BEST ESTIMATE SBLOCA ANALYSIS OF CANDU 9 SYSTEMS

PART I: CATHENA MODELLING

Fang Bao
McMaster University, Ontario, Canada

Abstract

To quantify the uncertainty and seek a better understanding of small break loss-of-coolant-accident (SBLOCA) as a design-basis accident (DBA) of the CANDU 9 reactor system, the best estimate analysis approach is proposed in this paper. After a general review of this approach, a particular best estimate methodology to study the SBLOCA is introduced. A combination CATHENA/WIMS/RFSP is selected to be the toolset for the analysis. A detailed thermal-hydraulic modeling with CATHENA as well as some preliminary steady state results are given. The possibility of using this model to generate a Phenomena Identification and Ranking Table (PIRT) is also briefly discussed as a part of the future work.

1. Introduction

The small break loss-of-coolant-accident (SBLOCA) can become a significant safety concern as Nuclear Power Plants (NPPs) age, are uprated, and continue to seek improved operating efficiencies. Hence, as a design-basis accident (DBA) of a CANDU system, SBLOCA determines many of the safety-related operating limits for a plant. An example of SBLOCA could be the rupture of reactor inlet feeder (RIH).

Historically, nuclear safety analyses of LOCAs adopted conservative methodologies, assuming that choosing conservative values of key parameters will ensure that the outcome of analysis would envelope those expected during a postulated accident. However, this does not provide the knowledge on the true response of the plant, thus it is impossible to quantify the exact margins to safety limits. The United States Nuclear Regulatory Commission (USNRC) has already revised its acceptance criterion for Emergency Core Cooling Systems (ECCS) response to allow for use of best estimate methods [1, 2]. The Canadian Nuclear Safety Commission (CNSC) has recognized that best estimate predictions of plant response, along with accurate assessments of uncertainties, is an acceptable alternative to more limiting and bounding analyses for demonstrating safety system response [1, 3]. Therefore, accurate quantification of uncertainties is crucial in the best estimate analysis of SBLOCA. Several best estimate methods have been developed for this purpose, for example Code Scaling, Applicability and Uncertainty（CSAU）methodology in the United States and Best Estimate and Uncertainty (BEAU) in Canada.

The initial concern of the best estimate method is to select codes that can appropriately model the physical processes that have the most consequences in the accident scenario. During a SBLOCA of a CANDU 9 system, the lost coolant inventory would change the thermal-hydraulic behaviour of the system; some of the thermal-hydraulic parameters like coolant density and temperature would affect the neutron power of the reactor core. CATHENA and WIMS/RFSP are believed to contain the models and correlations that could, based upon initial code validation studies, appropriately represent most of these SBLOCA processes for a CANDU system, and together they have become the standard system thermal-hydraulic and reactor physics code respectively for safety analysis in Canadian nuclear industry.

Additionally, for SBLOCA of CANDU (especially very small break LOCA), during the initial stage the Reactor Regulating System (RRS) would be able to maintain the reactor power through Liquid Zone Control (LZC) and/or Mechanical Control Absorber (MCA). This will be modelled with MATLAB.

If every component is included, the main flow chart of the coupled model of an SBLOCA can be shown in Figure 1.

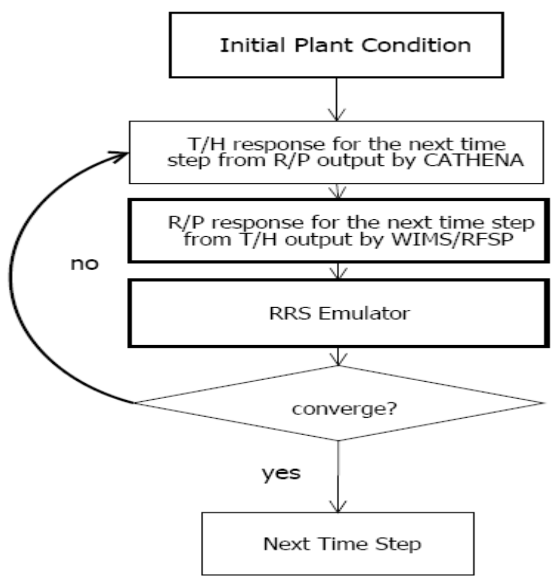

Figure 1 Main Flow Chart

In this paper, however, I will only discuss the thermal-hydraulic part of the work.

