STABILITY ANALYSIS OF SUPERCRITICAL-PRESSURE LIGHT WATER-COOLED REACTOR IN CONSTANT PRESSURE OPERATION

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Key word: SCLWR, SCFR, Thermal-hydraulic stability, Thermal-nuclear coupled stability, Frequency domain analysis, Decay ratio, Power distribution, Sensitivity analysis.

The purpose of this study is to evaluate the thermal-hydraulic and the thermal-nuclear coupled stabilities of a supercritical pressure light water-cooled reactor. A stability analysis code at supercritical pressure is developed. Using this code, stabilities of full and partial-power reactor operating at supercritical pressure are investigated by the frequency-domain analysis. Two types of SCRs are analyzed; a supercritical light water reactor (SCLWR) and a supercritical water-cooled fast reactor (SCFR). The same stability criteria as Boiling Water Reactor are applied. The thermal-hydraulic stability of SCLWR and SCFR satisfies the criteria with a reasonable orifice loss coefficient. The decay ratio of the thermal-nuclear coupled stability in SCFR is almost zero because of a small coolant density coefficient of the fast reactor. The evaluated decay ratio of the thermal-nuclear coupled stability is 3.41 ~ 10^{-V}at 100% power in SCFR and 0.028 at 100% power in SCLWR. The sensitivity is investigated. It is found that the thermal-hydraulic stability is sensitive to the mass flow rate strongly and the thermal-nuclear coupled stability to the coolant density coefficient. The bottom power peak distribution makes the thermal-nuclear stabilit y worse and the thermal-nuclear stabili ty better.

Introduction

The concept of a supercritical pressure light water-cooled reactor (SCR) has been developed (Oka et al, 1996,1998,2000). A cooling system of SCR is a once through direct cycle in which the whole core coolant flows to the turbine. The plant system is compact and the thermal efficiency is high. The core coolant flow rate is approximately one eighth of that of BWR because of the high enthalpy difference and no recirculation. Although there is no phase change at supercritical pressure, the coolant density greatly changes in SCR core. The continuous oscillation due to a disturbance may occur as the thermal-hydraulic and the thermal-nuclear coupled feedbacks increase. This periodic phenomenon can have effects on system performance. Plant start-up was studied and two types of start-up sequences have been studied (Nakatsuka et al.1998). In the case of the constant pressure start-up sequence, the nuclear heating starts at supercritical pressure. In the beginning stage of the reactor start-up, the power should be raised so that the cladding temperature does not exceed the limit. The flow rate is kept high enough in proportion to the power. Thus, the various combinations between the power and the flow rate appear at the partial power operations.

This study deals with the operation at which the pressure in supercritical. The purposes of this study are as follows:

1. to develop a stability analysis code at the supercritical pressure,
2. to evaluate the stability for full-power and partial-power operations by the frequency-domain analysis, using this code,
3. to investigate the sensitivity.

In this study, two types of SCRs are analyzed: a high-temperature supercritical light water reactor (SCLWR-H) and a high-temperature supercritical water-cooled fast reactor (SCFR-H).

Stability Analysis Code

reactor modeling

The SCR reactor model is shown in Figure 1. The model consists of four parts; a core thermal-hydraulic characteristic model, a fuel rod heat transfer characteristic model, a nuclear characteristic model and an ex-core circulation system model.

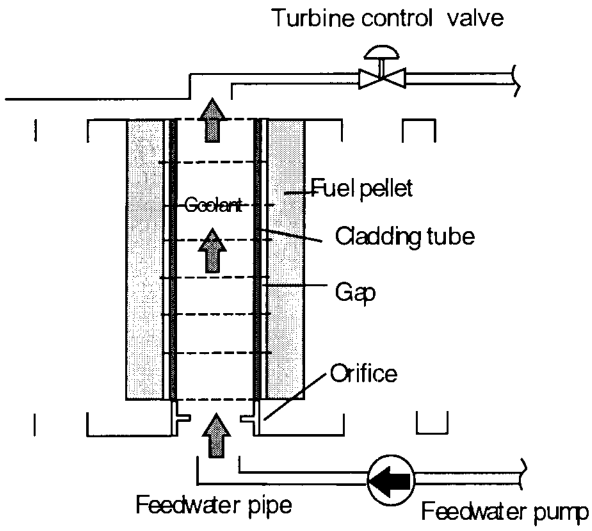


Fig.1 Reactor model

mathematical model

core thermal-hydraulic model

A single channel represents a fuel channel. A mass conservation equation, an energy conservation equation and a momentum conservation equation are used as shown below. The Blausius friction factor,f is used (Eq.4).

