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Natural Circulation System for Advanced Fast Reactor with Lead-Bismut as a Coolant

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ABSTRACT

This paper is an outome of a study on the development of the conceptual design of advanced fast reactors that have passive and inherent abilities. Safety performance indicators are evidenced by the ability of the reactor to remain under control and can operate on a new equilibrium shortly after an accident. Simulations are performed when an accident due to unprotected loss of flow (ULOF) and unprotected loss of heat sink (ULOHS) occurs at the reactor. The passive safety system is realised by the reactor vessel auxiliary cooling systems (RVACS), before its performance is evaluated. The simulation results show that the current power reactors can survive the ULOF and ULOHS accident. The optimisation of the reactor and the RVACS results in higher natural circulation level, indicating the reactors satisfy the inherent or passive safety standards. The results from the 'accident' show there is a large safety margin to the maximum temperature within the fuel, cladding, and coolant.

Keywords: Accident analysis, advance reactor, heat sink system, NPP, RVACS, ULOF, ULOHS

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INTRODUCTION

One of the most important aspects in the nuclear reactor design process is the safety aspect. Advanced and accurate safety simulation must be performed before it can be constructed. Complex calculation stages are needed to create a reliable simulation model, begins with the calculation of multi group diffusion equations (Ceolin, Schramm, Vilhena, & Bodmann, 2015; Ayyoubzadeh, Vosoughi, & Ayyoubzadeh, 2012; Quintero-

Leyva, 2012), reactor kinetic (Vyskocil & Macek, 2014; Xia, Jiang, Javidnia, & Luxat, 2012) and thermal hydraulic analysis (Giannetti, Di Maio, Naviglio, & Caruso, 2016; Liu, Zhang, Lu, Wang, & Qiu, 2016; Sun, Wang, Zhang, & Su, 2016; Mesquita & Gomes Do Prado Souza, 2014). Increased safety aspects of the advanced reactors can be realized by reducing the variety of active devices. The role of natural circulation that does not depend on electrical power becomes very important (Guo, Sun, Wang, & Yu, 2016; Wang et al., 2016; Zhao, Li, Chen, Zheng, & Chen, 2015). Two commonly used terminology associated with natural circulation are passive safety systems (PSS) and inherent safety systems (ISS) (Bochkarev, Alekseev, Korsun, & Kharitonov, 2016).

The research began with designing a reactor by adopting a two-dimensional geometry r-z of SPINNOR reactor (Small Pb-Bi Cooled Non-Refueling Nuclear Reactors), with 300 MWth powers, Pb-Bi liquid metal coolant, mixed oxide fuel (UO_2-PuO_2), as well as satisfying passive and inherent safety criteria in accordance with the standard of generation IV reactor designs (Yan & Sekimoto, 2008). The high fuel temperature safety margin as well as high coolant boiling point of Pb-Bi (~ 1700°C) made it possible to design reactors with inherent safety capabilities. The details of reactor design specifications are shown in Table 1. Preference of MOX fuel (UO_2-PuO_2) on this design is based upon the consideration that the fuel has a high Doppler coefficient, a high melting point and the availability of adequate in nature. MOX fuel can easily be applied in large power levels. At low power levels, MOX fuel requires a fraction of the volume of the fuel is high. The rationales for selecting of materials Pb-Bi as coolant because it has the characteristics of a good nuclear analysis in neutronic, void reactivity, and the breeding ratio. It is possible to design a reactor core with a long operating time without refueling (Su'ud & Sekimoto, 1995).

The design of advanced reactors is a conceptual design researched and developed. The reactor operates at a higher temperature than the previous generation, even several designs are intended for the production of hydrogen. The design is expected to represent excellence in enhancing the safety of nuclear reactors, improve resistance to proliferation, minimize waste and use of natural resources, and reduce costs through the development and operation (Su'ud, 2007). The advanced reactor must have the following criteria: innovative design that is compact, simple, modular, and more dependent on passive components (Butt, Ilyas, Ahmad, & Aydogan, 2016).

Reactor accident analysis can be realized by modifying the RETRAN code, in the case of accidents due to the increase in reactivity (Kim & Yang, 2005). The ULOF and UTOF accident simulation, also experimented on a small reactor with Pb-Bi as coolant, metal uranium and MOX as fuel. Natural circulation has succeeded in cooling the reactor. Changes in coolant temperature, fuel temperature, reactivity feedback, play an important role to achieve the capabilities' inherent safety (Su'ud, 2008). Numerically, the simulation of ULOF accident began with calculating of points kinetic equation and analyzing one phase thermal hydraulic and heat conduction calculation (Kazeminejad, 2008). Several conceptual designs that have implemented the concept of passive safety include: Material Testing Reactors (MTR), High-Temperature Gas Cooled Reactor (HTGR), and CANDLE fast reactor (Gaheen, Elaraby, Aly, & Nagy, 2007; Hossain, Buck, Said, Bernnat, & Lohnert, 2008; Yan & Sekimoto, 2008).

Other reactors that have implemented the concept of PSS are fast reactors namely Flexible Conversion Reactor (FCR). This reactor can be described as a version of the actinide burner reactor (ABR), or minor actinide burner reactor (MABR), or actinide burner reactor with Thorium (ABRT). Each type differed only in the configuration of the reactor core and the number of control rods. The overall design of the reactor is Lead coolant and has a power 700MWth and 300MWe (Nikiforova, Hejzlar, & Todreas, 2009). The coolant of this reactor, contained completely in the reactor vessel, and the decay heat removal system using RVACS, equipped with perforated plates to improve heat transfer. Design is not equipped with a seismic isolation system, then to handle the weight of the coolant during the earthquake, the reactor vessel is thickened to 25 cm. Thickening the walls of the reactor vessel will cause a reduction in heat dissipation rate, this is corrected by adding the liquid metal in the gap between the reactor vessel and the walls of the reactor vessel. Others are related to reactor designs that implement PSS is Korean Lead-cooled fast reactor (KLFR). KLFR has a power rating 900 MWth, 372 MWe, and using the RVACS design modifications for decay heat removal systems. KLFR using the liquid metal is Lead, as the gap filler material between the reactor vessel and the walls of the reactor vessel. Although the reactor heat removal system is the same as the reactor FCR, the selection of a liquid metal of lead is still widely questioned. The PSS performed on PWR reactors, 300 MW power by using software RELAPS5 / MOD.3.4. The focus from the study is to conduct a comparison type of water and the air coolant. Two types of coolant successfully made stable natural circulation system, decay heat can effectively be removed through the primary loop. For long-term cooling, the air conditioning is better than water cooling. The heat-transfer capability of a water coolant is stronger from the beginning of time (Wang, Tian, Qiu, Su, & Zhang, 2013).