2. The thermal-hydraulic modelling of CANDU 9 by CATHENA

The thermal-hydraulic part of the CANDU system is modelled by CATHENA version MOD-3.5d/Rev 2. The major focus of the analysis is the period between the accident initiation and the reactor trip. Nevertheless, the ECCS is also modelled for the completeness. The accident sequence that will be produced by the model is based on the assumptions that

1. Moderator temperature is held constant.

2. No operator interventions are credited.

3. Class IV power is available.

4. ECCS is available.

5. Main and auxiliary feedwater is available.

2.1 CATHENA code

The Canadian Algorithm for THErmal-hydraulic Network Analysis (CATHENA) thermal-hydraulic code was developed by Atomic Energy of Canada Limited (AECL) for the use in the analysis and design of CANDU systems [4]. The thermal-hydraulic model in CATHENA is a one-dimensional, non-equilibrium two fluid model with special consideration for horizontal pipes. It also simulates heat transfer of solid surfaces [4]. Theses features make the simulation of the stratified flow condition in a horizontal CANDU fuel channel during LOCA more accurate. Hence, it is one of the more appropriate codes to analysis CANDU system.

2.2 The detailed quadrants modelling of CANDU 9

The Primary Heat Transfer (PHT) system of CANDU is two loops with the shape of a figure-of-eight. It has four identical quadrants. In industry they are usually designated as North East (NE), North West (NW), South East (SE) and South West (SW). The Secondary Side (SS) has four quadrants with the same designation as well. In addition, there are two other sub-systems Pressure Inventory Control system (PCI) and ECCS connected to the PHT.

2.2.1 Primary heat transfer

The primary heat transfer system collects the heat from the core of the reactor and transport it to the boiler. The major components include the core (from inlet header to outlet header), pumps, boilers (tube side) and piping. Figure 2 is the detailed node-link diagram of the PHT. In the figure, the rectangular nodes do not have heat transfer between the coolant fluid and the surface of the pipe wall while the rounded nodes do. The boiler-tube-side type of nodes (i.e. the steam generator heater and pre-heater) transfers the heat in the coolant to the corresponding boiler-shell-side nodes of the secondary side.

