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सारांश

उप-प्रवाह क्षेत्र विधि (Subchannel Analysis) वह तकनीक है जिसके माध्यम से रिएक्टर ईन्धन समूह के उष्म-जलीय बर्ताव का पता लागाया जाता है। छड़ समूह के बीच के सम्पूर्ण प्रवाह क्षेत्र को कई समानान्तरीय अन्तरसम्बन्धित उप-प्रवाह क्षेत्रों में विभाजित किया जाता है। उप-प्रवाह क्षेत्रों के अन्तरसम्बन्धो को ध्यान मे रखते हुये प्रत्येक उप-प्रवाह क्षेत्र के संरक्षण समीकरणों का हल किया जाता है। उप-प्रवाह क्षेत्र विधि द्वारा प्रगत भारी पानी रिएक्टर के डी-५ (D-5) ईन्धन समूह के विभिन्न स्थानो पर ताप-शोषक जल (Coolant) की उष्म-जलीय परिस्थिति और ईन्धन ताप का अनुमान लगाया गया है और सबसे गर्म ताप वाले स्थानो का भी पाता लागाया गाया है। लम्बे समय की वजह से ताप-शोषक नली में हुई विकृति (Creep) के चलते ईन्धन की बदली कार्यक्षमता का भी अध्यन किया गया है। जन्सेन-लेवी संबन्ध द्वारा निम्न सीमान्त उष्मा अनुपात (MCHFR) की गणना की गई है। कुछ उप-प्रवाह क्षेत्र राशियो का सही-सही ज्ञान नही था परन्तु सम्वेदनशीलता विधि (Sensitivity Analysis) द्वारा प्रमाणित किया गया कि गणना पर उनका प्रभाव नगण्य है। इस लेख में उप-प्रवाह क्षेत्र की कई अन्य संगणकीय विधिया भी वर्णित है। COBRA-IIIC और COBRA-IV-I मे अपनायी गयी विधियो और समीकरणों का भी संक्षिप्त विवरण यहाँ दिया गया है।

Abstract

Subchannel analysis is a technique used to predict the thermal hydraulic behavior of reactor fuel assemblies. The rod cluster is subdivided into a number of parallel interacting flow subchannels. The conservation equations are solved for each of these subchannels, taking into account subchannel interactions. Subchannel analysis of AHWR D-5 fuel cluster has been carried out to determine the variations in thermal hydraulic conditions of coolant and fuel temperatures along the length of the fuel bundle. The hottest regions within the AHWR fuel bundle have been identified. The effect of creep on the fuel performance has also been studied. MCHFR has been calculated using Jansen-Levy correlation. The calculations have been backed by sensitivity analysis for parameters whose values are not known accurately. The sensitivity analysis showed the calculations to have a very low sensitivity to these parameters. Apart from the analysis, the report also includes a brief introduction of a few subchannel codes. A brief description of the equations and solution methodology used in COBRA-IIIC and COBRA-IV-I is also given.

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1. INTRODUCTION

Subchannel analysis is used for the prediction of the thermal hydraulic conditions in the subchannels of reactor fuel assemblies. In subchannel analysis, the coolant flow path is considered to be subdivided into a number of parallel interacting flow subchannels. The equations of mass, momentum and energy conservation are solved to give radial and axial variations in fluid enthalpy and mass velocity. The approach of the rods in the assembly to CHF is then assessed by means of appropriate CHF correlation applied to subchannel. Recently developed codes like CAPE-BWR[1] have used liquid film model (a mechanistic model), for dryout prediction. This method is based on tracking film thickness till it becomes zero. The power at which liquid film ceases to exist is the critical power.

It is desirable to determine flows and enthalpies in all subchannels subject to specified conditions at the inlet and outlet of subchannels. This can be considered a multi-point boundary value problem in the space domain. Subchannel codes, suitably adapted for reactor channel conditions, can be used for this problem, but because of numerical difficulties, a number of codes solve an approximation of the problem where inlet flows are assumed to be known or set by some simple criterion. This simplified problem can be considered as an initial value problem. Once steady state conditions are established, subchannel behavior during a given transient can be solved as an initial value problem in the time domain.

The main purpose of this report is to present and discuss the results obtained from the subchannel analysis of AHWR fuel cluster. Apart from that, a short review of the various popular subchannel codes is also given. COBRA-IIIC and COBRA-IV-I: the codes used for the present analysis have been discussed in greater detail.

2. A REVIEW OF FEW SUBCHANNEL CODES[2]

The various codes used for subchannel analysis are: COBRA[3,4], THINC[5], FLICA[6], CISE[7], HAMBO, MIXER etc. The mathematical models used in the different codes are essentially similar; the area of uncertainty is the interaction between the subchannels. This interaction can be divided into: crossflow (mass transfer between subchannels due to radial pressure gradient), turbulent mixing (mixing process independent of pressure gradient), and void drift (mixing to account for the flow resulting from tendency of two-phase flow to attain equilibrium). The latter has been used in newer codes such as CAPE-BWR.

Generally all subchannel codes adopt a coolant-centered methodology; the subchannels are defined by lines joining rod centers, or if against outer pressure vessel, by the line through the rod center normal to the outer surface. CISE, an Italian code however differs in this respect by having a rod-centered approach for subchannels. The subchannel boundaries are defined by what are called "Lines of Zero Shear Stress" around the rods. This scheme can have advantages in annular film two-phase flow regions where there is a tendency of liquid redistribution around the rod. However in this approach, it is difficult to:

1) Define lines of zero shear stress when there is a superimposed crossflow. And,
2) Quantify interactions between subchannels.

One of the inherent simplifications in subchannel analysis is that, no radial gradients of flow and enthalpy are considered in subchannels but only across subchannel boundaries. Incorporating buffer regions as in French code FLICA can smoothen the abrupt transition. This leads to increased complexity and greater calculation times.

