

ULOF Accident Analysis for 300 MWth Pb-Bi Cooled MOX Fuelled SPINNOR Reactor

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Abstract. In This study the result of investigation through simulation of unprotected loss of flow accident for 300 MWth MOX fuelled small Pb-Bi Cooled non-refuelling nuclear reactors (SPINNOR) are discussed. During the analysis 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 accross the core, hot and cool pool, and also steam generator. Then the accident started by the loss of pumping power. The sequence 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 .

INTRODUCTION

Since accident of Chernobyl and TMI II, safety aspect of nuclear reactor is attention focus of the many scientists. Development of safety analysis go to inherent safety claims simulation system which more sophisticated. Wherever practicable, inherent safety characteristics have been incorporated into the design of systems important to safety, particularly those that fulfil the three basic safety functions of reactor shutdown, decay heat removal and the containment of radioactive materials [4].

The SPINNOR (Small Power Reactor, Indonesia, No On-Site Refuelling) 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 refuelling.

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.

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

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.

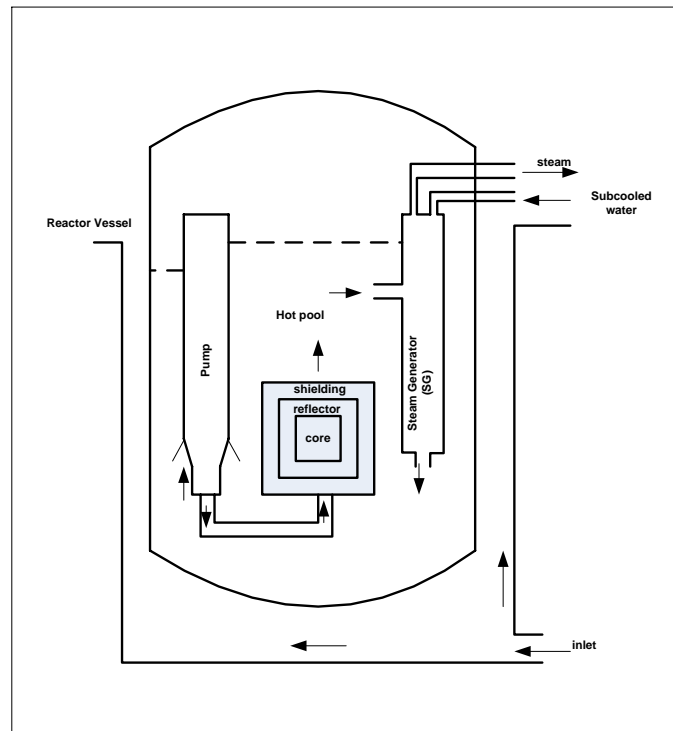


FIGURE 1. Overview of the SPINNOR reactor

CALCULATION MODEL

Calculation of distribution fluxes neutron

If we evaluate reactor core in the situation steady state, then time variable can be eliminated, and assumed material core region homogen, diffusion equation is adopted in this model [6]:

$$-\nabla \cdot D_g \nabla \phi_g(r) + \Sigma_{rg} \phi_g(r) = \frac{\chi_g}{k_{eff}} \sum_{g^1=1}^G \nu \sum_{fg^1} \phi_{g^1}(r) + \sum_{g^1=1}^G \Sigma_{sg^1 \rightarrow g} \phi_{g^1}(r) \quad (1)$$

- 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
- k_{eff} = Effective criticality factor
- ν = Number of neutron produced per fission
- $\Sigma_{sg^1 \rightarrow g}$ = Macroscopic cross-section scattering from group g^1 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 :

$$-\bar{\nabla} \cdot D_g \bar{\nabla} \Psi_g(\bar{r}, t) + \Sigma_{rg} \Psi_g(\bar{r}, t) = \frac{\chi_g}{k_{eff}} \sum_{g^1=1}^G \nu \sum_{fg^1} \Psi_{g^1}(\bar{r}, t) + \sum_{g^1=1}^G \Sigma_{sg^1 \rightarrow g} \Psi_{g^1}(\bar{r}, t) \quad (2)$$

Diffusion equation of multigroup is solved in numerical with method SOR (Successive Over Relaxation).

Neutron population in reactor core during transient process determined by solving of point kinetics equation . If transformation of distribution spacial can be eliminated, then level of reactor power as function time $p(t)$ obtainable with solving of point kinetics equation as follows, (Duderstadt, 1978) [6]:

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

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

- ρ = Reactivity (dk/k)
- β = Effective delay neutron fraction
- λ = Delayed neutron fraction
- C_i = Delay neutron precursor concentration
- Λ = Neutron generation time

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.

