# SIMULATION OF FEED WATER TEMPERATURE DECREASE ACCIDENT IN NUSCALE REACTOR

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## ABSTRACT

**SIMULATION OF FEED WATER TEMPERATURE DECREASE ACCIDENT IN NUSCALE REACTOR**. Study on thermal hydraulic behavior of the NuScale reactor during secondary system malfunction that causes a feed water temperature decrease has been conducted using RELAP5 code. This study is necessary to investigate the performance of safety system and design in dealing with an accident. The method used involves simulation of reactor transient through numerical modeling and calculation in RELAP5 code covering primary and secondary system, including the decay heat removal system (DHRS). The investigation focuses on the flow and heat transfer characteristics that occurs during the transient. The calculation result shows that at the beginning, core power increases up to trip set point of 200 MW which is driven by positive feedback reactivity of coolant overcooling and automatic control rod bank adjustment. Meanwhile, the core exit coolant temperature increases up to 600 K. and primary system circulation flow rate speeds up to 556 kg/s. After that, the reactor trips and power drops sharply, followed by opening of DHRS valves and closing of steam line and feed water isolation valves. The simulation shows that, the DHRS are capable to transfer decay heat to the reactor pool and as a result the primary system temperature and pressure decreases. The reactor could stay in safe shutdown state afterward.

Keywords: NuScale, RELAP5, feed water, decay heat, simulation

## *ABSTRAK*

***SIMULASI KECELAKAAN PENURUNAN TEMPERATUR AIR UMPAN DI REACTOR NUSCALE**. Studi tentang perilaku termalhidraulik reaktor NuScale saat terjadi kerusakan sistem sekunder yang menyebabkan penurunan suhu air umpan telah dilakukan dengan menggunakan kode RELAP5. Penelitian ini penting untuk menyelidiki kinerja disain dan sistem keselamatan reaktor dalam menghadapi kecelakaan. Metoda yang digunakan melibatkan simulasi transien reaktor melalui pemodelan dan kalkulasi numerik dengan RELAP5 yang meliputi sistem primer dan sekunder serta sistem pembuangan panas peluruhan (DHRS). Investigasi berfokus pada aliran dan karakteristik perpindahan panas yang terjadi selama transien. Hasil perhitungan menunjukkan bahwa pada awalnya, terjadi peningkatan daya teras hingga mencapai titik seting pemadaman (trip) 200 MW, sebagai akibat dari umpan balik reaktivitas positif dari pendinginan fluida sistem primar dan respon otomatis penaikan batang kendali. Sementara itu, suhu keluaran teras meningkat menjadi 600 K serta laju aliran sirkulasi sistem primer meningkat menjadi 556 kg/s. Setelah itu, reaktor padam dimana daya menurun tajam dan diikuti pembukaan katup DHRS dan penutupan katup pada jalur uap dan air umpan. Simulasi ini menunjukkan bahwa, DHRS mampu membuang panas ke kolam reaktor, dimana suhu serta tekanan sistem primer menurun. Reaktor tetap dalam keadaan shutdown aman sesudahnya.*

*Kata kunci: NuScale, RELAP5, air umpan, panas peluruhan, simulasi*

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## INTRODUCTION

The NuScale reactor is an innovative small modular reactor having thermal power rating about 160 MWt (~50 MWe). Its design is being under review by the US Nuclear Regulatory Commission[1]. As new comer in nuclear industry, this reactor has been designed with unique features that contrast with most existing PWR designs. Among the distinguishing features are; (1) Integrated steam supply system where the core, steam generator and pressurizer, is placed inside a single compact pressure vessel, (2) The use of gravity force for naturally circulating the primary system coolant or in other words, the system does not need reactor coolant pump (RCP) to circulate the coolant and (3) The use of vacuumed and relatively small sized metal containment system. With all these features, it is believed that the reactor would have more superior safety performance compared to conventional reactors design, such as a lower probability of loss of coolant accident, elimination of loss of flow accident and more reliable on core cooling system in the case of severe accident [2, 3].

