INVESTIGATION OF NATURAL CIRCULATION IN PRIMARY SYSTEM OF NUSCALE-SMR BY USING RELAP5

PHAM TUAN NAM, VO THI HUONG

Institute for Nuclear Science and Technology, 179 Hoang Quoc Viet str., Hanoi.

Email: ptn2910@gmail.com

Abstract: The natural convection method is used in many advanced nuclear reactors, such as Generation IV and Small Modular Reactors to improve the safety and reliability. In order to investigate the natural circulation (NC) behavior in A NuScale Power Module Reactor, a numerical study on the NC in primary system for such kind of reactor was obtained by using RELAP5. The calculation was performed under steady state conditions. As the results, the behavior of pressure, coolant temperature, velocity, mass flow rate, density, power, heat flux under steady state condition were discussed. In comparison with the Final Safety Analysis Report of NuScale, the calculation results had the difference ranged from 0.0% to 5.9 %.

Keywords: Natural circulation, NuScale, RELAP5.

I. INTRODUCTION

Small Module Reactors (SMRs) are 4th generation reactors with up to 300 MWe power, in which components and systems can be manufactured and transported as modules to the sites for installation [1]. Most of SMR designs adopt advanced or inherent safety features from the advantages of conventional high-power reactors, such as flexible power generation for a wider range of users and applications, enhancing safety performance, and offering better economic affordability.

Currently there are more than 50 SMR designs from many countries and companies. They are being developed with different level of extensiveness, from under conceptual design, design certification process to under construction, in which, natural circulation is an developing trending, as listed in Table 1. With the NC feature, the designs are simplified by removing reactor coolant pump, tubes connect from reactor vessel to steam generators (SGs), excluding the loss of flow accident, become more safety.

RELAP5 is the most extensively used best estimate code for thermal-hydraulic safety analysis of light water reactors [2]. There are many researches focus to capability of RELAP in simulating the natural circulation in the passive safety systems and facilities [3]. Sabundjian et al. (2007) [4] studied the natural circulation phenomena in one and two-phase regimes using RELAP5. The comparison between theoretical and experimental results for one and two phase flows showed a good agreement. D'Auria et al. (1997) [5] discussed the RELAP5 capability in predicting instabilities in single-phase natural circulation. The results describe that the stability was strongly affected by the initial conditions and by the overall dynamics of the system, such as thickness of wall, heat losses to the environment, and modalities for supplying or removing power to the loop (heater or cooler). Misale et al. (1999) [6] reported the analysis of single-phase natural circulation experiments by RELAP5. A series of results were compared with experimental data gathered in the MTT-1 loop, and showed poor agreement. Lakshman and Pandey (2010) [7] have studied numerically the behavior in a parallel-channel natural circulation boiling water reactor under low-pressure low-power startup condition and compared with its scaled facility. The effect of flow resistance on oscillations was explored but for nonidentical condition, the flow fluctuation in the parallel-channel system were found to be out-of-phase. Martin et al. (1992) [8] carried out the benchmarking assessment of RELAP5/MOD3 for low flow and natural circulation experiments, the tests were conducted at system pressures of 14.7 and 17.0 psia. The analysis of the test yielded general agreement with experimental data in prediction of forced flow and natural circulation trends, however the discrepancies exist because of deficiencies in geometric, mul-dimensional and form loss effects.

Susyadi et al. (2016) [9] simulated the thermal-hydraulic behavior in NuScale-SMR design, but mainly focused on the qualitative behavior due to limited source of the data. The Application Documents for the NuScale-SMR Design [10] were submitted the latest version (Revision 3) to United States Nuclear Regulatory Commission in August, 2019. The data was used to verify the natural circulation simulation capability of RELAP5 in the NuScale's primary system.

