Research Article

Analysis of Control Rod Drop Accidents for the Canadian SCWR Using Coupled 3-Dimensional Neutron Kinetics and Thermal Hydraulics

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The Canadian Supercritical Water-cooled Reactor (SCWR), a GEN IV reactor design, is a hybrid design of the well-established CANDU™ and Boiling Water Reactor with water above its thermodynamic critical point. Given the batch fueled design, control rods are used to manage the reactivity throughout the fuel cycle. This paper examines the consequences of a control rod drop accident (CRDA) for the Canadian SCWR. The asymmetry generated by the dropped rod requires an accurate 3-dimensional neutron kinetics calculation coupled to a detailed thermal-hydraulic model. Before simulating the CRDAs, the proper implementation of the 3D reactivity feedback was verified and various sensitivity studies were performed. This work demonstrates that the proposed safety systems for the SCWR core are capable of terminating the CRDA sequence prior to exceeding maximum sheath and centerline temperatures. In one instance involving a rod on the periphery of the core, the proposed trip setpoint (115% FP) was not exceeded and a new steady state was reached. Therefore it is recommended that the design also include provisions for a high-log rate and/or local Neutron Overpower Protection (NOP) trips, similar to existing CANDU designs such that reactor shutdown can be assured for such spatial anomalies.

1. Introduction

The Canadian Supercritical Water-cooled Reactor (SCWR) design is the result of the fusion of a CANDU reactor and a Boiling Water Reactor (BWR) [1]. The vertical orientation of the core and large light water density variation along the channel resembles the latter while the pressure-tube technology and separate low-pressure (0.3 MPa) heavy-water moderator are inspired from the CANDU design (Figure 1). It is important to note that the light water coolant (25 MPa, 350°C inlet) is in the supercritical condition and therefore there is no phase change as the coolant heats up along the channel (625°C outlet). The batch refueling design necessitates a large excess reactivity at beginning of cycle; therefore two reactivity control systems were designed for the Canadian SCWR. A burnable absorber (Gd_2O_3) was integrated into a selected number of fuel pins and 89 adjustable cruciform control rods can be inserted from the bottom of the core into the low-pressure moderator [2]. The core is composed of 336

fuel assemblies which generate a thermal power of 2032 MW leading to approximately 1000 MW_{el} (~48% efficiency).

Nuclear reactors can be subjected to a wide variety of conditions ranging from normal operations to postulated accidents. As a consequence, safety analyses are performed to evaluate the impact of such transients and the performance of the special safety systems and to ensure that safety limits are not exceeded. Evaluation of the conceptual designs of several GEN IV SCWRs has been performed as outlined below.

The High Performance Light Water Reactor (HPLWR), a European design, has already been subjected to extensive analyses of turbine trips, loss of coolant accidents (LOCAs), and loss of flow accidents (LOFAs) [3]. The transient created by a turbine trip/load rejection was expected to be relatively severe; however the detailed analysis showed large margins. Analysis of LOCA scenarios demonstrated cladding temperatures up to 849°C which is well below the 1204°C safety limit. Finally, LOFAs without reactor SCRAM did not show any issues as the thermal limits were not exceeded; however,

FIGURE 1: Canadian SCWR concept.

the author recommended a 3D kinetics model in future studies to confirm the validity of the point-kinetics model. All these thermal-hydraulic driven accidents showed acceptable consequences but more challenging results were observed during reactivity initiated accidents (RIAs). Uncontrolled withdrawal of control rods and control rod ejections without SCRAM were investigated in [3, 4]. The maximum allowed cladding and fuel temperature were not exceeded; however the margins in some cases were very small. Additional studies of uncontrolled withdrawal of control rods without SCRAM, loss of feedwater heating, and control rod malfunction [5] showed that the last two transients have large margins to the safety limits; however the control rod withdrawal leads to fuel melting and emphasizes the importance of RIAs analyses. Finally, potential xenon oscillations also have to be investigated despite being a much slower transient [6]. Xenon stability analysis was then performed for the HPLWR and concluded that xenon oscillations are nonexistent [7].

The Super Light Water Reactor (Super LWR), a Japanese SCWR concept, evaluated the impact of a small and large LOCA, a LOFA, a loss of turbine flow, a pressure control system failure, and an uncontrolled CR withdrawal at normal operation [8]. While the maximum cladding surface temperature (MCST) always remained within the acceptable range, the maximum fuel centerline temperature (MFCLT) was not evaluated [9]. Given the importance of MFCLT in RIA transients it is difficult to assess the performance under these conditions. A mixed spectrum SCWR (SCWR-M) proposed by Cheng et al. [10] investigated LOFAs and met the safety criteria as long as few design features were respected [11]. In addition, LOCAs were carried out for many break sizes and in the most detrimental case, the MCST largely exceeded the safety criterion which will lead to further core optimization [12].

