ULOF Accident Analysis for 300 MWt Pb-Bi Coolled MOX Fuelled SPINNOR Reactor

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Abstract

In this study the result of investigation through simulation of unprotected loss of flow accident (ULOF) for 300 MWth MOX fuelled small Pb-Bi Cooled non-refuelling nuclear reactors (SPINNOR) are discussed. The two dimensional diffusion calculation combined with transient thermal hydraulic analysis has been employed. The reactor is tank type Pb-Bi cooled fast reactors with steam generator included inside reactor vessel. The simulation begin with steady state calculation of neutron flux, power distribution and temperature distribution across the core, hot and cool pool, and also steam generator. The accident analysis begin with the loss of pumping power. The sequence of analysis is then the transient flow calculation across the core, the core temperature distribution, hot pool and cool pool fluid dynamic calculation and temperature change simulation, and the transient flow and temperature calculation across the steam generator. Then the reactivity feedback calculation is conducted, followed by kinetic calculation, and then the sequence repeated. The results show that the SPINNOR reactor has inherent safety capability against this accident.

Keywords: ULOF accident, Pb-Bi Cooled, SPINNOR reactor

1. Introduction

Since accident of Chernobyl, TMI II and the latest one Fukushima, the attention focuses on safety aspect of nuclear reactors. Therefore, development of safety analysis refers to inherent safety that claims simulation system which is more sophisticated.   Wherever practicable, inherent safety characteristics have been incorporated into the design of systems important to safety, particularly that fulfill the three basic safety functions of reactor shutdown, decay heat removal and the containment of radioactive materials (Zaki at al., 1996) .

The SPINNOR (Small Power Reactor, Indonesia, No On-Site Refueling) are concepts of small lead-bismuth cooled nuclear power reactors with fast neutron spectrum that could be operated for more than 15 years without on-site refueling. They are based on the concept of a long-life core reactor developed in Indonesia since early 1990 in collaboration with the Research Laboratory for Nuclear Reactors of The Tokyo Institute of Technology (Zaki, 2007).

The reactor is used in relatively isolated areas, preferably Small Islands, and operated to the end of its life without refueling or fuel shuffling. Some important requisite characteristics are easy operation, easy maintenance, transportability, inherent/passive safety and nuclear proliferation resistance. In this paper we analize the safety performance of SPINNOR reactor. By the term safe here we mean that reactor is able to survive ULOF (Unprotected Loss of Flow) accident without reactor scram or the help of the operator.

2. Design Concept of SPINNOR Reactor

A schematic view of the system being analyzed is shown in Fig.1. The overall system includes a reactor core, hot pool, cool pool, steam generator and pump. The intermediate heat exchanger (IHX) is eliminated, and heat from primary coolant system is transferred directly to the steam-water loop through the steam generator. The coolant flows trough the core removes heat generated in the core, and then flows up to the hot pool. From the hot pool, coolant flows into the steam generator, transferring the heat into the steam-water loop, and then goes down to the cool pool. From the cool pool the coolant is pumped back to the core (Zaki, 2007).

3. Calculation Model

3.1 Calculation of Neutron Distribution Flux

If we evaluate reactor core in the steady state, then time variable can be eliminated. We assume homogeneous material core region and adopt diffusion equation in this model (Duderstadt at al.,1976):

$$-\nabla , D_g \nabla \varphi_g (r) + \Sigma_{rg}\, \varphi_g (r) = \frac{x_g}{k_{eff}} \sum_{g^1=1}^{G} \nu \sum_{fg^1} \varphi_{g^1}(r) + \sum_{g^1=1}^{G} \Sigma_{sg^1 \to g}\, \varphi_{g^1}(r)$$

(1)

D_g = Diffusion constant for group g

ϕ_g = Neutron flux for group g

Σ_{rg} = Cross-section removal for group g

χ = Fission spectrum fraction

$keff$ = Effective critically factor

υ = Number of neutron produced per fission

$\Sigma_{sg' \to g}$ = Macroscopic cross-section scattering from   group g' to group g

After some simplification one of reasonable model for the fast reactor safety analysis is the adiabatic mode with the shape function becomes :

$$-\vec{\nabla} , D_9 \vec{\nabla} \Psi_9 (\vec{r},t) + \Sigma_{rg}\, \Psi_g (\vec{r},t) = \frac{x_g}{k_{eff}} \sum_{g^1=1}^{G} \nu \sum_{fg^1} \Psi_{g^1}(\vec{r},t) + \sum_{g^1=1}^{G} \Sigma_{sg^1 \to g}\, \Psi_{g^1}(\vec{r},t)$$

(2)

Diffusion equation of multi group is solved in numerical with method SOR (Succesive Over Relaxation).