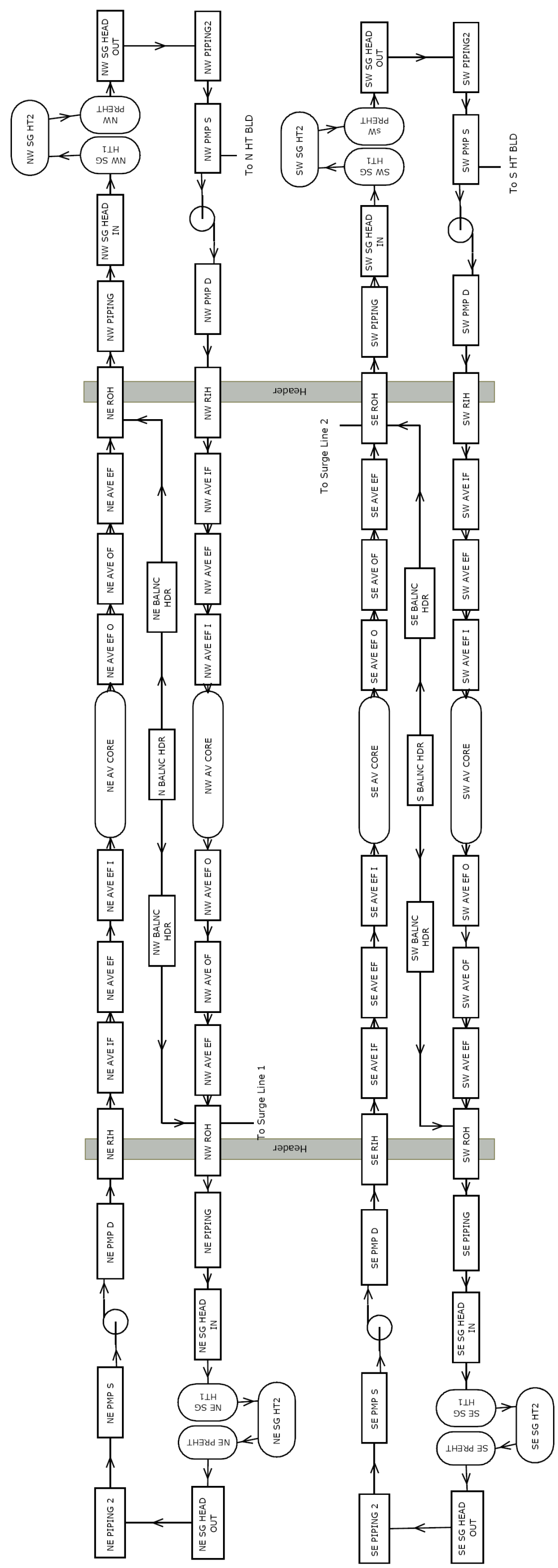

Figure 2 Node-link diagram of the PHT

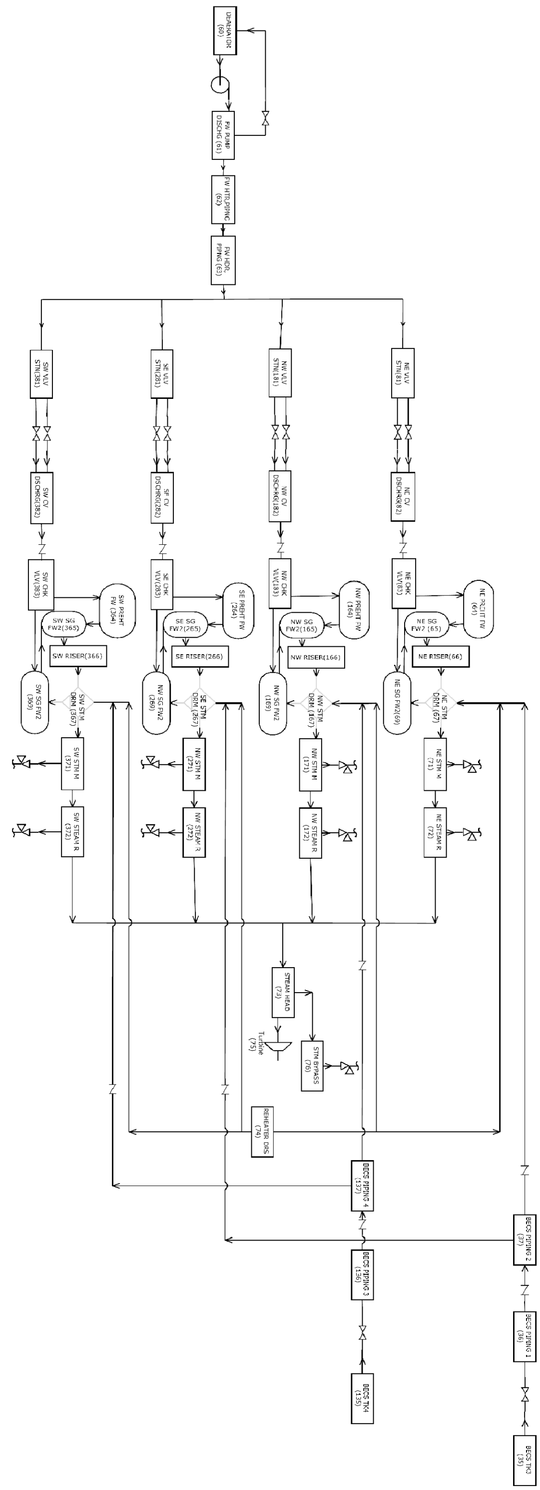

Figure 3 Node-link diagram of the secondary side

The core of a CANDU 9 system has 480 channels, 120 channels in each quadrant. In my model, I assume all 120 channels are identical to reduce the complexity i.e. I just have one channel group. The thermal power of a CANDU 9 system is around 2700 MW; hence my channel power is 5.625 MW. Due to the flexibility of CATHENA, adding more channel groups into the modelling is easy. In industry, around 10 ~ 20 channel groups are used to achieve more accurate modelling. In my future work, I plan to have 14 channel groups.