Thus far only thermal-hydraulic driven accident scenarios (LOCA, LOFA) have been investigated for the Canadian SCWR. While the coolant void reactivity (CVR) for the design was negative, studies demonstrate a positive reactivity

insertion under nonequilibrium voiding conditions [13, 14]. As can be seen in Figure 1, the coolant enters at the top of the core from the inlet plenum through channel specific orifices and travels downwards in the central tube. When the coolant reaches the bottom of the channel it turns around and travels upwards picking up the heat from the 64 (Pu,Th)O₂ fuel pins. The positive reactivity insertion happens when the coolant first voids around the fuel pins and was studied using coupled 3D neutron kinetics and thermal-hydraulic simulations for thermal-hydraulic driven transients [15]. The inlet LOCA resulted in a flow reversal leading first to a coolant density decrease around the fuel which generated a power pulse before being terminated by the fuel Doppler effect as well as the large central tube coolant density feedback. An inlet LOCA with loss of emergency core cooling systems was also investigated in [16] and evaluated a MCST up to 1278°C which is just above the 1260°C safety limit.

The work carried out in this study presents the first reactivity initiated accidents for the Canadian SCWR. Given the importance of RIA in the literature review of the HPLWR, it is important that such accidents be analyzed for the Canadian design. Such accidents may involve highly localized phenomena and reactivity feedback, thus necessitating a 3-dimensional neutron kinetics analysis coupled to thermal hydraulics simulations. A very fast control rod ejection is not considered for the Canadian SCWR as the control rods are inserted in the low-pressure moderator which is different from Light Water Reactor (LWR) technologies where the rods are surrounded by high pressure coolant. However, a control rod can fall under the action of gravity and lead to a control rod drop accident (CRDA). The initiating event, which is the separation of a control rod from its driving mechanism, is considered identical to a BWR's as the control rod design is similar [17]. The incident can occur at any time of the fuel cycle and for any control rod, leading to a local reduction in neutron absorption which in turn increases the fission rate causing a fast and localized power excursion in the

FIGURE 2: Fuel lattice cell meshing (a) and materials (b).

reactor core. Such local phenomena cannot be assessed using point-kinetics approximations. Moreover, neutron physics and thermal-hydraulic parameters are intrinsically coupled and play an important role during the transient by providing reactivity feedback mechanisms [18–20]. As a consequence, a 3D neutron kinetics model was built in PARCS [21] and has been coupled to RELAP/SCDAPSIM/MOD4 [22] through a series of external python scripts to provide a comprehensive prediction of the transient core response during RIA events.

2. Canadian SCWR Core Modelling

This section presents the reactor physics and thermal hydraulics models built to simulate the Canadian SCWR.

2.1. Lattice Physics Modelling

2.1.1. Fuel Cell Lattice. The fuel channel geometry and materials specifications for the Canadian SCWR lattice are summarized in Table 1. A total of 64 (Pu,Th) O_2 fuel pins are surrounded by a supercritical light water coolant contained in a pressure tube sitting in the heavy-water moderator. TRITON [23] was used to simulate the fuel lattice cell with depletion using the model shown in Figure 2. Reflective boundary conditions were applied on the sides of the cell and the 238-energy group neutron transport equation was solved to obtain the flux distribution within the cell. Using a flux weighting process, the cross sections were homogenized and condensed in 2 energy groups with an energy cutoff of 2.57 eV [24, 25]. The fuel materials were then depleted with ORIGEN [23] and this entire process was repeated until a burnup of approximately 70 MW·d/kg. Given the fuel enrichment and burnable absorber grading in the axial direction as well as changing thermal-hydraulic conditions along the channel, fuel lattice calculations were performed at 20 axial elevations (see Tables 2 and 3). The thermal-hydraulic conditions (coolant temperature and properties, material temperatures, and properties) were taken from [26]. Finally, the kinetics parameters (neutron velocity, delayed neutron fractions, and lambdas) were evaluated at each depletion step for the 20 elevations as recommended in [27],

hence providing the model with locally varying kinetics parameters. These runs represent the reference state of the lattice calculations. Additional sensitivity calculations were performed to assess the impact of lattice cell meshing to ensure accurate lattice cell cross sections are determined.

Branch calculations have also been performed to evaluate the sensitivity of the reference lattice homogenized cross sections to important thermal-hydraulic and control system perturbations. The branches listed in Table 3 were simulated at every axial elevation and several burnup steps by perturbing the reference thermal-hydraulic parameter (see Table 2) as well as simulating the insertion or removal of the control rod. It has to be noted that even though the coolant in the central tube and between the fuel pins is the same, their impact on the lattice physics phenomena is very different [14]. Since nonequilibrium voiding transients may occur each coolant region is treated separately by having separate branches for the inner and outer coolant regions. These branch calculations were used by PARCS to interpolate new cross sections depending on the local thermal-hydraulic conditions calculated by RELAP/SCDAPSIM/MOD4 and based on control rod positions. In [26], the cross sections interpolation technique was demonstrated to perform well for the studied transient.

Finally, the reflector cross sections were generated through a multicell geometry as shown in Figure 3. Sensitivity studies demonstrated that the fuel type and thermal-hydraulic conditions of the four fuel assemblies did not significantly change the reflector properties. Therefore, the reflector's cross sections were evaluated using the fuel assembly's thermal-hydraulic conditions at the bottom of the core and for fresh fuel. Moreover, it was found that homogenizing the entire reflector region in the multicell to generate the reflector's cross sections and applying it to every reflector cell in the diffusion model were also a good approximation.

