EVALUATION OF THE RELAP5/SCDAP ACCIDENT ANALYSIS CODE APPLICABILITY TO CANDU NUCLEAR REACTORS

Mirea MLADIN[1], Daniel DUPLEAC[2], Ilie PRISECARU[3]

Codul de calcul SCDAP/RELAP5 a fost dezvoltat în US pentru analiza termohidraulică a reactorilor de tip LWR, fiind rezultatul cuplării dintre RELAP5 pentru analiza termohidraulică şi SCDAP pentru modelarea accidentelor severe. Lucrarea trece în revistă rezultatele aplicării codului RELAP5 la reactorii de tip CANDU şi prezintă rezultatele obţinute pentru analiza fazei iniţiale a unui accident sever pentru un reactor de tip CANDU.

SCDAP/RELAP5 code has been developed in US for best-estimate simulation of light water reactors transients during nuclear accidents. It is the result of the coupling between RELAP5, modelling thermalhydraulic, and SCDAP, modelling the behavior of the reactor core during severe accidents. RELAP5 applications to CANDU will be reviewed in this paper in the frame of evaluation of the thermalhydraulic analysis capabilities of the code. Particular SCDAP early phase models as well as late phase models covering a wide range of phenomena can be applied to CANDU, as proven in the paper.

Keywords: RELAP5, SCDAP, Nuclear Reactors, Accidents.

1. Introduction

SCDAP/RELAP5 is a tool for nuclear reactors accident behavior simulation; it includes models for thermalhydraulic and the control system (RELAP5), and for severe accidents conditions with the eventual and undesired core melt and fission products release (SCDAP).

Although developed for light water reactors (LWR), the code is a flexible tool for computerized simulation as its approach allows to models as much as needed of a particular thermalhydraulic system, with use both for anticipated transients of nuclear power plants or of research reactors, and also for small scale test facilities.

It is generally known that design peculiarities of CANDU type reactors, especially the horizontal fuel channel and the moderator separated from the coolant do not allow a straightforward application of the advanced core degradation models existing in computer codes such as SCDAP/RELAP5,

[1] Ph.D. student, Institute for Nuclear Research, Piteşti, Romania
[2] Prof., Power Plant Engineering Faculty, University POLITEHNICA of Bucharest, Romania
[3] Prof., Power Plant Engineering Faculty, University POLITEHNICA of Bucharest, Romania

MELCOR, ICARE/CATHARE or ATHLET. But the analysis of design basis accidents and the modelling of experiments in specially designed facilities can be successfully performed as is outlined in the paper. Moreover, the early phase of the accident, including heatup due to voiding and oxidation, as well as, to a certain extent, other particular phenomena associated with the loss of geometrical integrity in course of a LOCA type accident coincident with ECCS, can be successfully modeled.

The paper aims at exemplifying all this range of applications, for promoting alternative calculation tools to the dedicated accident analysis code package (CATHENA + CHAN + MAAP), at least concerning the accident phenomenology inside the pressure vessel, where the CANDU peculiarities make it necessary the approach by means of codes containing specific models.

2. RELAP5 application to RD-14M

In the frame of IAEA Coordinated Research Project: „Intercomparison and validation of computer codes for thermalhydraulic safety analysis of heavy water reactors”, a benchmark problem consisting in test B9401 on RD14-M has been offered to a number of six participating countries. The experiment simulated the response to LOCA of HWRs [1].

2.1 Description of RD-14M facility

RD-14M, shown schematically in Fig.1, is a pressurized water loop having similar characteristics to the primary heat transport system for CANDU. It is designed in such a way that typical reactor conditions, as fluid mass flux, transit time, pressure and enthalpy can be obtained in the primary system both in forced cooling and in natural circulation. The project realizes the „figure 8” geometry of a CANDU reactor, with five horizontal channels per pass and a 1:1 representation of vertical elevations. The channels, each 6 meters long, contain seven electrically heated fuel element simulators, connected to end fitting simulators. The volumes, areas and metal masses are scaled in accordance with the channels. The fuel simulators have similar thermal characteristics to CANDU fuel with respect to power density, heat flux and heat capacity. The piping arrangement at inlet and outlet feeders represents the feeders from Darlington power plant. The geometry with 5 channels and their corresponding feeders experimentally simulates three middle channels, one top channel and one bottom channel. Preserving the vertical scale of 1:1 maintains hydrostatic pressures similar between RD-14M and a typical reactor.

2.2 Test B9401 in RD-14M

This test consists in a 30 mm diameter break in the inlet header, when the high pressure injection is available. The test was conducted in 1994. Its purpose was to investigate the response of the primary system loop to loss of coolant through the break with ECCS injection, according to the suppositions of a design basis accident.

