A SIMPLIFIED MODEL FOR THE ANALYSIS OF BLOWDOWN PHASE OF LOSS OF COOLANT ACCIDENTS IN PRESSURIZED WATER REACTORS

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ABSTRACT

The physical and thermal phenomena governing the course of large break loss of coolant accidient in a powr reactor are modeled and analysed according to the Nuclear Requlatory Commission (NRC) recommendations. The loop under investigation is divided into massive (fixes dimension) control volume with the addition of source flow and energy terms to the conservation equations of mass and energy respectively to avoid the treatment of connected components such as pressurizer and accumulators. The new contribution of the work is that is offers a simplified computer code for analysis of the blowdown condition of LOCA; which is sometimes useful more than the complicated and sophisticated codes in cases where rapid and survey results are required. The results are satisfactory compared with the reported results.

INTRODUCTION

The analysis done, includes the primary loop thermohydraulics, hydrodynamics and a complete physical-thermal analysis of the reactor core. With the use of the nodel flowpath concept [1,2], the primary loop is divided into fixed dimensions control volumes.

Energy and mass conservation equations are solved in the control volumes using an explicit difference technique. The momentum equation governing the system pressure changes and the corresponding mass flow rates are solved in the junctions, with the aid of a suitable set of constitutive equations and reliable Heat Transfer Correlations routine [HTC] and Critical Heat Flux routine (CHF) [3]. The break flow is evaluated using Moody Model [4]. The American National Standard (ANS) for decay heat [5] is used to evaluate the heat generation after reactor shutdown.

HYDRODYNAMICS AND FIELD EQUATIONS

The equations of conservation of mass and energy are solved in the nodes (control volumes) and the one dimensional momentum equation is solved in the flowpaths (Junctions). By solving the mass, energy, and momentum equations in this manner, one value of the total mass and energy is assigned for each node and an average flow is assigned for each flowpath. Fig. (1) illustrates the schematic representation of the Node-flowpath concept applied to the primary loop under LOCA conditions.

CONSTITUTIVE EQUATIONS

The definition of the fluid model in the present work as a homogeneous (one fluid) equilibrium (one temperature) model implies the discussion of the wall shear model and the heat transfer correlations (wall-to-fluid). [6]

CRITICAL FLOW MODEL

The coolant escape from the break is initially single phase till reaching saturation. The flow is then a two-phase flow. The flow in both cases is independent of the down stream pressure. The following are the main assumptions used in developing the model:

1. The break is double ended accuring in the cold leg at a fixed distance from the core inlet.
2. The rupture occurs very rapidly (in zero time),
3. The reactor is being scramed instantaneously with the accident initiation,
4. The ECCS is not actuated,
5. The fuel is U02 with Zr-IV cladding with a gap in between. The fuel rod is lumped radially into two regions (U02 and Zr IV).

6. The coolant fluid is homogeneous and thermodynamic equil^Xxrium exists between them.
7. The steam generator is adiabatic.

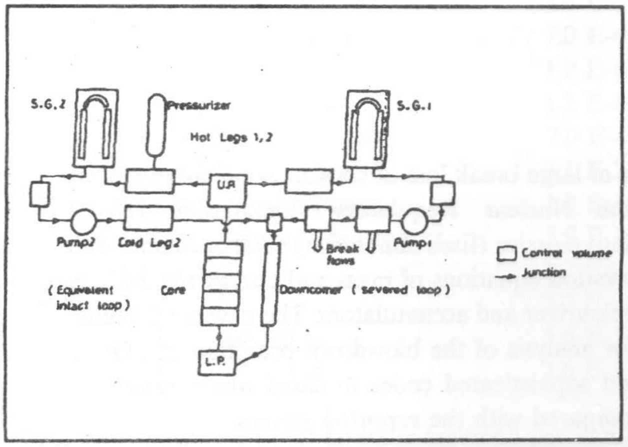

Figure 1. Nodaization seheme of the primary loops under LOCA condition. (LBLOCA in cold leg).