2.1.2. Full-Core Diffusion Model. The 336 fuel channels are surrounded by a 100 cm thick radial reflector and 75 cm axial reflector at the top and bottom of the core. The fuel channel is divided into 20 axial positions using the corresponding

TABLE 1: Fuel lattice cell specifications.

Component	Dimension	Material	Composition [wt%]	Density [$\text{g}\cdot\text{cm}^{-3}$]
Centre tube coolant	4.6 cm radius	Light water	100% H ₂ O	Variable
Centre flow tube	4.60 cm inner radius; 0.1 cm thick	Zr-modified 310 stainless steel	C: 0.034; Si: 0.51; Mn: 0.74; P: 0.016; S: 0.002; Ni: 20.82; Cr: 25.04; Fe: 51.738; Mo: 0.51; Zr: 0.59	7.9
Inner fuel pins (24)	0.415 cm radius; 5.4 cm pitch radius	0–3 m 12.8 wt% PuO ₂ in ThO ₂ 3–4 m 12.3 wt% PuO ₂ in ThO ₂ 4–5 m 11.5 wt% PuO ₂ in ThO ₂	Pu: 11.293; Th: 76.632; O: 12.075 Pu: 10.851; Th: 77.072; O: 12.077 Pu: 10.146; Th: 77.775; O: 12.079	9.91
Outer fuel pins (24)	0.44 cm radius; 6.575 cm pitch radius	0–3 m 10.24 wt% PuO ₂ in ThO ₂ 3–4 m 9.84 wt% PuO ₂ in ThO ₂ 4–5 m 9.20 wt% PuO ₂ in ThO ₂	Pu: 9.034; Th: 78.882; O: 12.084 Pu: 8.682; Th: 79.233; O: 12.085 Pu: 8.117; Th: 79.796; O: 12.087	9.87
Inner fuel pins with burnable absorber (8)	0.415 cm radius; 5.4 cm pitch radius (evenly spaced, $\pi/8$ angular offset)	0–3 m 12.8 wt% PuO ₂ in ThO ₂ ; 3 wt% Gd ₂ O ₃ in PuO ₂ /ThO ₂ 3–4 m 12.3 wt% PuO ₂ in ThO ₂ ; 2 wt% Gd ₂ O ₃ in PuO ₂ /ThO ₂ 4–5 m 11.5 wt% PuO ₂ in ThO ₂ ; 2 wt% Gd ₂ O ₃ in PuO ₂ /ThO ₂	Pu: 10.954; Th: 74.333; O: 12.11; Gd: 2.603 Pu: 10.635; Th: 75.53; O: 12.10; Gd: 1.735 Pu: 9.943; Th: 76.219; O: 12.103; Gd: 1.735	9.82 9.85 9.85
Outer fuel pins with burnable absorber (8)	0.44 cm radius; 6.575 cm pitch radius (evenly spaced, no angular offset)	0–3 m 10.24 wt% PuO ₂ in ThO ₂ ; 3 wt% Gd ₂ O ₃ in PuO ₂ /ThO ₂ 3–4 m 9.84 wt% PuO ₂ in ThO ₂ ; 2 wt% Gd ₂ O ₃ in PuO ₂ /ThO ₂ 4–5 m 9.20 wt% PuO ₂ in ThO ₂ ; 2 wt% Gd ₂ O ₃ in PuO ₂ /ThO ₂	Pu: 8.764; Th: 76.515; O: 12.118; Gd: 2.603 Pu: 8.508; Th: 77.649; O: 12.108; Gd: 1.735 Pu: 7.955; Th: 78.20; O: 12.11; Gd: 1.735	9.79 9.81 9.81
Cladding	0.06 cm thick	Zr-modified 310 stainless steel	As above	7.9
Coolant		Light water	100% H ₂ O	Variable
Liner tube	7.2 cm inner radius; 0.05 cm thick	Zr-modified 310 stainless steel	As above	7.9
Insulator	7.25 cm inner radius; 0.55 cm thick	Yttria stabilized Zirconia	Zr: 66.63; Y: 7.87; O: 25.5	5.37
Outer liner tube	7.8 cm inner radius; 0.05 cm thick	Excel alloy	Sn: 3.5; Mo: 0.8; Nb: 0.8; Zr 94.9	6.52
Pressure tube	7.85 cm inner radius; 1.2 cm thick	Excel alloy	Sn: 3.5; Mo: 0.8; Nb: 0.8; Zr 94.9	6.52
Moderator	25 cm square lattice pitch	Heavy water	99.833 D ₂ O; 0.167 H ₂ O	1.0851
Plutonium isotopics		Reactor grade plutonium	Pu238: 2.75; Pu239: 51.96; Pu240: 22.96; Pu241: 15.23; Pu242: 7.10	
Control blade body	Span 48.4 cm; thickness 1 cm;	Excel alloy	Sn: 3.5; Mo: 0.8; Nb: 0.8; Zr: 94.9	6.52
Control blade absorber	Position on the blade's arm: 16–24 cm; thickness 0.8 cm	Stainless steel 304		7.94

2-group cross sections evaluated at the lattice level (Figure 3). Zero incoming current boundary conditions were applied on the edge of the reflector. The control rods are inserted through the bottom of the core similar to a BWR and their location is depicted in Figure 4 (for 1/8th of the core due to the symmetry). A separate module, VERAview [28], was used for visualization of the transients result.