As consequence of a new design approach, the reactor's response in dealing with an incident or accident that may happen during its lifetime is also improved. For example, in conventional reactor design, when there is a malfunction in the feed water system, either a decrease in feed water temperature or a failure in feed water pump, the flow rate in primary system does not change because the RCP keeps working. However, in this new design any change in feed water condition (temperature or flow rate) would directly affect the flow rate in primary system. For this reason, analysis on this event is needed to understand behavior of the reactor during this transient and its implication to reactor safety. Currently, for the case of feed water temperature decrease event, work from independent analysts on this medium frequency event has not been performed. Publications related to this reactor mostly discussed the superiority of the design in dealing with severe accident and extreme event [3, 4]. Besides, the response of containment system using simplified model has also been investigated [5].

In order to perform analysis, RELAP5 code is used to numerically simulate the system response. This computer program is a general tool that permits users to model the combined behavior of the reactor coolant system and the core for many operational transients and postulated accidents that might occur in a nuclear reactor. RELAP5 (Reactor Excursion and Leak Analysis Program) can be used for reactor safety analysis and design, operators simulator training, and as an educational tool by universities [6-9]. BATAN has also been using this tool for several reactor safety analyses, both for research reactors and power reactors [10-13] . In the method, the NuScale reactor system and component is modeled using generic models available in the code, which then executed for transient simulation.

Although this work is very closely related to the safety analysis of the reactor being investigated, this research is mainly focused on the qualitative behavior during the transient, as some of the calculation parameter is generated using approximation from similar type water cooled reactor due to limited source of the data. By knowing the behavior of this reactor, better insight on the characteristic of this distinctive reactor design will be achieved.

## THE NUSCALE REACTOR DESCRIPTION

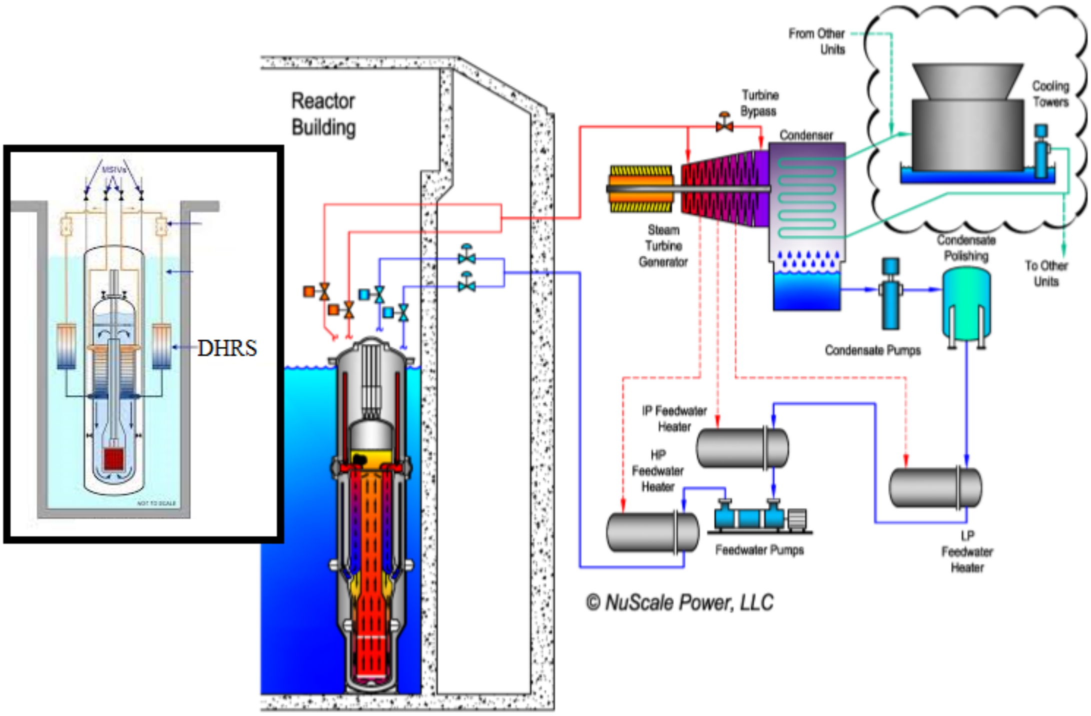


Figure 1. Schematic of a single NuScale power module and associated secondary equipment [14].