$$\frac{d}{dt} \quad \frac{dG}{dz} \quad 0 \qquad (1)$$

$$\frac{d\ h}{dt} \quad \frac{dhG}{dz} \quad \frac{P_e}{A} q\ (r_c, t) \qquad (2)$$

$$\frac{dP}{dz} \quad \frac{dG}{dt} \quad \frac{d}{dz}(\ u^2) \quad g\cos \quad \frac{2f}{D_h}\frac{G^2}{\ } \qquad (3)$$

$$f \quad 0.0791 \quad \mathrm{Re} \quad ^{0.25} \qquad (2100<\mathrm{Re}<10^5) \qquad (4)$$

nuclear kinetic model

Point kinetics equations with six delayed neutron groups are used to calculate the core power. We consider the Doppler effect and coolant density coefficient of reactivity simultaneously. The reactivity is obtained as the average in the axial direction weighted by the square of the power distribution. The value of and are summarized in the Appendix for SCLWR-H and SCFR-H.

$$\frac{dn(t)}{dt} \quad \frac{(t)}{\qquad} n(t) \quad \sum_{i\ 1}^{6} \ {}_iC_i(t) \qquad (5)$$

$$\frac{dC_i(t)}{dt} \quad \frac{{}_i}{\ } n(t) \quad \sum_{i\ 1}^{6} \ {}_iC_i(t) \qquad (6)$$

$$
\begin{bmatrix} 0 & 0 & 0 & 0 \\ \frac{u_{i\ 1}}{z} & \frac{\ \ _{i\ 1}}{z} & 0 & 0 \\ \frac{u_{i\ 1}^2}{z} & _A & \frac{1}{z} & 0 \\ \frac{h_{i\ 1}u_{i\ 1}}{z} & \frac{h_{i\ 1\ \ i\ 1}}{z} & 0 & \frac{\ _{i\ 1}u_{i\ 1}}{z} \end{bmatrix} \begin{bmatrix} \hat{\ }_{i\ 1} \\ \hat{u}_{i\ 1} \\ \hat{P}_{i\ 1} \\ \hat{h}_{i\ 1} \end{bmatrix} + \begin{bmatrix} s & \frac{\ _i}{z} & \frac{\ _i}{z} & 0 \\ & _B & _i s\ \ \frac{2u_{i\ \ i}}{z} & \frac{1}{z} \\ h_i s & \frac{h_i u_i}{z} & \frac{h_{i\ \ i}}{z} & 0 \end{bmatrix} \begin{bmatrix} 1 & 0 & \frac{d\ _i}{dP_i} & \frac{d\ _i}{dh_i} \\ & & 0 & 0 \\ & & & 0 \\ & & & _i s\ \ \frac{u_{i\ \ i}}{z} \end{bmatrix} \begin{bmatrix} \hat{\ }_i \\ \hat{u}_i \\ \hat{P}_i \\ \hat{h}_i \end{bmatrix} \begin{bmatrix} 0 \\ 0 \\ 0 \\ \frac{Pe}{A}\hat{q}_i \end{bmatrix}
$$

$$
_A \quad \frac{2u_{i\ 1\ \ i\ 1}}{z} \quad \frac{4f}{D_h}\lfloor u_{i\ 1\ \ i}
$$

$$
_B \quad u_i s \quad \frac{u_i^2}{z_{\ i}} \quad g\cos \quad \frac{2f}{D_h}\lfloor u_{i\ 1}^2 \tag{16}
$$

Combing three equations by the matrix form, we obtain in the core hydraulic model as Eq.16. Block diagrams of the thermal-hydraulic and the thermal-nuclear coupled stability given in Figures 2 and 3.

