Coupling Between DONJON and CATHENA Using a Bash Script for RIH Break Opening Time Assessment

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Abstract

More and more safety analyses are performed with coupled safety codes to assess in detail the reactor behaviour under abnormal conditions. The Canadian industry gives a particular attention to script-based or PVM-based coupling of reactor physics (RFSP) and thermalhydraulics (CATHENA or TUF) codes.

This work presents a first step to develop independent simulation capabilities using a script-based coupling of DONJON and CATHENA codes. This work aims to provide CNSC with a tool allowing for have independent insights related to CANDU reactors behaviour under different anticipated transients or accidents involving reactor physics and thermalhydraulics. A particular interest is being given to the assessment of the impact of RIH break opening time on the power pulse for a generic CANDU-6 reactor.

1. Introduction

The use of coupled reactor physics and thermalhydraulics simulations becomes increasingly important in the safety analysis where the nuclear industry tries to demonstrate safety margins availability for aging plants or power uprates from current derated states. The different set of accidents for which a coupled simulation is used, includes large break LOCA, small break LOCA, and LOFA. The reactor physics and thermalhydraulics codes that are typically used in the simulation Canadian CANDU 6 nuclear generating stations are RFSP [1] and CATHENA [2] with a script-based program which performs coupling of both codes [3].

In this work, we have developed a macroscopic cross-section database, calculated with the transport code DRAGON [4] for 37-element bundles with the FBM [5] and for the safety and control devices. The diffusion code DONJON [6] has been used to model a typical CANDU 6 reactor core with the shutdown system one, the reactor regulating system, and the platinum, vanadium and ion chamber detectors. CATHENA has been used to model the PHTS with 28 representative channels, the emergency core cooling system and a portion of the secondary side. The coupling has been made using a bash script and has been applied to assess the impact of a reactor inlet header break opening time on the reactor power pulse.

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2. Cross Section Database Generation with DRAGON

The cross sections database used to perform the core calculations has been developed using DRAGON [4]. 89-group ENDF-B V and ENDF-B VI libraries have been used for all of the following cell calculations. The calculations here have been homogenized over the cells and condensated over 2 groups with a separation at 0.625eV.

2.1. Cell Calculations

The cross sections database for the fuel cell and the reflector has been created with the FBM [5] which requires performing a nominal cross sections calculation and 20 perturbation calculations. Those perturbations include fuel and coolant temperature, coolant density, moderator temperature and density, coolant and moderator purity, and a boron concentration. Table 1 provides a summary of the nominal and perturbed values of the different parameters. Although bundle power perturbations are performed, associated cross sections have not been used in the core calculations due to inappropriate interpolations of the FBM for power. Therefore, going forward only the nominal bundle power is used. The perturbation calculations have been performed at each burnup step based on the nominal parameter values cross sections calculation. Thirty burnup steps have been selected uniformly on a log scale covering a burnup range up to 35000 MWd/t.

Table 1 Nominal and perturbed values for the FBM

Case	T_{fuel} (K)	T_{Cool} (K)	T_{Mode} (K)	D_{Cool} (g/cm^3)	D_{Mod} (g/cm^3)	P_{Cool} (%)	P_{Mode} (%)	Bor (ppm)	Xe (ppm)	Sm (ppm)	Np (ppm)	P (kW/kgU)
1 & 22	***960.15***	***562.15***	***348.15***	***0.8064***	***1.08288***	***98.6938***	***99.966***	***10^{-20}***	***10^{-20}***	***10^{-20}***	***10^{-20}***	***31.9713***
2	+1113											
3	-657											
4		+461										
5		-259										
6 & 23			+16									
7 & 24			-43									
8				x1.5								
9				x0.00001								
10 & 25					x1.1							
11 & 26					x0.7							
12 & 27								15				
13 & 28							-0.486					
14									+0			
15										+0		
16											+0	
17	+1113			x0.00001								
18		+461		x0.00001								
19												49.3866
20												27.2926
21												3.4656

With respect to the cell geometry, a non-crept pressure tube and 35 various crept pressure tube diameters have been used with the FBM to develop the cross section database. Figures 1 and 2 show the cell geometry with a non-crept pressure tube and another one with the maximum creep diameter based on the creep conditions at the reactor’s end of life. It should be noted that, for the cross-section calculations with crept pressure tubes, the bundle is not located in the centre of the cell as shown in Figure 2. A sensitivity assessment has demonstrated that tracking options using 11 quadrature angles and 15 lines/cm for self-shielding calculations or 15 quadrature angles and 30 lines/cm for cross sections calculations is sufficient.

