Progress in Nuclear Energy 110 (2019) 274–288

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

A two-phase model for simulation of MTR type research reactor during protected and unprotected LOFA

S. Talebi*, P. Najafi

Department of Energy Engineering and Physics, Amirkabir University of Technology (Tehran Polytechnic), 424 Hafez Avenue, P.O. Box 15875-4413, Tehran, Iran

ARTICLE INFO

Keywords:
Research reactor
LOFA transient
Two-phase model
10 MW IAEA reactor

ABSTRACT

An optimized, accurate and fast model is investigated to simulate the transient behavior of a research reactor during fast and slow LOFA (loss of flow accident). Simulation is performed in two states when the reactor scram system is initiated at critical core power and when it doesn't work properly. The latter case leads to boiling condition and hence the reactor core will be encountered with two-phase flow condition. A lumped heat conduction model for fuel and clad accompanied with a two-phase flow model for coolant are coupled with point kinetic equations to simulate reactor behavior during the accident. A combination of Runge-Kutta method and up-wind finite difference technique are adopted for the solution to the set of differential equations. The calculation performed for IAEA 10 MW benchmark problem in HEU and LEU fuel. Some important parameters such as the peak temperatures of fuel and clad are obtained for both HEU and LEU conditions during transients. Results of numerical simulations, at the first, agree well with the general physical rules and also verified through comparison with the other computer code results. According to present simulation, under both protected and unprotected LOFA transient, the peak clad surface temperatures remain in safe mode.

1. Introduction

There are many different types of research reactors to produce neutron with specific objectives and most of them use plate type fuels. The analysis of the transient behavior of these kinds of reactors during the accident and abnormal condition are one of the main concerns when designing and operating. This subject has received promptly as well as recent attention because of its relevance in determining the limits imposed by clad melting temperature. On the other hand, almost all the analysis of research reactor transients are performed using complicated and large code systems. Codes such as RETRAC (Bousbia and Hamidouche, 2005) RELAP (RELAP5, 1999) and CATHARE (Geffraye, 2011) have been developed for the analysis of some anticipated transients and accidents concerned about nuclear reactors. PARET code is another example which is designed for using in prediction the course and consequence of not-destructive transients in small reactor core (Obenchain, 1969). However, the use of a large code, usually need to consider input preparation and output processing effort and skill. This problem is more common in research reactor. Where, operators need to analyze a transient several times, to achieve optimum safety level.

It is noteworthy that sometimes code system developed for power reactors have been adapted for research reactor application (for example COBRA (Basile et al., 1999) and RELAP). As was explained at the beginning, the use of such codes is associated with difficulties. Therefore, it is desirable to submit a simple and efficient mathematical model to transient estimations in a research reactor, without a full simulation with a large code.

One of the most important accidents that commonly has been analyzed in the research reactor is LOFA. LOFA is a design basis accident (DBA) that may occur in each reactor. The main reason for LOFA even in normal operation is pump fails.

Heretofore, several efforts were carried out to investigate the transient behavior of MTR research reactor during LOFA accidents. Some of these studies were performed by commercial and accepted codes such as RELAP, PARET and RETRAC. On the other hand, some computational codes with lower complexity and fewer details were developed for the simulation of research reactors during a LOFA (Housiadas, 2002; Kazeminejad. 2008; El-Mordhedy, 2012; Ardahan and Zaferanlouei, 2013). Recently, a number of authors have used CFD (Computational Fluid Dynamic) analysis code to simulate LOFA in research reactors (Salama, 2011; Khater et al, 2015). Generally, in the previous works, especially in computational codes, it is assumed that reactor shutdown system works properly. But there is another group of accident that may be associated with a defect in reactor shutdown system. These accidents are known as ATWS (Anticipated Transient Without Scram). The intent

* Corresponding author.
E-mail address: sa.talebi@aut.ac.ir (S. Talebi).

https://doi.org/10.1016/j.pnucene.2018.10.004
Received 22 April 2017; Received in revised form 1 October 2018; Accepted 5 October 2018

