HEFAT2010
7th International Conference on Heat Transfer, Fluid Mechanics and Thermodynamics
19-21 July 2010
Antalya, Turkey

A PRACTICAL TOOL FOR QUICK LIGHT WATER REACTOR CORE DESIGN CALCULATIONS – MODIFIED ZEBRA CODE

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ABSTRACT

In nuclear reactor cores, coolant fluid to remove heat generated in the fuel rods is used. When coolant passes through the fuel bundles, it loses some fraction of its pressure because of acceleration, elevation and friction. Besides, spacer grids between fuel rods also cause pressure drop in the core since they behave like a barrier to the flow. Pump in the coolant loop should compensate this pressure loss to send adequate coolant to the core to remove heat from the system. Pump power required can be estimated by calculating the pressure loss in the core. In this study, ZEBRA code is modified and the spacer grid pressure drop calculation feature is added to its solution scheme. The modified code is used to model heat transfer experiments and the code solutions are compared with the experimental measurements. The comparisons show that the code results and the experimental data for the pressure drop are in good agreement. In addition, a closed channel of a Boiling Water Reactor (BWR) is modelled with ZEBRA and the effect of spacer grids on pressure loss is investigated.

INTRODUCTION

Nuclear power plants provide about 15% of the world's electricity [1]. In nuclear reactors, heat is generated in the fuel rods and it is removed by a coolant fluid for generating shaft work. Shaft work is used to generate electricity in the generator.

Light water is the most commonly used coolant. When water passes through the fuel bundles, it loses some fraction of its pressure. Pump in the coolant loop compensates this pressure loss to send adequate coolant to the core for heat removal. Pump power required can be estimated by calculating the pressure loss in the core. In order to keep the system stable and safe, the flow rate that must be supplied by the pumps should be determined. For this, pump power required should be calculated accurately for the preliminary design of a nuclear reactor.

Coolant water in the reactor core loses some fraction of its pressure because of change in the hydrostatic pressure due to the increase in level, the acceleration of the coolant and friction and form losses in channels [2]. Besides, spacer grids that are used to maintain fuel rod spacing between fuel rods also cause pressure drop in the core since they change the flow area causing contraction then expansion of the flow. The spacers can contribute up to 50% of the pressure drop in the core [2]. Kinetic energy of the fluid is partially lost through the spacer grid because of viscous dissipation [3]. The majority of this loss is due to inertia effects that are eventually dissipated by the shear stresses within the fluid [3] and this leads to unrecoverable pressure drop. A general method of predicting spacer grid pressure drop is expressed by Rehme as follows [2]:

$$\Delta P_{spacer} = -C_v \varepsilon^2 \frac{G^2 v}{g_c} \quad (1)$$

ε is a geometry-dependent parameter representing the ratio of the projected grid cross-section area in the rod bundle, to the undisturbed rod bundle flow area. C_v is spacer form loss coefficient, and values of C_v are determined experimentally for different spacer designs.

For the preliminary design analysis of the nuclear reactor cores, simple, one dimensional, steady-state closed channel analysis calculations are performed. For these analyses computer codes are used to simulate the thermal hydraulic behaviour of the reactor core channel. The ZEBRA code used in this study is an example of such closed channel analysis tools used basically for educational purposes and quick light water reactor core design analysis. The code uses a one-dimensional single or two phase convection in the axial direction and one-dimensional heat conduction analysis in the radial direction to calculate coolant and fuel rod temperatures and pressure drop in the channel. The code contains models to calculate the temperature and pressure dependent properties of the coolant and reactor materials.

The objective of this study is to modify ZEBRA code such that the spacer grid pressure drop calculation feature is implemented to the solution scheme. Columbia University's two-phase flow heat transfer experiment's data is used for validation [4].

NOMENCLATURE

C_v		Loss coefficient
ε		Ratio of the projected grid cross-section area
G	[lb/hr-ft^2]	Mass flux
ΔP	[psi]	Pressure drop
v	[lb/ft^3]	Specific volume

METHODOLOGY

Methodology of this study includes implementation of spacer grid pressure drop capability into ZEBRA code and validation of the modified program with using experimental data in literature. The procedure performed with the scope of this study is explained in the following sections.

