Journal of the Korean Nuclear Society
Volume 19, Number 3, September, 1987

Development of Transient Simulation Code for Pressurized Water Reactors

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(Received July 8, 1987)

가압경수형 원자력발전소의 과도현상 모의코드 개발

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Abstract

A plant simulation code, MCSIM(Micro-Computer SIMulator), has been developed to simulate plant transient accidents for pressurized water reactors. Reactor coolant system is modeled using decoupled energy and momentum equations, drift flux two-phase flow model and integral momentum equation. A two-fluid pressurizer model is used to simulate the pressurizer dynamics. Pot Boiler model is used for steam generator, steady-state decoupled energy and momentum equations for secondary side system, and point kinetics equations for nuclear power calculation. For test of the present version of MCSIM, complete loss of flow and RCCA withdrawal accidents are calculated with MCSIM. The results are compared with those in FSAR of KNU 5 & 6.

요 약

발전소 과도현상과 비냉각재 상실사고를 모의할 수 있는 가압경수로발전소 모의코드 MCSIM을 개발하였다. 원자로 냉각재계통은 에너지 방정식과 운동량 방정식을 분리 취급하면서 Drift Flux 2상 유동모델, 적분 운동량 방정식 등을 사용하여 모델링하였다. 증기발생기의 모사는 Pot Boiler 모델을 사용하였고, 2차계통을 위해서는 분리 취급된 정상상태 에너지 방정식과 운동량방정식을 핵출력 계산을 위해서는 점 동특성 방정식을 사용하였다. 현재의 코드성능을 시험하기 위해 완전 냉각재 유동상실사고와 제어봉 집합체 인출 사고를 계산하여 그 결과를 원자력 5/6호기 최종 안전 보고서의 결과와 비교하였다.

Nomenclature

A =flow crossectional area, m^2
E =total internal energy, kJ
f =friction factor
H =enthalpy, kJ/kg
h =specific enthalpy, kJ/kg
K =form loss factor
L =flow length, m
M=mass, kg
P =pressure, Pa
Q =heat transfer rate, kW
T=temperature, K
V=volume, m

W =water or steam flow rate, kg/s
α =shell side void fraction
β =effective fraction of delayed neutron
=density, kg/m
=reactivity

Subscripts

F =fuel
fd =feed water
I =impulse stage
i =inside control volume
M =moderator
l, v=liquid, vapor
g, f=gas, fluid
SL=steam line

1. Introduction

After the TMI accident, the necessity of real-time simulators has been emphasized to train operators and for helping operators diagnose or mitigate accidents. For those simulators, a fast running computer code with sufficient accuracy is required. To meet this requirement codes, for example [1] and [2], are under development.

The present work also contributes to the above mentioned research field. A plant simulation code named MCSIM (Micro-Computer SIMulator) has been developed for pressurized water reactor plants. The code can be used to analyze transients. Fast running capability is considered by use of decoupled energy and momentum equations, integral momentum equation and large control volumes. But, main effort has been made to achieve sufficient accuracy of the code.

In the present work, palnt system models adopted for MCSIM are described. Finally, some application results of the code are discussed comparing with results in FSAR for KNU 5 & 6.

2. Model Description

2.1. Nuclear Power Generation

Reactor kinetics is simulated by introducing the point kinetics model with prompt jump approximation[3]. The point kinetics equations can be derived from the continuous space-time-energy dependent diffusion equations with integration concept. The general configurations of these equations are given as,

$$\frac{dP_k(t)}{dt}=\frac{P(t)-\beta}{\Lambda}P_k(t)+\sum_i \lambda_i C_i(t) \quad (1)$$

$$\frac{dC_i(t)}{dt}=\frac{\beta_i}{\Lambda}P_k(t)-\lambda_i C_i(t) \quad i=1,\cdots,6, \quad (2)$$

The time dependent reactivity term makes the equations to be a nonlinear ordinary differential equation set, and so, some numerical techniques are needed to solve them. To avoid the numerical instabilities due to the wide range of time constants in these equations, the prompt jump approximation is adopted. This can be accomplished by only making the time derivative of neutron power term equal to zero. This approximation removes the effect of very short time constant, which may cause some numerical instabilities. Using this approximation, the point kinetics equations become,

$$P_k(t)=\frac{\Lambda}{\beta-\rho(t)}\sum_i \lambda_i C_i(t) \quad (3)$$

$$\frac{dC_i(t)}{dt}=\frac{\beta_i}{\beta-\rho(t)}\sum_i \lambda_i C_i(t)-\lambda_i C_i(t). \quad (4)$$

