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Control-rod, Pressure and Flow-Induced Accident and Transient Analyses of a Direct-Cycle, Supercritical-Pressure, Light-Water-Cooled Fast Breeder Reactor

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ABSTRACT

The features of the direct-cycle, supercritical-pressure, light-water-cooled fast breeder reactor (SCFBR) are high thermal efficiency and simple reactor system. The safety principle is basically the same as that of an LWR since it is a water-cooled reactor. "Maintaining the core flow" is the basic safety requirement of the reactor, since its coolant system is the one through type. The transient behaviors at control rod, pressure and flow-induced abnormalities are analyzed and presented in this paper. The results of flow-induced transients of SCFBR were reported at ICONE-3, though pressure change was neglected. The computer code has been improved to handle the pressure change inducing the pressure regulation system using the turbine control valve. The change of fuel temperature distribution is also considered for the analysis of the rapid reactivity-induced transients such as control rod withdrawal. Total loss of flow and pump seizure are analyzed as the accidents. Loss of load (with and without opening turbine by-pass valves), control-rod withdrawal from the normal operation, loss of feedwater heating, inadvertent start of an auxiliary feedwater pump, partial loss of coolant flow and loss of external power are analyzed as the transients. The behavior of the flow-induced transients is not so much different from the analyses assuming constant pressure. Fly wheels should be equipped with the feedwater pumps to prolong the coast-down time more than 10s and to cope with the total loss of flow accident. The coolant density coefficient of the SCFBR is less than one tenth of a BWR in which the recirculation flow is used for the power control. The over pressurization transients at the loss of load is not so severe as that of a BWR. The power reaches 120%. The minimum deterioration heat flux ratio (MDHFR) and the maximum pressure are sufficiently lower than the criteria; MDHFR above 1.0 and pressure ratio below 1.10 of 27.5MPa, maximum pressure for operation. Among the reactivity abnormalities, the control rod withdrawal transient from the normal operation is analyzed. The maximum fuel enthalpy is 104.9cal/g and the smallest MDHFR is 1.66 which are sufficiently lower than the criteria; fuel enthalpy below 170cal/g and the MDHFR above 1.0. In conclusion, all the transients and accidents satisfy the safety criteria.

INTRODUCTION

The advantages of the supercritical-water-cooled reactors are high thermal efficiency, simple reactor system and breeding capability[1,2]. The core can be designed as thermal reactor[3] (SCLWR), fast breeder[4] (SCFBR) and fast converter[2] (SCFR). The plant system is identical among them. The power output can be maximized in the fast converter. The supercritical water does not exhibit a change of phase. The recirculation system, steam separator, and dryer of a boiling water reactor (BWR) are unnecessary. Roughly speaking, the reactor pressure vessel and control rods are similar to those of a pressurized water reactor (PWR), the containment is similar to a BWR, and the balance of plant is similar to a supercritical-pressure fossil-fired power plant (FPP). The number of coolant lines is only two because of the high coolant enthalpy. Containment volume is much reduced. The thermal efficiency is improved by 24% over a BWR[2]. The coolant void reactivity is negative by placing thin zirconium-hydride layers between seeds and blanket[5,6]. The power costs will be much reduced compared with those of a LWR and a liquid-metal fast breeder reactor (LMFBR). The fundamental requirement for safety should be different from the current LWR's.

This paper deals with the safety design and analyses of control rod, pressure and flow-induced accidents and transients of the direct-cycle supercritical-water-cooled fast breeder reactor (SCFBR). The SCFBR core characteristics are summarized in Table 1. Preliminary analyses of the flow-induced events of the SCFBR were reported by assuming constant pressure and steady state fuel temperature distribution[7]. These assumptions are not always conservative. Therefore the computer code has been improved to handle the pressure change inducing the pressure regulation system using the turbine control valve and the change of fuel temperature distribution.