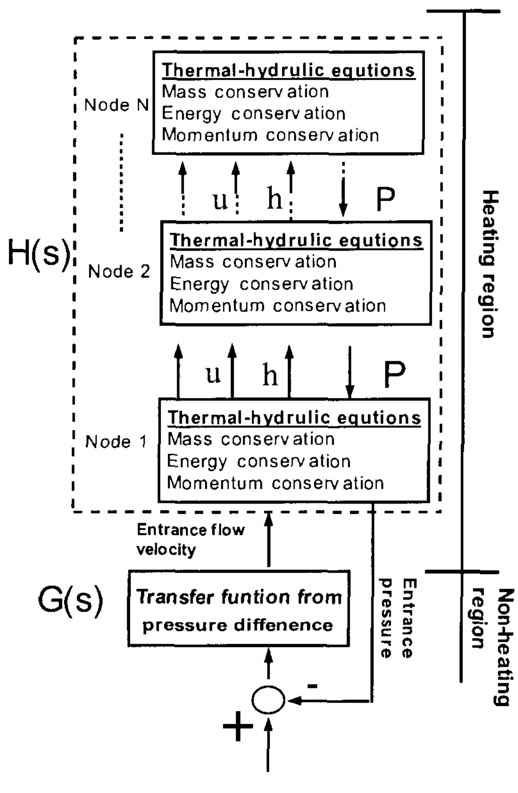


Fig. 2 Diagram of the thermal-hydraulic stability

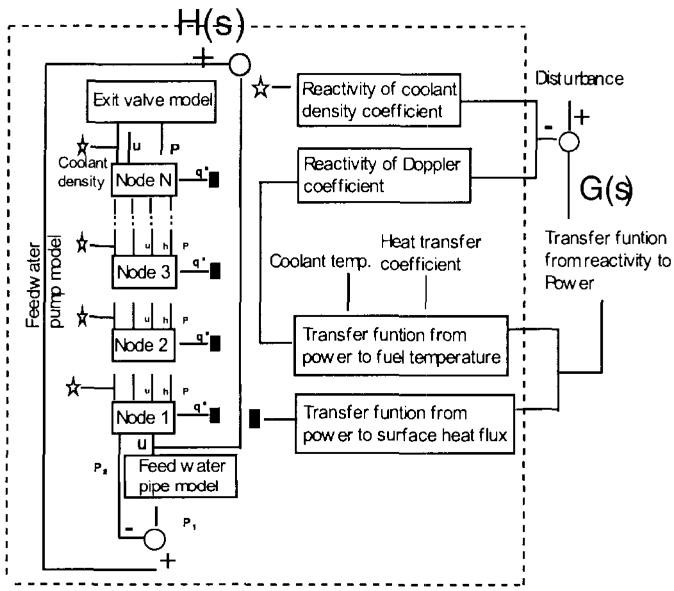


Fig. 3 Diagram of thermal-nuclear coupled stability

decay datio

By using those equations, the transfer function is obtained. The stability analysis code is verified by frequency responses at a steady state. The characteristic of the closed-loop transfer function is expressed as,

$$\frac{G(s)}{1 \quad G(s)H(s)} \tag{17}$$

The zeroes of the characteristic equation (1+G(s)H(s)=0) from Equation 17 give the poles of the closed-loop transfer function. Among the poles, the nearest pole to the complex axis represents the response of the system, which is used to calculate the decay ratio, x_2 / x_0 as shown in Figure 4. The decay ratio is defined as the proportion by which the amplitude decays in one cycle for a given step disturbance. When the mesh size changes in the axial direction, the frequency response varies a little. Therefore, the decay ratio is calculated as extrapolation at the zero mesh size obtained by the Least squares. For instance, the decay ratios are illustrated in Figure 5 depending on the pressure loss of orifice, ˜ .

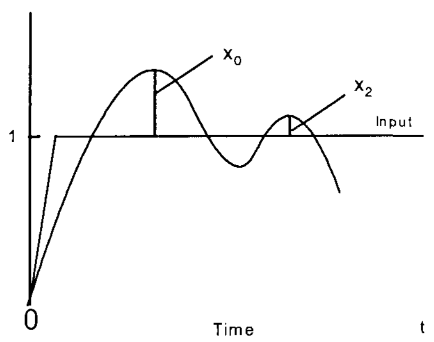


Fig.4 Decay ratio

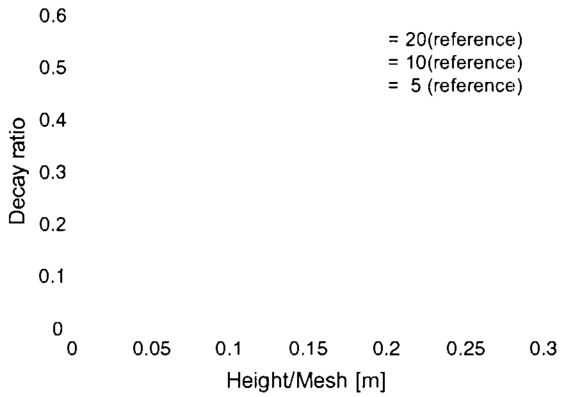


Fig.5 Change of decay ratio with axial mesh width

Stability analysis

The stability criteria of SCRs are applied the same as BWRs. The criteria are summarized in Table 1. With respect to the various operation situations, the decay ratio should be lower than 1 for maintaining its stability.

SCLWR-H

The high-temperature supercritical light water reactor (SCLWR-H) is a thermal reactor that has a large coolant density coefficient. Hence, thermal-nuclear coupled stability may be significant in SCLWR-H. The characteristics of SCLWR-H are summarized in Table 2.

To investigate the stability of the constant pressure start-up, the rated and the partial power operations are analyzed. The mass velocity decreases in proportion to the power decrease, and the inlet and the outlet temperatures are kept constant. The axial power distribution is assumed to be a cosine function.

Table 1 Criteria of Stabilit

A	ll operation		Normal operation
Thermal-hydraulic stability	$x_2/x_0<1.0$	x	$_2/x_0<0.50$
Thermal-nuclear coupled stability	$x_2/x_0<1.0$	x	$_2/x_0<0.25$

Table 2 Characteristic of SCLWR-H

Thermal/Electric Power	[MW]	**3568/1570**
Core height	[m]	**4.2**
Pressure	[MPa]	**25.0**
Fuel/Cladding material		**UO_2/Inconel**
Fuel rod diameter/ Cladding thickness/Pitch	[mm]	**8.0/0.40/9.5**
Inlet/outlet coolant Temp	[C]	**280/508**
Doppler coefficient	[(pcm/K)]	**-2.50**
Coolant density coefficient	[(dk/k)/(g/cc)]	**0.40**
Maximum/Average linear power	[W/m]	**390/246.5**
Mass velocity	[kg/ m^2s]	**1310**

thermal-hydraulic stability

The maximum power channel and average power channel are analyzed for the thermal-hydraulic stability. The frequency responses of the full power of average channel at orifice loss coefficient ˜ =10 are shown in Figure 6 .