The major parameters which will affect the reactivity changes are the moderator and fuel temperature feedback effects and the control rod motion. These quantities can be obtained through the following equations;

$$\rho_M(t)=\alpha_M(T_M(t)-T_{M0}), \quad (5)$$

$$\rho_F(t)=\alpha_F(T_F(t)-T_{F0}) \quad (6)$$

Then, the total reactivity insertion at any time is given as,

$$\rho(t)=\rho_F(t)+\rho_M(t)+\rho_{EX}(t) \quad (7)$$

Because of the feedback reactivities coupled implicitly with the neutron power, the reactor power at each time step is determined by iterative way.

Lumped parameter model[4] is used to evaluate the fuel rod temperature changes with the generated power and the moderator temperatures are calculated by considering the moderator region as a single node. The solutions of the ordinary differential equation set, equation(3) and (4), are obtained by using the Runge-Kutta fourth order formula. If we represent the equation (3) and (4) as

$$\frac{dC_i(t)}{dt}=f_i(C_j(t),\ t),\ j=1,\cdots,6 \tag{8}$$

$$P_k^{n+1}=\frac{\Lambda}{\beta-\rho^{n+1}}\sum_i \lambda_i C^i{}_{n+1}. \tag{9}$$

The stability and convergence of Runge-Kutta method [5] are well proved, and the truncation error of the fourth order formula is known as about $0(t)$.

2.2. Reactor Coolant System

The following components are modeled as

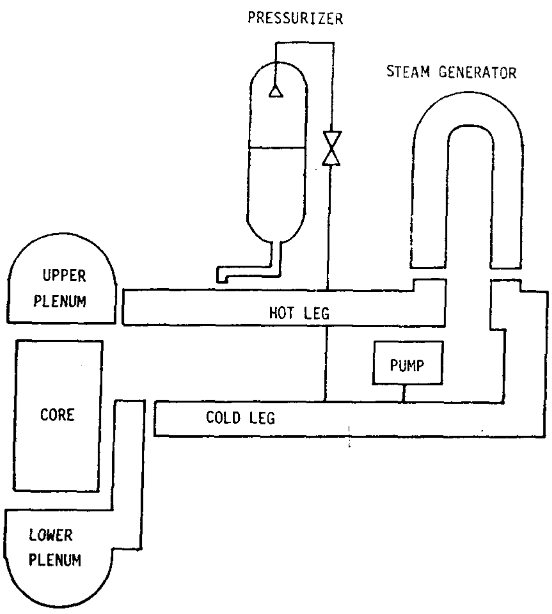

Fig. 1. Reactor Coolant System Control Volume Noding Scheme

reactor coolant system and shown in Fig. 1; reactor core, reactor vessel upper plenum, hot leg, steam generator tube side, cold leg, reactor vessel lower plenum, reactor coolant pump and pressurizer. The momentum equation is decoupled from the energy equation to get integral momentum model [7].

$$\frac{LdW}{Adt}=\Delta P_{\text{pump}}-\frac{fL}{2\rho D_h A^2}W|W| -K\frac{W|W|}{2\rho A^2}+\phi\rho \underline{g}\cdot d\underline{s} \tag{10}$$

Here, D_h and $\underline{s}$ mean the hydraulic diameter and position vector, respectively. Pump head is calculated using pump characteristic equations [9]. The mass and energy balance equations of the control volume i are:

$$\frac{dM_i}{dt}=W_{i-1}-W_i+W_s, \tag{11}$$

$$\frac{dE_i}{dt}=(WH')_{i-1}-(WH') +(WH')_s+Q_i \tag{12}$$

Equations (11) and (12) can be combined by use of the average mixture enthalpy of control volume i. The resulting energy equation can be simplified by assuming uniform loop flowrate. For further development of the equation, the linear enthalpy profile within control volumes is assumed. In a saturated two-phase flow condition, drift flux model is used to obtain a relation between mixture enthalpy and mixing-cup enthalpy H' [6]. The boundary conditions are spray flow rate and enthalty, W_{sp} and H_{sp}, relief valve flow rate, W_{rv}, surge flow rate and enthalpy, W_{su} and H_{su}, and heater power. The mass and energy balance equations are for vapor

$$\frac{dM_v}{dt}=W_{fl}-W_{r0}-W_{sc}-W_{rv}-W_{wc}, \tag{13}$$

$$\frac{d(Mh)_v}{dt}=W_{sp}(h_{sp}-h_f)+W_f h_g -(W_{r0}+W_{sc})h_f-W_{rv}h_v -W_{wc}h_v+V_v\frac{dp}{dt}, \tag{14}$$

and for liquid;

$$\frac{dM_l}{dt}=M_{su}+W_{sp}+W_{sc}+W_{wc}+W_{r0}-W_{fl}, \qquad (15)$$

$$\frac{d(Mh)_l}{dt}=(Wh)su+(W_{sp}+W_{sc}+W_{wc}+W_{r0})h_f,-W_{fl}h_g+Q_h+V_f\frac{dp}{dt} \qquad (16)$$