The test procedure was as follows: the loop was operated until reaching the steady-state monophasic conditions and then the blowdown valve was opened at inlet header no. 8 to simulate the break. At 2 seconds after break initiation, the power was reduced to represent the residual power and the rotation speed of the pump was reduced to simulate the loss of class IV power supply. The ECC isolation valves were opened at 20.6 s and the pressurizer was manually isolated at 22.8 s. The test was terminated after an extended residual power level.

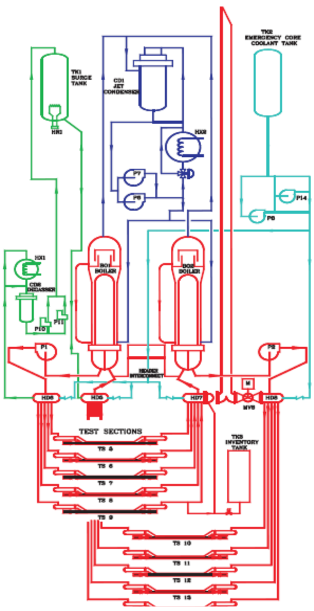

Fig.1. Sketch of the RD-14M facility [1]

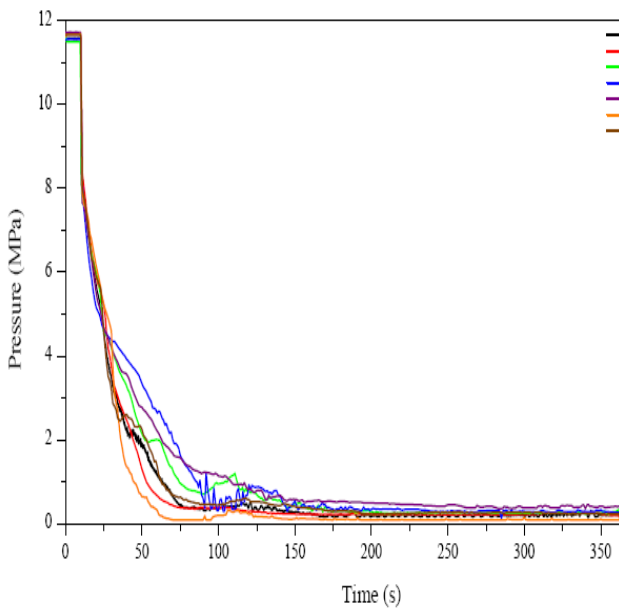

Fig.2. Depressurization in header no. 8 [1]

2.3 Computer codes and results of B9401 test simulation

The computer codes used for simulations by the participants from the six countries were:

a) FIREBIRD III MOD1-77 (Romania, Argentina);
b) CATHENA (Canada);
c) RELAP5 mod3.2 (India);
d) RELAP5 mod3.2.2g (Italia)
e) RELAP5/CANDU (Korea)

From the above list only CATHENA and FIREBIRD are CANDU dedicated tools, the rest being represented by different RELAP5 versions, among which one, the Korean, having adaptations for CANDU. Below are summarized the areas in which RELAP5/CANDU has specific developments/adaptations:

1. Critical flow model;
2. Kinetics model;
3. Heat flux model;
4. Core control model;
5. Developments in the horizontal flow regime map;
6. Horizontal channel heat transfer model.

It is worth mentioning that some of these adaptations had been already adopted by the MOD3.2 version of RELAP5, before the drafting of the final report of the CRP (2004).

Results of codes applications are illustrated, reflecting, on the one hand, the quality and the suitability of the thermalhydraulic models, and on the other, the user effect generally present in computer codes simulations.

Fig.2 presents the evolution of pressure in header no. 8, calculated by the participants against the experimental trend. In general, the depressurization rate in the header is strongly influenced by the discharge rate through the break, the later being influenced by the ECC injection. In B9401 test, the ECC injection was directed into four headers. The header no. 8 is the one with the break, and consequently it has the highest depressurization rate.

