7th International Conference on Modelling and Simulation in Nuclear Science and Engineering (7ICMSNSE)
Ottawa Marriott Hotel, Ottawa, Ontario, Canada, October 18-21, 2015

7ICMSNSE-46

Analysis of a SBO in a CANDU using RELAP/SCDAPSIM/MOD3.6

Elena Dinca[1], Daniel Dupleac[2], Roxana-Mihaela Nistor-Vlad[2], Analia Bonelli[3], L. J. Siefken[4], C. M. Allison[4], J. K. Hohorst[4]

[1]CNCAN National Commission for Nuclear Activities Control, Romania
[2]Politehnica University of Bucharest, Romania
[3]Licensing, Nuclear Safety and Core Design, UG-CNAII-IVCN Nucleoeléctrica Argentina S.A, Lima (B2806AEL) Buenos Aires, Argentina
[4]Innovative Systems Software, 1242 South Woodruff Avenue, Idaho Falls, Idaho 83404 USA
elena.dinca@cncan.ro, ddupleac@yahoo.com, roxanamihaelanistorvlad@gmail.com, abonelli@na-sa.com.ar, lmsiefken@gmail.com, iss@srv.net, jkh@srv.net

Abstract

A new experimental version, RELAP/SCDAPSIM/MOD3.6, is being developed to support the analysis of Pressurized Heavy Water Reactors (PHWRs) under severe accident conditions. It is a derivative of RELAP/SCDAPSIM/MOD3.5, which has the most advanced fuel and severe accident modeling options for LWRs. This paper summarizes the verification and development of input models and minor changes to the code for a RELAP/SCDAPSIM/MOD3.6 analysis and the recommendations for code improvements to develop a robust code for severe accident analysis of a CANDU reactor. The CANDU-specific modeling improvements which include (a) the development of a single channel fuel bundle model, (b) improvements to the SCDAP fuel rod and shroud component models for horizontal fuel bundles, calandria tubes, and pressure tubes, and (c) improvements to the COUPLE porous media module to predict the latter stages of the accident. This paper will also discuss the verification testing of the models by comparing predicted results to LWR experiments and CANDU specific codes, and the analysis of a station blackout in a CANDU NPP.

Keywords: Thermal hydraulics, Fuel and Fuel Channels, Analysis, and Code Validation, etc.

1.0 Introduction

RELAP/SCDAPSIM, designed to predict the behavior of reactor systems during normal and accident conditions, is being developed at Innovative Systems Software (ISS) as part of the international SCDAP Development and Training Program (STDP). RELAP/SCDAPSIM uses the publically available SCDAP/RELAP5 [1,2] models developed by the US Nuclear Regulatory Commission (NRC) in combination with proprietary (a) advanced programming and numerical methods, (b) user options, and (c) models developed by ISS and other STDP members [3].

RELAP/SCDAPSIM is designed to describe the overall reactor coolant system (RCS) thermal hydraulic response and core behavior under normal operating conditions or under design basis or severe accident conditions. The RELAP5 models calculate the overall RCS thermal hydraulic response, control system behavior, reactor kinetics, and the behavior of special reactor system components such as valves and pumps. The SCDAP models calculate the behavior of the core and vessel structures under normal and accident conditions. The SCDAP portion of the code includes user-selectable reactor component models for LWR fuel rods, Ag-In-Cd and B_4C control rods, BWR control blade/channel boxes, electrically heated fuel rod simulators, and general core and vessel structures. The models calculate the damage progression in the reactor core: heat-up, oxidation and meltdown of fuel rods and control rods, ballooning and rupture of fuel rod cladding, release of fission products from fuel rods and disintegration of fuel rods into porous debris and molten material. The SCDAP portion of the code also includes models to treat the later stages of a severe accident including debris and molten pool formation, debris/vessel interactions, and the structural failure (creep rupture) of vessel structures. The latter models are automatically invoked by the code as the damage in the core and vessel progresses.

RELAP5 has been widely used for the analysis for LWRs, but the unique design of the CANDU reactor has presented a challenge for the code. Since the late 1990s when Professor J. T. Rogers at Carleton University in Ottawa, Canada, and several graduate students developed an input model for the Calandria Vessel using the COUPLE module in RELAP/SCDAPSIM, code improvements and input models for CANDU reactors have been developed by ISS and the "Politehnica" University of Bucharest in Romania.

A new experimental version of RELAP/SCDAPSIM/MOD3.6, is being developed to support the analysis of Pressurized Heavy Water Reactors (PHWRs) under severe accident conditions. This version is a derivative of RELAP/SCDAPSIM/MOD3.5, which has most advanced fuel and severe accident modeling options for LWRs. This paper will give a summary CANDU analyses performed using RELAP/SCDAPSIM and the validation of the code results against LWR experiments and other codes, such as CATHENA, MAAP CANDU, and/or ISSAC; a brief summary of the new code models and their impact on the analysis of heavy water moderated fuel channel reactor; and finally the conclusions. LWR experiments were used to verify the new models due to the lack of CANDU specific experiments.

2.0 Input Model Development and Validation

The development of input models for RELAP/SCDAPSIM for the safety analysis of CANDU reactors began in 2004 by the Politehnica University, Bucharest, Romania, in an effort to demonstrate that the code could adequately predict CANDU reactor behaviour. An input model for a CANDU reactor, developed in Romania, has been used in the analysis and verification of CANDU models with only slight variations up to the present. The RELAP5 nodalization used for analysing design and beyond design basis accidents is shown in Figure 1.

In 2005 Prisecaru, Dupleac and Biro [4] simulated several postulated transients using RELAP/SCDAPSIM to determine if the best-estimate code RELAP/SCDAPSIM, with some CANDU modifications, could be used to calculate major design basis accidents during the licensing review and/or the safety evaluation of operational transients and incidents in the CANDU reactors in Romania. The calculations reported in this paper included 1) a steady state natural circulation in the heat transport system, 2) a 35% inlet header break, and 3) a 100% steam header break. Results from these calculations were compared to results from similar sequences analysed with CATHENA, a CANDU specific code. Table 1 compares the RELAP calculated values for steady state to those calculated by CATHENA and the design values.

Results from these calculations show that a mechanistic code such as RELAP/SCDAPSIM can be used to predict CANDU reactor behaviour. Figures 2 and 3 show representative results from these calculations.