With regards to subcooled boiling, COBRA uses the Levy model and HAMBO and some versions of THINC modified versions of Bowring model. Both COBRA and HAMBO use equal mass model for turbulent interchange. MIXER and ASSERT (A Canadian code) use equal volume model. Recent experiments indicate that the latter assumption is more realistic. A short summary of few well known codes is given below:

2.1 COBRA[3,4]

COBRA is an acronym for COolant Boiling in Rod Arrays. These codes are the best documented of all subchannel analysis procedures. These codes have undergone considerable evolution, the latest version being COBRA-IV-I. Levy's model for subcooled boiling is used. It uses an equal mass model for turbulent mixing. It can also consider variations in subchannel areas and gap spacing. The other important features of COBRA, since COBRA-IIIC onwards are a fuel heat transfer model, a more complete lateral momentum balance equation, ability to impose a constant outlet pressure as a boundary condition and provision for forced crossflow mixing. COBRA-IV additionally incorporates inlet pressure also as the boundary condition so that requirements of constant inlet and outlet pressures are met. Reverse flows can also be accommodated. While COBRA-IIIC uses Dittus-Boelter correlation for calculating heat transfer coefficient, COBRA-IV has specific correlations for calculating two-phase heat transfer coefficient.

2.2 HAMBO

The physical model used in HAMBO is similar to that used in COBRA. There are, however a few differences e.g., HAMBO uses Bowring model for subcooled boiling and variations in subchannel areas and gap spacing are not considered in HAMBO. The equal mass model used in HAMBO includes a coupling between mixing and crossflow i.e., the mixing contribution is reduced in the presence of a strong crossflow. In HAMBO, a central differencing scheme is used for solving the enthalpy equation. HAMBO and COBRA give similar, though not same results to the same problem.

2.3 FLICA[6]

This French code was written to study the behaviour of reactor as a test loop. An important feature of this code is the presence of gap buffer regions between main subchannels. It also includes the effect of neutronic coupling and allows for recirculating pipework linking one or a number of channels with a bypass in parallel. This subchannel code, for steady-state and transient thermal analysis of water cooled reactors is widely used in France by CEA, Framatome, EDF and Technicatome.

2.4 THINC[5]

In the model for this code, void generation is assumed to occur in three stages. In the highly subcooled region, Maurers's correlation is used; in the slightly subcooled region, voids are obtained from Tong and Weisman's modification of the correlation of Thom et al. or a modified form of Bowring's correlation. When the liquid reaches saturation temperature, all of the additional heat is used to produce vapor. Pressure drops are calculated assuming homogenous flow. Mixing is defined by a single parameter that does not vary with quality.

Initially THINC II assumed that no pressure gradient exists within the hot assembly. The reasoning given was that due to close coupling between channels, there could be only a very low, pressure gradient in the assembly causing only very small differences in axial flows. The change in axial flow in a given channel was determined by the requirement that the pressure drop be the same across each control volume at a given elevation. THINC IV abandons this assumption and the lateral momentum balance incorporates both frictional and inertia terms. It is also capable of dealing with reverse flows.

2.5 VIPRE[8]

VIPRE (Versatile Internals and Component Program for Reactors; EPRI) was developed by EPRI for nuclear power utility thermal-hydraulic analysis applications. It was designed to help evaluate nuclear reactor core safety limits including minimum departure from nucleate boiling ratio (MDNBR), critical power ratio (CPR), fuel and clad temperatures, and coolant state in normal operation and assumed accident conditions.

2.5.1 VIPRE-01

VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. VIPRE was developed on the strengths of the various COBRA codes and has been tailored to the analytical needs of nuclear utilities. It has improved flexibility of use, upgraded capabilities, and improved numerical solution schemes. Some of the more important improvements are an expanded choice of correlations for critical heat flux, critical power ratio, two-phase flow and heat transfer for reload and safety analysis, one-pass hot-channel analysis capability, automatic iteration for set point analysis, subcooled voiding capability, and the ability to compute bypass channel flow for boiling water reactor (BWR) applications.

VIPRE-01 predicts the three-dimensional velocity, pressure, enthalpy fields, and fuel rod temperatures for single- and two-phase flow in pressurized water reactor (PWR) and boiling water reactor (BWR) cores. It solves the finite-difference equations for mass, energy, and momentum conservation for an interconnected array of channels, assuming incompressible thermally expandable homogeneous flow. The equations are solved with no time-step or channel size restrictions for stability. Although the formulation is homogeneous, non-mechanistic models are included for subcooled boiling and vapor/liquid slip in two-phase flow.

2.5.2 VIPRE-02

VIPRE-02 is a thermal-hydraulic analysis code designed to model steady-state conditions and operational transients in LWR cores and vessels. It uses a two-fluid representation of two-phase flow, solving conservation equations for mass, momentum and energy for each phase. This six-equation model is solved implicitly, using a modified Gauss-Seidel iteration procedure, and has no time-step size limitation for stability. Models for phase interaction based on flow regime mapping are provided, using semi-empirical interfacial correlations for heat and mass transfer, and vapor generation. In addition, the code contains as an option a dynamic flow regime model, which uses an interfacial area transport equation to determine the phase interaction terms.

For core analysis, VIPRE-02 uses a subchannel formulation of the conservation equations; for PWR vessel models, it contains options for a fully three-dimensional representation of the lower plenum. For core analysis, boundary conditions can be specified using the inlet flow or the overall core pressure drop. For vessel models, the hot and cold leg boundaries can be represented using mass sources or pressure sinks in appropriate modes, with local flow blockages to model vessel internal structures and the lower plenum and upper head domes. The input specification is designed to be as flexible as possible, and still remain relatively user friendly

3. COBRA IIIC and COBRA-IV-I: THE CODE DETAILS AND THE MODELING USED

In the present work, COBRA IIIC and COBRA-IV-I have been used for carrying out subchannel analysis from given data on inlet mass flux, inlet temperature/enthalpy, inlet crossflow and exit pressure. COBRA allows the user to specify his/her own correlations or use those available in COBRA. The basic assumptions in COBRA are:

1) One-dimensional, two-phase, slip flow exists in each subchannel during boiling.
2) The two-phase flow structure is fine enough to allow specification of void fraction as a function of enthalpy, pressure, flow rate, axial position and time.
3) A turbulent crossflow exists between adjacent subchannels, which cause no net flow redistribution (equal mass model).
4) The turbulent crossflow may be superimposed upon a diversion crossflow between subchannels that results from flow redistribution. This may occur artificially from devices that force diversion crossflow.
5) Sonic velocity propagations are ignored. (COBRA can't handle fast transients)
6) The diversion crossflow velocity is small compared to the axial velocity within a subchannel.