A peak of the gain due to a resonance is large at =3, which is consistent with the time when the coolant flows through the core. The peak is due to the presence of feedback. Figure 7 shows the relation between the power and the decay ratio.

As the power decreases, the core tends to be unstable till 30% of the full power. However, the stability is improved when the power is below 30% of the full power. The relation between the orifice loss coefficient and the maximum decay ratio from Figure 7 is depicted in Figure 8. The stability increases as the pressure loss at the orifice increases. Because the reactor start-up is a kind of the transient, the criterion for normal operation is applied to analyze the thermal-hydraulic stability. As a result, when the orifice loss coefficient is greater than 6.8(the pressure loss is 0.17[bar]), the thermal-hydraulic stability satisfies the criteria of a normal operation. It is not difficult to obtain the large orifice pressure coefficient in SCLWR-H because the bottom structures of the reactor as well as the orifice at the entrance make it possible. The present design of SCLWR-H has to adopt the orifice loss coefficient more than 6.8. The criterion of the partial power operation is also satisfied at this pressure loss. The SCLWR-H is to be stable both full power and partial-power operations.

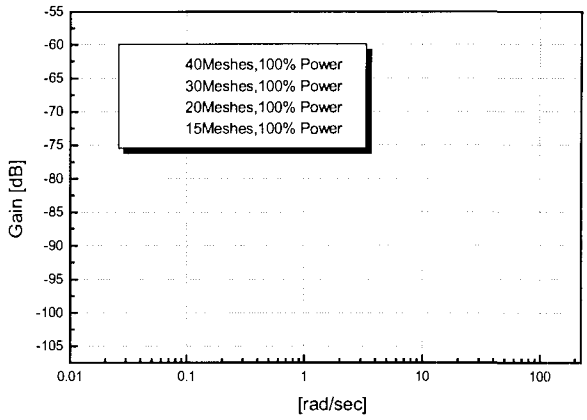


Fig.6 Frequency responses of Closed-loop transfer function (SCLWR-H)

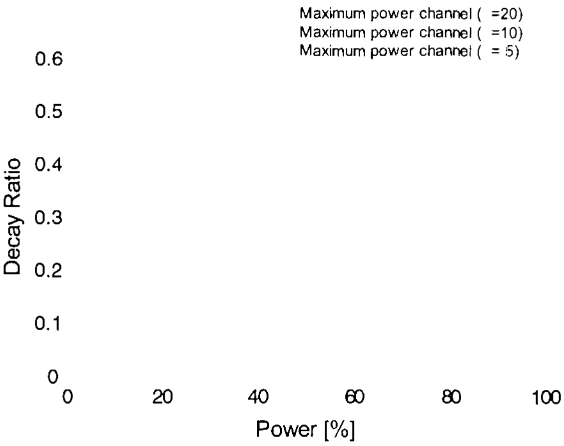


Fig.7 Relation between power and decay ratio of thermal-hydraulic stability(SCLWR-H)

thermal-nuclear coupled stability

The average power channel is analyzed in the thermal-nuclear coupled stability. Figure 9 shows the relation between the power and the decay ratio. The present calculation result shows that the stability becomes worse when the power is reduced. When the mass velocity decreases, the time that the coolant passes though the core is longer. This makes the resonance response time longer.

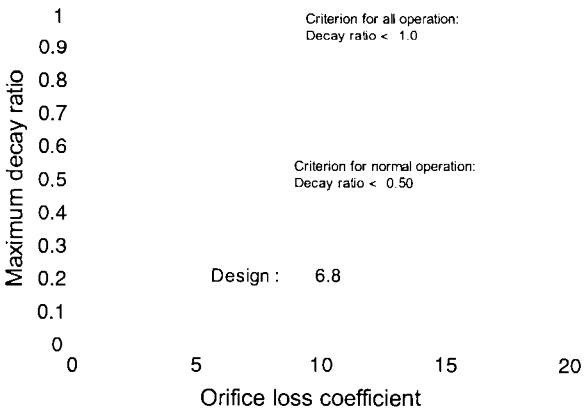


Fig.8 Relation between ˜ and maximum decay ratio of thermal-hydraulic stability(SCLWR-H)

Therefore, the thermal-nuclear coupled stability becomes worse. The decay ratio is 0.028 at the normal operation, which satisfies the criterion. The maximum decay ratio is 0.178 at the partial- power operations, which satisfies the criterion of the decay ratio for all operations. Consequently, the thermal-nuclear coupled stability of SCLWR-H is maintained.

