Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 22, 2009

Evaluation of Power Pulse in Large LOCA Using the IST Code for CANDU6 NPP

Park Dong Hwan and Kim Yun Ho
Nuclear Power Generation Lab, KEPRI. 103-16 Munjidong Yuseong-gu Daejeon
dhpark@kepri.re.kr

1. Introduction

Large Loss of Coolant Accident (LLOCA) is typical limiting design basis accident in CANDU plants in terms of discharged coolant inventory and fuel failure. In general, void in CANDU6 plant give positive effect on the core reactivity. Due to the positive void reactivity, Large LOCA induces very rapid increase of void and power within 5 seconds from the start of break which is called power pulse. The amount of power pulse is dependent on the break size and location. Usually break size much larger than feeder size is called large break LOCA.

The main objective of this study is to re-evaluate the break type and size which is the most deposited enthalpy using the industry standard toolset (IST) code system during LLOCA for general CANDU6.

This paper was limited to a study of the effectiveness of Shutdown System No. 1. All Cases analyzed were shutdown by insertion of the shutoff rods(SOR).

2. Analysis Methods

These feedback effects are modeled by external coupling calculation between physics and thermal-hydraulics. Physics calculation is performed with WIMS-AECL[1] based RFSP[2] code and thermal-hydraulics is performed with CATHENA[3] code. RFSP and CATHENA code is coupled by RFSPCB[4]. The input parameter of basis case is summarized in Table 1.

2.1 Reactor Physics Model

Conventionally the calculation of CANDU6 reactor physics including the void-induced positive reactivity has been performed by the code 1.5 group POWDERPUFS-V(PPV). But PPV is known to underestimate void reactivity after recent experiments. Therefore PPV is substituted to WIMS-AECL. Power transient in large LOCA, WIMS-AECL based 2-group SCM method is used in this study.

In this study two types of core state are modeled for plutonium peak and equilibrium core. For plutonium peak core, a set of one fuel type was just required. But lattice parameters for crept pressure tube were applied for equilibrium core in which 35 types crept tube – 7 radial and 5 axial regions - was modeled.

The incremental cross sections for all the reactivity devices and structural materials are modeled and generated by the code DRAGON-IST[5], which is a computer code that solves the three-dimensional neutron transport equation using collision probability techniques. The cross sections of a supercell (composed of fuel, tubes, coolant, moderator, and reactivity device) are provided by the WIMS-AECL. The final products of the supercell calculations are a set of incremental cross sections for an individual device to be added to the standard cell cross sections in the regions of the core over which the device has been smeared. These incremental cross sections were also modeled and generated for two core conditions.

Core physics model and calculation was performed by the code RFSP-IST which based on the WIMS-AECL SCM approach using the 2 energy group. Two models by core conditions were considered such as plutonium peak core, and equilibrium core which was included the crept pressure tube. All of core states were modeled ‘rapid startup after long shutdown’ in order to maximize inherent positive reactivity at LLOCA.

The spatial flux shapes of the core are depend on thermalhydraulics conditions. Kinetic calculation is needed. The CERBERUS module in RFSP solves the time-dependent two-energy-group neutron diffusion equation by means of the IQS (Improved Quasi-Static) method. The simulation proceeds by consecutive executions of the modules of the module. The CERBERUS module is coupled to a thermalhydraulics code and expected data which are coolant density, coolant temperature, and fuel temperature for 28 channel groups and 12 bundles wise

2.2 Thermalhydraulics Model

For the coupled thermalhydraulics-physics power pulse calculations, the CATHENA code was used. It is a two-fluid thermalhydraulics computer program and developed by AECL for analysis of flow transients in reactors and piping networks. The code CATHENA transfers the coolant temperature and density, fuel temperature to core physics calculation. Also each channel group is divided by 12 axial nodes corresponding to the 12 bundle locations. The pressure tube creep was consistently applied with the core physics model at equilibrium core.

2.3 Neutron Physics-Thermalhydraulics Coupling

Figure 1 is the procedure for coupling calculation within RFSP and CATHENA. For this procedure, RFSPCB was performed. First few steps are coupled between physics and thermalhydraulics, and the TRIP_TIME module of RFSP calculates the actuation time of shutdown system 1 through comparing the neutronic overpower flux as each detector position and high log rate power. After trip, the insertion model of SOR is also simulated.

3. Results

For 21 power transients following core state and tilt, break position and size have been performed.

In point of view actuation time, reactor inlet header (RIH) 100% break at equilibrium core with tilt case has the shortest time, and reactor outlet header(ROH) 80% break at plutonium peak core has the longest time to enter the SOR.

Enthalpy was highest deposited at 100% RIH break at equilibrium core state. The maximum bundle powers and deposited enthalpy are shown as Figure 1, 2, and 3.

Plutonium peak core has more inherent positive reactivity than equilibrium core. But power pulse spectrum of Pu peak core is lower than equilibrium core due to initial void fraction.

