***M*ulti-step *A*pproach to *C*ode-coupling for *P*rogression *I*nduced *S*evere *A*ccidents in *CANDU* NPPs (MACPISA-CANDU)**

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ABSTRACT

This paper reviews the progression of severe accidents, describes computer codes currently employed for analysis of severe accidents and outlines a new methodology to modelling the progression of severe accidents in CANDU nuclear power plants (NPPs) called the *M*ulti-step *A*pproach to *C*ode-coupling for *P*rogression *I*nduced *S*evere *A*ccidents in *CANDU* NPPs (MACPISA-CANDU). The MACPISA-CANDU methodology was used to couple the U.S. NRC codes SCDAP/RELAP5 (RELAP/SCDAPSIM Mod 3.4) and MELCOR (1.8.5) in order to model a small break loss of coolant accident with loss of emergency coolant injection (SBLOCA-LOECI) under natural circulation in a CANDU 6 NPP. Using this model it was shown that the sheath temperature did not exceed the zirconium melting temperature of 2098 K and hence the progression of the severe accident was terminated as expected.

NOMENCLATURE

Symbol	Description	Units
C	Enhanced heat transfer multiplier due to sagging	
k_{CO2}	Thermal conductivity of carbon dioxide	W/m·K
k'_{CO2}	Enhanced thermal conductivity of carbon dioxide	W/m·K
Tp	Temperature of pressure tube	K
ψ_o	contact angle	Degrees

1. INTRODUCTION

This paper outlines a modern method of modelling the progression of severe accidents in CANDU 6 nuclear power plants (NPPs). The method is called the *M*ulti-step *A*pproach to *C*ode-coupling for *P*rogression *I*nduced *S*evere *A*ccidents in *CANDU* NPPs (MACPISA-CANDU) and is discussed in further detail in section 7. The method was used to couple the U.S. NRC codes SCDAP/RELAP5 (RELAP/SCDAPSIM Mod 3.4) and MELCOR (1.8.5) in order to model a small break loss of coolant accident with loss of emergency coolant injection (SBLOCA-LOECI) under natural circulation in a CANDU 6 NPP.

Modelling of severe accidents is an increasingly important part of reactor safety analysis that conforms to modern international standards.[1] Severe accidents have a very low frequency of occurrence, but may have significant consequences resulting from degradation in cooling of nuclear fuel. The infrequent nature of some severe accident sequences has led the nuclear industry to use overly conservative guidelines in some scenarios. Nevertheless, in the recent past the Canadian nuclear industry has moved towards risk-informed decision making and towards best estimate code development.[2]

Computer codes are essential tools for understanding how the reactor and its containment might respond under severe accident conditions. Codes are used as tools to support engineering judgment, based on which specific measures are designed to mitigate the effects of severe accidents. They are also used to aid severe accident management guidelines (SAMGs) and probabilistic safety assessment (PSA).[3, 4]

2. PROGRESSION OF SEVERE ACCIDENTS IN CANDU NUCLEAR REACTORS

There is significant commonality in how severe accidents progress in CANDU NPPs. A convenient way to represent severe accident progression is to define a small number of core damage states (CDSs). A CDS is a quasi-steady state during which the decay heat is absorbed into its surrounding environment. Although the timing of progression from one state to the next can be affected by the initiating event, the detailed design and operator actions CDSs are independent of the initiating event and generally independent of station design.[4] There are five such typical CDSs for CANDU 6 NPPs that are listed in Table 1 below.

Table 1: 5 core damage states for a CANDU NPP and description of events [4]