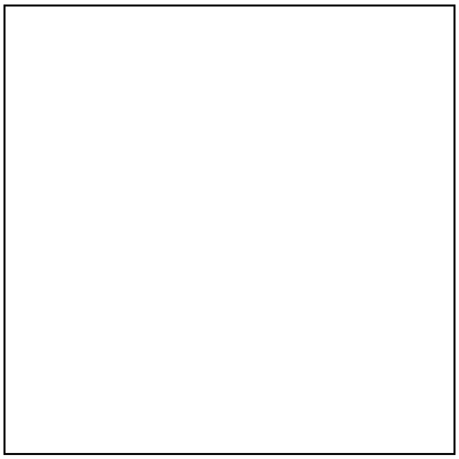

Figure 1 37 element CANDU bundle without creep

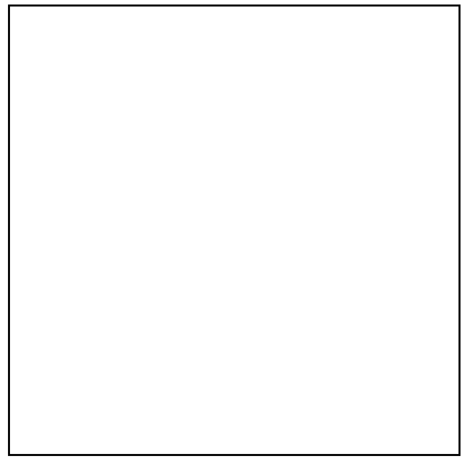

Figure 2 37 element CANDU bundle with maximum creep

2.2. Devices Calculations

The cross sections calculations for the devices have been performed with non-crept pressure tube and average crept pressure tube diameters at nominal local conditions using fresh fuel and fuel with average burnup. For the crept pressure tube cross section calculation, the bundles have been positioned in the centre of the cell. A sensitivity assessment was performed with a 2D cross-section cell calculation and had shown that the impact of bundle shift due to the creep on the cross sections is negligible. This allows to reduce the time calculations by using the symmetries where possible. The devices cross sections database has been generated with exact geometries for the different types of LZC and ADJ, but also for the SOR (or MCA). Figures 3 and 4 provide examples of the exact geometries used in the devices cross sections calculations.

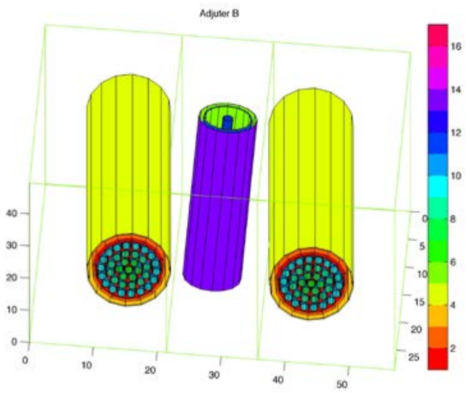

Figure 3 Type B Adjuster

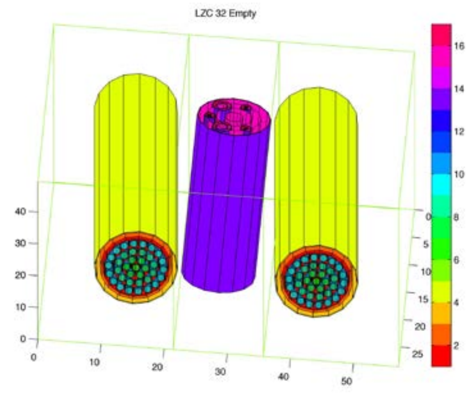

Figure 4 Type 32 Liquid Zone Controller

For these 3D calculations, a sensitivity assessment has shown that 4 quadrature angles and 2.5 lines/cm for the self-shielding, and 12 quadrature angles and 10 lines/cm for the cross sections calculations are sufficient.

3. CANDU-6 Core Modelling with DONJON

In this section, we have modelled a generic CANDU-6 reactor core with 14 LZC, 21 ADJ, 4 MCA and 28 SOR. Figure 5 shows the localization of the safety and control devices. In addition, 34 platinum detectors and 3 ion chambers for SDS-1, and 14 platinum detectors, 102 vanadium detectors and 3 ion chambers for RRS have been modelled.

