Available online at www.sciencedirect.com

ScienceDirect

Nuclear Energy and Technology 2 (2016) 294–298

www.elsevier.com/locate/nucet

Modeling of natural circulation for the inherent safety analysis of sodium cooled fast reactors

A.S. Bochkarev[a,*], P.N. Alekseev[b], A.S. Korsun[a], V.S. Kharitonov[a]

[a] *National Research Nuclear University "MEPhI", 31 Kashirskoye shosse, Moscow 115409, Russia*
[b] *National Research Centre "Kurchatov Institute", 1 Akademika Kurchatova Sq., Moscow 123098, Russia*

Available online 1 December 2016

Abstract

The paper discusses a set of developed integrated one-dimensional models of thermal-hydraulic processes that contribute to the removal of decay heat in a BN-type reactor. The assumptions and constraints involved in one-dimensional equations of unsteady natural convection in closed circuits have been analyzed. It has been shown that the calculated values of the primary circuit sodium temperature and flow rate in conditions with a loss of heat sink and with a forced circulation of the primary coolant are in a reasonable agreement with the results of a benchmark experiment in the PHENIX reactor. The model makes it possible to assess the effects general thermophysical and geometrical parameters and the selected technology have on the efficiency of passive heat removal by the natural coolant convection in the reactor tank and in the emergency heat removal system's intermediate circuit and by the heat transfer through the reactor vessel. The model is a part of an integrated algorithm used to assess the inherent safety level of advanced fast neutron reactors and is intended primarily to develop, at the early conceptual design stage, the recommendations and requirements with respect to the reactor equipment parameters leading to an increase in the reactor inherent safety. The model will be used to identify the set of quantitative thermal-hydraulic criteria that have an effect on the dynamics of emergency transients leading to a potential loss of integrity by the reactor safety barriers, and to formulate such limits for the defined criteria as would cause, if observed, the requirement for the safety barrier integrity to be met under any combination of the accident initiating events.

Keywords: Fast reactor; BN; PHENIX; Emergency heat removal system; Inherent safety; Decay heat removal; Natural circulation; Pump.

Introduction

Safety is one of the most important requirements to advanced nuclear power plants (NPP) [1]. The safety of NPPs is achieved through a number of organizational, technical and legal measures. When it comes to technical measures, advanced NPP designs are specifically required to possess a high level of inherent safety [2,3]. The development of inherent safety in a nuclear reactor has the purpose of giving the reactor such safety status as would prevent any combination of internal emergency-initiating events from leading to a loss of integrity of the plant's safety barriers. Since, as stated in [2], "with respect to beyond design basis accidents, other than avoidable through the reactor inherent safety properties and design philosophy, independent of how probable they are, organizational measures shall be developed for managing such accidents", the need arises for such unlikely accidents and their consequences to be computationally simulated. It is also necessary to design and introduce additional components requiring computational justification, which leads to more time and material resources consumed. Importantly, a higher level of inherent safety allows many beyond design basis accidents and their combinations to be avoided [4–6], which, in turn, is expected to eliminate the need for extra organizational, technical and legal measures to be taken with respect to such

* Corresponding author.
E-mail addresses: alseboch@mail.ru (A.S. Bochkarev), Alekseev_pn@nrcki.ru (P.N. Alekseev), askorsun@bk.ru (A.S. Korsun), VSKharitonov@mephi.ru (V.S. Kharitonov).
Peer-review under responsibility of National Research Nuclear University MEPhI (Moscow Engineering Physics Institute).
Russian text published: Izvestiya vuzov. Yadernaya Energetika (ISSN 0204-3327), 2016, n.3, pp. 129–138.

http://dx.doi.org/10.1016/j.nucet.2016.11.011

situations not only at the plant operation stage but also in the process of design.

This can be effectively achieved through the use of a methodology that defines the quantitative characteristic of the inherent safety level for the reactor under investigation, taking into account only the reactor core physical properties and the primary circuit layout. An integrated algorithm based on such methodology is employed in the DYANA code [7,8]. DYANA can be used to analyze both steady-state asymptotic conditions of the reactor and transients initiated by perturbations of the reactor's key operating parameters. DYANA allows complex one-dimensional neutronic and thermohydraulic simulations of a liquid metal cooled fast neutron reactor limited to the primary circuit. Analogs of such methodology are described in [9–15]. Such approach does not however make it possible to simulate in an adequate way the operation of passive heat removal systems as applied to safety analyses of advanced liquid metal cooled fast reactors. This dictates the requirement for new models to be additionally developed to calculate the level of natural circulation in emergency heat removal systems.

An improved thermohydraulic one-dimensional model of the primary circuit and a model of the emergency heat removal system (EHRS) circuits of a BN-type reactor have been developed for the DYANA code. The model includes the capability for heat transfer to the cooling air through the reactor vessel, and a thermohydraulic model of the pump has been developed.