Core Damage State	Description of Events
CDS1	• Fuel channels have lost water inventory, dried out and heated up • Fuel sheath is oxidized and pressure tubes have ballooned or sagged into contact with calandria tubes • Moderator removes most of decay heat • Terminal state of a LOCA plus loss of ECCS • Sustainable as long as the moderator level can be maintained
CDS2	• Moderator level has dropped exposing several upper channels (due to moderator rupture disk bursting due to boiling or in-core LOCA) • Exposed channels have heated up, sagged, oxidized and broken apart collapsing onto lower submerged channels or dropping to bottom of calandria vessel • Most of decay heat is removed from submerged channels as well as some of decay heat of the collapsed fuel channels that are now submerged • Adding water to calandria vessel can prolong this state
CDS3	• Moderator inventory is gone (moderator boiled off slowly or drained quickly due to type and location of break) • All channels have heated up, sagged, oxidized and broken apart leaving a rubble pile of 'corium' (mix of fuel and core structural materials) at bottom of the calandria vessel • Steel calandria vessel and surrounding biological shielding materials (water and steel in the shield tank, or concrete) remove some of decay heat • Structure is not capable of removing all decay heat and corium will eventually melt through; however, adding water to calandria vessel can prolong this state
CDS4	• Corium has penetrated through calandria vessel, biological shield and is on concrete floor • Accumulated water will quench molten corium
CDS5	• Due to lack of water or insufficient contact area for boiling, or due to formation of an upper crust, corium attacks concrete referred to as molten core concrete interaction • Ablation of concrete produces steam, H_2, CO and CO_2 • Degree to which the molten core concrete interaction can be terminated depends on decay heat (diminishes with time), surface area of the melt (affects rate of cooling by a water layer, limited by critical heat flux) and availability of water • Rate of ablation is slow (about 2 cm/hour with decay power at 1%) and the basemat is thick (>1m), and decay power diminishes with time, basemat penetration is unlikely

3. FULLY-INTEGRATED SEVERE ACCIDENT CODES

There are only 2 fully-integrated severe accident codes currently used for severe accident analysis of CANDU NPPs. These codes are MAPP4-CANDU and ISAAC and are discussed in this section. A modern multi-step approach is discussed in Section 4.

3.1 MAPP4-CANDU CANDU 6 Model

The MAAP4-CANDU (Modular Accident Analysis Program for CANDU Nuclear Generating Station) code was developed from the MAAP4 code used for PWRs (pressurized water reactors) and BWRs (boiling water reactors). To model a CANDU 6 nuclear reactor, simplified fuel channels were used in the core. The 22 rows by 22 columns of a CANDU 6 nuclear reactor were divided into 6 vertical nodes and 6 horizontal nodes. Dividing the reactor core down the vertical center, the 3 horizontal nodes on left are one of the primary heat transfer system (PTHS) loops and the 3 horizontal nodes on the right are the other PTHS loop. Therefore, the 380 fuel channels have been reduced to

36 representative channels and 18 representative channel groups for each PHTS loop. The 12 fuel bundles have been divided into 5 axial nodes. Thus, each PTHS loop is modelled as 90 nodes in 3 spatial dimensions. The 37-element fuel bundle was converted into elements arranged in 4 concentric rings (i.e. central, inner, intermediate and outer rings). The PHTS was modelled to contain 2 steam generators, 2 pumps, 2 reactor inlet headers, and 2 reactor outlet headers. The feeders connect the inlet and outlet end of fuel channels to reactor inlet and reactor outlet headers respectively. The flow through one PTHS loop follows the shape of a figure-of-8 pattern with some channels carrying the flow inward and others outward from the reactor face.[4]

3.2 ISAAC CANDU 6 Model

The ISAAC (Integrated Severe Accident Analysis code for CANDU plants) code was developed from the MAAP4 code as well and in general it employs most of the MAAP4 models for severe accident phenomena. The use of ISAAC allows the simulation of accident scenarios that could lead to a damaged core and eventually to containment failure at the Wolsong NPPs of the Republic of Korea. For the CANDU 6 model developed in ISAAC, a simplified fuel channel model was used in the core. The 380 fuel channels were reduced to 28 (up to 74) representative fuel channels or 14 (up to 37) representative channel groups for each loop. As for the 12 fuel bundles, they are divided into 12 axial nodes. The fuel channel, which consisting of the calandria tube, pressure tube and 37-element fuel bundles was converted into a single fuel pin surrounded by the pressure tube and calandria tube. The PHTS has been model with two independent figure-of-8 loops. Both loops and all 4 steam generators/pumps are modelled individually.[4]