Implementation of Spacer Grid Pressure Drop Calculation

The implementation of spacer grid pressure drop capability into ZEBRA code contains a series of programming issues and validation. Modifying a code requires careful investigation of the code structure and the calculational scheme. Calculations of the core design parameters performed by ZEBRA are tightly coupled with the calculation of each one of them, therefore the proposed modification was performed after a detailed examination of the code and proper calculational scheme was generated for the pressure loss due to spacer grids.

The principal part of the ZEBRA computer program consists of a preliminary calculation section and a primary reiterative loop or do-loop. There are several supporting subroutines that contain expressions and processes to perform the closed channel analysis. A new subroutine (DP_GR_s.FOR) is encoded which includes expression in equation (1) to perform spacer grid pressure drop calculation.

Input deck of the program includes input parameters which specify the properties of a nuclear power plant or a test facility. Information in the input deck is read by the main program and calculations are done using this information. The required information for ZEBRA is arranged into three major groupings as listed below:

1. Physical parameters
2. Option selection
3. Data required by options

The physical parameters include data used for the geometry, heat generation and thermal hydraulic properties. The options and data required for the options include the details of the special models used in the code. A new option is added to the input deck and the code and input deck are rearranged for additional parameters.

There are different types of spacer grids located on a fuel bundle. Therefore, the code and input deck are designed to be capable of performing spacer grid pressure drop calculations for a fuel bundle which is assembled with more than one type of spacer grid. All new parameters added to the code are defined as *allocatable* to offer the usage of different spacer grids with one fuel bundle. Parameters that specify the types and locations of spacer grids in the input deck are as follows:

Spacer Grid Data Cards

- First spacer grid data card: Number of Spacer Grid Types (ntype)
- Second spacer grid data card: Number of Spacer Grids in Type i (i = 1, ntype)
- Third spacer grid data card: C_v, ε, node 1, node 2,…., node n.

The parameters are described further on Figure 1 which presents the input deck of modified ZEBRA. The input deck shown contains information about the Columbia University loop.

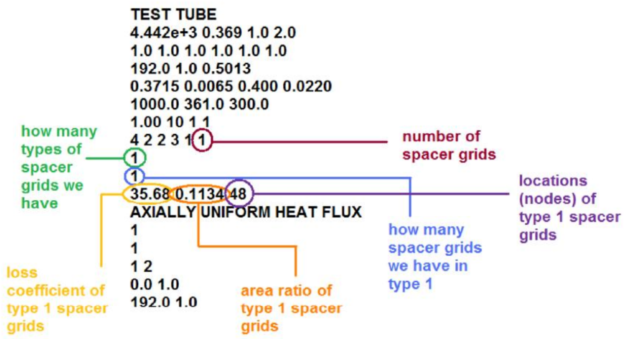

Figure 1. Input deck for Columbia University's loop

During the execution of the program, when the pressure loss due to spacer grid placed at specified axial location is required, the added subroutine is called and spacer grid pressure drop is calculated for the spacer grid at the specified location. Then system pressure and thermophysical properties are recalculated and all other calculations in the program are affected by this calculation.

Validation of the Modified Program

Data obtained by Columbia University's two-phase flow heat transfer experiment is used for the validation of modified ZEBRA. As a result of the experiment, 200 critical heat fluxes obtained with boiling water flow in 0.400-in-id tubes, uniformly heated over lengths of 8, 12, and 16 ft, at mass velocities between 0.9 and 7.2 x 10^6 lb/hr-ft^2 [4]. Swirl flow, induced by a full-length tape or by two tape inserts, 2 in. long and 8 ft apart, and axial flow were studied [4]. The schematic of Columbia University loop is shown in Figure 2.

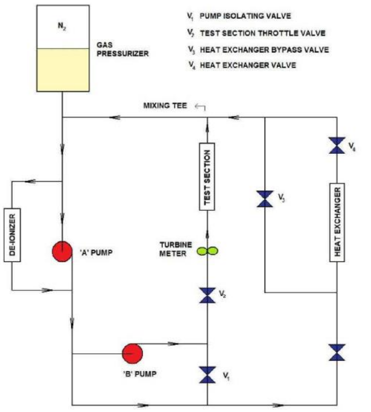

Figure 2. Flow schematic of Columbia University loop [4]

In the experiment, there are swirl tapes which cause form losses similar to form losses caused by the spacer grids. Data obtained with twisted tape segments 2 in. long and 0.015 in. thick with a 180-deg. twist are used for the validation. Some of the data used are tabulated in Table 1. The data listed in Table 1 are selected to cover the data range of different runs of the experiment.

