Simplified Technique for 3-Dimensional Core T/H Model in CANDU6 Transient Simulation

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Abstract

Simplified approach has been adopted for the prediction of the thermal behavior of CANDU reactor core during power transients. Based on the assumption that the ratio of mass flow rate for each core channel does not vary during the transient, quasy-steady state analysis technique is applied with predicted core inlet boundary conditions(total mass flow rate and specific enthalpy). For restricted transient case, the presented method shows functionally reasonable estimation of core thermal behavior which could be implemented in the fast running reactor simulation program.

1. Introduction

Because thermal behavior of reactor core during reactor operation affects the overall system behavior and safety characteristics of reactor core itself, detailed thermal-hydraulic analysis on the reactor core is required in the steady state and transient state cases. Also, the realistic simulation of nuclear power plant for operational transients needs treatment of detailed core thermal analysis for coupling with reactor kinetic behavior, i.e., reactivity feedback.

This study aims at developing a simplified technique for predicting regional core thermal behavior of CANDU system during operational transients, which can be used in reactor kinetics calculation. Considering application range, various assumptions are used for fast running simulation.

2. Model Description

0 Reactor Core Geometrical Model

CANDU type reactor, with its separate core flow channel characteristics, needs channel grouping, in which each group is assumed to have same thermal-hydraulc characteristics and is represented as a typical single channel. In the present application, coupling with reactor kinetics, the core is horizontally divided into seven regions equivalent to the control zones in CANDU core, and included channels in each zone is analyzed as a single flow channel. The core is also axially divided into several nodes for thermal-hydraulic calculations. In addition, one half of the core is analyzed by considering the core is symmetric(Also, the applied system code can only treat one loop.) The core geometrical model is showen in Fig.1.

0 Thermal-hydraulic Model

The development of present model is not to analyze detail aspects of the core thermal behavior, but to examine overall space-dependent behavior of core with fast prediction capability. Therefore, the following simplifying assumptions are adopted.

- For slow transient without significant change of properties with time, quasi-steady state approach is acceptable.
- For operational transients, especially power maneuvering cases, the realative flow rates among the channels seem not to vary significantly, which leads to assume that fuel channel flow distribution ratios at steady state are assumed to be maintained at transient
- In a real situation of reactor core, coolant density change across the channel exists and some boiling may occur in normal power operation. However, in operational transient range, the density change and boiling are usually negligible for the calculation results. Therefore, single phase approximation seems to be acceptable for present use, i.e., mass flow rate is assumed to be constant along the channel

Using the above assumptions, in a given fuel channel, simple energy balance equation is applied for each node with the inlet enthalpy obtained from system code at each time step. The pressure in a channel is uniform along the channel at a average value of inlet and outlet, which are also evaluated from system code. Evaluation of fuel and sheath temperature is performed with FEULPIN model, which uses lumped parameter technique and now is being used in system code.

0 Prediction of Zone Average Values

Primary heat transport system of CANDU6 consists of two independent loop and in a given loop core pass is bidirectional(figure-of-eight). Also, in the core, adjacent fuel channels have different flow directions. Therefore, each horizontally divided zone is represented as two bidirectional fuel channels, each representative in that direction. Therefore, thermal calculation in any zone is performed for each direction separately, and the zone average value is obtained by averaging.

3. Test and Results

The devised model has been incorporated into CANDU simulation computer code (DSNP) which covers operational transients with appropriate control logics. Test case of setback, power decrease from 100% full power to 60% full power with ramp rate of 0.4%/sec, was selected for functional validation of the present technique. The predicted core thermal behavior is shown in Fig.2, in which time dependent average fuel temperatures at central node are shown for zone 3 and zone 4, respectively.

The core average temperature variation, evaluated with single core average fuel channel, is also shown.

Even though the quantitative validation of the present model is not possible due to the lack of available data, the trends of temperature variation in the figures indicate that the present method is physically acceptable and seem to be applicable. Also, comparison between core average values from single channel approach with those from 3-D approach shows a reasonable agreement.

4. Conclusion and Discussion

The proposed simplified technique for 3-dimensional core T/H model for CANDU transient simulation seems to be physically reasonable in that it predicts the trend of core temperature variation in reasonable manner considering time dependent power change. Therefore, no reason why the present model could not applyed to the fast running CANDU transient simulation of operational transients is found now. The quantitative validation of the present model will be performed with T/H - reactor kinetics coupled scheme and with related data provided.

Some of the assumptions used in thermal-hydraulic model seem to be oversimplified, which may cause the predictions unrealistic in some transients. Therefore, it is recommended that the following items should be investigated further with additional data in parallel with the continuous validation efforts;

- incorporating momentum conservation equation
- adequate treatment of two-phase flow condition

References

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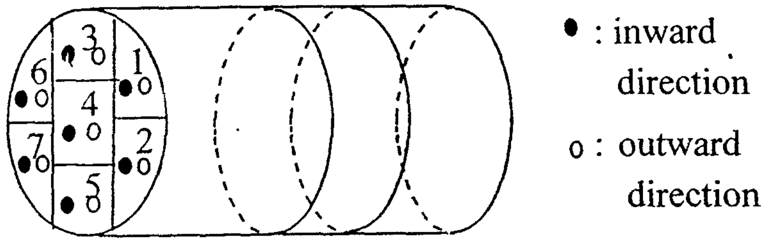

Fig.1 Geometrical Model of CANDU Core

Fig.2 Fuel Temp. Variation with Reactor Power Change at Central Node