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Development of a thermal–hydraulic safety analysis code RETAC for AP1000

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A B S T R A C T

In the present study, a thermal–hydraulic safety analysis code for AP1000 named RETAC (REactor Transient Analysis Code) has been developed using FORTRAN 90 language. A point reactor neutron kinetics model with six groups of delayed neutrons was adopted to describe the core thermal power transient. A distributed parameter model with two-phase drift flux model was used in the U-tube steam generator simulation. In the pressurizer simulation, the RETAC code was equipped with three-region non-equilibrium model and multi-region non-equilibrium model respectively. Similar to current large commercial codes such as RELAP5 and RETRAN series, the four-quadrant analogy curves were adopted for the solution of the transient behaviors of the main coolant pumps. In this paper, a new and reasonable model for the passive residual heat removal heat exchanger (PRHR HX) was proposed based on basic equations of mass, momentum and energy. Gear method and Adams predictor-corrector method were adopted alternately for a better solution to ill-condition differential equations corresponding to detail processes.

The PRHR HX inadvertent operation accident and the ADS (automatic depressurization system) inadvertent operation accident were chosen in the transient accident analysis. Furthermore, the simulation results obtained by RETAC were compared with that by Westinghouse-developed LOFTRAN code. The comparison results showed a good agreement and thus proved the accuracy and reliability of the RETAC code. With the adoption of modular programming techniques, the RETAC code can be easily modified and applied to higher power passive safety reactors in the future.

1. Introduction

AP1000 is a two-loop, 3400 MWt Westinghouse-designed PWR which received Final Design Approval by the United States Nuclear Regulatory Commission (NRC) in 2004 and Design Certification by U.S.NRC in 2005 as a Generation III+ reactor (Schulz, 2006). It employs a series of passive safety systems to improve its safety performance. The schematic diagram of AP1000 is shown in Fig. 1. The reactor coolant system consists of two heat transfer circuits, each with one steam generator, two reactor coolant pumps, one hot leg and two cold legs for transporting reactor coolant (Westinghouse, 2004). Meanwhile, the passive safety system is composed of two full-pressure core makeup tanks, two accumulators, a passive residual heat removal heat exchanger (PRHR HX) immersed in the in-containment refueling water storage tank (IRWST) and the automatic depressurization system (ADS) (Wright, 2007).

In support of the AP1000 design, a series of thermal–hydraulic analysis codes were developed by Westinghouse, including LOFTRAN/LOFTTR2 for non-LOCA transients, NOTRUMP for small-break LOCAs, WCOBRA/TRAC for large-break LOCAs and long-term cooling analysis and WGOTHIC for containment systems performance analysis (U.S. NRC, 2004; Westinghouse, 2004). However, all the codes mentioned above are proprietary and the thermal–hydraulic characteristics of AP1000 under steady state and transient accident conditions have not been described in detail in open literature and further studies are still required.

In the present work, a thermal–hydraulic safety analysis code RETAC (REactor Transient Analysis Code) has been developed to evaluate the transient thermal–hydraulic behaviors of AP1000. Up to now, many thermal hydraulic and safety analysis codes have been developed for China Advanced Research Reactor (CARR), Molten Salt Reactor (MSR) and Chinese Advanced Pressurized Water Reactor (AC-600) by Nuclear Thermal–hydraulic Research Laboratory of Xi'an JiaoTong University (XJTU-NTRL) and the RETAC code is a derivative of these codes (Tian et al., 2005, 2007; Zhang et al., 2008, 2009; Wu et al., 2010). For the transient analysis, the behavior of the PRHR HX inadvertent operation accident and the ADS inadvertent operation accident were obtained and discussed.

