Typical Accidents Analysis on the Loss of Flow and External Neutron Source Changing Accidents for A 800 MWth ADS

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ABSTRACT

In this paper, preliminary safety studies on the 800MWth accelerator-driven system (ADS) proposed by Xi'an Jiaotong university are presented. The system is a pool type facility coupling a proton accelerator with current in the range of 17 to 23mA and a sub-critical core by means of a spallation target. Here, the RELAP5/MOD3.3 code is selected as a base tool and to meet the requirement of ADS analysis, the point kinetics model is modified and the property of lead-bismuth is implemented. The paper focuses on the assessment of its response to the loss of flow events, the former is originated from the failure of the pump and the latter derives from the significant flow blockage at a fuel assembly inlet. The reactivity insertion accidents are caused by the change of the proton beam current. The results show that the safety and criteria are satisfied and the system is tolerant to the loss of flow accidents and proton beam doubled accident and is sensitive to the external neutron changing.

Key words: ADS, LOFA, reactivity insertion, RELAP5.

Nomenclature

t	Time(s)
n	Neutron density (neutrons/m^3)
C_i	Delayed neutron precursor concentration in group i (nuclei/m^3)
β	Effective delayed neutron fraction
Λ	Prompt neutron generation time (s)
ρ	Reactivity
f_i	Fraction of delayed neutrons of group i
β_i	Effective delayed neutron precursor yield of group i
λ_i	Decay constant of group i (1/s)
S	Source rate density (neutrons/m^3·s)
S'	Source rate (neutrons/ s)
P_{f}	Immediate (prompt and delayed neutron) fission power (MeV/s)
P_{rf}	Rated immediate fission power (MeV/s)
Q_{f}	Immediate fission energy per fission (MeV/fission).
N_{d}	Number of delayed neutron precursor groups.
ν	Neutrons released per fission (n fission−1)
I_{prot}	External proton beam intensity (mA)
η_{prot}	Number of spallation neutrons released per proton

1. INTRODUCTION

In an accelerator-driven subcritical system (ADS), the chain reaction is kept by an accelerator which provides the beam currents. The power of the system is adjustable by changing the beam currents and it is a more efficient and reliable way than adjusting the control rod (Masood et al., 2003). The fission chain reaction can't be sustained without the external supply of neutrons produced in the spallation target which means that the system can be shut down immediately once the accelerator stops working. The ADS also offers potential solutions towards the spent fuel problem caught public's eye for decades. The largest part of the long-lived nuclear waste may be eliminated by transmutation through the system. Though the problem of the closure of the nuclear fuel cycle is not completely solved, the ADS is a mitigatory measure to handle the nuclear waste (Stanculescu, 2013). Due to the above advantages, a lot of researches have been done during the past decades.

Since the early 1990s, more and more researchers have devoted themselves to the developing of the ADS. Following a preliminary design developed in 1998, which was based on the energy amplifier concept proposed by CERN, the reference configuration of a Lead-Bismuth Eutectic-cooled Experimental ADS (LBE-XADS) was desinged from 1999 to 2001 (Aliotta et al., 2002; Cinotti et al., 2004). The feasibility of a 80 MWth ADS was assessed and the accidents of loss of flow was analyzed (Aliotta et al., 2002; Cinotti et al., 2004). In Korea, The Korea Atomic Energy Research Institute (KAERI) has been performing accelerator driven system related research and development called HYPER (HYbrid Power Extraction Reactor) which is 1000MWth. In the research, the main work was focused on the core design (Park et al., 2000). In China, according to the demand of the research plan of "973" project, the beam transient change accident of ADS is extensively analyzed (Yu, 2005) and researchers from School of Nuclear Science and Technology in Xi'an Jiaotong University have been working on the 800MWth ADS (Zhou et al., 2014) since 2011 and the physical part of the system has already been designed which is supported by National Natural Science Foundation of China.

In this paper, in order to do the accident analyses, the modifications of the RELAP5/MOD3.3 code are performed, which include the LBE thermophysical property and the point kinetics model. For fast reactor, loss of flow accident is design basis accident and here, it contains totally loss of flow in the core and partial obstruction of a single fuel assembly caused by inlet flow blockage. Since the subcritical system is kept by the proton beam driven by the accelerator and the instability of the proton beam would affect the power of the core. It is necessary to analyze the external neutron source intensity changing accidents. In this paper, the accidents analyses, including the loss of flow accident and external neutron source changing accident, are conducted for the 800MW ADS. The result shows that with the effect of feedback, the system would keep safe in both loss of flow accident and external neutron source intensity doubled accident.

