Safety Analysis of Pb-208 Cooled 800 MWt Modified CANDLE Reactors

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Abstract. Safety analysis of 800MWt Pb-208 cooled fast reactors with natural Uranium as fuel cycle input employing axial-radial combined Modiified CANDLE burnup scheme has been performed. The analysis of unprotected loss of flow(ULOF) and unprotected rod run-out transient overpower (UTOP) are discussed. Some simulations for 800 MWt Pb-208 cooled fast reactors has been performed and the results show that the reactor can anticipate complete pumping failure inherently by reducing power through reactivity feedback and remove the rest of heat through natural circulations. Compared to the Pb-nat cooled long life Fast Reactors, Pb-208 cooled reactors have smaller Doppler but higher coolant density reactivity coefficient. In the UTOP accident case the analysis has been performed against external reactivity up to 0.003dk/k. And for ULOHS case it is assumed that the secondary cooling system has broken. During all accident the cladding temperature is the most critical. Especially for the case of UTOP accident. In addition the steam generator design has also consider excess power which may reach 50% extra during severe UTOP case..

1. Introduction

As the respons to the TMI II and Chernobyl accident Innovative Nuclear Power Plants with inherent safety capability have been widely developed with Lead cooled fast reactors as one of them. The key factors for lead cooled fast reactors inherent safety capability are their negative reactifity feedbacks and their natural circulation capability. There are four important feedback in lead cooled fast reactors: Doppler, fuel axial expansion feedback, core radial expansion feedback, and coolant density feedback. In case of sodium cooled fast power reactors which more widely developed in many countries the coolant density reactivity feedback is positive except in some special cases. On the other hands, most lead cooled fast reactors have negative coolant density reactivity feedback. In large lead cooled fast reactor core it may be positive but the absolute value is not so large.

Pb-208 has low capture cross section in the fast region, low inelastic capture cross section in the fast energy region which give harder spectrum, potential to improve coolant void coefficient, and potential to improve neutronic performance of small long life liquid lead cooled fast reactors. In general the use of Pb-208 as coolant give better coolant density reactivity feedback than that of natural lead. It also give better criticallity than that of natural lead. Therefore the usage of Pb-208 as coolant in this study can be expected will contribute significantly to the overall safety performance.[1-2]

In this study, accident analysis has been performed for Pb-208 cooled 800 MWt Modified CANDLE based fast reactors[3-5]. The accident analysis include unprotected loss of flow accident (ULOF accident) and unprotected rod run-out transient overpower accident (UTOP accident). In this analysis coupled neutronic and thermal hydraulic analysis which include adiabatic model in nodal approach of time dependent multi-group diffusion equations has been adopted. The thermal hydraulic model include transient model in the core, steam generator, and related systems. Natural circulation based heat removal system is important to ensure inherent safety capability during unprotected accidents. Therefore the system similar to RVACS (reactor vessel auxiliary cooling system) is one of the important part to ensure the inherent safety feature of these reactors.[6-13]

2. Methodology[1,6-14]

The detail model and mathematical formulations for the current safety analysis can be obtained in the following references. Here the algorithm of the simulation will be discussed. After calculating effective macroscopic group constant then steady state multi-group diffusion calculation and steady state thermal hydraulic calculation are performed. The accident simulation begins with the accident initiator such as withdrawal of all control rods in case of UTOP accident and total loss of pumping power in the primary system for the case of ULOF accident. Then the calculation of total coolant flow-rate and flow distribution across the reactor core are performed which followed by the calculation of coolant and fuel temperature distribution, the calculation of energy and mass balance in the steam generator. Next, we solve the kinetic equation after calculating current feedback reactivities (Doppler, coolant density, core radial expansion and fuels axial expansion reactivity feedbacks) which then used in the calculation of amplitude and shape functions. The calculation is repeated by going back to the calculation of total coolant flow-rate and flow distribution across the core until reach the end of the simulation time.

3. Results and discussions

Table 1 shows main paramters for sample parameters for the current simulations. The reactor power is 800 MWt and this is a medium sized Modified CANDLE burn-up system.

