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Role of RELAP/SCDAPSIM in Nuclear Safety

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ABSTRACT

The RELAP/SCDAPSIM code, designed 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). SDTP consists of nearly 60 organizations in 28 countries supporting the development of technology, software, and training materials for the nuclear industry. The program members and licensed software users include universities, research organizations, regulatory organizations, vendors, and utilities located in Europe, Asia, Latin America, and the United States. Innovative Systems Software (ISS) is the administrator for the program. RELAP/SCDAPSIM is used by program members and licensed users to support a variety of activities. The paper provides a brief review of some of the more important activities including the analysis of research reactors and Nuclear Power Plants, design and analysis of experiments, and training.

1 INTRODUCTION

The RELAP/SCDAPSIM code, designed 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) [1,2]. SDTP consists of nearly 60 organizations in 28 countries supporting the development of technology, software, and training materials for the nuclear industry. The program members and licensed software users include universities, research organizations, regulatory organizations, vendors, and utilities located in Europe, Asia, Latin America, and the United States. Innovative Systems Software (ISS) is the administrator for the program.

Three main versions of RELAP/SCDAPSIM, as described in Section 2, are currently used by program members and licensed users to support a variety of activities. RELAP/SCDAPSIM/MOD3.2 and MOD3.4 are production versions of the code and are used by licensed and program members for critical applications such as research reactor and nuclear power plant applications. The most advanced production version, MOD3.4, is also used for general user training and for the design and analysis of severe accident related experiments such as those performed in the Phebus and Quench facilities. In turn, these experiments are used to improve the detailed fuel behaviour and other severe accident-related models in MOD3.4 and MOD4.0. MOD4.0 is currently available only to program members and is used primarily to develop advanced modelling options and to support graduate research programs and training.

2 RELAP/SCDAPSIM

RELAP/SCDAPSIM uses the publicly available RELAP/MOD3.3[3] and SCDAP/RELAP5/MOD3.2[4] models developed by the US Nuclear Regulatory Commission in combination with proprietary (a) advanced programming and numerical methods, (b) user options, and (c) models developed by ISS and other members of the SDTP. These enhancements allow the code to run faster and more reliably than the original US NRC codes. MOD3.4 and MOD4.0 can also run a much wider variety of transients including low pressure transients with the presence of non-condensable gases such as those occurring during mid-loop operations in LWRs, in pool type reactors, or in spent fuel storage facilities.

The most advanced version of the code, RELAP/SCDAPSIM/MOD4.0[5], is the first version of RELAP or SCDAP/RELAP5 completely rewritten to FORTRAN 90/95/2000 standards. This is a significant benefit for the program members that are using the code for the development of advanced models and user options such as the coupling of the code to other analysis packages. Coupled 3D reactor kinetics and coupled RELAP/SCDAPSIM-SAMPSON [6] calculations are examples where MOD4.0 is used because of a significant reduction in the code development effort and expense to link the packages. MOD4.0 also includes advanced numerical options such as improved time advancement algorithms, improved water property tables, and improved model coding. As a result the code can reliably run complex multi-dimensional problems faster than real time on inexpensive personal computers. Plant simulation and integrated uncertainty analysis are among the most important applications benefiting from the improved speed and reliability of MOD4.0. MOD4.0 includes many enhanced user options to improve the accuracy of the code or to offer new options for the users. For example, the addition of an alternative material property library designed for Zr-Nb cladding materials is important for VVER and CANDU reactor designs, particularly for severe accident related transients. The addition of an advanced water property formulation is important for many transients, in particular those involving super critical water applications.

3 REVIEW OF REPRESENTATIVE APPLICATIONS

RELAP/SCDAPSIM is being used for a variety of applications. As described in Section 3.1, the code is used for the analysis of research reactors and nuclear power plants. The research reactors analysed by the code include TRIGAs, MTR-plate designs, and well as other unique designs. Nuclear plants analysed include Western designed PWRs and BWRs, Russian designed VVERs and RMBKs, Canadian and Indian designed CANDUs and HWRs, and Chinese designed PWRs. The analysis of experiments, as discussed in Section 3.2, is also an important application of the code. This application includes the design of the experiments, the assessment and development of modelling improvements, and finally advanced user training. The application of the code to support the development of improved models and analytic capabilities is discussed in Section 3.3. Section 3.4 presents an overview of the application of the code for training.

3.1 Research Reactor and NPP Applications

A combination of RELAP/SCDAPSIM/MOD3.2 and MOD3.4 is being used to analyze research reactors. A brief summary of the early work by several countries was given in Reference [6]. The research reactors noted in this paper include (a) the LVR-15 reactor located at the Nuclear Research Institute in Rez, Czech Republic, (b) the CARR reactor being built in Beijing, China by the China Institute of Atomic Energy, and (c) TRIGA reactors located at the Atomic Energy Research Establishment in Dhaka, Bangladesh and National

Nuclear Energy Agency in Bandung Indonesia. LVR-15 is a light-water moderated and cooled pool type reactor with a nominal thermal power of 15 MW. The pool operates at atmospheric pressure with an average coolant temperature in the core of 320 K. The reactor also has closed high pressure/temperature loops suitable for testing of materials under PWR and BWR conditions. The reactor core is composed of several fuel assemblies of Al-U alloy arranged in square concentric tubes. CARR is a tank-in-pool design, cooled and moderated by light water and reflected by heavy water. The rated power is 60 MW. The core consists of plate-type fuel assemblies of Al-U alloy. The Indonesian and Bangladesh TRIGA reactors are pool type reactors with 2 MW and 3 MW thermal power respectively. The reactor cores are composed of solid U-ZrH fuel rods arranged in a hexagonal array and are cooled by water in either forced or natural circulation, depending upon the conditions.

More recently, the analysis of two additional reactor types have been reported in References 7-9. The first is for the SAFARI-1 research reactor located in South Africa [7,8]. The second is the University of Missouri Research Reactor located in the United States [9]. The SAFARI-1 research reactor is a tank-in-pool type reactor operated at a nominal core power of 20MW. The core is cooled and moderated by forced circulation of light water. The reactor core can be operated in a variety of configurations from 24 to 32 fuel assemblies. Figure 1 shows an example of one such configuration. The fuel is U-Si-Al plate-type fuel elements. MURR is a 10 MW pool type reactor design with a pressurized primary coolant loop to cool the fuel region. The pressurized primary system is located in a pool allowing direct heat transfer during normal operation and transition to natural convection under accident conditions. The reflector region, control blade region, and center test hole are cooled by pool water (natural convection).

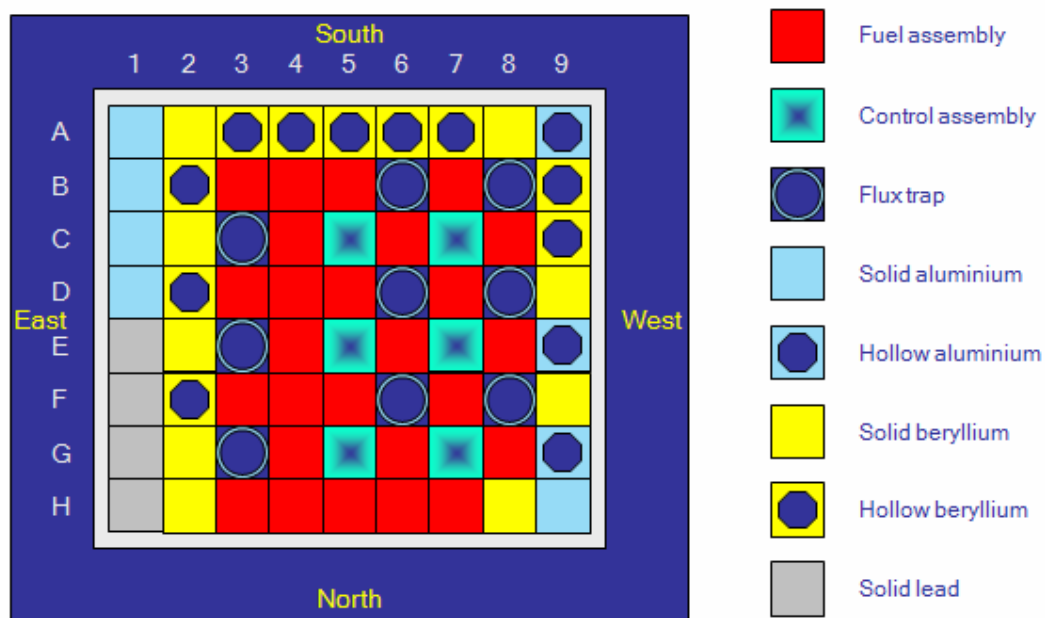


Figure 1: SAFARI Reactor Core Configuration.

