of safety systems in proposed SMR design at the initial phase of postulated scenarios; (a) CWL above the core, and (b) total heat removal rate.

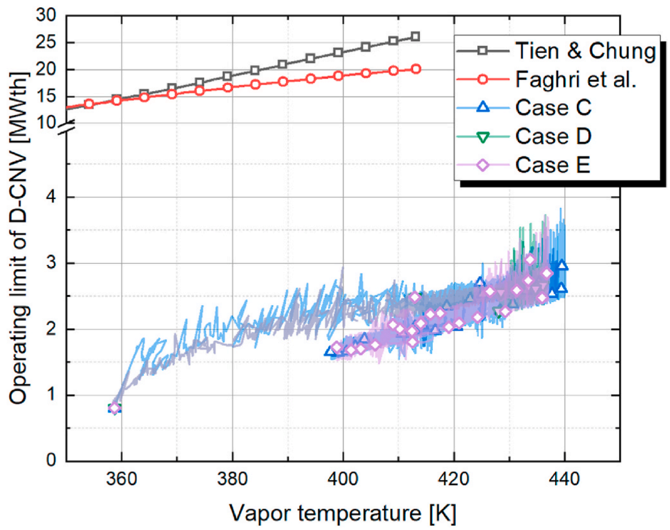

Fig. 16. Comparison of the maximum heat removal capacity of the heat pipe type D-CNV according to the flooding limit calculation results.

phase heat transfer. The water inventory volume is reduced by approximately 81.44 % compared to the existing pool design, facilitating the alleviation of the concrete building costs for the reactor containment. From the perspective of reactor safety, the utilization of the heat pipe type D-CNV not only ensures the existing heat removal performance but also extends the golden time until core damage occurs under specific transient conditions despite a significant reduction in the height of the pool and the volume of stored water.

To verify the heat removal performance of the heat pipe type D-CNV, the research investigated the maximum heat removal capabilities particularly focusing on the flooding limit. The study established that the proposed heat pipe-based system offered a significant design margin by employing empirical correlations and theoretical models, even with its unique geometrical characteristics. This margin ensured that the D-CNV could operate as a robust passive decay heat removal system.

The major findings of the current work can be summarized as follows.

1. The application of the heat pipe type D-CNV was numerically analyzed for its feasibility and decay heat removal performance as an innovative passive safety system with internal two-phase natural circulation heat transfer.
2. Based on the compact geometrical characteristics and an efficient heat removal rate based on the phase change phenomena in the D-CNV, the submerged type SMR can be improved with the reduction of a large volume of stored water.
3. Furthermore, the reduction of the requirement of the large pool structure is anticipated as the solution for concerns and economic challenges for the construction of SMR.
4. This study is anticipated to offer significant insights into the feasibility of heat pipe-based systems in SMR designs and the improvement of the practicality of the small modular NPP as a representative energy source in the future.

However, improving the UHS auxiliary system ultimately involves modifications to the heat sink, which could impact one of its strengths: long-term cooling performance. The application of the proposed heat pipe type D-CNV in this study should not compromise the existing passive decay heat removal performance, so further research is imperative. From this perspective, an investigation of the passive cooling performance of over 30 days without operator intervention under the SBO transient conditions will be included in the comprehensive framework of the research. Analyzing the impact under various transient conditions in the improved UHS structure and simulating additional solutions aim to enhance the feasibility of SMR construction, anticipating insightful contributions in the nuclear industry.

CRediT authorship contribution statement

Ju Hun Jung: Conceptualization, Data curation, Formal analysis, Investigation, Methodology, Validation, Writing – original draft, Writing – review & editing. **Ji Yong Kim:** Conceptualization, Investigation, Methodology. **Dong Hun Lee:** Conceptualization, Investigation, Methodology. **In Cheol Bang:** Conceptualization, Investigation, Methodology, 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.

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