Neutron population in reactor core during transient process is determined by solving of point kinetics equation. If transformation of spatial distribution can be eliminated, then we can obtain level of reactor power as function time p(t) by solving of point kinetics equation   as follows, (Duderstadt at al.,1976):

$$\frac{dp(t)}{dt} = \left[\frac{\rho(t) - \beta}{\Lambda} \right] p(t) + \sum_{i=1}^{6} \lambda_i C_i \tag{3}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} p(t) - \lambda_i C_i(t) \tag{4}$$

ρ = Reactivity (dk/k)

β = Effective delay neutron fraction

λ = Delayed neutron fraction

C_i = Delay neutron precursor concentration

Λ = Neutron generation time

3.2 Thermal Hydraulic Calculation

Calculation of thermal hydraulic covers calculation distribution of temperature in all parts of reactor that is in reactor core and in steam generator (SG) that is fuel temperature, coolant temperature and temperature cladding and gap, so pressure in reactor and all coolant circulation aspects in reactor.

Figure 2 shows thermal hydraulic model in this calculations. Reactor is divided into concentric ring, where the cross-flow between two adjacent rings is assumed zero. For coolant, calculation used mass and energy conservation equation:

$$\rho c_p \frac{\partial T}{\partial t} - w c_p \frac{\partial T}{\partial z} = Q''' \tag{5}$$

ρ = Mass density

w   = coolant mass flow rate

c_p = Specific heat at constant pressure

Q''' = Power density

Pressure drop calculation is solved momentum conservation equation.

$$\frac{\partial G}{\partial t} = -\frac{\partial P}{\partial z} - \frac{\partial}{\partial z} \frac{G^2}{\rho} - \frac{fG^2}{2D_e \rho} - \rho g \tag{6}$$

G = Total mass flow

P = Pressure

f = friction factor

D_e = Effective hydraulic diameter

g = gravitational acceleration

3.3 Hot pool and Cool Pool

During this analysis, average temperature in both hot pool and cool pool are used .

$$T_{hp} = [(h_{hp} A_{hp} \rho_{hp} - G_{psg} \Delta t) C_{php} T_{hp}^{old} + G_{core} \Delta t C_{pcore} T_{core}] / [(h_{hp} A_{hp} \rho_{hp} - G_{psg} \Delta t + G_{core} \Delta t) C_{php}] \tag{7}$$

$$h_{hp}^{i+1} = (h_{hp}^{j} A_{hp} \rho_{hp} + (G_{psg} - G_{core}) \Delta t) / (\rho_{hp} A_{hp}) \tag{8}$$

T_{hp} = Temperature of hot pool

h_{hp} = Height of hot pool

A_{hp} = Area of hot pool

ρ_{hp} = Mass density of hot pool

G_{psg} = Total mass flow rate in primary SG

C_{php} = Specific heat of hot pool

G_{core} = Total mass flow rate of core

C_{pcore} = Specific heat of core

Similarly with average hot pool, temperature cool pool is solved these equations:

$$T_{cp} = [(h_{cp}A_{cp}\rho_{cp} - G_{psg}\Delta t)C_{pcp}T_{cp}^{old} + G_{core}\Delta tC_{pcore}T_{core}]/[(h_{cp}A_{cp}\rho_{cp} - G_{psg}\Delta t + G_{core}\Delta t)C_{pcp}] \tag{9}$$

$$h_{cp}^{i+1} = (h_{cp}^{j}A_{cp}\rho_{cp} + (G_{psg} - G_{core})\Delta t)/(\rho_{cp}A_{cp}) \tag{10}$$

T_{cp}　= Temperature of cool pool

h_{cp}　= Height of cool pool

A_{cp}　= Area of cool pool

ρ_{cp}　= Mass density of cool pool

3.4 Feedback Calculations

$\rho(t)$ in Eq. (3) is summed external reactivity and feedback reactivity. Feedback reactivity is included Doppler, axial expansion, radial expansion, and void reactivity.

Feedback reactivity is the effect of the core temperature *T*. We can express temperature coefficient of reactivity:

$$\alpha \equiv \frac{\partial\rho}{\partial T} \tag{11}$$

α is a negative value that we inserted in calculation since increasing temperature will cause decreasing ρ. T is average temperature in core (fuel and coolant).