SCFR-H

The stability of a high-temperature supercritical water-cooled fast reactor (SCFR-H) is studied. The maximum power channel is considered for both the thermal-hydraulic stability analysis and the thermal-nuclear coupled stability analysis. Procedure of the analysis is the same as that of SCLWR-H. The characteristics of SCFR-H are summarized in Table 3. Note that the coolant density coefficient of SCFR-H (0.015[(dk/k)/(g/cc)]) is substantially smaller than that of SCLWR-H (0.40[(dk/k)/(g/cc)]).

thermal-hydraulic stability

The frequency responses of the transfer function (˜ = 10) of the maximum power channel at the full power operation are similar with the SCLWR-H. The relation between the power and the decay ratio is shown in Figure 10. When the orifice loss coefficient, ˜ is 5 at the normal operation, the decay ratio is 0.37.Compared to that of SCLWR-H, the thermal-nuclear coupled stability is 1.6 times better than that of SCLWR-H.

Figure 11 shows the relation between the orifice loss coefficient and the decay ratio from Figure 10 at the maximum power channel. The value of ˜ that satisfies the operating criterion should be greater than 3(the pressure loss is 0.025[bar]. The equivalent hydraulic diameter of SCFR-H, 0.0041[m^2] is smaller than that of SCLWR-H, 0.0044[m^2].

Therefore, the pressure loss of SCFR-H, 1.175[bar] is larger than that of SCLWR-H 0.98[bar]. This makes the coolant passing time through the core fast. As a result, the thermal-hydraulic stability in SCFR-H is slightly stable than that of SCLWR-H. The thermal-hydraulic stability criteria at the normal operation are satisfied.

thermal-nuclear coupled stability

Figure 12 shows the frequency responses of the transfer function at the maximum channel when the mesh number is 40.

Due to Doppler and coolant density feedback effect at the lower frequency region, the gain is decreased. Moreover, it can be seen that the frequency responses

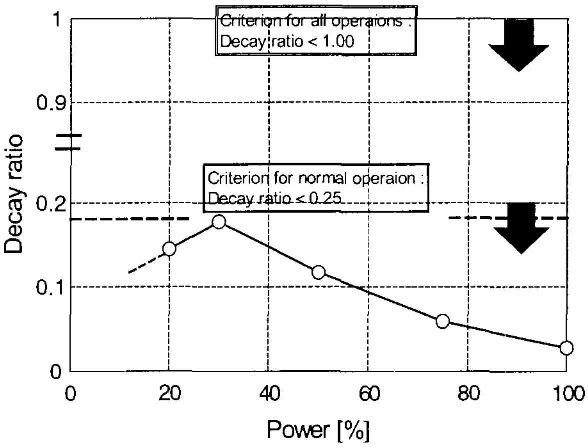


Fig.9 Relation between power and decay ratio of thermal-nuclear coupled stability (SCLWR-H)

Table 3 Characteristic of SCFR-H

Thermal/Electric Power	[MW]	**3862/1689**
Core height	[m]	**3.36**
Pressure	[MPa]	**25.0**
Fuel/Cladding material		**MOX/Inconel**
Fuel rod diameter/ Cladding thickness/Pitch	[mm]	**10.2/0.54/11.5**
Inlet/outlet coolant Temp.	[C]	**332/589**
Doppler coefficient	[(pcm/K)]	**-0.26**
Coolant density coefficient	[(dk/k)/(g/cc)]	**0.015**
Maximum linear power	[W/m]	**382**
Mass velocity	[kg/ m^2s]	**1076**

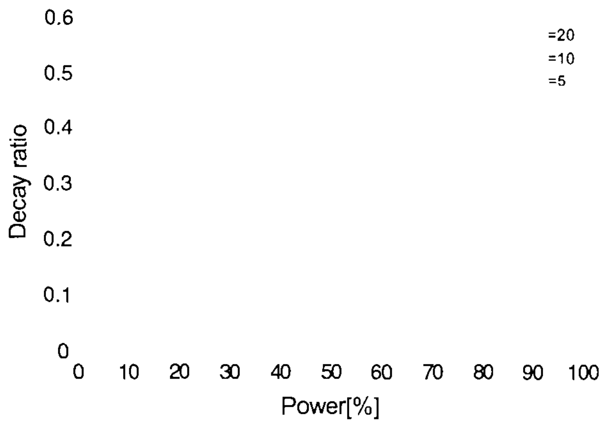


Fig.10 Relation between power and decay ratio of thermal-hydraulic stability(SCFR-H)

approach to the one point approximation because of decreasing feedback effects. Compared to that of SCLWR-H, the gain peak is smaller. If the power is high, the feedback effects play a role till the high frequency region, which represent the real reactor behavior. The relation between the power and the decay ratio is shown in Figure 13. The decay ratio of thermal-nuclear coupled stability in the SCFR is almost zero because of the small coolant density coefficient of fast reactor. It is $3.41 \sim 10^{-V}$at 100% power and the criterion is satisfied. Consequently, the thermal-nuclear coupled stability of SCFR-H is not a matter of concern.