3.1 Equations used in COBRA

The basic equations solved in COBRA IIIC are continuity, energy, axial momentum and transverse momentum along with an equation of state. These equations for a subchannel i interacting with subchannel j can be written as:

Continuity equation:

$$A_i \frac{\partial \rho_i}{\partial t} + \frac{\partial m_i}{\partial x} = - w_{ij}$$

Energy equation:

$$\frac{1}{u_i}\frac{\partial h_i}{\partial t} + \frac{\partial h_i}{\partial x} = \frac{q_i'}{m_i} - \left(h_i - h_j\right)\frac{w_{ij}'}{m_i} - \left(T_i - T_j\right)\frac{c_{ij}}{m_i} + \left(h_i - h^*\right)\frac{w_{ij}}{m_i}$$

The third term on the right hand side of energy equation accounts for the conduction between adjacent subchannels. The temperature is related to enthalpy via equation of state.

Axial Momentum Equation:

$$\frac{1}{A_i}\frac{\partial m_i}{\partial t} - 2u_i\frac{\partial \rho_i}{\partial t} + \frac{\partial p_i}{\partial x} = -\left(\frac{m_i}{A_i}\right)^2\left[\frac{v_i f_i \phi_i}{2D_i} + \frac{k_i}{2\Delta x} + A_i\frac{\partial\left(v_i'/A_i\right)}{\partial x}\right] - \rho_i g\cos\theta$$

$$-\frac{f_i}{A_i}(u_i - u_j)w_{ij}' + \frac{1}{A_i}(2u_i - u^*)w_{ij}$$

Transverse momentum equation:

$$\frac{\partial w_{ij}}{\partial t} + \frac{\partial(u^* w_{ij})}{\partial x} = \frac{s}{l}(p_i - p_j) - F_{ij}$$

Fuel model in COBRA

The fuel heat transfer model calculates the internal temperature distribution within the fuel pin using the fundamental heat conduction equation:

$$\rho c\frac{\partial T}{\partial t} = k\left(\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r}\right) + q'''$$

The fuel model is interfaced with fluid thermal hydraulic solution using surface heat transfer coefficients.

The major difference between COBRA-IIIC and COBRA-IV-I for steady state cases arises in the availability of heat transfer coefficients. While, COBRA-IIIC uses Dittus-Boelter equation, COBRA-IV-I uses a complete boiling and non-boiling correlation package similar to that of the RELAP-4 computer code. Hence temperature predictions of COBRA-IV-I are better than COBRA-IIIC. Apart from this difference, COBRA-IIIC and COBRA-IV-I have the same set of conservation equations and same solution scheme for steady state problems.

3.2 Solution scheme in COBRA-IIIC and COBRA-IV-I

In COBRA-IIIC, the conservation equations are solved as a boundary value problem by using a semi-explicit finite difference scheme. The boundary conditions are inlet values of enthalpy, flow and crossflow, and exit pressure. The energy equation is solved first to determine enthalpy values at the next axial location. This is followed by flow and pressure calculations. The pressure calculation scheme allows the downstream pressure variations to be felt at upstream locations. For the fuel rod temperature calculations, COBRA-IIIC uses a finite difference scheme for time, radial and axial derivatives.

The COBRA-IV-I solution scheme is same as the COBRA-IIIC scheme for steady state problems. Hence subchannel thermal hydraulic parameters obtained from both the codes is identical. For transient calculations, however, COBRA-IV-I has options for both explicit as well as implicit (also available in COBRA-IIIC) solution schemes. Apart from inlet flow boundary condition, COBRA-IV-I also has the option for pressure drop boundary condition. Here unlike COBRA-IIIC, Method of Weighted Residuals (MWR) is used for obtaining fuel *temperature distribution in radial direction*. This method affords higher order of accuracy by using the roots of orthogonal polynomials as the nodal positions where the solution is evaluated. Time and axial derivatives are obtained by finite difference method.

3.3 CHF Prediction in COBRA

MCHFR calculation gives an estimate of the safety margin available during reactor operation. For prediction of CHF Babcock and Wilcox (B&W-2) and Westinghouse (W-3) correlations are available in COBRA-IIIC and COBRA-IV-I. Additionally, Jansen-Levy, Reddy and AECL's 1996 CHF look-up table have been added in COBRA-IIIC and Jansen-Levy has been added in COBRA-IV-I.

4. SUBCHANNEL ANALYSIS OF AHWR D-5 CLUSTER

Advanced Heavy Water Reactor (AHWR) is a 920 MWth channel type boiling water reactor, in which natural circulation is employed for coolant flow through the vertical core. It employs light water as coolant and heavy water as moderator.

Thorium utilization is one of the main objectives kept in mind while designing AHWR fuel. Thorium is a fertile material which converts to U233-a fissile material. The reactor is designed to be self sufficient in its U^{233} requirements under equilibrium core conditions. The fuel bundle of AHWR consists of 54 fuel pins arranged in three concentric rings of 12, 18 and 24 pins (fig. 1). The fuel pin consists of fuel pellets loaded into Zircaloy tube of 11.2 mm OD and 0.6mm thickness. The fuel pin has a pellet stack length of 3500 mm and a plenum volume at top to provide space for accumulation of fission gases. A central dysprosium rod is present for providing negative void coefficient. The details of fuel used in AHWR are as follows:

1) $(Th\text{-}Pu)O_2$ in outer ring pins. There are 24 such pins.
2) $(Th\text{-}U^{233})O_2$ in intermediate (18 pins) and inner ring pins (12 pins).

The enrichment (% wt.) of the fuel is as follows:

1) Pu = 3.25% in outer ring pins.
2) U^{233} = 3.75% in intermediate ring pins.
3) U^{233} = 3.0% in inner ring pins.

In this work, a steady state analysis of the AHWR fuel bundle has been carried out to determine the flow and enthalpy distributions, MCHFR and the effect of channel creep on MCHFR.

The fuel enrichment values mentioned above are averaged over the length of the bundle. The effect of axially graded fuel is seen in the axial heat flux profile (fig. 6) used for calculations. Radial power profile within the fuel cluster has also been considered. The thermal conductivities used for determination of fuel temperature distribution are based on the enrichment and composition of fuel pellets.

4.1 Subchannel model for AHWR fuel Bundle

A $1/12^{th}$ symmetry sector of the AHWR fuel cluster (Fig. 1) has been considered for subchannel analysis. This model consists of 4.5 fuel pins and 12 coolant centered subchannels.