The above 4 equations can be solved along reactor coolant system with mass and energy equations (11) and (12). The computational flow chart of reactor The computational flow chart of reactor coolant system is shown in Fig. 2. First, momentum and pump characteristic equations are solved, then pressurizer and flow path state equations are solved to get updated properties within the control volumes of Fig. 1. Boundary conditions are nuclear power and shell side steam generator pressure.

2. 3. Steam Generator Shell Side

Pot Boiler model [8] is used to simulate shell side dynamics of U-tube steam generator. The control volume is the entire shell side. Assuming complete mixing and thermal equilibrium, energy and mass equations become;

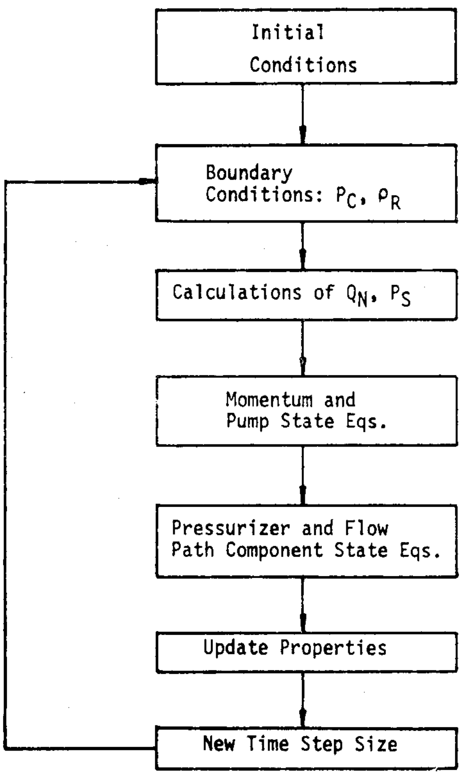

Fig. 2. Computational Flow Chart

$$V_s(\rho_g-\rho_f)\frac{d\alpha}{dt}+V_s\left\{\alpha\frac{d\rho_g}{dp}+(1-\alpha)\frac{d\rho_f}{dp}\right\}\frac{dp}{dt}=W_{fd}-W_{st}, \qquad (17)$$

$$V_s(h_g\rho_g-h_f\rho_f)\frac{d\alpha}{dt}+V_s\left\{-1+\alpha h_g\frac{d\rho_g}{dp}+\alpha P_g\frac{dh_g}{dp}+(1-\alpha)h_f\frac{d\rho_f}{dp}+(1-\alpha)\rho_f\frac{dh_f}{dp}\right\}\frac{dp}{dt}=h_{fd}W_{fd}-h_gW_{st}+Q. \qquad (18)$$

Since reactor coolant system modeling includes tube side dynamics, the above pot boiler model is used to calculate steam generator pressure and level at given conditions that is feedwater flow rate and enthalpy.

2. 4. Secondary Side

The major part of the secondary side modeling is to obtain the thermohydraulic properties of the main steam and main feedwater. The properties can be calculated with boundary conditions; the pressure and temperature of steam generator and condenser. The model consists of steam line, impluse chamber, high and low pressure turbine, moisture separator and reheater, low and high pressure preheater, main feedwater pump, condensate pump and condenser. Two control systems are considered in this model, one is the steam generator water level control system, the other is the feedwater pump speed control system. Schematic diagram of the secondary side of PWR plant is shown in Fig. 3. Steam line model simulates the dynamics of the steam line in the secondary side. The line are modeled with a single control volume, which is total pipe volume from the main steam control valves to the common header, with three inlet and one outlet. The model is based on the assumption that the steam is in saturated condi-