Table 1 Main paramters used in this simulations

Parameter	Value
Power (MWth)	800
Core Geometry	2-D Cylinder
Refueling Periode(years)	10 years x10 batch
Fuel/cladding/coolant type	UN and PuN/SS316/Pb-208
Active core radius/height	117 cm/237.5cm
Reflector width (Pb-208)	70 cm
Fuel/structure/coolant volume fractions	60/12.5/27.5

The simulation results for Unprotected Loss of Flow (ULOF) accident case are shown in figures 1-4. Figure 1 shows coolant flow-rate in the core during unprotected loss of flow accident. Figure 2 shows power change with time during this accident. Fig. 3 shows The coolant decrease triggers imbalance between produced heat and coolant flow-rate causing temperature increase in coolant and fuel. Figure 3 shows coolant, cladding and fuel temperature change with time during the accident and finally Figure 4 shows reactivity feedback change with time during the accident. Temperature increase in coolant and fuel produces negative reactivity feedback which drive reactor power to decrease. After

400 seconds the coolant flow rate and the reactor power are approaching asymptotic levels and the total reactivity approaching zero. The maximum temperature in fuel and coolant are still far below safety limit.

The flow-rate decrease basically influenced by natural circulation level of the thermal hydraulic system of the NPP. About 200s after the accident begins, the core and primary side SG flow-rate ave approached natural circulation levels and after 400 seconds the flow-rate change become relatively slow. In case of the reactor core power level, it also decreases following the decrease of the core flow-rate. However the asymptotic level is still at higher percentage compared to that of flow-rate.

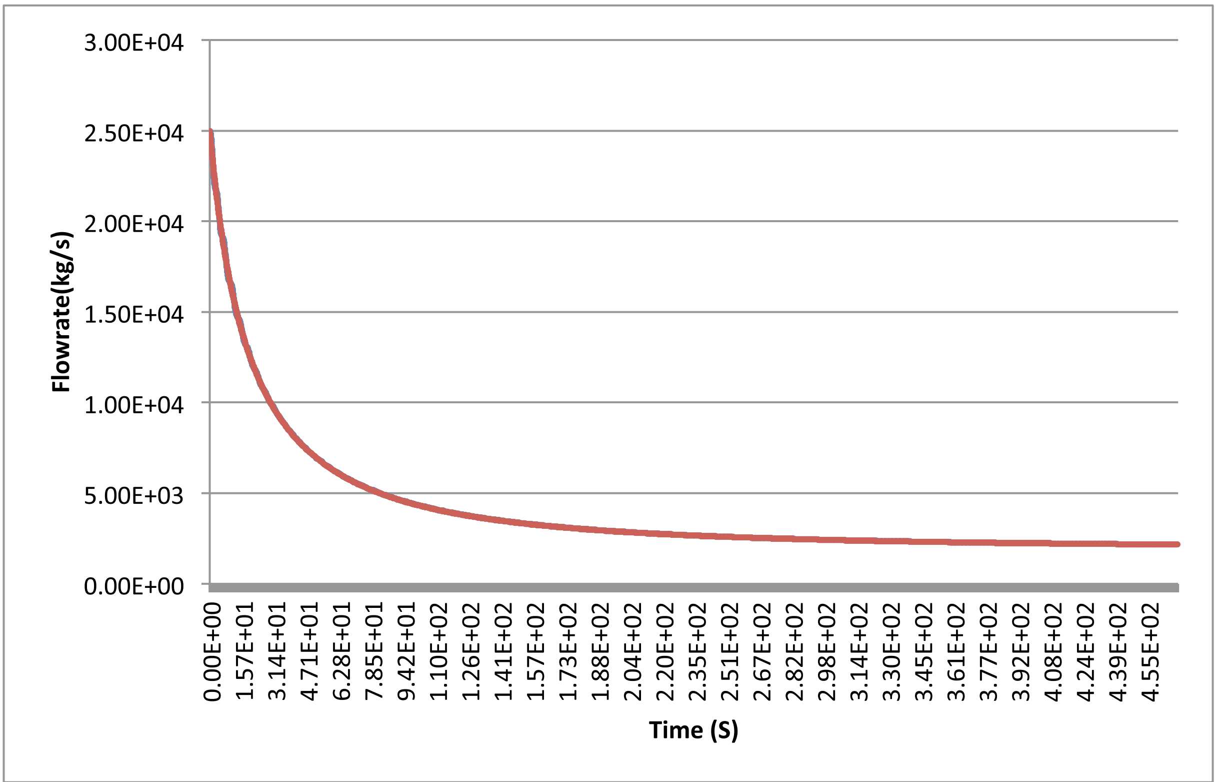

Figure 1 Coolant flow-rate change with time during loss of flow accident

After the decrease of the coolant flowrate, the fuel, cladding and cooolant temperatures initially increase and after reaching the peak then decrease and move toward new equilibrium level. The fuel temperature experience more significant reduction compared to the coolant and cladding temperature. This situation can be explained due to relatively large coolant-center fuel temperature different in the fuel pin. Due to the change of the fuel to coolant temperature difference during the accident the position of maximum fuel, cladding and temperatures are basically changing with the time and may not in the same positions for each coolant maximum temperature, cladding maximum temeprature, and fuel maximum temperature.