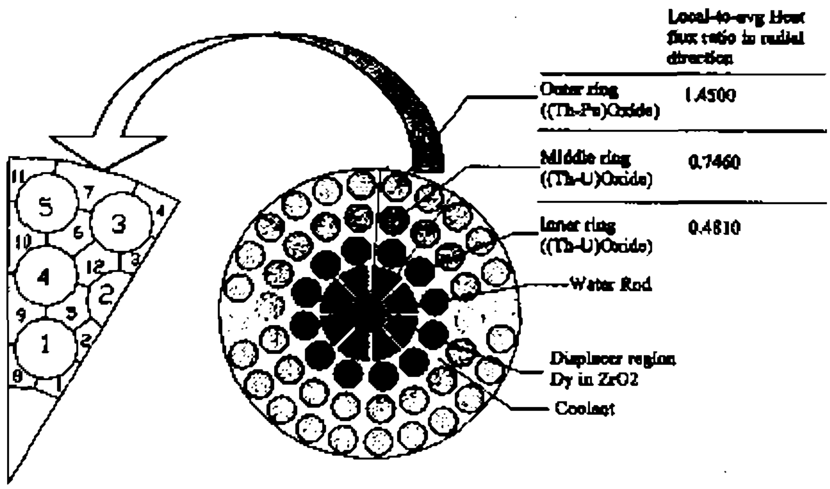


Fig. 1 1/12[th] symmetry sector of AHWR fuel cluster considered for analysis. The subchannel and fuel rod numbering have also been shown.

AHWR fuel bundle has a total length of 3.80m of which 3.50m consists of active fuel and the rest is unheated. Subchannel analysis has been carried out for normal power and 20% over-power conditions for both average power and maximum power channels. Subchannel analysis has also been carried out for crept channels.

4.2 Input data for various cases

The input data used for the calculations has been summarized in table 1

Table1: Input data

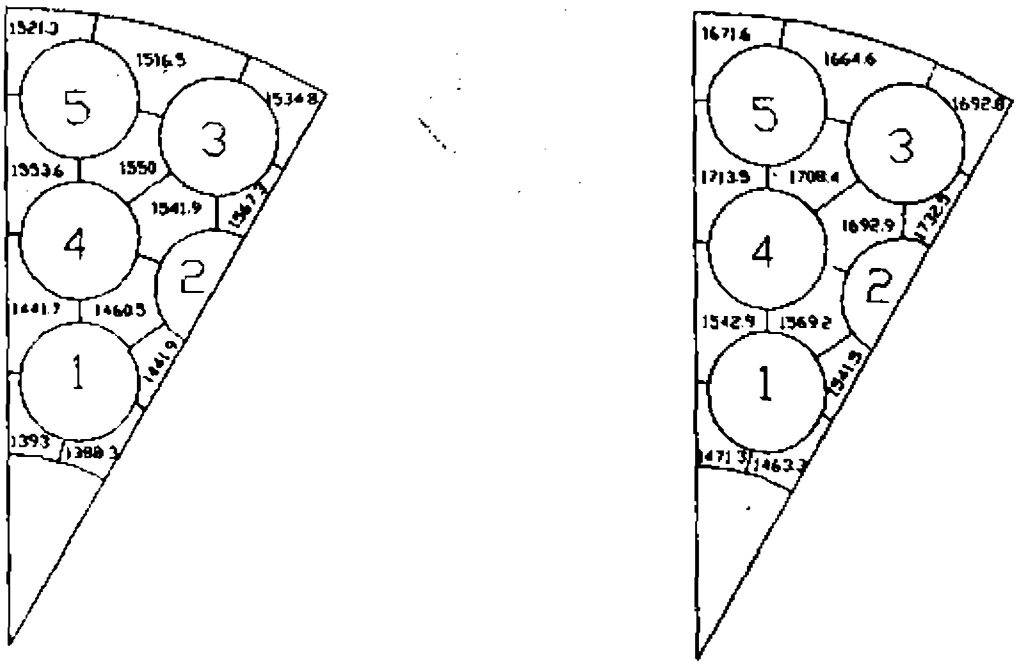
	Normal power conditions		20% over power conditions	
	Average power Channel	Max. power Channel	Average power Channel	Max. power Channel
Active coolant power (MW)	2.038	2.600	2.446	3.120
Average heat flux (clad)* (W/m^2)	308378	393404	370056	472096
Coolant mass velocity ($kg/m^2.s$)	1064.53	997.99	1015.27	939.9
Clad thickness (mm)	0.6			
Clad conductivity (W/m.K)	13.9			
Gap conductance ($W/m^2.K$)	8517.4 (1500 $Btu/hr.ft^2.F$)			
System pressure	70 bar			
Inlet temperature	259.3^{0}C			
Radial peaking factors	Pins in inner ring		0.4810	
	Pins in middle ring		0.7460	
	Pins in outer ring		1.4500	
Maximum axial peaking factor	1.4268	1.3874	1.4268	1.3874

4.3 *Results and discussions*

The Advanced Heavy Water Reactor (AHWR) has a bottom peaking axial profile. This has been done to achieve a more or less constant MCHFR along the bundle thereby increasing the power that can be extracted from the fuel bundle.

4.3.1 Exit enthalpy and void fraction distribution

The calculated mixture enthalpy at the exit of coolant channel is shown in fig. 2.

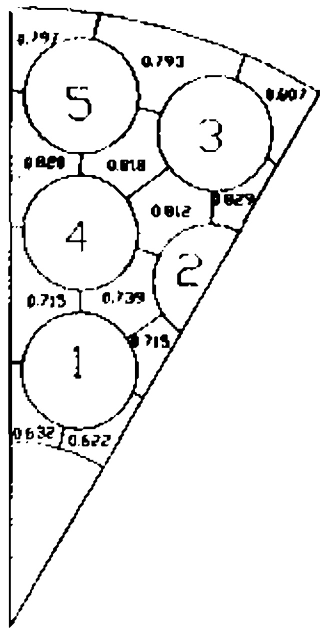


(a) Average power channel

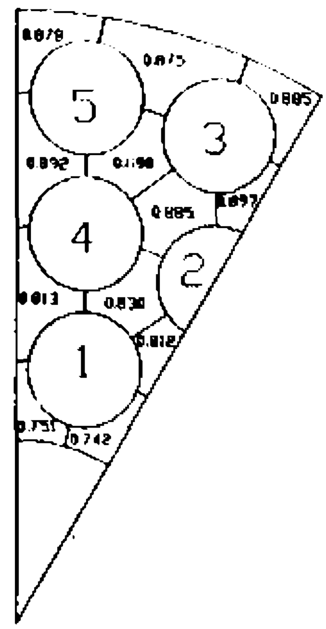
(b) Maximum power channel

Fig 2. Exit enthalpy (kJ/kg) under full power conditions for (a) average and (b) maximum power channels.

