A SIMPLE APPROACH FOR PRE-LOCA ANALYSIS OF MTR TYPE RESEARCH REACTORS

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Abstract

In this study, it is intended to analyze early phases of a protected loss of coolant accident (LOCA) for TR-2 Research Reactor at Istanbul, and to show the applicability of the presented model to the other similar types of research reactors. The transient situation since the time when coolant is beginning to be lost throughout one or more of the main coolant pipes which were supposed to be broken guillotine-like to the time when the core is totally uncovered is investigated.

The modeling of the problem was separated into two phases; in the first phase when the water level of the pool is being decreased in a pre-estimated time-dependent way calculated by using Modified Bernoulli Equation, the conservation equations are solved by a usual implicit finite difference algorithm. The later phase, when water level reaches to the top level of fuel plates and begins to decrease until the bottom of the core, needs some modifications to the approach used for the first phase. Because, the coolant channels among fuel plates are filled with air when the level goes below, and the fuel plates are being cooled by air above the water level. This complexity is resolved using a moving boundary approach in the numerical solution. A Lagrange type interpolation approximation for the derivatives along with interface conditions in the neighborhood of the air-water interface was imported to the numerical algorithm. The analysis is performed for a nominal channel.

1. Introduction

As a result of the work carried out worldwide aiming at the conversion of research reactors from HEU to LEU, remarkable effort has been spent on re-evaluation of the safety of the current research reactors. Most of the reports concerning safety assessments were published by IAEA in a special edition of 5 volumes [1]. An important part of safety assessment of research reactors is the safety calculations requiring the simulation of events belonging to the two wide categories of accidents; reactivity insertion accidents and loss of flow accidents.

It has been customary to consider two transient cases in the analysis of reactivity accidents; protected transients in which the reactor was not shut down by the scram signal and unprotected or self-limited transients in which the reactor was shut down by the scram signal. Unprotected reactivity transients have been matter of interest in the past [2] and recently [3,4] since they are determinant of the reactivity insertion limits for clad melting temperature. However, simulation of unprotected loss of flow transients for pool type research reactors is a rather new research interest [5,6].

The most important accident among the inventory of research reactors is the loss of coolant accident (LOCA). The former studies on LOCA were direct accident experiments during the years when first research reactors had been built. Spert experiments are of these kind of studies [7]. In 1980s, within the framework of IAEA supported reduced enrichment project for test and research reactors, a computer code (3D-AIRLOCA) was developed for MTR type reactors' LOCA analysis [8]. As another IAEA supported research project, 3D-AIRLOCA was adapted to THEAP-I computer code in Democritus Nuclear Research Centre, which could only be used in LOCA analysis of MTR type research reactors [9]. THEAP-I code was used for Turkish TR-2 Reactor and given results in conformity with the experiments [10].

Most of the work concerning LOCA of research reactors in the literature has been done for the latest phase of the accident where the core was totally uncovered and being cooled by natural circulation of air. Whereas, partial LOCA analysis could be especially important for high flux and power MTR type reactors when reactor core is partially covered with water [11]. There is a gap in the literature for the analysis of the phase until the core is totally uncovered. For our purpose, Turkish TR-2 Reactor is taken as an illustrative case. It is our aim to investigate the transient situation since the time when coolant is beginning to be lost throughout one or more of the main coolant pipes to the time when the core is totally uncovered.

2. Modeling

2.1. Loss of pool water

TR-2 Reactor is a 5 MW MTR type research reactor which is inside a partitioned pool along with dismantled TR-1 Reactor. Two primary cooling pipes of the reactors and one common diffuser pipe back to reactor pool penetrate pool wall. One or more of these three pipes is supposed to be broken guillotine-like at the outside of the pool wall (as a result of expected earthquake in Istanbul or a terrorist bombing, etc.). For the pipe under consideration, Extended Bernoulli Equation could be written throughout the streamline, which is from pool surface to the point where the pipe is broken outside the pool as

$$\int_1^4 \frac{\partial v}{\partial t} ds + \frac{P_4}{\rho_4} + \frac{v_4^2}{2} + gz_4 = \frac{P_1}{\rho_1} + \frac{v_1^2}{2} + gz_1 - \Delta P_t \tag{1}$$

where subscripts 1 and 4 denote respectively pool surface and break point, *t* is time, *v* is coolant velocity, *P* is pressure, ρ is density, *g* is gravity acceleration, z is altitude from pool base. The last term on the right hand side of the above equation is the total pressure drop along the streamline consisting of core pressure drop and pipe pressure drop

$$\Delta P_t = \Delta P_c + \Delta P_b \tag{2}$$

where each value is determined experimentally as a function of flow-rate [12]. Noticing

$P_4 = P_1 = P_{atm}$, $\rho_1 \cong \rho_4, v_1 \cong 0$ (reactor pool surface velocity is too low),