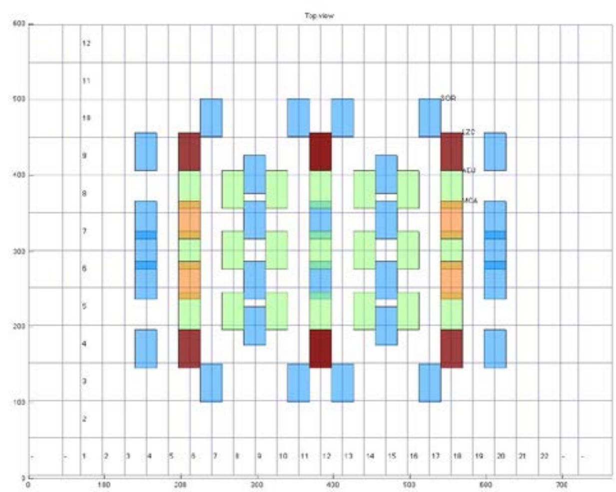

Figure 5 Top View of the Reactor with Safety and Control Devices

Several options have been introduced in the core calculations. The first one is that the channels can be modelled either with creep or without creep. For the creep modelling, to each cell is assigned one of the 35 FBM database for crept pressure tube. Another option is the possibility to search for an initial core criticality by adjusting the boron concentration when the burnup distribution in the core given, or by adjusting the exit burnup of different burnup regions during core time-average calculation and applying an age pattern [7]. The transient calculations can be performed either by using an improved quasi-static method or by using a theta method. Typically, a fully implicit theta method is suggested due to its stability. For this generic CANDU-6 core, a 2 out of 3 logic for the instrumentation channels has been modelled to trip the reactor during the transients. The neutronic trip setpoints are provided in Table 2 and have been used with an aged reactor (crept pressure tubes) in the following simulations.

Table 2 Neutronic Trip Setpoints

High Power	108.0%FP
High Log Rate	10.0%/sec

Verifications of the control and safety devices reactivity worth have been performed by comparing to the literature [8] to ensure a certain confidence in the assessments.

4. CATHENA Modelling

The thermalhydraulic modelling of the generic CANDU-6 reactor consists of the PHTS, the ECCS, and a part of the secondary side. The modelling includes the plants overall control with the associated setpoints. Since we will be assessing only the first few seconds of the large break LOCA later on, the PHTS components presented in Figure 6 are the main reactor behaviour players during the assessed period of time.

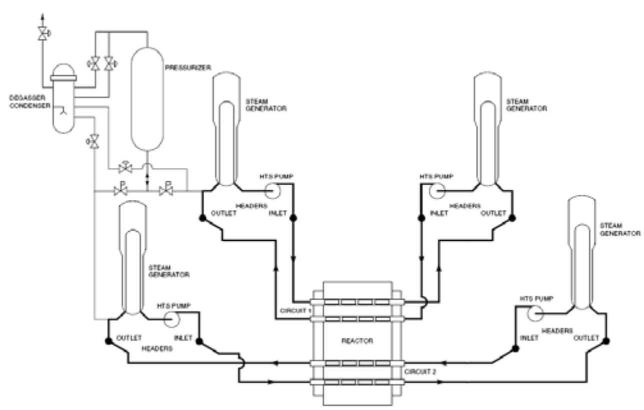

Figure 6 PHTS Modelled in CATHENA

In this generic CANDU-6 model, we have used 7 channels per pass for the 4 passes. The 28 channels modelled have been chosen to represent a group of channels based on their power and flow rate. The main conditions that impose boundary conditions for the aged PHTS are defined in Table 3.

Table 3 PHTS Main Conditions

Pressurizer setpoint (ROH pressure)	10.0MPa
RIH temperatures	267.3°C

5. Coupling Methodology of DONJON and CATHENA

The coupling between DONJON and CATHENA codes is based on a bash script. The coupling scheme is presented in Figure 7. The bash script for coupling DONJON and CATHENA uses the input deck templates where keywords are replaced with restart file name, initial and final time. The process starts with a DONJON initial criticality search in which the local thermalhydraulic conditions are fixed to determine a first estimate for the power distribution. Based on the power distribution, a group-averaged bundle power is calculated for each bundle of the representative channels. These bundle powers are then used to perform the first time step with CATHENA and to determine the local thermalhydraulic conditions. The next DONJON calculations, which can either be steady-state calculation with criticality search or kinetic calculation with RRS to maintain the power constant, are performed to reach an initial steady-state with all thermalhydraulic conditions from CATHENA. Similarly, the subsequent CATHENA calculations are used to achieve an initial steady-state. Once the steady state is achieved, DONJON uses the kinetic calculation for the transient and CATHENA continues to run as previously. The bash script ensures the appropriate exchange of information between the two codes at each time step, but does not perform mathematical calculations with the reactor physics or thermalhydraulics parameters. This means that the coolant densities, fuel and coolant temperatures for each bundle of the representative channels calculated by CATHENA is sent to DONJON, while the power of each group-average bundle calculated by DONJON is sent to CATHENA.