2. DESCRIPTION OF THE 800MW ADS

The newly designed 800 MW ADS mainly consists of the accelerator, the spallation target and the reactor core. The external proton beam produced in the accelerator reaches the spallation target central through the beam pipe. The function of the spallation target in the ADS is to convert the high energy particle beam to low energy neutrons. LBE is used as the target material and the coolant to remove the heat produced by the spallation reaction.

A hexagonal type of fuel array is considered for the compact core design. One-sixth of the reference core configuration is shown in Figure 1. The detailed design parameters are presented in Table 1

Table 1 Parameters of the fuel assembly

Parameter	Value
Fuel	(TRU-10Zr)-*Zr
Cladding/assembly wrapper	HT-9
Coolant	LBE
Fuel pin diameter	7.20 mm
Cladding thickness	0.50 mm
Pitch/diameter ratio	1.48
Number of fuel pins per assembly	271
Assembly pitch	177.8 mm
Assembly wrapper thickness	2.0 mm

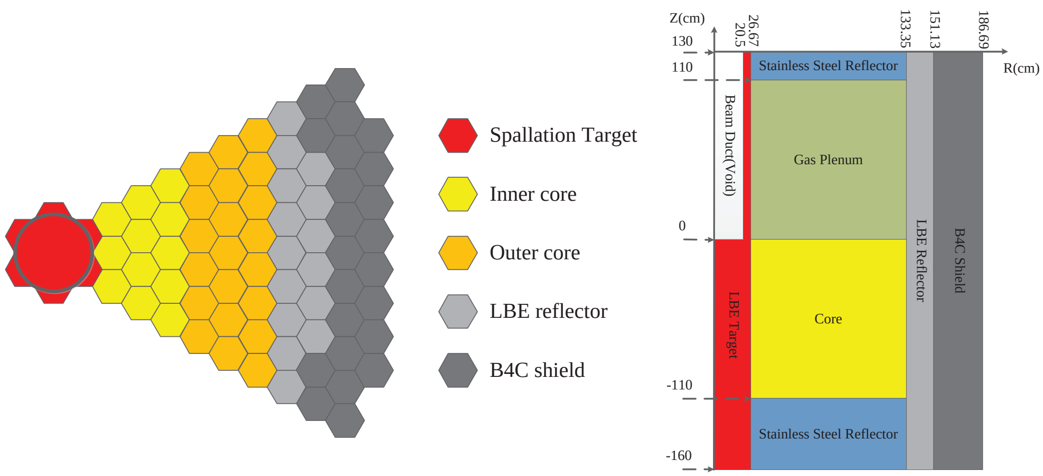


Figure 1 Arrangement of assemblies and R–Z model

The height and the effective diameter of the active core are 1.1 m and 3.73 m, respectively. The reflector assemblies cooled by liquid Pb-Bi are located at the core perimeter. The shielding assemblies composed of B_4C are located at the outermost core perimeter to prevent excessive irradiation damage to the reactor structures and components surrounding the core (Zhou et al., 2014).

A multiplication factor of 0.97013 (at core rated power, BOL) is considered to be sufficiently low to ensure the safe operation of the system without need of shutoff rods (Zhou et al., 2014). Liquid Pb-Bi eutectic alloy, used as the spallation target material, is adopted as the coolant due to its excellent nuclear and heat transfer characteristics as well as the good compatibility with other materials. Meanwhile, LBE is filled in the gap between the fuel and the cladding to reduce the fuel temperature due to its high heat conductivity and high boiling temperature.

The primary side is a pool-type configuration, which refers to the design of the pool-type sodium cooled fast reactor system. The spallation target, the reactor core, the main pump and the core structures as well as the intermediate heat exchangers (IHX) are immersed in the LBE pool of the main vessel. The cross-section figure of ADS primary side is shown in Figure 2 (Cinotti and Gherardi, 2002). The thermal inertia of the pool-type system is higher than that of the loop-type system, which may prevents the pool temperature from increasing to a high level during the loss of flow accident. Meanwhile, the pool-type design is beneficial to the biological shielding. Due to the large thermal power of the ADS, the forced circulation by the electromagnetic pumps other than the natural circulation is adopted to circulate the LBE coolant in the primary side.