4. MULTI-STEP APPROACH

Recently a multi-step approach has been employed to analyze the core damage of pressurized heavy water reactors (PHWR). In the multi-step approach, the computed data from one step is used as a boundary condition for the next step; where each step addresses specific phenomena. This procedure has been shown to help reduce computational time and numerical problems, while increasing the ability to select specific requirements for each step.[4] This fits well with the sequential nature of CDSs, in that each specific code can sequentially be employed. Specifically, the codes SCDAP/RELAP5, ASTEC and ANSYS have been used in this manner to analysis severe accidents.[4]

The SCDAP/RELAP5 and MELCOR codes have been chosen to model a small break loss of coolant accident with loss of emergency coolant injection (SBLOCA-LOECI) under natural circulation in a CANDU 6 nuclear power plant. When selecting codes to analyze an accident sequence it is import the all relevant phenomena for the given reactor are modelled. The capabilities of SCDAP/RELAP5 and MELCOR to model the relevant phenomena for a SBLOCA-LOECI event are listed in Table 2 below.

Table 2: SCDAP/RELAP5 and MELOCR capabilities of modelling SBLOCA-LOECI phenomena[4, 5, 6]

Phenomena for SBLOCA-LOECI in CANDU 6	SCDAP/RELAP5 Capability	MELCOR Capability
• Single phase and two phase thermal-hydraulics in the PHTS and containment thermalhydraulics	Yes	Yes
• Reactor header flow stratification and vapour pull through and reactor	Yes	Yes
• Channel flow stratification	Yes	Yes
• Radiation heat transfer among the fuel pins, pins to the pressure tube and also from pressure tube to the calandria tube	Yes	No
• Pressure tube deformation by sagging or by symmetric/asymmetric ballooning	Yes	No
• Calandria Tube outer surface boiling heat transfer	Yes	Yes
• Fuel element heat up and metallurgical deformations and release of fission gas from fuel matrix	Yes	Yes
• Ballooning of the fuel pin and its failure	Yes	No
• Bundle behaviour during asymmetric fuel pin heating	Yes	No
• Steam-zircaloy-UO2 reaction and formation of the eutectic of U-Zr alloy and hydrogen generation	Yes	No
• Transportation of the radioactive material in the PHTS	No	No
• Ex-channel molten fuel-coolant (moderator) interaction	No	No
• Debris bed–molten pool behaviour	No	No

5. SCDAP/RELAP5 CANDU 6 MODEL

Similar to the approach use by Mladin et al.[7, 8], a hydrodynamic representation of single, horizontal fuel channel modelled in SCDAP/RELAP5 can be seen in Figure 1 below.

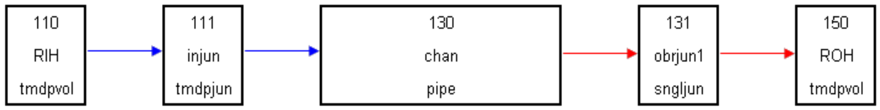

Figure 1: SCDAP/REALAP5 CANDU Fuel Channel

The single fuel channel model uses 2 time-dependent volumes, 1 time-dependent junction, 1 pipe component and 1 single junction. The 2 time-dependent volumes represent the reactor inlet header and the reactor outlet headers and simulate pressure and temperature for steady state runs and pressure and static quality for transient runs. The inlet time dependent junction facilitates flow from the reactor inlet header and controls the mass flow rate boundary condition for steady state runs and controls the mixture of water and steam for transient runs. The pipe component behaves as the fuel channel and the single junction facilitates the flow from the channel to the reactor outlet header. Each internal junction of the fuel channels (5 in total) have local pressure loss coefficients summing up the contributions of the end plates and of the mid-plane spacers were assumed to be 0.82 for the channel.

The fuel was modelled continuously over length of the fuel channel where 2 fuel bundles are represented by 1 axial node (i.e. 6 axial nodes in total). The individual fuel rods have the same geometry, power history, axial power profile, radial power profile, void volume and internal pressure as individual elements in a 37-element fuel bundle. The pressure tube (PT), carbon dioxide filled gap and calandria tube (CT) are modelled using the SCDAP shroud component. The inner face of the shroud component (i.e. inner face of the PT) is thermally connected with the hydrodynamic channel and exchanges energy by radiation with the outside of the fuel bundle. The outer face of the shroud component (i.e. outer face of the CT) is thermally connected with the moderator which is modelled as a large stagnant heavy water volume at 1.22 bar and 71ºC.