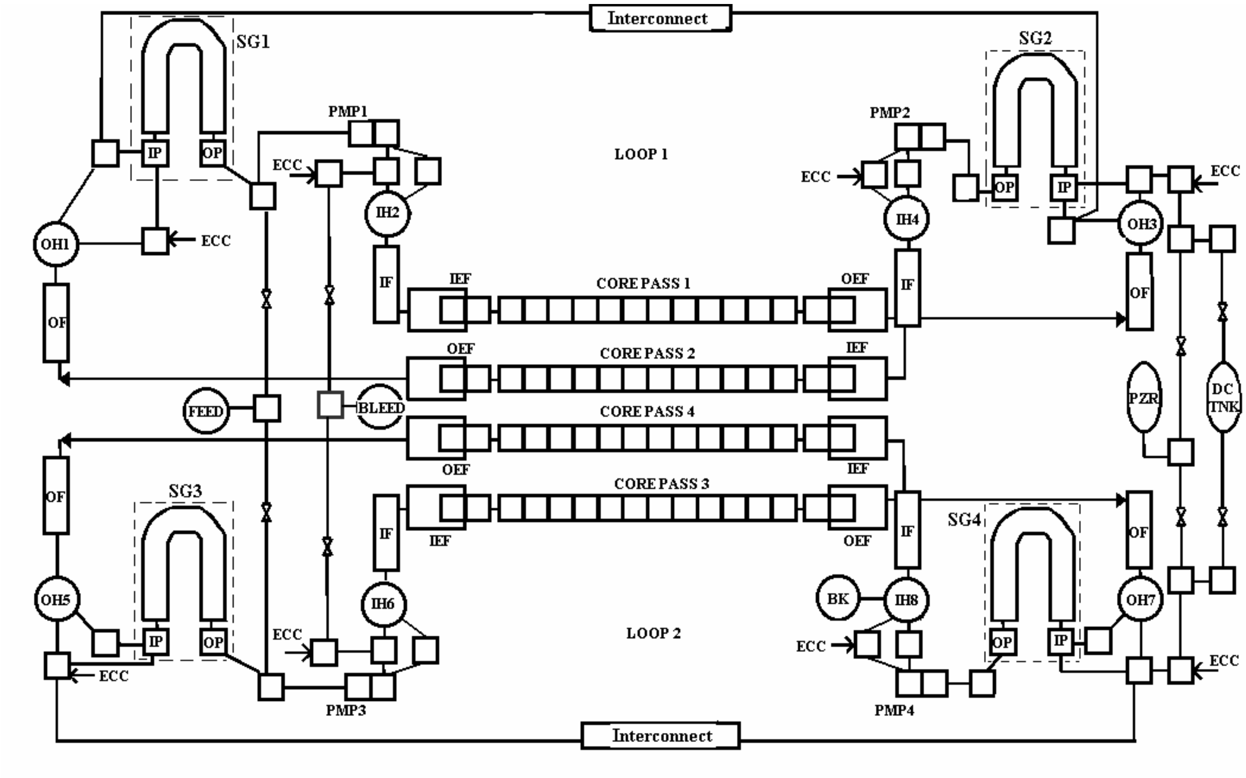

Figure 1: CANDU 6 Heat Transport System nodalization

In 2008 Dupleac etal [5] presented a paper at ICAPP 2008 assessing the ability of the COUPLE module in RELAP/SCDAPSIM to predict the coolability of a severely degraded CANDU6 core. The results from two loss-of coolant sequences, one calculated by ISSAC and the other by MAAP-CANDU were used as input to the COUPLE module in RELAP/SCDAPSIM for this study. The case calculated by ISSAC used for this study was a loss-of coolant sequence without any recovery action [6] and the MAAP-CANDU case was a steam generator multiple tube rupture accident [7]. The COUPLE module in RELAP/SCDAPSIM, using the no-slump option, can perform independent lower head thermal and mechanical analyses, thus avoiding the necessity of performing a full plant calculation.

The authors concluded in this work that RELAP/SCDAPSIM could predict results that are consistent with the results from the CANDU specific codes, but additional work will continue on developing an accurate calandria couple model and modifying the COUPLE module in the code to accurately predict calandria behaviour. The case using the conditions in the reactor at the time the core collapses into the calandria vessel calculated by ISSAC as input into the RELAP/SCDAPSIM COUPLE model, the code predicted the CV failure to be about 126,300 s after initiation of the accident, whereas ISAAC predicted CV failure at 127,080 s. For the case using the conditions when the core collapses into the calandria vessel calculated by MAAP-CANDU for input into the RELAP/SCDAPSIM COUPLE module the code predicted CV failure at about 140,100 s after initiation of the accident, whereas MAAP4-CANDU predicted CV failure at 130,557 s.

Table 1: Design values and calculated values of some important parameters used in RELAP and CATHENA steady-state calculations

Parameters	Design values of CANDU 6	RELAP5 calculated values	CATHENA calculated values
Reactor Inlet Header Pressure (MPa) Temperature (°C)	 11.35 266.6	 11.34 265.6	 11.36 268
Reactor Outlet Header Pressure (MPa) Temperature (°C) Flow Quality (%)	 9.99 310 < 4	 9.995 310 2.75	 10.03 310.5 3.7
Core pass Flow (kg/s)	1900	1973.5	1913
Primary Pump Suction Pressure (MPa)	9.54	9.53	9.59
Primary Pump Head (MPa)	1.84	1.82	1.80
Steam Generator Steam Pressure (MPa) Temperature (°C) Level (m) Steam Flow (kg/s) (per SG)	 4.695 260 14.222 266	 4.68 260 14.192 260.34	 4.70 260 14.007 258.6

.