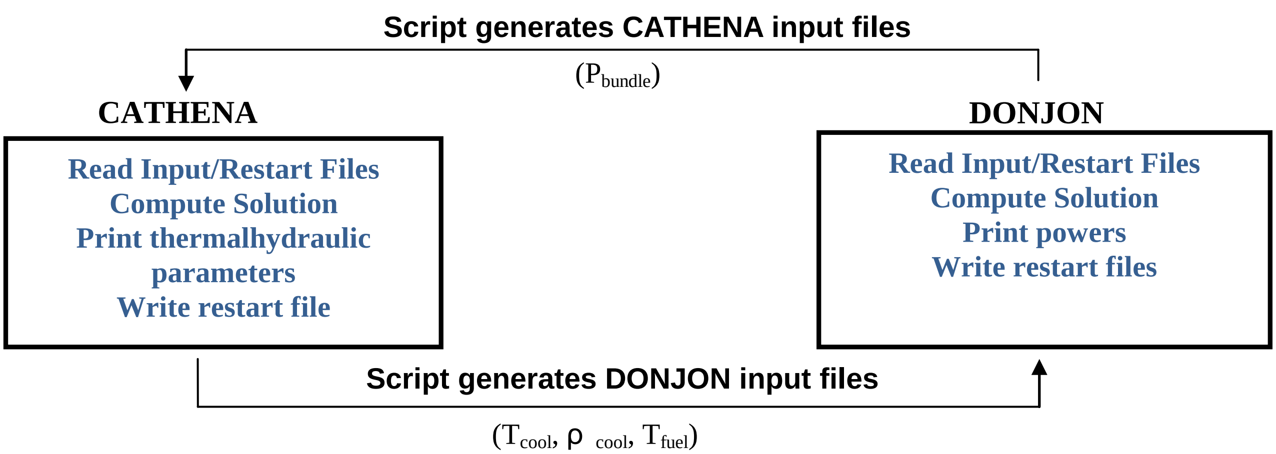

Figure 7 Codes Coupling Scheme

6. Simulation of Impact of Break Opening Time on Large Break LOCA Power Pulse

Stylized deterministic large LOCA safety analysis assumes a RIH break opening time of 1ms to be within the design basis accidents. This instantaneous double ended guillotine break is considered to be extremely conservative. Therefore, there is an increased interest in performing a more realistic assessment of large break LOCA with the assumption of a more realistic break opening time. In this section, we have performed a sensitivity analysis with respect to the RIH break opening time to confirm the consistency of our modelling results with the literature [9] and to have insights regarding the impact of a break opening time on the power pulse.

Table 4 Break Opening Time Cases

Case 1	200% BR in 1.0ms
Case 2	200% BR in 0.5s
Case 3	200% BR in 1.0s
Case 4	200% BR in 2.0s
Case 5	200% BR in 5.0s
Case 6	200% BR in 9.0s
Case 7	17% in 5ms followed by a 200% in 5.0s
Case 8	17% in 5ms followed by a 100% in 5.0s

Table 4 provides a summary of the break opening characteristics which are used as sensitivity cases. The last two cases were performed in order to assess the impact of a two stages break model which may represent a more realistic break opening characteristic. These two cases are more in line with current industry investigation to define a more realistic accident progression. Figure 8 shows the sensitivity of the power pulse to the break opening characteristics. All the transients in Figure 8 are terminated with the first SDS-1 trip. The results for a stylized large break LOCA assessment show consistency with the literature [9] which provides us with a certain confidence with respect to our results. The assessment of a large break LOCA with a more realistic break opening time shows a major reduction of the power pulse. A sensitivity analysis has been performed to assess the optimal time step applied for all the simulations which ended up being 50 ms for DONJON, and between 0.1 ms and 50 ms for CATHENA.

