DEVELOPMENT OF A CANDU-6 PLANT MODEL USING GRAPE

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ABSTRACT

RELAP/SCDAPSIM, design to predict the behaviour of reactor systems during normal and accident conditions, is being developed as part of the international SCDAP Development and Training Program (SDTP). GRAPE (Graphical RELAP/SCDAPSIM Analysis Platform for Education and engineering) is a platform for an educational simulator which can provide simple and easy operation experiences where reliable plant models can be treated with RELAP/SCDAPSIM that has been used for licensing applications. The present system automatically setup analysis and visualization environments only with a simple operation.

This paper presents: 1) the CANDU-6 plant model for GRAPE, based on the RELAP/SCDAPSIM noding diagram developed at University Politehnica of Bucharest, 2) the analysis and graphical display for a LOCA transient in a CANDU-6 primary heat transport system, 3) some suggestions for additional improvements and testing to represent the latest detailed core model of a CANDU-6 reactor.

1 Introduction

Unlike Light Water Reactors (LWRs) in which the H_2O contained in the Reactor Pressure Vessel serves as both moderator and coolant, the CANDU Pressurized Heavy Water Reactor (PHWR) incorporates two completely independent systems for the moderator and the coolant: the high pressure and high temperature heat transport system (reactor coolant system) that circulates D_2O coolant through the fuel channels to remove the heat produced by fission in the fuel, and the cool low pressure moderator system that circulates the D_2O moderator surrounding the fuel channels through heat exchangers to remove the heat generated in the moderator (1).

Figure 1 shows a simplified schematic of the CANDU heat transport system (HTS). The HTS circulates pressurized heavy water coolant through the fuel channels to remove the heat produced by fission in the nuclear fuel. The coolant transports the heat to the steam generators, where it is transferred to light water to produce steam to drive the turbines. Two parallel HTS coolant loops are provided in a CANDU-6. Each loop has one inlet and one outlet header at each end of the reactor core. Heavy water is fed to each of the fuel channels through individual feeder pipes from the inlet headers and is returned from each channel through individual feeder pipes to the outlet headers (2).

Each HTS loop is arranged in a ‘Figure 8’, with the coolant making two passes, in opposite directions, through the core during each complete circuit, and the pumps in each loop operate in series. The coolant flow in adjacent channels is in opposite directions. The pressure in the HTS is controlled by a pressurizer connected to the outlet headers at one end of the reactor (3), (4).

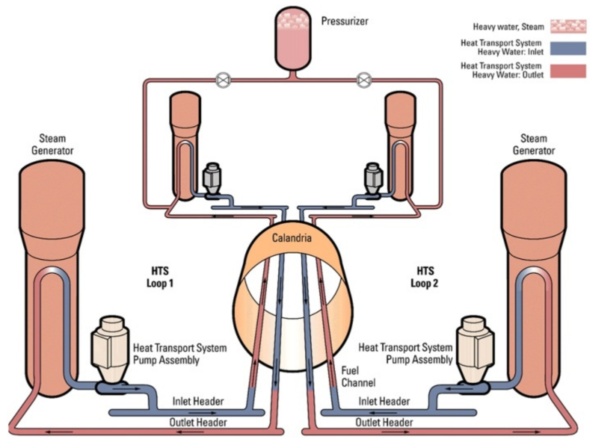

Figure 1 CANDU 6 Heat Transport System

The moderator system of CANDU-6 reactor is separated from the coolant system (Figure 2) unlike that in a light water pressurized reactor. The moderator fluid is the heavy water (D_2O) same as the coolant. The moderator is contained in a large tank called the Calandria (2).

The main moderator system is a low pressure, low temperature, closed D_2O circuit that is operated independently of the high pressure, high temperature primary heat transport system. In a CANDU-6 moderator system, cool D_2O moderator is supplied to the Calandria through nozzles that penetrate the wall of the Calandria shell just below its mid-height. The moderator flow is heated in the Calandria vessel, and extracted through two outlet ports at the bottom of the Calandria vessel. It can also be seen that the moderator which flows from the two outlets is mixed in a header and passes through either of two 100% capacity moderator pumps, subsequently being cooled via two parallel 50% capacity heat exchangers. The cooled flow is returned to the Calandria through inlet nozzles (4).

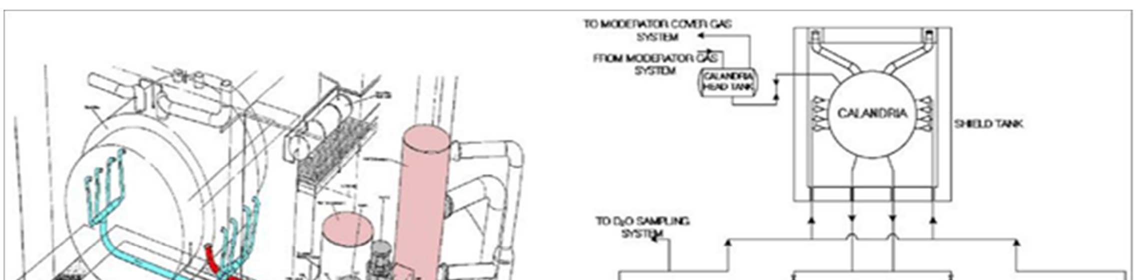

Figure 2 CANDU-6 Moderator system

2 RELAP/SCDAPSIM and GRAPE for CANDU-6 reactors

2.1 RELAP/SCDAPSIM for CANDU-6

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 (SDTP). RELAP/SCDAPSIM uses the publically available SCDAP/RELAP5 (6), (7) 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 (8).