HEAT TRANSFER CORRELATION (HTC) SELECTION LOGIC SCHEME

During a postulated LOCA, an extremely wide range of flow conditions is encountered which make it difficult to select the suitable HTC. The heat transfer coefficient slection logic scheme is given in Figure (2), and it is based on two factors, local channel conditions and fluid conditions. The appropriate region of the boiling curve is determined and its associated proper correlation.

HEAT GENERATION MODEL

All heat sources during LOCA which are recommended by the USNRC are considered. These are decay heat of fission products and actinides and the secondary heat source due to metal-steam reaction.

The 1979 ANS stanadrd for decay heat is used through the a[lication of cathcart-model [5].

HEAT CONDUCTION AND FUEL ROD MODELING

The following assumptions are carried out while modeling the fuel rod.

1. No axial heat flow, (infinite thermal resistance in the axial direction).
2. At each axial node, the temperature profile is flat in each region.
3. The gap is represented by an additional thermal resistance to the fuel and no thermal capacitance, i.e. no heat storage in the gap.
4. The cladding is a thermal capacitance only.
5. Sources of heat are represented as explicit functions of time.

The fuel channel nodelization scheme and its associated temperature profile is shown in Figure (3).

Accordingly, the heat balance equation for the fuel (UO2) region can be written for node K as follows.

$$q_n^k(t).\Delta L = C_1^k \frac{dT_1^k}{dt} + \frac{T_1^k - T_2^k}{R_1^k} \tag{1}$$

where:

$q_n^k(t)$	$\equiv$	decay heat power per unit length,
C_1^k	$\equiv$	thermal capacitance of U02 given by
C_1^k	$\equiv$	$\pi\ r^2_1 \Delta L \rho_1 C_{p1}$
r_1	$\equiv$	pellet radius,
ΔL	$\equiv$	axial mesh length,
ρ_1	$\equiv$	U02 density
C_{p1}	$\equiv$	U02 specific heat,
T_1	$\equiv$	U02 temperature
T_2^k	$\equiv$	clading temperature,
R_1^k	$\equiv$	thermal resistance of U02 and gap, given by

$$R_1^k = \frac{1}{4\pi\ K_1 \Delta L} + \frac{1}{2\pi\ r_1 h_g \Delta L}$$

R_1	$\equiv$	U02 thermal conductivity, and
h_g	$\equiv$	gap conductance.

and the heat balance for the cladding region is

$$q^k_{MW} \Delta L + \frac{T_1^k - T_2^k}{R_1^k} = c_2^k \frac{dT_2^k}{dt} + \frac{T_2^k - T_3^k}{R_2^k} \tag{2}$$

where,

q^k_{MW}	$\equiv$	steam-cladding reaction heat power per unit length
T_3^k	$\equiv$	coolant temperature,
c_2^k	$\equiv$	cladding thermal capacitance, given by
c_2^k	$\equiv$	$2\pi\ r_2 \Delta L \rho_2 c_{p2}$ and
R_2^k	$\equiv$	coolant film thermal resistance at cladding surface, given by