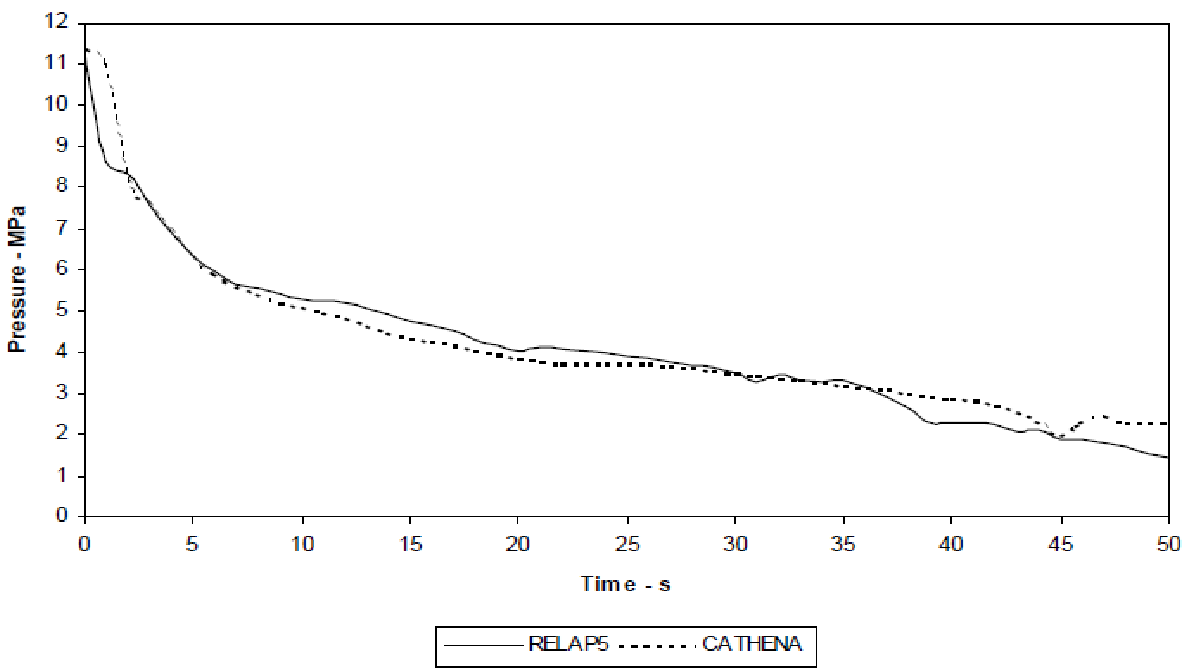

Figure 2. Broken inlet header pressure

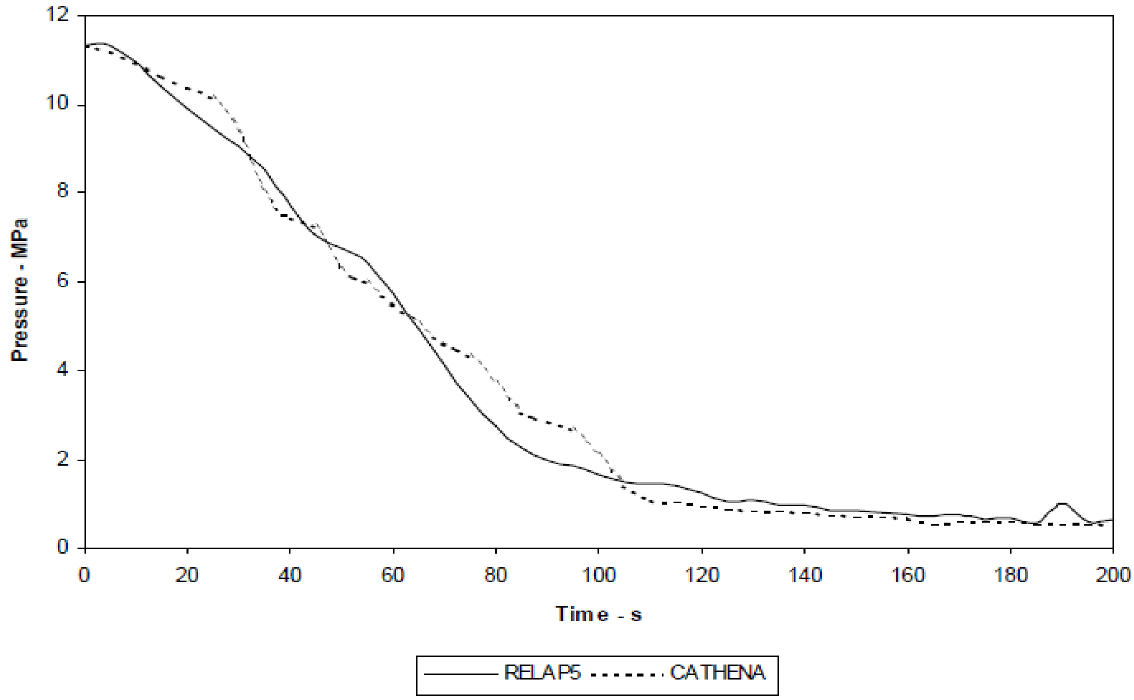

Figure 3. Inlet header pressure transients

In 2008 Mladin and Dupleac published a paper in Nuclear Engineering and Design [8] that indicated that RELAP/SCDAPSIM gives results similar to those obtained using CANDU specific codes. The deficiencies in the code regarding CANDU specific geometry and behaviour modelling were listed as well as where code or model improvements specific to CANDU or channel type reactors could be made.

In 2009 results of a LBLOCA analysis, with the assumed unavailability of several critical safety systems in a CANDU6, performed by RELAP/SCDAPSIM and a comparison of these results to those obtained with ISSAC and MAAP-CANDU was presented by Dupleac at ICONE-17 [9]. The study concluded that 1) RELAP/SCDAPSIM could adequately calculate a severe accident in a CANDU plant; 2) that the SCDAP core components can be used to model CANDU fuel channels; 3) that the COUPLE module needed to be used independently to calculate calandria behaviour; 4) identified areas where code model improvements should be made; and 5) the comparison of the RELAP/SCDAPSIM results to ISSAC and MAAP-CANDU showed differences in the timing of events in the LBLOCA. It was determined that the differences resulted from a more detailed thermal hydraulic model to simulate the PHTS and the more detailed nodalization of the reactor when RELAP/SCDAPSIM was used for the calculation. The comparison of code results is shown on Table 2.

Also in 2009 Mladin [10] developed several new CANDU specific models for RELAP/SCDAPSIM including modifications to the subroutine LIQSOL to account for the horizontal geometry of the CANDU reactor and a horizontal relocation model for metallic melt and presented test results at ICONE-17 [11]. Calculated results using a version of RELAP/SCAPSIM where these models were implemented were compared to the results of the CANDU fuel deformation experiments which were performed using a single CANDU fuel bundle with no internal heat generation. Comparisons were made to the measured temperatures, hydrogen accumulation, and to the post-test visual inspections of the bundle. Results from these calculations were presented at ICONE-17.