Table 1. Water-cooled SMRs designs available for near and mid-term deployment [11]

Design	**Type, Power output (MWe), Designers**	**Status**
CAREM-25	Integral PWR, natural circulation, 27, CNEA, Argentina	One unit prototype under construction near Atucha-2 site
IMR	Integral modular PWR, natural circulation, 335, MHI, Japan	Conceptual design
ABV-6M	Integral PWR, natural circulation, 8.6, OKBM Afrikantov, Russian Federation (RF)	Detailed design
UNITHERM	Very Small, Integral PWR, with Natural Circulation, 2.5, RDIPE, RF	Conceptual design
NuScale	Integral natural circulation PWR, 570, NuScale Power, USA	Detailed design

II. NUSCALE-SMR DESIGN OVERVIEW

The NuScale-SMR plant incorporates several features that reduce complexity, improve safety, enhance operability, and reduce costs [12]. The design goals are to achieve a high level of safety and asset protection while providing an affable approach to nuclear power that gives the plant owner the maximum flexibility in construction, operation and application.

The NuScale SMR plant is scalable, such that from one (1) to twelve (12) NuScale Power Modules (NPM) operate within a single reactor building [10]. A NPM is a composed of reactor core, a pressurizer (PRZ) and two steam generators integrated within a reactor pressure vessel and housed in a compact steal containment vessel. A diagram of the NPM is shown in Figure 1. Thermal power of the reactor is 160 MWt, the nuclear core consisting of 37 fuel assemblies and 16 control rod clusters. The fuel assemblies contain a 17x17 array of zircalloy-clad, low-enriched UO2 fuel similar to traditional pressurized Light Water Reactors. The reactor vessel is approximately 64.8 ft tall and 9.6 ft in diameter. Above the core is a central hot riser tube, a helical coil steam generator surrounding the hot riser tube, and a pressurizer. The helical coil steam generator consists of 21 tube columns. Table 2 provides the main characteristics in a NPM. In this report, British-American System of Units were used to compare our simulation results with FSAR data.

In the primary system, the reactor coolant is heated by fission reaction in the core to produce the convection flow that is upward to riser. The heated coolant moves from core to upper plenum, then flows through the steam generator and is cooled down. The lower temperature and higher density coolant is drawn back to lower plenum by gravity, is heated again in the reactor core to have the natural circulation in primary system. The NuScale-SMR design is PWR type, the reactor coolant is kept separate from the secondary circuit. The heat of the primary circuit is transferred to secondary side via 1380 helical tubes in SGs. In the secondary side, feed water is pumped into the SG's tubes where it boils to generate superheated steam, and is circulated to the turbine-generator system.

The entire nuclear steam supply system is enclosed in a steel containment that is 80 ft tall and 15 ft in diameter. NuScale-SMR has two passive systems, Decay heat removal System (DHRS) and Emergency Core Cooling System (ECCS), which provide for safety performance by natural convection principles. If the normal feedwater system of the secondary circuit is not available, DHRS is able to remove decay heat for a minimum of 3 days without pumps or power. If DHRS is also not available, ECCS uses the reactor vent valves located on the sides of the reactor vessel, working in conjunction with the containment heat removal system to remove heat from the core [13].

Table 2. Thermal-hydraulic components for RELAP5 modeling [10]

Parameter	**Value (Britsh)**
Reactor core	
Diameter of active core	4.94 ft
Height of active core	6.57 ft
Core thermal output	160 MWt
System pressure	1850 psia
Heat transfer area on fuel surface	6275.6 ft^2
Core flow area	9.79 ft^2
Number of fuel assemblies	37
Steam generator	
Total number of helical tubes per NPM	1380
Steam flow (full power)	532100 lbm/hr
Tube wall outer diameter	0.625 inches
Tube wall thickness	0.050 inches
Total heat transfer area	17928 ft^2
Normal steam pressure	500 psia
Normal steam temperature	575◦F
Normal feedwater temperature	300◦F

The NuScale-SMR detail data is provided in the latest version (Revision 3) FSAR to United States Nuclear Regulatory Commission in August, 2019, that basic to simulate and verify the RELAP5.

III. RELAP5 NODILIZATION FOR NPM

Figure 2 shows RELAP5 nodilization scheme for primary system in NPM. The main components consist of Reactor core PIPE 200 (average channel) and PIPE 210 (hot channel), Riser 290, 300, and 310, Lower plenum 100, Upper plenum 400, Pressurizer 500 and 510, Primary side of steam generator 600, Downcomer 700; Secondary system included Feed water source 608; Secondary side of steam generator 610; and Steam sink 612; detail shows in Table 3 and Table 4.

Table 3. Hydraulic components and heat structures of NPM modeling.