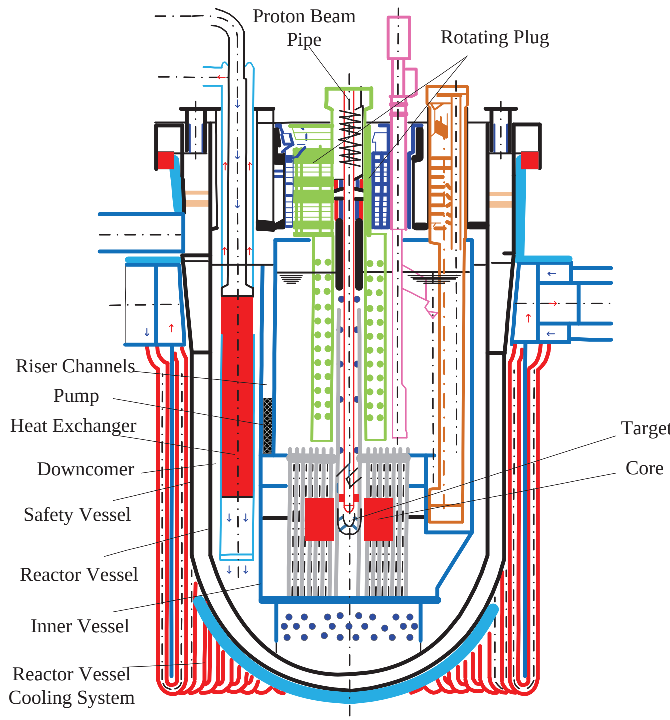


Figure 2 The cross-section figure of ADS primary side

The two loops secondary system is thermally coupled to the primary loop through two IHX's so to transfer the operational core heat to the outer environment. In this paper, the main work is focused on the primary side and the secondary side could be simplified when assuming the heat transfer is sufficient. And in the transient assessments, the secondary side is also simplified, for example, in the protect loss of flow accident, after shutting down the core, the temperature in the secondary side would slowly reduce. However, in other accidents, the change of the secondary side is ignored, since the power almost remains the same (in the unprotected loss of flow accident assessment) or the time of the transient is too short (in the analysis of beam trip accident of ADS). In this paper, the secondary side is filled with water, and the inlet temperature of the IHX is 500K.

3. COMPUTER CODE, SYSTEM MODELS & DATA

3.1 Modification of The RELAP5/MOD3.3 Code

The RELAP5/MOD3.3 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents (Siefken et al., 2001). In addition, it can be used for simulation of a wide variety of hydraulic and thermal transients in nonnuclear systems involving mixtures of steam, water, noncondensable, and solute. When the RELAP5/MOD3.3 code is used to perform the transient analysis of ADS system, the modification of the code is necessary. In this paper, the point kinetics model and the thermal property package are modified to simulate the transients of the 800MW ADS.

The point kinetics equations are:

$$\frac{\mathrm{d}n(t)}{\mathrm{d}t}=\frac{\left[\rho(t)-\beta\right]}{\Lambda}n(t)+\sum_{i=1}^{\mathrm{N_d}}\lambda_i C_i(t)+S \tag{1}$$

$$\frac{\mathrm{d}C_i(t)}{\mathrm{d}t}=\frac{\beta f_i}{\Lambda}n(t)-\lambda_i C_i(t) \qquad i=1,2,...,\mathrm{N_d} \tag{2}$$

When the point kinetics model of the original RELAP5/MOD3.3 code is utilized to simulate a subcritical system, the neutron source rate density S in normal condition is calculated by the source rate S', which is obtained by

$$S'=\frac{P_{\mathrm{rf}}\nu}{Q_{\mathrm{f}}}\left(\frac{1-k_{eff}}{k_{eff}}\right) \tag{3}$$

Form equation(3), the neutron source rate density will be kept constant once it is determined and the other parameters such as power and the reactivity are then calculated by using Runge-Kutta method. In this way, the code cannot be utilized to simulate the external neutron source intensity changing accident. The point kinetics model should be modified to simulate the accidents caused by changing the accelerator intensity. Therefore, equation (3) is modified as follows:

$$S'=6.24\times10^{15}\cdot I_{prot}\cdot\eta_{prot} \tag{4}$$

Where the proton beam intensity is calculated by:

$$I_{prot}=\frac{P_{\mathrm{f}}\nu}{0.20\eta_{prot}}\left(\frac{1-k_{eff}}{k_{eff}}\right) \tag{5}$$

For Pb-target and1.2GeV proton beam, η_{prot} is about 20 n per proton. The constant 0.20 in equation(5) is obtained through multiplying $6.24\times10^{15}\,\mathrm{protons}\cdot\mathrm{s}^{-1}\cdot mA^{-1}$ by $3.2\times10^{-17}\,MW\cdot s\cdot fission^{-1}$ (Schikorr, 2001).