There are 3 modes of fuel channel deformation to consider for CANDU:

1. PT ballooning – Occurs at high pressure PT balloons into uniform contact with CT
2. PT sagging – Occurs at low pressure PT sags into local contact with contact angle, ψ_o
3. Fuel bundle slumping – Occurs for both ballooning and sagging

Given that sagging is the limiting case, because of higher temperatures (a lower limit of 850 ºC), this is the phenomenon that was modelled. Once the channel sags and the PT comes into contact with the CT heat transfer is enhanced. The enhanced heat transfer due to PT sagging into contact with the CT was calculated as follows:

$$k'_{co2} = C \cdot k_{co2}$$

Where;

$$C = a + b \cdot \psi_o + c \cdot \psi_o \cdot T_p + d \cdot T_p$$

$$a = 0.03, b = -0.964, c = 0.000355, d = 0.00437$$

For a contact angle of 5º, a typical value of C is 6, which used above the sagging threshold. This is model of enhance heat transfer was used to compare against the approached used by Mladin et al.[7, 8]. For comparison purposes, the temperature dependence of k'_{co2} was linearly introduced starting at 873 K and in full effect by 1083 K. At sheath temperatures of 1473 K to 1673 K the fuel bundle slumping occurs for both ballooning and sagging which leads to ‘close-packed configuration’. It should be noted that this method of modifying the heat transfer is not representative of post-contact sagged PT heat transfer behaviour. Experimental studies have shown that high temperature sagged pressure tubes contacting calandria tubes shows that they contact over a 120 degree angle due to plastic creep deformation associated with bundle loading.[9]

6. MECLOR CANDU 6 MODEL

6.1 Fuel Model

In order to aid the nodalization of a typical 37-element CANDU 6 fuel bundle in MELCOR, the fuel bundles geometry was change to a single fuel pin using a heat structure which is same as the approach using the ISAAC code. (Figure 2)

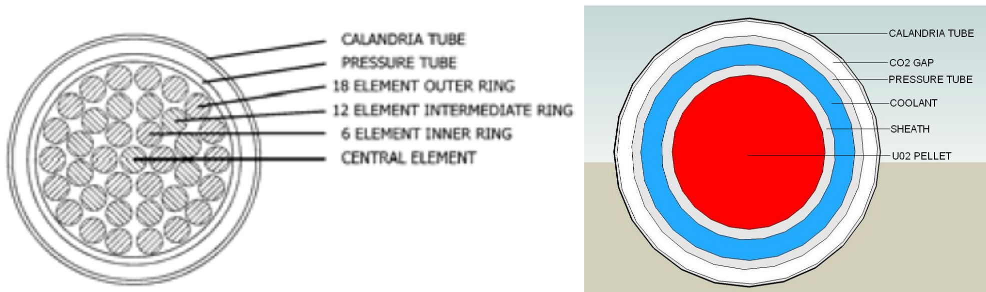

Figure 2: Transformation of 37 element fuel bundle into single fuel pin

6.2 Reactor Core and Moderator System Model

For the reactor core a simplified fuel channel model was used. The model used is similar to the approaches taken in the MAAP4-CANDU and ISAAC model discussed in sections 3.1 and 3.2 respectively. The 380 fuel channels were reduced to 36 representative fuel channels or 18 representative channel groups per each loop as seen in Figure 3. Each representative fuel channel represents 10.56 actual channels. This takes the form of 6 rows by 6 columns in the calandria vessel. Each of the PHTS loops is composed of 6 rows and 3 columns and they are separated from each other by the central vertical plane of the reactor. The flow through the core is in a typical ‘checker board’ pattern and flows through the primary heat transfer systems in a figure-of-8. The 12 fuel bundles per channel were only coarsely nodalized into 1 representative axial node within the MELCOR model.

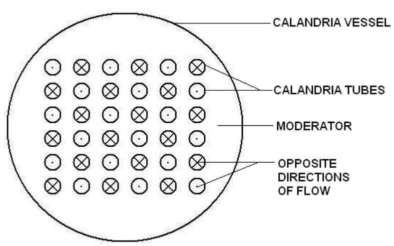

Figure 3: Moderator system model

The moderator system was modelled as large stagnant volume of water at 122 kPa and 71ºC. The calandria tubes are in contact with the moderator and are a source of heat input into this system.