The PSS on BWR reactor can be done without operator assistance, AC power supply, and the injection pump. By utilising the effect of gravity, the components of the gravity-driven cooling system (GDCS) inject water into the reactor and the reactor core. Component's isolation condenser system (ICS), and passive containment cooling system (PCCS) are used to eliminate the residual heat. Analysis of the passive safety system response to the situation loss of coolant accidents (LOCA) can be performed by simulation (Lim et al., 2014). Thermal hydraulic analysis of LOCA accident on advanced heavy water reactor (AHWR), 920 MWT power, is also conducted. This study discusses the impact of high pressure injection in the process of cooling the reactor after the accident. Passive safety features through the natural circulation occur in primary coolant with no pump. This reactor is equipped with the emergency core cooling system (ECCS) and isolation condensers (ICs) to remove the decay heat. This ECCS provides cooling for the fuel in passive mode for 15 minutes after the accident. Cooling is then performed within three days through a gravity-driven water pool (GDWP) (Tyagi, Kumar, Lele, & Munshi, 2013). Inherent and passive safety system is an innovative design to improve the reliability of nuclear power plants. CPR1000 design is equipped with a secondary cooling system of passive residual heat removal system (PRHRS) and system core makeup tank (CMT). The PRHRS effectively manages to eliminate the residual heat while the CMT is used to inject Boron water into the reactor core during an LOCA accident (Wang et al., 2013).

The concepts of integrated PSS are developed to perform various functions of passive safety. This design has a variety of functions, including: system of residual decay heats sinks, the injection of PSS, passive cooling containment system, cavitation system, passive system retention vessel, and filtering through the ventilation system to control the pressure. The safety systems are enhanced in cases when the situation reactor experiences a blackout (Chang, Kim, & Choi, 2013). In the transient state, the PSS is designed to cool down the reactor coolant from the normal state to the shutdown conditions through a natural circulation system (Dai, Jin, Wang, & Chen, 2013). Thermal-hydraulic research advances have significantly improved operating performance of nuclear reactors. Research and development aspects of thermal hydraulics are carried out using experimental and computational approaches. The use of fine-scale computational models of multi-physics aims to increase the computing power (Saha et al., 2013).

The new concept of offshore nuclear reactor enhances security features by implementing security structure based gravity (gravity-based structures). This structure is applied to the APR1400 reactor, which is the latest model of the nuclear power plant in the Republic of Korea. There are new features proposed and can be directly applied to a large-scale nuclear power plant. Such features include emergency passive containment cooling system (EPCCS) and emergency passive reactor-vessel cooling system (EPRVCS). Both features use the seawater as a coolant and are also equipped with security features to withstand earthquakes, tsunamis, storms and collisions at sea. The offshore NPP is more powerful than a nuclear power plant, and conventional land has the strong potential to provide opportunities in the nuclear-power industry (Lee, Lee, Lee, Jeong, & Lee, 2013).

This study aims to provide a comprehensive safety analysis of NPP. Simulations are performed on advanced liquid metal-cooled reactor. The focus is on how to improve the reactor design performance and increase natural circulation so that it meets the criteria of passive safety. The object of study is a small reactor that has a 225 MWth; Pb-Bi is a coolant, and mixed oxide fuel (UO2- PuO2). Pb-Bi chosen because it has a heat transfer coefficient and since its boiling point is very high, it is possible to design a reactor with inherent safety capabilities.

The simulation of the NPP accident was done using quasi-static method (Wei, Sui, Ma, Jing, & Shan, 2014). The accident was caused by the failure of the technical components in the reactor. Simulations are done on unprotected loss of flow (ULOF) and unprotected loss of a heat sink (ULOHS).

METHODS

Table 1
Specification of fast reactor design

Parameters		Specification
Thermal power		225 MWth
Coolant		Pb-Bi (Pb 45%-Bu 55%)
Shielding materials		B_4C + *Stainless steel*
Fuel		MOX (UO_2-PUO_2)
Fuel enrichment		10%~13% (PuO_2)
Reactivity Swing		Max. 0.002 (0.5$)
Reactor Core	Diameter of fuel pin	1.0 cm
	Cladding Thickness	0.05 cm
	Pin pitch of fuel	1.2 cm
	Average of temperature input	400°C
	Average of temperature output	800°C
	Volume fraction : Fuel	60%
	Structure	10%
	Coolant	35%
Steam Generator	Height of SG	4.0 m
	Pipe diameter (*inner/outer*)	1.7 / 2.0 cm
	Pitch of pipe	3 cm
	Primary flowrate	8000kg/s
	Secondary flowrate	100 kg/s
	Water temperature	225°C

The reactor designs used as research subjects are based on the basic concept of the reactor SPINNOR (Small Pb-Bi cooled No on-site refueling Nuclear-Power Reactors). The SPINNOR aresmall-sized nuclear reactors with a fast neutron spectrum that can be operated more than 15 years without replacing the fuel. The SPINNOR has a long life shelf live and uses little power. It was developed in Indonesia in 1990 in collaboration with the Research Laboratory for Nuclear Reactors of the Tokyo Institute of Technology. This reactor is suitable in isolated areas and small islands. It is easy to operate and maintain, transport, and meet the inherent safety/ passive, and resistant to nuclear proliferation (Su'ud, 2008). The SPINNOR reactor design adds a component RVACS to protect and enhance the reactor cooling system that operates naturally.

