Simulation of LOCA Power Transients of CANDU6 by SCAN/RELAP-CANDU coupled code system

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1. Introduction

As can be seen in the standalone application of RELAP-CANDU for LOCA analysis of CANDU-PHWR [1], the system thermal-hydraulic code alone cannot predict the transient behavior accurately. Therefore, best estimate neutronics and system thermal-hydraulic coupled code system is necessary to describe the transient behavior with higher accuracy and reliability.

The purpose of this research is to develop and test a coupled neutronics and thermal-hydraulics analysis code, SCAN (SNU CANDU-PHWR Neutronics) [2] and RELAP-CANDU, for transient analysis of CANDU-PHWR's. For this purpose, a spatial kinetics calculation module of SCAN, a 3-D CANDU-PHWR neutronics design and analysis code, is dynamically coupled with RELAP-CANDU, the system thermal-hydraulic code for CANDU-PHWR.

The performance of the coupled code system is examined by simulation of reactor power transients caused by a hypothetical loss of coolant accident (LOCA) in Wolsong units, which involves the insertion of positive void reactivity into the core in the course of transients. Specifically, a 40% reactor inlet header (RIH) break LOCA was assumed for the test of the SCAN/RELAP-CANDU coupled code system analysis.

2. Models of SCAN

2.1 Kinetics Calculation Model

The kinetics calculation model of SCAN code is composed of both UNM (Unified Nodal Method) [3]-based CMFD (Coarse Mesh Finite Difference) and FDM (Finite Difference Method) solutions to the time-dependent two-group (2G) diffusion equations. Two UNM options: NEM (Nodal Expansion Method) and ANM (Analytic nodal method), are implemented in the SCAN code. For the temporal discretization of the diffusion equations, the theta method with exponential transformation of the flux is applied.

2.2 Data Exchange Model

The dynamic linked library code, SCAN DLL, has been generated to build the coupled code system of SCAN and REALP-CANDU. For the generation of SCAN DLL, the SCAN code was modified to be workable for the Windows platform and re-formulated as a sub-routine of RELAP-CANDU code. The Adapter DLL has been generated for data exchange between these two individual codes. Fig. 1 illustrates the schematics of data exchange between SCAN and RELAP-CANDU codes. The fundamental data exchange model of the coupled code system is similar to the previous works, MARS/MASTER coupled code system for PWR [4].

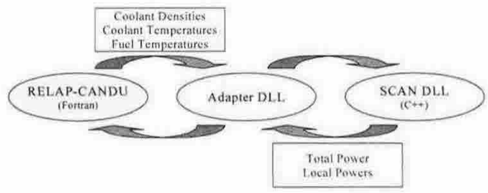

Figure 1. Feedback Data Exchange in SCAN/RELAP-CANDU

2.3 Data Mapping Model

In a CANDU6 reactor, the reactor core consists of 380 channels with 12 bundles in each channel. A total of 21,696 nodes are used for the full core power calculation with standard mesh configuration of 42x34x20 model in SCAN code. In contrast, a total of 48 nodes are used for thermal-hydraulic analysis by RELAP-CANDU. This makes it necessary to develop a node-to-node mapping module between thermal-hydraulic node and neutronics nodes in order to conduct exchange of feedback data between the two individual codes.

2.4 Reactor Trip Model

Power transient initiated by coolant voiding is terminated with insertion of shutoff rods (Shutdown System 1) or liquid poison (Shutdown System 2) as a result of trip signal. To estimate the time of the trip signal to be used, the rate-of-log power calculation module is developed using the ion chamber circuitry equations [5]. This module enables SCAN code not only to calculate the reactor trip time but also to initiate automatic drop of shutdown system 1

3. Benchmark Verification

3.1 Benchmark Problem

To qualify the performance of SCAN/RELAP-CANDU coupled code system, 40% RIH break LOCA at thermal-hydraulic channel 4 is analyzed. The homogeneous lattice and incremental cross sections are generated by WIMS-AECL and DRAGON codes using two-group model with ENDF/B-VI library, respectively. The power transients are estimated by using standard geometry of 42x34x20 model and measured data of SDS1 drop curve [6] for the pre-equilibrium core condition of Wolsong units.

3.2 Numerical Results

Fig. 2 compares the coolant density estimation of the RELAP-CANDU code with that of CATHENA code on condition that the applied transient power shape is the same between these two codes. In the simulation of LOCA power transients of CANDU-PHWR, the coolant density is the most influencing feedback parameter on the reactor power shape. The transient power behavior in Fig. 3 reflects this effect.

In Fig. 3, the transient power curve of SCAN/RELAP-CANDU coupled code system is compared with that of RFSP/CATHENA coupled code system. It is shown that the SCAN/RELAP-CANDU predicts power shape similar to RFSP/CATHENA with about 0.06 sec power delay for 40% RIH break LOCA. The trip initiation times of SDS1 calculated by SCAN/RELAP and RFSP/CATHENA code systems are 0.482 second and 0.425 second, respectively. It is also shown that NEM slightly underestimates the powers in comparison with the FDM after SDS1 insertion started.

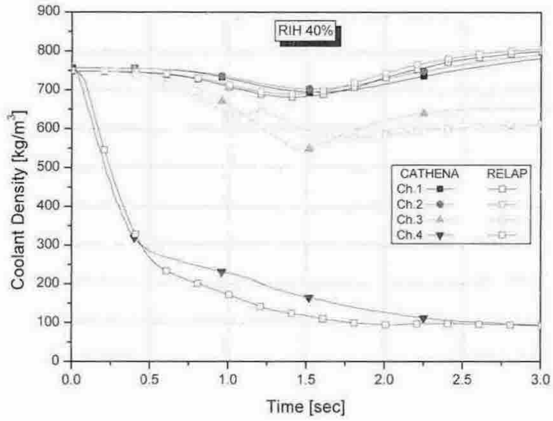

Figure 2. Coolant Density Variation for 40% RIH Break

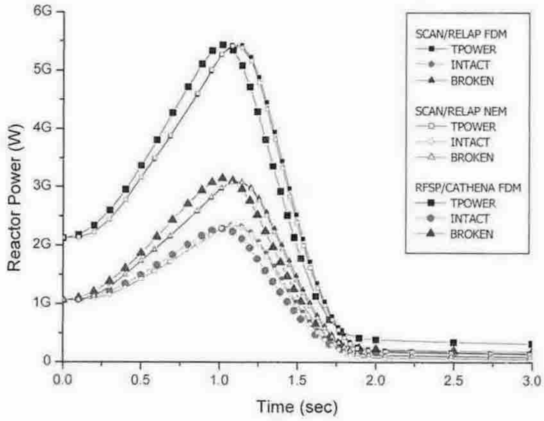

Figure 3. Transient Power Curve for 40% RIH Break

4. Conclusions

To make more accurate simulation for power transients of CANDU-PHWR, SCAN/RELAP-CANDU coupled code system is developed and its performance is tested in terms of the simulation of the positive void reactivity driven 40% RIH break LOCA. As was shown from the numerical results, the SCAN/RELAP-CANDU coupled code system may qualify as a transient analysis tool of CANDU-PHWR's. Further studies need be performed for verification and validation of the SCAN/RELAP-CANDU coupled code system.

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