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 .

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

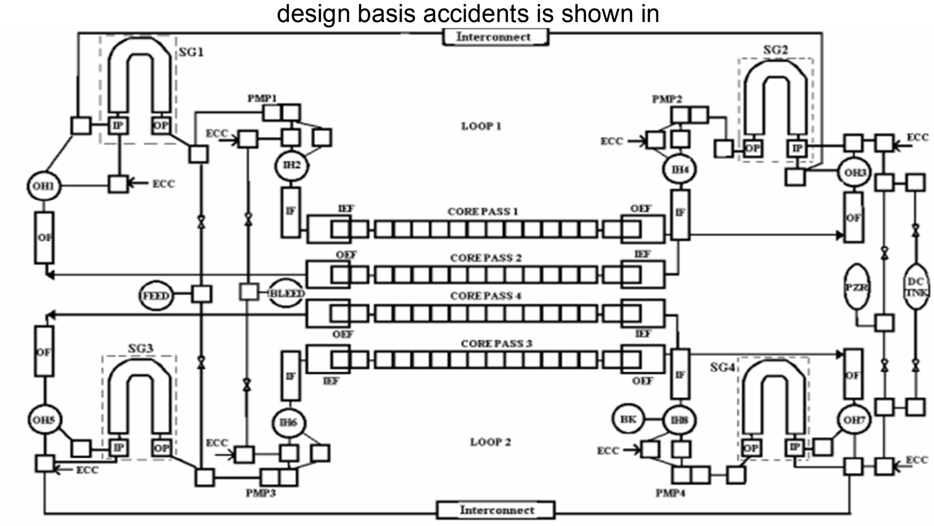

Figure 3.

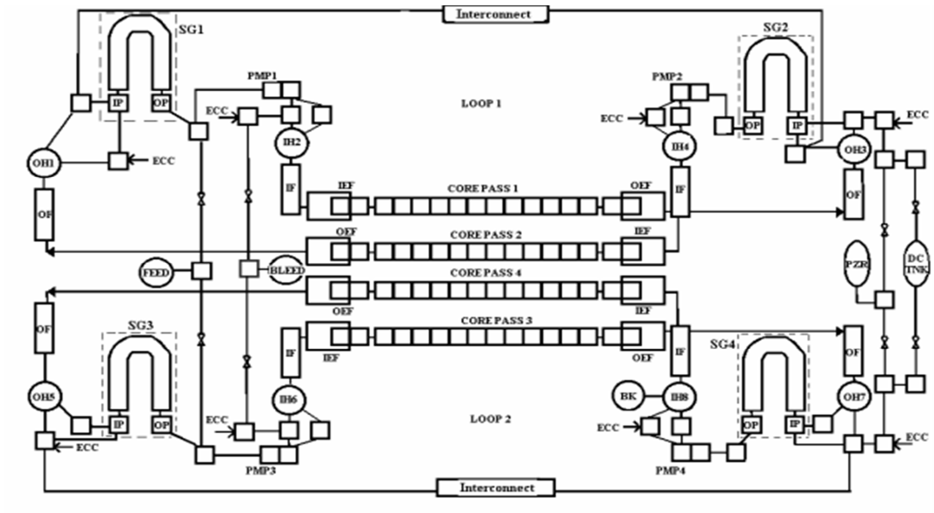

Figure 3 CANDU-6 nodalization for RELAP/SCDAPSIM

In 2005 I. Prisecaru, D. Dupleac and L. Biro (9) 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: a steady state natural circulation in the heat transport system, a 35% inlet header break, and a 100% steam header break. Results from these calculations were compared to results from similar sequences analysed with CATHENA, a CANDU specific code. Results from these calculations show that a mechanistic code such as RELAP/SCDAPSIM can be used to predict CANDU reactor behaviour.

2.2 GRAPE (Graphical RELAP/SCDAPSIM Analysis Platform for Education and Engineering)

GRAPE is a platform for an educational and engineering simulator which can provide a simple and easy operation experience where reliable plant models can be treated with RELAP/SCDAPSIM that has been used for licensing applications. With easy-to-understand interactive operations for specifying analysis conditions and rich visualization capabilities, it can be easily used for novice users to start learning of basic principles on nuclear power plant behaviors without deep knowledges on a calculation code itself. It can cover a wide range of educational or engineering needs with a nature of flexibility and extensibility (10).

GRAPE is designed and implemented by taking advantage of recent technologies of the Web such as HTML5, CSS and JavaScript in combination with the modern software architecture. Intuitive graphical user interface was established to meet user needs while maintaining flexibility, extensibility and maintainability. It is also effective for study and research projects as visualization and/or sensitivity platforms since GRAPE can import existing RELAP-based calculation results using a converting utility program to visualize the calculation results. A macro language is available to automate the workflow in sensitivity analysis and make the process more efficient (11).