6.3 Primary Heat Transfer System Model

The PHTS model nodalization (Figure 4) is similar to the approaches taken in the MAAP4-CANDU and ISAAC model discussed in sections 3.1 and 3.2 respectively. Each loop was modelled to contain 2 steam generators, 2 pumps, 2 inlet headers, and 2 outlet headers. The feeders connect the inlet and outlet end of fuel channels to reactor inlet and reactor outlet headers respectively. The flow through one PTHS loop follows the shape of figure-of-8 with some channels carrying the flow inward and others outward from the reactor face. Each figure-of-8 loop consists of 14 hydrodynamic control volumes and 14 flow paths.

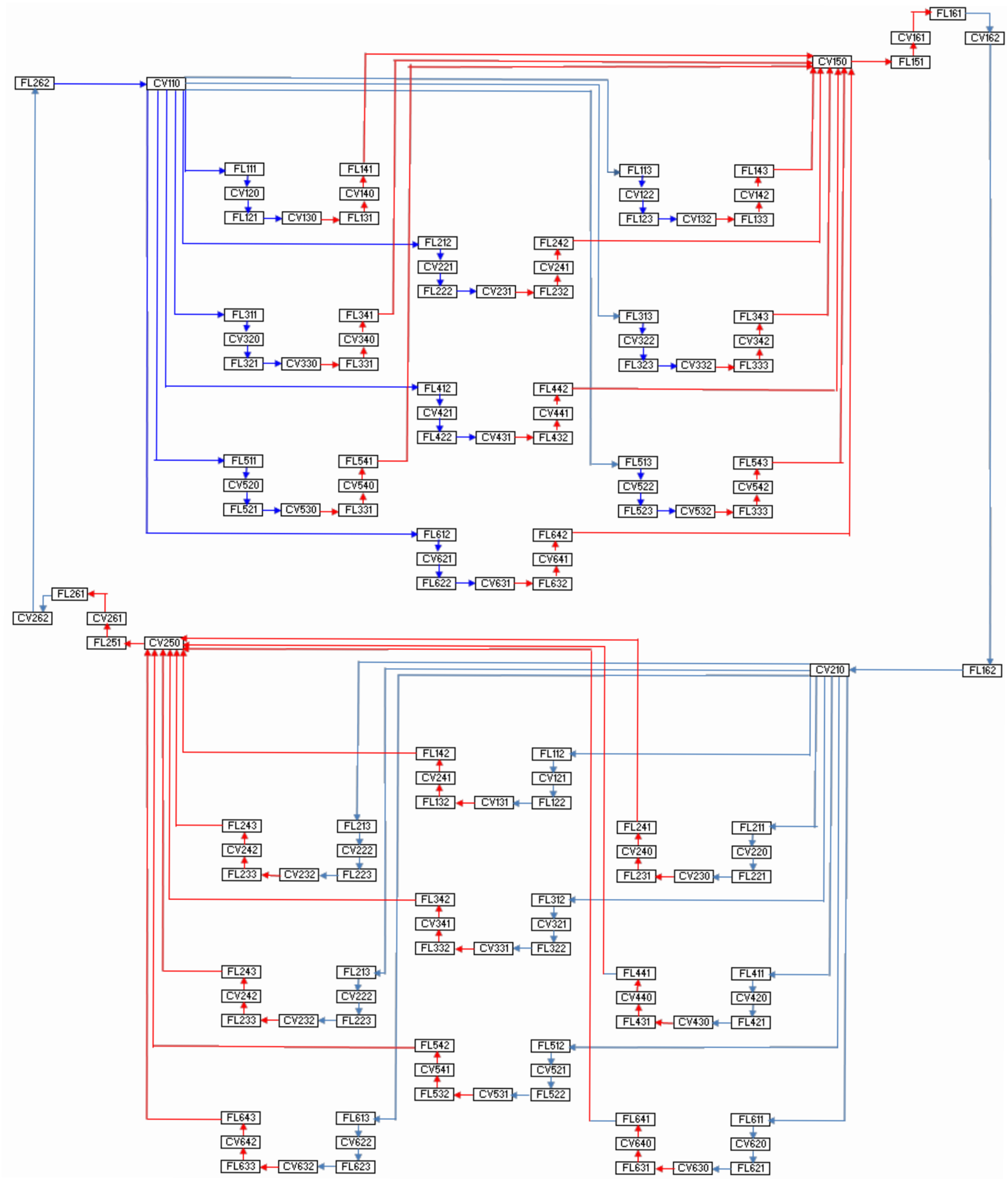

Figure 4: Figure-of-8 nodalization of single loop in CANDU 6 primary heat transport system

7. MACCPISA-CANDU METHODOLOGY

The *M*ulti-step *A*pproach to *C*ode-coupling for *P*rogression *I*nduced *S*evere *A*ccidents in *CANDU* NPPs (MACPISA-CANDU) is a generalized methodology for modelling severe accidents in a CANDU NPP. The logic for this methodology as applied to coupling SCDAP/RELAP5 and MELCOR is presented as a flow chart in Figure 5. Based on the characteristics of a given CANDU reactor design, there are a number of phenomena that can occur. The computer codes selected to analyze a particular sequence depends on the type of phenomena that are encountered in that sequence. Within the computer codes there may be more than one model available for simulating a particular phenomenon. The model selection is based on the progression of a severe accident. In the case of severe accident progression in CANDU NPP, these phenomena can be seen to take place sequentially in a 5 quasi-static core damage states (CDSs) as seen in Table 1. This allows each phenomenon to be modelled in sequence with separate codes. The results from one code are the boundary conditions for another until a terminal core state is reached. The terminal state is reached when there is no further deterioration in the state of the plant.