Table1 Characteristics of SCFBR

core	
Core diameter/height(m)	2.52/3.50
Coolant inlet/outlet temperature(℃)	310/431
Coolant inlet/outlet density(g/cm^3)	0.725/0.1215
Coolant inlet/outlet velocity(m/s)	Maximum 3.9/23.3
Fuel/rod diameter/pitch(cm)	Mox/0.88/1.01
Cladding/thickness(cm)	SS/0.052
Plutonium fissile enrichment, inner/outer seed(%)	14.98/16.52
Average discharge burnup(GWd/t)	77.7
Coolant density coefficient BOIEC/EOIEC($\Delta k/k \cdot (g/cm^3)^{-1}$)	0.0526/0.032
Doppler coefficient BOIEC/EOIEC($\Delta k/k/℃$)	$-2.5\times10^{-5}/-2.0\times10^{-5}$
Maximum linear power(W/cm)	400
Power density maximum/average (W/cm^3)	453/172
Fuel centerline temperature(℃)	Maximum 1995
system	
Thermal/Electrical power(MW)	3000/1245
Thermal efficiency	0.415
Pressure(MPa)	25.0
Main steam line number	2
Feedwater flow rate(kg/s)	2048

BOIEC:Begining of initial equilibrium cycle
EOIEC:End of initial equilibrium cycle

PLANT SYSTEM AND SAFETY DESIGN

Concept of SCFBR safety system

The plant system of SCFBR is depicted in Fig 1. The main coolant system consists of two lines, each of which has 50% capacity of the steady state mass flow rate. The main feedwater pumps are driven by turbines and supply coolant to the core at 25.0MPa. The whole outlet coolant flows to the turbine. The operating pressure is kept by means of turbine control valve.

The safety system should be designed to maintain the fuel cladding integrity under any condition of anticipated transients. This is accomplished by removing the heat generated in the core before the cladding temperature rises excessively. Since the SCFBR has no recirculation line nor primary loop, the coolant flow in the core should be directly maintained. If the following two requirements are satisfied under any condition, the excessive increase of the cladding surface temperature is avoided:

(1) To keep the feedwater flow from the coldleg
(2) To keep the coolant outlet open at the hotleg

Abnormal events are classified into transients when their frequencies are relatively high, or into accidents when their frequencies are low. Auxiliary systems are designed to keep the safety when abnormal transients occur. As far as the plant safety is kept by the auxiliary systems, the plant can quickly return to the normal operation. When accidents occur, engineered safety systems are actuated.

To keep the feedwater, the flow rate is measured and the following systems are actuated according to its abnormality level:

Flow level 1 : reactor scram,
Flow level 2 : actuation of auxiliary feedwater system (AFS).
Flow level 3 : actuation of automatic depressurization system (ADS) and the low pressure coolant injection system (LPCI).

To keep the coolant outlet open, the core pressure is measured and the following valves are operated according to the abnormality level:

Pressure level 1 : turbine control valve,
Pressure level 2 : turbine bypass valves,
Pressure level 3 : safety relief valves (SRV) or automatic depressurization system (ADS).

Turbine control valve are used when the reactor pressure stays between 24.0MPa and 26.5MPa. Turbine bypass valves are open if the turbine control valve are closed; for instance, due to the loss of load. SRVs are open if the reactor pressure rises above 26.5MPa, while ADS is operated if the pressure decreases below 24.0MPa.

One of the advantages of the present safety concept is that the detection of abnormality is straightforward with the basic requirements. The inlet feedwater and the opening of outlet are directly recognized by the operators with the measurements of the flow rate and the pressure, respectively. Besides, the flow path in the core is unique, since the feedwater is always at the coldleg side and the outlet is always at the hotleg side. This simple relation is good to reduce risks of misunderstanding and misoperation in the control-room. This straightforward and simple safety system will enhance both mechanical and human reliability, though it always needs active components and relatively quick actuation. The probabilistic safety assessment is remained for the future study.

If AFS fails to keep enough feedwater or the incident is beyond the AFS capability, the ADS+LPCI system is actuated. When the reactor pressure is not kept above 24.0MPa, ADS+LPCI should be used as well to avoid the operation near the critical pressure (22.1MPa) where the heat transfer is markedly deteriorated. When the relief valves are open, the coolant stored in the lower plenum, downcomer and coldlegs evaporates and flows through the core. This ensures cooling during the depressurization period. The core flow is maintained by LPCI after the reactor pressure is reduced to around atmospheric pressure. Once the reactor pressure falls down, it needs a long time to return to the normal operation. Thus the probability of the actuation of ADS+LPCI should be small enough to enhance the load factor.