Because of the unique reactor designs, the RELAP/SCDAPSIM input models were developed separately by each organization and include a range of different nodalizations as presented in the reference papers. However in general terms, the RELAP/SCDAPSIM input models include all of the major components of each reactor system including the reactor tank,

the reactor core and associated structures, and the reactor cooling system including pumps, valves, and heat exchangers. The secondary sides of the heat exchanger(s) are also modelled where appropriate. These input models were qualified through comparison with reactor steady state data, with original vendor safety analysis calculations where available, and with experiments in a limited number of cases. Figures 2 through 4 give examples of the nodalization used for MURR. Figures 2 and 3 show the detailed hydrodynamic nodalization for the pressurized primary cooling system and the bulk pool and pool cooling system, respectively. Figure 4 shows the nodalization of the fuel plates. This input model is also somewhat unique in that all 24 fuel plates were modeled using RELAP5 heat structures.



Figure 2: MURR RELAP/SCDAPSIM Nodalization of the MURR Pressurized Primary Cooling System.

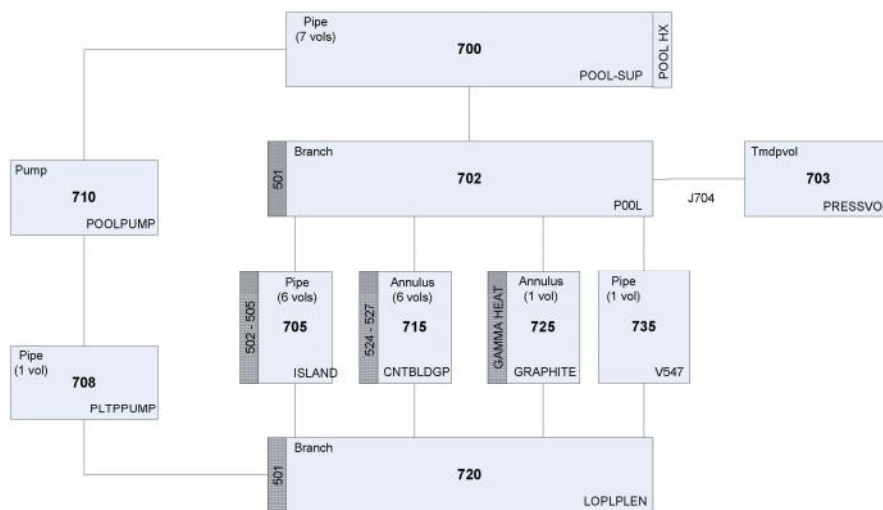


Figure 3: RELAP/SCDAPSIM Nodalization of the MURR Bulk Reactor Pool and Pool Coolant Loop.

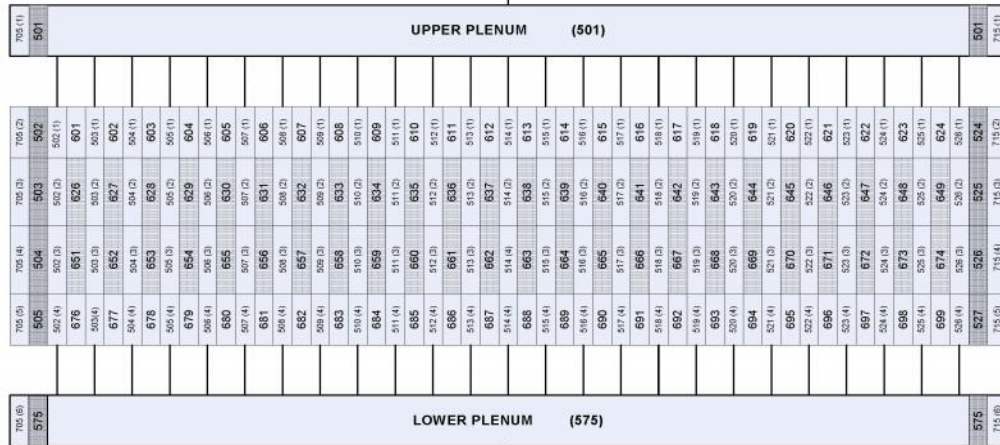


Figure 4: RELAP/SCDAPSIM Nodalization of the MURR 24 Fuel Plate Core.

Figure 5 show the nodalization used for the SAFARI research reactor. Figure 5 shows the overall system hydrodynamic nodalization with the upper right corner of the figure showing the core nodalization. Note from the insert of the core nodalization diagram that the bypass or unheated channels were modelled separately from the heated fuel assembly channels. The core nodalization also included two separate hot plate channels located on each side of the hottest plate.

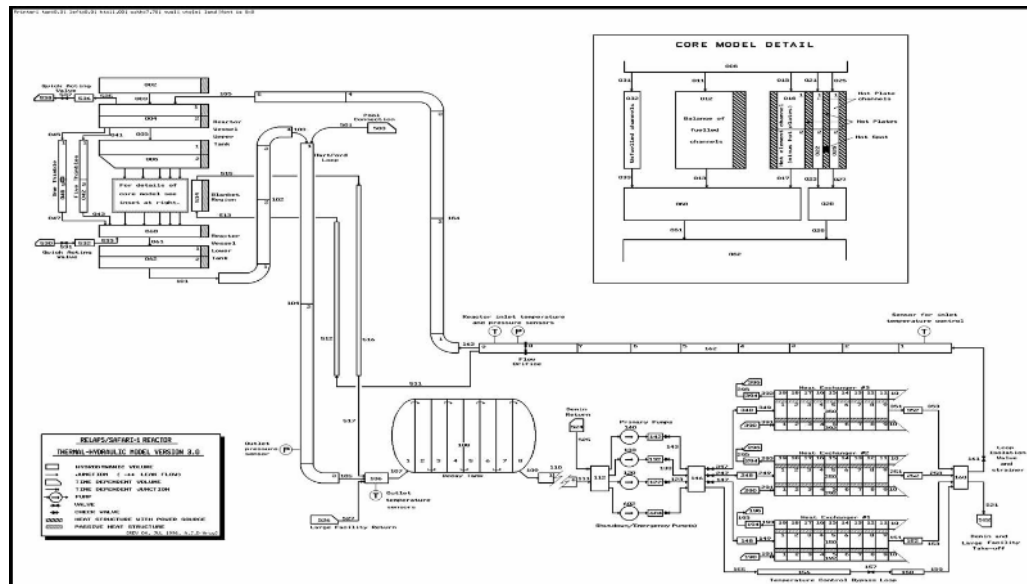


Figure 5: RELAP/SCDAPSIM Nodalization of the SAFARI Cooling System.

A wide variety of transients have been analyzed using the code. Examples are included in the references and include reactivity initiated power excursions and loss of flow or coolant

transients. Figure 6 shows one such example for MURR. The figure shows the centerline temperatures for the 24 fuel plates during a cold leg LOCA, indicating that the fuel transients remained well below the assumed fuel damage limits of 900°F.