Component number	Name	Type	Volumes
100	Lower Plenum	Branch	1
210	Core (Average channel)	Pipe	8
220	Core (Hot channel)	Pipe	8
290	Low riser	Branch	1
300	Middle Riser	Pipe	6
310	Upper Riser	Pipe	25
400	Upper Plenum	Branch	1
500	Pressurizer	Branch	1
510		Time Dependent Volume	1
600	Steam Generator (Primary side)	Pipe	15
610	Steam Generator (Secondary side)	Pipe	15
700	Down comer	Pipe	20
608	Feed water Source	Time Dependent Volume	1
612	Sink	Time Dependent Volume	1
10	Fuel Rods (Average channel)	Heat Structure	8
11	Fuel Rods (Hot channel)	Heat structure	8
20	SG - Helical tubes	Heat Structure	15

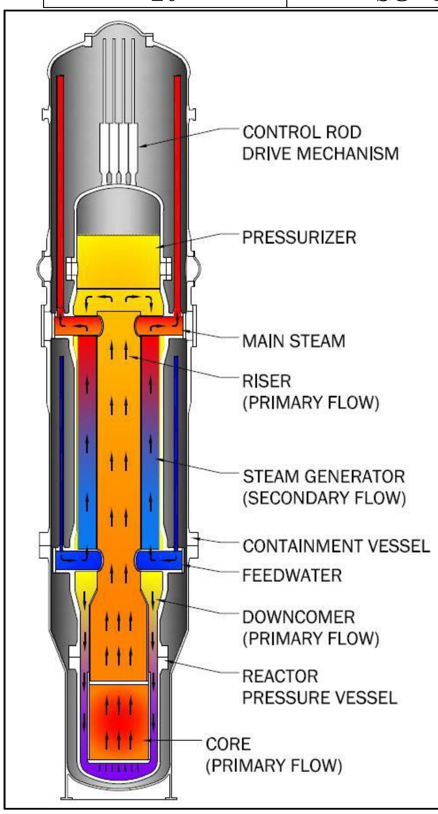


Figure 1. Schematic of a NuScale power module [10]

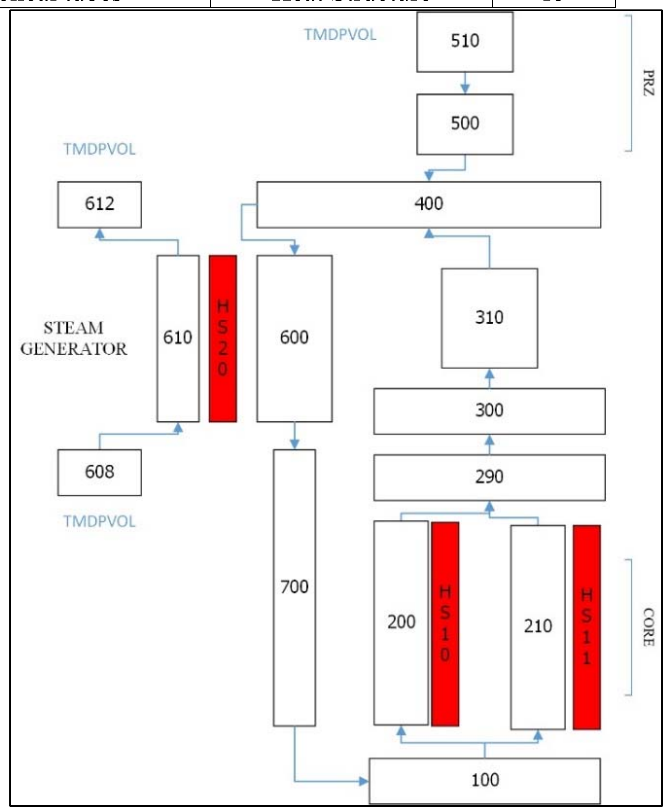


Figure 2. Nodalization of NPM thermal hydraulic system in RELAP5

Reactor core

In normal condition, NPM power is 160 MWt in 37 fuel assemblies, in which a fuel assembly with 1.5 power coefficient is hot channel (PIPE 210) and 36 fuel assemblies is average channel (PIPE 200), correspond to Heat Structure 11 (HS11) and 10 (HS10). The heat structure materials are UO_2 and fuel cladding, total heat transfer area on fuel surface is 6275.6 ft^2. The coolant from the hot and average channels is mixed in BRANCH 290, and convective up to RISER 300 and 310. The primary system pressure is remained at 1850 psia by pressurizer 510 and 500. The reactor core consists 8 nodes in axially, the axial power distribution is showed in Figure 3 from FSAR data.