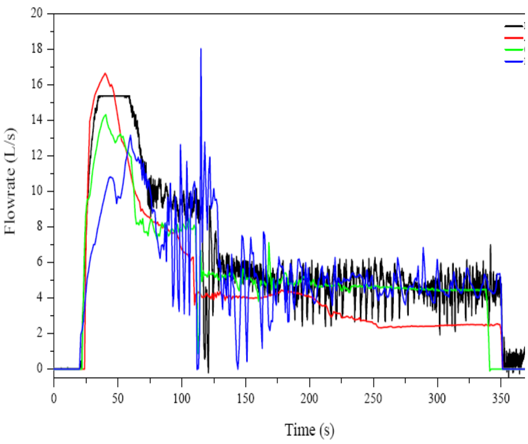

Fig.3. Void fraction at inlet of boiler 2 [1]

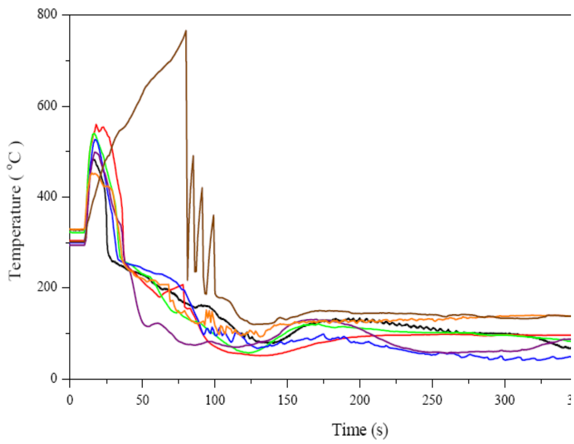

Fig.4. Maximum temperature of the fuel [1]

Fig.3 presents the void fraction at boiler 2 inlet. The boilers are heat sinks in the first part of the transient while later they can become heat sources. The void fraction at boilers inlet, together with the pumps differential pressures can provide important results concerning the entering of the flow in the primary heat transport system and the subsequent steam production. The comparison of void fraction at boilers inlet was performed at one chosen node. The discrepancies were explained on the basis of choked flow in the feeders, which is not always modeled very well.

Fig.4 shows the maximum of the temperature in the broken loop is presented. The differences appearing with the FIREBIRD-Romania results seem to be a user effect, as the same version of the code (III Mod1-77) was used by Argentina, too.

3. RELAP5 plant applications

In Romania, two RELAP5 plant CANDU models have been developed. One was developed in University "Politehnica" of Bucharest [2, 10], and the other in the Institute for Nuclear Research, Pitesti [3].

Fig.5 presents the nodalization in RELAP5 from [2]; 390 volumes and 413 junctions were used. With this model, a series of transients was performed:

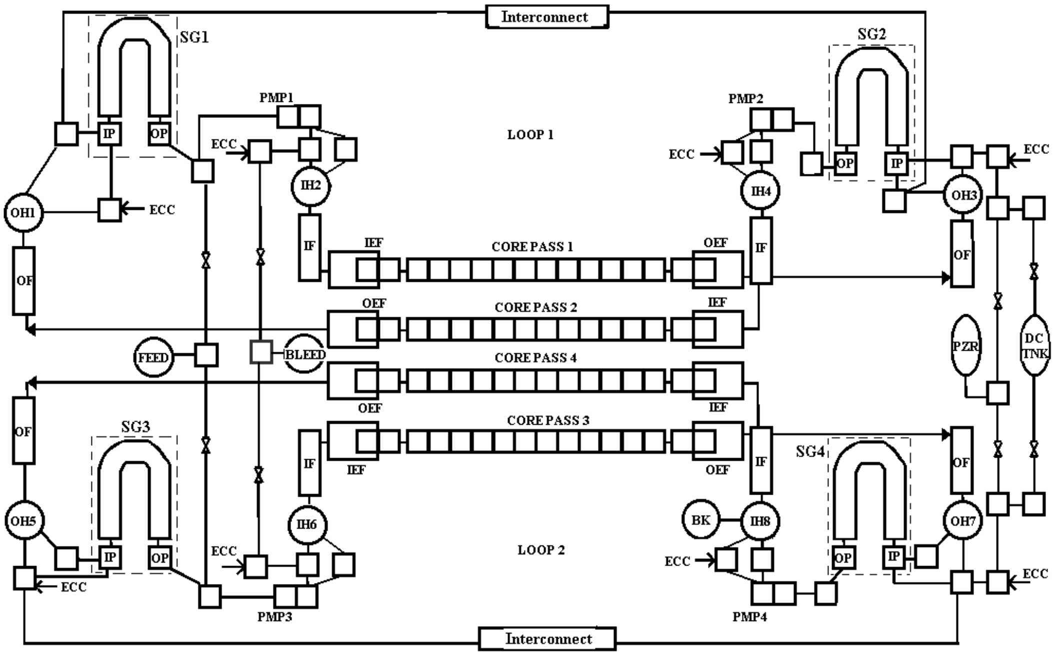

Fig.5 Nodalization created in RELAP5 model for CANDU 6 [2].