Sensitivity analysis

Sensitivity of several principal parameters is studied. The reference core is SCLWR-H at 100% power operation of SCLWR-H. The characteristics are shown in Table 4. They are the core height, the linear power, the mass velocity, the coolant density coefficient and the axial power distribution.

core height

Sensitivity of core height to the thermal-hydraulic stability is investigated. The linear power and the inlet and outlet temperatures are fixed. Figure 14 shows the relation between the core height and the decay ratio. As shown in Figure 14, the thermal-nuclear stability is not sensitive to the core height.

linear power

Sensitivity of linear power to the thermal-nuclear coupled stability is examined.

The mass rate is kept constant and the inlet/outlet temperature is varied. The relation between the linear power and the decay ratio is shown in Figure 15. The stability does not change as the linear power increases.

mass velocity

When feed water flow rate is lost, the mass flow rate changes largely. The sensitivity of the mass flow in the thermal-hydraulic stability is investigated. As the calculation condition, the linear power is kept constant, and the inlet and outlet temperatures change with the mass velocity. Figure 16 shows the relation between the mass velocity and the decay ratio.

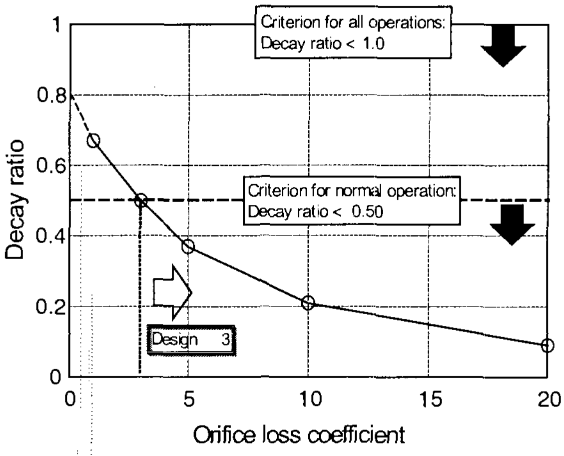


Fig.11 Relation between ~ and decay ratio of thermal-nuclear coupled stability (SCFR-H)

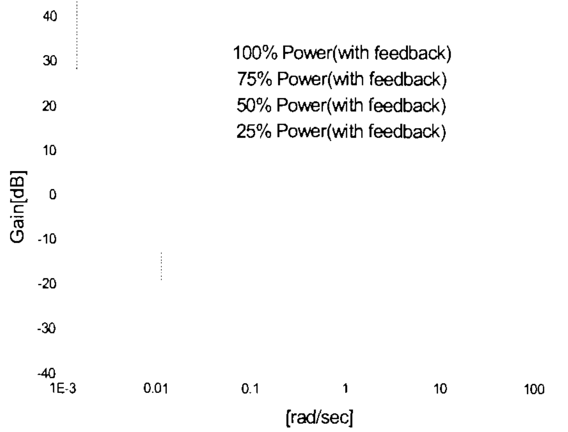


Fig.12 Frequency responses of Closed-loop transfer function (SCFR-H)

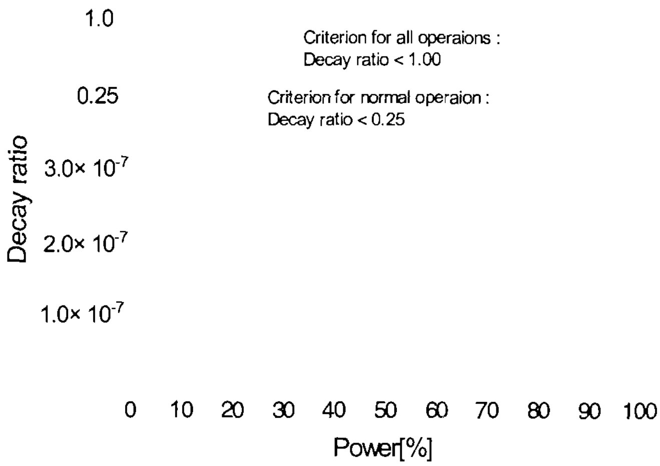


Fig.13 Relation between power and decay ratio of thermal-nuclear coupled stability (SCFR-H)

Table 4 Reference core characteristics of SCLWR-H for sensitivity analysis

Core height [m]	Mass velocity [kg/ m^{2}s]	Linear power [W/cm]	Coolant density coefficient [(dk/k)/(g/cc)]
4.2	**1310**	**246.5**	**0.40**