The NuScale is an integral pressurized water reactor (iPWR) that is designed to produce thermal power of 160 MWt and electrical output of 50 MWe. Component integration of the NuScale design is reflected in the simplification of equipment and elimination of some traditional key equipment. For example, reactor coolant pumps are eliminated in the design so the nuclear core is cooled entirely by natural circulation during normal operation as well as transient or accident conditions. In the vessel, the nuclear core heats up water to produce a low density fluid that moves upward through the hot leg riser. The density difference acting over an elevation difference between hot leg and cold leg results in a buoyancy force that drives the fluid flow around the primary loop. A once-through helical-coil typed steam generator is used to generate superheated steam and placed in the annular space between the hot leg riser and the reactor vessel's inside wall. Feed water enters the tubes at the bottom and superheated steam exits at the top. Two redundant, independent arrays of steam generator tube banks reside in the steam generator region. The NuScale design implements well-established LWR technology, including a standard LWR fuel in 17 × 17 configuration with enrichment of 4.95% $^{235}U$, proven codes and methods, and existing regulatory standards. Besides, it is supported by a one-third scale, electrically heated integral test facility which operates at full pressure and temperature[15, 16] , which will support the licensing within the existing LWR regulatory framework.

Each NuScale module has two redundant passive safety systems to allow decay heat to reach final heat sink of the containment pool, i.e. the decay heat removal system (DHRS) and the containment heat removal system (CHRS). These two systems do not need external power for actuation. To essentially diminish radiation release, NuScale has seven level of defense between fuel and environment. They are fuel pellet and cladding, reactor vessel, containment as in conventional nuclear plants, and water in reactor pool, stainless steel lined concrete reactor pool, biological shield covers each reactor, and reactor building. Schematic of NuScale power module is shown in Figure 1.

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## METHODOLOGY: RELAP5 MODELING

Based on the system shown in Figure 1, a calculation model is developed in RELAP5 as shown in Figure 2. It consists of several generic models available in RELAP5 which is arranged so that representing the NuScale reactor system. From the figure, it can be seen that there are three models of heat structure (see red color). On the left is the model for reactor core. In this part, the core that consists of 48 of 17x17-pin fuel bundles are represented into two channels, i.e average channel and bypass channel. The average channel includes the heat source of and removal from the fuel element while the bypass channel allow about 8% of the total core mass flow unheated [14]. In this model, decay heat schema of ANS73 standard fission product is used. For values related with feedback reactivity is approximated by referring to Tsuruga PWR-4 Loop (NSAC-91) as shown Table 1 and Table 2. In the middle, the heat structure model represents two circular steam generator modules which are interfacing the primary and secondary systems. On the right, the heat structure represents the decay heat removal system (DHRS). This system is crossing point of the secondary side with reactor pool. The system is connected to the main steam line and feed water line. This will be activated by opening the valves.

Table 1. Density reactivity table.

| Density (kg/m3) | Reactivity (dollars) |
|---|---|
| 500.0 | -6.3208 |
| 550.0 | -3.4818 |
| 600.0 | -2.1083 |
| 650.0 | -0.8764 |
| 699.5 | 0.0 |
| 700.0 | 0.0069 |
| 800.0 | 1.1042 |

Table 2. Doppler reactivity table.

| Temperature (K) | Reactivity (dollars) | Temperature(K) | Reactivity (dollars) |
|---|---|---|---|
| 273.15 | 2.55164 | 737.74 | 1.33550 |
| 505.03 | 2.55164 | 762.24 | 1.22245 |
| 534.91 | 2.39767 | 799.20 | 1.04013 |
| 552.79 | 2.29809 | 831.01 | 0.898686 |
| 570.21 | 2.21553 | 866.57 | 0.733490 |
| 588.07 | 2.10533 | 901.43 | 0.574738 |
| 617.49 | 1.95567 | 935.82 | 0.418159 |
| 640.77 | 1.82365 | 975.41 | 0.244033 |
| 661.79 | 1.72582 | 1017.6 | 0.0696194 |
| 693.01 | 1.55264 | 1032.4 | 0.00644376 |
| 715.39 | 1.45255 | 1082.6 | -0.213415 |