Table 1. Columbia University Loop Experimental Data [4]

Run No	Inlet Temp. (˚F)	Power Input (MW)	Mass Flux 10^6 lb/hr-ft^2	Pressure Drop (psi)
246	442	0.137	1.05	18
169	531	0.152	2.08	45
174	527	0.194	3.12	74
259	358	0.461	7.04	91
235	512	0.203	2.93	68
250	518	0.234	3.98	94
236	497	0.25	3.62	85
255	361	0.369	5.09	61

The modified code is executed as if there are no swirl tapes for the conditions tabulated in Table 1. The difference between experimental pressure loss and pressure loss without swirl tapes calculated by the program is evaluated. This pressure loss difference originates from swirl tapes. Since the size of the tape and channel cross-section area are definite, ratio of the projected grid cross-section area is calculated as 0.1134. The loss coefficient of the tape is calculated for each different mass flow rate-inlet temperature cases by using Equation (1).

Table 2. Modelled and Experimental Pressure Drop Values and Loss Coefficients

Run No	Modelled PD without SG (psi)	Experimental PD (psi) [4]	Loss Coefficient
246	15	18	1176.3
169	38	45	699.42
174	67	74	312.3
259	81	91	95.151
235	60	68	404.71
250	93	94	27.417
236	80	85	165.71
255	59	61	35.683

The modified code is executed with and without spacer grids for the cases described in Table 1 by using the calculated loss coefficients and the ratio of the projected grid cross-section area. The location of the spacer grid is defined as the midplane of the channel.

The physical parameters and options used for modelling Columbia University loop are listed below:

- Fraction of total heat generated within the fuel rod=1.00
- Size of axial nodes=2 inches
- Nuclear hot channel factors=1.00
- Engineering hot channel factors=1.00
- Total length of active core=192 inches
- Number of fuel rods=1
- Fuel rod pitch=0.5013 inches
- Fuel pellet diameter=0.3715 inches
- Fuel-clad gap size-cold dimension=0.0065 inches
- Cladding thickness=0.0220 inches
- Core coolant pressure=1000 psi
- Fuel rod internal gas pressure=300 psi
- Fraction of the total reactor power to be used in analysis=1.00
- Number of fuel divisions to be used in the fuel centreline temperature analysis=10
- Square lattice
- Cladding: Zircaloy-2
- Axially uniform heat flux
- Seider-Tate correlation for subcooled forced convection
- Ross & Stoute correlation for fuel-to-clad gap conductance
- Janssen-Levy (GE-1962) correlation for critical heat flux
- Number of axial locations for spacer grids=1

Core inlet temperature, core coolant flow and heat output values change according to the experimental data tabulated in Table 1.

RESULTS

Columbia University's loop is modelled by using the modified program for cases with and without spacer grids using the loss coefficients listed in Table 2. One sample result of the calculations is presented in Figure 3. The location and effect of the spacer grid on pressure distribution (x=8 ft) are obvious and can clearly be seen on graph shown in Figure 3.

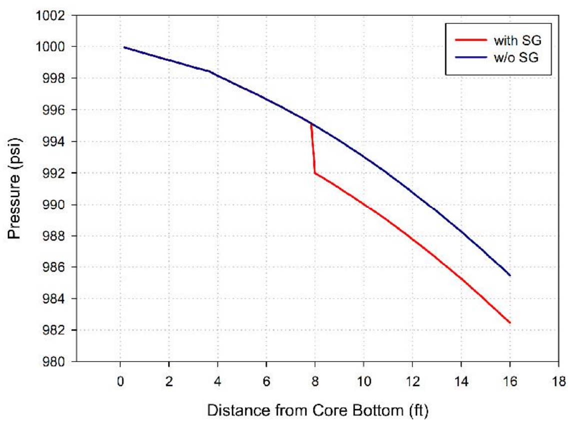

Figure 3. Pressure distributions with and without spacer grids

Experimental versus modelled pressure drop values are depicted in Figure 4. Figure 4 indicates that once spacer grid model is added to the code, the agreement between the experimental data and the results of the code calculations improves.

