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The Loss of Coolant Flow Accident Analysis in Kori-1

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고리1호기 원자로 냉각재 유량상실사고 해석

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Abstract

The loss of coolant flow accident is analyzed for the pressurized water reactor of Korea Nuclear Unit-1. The loss of coolant flow accident is classified into three types in accordance with its severity; partial loss of coolant flow, complete loss of coolant flow and pump locked rotor accident. Analysis has been carried out in three stages; system transient and average core analysis, DNBR calculation and hot spot analysis. The purpose of developing KTRAN is to simulate the transient fast. For the DNBR calculation, the thermal hydraulic codes, SCAN and COBRA Ⅳ-I, are adopted. And for the hot spot analysis, the fuel thermal transient code LTRAN is employed.

This code system should be fast responding to the transient analysis. In case the transient occurs, severity comes within a couple of seconds. So response should be fast to accomodate the following sequence of the accident. Unfortunately this purpose could not be achieved by KTRAN. However, the calculated results are well comparable with FSAR results in range. Thereby, the effectiveness of KTRAN code analysis in this type of accident is proven.

요 약

냉각재 유량상실 사고가 가압경수형 원자로인 고리 1호기에 대하여 해석되었다. 냉각재 유량 상실 사고는 그 심각도에 따라 다음과 같이 3가지로 분류된다. 즉, 일부 유량 상실사고, 완전 유량 상실 사고, 그리고 펌프 축 고착 사고이다. 사고 해석은 계통 과도 현상 및 평균 노심분석, DNBR 계산, 그리고 고온점 분석의 3단계로 수행된다.

원자로 계통 과도 현상 코드인 KTRAN이 본 사고를 빠른 시간에 모사할 수 있도록 개발되었다. DNBR 계산을 위해서는 열수력학 코드인 SCAN 및 COBRA Ⅳ-I가 채택 되었으며, 고온점 분석을 위해서는 연료봉 과도 현상 코드인 LTRAN이 쓰였다.

이러한 전산코드 시스템은 과도 현상 해석에 빨리 응답하여야 한다. 왜냐하면 사고가 발생한 후 수초안에 심각한 상태에 이르기 때문이다. 불행히도 KTRAN코드에 의하여 이러한 목적은 충족되지 않았다. 그러나 다른 계통 해석 코드에 비하여 짧은 계산 시간에도 불구하고 KTRAN에 의한 계산 결과는 FSAR의 결과와 전반적으로 잘 일치 함으로써 KTRAN코드가 사고 해석에 유용함이 밝혀졌다.

I. Introduction

The nuclear reactor accidents are mostly caused by power cooling mismatch (PCM)[1]. PCM is anticipated to occur when the primary or secondary system fails to remove the heat appropriately from the reactor core.

The loss of coolant flow caused by reactor coolant pump (RCP) malfunction can result in PCM.

The loss of coolant flow accident may be classified into three types; partial loss of coolant flow, complete loss of coolant flow and pump locked rotor accident. These accidents are considered to be American Nuclear Society (ANS) condition Ⅱ, Ⅲ and Ⅳ accident, respectively.

If the reactor is at power operation at the onset of this accident, the immediate effects of accident are rapid increases in coolant temperature and pressure. These increases may result in departure from nucleate boiling (DNB) with subsequent fuel damage unless the reactor is tripped promptly. Fortunately, the reactor core is tripped by pertinent protection system after the initiation of accident. So the core power is decreased rapidly and the heat is removed by intact loop flow or coastdown flow by pump flywheel with large inertia. Thereby the reactor will be shutdown without causing DNB or fuel damage at hot spot. And decay heat is removed orderly by residual heat removal system (RHRS) after shutdown. The DNB is not expected to occur in partial loss of coolant flow accident and complete loss of coolant flow accident, but in pump locked rotor accident, it may occur. In this paper, the loss of coolant flow accident analysis is performed on Korea Nuclear Unit-1 (KNU-1). For this analysis, the reactor system transient code KTRAN[2] which can simulate reactor core, primary loop as well as secondary loop, is developed. For the DNBR calculation, single channel thermal hydraulic code SCAN[3] and multichannel thermal hydraulic code, COBRA Ⅳ-I[4] are adopted. And for the hot spot analysis, fuel rod thermal transient code, LTRAN[5] is employed.

The severity of this accident comes within few seconds, so this code system is developed for the fast simulation of the transient.

The results of this analysis are compared with the results in FSAR[6] of KNU-1 to show effectiveness of KTRAN code, taking short computing time and small memory size in this type of accident analysis.

II. Accident Description

The loss of flow accident may occur by mechanical or electrical malfunction of RCP or failure of power supply to RCP. This accident at power operation makes rapid rise of coolant temperature and pressure which could result in DNB with subsequent fuel damage without pertinent reactor trip. And fuel damage can result in the fission product release to the environment.

