CONF-871101--37

BNL-NUREG-39933

IN-VESSEL FLOW CHARACTERIZATION UNDER SEVERE ACCIDENT CONDITIONS

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BNL-NUREG--39933

DE87 011921

The phenomenological uncertainties associated with predicting the progression of severe accidents in nuclear reactors has been the subject of several recent studies.[1,2]

In the event of a severe reactor accident involving core uncovery at high primary system pressures, natural circulation of hot gases in the reactor vessel and the hot leg could lead to reactor coolant system failure prior to bottom head melt-through. This could lead to pressure relief before molten core material is expelled from the vessel, thus reducing, and possibly eliminating the potential for high pressure melt ejection-induced direct containment heating in Pressurized Water Reactors (PWRs).

The purpose of the current study is to provide a parametric framework for characterization of flow and heat transfer regimes and their associated phenomenological uncertainties following severe accidents using a two dimensional, heterogenous, porous media formulation. This approach extends the understanding of buoyancy-induced flow characteristics in the uncovered region of the reactor core and the upper plenum of a PWR vessel. The results of this study can be used to augment the boil-off steam flow in integrated one-dimensional severe accident codes such as the Source Term Code Package (STCP).[3]

The model configuration and the coordinate system are shown in Figure 1.

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.

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The flow is assumed to be steady, two-dimensional, and axisymetric. A volumetric heat generation q is assumed in the uncovered portion of the core while internal structures with specific surface area S (surface area per unit volume) and temperature T_u are assumed in the upper plenum. Saturated steam enters the uncovered region of the core (at the froth level) and superheated steam leaves the upper plenum through the hot legs. Invoking Brinkham-Forchheimer extended Darcy's law, the dimensionless form of the governing equations are

$$\nabla^* \cdot \vec{U}^* = 0 \tag{1}$$

$$\nabla^* P^* = \frac{Ra_E}{Pr} \frac{\theta}{1 + \theta\eta} \vec{k} - \left[\frac{B}{\sqrt{Da}} \left|\vec{U}^*\right| + \frac{1}{Da}\right] \vec{U}^* + \nabla^{*2}\vec{U}^* \tag{2}$$

$$\vec{U}^* \cdot \nabla^*\theta = \frac{\nabla^{*2}\theta}{Pr} + \frac{Ra_I}{Ra_E Pr} F(r^*) - \frac{Nu\ S^*(\theta-1)}{Pr} \tag{3}$$

where:

$$\vec{U}^* = \frac{\rho\vec{U}D}{\mu}, \qquad P^* = (P + \rho_o g z)D^2\rho_o/\mu^2, \qquad \Theta = \frac{T-T_o}{T_u - T_o},$$

$$\eta = (T_u-T_o)/T_o, \qquad Ra_E = \frac{g(T_u-T_o)D^3\rho_o^2C_p}{T_o\mu k}, \qquad Ra_I = \frac{g\bar{q}D^5\rho_o^2C_p}{T_o\mu k^2},$$

$$Pr = \frac{\mu C_p}{k}, \quad Da^o = \frac{K^o}{D^2}, \quad S^* = S \cdot D, \quad r^* = \frac{r}{D},$$

$Z^* = \frac{Z}{D}$, and $\vec{k}$ = unit vector in Z^* direction.

In the above equations K^o is the permeability tensor and B is the inertia coefficient and both are functions of the microstructure of the porous media. The effect of radial variation of heat generation is incorporated through a shape factor $F(r^*)$.

The governing Eqs. (1) through (3) were solved by the SIMPLER algorithm.[4] A parametric study was performed to investigate the influence of various parameters, such as fraction of core uncovered, system pressure and decay heat on the flow characteristics. Figure 2 shows the isotherms and the velocity vectors for the typical values of parameters at the time of 70% core uncovery during a station blackout (TMLB') accident scenario in a PWR. For these conditions a two-dimensional circulation pattern is developed in the upper plenum which reaches down into the core. Steam exits the core at high temperature, flows upward where it is cooled in the upper plenum region and re-enters the core at the outer radial regions.