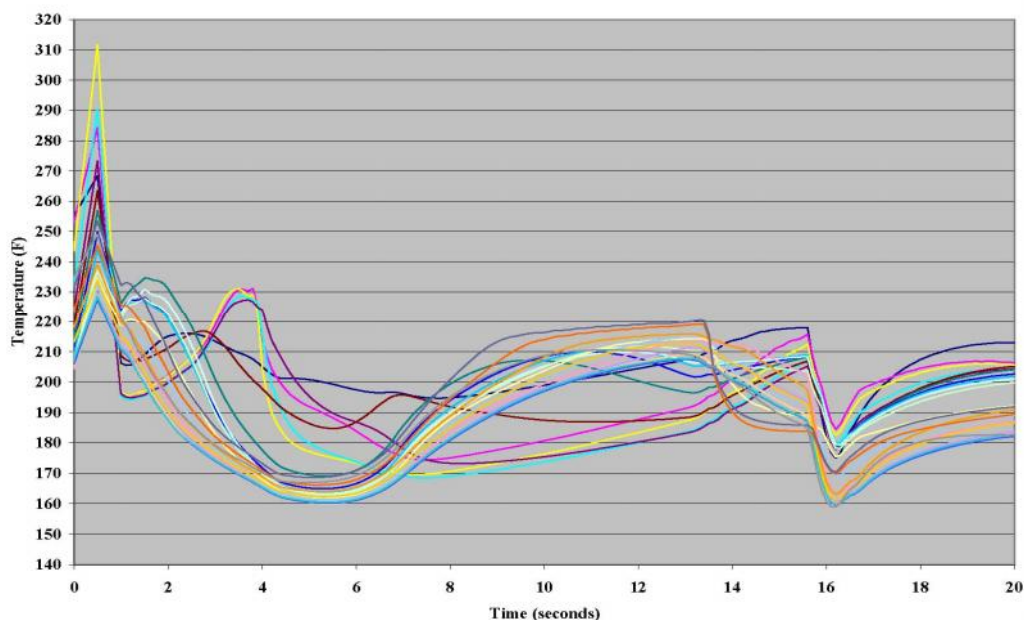


Figure 6: Example of RELAP/SCDAPSIM Calculated Fuel Plate Centerline Temperatures for the 24 Fuel Plates in MURR for a Cold Leg Large Break Transient.

All three versions of RELAP/SCDAPSIM have been used to analyze a variety of nuclear power plant designs. The applications have included RELAP5-only input models for normal operating or transient conditions where core damage is not expected as well as combined RELAP-SCDAP input models that included the possibility of transients with the loss of core geometry. A few representation examples are discussed in more detail in the remainder of this subsection.

Analysts at the Paul Scherrer Institut (PSI) in Switzerland have applied the code to the TMI-2 accident [10], an analysis of a LOCA during cooldown in the Beznau Westinghouse type 2-Loop PWR [11], and an analysis of a station blackout transient in the Gösigen Nuclear Plant [12]. The TMI-2 calculations included comparisons with the limited data available from the accident as well as comparisons with the MELCOR [13] and SCDAP/RELAP5[14] codes. The Beznau analysis paper summarized the results of the analyses of postulated LOCAs in the Beznau (KKB) PWR, occurring during hot (HS) and intermediate (IS) shutdown with emphasis on large break LOCAs during hot shutdown. The large break LOCA during HS posed the greatest challenge to the plant safety systems. The analysis of the station blackout transient in the Gösigen Nuclear Plant focused on the impact of a potential failure of the depressurization system. In particular, the analysis focused on the timing of the heatup and failure of the RCS piping relative to the relocation of melt into the lower plenum and failure of the lower head. MELCOR, RELAP/SCDAPSIM, and SCDAP/RELAP5 were also used in both Beznau and Gösigen analyses.

The TMI-2 RELAP/SCDAPSIM and SCDAP/RELAP5 nodalization, as shown in Figure 7, used a 2 dimensional representation of the core region with a detailed SCDAP components being used to describe the behavior of the fuel rods and other core structures within each of the five representative flow channels in the core. The transition from the initial

intact core geometry to a damaged state is automatically handled by the SCDAP models including the initial failure of the control rods, liquefaction and relocation of the metallic U-O-Zr fuel rod material, formation and growth of a ceramic [U-Zr]-O₂ molten pool, and relocation of the molten ceramic into the lower plenum. Figure 8 shows one of the set of representative calculations presented in the paper. The figure shows the variation in predicted system pressure for a range of modeling parameters for the relocation of the metallic U-O-Zr fuel rod material for both RELAP-based codes. It was noted in the paper that both RELAP-based codes correctly calculated an in-core molten pool, of which two RELAP/SCDAPSIM cases predicted relocation to the lower head (via the bypass, as observed), while only one MELCOR case did so. It was further noted that the RELAP-based codes correctly calculated that lower head failure did not occur.

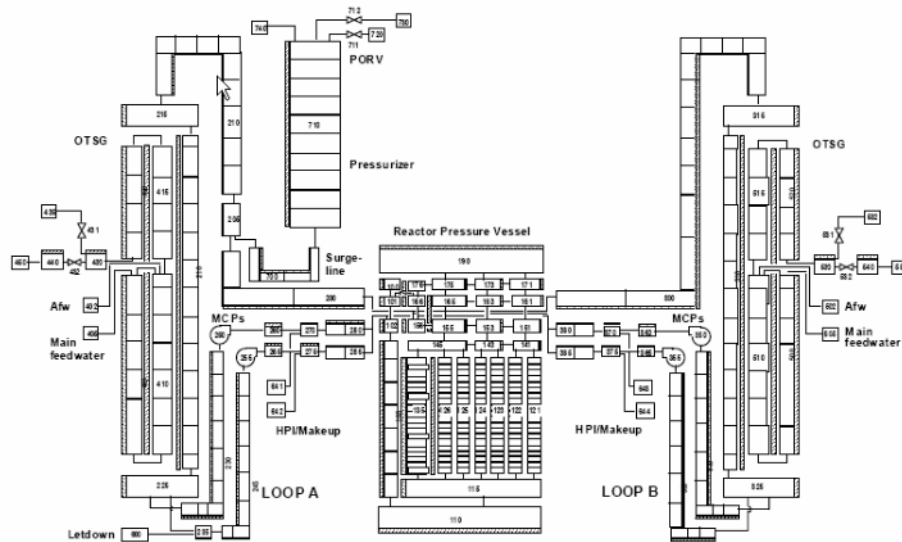


Figure 7: TMI-2 Nodalization Used for RELAP/SDAPSIM and SCDAP/RELAP5.

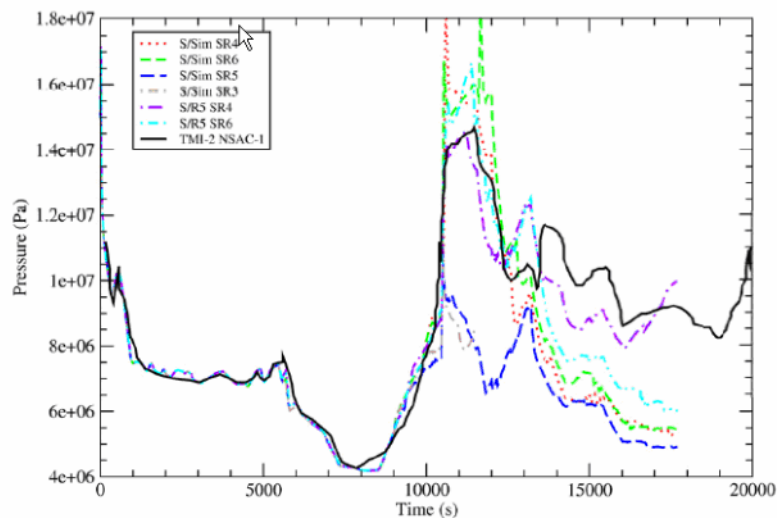


Figure 8: Example of RELAP/SDAPSIM and SCDAP/RELAP5 TMI-2 Calculated Results for System Pressure for a Range of Metallic Fuel Rod Material Relocation Modeling Parameters.