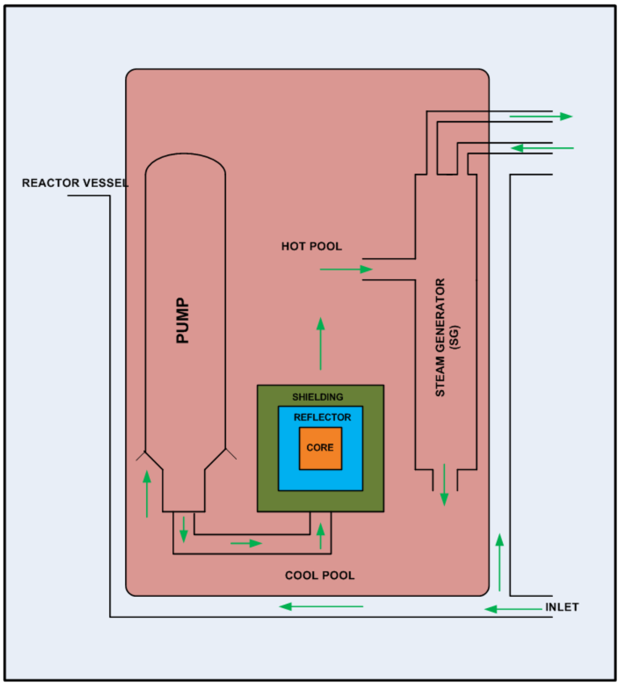


Figure 1. A schematic diagram of SPINNOR

A schematic diagram of the reactor SPINNOR is shown in Figure 1. The system consists of a reactor core, hot pool, cool pool, steam generators and pumps. In this system, there is no intermediate heat exchanger (IHX), so the flow heat (green arrow) from the primary cooling system was directly transferred to the steam generator. The working principle of the heat flow reactor can be explained as follows: the coolant flow passes through the reactor core to take the heat from the reactor core, after that moves up into the hot-pool. From the hot-pool, the coolant flow enters the generator steam and heat transfer, and then down to the cool pool. Cool-pool of coolant is pumped back into the reactor core. The thermal hydraulics system of SPINNOR has two flow circulation cooling system, which is the primary coolant and secondary coolant. Figure 2 shows a schematic flow of coolant. There is a pump that is used to provide forced circulation to cool the reactor core. When the reactor faces loss of cooling capability, its pump power driving cooling becomes inhibited.

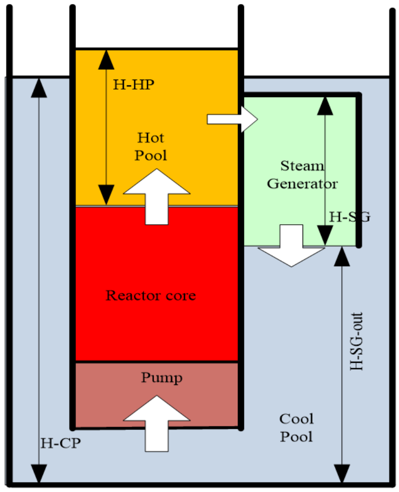


Figure 2. A scheme of coolant flow rate. From "Safety analysis of small long life CANDLE fast reactor", by Yan and Sekimoto, 2008, *Annals of Nuclear Energy, 35*(5), pp. 813-828. Copyright 2008 by Elsevier

The reactor design processes begins with neutronic analysis to get the neutron flux distribution and power distribution in a steady-state nuclear reactor. The calculation is performed using a multigroup diffusion equation. The discretisation processing diffusion equation is carried out using the finite difference method, and then solved using SOR (Successive Over Relaxation) method and inner-outer iteration scheme (Yulianti, Su'ud, Waris, & Khotimah, 2009). The calculation result of this neutronic analysis would provide information about reactor criticality level and neutron flux distribution.

The dynamics of neutron population in the core during the transient process is determined by using nuclear reactor kinetics. The kinetics calculations give time dependent power distributions pattern during some transient event of the reactors. Then, the calculations of the thermal hydraulic are done, including temperature distribution in each part of the reactor consisting of a reactor core and steam generator, i.e. fuel temperature, coolant temperature, cladding temperature, and all aspects of coolant circulation in the reactor.

The ULOF accident simulations are performed to investigate inherent safety performance with the following processes: a reduction of flow rates due to the loss of the pumping power. This then causes coolant temperature to increase due to an imbalance between the power level and the coolant flow rate. The increase of fuel and coolant temperature triggers negative feedback that forces the decrease in the power level. The power reduction causes a decline in fuel temperature which then triggers positive feedback, and the system would reach a new equilibrium if the absolute value of negative reactivity feedback due to coolant temperature rise equals to the positive reactivity feedback due to the decrease in fuel temperature. The final equilibrium temperatures depend on the coefficient of feedback reactivity as well as its thermal characteristics, especially heat capacity of the coolant and thermal conductivity of the fuel.

The ULOHS accident simulations are performed initially by investigating RVACS (Reactor Vessel Auxiliary Cooling System) performance and its optimisation. Passive safety capability during ULOHS accident strongly depends on the portion of heat which can be removed through RVACS during the accidents. The analysis includes heat transfer i.e. radiation, convection, and conduction. The first step begins with the design process of RVACS construction and numeric calculations of RVACS performance optimisation at steady state using Newton-Raphson method. Indicators of passive safety performance are shown in percentage quantity of RVACS system in removing heat from the reactor vessel and vessel wall. The RVACS system is expected to remove the heat using natural circulation of the atmosphere around the reactor vessel at a steady state. The systems of heat transfer involved in the analysis are radiation, conduction and natural convection. The heat transfer calculations ae conducted on the reactor vessel elements, outside wall of the reactor vessel, and the separator plate (Abdullah & Nandiyanto, 2016).

The current reactor designs eliminate IHX (intermediate heat exchanger) so that the heat is transferred directly from the core to the steam generator loop by a Pb-Bi coolant. The ULOHS accident is triggered by the loss of cooling capability of steam generators. The accident sequence begins with the loss of the reactor's ability to remove heat from the secondary cooling loop. During the accident, the heat dissipation rate decreases whereas the coolant inlet (flow input) temperatures increase till a new equilibrium level is reached.