Design of SCFBR safety system

AFS consists of four lines; two of them are driven by turbines and the others are driven by electric motors. Each line possesses 10% capacity of the steady-state mass flow rate. This system is actuated when the flow rate is lower than the level 2. LPCI is in operation when the flow rate falls below the level 3. The capacity of LPCI was determined by the analysis of large break loss-of-coolant accident (LOCA)[8]. LPCI has four lines, each of which has capacity 805 kg/s. In the LOCA analysis, two of four lines are assumed to lose the function; one is connected to the break line and the other fails to start up. When LPCI is used, ADS is actuated simultaneously. Actually, LPCI is not considered in any case in the present study, since all the events analyzed are managed to satisfy the safety criteria without ADS+LPCI.

In the present analyses, the following conditions of actuation are employed:

-Actuation of the reactor scram:
(1) mass flow rate at cold leg below 90% of the steady state,
(2) reactor power over 120% of the steady state,
(3) loss of external power,
(4) rapid closure of the turbine control valve,
(5) reactor period below 10sec.

The scram is assumed to be completed in 3.7sec including the delay time of signal processing and motion of the control rods. The scram reactivity is \$12.7.

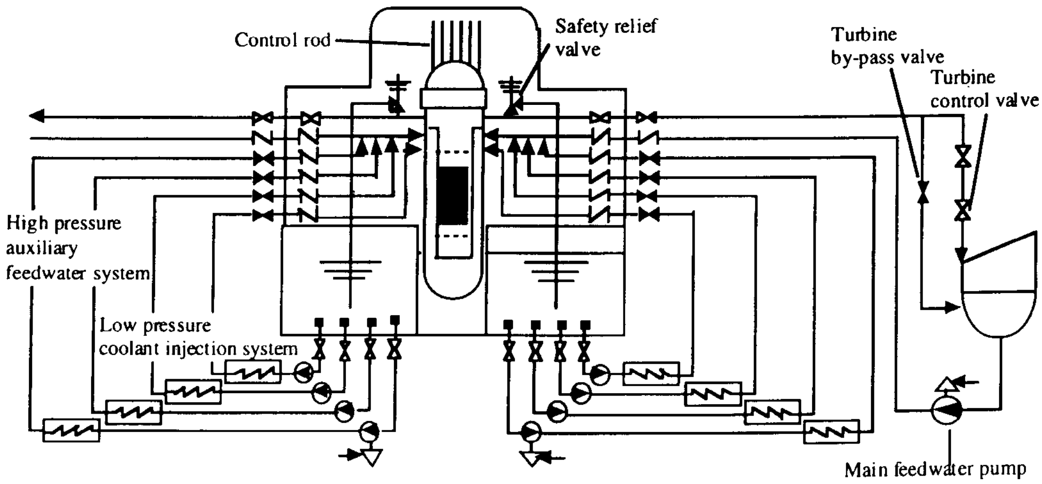

Fig.1 Plant system of SCFBR

-Actuation of the auxiliary feedwater system (AFS):
(1) mass flow rate at cold leg below 20% of the steady state,
(2) trip of the main feedwater pumps.

It is assumed that two of four AFS lines are actuated after 5.0sec because of the delay time of signal processing and inertia of the coolant.

-Actuation of the turbine bypass valves:
(1) rapid closure of the turbine control valve.

Turbine bypass valves are opened after 0.1sec because of the delay time of signal processing. Capacity and control of the turbine bypass valves are the same with those of the turbine control valve.

-Actuation of the safety relief valves (SRV):
(1) setting of pressure values are shown in Table 2.

Table2 Set point and number of safety relief valves

open(MPa)	close(MPa)	number
26.5	25.5	2
26.7	25.7	3
26.9	25.9	10
27.1	26.1	10

In the present study, the following pressure and flow-induced accidents and transients at the beginning of initial equilibrium cycle (BOIEC) and the end of initial equilibrium cycle (EOIEC) are analyzed. Since at the BOIEC the fuel temperature coefficient is smaller and the coolant density coefficient is larger, more severe results are expected at the BOIEC.

The analyzed accidents are as follows:
(1) Total loss of reactor coolant flow;
(2) Reactor coolant pump shaft seizure;
The analyzed transients are:
(3) loss of feedwater heating;
(4) inadvertent start-up of auxiliary feedwater system;
(5) partial loss of reactor coolant flow;
(6) loss of external power;
(7) loss of load (turbine bypass valves are opened);
(8) loss of load (turbine bypass valves can not be opened);
(9) control rod withdrawal (from normal operation);

CALCULATION MODEL

The calculation code is developed based on the following assumptions.