The present system is equipped with two plant models (10): 3-loop PWR plant model (Surry model, Figure 4) and BWR-5 model (Laguna Verde model, Figure 4) and the projects stored in the default data directori of GRAPE contain a LOCA for BWR model and a SBO for PWR model (Tab 1: Projects stored in the default data directory of GRAPEE (Tab 1).

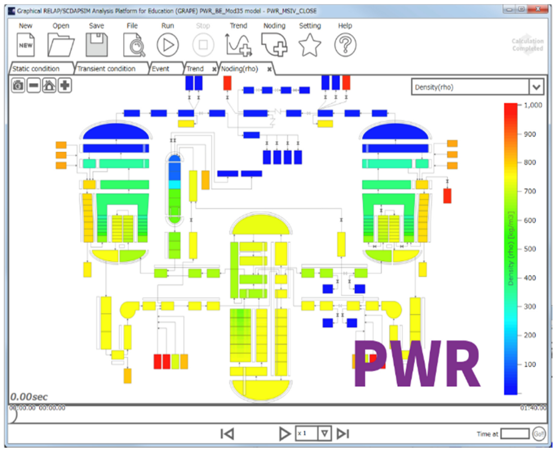

Figure 4 PWR 2loop (Surry model)

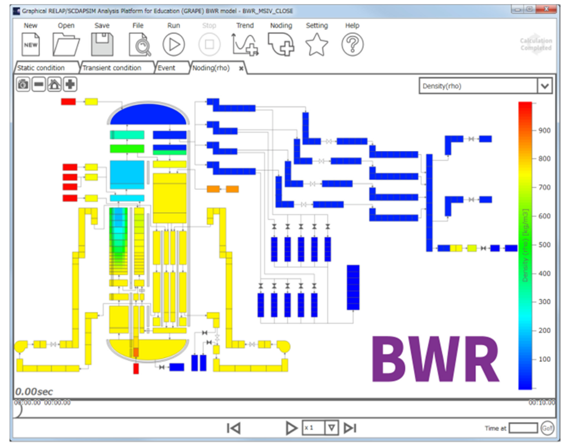

Figure 5 BWR-5 model (Laguna Verde)

Project	Scenarios
BWR_LOCA	Recirculation line break scenario in BWR plant
PWR_SBO	Station black out scenario in PWR plant

Tab 1: Projects stored in the default data directory of GRAPE

2.3 Development of CANDU-6 model for GRAPE

Since the present system was equipped with only two plant models: 3-loop PWR plant model (Surry model) and BWR-5 model (Laguna Verde model), Politehnica University of Bucharest joint forces with Innovative Systems Software (United States) and NEL (Nuclear Engineering, Ltd. in Japan) to develop a CANDU-6 model for GRAPE based on the nodalization used in RELAP/SCDAPSIM (Figure 3).

The first step in developing this model in GRAPE consisted in building the noding diagram for the primary heat transport systems and the secondary system using Adobe Illustrator, and joining both models into a single one.

