Assessment of LOCA with Loss of Class IV Power for CANDU-6 Reactors Using RELAP-CANDU/SCAN Coupled Code System

S. H. Hwang[a], M. W. Kim[a] and H. J. Kim[a], I. S. Hong[b] and C. H. Kim[b]
[a] *Korea Institute of Nuclear Safety, P.O.Box 114, Yuseong, Daejeon, Korea,* femoth@freechal.com
[b] *Seoul National University, San 56-1, Shinlim-dong, Kwanak-Gu, Seoul , Korea*

1. INTRODUCTION

Recently, there is an effort to improve the accuracy and reality in the transient simulation of nuclear power plants. In the prediction of the system transient, the system code simulates the system transient using the power transient curve predicted from the reactor core physics code. However, the pre-calculated power curve could not adequately predict the behavior of power distribution during transient since the coolant density change has influence on the power shape due to the change of the void reactivity. Therefore, the consolidation between the reactor core physics code and the system thermal-hydraulic code takes into consideration to predict more accurate and realistic for the transient simulation

In this regard, there are two codes are developed to assess the safety of CANDU reactor. RELAP-CANDU is a thermal-hydraulic system code for CANDU reactors developed on the basis of RELAP5/MOD3 in such a way to modify inside model for simulating the thermal-hydraulic characteristics of horizontal type reactors. SCAN (SNU CANDU-PHWR Neutronics) is a three dimensional neutronics nodal code to simulate the core physics characteristics for CANDU reactors. To couple SCAN code with RELAP-CANDU code, SCAN code was improved as a spatial kinetics calculation module in such a way to generate a SCAN DLL (dynamic linked library version of SCAN). The coupled code system, RELAP-CANDU/SCAN, enables real-time feedback calculations between thermal-hydraulic variables of RELAP-CANDU and reactor powers of SCAN.

To verify the reliability of RELAP-CANDU/SCAN coupled code system, an assessment of 40% reactor inlet header (RIH) break loss of coolant accident (LOCA) with loss of Class IV power (LOP) for Wolsong Unit 2 conducted using RELAP/CANDU-SCAN coupled system. The LOCA with LOP is one of GAI (Generic Action Items) for CANDU reactors issued by CNSC (Canadian Nuclear Safety Commission) and IAEA (International Atomic Energy Agency).

2. METHODS AND RESULTS

2.1 RELAP-CANDU/SCAN Coupled System

The coupled RELAP-CANDU and SCAN codes are developed to operate under the Windows operating system [1]. During the transient calculation, the thermal-hydraulic variables of the reactor core, such as coolant density, temperature and fuel temperature distributions, are determined at each calculating time step by RELAP-CANDU code and transferred in real-time based to SCAN code so that they can be used to update the group constants inside SCAN code. The power distribution newly obtained by SCAN thereafter and is transferred to RELAP-CANDU code in turn so as to calculate the thermal-hydraulic variables at the next time step.

2.2 Thermal-hydraulic Modeling[2]

The heat transport system (HTS) of Wolsong Unit 2 consists of two closed loops. The reactor core consists of 380 fuel channels including pressure tubes and the coolant flows with counter-current direction between neighboring fuel channels. To simulate thermal-hydraulic behavior, the reactor core is modeled with 4 simplified fuel channels, 4 coolant pumps, core inlet and outlet feeder pipe, feeder pipe header and inlet pipe of coolant as shown in Fig. 1.

Each simplified fuel channel represents 95 fuel channels. Steam generator (SG) tubes are modeled with heat structures to simulate heat transfer between primary and secondary side. The secondary side of SG consists of downcomer, riser, moisture separator, recirculation path and dome. Emergency core cooling system (ECCS) is modeled in three modes such as high pressure, intermediate pressure and low pressure injection

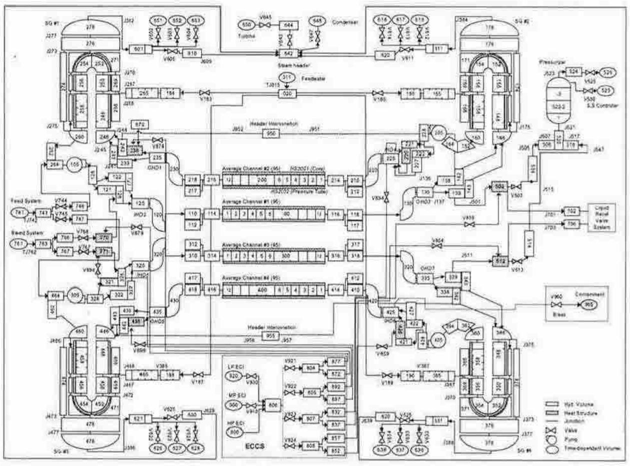

Fig. 1 RELAP/CANDU Nodalization of Wolsong Unit 2

The break takes place at the reactor inlet header (IHD) 8 in the Channel 4 and is modeled with trip valve and containment volume.