In the ADS, the delayed neutron fraction is:

$\beta\approx0.0021813=0.21813\%=218.13pcm$, and here $k_{eff}=0.97$ is defaults

According to equation(1) to equation(5), the relationship between the external source density and proton beam is clear, the external neutron source density is not a constant and would change with the external proton beam intensity I_{prot}. Therefore, the accidents of changing the accelerator operation condition can be simulated.

In the past, the best-estimate thermal hydraulic analysis codes originally developed for water cooled reactors were modified to be used for the liquid metal cooled reactors by embedding the thermophysical properties of liquid metals, such as sodium, lead, bismuth and LBE into them. In 2010, the thermo-physical properties and thermal-hydraulic characteristics of lead-bismuth eutectic were added to the CATHARE code. In the RELAP5-3D code, the thermophysical properties of lead-bismuth, sodium, lithium are include into the property package. In this paper, in order to simulate the ADS system, the thermophysical properties of LBE are established based on the database of thermophysical properties of liquid metal coolants, such as sodium, lead, bismuth and LBE for GEN-IV reactors developed by Belgian Nuclear Research Centre (BNRC) (Sobolev, 2011).

Referring to the format of the thermophysical property table of water in RELAP5/MOD3.3, the LBE property package was established. The property package contains five tables: table 1 is the temperature matrix, table 2 is pressure matrix, table 3 is the saturation properties table as a function of temperature, table 4 is the saturation properties as a function of pressure and table 5 is the single phase properties. Specific volume, internal energy, enthalpy, coefficient of isobaric thermal expansion, coefficient of isothermal compressibility, specific heat at constant pressure and entropy are contained in the binary table besides temperature and pressure. After the tables were generated, the RELAP5/MOD3.3 code was modified to read and call the property package.

3.2 The System Models &Data

The main design parameters of the 800MW ADS at nominal operating condition are given in Table 2. The effect of the fuel expanding should be taken into consideration. In this system, the feedback of axial expanding is considered by fuel temperature feedback factor, -0.383629 pcm/K. While for radial expanding, the factor is considered by coolant density feedback factor, -0.653724 pcm/K based on the core design analysis.

Table 2 Main design parameters

Fission power, MW	800
Primary coolant flow rate, kg/s	24130
Core flow rate, kg/sec	21960
Core inlet temperature, °C	300
Core outlet temperature (at rated power), °C	530
β	218.13pcm
k_{eff}	0.97
Doppler feedback factor, pcm/k	-0.042806
Coolant density feedback factor, pcm/1%	-41.741871
Secondary coolant inlet temperature, °C	227
axial expanding feedback factor, pcm/k	-0.383629
radial expanding feedback factor, pcm/k	-0.653724

The nodalization of the ADS system is shown in Figure 3, essentially simulates the region inside the main vessel. Most of its volume constitutes the primary coolant circulation flow path with only minor fractions of it being filled with quasi stagnant lead-bismuth. For simplification, the secondary side of the IHX (181/182 in Figure 3) is modeled with proper inlet and outlet boundary conditions.

