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Investigation of the Heat Removal Capability of the Hybrid Control Rod in Natural Circulation-type SMR under Station Black Out Accident with MARS-KS code

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1. Introduction

There has been increasing interest in low-carbon emission energy sources due to climate change issues. The small modular reactor (SMR) which shows improved safety features compared to the existing large nuclear power plant is considered as most promising solution as the low-carbon energy source worldwide.

The integral pressurized water reactor type SMR (i-SMR) design with passive safety systems shows improved safety features compared to the existing large nuclear power plant [1]. The i-SMR design with passive working principles makes the entire system simpler and more reliable with an enhanced safety margin [2].

The evolutionary design feature of natural circulation type i-SMR was evaluated by several researchers. Skolik et al. analyzed the transient behavior of natural circulation type i-SMR under the inadvertent opening of one reactor vent valve without ECCS and DHRS by RELAP/SCDAPSIM code [3]. The core meltdown was observed in this study and the author emphasized the importance of ECCS. The transient behavior of the i-SMR under the station blackout accident (SBO) was analyzed by Fakhraei et al. [4]. During SBO transient, i-SMR experiences the two-phase density wave oscillation type instability due to the loss of the natural circulation path. The failure of ECCS during the SBO accident prolonged the time duration of two-phase flow instability. In the following study, Fakhraei et al. propose the two approaches (safety injection tank, riser valve) to reduce the two-phase flow instability [5]. The analysis results show that the proposed methods effectively reduce the two-phase flow instability by ensuring the natural circulation path during the SBO accident.

The previous safety analysis research on natural circulation type SMR reveals the importance of passive safety systems. Various passive safety system designs including PINCs (passive in-core cooling system) was proposed to improve the safety margin of the i-SMR design. The novel concept of PINC which contains the hybrid heat pipe (H-HP) was proposed by the UNIST research group [6]. The H-HP takes the hybrid form of the wickless heat pipe (thermosyphon) and conventional control rod. The separate effect tests of the H-HP were conducted in previous literature to evaluate the heat removal performance [7, 8]. However, the integral effect of the H-HP on the reactor system was not validated. In the current study, the feasibility and the heat removal capability of the hybrid control rod were investigated in natural circulation type i-SMR with MARS-KS code.

2. System Modeling with MARS-KS

The NuScale reactor is considered as a reference reactor design of natural circulation type i-SMR design in the current study. The MARS-KS code was utilized to simulate the transient behavior of i-SMR and H-HP under designated accidental conditions. The SBO accident was simulated in the current study which accompany the loss of the AC power in nuclear power plant to analyze the passive safety features of proposed H-HP (PINCs).

2.1 Design Characteristics of the NuScale

The NuScale reactor is a natural circulation type integral PWR reactor with 160MW thermal power [9]. Schematics of the NuScale design with passive safety systems are depicted in Fig. 1.

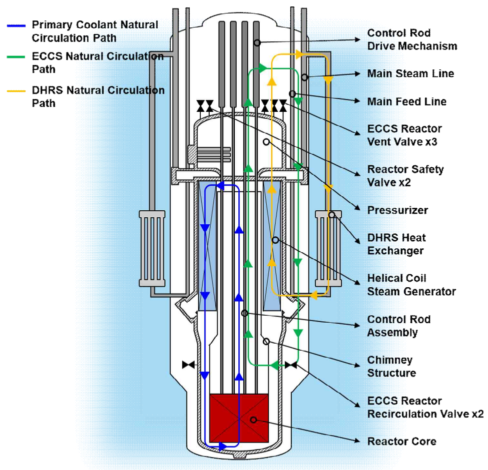

Fig. 1. Schematics of NuScale reactor design features.

The reactor pressure vessel (RPV) of the NuScale is located inside the compact containment vessel (CNV) which is submerged in the ultimate heat sink pool (UHS). The major reactor components including the reactor core, helical coil steam generator (SG), and chimney structure are located inside the RPV. The primary coolant of NuScale is circulated by the natural circulation driven by buoyancy force and gravity head. The ECCS and DHRS are equipped in NuScale design to remove the decay heat from the reactor core during the accidental transient conditions. The ECCS of

NuScale consists of 3 reactor vent valves (RVV) and two reactor circulation valves (RRV). The DHRS of the NuScale consists of a DHRS heat exchanger which submerged in the UHS.

2.3 Reactor Modeling

The MARS-KS nodalization of the NuScale reactor was made based on the published design value information of the NuScale with several assumptions [9].