- breaks in the pipes of the primary heat transport system: RIH 20%, 25%, 30%, 35%, 40%, 45% and 100% , ROH 100%;
- steam line break in the secondary system;
- natural circulation in primary heat transport system;
- ECC injection exercise.

The results are illustrated in Fig.6, where the evolution of maximum temperature is presented for the headers ruptures mentioned above, and in Tab.1, displaying a comparison RELAP5 – CATHENA concerning the maximum temperature and the time from LOCA initiation at which this maximum is obtained.

A similar thermalhydraulic model for PHWR reactors was realized in India. [4].

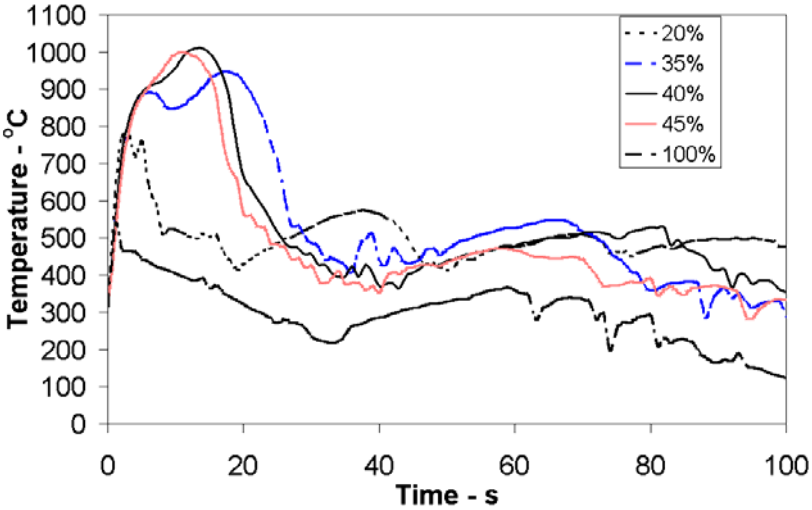

Fig.6 Maximum temperature of the fuel with RELAP5 in CANDU plant calculations [2].

Table 1

Maximum temperature during LOCA and the moment of reaching it; RELAP5 vs. CATHENA [2]

	Break		
	30%	35%	40%
RELAP5	862 °C 23 s	948 °C 18 s	1011 °C 13 s
CATHENA	1001 °C 23 s	1035 °C 18 s	975 °C 13 s

4. Fuel channel RELAP5/SCDAP application during LOCA/LOEC

These calculations, performed at the Institute for Nuclear Research [5], aimed at using SCDAP/RELAP5/MOD3.4(bi7) to the early phase of a LOCA accident coincident with LOECC, when the heatup and the loss of the initial geometry for the fuel bundles takes place. The fuel channel analysis, performed with moderator still existing outside the calandria tube, was compared to CHAN-II results which is the channel analysis Canadian code. [5]. The hydrodynamic model constructed is presented in Fig.7, while the schematic of SCDAP components that simulate the fuel and the pressure and calandria tubes is shown in Fig.8.

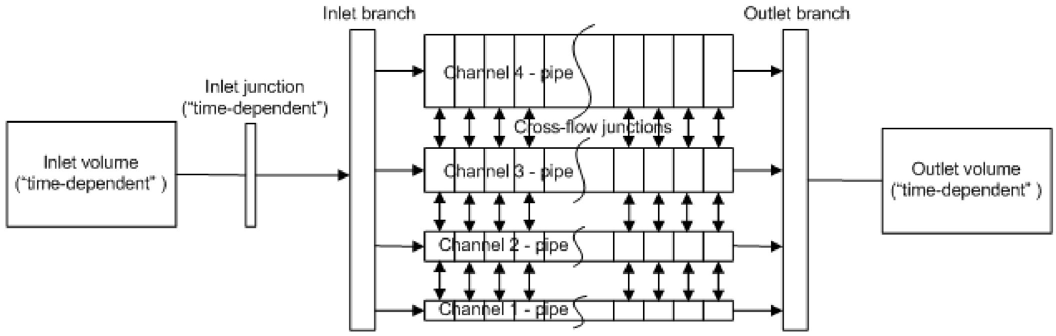

Fig.7. Hydrodynamic volumes and junctions in the CANDU channel analysis with SCDAP/RELAP5

