Delayed Neutrons Effect on Power Reactor with Variation of Fluid Fuel Velocity at MSR Fuji-12

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Abstract. As the nuclear reactor operate with liquid fuel, controlling velocity of the fuel flow on the Molten salt reactor very influence on the neutron kinetics in that reactor system. The effect of the pace fuel changes to the populations number of neutrons and power density on vertical direction (1 dimension) from the first until fifth year reactor operating had been analyzed on this research. This research had been conducted on MSR Fuji-12 with a two meters core high, and LiF-BeF$_2$-ThF$_4$-233UF$_4$ as fuel composition respectively 71.78%-16%-11.86%-0.36%. Data of reactivity, neutron flux, and the macroscopic fission cross section obtained from ouput of SRAC (neutronic calculation code has been developed by JAEA, with JENDL-4.0 as data library on the SRAC calculation) was being used for the calculation process of this research. The calculation process of this research had been performed numerically by SOR (successive over relaxation) and finite difference methode, as well as using C programing language. From the calculation, regarding to the value of power density resulting from delayed neutrons, concluded that 20 m/s is the optimum fuel flow velocity in all the years reactor had operated. Where the increases number of power are inversely proportional with the fuel flow speed.

1. Introduction

Molten Salt Reactor is more unique than other Generation IV nuclear reactors concepts, because the fuel state of this reactor is liquid, and the reactor system based on the circulation of the molten salt in the core reactor and the heat exchanger. In consequence, the neutron kinetics calculation analysis of the MSR is difference with the other type of reactor.

The neutron kinetics calculation study of MSR has been discussed in the literature [1-3]. On these publications have described that the neutron kinetics calculation procedure of MSR is different from that of the conventional reactors are use solid fuel.
The number of delayed neutron precursors influenced by the fuel velocity. However, under the steady state condition, the fuel salt velocity has little effect on the fast and the thermal fluxes [2]. Study on effect during transient condition, respectively; the pump coastdown transient, rods drop transient, and the inlet temperature drop transient against the relative power had been calculated by the code DRAGON [2]. Delayed neutron and temperature feedback is useful in providing an estimate of the transient behavior of the reactor power [3, 4].

Many scientist have focused in short time scale of the neutron kinetics study, to analyze the safety system reactor, such as disturbance or accident; reactor start up; and short modeling experiment [2-5]. However, study about neutron kinetics in long time scale is also important on this research area, because it has related with nuclear fuel cycle study; fuel burnup and waste management. The ability to reduce nuclear waste and nuclear weapon materials is one advantages by MSR [6, 7]. This paper describes the solution of the neutron kinetics study, especially delayed neutron and power reactor have produced when the reactor operate during five years.

2. MSR FUJI-12 Description
The main features of MSR Fuji-12 is shown at Table 1. Fuji-12 is the smallest MSR has been developed for energy production with total power output around 100 MWe [8]. The core size is shown in the third column of Table 1. Fuji-12 core consists from hexagonal cylinders assembly with flow channel for fuel in the middle, and it has made from graphite with number density is 1.84 g/cm³ [9]. Equivalent diameter of the assembly is 20 cm, and the flow channel diameter is 10.95 cm. Graphite assembly is also functioned as moderator with several conditions; low absorption cross sections, resistant to height temperatures, high thermal conductivity, and has fine mechanical strength.

Figure 1 interpretes that the lifetime of this sistem is more than 15 years, it is depended by the strength of the graphite. Refueling times is influenced by the reactivity of the reactor, where the fissile materials composition has important role to the number of reactivity [10].

| Design Parameters          | Specifications |
|----------------------------|----------------|
| Thermal output             | 350 MW         |
| Efficiency                 | 40%            |
| Core Geometry              |                |
| High                       | 400 cm         |
| Diameter                   | 400 cm         |
| Fuel Salt Composition      |                |
| LiF                        | 71.78 mol %    |
| BeF₂                       | 16.00          |
| ThF₄                       | 11.86          |
| ²³⁵UF₄                     | 0.36           |
| Fuel Salt Temperature      |                |
| Inlet                      | 833 K          |
| Outlet                     | 973 K          |
| Refueling Time             | 5 years        |
| Average Power Density      | 7 kW/ liter    |
3. Theory and Calculation

On the molten salt reactor, delayed neutron precursors drift with the fuel salt flow. This conditions make the neutron kinetics characteristics of the molten salt reactor difference than others solid-fuel reactors. The neutron kinetics equations with one group of flux neutron are given:

\[
\frac{\partial n}{\partial t} + \nabla \cdot \left( U n \right) = \beta \nu \sum_f \phi (r, z, t) + \sum_{i=1}^{6} \lambda_i C_i (r, z, t) \quad (1)
\]

\[
\frac{\partial C_i}{\partial t} + \nabla \cdot \left( U C_i \right) = \beta_i \nu \sum_f \phi (r, z, t) - \lambda_i C_i (r, z, t) \quad (2)
\]