RESULTS AND DISCUSSION

Analysis of RVACS Performance

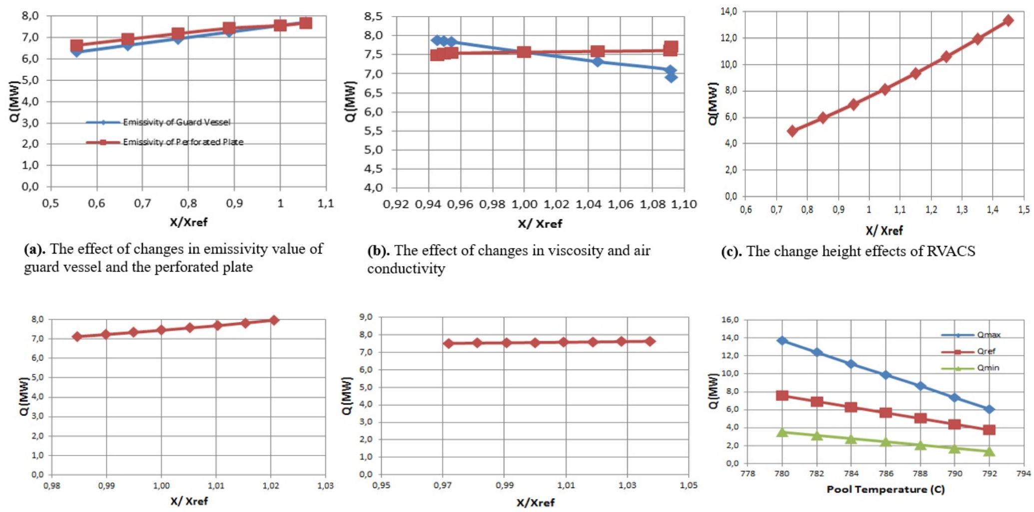
The paper analyses the performance of reactor designs in removing decay heat. It also evaluates the performance of reactor design as a result of accidents that leads to cooling capability loss. Reactor accident simulation analysis is applied in the case ULOF and ULOHS accident. The calculation of the steady-state, such as the calculation of neutron flux distribution, power distribution and reactor multiplication factor are not discussed in this paper. Calculation of RVACS performance is discussed in detail as it is used to prove the hypothesis that passive safety system in this SPINNOR reactor design will run properly.

Before the simulation is done, it is necessary to arrange the total decay heat as initial assumptions to prepare the PSS; the latter is developed by utilising the natural circulation of the atmosphere surrounding the reactor by using RVACS. The RVACS design analysis is conducted to obtain information about how much residual heat can be removed by RVACS. The optimisation is realised to receive a large maximum heat transfer so that it can boost the performance of RVACS in removing heat. Two aspects are examined: First, the characteristic aspects of thermal emissivity covering the walls of the reactor vessel to the separator plates, as well as the effect of viscosity to the conductivity of the air. Second, optimising the geometry aspects which include the impact of changes RVACS height, diameter and channel gap. The reference design is used as shown in Table 2. The analysis only covers most of the coolant temperature. If the analysis produces a cooling mass below the maximum temperature of the cladding, then the design will be considered successful, because the power level of decay across the cladding is very small and there is little thermal resistance between the cladding and the primary coolant.

Table 2
Main parameters of RVACS design

High of RVACS	40 m
Downcomer gap	5,86 m
Radius of reactor vessel	4,62 m
The thickness of reactor vessel	0,4 m
The thickness of guard vessel	0,1 m

The simulation begins with calculating the total heat transfer in the reference design. The RVACS can remove residual decay heat of 4.39 MW, or approximately 1.95% of the thermal power reactors. Decay heat removal is obtained through a process of natural circulation. Simulations are performed when the reactor is a steady state. Heat-transfer coefficient and friction factor are assumed in the case of a turbulent flow.



(a). The effect of changes in emissivity value of guard vessel and the perforated plate

(b). The effect of changes in viscosity and air conductivity

(c). The change height effects of RVACS

(d). The change diameter effect of RVACS

(e). The change channel gap effect

(f). Heat removal capabilities of RVACS

Figure 3. The result of optimisation of RVACS performance

Figure 3(a) and Figure 3(b) show RVACS performance in optimising emissivity of the reactor vessel wall and separator plates, and optimisation aspects of viscosity and air conductivity. Optimising emissivity is intended to clear the effect on the ratio in the energy radiated by the walls of the reactor vessel and the perforated plate with the energy radiated by a black body at the same temperature. This emissivity value shows the ability of the reactor vessel material and perforated plate to radiate absorbed energy. Optimising the emissivity value shows an increase in performance RVACS if the value ends at 1, while optimising the amount of viscosity and conductivity of the air does not contribute significantly to improving the heat-transfer of RVACS. The viscosity and conductivity of the atmosphere around RVACS is relatively unchangeable and is at room temperature. The simulation results will provide different conclusion if RVACS applied to the reactor is placed at the location that has an extreme climate.

Figures 3(c), 3(d) and 3(e) show the results from the optimisation of the geometry aspects. Our simulation results show that the factor altitude will affect the RVACS performance. Optimising the diameter and width of the channel gap does not alter the performance of RVACS. The heat-transfer is similar with the simulation results of reference design. Although with changing altitude of RVACS will give excellent results, a maximum limit must be determined, because it is limited by the ability of the construction of the reactor. The width of the channel gap cannot be extended more generally related to the fluid nature of gap fillers that require a very fast time to melting.

Figure 3(f) shows a result summary for the optimisation of minimum design, reference designs and a maximum design. The simulation results conclude the RVACS Design can remove the heat between 0.66% to 2.12% of the reactor's thermal power. Obtaining value of heat transfer is a result of the role of natural circulation of air around. If an accident occurs at the reactor, the PSS can remove the heat. The calculations show that the heat dissipation is at sufficient level to prevent the high temperatures provided RVACS is reliable and efficient.

Analysis of ULOF Accident

The ULOF accident occurs due to a failure in the primary loop triggering failure; the loss of pump power (coast down) P (t) is calculated using the formula (Yan & Sekimoto, 2008):

$$P(t) = \frac{P_o}{\left(1 + t/\tau\right)}$$

Where

t : time after accident (s).

P_o : steady state pump power.

τ : pump coast down halving time (s)

To evaluated to effect of halving time τ, investigated the two experiments on τ=20 s (case ULOF A) and τ=12 s (case ULOF B).

Figure 4 shows the change in the mass flow rate coolant through the reactor core for both cases. In the first 30 seconds after the accident, the decrease in coolant flow rate is very drastic. At 40 second to the end, the mass flow rate slowly declines to almost constant. In the transient state, the case ULOF-A has a mass flow rate is greater than in the case ULOF-B. When the pump power decreases from steady-state value, mass flow rate in reactor core still survives in the range of 32.5% up to 37.25%. This is due to natural circulation factor.