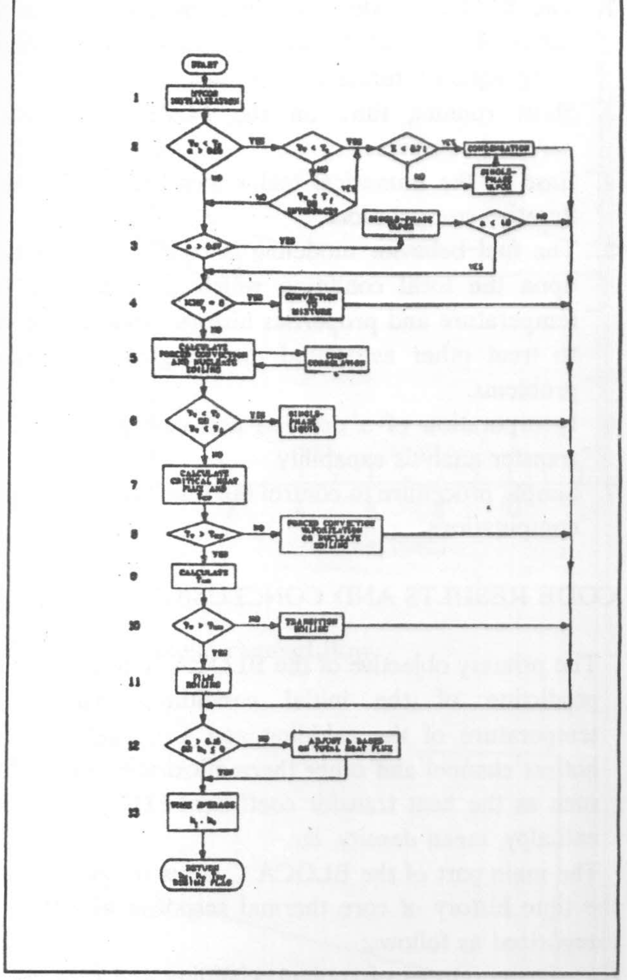

Figure 2. HTC corrrelation selection logic.

$$R_2^k = \frac{1}{2\pi\, r_2 h \Delta L}$$

r_2 ≡ outer radius of cladding

r ≡ clading thickness, and

h ≡ the heat transfer coefficient

The heat balance of coolant is given by

$$\frac{T_2^k - T_3^k}{R_2^k} = 2\,\pi\, r_2\, \Delta L.\ HF \tag{3}$$

where HF is the heat flux

The lumped finite difference method is utilized and the problem is treated in a quasi-static one dimensional fashion. The time derivative of temperature will be approximated as follows:

$$\frac{dT}{dt} = \frac{T(t)-T(t-\Delta t)}{\Delta t} \tag{4}$$

where Δt is the time step.

Introducing the finite difference approximation into the balance equations (1) and (2), and solving for $T_1(t)$ and $T_2(t)$ yields

$$\begin{aligned} a_1 T_1^k(t) + b_1 T_2^k(t) &= C_{11} \\ a_2 T_1^k(t) + b_2 T_2^k(t) &= C_{22} \end{aligned} \tag{5}$$

Equation (5) are two equations in T_1 and T_2 in which

$$a_1 = \frac{C_1^k}{\Delta t} + \frac{1}{R_1 k}$$

$$a_2 = \frac{1}{R_1^k}$$

$$b_1 = \frac{1}{R_1^k}$$

$$b_2 = -\left(\frac{1}{R_1^k} + \frac{C_2}{\Delta t} + \frac{1}{R_2^k}\right)$$

$$C_{11} = q_n^k(t)\Delta L + \frac{C_1}{\Delta t}\ T_1^k\,(t-\Delta t)$$

$$C_{22} = q_{MW}^k(t)\Delta L + \frac{C_2}{\Delta t}\ T_2^k\,(t-\Delta t) + \frac{1}{R_2^k}\,T_3^k$$

INPUT DATA AND CASE STUDY

The runs of BLOCA code for typical 1000 MWe; 4 Loops KWU pressurized Water Reactor [6] reveals satisfactory results along with the available results from experimental facilities and best estimate codes.

CALCULATION PROCEDURE

The calculation procedure can be summarized as:

1. Read the input data,
2. Calculate the initial conditions in the core,
   For the following time steps and utilizing the initial and boundary conditions, the main loop parameters are updated as in the subsequent steps
3. Evaluate the derivatives of total mass , dM/dt, total energy, dU/dt, and the flow, dW/dt.

