Nuclear Engineering and Design 240 (2010) 112–122

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Development of a thermal-hydraulic analysis software for the Chinese advanced pressurized water reactor

Y.W. Wu, G.H. Su*, S.Z. Qiu, C.J. Zhuang

State Key Laboratory of Multiphase Flow in Power Engineering, Department of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an City 710049, China

ARTICLE INFO

Article history:
Received 16 March 2009
Received in revised form 5 October 2009
Accepted 13 October 2009

ABSTRACT

A point reactor neutron kinetics model, a drift-flow U-tube steam generator model, a non-equilibrium three-region pressurizer model and other models were established and a transient analysis code with Visual Fortran 6.5 has been developed to analyze the thermal-hydraulic characteristics of the Chinese advanced pressurized water reactor (AC-600). Visual input, real-time processing and dynamic visualization output were achieved with Microsoft Visual Studio.NET 2003, which greatly facilitate applications in the engineering. The software were applied to analyze the transient thermal-hydraulic characteristics of the loss of feed-water accident, the double loops loss-of-flow accident, the reactivity insertion accident, the sudden increase of feed-water temperature accident and the loss of offsite power accident for the Qinshan nuclear power plant in China. The obtained analysis results are significant to the improvement of design and safety operation of the plant.

1. Introduction

Along with the rapid development of computing and computer technologies, many safety analysis codes for nuclear power system accident or hypothetical accident have been developed. At present, the best large thermal-hydraulic analysis codes for the light water nuclear reactor systems are RELAP series, RETRAN series, TRAC series, etc. Among them many have become indispensable and important tools for the design, evaluation and safety analysis of nuclear power system. These large thermal-hydraulic analysis codes have considered many factors, as well as have complex and detailed mathematical models and comprehensive functions. Therefore these large codes require users to have high professional technology and program technique relatively. In addition, it is very difficult to make modifications and supplements to large codes because they have complex structure, high data storage and transmission techniques. The operation speed of large codes is relatively slow and cannot realize the real-time simulation. The above factors have gravely limited the applications of large codes.

In this study, a thermal-hydraulic analysis software named TTHAsoft 1.0 for the Chinese advanced pressurized water reactor (AC-600) with two loops has been developed. Compared with RELAP5 and other commercial codes, the TTHAsoft 1.0 needs less computational time and can realize the real-time calculation. Unlike other existing real-time simulation tools with dynamic visualization (e.g. plant simulators) which were generally developed on the basis of large amounts of experimental data or theoretical calculation data, the TTHAsoft 1.0 is based on fundamental conservation principles: the mass, momentum and energy conservation equations. In comparison with the complicated preparation work on the input card of RELAP5, the pre-process is much more convenient; and in comparison with the output results of RELAP5, the post-process is much more advanced for its real-time processing and dynamic visualization output. At the mean time, the TTHAsoft 1.0 is highly modularized using Fortran 90 language; therefore it is much easier for the user to add new mathematical models and physical models for further development. For instance, more accurate heat transfer or flow friction correlations could be easily adopted by the TTHAsoft 1.0 by adding the corresponding subroutine program module.

The safety analysis of the Qinshan nuclear power plant transient thermal-hydraulic behaviors have been conducted using this software. All analyses are aiming at verifying whether the plant is safety in operation.

2. Mathematical models

The object of this study is AC-600 which is a single reactor with two loops nuclear power system. The basic field model of AC-600 is based on fundamental conservation principles: the mass, momentum and energy conservation equations. With the assumption of

* Corresponding author. Tel.: +86 29 82663401; fax: +86 29 82663401.
E-mail address: ghsu@mail.xjtu.edu.cn (G.H. Su).

0029-5493/$ – see front matter © 2009 Elsevier B.V. All rights reserved.
doi:10.1016/j.nucengdes.2009.10.020

Nomenclature

A	cross sectional area (m^2)
c	specific heat (kJ/(kg K))
C_i	neutron precursor concentration of group i
DNBR	departure from nucleate boiling ratio
F	surface area (W/(m^2 K))
H	heat transfer coefficient (W/(m^2 K))
h	enthalpy (kJ/kg)
k	feedback coefficient
L	water level (m)
M	mass (kg)
N	kinetic power (MW) or number of the control volume in the primary loop
N–S	Nassi and Shneiderman
p	pressure (MPa)
q	heat flux (W/m^2)
T	temperature (K)
$\bar{T}$	average temperature (K)
t	time (s)
u	velocity (m/s)
v	specific volume (m^3/kg)
V	volume (m^3)
W	mass flow rate (kg/s)
x	quality
z	special coordinate (m)