The enthalpy values in subchannels 1 and 8 are the least due to the radial power profile. Subchannels 3, 6, 10 and 12 are bounded by the fuel pins generating maximum power also the flow area in these subchannels is smaller and the inter subchannel gaps are constricted hence these are the hottest subchannels. It is found that these subchannels are the closest to critical heat flux condition at most of the axial locations. Due to higher enthalpy, there is greater void fraction in these subchannels. This is clear in fig. 3



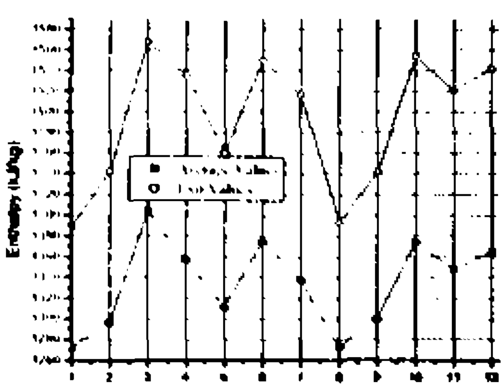
(a) Average power channel



(b) Maximum power channel

Fig 3. Exit void fraction in full power conditions for (a) average and (b) maximum power channels.

It can be seen from these figures that rods 3 and 5 are most susceptible to approaching critical heat flux. This inference is vindicated by MCHFR calculations.



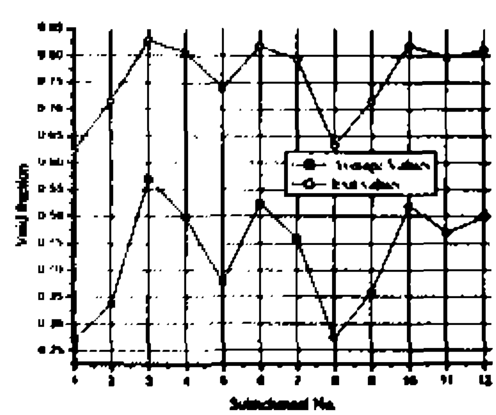


Fig.4 Comparison of Subchannel average and exit values of enthalpy and void fraction for average power channel under full power conditions.

In fig.4, the exit and average (along the length of a subchannel) values of enthalpy and void fraction are shown for the subchannels of an average power channel under normal power conditions. The trend for both average and exit values are the same. The curves reiterate the high enthalpy values in subchannels 3, 6, 10 and 12. It hence turns out that rod number 3 and 6 are most prone to approaching CHF condition.

4.3.2 Axial distribution of thermal hydraulic parameters

The pressure drop for the AHWR fuel bundle is shown in fig.5. As expected, pressure drop increases with channel power. Hence in the cases studied, pressure drop is highest for maximum power channel in 20% over-power conditions. The pressure values are plotted with reference to system pressure (pressure in steam drum). The pressure drop profile shows small *ripples due to spacers*. Towards the exit, pressure drop is higher and a marked drop is seen at the location of top tie-plate.

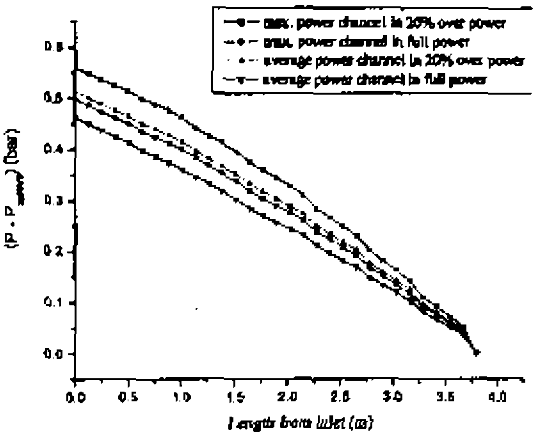


Fig. 5 Axial pressure variation (bundle average values)

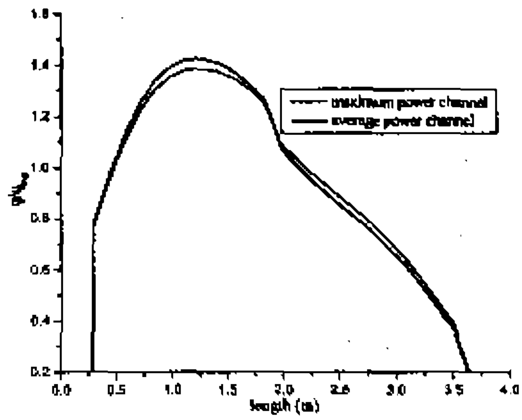


Fig. 6 Axial power variation (heated length =3.5m)

The axial power profile has a very important bearing on the axial distributions of thermal hydraulic parameters. In AHWR, a bottom peaking Heat flux profile is achieved by differential axial enrichment of the fuel. This kind of profile leads to a flat MCHFR profile (fig.11) in AHWR which is desirable as it increases the operating power. Fig. 6 shows the axial power profile for average power and maximum power channels of AHWR. It must be noted that the heated length is 3.5m.

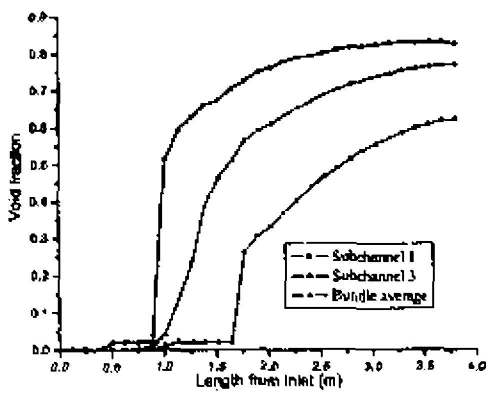


Fig. 7 Void fraction variation along length

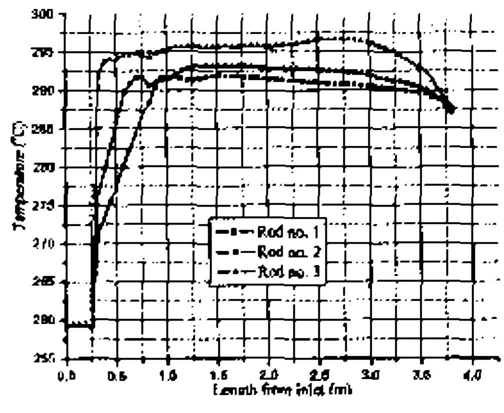


Fig. 8 Clad surface temperature variation along length

The void fraction variation with length for the channels with maximum and minimum void fractions is shown along with the bundle average values in fig. 7. The results shown are for average power channel at full power conditions. Due to the bottom peaking axial power profile, bulk boiling starts very near to inlet. This helps in extracting more power from the fuel as the heat transfer coefficient for two-phase flow is much higher than single phase flow.