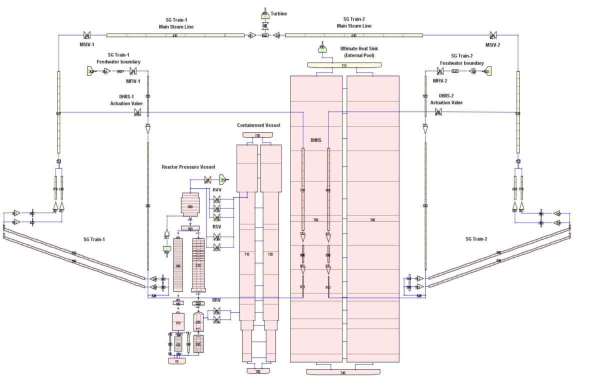

Fig. 2. Nodalization of NuScale reactor design for the MARS-KS code.

The reactor core is modeled with three parallel pipes components which are hot channel, average channel, and bypass channel. The heat generation rate of core was modeled based on the BOC core of the NuScale. The crossflow junction between the hot and the average channel was not considered in the current study to have a conservative analysis.

The inner flow channel of the chimney structure called riser (~ hot leg of conventional PWR) was modeled by a pipe component. The down comer region (~ cold leg of conventional PWR) of the nodalization was modeled by annulus component. The PZR level controller and the pressure controller were added to set the initial conditions.

The secondary system is modeled in 2 trains. Each train contained two helical coil steam generator modules. The helical coil steam generator module is modeled as a single pipe with the equivalent cross-sectional area and hydraulic diameter. The helical coil steam generator of the NuScale was modeled with an inclined pipe with 16.5^{o}. The main feed and steam line were modeled. The main steam isolation valve (MSIV) and main feedwater isolation valve (MFIV) were modeled with the trip valve component.

The CNV and UHS were modeled as parallel pipe components to well simulate the natural circulation phenomena inside. The branch components were used at the upper and lower end of each parallel component. The two DHRS trains were modeled in current nodalization. Each heat exchanger was connected to the secondary system with DHRS actuation valve (trip valve). The DHRS heat exchanger was modeled as single pipe with equivalent cross-sectional area and hydraulic diameter. The heat structure between RPV-CNV, and CNV-UHS was modeled in the current study.

2.2 Hybrid Heat Pipe Modeling

In the original concept of the H-HP, the working fluid (water) is injected into the control rods. The working fluid pass through the annular fluid path formed between control rod cladding and neutron absorber pellets (B_4C). However, the long height of the natural circulation type i-SMR imposes the limitation of the direct application of the original concept of H-HP. In case of the NuScale design, the RPV height is ~17m, CNV height is ~23m. Therefore, the slight design modification of the H-HP was made in the current study to apply the H-HP in SMR. In the modified H-HP design for the i-SMR, the control rod assembly drive shaft acts as a wickless thermosyphon, and control rods are attached below to control the power level of the reactor core. The modified design of the H-HP for i-SMR is depicted in Fig. 3 (b). The total length of the H-HP considered in the current study is 19.38m. The heated section length is 10.25m (~ riser region of RPV), adiabatic section length is 3.83m (~ PZR region of RPV), and condenser section length is 5.3m (~ upper region of CNV above PRV).

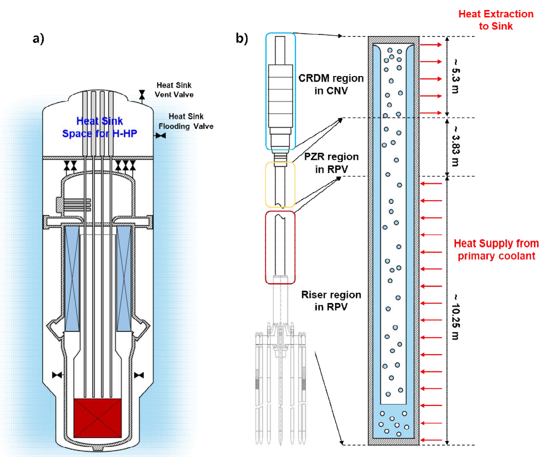

Fig. 3. Modified NuScale design with PINCs (a), and schematics of the hybrid heat pipe design for the small modular reactor (b).