(1) The reactor pressure is kept at 25.0MPa by the turbine control valve within the pressure perturbation between 24.0MPa and 26.5MPa.
(2) The heat transfer coefficient from cladding surface to the coolant is calculated by Dittus-Boelter correlation.
(3) The reactor power is calculated by point kinetics equation with six delayed neutron groups, while the decay heat is calculated using two group approximation of ANS+20% formula.
(4) The axial reactor power distribution is assumed to follow the cosine distribution.
(5) Doppler and coolant density feedback are considered.
(6) The reactor system is divided into three parts: core, upper and lower plenums, as shown in Fig.2.
(7) The hottest single channel is analyzed.

The core is expressed by single channel model and divided into five nodes axially. The lower plenum which includes down comer is modeled by a single cell. The upper plenum which includes main steam lines is divided into five nodes. It is impossible to calculate the density change in the upper plenum with a single cell, since it has large volume. In the present analysis, the calculation proceeds from the core inlet to the outlet. The boundary conditions are the inlet coolant flow and the inlet coolant temperature. The heat transfer coefficient is calculated from Dittus-Boelter formula which gives a conservative heat transfer coefficient at pseudocritical temperature.

The flowchart of the calculation code is shown in Fig.3. The calculation consists of the thermal hydraulic, reactor pressure, reactivity feedback and nuclear calculation. The input data are the inlet coolant flow and the inlet coolant temperature.

Thermal hydraulic and reactor pressure calculation

The energy and mass conservation equations are as follows:

$$v\frac{\partial}{\partial t}(\rho H) = Qout - \nabla(W \cdot H) \quad (1)$$

$$v\frac{\partial \rho}{\partial t} = -\nabla \cdot W \quad (2)$$

where,
H : coolant enthalpy(J/kg).

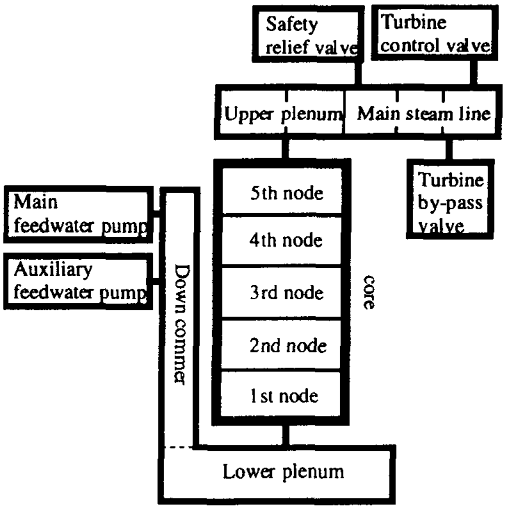

Fig.2 Calculation model

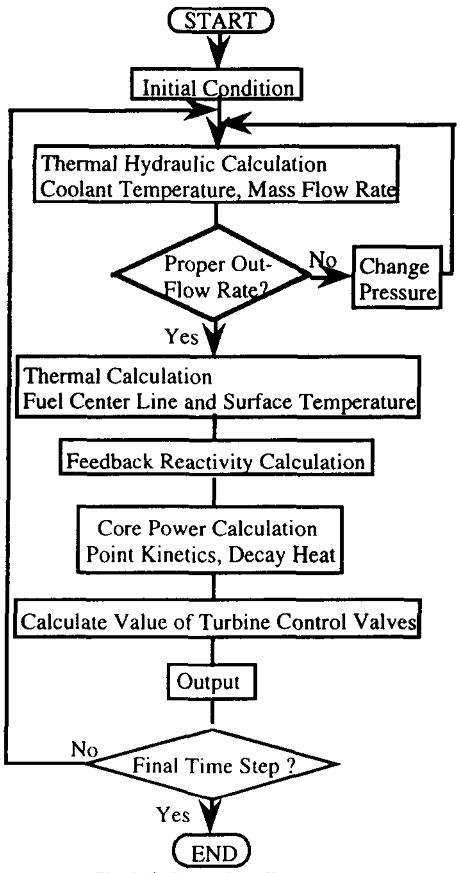

Fig.3 Calculation flowchart

Qout : heat transfer from fuel surface to coolant(W),
t : time(sec),
v : cell volume(m^3),
W : mass flow rate(kg/s),
ρ : coolant density(kg/m^3).