Event	ISAAC	MAAP4-CANDU	RELAP5/SCDAP
ROH guillotine rupture	0.0	0.0	0.0
LOCA signal	8.28	6	7
HPECC initiation		6	14
HPECC terminated & MPECC initiation	180	98	229
MPECC terminated	2,340	1,100	1170
Steam generator is dry (intact loop)	3,420	2,614	3,320 and 3,584
Fuel bundles are uncovered inside fuel channels		2,860	1,954
Steam generator is dry (broken loop)	19,440	14,386	Not dry
At least one channel is dry		17,940	2,482
CV water pool is saturated		19,826	5,990
Fuel bundles are uncovered inside fuel channels in intact loop		22,060	7,620
At least one channel is dry in intact loop		25,180	8,500
CV rupture disk is open		26,236	7,100
PT and CT rupture, intact loop	9,720	35,066	8,900
Beginning of the core disassembly		36,381	10,969
Core collapse onto the CV bottom	17,640	50,781	12,485
Water is depleted inside CV	39,240	68,068	21,540
CV bottom wall failed due to creep	146,160	199,308	152,910

Table 2. Time Sequence of Significant Events for Large Break LOCA (s)

The results of the comparison to the experiments indicated that acceptable temperature results were obtained from RELAP/SCDAPSIM taking into account the lack of a transversal plate model that could be used to model the end plates and a homogeneous shroud model. A significant difference was noted between the calculated hydrogen production and the measured hydrogen production for the case that best matched the measured temperatures (the 1327^{o}C preheated steam) which could be explained in part by the experimental increases in the steam temperature in small increments at unknown time intervals, not modelled in the code's input deck. The code's calculated horizontal relocation was similar to that observed in the experiment. Results from these experiments indicated additional models were needed to accurately represent a CANDU.

At Top Fuel in 2009 Dupleac [12] presented a paper on the effects of how the fuel bundle was modelled for a severe accident calculation using RELAP/SCDAPSIM. Four different input models were used to characterize the Stage2 severe accident where the fuel channels remain uncovered by the moderator. The results obtained were compared in terms of temperature distribution across the fuel bundle, hydrogen production and the timing to reach the criteria for bundle and fuel channel disassembly. Figure 4 shows how the fuel channel was modeled from a very simple model to a more detailed model. The figure shows the two single channel modes and the two four channel models that were used for this study. Model 1, the simplest, assumed that all fuel elements in the bundle behaved in the same manner, thus only one SCDAP fuel rod model was used. Model 2 used four different SCDAP fuel rod components to model the fuel bundle in four rings, therefore the actual power peaking factors could be input, but since the bundle was modeled as a single pipe each fuel element was subjected to the same thermal hydraulic conditions. Model 3 used four SCDAP fuel rod components as the second model with four pipe components with cross flow junctions connecting the pipes to model the fuel channel with an inter-channel simulated, therefore different thermal hydraulic conditions could be simulated at each fuel component boundary. Model 4 was the same as Model 3 but employed the newly developed models that allowed bundle slumping in the fuel channel.

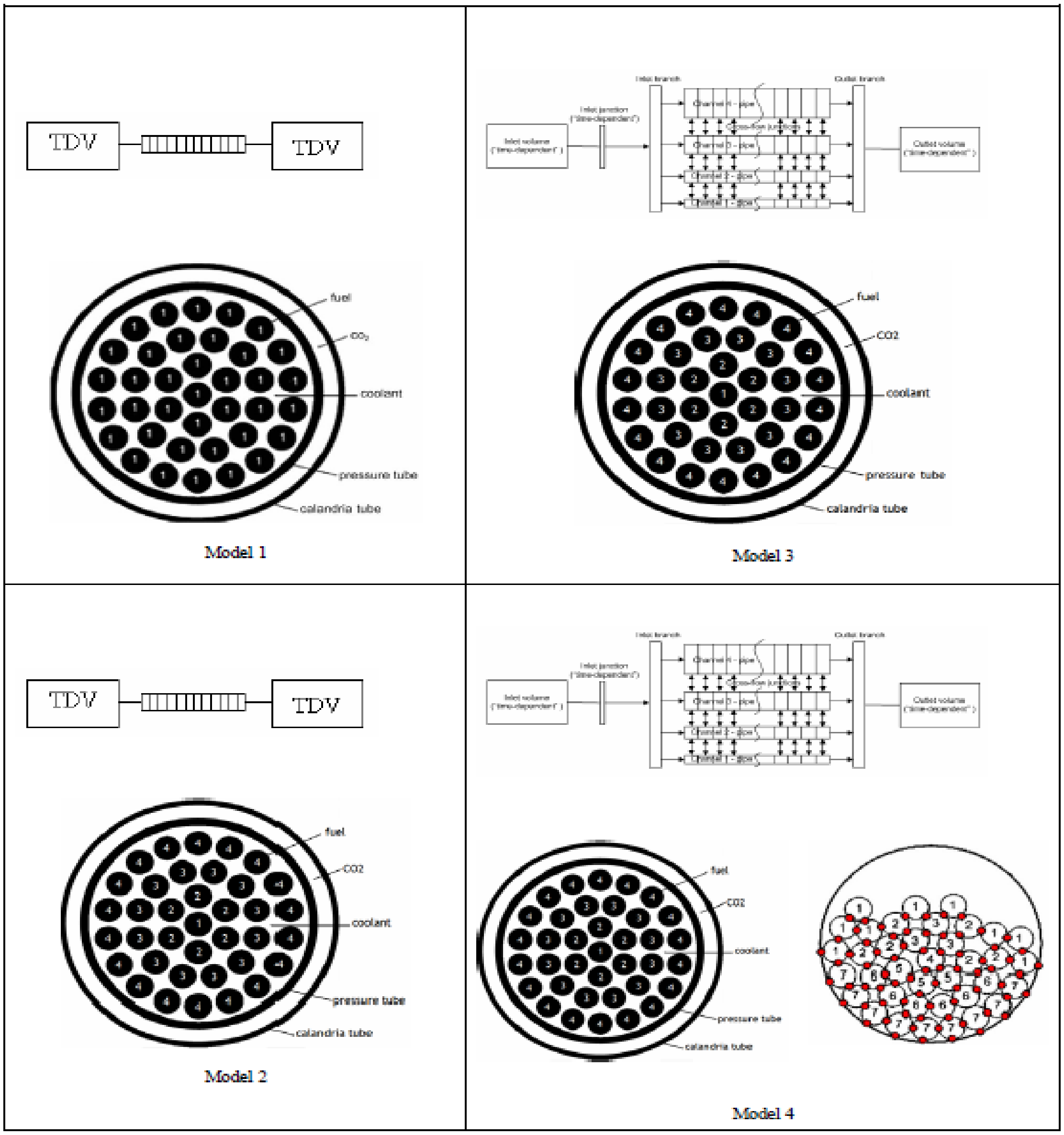

Figure 4. Fuel Bundle and Fuel Channel Model

Results from this analysis indicated 1) that the model used to simulate the fuel channel and the channel's power are irrelevant to the time at which channel disassembly and the collapse of the fuel begins; 2) the quantity of fission products released is only slightly influenced by the model used; 3) the amount of hydrogen generated is influenced by the model used and the power input with the most detailed model giving the best results; 4) for fast transients such as a LBLOCA Model 3 and Model 4 give similar temperature and hydrogen generation results, except that Model 4 predicts the relocation of the fuel into the fuel channel; 5) for a SBO all models give similar results; and 6) that additional sensitivity studies should be performed.

