A DYNAMIC ANALYSIS OF CIRCULATING FUEL REACTOR WITH ZERO DIMENSIONAL MODELING

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INTRODUCTION

The Generation IV International Forum (GIF-IV) [1] has identified the Molten Salt Reactor (MSR) as one of the innovative nuclear reactor design which promises to serve various purposes, such as breeding of nuclear fuel, actinides transmutation and high temperature operation for electricity and hydrogen production [2, 3]. These attractive features result in a renewed interest in circulating fuel reactor especially in the Molten Salt reactor. Transient study of circulating fuel based system requires modification to conventional reactor simulation tool or development of dedicated computational method to take into account drift of delayed neutron precursor (DNP) and the subsequent decay in out of core external primary loop [4]. It is also possible that the delayed neutron precursors re-entering the core depending upon the flow rate and result in the feedback. This pose a new challenge from the perspective of mathematical and numerical schemes for simulating the dynamic behavior of nuclear power plants [5]. The circulation of fuel salt serves the purpose of coolant as well as fuel simultaneously. This results in strongly coupled neutronics and thermo-hydraulics system. These special features require a modification to mathematical models applicable to static fuel system. In this work, a simplified approach is adopted based on the modified space-independent point kinetics equations which take into account the effect of the fuel salt circulation along with zero-dimensional thermo-hydraulic equations [6]. Six group delayed neutron approximation results in seven point kinetics equations. The zero-dimensional thermal-hydraulic model consists of two equations describing the simple energy balance for the molten salt and the graphite. These two equations reduce to one if MSR core does not have any graphite moderator (For example: Molten Salt Fast Reactor). A program has been developed to solve these first order coupled set of equations using delay differential equation solver (dde23) from pydelay [6]. The MSRE design data for both $^{232}$Th-$^{233}$U and $^{238}$U-$^{235}$U fertile-fissile based fuel has been used to validate the developed simplified simulation tool. The theoretical and experimental data from Oak Ridge National Laboratory (ORNL) [7] provided from the Molten Salt Reactor Experiment (MSRE) operation is used as reference to validate the computational tool.

The point-kinetics model used in the present work has two additional terms $\frac{c_i}{T}$ and $\frac{\frac{c_i(t-t_0)}{T}}{T}$ in order to account for delayed neutron precursors are leaving and entering the reactor core [8].

\begin{equation}
\frac{dn(t)}{dt} = \frac{(\rho-\beta)}{\lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \tag{1}
\end{equation}

\begin{equation}
\frac{dC_i(t)}{dt} = \frac{\rho_i}{\lambda} n(t) - \lambda_i C_i(t) - \frac{c_i}{T} + \frac{c_i(t-t_0)}{T} e^{-\lambda_i t} \tag{2}
\end{equation}

Using equations (1) and (2) loss of reactivity due to drift of DNP is evaluated and given by equation (3). This loss of reactivity needs to be supplied through withdrawal of control system or fuel feed in order to maintain criticality. It depends on flow rate of fuel salt which couples the thermal-hydraulic and neutronics.

\begin{equation}
\rho_d = \beta - \sum_{i=1}^{6} \frac{\rho_i}{\lambda_i + \frac{1}{\lambda_i} e^{-\lambda_i t}} \tag{3}
\end{equation}

The zero-dimensional thermal-hydraulic model for the molten salt and the graphite is presented by equations (4) and (5).

\begin{equation}
\frac{dT_s(t)}{dt} = \frac{Yu(t)}{M_s \rho_p s} + \frac{\rho_s}{M_s \rho_p s} (T_g(t) - T_s(t)) + \frac{\rho_s}{\tau_c} \frac{\theta_{out} - \theta_{in}}{\tau_c} \tag{4}
\end{equation}

\begin{equation}
\frac{dT_g(t)}{dt} = \frac{(1-\gamma) p(t)}{M_g \rho_p g} - \frac{\rho_s}{M_g \rho_p g} (T_g(t) - T_s(t)) \tag{5}
\end{equation}

\begin{equation}
T_s(t) = \frac{\theta_{in} + \theta_{out}}{\tau_c} \tag{6}
\end{equation}
RESULTS AND DISCUSSION

The loss of reactivity at reference velocity of fuel salt for $^{233}$U and $^{235}$U has been computed and compared with the reference values from MSRE [7] and found a reasonable agreement. Next, transient analysis have been carried out for step reactivity insertion up to 300 pcm (per cento mile=10^{-5}) at nominal power (8MW) with nominal fuel flow rate (0.24cm/sec). It is found that the peak power computed by the point model results in higher value as compared to the reference result [4].

The temperature transient of graphite and fuel salt is evaluated for 300 pcm step reactivity and compared with reference result [4]. For the temperature transient, both trends and final values are slightly different. It can be seen that the power transient with step reactivity insertion is inherently self-limiting by virtue of the negative temperature co-efficient of reactivity of the system.

CONCLUSIONS

In this paper, power and temperature transients of molten salt reactor with $^{232}$Th-$^{233}$U and $^{238}$U-$^{235}$U based fuel salt have been analyzed using a zero-dimensional model of neutronics and thermal hydraulics. This gives a useful estimate of power and temperature peak values during transient. However, these models are inadequate to predict the accurate results and design analysis requires development of multi dimensional and multi physics models.

NOMENCLATURE

$n(t)$: neutron density; $C_i(t)$: ith DNP group density; $\lambda_i$: ith group DNP decay constant; $\rho_0$: loss of reactivity in steady state; $\rho$: reactivity; $\beta_i$: ith group delayed neutron fraction; $\beta$: total delayed neutron fraction; $\tau_1$: Loop transit time(s); $\tau_2$: Core transit time(s); $t$: time (s); $\Lambda$: neutron generation time (s); $\gamma$: fraction of power released in the salt; $P(t)$: core power (W); $T$: Temperature (K); $U$: overall heat transfer coefficient between salt and graphite (W/K); $M$: Mass(kg); $c_p$: specific heat (Jkg^{-1}K^{-1}); $F$: mass flow rate (kg/sec).

REFERENCES