The reactivity feedbacks move in more complex situations. The Doppler and fuel axial expansion at the beginning give strong negative feedback, however, after reaching maximum temperature then their absolute values decerase and finally move toward positive direction. On the other hand, the core radial expansion and the coolant density reactivity feedback are definitely negative. However their values also decrease after reaching their maximum peak(absolute).

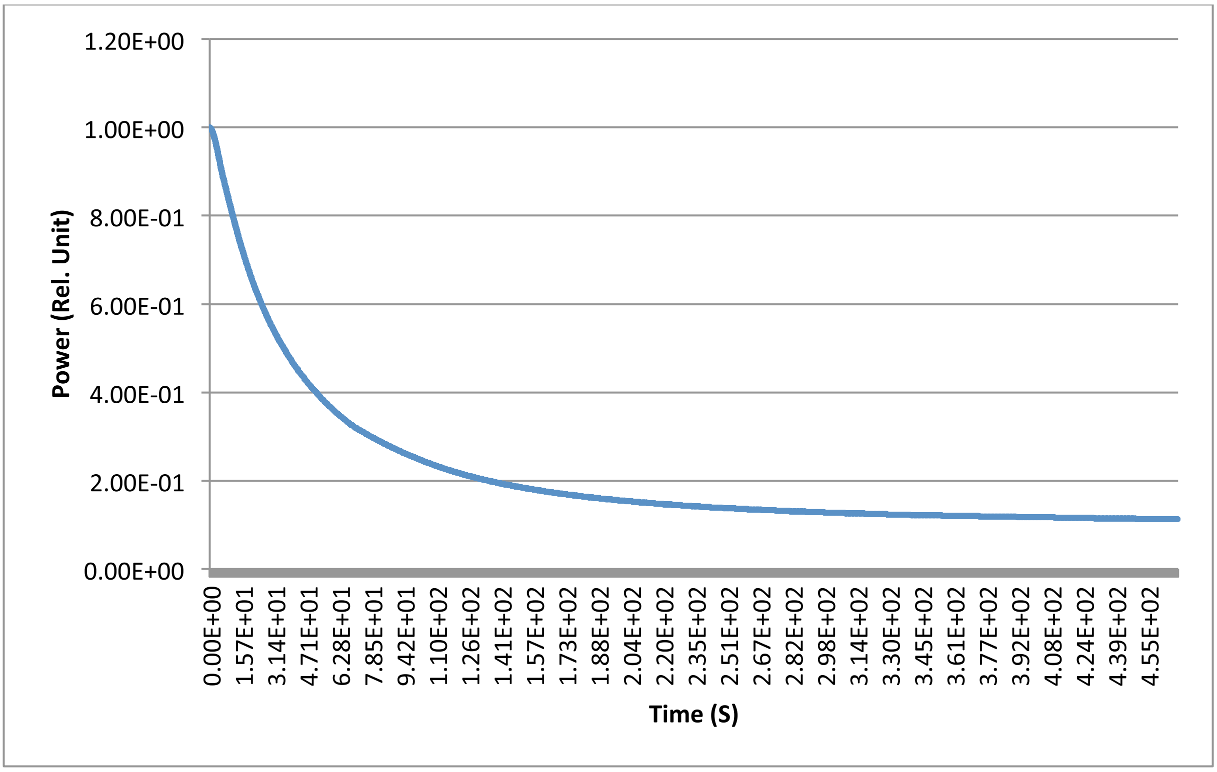

Figure 2 Power change with time during loss of flow accident

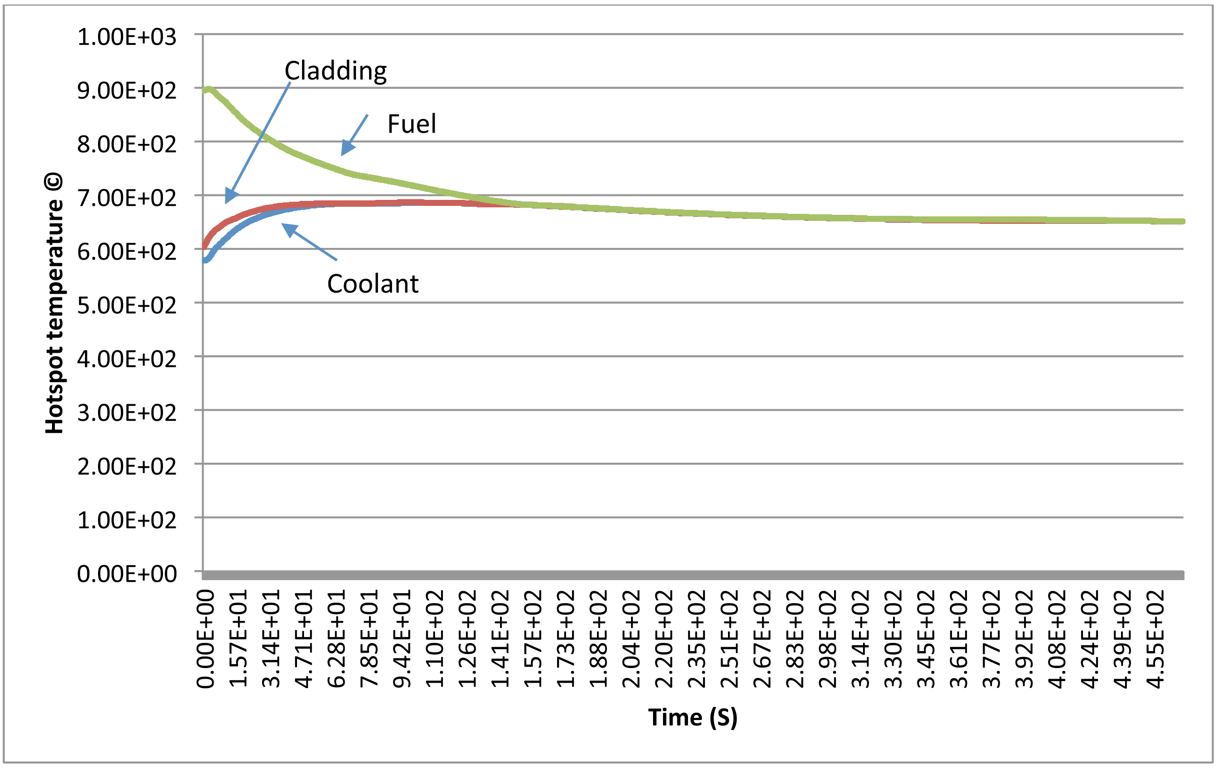

Figure 3 Hot spot temperatures change with time during loss of flow accident

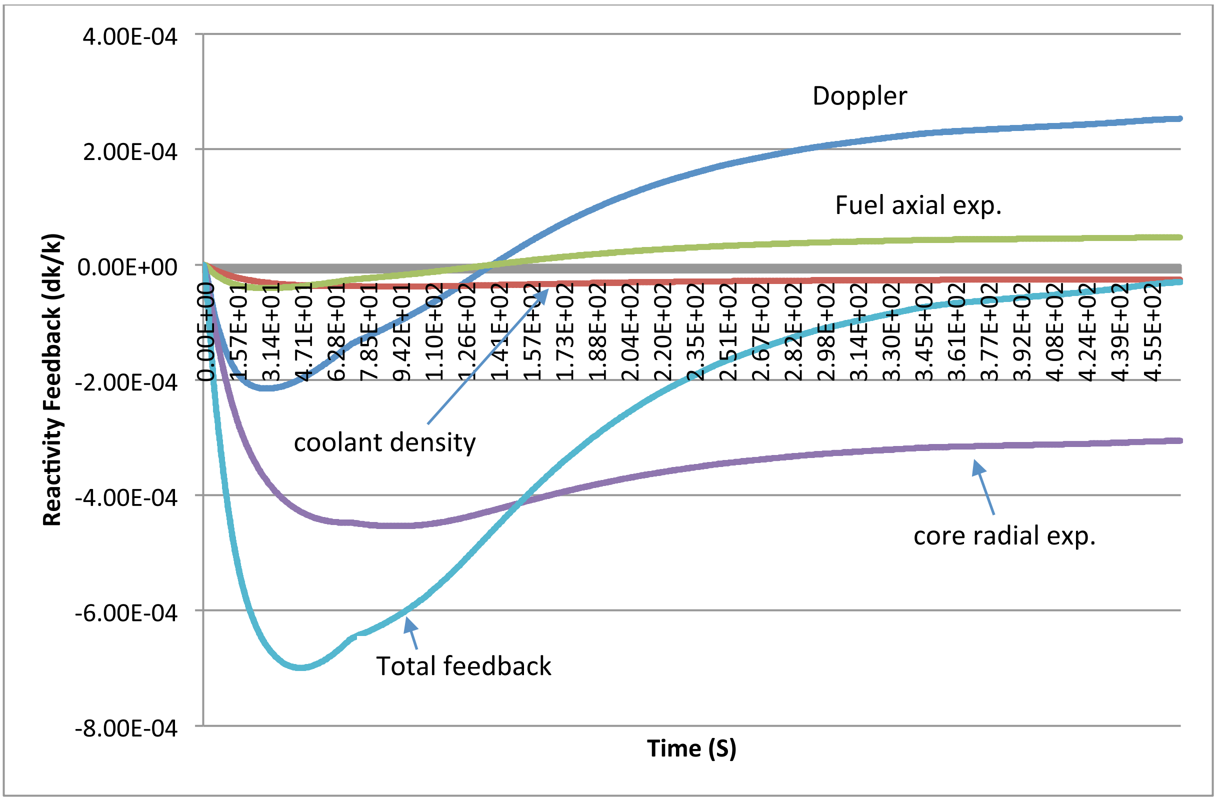

Figure 4 Reactivity feedback change with time during loss of flow accident

The accident analysis simulation results for Unprotected rod run-out Transient Over Power (UTOP) accident are shown in figures 5-8. As the accident intitator, the positive external reactivity triggers power increase which then causes temperature increase in coolant, cladding and fuel. Temperature increase in coolant, cladding and fuel produces negative reactivity feedback which is used to gradually compensate external positive reactivity. After 200 seconds the power level approaching asymptotic levels and the total reactivity approaching zero. Doppler and Core radial expansion feedback play dominant role in UTOP accident. The temperature increase in fuel and coolant is still far below safety limit.

There are overshoot pattern of power following fast withdrawal of the control rod. Therefore this pattern influence the pattern of hotspot fuel, cladding and coolant temperature change with time and also the feedback. The most important difference between ULOF and UTOP cases are the fact that in the UTOP accident the fuel, cladding and coolant temperatures are continuously increase till approaching the asymptotic new power levels while in the ULOF case there are peak values of fuel coolant and cladding temperature as shown in Figures. 3 and 7. In the UTOP case the power is also increases till reaching asymptotic level while in the ULOF case decreases and then approaching asymptotic new power level as shown in Figs.3 and 6. In case of the UTOP accident the position change of maximum temperature for coolant, cladding and coolant are not significant.