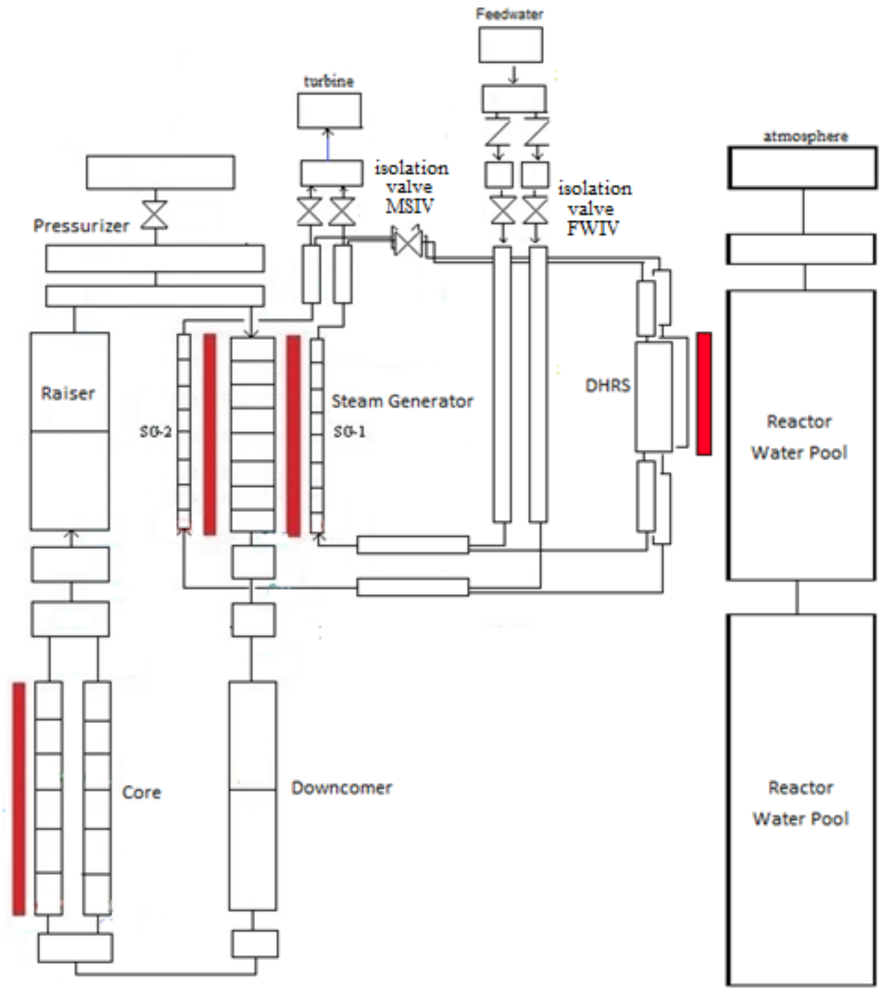


Figure 2. RELAP5 Nodalisation of NuScale Reactor System.

## Event Scenario

Several events are expected to occur with moderate frequency during plant operation. One of them is feedwater system malfunction that results in a decrease in feedwater temperature. In this analysis, the temperature is assumed to drops from 422.15 K into 373.15 K within 160 seconds [14]. This decrease leads to a reduction in moderator temperature that causes positive reactivity feedback, and also automatic control rod bank adjustment, bringing the power reaches trip set point. When trip occur, DHRS valves would open and main steam line and feed water isolation valves ( MSIV and FWIV) would close. For this event scenario the reactor is assumed to run at steady state nominal power level with 2% additional for uncertainty or about 162 MWt.

## RESULT AND DISCUSSION

In performing this analysis of a decrease in feedwater temperature, the RELAP5 input deck representing NuScale integral pressurized water cooled reactor system is firstly executed for steady state condition. In this steady calculation the fuel and moderator feedback reactivity is taken into account. Some adjustments were needed to get acceptable result, such as fouling parameter for the input of heat structure of the steam generator and a proper initial value of the input deck. The result is shown in Table 3. For benchmarking purpose, data from reference is also presented. It can be seen from the table that the calculation result from RELAP5 modeling used is relatively close with data from the reference, meaning that the model can be used for transient calculation.