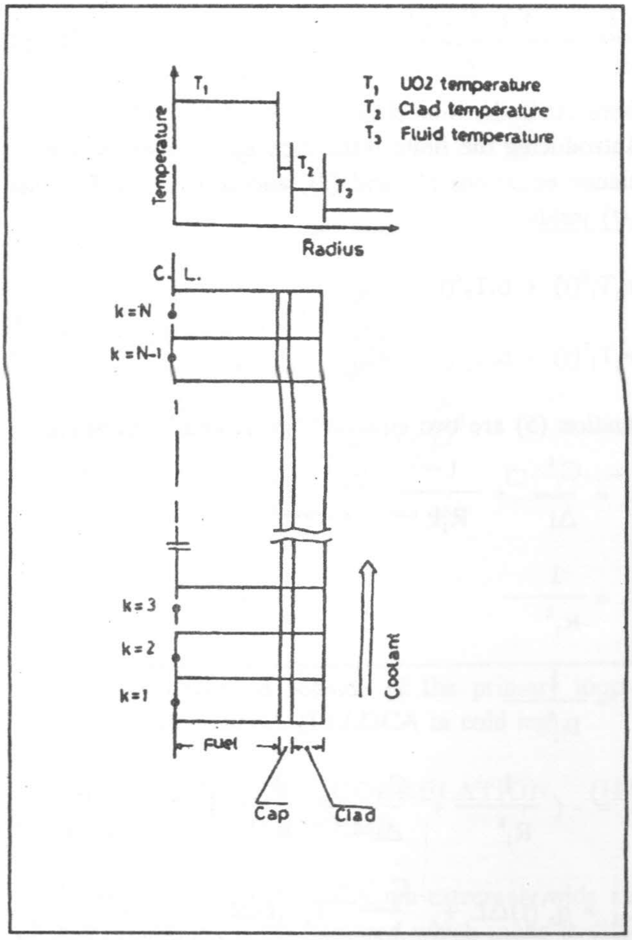

Figure 3. Fuel channel nodalization scheme & the associated temperature profile.

4. Integrate to update the values of total mass, energy, and flow.
5. Compute pressure and local core conditions, T_F, HTC, CHF and DNER.
6. Calculate the reactor power, (Nuclear and chemical energy)
7. Calculate the core temperature.
8. Calculate the break flow.

BLOCA CODE CHARACTERISTICS

The BLOCA code has many good features which makes it capable to treat so many different problems and is planned to be modified to achieve the level of common standard codes. Among the features of BLOCA Code are the following:

1. The ability to analyze the entire accident phenomena including the evaluation of intial conditions,
2. The BLOCA Code is modular by function, and the major phenomeno logical calculations are performed using separate functional routines,
3. Short running time on the PDP11 machine and personal computers
4. Ease of the numerical techniques used and reduced iterative computations,
5. The fuel behavior modeling is completely dependent upon the local condition which gives accurate fuel temperature and properties histories that can be used to treat other associated metellurgical and physical problems.
6. Incorporation of a detailed and comprehensive heat transfer analysis capability.
7. Simple procedure to control the time increment during computations.

CODE RESULTS AND CONCLUSIONS

The primary objective of the BLOCA code is simply the prediction of the initial conditions; namely, the temperature of the caldding and U02 fuel along the hottest channel and other thermalhydraulic information such as the heat transfer coefficient HTC, the coolant enthalpy, mean density, etc.

The main part of the BLOCA Code is the prediction of the time history of core thermal response which can be categorized as follows:

(i) Temperatures of coolant, cladding and fuel and the associated heat fluxes.
(ii) Thermohydraulic response including core flow, HTCs, core quality and void fraction, critical flux, slip ratios and the DNBR.
(iii) Physical response of the core including the determination of the core thermal power, core material properties, steam cladding-reaction parameters, etc.

The average channel flow is shwon in Figure (4). Upon the accident initiation the flow decreases sharply and reaches negative values during the first second and oscillates during the course of the accident according to the pressure differential between the upper & lower core plena. The system depressurization is shown in Figure (5). The time history of pressure predictated by BLOCA code takes on the same overall shape as predicted by standard codes like RELAP [7] and TRAC. [8].