The coolant enthalpy and mass flow rate in the nodes are calculated from the above equations.

The opening ratio of turbine control valve is calculated from the following equations:

$$V = G(s) \cdot Vr, \tag{3}$$

$$G(s) = \frac{1+2s}{1+5s}, \tag{4}$$

where,

G(s) : transfer function,
V : angle that is defined as valve capacity ratio between the present and the full one, of turbine control valve(%),
Vr : required angle of turbine control valve(%).

The change of pellet average temperature is calculated from the following equation:

$$\frac{\partial}{\partial t} Tave = \frac{Qpellet - Qour}{Cp \cdot \rho \cdot v}, \tag{5}$$

where,

Cp : heat capacity(J/kg℃),
Qpellet : generated heat in the pellet(W),
Tave : pellet average temperature(℃).

The pellet centerline and surface temperatures are related to the pellet average temperature and the heat transfer to the coolant as follows:

$$Tcenter = Tave + \frac{r_f^2 q''}{8\bar{K}_f} - \frac{\pi \rho r_f^2 Cp}{8\bar{K}_f} \Delta Tave, \tag{6}$$

$$Tsurface = Tcenter - \frac{r_f^2 q''}{4\bar{K}_f} + \frac{\pi \rho r_f^2 Cp}{4\bar{K}_f} \Delta Tave, \tag{7}$$

where,

$\bar{K}_f$: average thermal conductivity of fuel(W/m℃),
q'' : power density(W/m^3),
r_f : fuel radius(m),
Tcenter : pellet centerline temperature(℃),
Tsurface : pellet surface temperature(℃).

The cladding surface temperature is calculated from the following equation:

$$Tclad = Tcoolant + \frac{Qout}{\pi \cdot D \cdot h}, \tag{8}$$

where,

D : fuel rod diameter (m),
h_s : heat transfer coefficient (W/m^2℃),
Tclad : cladding surface temperature (℃),
Tcoolant : coolant temperature (℃).

The heat transfer coefficient is calculated from Dittus-Boelter formula.

Reactivity feedback calculation

The Doppler feedback and coolant density feedback reactivities are calculated from the following equations:

$$k_{Doppler} = \alpha_{Doppler}(\overline{Tave}, \overline{D}, Q) \cdot \Delta\overline{Tave}, \tag{9}$$

$$k_{density} = \alpha_{density}(\overline{D}) \cdot \Delta\overline{D}, \tag{10}$$

where,

$\overline{D}$: average coolant density of all five cells weighted by the power distribution (kg/m^3),

$\Delta\overline{D}$: difference between the average and initial coolant density (kg/m^3),

$k_{Doppler}$: Doppler feedback,

$k_{density}$: coolant density feedback,

$\overline{Tave}$: average fuel temperature of all five cells weighted by power distribution (℃),

$\Delta\overline{Tave}$: difference between the average and initial fuel temperature (℃),

$\alpha_{Doppler}$: function that gives fuel temperature coefficient,

$\alpha_{density}$: function that gives coolant density coefficient.

The effective multiplication factor is calculated from the Doppler feedback and the coolant density feedback as follows:

$$keff = k_{Doppler} + k_{density} + 1.0, \tag{11}$$

where,

keff : effective multiplication factor.

Nuclear calculation

Core power is calculated from the point kinetic equation. Decay heat is calculated by two group approximation of the ANS+20% evaluation as shown in the following equation.

$$\frac{d}{dt}D_i = \lambda_i(a_i \cdot Q - D_i) \quad \{i=1,2\}, \tag{12}$$

where,

a_i : initial power ratio of group i of decay heat (constant),

D_i : power of group i of decay heat (W),

Q : reactor power (W),

λ_i : time constant of group i of decay heat (/s).

SAFETY CRITERIA

The safety philosophy is that (a) large core damage does not occur under the postulated accidents, and (b) fuel damage does not occur and the plant can be returned to the normal condition under the postulated transients. The safety criteria are made in referring to those of the LWR:

Accidents

(1) Stainless Steel (SS) cladding surface temperature below 1260℃.