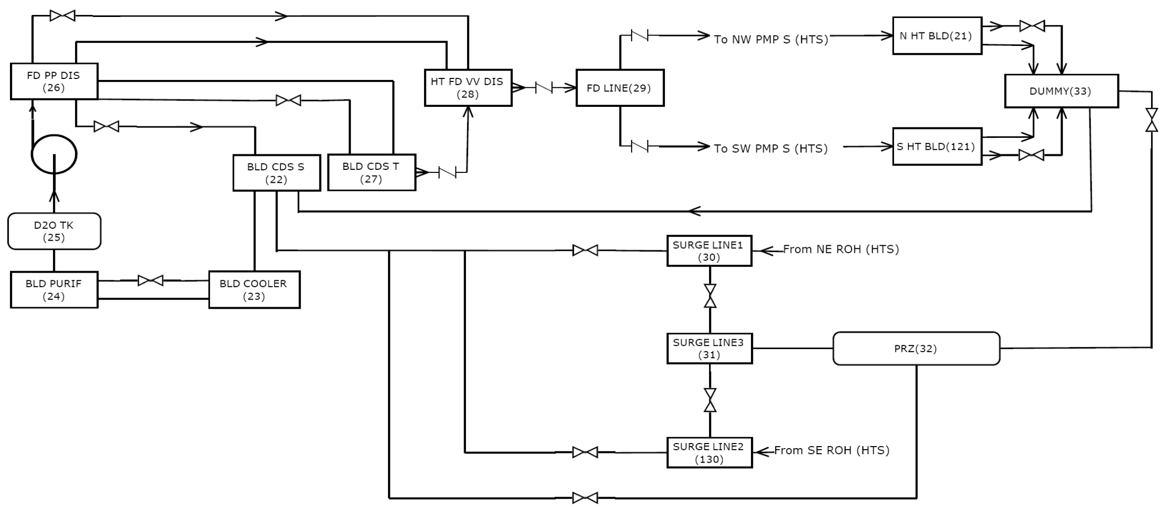

Figure 4 Node-link diagram of PIC

2.2.2 Secondary side

The CANDU 9 system uses an indirect-cycle; therefore, the system has a secondary side. As mentioned above the only connection between PHT and SS is the heat transfer between boiler-tube-side nodes of PHT and boiler-shell-side nodes of SS. Figure 3 is the detailed node-link diagram of the secondary side. Same as in the PHT, the rounded nodes are heat-transfer nodes and rectangular nodes are non-heat-transfer nodes.

Both the beginning of the secondary side, deaerator, and the end of the secondary side, turbine, are modelled as boundary conditions. The major components of the model include the feed water pump, steam separators, boilers (shell side), boiler emergency cooling system and piping.

2.2.3 Pressure inventory control system

The PIC controls the inventory and pressure of the heat transport system. The major components in the PIC include feed and bleed, feed pump, bleed condenser, D_2O tank, pressurizer, and surge lines. The detailed node-link diagram is shown in Figure 4.

2.2.4 Emergency core cooling sysyem

The major purpose of ECI system is to provide cooling to the fuel after a LOCA. CANDU 9 uses a direct cooled system. The detailed node-link diagram is shown in Figure 5.

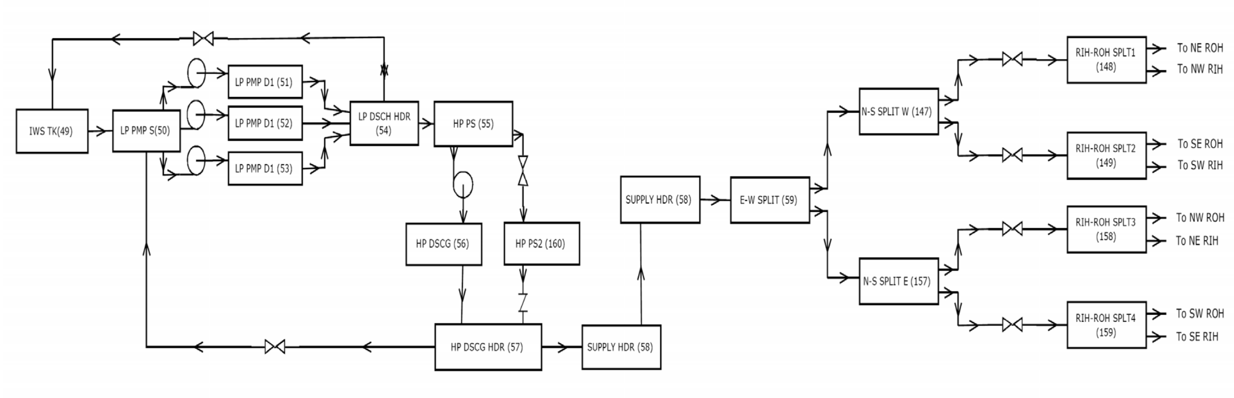

Figure 5 Node-link diagram of ECCS

2.2.5 Boundary condition for the steady state generation

A steady state with normal operation conditions is required as the initial condition for LOCA analysis. In order to generate this steady state, several boundary conditions need to be introduced. Table 1, Table 2, Table 3 and Table 4 list all the boundary conditions for PIC, PHT, SS and ECCS respectively.