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Table 3. Steady state calculation results.

| Parameter | Unit | Calculation | Reference[14] |
|---|---|---|---|
| Power | MWt | 160 | 160 |
| Reactor pressure | MPa | 12.75 | 12.755 |
| Coolant flow rate | kg/s | 536 | Range of 535 - 670 |
| Core Inlet tempereature | K | 538 | 535 |
| Core outlet temperature | K | 594 | 592 |
| Feedwater temperature | K | 422.15 | 422.15 |
| Fedwater flow rate | kg/s | 68 | 68 |
| Secondary side pressure | MPa | 3.5 | 3.5 |
| Reactor pool temperature | K | 305 | Range of (305- 322) |

**Transient Analysis**

The simulation of malfunction of feedwater system causing feed water temperature decrease was started at t= 100 seconds where the feedwater temperature entering steam generator drops within 160 seconds. The result of this calculation can be seen in Figure 3 to Figure 5. Figure 3 shows temperature and pressure in the system, Figure 4 presents reactor power and coolant flow rates, and Figure 5 displays heat transfer occurring in the core, steam generator and DHRS.

In Figure 3, it can be seen that the temperature of feed water at steam generator inlet drops from 422.15 K to 373.15 K. In conjunction with that, reactor power increase from 162 MW to 200 MW (see Figure 4). This power increase occurs due to two factors. Firstly, it is as a result of feedback reactivity from the water moderator and fuel. When the coolant temperature decreases, the density increases so that capability to moderate neutron from fast to thermal is increasing which is equivalence with a positive reactivity insertion to the core. The second factor is due to automatic respond from control rod in compensating the coolant temperature drop. The control rod performs adjustment to put the temperature back to normal so it inserts reactivity to increase power. These two conditions are taken into account in the calculation modeling. As a result, the power is increasing to reach the level of trip set point. For the other parameters, the core inlet temperature decreases to 530 K at the early stage of transient and then fluctuates at value around (535-537) K at the later stage. Meanwhile the core outlet temperature decreases at early stage to 592 K and then increase to 600 K.

