STEADY STATE THERMAL HYDRAULIC ANALYSIS OF A BOILING WATER REACTOR CORE, FOR VARIOUS POWER DISTRIBUTIONS, USING COMPUTER CODE THABNA

by

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Abstract

The core of a Boiling Water Reactor may see different power distributions during its operational life. How some of the typical power distributions affect some of the thermal hydraulic parameters such as pressure drop, Minimum Critical Heat Flux Ratio, void distribution etc. has been studied using computer code THANMA. The effect of an increase in the leakage flow has also been analysed.

NOMENCLATURE

P_f - axial power factor
X - Steam quality, weight fraction
α - Void fraction
CHF - Critical Heat Flux

I. INTRODUCTION

In a Pressurised Water Reactor the axial neutron flux distribution is basically sinusoidal, symmetrical about the central plane of core. In a Boiling Water Reactor with upward coolant flow, voids increase with height. This causes the neutron flux and hence heat flux to peak below the core mid plane. This flux distortion can be partially compensated by the insertion of control rods from the bottom. Depending on the amount of insertion of control rod, the axial power peaking may occur above the core mid plane or below the core mid plane. The effect of these flux distributions on the thermal hydraulic parameters of the Tarapur reactor, has been studied using the computer programme THABNA and the results are presented in this paper. The affect of an increase in the leakage flow fraction, i.e. the fraction of total flow which bypasses the fuel assembly, on the thermal hydraulic performance of the core has also been studied.

II. DESCRIPTION OF THE REACTOR CORE

Each of the two Boiling Water Reactors at Tarapur has 284 fuel assemblies housed in a pressure vessel. The cross-section of a typical fuel assembly along with a control rod is shown in fig. 1a. Each fuel assembly has 36 equal diameter fuel rods arranged in square lattice within a shroud of square cross section. These rods are attached to an upper and a lower tie plate with seven spacers in between. The assembly rests on support casting as shown in fig. 1b. The support casting is orificed to reduce the flow dependency on power by increasing the fraction of fuel assembly pressure drop which is power independent (single phase pressure drop at inlet). Peripheral fuel assemblies which generate less power, have more restrictive orifices than the interior ones. The total coolant flow which enters the core, distributes between the various fuel assemblies on the basis of equal pressure drop in all the assemblies. Because of the clearances and tolerances of various components in the flow path between the pressure vessel inlet and inlet to fuel assemblies, part of the coolant bypasses the fuel assemblies as shown in fig. 1b. This is called leakage flow. This leakage flow mixes with the active coolant (i.e. coolant flowing through fuel assemblies) at the core outlet. The leakage flow may change with time because of change in clearances caused by in-reactor creep.

III. COMPUTER PROGRAMME THABNA

The analysis has been carried out using the computer code THABNA. A detailed description of the code is given in reference 1. For the analysis, the fuel assemblies in the core are divided into a number of channel types having same axial and radial power factors and other thermal hydraulic characteristics. The code has now been modified to accommodate a maximum of 12 channel types and 24 axial nodes for each channel. Results of analysis, carried out using the code THABNA, are fed as input into a computer programme to evaluate physics parameters. These calculated values of physics parameters for some typical cases were compared with the values obtained from direct in-core measurements. A good agreement between the two values was observed. A beief description of the code follows.

Either the total core flow or the core pressure drop can be specified as input for the code. When one of the two is specified as input the other is calculated. The coolant flow distribution between the various fuel assemblies is estimated on the basis of equal pressure drop in all the assemblies. For each channel type, the code carries out a node by node calculation from bottom to top. The following pressure drop components are taken into account in calculating the total pressure drop.

1. Friction pressure drop (in non-boiling and boiling region)
2. Local pressure drop (in inlet section, spacers and upper tie plate)
3. Acceleration pressure drop
4. Elevation pressure drop

The coolant flow through the various channel types is adjusted so that the core pressure drop or the total core flow agrees with the specified input value, within the convergence limits specified. The steam quality, void fraction, Critical Heat Flux (CHF) etc. are calculated by the code.

IV. ANALYSIS

The present analysis has been carried out for a core power of 700 MWt. Twelve channel types, each with 20 nodes have been considered. The average pressure of coolant is 1030 psia and inlet enthalpy is 506.1 Btu/lb. For a total core flow of 2.92×10^7 lbs/hr, the analysis has been carried out to study the effect of (1) change in axial power distribution and (2) change in leakage flow. Keeping the leakage flow constant at 10% of total flow, the effect of changing the axial flux distribution in such a way that the peak shifts from near the bottom of the core to near the top has been analysed. For both the cases the total core flow is assumed to be the same. This assumption is justified because, as indicated in the next section, the differences between the pressure drops for the two cases are insignificant when compared to the total pressure drop in the circuit. Next, for a bottom-peaking axial flux distribution, the effect of a change in the leakage flow from 10% to 20% and 30% was examined. The data used for the analysis is given in Table 1. The results of the analysis are discussed in the next section.