1. Partial loss of coolant flow

A partial loss of coolant flow can result from a mechanical or electrical failure in one of the RCPs or from a fault in the power supply to the pump. This accident is classified as ANS condition Ⅱ accident. Although an electrical generator is tripped, the buses remain connected with external grid and the pump works continuously to supply coolant flow to the core. And the reactor coolant pump has the flywheel with large inertia, which provides long coastdown time. The necessary protection for partial loss of coolant flow accident is actuated by low coolant loop flow signal. The additional protections are overtemperature $\Delta T(OT\Delta T)$ and overpower $\Delta T(OP\Delta T)$ trip signals. After the initiation of the accident, coolant flow is maintained with one intact loop and one coastdown flow.

2. Complete loss of coolant flow

The complete loss of coolant flow accident may result from simultaneous loss of power supplies to all RCPs. This accident is classified as ANS condition Ⅲ accident, and coolant flow is maintained with coastdown flow. The increasing rate of temperature and pressure is more rapid than partial loss of coolant flow. The protection for complete loss of coolant flow accident is provided by undervoltage or underfrequency signal or low coolant loop flow signal (2/3 coincidence).

3. Pump locked rotor accident

This event is initiated when one pump motor is locked by mechanical failure. In this accident, the failed loop flow is decreased very rapidly and reverse flow occurs. Hence the core flow is maintained only by intact loop flow. The 2/3 coincidence low loop flow signal provides protection for the reactor. Because of very rapid flow decrease, the DNB occurs in spite of pertinent trip. The rapid expansion of the coolant in the core causes surge into the pressurizer. The surge into the pressurizer and pressure increase actuate the spray system and open the relief valves in sequence and pressure rise is reduced.

III. Accident Analysis

1. Method of analysis

A loss of coolant flow accident analysis is carried out through the following three stages; the system transient and average core analysis, DNBR calculation and hot spot analysis. The computer codes used in this analysis are shown in Table 1, and the data flow between computer codes is shown in Fig. 1. The LOFTRAN[8] code used in FSAR simulates reactor coolant system, pressurizer, steam generator using point kinetics theory. The LOFTRAN can compute plant variables including coolant temperature, coolant flowrate, coolant pressure, nuclear power and average core heat flux. The THINC[9], multichannel thermal hydraulic code, was used in computing the DNBR in FSAR. And the radial temperature distribution of fuel rod and heat

Table 1. The Computer Codes Used in Anaysis

CALCULATION	CODE IN FSAR	USED IN CALCULATION
Avg. Core & System Transient Analysis	LOFTRAN	KTRAN
Avg. Core Heat Flux	FACTRAN	LTRAN
DNBR Calculation	THINC	SCAN & COBRA
Hot Spot Analysis	FACTRAN	LTRAN

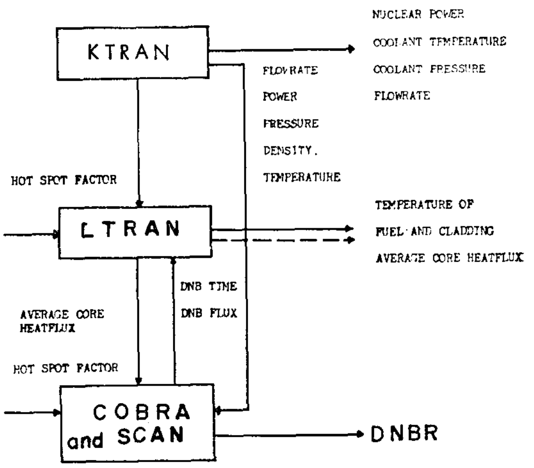

Fig. 1. Data Flow between Computer Codes

flux at cladding surface was obtained as time dependent variable with the FACTRAN[10] in FSAR.

1. 1. System transient and average core analysis

The reactor system transient code KTRAN can simulate reactor core, primary and secondary systems, pressurizer, steam generator, safety injection and protection systems. The KTRAN is used to calculate nuclear power, coolant temperature, pressure and flowrate throughout the transient. And the average core heat flux can be deduced by LTRAN using data from KTRAN calculation.

1. 2. DNBR calculation

The DNBR calculation at hot channel is performed by SCAN, single channel thermal hydraulic code, and COBRA Ⅳ-I, multichannel thermal hydraulic code. The W-3 correlation is used in this calculation. The coolant flow, temperature, pressure and heat flux of average core are used in DNBR calculation as input data.