2.2. Thermal-Hydraulic Model. The original RELAP/SCDAPSIM/MOD4 (RSM4) [22] source code was previously modified to correct for the water properties at supercritical

conditions in [29]. Each of the 336 fuel channels is simulated in this work; however due to limitations in RSM4 not all channels are simulated simultaneously. Therefore the core was divided into eight identical zones and RSM4 simulations were performed for each of the eight regions successively (Figure 5). Each of the eight separate RSM4 models has a fixed inlet boundary condition (25.8 MPa, 350°C), an inlet plenum, 46 parallel channels with individual static orifices, an outlet plenum, and a 25 MPa fixed outlet boundary condition (Figure 6). The power in each channel was split

TABLE 2: Thermal-hydraulic conditions for fuel lattice simulation.

Position (mm)	Upward coolant density ($\text{kg}\cdot\text{m}^{-3}$)	Upward coolant temperature (K)	Cladding temperature (K)	Liner temperature (K)	Insulator temperature (K)	Pressure tube temperature (K)	Downward coolant temperature (K)	Downward coolant density ($\text{kg}\cdot\text{m}^{-3}$)	Fuel temperature (K)	Center tube temperature (K)
4875	67.7	898	1028	869	679	454	625	621.1	1437	761
4625	70.7	875	1015	847	665	450	626	615.6	1443	751
4375	74.4	849	996	823	651	446	628	610.5	1437	738
4125	78.9	823	970	798	635	441	629	606.0	1410	726
3875	84.3	796	940	773	620	436	630	602.1	1371	713
3625	90.8	771	909	750	606	432	631	598.6	1328	701
3375	98.5	748	877	729	593	428	632	595.6	1284	690
3125	108.0	727	845	710	581	424	632	593.1	1242	680
2875	119.0	709	814	694	571	421	633	590.9	1200	671
2625	132.7	694	784	682	563	419	633	589.1	1159	664
2375	149.3	682	756	672	557	417	634	587.6	1120	658
2125	169.7	673	733	665	553	415	634	586.2	1083	654
1875	194.8	667	713	660	550	414	634	585.1	1053	651
1625	225.7	663	697	658	548	414	635	584.0	1033	649
1375	263.4	660	685	656	547	414	635	582.9	1017	648
1125	309.3	659	679	655	547	413	635	582.0	1006	647
875	363.7	657	680	654	546	413	635	581.1	995	646
625	423.8	655	685	651	544	412	636	580.3	976	645
375	483.4	651	689	646	541	411	636	579.7	947	643
125	535.8	644	684	638	536	410	636	579.4	907	640

TABLE 3: Branches definitions for cross-sections interpolation.

Branches	Control rod	Central tube		Fuel	Coolant around fuel pins	
	0 = out 1 = in	Density ($\text{g}\cdot\text{cm}^{-3}$)	Temperature (K)	Temperature (K)	Density ($\text{g}\cdot\text{cm}^{-3}$)	Temperature (K)
0 (reference)	0	$\rho_{\text{ref_cool1}}$	$T_{\text{ref_cool1}}$	$T_{\text{ref_fuel}}$	$\rho_{\text{ref_cool2}}$	$T_{\text{ref_cool2}}$
1	1	-	-	-	-	-
2	0	0.02	-	-	-	-
3	0	0.70	-	-	-	-
4	0	-	600	-	-	-
5	0	-	1000	-	-	-
6	0	-	-	600	-	-
7	0	-	-	2000	-	-
8	0	-	-	-	0.02	-
9	0	-	-	-	0.70	-
10	0	-	-	-	-	650
11	0	-	-	-	-	1000
12	1	0.02	-	-	-	-
13	1	0.70	-	-	-	-
14	1	-	600	-	-	-
15	1	-	1000	-	-	-
16	1	-	-	600	-	-
17	1	-	-	2000	-	-
18	1	-	-	-	0.02	-
19	1	-	-	-	0.70	-
20	1	-	-	-	-	650
21	1	-	-	-	-	1000

FIGURE 3: Reactor physics modelling process (2D view).

FIGURE 4: Control rods location (1/8th symmetric).

FIGURE 5: Channels splitting for RSM4.

between the two fuel rings using a constant ratio (53% and 47% for the inner and outer ring, resp.) based on fuel lattice calculations over the fuel cycle. While this ratio varies with burnup, the values were selected such that the peak power occurs on the limiting elements and hence it is conservative for the MCST and MFCLT calculations. The power imbalance within a fuel ring due to the presence of burnable absorbers in few pins has not been considered here. As a consequence, the MFCLT is not completely conservative.