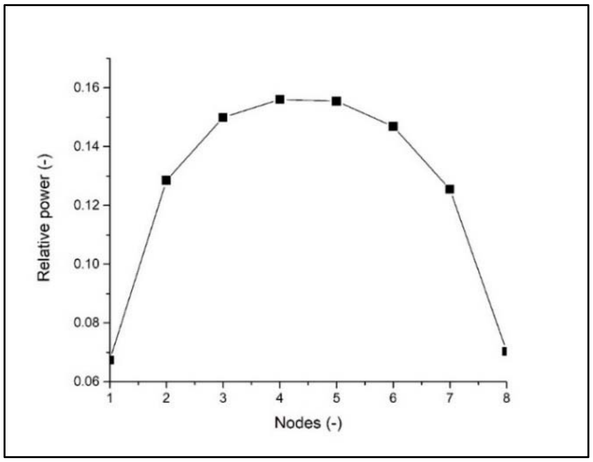


Figure 3. Axial relative power distribution in RELAP5 simulating [10]

Steam generator

The heat from reactor core is removed by Steam Generator, the heat balance is carried by injecting of the feed water from Time-dependent Volume (TMDPVOL) 608 (500 psia, 300 ^{o}F) to SG, the mass flow rate is kept at 147.9 lbm/s by the Time-dependent Junction (TMDPJUN) 609. The heat transfer between primary and secondary sides is simulated by HS20, total heat transfer area is 17928 ft^2, the SG tubes are fabricated with SB-163 Alloy 690 (UNS N06690) [10].

Table 4. Geometry data of the NuScale-SMR primary system [10]

Component	Total volume (ft^3)	Sub-Component	Average Flow Area (ft^2)	Length (ft)
Riser	635	Lower riser	24.9	9.4
		Upper riser	15.4	26.0
Downcomer	1199	Downcomer and SG	25.7	46.0
Core	89	Fuel assemblies	10.3	7.9
		Reflector	0.9	7.9
Pressurizer	578	Main steam plenum	36.1	1.7
		Cylin. pressurizer	61.4	6.9
		Pressure vessel head	41.2	2.2

IV. RESULT AND DISCUSSION

The NuScale's thermal-hydraulic behavior was investigated in 4600 seconds by RELAP5, NPM modeling reaches to steady-state, this means the calculated parameter change is negligible. The initial and boundary conditions was created from the FSAR data. The most important calculated parameters are pressure, temperature, void fraction, and mass flow rate, these parameters will be discussed in this section to assess the NPM modeling.

IV.1 The reactor core

Total reactor power is 160 MW, he coolant temperature increases from 502^{o}F to 600^{o}F when move through the core (Figure 4) with the total mass flow rate is 1218 lbm/s (Figure 6). PRZ sets at 1850 psia in the top of NPM, the core pressure is higher to make the coolant flow, as show in Figure 7. The pressure difference between the core and PRZ is 15 psia.