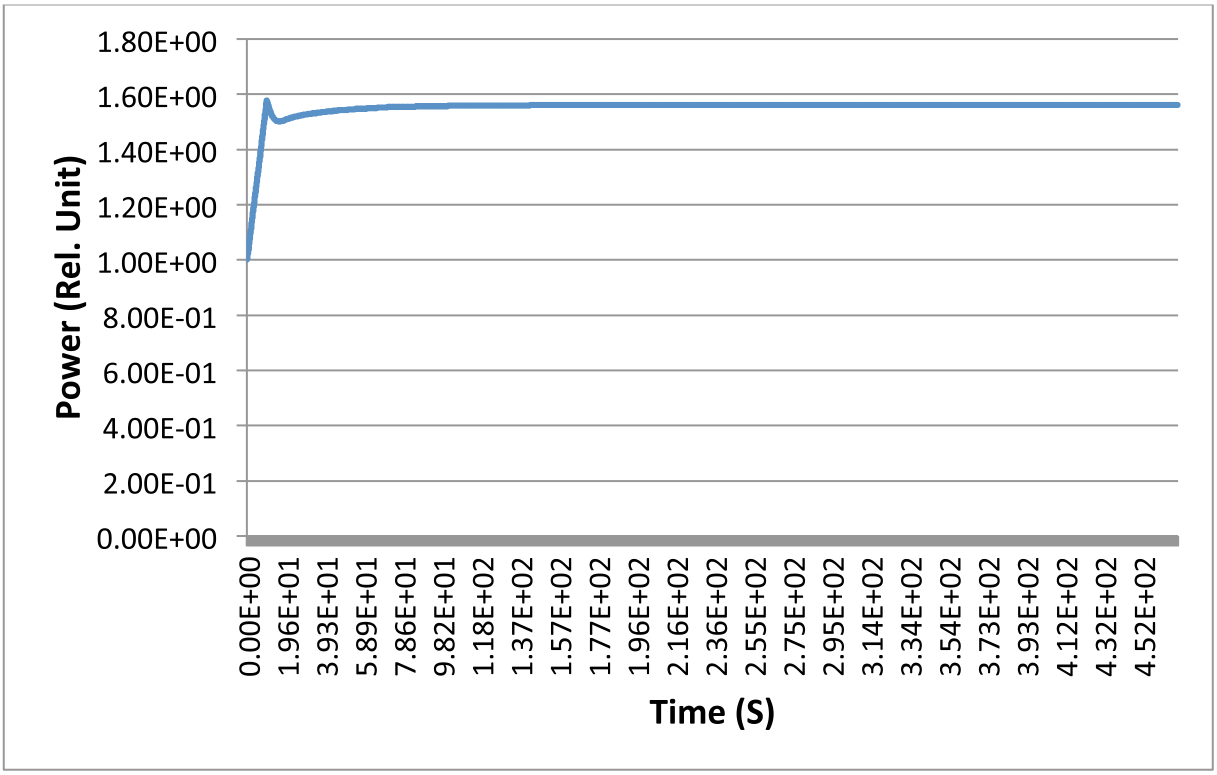

Figure 5 Power change with time during UTOP accident

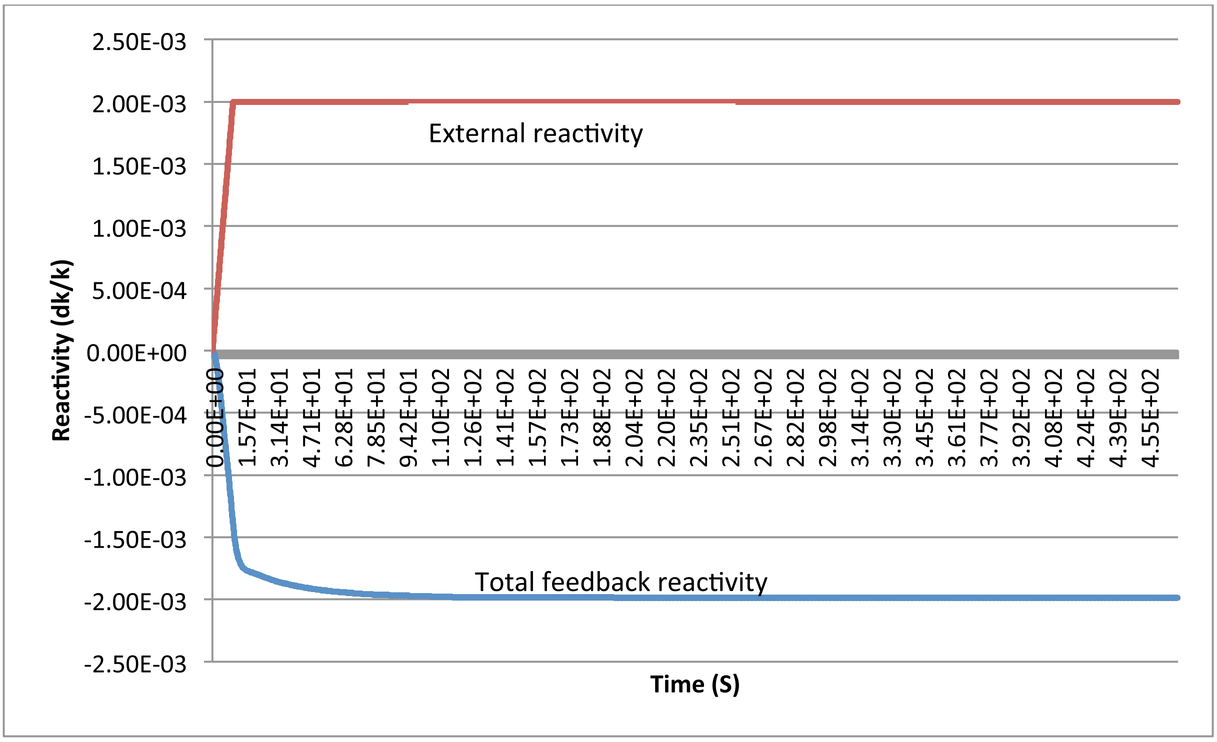

Figure 6 External and total reactivity feedback change with time during UTOP accident