1. 3. Hot spot analysis

The fuel rod thermal transient code, LTRAN, computes the radial temperature distribution of fuel rod and heat flux at the cladding surface from coolant flowrate, temperature, pressure, power level, hot spot factor and time that DNB occurs or DNB flux. Before DNB, the LTRAN uses Dittus-Boelter correlation[11] at subcooled convection heat transfer and Jens-Lottes' correlation[12] at nucleate boiling heat transfer region. After DNB, Bishop-Sandberg-Tong correlation[5] is utilized to determine film boiling heat transfer coefficient.

2. Calculation model of KTRAN

The KTRAN is the multiloop PWR system transient analysis code which can simulate reactor core, coolant system, pressurizer, steam generator, main steam line system, safety injection and protection systems.

The base of code is similar in layout to other larger system code, RELAP[16] or RETRAN, but

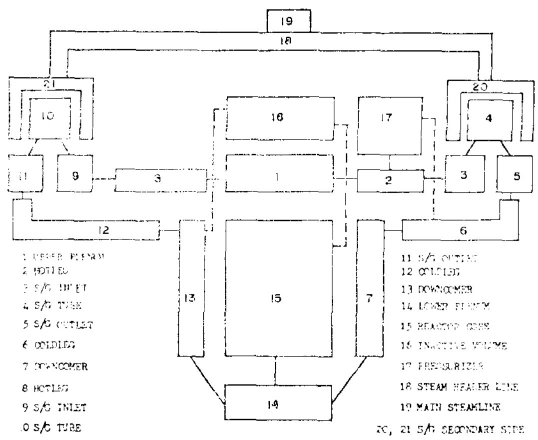

Fig. 2. System Nodalization

is simpler in modeling and has reduced flexibility. The modeling of volumes, heat conductors and other components are fixed in code. The schematic nodalization of reactor system in KTRAN is shown in Fig. 2.

The KTRAN can solve the energy and mass balance equations simultaneously using finite difference methods. The coolant flow calculation is performed separately via the momentum balance equation. This is the difference from RELAP or RETRAN model and makes short computational time.

2. 1. Kinetics model

The point kinetics model is used for the power response during transient. The major feedback mechanisms; Doppler, moderator, boron reactivities are included in evalution of K_{eff}. In addition, reactivity insertions due to control rod motion and initial reactivity are taken into account. Reactivity insertions by Doppler, moderator feedback and boron concentration changes are considered by corresponding reactivity coefficients.

2. 2. Heat transfer model

The heat transfer calculation is performed two stages, fuel rod heat conduction[13] and coolant heat transfer. The fuel rod is divided into M equal pellet nodes and two equal cladding nodes.

And coolant channel can be represented as maximum 12 axial nodes. The heat capacity and conductivity of uranium oxide are calculated as a function of temperature. The solution of heat balance equation in coolant channel reflects axial heat profile.

2. 3. Reactor coolant loop model

The KTRAN code first solves the mass and energy equations simultaneously, and then solves the momentum equation separately. The conservation of mass and energy equations for a control volume i are based on the following differential equations.

$$\frac{d\rho}{dt}=-\frac{\partial}{\partial z}\left(\frac{W}{A}\right) \quad (1)$$

$$\rho\frac{dh}{dt}=-\left(\frac{W}{A}\right)\frac{\partial h}{\partial z}+\frac{1}{J}\frac{dp}{dt}-Q \quad (2)$$

These equations are integrated over a fixed control volume i to yield;

$$\frac{dh_i}{dt}=\frac{W_i(h_i^{\text{in}}-h_i^{\text{out}})}{m_i}+\frac{dp}{dt}\frac{\nu_i}{J}-\frac{Q_i}{m_i}+\frac{(h_{\text{surge}}-h_i)}{m_i}\frac{dm_i}{dt} \quad (3)$$

where h; enthalpy, p; pressure, Q; heat generation rate, m; coolant mass, W; mass flowrate, t; time.

The inlet and outlet enthalpies are based on the average control volume enthalpies. The reactor coolant system pressure is obtained from pressurizer model, and the pressure of pressurizer is calculated from the specific volume change in the pressurizer void region with the assumption of isentropic process. The specific volume change is calculated from liquid mass change in RCS. The pressure changes in reactor core coolant loop and pump can be calculated by solving the momentum equation separately in each loop. The schematic diagram of loop configuration is shown in Fig. 3.