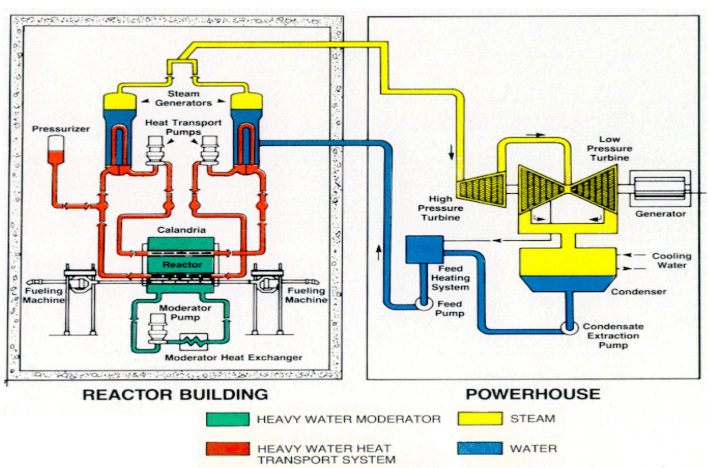

Figure 6 CANDU-6 systems

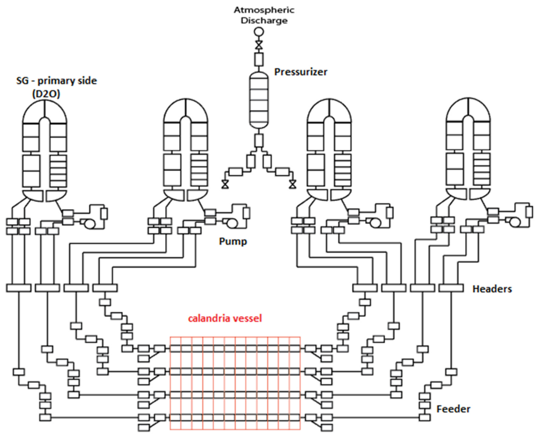

Figure 7 PHTS&Calandria Vessel

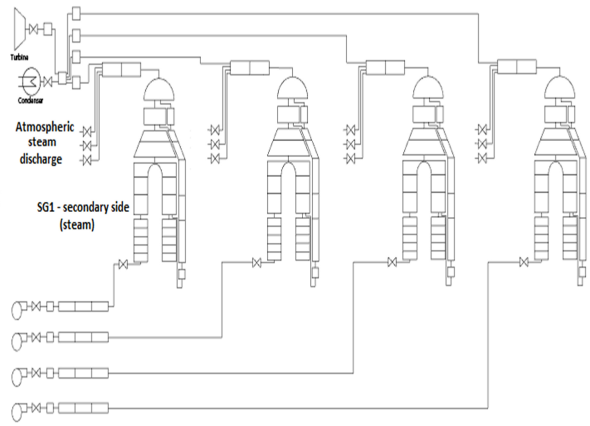

Figure 8 Secondary system

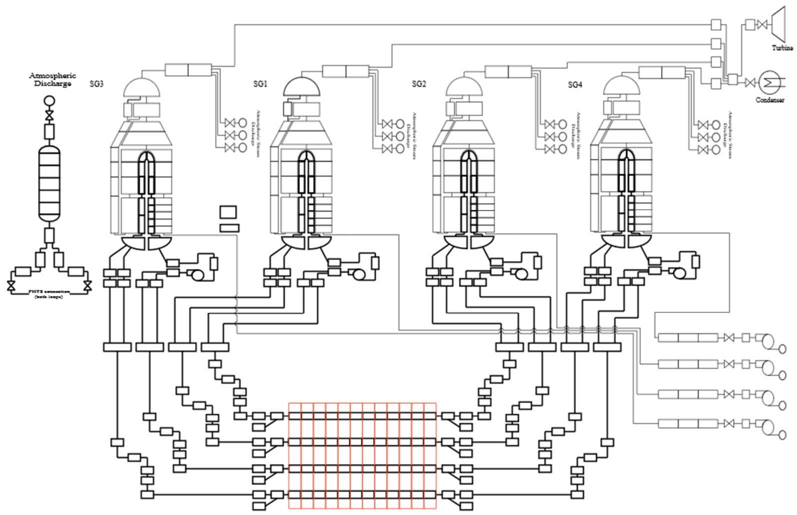

Figure 9 CANDU-6 noding diagram for GRAPE

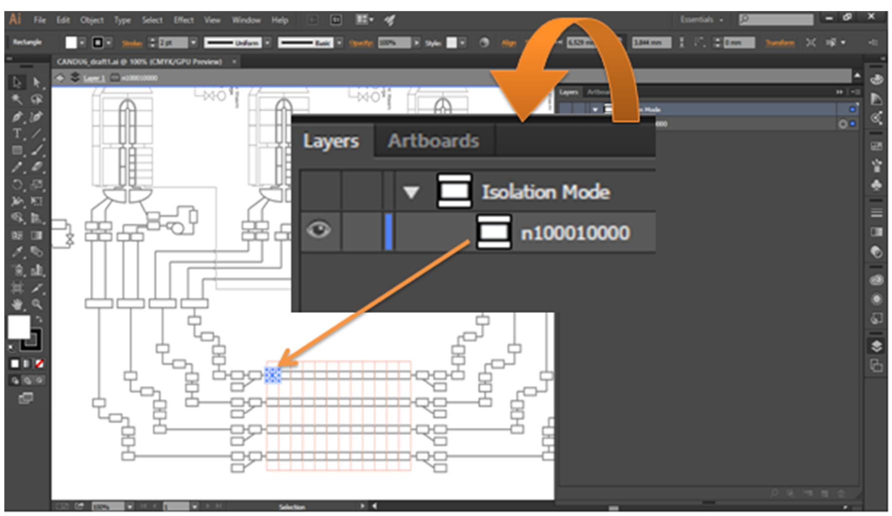

Figure 10 Assigning component number

Once the two systems have been grouped together and adjusted in order to provide the user a realistic view of the plant systems, the model configuration has been started. Based on the input deck for CANDU-6 steady state analysis, the user has assigned the specific component number from RELAP/SCDAPSIM to each closed volume drawn in Adobe Illustrator. For example, one of the fuel channels was described through component number 100 in RELAP/SCDAPSIM input model, and it was divided into 12 axial volumes specific to each fuel bundle inside the channel. According to this, the first volume of one of the channels drawn in Adobe Illustrator will receive the name "n100010000" (Figure 10Figure 10), which means that this volume behaviour will be computed from RELAP/SCDAPSIM input deck from component number 100, volume 01.

3 Steady state in CANDU-6 with GRAPE

The steady-state analysis shows the circulation pattern and temperature distribution within the calandria vessel durin full power operating conditions ehich is dictated by the interaction between momentum and buoyancy forces. Momentum forces are generated by the incoming jets and buoyancy forces are generted from heat addition within the core and reflector regions (causing density differences).