In 2011 Dupleac [13] reported on a series of parametric studies performed to quantify the effect of several identified sources of uncertainty on calandria vessel failure with RELAP/SCDAPSIM. The main outcome of the calculations was the identification of possible sources of uncertainty and where new or modified models were needed to improve the accuracy of results. Also the results obtained confirmed that heat removal from the Calandria vessel wall by the Reactor vault water is possible as long as a liquid phase is present in the tank.

The analysis gave valuable information for the improvement of the RELAP/SCDAPSIM code models to better simulate a severe accident in CANDU plants. However, in the absence of integral experiments to study the core collapse into the CV and its subsequent dry-out and heat-up, uncertainty in modeling these phenomena will remain.

In 2012 a Coordinated Research Project (CRP) was completed to benchmark severe accident analysis computer codes used for consequence analysis of HWRs. The purpose of the benchmark was to compare the integrated effects of embedded models in the codes, gain an understanding of their limitations, assess the level of uncertainties, and increase the confidence in severe accident code predictions [14]. Seven organizations took part in the benchmarking study, among them: “Politehnica” University of Bucharest (PUB), which used RELAP/SCDAPSIM Mod 3.4, Korea Atomic Energy Research Institute (KAERI), which used ISAAC 4.02, and Atomic Energy of Canada Limited (AECL) which used MAAP4-CANDU v4.0.6A.

The main outcome of the coordinated research project was good agreement on timing of events among participants at the less complex, early stages of the sequence, whereas the processes involving more complex latter stages of the sequence had poor agreement and that more complex phenomena like core disassembly, debris oxidation and corium-vessel interactions require further research to improve the understanding [14]. However, as is shown in Table 3 and in Figure 5 there is a reasonable agreement between results obtained by PUB using RELAP/SCDAPSIM Mod 3.4 and the CANDU severe accident specific codes ISAAC and MAAP4-CANDU

.In 2013 Dinca [15] investigated the influence of fuel channel modelling on hydrogen production during a simulated severe accident in a CANDU reactor. It was determined that the model used to represent the fuel channel had little influence on the start or finish of fuel disassembly validating the results of an earlier study by Dupleac. Results also indicated that it was necessary to represent the whole core as the power distribution effected hydrogen generation. Looking at transients such as a SBO or a LBLOCA the study verified that for slow transients, such as a SBO, the model used had little effect on hydrogen generation, whereas for a fast transient such as a LBLOCA the quantity of hydrogen generated is lower than for a SBO and that a more detailed model is necessary for an accurate calculation

Event	Event Timing (h)		
	PUB	AECL	KAERI
Class IV and Class III Power loss	0	0	0
LRVs open for the first time	2.41	2.31	1.73
SG secondary sides are dry	2.08	2.65	2.07
Pressurizer empty	3.05	4.04	2.83
At least one channel is dry (complete boil-off)	3.22	4.92	2.94
Pressure and calandria tubes are ruptured	3.63	4.97	2.83
Calandria vessel rupture disks #1-4 open	3.63	5.02	2.83
Moderator reaches saturation temperature	3.63	5.04	3.33
Beginning of the core disassembly	5.02	5.51	4.40
Core collapse to calandria vessel bottom	7.85	14.63	42.0
Water is depleted inside calandria vessel	11.58	14.62	10.6
Water in calandria vault reaches saturation temperature	15.75	14.87	14.5
Calandria vessel failed	50.55	55.6	44.7
Water is depleted inside calandria vault	51.5	63.78	49.3

Table 3. Time Sequence of Significant Events for SBO (s) (14)