The conductivity and heat capacity for the inner ring fuel, outer ring fuel, cladding/central tube material, liners, insulator, and pressure tube were taken from [30]. This work used the Dittus-Boelter correlation for convective heat transfer; however many studies have examined that its error may be very large in the vicinity of the pseudocritical temperature. To observe the sensitivity of the MCST to the convective heat

transfer coefficients three separate sensitivity studies were performed. In all sensitivity studies, a null-change transient was performed until the diffusion and thermal-hydraulic models converged to a new steady state. In the first test the convective heat transfer was reduced uniformly by 10% across the entire core and an increase of 20°C was observed. In the second study, the heat transfer coefficient was reduced by 50% for all RSM4 nodes where the coolant temperature was near the pseudocritical point (where deterioration may occur) and the resultant MCST only increases by 1°C. Finally, the conductivity and heat capacity of the fuels were increased by 10% and resulted in a 160°C decrease in MFCLT.

2.3. Control Rod Optimization. A genetic algorithm was used to optimize the insertion depth of the control rods during the batch cycle similar to the optimization performed in [26]. The goal was to obtain an equilibrium cycle with a near-critical core while minimizing the MCST and the MFCLT. The simulations assume a four batch fueling scheme and a shuffling pattern and thermal-hydraulic response surface similar to [26] are used. A series of successive fuel cycles is simulated with the rod optimization performed at each burnup step in each successive fuel cycle. The iterations are terminated when the control rod pattern is constant from cycle to cycle. The insertion of the control rods at beginning of cycle (BOC) and end of cycle (EOC) for the equilibrium case is shown in Figure 7.

A coupled steady state between PARCS and RSM4 was then evaluated at BOC and EOC to obtain a consistent 3D power distribution with the thermal-hydraulic conditions and these coupled states are used as the initial conditions for the transients below.

2.4. PARCS/RSM4 Coupling for Transient Simulation. The synchronization in time between PARCS and RSM4 is of prime importance during a fast transient. A quasistatic process involving very small time steps was used where thermal-hydraulic conditions are assumed constant during a PARCS time step and conversely, the local powers are assumed constant during a RSM4 time step.

The neutronic simulation leads the coupling as it will initiate the reactivity transient so PARCS is first run for a small time step and the new 3D power distribution is extracted. Then, RSM4 simulations are performed for the updated 3D power distribution for the same small time step and the new 3D thermal-hydraulic conditions are extracted. In fact 8 RSM4 simulations are performed for the time step each with 46 channels so that no channel averaging is needed. The extracted powers are then used by PARCS to update all cross sections based on the partial derivatives in each branch calculation done in SCALE (see Section 2.1.1). A diagram of the coupling process is shown in Figure 8. Sensitivity simulations were performed for time steps ranging from 0.005 to 0.1 s with no observed impact on the conclusions.

The control rod drop accident is a localized and fast power excursion transient. Therefore, every fuel channel had to be simulated in RSM4 to obtain accurate predictions in the entire core. Channel grouping is not used in the thermal-hydraulic simulations [31].

FIGURE 6: Canadian SCWR model for thermal-hydraulic studies in RELAP/SCDAPSIM/MOD4.

FIGURE 7: Control rods insertion at BOC (a) and EOC (b).

3. Control Rod Drop Accident

3.1. Transient Setup. The fully inserted, or close to full insertion, control rods at BOC and EOC have been fully removed one by one to calculate their reactivity worth (Table 4). Conversely, the control rods sitting out of the core or barely inserted have been fully inserted one by one to find to highest worth rod (bold values in Table 5). The most effective rod is assumed stuck during simulations where reactor trip is credited as per the International Atomic Energy Agency (IAEA) standards [32].

The drop time of the control rod was evaluated based on its weight and the small pressure difference between the moderator (0.3 MPa) and the containment (atmospheric pressure of 0.1 MPa). The control rod's weight was calculated

to be 334 kg based on its dimensions and material properties. The drop time was evaluated to be 0.8 s using Newton's second law of motion (the hydraulic resistance due to viscosity was neglected in order to provide a conservative estimate on the drop time). The sensitivity to rod drop was assessed by changing the drop time by 50%.

The safety limits of interest are the integrity of the cladding ($T < 1533$ K [29]) and the fuel ($T < 3100$ K). The melting point of thorium-plutonium fuels can be estimated from earlier studies which evaluated the melting point of such fuels using different molar fraction ratios. Fresh fuel's melting point can be estimated to be around 3400 K but will likely decrease as the fuel gets irradiated [33]. Hence 3100 K is used as a conservative estimate of the fuel melting temperature.

TABLE 4: Control rod reactivity worth when fully removed from the core.

Control rod number	BOC (beta = 2.959 mk)									EOC (beta = 3.153 mk)				
	1	2	4	6	8	10	12	15	16	4	5	7	12	15
Reactivity worth (mk)	0.64	0.59	0.60	0.37	0.76	0.84	0.75	0.89	0.55	0.69	0.82	0.67	0.74	0.83
Reactivity worth (\$)	0.21	0.20	0.20	0.12	0.25	0.28	0.25	0.30	0.18	0.23	0.27	0.22	0.25	0.28

TABLE 5: Control rod reactivity worth when fully inserted into the core.