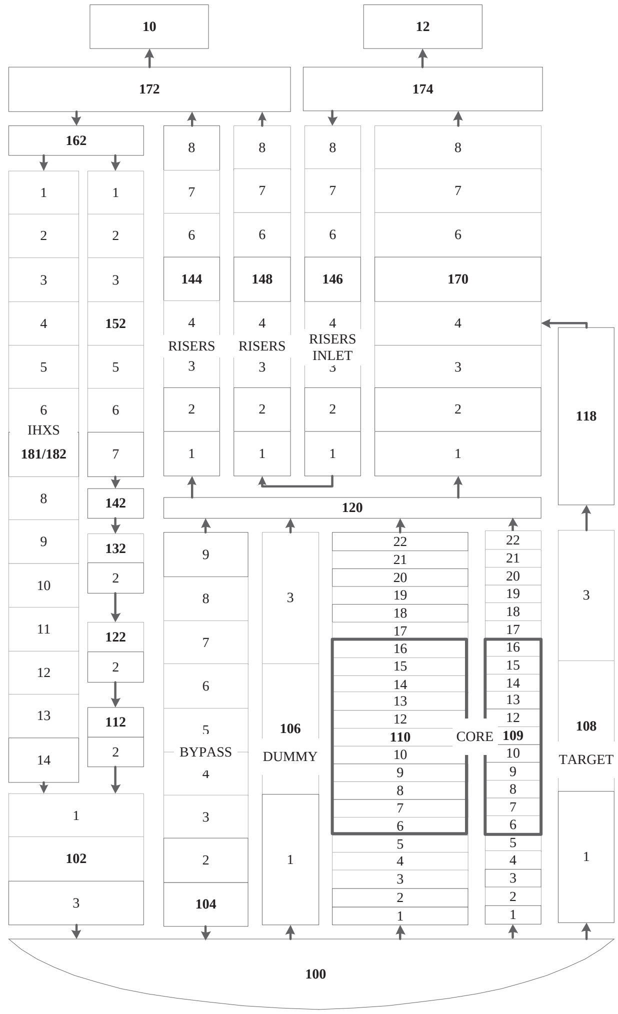


Figure 3 The model of ADS in RELAP5

The main circulation flow path includes: CV 110 (Core region) – CV 120 (Upper plenum) –CV 144 (Main Risers) –CV 172, 162 (Downcomer) –CV 181/182 (IHX) –CV 102 (Downcomer) –CV 100 (Lower plenum).

The core of the system is consisted of CV 110, CV 109, CV 104 and CV 106. CV 110 is the average fuel channel and CV 109 is hot fuel channel, the power of which are 793.3MW and 6.68MW, respectively. The inlet flow area reduction takes place in the hot assembly when the blockage accident is analyzed. As for the remaining two pipes, CV 106 simulates the bypass coolant flowing through the interspace between the affected assembly and the adjacent ones, whereas CV 104 simulates the flow pertaining to all the remaining fuel assemblies interspaces. Then, CV 108 and CV 118 represent the target part of the core in which the coolant flows and takes away the heat generated by the spallation target and the power of CV 108 is 30MW. The power peaking factor is shown in Figure 4.

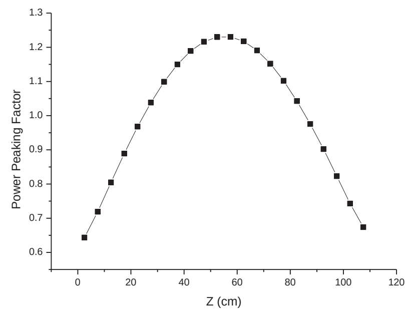


Figure 4 Core power peaking factor

4. ANALYSES OF THE ACCIDENTS

4.1 Loss of Flow Accident

4.1.1 Total loss of flow accident

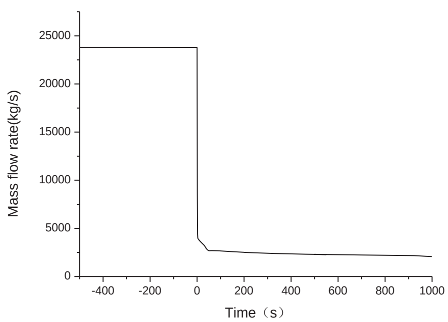


Figure 5 Core flow rate in loss of flow accidents

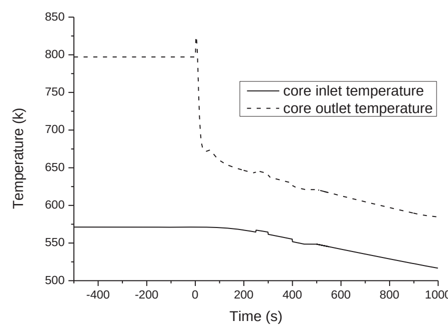


Figure 6 Core inlet & outlet temperature in loss of flow accidents

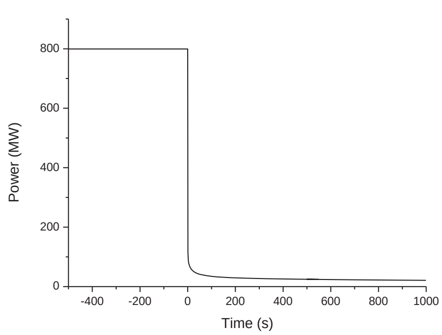


Figure 7 Core power in loss of flow accidents

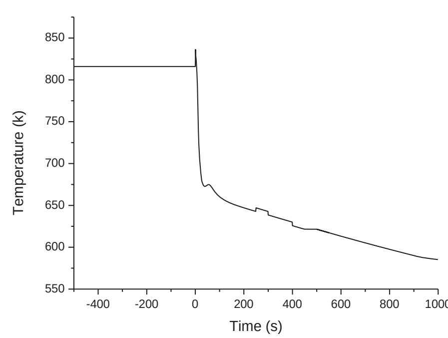
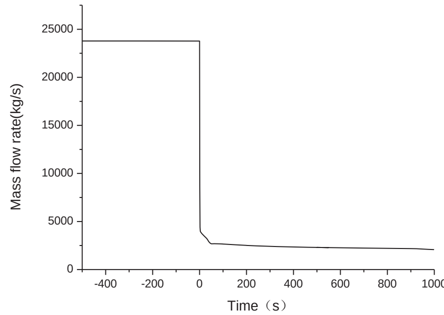