Location	Type	Value
N HT BLD to DUMMY33	valve	0.065
BLEED COND S	pressure	1720 kPa
BLEED COND S	enthalpy	838.3 k J/kg
BLD COOLER to BLD PURIF	valve	0.283
D2O TK	pressure	135.8 kPa
D2O TK	enthalpy	213.1 kJ/kg
FD PP DIS to BLD CDS S	valve	Closed
FD PP DIS to BLD CDS T	valve	0.075
FD PP DIS to HT FD VV DIS	valve	0.22
S HT BLD to DUMMY33	valve	0.065
DUMMY33 to PRZR	valve	Closed
SURGE LINE1 to BLD CDS S	valve	Closed
SURGE LINE1 to SURGE LINE3	flow	0 kg/s
SURGE LINE1 to SURGE LINE3	valve	Fully Open
SURGE LINE2 to BLD CDS S	valve	Closed
SURGE LINE2 to SURGE LINE3	flow	0 kg/s
SURGE LINE2 to SURGE LINE3	valve	Fully Open
SURGE LINE3 to PRZR	flow	0 kg/s
PRZR to BLD CDS S	valve	Closed

Table 1 Boundary conditions of PIC

Location	Type	Value
NE ROH	Press	9.95E+03kPa

Table 2 Boundary conditions of PHT

Location	Type	Value
DEAERATOR	pressure	5.71E+02 kPa
DEAERATOR	enthalpy	6.53E+02 kJ/kg
FW PMP DISCHG to DEAERATOR	valve	Closed
NE VLV STN to NE CV DSCHRG	valve	79.60%
NE CHK VLV	flow	5.50E+01 kg/s
NE STM DRM	pressure	4.87E+03 kPa
NW VLV STN to NW CV DSCHRG	valve	79.80%
NW CHK VLV to NW SG FW2	flow	5.50E+01 kg/s
STM Bypass	flow	6.24E+01 kg/s
STEAM HEAD to Turbine	valve	50.00%
SE VLV STN to SE CV DSCHRG	valve	79.70%
SE CHK VLV to SE SG FW2	flow	5.50E+01 kg/s
SW VLV STN to SW CV DSCHRG	valve	79.90%
SW CHK VLV to SW SG FW2	flow	5.50E+01 kg/s
Turbine	pressure	4.95E+03 kPa
Turbine	enthalpy	2.79E+03 kJ/kg
REHEATER DRS	pressure	5.14E+03 kPa
REHEATER DRS	enthalpy	1.14E+03 kJ/kg
REHEATER DRS to NE STM DRM	flow	1.56E+01 kg/s
REHEATER DRS to NW STM DRM	flow	1.56E+01 kg/s
REHEATER DRS to SE STM DRM	flow	1.56E+01 kg/s
REHEATER DRS to SW STM DRM	flow	1.56E+01 kg/s
BECS PIPING 2 to NE STM DRM	valve	Closed
BECS PIPING 2 to SE STM DRM	valve	Closed
BECS PIPING 4 to NW STM DRM	valve	Closed
BECS PIPING 4 to SW STM DRM	valve	Closed

Table 3 Boundary conditions of SS

Location	Type	Value
N-S SPLIT W to RIH-ROH SPLT1	valve	Closed
N-S SPLIT W to RIH-ROH SPLT2	valve	Closed
N-S SPLIT E to RIH-ROH SPLT3	valve	Closed
N-S SPLIT E to RIH-ROH SPLT4	valve	Closed

Table 4 Boundary conditions of ECCS

2.2.6 Reactor physics model

A 3-D RFSP reactor physics model will be produced as the reactor physics side of the work. Yet in order to produce some testing result for the thermal-hydraulic model, a point kinetics model provided by CATHENA is include in the model.

2.2.7 Reactor Trip

The reactor trips simulated in this model are Heat Transport Low Pressure, Heat Transport Low Flow, Pressurizer Low Level, High Log Rate power and Neutron Over Power trips. The last two trips are based on the power calculated by point kinetics.