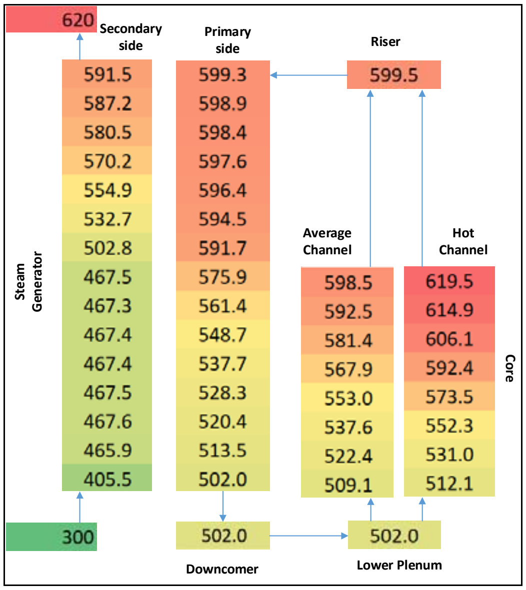


Figure 4. Temperature distribution in steady-state

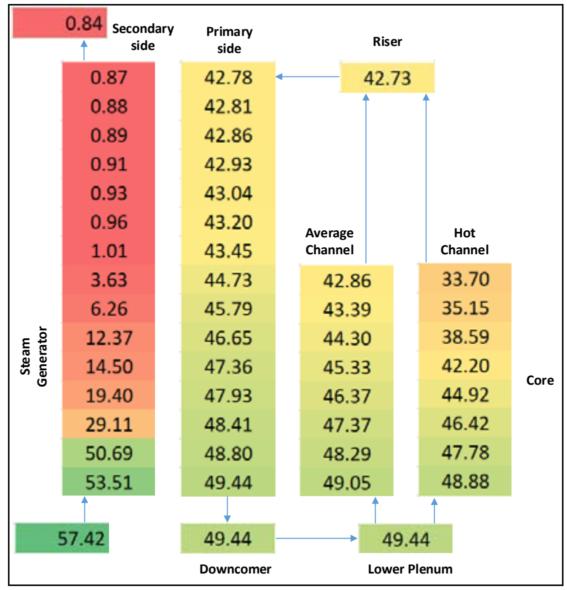


Figure 5. Density distribution in steady-state

The coolant velocity increases along the core length from 2.52 ft/s to 2.88 ft/s corresponding to the core inlet and outlet (Figure 8), average coolant velocity is about 2.7 ft/s. The heat fluxes are 44.7 and 29.8 Btu/s.ft^2 in the node 4 of the hot and average channels (Figure 9) because the hot channel factor is 1.5.

The density difference is the driving force in natural circulation and leads to the movement of the reactor coolant in primary system. The coolant density decreases from the core inlet (49.44 lb/ft^3) to the core outlet (42.73 lb/ft^3), as show in Figure 5. RELAP5 results simulated the natural convection phenomena in the NPM reactor core, the parameter difference will be compared with FSAR data in the following section.

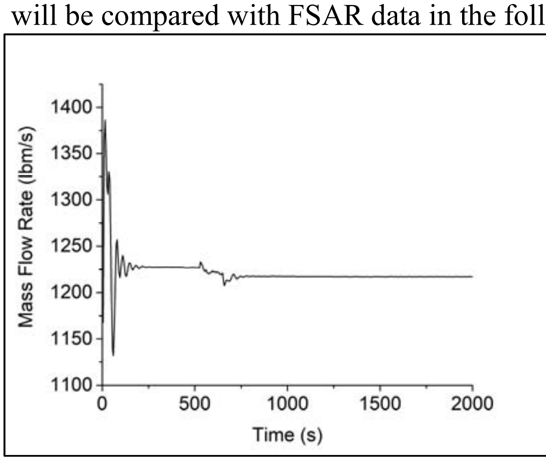


Figure 6. The coolant mass flow rate in the core.

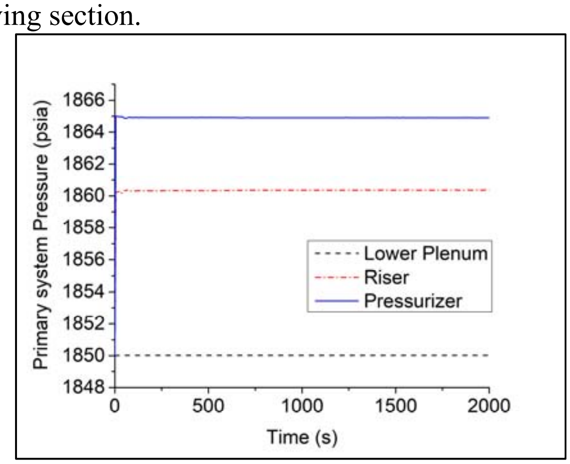


Figure 7. Pressure in the primary system.