Control rod number	BOC (beta = 2.959 mk)								EOC (beta = 3.153 mk)									
	3	5	7	9	11	13	14	1	2	3	6	8	9	10	11	13	14	16
Reactivity worth (mk)	-0.54	-0.44	-0.40	-0.54	-0.20	-0.47	-0.46	-0.43	-0.45	-0.52	-0.25	-0.52	-0.55	-0.50	-0.24	-0.54	-0.46	-0.32
Reactivity worth (\$)	-0.18	-0.15	-0.13	-0.18	-0.07	-0.16	-0.15	-0.14	-0.15	-0.17	-0.08	-0.17	-0.18	-0.17	-0.08	-0.18	-0.15	-0.11

FIGURE 8: Coupling process for transient simulation.

3.2. *Transients Results.* A null transient was performed to ensure that the coupled initial steady-state kinetics and thermal hydraulics simulations were converged.

3.2.1. *Reactivity Feedback Verification.* The control rod drop accident (CRDA) of the central control rod (number 1 in Figure 4) was investigated for several cases to ensure the feedback models were properly implemented. The CRDA was first simulated without reactivity feedback and then simulated with each of the five types of reactivity feedback separately. Finally all feedback is included such that the integrated response can be assessed. Figure 9 shows the relative power change for each case.

When no reactivity feedback are modelled the prompt jump can be initially observed followed by a slow power increase which is expected as no phenomena are included to alter the power trajectory and the power will end up increasing exponentially. The impact of each reactivity feedback is then evaluated individually and they can be classified

in two categories: negative feedback (fuel temperature and inner coolant density) or positive feedback (outer coolant density/temperature and the inner coolant temperature).

- (i) The fuel temperature Doppler effect shows a strong negative feedback and is near-instantaneous. For the case where the fuel temperature is included as a feedback the reactor power rapidly decreases and stabilizes to approximately 2% higher than the steady-state power.
- (ii) The central tube coolant density feedback also introduces a negative reactivity as shown in Figure 9 although the feedback is smaller and at a slower rate than the fuel temperature case. Here the power oscillates before converging to a power 14% higher than the steady-state power. These oscillations are due to the heat transfer phenomena which are proceeding at a slower time scale than the initial rod perturbation. The heat generated in a fuel pin needs to be

TABLE 6: MCST, MFCLT, and maximum relative power during the CRDA with reactor SCRAM.

(a)										
BOC	Initial	CR1	CR2	CR4	CR6	CR8	CR10	CR12	CR15	CR16
MCST (K)	1119.7	1129.0	1133.7	1129.8	1334.0	1136.6	1141.6	1137.1	1149.3	1142.5
MFCLT (K)	2436.9	2516.5	2549.2	2530.0	3307.9	2570.5	2578.9	2533.5	2536.2	2512.5
Max relative power	1.0	1.208	1.194	1.202	1.121	1.261	1.321	1.261	1.344	1.191

(b)						
EOC	Initial	CR4	CR5	CR7	CR12	CR15
MCST (K)	1168.9	1187.0	1186.4	1190.8	1187.9	1185.8
MFCLT (K)	2447.8	2536	2534	2552	2535	2528
Max relative power	1.0	1.229	1.293	1.214	1.250	1.302

FIGURE 9: Impact of individual reactivity feedback on the power during a control rod drop accident.

transferred through the pin, the cladding, the coolant between the fuel pins, and the central tube, before finally reaching the central tube coolant and causing a perturbation. This delay causes the power transient to initially exceed its new steady-state power during the portions of the transients where power is increasing and vice versa for periods of decreasing power.

- (iii) The central flow tube temperature feedback is positive as higher temperature moderators tend to up-scatter neutrons into the low-energy Pu fission resonances. This feedback is slightly delayed relative to the two other positive mechanisms due to the time delay involved in the heat transfer from the fuel pins through the central flow liner and to the central coolant (similar to the time delay involved in the central density feedback).
- (iv) The coolant in the fuel region also has a positive feedback contribution due to its propensity to up-scatter neutrons into the low-lying Pu fission resonances, although the feedback has an earlier onset since the thermal lag to the coolant in this region is less than that to the central flow tube.
- (v) Finally the fuel region coolant density feedback is also strongly positive and on a time scale similar to coolant temperature changes in this region of the channel.

FIGURE 10: Relative power, MCST, and MFCLT evolution during the CRDA of control rod 15.

These results are consistent with the benchmark study in [14] and confirm the proper implementation of the reactivity feedback.