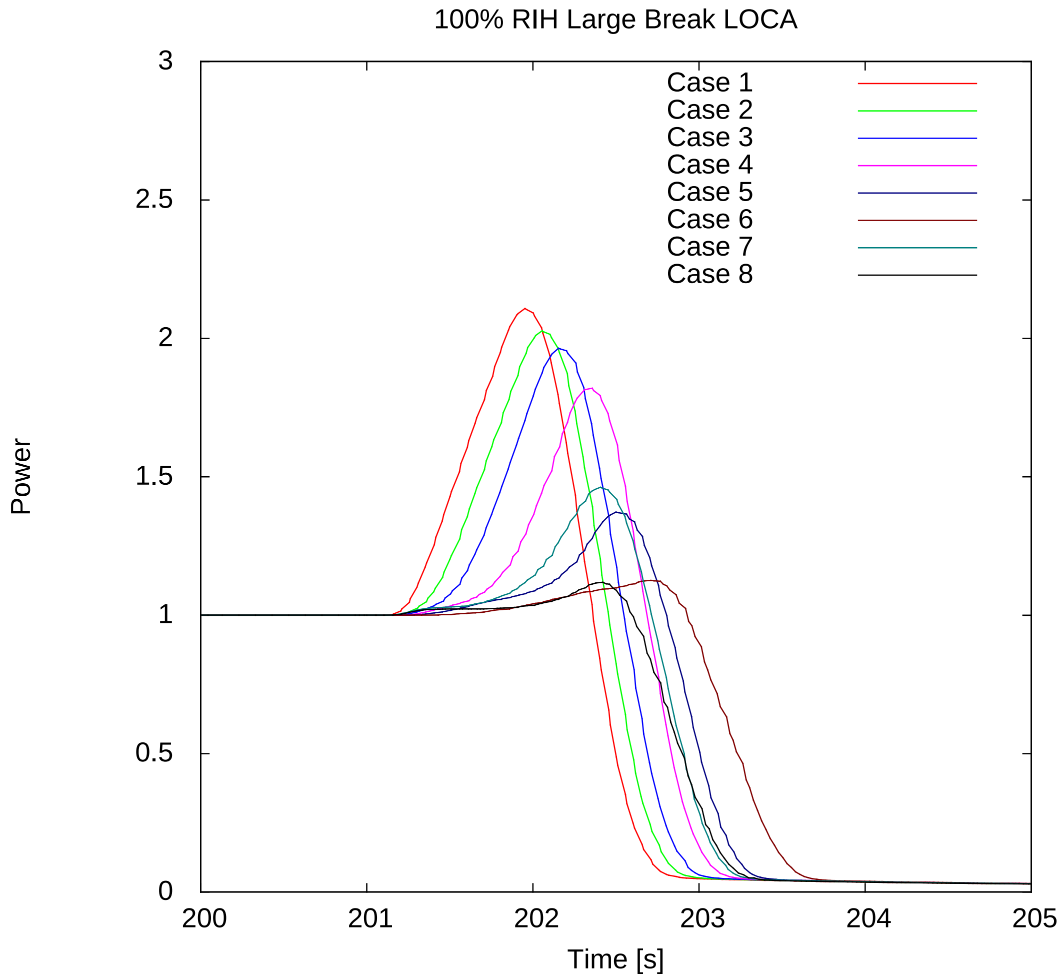

Figure 8 Power Pulse for Different Break Opening Times

7. Conclusion

In this work, we have developed an independent capability to perform coupled reactor physics – thermalhydraulics simulations using a bash script. Indeed, we have used the DRAGON code to develop a cross-section database for DONJON code. We have developed a generic CANDU-6 core modelling with DONJON. Finally, we have used a generic CANDU-6 CATHENA modelling to perform a script based coupling. Comparisons with literature have shown that our results are consistent. We have applied this new simulation capability to explore the impact of a realistic break opening time on the power pulse on a generic CANDU-6.

8. Acronyms

ADJ	Adjuster
BR	Break Ramp
CANDU	CANadian Deuterium Uranium
ECCS	Emergency Core Cooling System
FBM	Feedback model

LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LZC	Liquid zone controller
MCA	Mechanical control absorber
PHTS	Primary Heat Transport System
PVM	Parallel Virtual Machine
RIH	Reactor inlet header
ROH	Reactor outlet header
SDS	Shutdown system
SOR	Shutoff rod

9. Acknowledgements

The authors are very thankful to Valérie Fougère and Vali Tavasoli for their support. This work has been performed at the CNSC with the support of École Polytechnique de Montréal.

10. Disclaimer

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