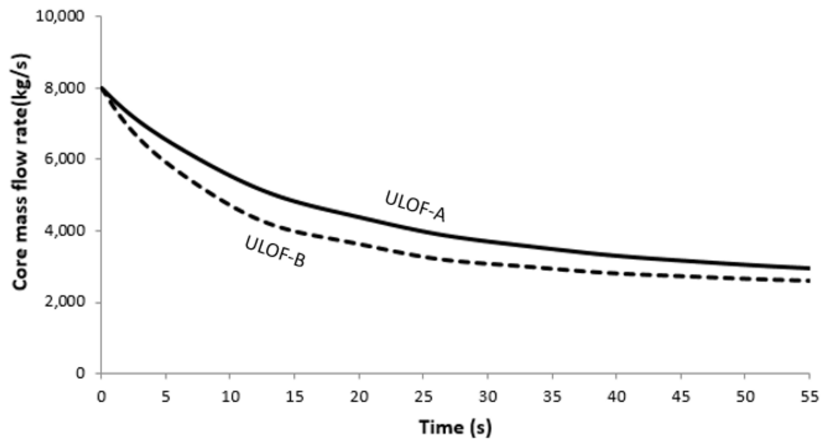


Figure 4. Change of mass flow rate at reactor core in ULOF-A and ULOF-B

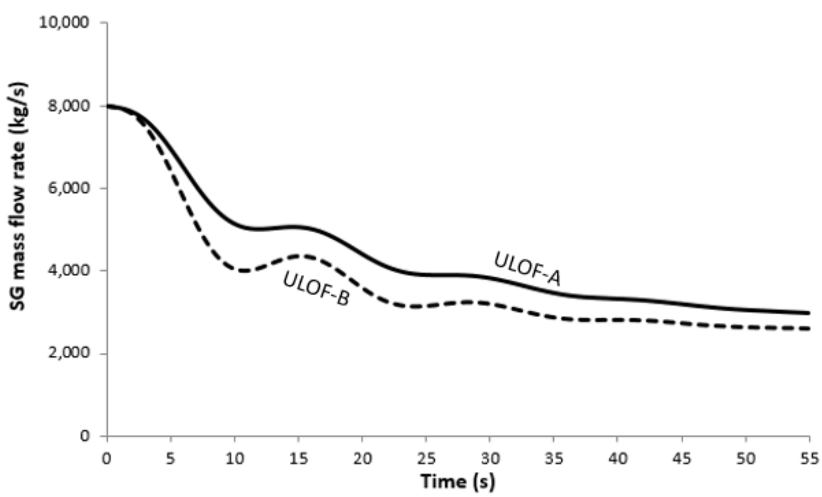


Figure 5. Change of mass flow rate at steam generator in ULOF-A and ULOF-B

Figure 5 shows the change of mass flow rate coolant in the steam generator (SG). The simulation results are almost identical to changes in mass flow rate coolant through the reactor core, but the flow rate of coolant in SG experiences a small oscillation that cause degeneration in the flow of the main pump that controls the level and coolant temperature. Natural circulation can increase the level of coolant circulation flow and the mass flow rate to survive the value of 32.25% up to 37% of the normal operating state.

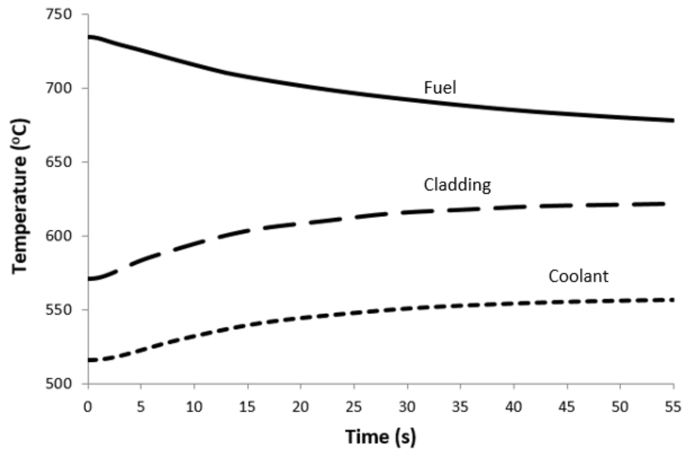


Figure 6. Hot spot temperature from fuel, cladding and coolant of case ULOH-A

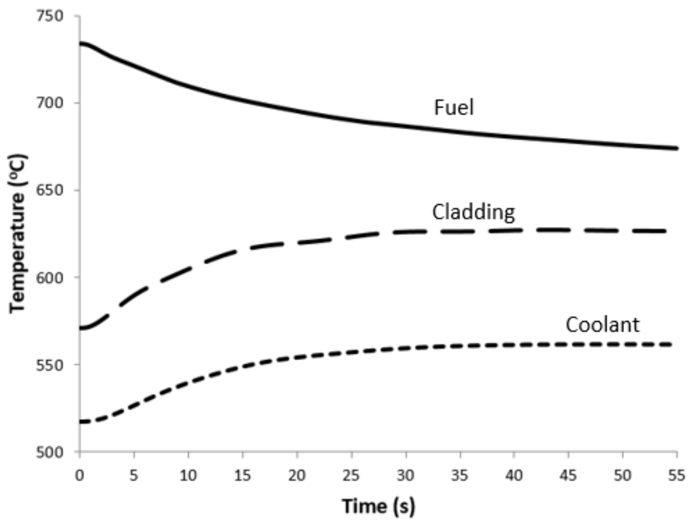


Figure 7. Hot spot temperature from fuel, cladding and coolant of case ULOH-B

Figure 6 shows the temperature profile of fuel, cladding and coolant in ULOF-A. In this case, the maximum temperature of the fuel gained 734.10°C, then declined slowly towards a constant value 678.20°C. Cladding temperature and coolant temperature respectively reach 621,64°C and 556,19°C. The temperature of the fuel, cladding, and cooling in the case or ULOF-A and ULOF-B still has a very large margin, so the reactor can survive the conditions of this accident. The threshold allows melting point of the fuel, cladding and cooling respectively at 2865°C, 2519°C and 1700°C. Usually the larger accident is caused by a high reactivity value, due to the failure of the control rod operation.