REFERENCES

1. S. Douglas, "WIMS-AECL Release 2-5d Users Manual", RC-1176/COG-94-52 (Revision 4)/FFC-RRP-299, 2002 July.
2. P. Schwanke, “RFSP-IST Version REL_3-04:User’s Manual, SQAD-06-504/153-117360-UM-001, Rev.0, 2006 December.
3. “CATHENA MOD-3.5d/Rev 2 Input Reference Revision 0 : 153-112020-UM-001 August 2005
4. R. McArthur, “RFSPCB Version 2.0.0 - Application Manual: Software User’s Manual”, CW-117370-UM-001, Rev. 0, 2007 October.
5. G. Marleau, A. Hébert and R. Roy, “A User’s Guide for DRAGON”, Report IGE-174 Revision 3, École Polytechnique de Montréal, 1998 September.

Table 1
Basic Parameter for LLOCA Analysis of CANDU6

Parameter	Basic Parameter
Initial Reactor Power(%FP)	103
Core Condition	Pu Peak/Eq.
Cell Code/CVR Bias(mk)	WIMS-AECL /-1.6
Initial Side-to-Side Power Tilt(%)	4%
Moderator Purity (atom%)	99.833
Moderator Purity (atom%)	98.50
Moderator Poison(ppmB) – Pu peak/Eq	6.192/2.611
Moderator Temperature(℃)	69
Number of T/H Regions	28
Pressure Tube Creep at Eq. Core(%)	Max. 2.259%
Credited Trip / System	High Log-rate Neutron Power/ SDS1
No. of Logic Channels to Trip	3/3
No. of SDS1 ROP Detectors	34
No. of SDS1 Ion Chambers	3
Total No. of SORs	28
Unavailable SORs	SOR01, SOR05

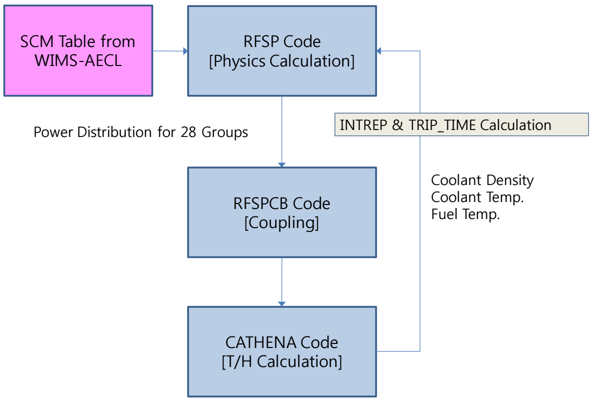

Figure 1
Program Interface for LLOCA Transient Analysis

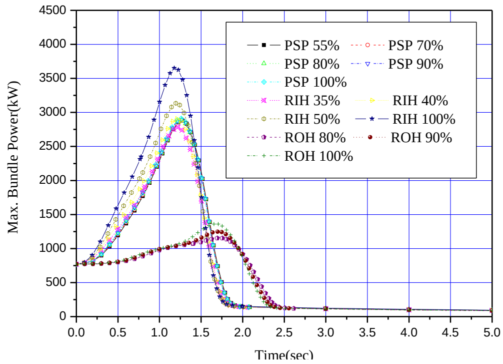

Figure 2
Maximum Bundle Power Transients for Eq. Core

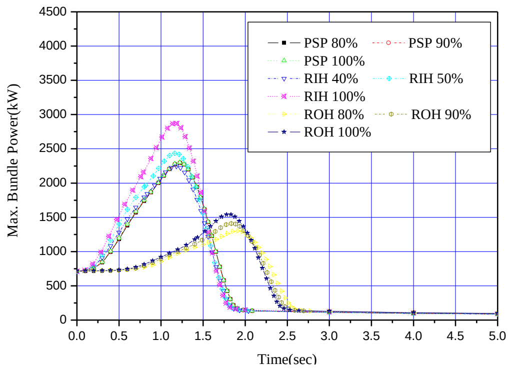

Figure 3
Maximum Bundle Power Transients for Pu. Peak Core

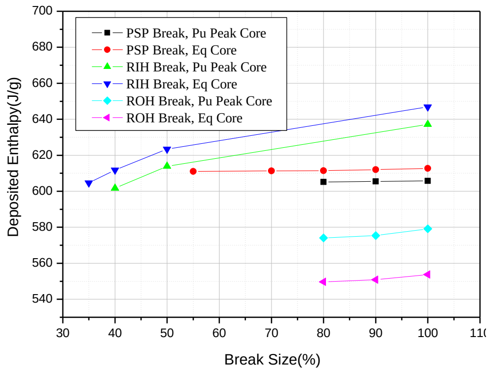

Figure 4 Deposited Enthalpy in Fuel Bundle