The Beznau RELAP/SCDAPSIM and SCDAP/RELAP5 nodalization, as shown in Figure 9, also used a 2 dimensional representation of the core region with detailed SCDAP components used to describe the behavior of the fuel rods and other core structures within each of the five representative flow channels in the core. Figure 10 shows one of the set of representative calculations presented in the paper. The figure shows the peak core temperatures calculated by MELCOR and the RELAP5-based codes for different assumptions regarding the activation of the Safety Injection pumps including the number of pumps and delays in the actuation of the pumps. The paper concludes that all three codes predict that in the limiting large break case the core is readily quenched without damage, by the nominal operation of the system injection system. However, it was noted that the more mechanistic RELAP-based calculations demonstrated that a larger margin existed (relative to that predicted by MELCOR) with recovery being possible even if only one pump operates after some delay.

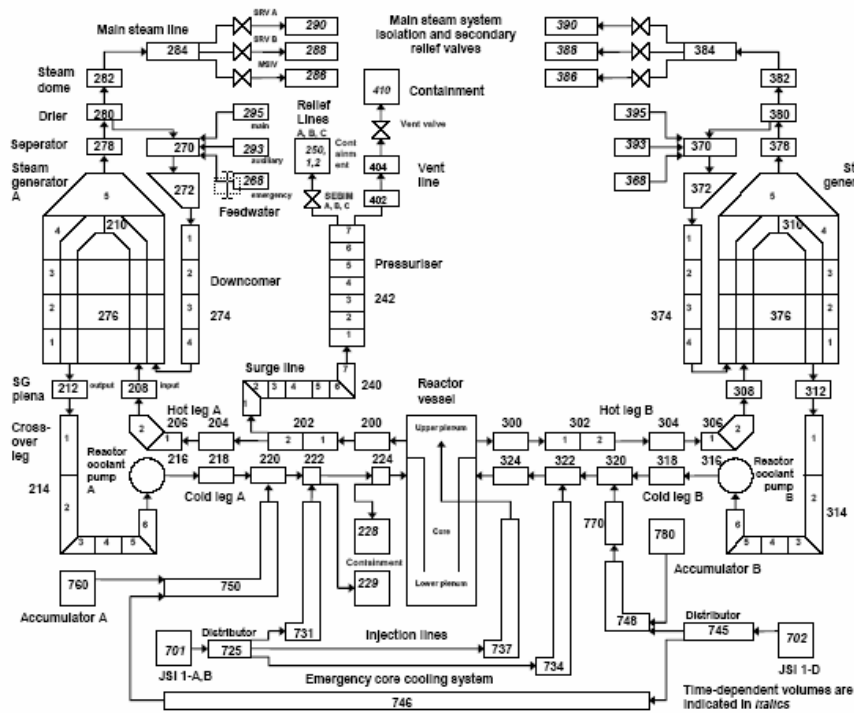


Figure 9: Beznau Nodalization Used for RELAP/SDAPSIM and SCDAP/RELAP5.

The RELAP/SCDAPSIM and SCDAP/RELAP5 nodalization used for the Gösgen analysis also included a detailed core nodalization as described previously. However, the calculations also looked at the effect of hot leg nature circulation using a split hot leg model as shown in Figure 11. The split channel model allows the hotter vapor to move from the vessel to the steam generators along the top of the hot leg and cooler vapor to return along the bottom of the hot leg. The influence of the split hot leg input model relative to a single channel hot leg (which does not allow countercurrent flow of the vapor within the hot leg) is shown in Figure 12. As shown in the figure the split hot leg model predicted a more gradual heatup of the core but both single channel and split channel models still predict rupture of the surge line or hot leg piping before any molten core material relocates into the lower head.

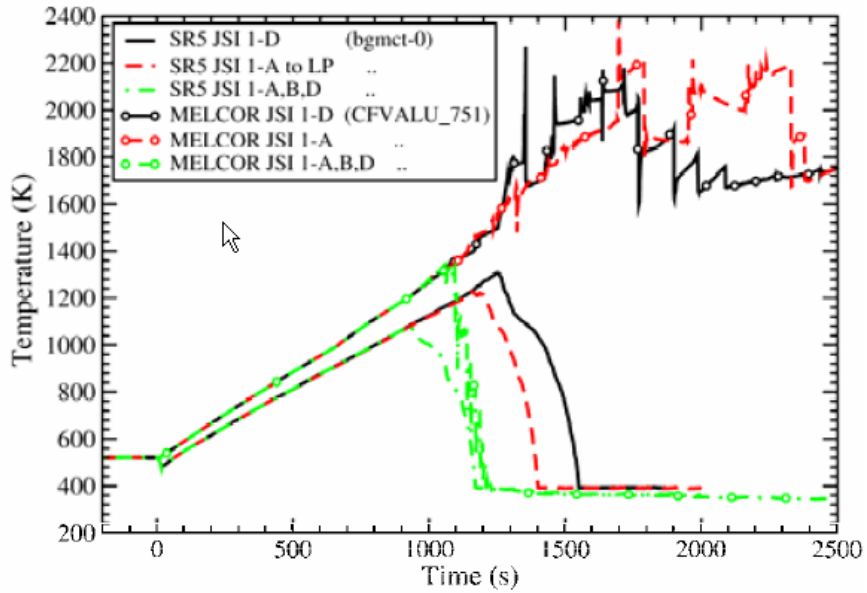


Figure 10: Example of RELAP-based and MELCOR Beznau Calculated Results for Peak Core Temperature for a Range of Safety Injection Assumptions.

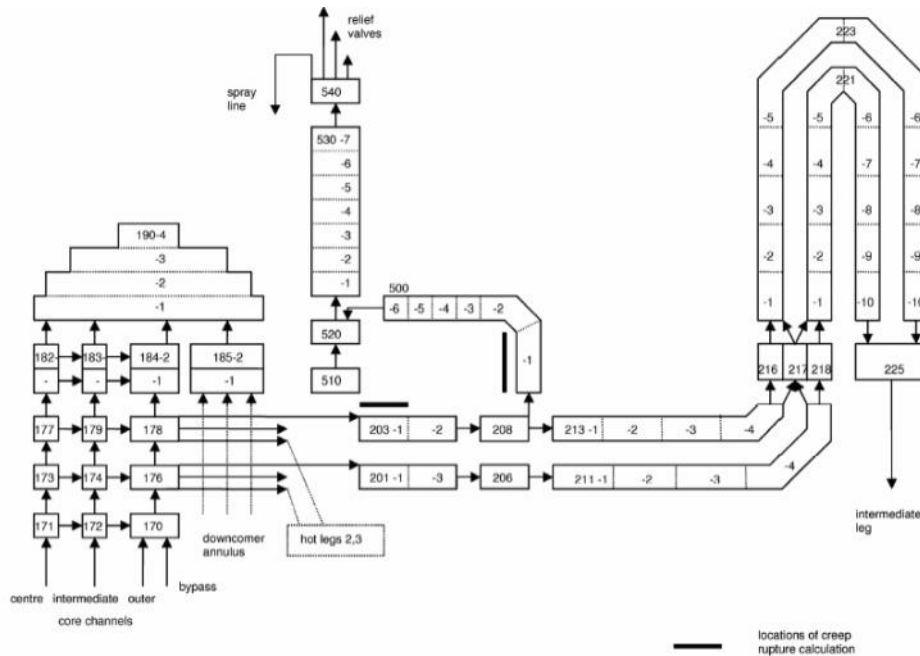


Figure 11: Gösgen Split Hot Leg Nodalization Used for RELAP/SDAPSIM and SCDAP/RELAP5.

