Comparatives Studies Of A Safety Analysis For Molten Salt Reactor (MSR)

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Abstract. The Generation IV nuclear power system has been developed by the Generation IV Forum (GIF) with its benefits such as highly economical, enhanced safety, low waste production, and proliferation resistant. MSR is one of Generation IV nuclear power system which uses molten salt as a fuel and a coolant, therefore its technology differs from the conventional solid-fuel reactors. In this study, the author focus on the safety characteristic analysis of the MSRs, in which a point kinetic equations and core heat transfer have been modeled by developing a mathematical model for MSRs. The founded model is applied to analyze the safety characteristics of the molten salt actinide recycler and transmuter system (MOSART) by simulating transient basic condition, and that is the unprotected loss of flow (ULOF) and unprotected overcooling accident (UOC). The result of reactor accident simulation is power distribution and temperature in the core reactor toward the time and showing that MOSART conceptual design is a reactor design which is stably inherently. This research purposes to give basic understanding about safety characteristic of the MSR.

1. Introduction

At present the development of nuclear power plant technology reaches the stage of research and development of Generation IV nuclear power plants (advanced reactor systems) which are innovative developments from previous generation nuclear power plants. There are thirteen countries which are incorporated in the Generation IV International Forum (GIF), and they are: Argentina, Brazil, Canada, China, Euratom (European Atomic Energy Community), France, Japan, South Korea, Russia, South Africa, Switzerland, the United Kingdom and the United States, has evaluated and reviewed about 100 types of reactor types which may be suitable for the 4th Generation Nuclear Energy System application (Gen IV Reactor). At the end of December 2002, 6 types of potential reactor candidates were decided to be built in 2030, namely: Very High Temperature Reactor (VHTR), Sodium cooled Fast Reactor (SFR), Gas-cooled Fast Reactor (GFR), Lead cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), and Super Critical Water-cooled Reactor (SCWR). The typical characteristics of generation IV nuclear power plants generally are sustainability, economy, safety and reliability, and prevention of the use of nuclear weapons and physical protection. The focus of the sustainability aspect is the use of fuel and waste management, the focus of economic goals is competitive plant operating costs such as energy production costs and financial risks, the focus of the safety aspect is safe and reliable operation, and focuses on the prevention aspects of nuclear weapons and protection physical control and security of nuclear materials and nuclear facilities\cite{1}.

Molten Salt Reactor (MSR) is one of six types of IV generation nuclear reactors that are fluid-fueled so that the used technology has a difference from other conventional reactors. Such as in reactor design, reactor safety systems, and systems specifically related to fuel from MSR. In this study is conducted further learning about the safety system from MSR. Several MSR concepts have been proposed such as the small molten salt reactor
(SMSR)[2], the actinides molten salt transmuter (AMSTER)[3], MOSART[4], the molten salt fast reactor (MSFR) and thorium molten salt reactor (TMSR)[5].

Modeling a liquid fuel system is something that must be done to assess the performance of the MSR, and it is something interesting in reactor physics because it can review the effects of the flow of liquid salt fuel. For MSR safety analysis, the fuel moving transient simulation is carried out by using the reactor kinetic point model for the reactor with liquid fuel. The reactor kinetic point model is very useful as an approach to the analysis of safety system and MSR stability[6].

In this study, the reactor kinetic point model that calculates the effect of fuel flow for liquid fuel reactors is obtained without an approach of time-dependent equations of neutrons and delayed neutron precursors. The developed spatial kinetic model was applied to calculation of numeric unprotected loss of flow (ULOF) and unprotected overcooling accident (UOC) based on MOSART's conceptual design, which was developed to burn used nuclear fuel. ULOF results by all models show that the behavior of relative power, liquid salt fuel temperature, graphite temperature, and reactivity feedback are similar due to strong negative reactivity feedback (not only temperature but also flow rate)[7]. The method used in the calculation is the numerical method, by combining the reactor's kinetic equations and heat transfer.

2. Matemathic Model

Model Point Kinetic for Liquid Fuel Reactors

The kinetic model approach for calculating basic transients in liquid fuel reactor uses the modified point kinetic model (MPKM) which are derived by MacPhee by considering the precursor neutron decay outside the primary loop is[8]

\[
\frac{dp(t)}{dt} = \frac{1}{A} p(t) + \sum_{i=1} \lambda_i c_i(t) 
\]

(1)

\[
\frac{dc_i(t)}{dt} = \beta_i p(t) - \lambda_i c_i(t) - \frac{1}{\tau_c} c_i(t) + e^{-\lambda_i \tau_l} c_i(t - \tau_l) 
\]

(2)

MPKM was chosen because MPKM considers the flow effects by approximation, while the Point Kinetic Method (PKM) does not consider the flow effects. Equation (2) shows the neutron decay of precursors outside the primary loop, \(\tau_c\) time is the precursor flow in the core and \(\tau_l\) is the time of flow outside the primary loop (outside the core).