4.3.3 Fuel temperature distribution

COBRA-IIIC uses the Dittus-Boelter correlation for calculating heat transfer coefficient at clad surface in two-phase region, this leads to higher than actual values for fuel temperatures. This problem is overcome in COBRA-IV-I, which has inbuilt correlations for calculation of heat-transfer coefficient and basic selection logic for choosing among the correlations according to the region in which the system lies. Seven regions are considered: forced convection, subcooled and nucleate boiling, forced convective vaporization, transition boiling, transition pool boiling, film boiling and pool film boiling. Hence, for fuel temperature calculations, COBRA-IV-I has been used. It should however be noted that there is no difference in the predictions of COBRA-IIIC and COBRA-IV-I as regards coolant thermal-hydraulic properties at steady state. The clad surface temperatures for inner, middle and outer rods are plotted in fig 8. Since fuel temperatures are limiting criteria in reactor operation, the maximum power channel values have been plotted. Due to boiling initiation at the very beginning of the channel, clad temperatures are high all along the length, except for the region where boiling has not initiated. Maximum clad surface temperature is approximately 296°C in rods 3 and 5.

Axial and radial variations of temperature within the fuel pin have also been calculated and are presented in fig. 9 as contour plots. The details of analysis for fuel temperature prediction are described in ref. 9. It can be seen that outer pins are the hottest. This can be attributed to higher heat generation in outer pins and their lower thermal conductivity. The maximum centerline temperature is 1020°C in rods 3 and 5.

The results for rods 3 and 5 are different from those in ref. 9. This is because, in the report, the thermal conductivity of outer rods was conservatively taken to be that of UO_2 whereas in the present case, it is taken as that of $(Th\text{-}4\%Pu)O_2$.

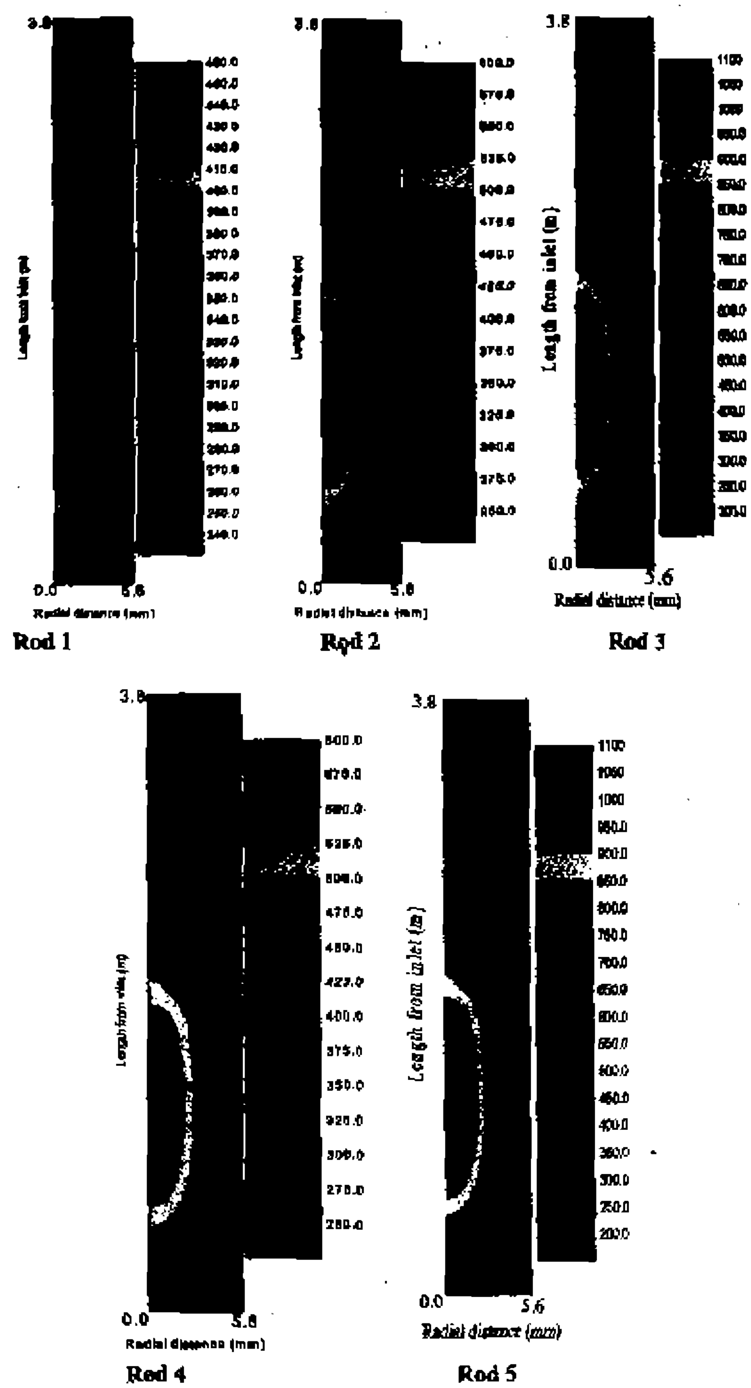


Fig.9 Contour plots of fuel temperature (°C) (Results for maximum power channel in full power conditions.).

Radial variation of fuel temperature for the inner ring (rod 1), middle ring (rods 2, 4) and outer ring (rods 3, 5) of rods has been shown in fig. 10. These results are plotted at the axial location having maximum fuel centerline temperature.