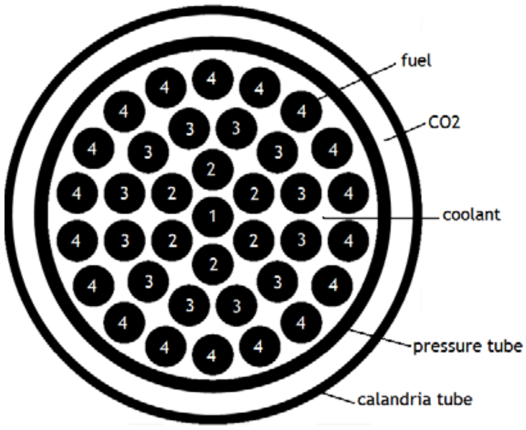

Fig.8. SCDAP components inside the fuel channel

The results are illustrated in Fig.9 and Fig.10, consisting in the temperatures of cladding and hydrogen accumulation for 11.7 g/s steam, which is the flow rate (constant) determining the maximum of the hydrogen production; both compared to CHAN-II similar case [6].

The results of RELAP5/SCDAP and CHAN-II are in acceptable agreement. Differences could possibly be due to user effect (details of CHAN-II input deck were not available), and/or to the global performance of the two codes.

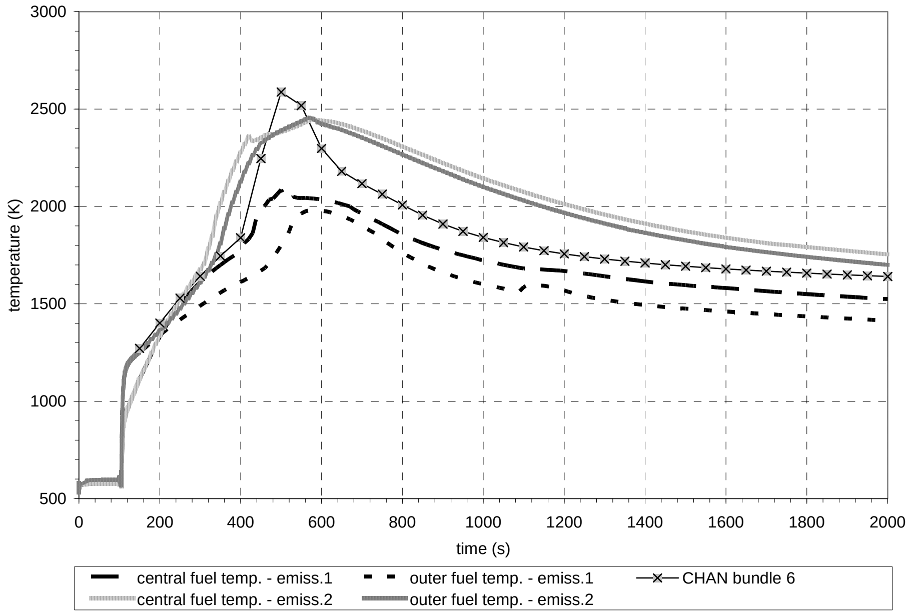

Fig.9 Maximum clad temperature for 11.7 g/s steam; SCDAP/RELAP5 vs. CHAN

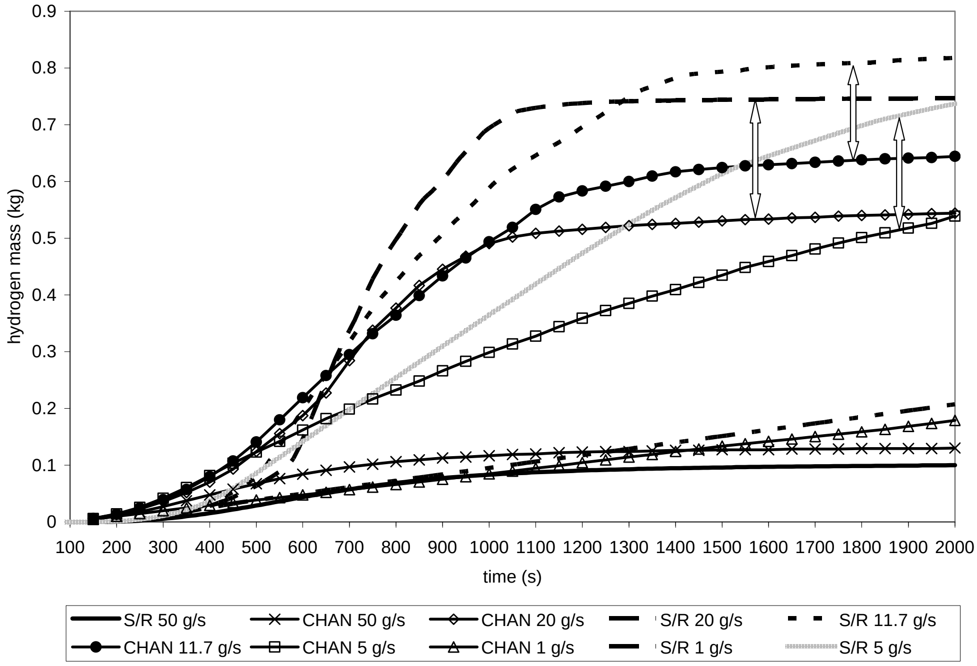

Fig.10 Hydrogen production per average channel; SCDAP/RELAP5 vs. CHAN

5. In-vessel melt retention using RELAP5/SCDAP (COUPLE)

This application of the code uses the two-dimensional debris and surrounding structures model (COUPLE) to investigate the coolability of severely degraded CANDU 6 core. It has been recently realized at University "Politehnica" [7] and studies the late phase of a severe accident that progresses up to debris bed formation on the bottom of Calandria vessel (CV). Progression of the severe accident can be briefly described as follows. During the LOCA sequence, the fuel bundles become uncovered inside the fuel channels as the result of a combination of the following phenomena: (1) decay heat from core to boil off coolant, (2) loss of coolant through the break, (3) loss of heat sink in the steam generators because the secondary side steam generator (SG) inventory is lost, and (4) loss of coolant through pressurized heat transport system (PHTS) liquid relief valves (LRV) as PHTS pressure increases after SGs dry out.