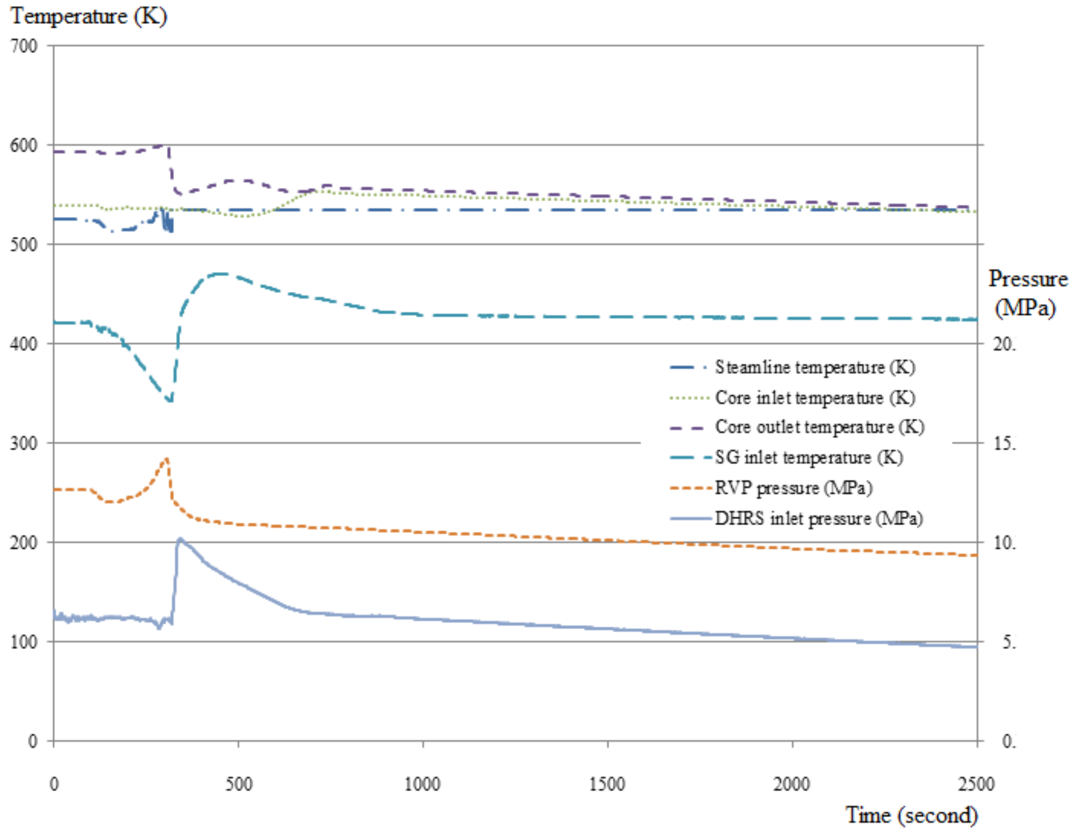


Figure 3. Temperature and pressure in the system.

As the primary system coolant circulates naturally, or without pump driving, the change of power and temperature would change its flow rate. The profile of primary system flow can be seen in Figure 4. The figure shows that at the beginning the flow increase from 535 kg/s to about 556 kg/s. Meanwhile, at the secondary side, the steam temperature is also affected by the decrease of feed water temperature. The figure shows that the steam temperature fluctuated from 526 K to 514 K and then back to increase again. The chart trend is similar with the temperature in the primary system.

When power reaches trip set point of 200 MW within 203 seconds after the initiation, the reactor is tripped. The power then decrease drastically to 16 MW in 28 seconds, which is keep decreasing gradually afterward. Following that, all thermal hydraulic parameter changed accordingly. In primary side, the core outlet drops from 600 K into 552 K and the core inlet temperature from 537 K into 530 K with different time span. These two temperatures, determine the density between cold-leg and hot-leg of the primary system which affects the natural circulation flow. The flow rate profile as shown in Figure 4 drops from 556 kg/s to about 45 kg/s at the lowest. At t = 425 seconds, the flow starts back to increase again to 204 kg/s following the temperature condition between hot-leg and cold-leg. After about 550 second from the event initiation, the temperature of primary coolant tend to keep decreasing consistently which shows that the system is going to reach a stable shutdown.

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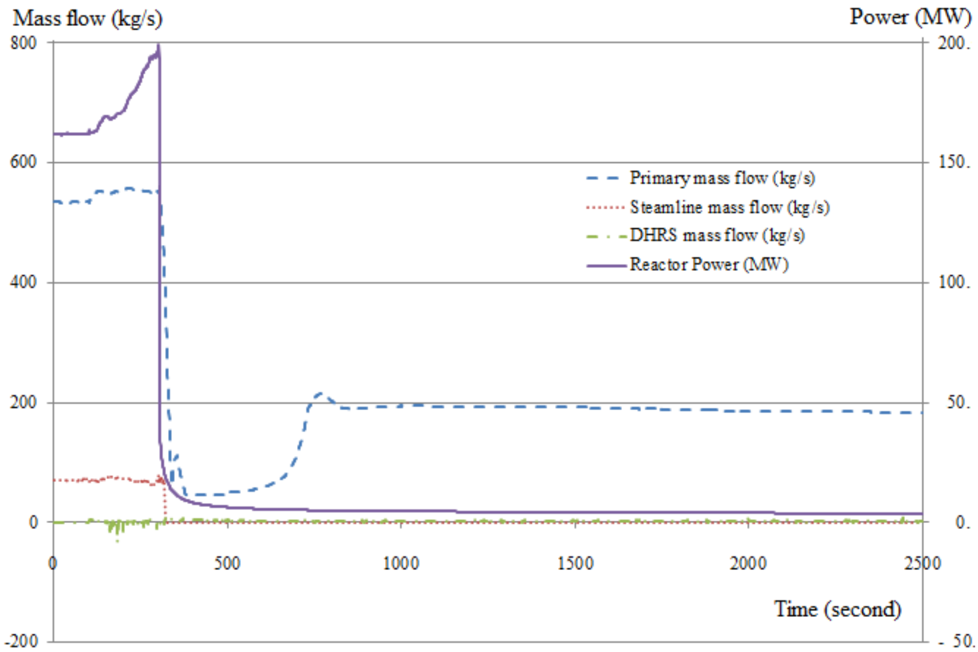


Figure 4. Reactor power and flow rate in the system.