This type of natural circulation flows between the core and upper plenum will redistribute, more effectively than the boil-off flow, the core decay power to structural materials in the upper plenum. This enhanced cooling of the core has significant implications concerning; delays in core degradation, higher hydrogen generation rates, fission product transport and possibility of reactor coolant system failure prior to bottom head melt-through. The present approach will provide an excellent means of performing repeated calculations for a wide range of governing parameters, to correlate steam velocities through the reactor vessel, which in turn can provide a reasonable estimate for augmenting the boil-off flow by natural circulation for use in the existing integrated severe accident codes such as the STCP.[3]

References

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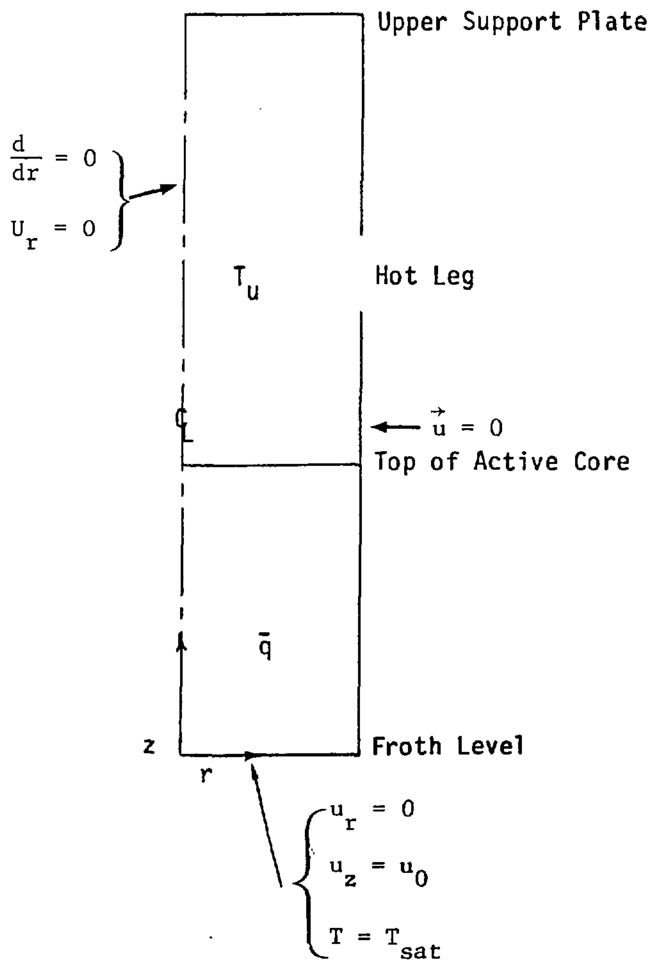

Figure 1: Schematic Diagram of the 2-D Uncovered Core and Upper Plenum

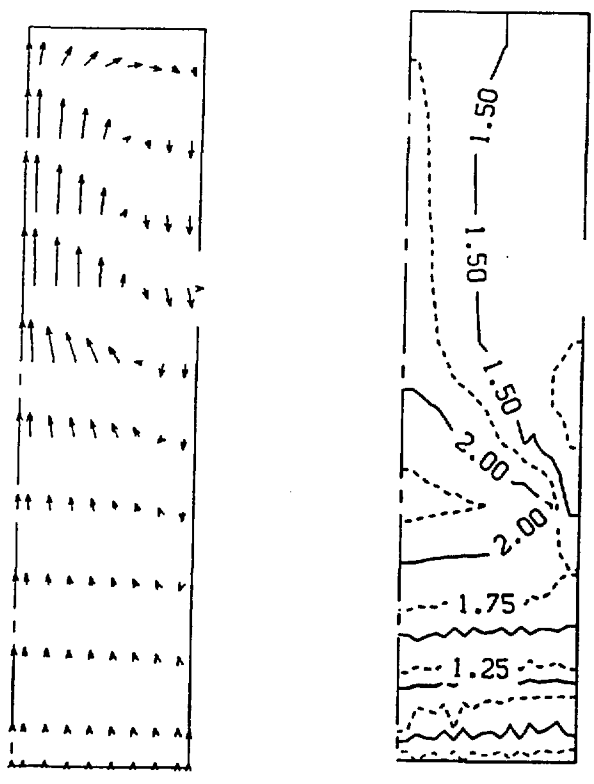

Figure 2: Isotherms and Velocity Vectors for $U_0^* = 8 \times 10^4$, $Ra_I = 6.4 \times 10^{20}$, $Ra_E = 8 \times 10^{14}$