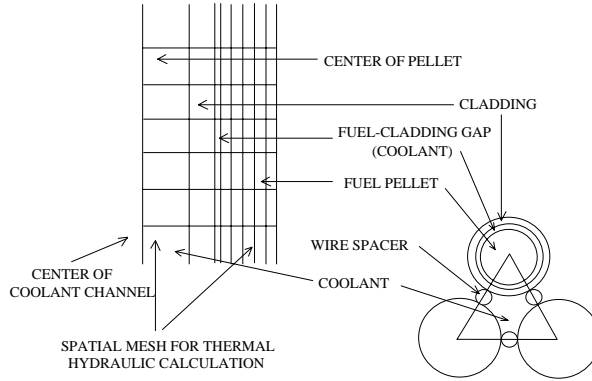


FIGURE 2. Calculation model for coolant channel

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:

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

- 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.

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

- G = Total mass flow
- P = Pressure
- f = friction factor
- D_e = Effective hydraulic diameter
- g = gravitational acceleration

Hot pool and Cool Pool

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

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

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

- 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:

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

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

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

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 [6]. We can express temperature coefficient of reactivity:

$$\alpha \equiv \frac{\partial \rho}{\partial T} \quad (11)$$

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).

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 given in Fig. 3.

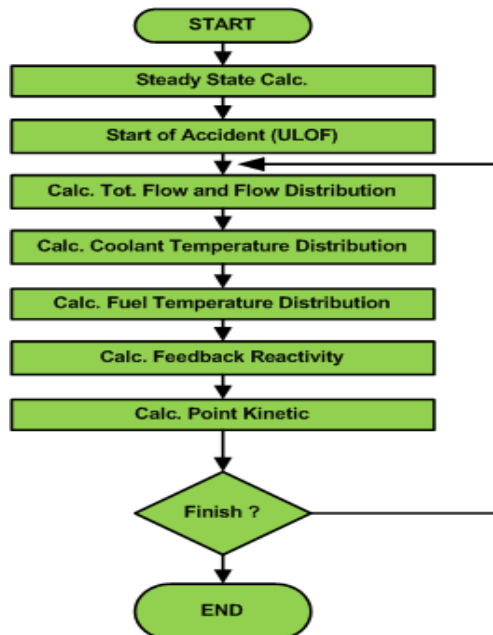


FIGURE 3. Calculational flow diagram

For the first step, the steady state neutronic and the thermal-hydraulic calculation are performed, then the accident condition are 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.

RESULT AND DISCUSSION

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 Flowrate	12000 kg/s

Table 1. show the main reactor design parameter. In this section we will discuss about calculation results includes power reactor, reactivity feedback, max of fuel and coolant temperature and total flowrate.

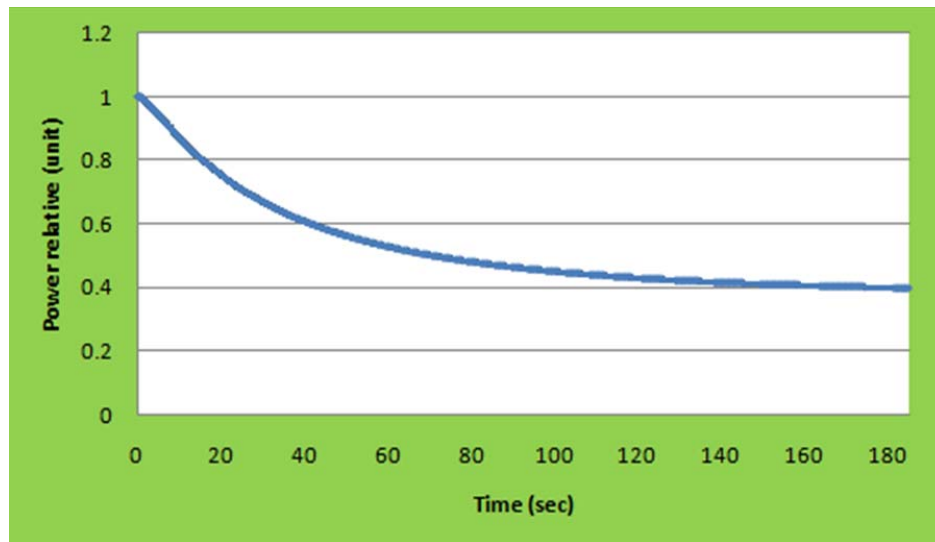


FIGURE 4. Power change during ULOF accident.

Fig. 4 show the reactor power change pattern during accident. The power increase at the beginning of the accident but then decrease and goes towards a new stable about 40% of initial power.