$g(z_1 - z_4) = h$

first term of the left hand side of equation (1) could be arranged as

$$\int_1^4 \frac{\partial v}{\partial t} ds = \int_1^2 \frac{\partial v}{\partial t} ds + \int_2^3 \frac{\partial v}{\partial t} ds + \int_3^4 \frac{\partial v}{\partial t} ds \cong \int_2^3 \frac{\partial v}{\partial t} ds + \int_3^4 \frac{\partial v}{\partial t} ds = L_c \frac{dv_c}{dt} + L \frac{dv_4}{dt} \tag{3}$$

where subscripts 2 and 3 denote respectively core inlet , core outlet, L_c is the height of the core, v_c is the mean water velocity in the core, L is the total length of the broken pipe, v_4 is the mean water velocity in the pipe. Inserting (3) into (1) we get

$$L_c \frac{dv_c}{dt} + L \frac{dv_4}{dt} + \frac{v_4^2}{2} = gh - \Delta P_t \tag{4}$$

Continuity of mass could be written as the equality of the flowrates along the core and the pipe as following

$$v_c A_c = v_4 A_4 \Rightarrow v_c = \frac{v_4 A_4}{A_c} \tag{5}$$

where A_c is the total flow area of the core and A_4 is the pipe cross sectional area.

Equation (4) could be arranged for v_4 using (5)

$$(L_c . \frac{A_4}{A_c} + L) \frac{dv_4}{dt} + \frac{v_4^2}{2} = gh - \Delta P_t \tag{6}$$

In order to calculate mean water velocity in the core (v_c) and height of water level(*h*), non-linear equation(6) was solved together with equations (2) and (5) using below discretization in time

$$(L_c.\frac{A_4}{A_c}+L)\frac{v_4(t+\Delta t)-v(t)}{\Delta t}+\frac{v_4^2(t)}{2}=g.h(t)-\Delta P(t) \qquad (7)$$

with the initial condition *h(0)=initial level of the pool* and using an iterative method. As an example of results of the above calculations, change of the pool level and the mean water velocity in the core with time are given in Figure 1 and Figure 2 respectively.

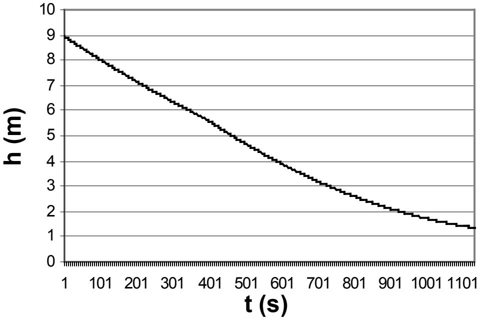

Figure 1. Variation of pool level with time for the case of break of the difussor pipe

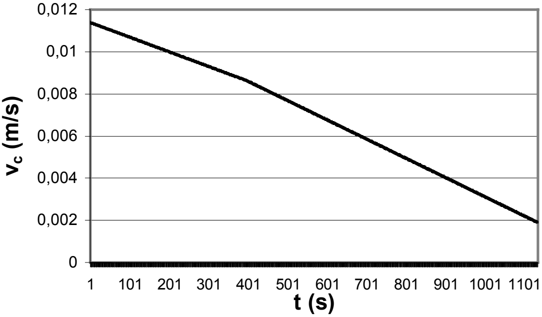

Figure 2. Variation of mean core water velocity with time for the case of break of the difussor pipe

2.2. Thermal-Hydraulics Model of Coolant Channels

Consider a typical pool-type research reactor with MTR-type fuel elements of rectangular geometry, cooled and moderated with light water. A one-dimensional core is considered, consisting of a coolant channel of width l_w and a fuel plate of width l_p. Incompressible slug flow of velocity v is assumed to take place on the plate surface through a heat transfer coefficient h.

The equations for local coolant temperatures T_b and fuel surface temperature T_w [13] are

$$q''(z,t) = \rho_p l_p C_p \frac{\partial T_w}{\partial t} + 2h(T_w(z,t) - T_b(z,t)) \qquad (8)$$

$$2h(T_w(z,t) - T_b(z,t)) = \rho \frac{W}{W'} l_w C_w \frac{\partial T_b}{\partial t} + \rho v \frac{W}{W'} l_w C_w \frac{\partial T_b}{\partial z} \qquad (9)$$

Where t is time, z is axial coordinate, $q''(z,t)$ is the local heat flux, ρ is density, c is specific heat, and the subscripts b and f denote respectively the coolant and the fuel.

2.3. Numerical Solution

Steady state calculations are performed for the above equations by taking time-derivative terms as zero and using a standard finite difference scheme. For this purpose, axial coolant channel is divided into 60 meshes.

First Phase

Postulated accident begins with the reactor's shut down by scram signal when one or more pipes are broken guillotine-like. In the first phase of the accident when the pool level is being decreased in a pre-estimated time-dependent way as calculated before, equations a usual implicit finite difference algorithm solves (8) and (9).

Second Phase

In the later phase, when water level reaches to the top level of fuel plates and begins to decrease until the bottom of the core, the coolant channels among fuel plates are filled with air and the fuel plates are being cooled by air above the water level (Figure 3). This complexity is resolved using a moving boundary approach in the numerical solution. A Lagrange type interpolation approximation for the derivatives along with interface conditions in the neighborhood of the air-water interface was imported to the numerical algorithm.