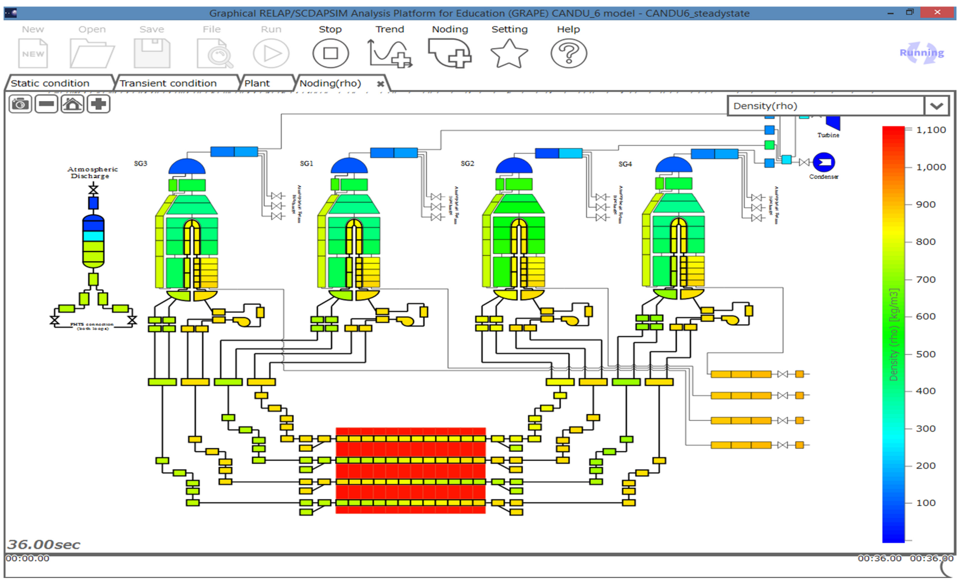

Figure 11 Steady state display of a CANDU-6 reactor systems analysis (GRAPE)

4 Large Break Loss Of Coolant Accident in CANDU-6 reactor with GRAPE

A large LOCA in a CANDU is conventionally defined as one where the break area is larger than twice the cross-sectional area of the largest feeder pipe. Because there are two feeder pipes connected to each channel, there is a lot of small-bore piping in CANDU - hence the probability of a pipe break drops by about two to three orders of magnitude for break area exceeding twice that of the largest feeder pipe. Thus any “large LOCA” can only be located in the large piping above the core, and is analyzed separately from small LOCAs. There are three representative locations: reactor inlet header (RIH), reactor outlet header (ROH), and pump suction line (PSH). Other locations are lines connected to the pressurizer, shutdown cooling lines and the header interconnect lines.

This paper shows the displays in GRAPE for the both primary heat transport system and secondary system behaviour during a LB-LOCA . In this analysis a large LOCA scenario is initiated by a 35% rupture in the RIH (Reactor Inlet Header).

The initial phase of the accident (0-5 s) is characterised by a short power transient, which is terminated by either a neutron or process trip.

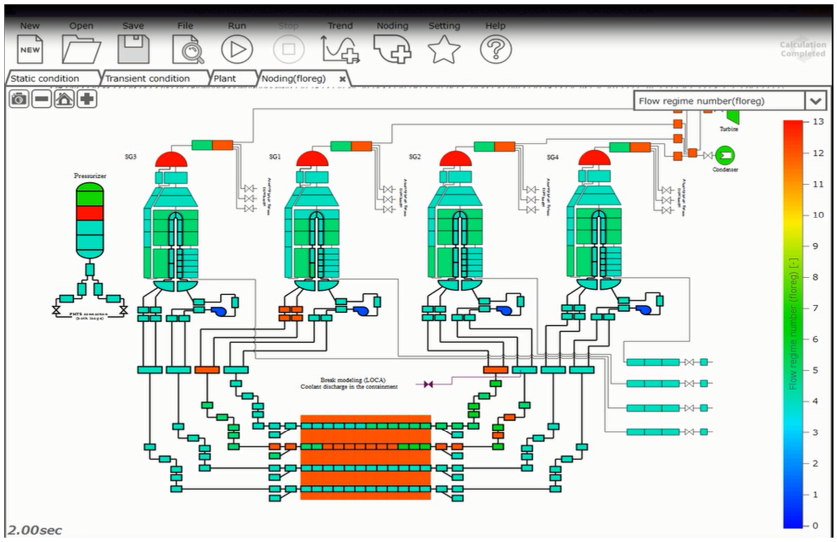

Figure 12 Flow regime number after 2s of the initiating event (LB-LOCA)

The second phase of the accident (5-30s) is characterised by the blowdown and depressurisation of the fuel channel prior to ECC injection. Despite the reactor shutdown, fuel temperatures may remain high due to the degradation in cooling, decay heat and oxidation of the fuel sheath. The fuel sheath may undergo significant deformation and may fail releasing fission products to the fuel channel and subsequently to containment. During this phase, the temperature of the pressure tube also rises and the pressure tube deforms into contact with the calandria. Once the pressure tube is in contact with the calandria tube, the moderator acts as a heat sink, cooling the pressure tube and preventing failure of the fuel channel.