V. DISCUSSION OF RESULTS

1. Effect of Variation in Axial Heat Flux Distribution: The analysis indicates that for a given total core flow, the change in heat flux from bottom peaking to top peaking has only a very minor influence on the coolant flow distribution through the various channels. For channel types 1 to 7, the flow marginally increases whereas for channel types 8 to 12 it decreases slightly. This results in a corresponding decrease in exit steam quality for channel types 1 to 7 and increase in quality for channel types 8 to 12. The variation of steam quality, void fraction and Critical Heat Flux Ratio (CHFR), i.e. ratio of CHF to surface heat flux, along the fuel channel for top peaking axial flux distribution is compared with that for bottom peaking flux in fig. 2. The curves in fig. 2 are for channel type 1. The curves for other channel types also exhibit similar trends.

The non-boiling length, component pressure drops and Minimum Critical Heat Flux Ratio (MCHFR) for the two cases are compared in Table II. For top peaking flux distribution, the non boiling length increases considerably. This has the effect of reducing the voids and increasing the average coolant density in the core. Consequently, the elevation pressure drop also increases. However, the friction pressure drop and local pressure drop decrease. This is because of reduction in boiling length (over which two phase flow occurs). For top-peaking flux, the total core pressure drop reduces to 25.2 psi as against 26.67 psi for bottom peaking case. The MCHFR also reduces for the top-peaking case. Also the location of MCHFR shifts closer to the fuel channel exit.

2. Effect of Variation in Leakage Flow

For a total coolant flow rate of 29.2×10^6 lbs/hr, the effect of variation of leakage flow rate from 10% to 20% and 30% has been analysed for bottom peaking axial flux. Since, the total core flow is assumed to be constant as the leakage flow increases, the amount of coolant which is available for cooling the fuel rods decreases. Consequently, the steam quality at various axial nodes as well as at fuel channel exit increases as the leakage flow increases. This has the effect of reducing the MCHFR for all the channel types as well as the overall MCHFR for the core as a whole.

As leakage flow fraction increases both friction and local pressure drop components decrease. This is because the reduction caused by reduced coolant velocity through the fuel assemblies more than offsets any increase due to the higher values of two-phase pressure drop multiplier caused by increase in steam quality. The elevation pressure drop also decreases because of reduced average density of coolant over the channel. Since, all the component pressure drops decrease, as leakage flow increases, the total core pressure drop also decreases. As a consequence of increased leakage flow, even though the steam quality at fuel channel exit increases, the quality at core exit remains constant since the total core flow remains the same. The total core pressure drop, core MCHFR and core exit quality for various leakage flows are given in Table III.

REFERENCE

1. V. Venkat Raj, A.K. Anand, D. Saha, "THABNA-A Computer Programme for the Thermal Hydraulic Analysis of Boiling Nuclear Assemblies". Paper No. C-4, Second National Heat and Mass Transfer Conference, I.I.T., Kanpur.

TABLE-I

INPUT DATA

Channel type	1	2	3	4	5	6	7	8	9	10	11	12
No.of channels in each type	4	28	4	28	4	52	4	60	4	4	4	88
Radial Power Factor	1.467	1.433	1.432	1.384	1.368	1.144	1.324	1.129	0.709	0.669	0.779	0.533
Heated Length (ft)	11.8542	11.8542	12.0	12.0	11.8542	11.8542	12.0	12.0	11.8542	11.8542	12.0	12.0
Unheated Length(ft)	0.9806	0.9806	0.8348	0.8348	0.9806	0.9806	0.8348	0.8348	0.9806	0. 9806	0.8348	0.8348

The axial flux distribution for channel type 1 is given in fig.2.
The axial flux distribution for other channel types is more or less similar.