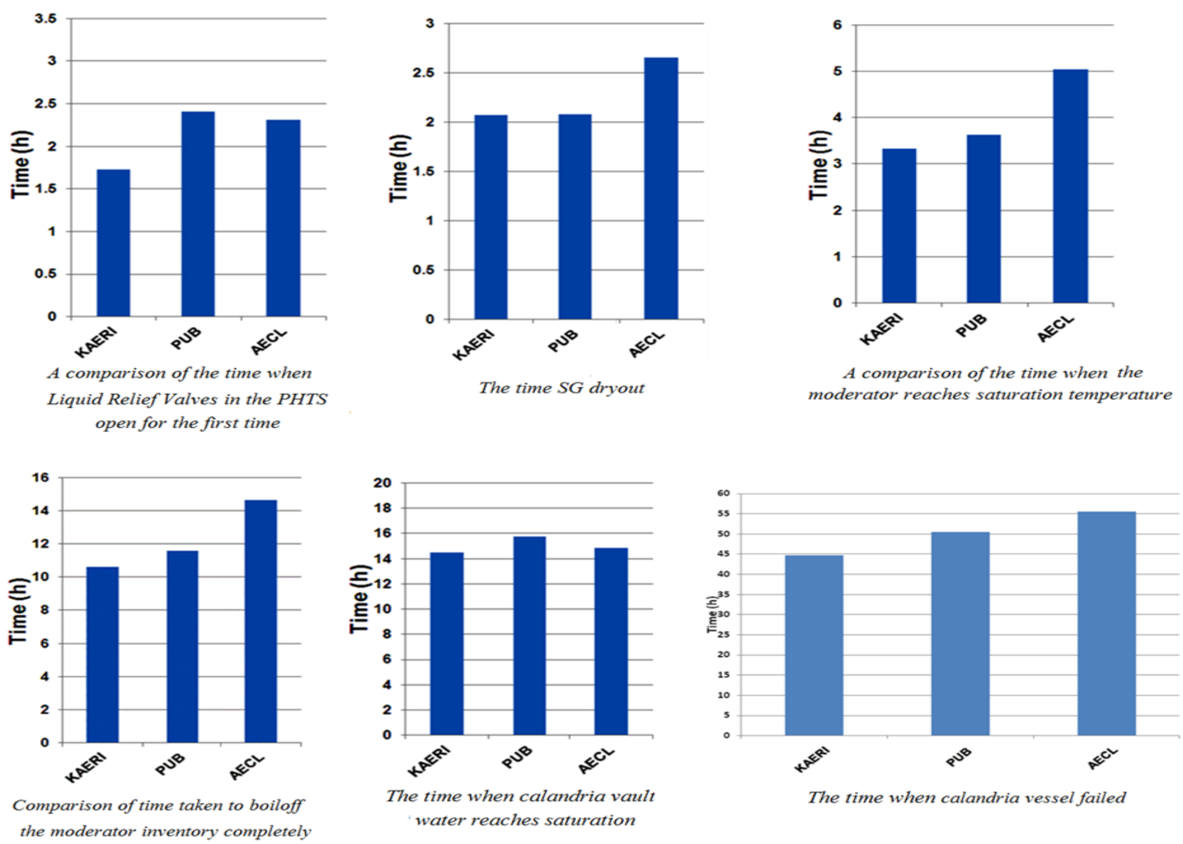

Figure 5. Comparison of various event time calculated by participants

3.0 New Models Specific to Heavy Water Fuel Channel Reactors.

During work for the safety analysis of ATUCHA2 in Argentina the ability to model a heavy water moderated reactor (PHWR) with separate fuel channels containing multiple fuel rods was implemented into the code. (16) The fuel channel model later was adapted for the horizontal fuel channels in CANDU reactor. A CANDU fuel channel contains 12 short (0.5m) fuel bundles with 37 fuel rods in each bundle. Each fuel channel consists of a 104 mm in diameter, 4.3 mm thick Zirconium-Niobium pressure tube inserted into a slightly larger calandria tube (thinner than the Zirconium pressure tube), with two stainless steel fittings at the ends of the fuel channel. The COUPLE module was adapted to predict the layering of melt in the calandria vessel. The extensions to the code implemented for separate fuel channels are describe in the following section.

3.1 Extensions to RELAP5/SCDSIM for analysis of CANDU reactor

Several extensions and modifications were made to the RELAP5/SCDSIM code for the accident analysis of CANDU reactors. These extensions allow modeling of damage progression unique to CANDU reactors, such as (1) ballooning of pressure tube into calandria tube; (2) rupture of pressure tube and calandria tube and the resulting flow of fluid from inside the pressure tube to the calandria vessel; (3) melted cladding material slumping perpendicular to longitudinal axis of the fuel rods; and (4) extensions to the COUPLE module to handle detailed heat transfer of relocated material in a flat and thin in height shape to the walls of a lower head or calandria vessel. Future extensions will include the modeling of the effect of sagging of fuel rods and sagging of calandria tubes.

First, modeling was extended to calculate the onset of the ballooning of the pressure tube into the calandria tube. The onset of ballooning is determined when the threshold temperature for ballooning and pressure of the fluid in the pressure tube is attained. At this time the values for the threshold temperature and pressure are defined by input to the code by the user. As experimental data describing the relation of the onset of ballooning of the pressure tube with respect to the temperature and internal pressure of the pressure tube becomes available, then this relationship will be applied and the input data omitted.

If ballooning of the pressure tube is calculated to occur, then the thermal resistance of the gap between the pressure tube and calandria tube is lost. The loss of the gap thermal resistance between the pressure tube and the calandria tube is calculated by the difference between the gap conductance using the user defined gap and the composition of the gas in the gap between the pressure tube and calandria tube, taking into account radiation heat transfer, prior to ballooning, and the gap conductance between the pressure-calandria tube interface.

Second, modeling was extended to calculate the possibility of rupture of the combined pressure tube and calandria tube and the consequence of the rupture. The code compares the calculated temperature at the inner surface of the pressure tube and the pressure of the fluid in the pressure tube to the rupture temperature and pressure of the tube and if the rupture temperature and pressure is exceeded the tubes rupture. If the code indicates that a rupture has occurred by setting an index equal to 1, a trip valve connecting the pressure tube to the calandria tube opens.

Third, the modeling was extended to calculate radiation heat transfer between the outer surface of the CANDU pressure tube and the inner surface of the CANDU calandria tube. During an accident, this heat transfer accelerates the heat-up of the calandria tube and subsequently oxidation on the outer surface of the calandria tube. A model has been added to the code to calculate the radiation heat transfer occurring across the gap between the pressure tube and calandria tube.

A SCDAP shroud component identifies the location of the gap between the pressure tube and calandria tube by reviewing the user-defined materials for each radial mesh of the component representing the nested pressure tube and calandria tube. Radial meshes with a density of 10 kg/m^3 or less are considered to be composed of gas and thus these radial meshes are overlaying the gap between the pressure tube and calandria tube. In this sequence of meshes, the thermal conductivity of the gas is increased beyond the thermal conductivity of the gas in the meshes to account for the radiation heat transfer across the meshes (gap). The effective conductivity for each radial mesh at each axial node overlaying the gap between the outer surface of the pressure tube and the inner surface of the calandria tube is calculated as are the emissivities of the outer surface of the pressure tube and the inner surface if the calandria tube. The emissivities are calculated using a material properties subroutine taking into account the temperature of the surfaces and a nominal oxide thickness of 1 micron.

Fourth, in the event of melting of the pressure tube and its surrounding calandria tube at an axial node, radiation heat transfer is not calculated between the enclosed fuel rods and the surrounding pressure tube. The pressure and calandria tube material at that axial location are assumed to have slumped into the calandria vessel. At this axial node, the temperatures of the radial mesh points for the pressure and calandria tube are set to the temperature of the fluid inside the pressure tube at that axial node.

Fifth, an extension in modeling was made for the calculation of the oxidation on the outside surface of the calandria tube. In some scenarios as in a CANDU SBO, the oxidation of the outer surface of the calandria tube may occur and significantly affect damage progression. This oxidation and the heat generation caused by oxidation at any location ceases when the Zr at that location slumps away. The code was extended to model this slumping into the calandria vessel.

4.0 Results from a SBO analysis using the new models

To validate the new models implemented in RELAP/SACDAPSIM/MOD3.6 a Station Blackout calculation was performed up until the time of water depletion in the calandria vessel using the nodalization shown in Figure 1. Table 4 summarizes the timing of predicted events occurring during the SBO as calculated by the code. Several representative calculations will be described in the following paragraphs.

At the start of the Station Blackout (SBO) initiated by a reactor trip, the Primary Heat Transfer System (PHTS) loop isolation valves remained open. Initially the pressure in both reactor loops decreased with time as heat extracted from the coolant at the steam generators was greater than the heat input into the coolant. Heat transfer from the PHTS to the steam generators determined the boil off of fluid in the steam generators. Figure 6 shows the predicted water level in the steam generators during the accident.