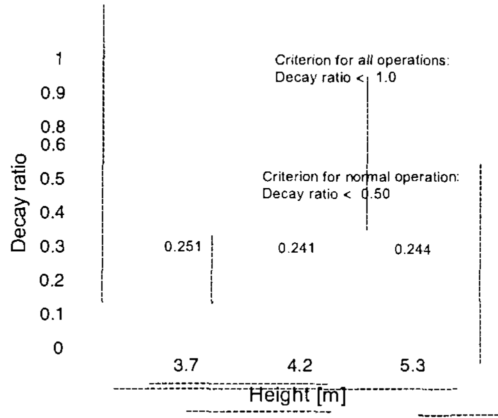


Fig.14 Relation between core height and decay ratio

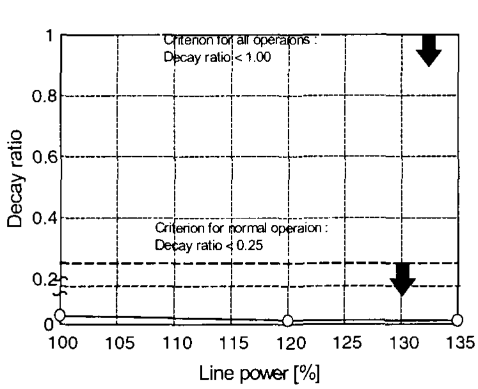


Fig.15 Relation between linear power and decay ratio

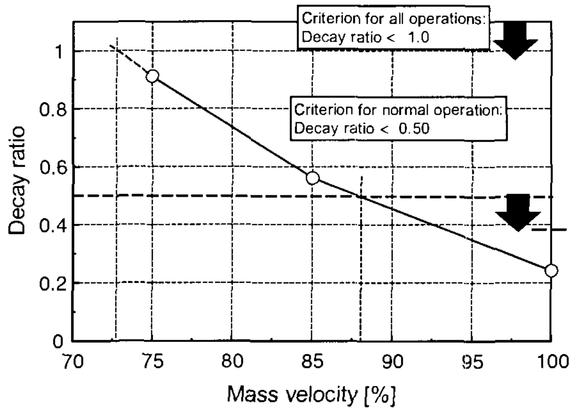


Fig.16 Relation between mass velocity and decay ratio

The decay ratio increases as the mass velocity decrease. The stability becomes worse when the flow velocity changes. However, it is not serious because this situation is limited during transients.

coolant density coefficient

Sensitivity of coolant density coefficient to the thermal-nuclear coupled stability is examined. The liner power, the mass rate and the inlet and outlet temperatures are unchanged. Figure 17 shows the relation between the coolant density coefficient and the decay ratio. As shown in Figure 17, the criterion for the normal operation is satisfied till the coolant density coefficient increases to 200~225 times as large as the reference value, 0.40[(dk/k)/(g/cc)].

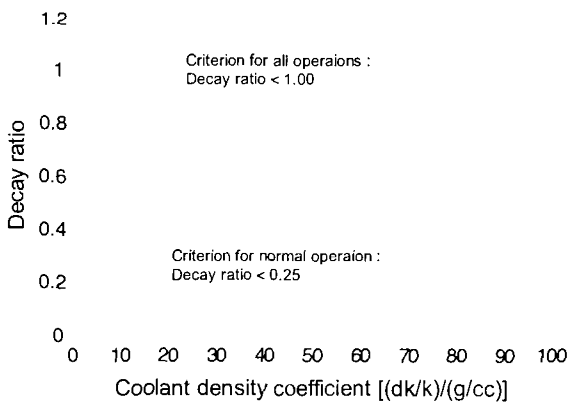
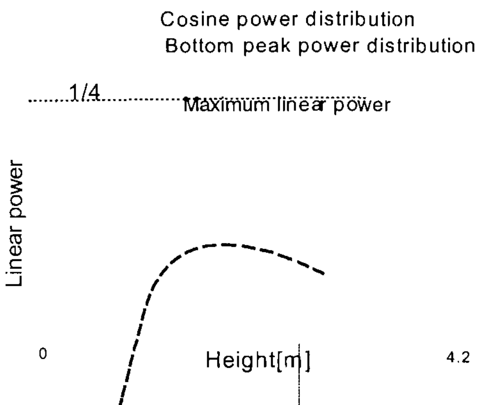


Fig.17 Relation between coolant density coefficient and decay ratio

axial power distribution

It is reported that the thermal-hydraulic stability decreases when the power having the bottom peak. Because the occurrence position of void comes to near the channel entrance, the two-phase pressure loss becomes large. The sensitivity of bottom peak of power distribution as shown in Figure 18 is studied. The total power is the same as the cosine power distribution.



Fig/18 bottom power distribution

thermal-hydraulic stability

The decay ratios on the normal operation are shown in Figure 19. From the present result, the stability of bottom power distribution is worse than that of cosine power distribution..

This is because the pressure drop (1.03[bar]) in the core on the bottom peak power distribution becomes smaller than that of the cosine power distribution (1.94[bar]). It also corresponds to a large peak of gain at the 3[rad/sec] due to the feedback effect of the bottom peak power distribution as shown in Figure 20.