3.2.2. CRDA with SCRAM. All the fully inserted, or near full insertion, control rods were considered for rod drop analyses at both BOC and EOC (Table 6). A minimum time step of 0.1 s (maximum time step of 0.5 s) was used during these transients based on a parametric study presented below (Table 7). Each control rod starts dropping two seconds after the beginning of the simulation from its initial position and is removed until the out-of-the-core position in 0.8 s (simulation time reaching 2.8 s) by assuming a constant speed fall. The reactor trip signal occurs when the total power reaches 115% full power (FP) and the reactor SCRAM delay is 0.6 s, consistent with LWR studies [34, 35]. During the reactor SCRAM, all the control rods (no shutoff rods were modelled) are linearly inserted into the core in 3 s (approximately speed of 1.6 m/s) with the exception of the rod dropped and the assumed unavailable rod (the most effective rod). A typical transient for the changes in relative power, MCST, and MFCLT as a function of time for the drop of control rod 15 at BOC have been plotted in Figure 10. Note that the MCST

TABLE 7: Parametric study on the CRDA of control rod 15 at BOC.

CR15 at BOC	Time step (s)	Scram signal (% FP)	Scram delay (s)	Drop time (s)	Max relative power	MCST (K)	MFCLT (K)
Case 0: reference	0.02	115	0.6	0.8	1.332	1148.7	2535.6
Case 1: time step decrease	0.005	115	0.6	0.8	1.331	1147.6	2533.1
Case 2: scram signal	0.02	120	0.6	0.8	1.332	1152.4	2544.4
Case 3: scram delay	0.02	115	1.2	0.8	1.332	1173.2	2587.5
Case 4: drop time	0.02	115	0.6	0.4	1.3893	1148.4	2540.1
Case 5: time step increase	0.1	115	0.6	0.8	1.344	1149.3	2536.2

FIGURE 11: Normalized assembly power distribution before the accident (a) and at core maximum power (b) during the CRDA of control rod 15.

and MFCLT were normalized relative to the hot zero power condition (623 K) and slowly converge towards it as the power converges to zero. The normalized radial power distribution has also been plotted before the accident and when the core reaches its maximum power (0.8 s after the beginning of the drop) (Figure 11). The local power excursion near the dropped control rod can be clearly observed with the relative assembly power going from 1.08 up to 1.86 for assembly number 50 (see Figure 3 for assembly numeration). Figure 12 is an axial cut of row 4 at four different times during the transient. At 2.4 s, half of the control rod is still inserted into the core leading to a distorted power distribution towards the top of the core, especially where the control rod is removed (column P and Q). It can be noted that the normalized power remains locally high after the SCRAM ($t = 10$ s) because the dropped rod and the most effective rod are assumed stuck; however, the core total power is significantly reduced with the SCRAM (6.7% FP). Figure 13 shows an axial cut of row 4 for the cladding and fuel centerline temperature at steady state and when they reach their highest value. A slight increase and asymmetry can be observed.

At BOC condition a CRDA involving control rod number 6 never reaches the overpower signal of 115% full power. Therefore, the reactor trip does not register and the core reaches a new steady state with excessively high cladding and fuel temperature. Further analysis of such cases is provided in the next section which considers cases where reactor trip is unavailable.

For all other CRDA cases the safety limits were met. The maximum reactivity insertion was 0.3\$ (see Table 4); therefore supercriticality is avoided. The power excursion can be terminated by the reactivity feedback alone but the reactor SCRAM safely shut down the reactor and limits the MCST increase to 10–30°C and the MFCLT rise to 80–150°C.

Parametric studies on the minimum time step, control rod drop time, overpower signal, and SCRAM delay were carried out to evaluate the sensitivity of the results to those parameters. The CRDA for control rod 15 at BOC was simulated with the modified parameters and the results are shown in Table 7.

Cases 0, 1, and 5 demonstrate the minimal impact of the time step on the studied parameters. A higher time step

FIGURE 12: Snapshots of the axial power distribution of row 4 for the CRDA of CR15 with reactor SCRAM.

tends to overestimate the MCST, MFCLT, and maximum power albeit only to a small extent. Therefore any time step between 0.005 s and 0.1 s is considered acceptable. Increasing the SCRAM signal from 115% to 120% full power also has negligible effects on the transient. The MCST and MFCLT are more sensitive to the time delay associated with the SCRAM signal, but they remain far below the safety limits. Finally, the sensitivity to rod drop timing was examined by decreasing the drop time by 50% and showed that the power increase is more substantial but the MCST and MFCLT are not significantly affected.

3.2.3. CRDA without SCRAM. The CRDAs discussed above were reexamined assuming a rod drop without SCRAM. Such

events involve changes in power, fuel temperature, coolant properties, and so on over the full core. In all cases the power was redistributed within the core and a new steady-state power was reached. The MCST and MFCLT were evaluated to assess the performance of the core and the results summarized in Table 8. For almost all cases the fuel centerline limit was exceeded; hence reactor SCRAM is required for this accident scenario.

The relative core power, MCST, and MFCLT change for the drop of control rod 15 at BOC are presented in Figure 14. This scenario presents the highest cladding temperature increase as well as the highest power increase.

The normalized radial power distribution has been plotted for three states: before the accident (0 s), at maximum

FIGURE 13: Snapshots of the cladding and fuel centerline temperature for the CRDA of CR15 with reactor SCRAM.

core power (2.8 s), and when the MFCLT reached the melting point of 3100 K (9.1 s). As the two first states are identical to the transient without SCRAM they are not plotted here again and only the latter state is shown in Figure 15. After reaching a maximum at 2.8 s, the radial power tilt decreases due to the reactivity feedback. For instance, the relative assembly power in assembly 50 increases from 1.08 to 1.86 before decreasing to 1.61.

