RELAP5 EXPERIENCES AND APPLICATIONS FOR HEAVY WATER REACTOR TECHNOLOGY AT NINE

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Abstract

The present paper summarizes the experiences and the applications of RELAP5 code for performing thermal-hydraulics analysis in heavy water reactor technology. In particular three analyses are discussed in detail:

- The safety analysis carried out to support the licensing process of the Atucha-2 Nuclear Power Plant in Argentina;
- The transient analysis performed to simulate the Loss-Of-Flow accident in Darlington Nuclear Power Plant;
- The simulation of the core of OPAL Research Reactor in Australia to evaluate the consequence of a possible increase of the analytical limit for the Reactor Protection System overpower trip.

Additionally to the performed analyses, the paper provides insights in relation the methodology adopted for qualifying a system thermal-hydraulic code calculation, and in particular in respect to the demonstration of the geometrical fidelity, demonstration of the steady state achievement and the qualification at on-transient level.

1. Introduction

Since early '60s, until today, the thermal-hydraulic system codes have undergone deep changes and substantial improvements. This has imposed among other things, a continuous assessment process leading to the latest released code versions. In this period, a large number of facilities, that have been classified as Integral Test Facilities (ITF) and Separate Effect Test Facilities (SETF) have been designed and put into operation all over the world. This has led to the availability of a huge amount of experimental data that was used for qualifying the codes and for identifying and characterizing new phenomena, thus requiring additional code improvements. This process used large amounts of resources till the beginning of'90s. Nowadays, the development and validation of the system thermal-hydraulic codes have reached a high degree of maturity, through the consideration of the ITF and SETF experiments and advanced numerical models, and are regularly used in the areas of design, operation, licensing and safety of Nuclear Power Plants (NPPs).

However the applications of these computational tools still require to properly address through appropriate and systematic methodology [1] the qualification issues which deal not only with code model limitations but also with nodalization inadequacies and/or lack of user qualification. In other words, although huge amounts of financial and human resources have been invested for the development and improvement of codes, the calculation results are still affected by errors, and in the sophisticated nuclear technology, design and safety of NPP, these errors must be quantified. As a consequence, since middle of ’90s an important part of the research in the area of system thermal-hydraulic code has been dedicated to the development and the qualification of the simulations errors of a NPP related transient scenarios [1], [3], [4], [5]. Several Uncertainty Methodologies (UM) have been developed starting from middle of’90s and coupled with the Best Estimate (BE) codes in order to allow for the application of Best-Estimate Uncertainty Method (BEPU) in safety analysis. More details about the proposed NINE methodology [1] are out of the goal of the present paper and only a short description of the procedures to qualify system thermal-hydraulic codes is summarized in Section 2.

A detailed description of the thermal-hydraulics activities performed at NINE in the area of Code Assessment, Quantification of the Accuracy and Safety Analysis of NPP can be found in [6]. In the present paper, the attention is focused on the experiences and applications of RELAP5 code for performing thermal-hydraulics analysis in heavy water reactor technology. In particular three analyses are discussed in detail:

- The safety analysis carried out to support the licensing process of the Atucha-2 Nuclear Power Plant in Argentina (Section 3);
- The transient analysis performed to simulate the Loss-Of-Flow accident in Darlington Nuclear Power Plant (Section 4);
- The simulation of the core of OPAL Research Reactor in Australia to evaluate the consequence of a possible increase of the analytical limit for the Reactor Protection System overpower trip (Section 5).

2. Qualification Process of System Thermal-Hydraulic Code Calculations

The qualification process of a thermal-hydraulic system code calculation has the goal to demonstrate that the code results (obtained by the application of the code with the developed nodalization) constitute a realistic approximation of the reference plant behaviour.

A brief description of the procedure developed and applied to qualify a thermal-hydraulic system code calculation is given hereafter. More in general, the procedure is part of SCCRED (Standardized Consolidated Calculated Reference and Experimental Database) [1] a multi-objectives methodology which provides a set of quality assurance measures to be applied to the collection and verification of plant data and experiments, to the verification of computer input deck and to the validation of plant models. The flow chart of the qualification process is given in Figure 1.

A nodalization representing an actual system (ITF or NPP) is qualified when:

- It has a geometrical fidelity with the involved system;
- It reproduces the measured nominal steady state condition of the system;
- It shows a satisfactory behavior in time dependent conditions.

Based on the above, two main phases of the qualification process can be distinguished:

1. The Qualification at “Steady State Level”, which includes the sub-steps:
   - 1a. Demonstration of the Geometrical Fidelity of the developed nodalization;
   - 1b. Demonstration of Steady-State Achievement;
2. The “On-Transient” Qualification.