The simulation of results shows that when the flow rate of cooling declines, the coolant temperature average will increase, while the fuel temperature average declines. This occurs because of an imbalance between power and a coolant material. However, the temperature difference between a cooling material with the fuel decreases. This situation raises negative feedback causing a decrease in fuel temperature. The system will re-balance when the reactivity of the negative feedback due to the rise in coolant temperature is compensated by a positive reactivity feedback.

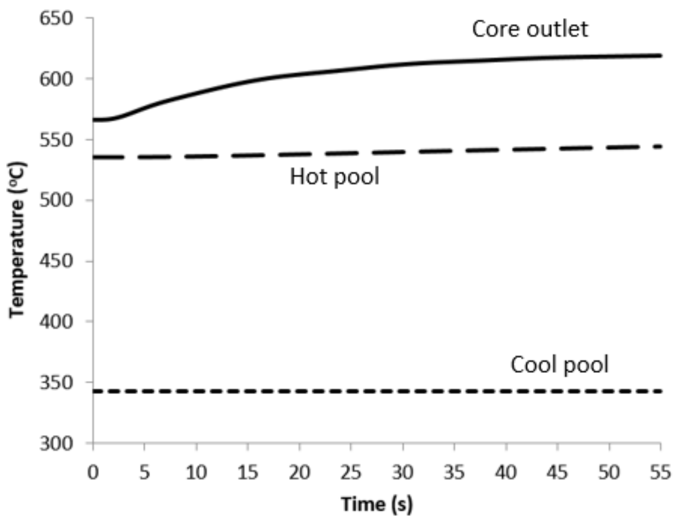


Figure 8. Temperature of hot pool, cool pool and core outlet of ULOF-A

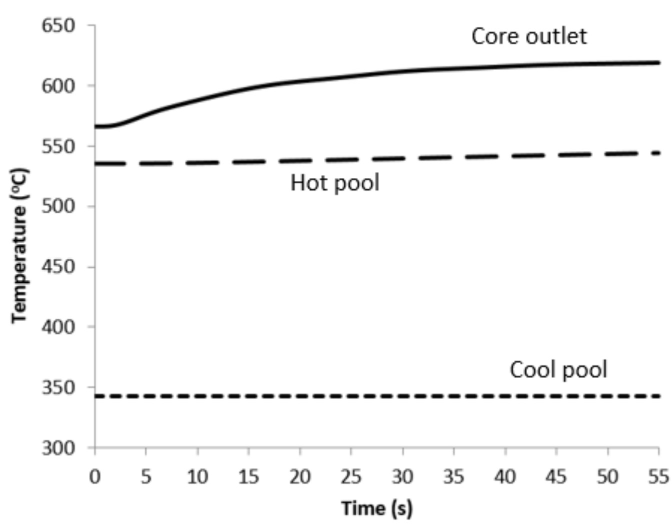


Figure 9. Temperature hot pool, cool pool and core outlet of ULOF-B

Figure 8 and Figure 9 show a thermal reactor system behaviour that occurs in the hot pool, cool pool, and core outlet in case ULOF-A and ULOF-B. The figures show the changes in the value of τ affects the transient process, while the temperature of the system is always convergent at the same value. Core outlet temperature changes tend to be faster and more intense than the temperature of the hot pool and a cool pool. Temperatures at a cool pool is relatively stable because the steam generator continues to work normally.

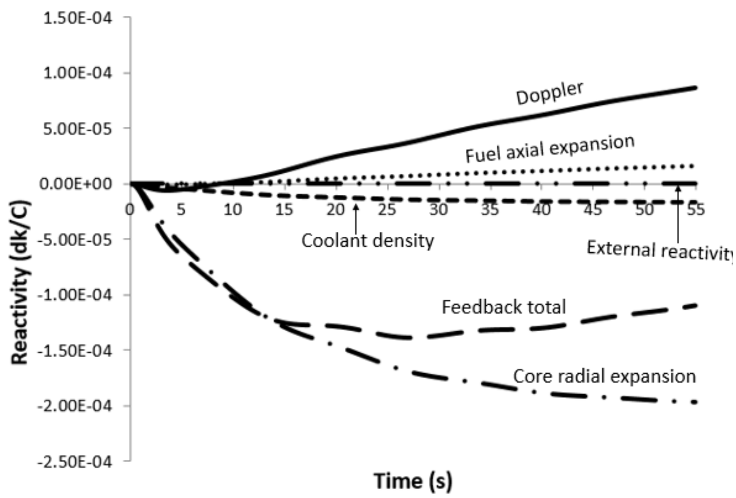


Figure 10. Reactivity change in case ULOF-A

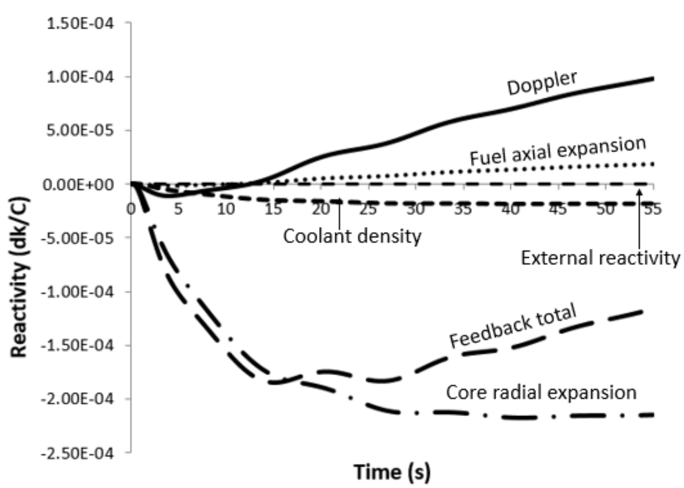


Figure 11. Reactivity change of case ULOF-B

Figure 10 and Figure 11 show the change in reactivity. Reactivity axial expansion provides the largest contribution, while reactivity contributes the smallest radial expansion. At the end of the accident, they will compensate positive reactivity of the Doppler effect and changes in coolant density. Doppler positive effect comes from the decreasing temperature cool pool.