The conservation of momentum equation[14] for the loop i is given by

$$\Delta P_i=-\Delta P_{\text{pump}i}+\frac{L}{A}\frac{dW}{dt}+\frac{K_{RV}W_T|W_T|}{2\rho_i}+\frac{K_iW_i|W_i|}{2\rho_i} \quad (4)$$

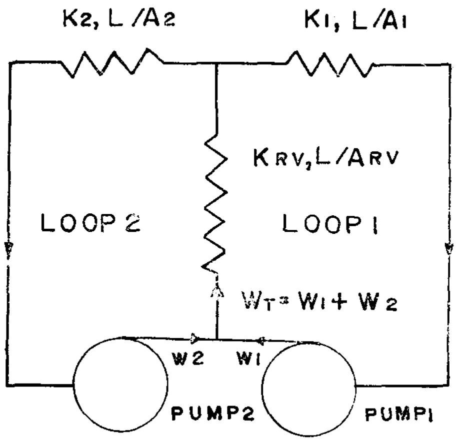

Fig. 3. Loop Configuration

where ΔP_i is the total pressure drop, K is the pressure loss coefficient and W is the mass flowrate. The relation between torque and angular speed of pump is

$$I_i\frac{d\omega_i}{dt}=T_i \quad (5)$$

And the total pump torque is given by

$$T_i=T_{mi}-T_{hi}-T_{fi}-T_{wi} \quad (6)$$

where T_{mi}: motor torque

T_{hi}: hydraulic torque

T_{fi}: friction torque

T_{wi}: windage torque

The friction and windage torques are given by[15]

$$T_{fi}+T_{wi}=1202\left(\frac{\omega}{\omega_R}\right)^{1.15}+46\left(\frac{\omega}{\omega_R}\right)^2 \quad (7)$$

And hydraulic torque is found from pump characteristic curves. The pump characteristic curves are represented by the dimensionless parameters which are given in terms of the rated values;

$\alpha=\omega/\omega_R$ angular speed

$\nu=Q/Q_R$ volumetric flow

$h=H/H_R$ head

$b=T/T_R$ torque

The pump head and hydraulic torque are thus found by; computing α and ν for given condition; finding either h/α^2 and b/α^2 or h/ν^2 and

Table 2. Pump Parameters Used in Analysis

PARAMETERS	VALUE
Rated Volumetric Flow	89,000 GPM
Rated Pump Torque	24,770 lbf-ft
Rated Pump Speed	1,190 RPM
Rated Pump Head	262 ft.
Pump Inertia	82,000 lb-ft^2

b/ν^2; calculating h and b; and using these results

$$H=H_R\cdot h$$

$$T_h=T_R\cdot b\cdot\rho/\rho_R$$

So the pump pressure rise is calculated from

$$\Delta P_{pump}=H\cdot\rho$$

The characteristic curves and pump parameters used in analysis are shown in Fig. 4. and Table 2.

2.4. Numerical method

The KTRAN utilizes the Runge-Kutta method as basic numerical scheme. The differential equations for the fuel rod temperature, coolant channel enthalpies, point kinetics are solved using Runge-Kutta method with variable time step.

The Runge-Kutta method is also used in solving the loop momentum equation. The loop flows obtained from the loop momentum equation are updated continuously for use in the RCS mass and energy equation. In the KTRAN code, the time steps are selected automatically based on the truncation error. If the maximum relative error exceeds the user specified accuracy limit, the time step is halved. And the maximum time step is specified by user.

3. Initial conditions and assumptions

In this analysis, assumptions and initial conditions are adopted for the conservative analysis as following;

a. Reactor condition is at the beginning of life (BOL) of first cycle on KNU-1.

b. Initial operating conditions are the severest with respect to DNB margin.

—maximum 102% steady state power including 2% uncertainty

—maximum steady state coolant temperature 578.2°F including measurement error 4°F

—minimum steady state pressure 2220 psi including instrumentation error 30 psi

—maximum pressure 2280 psi including the instrumentation error 30 psi for the pressure transient calculation of locked rotor accident

c. The Doppler and moderator reactivity coefficients are assumed to be −2.2 pcm/°F and 0 pcm/°F respectively to minimize the reactor power drop after shutdown.

d. The shutdown reactivity of 4% ΔK is assumed, which is conservative value compared with calculated RCCA worth. The fully insertion time of RCCA is conservatively assumed to be 2.2 seconds.

e. The reactor trips assumed in this analysis are given by

partial loss of coolant flow........... low loop flow signal

complete loss of coolant flow........... undervoltage signal

pump locked rotor ....low loop flow signal

Table 3. shows the trip set points and the trip delay times assumed in this analysis.