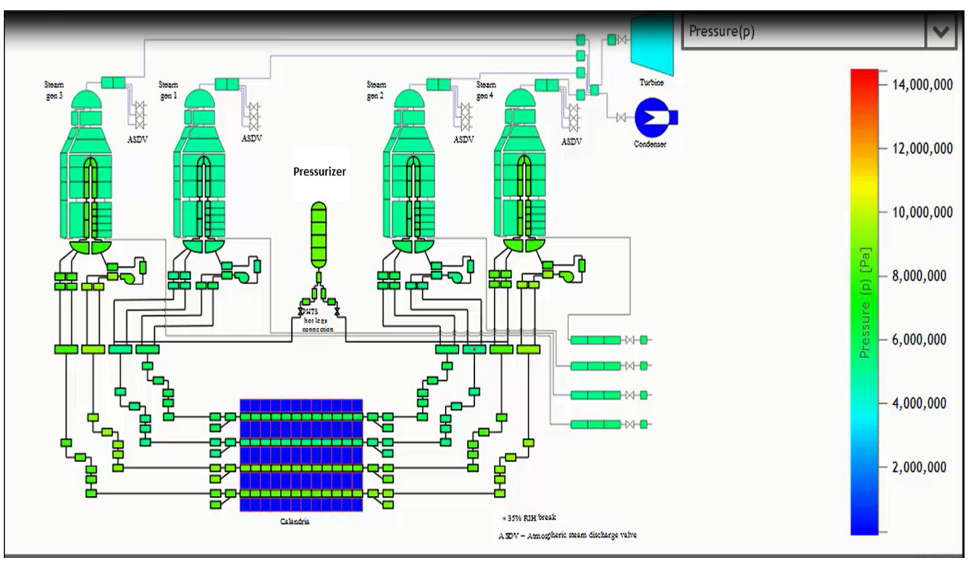

Figure 13 Reactor systems pressure (5 seconds after the initiating event)

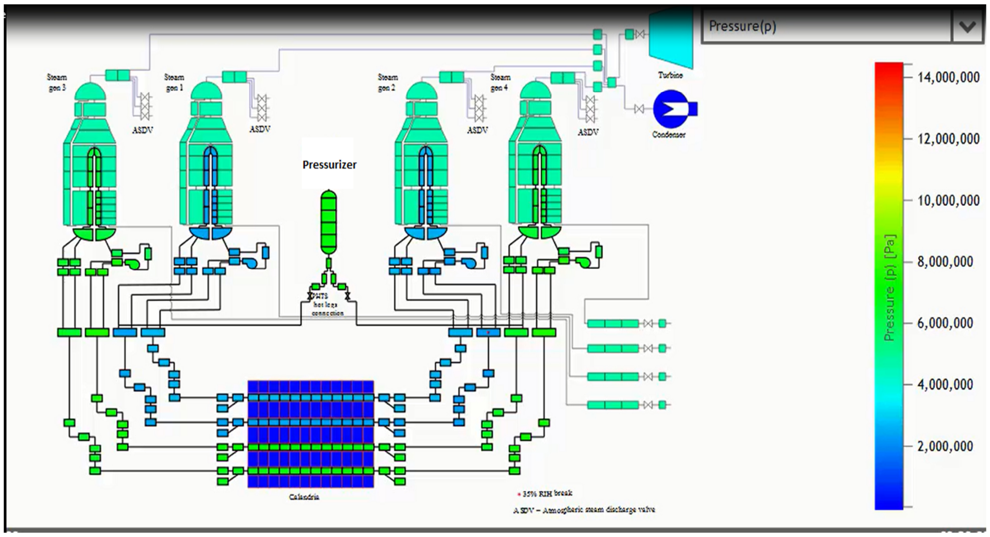

Figure 14 Reactor systems pressure (30 seconds after the initiating event)

The third phase of the accident (30-200 s) is characterised by the initiation of ECC. During this period, ECC is being injected into the primary heat transport system, but has not yet reached sufficient levels to effectively cool the fuel. Depressurisation of the heat transport system continues (Figure 15) and stored heat and decay heat from the fuel is radially removed to the moderator through the pressure tube and calandria tube. Fuel failures are likely during this stage of the accident.

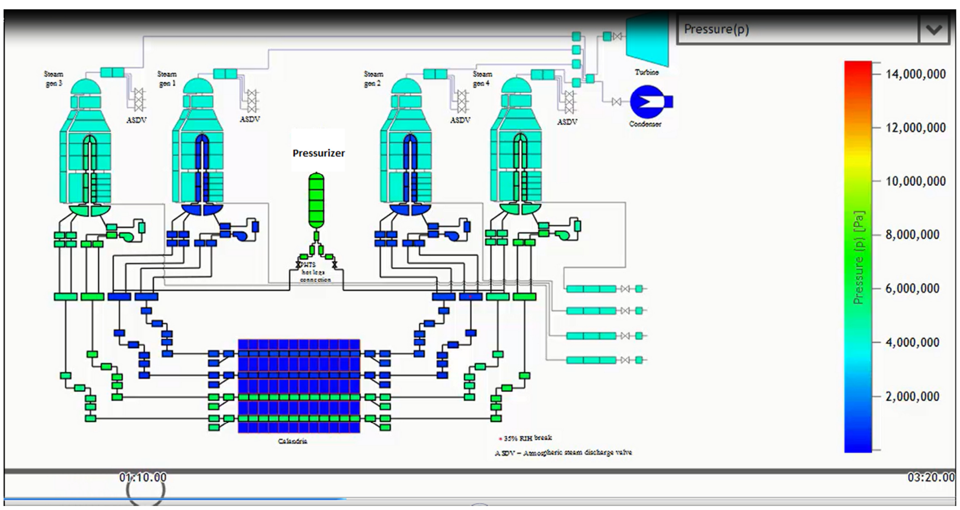

Figure 15 Reactor systems pressure (70 seconds after the initiating event)

During the fourth and final phase of the accident (>200 s) the injection of ECC has reached a level where it can effectively cool the fuel. The heat transport system pumps have tripped, refill of the channels in the core proceeds and a quasi-steady state is attained.