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Nomenclature

A	cross-sectional area or heat transfer area (m^2)
C	specific heat (kJ/(kg °C))
D	diameter (m)
f	friction coefficient
G	mass velocity (kg/(m^2 s))
g	gravity acceleration (m/s^2)
h	specific enthalpy (kJ/kg)
k	thermal conductivity(W/(m k))
L	length of control volume (m)
M	mass (kg)
m	total number of control volumes
N	kinematic power (MW)
n	total number of control volumes
P	pressure or driving pressure head (MPa)
Q	electric heating power (W)
S	slip ratio
s	specific entropy (kJ/(kg K))
T	temperature (K)
t	time (s)
V	volume (m^3)
W	mass flow rate (kg/s)
ADS	automatic depressurization system
CMT	core makeup tank
IRWST	in-containment refueling water storage tank
NRC	Nuclear Regulatory Commission
N–S	Nassi and Shneiderman
PRHRS	passive residual heat removal system
PRHR HX	passive residual heat removal heat exchanger

Greek symbols

α	heat transfer coefficient (W/(m^2 K))
β	β-ray or surge coefficient in the pressurizer
γ	γ-ray
Δt	temperature difference (K)
ρ	density (kg/m^3)
v	specific volume (m^3/kg)

Subscripts

0	normal working condition
1	the primary side
2	the IRWST side
br	bubble
df	falling liquid droplet
e	equivalent parameter
f	saturated liquid
fg	saturated liquid–vapor difference
g	saturated vapor
i	number of control volume
in	inlet
l	liquid
loci	local resistance
p	constant pressure
pump	pump
rsv	relief valve and safety valve
sat	saturated condition
sc	spray condensation
sp	spray
su	surge water
v	vapor
w	wall

2. Mathematical and physical models

The mathematical and physical models of RETAC for AP1000 were based on fundamental conservation principles, i.e., the mass, momentum and energy conservation equations and will be discussed as follows.

2.1. Core model

The single channel and thermal channel model were chosen for the reactor core thermal–hydraulic calculation. Point reactor neutron kinetics model with six groups of delayed neutrons was adopted for the solution of the core fission power. The reactivity feedback of the moderator temperature, the fuel temperature and the moderator density (in the case of rapid depressurization transient) were considered.

After reactor shutdown, the core power consists of two parts: the core fission power and the core decay power. The core decay power could be calculated from a relatively accurate and conservative formula applicable to low-enrichment uranium reactor proposed by Untermeyer and Weills (1952) as follows:

$$\frac{N_{\beta,\gamma}}{N_0} = 0.1\Big\{\Big[(t+10)^{-0.2} - (t+t_0+10)^{-0.2}\Big] - 0.87\Big[\Big(t+2\times10^7\Big)^{-0.2} - \Big(t+t_0+2\times10^7\Big)^{-0.2}\Big]\Big\} \quad (1)$$

where $N_{\beta,\gamma}$ is the core decay power, N_0 is the core operation power under normal working condition, t is the time after reactor shutdown and t_0 is the reactor operation time under the power of N_0.

2.2. U-tube steam generator model

A distributed parameter model with two-phase drift flux model was adopted in the simulation of U-tube steam generator and the following simplification assumptions were given (Guo, 1994):

(1) One-dimension flow was assumed in steam generator and there was no heat conduction along the axial direction of U-type tubes;
(2) The pressure of steam generator working fluid had the same spatial characteristics in the primary and secondary side respectively and varied with time only;
(3) The heat capacity in steam generator except the U-type tubes was ignored;
(4) There was no heat transfer between the riser and downcomer of steam generator and there was no boiling in the downcomer.

As shown in Fig. 2, the steam generator was divided into several parts (Wu et al., 2010): the primary fluid channel, the feedwater plenum, the downcomer, the U-tube wall, the subcooled and boiling section of the heat transfer zone, the riser and the steam separator, the steam space, the inlet and outlet plenum of the primary fluid.

It can be inferred that the distributed parameter model adopted by RETAC can simulate the dynamic characteristics of steam generator more accurately compared to a lumped parameter model adopted by the safety analysis code LOFTRAN (Westinghouse, 2004). In LOFTRAN code and its updated version LOFTTR2, the secondary side of the steam generator uses a homogenous, saturated mixture for the thermal transient, and a two-region steam generator model in the secondary side was adopted. A similar simplified two-region model is also used in the safety analysis codes (Auh et al., 1987; Han, 2000) developed by Korea Atomic