3.5 General Calculation Algorithm

Calculation method employed in the present study is two dimensional r-z geometry for diffusion calculation is carried out every year to get neutron flux distribution and power distribution.　The flow diagram of the general calculation algorithm is given in Fig. 3. For the first step, the steady state neutronic and the thermal-hydraulic calculation are performed, and then the accident condition is taken. The ULOF accident is started by failure of the primary pumping system. Temperature in core will be change depend on how much flow rate has inserted. Changing of core temperature will change average hot pool temperature and steam generator temperature. In this research reactivity external of reactor is assumed constant in every time. Heat will be transferred to water circulation in steam generator.

4. Result and Discussion

Table 1 shows the main reactor design parameter. Fig. 4 shows the reactor power change pattern during accident. The power increases at the beginning of the accident, then gradually decreases and stabilizes around 40% of initial power. Fig.5 shows temperature of fuel, cladding and coolant. The maximum fuel temperature was about 800,23　^{o}C, the maximum cladding temperature about 634,07^{o}C, and the maximum coolant temperature about 627,89^{o}C. Fuel, cladding and coolant temperature are still below the maximum temperature limits. Fig. 7 Show flow rate change during ULOF accident, there are also small oscillation in total core flow and primary SG flow rate, which are mainly caused by the fact that flow rate in the core and primary side SG in the absence of primary pumping is mainly controlled by level and temperature of coolant is the hot and cool pool which in turn depend on the flow rate of the core and primary SG.

5. Conclusions

The conclusions are summarized as follow,

1). In general the reactor can be survived the ULOF accident inherently.
2). The relatively high natural circulation component contribute important factor to this ability to survive ULOF accident.
3). Accident analysis result show of the case maximum coolant, fuel and cladding temperature are still below each temperature limit, and margin to coolant and fuel are large.

References

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Table 1**.** Main reactor design parameter

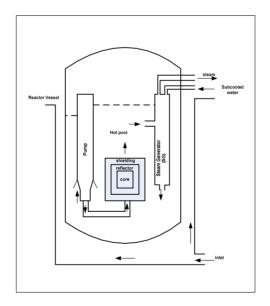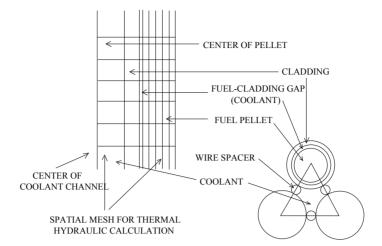| No	Parameter	Value
1	Reactor Power	300 MWth
2	Fuel	UO_2- PUO_2
3	Shielding Material	B4C
4	Coolant	Pb-Bi
5	Pin/Pitch diameter	1.2 cm
6	Pin diameter	1.0 cm
7	Cladding Thickness	0.8 cm
8	Total Flow rate	12000 kg/s

Figure 1. Overview of the SPINNOR reactor
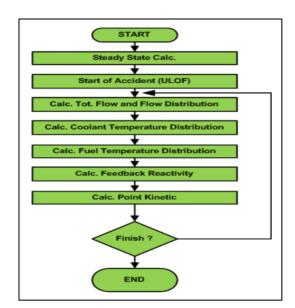
Figure 2. Calculation model for coolant channel
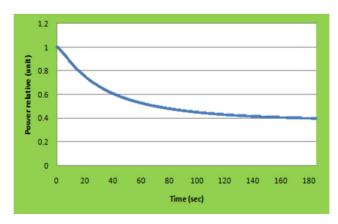

Figure 3. Calculational flow diagram

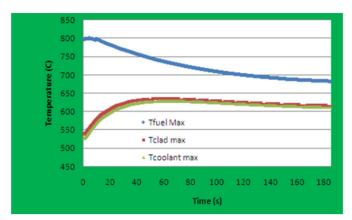Figure 4. Power change during ULOF accident

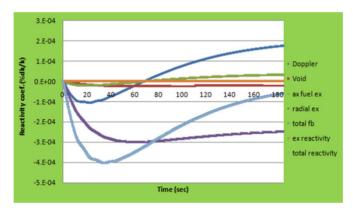Figure 5. Hot spot temperature during ULOF accident
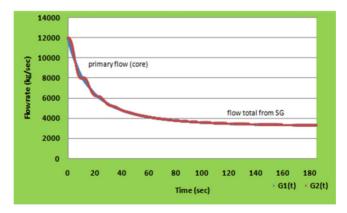

Figure 6. Reactivity component change during ULOF accident

Figure 7. Flow rate change during ULOF accident