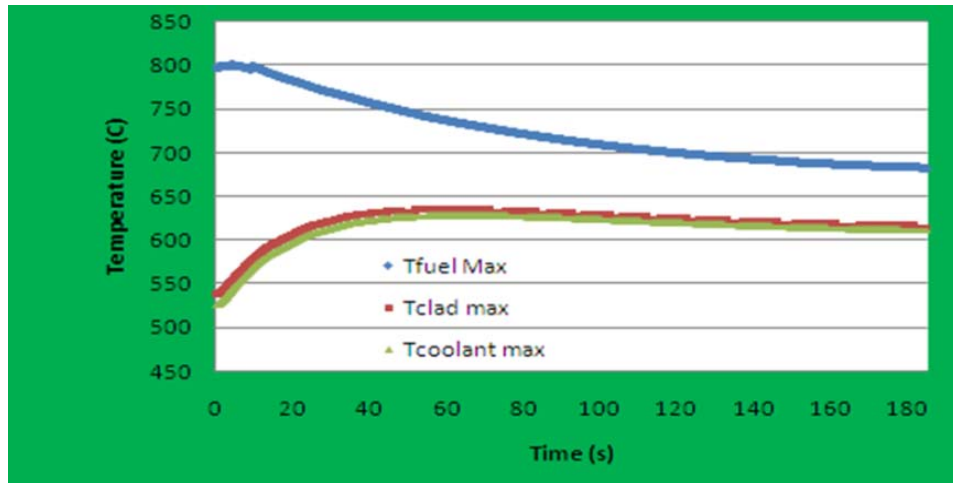


FIGURE 5. Hot spot temperature during ULOF accident.

Fig.5 show temperature of fuel, cladding and coolant. The maximum fuel temperature about $800,23^{\circ}\text{C}$, the maximum cladding temperature about $634,07^{\circ}\text{C}$, and the maximum coolant temperature about $627,89^{\circ}\text{C}$. Fuel, cladding and coolant temperature are still below the maximum temperature limits.

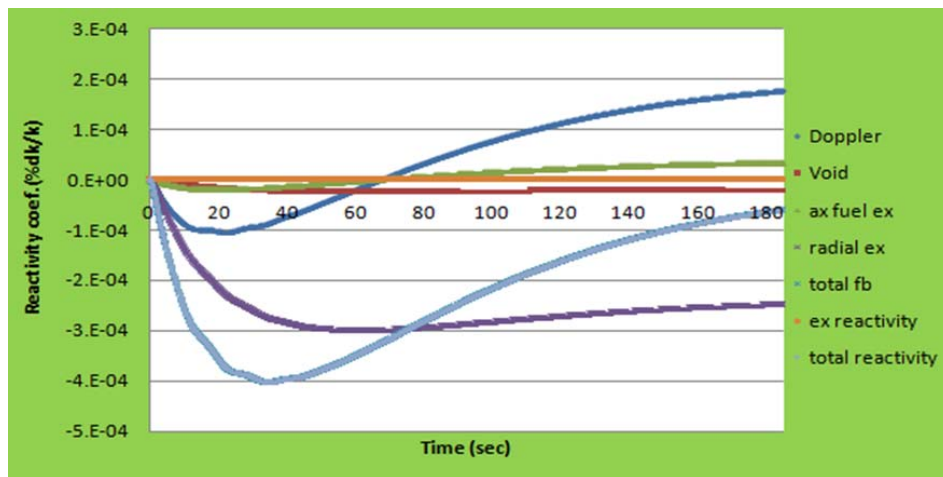


FIGURE 6. Reactivity component change during ULOF accident

Fig. 6 show of reactivity component change during ULOF accident, the most important feedback in this accident is core radial expansion. At the beginning stage of the accident the Doppler component is negative but then move to positive value near the asymptotic condition. At the asymptotic condition a new equilibrium condition is at reduced power level (40% of nominal power) which balanced natural circulation level.

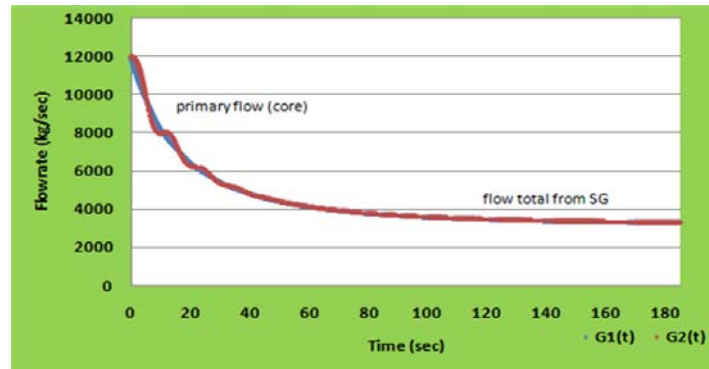


FIGURE 7. Flowrate change during ULOF accident

Fig. 7 Show flowrate change during ULOF accident, there are also small oscillation in total core flow and primary SG flowrate, 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 flowrate of the core and primary SG.

CONCLUSIONS

The conclusions are summarized as follow,

1. In general the reactor can survive 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.

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