Once the steam generators dried out, reactor decay heat boils off the coolant in the fuel channels leading to dry out. Once the fuel channels have dried out a rapid increase in temperature is observed leading to rupture of the pressure and calandria tubes. Once these tubes rupture, a blowdown from the

PHTS into the calandria vessel is predicted with a rise in temperature leading to oxidation of fuel cladding and the release of Hydrogen (Deuterium). Figure 7 shows the predicted PHTS D_2O inventory and Figure 8 shows the predicted rate of deuterium release and the total release of deuterium in Kgs.

Event	Event Timing (h)	Event Timing (s)
Class IV, Class III Power loss (Emergency Power System unavailable)	0.0	0
Reactor trips	0.0	0
SG secondary sides are dry	2.13	7670
LRV's open for the first time	2.525	9090
Pressure and calandria tubes are ruptured	4.214	15171
Calandria vessel rupture disks #1-4 open	4.214	15171
Beginning of the core disassembly	5.255	18917
Core collapse to calandria vessel bottom	7.875	28351
Water is depleted inside calandria vessel	11.072	39862

Table 4. Predicted timing of events in a Station Blackout calculation

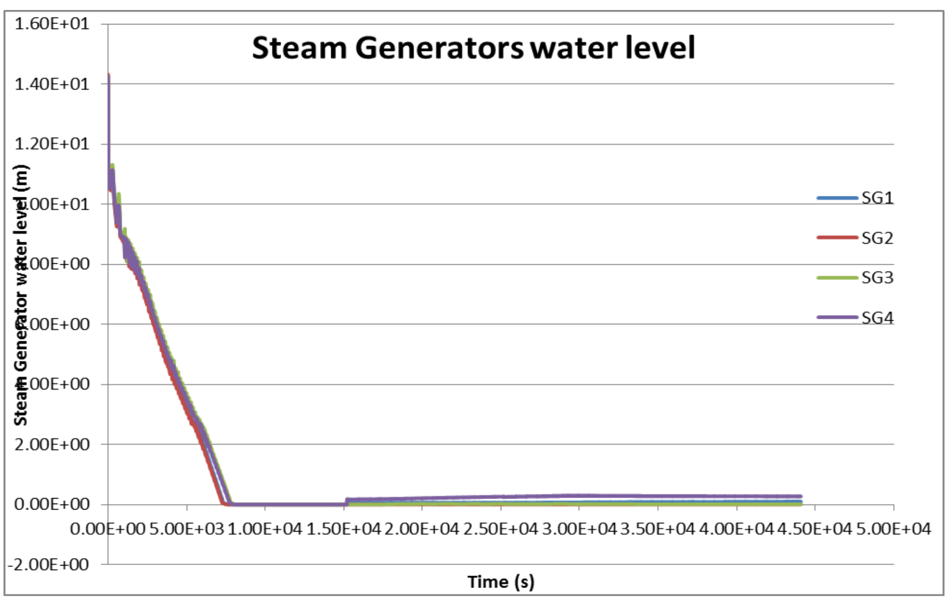

Figure 6. Predicted water level in the steam generators

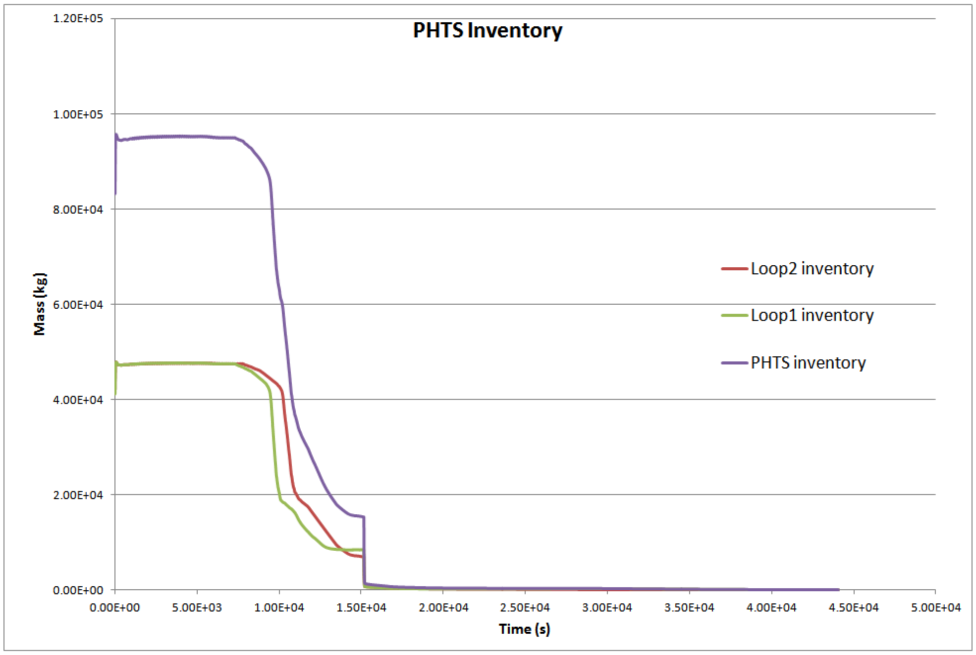

Figure 7. Predicted D_2O Inventory in the Loops and PHTS

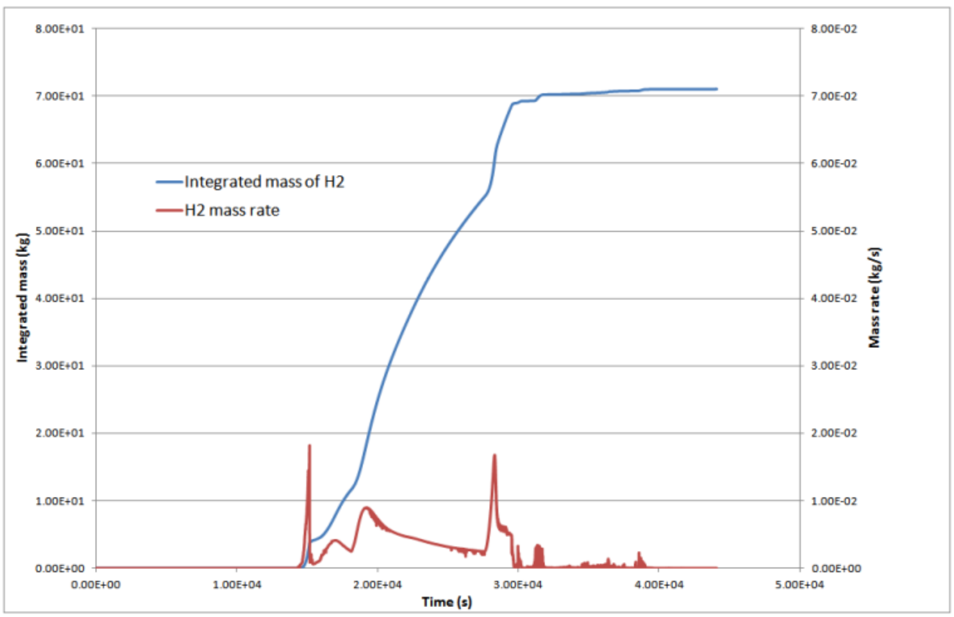

Figure 8. Predicted H_2 ($D_{2)}$ Generation Rate and Integral D_2 Production.

5.0 Conclusions

The modelling improvements to RELAP/SCDAPSIM recommended through the analysis of severe accidents in CANDU reactors using an earlier versions of the code identified areas where new or improved models were needed. The new models implemented into the code along with more detailed input models improved the codes ability to predict the damage progression during a simulated severe accident in a CANDU reactor. Due to the lack of CANDU specific experiments, validation of the new models was limited to the comparison of predicted results to LWR experiments, such as PHEBUS and QUENCH, and to the predicted results for a similar accident obtained from CANDU specific computer codes, such as CATHENA and MAAP-CANDU. The new models included ballooning of the pressure tube into the calandria tube, rupture of the pressure and calandria tubes with fluid flow from the calandria tube into the pressure tube, improvements to the shroud model to model the gap between the pressure and calandria tubes, the slumping of cladding material into the calandria vessel with Zircaloy oxidation as the temperature of the relocated material rises, and extensions to the COUPLE models. It was shown that the code could predict an SBO in a CANDU reactor. As results from more CANDU specific experiments become available, the results from these experiments will be used to validate the code for CANDU reactors and a more detailed SBO analysis will be performed.

6.0 Acknowledgement

The authors would like to acknowledge the early work by J. T. Rogers and his graduate students at Carleton University, Ottawa for developing the initial COUPLE model for the calandria vessel and his faith that COUPLE could be used for the calandria vessel.