Following the initiating event, the moderator temperature and pressure in calandria vessel (CV) increase as a result of loss of moderator cooling and heat transfer from the core. The moderator in the CV reaches the saturation temperature, steam generated by moderator boiloff causing the pressure inside calandria vessel to increase. The pressure inside the CV reaches the set point of rupture disk, failing the rupture disks and thus, resulting in moderator expulsion through the relief ducts. The moderator continues to discharge into the containment resulting in a further gradual decrease of CV water level.

As the CV water level decrease, the fuel channels become uncovered; channels overheat and disintegrate being relocated to the CV bottom. The terminal debris bed is initially submerged in water and so is quenched [8].

After water in the CV is depleted, the core debris in CV begins to heat up. This is the starting point of the performed study: a quenched debris bed relocated to a bottom of a depleted CV and externally cooled by reactor vault (RV) water, as shown in Fig.11.

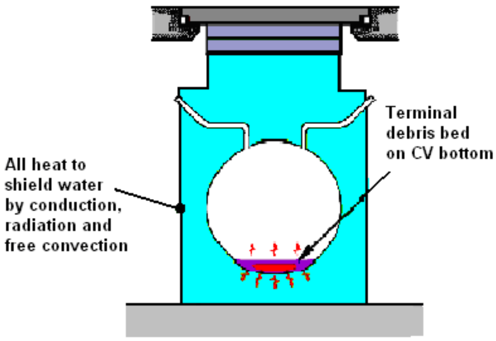

Fig.11 Debris Bed Relocated on CV Bottom, Externally Cooled by RV Water

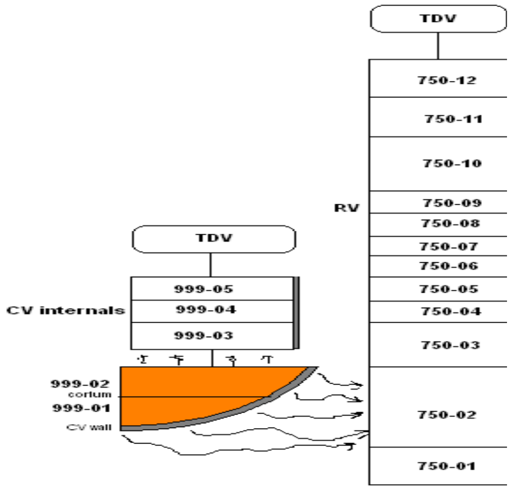

Fig.12 Simplified SCDAP/RELAP5 model of the CV thermal-hydraulics

The schematic of the RELAP5 nodalization is presented in Fig.12. A 2D finite element mesh is generated by COUPLE based on the coordinates input of a selected number of nodes. When using the 'no-slumping' option, the elements of the mesh are filled by the code user with debris material and structure material at the desired locations and timing. In this way, the calculation of the severe accident sequence prior to debris bed formation is avoided.

Two cases, defined in Tab.2 by previous analyses with MAAP-CANDU and ISAAC codes, were calculated using the SCDAP/RELAP5 model.