Figure 8 Cladding inner temperature in loss of flow accidents

In the loss of flow accident, the pump stops working and is totally seized. The core coolant mass flow rate declines from the nominal value of 23775 kg/s down to a value of 3900 kg/s in



5 seconds (16.4% of nominal, Figure 5) due to the transition from forced to natural circulation. Hence, the primary coolant temperature at the core outlet suddenly increases (Figure 6) and reaches the set point of 823K (550°C) at 4 s which will trip the proton beam. Then the core power decreases to the decay heat level as shown in Figure 7. As a consequence, the core outlet temperature quickly decreases. Later on, the mismatch between the power removed by the secondary system and the core power causes the decrease of the core inlet temperature and the temperature of the system decreases slowly. The highest cladding temperature is 836K which is smaller than the criterion of 1055K (Figure 8).

4.1.2 Flow blockage in a single fuel assembly

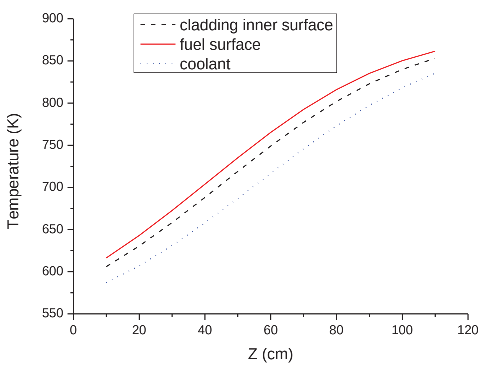


Figure 9 Hot assembly axial temperature for A/A0 = 1.0 in flow blockage assessment

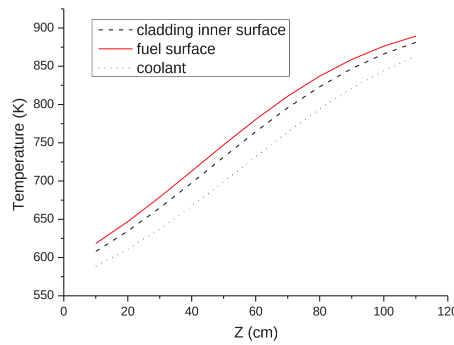
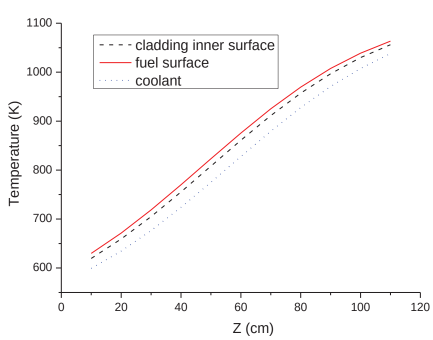


Figure 10 Hot assembly axial temperature for A/A0 = 0.35 in flow blockage assessment



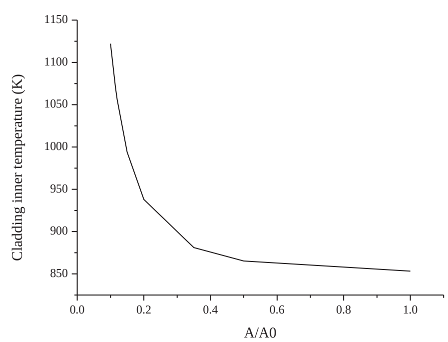


Figure 11 Hot assembly axial temperature for A/A0 = 0.12 in flow blockage assessment

Figure 12 The highest inner cladding temperature in hot assembly for different A/A0

Because of the oxidizing reaction of LBE, agglomeration of particles from various part of the system can lead to partial flow obstruction or blockage in a region of the reactor core. In this paper, the accident caused by partial flow obstruction which takes place in the hot assembly is analyzed.

When analyzing the accident of flow blockage in a single assembly, it is assumed that the flow area reduction takes place in the hottest assembly and the mixing between the hottest channel and the average channel is not considered for conservatism. And in the simulation, all other assemblies are well working and the core power is kept to be the rated value during the accident which means that the total core flow rate and the outlet core temperature almost keep the same and the power of the core does not change.