Figure 1. Reactor Configuration of MOSART[4]

Heat Transfer Model

The formed spatial kinetic model is for the analysis of ULOF and UOC on MOSART. To calculate ULOF and UOC, a mathematical model of heat transfer calculation is needed. The parameters are adopted from the calculation of heat transfer inside the core based on the principle of energy conservation [9]. Energy balance for liquid fuels is formulated by:

\[
M_f c_{p,f} \frac{dT_f}{dt} = F_f (p'(t) - 2m(t)c_{p,f} (T_f - T_{in}) + h_{fg} A_{fg} (T_g - T_f)) 
\]

(3)
The equation for material structure (graphite in MOSART) is

$$M_{g} \frac{d\rho_{g}}{dt} = F_{g} p(t) + h_{fg} A_{fg} (T_{f} - T_{g}) \tag{4}$$

Where $M_{f}$ and $M_{g}$ are the period of liquid fuel and graphite inside the core, $F_{f}$ and $F_{g}$ are common fraction energy in liquid fuel and graphite, $m$ is the mass flow rate of liquid fuel, $h_{fg}$ is the coefficient of heat transfer between liquid fuel and graphite, and $p(t)$ is the thermal reactor power. This is a calculation for the temperature of liquid fuel and graphite:

$$T_{f} = \frac{\Sigma M_{f,i} \rho_{f,i}}{\Sigma M_{f,i}} \tag{5}$$

$$T_{g} = \frac{\Sigma M_{g,i} \rho_{g,i}}{\Sigma M_{g,i}} \tag{6}$$

**General Description of MOSART**

The MOSART concept was proposed by RRC-KI, and was studied in the coordinated research project (CRP) within the IAEA framework to examine and demonstrate the feasibility of MSR which could also reduce the toxicity of long-lived waste and produce electricity efficiently in closed cycles. The fuel flow in MOSART and reactor configuration is presented in reference[4]. in Figure 1 shows the configuration of the MOSART reactor, and in Table 1 and Table 2 shows the parameters of the MOSART concept.

| Table 1. Basic Parameters of MOSART [7] | Nilai |
|----------------------------------------|------|
| Core diameter/height (m)               | 3.4/3.6 |
| Core nuclear power (MW)                | 2.400 |
| Fuel salt mass in the core (kg)        | 10.000 |
| Fuel salt mass in the core (kg)        | 69.914 |
| Fuel salt mass in the external loop (kg)| 39.376 |
| Graphite mass in the core (kg)         | 20.000 |
| Fuel salt volume in the core (m$^3$)   | 32.67 |
| Fuel salt volume in the external loop (m$^3$) | 18.4 |
| Core inlet temperature (°C)            | 600 |
| Core outlet temperature (°C)           | 715 |
| Graphite temperature (°C)              | 770 |
| Reactivity coefficient of the fuel temperature (pcm/K) | -4.125 |
| Reactivity coefficient of the graphite temperature (pcm/K) | -0.04 |

In this study, other parameter is needed to complete the calculation, that is the delayed parameters of precursor neutrons for MOSART which is shown in Table 2. In the Table 2 there are delayed neutron fractions $\beta_{i}$, decay constants$\lambda_{i}$, and $\tau_{i}$ lifetime family delayed precursor neutrons of MOSART, and total $\beta$ which has been calculated by adding $\beta_{i}$ to 340 pcm.

| Table 2. Static parameters of delayed neutron precursors from MOSART [7] |  |
|--------------------------------|--------|--------|--------|--------|--------|--------|
| $\beta_{i}$ (pcm)               | 7.786  | 77.248 | 54.944 | 118.150| 61.030 | 20.842 |
| $\lambda_{i}$ (s$^{-1}$)        | 0.0129 | 0.0301 | 0.1216 | 0.3350 | 1.2930 | 3.2070 |
| $\beta_{i}/\lambda_{i}$         | 603.57 | 2566.38| 451.84 | 352.69 | 47.20  | 6.50   |
| $\tau_{i}$ (s)                  | 77.5194| 33.2226| 8.2237 | 2.9851 | 0.7734 | 0.318  |
3. Results And Discussion

In this study, the calculated basic transient simulations is ULOF and UOC with parameters on MOSART.