Table 2

Key input parameters for CANDU 6 debris bed problem [7]

Parameter	**ISAAC Case #1**	**MAAP4-CANDU Case #2**
Thermal reactor power, MW	2140	2064
Reactor vault inventory, kg	360,000	465,324
UO_2 mass in the core, kg	99,000	98,815
Zircaloy mass in the core, kg	37,820[a]	38,647
Core collapsed onto the CV bottom, s	8,280[b]	13,200
Water is depleted inside calandria vessel, s	32,400	20,800

For the case #1, the SCDAP/RELAP5 model predicts the CV failure at about 126,300 s after accident initiation, whereas ISAAC predicts CV failure at 127,080 s.

For the case #2, the SCDAP/RELAP5 model predicts the CV failure at about 140,100 s after accident initiation, while MAAP4-CANDU predicts CV failure at 130,557 s.

As for the MAAP4-CANDU and ISAAC analysis, the general conclusion of SCDAP/RELAP5 application is that CV failure may result only if the RV water inventory becomes depleted.

6. Complete RELAP5 analysis of LOCA/LOECC accident

This kind of application was realized in India for the 220 MW PHWR plant [9]. The approach is to make coupled calculations SCDAP/RELAP5 – ANSYS. Actually, the whole method relies on RELAP5 concerning thermalhydraulics, SCDAP code (present in the coupled version) not being used. In the first step, a plant model is realized: primary circuit, secondary circuit, point kinetics (as the method for describing the reactor power) and moderator (including the model of CV as a vertical pipe) with RELAP5. The purpose of this stage was finding the evolution of the moderator level in the CV. The graph of this evolution is shown in Fig.13. Out of this, one finds the time at which the level decreases below the fuel channels, and calculates the number of uncovered fuel channels (Fig.14). Then, comes the determination of the moment at which the channels fail by creep rupture with an ANSYS analysis, imposing the mechanical loads on the calandria tube and the criteria for channel disassembly.

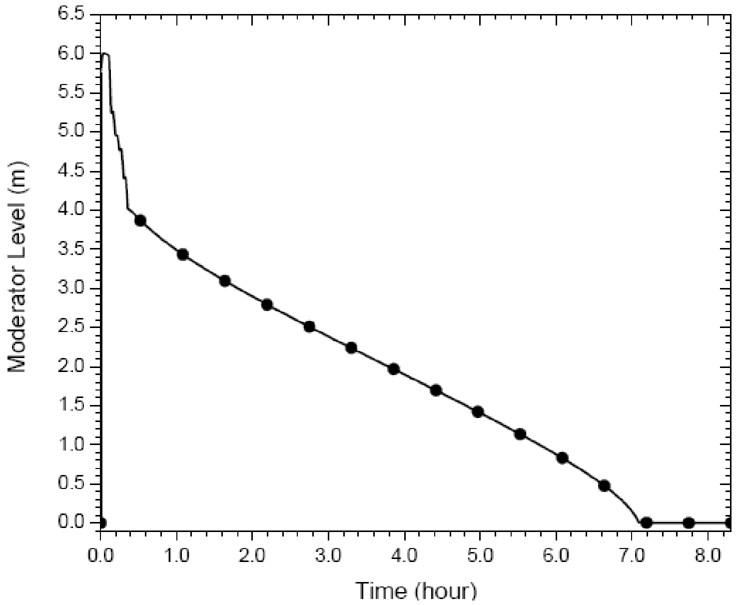

Fig.13 Moderator level inside CV [9]

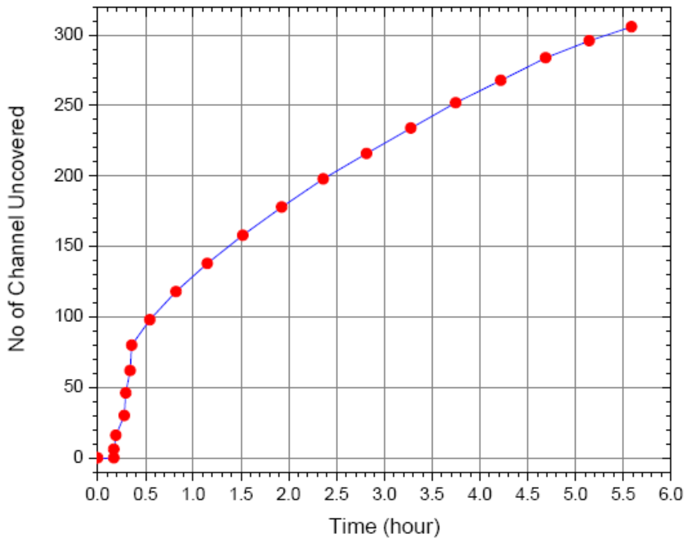

Fig.14 Number of uncovered channels [9]

When the whole core collapsed, the thermalhydraulic conditions in the moderator are taken and the corium composition is introduced in a separated model. Modelling the thermal evolution of corium and calandria vessel is done for rectangular geometry, by means of RELAP5 heat structures, heat transfer being made from the corium to the calandria vessel wall and further to the shield tank water. The evolutions of corium and calandria vessel temperature are shown in Fig.15 and Fig.16.