Table 3. Trip Point and Delay Time Used in Analysis

TRIP FUNCTION	TRIP SET POINT	DELAY TIME
1. Power range high neutron flux		sec.
—high setting	118%	0.5
—low setting	35%	0.5
2. Overtemperature ΔT		6.0
Overpower ΔT		6.0
3. High PZR pressure	2385 psig	2.0
Low PZR pressure	1865 psig	2.0
4. High PZR level	100% of PZR level	2.0
5. Low reactor coolant flow	87% loop flow	1.0
6. Undervoltage	68% of nominal	1.2
Underfrequency	58 Hz	0.6
7. Low low S/G level	0% of narrow range span	2.0

f. The delayed neutron fraction and neutron life time are assumed to be $\beta=0.0075$ and $1^{*}=1.844\times10^{-5}$ at BOL.

g. The pressurizer relief valve actuates at 2350 psi in the average core analysis. But, in the pressure transient calculation of the pump locked rotor accident, the pressurizer relief valve is not considered to obtain conservative result (high pressure).

h. In hot spot analysis, Bishop-Sandberg-Tong correlation for film boiling heat transfer coefficient calculation is employed. And the initial values of pressure and bulk density are used throughout the transient since they are the most conservative with respect to cladding temperature response.

i. The pump characteristic curves for calculation of flowrate are adopted from RELAP-4[16] built-in curves of Westinghouse type pump. These are shown in Fig. 4.

IV. Results and Discussion

1. The system transient and average core analysis

In this analysis, flowrate, coolant temperature, system pressure, nuclear power and heat flux are calculated. As shown in Fig. 5. through Fig. 7, flowrates are in good agreement with those given in FSAR.

In the partial loss of coolant flow accident, the failed loop flow coasts down by flywheel inertia, and the intact loop flow is increased to

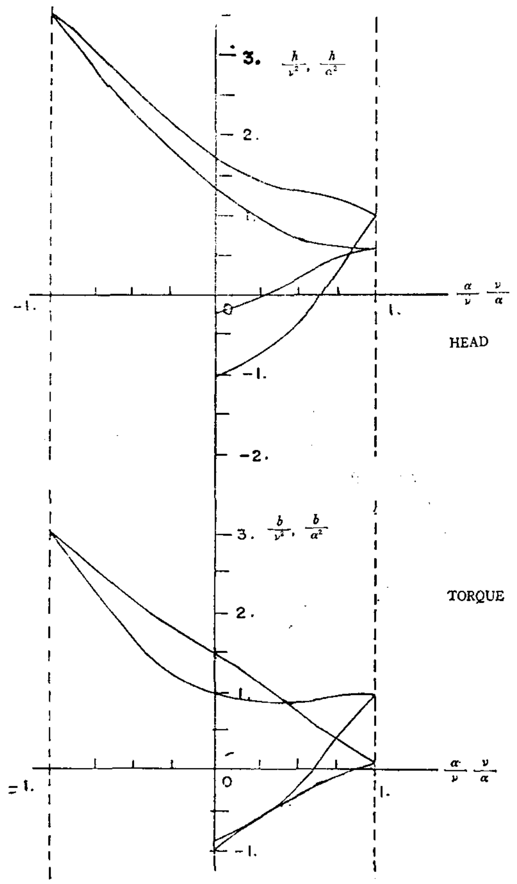

Fig. 4. Pump Characteristic Curve (Westinghouse Type)

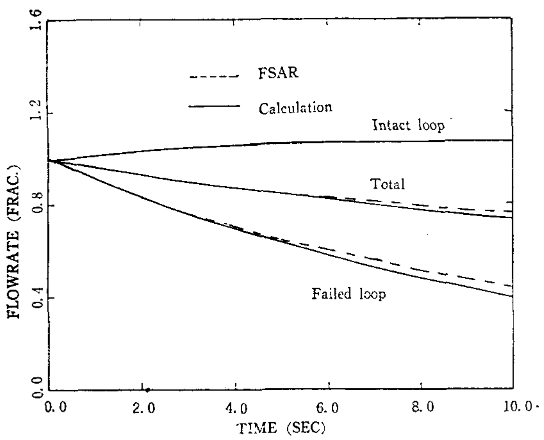

Fig. 5. Flowrate (Partial Loss of Flow)

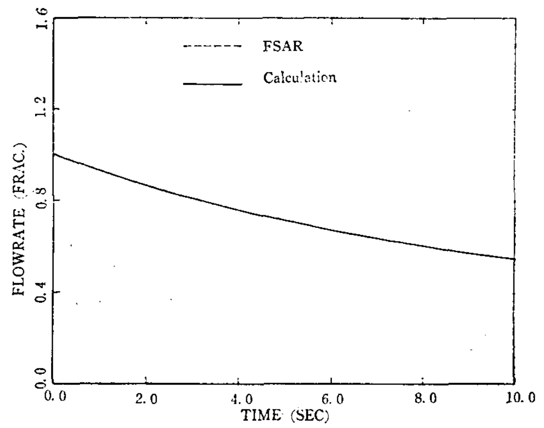

Fig. 6. Flowrate (Complete Loss of Flowrate)