**Unprotected Loss Of Flow (ULOF) Analysis**

Initially, assumption of ULOF accidents were begun with loss of cooling circulation in the primary system due to pump failure, so that the initial cooling flow rate of 10,000 kg/s dropped to a value of 4% within 7 seconds, which is illustrated in Figure 2. During the initiation, temperature on fuel remains constant.

The ULOF initiation as loss of cooling circulation in the primary system produces two effects, the first is reduction of disintegration in delayed neutron on the terrace which causes insertion of reactivity, the second is increasing fuel temperature on the terrace shown in Figure 3. From the graph in Figure 3, it can be seen that the fuel initially has a temperature of around 660°C, then along with the loss of the cooling flow rate in the primary system, the temperature rises to around 742°C at ± 35s. Then the temperature of the fuel decreases to around 675°C with stable conditions and experiencing a slight increase of ± 2-3°C, along with the safety system that has been active at MSR. From the perspective of the safety system on the melting of fuel that has already occurred, the most dangerous temperature of the fuel is 1400°C, in this simulation the highest fuel temperature has a value of about 745°C. this shows that the design of the MOSART concept still has a good safe threshold.

Next in Figure 4 and Figure 5, show the relative power during ULOF and occurring the fuel feedback reactivity coefficient during ULOF. The power value during the ULOF process shows a drastic reduction in power from 100% to 4%. This occurs because of the influence of the feedback reactivity coefficient of the fuel. For the feedback reactivity coefficient of the fuel, it is affected by the temperature of the fuel itself. The change
in the feedback reactivity coefficient of the fuel has the same characteristics as the change in fuel temperature with the opposite value.

![Figure 4. Graph of Power During ULOF on MOSART](image1.png)

When the fuel temperature rises, the feedback reactivity coefficient of the fuel drops. And so on until the condition is stable. From the feedback reactivity coefficient of this fuel, it affects the number of delayed neutrons on the terrace, and it is relevant with the increase in fuel temperature. The effect is on variations in reactor power during transients. Negative reactivity feedback which is affected by the fuel temperature increases while the insertion of positive reactivity by delayed neutrons is inhibited, so there is a power decrease according to Figure 4.

Furthermore in Figure 6 and Figure 7, they show the characteristics of changes in graph temperature and the feedback reactivity coefficient of graphite during transients. The characteristics of graphite temperature changes and the feedback reactivity coefficient of graphite shows similarities with the opposite values.
Figure 6. Graph of Temperature of Graphite During ULOF on MOSART

Figure 7. Graph of the Reactivity Coefficient of Feedback from Graphite During ULOF on MOSART

Unprotected Overcooling Accident (UOC) Calculations

In this case, the fuel temperature is assumed to decrease by 100°C in 60 seconds as shown in the graph in Figure 8 (black lines). Whereas the red line shows the temperature changes that occur in graphite.
Because the reactivity coefficient of the fuel is negative, the reactivity feedback from the fuel is positive, and it causes the power (black lines) to rise to 2.7 as shown in Figure 9. The red line in Figure 9 shows the period of the cooling flow rate, which is constant during the UOC transient.

In Figure 8 it can also be seen that the temperature of graphite rises gradually, which inserts negative reactivity feedback to balance the positive reactivity which is provided by the fuel which has little effect on the power and causes the temperature of the fuel to stabilize. Figure 10 shows changes in reactivity feedback due to temperature changes (grey lines) and graphite (black lines).
4. Conclusions

In this study kinetic models for liquid fuel reactors were applied. The simulated basic transients are ULOF and UOC with parameters in the MOSART design. The period of cooling flow rate is considered in ULOF transient initiation. In ULOF transients the loss of cooling circulation in the primary system produces two effects, the first is reduction of disintegration in delayed neutron on the core which causes insertion of reactivity, the second is increasing fuel temperature on the core. In the UOC transient the reactivity coefficient of the fuel is negative, then the reactivity feedback from the fuel is positive, it causes the power rise to 2.7 and the graphite temperature rises gradually, which inserts negative reactivity feedback to balance the positive reactivity which is provided by fuel which has little effect power and causes the fuel temperature to stabilize. The results of the reactor accident simulation are the power distribution and temperature on the reactor core to time and show that the MOSART conceptual design is an inherently stable reactor design. The calculation result also shows the conformity with previous the references (Zhang D, et al., 2009 & 2015).

Acknowledgements

ITB reasearch program and research program of ministry of research technology and higher education for the research activities and publication supports.

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