(2) Reactor pressure below 110% of 27.5MPa, maximum pressure for operation.

Transients

(1) Minimum deterioration heat flux ratio (MDHFR) over 1.00.

(2) Reactor pressure below 105% of 27.5MPa, maximum pressure for operation.

(3) Maximum fuel enthalpy below 170cal/g(7.12 × 10^{5}J/kg).

The stainless steel cladding temperature criterion is based on the criterion developed for the LWRs with SS cladding by U.S. Nuclear regulatory commission[9].

The MDHFR is defined as the ratio of the cladding surface heat flux to the deterioration heat flux where the heat transfer deterioration occurs. At the supercritical pressure, water density changes continuously and boiling phenomena do not exist. The specific heat shows a sharp peak at the pseudocritical temperature. The heat transfer deterioration occurs nearby this temperature. The wall temperature increases locally and continuously where the deterioration occurs. This behavior is much milder than the burnout. The MDHFR is used in a similar way like the minimum critical heat flux ratio (MCHFR) for the BWR. In this study, Yamagata's correlation is used to evaluate the deterioration heat flux[10]:

$$q_c = 0.2G^{1.2}, \tag{13}$$

where,

G : mass velocity (kg/m^{2}s),

q_c : critical heat flux (kW/m^2).

The reasons for choosing the limit MDHFR=1.00 are:

(a) The cladding temperature does not increase sharply due to the continuous change of physical properties of supercritical water, even if the deterioration occurs.

(b) Yamagata's correlation does not include the effect of channel geometry and grid spacer, which certainly enhances the critical heat flux in the reactor. In fact, the deterioration was not observed in the rifle tube, when the heat flux was doubled[11].

The maximum fuel enthalpy criterion is applied to reactivity abnormality transients. The pressure of the gas plenum is kept lower than the coolant pressure at any burnup.

RESULTS

Total loss of reactor coolant flow accident

All feedwater pumps are suddenly tripped. The coolant flow at the cold leg decreases linearly with the flow coast down of 10sec. The result at the EOIEC is shown in Fig.4. The scram is completed at 4.7sec and AFS starts at 5.0sec. Since the coolant density decreases due to the reduction of core flow rate, the reactor power decreases through the coolant density feedback, while the cladding surface temperature increases. After the scram, the power stays at the decay heat level and the cladding surface temperature begins to decrease. The coolant flow decreases further and the cladding surface temperature increases again reaching the maximum at 12.0sec. After 12.0sec, the temperature decreases since the core flow rate increases. The maximum cladding surface temperature is 600℃ (BOIEC) and 634℃ (EOIEC). The criteria are satisfied. The flow coast down time of current LWRs is about 5.0sec. The coolant temperature is over 800℃ if the flow coast down time is 5.0sec in SCFBR.

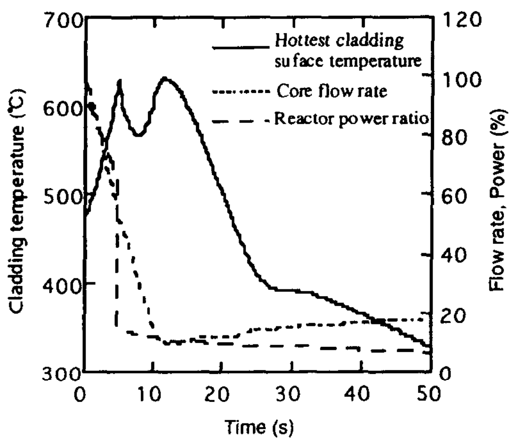

Fig.4 Total loss of reactor coolant flow (EOIEC)

Loss of feedwater heating transient

One stage of feedwater heaters is lost. This causes reduction of 35℃ of the inlet coolant temperature. The reduction of the feedwater coolant temperature is assumed to be 55℃, the same as for a BWR. The result at the BOIEC is shown in Fig.5. The coolant density at the lower plenum increases since cold coolant enters. This causes the decrease of core flow rate for a short period. During the core coolant flow rate stays low level, the coolant temperature increases and its density decreases. Then the reactor power decreases due to the coolant density feedback. The cladding temperature increases since the heat transfer from cladding to coolant decreases. After 1sec, the reactor power begins to increase up to 117% (BOIEC) and 114% (EOIEC) because cold coolant flows into the core, and the cladding temperature decreases. The reactor scram signal is not released. The MDHFR decrease to 1.17 (BOIEC) and 1.18 (EOIEC). The criteria are satisfied. It was expected that this transient would give a severe result since the SCFBR does not have a recirculation system. The cold feedwater directly flows in the core without mixing with the recirculation coolant. The obtained result is, however, not so severe due to the smaller coolant density coefficient than that of a BWR.