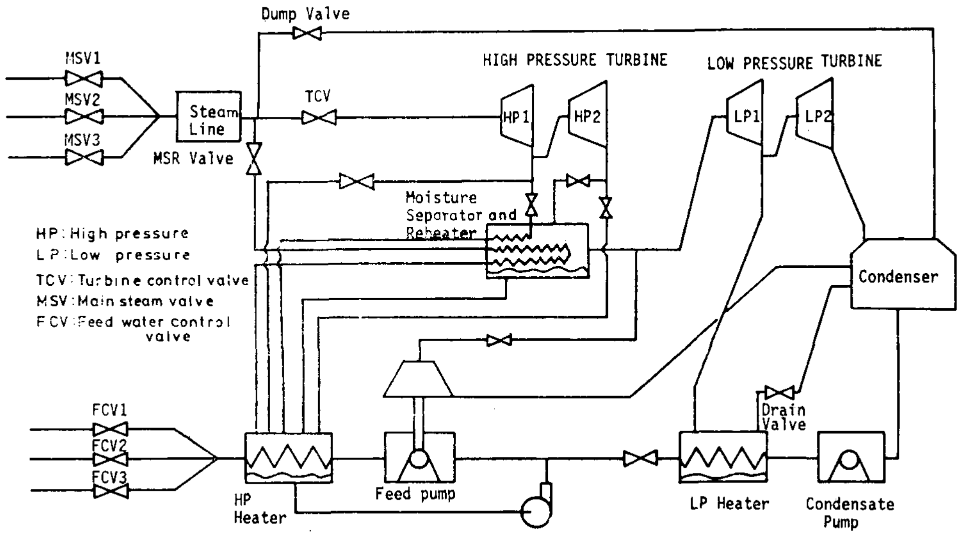

Fig. 3. Schematic Diagram of Secondary Side in MCSIM

tions in the control volume. Continuity equation for the control volume can be given by;

$$\frac{dP_{SL}}{dt}=\frac{\Sigma W_i - W_0}{V\frac{d\rho}{dp}} \quad i=1, 2, 3. \tag{19}$$

Flow rate from steam generators can be calculated by;

$$W_i=K_i\sqrt{\rho_i(P_{SG,i}-P_{SL})} \tag{20}$$

Total steam flow leaving the steam line is divided into three pathes, dump flow, bypass flow to the turbines through the control valves and to the high pressure feedwater preheater through the MSR(Moisture Separator and Reheater). Flow through the impluse chamber is assumed to be sonic under all operating conditions. The flow rate is given by;

$$W_I=\frac{C_s\times P_s}{\sqrt{T_s+460}} \tag{21}$$

The pressure at the exit of the impulse chamber is given by;

$$P_I=\frac{W_I}{C_I}\sqrt{T_I+460} \tag{22}$$

Turbine model consists of high and low pressure turbine. The exit pressure of each turbine is determined by using of above equation. Turbine output power is easily obtained, because the expansion process in each turbine stage is assumed as isentrophic [10]. For the high and low pressure preheater model, a counter flow in tube and shell side is assumed. The flow in the tube side is main feedwater and that in the shell side is the steam expanded from each turbine stage. Each preheater is simulated by dividing into condensate and drain section. Residual heat of the steam transfer to the feed water in condensate section and heat transfer in the drain section is completed by temperature difference between condensate water and feedwater. The exit temperature and exchanged heat in the preheater can be calculated with the two equations which are derived from energy balance equation of each section. The heat transfer rate of the MSR is assumed to be proportional to the generator power. The feedwater flow rate to the steam generators can be obtained by using momentum equation.

3. Results of Application

To test the capability of MCSIM code developed up to now, two relatively short transients, namely complete loss of flow from 102% power operation and uncontrolled RCCA (Rod Control Cluster Assembly) withdrawal from full power operation were analyzed by MCSIM for KNU 5 & 6. Herein, some conservative boundary and initial conditions used in FSAR analyses were considered by input preparation for MCSIM.

3.1. Complete Loss of Flow (CLOF) from 102% Power Operation

This accident is analyzed to verify the RCS

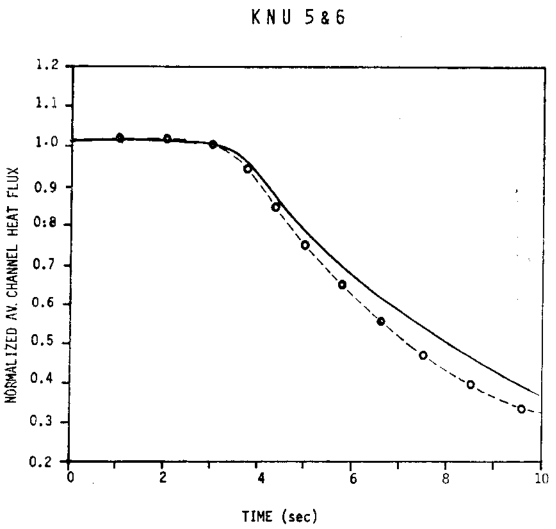

Fig. 4. Complete Loss of Flow from 102% Power
(MCSIM: -o-, FSAR —)