5 Suggestions for additional improvements and testing

5.1 3D model describing the core of a CANDU-6 reactor

Based on the latest thermal hydraulic model in RELAP/SCDAPSIM which describes the representative fuel channels and the moderator system, where thermal hydraulic channels in the core were grouped in 56 from 380, considering the radial power distribution at the core elevation and the core pass (Figure 16) and the calandria vessel which was modelled as 12 parallel pipe components with eight sub-volumes having a vertical orientation. Each pipe component simulates a section of the calandria volume representing the moderator surrounding each first sub-volume of the fuel channels of PHTS. The analogous volumes of the 12 parallel pipes are connected by cross flow junctions.

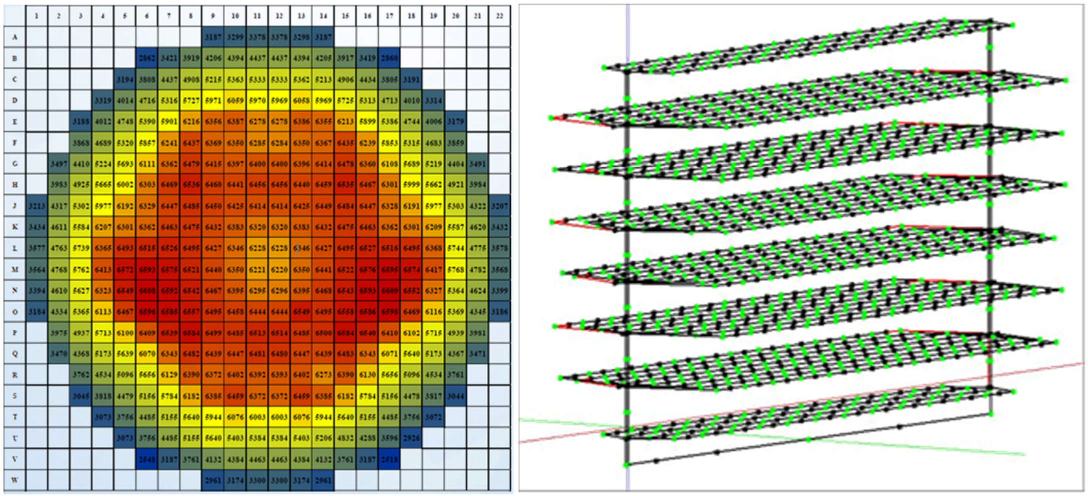

Figure 16 Radial power distribution in a CANDU-6 core and the nodalization sketch in RELAP/SCDAPSIM

The determination of flow rates, and flow quality, from inlet headers to reactor coolant channels through feeders is very important for the safety of the reactor. These flow rates are influenced by geometry, configuration of connecting feeders, system pressure and vortex formation. Because this system of pipes is complex and the connection fuel channels-feeders-headers is also complex taking in count the flow going in opposite directions in adjacent fuel channels, a 3D visualisation screen in GRAPE is needed.

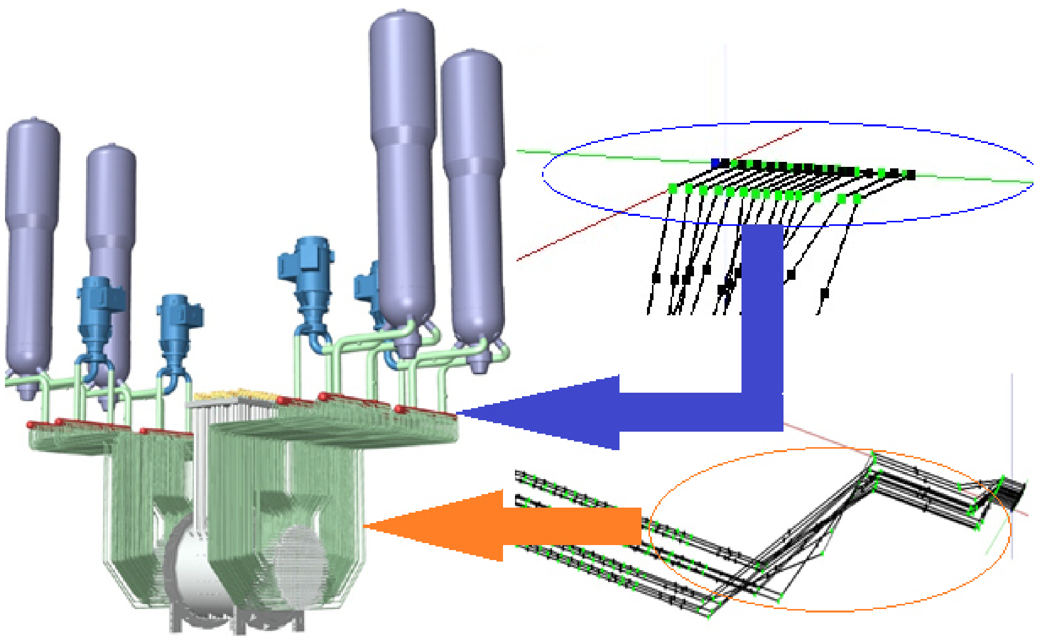

Figure 17 Fuels channels-feeders-header connections model in RELAP/SCDAPSIM (11)

5.2 3D model describing the Calandria Vessel

Flow and temperature patterns inside the calandria vessel both axial and radial directions will be needed as soon as the initial model with 12 parallel pipes will be tested and the accident sequence is runned.