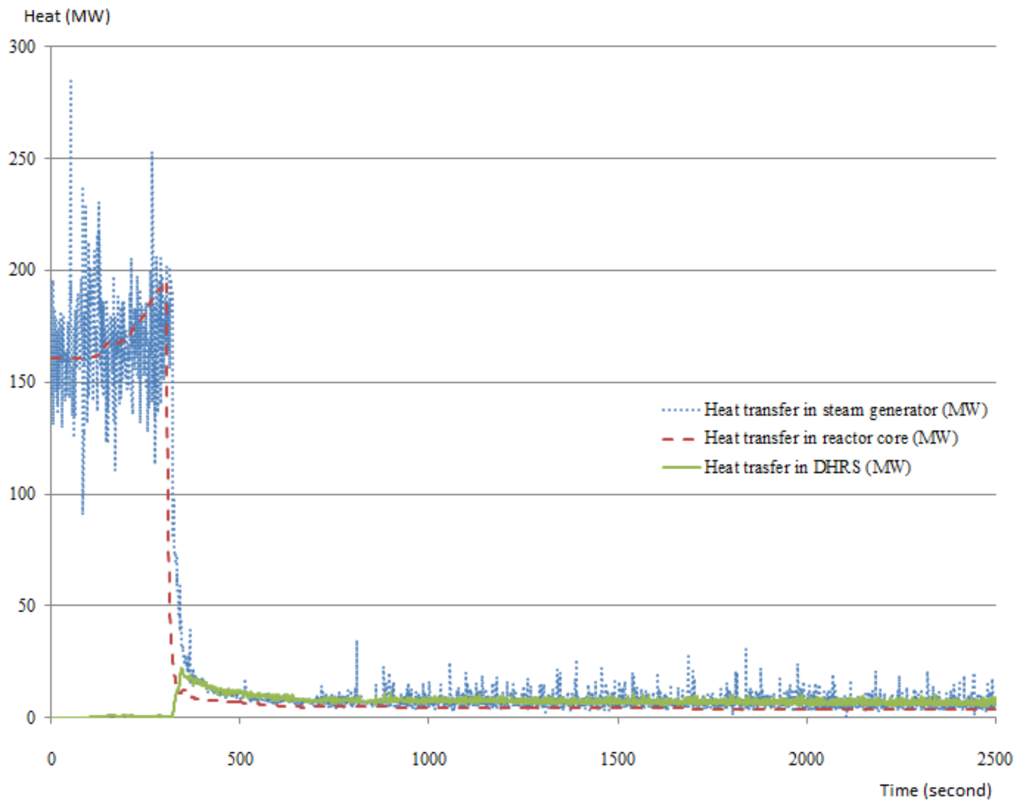


Figure 5. Heat transfer in the core, steam generator and DHRS.

On the secondary side, after the trip the DHRS valves are opened, followed by closing of feed water and steam line isolation valves (FWIVs and MSIVs), which will stop feed water supply and isolate the system from the turbine. The opening of DHRS valves allows steam to go into decay heat removal system, heat exchanger units that are submerged in the reactor pool and installed on the outside wall of the containment vessel. This DHRS system will condense the steam produced by decay heat and transfer the heat to the reactor pool. The condensate is then redirected to enter the steam generator. This natural circulation will maintain continuous heat removal and keep cool the reactor. The heat transfer profile in the steam generator and DHRS can be seen in the Figure 5. The figure shows that at the beginning the heat transfer in steam generator fluctuates around 162 MW. The fluctuation is due to two phase flow occurs inside steam generator tube. After reactor trip, the heat transfer in the steam generator drops sharply, following the trend of reactor power. In the meantime, it can be seen also that heat removal in DHRS starts to occur at the time of valve opening. At the time about 260 seconds after the transient initiation, the balance between heat transfer in steam generator and heat removal in DHRS is reached. At the same time, reactor power from decay heat is slightly lower. The heat transfers in both systems tend to be decreasing afterward. With that situation, it can be seen that the DHRS is capable to remove residual heat, and this will guarantee that the temperature in the primary system will be lower subsequently. This shows that reactor is in a safe condition.