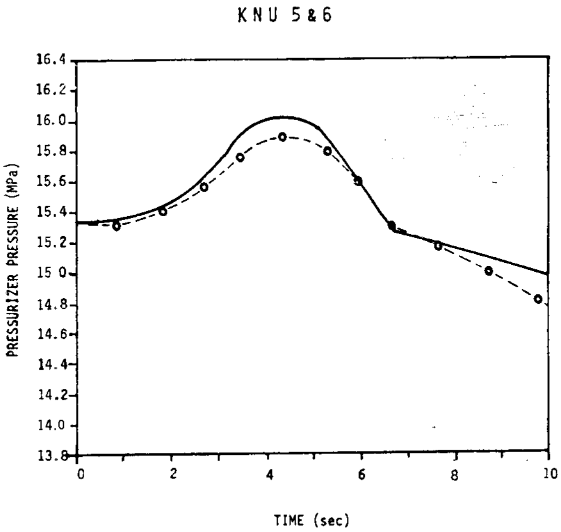

Fig. 5. Complete Loss of Flow from 102% Power
(MCSIM: -o-, FSAR —)

(Reactor Coolant System) model in MCSIM. Because reactor control and protection system has not been modeled in present MCSIM, reactivity change corresponding to the nuclear power change in FSAR was given as boundary condition. The change of average channel heat flux and pressurizer pressure calculated with MCSIM is compared with result of FSAR in Fig. 4 and Fig. 5, respectively. Pump trip occures at 0 sec., and the reactor trip follows 2.5 sec. after it. A satisfactory agreement can be observed between result of MCSIM and FSAR.

3.2. Uncontrolled RCCA Withdrawal from Full Power Operation

To test the reactor core model in MCSIM, this accident was chosen. The model can describe the influence of reactivity change on thermohydraulic properties of RCS. The same variables as in the analysis of complete loss of flow are compared with FSAR result in Fig. 6 and Fig. 7. The variation of these two variables throughout entire computation time has similar trend with that in FSAR.

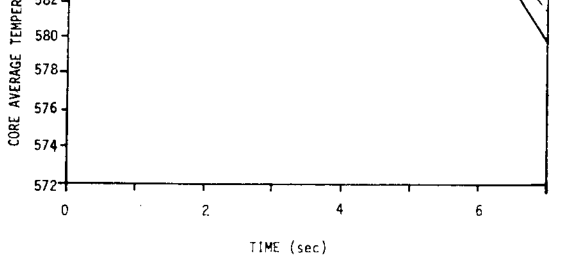

Fig. 6. Uncontrolled RCCA Withdrawal from Full Power Condition
(MCSIM: -o-, FSAR: —)

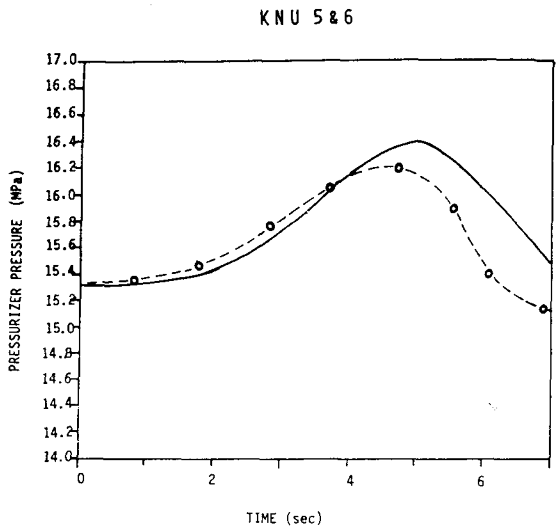

Fig. 7. Uncontrolled RCCA Withdrawal from Full Power Condition
(MCSIM: -o-, FSAR: —)

4. Conclusion

For transient analyses of a PWR power plant, MCSIM code with NSSS model and BOP model has been developed. The application results of the present version of MCSIM for KNU 5 & 6 show reasonable agreement with those in FSAR. Thereby, MCSIM has demonstrated it's capability to simulate plant transients. Verifications and improvements of the each model in MCSIM will be continued by KAERI.

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