Enhancement of Safety Analysis Reliability For A CANDU-6 Reactor Using RELAP-CANDU/SCAN Coupled Code System

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I. Introduction

In LOCA analysis of the CANDU reactor, the system thermal-hydraulic code, RELAP-CANDU [1], alone cannot predict the transient behavior accurately. Therefore, the best estimate neutronics and system thermal-hydraulic coupled code system is necessary to describe the transient behavior with higher accuracy and reliability. To perform on-line calculation of safety analysis for CANDU reactor, a coupled thermal hydraulics-neutronics code system was developed in such a way that the best-estimate thermal-hydraulic system code for CANDU reactor, RELAP-CANDU, is coupled with the full three-dimensional reactor core kinetic code,

II. RELAP-CANDU/SCAN Coupled Code System

1. Thermal-hydraulic Model Improvement

Because the RELAP5 computer code has been developed for best-estimate transient simulation of a pressurized water reactor and its associated systems, it is difficult to simulate properly the thermal-hydraulic behaviors of the horizontal multi-channels of a CANDU reactor. Therefore, there are a lot of studies to improve the RELAP5 code applicable to a CANDU reactor such as: (i) the addition of correlations for horizontal fuel bundle, (ii) the addition of a pressure tube (PT) deformation model with PT burst criterion, and (iii) the addition of a CANDU pump homologous curves.

2. Kinetics Calculation Model

The kinetics calculation model of SCAN code [2] is composed of both UNM (Unified Nodal Method)-based CMFD (Coarse Mesh Finite Difference) and FDM (Finite Difference Method) solutions to the time-dependent two-group (2G) diffusion equations. And two UNM options of 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.

3. 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. As the trip signal used to be generated from the code prediction by fluxes at detector positions, the rate-of-log power calculation module is developed using the ion chamber circuitry equations. Using this module, SCAN code can not only initiate automatic drop of shutdown system 1, but also calculate the reactor trip time.

4. 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 adequate for the Windows platform and re-formulated as a sub-routine of RELAP-CANDU code. And the Adapter DLL has been generated for data exchange between those two individual codes. Fig. 1 illustrates the feedback data exchange schematic of RELAP-CANDU/SCAN coupled code system.

5. Data Mapping Model

In a CANDU6 reactor, the reactor core consists of 380 channels with 12 bundles in each channel. And the total of 21,696 nodes is used for the full core power calculation with standard mesh configuration of 42x34x20 model in SCAN code. In a meanwhile, the total of 48 nodes is used for thermal-hydraulic variable estimation in RELAP-CANDU conventionally. Therefore, the development of the general node-to-node mapping module between nodes for thermal-hydraulic calculation and nodes for neutronic calculation is conducted for reliable exchange of feedback data between those two individual codes.

III. BENCHMARK VERIFICATION

To verify the reliability of RELAP-CANDU/SCAN coupled-codes, a series of benchmark problems such as 40% RIH break, 100% ROH break, and 50% PS (Pump Suction) break were performed in such a way to compare with those of CATHENA-RFSP codes [3, 4]. For the 40% RIH break, the changes of coolant densities for each pump model are similar as shown in Fig. 2. In the case of

100% ROH break, the transient reactor power calculated by SCAN is compared with that of RFSP in Fig. 3.

IV. Sensitivity Study For The Coupled Code System

To explore the sensitivity of RELAP-CANDU/SCAN coupled code system, assessment of 35% RIH break [5] has conducted. Fig. 4 shows the fuel sheath temperature of critical path in the broken loop for two cases: (i) the results calculated by RELAP-CANDU/SCAN coupled code system (Case 1) and (ii) the results by RELAP-CANDU with pre-calculated power (Case 2).

IV. CONCLUSIONS

To get more accurate predictions for power transients of CANDU reactor, the coupled RELAP-CANDU/SCAN code system is developed and verified through the simulation of the positive void reactivity driven RIH 40%, ROH100% and PS50% break LOCA. As the results, the RELAP-CANDU/SCAN coupled code system may possibly be used as a specific transient analysis tool for CANDU reactor.

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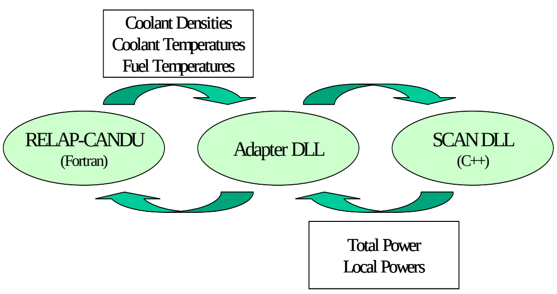

Figure 1. RELAP- CANDU/SCAN Coupled Code Model

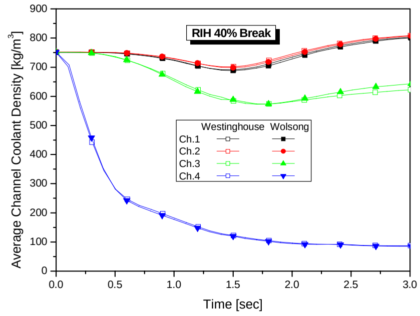

Figure 2. Coolant densities for RIH 40% break

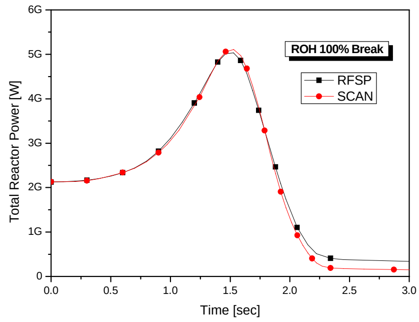

Figure 3. Reactor power for ROH 100% break

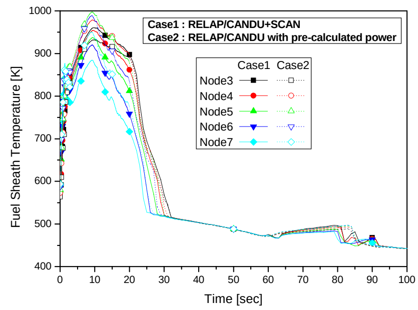

Figure 4. Fuel Sheath Temperature in the critical path