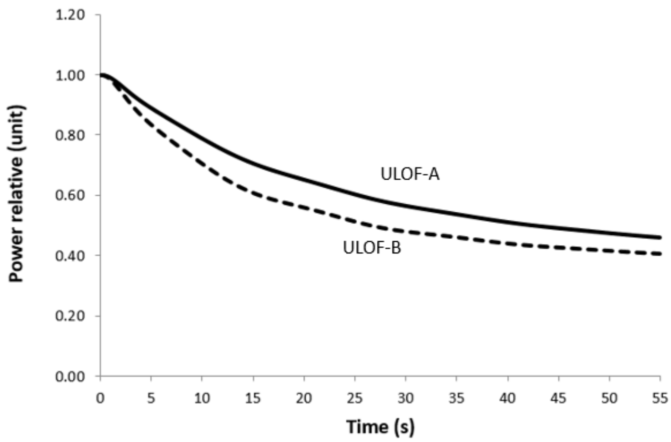


Figure 12. Thermal power relative change of case ULOF-A and ULOF-B

The change in thermal power is shown in Figure 12. Thermal power declines relatively rapidly in the first 30 seconds and then gets more slowly. Thermal power eventually converges at the value of 40.61% for cases ULOF-B and 45.87% for cases ULOF-A. The mass flow rate in the reactor core is lower, i.e. in the range of 32.5% to 37.25%, and the mass flow rate in the steam generatoris only 32.25% up to 37% of the steady state. This proves that the ability of natural circulation has an important role in this ULOF accident case. Overall, the results from this simulation show that the proposed reactor designs survive ULOF accidents. The increase in coolant temperature and outlet temperature does not pass the thermal limitations of a cooling material, structural material and fuel.

Analysis of ULOHS Accident

The reactor accident of ULOHS case is triggered by the loss of cooling capability in the steam generator (SG). After the accident occurs, the cooling capability of theSG can be observed. Figure 13 shows the mass flow rate of coolant at the reactor core and the SG. Shortly after the accident, the flow of coolant at the reactor core is less affected, but the mass flow rate is only slightly decreased. Figure 13 shows the mass flow rate of coolant through the reactor core and the steam generator. Shortly after the accident occurs, the flows of coolant through the reactor core are less affected, but the mass flow rate is only slightly decreased. The mass flow rate of the coolant at the SG looks unstable; proving that there are very large oscillations. In the first 40 seconds, a coolant mass flow rate is unstable but moments later it returns to stability, and achieves a new equilibrium. The oscillation phenomenon can be mitigated by increasing the volume of a cool pool.

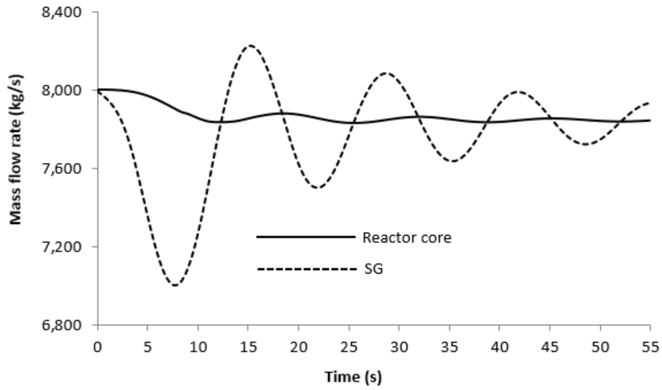


Figure 13. Changes at the coolant mass flow rate and the reactor core SG on ULOHS accident

Figure 14 shows the effect of changes in the volume of cool pool to the oscillation of the coolant flow that passes through the steam generator. Simulations are carried out by increasing the volume of cool pool randomly (the case of ULOH-B and ULOH-C). This is done to prove the increase in the volume of cool pool in the reactor will affect the rate of coolant flow oscillations at the SG. Referring to the previous RVACS design where the heat can be discharged approximately 4.39 MW or 1.95% of thermal power reactors, the results of these calculations can be used as a reference RVACS capability to remove heat from an accident ULOHS.

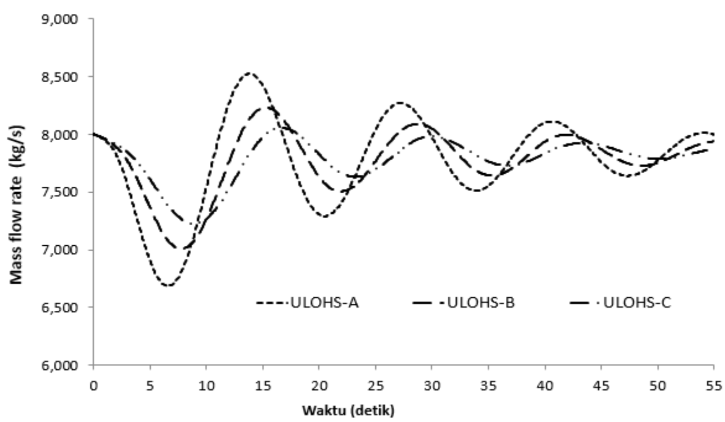


Figure 14. Changes of coolant mass flow rate at SG with various cool pool volume

Figure 15 shows changes of coolant temperature at hot pool and cool pool. Accidents cause loss of cooling capability from SG impacting on the coolant temperature at a cool pool. Shortly after an accident (for 10 seconds), the coolant temperature at a cool pool is still at normal state. But shortly afterwards, cool pool temperature increases, which leads to incapability of the SG to do the cooling. Because the design of SPINNOR reactor with cool pool volume has a large impact on ULOHS, accident can only be sensed in a expanded time, and to obtain thermal equilibrium takes a long time. Increased temperatures of cool pool impacts the rise in temperature at a hot pool. Coolant temperature goes to the reactor core value is higher than when the reactor is in steady state.