TABLE-II

Channel Type	Non-boiling length(ft)		Friction pressure drop(psi)		Local Pressure drop (psi)		Acceleration pressure drop (psi)		Elevation pressure drop(psi)		Total pressure drop(psi)		MCHFR	
	Bottom	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom	Top
1	0.483	2.626	8.331	6.61	15.14	14.99	0.572	0.572	2.62	3.03	26.67	25.2	3.12	2.34
2	0.554	2.868	8.20	6.45	15.27	15.13	0.545	0.544	2.66	3.08	26.67	25.2	3.14	2.44
3	0.682	2.974	8.06	6.32	15.40	15.25	0.544	0.544	2.66	3.09	26.67	25.2	3.25	2.52
4	0.741	2.878	7.71	6.18	15.69	15.40	0.5053	0.504	2.76	3.12	26.67	25.2	3.42	2.67
5	0.791	3.195	7.95	6.25	15.47	15.29	0.489	0.487	2.76	3.17	26.67	25.2	3.33	[illegible]
6	1.36	4.422	5.68	5.32	16.61	16.17	0.268	0.267	3.11	3.44	26.67	25.2	4.03	3.55
7	0.896	3.320	7.53	5.95	15.85	15.58	0.4520	0.449	2.83	3.22	26.67	25.2	3.49	2.67
8	1.622	4.298	6.39	5.21	16.86	16.30	0.255	0.257	3.17	3.44	26.67	25.2	4.32	3.83
9	1.155	3.791	2.28	1.71	21.37	20.11	0.123	0.124	2.89	3.26	26.67	25.2	4.90	4.87
10	1.278	4.159	2.14	1.60	21.45	20.15	0.105	0.107	2.98	3.34	26.67	25.2	5.09	5.08
11	1.023	3.256	2.49	1.88	21.24	20.03	0.154	0.154	2.78	3.14	26.67	25.2	4.58	4.50
12	2.937	5.104	1.56	1.28	21.67	20.30	0.039	0.043	3.40	3.57	26.67	25.2	7.09	7.05

TABLE-III

Leakage flow (%)	Total core Pressure Drop(psi)	Core MCHFR	Core exit quality
10	26.67	3.12	0.0666
20	22.75	2.99	0.0666
30	19.08	2.87	0.0666

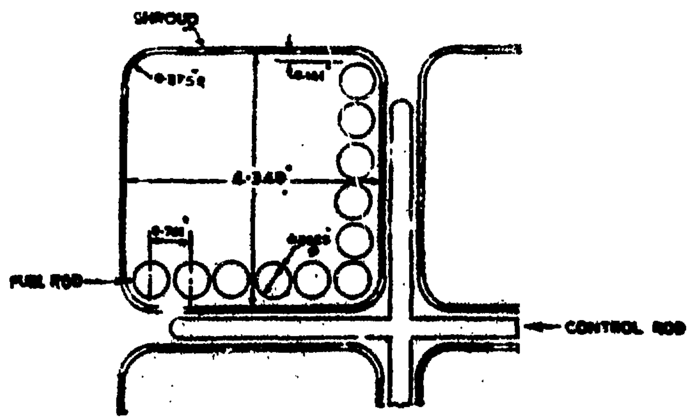
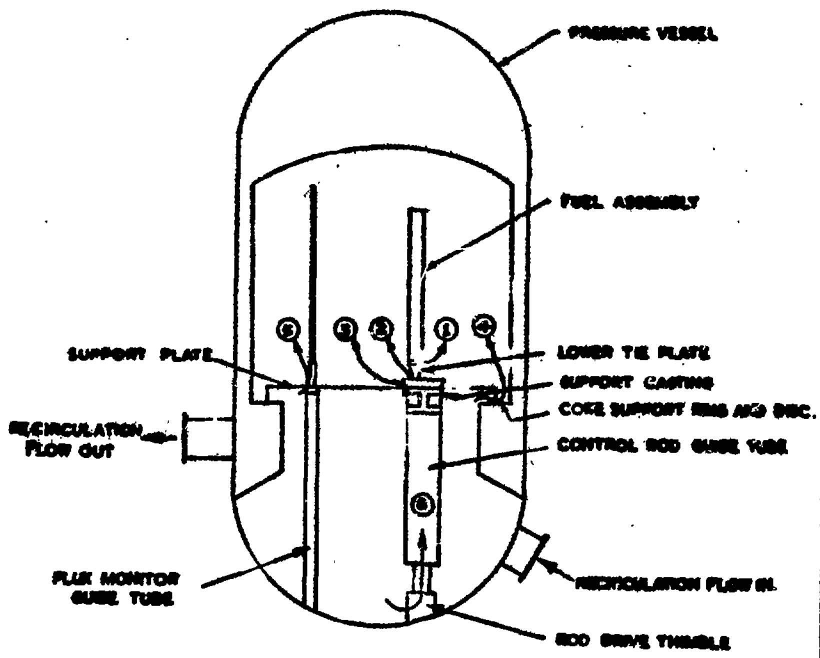


FIG.1a. FUEL ASSEMBLY AND CONTROL ROD



ARROWS (1 THRU 6) INDICATE LEAKAGE FLOW PATHS

FIG.1b. LEAKAGE FLOW PATHS

AXIAL VARIATION OF POWER FACTOR, EQUILIBRIUM QUALITY, VOID FRACTION AND CRITICAL HEAT FLUX RATIO IN CHANNEL-1

FIGURE-2