Problem statement for nonstationary natural circulation in the circuit

According to [16], the flow of a liquid medium with a temperature-dependent density in an arbitrary channel with a variable cross-section area, $S(x)$, is described by the equations:

$$\frac{\partial \rho}{\partial \tau} + \frac{\partial \rho u_i}{\partial x_i} = 0, \tag{1}$$

$$\frac{\partial \rho u_i}{\partial \tau} + \frac{\partial \rho u_i u_j}{\partial x_j} = \rho g - \frac{\partial p}{\partial x_i} + \frac{\partial S_{ji}}{\partial x_j}, \tag{2}$$

where S_{ji} is the stress tensor taking into account the momentum transport due to turbulent and molecular viscosities; p is pressure; ρ is the medium density; u is the flow velocity; and τ is time.

Let us consider a one-dimensional approximation with respect to cross-sectional average velocity, pressure and density values, under a no-slip boundary condition. We neglect the momentum transport by diffusion along the channel, as compared to convective transport, and the medium compressibility effects ($\partial\rho/\partial\tau = 0$). In the event of a closed circuit along which the liquid temperature is not constant, we shall derive from systems (1) and (2), if the pressure in the circuit is continuously variable, an equation of motion [17]:

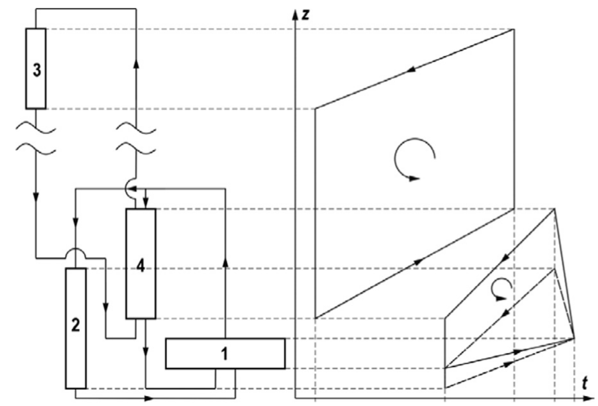

Fig. 1. Schematic of the primary circuit and the EHRS intermediate circuit and their temperature diagrams. 1 – reactor core; 2 – intermediate heat exchanger (IHX); 3 – air-cooled heat exchanger; 4 – autonomous heat exchanger (AHX).

$$\Delta p_{nc} + \sum_i \Delta p_i = \Delta p_{res} + l^{*-1}\frac{dG}{d\tau} = \Delta p_{res} + \oint \frac{dl}{S(l)}\frac{dG}{d\tau} \tag{3}$$

where G is the mass flow rate. As shown in (3), the driving head is formed by the pump head Δp_i and the natural circulation head Δp_{nc}. The total driving head accelerates the liquid flow and helps overcome the circuit's total resistance. The parameter l^{*-1} in (3) characterizes the inertia degree of the coolant flow rate variation in the circuit.

Natural circulation of sodium in the reactor primary circuit

Normally, BN-type reactor designs allow decay heat removal by natural coolant circulation in the primary circuit and in the emergency heat removal systems (EHRS and through the reactor vessel) [18–20].

Fig. 1 shows the schematic of the simulated circuits and the fluid temperature distribution. As the primary coolant flow rate decreases to below a certain value, the EHRS loops are engaged and a natural coolant circulation is established in the primary circuit and in the EHRS circuits. The coolant with the temperature $t_{\text{in}} = (T_{\text{ahx}} g_{\text{ahx}} + t_{\text{ihx}} G_{\text{ihx}})/(G_{\text{ahx}} + G_{\text{ihx}})$ and the flow rate $G = G_{\text{ahx}} + G_{\text{ihx}}$ is fed into the core. The natural circulation in the investigated closed circuits is described by Eq. (3). And the total resistance is $\Delta p_{\text{res}} = \Delta p_{\text{res1}} + \Delta p_{\text{res2}} + \Delta p^{\text{rc}}{}_{\text{res}}$, where $\Delta p_{\text{res1}} \sim G_{\text{ahx}}{}^2$; $\Delta p_{\text{res 2}} \sim G_{\text{ihx}}{}^2$; $\Delta p^{\text{rc}}{}_{\text{res}}, \sim G^2$. The total head of the buoyancy forces $\Delta p_{\text{nc}} = \Delta p_{\text{nc1}} + \Delta p_{\text{nc2}}$, where the indices 1 and 2 denote the IHX and AHX circuits respectively.

The reactor tank coolant temperature t_t, with all decay heat removal channels taken into account, is determined by the following heat balance equations, with regard for the coolant mixing in the volume above the core:

$$\rho c_p V\, dt_{\text{t}}/d\tau = G_{\text{rc}} c_p \mu_{\text{ic}} \left(t^{\text{out}}{}_{\text{rc}} - t^{\text{in}}{}_{\text{ahx}}\right) - Q^*{}_{\text{ehrs}} - Q_{\text{ves}} \tag{4}$$