Main Steam Safety Valves are actuated with LOCA signal to be open to conduct the crash cooldown of primary system. LOCA signal is generated with the

synchronous actuation of HTS low pressure (less than 5.25MPa) during LOCA transients.
When the turbine is tripped (LOCA signal with delay time 5sec), the loss of class IV power occurred. Then the HTS pump and main feedwater pump (MFWP) are stopped and after 3 minutes the auxiliary feedwater pump (AFWP) is activated.

2.3 Initial Steady-State Conditions

In the steady-state calculation, the initial power is fixed and the axial power distribution is calculated by SCAN code. The initial thermal-hydraulic properties are well agreed with the design values for 103% power operation condition in the Wolsong Unit 2 Final Safety Analysis Report [3].

2.4 Transient Simulation

After the break is initiated, the HTS depressurized rapidly due to blow-down of the coolant through break pipe. The void is generated in the core due to the depressurization and makes an increase of the reactivity. Fig. 2 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). In Case 2, the axial power distribution shape is given as an input and is not changed throughout the calculation. But in Case 1 the realistic axial power distribution is calculated at each calculating time step. The calculated maximum fuel sheath temperature is 960K at 9.7sec in node 3 for Case 1 and 997.3K at 9.3sec in node 5 for Case 2, respectively. It means that the pre-calculated power methodology is more conservative rather than the present proposed methodology in terms of the safety margin for the fuel channel integrity.
The variation of axial power distribution for Case 2 is shown in Fig. 2. It shows that the axial power distribution varies according to void reactivity as time progresses. At initial condition, the axial power distribution shape has maximum value at the center and minimum values at the each end. But after break, the axial power distribution shape in the intact loop is flatted except for the each end. While the peak power in the broken loop is shifted near to the core entrance due to void reactivity effect.
Using RELAP-CANDU/SCAN coupled code system, the transient analyses are more accurate and realistic.

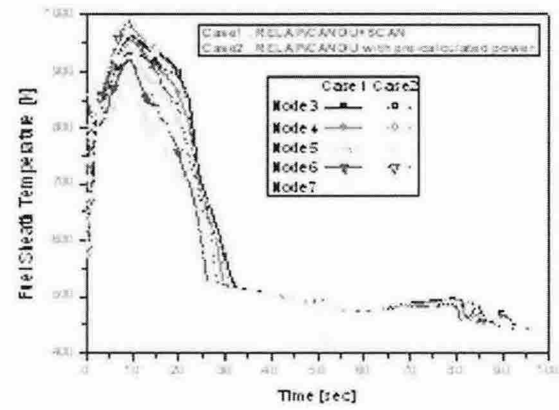

Fig. 2 Fuel Sheath Temperature in Ch. 4

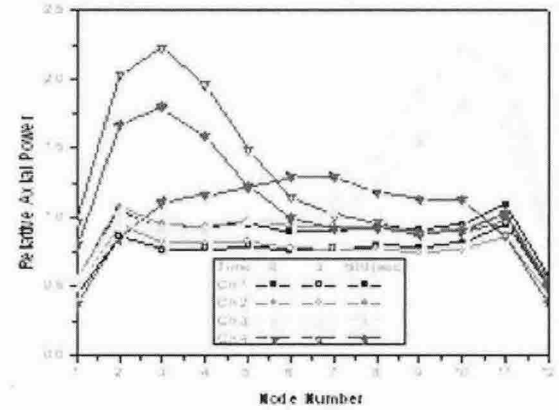

Fig. 3 Relative Axial Power Distributions in Case1

3. CONCLUSION

The RELAP-CANDU/SCAN coupled code system is developed to predict more accurate and realistic of the thermal-hydraulic characteristics in the transient condition. Using this coupled system, the generic safety issue (GSI),*40% RIH break LOCA with loss of class IV power* for Wolsong Unit 2 is simulated. In conclusion, the present proposed methodology can represent the realistic power distribution due to the void reactivity effect and be useful for various transient analyses.

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