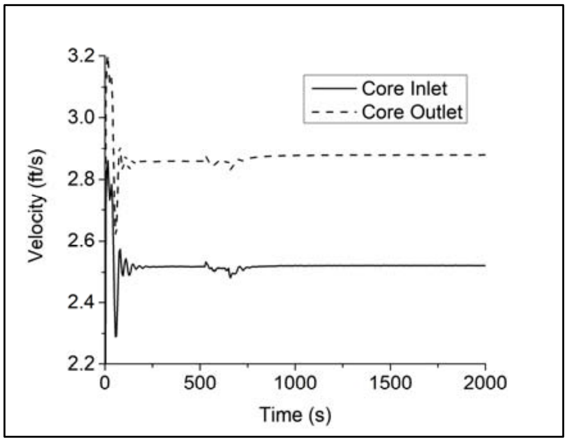


Figure 8. The coolant velocity in the core.

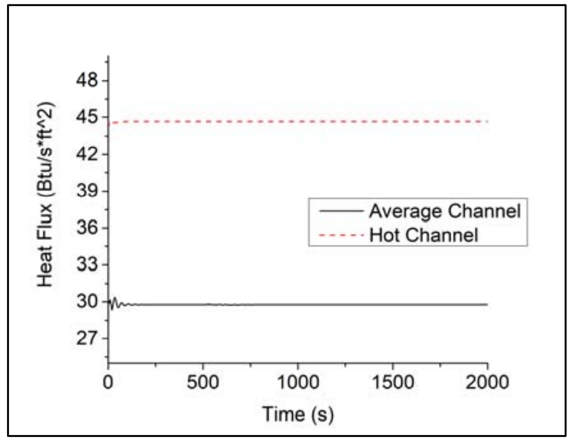
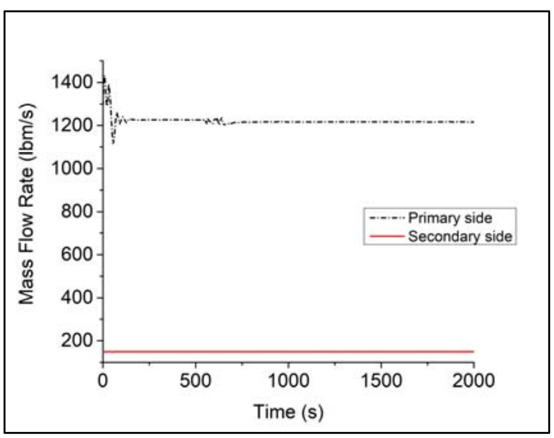


Figure 9. Heat flux between fuel and coolant

IV.2 Steam generator

The heat is removed from the primary side (PIPE 600 – Left boundary) to the secondary side (PIPE 610 – Right boundary) by Heat Structure 20 in steam generator. The feed water flow rate is constants of 148.5 lbm/s by TMDPJUN 609 (Figure 10), SG operates in the pressure of 1850 psia and 500 psia correspond to the primary and secondary sides (Figure 12). The reactor coolant temperature decreases from 599.3 ^{o}F to 502.0 ^{o}F (Figure 4), while the feed water temperature increase and produce the vapor in the SG secondary side. The heat flux in the middle of SG is 22.4 Btu/s.ft^{2}, balance between the left and right boundaries, as show in Figure 11. The right boundary heat flux fluctuates more strongly than the left boundary because the steam is generated in the secondary side while there is only one phase flow in the primary system.

The void fraction increases in the SG's secondary side (PIPE 610), reaches to steady state at 1000 second (Figure 11). As showed in Figure 13, steam occurs at node 3 when the void fraction is larger than zero; in the following nodes, the void fraction parameter increases and reaches to the maximum value (1.0) at node 9. From node 9, the entire fluid transfers from water to steam in the control volume (CV) and the steam temperature increases until the end of SG heat transfer tube (Figure 4).



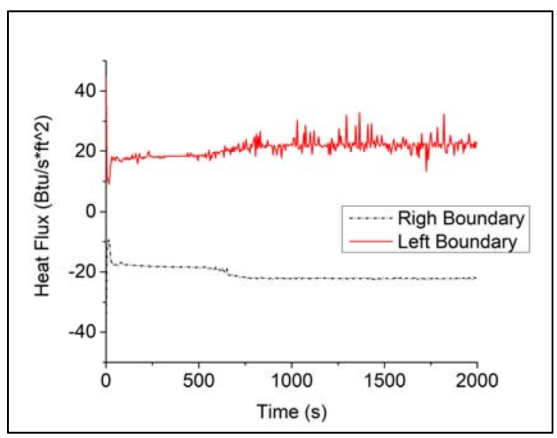


Figure 10. The mass flow rates in the primary and secondary sides.