On the steady state conditions, time has no affect on the neutron density and the average density of delayed neutron precursors, so that we can assume as: \( \frac{\partial n}{\partial t} = 0 \) and \( \frac{\partial C_i}{\partial t} = 0 \). Moreover, those equations can be simplified because the calculation method of this research is only considering fuel displacement at vertical direction (z axial). Where the fuel salt is moving form bottom to top of channels at the core reactor. Respectively, the equation are given:

\[
U_z \frac{\partial n}{\partial z} = \beta \nu \sum_f \phi (x) + \sum_{i=1}^{6} \lambda_i C_i (z) \quad (3)
\]

\[
U_z \frac{\partial C_i}{\partial z} = \beta_i \nu \sum_f \phi (z) - \lambda_i C_i (z) \quad (4)
\]

Boundary condition a given from bottom to top by the core reactor. When the fuel salt is circulating with velocity \( (U_z) \), circulate time of fuel salt in the external loop can be defined as \( (\tau_L) \) [5,9]. Reduction the average density of delayed neutron precursors at the fuel salt in the external loop is depended by decay constant and external loop transit time, mathematically can be difined at Eq. (5). For the neutron flux, boundary condition can be assumed as \( \phi (z) = 0 \) at the bottom and top of the core reactor.

\[
\frac{dC_i (\tau_L)}{d\tau} = \lambda C (\tau_L) \quad (5)
\]

When the high of active core is 2a, accordingly the boundary condition for neutron flux and density of delayed neutron precursors read as:

\[
\phi (-a) = \phi (a) = 0 \quad (6)
\]
| Group | $\beta_i$ | $\lambda_i$ |
|-------|----------|----------|
| 1     | 2.58     | 0.0129   |
| 2     | 6.96     | 0.0347   |
| 3     | 5.46     | 0.119    |
| 4     | 10.9     | 0.288    |
| 5     | 3.59     | 0.805    |
| 6     | 1.26     | 2.47     |

\[ C(-a) = C(a)e^{-\lambda L} \quad (7) \]

When the reactor operate on the thermal neutron with $U^{233}$ as fuel, the power produced from delayed neutron can be calculated from the following equation [14];

\[ P(W/cm^3) = \frac{\Sigma f W n}{3.17 \times 10^{10}} \quad (8) \]

4. Results and Discussion

As same as with prompt neutron, fission reaction is also occure on delay neutron. Analysis of the delayed neutron in the core reactor is assisting in investigating the characteristic of the core reactor. Changes velocity of the fuel flow affected on the power had been generated by delayed neutron. Power dropped exponentially when the velocity of fuel flow was being increased. Shown in figure 2. The reactor had operated in five years, and generally all graphics of the power for every years dropped exponentially. One of the interesting points of the graphics is when the reactor operate at 0.2 m/s of fuel flow velocity. The power significantly decreas at 0.05 m/s to 0.2 m/s of fuel flow velocity, and slopingly at 0.2 m/s to 0.5 m/s.

Basically, changes of the power was affected by the number of neutrons at the core reactor. Where the number of neutrons was affected by fluid fuel velocity.

![Figure 2](image_url)
Distribution of the neutrons is shown at figure 3. The distribution number of delayed neutron between fourth and fifth years operation when the reactor had operated on 0.05 m/s fuel flow velocity are similar. But, it is look difference for distribution number of delayed neutron has operated on 0.2 m/s.

On the SRAC calculation was found that the critically condition occur on the fourth year reactor operated with number multiplication effective is 1.01547, and the number multiplication effective on the fifth year reactor operated is 0.98527. It because reduction number of fissile materials impacting on reduction number of neutrons (prompt and delayed). So that, when the reactor operated on the subcritical condition (on the fifth year operation), it status can be increased on critical condition with reducing velocity of the fuel flow, and assuming the availability of fissile materias are still quite.

On the other hand, can be conclude that increasing the reactor status from subcritical to critical condition on the fifth year reactor operated will increasing the utilization of fertile materials (more efficient). So that, the number of fertile materials on the spent fuel will decrease.

5. Conclusion
The result of this research show that velocity of the fuel salt has effect on number of delayed neutron distribution at core reactor and the power had been produced. Optimization the utilization of fuel materials (fissile materials) is a benefit from reactor with fluid fuels. From the result can be predicted that the 0.2 m/s is the normal condition by velocity of fluid fuel during reactor has operated.

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