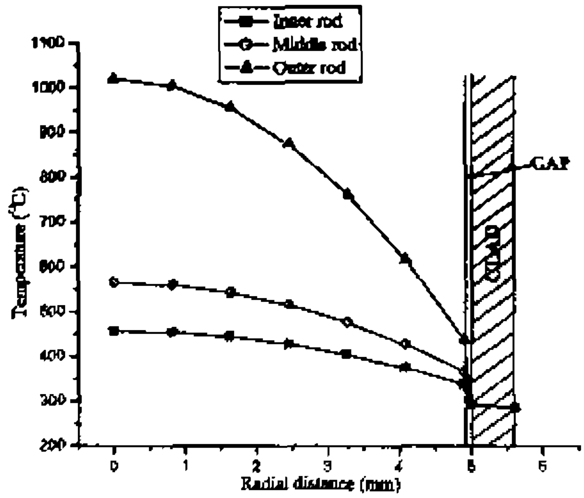


Fig 10 Radial variation of fuel temperature at the axial location having maximum centerline temperature

4.3.4 Minimum Critical Heat flux Ratio (MCHFR)

Meeting the MCHFR criteria is an essential requirement for reactor safety. Accurate calculation of CHF requires an accurate estimation of thermal hydraulic parameters around the fuel rods. Hence, CHF calculation is affected by the accuracy of the subchannel model. Uncertainty in CHF prediction also arises from the correlation used for CHF calculation. For safety purposes, the values of MCHFR furnished are for the maximum power channel at 120% full power conditions. Fig. 11 plots the MCHFR among fuel elements at a certain axial location against the axial distance from inlet of coolant channel. The effect of bottom peaking heat flux profile (fig.6) can be markedly seen as a flat axial profile of MCHFR is obtained. There is a sudden drop in MCHFR values at around 1 m from inlet due to inception of boiling. Janssen-Levy correlation has been used for estimating CHF values based on local conditions provided by subchannel analysis. The MCHFR hence obtained is 1.161.

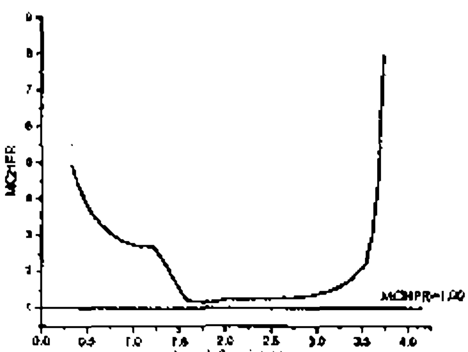


Fig. 11 Axial variation of MCHFR at 120% full power

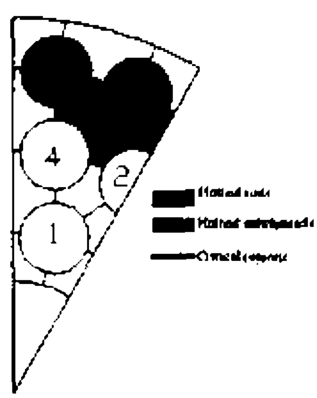


Fig. 12 Hottest regions in AHWR fuel bundle (CHF considerations)

Rods 3 and 5 (refer fig. 1) have been found to be most susceptible to approaching CHF. Subchannels 3, 6 and 12 are the hottest subchannels. The predicted regions of CHF occurrence are identified in fig. 12.

4.3.5 Influence of channel creep

Since the coolant circulation in AHWR is by natural circulation, the coolant flow and coolant inlet conditions adjust according to flow resistance offered by the system. In the present study, results of 3% and 6% concentrically crept channels have been compared with uncrept channel. Maximum power channel under normal power conditions has been analyzed; the inlet temperature and mass-flux are given below:

Table2: Inlet mass-flux and temperature for crept channels

	Uncrept	3% creep	6% creep
Inlet Temperature (°C)	259.3	260.65	261.38
Inlet Mass-flux (kg/m^2.s)	987.9205	907.4914	836.7169

The exit values of mass-flux are compared for the different subchannels (fig. 13). It can be expected that the average values follow the similar trend (see fig. 4). Firstly, from the table it is clear that mass-flux reduces with creep. Secondly, it is seen (fig. 13) that the mass flux in outer channels (4, 7 and 11) increases substantially with creep as more open space is available for flow of coolant. Mass-flux in other subchannels is thereby reduced. The third row of subchannels (3, 6, 10 and 12), which are the hottest; become hotter; this leads to lowering of MCHFR.

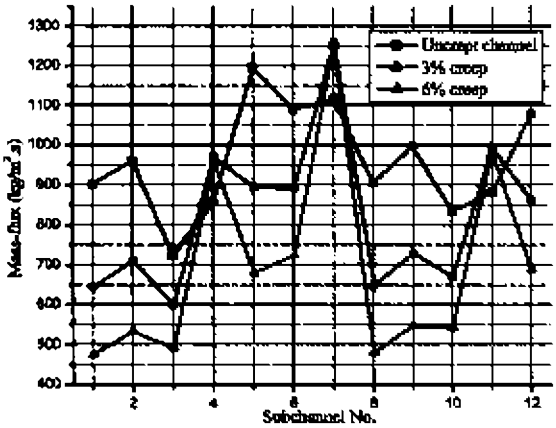


Fig 13 Subchannel exit mass-flux variation with creep

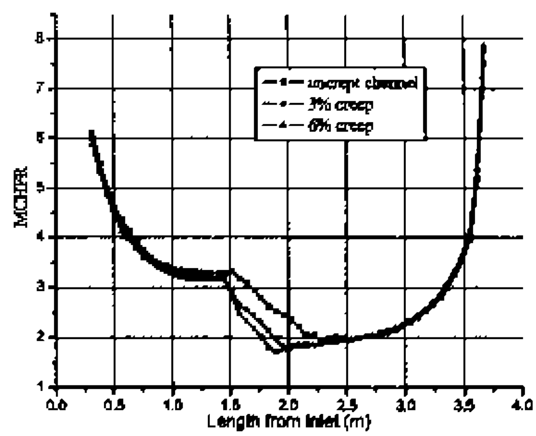


Fig. 14 MCHFR variation for different values of channel creep (full power conditions)

The MCHFR variation along length is plotted in fig. 14. It is seen that MCHFR values lower with creep and the location of CHF occurrence shifts upstream as creep increases. These MCHFR values are significantly higher than those given in fig. 11 because here the analysis has been carried out at full power conditions.

4.3.6 Sensitivity analysis

The transverse momentum equation in COBRA-IIIC consists of parameters like (s/l) and the transverse friction coefficient, K_{ij} (this term occurs in the simplification of F_{ij}). Since the exact values of these terms are unknown, their effect needs to be studied. Sensitivity analysis was done for these two parameters, while considering the case for maximum power channel at reactor full power conditions. Results plotted are for subchannel 6 (figs. 15 and 16).