Nomenclature

a	Reactivity feedback coefficient
c	Heat capacity (J/Kg.K)
q_w	Heat flux (W/m^2)
x_{tt}	Martinelli factor
h	Enthalpy
(J/kg)	heat transfer coefficient (W/m^2.K)
C	Precursor density
D	Channel hydraulic diameter
F	Reynolds number factor
G	Mass flux (kg/m^2.s)
P	Power
Pe	Peclet number
Pr	Prandtl number
Re	Reynolds number
T	Temperature (K)
V	Volume
g	Gravity acceleration (m/s^2)
k	Thermal conductivity (W/m.K)
t	Time (s)
v	Coolant velocity
x	Vapor quality
z	Axial direction (m)

Greek symbols

β_i	*ith* group of delayed neutron fraction
λ_i	*ith* group decay constant
Λ	Prompt neutron generation time
α	Void fraction
β	Total delayed neutron fraction
δ	Thickness
μ	Viscosity (Pa.s)
ρ	Reactivity or density

subscripts

$1ph$	Single phase
$2ph$	Two-phase
T	Total
avg	Average in z direction
c	Clad
f	Fuel
fc	Forced convection
fg	Difference between liquid and gas
g	Gas phase
l	Coolant or Liquid phase
m	Mixture
n	Natural
nb	Nucleate boiling
p	Pool
sat	Saturation
x, 0	Steady state

Superscribe

s	Saturation

of the present work is to provide a mathematical model for simulation of LOFA in two situations. Therefore, it is customary to consider the protected and unprotected transients, which refer to reactor scram and failure, respectively. The simulation is limited to research reactors with plate-type fuel elements.

The idea of the developed model lies in the use of a lumped parameters modeling fuel heat transfer (Tong and Weisman, 1970). The model consists of point kinetic model for power calculation and fuel and coolant temperature feedback model (Henry, 1982). The one-dimensional energy equation is considered to the simulation of the coolant temperature in single phase and to find vapor quality in the two-phase region between parallel plates.

Generally, transient two-phase flow problem can be formulated by using a two-fluid model (Ishii, 1975; Todreas and Kazimi, 1990) or a drift-flux model (Ishii, 1977; Todreas and Kazimi, 1990). These models depend on the degree of the dynamic coupling between the phases. In the two-fluid model, each phase is considered separately; hence the model is formulated in terms of two sets of conservation equations governing the balance of mass, momentum, and energy of each phase. Complex mathematics, uncertainties in specifying interfacial interaction terms between the two phases and common numerical instabilities are disadvantages for two phase model. These difficulties associated

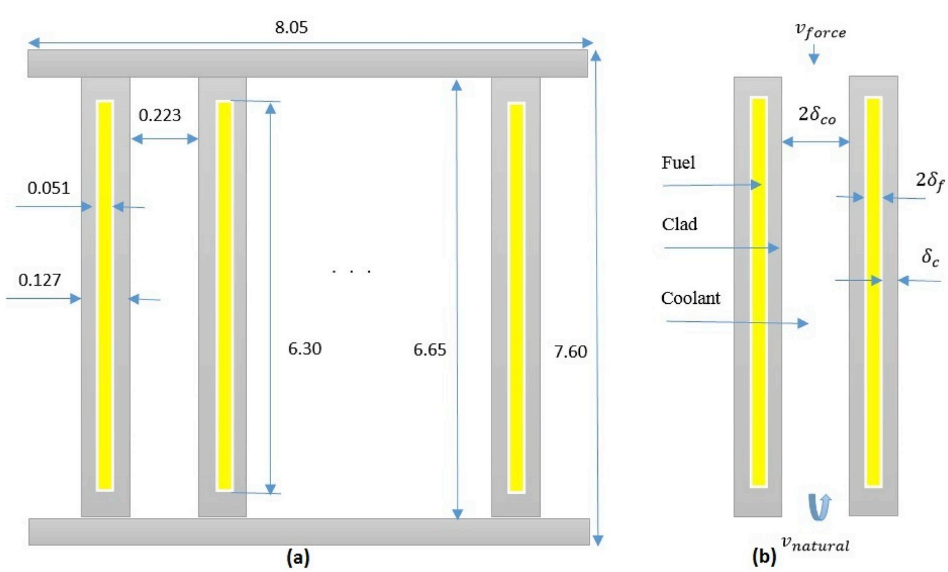

Fig. 1. (a) Unit cell for 10 MW benchmark MTR fuel element (dimensions in Cm). (b) Flow model.