## CONCLUSION

The simulation of reactor transient during feed water temperature drops due to malfunction in the Nuscale reactor system has been performed. The simulation is conducted using numeric calculation by modeling the reactor in RELAP5. After the initiation of the event, the temperature of the primary system was drop slightly before it goes up again due to power increase as a result of feedback reactivity and reactivity insertion from the control rod. As reactor power reaches the maximum set point, reactor trip and then the heat removal is conducted by DHRS. Calculation shows that DHRS is capable to remove decay heat and maintain coolant temperature. From this simulation, it can be concluded that the DHRS design is able to maintain safe shutdown condition.

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## REFERENCES

1. Cummins W.E., Matzie R. Design evolution of PWRs: Shippingport to generation III+. Progress in Nuclear Energy. 2018. **102**:9-37.
2. Hidayatullah H., Susyadi S., Subki M.H. Design and technology development for small modular reactors–Safety expectations, prospects and impediments of their deployment. Progress in Nuclear Energy. 2015. **79**:127-35.
3. Johnson R., Pottorf J., Leonard M., Modarres M., Corradini M., Dhir V., et al. Severe Accident Prevention and Mitigation Features of the NuScale PWR Design. in: *Proc. ANS Winter Meeting*. 2009. pp. 15-19.
4. Reyes Jr J.N. NuScale plant safety in response to extreme events. Nuclear Technology. 2012. **178**(2):153-63.
5. Susyadi S., Tjahjono H., Tjahyani D.S. Numerical Study on Condensation in Immersed Containment System of Advanced SMR During Uncontrolled Depressurization. Tri Dasa Mega. 2017. **19**(3):149-58.
6. Antariksawan A., Widodo S., Juarsa M., Haryanto D., Kusuma M., Putra N. Numerical study on natural circulation characteristics in FASSIP-02 experimental facility using RELAP5 code. in: *IOP Conference Series: Earth and Environmental Science*. 2018. p. 012090.
7. Skrzypek M., Laskowski R. Thermal-hydraulic calculations for a fuel assembly in a European Pressurized Reactor using the RELAP5 code. Nukleonika. 2015. **60**(3):537-44.

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(Masa Berlaku s/d Vol. 22 No. 3 Tahun 2020)

8. Shaoxin Z., Wei S., Jianping J., Jieru A., Chunming Z. Transient Analyses of Main Steam Line Break Accident for High-Power Passive Reactor. in: *International Confernece Pacific Basin Nuclear Conference*. 2016. pp. 365-73.
9. Ni X., Zheng J., Hou E., Hao J., Bian B., Li N. Simulation of early phase radioactivity of CPR1000 plant under LOCAs based on RELAP5-3D core engineering simulator. Progress in Nuclear Energy. 2016. **93**:47-58.
10. Andi S.E., Surip W. Modeling on Passive Containment Cooling System using RELAP5. Tri Dasa Mega. 2012. **14**(3):137-45.
11. Antariksawan A.R., Widodo S., Tjahjono H. Parametric study of LOCA in TRIGA-2000 using RELAP5/SCDAP code. Tri Dasa Mega. 2017. **19**(2):59-70.
12. Antariksawan A.R., Wahyono P.I. Steady state and LOCA analysis of Kartini reactor using RELAP5/SCDAP code: The role of passive system. in: *Journal of Physics: Conference Series*. 2018. p. 012048.
13. Ekariansyah A.S., Hastuti E.P., Sudarmono S. RELAP5 Simulation For Severe Accident Analysis of RSG-GAS Reactor. Tri Dasa Mega. 2018. **20**(1):23-34.
14. NuSCale. NuScale Final Safety Analysis Report. Corvallis, Oregon 97330 NuScale Power LLC; 2016. Available from: https://www.nrc.gov/reactors/new-reactors/design-cert/nuscale/documents.html
15. Butt H.N., Ilyas M., Ahmad M., Aydogan F. Assessment of passive safety system of a Small Modular Reactor (SMR). Annals of Nuclear Energy. 2016. **98**:191-99.
16. Yoon D.S., Jo H., Fu W., Wu Q., Corradini M.L. MELCOR Analysis of OSU Multi-Application Small Light Water Reactor (MASLWR) Experiment. Nuclear Technology. 2017. **198**(3):277-92.