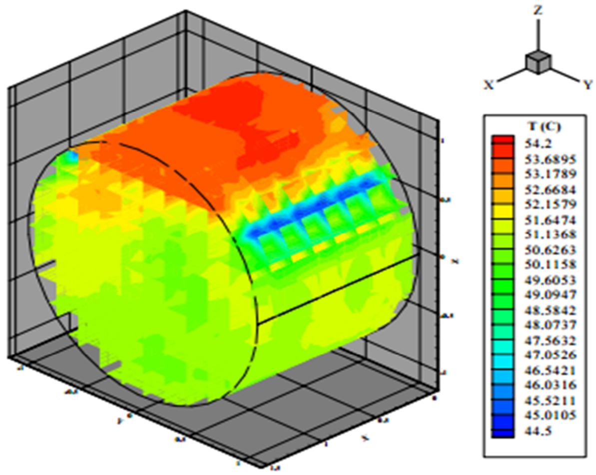

Figure 18 Assumed temperature distribution inside the calandria vessel (12)

This becomes particularly important during a core uncovery scenario where a complex temperature distribution can develop both vertically because of the vertical variation in power and water level and horizontally (parallel to the face of the calandria vessel) because of the horizontal variation in power. This in turn can lead to even more complex 3D patterns of oxidation and resulting heating on the exterior of the channel walls.

The 2D model is not able to give an integral view of the core collapse or the transition phase from where the intact fuel channels start to sagg, oxidize, and collapse to the bottom of the calandria vessel. Complex temperature distribution (vertically and horizontally) results in 3D patterns of oxidation.

6 Conclusions

This section will be completed by January 20th, the mentioned deadline for the final paper submission.

Acknowledgement

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References

1. *Moderator, Heat Transport and Steam Systems.* **G. Hongqi, R.S. Hart.** 6, s.l. : China Journal of Nuclear Power Engineering, 1999, Vol. 20.

2. **Hart, R.S.** *CANDU Technical Summary.* s.l. : AECL, 1997.

3. **J.H. Kim, D.S. Jin, S.H. Chang.** Development of safety analysis methodology for moderator system failure of CANDU-6 reactor by thermal-hydraulics/physics coupling. *Nuclear Engineering and Design.* 2013, Vol. 263.

4. **Projects, CITON - Center of Technology and Engineering for Nuclear.** *Wolsong Nuclear Power Plant Unit No 2. Preliminary Safety Analysis Report, Volume 3, Chapter 5 - Reactor Process Systems.* Magurele : s.n., 1991.

5. **L. Siefken, E. Coryell, E. Harvego, and J. K. Hohorst.** *SCDAP/RELAP5/MOD3.3 Code Manual,.* s.l. : NUREG/CR-6150 – INEL-96/0422 – Revision 2., 1996.

6. RELAP5&RELAP/SCDAPSIM Software. *RELAP5 Thermal-Hydraulic Safety Analysis Software.* [Online] Innovative Systems Software, 2016. [Cited: 20 November 2016.] www.relap.com.

7. *Role of RELAP/SCDAPSIM in Nuclear Safety.* **C. M. Allison, J. K Hohorst.** Dubrovnik, Croatia : TopSafe, 2008.

8. *A study of RELAP5 capability to perform the analysis of CANDU 6 reactors accidents.* **I Prisecaru, D Dupleac, L Biro.** Avignon, France : The 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), 2005.

9. *GRAPE User's Guide (Version 1.1.0).* s.l. : Nuclear Engineering, Ltd., 2015. BTN-1510036.

10. *GRAPE: Graphical RELAP/SCDAPSIM Analysis Platform for Education and Engineering.* **Masahiro Tatsumi, Kosuke Tsujita, Kotaro Sato, Yohei Tamari.** Charlotte, North Carolina, USA : 24th International Conference on Nuclear Engineering, 2016. ISBN: 978-0-7918-5004-6.

11. *Development of Detailed RELAP/SCDAPSIM/MOD3.6 Input Models and User Guidelines for the Analysis of a CANDU Core During Core Uncovery Scenarios.* **R. M. Nistor-Vlad, C. M. Allison, J. K. Hohorst.** San Francisco, CA, USA : International Congress on Advances in Nuclear Power Plants, 2016.

12. *Velocity and Temperature Distribution Measurements Inside a Scaled Calandria in Support of CANDU X Passive Moderator Cooling System Design .* **Hussam F. Khartabil, Romney B. Duffey.** Kyoto, Japan : International Conference on Global Environment and Advanced Nuclear Power Plants (GENES4/ANP2003), 2003. Paper 1205.