Figure 11. Heat flux in the middle of SG.

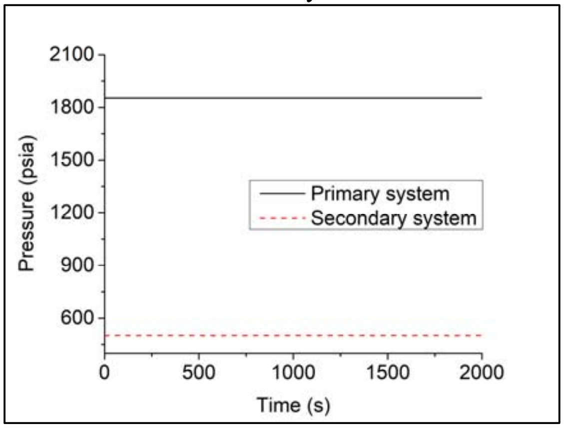


Figure 12. The pressure difference between the secondary and primary sides.

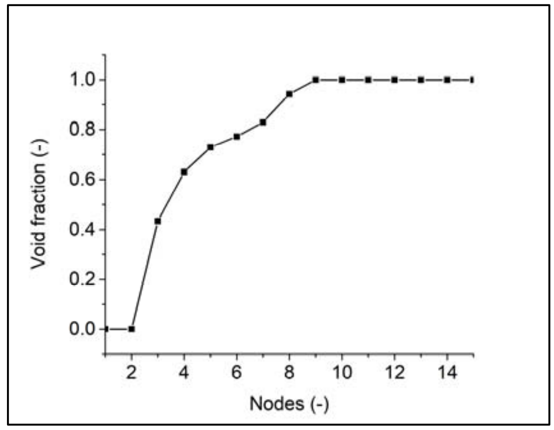


Figure 13. Void fraction in the SG's secondary side.

IV.3 Discussion

In general, the natural circulation in the NPM primary system was simulated by RELAP5, the driving force is density changes in the core heating and SG heat-removing processes. The natural circulation is reach to the steady-state in the primary, and the secondary removed the heat from the reactor core, ensure the energy balance in the thermal hydraulic systems. For quantitative evaluation, the most important parameters are showed and compared with FSAR data in Table 5. The total power and pressure in the primary system is set as input of the modeling, hence there are no differences in the parameters.

Table 5. The parameter comparisons between simulation results and FSAR

Parameters	**RELAP5**	**Design value [10]**	**%**
Core			
Power, MW	160	160	0.0
SG Pressure, psia	1850	1850	0.0
Inlet temperature, ^oF	502	497	1.0
Outlet temperature, ^oF	600	597	0.5
Coolant velocity, ft/s	2.70	2.70	0.0
Core inlet velocity, ft/s	2.52		
Core outlet velocity, ft/s	2.88		
Mass flow rate, lbm/hr	4.38E+06	4.66E+06	5.9
SG's secondary side			
Pressure, psia	500.0	500.1	0.0
Inlet temperature, ^oF	300.0	299.7	0.1
Outlet temperature, ^oF	591.6	584.4	1.2
Mass flow rate, lbm/s	148.5	147.9	0.4

The predicted core inlet temperature is 5^oF higher, core outlet temperature is 3^oF higher, and SG outlet temperature is 7.2^oF higher; these differences are smaller than 2%. The core inlet and outlet temperatures are 2.52 ft/s and 2.88 ft/s, it is suitable for the average velocity in the reactor core in FSAR. The biggest difference is in mass flow rate in the core (5.9%), because the complex core geometry is difficult to set the thermal hydraulic parameters, such as the friction coefficients; the heat transfer hydraulic diameter in heat structures (4*flow area/heated perimeter); and the hydro diameter in hydrolic components.

5. CONCLUSIONS

The NPM primary system is modeled using RELAP5 computer code, the results identify that: RELAP5 code is able to simulate the natural circulation in NPM primary system, the steady-state condition is obtained; The thermal hydraulic simulated values are suitable to the

FSAR data, the differences are acceptable from 0% to 1.3%, only mass flow rate of 5.9%; all differences are less than 10%.