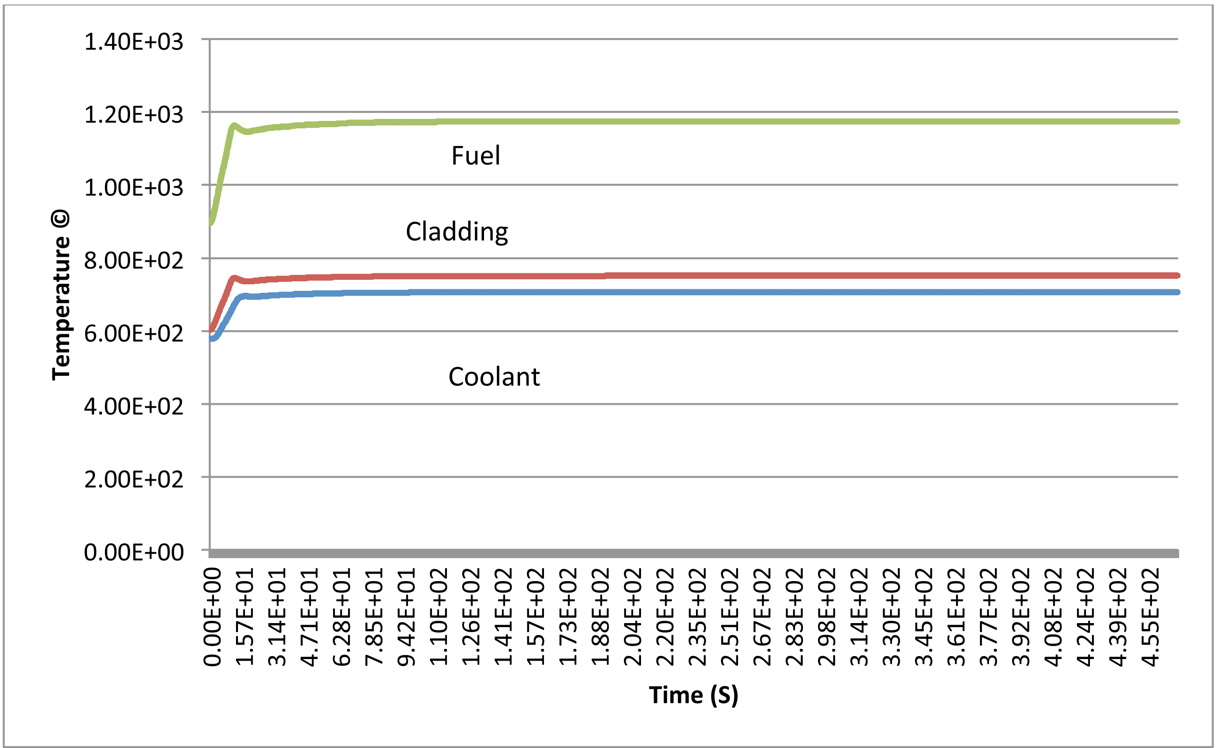

Figure 7 Hot spot temperatures chhange with time during UTOP accident

Figure 8 shows teh change of external and feedback reactivities during UTOP accident simulations. As shown in this figure, the new equilibrium conditions can be reached after total feedback reactivity can completely compensate external reactivity. From figure 8 it is shown that after 400 seconds from the accident begin the asymptotic level have been approached.