3. Preliminary Results

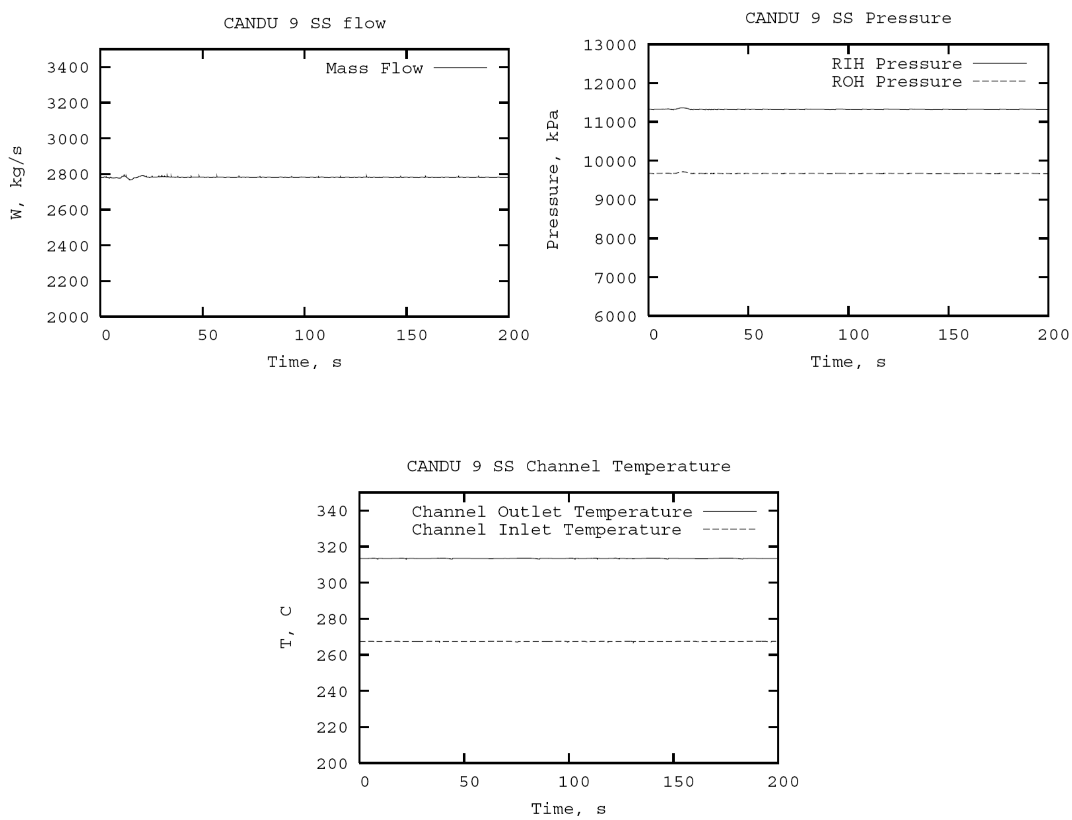

Figure 6 Major parameters of CANDU 9 at steady state

With the nodelization and boundary conditions described in the last section, a steady state is produced. Figure 6 shows some major parameters of the system at steady state, such as inlet and outlet header pressure, channel temperature and flow.

4. Future Work

The Best Estimate method requires the development of a Phenomena Identification and Ranking Table (PIRT). Because the PIRT identifies, for a given accident scenario, those physical processes that are expected to be dominant contributors to uncertainty [5]. To construct a PIRT, parameters of several categories are to be studied. These categories include but are not limited to, Steady State initial condition (e.g. pump head, steam generator level, inlet temperature, initial power and etc.), correlations used in the model (e.g. friction correlation, void correlation, critical heat flux correlation, break discharge coefficient, two phase multipliers and etc.), and safety system parameters (e.g. shut down system response time, instrument response time and etc.).

In addition, different break sizes and break locations will also be studied.

Acknowledgement

This work is supported by a grant from the University Network of Excellence in Nuclear Engineering (UNENE). The author would also like to thank Atomic Energy of Canada Limited (AECL) for providing the software.

References

1. D.R. Novog, F. Bao, J. Luxat "Best Estimate Analysis of Loss of Flow Events at CANDU Nuclear Power Plants" 28th Annual CNS Conference, 2007
2. Canadian Nuclear Safety Commission, Draft Regulatory Guide G-144, Guidelines for Establishment of Shutdown System Trip Parameter Effectiveness, March 2005
3. Technical Program Group, "Quantifying Reactor Safety Margins – Application of CSAU Methodology to a LBLOCA", EG&G Idaho, Inc., NUREG/CR-5249, December 1989
4. B.N. Hanna "CATHENA: A thermalhydraulic code for CANDU analysis" Nuclear Engineering and Design 180 (1998) 113–131
5. S. M. Bajorek, N. Petkov, K. Ohkawa, R. M. Kemper, A. P. Ginsberg, "Realistic Small-and-Intermediate-Break Loss-of-Coolant Accident Analysis Using Wcobra/Trac" Nuclear technology 136(2001) 50-62