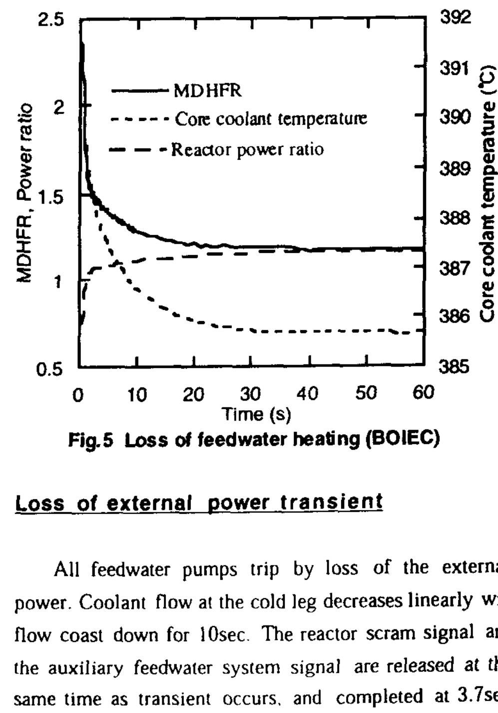

Fig.5 Loss of feedwater heating (BOIEC)

Loss of external power transient

All feedwater pumps trip by loss of the external power. Coolant flow at the cold leg decreases linearly with flow coast down for 10sec. The reactor scram signal and the auxiliary feedwater system signal are released at the same time as transient occurs, and completed at 3.7sec and 5.0sec. The result at the BOIEC is shown in Fig.6. This transient behavior is similar to the total loss of reactor coolant flow accident. The smallest MDHFR is 1.10 (BOIEC) and 1.13 (EOIEC) at 10sec. The criteria are satisfied. This transient gives the smallest MDHFR among the analyzed transients. It is because SCFBR has a small coolant inventory and does not have the recirculation system.

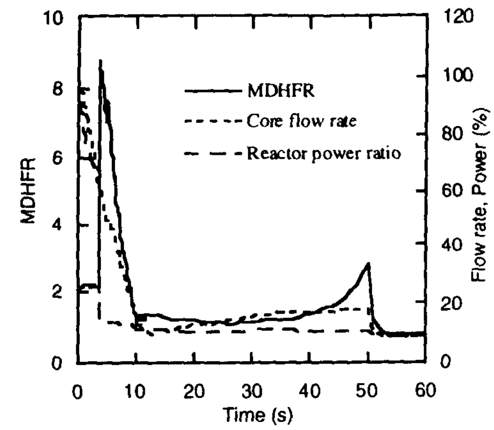

Fig.6 Loss of external power (BOIEC)

Loss of load (turbine bypass valves cannot be opened) transient

The turbine control valve is quickly closed in 0.07sec because of the loss of load. The reactor scram signal and turbine bypass valve signal are released at the same time as transient occurs. It is assumed that the turbine bypass valves are not opened. The reactor pressure and the reactor power increase until the reactor pressure reaches the set point of safety relief valves (Table 3.). The result at the BOIEC is shown in Fig.7. At 0.3sec, safety relief valves are opened, so that the reactor pressure and the reactor power decrease. The reactor pressure and the reactor power keep high level until the scram. After the scram, the power stays at the decay heat level and the reactor pressure decreases. The maximum reactor pressure is 27.1MPa (BOIEC,EOIEC) at 0.4sec. The smallest MDHFR is 1.61 (BOIEC) and does not drop below the steady state value (EOIEC). The maximum reactor power is 120% (BOIEC) and 110%(EOIEC). The criteria are satisfied. Compared