The design modification of the NuScale was made to make the heat sink of the H-HP in natural circulation type i-SMR. The upper part of the CNV above the RPV is isolated by a horizontal plate as depicted in Fig. 3 (a). The lower parts of the CNV divided by horizontal plate are acts as ECCS systems while the upper part of the CNV acts as a heat sink for the H-HP. In normal operating conditions, the upper part of the CNV is remains vacuum to minimize the heat loss through the H-HP. The heat sink flooding valve and vent valve are attached to the upper part of the CNV to flood the cool water from the UHS under accidental conditions.

The SCH80 1-1/2” pipe was considered as a container of working fluid of the H-HP. The total number of H-HP in NuScale is 37 (corresponds to the number of NuScale fuel assembly). The H-HP is modeled as single pipe with equivalent cross section and hydraulic diameter. The CCFL model was applied inside the H-HP in MARS-KS calculation. The modified nodalization of the natural circulation type i-SMR reactor with H-HP for the MARS-KS code calculation is depicted in Fig. 4.

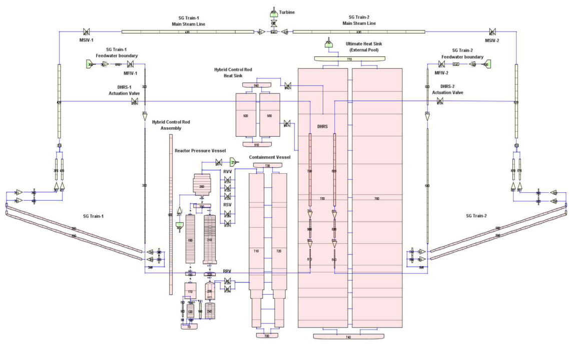

Fig. 4. Nodalization of modified NuScale reactor design with PINCs for the MARS-KS code.

3. Results and Discussion

3.1 Steady State Analysis Results

The steady-state analysis of natural circulation type i-SMR with and without H-HP was conducted by MARS-KS code. The calculation results are summarized in Table 1. The steady-state calculation results show a good agreement with the design value of the reference reactor (NuScale) [10] with a relative error of less than 1% except for the SG feedwater flow rate. Therefore, the further transient calculation was conducted based on the steady-state calculation results.

Table I: Summary of steady state analysis results of i-SMR.

Parameter	DCA [10]	NuScale (MARS-KS)	NuScale+H-HP (MARS-KS)	Error (%)
Reactor power [MW_{th}]	160.0	160.0	160.0	0.0
Core outlet T [K]	587.04	583.76	583.75	0.56
Core inlet T [K]	531.48	532.07	532.07	0.11
Coolant flowrate [kg/s]	587.15	587.46	587.47	0.05
Core bypass [kg/s]	42.86	42.46	42.46	0.95
PZR P [MPa]	12.76	12.76	12.76	0.0
SG feed T [K]	422.04	421.91	421.91	0.03
SG steam T [K]	547.8	572.69	572.68	0.37
SG steam P [MPa]	3.45	3.45	3.45	0.06
Feed flowrate [kg/s]	67.07	68.80	68.80	2.58
H-HP initial P [MPa]	-	-	9.89	-
H-HP heat loss [MW]	-	-	6.51E-3	-

3.2 SBO Accident Analysis Results with and without Hybrid Heat Pipe

The transient was initiated by loss of AC power at 0s. The feedwater flow rate rapidly decreased due to the loss of AC power after SBO initiation. The RPV pressure starts to increase due to the decreased heat removal rate by SG in both NuScale and modified NuScale design with H-HP. The reactor trip signal is generated at 17s after the SBO initiation in both cases. The reactor was tripped and shut down 2s after the trip signal generation. The ANS-73 decay heat curve model was applied to simulate the decay heat of the reactor core in the current study. The DHRS valve fully with the 30s of delay after trip signal generation in both cases. In the case of modified NuScale with H-HP, the heat sink flooding valves on the upper part of CNV opened simultaneously with DHRS valves.

After the opening of the DHRS valves, the decay heat removal system removes the heat from the primary system via SG. The H-HP starts to remove the heat from the primary system 750s after the opening of the heat sink flooding valve. The heat removal rate of the H-HP is less than the core decay heat level.

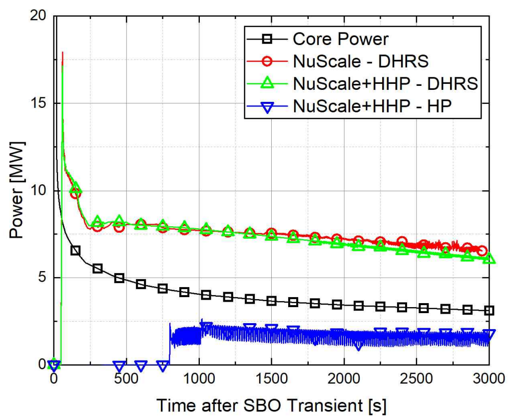

Fig. 5. Core decay heat and heat removal rate of DHRS and hybrid heat pipe under SBO accident.