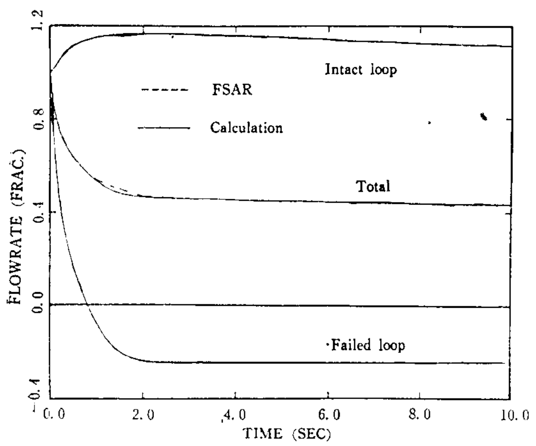

Fig. 7. Flowrate (Pump locked rotor)

107% of nominal flowrate by effect of failed loop. In case of complete loss, the coastdown flow is decreased to 54.5% of nominal flowrate in 10 seconds. In pump locked rotor accident, the reverse flow occurs in the failed loop, so flowrate is dropped rapidly to −24.8% of nominal flowrate. Meanwhile, the intact loop flow is increased to 116.6% of nominal flowrate.

After the reactor is tripped by the actuation of trip signal, the power and heat flux are decreased. These are shown in Fig. 8. to Fig. 10. The calculated trend of nuclear power in partial loss of flow accident is similar to FSAR, but there is time delay about 0.5 second between them. Thereby, it can be said that there is difference in trip delay time.

The calculated heat fluxes are lower than those of FSAR. In the calculation model, the fuel rod is divided into five equal nodes in the fuel region and two nodes in the cladding region, however, the LOFTRAN code used in FSAR calculation, considers only one radial node. Such difference in fuel rod nodalization can affect the heat flux calculation.

The core average temperatures are shown in Fig. 11. Increases in the coolant temperature are not significant except the pump locked rotor accident. A large increase in the coolant tem-

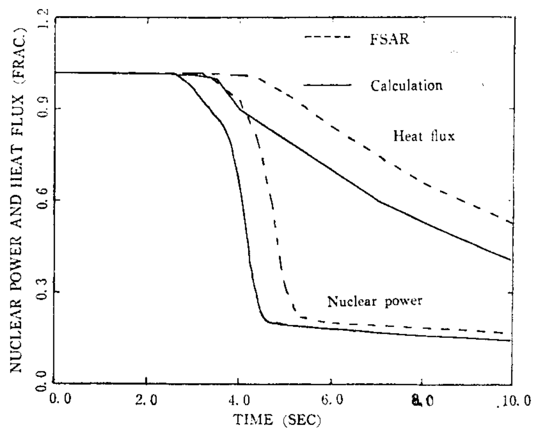

Fig. 8. Nuclear Power and Heat Flux (Partial loss of flow)

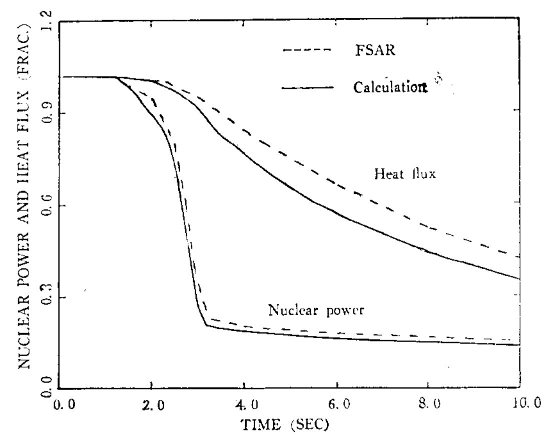

Fig. 9. Nuclear Power and Heat Flux (Complete loss of flow)

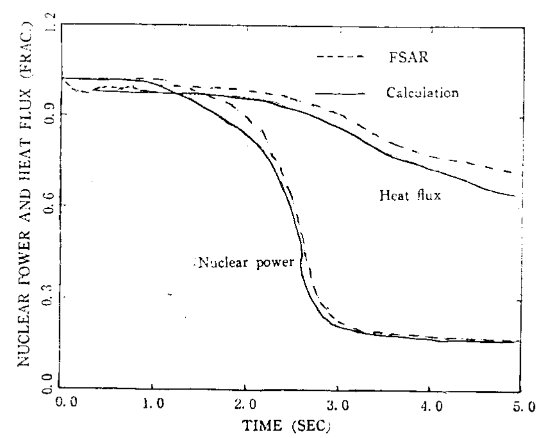

Fig. 10. Nuclear Power and Heat Flux (Pump locked rotor)