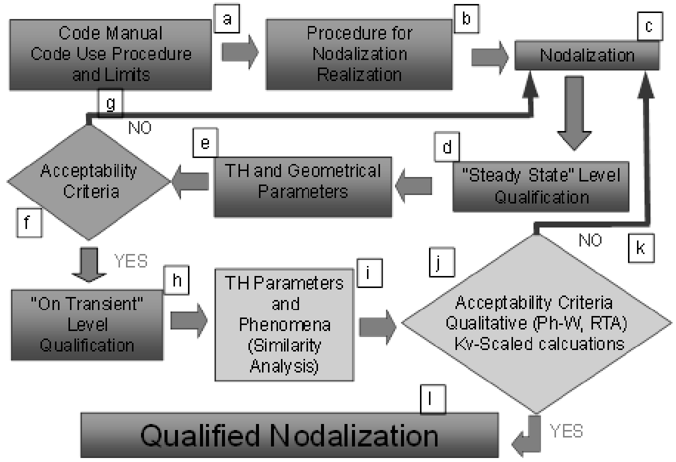


Figure 1 Flow Chart of the Nodalization Qualification Procedure.

The qualification at Steady-State level implies the following activities (steps d), e) and f)) in Figure 1):

- Verification of the geometrical fidelity between the RELAP5 nodalization and the plant data (see item 1a above), i.e. relevant geometrical parameters of the plant (e.g. volumes, heat transfer areas, elevations, etc.) are compared with the values implemented into the nodalization and the differences among them must be acceptably small (i.e. below the acceptability criteria set in Table I);
- Demonstration of the Steady-State Achievement (see item 1b above), i.e. the steady state calculated results are compared against data available from nominal stationary conditions measured in the plant system. This step includes:
  a) Analysis of the pressure drops distribution along the loops of the plant, i.e. quantification of the discrepancies between plant measurements and calculated results and verification that differences among them is acceptably small (i.e. below the acceptability criterion at item #8 in Table II);
  b) Selection of significant parameters (absolute pressures, fluid temperatures, rod cladding temperature, levels, flow rates, etc...) and comparison with plant measurements. The difference among calculated and plant data must be acceptably small (i.e. below the acceptability criteria listed in Table II);
  c) Verification of the achievement of a stationary behaviour, i.e. with reference to each selected parameter, a "null-transient steady-state" calculation must be performed until the solution is stable with an inherent drift < 1% / 100 s.

In addition to the criteria in Table I and Table II, it shall be considered that experimental measurements are typically available with an error band that must be considered when performing the comparison with the calculated results. No error is made if the calculated value is inside the experimental uncertainty bands. More in general, the error E between a measured Y_E and calculated Y_C values can be calculated by the following formulas where U_E is the measurement uncertainties:

$$\text{if} \quad Y_E - U_E \le Y_C \le Y_E + U_E \quad \rightarrow \quad E = 0$$
$$\text{if} \quad Y_C < Y_E - U_E \quad \rightarrow \quad E = \left(Y_E - U_E - Y_C\right)/Y_E$$
$$\text{if} \quad Y_C > Y_E + U_E \quad \rightarrow \quad E = \left(Y_C - Y_E - U_E\right)/Y_E$$

Table I "Steady State Qualification" Demonstration of Geometrical Fidelity: Acceptable Errors.

#	CATEGORY OF GEOMETRICAL PARAMETERS	ACCEPTABLE ERROR [1]
1	Primary circuit volume	1%
2	Secondary circuit volume	2%
3	Non-active structure heat transfer area (overall)	10%
4	Active structure heat transfer area (overall)	0.1%
5	Non-active structure heat transfer volume (overall)	14%
6	Active structure heat transfer volume (overall)	0.2%
7	Volume vs height curve (i.e. "local" volume per each circuit)	10%
8	Component relative elevation	0.01 m
9	Flow area of components like valves, pumps, orifices	1%
10	Generic flow area	10%

(1) The % error is defined as the ratio: $\frac{|\text{system hardware value} - \text{code model value}|}{|\text{system hardware value}|}$

Table II "Steady State Qualification" Steady-State Achievement: Acceptable Errors.

#	CATEGORY OF THERMAL-HYDRAULIC PARAMETERS [1]	ACCEPTABLE ERROR [2]
1	Primary Circuit Power	2%
2	Secondary Circuit Power	2%
3	Absolute pressure (PRZ, SG, ACC)	0.1%
4	Fluid temperature	0.5% [3]
5	Rod surface temperature	10 K
6	Pump velocity	1%
7	Heat losses	10%
8	Local pressure drops	10% [4]
9	Mass inventory in primary circuit	2% [5]
10	Mass inventory in secondary circuit	5% [5]
11	Flow rates (primary and secondary circuit)	2%
12	Bypass mass flow rates	10%
13	Pressurizer level (collapsed)	0.05 m
14	Secondary side or downcomer level	0.1 m [5]
15	Axial and radial power distribution	1%

(1) With reference to each parameter, the solution must be stable with an inherent drift < 1% / 100 s (acceptance criterion-SS).

(2) The % error is defined as the ratio: $\frac{|\text{reference or measured value} - \text{calculated value}|}{|\text{reference or measured value}|}$

The "dimensional error" is the numerator of the above expression.

(3) The acceptable error shall be consistent with the power error.

(4) 10% of the difference between the maximum and minimum pressure in the loop.

(5) The acceptable error shall be consistent with the other errors.