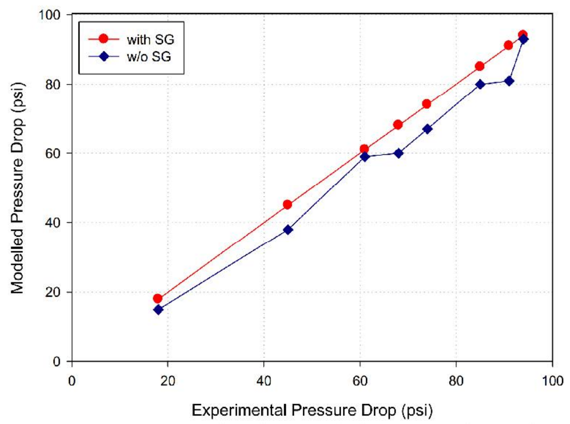

Figure 4. Experimental and numerical pressure drop values for various experimental data

The modified code is also used to simulate a closed channel of Susquehanna Boiling Water Reactor.

Coolant pressure distribution of Susquehanna BWR is depicted in Figure 5. Effect of spacer grids on coolant pressure can clearly be seen as abrupt decreases depicted in Figure 5.

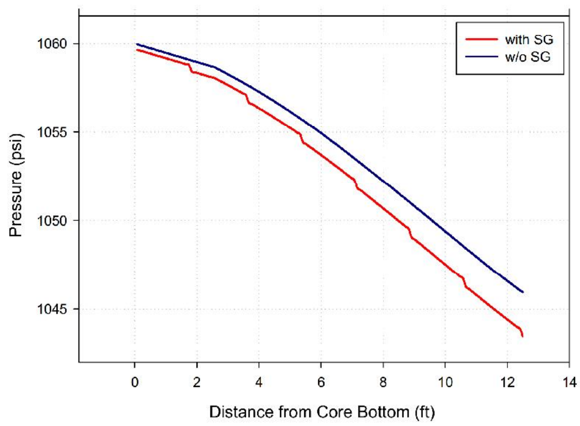

Figure 5. Pressure distribution of Susquehanna BWR modelled by ZEBRA code – with and without spacer grid

The total pressure drop calculated for the channel of Susquehanna with spacer grids is 17 psia. Without spacer grids, this value is 14 psia. Reported pressure drop for such a channel is about 26 psia [2].

CONCLUSION

The ZEBRA code is modified to implement spacer grid pressure drop calculations into the solution scheme of ZEBRA. Experimental data of the Columbia University loop is used for the validation of the modified program. The experimental and numerical results are compared and the comparisons show that the code results and the experimental data for the pressure drop are in good agreement. Modified ZEBRA yields more accurate results for thermal hydraulic analysis of a nuclear reactor. %11.5 higher accuracy is achieved for Susquehanna BWR. The difference between reported and calculated pressure drop values of Susquehanna BWR arises because of the deficit in the solution scheme of ZEBRA which excludes lateral flow between channels, computations of pressure losses in pipes, lower plenum and upper plenum of the core. Further modifications should be done in the future to increase accuracy of the ZEBRA code.

REFERENCES

[1] "Nuclear Power in the World Today", web address: http://www.world-nuclear.org/info/inf01.html, date retrieved: December 1st, 2009.

[2] Elements of Nuclear Reactor Design, Lecture Notes by L.E. Hochreiter, S. Ergun, G.E. Robinson, The Pennsylvania State University 2008.

[3] Munson B.R., Young D.F., Okiishi T.H., Fundamentals of Fluid Mechanics, 5th edition, John Wiley & Sons, pp. 438-439, 2006.

[4] Matzner B., Casterline J.E., Moeck E.O., Wickhammer G.A., Critical Heat Flux at 1000 psi With and Without Swirl Promoters, *ASME Publication*, 65-WA/HT-30.