Using Lagrange Interpolation Formula, we get the spatial derivatives for airside ($z < s(t)$) at time $j\Delta t$

$$\frac{\partial T_a}{\partial z} = \frac{1}{\Delta z}\left(-\frac{p}{p+1}T_{i-1,j} - \frac{1-p}{p}T_{i,j} + \frac{1}{p(p+1)}T_B\right), \quad z = i\Delta z \tag{10}$$

$$\frac{\partial T_a}{\partial z} = \frac{1}{\Delta z}\left(\frac{p}{p+1}T_{i-1,j} - \frac{p+1}{p}T_{i,j} + \frac{(2p+1)}{p(p+1)}T_B\right), \quad z = s(t) \tag{11}$$

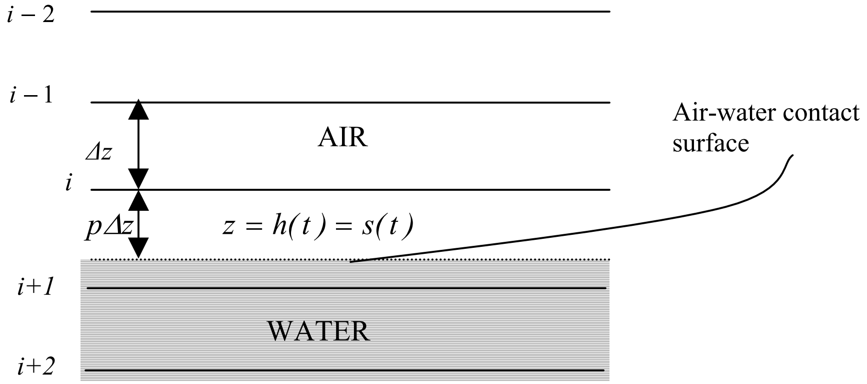

Figure 3. Discretization when water level falls between meshes *i* and *i+1*

Similarly for the water side $z > s(t)$

$$\frac{\partial T_b}{\partial z} = \frac{1}{\Delta z}\left(\frac{p}{p-1}T_{i+1,j} + \frac{1-p}{2-p}T_{i+2,j} - \frac{1}{(1-p)(2-p)}T_B\right), \quad z = (i+1)\Delta z \tag{12}$$

$$\frac{\partial T_b}{\partial z} = \frac{1}{\Delta z}\left(\frac{2p-3}{(1-p)(2-p)}T_B + \frac{2-p}{1-p}T_{i+1,j} - \frac{(1-p)}{(2-p)}T_{i+2,j}\right), \quad z = s(t) \tag{13}$$

For the meshes, which are not in the neighborhood of the interface a usual implicit finite difference scheme is used.

3. Results and conclusion

As an illustrative case of the results, the diffuser pipe is assumed broken guillotine-like at the exit of the pool wall. Calculation is based on taking initial pool water temperature and ambient air temperature as 35 °C and 30 °C respectively. Temperature distributions of coolant water and fuel surface temperature when the pool level is just on the top of the core which is the latest state of first phase are given in Figure 4. Figure 5 shows the temperature distributions of air and fuel surface as the latest state of the phase 2 when the water level is just below the core.

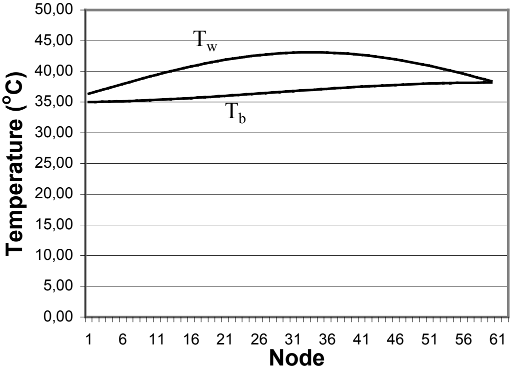

Figure 4. Temperature distributions of coolant water and the fuel surface when the pool level is just on the top the core

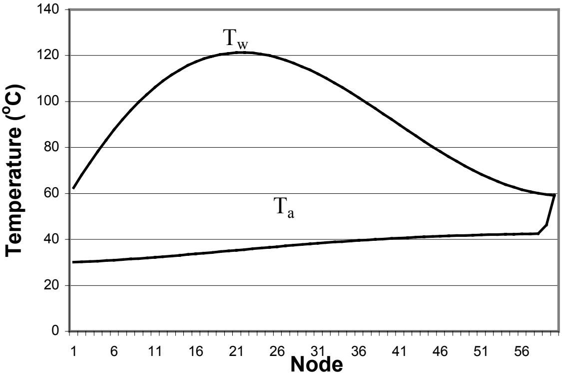

Figure 5. Temperature distributions of air and fuel surface when the water level is just below the core.

It could be observed from the Figures that a considerable increase in the fuel temperature as coolant water is replaced with the air when pool level falls. The considered model lets someone start with obtained data as an initial condition to analyze further stages of LOCA when the core is totally uncovered and the residual heat is removed via natural circulation of air.

Comparison of the predicted temperatures with experimental results is desired to evaluate the applicability of the model for analyzing the early phase of LOCA for MTR type research reactors.

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