As a partial validation of this methodology, the U.S. NRC codes SCDAP/RELAP5 and MELCOR have been chosen to model a small break loss of coolant accident with loss of emergency coolant injection (SBLOCA-LOECI) under natural circulation in a CANDU 6 nuclear power plant. Based on the characteristics of the CANDU 6 reactor design, there are a number of phenomena that occur for this event. The capability of SCDAP/RELAP5 and MELOCR to model these phenomena that occur for SBLOCA-LOECI is listed in Table 2. Given these capabilities, the initial part of the first core damage state (CDS1), was modelled in SCDAP/RELAP5 and the second part of CDS1 was modelled in MELCOR. CDS1 is the terminal state is the terminal state for a SBLOCA-LOECI event under natural circulations.

Not much work has been done to qualify the U.S. NRC codes SCDAP/RELAP5 and MELCOR for the purpose of analyzing CANDU type reactors. Nonetheless it is possible to derive partial results using the MACPISA-CANDU methodology. SCDAP/RELAP5 was used to determine the timing and values for key system parameters such for a single inlet feeder break up until the sheath temperature reached 1473 K, the point at which bundle slumping occurs. Bundle slumping is when the fuel elements relocate at the bottom of the fuel channel because they are no longer held in firm contact by the end plate. The values for the key system parameters were used as initial conditions for the MELCOR model. The MELCOR model was executed with the fuel relocated to the bottom of the fuel channel, until the fuel sheath surface temperatures reached nominal values.

Of course, if the accident sequence was altered in such a way as to degrade the cooling of the fuel even further additional codes would be required for the second, third, fourth and fifth core damage states (i.e. CDS2, CDS3, CDS4 and CDS5) as the movement of molten-fuel within a horizontal channel cannot be modelled within SCDAP/RELAP5 and MELCOR as indicated in Table 2.

8. RESULTS AND ANALSIS

The SCDAP/RELAP5 fuel channel model was subjected to a total loss of liquid flow and an increased vapour flow from 0g/s to 20 g/s within 5 seconds of an inlet feeder break. The preliminary results from the SCDAP/RELAP5 simulation as well as the coupled SCDAP5/RELAP5-MELCOR simulation are presented below.