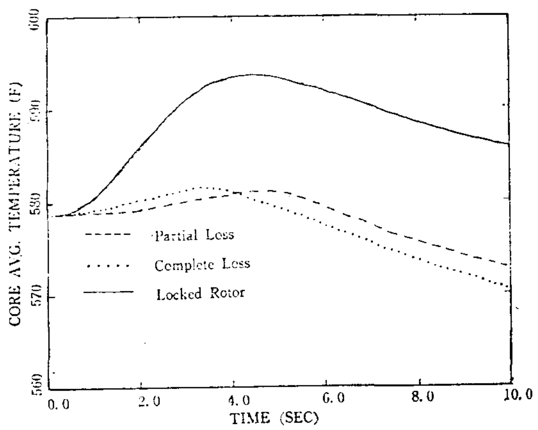

Fig. 11. Core Average Temperature

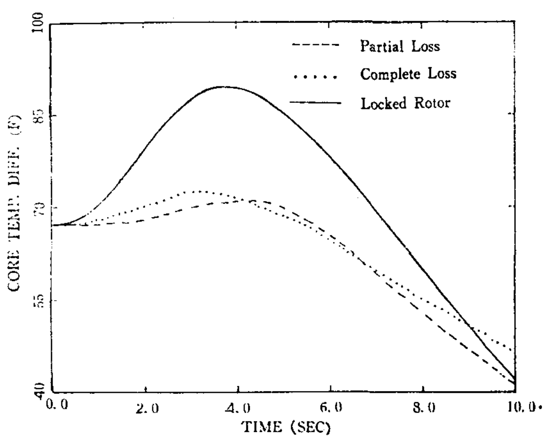

Fig. 12. Core Temperature Difference

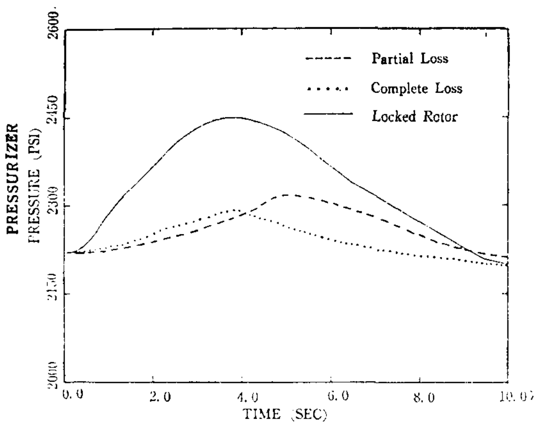

Fig. 13. Pressurizer Pressure

perature indicates the high probability of DNB. The temperature rise is ceased by rapid decrease of power and heat flux after reactor trip.

The Fig. 13. shows the pressure change during accident. As the coolant temperature is increased, the pressure is increased. The complete loss of flow accident is more severe than the partial loss of flow accident. But as shown in Fig. 13, the pressure in the partial loss of flow accident is higher than that of complete loss of flow accident. This is because of the difference in trip signal. The undervoltage trip signal is actuated at the initiation of accident, but the low flow trip signal is actuated at 1.6 second after accident in partial loss of flow accident. The steam generator model adopted in KTRAN considers the heat transfer to the secondary side by only one mode; nucleate boiling. But the change in heat transfer mode is expected in very fast transient like pump locked rotor accident. So the further study on the steam generator model is recommended.

The KTRAN takes 250 seconds in analysis of this accident by CDC 6000 computer. Such computation time is not sufficiently fast to accomodate the following sequence of this accident, however, it is relatively short computation time compared with other system transient codes.

2. The DNBR calculation

DNBR calculations are performed with the thermal hydraulic code SCAN and COBRA IV-I. In the pump locked rotor accident, DNB occurs throughout the transient. The DNB occurrence time versus hot spot factor calculated by SCAN is shown in Table 4.

Table 4. DNB Time vs. Hot Spot Factor in Locked Rotor Accident Calculated by SCAN

HOT SPOT FACTOR	DNB TIME
2.80	1.0 sec
3.00	0.8 sec
3.50	0.5 sec
4.00	0.2 sec
4.50	0.0 scc