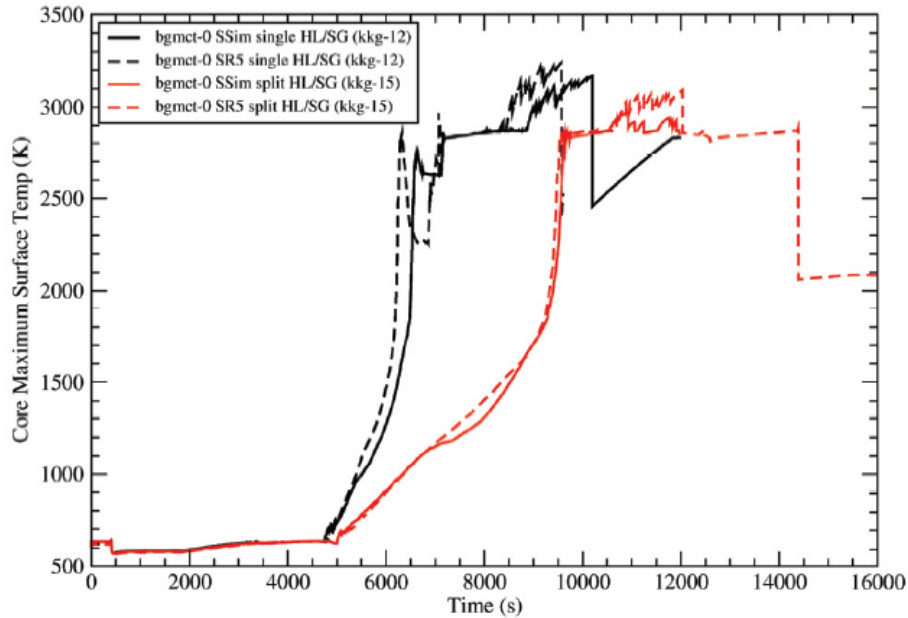


Figure 12: Example of RELAP/SCDAPSIM and RELAP/SCDAPSIM Calculated Results for Peak Core Temperature for Single and Split Hot Leg Model.

The Politehnica University, Institute for Nuclear Research, and National Commission for Nuclear Activities Control in Romania have used RELAP/SCDAPSIM/MOD3.4 for a variety of analyses of CANDU reactor designs. Reference [13] presents the analysis of reactor inlet header break, looking at the size of the break, the choked flow model employed, the emergency core cooling (ECC) performance and the core nodalization. The results were compared with the original safety analysis results. Reference [14] presents the analysis of the influence of the header manifold modeling for an inlet header break in a CANDU 6. The paper looked at a 35% inlet header break which was expected to produce the highest peak fuel cladding temperatures among all postulated break sizes. Reference [15] presents the analysis of a reactor outlet header break in a CANDU-6. The paper focused on a 100% reactor outlet header break which had the highest potential for fuel failure and release of radioactivity. The paper also compared the results to earlier calculations performed using the CATHENA code [16]. Reference [17] presents an analysis of the late phase of a severe accident in CANDU 6 where bed of dry solid debris or a molten pool of core material had already formed at the bottom of the calandria vessel and was externally cooled by shield-tank water. The study used the SCDAP COUPLE module and included comparisons with earlier results performed using the ISAAC [18] and MAAP4-CANDU [19] codes.

The general system thermal hydraulic nodalization for the CANDU system thermal hydraulic analysis analyses [13-15] is shown in Figure 13. Portions of this nodalization were varied somewhat depending on the analyses being performed. Figure 14 shows an example of the portion of the nodalization that was used in [14] for the study of the influence of the inlet header model. Figure 15 shows an example of some of the results from the inlet header break model study. In the figure, the reference curve is the results from a single average channel circuit model using a single manifold volume and Cases 1, 2, and 3 represent the break location in the multiple inlet manifold volumes in combination with multiple core channels. The curves shown in the figures are the maximum cladding temperatures in fuel bundles contained within the multiple flow channels.

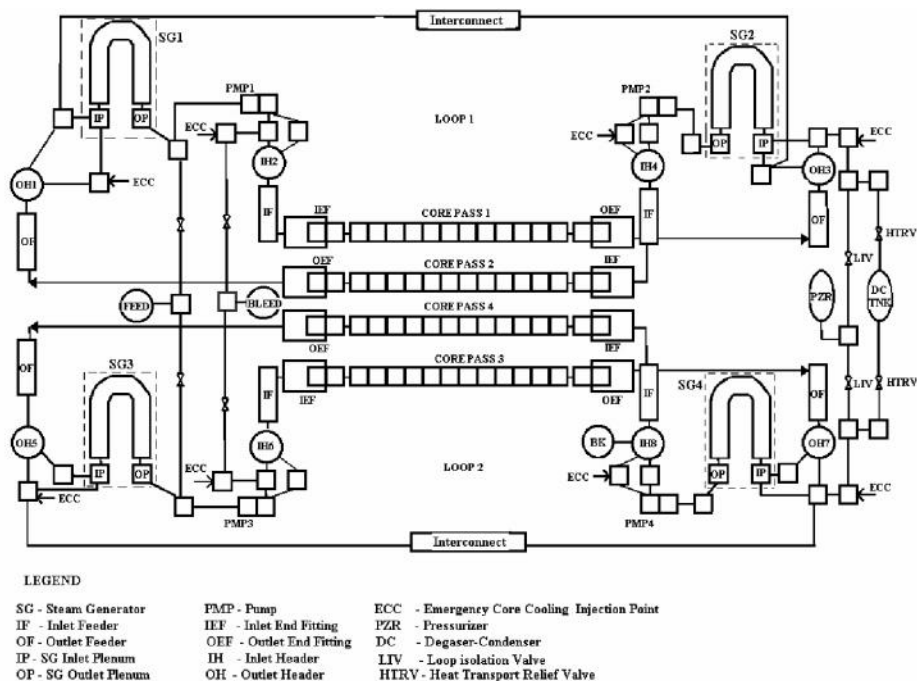


Figure 13: RELAP5 Thermal Hydraulic Nodalization for CANDU-6.

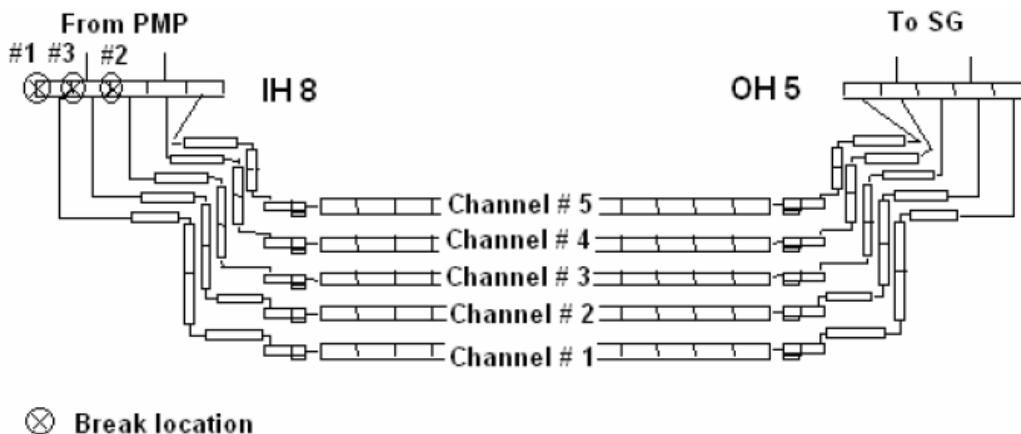


Figure 14: RELAP5 Thermal Hydraulic Nodalization of Manifold Headers for CANDU-6 Inlet Manifold Break Analysis.

Figure 16 shows the basic problem analyzed for late phase of a severe accident in a CANDU-6 where a debris bed is present in the bottom of the calandria vessel along with the RELAP5, SCDAP, and COUPLE nodalization used in the analysis. The RELAP5 thermal hydraulic volumes on the right of the figure represent the pool on the outside of the calandria vessel. The RELAP5 and SCDAP volumes above and within the COUPLE mesh provide initial and boundary conditions for the debris bed and calandria vessel wall. (The paper also included a more detailed RELAP5 nodalization of the outer pool at the elevations associated with the debris bed. The more detailed nodalization resulted in significantly lower pool containment pressures upon vessel failure due to the more accurate representation of the external cooling of the calandria vessel.)