7.0 References

1.0 L. Siefken, E. Coryell, E. Harvego, and J. K. Hohorst, SCDAP/RELAP5/MOD3.3 Code Manual, NUREG *NUREG/CR-6150 – INEL-96/0422 – Revision 2*.

2.0 www.relap.com

3.0 C. M. Allison, J. K Hohorst, "*Role of RELAP/SCDAPSIM in Nuclear Safety*", *TOPSAFE,* Dubrovnik, Croatia. September 30-October 3, 2008.

4.0 I. Prisecaru, D. Dupleac, and L. Biro, "*A Study of RELAP5 Capability to Perform the Analysis of CANDU 6 Reactors Accidents*" Proceeding of the 11th International Meeting on Nuclear Reactor Thermohydraulics (NURETH-11), Avignon, France, October 2-6, 2005.

5.0 D. Dupleac I Prisecaru, M. Mladin, and G. Negut, "*Investigation on Coolability of Severely Degraded CANDU6 Core – Preliminary Results",* Proceedings of ICAPP '08, Anaheim, CA USA, June 8-12, 2008.

6.0 S.Y. PARK, Y.H. Jin, and Y.M. Song, *An Investigation of an In-Vessel Corium Retention Strategy for the Wolsong Pressurized Heavy Water Reactor Plants*, Nuclear Technology vol. 158, pg. 109-115, (2007)

7.0 S.M. PETOUKHOV, B. Awadh and P.M. Mathews, *CANDU 6 Severe Core Damage Accident Consequence Analysis for Steam Generator Tube Rupture Scenario using MAAP4-CANDU V4.0.5A:Preliminary Results*, Proceedings of ICAPP '06, Reno, USA, June 4-8, (2006).

8.0 M. Mladin, D. Dupleac, I. Prisecaru, "*SCDAP/RELAP5 application to CANDU6 fuel channel analysis under postulated LLOCA/LOECC conditions*", Nucl. Eng. Des. (2008), j.nucengdes 20008.10.014.

9.0 D. Dupleac, M. Mladin, I. Prisecaru, "*SCDAP/RELAP5 Analysis of Progression of Severe Accident in CANDU 6 Plant*", Proceedings of 17th International Conference on Nuclear Engineering, ICONE-17, Brussels, Belgium, July 12-16, 2009.

10.0 M. Mladin et al., *Two New Models in SCDAP for CANDU Fuel Channel under Severe Accidents*, Topical Conference on Safety of Nuclear Installations, 30.09- 03.10.2008, Dubrovnik, Croatia.

11.0 M. Mladin, D. Dupleac, I. Prisecaru, "*Verification of Modified SCDAP/RELAP5 Against CANDU Fuel Deformation Experimental Results*", Proceedings of 17th International Conference on Nuclear Engineering, ICONE-17, Brussels, Belgium, July 12-16, 2009.

12.0 D. Dupleac, M. Mladin, I. Prisecaru, "*Effect of CANDU Fuel Modelling on Severe Accident Analysis*", Proceedings of Top Fuel 2009, Paris, France, September 6-10, 2009.

13.0 D. Dupleac, M. Mladin, I. Prisecaru, "*Sensitivity Studies on Uncertainty Parameters and Code Modelling of Calandria Vessel Integrity during Late Phase CANDU 6 Severe Accident*", Proceedings of ICAPP 2011, Nice, France, May 2-5, 2011.

14.0 Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications, IAEA TECDOC SERIES, TECDOC No.1727, International Atomic Energy Agency, Vienna, 2013.

15.0 E. Dinca, D. Dupleac, I. Prisecaru, "*Investigation on the Influence of Fuel Channel Modelling on Hydrogen Generation during a CANDU 6 Severe Accident*", Proceeding of SIEN 2013, Bucharest, Romania. November 10-14, 2013.

16.0 A. Bonelli, L. Siefken, O. Mazzanti, C. Allison, "*Summary of Severe Accident Assessment for Atucha 2 Nuclear Power Plant using RELAPSCDAPSIM Mod 3.6*", Proceedings of the 10th International Topical Meeting of Nuclear Thermal Hydraulics, Okinawa, Japan, December 14-18, 2014.