Six cases with different flow area blockage ratio, A/A0 (A0 is the initial flow area and A is that after blockage taking place) are simulated and the results are given in Figure 9 to Figure 11. Figure 12 shows that the highest inner cladding temperature increases with the increased blockage ratio. When the blockage ratio is larger than 0.35, the increase of the inner cladding temperature with the blockage ratio is rather small. However when the blockage ratio is less than 0.35, the inner cladding temperature increases rapidly as the blockage ratio increases. Meanwhile, when the blockage ratio is 0.12, the highest inner cladding temperature will exceeds the limiting criterion of 1055K. Therefore, appropriate control measure or scram signal should be designed to avoid the cladding temperature exceeding the criterion.

4.2 External neutron source failure accidents

4.2.1 External neutron source intensity doubled accident

The external neutron source intensity doubled accident is caused by doubling the power of the accelerator which is specific to ADS. The power reduction of the ADS due to the fuel burnup is usually compensated by adjusting the power of the accelerator. In addition, during a refueling cycle, the power of the accelerator could be doubled to compensate for the decrease of the reactivity.

For ADS system, the reactivity insertion accidents may be caused by unanticipated change of the external neutron source intensity or control rod withdrawl. In this paper, the accident of doubling the external neutron intensity without reactor scram is simulated. This accident may result in prompt critical of the ADS system, which would lead to the core out of control. And consequently, the core power becomes so high that the pressure boundary of primary loop may be damaged.

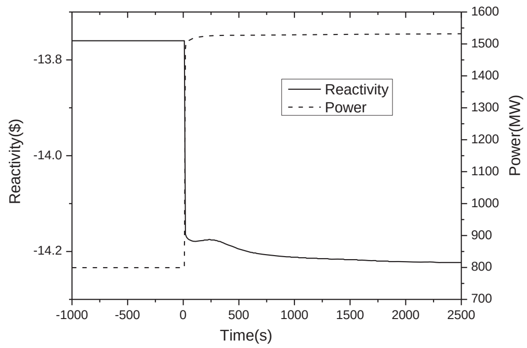


Figure 13 The Power & Reactivity in external neutron doubled assessment

Figure 13 shows the power and reactivity variation during the accident. The total power increase to be a new stable level of 1534MW which is about 190% of the initial value after the accident occurs. The reactivity reduces by 0.47 to be about -14.25 due to the reactivity feedback. The results indicated that the core power is not doubled due to the negative reactivity feedback.

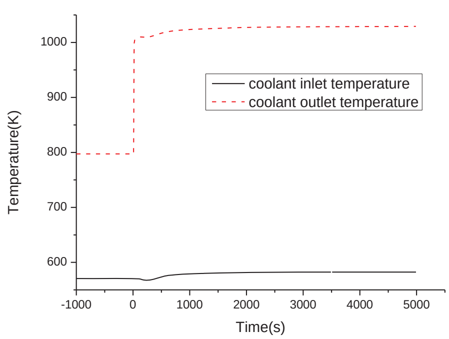


Figure 14 The coolant temperature in external neutron doubled assessment

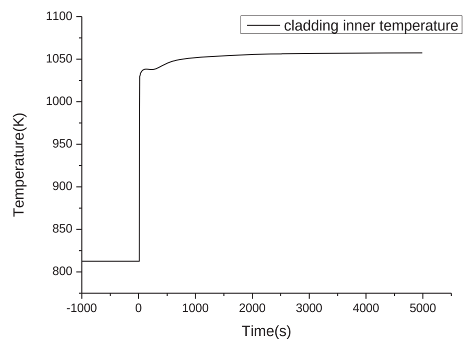


Figure 15 The highest cladding inner temperature in external neutron doubled assessment

As is shown in Figure 14, the core inlet temperature increases slightly to be a new steady-state level after the accident due to the inlet condition of the secondary side of the IHX is kept unchanged. The core outlet temperature of the normal assembly rises from 800K (527℃) to 1030K (757℃), which is still much lower than the saturation temperature and the heat transfer deterioration does not take place. The cladding inner temperature rises from 812K to 1057K as shown in Figure 15, 2 degree higher than the temperature limitation during normal working condition (1055K), but it is still much lower than the melting point of 1703K for HT-9. From the above analyses, it can be concluded that the system is safe in short time if the external neutron source intensity is doubled, but in the long run, high temperature and high irradiation environment would cause cladding embrittlement which would be harmful to the safety of the system.