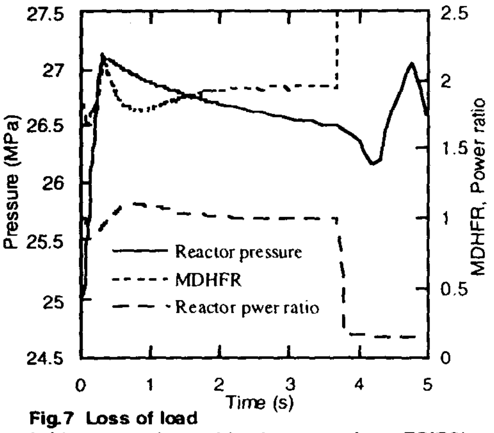

Fig.7 Loss of load (without opening turbine bypass valves, EOIEC)

with the BWR, the pressure abnormality transients of the SCFBR are not so severe because of the smaller coolant density coefficient.

Control rod withdrawal (normal operation) transient

One control rod is withdrawn continuously at normal operation. It has the highest reactivity of 3.0$ at BOIEC. The withdrawal speed is 0.091m/s. The result at the BOIEC is shown in Fig.8. Since the control rod reactivity at BOIEC is higher than that at EOIEC, only BOIEC condition is calculated. The reactor power increases due to the control rod withdrawal. The obstruction signal of control rod withdrawal is released at 11.8sec. The scram is completed at 15.5sec. The reactor power reaches 129%. The maximum fuel enthalpy is 104.9cal/g. The smallest MDHFR is 1.66. The criteria are satisfied.

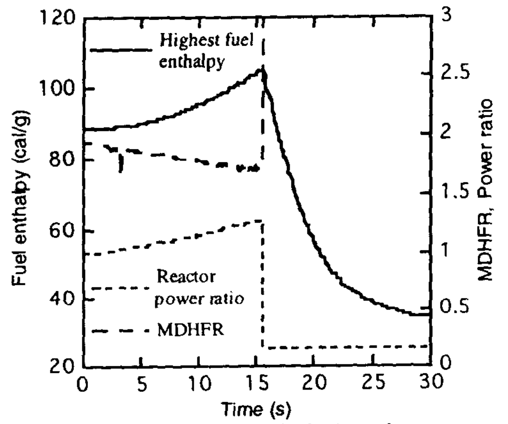

Fig.8 Control rod withdrawal (normal operation, BOIEC)

CONCLUSION

The accident and transient calculation code is developed for the analysis of the direct-cycle supercritical-pressure light-water-cooled fast breeder reactor. The effect of pressure change is considered in the code. The pressure and flow-induced accidents and transients of the SCFBR are analyzed. All analyzed accidents satisfy the criteria related to the maximum cladding surface temperature and the maximum reactor pressure. All analyzed transients also satisfy the criteria related to the minimum deterioration heat flux ratio and the maximum reactor pressure. The 'loss of external power' transient gives the most severe result among analyzed ones. The reactor safety is maintained by prolonging the pump coast down time over 10sec. The feedwater pumps will be equipped with the fly-wheels. Because of the small coolant density coefficient, the increase of core power is smaller than that of the BWR at the over-pressurized transient. The 1,245MWe reactor tolerates the control rod, pressure and flow-induced events, although there are only two coolant loops.

REFERENCES

1. Y.Oka et al., Nucl. Technol. **103**, 295-302 (1993).
2. Y.Oka et al., Nucl. Technol., **109**, 1-10 (1995).
3. Y.Okano et al., Ann. Nucl. Energy 21 , 601-611, (1994).
4. T.Jevremovic et al., Nucl. Technol. **108**, 24-32 (1994).
5. Y.Oka et al., Proc. Topl. Mtg. Advances in Reactor Physics, Knoxville, Tennessee, April 11-15, 1994, p.II-419, American Nuclear Society (1994).
6. T.Jevremovic et al., J. Nucl. Sci. Technol., **30**, 497-504 (1993)
7. Y.Okano et al., Proc. ICONE-3, Kyoto, Japan, 1995, 891-895
8. S.Koshizuka et al., Ann. Nucl. Energy, **21**, 177-187 (1994).
9. F.D.Coffman.Jr, NUREG-0065, U.S. Nuclear Regulatory Commission (1976).
10. K.Yamagata et al., Int. J. Heat Mass Transfer, **15**, 2575-2593 (1972).
11. J.W.Ackerman, J. Heat Transfer **92**, 490-498 (1970).