A vertical slice of row 4 of the core was used to plot the normalized axial power distribution at 4 specific times of the transient (Figure 16): before the accident (0 s), half-way through the rod drop (2.4 s), rod reaching the out-of-the-core position (2.8 s), and when the MFCLT reached the assumed

melting point (9.1 s). The cladding and fuel centerline temperature were also plotted before the transient and at 9.1 s. The power decreases after the positive reactivity insertion due to the negative power coefficient and stabilizes to a new level (104% FP in this case).

Initially, the MCST is located in this vertical slice (column Q in Figure 17 or assembly 51) but the MFCLT is in assembly 163. The MCST remains in the same location as the transient progresses because the MCST is already located around the failing rod. However, the MFCLT location changes to assembly 50 (column P in Figure 17) two seconds after the initiation of the transient.

TABLE 8: Results for CRDAs without SCRAM.

(a)										
BOC	Initial	CR1	CR2	CR4	CR6	CR8	CR10	CR12	CR15	CR16
MCST (K)	1119.7	1363	1358	1369	1333	1397	1450	1411	1493	1431
MFCLT (K)	2436.9	2855	3139	3097	3307	3478	3787	3379	3735	3577
Max relative power	1.0	1.203	1.189	1.198	1.119	1.255	1.312	1.256	1.333	1.189
New relative steady-state power	-	1.031	1.030	1.029	1.018	1.035	1.034	1.034	1.033	1.024

(b)						
EOC	Initial	CR4	CR5	CR7	CR12	CR15
MCST (K)	1168.9	1429	1450	1364	1400	1458
MFCLT (K)	2447.8	3233	3400	3122	3100	3350
Max relative power	1.0	1.223	1.285	1.209	1.243	1.295
New relative steady-state power	-	1.037	1.037	1.037	1.038	1.038

FIGURE 14: Normalized power, MCST, and MFCLT evolution during control rod 15 drop without SCRAM.

4. Conclusion

A system capable of simulating coupled neutron physics and thermal-hydraulic transients for the Canadian Supercritical Water-cooled Reactor has been developed with PARCS, RELAP/SCDAPSIM/MOD4, and external python scripts for data exchange. Control rod drop accidents have not previously been assessed for this core design and given the fact these accidents are localized, fast, and asymmetric, there is a need for fully coupled 3D neutron kinetic and thermal-hydraulic analysis. The PARCS spatial kinetics model was built using the cross sections and kinetics parameters evaluated as function of burnup and axial elevation at the lattice level and the RSM4 model includes all of the 336 fuel assemblies.

Five types of feedback were considered in addition to the effect of the rod drop. In order to verify the coupled model a series of separate effect feedback analyses were performed where each piece of feedback was considered in isolation. The

FIGURE 15: Normalized radial power distribution reached at $t = 9.1$ s without SCRAM.

strong fuel Doppler effect was found to be very stabilizing and quickly terminated the transient. The coolant density in the central tube also limits the power excursion but oscillations were observed due to the differences between neutronic and heat transfer time scales. While the three other pieces of feedback are positive, the power coefficient remains strongly negative.

The reactivity worth of each rod was evaluated at BOC and EOC when dropped. Even though the delayed neutron fraction is low ($\beta \sim 0.3\%$) due to the high plutonium content, the reactivity insertion remains less than prompt criticality for all drop scenarios. The first set of transients assumed an operational reactor trip when the core power exceeded 115% full power. After a delay of 0.6 second, all the control rods except the failed one and the highest

FIGURE 16: Snapshots of the normalized axial power distribution of row 4 during the CRDA of CR15 without reactor SCRAM.

reactivity worth rod were introduced from their current position to a fully inserted position using a conservative insertion characteristic time. All transients but one activated the reactor trip and safely shutdown without challenging the safety margins. Moreover, a parametric study was carried out on the minimum time step, overpower trip set point, SCRAM delay, and control rod drop time. The maximum relative power, MCST, and MFCLT were found to be fairly insensitive to those parameters. However, while the case which did not trip reached a new steady state and the MCST remained below the safety limit, fuel melting occurred in the assemblies around the dropped rod. As a consequence, other detection techniques and trip parameters are needed. For example, the

local Neutron Power Protection (NOP) system utilized in the existing CANDU fleet could be considered for this design.

While unacceptable from a consequence standpoint (i.e., fuel melting), cases without reactor SCRAM are interesting to assess the overall core properties. The study shows that almost all the transients reached a MFCLT above 3100 K; therefore, fuel melting would occur in most cases and show the necessity for reactor SCRAM demonstrating the need for special safety system action for these events.

Conflicts of Interest

The authors declare that there are no conflicts of interest regarding the publication of this paper

FIGURE 17: Snapshot of the cladding (a) and fuel centerline (b) temperature of row 4 during the CRDA of CR15 without reactor SCRAM.

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