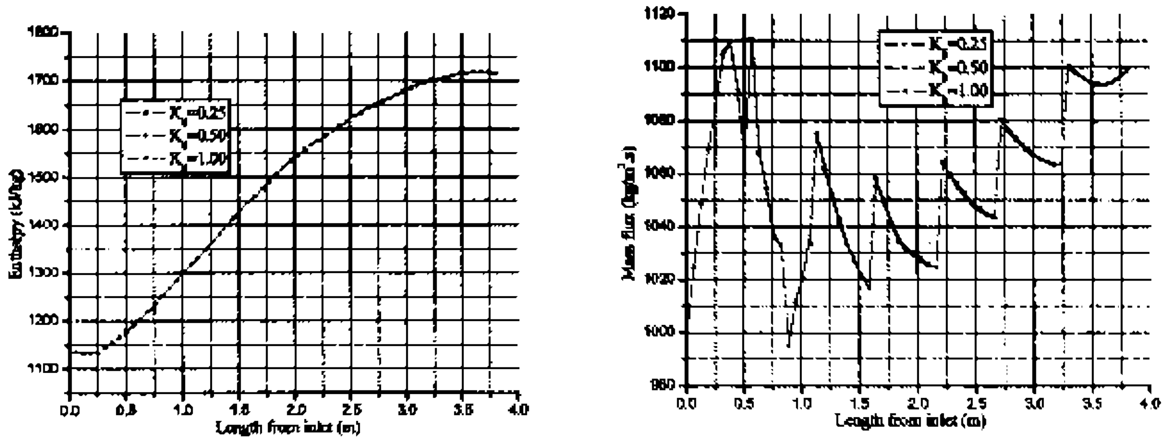


Fig. 15 Effect of variation of K_{ij}.

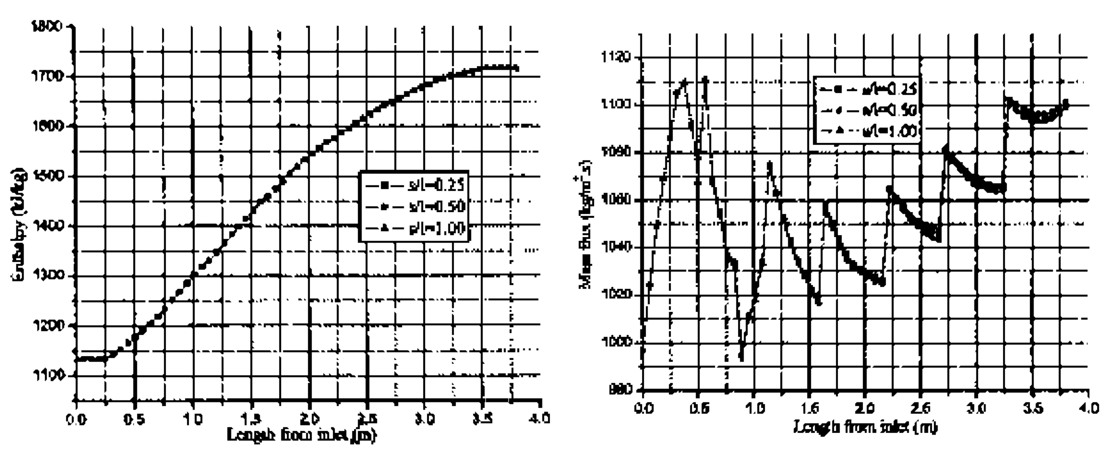


Fig. 16 Effect of variation of s/l.

It is clear from the sensitivity analysis that the effect of varying s/l and K_{ij} on the computed flow is small, hence their values need not be known precisely, only representative values corresponding to reactor conditions are sufficient. These terms do not have a great influence on flow solutions because the magnitude of crossflow is very small compared to axial flow.

5. CONCLUSIONS

The thermal hydraulic behavior of AHWR fuel cluster has been studied using the subchannel codes COBRA-IIIC and COBRA-IV-I. The conclusions can be summarized as follows:

1. The hottest subchannels (3, 6, 10 and 12), hottest rods (3 and 5) and the regions most susceptible to approaching critical heat flux have been identified.
2. It has been observed that AHWR fuel has a flat MCHFR profile over a significant length. This is due to a bottom peaking axial heat flux profile which is a direct result of differential axial enrichment of fuel.
3. The fuel temperatures were evaluated and it was found that maximum clad surface and maximum centerline temperatures occur in the outer ring of fuel pins. Due to low power density of AHWR as compared to other reactors like PHWR and PWR, the maximum fuel centerline temperature is limited to lower values (approx 1020 °C in AHWR).
4. Concentric creep was analyzed and it was found that MCHFR decreases with increasing creep. This is due to flow redistribution on favour of outer subchannels (4, 7 and 11) as a result of which, the third critical ring of subchannels (3, 6, 10 and 12) receives lesser flow and becomes hotter.
5. Sensitivity analysis with respect to s/l and K_{ij} (transverse friction parameter) has been carried out. The variation of thermal hydraulic parameters such as mass-flux, enthalpy, quality etc. found to be very less sensitive to these terms.

6. NOTATIONS

A = Subchannel area

c = Thermal conduction coefficient, Specific heat

D = Subchannel hydraulic diameter

f = Single phase friction coefficient

F = Transverse friction term

f_T = Transverse momentum parameter

h = Enthalpy

h^* = Effective enthalpy carried by diversion crossflow

k = Spacer loss coefficient, Thermal conductivity

K = Transverse friction coefficient

l = Centroidal distance between adjacent subchannels.

m = Mass flow rate

p = Pressure

q' = Heat flux to coolant

q''' = Rate of heat generation per unit volume

s = Gap length

t = Time

T = Temperature

u = Axial flow velocity

u^{**} = Effective velocity for enthalpy transport

u^* = Effective velocity carried by diversion crossflow

v = Liquid specific volume

v' = Mixture specific volume

w = Diversion crossflow

w' = Turbulent crossflow

x = Length along channel axis

Greek symbols:

ϕ = Two phase friction multiplier

ρ = Density

θ = Angle of inclination from vertical

Subscripts:

i = Subchannel identification number

j = Neighbouring subchannel of i

7. REFERENCES

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