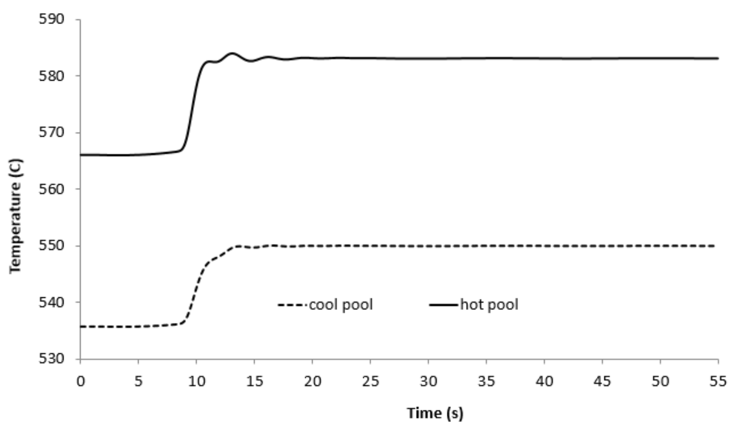


Figure 15. Changes of coolant temperature at hot pool and cool pool, after ULOHS accident

Changing temperatures of coolant, cladding and fuel after ULOHS accident does not affect a temperature hot spot in a short time, so the fuel temperature only increases slightly and temperatures tend to converge at a temperature of 815°C. Likewise, cladding and coolant temperatures do not change drastically. The RVACS is known to be capable of removing heat from the reactor vessel walls. Simulation results prove natural circulation in RVACS is able to function properly. There are differences in temperature change between the cladding and the cooler, but the difference is not so great, as shown in Figure 16. Ten seconds after the accident, temperature of coolant and cladding sudden increases, but then the temperature converge respectively t 583°C and 588°C. The temperature increase of the coolant, cladding or fuel, show they are in a safe margin, meaning that it is still far from the limit to the melting point of the material used.

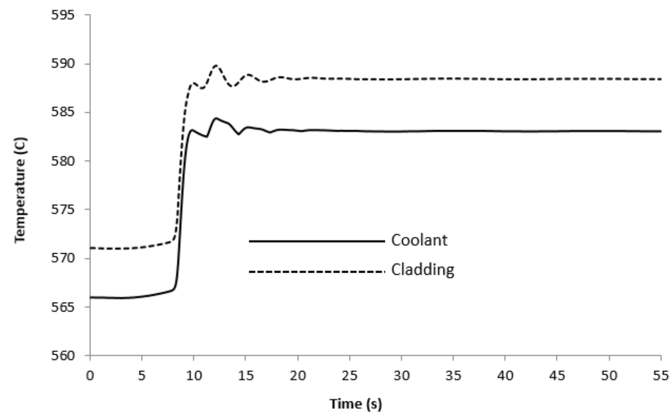


Figure 16. Changes of coolant and cladding temperature after ULOHS accident

Figure 17 show changes of thermal power relative in relation to the ULOHS accident. The thermal power at the reactor decreases quickly at 10s first before it slows down. Thermal power eventually converges towards the value of 57.16%. This proves that capability natural circulation has an important role in ULOHS accident.

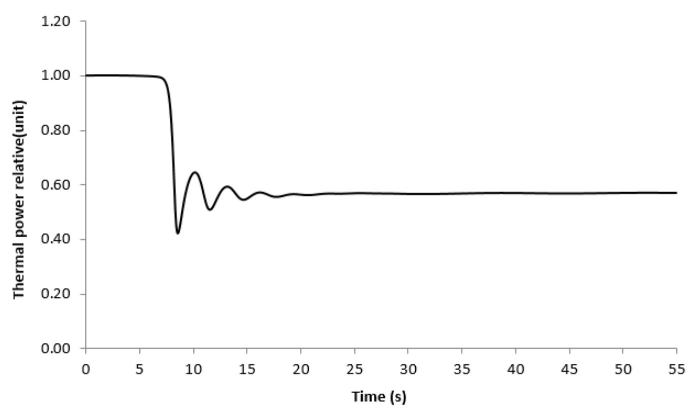


Figure 17. Changes in thermal power after ULOHS accident

Figure 18 shows changes of reactivity after ULOHS accident. This reactivity is dominated by Doppler Effect and reactivity feedback. This is due to the temperature of a cool pool increase, meaning that the temperature at the inlet also experiences substantial increases.

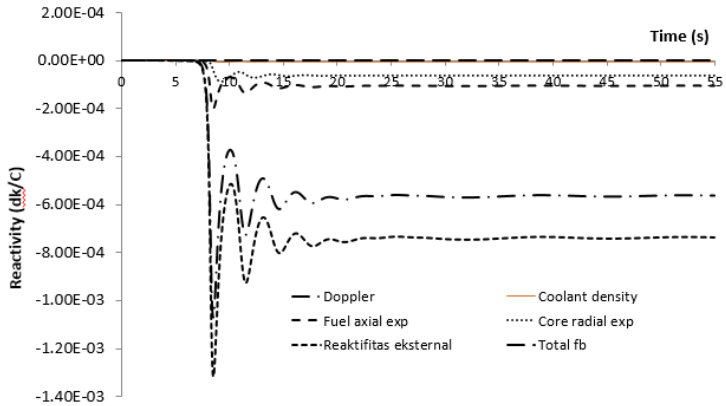


Figure 18. Changes of reactivity for case ULOHS accident

This study is only a simulation whereby ULOF accident simulations are performed to investigate inherent safety performance with the following processes: a reduction of flow rates due to the loss of the pumping power. This then causes the coolant temperature to increase due to an imbalance between power level and the coolant flow rate. Further, analysis for realistic condition will be examined by the reseachers in their future work.

CONCLUSION

In order to study the passive safety system, a system conceptual design of natural heat reduction is done using RVACS. In the reference design, RVACS can reduce 4.39 MW heat or around 1.95% of the total power of thermal reactor. This shows that when there is an accident in the reactor a while after it is shut down, the rest of the heat can be released by RVACS system. The improvement of RVACS works can actually be carried out by optimising several variables. The optimisation is done in two design aspects; first, thermal characteristic aspect that includes emissivity effect of the reactor's vessels and separate plates as well as the air viscosity and conductivity; second, optimisation on geometrical aspects covering the diameter effects, the height of RVACS, and channel gap effects. In general, simulation results show the reactor can easily survive by optimising the roles of natural circulation and passive safety system. Meanwhile, the results of accident analysis simulation due to the loss of reactor coolant show the maximum temperature, cladding, and coolant, have limitation in terms of melting points, meaning that the material reactor does not exceed its thermal limits.

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