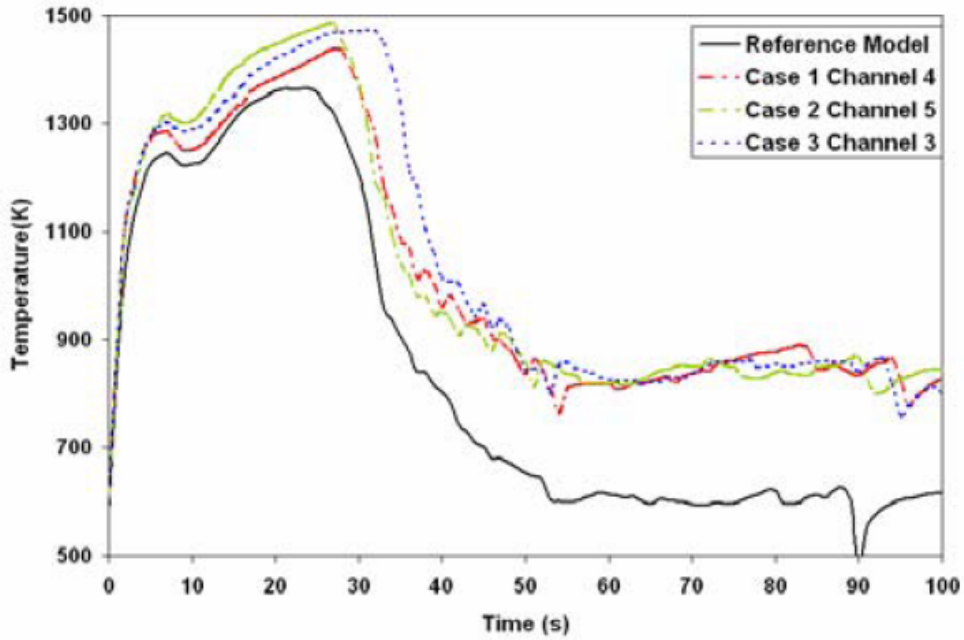


Figure 15: Maximum Fuel Bundle Cladding Temperatures for a CANDU-6 Inlet Manifold Break Analysis for Different Break Locations.

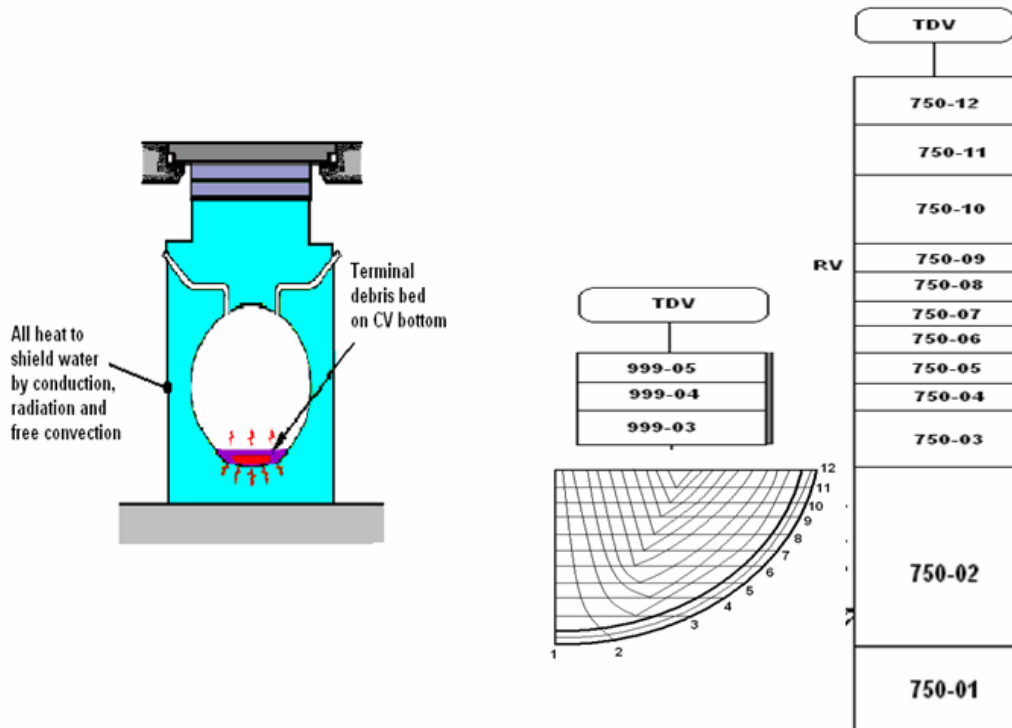


Figure 16: RELAP, SCDAP, and COUPLE Module Nodalization and Conceptual Sketch for CANDU-6 Calandria Vessel Analysis.

The papers indicated that RELAP/SCDAPSIM calculations gave comparable results to the CANDU-specific codes, CATHENA for system thermal hydraulics, and ISAAC and MAAP-CANDU for severe accidents. For system thermal hydraulic analysis, RELAP/SCDAPSIM, when using similar input models, provided similar trends as compared to the original safety analysis reports or comparable CATHENA calculations. However, the results were sensitive to the level of detail used in the nodalization, specifically the more detailed nodalization possible using RELAP/SCDAPSIM had a noticeable impact on the results for the inlet header manifold [14], the fuel channels (simulating the effects of horizontally stratified flow in the channels [13], and the outer pool (exterior to the vessel calandria) [17].

RELAP/SCDAPSIM has also been used to analyze VVER reactor designs although the calculations to date have been proprietary and have not been published in the open literature. Figures 17-18 shows a non-proprietary, but representative, input model for a VVER-1000. This representative VVER-1000 input model and associated detailed input model engineering handbook was provided by Risk Engineering in Bulgaria [20] as an in-kind contribution for use in SDTP-sponsored VVER training activities. The nodalization of the basic components of the reactor (vessel and internals) is presented in Figure 16 and includes two channels in the core, the hot and peripheral channels. This basic vessel nodalization would be replaced by a more detailed multi-dimensional model comparable to that used for the PWR calculations described previously for general applications where core damage transients might be considered. Four separate main circulation loops with their corresponding main coolant pumps, cold and hot circulation pipelines are also included in the representative input model as shown in Figure 17.

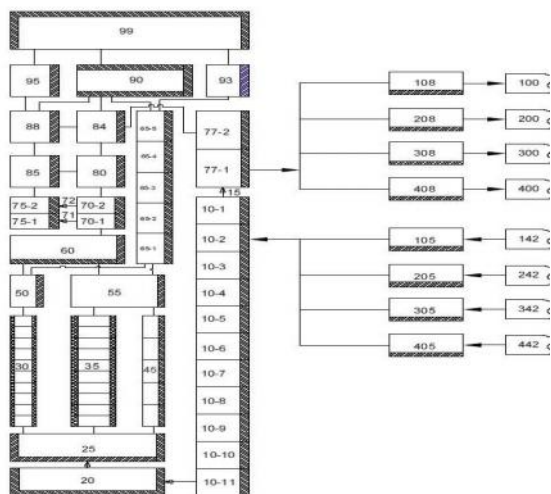


Figure 17: Representative Vessel Input Nodalization for VVER-1000 for Transients Not Including the Possibility of Loss of Core Geometry.

Analysts at the Lithuanian Energy Institute (LEI) have published a number of papers describing their use of RELAP/SCDAPSIM/MOD3.2 for the analysis of RBMKs. Recent references are cited as [21-23]. LEI also provided non-proprietary, but representative, input models and associated engineering handbooks for use in SDTP-sponsored training activities. Figure 19 shows the nodalization diagram used for the representative RBMK RCS input model.