Greek symbols

α	void fraction
β	total delayed neutron fraction
β_i	delayed neutron fraction of group i
Λ	neutron generation time
λ_i	decay constant of precursor group i
ρ	reactivity or density

Subscripts

afw	auxiliary feed water
b1	ascending bubble
c	coolant
ccd	coolant and clad
cd	clad
ce	interface net vaporization
cs	spray condensate water
dc	downcomer
dg	drift-flow
ex	externally introduced
f	fuel, saturated liquid or frictional
fcd	fuel and clad
fs	fuel surface
fw	feed water
g	saturated vapor
in	inlet
loc	local
m	tube wall
out	outlet
p	primary
pr	pressurizer
re	release valve
rir	riser
s	secondary
sa	safety valve
sd	steam space
sp	spray water
stm	steam
su	surge water
ws	wall condensate water

one-dimensional flow, these equations, including single-phase and two-phase conservative equations, can be easily found in reference (Collier and Thome, 1994). The characteristics of the mathematical models are introduced in detail as follows.

2.1. Core model

The core fission power is calculated through the solution of the point kinetic equations with six groups of delayed neutron. The power distribution in axial is assumed to be known.

$$\frac{dN(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda}N(t) + \sum_{i=1}^{6}\lambda_i C_i(t) \tag{1}$$

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

The reactivity feedbacks caused by the temperature variations of the moderator and the fuel are specially considered.

$$\rho(t) = \rho_0 + \rho_{\mathrm{ex}}(t) + k_{\mathrm{c}}\left(\bar{T}_{\mathrm{c}}(t) - \bar{T}_{\mathrm{c}}(0)\right) + k_{\mathrm{f}}\left(\bar{T}_{\mathrm{f}}(t) - \bar{T}_{\mathrm{f}}(0)\right) \tag{3}$$

In Eq. (3), on the right hand side, the first term (ρ_0) is the initial reactivity; the second term (ρ_{ex}) is the externally introduced reactivity; and the remaining terms are the various feedback contributions of the moderator temperature and the fuel temperature. The parameter T is the mass weighted average temperature of the fuel or the coolant; k is the feedback coefficient.

The single channel model is chosen for the reactor core thermal-hydraulic calculation. The fuel element is cylinder and the axial heat conduction is ignored. Thus, to the fuel element, the heat transfer equation can be defined as

$$\frac{dT_{\mathrm{f}}(t)}{dt} = \frac{-H_{\mathrm{fcd}} \cdot F_{\mathrm{fcd}}(T_{\mathrm{fs}} - T_{\mathrm{cd}}) + q_{\mathrm{f}}}{M_{\mathrm{f}} c_{\mathrm{f}}} \tag{4}$$

To the clad, the heat transfer equation can be defined as

$$\frac{dT_{\mathrm{cd}}(t)}{dt} = \frac{H_{\mathrm{fcd}} \cdot F_{\mathrm{fcd}}(T_{\mathrm{fs}} - T_{\mathrm{cd}}) - H_{\mathrm{ccd}} \cdot F_{\mathrm{ccd}}(T_{\mathrm{cd}} - T_{\mathrm{c}})}{M_{\mathrm{cd}} c_{\mathrm{cd}}} \tag{5}$$

To the coolant, the variation of the temperature can be defined as

$$\frac{dT_{\mathrm{c}}(t)}{dt} = \frac{H_{\mathrm{ccd}} \cdot F_{\mathrm{ccd}}(T_{\mathrm{cd}} - T_{\mathrm{c}}) - W_{\mathrm{c}}(\partial h/\partial z)}{M_{\mathrm{c}} c_{\mathrm{c}}} \tag{6}$$

2.2. Pressurizer model

It is very important to accurately simulate the dynamic characteristics of the pressurizer in order to improve the simulation accuracy of the whole nuclear power system. Many scholars have studied on the mathematical simulation for dynamic characteristics of the pressurizer since the 1960s. In one early model, the steam and water in the pressurizer were thought to be at saturation state with same thermodynamic characteristics and the pressurizer was thermodynamic analyzed as a closed system. The model was very simple and convenient; however, the prediction results were not satisfactory when it was applied in the rapid variation processes. Later, Redfoeld (1968), Baron (1973) and Abdallah et al. (1982), etc. proposed a two-region unbalanced model, namely, the pressurizer was divided into the steam region and water region, and the thermodynamic state was unbalanced. The unbalanced model was more close to the actual processes of the pressurizer than the balance model, but it came to its shortages when the stratification phenomenon appeared after the super-cooled water flowing into the pressurizer while there was a positive system pressure surge. Baggovra and Martin (1983) proposed a three-region model, namely, two liquid phase regions and a vapor phase region which considered vaporization and condensation dynamics phenomena. However, the disadvantage of this model was that the