4.2.2 Beam trip accident of ADS

Since the instability of the proton beam which is related with ion source and the accident of radio frequency system, loss of proton beam accident should be analyzed. This accident is simulated by cutting off the proton beam, which means cutting off the external neutron, under the rated power condition and then restarting it after ten seconds. The pump is normally working and the inlet condition of the secondary side is kept constant.

As shown in Figure 16 and Figure 17, after cutting the external neutron source, the power decreases to be 65MW immediately. And then, when the accelerator is restarted again after 10 seconds' stop, the power rises sharply to be 772MW and restores slowly to be the initial power level of 800MW. Simultaneously, the reactivity rises to be -12.5 after the accelerator stops, and decreases to be -14.16 when the accelerator is restarted again.

Figure **18** shows that the average inlet and outlet coolant temperature of the normal assembly. The inlet coolant temperature does not change so much after the accident while the outlet temperature falls from 800K to 590K.

This results show that since the system is subcritical, the safety and criteria are satisfied in this accident. In this accident, the temperature of the core changes sharply which would cause thermal fatigue damage of the structure and shorten the lifespan of the system.

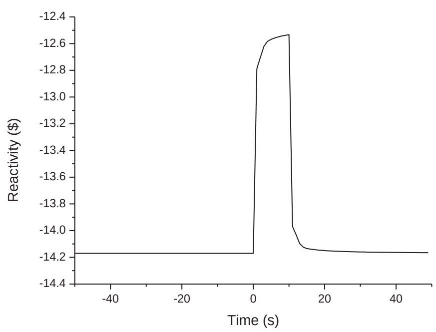


Figure 16 The reactivity in beam trip accident of ADS

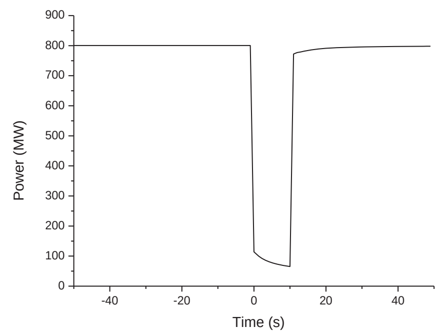


Figure 17 The power in beam trip accident of ADS

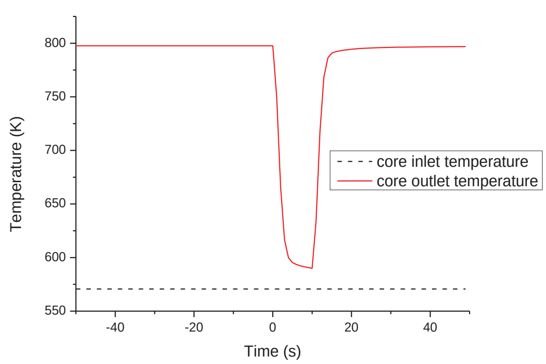


Figure 18 Core inlet/outlet coolant temperature in beam trip accident of ADS

5. CONCLUSION

In this research, through modifying the point kinetics and adding the properties of LBE, the RELAP5/MOD3.3 code is improved for the accident analyses of the ADS system. With the modified code, the loss of flow and external neutron intensity changing accidents of a 800MWth ADS were conducted. The conclusions are as follows:

1. Following a loss of primary flow since the pump stops working and is totally seized, the primary coolant mass flow rate decrease promptly causes high temperature difference across the core which actuates the plant protection system (proton beam trip) as the high core outlet coolant temperature set point is attained. And in this accident, the highest cladding temperature is 836K which is smaller than the criterion of 1055K.
2. In case of flow blockage at the inlet of the hot assembly, the simulations results indicate that the LBE-cooled ADS is very tolerant to flow blockage (no significant temperature rise up to a 65% blockage ratio). For higher flow blockage, the temperature is below the cladding temperature limitation for the normal working condition (1055K) when the blockage ratio is less than 88%. For higher flow blockage, appropriate control measure or scram signal should be designed to avoid the cladding temperature exceeding the criterion.
3. The accidents of changing the external neutron are analyzed and the result shows that in the case of external neutron source intensity doubled accident, with the effect of feedback, the system can keep safe in short time, but in the long run, high temperature and high irradiation environment are still harmful to the system. At the same time, the analysis of beam trip accident of ADS shows that the power changes rapidly as the external neutron, sent from the accelerator, increasing or decreasing. And the core would carry on working and recovering to the normal power once the external neutron returns to the normal value after lost proton beam in short time. And in this accident, the safety and criteria are satisfied and sharply changing temperature would cause thermal fatigue damage of the structure.

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