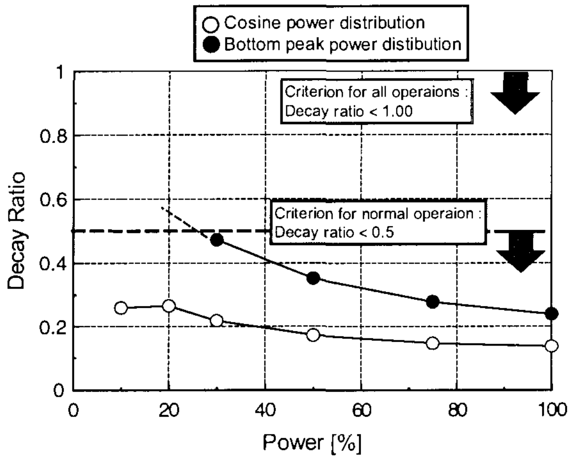


Fig.19 Comparison decay ratio of thermal-hydraulic stability between bottom peak and cosine power distribution

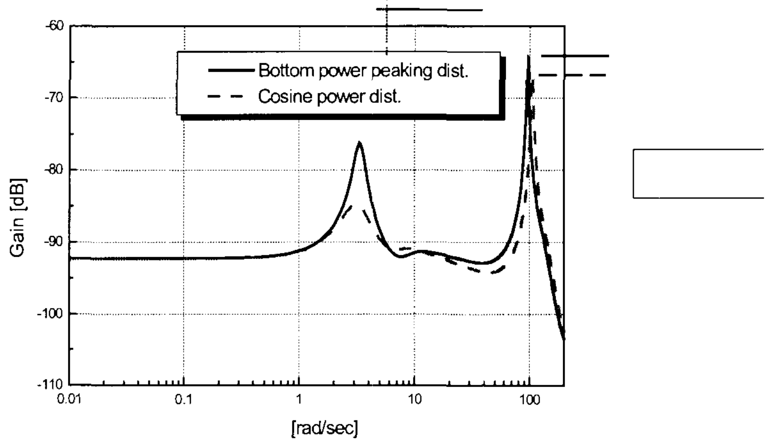


Fig.20 Comparison frequency response of power distribution in thermal-nuclear coupled stability

thermal-nuclear coupled stability

Figure 21 shows the decay ratios of the thermal-nuclear coupled stability on the normal operation. The thermal-nuclear coupled stability of the bottom power distribution is better than that of the cosine power distribution. This is because the reactivity feedback of the coolant density is smaller in the bottom peak power distribution.

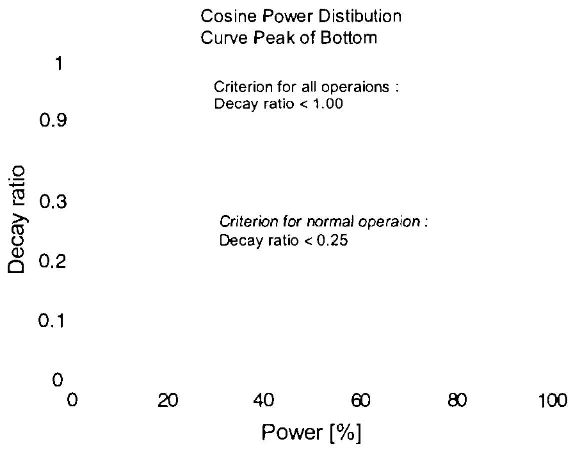


Fig.21 Comparison of decay ratio thermal-nuclear coupled stability between bottom peak and cosine powe distribution

Conclusion

The stability analysis code at supercritical pressure is developed. Using this code, a thermal-hydraulic stability and a thermal-nuclear coupled stability are analyzed. Both SCLWR-H and SCFR-H satisfy the thermal-hydraulic stability criteria when an orifice is equipped with the inlet. The thermal-nuclear coupled stability is also satisfied. The SCFR-H is greatly stable in the thermal-nuclear coupled stability than the SCLWR-H due to the small coolant density coefficient. The sensitivity of several principal parameters is investigated. The mass velocity is a dominant factor in the thermal-hydraulic stability. The thermal-nuclear coupled stability is sensitive to the coolant density coefficient. Furthermore, the bottom power peak distribution makes the thermal-hydraulic stability worse and the thermal-nuclear coupled stability better than that of the cosine power distribution.

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Appendix

Nuclear kinetic parameters of SCLWR-H

Prompt-Neutron generation time			1.34735E-05
Delayed neutron fraction			5.97062E-03
No.		Effective delayed neutron decay constants	Effective delayed neutron fraction constants
1	1	.28034E-02	0.0258
2	3	.16090E-02	0.2030
3	1	.23055E-01	0.1854
4	3	.27398E-01	0.3865
5	1	.40263E+00	0.1478
6	3	.85651E+00	0.0513

Nuclear kinetic parameters of SCFR-H

Prompt-Neutron generation time			1.4630E-05
Delayed neutron fraction			3.81611E-03
No.		Effective delayed neutron decay constants	Effective delayed neutron fraction constants
1	1	.243983E-02	0.0209
2	3	.050824E-02	0.2134
3	1	.114384E-01	0.1813
4	3	.013683E-01	0.3650
5	1	.136306E+00	0.1714
6	3	.013683E+00	0.0480