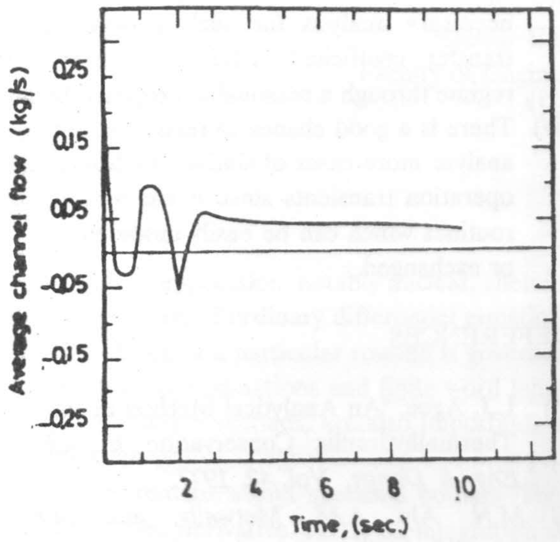

Figure 4. Average channel flow.

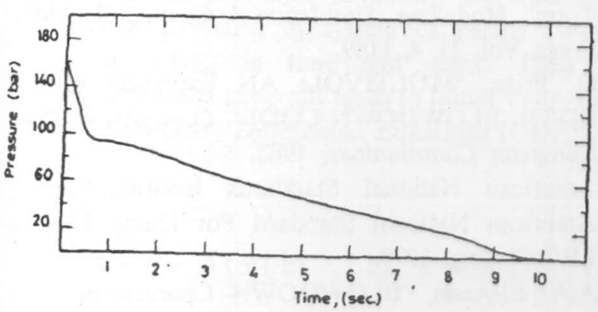

Figure 5. The system pressure at core inlet.

The steam quality and void fraction in the core are shown in Figure (6). The variation of quality and fraction in the core depends directly on the core depressurization.

As far as the clading-steam reaction is considered, the cladding unreacted thickness is the most important factor that must be considered. Figure (7) shows the variation the cladding uncreacted thicknes with time. Accordingly, the thickness deterioration starts at about 3 second from the accident intitation. It becomes significant at about 7 seconds due to the higher steam availability at these later seconds.

Of the same importance is the amount of heat generated due to the cladding-steam reaction which is a load on the cladding when considering the cladding integrity. According to Figure (8), the total heat generation rate in the hotest channel increases; as soon as the reaction starts; to significant values and reaches saturation values in the last considered seconds.

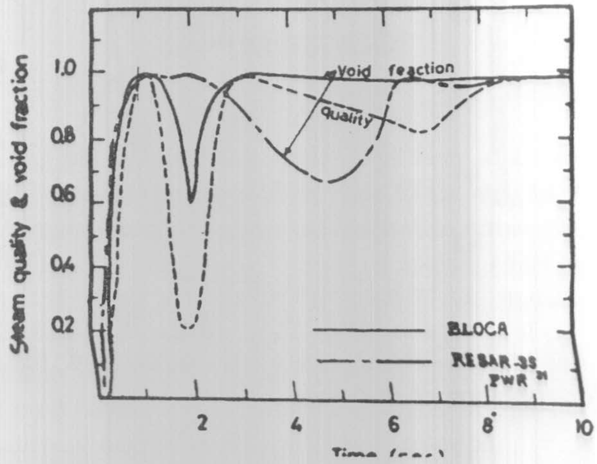

Figure 6. Steam quality & void fraction at core inlet (Av channel).

The theoretical analysis is an important part of the research programmes as it leads to the construction of reliable codes that are used successfully to conduct a complete physical and thermohydrodynamic analysis of NPPs to evaluate their performance before being lisenced.

The present study introduces a simple, fast running code that is based on a complete modeling of the most of the major phenomena covering the blowdown phase of a LBLOCA. The results of the blowdown BLOCA Code in conjunction with type standard reliable codes revealed great satisfaction. The code has an excellent capability to predict all physical, thermal, and hydrodynamic core variables such as thermal power, fuel properties, clading-steam reaction heat generation and cladding deterioration, core flow, system depressurization, as well as the core temperature distribution. The code also uses a set of reliable correlations to predict the heat transfer coefficient and the critical heat flux with the help of a reasonable selection logic scheme which defines the regimes involved in the analysis. All results are shown in Figures (4) to (8).