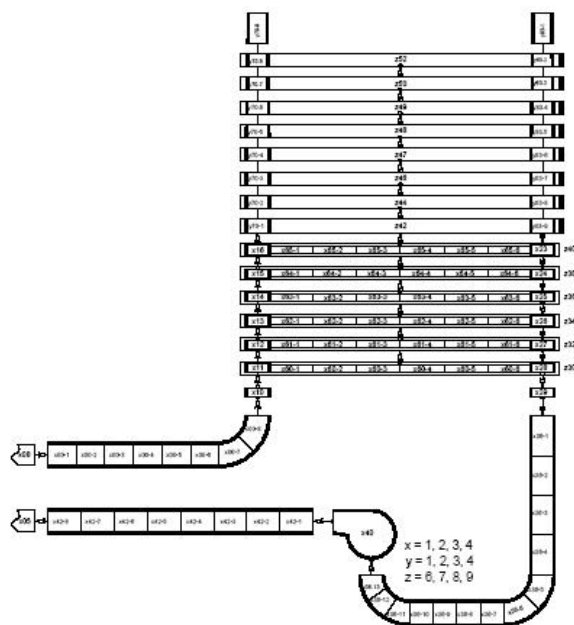


Figure 18: Representative Loop Input Nodalization for VVER-1000.

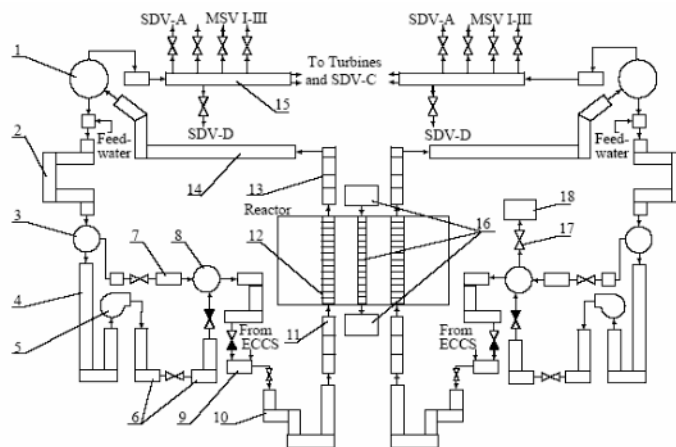


Figure 19: Representative RCS Nodalization for RBMK.

All three versions of RELAP/SCDAPSIM/MOD4.0 have been used to analyze BWRs although few results have been published in the open literature. Reference [24] describes some of the activities related to the use of the code for plant simulation and training for the Laguna Verde plants in Mexico. Figure 20 shows a representative input model, developed by the Nacional de Seguridad Nuclear y Salvaguardias (CNSNS), the Mexican regulatory authority, for the analysis of the Laguna Verde plants. This model is also used to support SDTP-sponsored BWR-specific training activities. See Section 3.4 and Reference 24 for more information on these activities in Mexico.

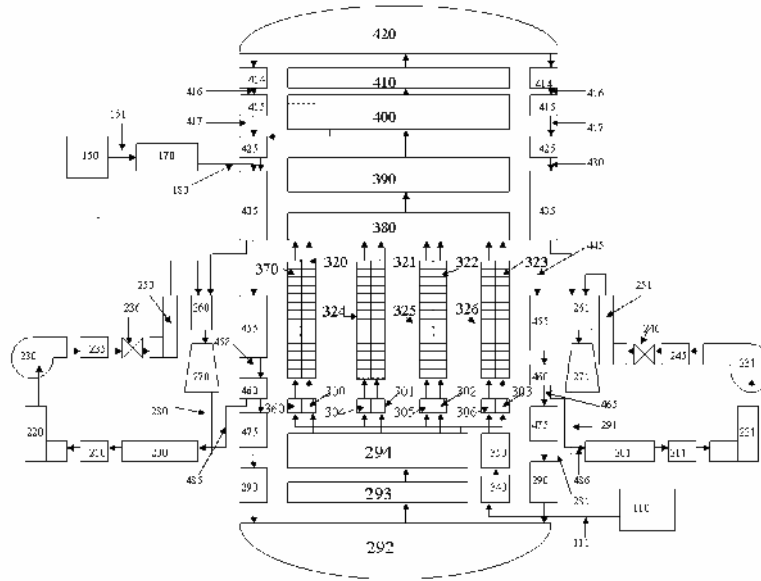


Figure 20: Representative RCS Nodalization for BWRs.

3.2 Experimental Analysis

RELAP/SCDAPSIM/MOD3.4 has been used by a number of organizations to help design experiments, to assess thermal hydraulic and severe accident models, and to support advanced user training. In recent years, the application of the code to experimental analyses have focused on European experimental programs including the German Quench experiments [25-29], French Phebus FPT experiments [30-33], and most recently Russian PARAMETER experiments [34].

The most detailed of the calculations have been involved in the design of new experiments. For example, as described in detail in [26], the design of new experiments requires the development of complex input models to describe the unique features of each experiment and in many cases, the development of specialized new models to treat features of the experiments not previously included in the code. In this example, the analysts from PSI and experimentalists from Forschungszentrum Karlsruhe (FzK) describe their use of RELAP/SCDAPSIM/MOD3.4 in conjunction with MELCOR and a special FzK-developed version of SCDAP/RELAP5 [35] to design and analyze three different experiments in the quench facility, Quench-10, Quench 11, and Quench 12. Quench 10 was a unique experiment in that it was the first integral test to look at the influence of air ingress on bundle heating and reflood. The design and analysis of this experiment required PSI to develop and incorporate special SCDAP models to treat the oxidation of Zircaloy in air/steam mixtures. The experimentalist also ran special small separate effects experiments to help develop the correlations that were then used in these new models. Quench 11 was a unique test for the Quench facility in that the test started with the bundle full of water and then the heat up transient was initiated by the boil-down of the water. (Previous Quench experiments used a mixture of steam and argon during the heat up of the bundle prior to reflood.) Although in this case, it was not necessary to modify any of the RELAP or SCDAP models, the modeling of the auxiliary heaters, added to the lower plenum of the experimental test train to provide

realistic boildown rates, proved to difficult because of relatively large heat losses in the lower plenum region. QUENCH-12 was unique in that it was designed to determine the influence of a WWER bundle configuration and cladding on heat-up, oxidation, and quench response. Previous Quench experiments used PWR or BWR configurations and cladding materials. The Quench 12 bundle was significantly modified with changes to cladding material (Zr/1%Nb instead of Zry-4), electrical heating, and geometry. Oxidation correlations for Zr/1%Nb in steam were introduced into SCDAP to support the design and analysis of this experiment. Figure 21 shows a schematic of the Quench facility along with the RELAP/SCDAPSIM nodalization diagram.

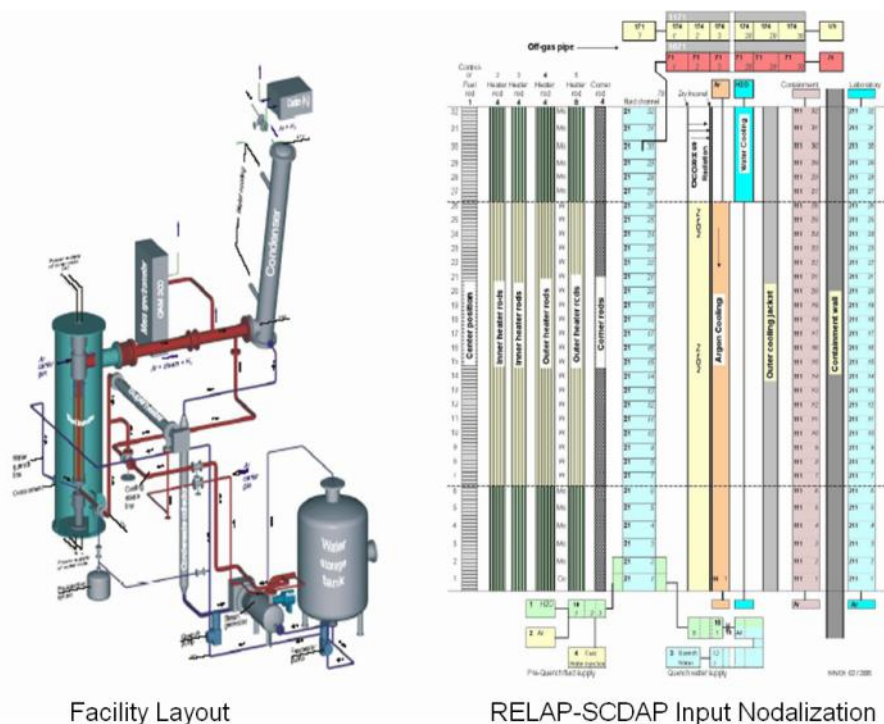


Figure 21: Example of Experiment Design – Quench Facility Layout and Input Nodalization.