8.1 Fuel Temperature in High Power Channel (7.3 MW)-SCDAP/RELAP5 Simulation of Single Channel

Figure 6 below shows the time evolution of fuel temperatures for the simulation of a feeder break for a single channel using the SCDAP/RELAP5 model. The increase in temperature is due to the inability of the slow moving vapour to remove the decay heat from the fuel quickly enough to cool it, in addition to the fuel oxidation taking place which is an exothermic reaction. Given that the coolant flow rates are not restored after the initial loss of flow the radial fuel temperatures continues to rise.

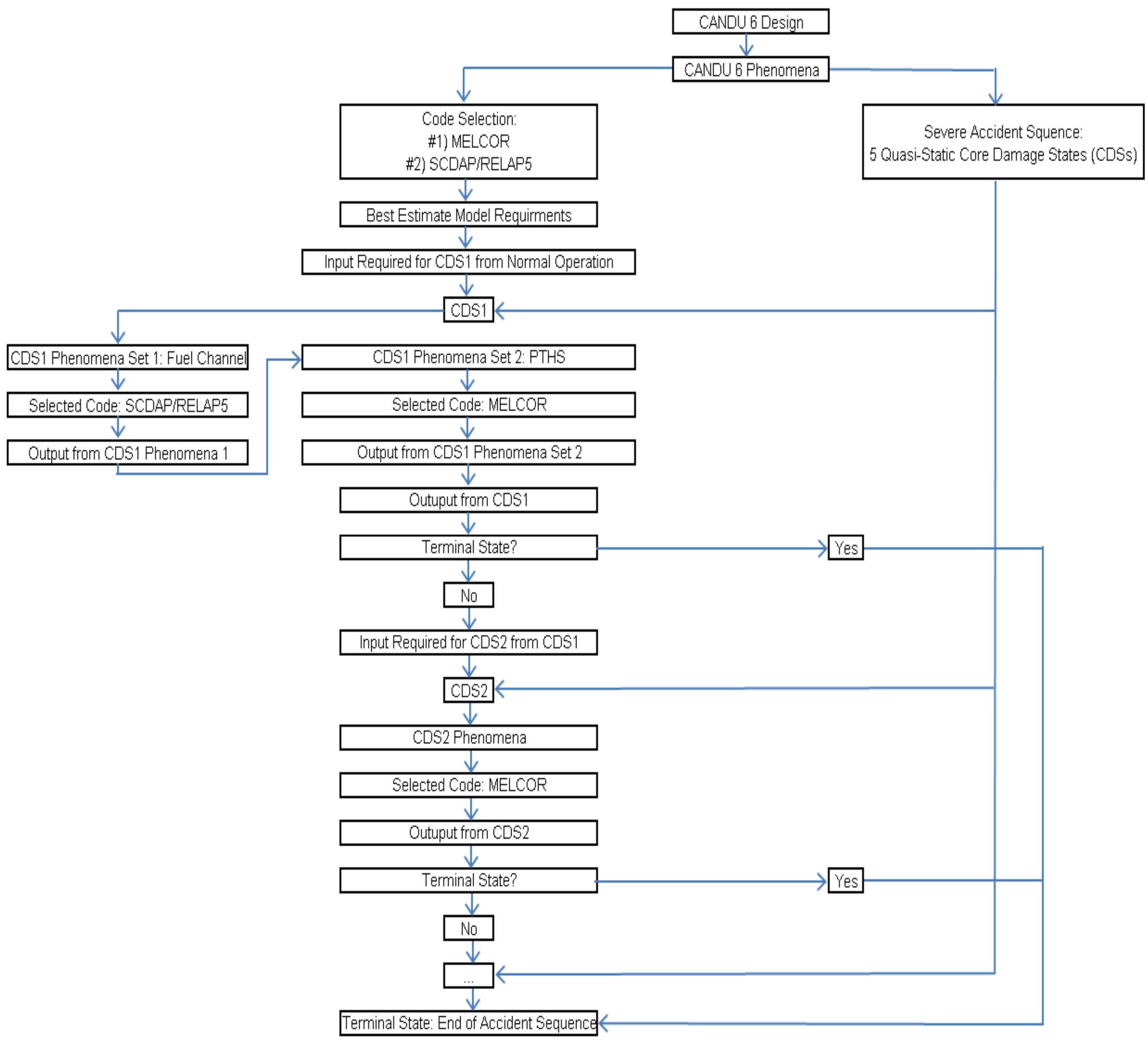

Figure 5: Multi-step Approach to Code-coupling for Progression Induced Severe Accidents in CANDU NPPs (MACPISA-CANDU)