This result is the basis to verify the capability of RELAP5 in the modeling and analysis for the new SMRs that use the natural circulation method, such as NuScale-SMR technology. The future works will complete the safety systems, such as Decay heat removal system or Emergency core cooling system, to modeling and analysis the NPM in the transient and accident conditions.

REFERENCES

[1] IAEA, Advances in small modullar reactor technology developments, 2018 Edition.

[2] G. L. Mesina, A History of RELAP Computer Codes, Nuclear Science and Engineering, vol. 182, v – ix, January, 2016.

[3] Amit Mangal et al., Capability of relap5 code to simulate natural circulation behavior in test facilities, Progress in Nuclear Energy 61 (2012) 1-16.

[4] Sabundjian, G., et al, 2007. RELAP5 simulation for one and two phase natural circulation phenomenon. In: International Nuclear Atlantic Conference-INAC.

[4] G. Sabundjian et al., A RELAP5 study to indentify flow regime in natural circulation phenomenon, International Nuclear Atlantic Conference - INAC 2011.

[5] D' Auria et al., System codes capabilities in predicting instabilities in single phase natural circulation, Proceeding of 4th Regional Meeting Nuclear Energy in Cetral Europe, 1997.

[6] Mario Misale et al., Analysis of single-phase natural circulation experiments by system codes, Int. J. Therm. Sci. (1999) 38, 977–983.

[7] S. P. Lakshmanan and Manmohan Pandey, Numerical Investigation of Startup Instabilities in Parallel-Channel Natural Circulation Boiling Systems, Hindawi Publishing Corporation, Science and Technology of Nuclear Installations Volume 2010, Article ID 574195, 8 pages.

[8] Martin, R.P., Taylor, B.K . Benchmarking assessment of RELAP5/MOD3 for the low flow and natural circulation experiment. The international topical meeting on nuclear reactor thermal-hydraulics (NURETH-5). Salt Lake City, UT, United States, 21-24. 1992.

[9] Susyadi et al., Thermal-Hydraulic Analysis of SMR With Naturally Circulating Primary System During Loss of Feed.

[10] NuScale Power LLC, NuScale Standard Plant Design Certification Application, revision 3, 2019.

[11] H. Hidayatullah et al., Design and technology development for small modular reactors - Safety expectations, prospects and impediments of their deployment, Progress in Nuclear Energy 79 (2015) 127-135.

[12] D. T. Ingersoll et al. NuScale Energy Supply for Oil Recovery and Refining Applications, Proceedings of ICAPP 2014, Paper 14337.

[13] JOSÉ N. REYES, Jr., NuScale plant safety in response to exxtreme events, Nuclear Technology, 178:2, 2012, 153-163.

KHẢO SÁT VÒNG TUẦN HOÀN ĐỐI LƯU TỰ NHIÊN TRONG HỆ SƠ CẤP LÒ PHẢN ỨNG NUSCALE-SMR BẰNG RELAP5

PHẠM TUẤN NAM, VÕ THỊ HƯƠNG

Viện Khoa học và Kỹ thuật hạt nhân, 179 Hoàng Quốc Việt, Hà Nội

Email: ptn2910@gmail.com

Tóm tắt: Nguyên lý đối lưu tự nhiên được sử dụng trong nhiều lò phản ứng cải tiến, như công nghệ thế hệ IV và lò công suất nhỏ dạng mô-đun, nhằm nâng cao tính an toàn và độ tin cậy. Để khảo sát diễn biến của vòng tuần hoàn đối lưu tự nhiên (NC) trong lò phản ứng NuScale, nghiên cứu này thực hiện mô phỏng NC trong hệ sơ cấp bằng chương trình tính toán RELAP5. Tính toán được thực hiện trong trạng thái vận hành ổn định. Các thông số quan trọng như áp suất, nhiệt độ chất làm mát, vận tốc, tốc độ dòng khối, công suất, thông lượng nhiệt từ kết quả mô phỏng được phân tích và so sánh với Báo cáo Phân tích An toàn của NuScale, chênh lệch nằm trong khoảng từ 0,0 đến 5,9%.

Từ khóa: Đối lưu tự nhiên, NuScale, RELAP5.