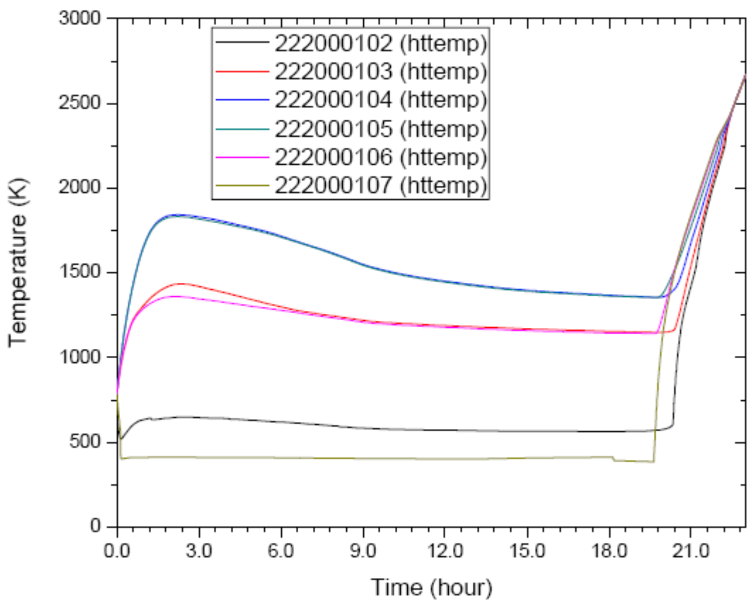

Fig.15 Corium temperature [9]

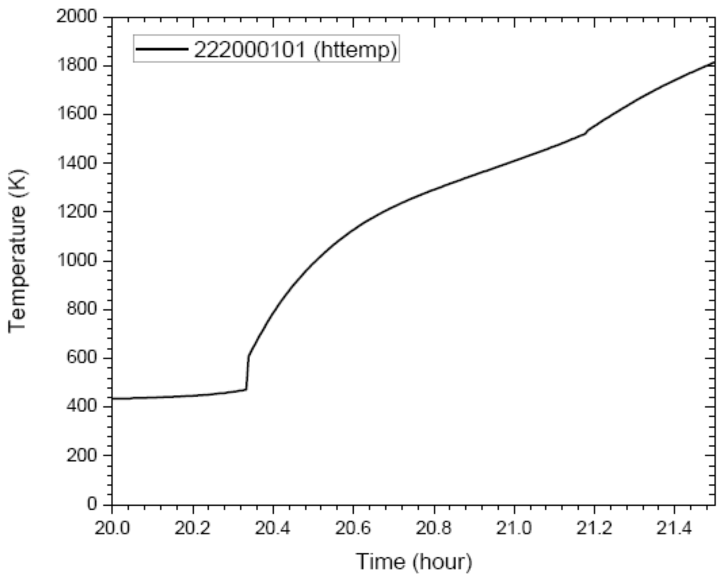

Fig.16 Calandria vessel temperature [9]

Failure of calandria vessel is calculated at 21.4 h from the start of the accident, when the temperature of the material exceeds 1450 ˚C.

7. Conclusions

The paper highlights and exemplifies the range of applications for CANDU analysis accident treatment using SCDAP/RELAP5. It encompasses many areas of nuclear accident analysis in CANDUs and proves that the code can be used, sometimes with advanced user skills application and together with other codes, on all phases of a CANDU severe accident. But this does not imply the direct calculation of a complete severe accident sequence, up to the failure of calandria vessel.

Generally, applying a non-dedicated code without enough precautions to a certain reactor design implies major drawbacks. These come from the complexity of the severe accident phenomenology and from the numerical difficulties of the major part of the codes when simulating certain physical conditions. Even for the dedicated types of reactors, for which the code is designed, this kind of difficulties are sometimes encountered.

The user effect can play an important part in two ways: on one hand the results of codes devoted to a certain reactor design can be negatively influenced by user options, and on the other hand, user experience and skills may allow obtaining consistent results in directions less explored before, as is exemplified in the paper.

As an extra argument in favor of the utilization of the code for CANDU, the existence of the heavy water library in the release packages of RELAP/SCDAPSIM versions from ISS-SUA can be mentioned.

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