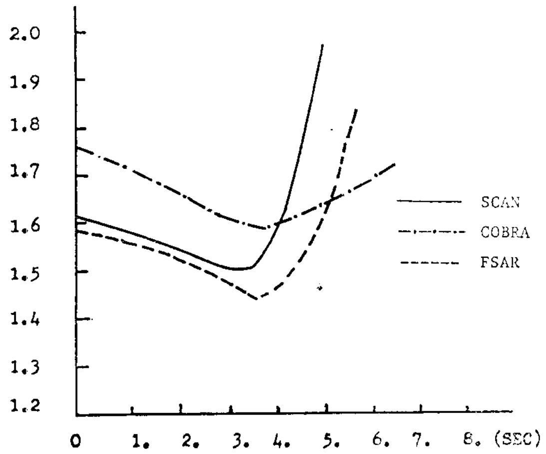

Fig. 14. DNBR Partial Loss of Flow

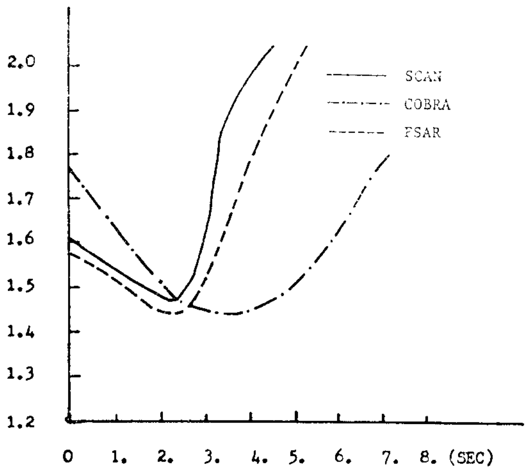

Fig. 15. DNBR Complete Loss of Flow

The DNBRs of partial loss of flow accident and complete loss of flow accident which are calculated by SCAN and COBRA IV-I areshown in Fig. 14 and Fig. 15. The figures show that the results of SCAN are more consistent the FSAR than those of COBRA IV-I. This results from the difference in coolant mixing evaluation between codes. Since the SCAN can not directly evaluate coolant mixing, it uses the bias curve which is calculated by THINC. The more rapid change of DNBR in SCAN than COBRA IV-I may be caused by overestimation of coolant mixing in SCAN. However, generally, the results of COBRA IV-I are more reliable than those of FSAR and SCAN.

3. The hot spot analysis

The hot spot analysis in partial loss of flow accident and complete loss of flow accident are omitted because the DNB is not expected to occur. In pump locked rotor accident the DNB occurs during the transient. The DNB occurrence time assumed in the FSAR of KNU-1 is 0.5 second and that of KNU-2 is 0 second. The corresponding hot spot factors are 3.5 and 4.5 respectively. The analysis in the pump locked rotor accident is performed using hot spot factors 3.5, 4.5 and 2.8. The hot spot factor 2.8 is the value at steady state including 20% uncertainty. The results are shown in Fig. 16. In the case of hot spot factor 3.5, the peak inner cladding temperature is lower than that of FSAR because the calculated heat flux is underestimated compared with FSAR. Even the worst case of hot spot factor 4.5 assumed in the FSAR of KNU-2, the results did not exceed the design limits; fuel melting point of 5080°F and peak cladding surface temperature 2200°F. Thus the core will remain in the place and intact without loss of

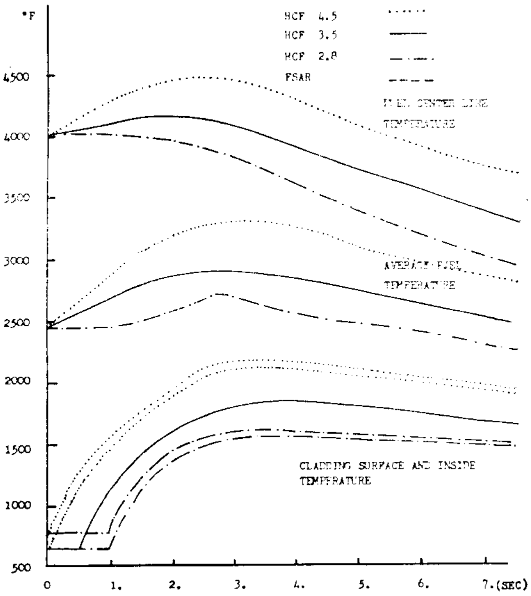

Fig. 16. Fuel and Cladding Temperature (Pump locked rotor)

core cooling capability.

V. Conclusion

The multiloop reactor system transient code KTRAN is developed and utilized in the analysis of loss of coolant flow accident on KNU-1. The initial conditions and assumptions are adopted from FSAR.

The purpose of this research is to develop the fast running transient code to respond to the loss of coolant flow accident. As the results show, its severity comes within a few seconds. Therefore, with present code system, operating personnel could not respond the accident quickly. So the fast running transient code system should be developed. Initially KTRAN is expected to do this job. Unfortunately this quick response was not achieved with KTRAN. However, results of KTRAN calculations are well comparable to the conservative values of FSAR. Accordingly the effectiveness of calculation models in KTRAN is at least verified. Further revision in couple of model would be expected to meet fast response.

In KTRAN analysis for the loss of coolant flow accident, important results can be summarized as the following;

In the partial loss of coolant flow accident and complete loss of coolant flow accident, the reactor is cooled by coastdown flow and shutdown to decay heat level without occurring DNB.

The DNB is expected to occur in pump locked rotor accident, however, the fuel and cladding temperatures did not exceed the design limits.

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