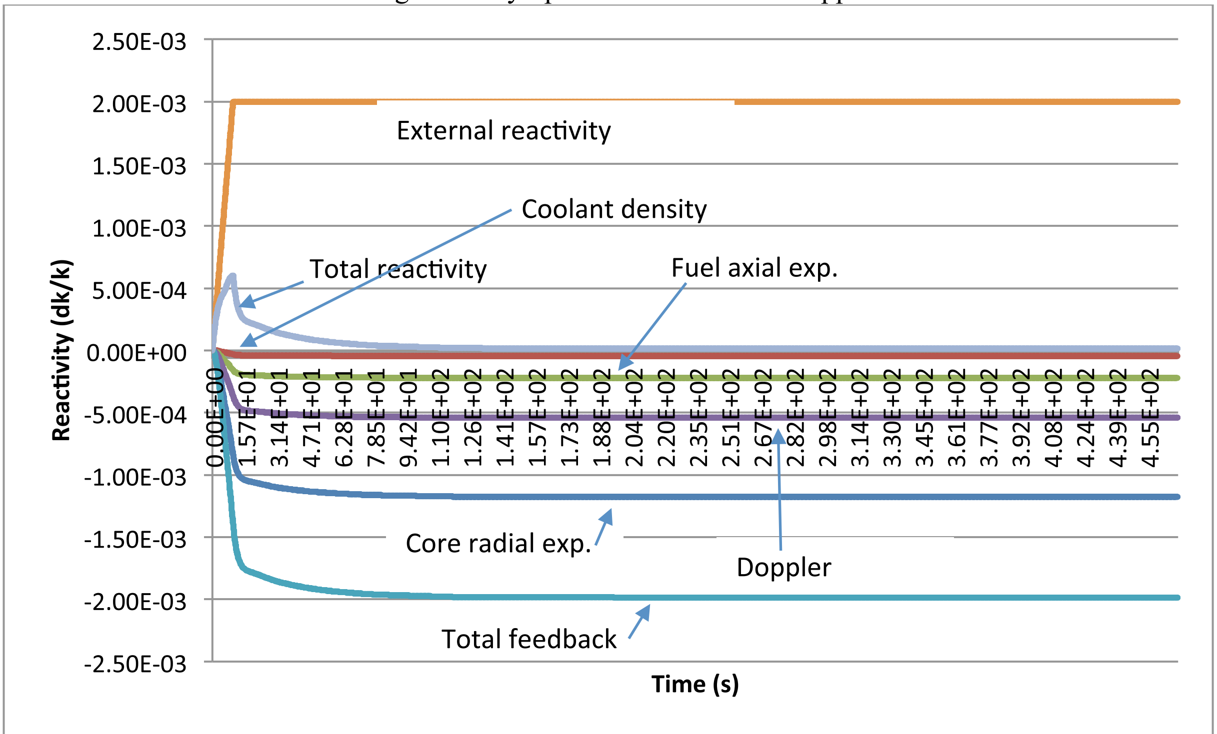

Figure 8 Reactivity change with time during UTOP accident

4. Conclusion

Safety analysis simulations have been performed for Pb-208 cooled 800MWt Modified CANDLE fast reactors especially against unprotected loss of flow (ULOF) and unprotected rod-run out transient overpower (UTOP). Pb-208 Cooled Small Modified CANDLE Reactors can survive ULOF and UTOP inherently. Natural circulation plays important role in the ULOF accident. Core radial expansion and Doppler reactivity feedback plays important role in the ULOF and UTOP accident.

References

[1] Zaki S,, 2008, Progress in Nuclear Energy, Neutronic Performance Comparation of MOX, Nitride and Metallic fuel based 25-100 MWe Pb-Bi Cooled Long Life Fast Reactors without on site Refuelling, Vol. 50, p. 276-278
[2] Zaki S, 2008, Progress in Nuclear Energy, Safety Performance Comparation of MOX, Nitride and Metallic fuel based 25-100 MWe Pb-Bi Cooled Long Life Fast Reactors without on site Refuelling, Vol. 50, p. 157-162
[3] Zaki Su'ud and H Sekimoto, 2010, IJNEST, Design study of long-life Pb-Bi cooled fast reactor with natural uranium as fuel cycle input using modified CANDLE burn-up scheme, Vol 5, No. 4 , p.347-368
[4] Su'ud, Z., and Sekimoto, H., 2012, IJNEST, Design study of medium-sized Pb-Bi cooled fast reactors with natural uranium as fuel cycle input using modified CANDLE burn-up scheme, Vol. 7 (1) , pp. 23-44.
[5] Zaki Su'ud and H Sekimoto, 2013, Annals of Nuclear Energy, The prospect of gas cooled fast reactors for long life reactors with natural uranium as fuel cycle input, Vol. 54 , pp. 58-66.
[6] S.K. Cheng and N.E. Todreas, 1986, Hydrodynamic models and correlationx for bare and wire-wrapped hexagonal rod bundles, Nucl. Eng. Des. 92, p 227.
[7] Zaki Su'ud and H. Sekimoto, 2010, IJNEST, Local Blockage analysis of Lead-Cooled Next Generation Nuclear Power Reactors, Vol. 5, No.2
[8] J.G. Guppy, 1993, Super System Code, NUREG/CR-3169.
[9] S. Kakac, R.K. Shah and W.Aung, 1987, Handbook of Single Phase Convective Heat Transfer, Wiley, pp.7.1-7.62.
[10] K.O. Ott and R.J. Neuhold, 1985, ANS, Nuclear Reactor Dynamics.
[11] H.Sekimoto and S. Zaki, 1995, Nucl. Technol. , Design study of lead and lead-bismuth cooled small long-life nuclear power reactors using metallic and nitride fuel, 109,p. 307-313.
[12] Zaki S. dan Sekimoto,H., 1996, Nucl. Eng. And Design, Accident Analysis of Lead or Lead-Bistmuth Cooled Small Safe Long-Life Fast Reactor using Metallic or Nitride Fuel, Vol. 162, Elsevier Science, pp.205-222.
[13] Zaki S., 1998, Progress of Nuclear Energy, Comparative Study on Safety Performance of Nitride Fueled Lead-Bismuth Cooled Fast Reactor with Various Power Level", Vol.32, No. ¾,, 571~577
[14] N. Nakagawa and K.Tsuchihashi, 1984, JAERI Memo, SLAROM: a code for cell homogenization calculation of fast reactor, (Japan Atomic Energy Research Institute, Ibaraki), JAERI M-1294