The primary system pressure and the coolant temperature decrease after the actuation of DHRS as depicted in Fig. 6 and 7. The pressure level inside the H-HP is also decreased along with primary system pressure. After starting up of H-HP, the primary pressure and coolant temperature decrease rapidly compared to the original NuScale design. These analysis results show that H-HP can acts as effective means of rapid depressurizing of the primary system without losing the primary coolant.

The natural circulation flow rate inside the primary system in both NuScale and modified NuScale with H-HP are shown in Fig. 8. As shown in figure, starting up of H-HP dose not affect the primary coolant flow rate at very initial state of the SBO transient. The primary coolant flow rate of the modified NuScale with H-HP starts to increase after ~2,000s after transient. The enhanced natural circulation flow in modified design is attributed to the rapid depressurizing of the RPV due to the operation of H-HP. The phase change occurred in upper part of the RPV due to the rapid depressurization which increase the driving force of the natural circulation in modified NuScale design.

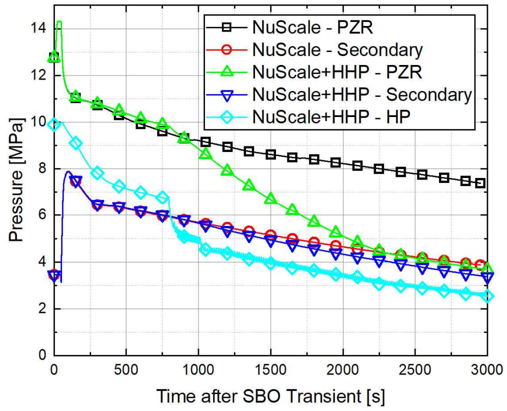

Fig. 6. Transient pressure behavior of NuScale and modified NuScale with hybrid heat pipe under SBO accident.

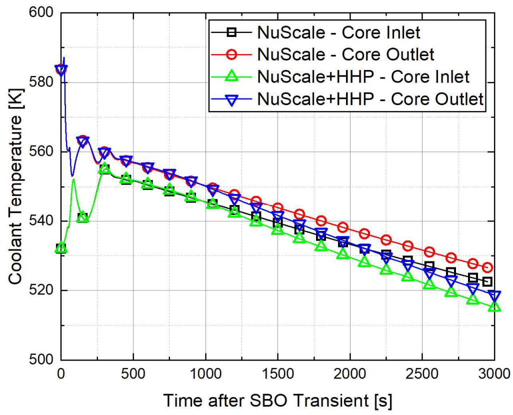

Fig. 7. Primary coolant temperature behavior of NuScale and modified NuScale with hybrid heat pipe under SBO accident.

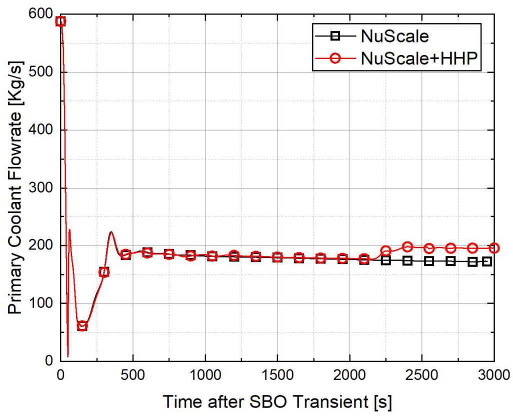

Fig. 8. Primary coolant temperature behavior of NuScale and modified NuScale with hybrid heat pipe under SBO accident.

3. Conclusions and Further Work

The feasibility and integral effect of the hybrid control rod concept on SMR was numerically investigated with MARS-KS code. The effect of the H-HP on the natural circulation type i-SMR under the initial stage of SBO transient was analyzed. The analysis results show that the application of the hybrid control rod can act as an effective means of rapid primary system depressurization without losing the coolant inventory. Also, the H-HP shows the moderate decay heat removal capability in the current study.

The long-term behavior of the H-HP on natural circulation type i-SMR will be further analyzed with a refined model based on the current study to investigate the safety margin of the proposed PINCs concept on the SMR design. The experimental integral effect test is required to validate and refine the safety analysis results with H-HP on SMR design.

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