The analysis of the German Quench and French Phebus experiments have also played a pivotal role in the assessment of RELAP5/SCDAPSIM, the development of new improved models as discussed in Section 3.3, and in advanced user training as discussed in Section 3.4. References 28, 29, 31-33 are examples of the analysis of these experiments to assess the accuracy of the code and to identify areas where the models could be improved. References 36 and 37 describe the use of these experiments to support advanced user training.

3.3 Development of improved models and analytic capabilities

The development of improved models and analytic capabilities is also an important part of the overall SDTP cooperative activities. In addition to the modelling improvements driven by large scale experimental programs in the Phebus and Quench facilities as discussed in the previous section, other model and code development activities have been driven by the needs of SDTP members and licensed software users. INSS (Institute of Nuclear Safety System), Japan, one of the original members of SDTP, developed and validated new RELAP/SCDAPSIM models to treat the heat transfer in the gap between a debris bed and the

lower plenum wall [38] and improved correlations for condensation in the presence of non-condensable gases [39]. The application of the improved correlations are described in [40]. IAE/NUPEC (Institute of Applied Energy/Nuclear Power Engineering) Japan, a long time member of SDTP, has been working with the code to develop improved analytic capabilities to support the Japanese nuclear industry. The merger of RELAP/SCDAPSIM/MOD4.0 with the IMPACT/SAMPSON package [41] is one of the most significant projects. However, IAE/NUPEC has also been using the code for a variety of other tasks including the development of enhanced analytical capabilities to analyse corrosive conditions in nuclear power plants using coupled system thermal hydraulics and CFD techniques along with corrosion modelling [42]. CNSNS and ININ (Instituto Nacional de Investigaciones Nucleares) in Mexico have added integrated BWR containment models to RELAP/SCDAPSIM/MOD4.0 and are now working on the possible integration of detailed integrated sub-channel and containment modules developed by IAE/NUPEC [43]. Nuclear plant analyser graphic packages including VISA [44], developed by KAERI (Korean Atomic Energy Research Institute) and RELSIM, developed by RMA (Risk Management Associates) have been linked to RELAP/SCDAPSIM/MOD3.4 and MOD4.0. Other activities by members and licensed users include the coupling of the code with 3D reactor kinetics packages.

The development of improved models and code capabilities for RELAP/SCDAPSIM/MOD4.0 by university members of SDTP has also been an important factor in the improvement of RELAP/SCDAPSIM/MOD4.0 [5]. The rewriting of the code to Fortran 90/95/2000 version of the code has made it significantly easier for university faculty and students to work with. MOD4.0 also provides a well characterized framework for university researchers and students to explore new modelling approaches since the tedious programming details associated with use of complex fluid/material properties libraries, reactor component models such as pumps and valves, input/output, and data base management for tasks such as dynamic data allocation are provided through a standard compile library maintained by ISS. The incorporation of integrated fission product transport models by Honaiser, University of Florida, USA [29] and ongoing work to add an integrated uncertainty analysis package by Perez, University of Catalunya, Spain [45,46], and CANDU-specific models for fuel channel failure by Mladin, Politechnic University, Romania [17] are good examples where university students are key contributors to the development of the code.

3.4 Training of analysts and model/code developers

RELAP/SCDAPSIM/MOD3.4 and MOD4.0 are also widely used to support SDTP-sponsored training activities. MOD3.4 is used for basic user and applications training. This includes (a) 1 to 2 week novice and advanced RELAP5 and SCDAP user training workshops and seminars, (b) longer term, 1 to 3 month, user and application training under IAEA and SDTP-sponsored training fellowships, and (c) IAEA-sponsored specialized missions on research reactor applications, severe accident management and others. For example, novice users will use the code to set up basic thermal hydraulic problems such as the flow of water in a pipe or the boildown and quenching of a representative fuel assembly and then move on to the optimisation or expansion of the input model to a representative full research reactor or NPP. More advanced students or participants in longer term training sessions will typically use the code to develop input models for their own facilities or more typically adapt existing input models to run more reliably or run a much wider variety of possible transients. MOD4.0, and to some extent MOD3.4, are also widely used by the SDTP member universities to support their graduate and faculty research programs. Section 3.3 gave some specific examples of university students that started out participating in SDTP-sponsored training activities using MOD3.4 and MOD4.0 and then going on to make significant

contributions to improvement of MOD4.0. Another good example of the use of the codes at universities is provided by Professor Manmohan Pandey and others from the Department of Mechanical Engineering of Indian Institute of Technology Guwahati (IIT-Guwahati), India in a report submitted as an in-kind contribution for their university membership [47].

IIT-Guwahati used RELAP/SCDAPSIM/MOD4.0 and the Nuclear Plant Analyser RELAP/SCDAPSIM-VISA package (ViSA-RS) for numerical simulations of a natural circulation boiling water reactor (NCBWR) and supercritical water cooled reactor (SCWR). Figure 22 shows the example of the NCBWR schematic and nodalization. Their applications included the following areas

- Parametric studies of the primary heat transport loop of NCBWR,
- Stability analysis of NCBWR,
- Stability analysis of SCWR, and
- Educational use of RELAP5 and VISA-RS.

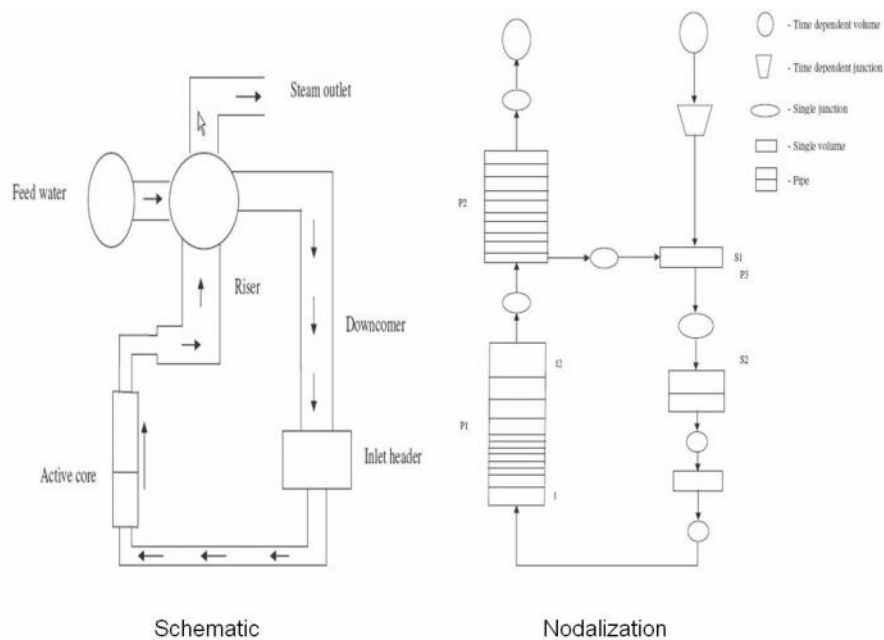


Figure 22: Example of University Applications - Natural Circulation Boiling Water Reactor Applications by IIT-Guwahti.

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