Referring to these results, the following conclusions are extracted:

(i) The fuel rod cladding temperatures do not reach the limite stated in many safety issues which indicates the inherent level of safety in PWR's unchanged;

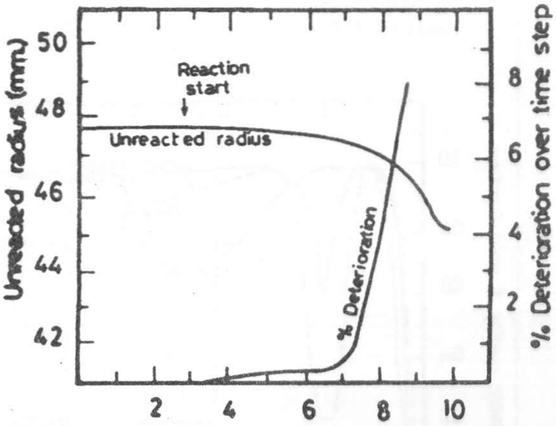

Figure 7. Cladding unreacted radius and the % deterioration of clad over.

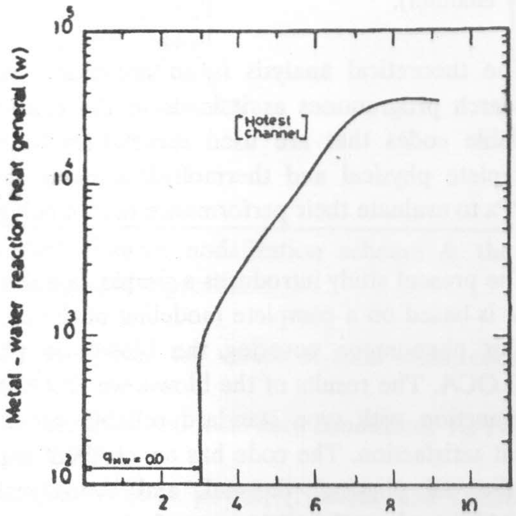

Figure 8. Total heat generated per second due to cladding-steam reaction.

(ii) The integrity of the reactor internals is kept unchanged; under the condition of LBLOCA which is one of the severest accident considered in the safety research programmes;

(iii) The code has an excellent capability to make the necessary analysis and it perdicts effectively the necessary analysis for such a case and the heat transfer coefficient, CHF, and defines the flow regime through a reasonable proposed boiling curve.

(iv) There is a good chance to revise the present code to analyze more cases of similar conditions and normal operation transients since it has a set of functional routines which can be easily modified or exchanged.

REFERENCES

[1] L.J. Agee, "An Analytical Method of Integrating the Thermalhydraulic Conservation Equations", *Nucl. Eng. & Design*, Vol. 42, 1977.

[2] M.N. Aly, A.M. Metwally, and M.K. Shaat, "Simplified Representation for thermal-hydraylic Network", *Modelling, simulation and control, B*, AMSE press, Vol.23, 2, 1989.

[3] M.N. Aly and A.M. Metwally, "DYTF-2 Dynamic Two-Phase Flow Paraeeters Modeule For PWR Cores", *Modelling, Simulation & Control, B*, AMSE Press, Vol. 23, 4, 1189.

[4] G. Frize, "MULTIVOL: AN Extension of The BIVOL BLOWDOWN CODE", *Commission of the European Communities*, 1983.

[5] American National Standards Institute, Inc., " American National Standard For Decay Heat in LWRS", Aug. 1979.

[6] A.A. El-Azab, "BLOWDOWN Characteristics for LWR Cores", M.Sc. Thesis, Alex. University, 1989.

[7] "RELAP 4/MOD '0'": Code Description, Vols 1-3 Idaho National Engineering Laboratory, May 1979.

[8] S. Fabic, "Review of Existing Codes for Loss-of-coolant Accident Analysis", *Advances in Nuc. Science & Tech.*, Vol. 10, 1977.

[9] J.G. Coolier, "Convective Boiling & Condensation", McGraw-Hill, NY, USA, 1972.

[10] M.Benedict and T.H. Pigford, "*Nuclear Chemical Engineering*", McGraw-Hill Book Co., 2-nd ed., 1981.