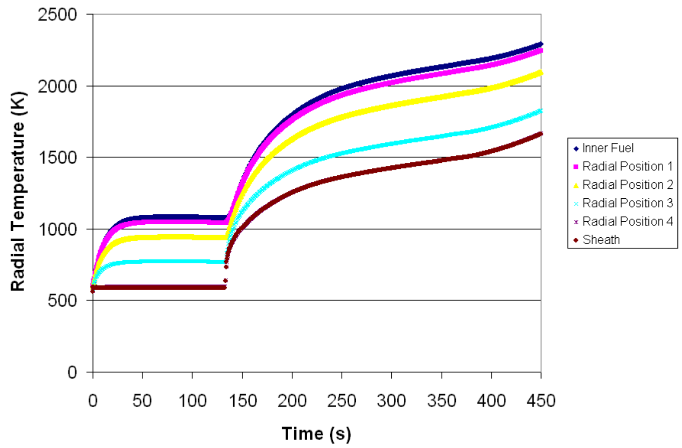

Figure 6: Fuel channel temperatures with SCDAP/RELAP5

8.2 Fuel Temperature in High Power Channel (7.3 MW)-Coupled Simulation

Figure 7 below shows the time evolution of fuel temperatures for the simulation of a feeder break for a single channel by using the coupled SCDAP/RELAP5-MELCOR models. Again the initial increase in temperature is due to the inability of the slow moving vapour to remove the decay heat from the fuel quickly enough to cool it, in addition to the fuel oxidation taking place which is an exothermic reaction. When, bundle slumping occurs at a sheath temperature of 1473 K, the moderator has already been established as a heat sink as well as the coolant flow is re-established through the core using the MELCOR model. Thus at 350 seconds, all the radial temperatures start to decrease and most rapidly at the sheath. Given that the sheath does not increase past the zirconium melting temperature (2098K) no fuel is released into the PHTS or moderator.

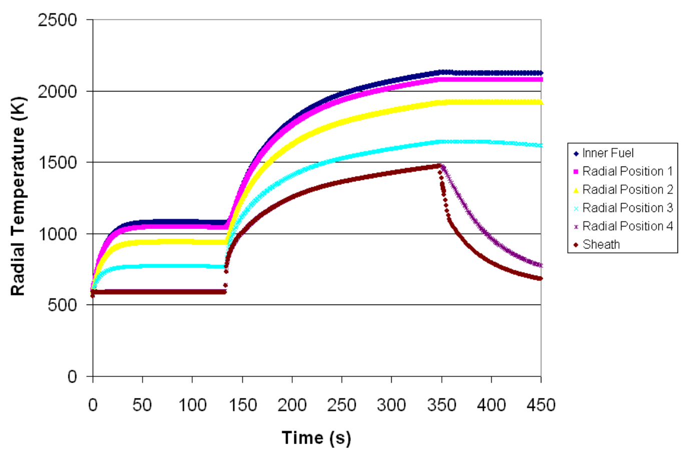

Figure 7: Fuel channel temperatures when SCDAP/RELAP5 coupled with MELCOR

10. CONCLUSION

The *M*ulti-step *A*pproach to *C*ode-coupling for *P*rogression *I*nduced *S*evere *A*ccidents in *CANDU* NPPs (MACPISA-CANDU) has been shown to yield expected results. More specifically, this methodology was used to couple the U.S. NRC codes SCDAP/RELAP5 (RELAP/SCDAPSIM Mod 3.4) and MELCOR (1.8.5) in order to model a small break loss of coolant accident with loss of emergency coolant injection (SBLOCA-LOECI) under natural circulation in a CANDU 6 nuclear power plant. The preliminary results show that the accident terminated at the first quasi-static core damage (CDS1) state as expected. Moreover, with further development of the individual models subsequent validation can take place. This method shows promise in allowing the use of codes that are not specifically developed and qualified for CANDU reactor in order to obtain results applicable for CANDU reactor analysis.

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