It should be noted that if the calculated data point lies within the experimental uncertainty band, there is no possibility to perform a better calculation and thus consistently with the qualification methodology and the "acceptable errors", the associated error is zero.

The values of the Acceptable Errors listed in Table I and Table II have been derived based on the engineering judgment of the developers of the methodology. Notwithstanding with the subjectivity of the derivation, the following has to be highlighted:

- The engineering judgment has been supported by the analysis of several tens of experiments and associated code predictions. During this process, it was found that when the value of one parameter (in Table I or II) was different respect to the reference more than a certain threshold, the code predictions start to differ from the experimental results in a considerable way;
- The values of the acceptable errors are not modified from one application to another; i.e. the engineering judgment has been frozen and does not change with the application.

In case the Steady-State qualification is not passed, the path "g" in Figure 1 shall be activated. Once the steady state qualification is finally satisfied, the on-transient level qualification has to be performed following the steps h), i) and j) in Figure 1. The path "k" must be activated if the on-transient qualification is not fulfilled.

Qualitative and quantitative accuracy evaluations are performed to evaluate the acceptability of the calculation on "transient level". The qualitative accuracy evaluation must be completed before any meaningful attempt to perform the quantitative evaluation. The following aspects are involved during the qualitative accuracy evaluation:

a. *Visual observation*: visual comparisons are performed between experimental and calculated relevant parameter time trends;
b. *Resulting Time Sequence of Events*: it means that the list of the calculated significant events with the corresponding calculated time of occurrences is compared with the measured events and values;
c. *Use of the phenomena specified in the CSNI validation matrix*: the relevant phenomena suitable for the code assessment, the relevance of the phenomena in the selected facility and in the selected test can be derived from the CSNI matrix. A judgment can be express taking into account the characteristics of the facility, the test peculiarities and the code results;
d. *Use of the Phenomenological Windows (PhW), Key Phenomena and Relevant Thermal-hydraulic Aspects (RTA)*: each test scenario (measured or calculated) shall be divided into phenomenological windows (i.e., time spans in which a unique relevant physical process mainly occurs and a limited set of parameters control the scenario). In each PhW, key phenomena and RTA must be identified. Key phenomena are attributed to a class of experiments. RTAs are defined as the characterization of the key phenomena for the specific transient and selected facility and are characterized by numerical values of significant parameters:
    - Single Valued Parameters, SVP (e.g., minimum level in the core);
    - Non Dimensional Parameters, NDP (e.g., Froude numbering the hot leg at the beginning of reflux condensation);
    - Time Sequence of Events, TSE (e.g., time when dryout occurs);
    - Integral Parameters, IPA (e.g., integral of break flow rate during subcooled blowdown);
    - Derivative Parameters, DPA (e.g. derivative of primary and secondary pressure).

Around 20 RTAs, characterized by more than 40 values of significant parameters, must be selected for the qualitative evaluation of the code calculation. The qualitative analysis is finally synthesized by the use of five subjective judgment marks, which are applied to the matrix of phenomena, to the visual observation of the time trends and to the list of RTAs:

- The code predicts qualitatively and quantitatively the parameter (Excellent - the calculation falls within the measured data uncertainty band).

- The code predicts qualitatively, but not quantitatively the parameter (Reasonable - the calculation shows only correct behavior and trends).
- The code does not predict the parameter, but the reason is understood and predictable (Minimal - the calculation does not lie within the measured data uncertainty band and does not have correct trends).
- The code does not predict the parameter and the reason is not understood (Unqualified - calculations do not show the correct trend and behavior, and reasons are unknown and unpredictable).
- Not applicable (-).

If the qualitative accuracy evaluation is acceptable, the accuracy of the code calculation can be quantified utilizing the Fast Fourier Transform Based Method (FFTBM) [7, 8]. This tool calculates a couple of values in the frequency domain from the comparison between each calculated and measured time trend: 1) the so-called "Average Accuracy" (AA) and 2) the "Weighted Frequency" (WF). The transformation from time to the frequency domain avoids the dependence of the error on the transient duration. Weighting factors are attributed to each time trend to make possible the summing up of the AA of each parameter and the achievement of a unique threshold for accepting a calculation (AA_{TOT}). A quantitative accuracy evaluation must be carried out following demonstration that the calculation is qualitatively acceptable. The same time trends selected for the qualitative analysis are utilized as input to the FFTBM software. Acceptability criteria have been set-up, their fulfillment ensure the achievement of a qualified calculation also from the quantitative point of view.

3. Safety Analysis for FSAR Chapter 15 of Atucha-II

The RELAP5-3D© code developed by INL has been adopted as Best Estimate (BE) SYS-TH code to perform the accident analyses related with the Chapter 15 of the FSAR of Atucha-II (CNA-II) NPP in Argentina. More details about the activity performed can be found in [9, 10]. Hereafter a short overview is provided and the results of one scenario, over the 83 analyzed, is discussed.

The complexity of a NPP and of the accident scenarios may put a challenge for a conservative analysis and may justify the choice for a Best Estimate Plus Uncertainty (BEPU) approach in the licensing process. This implies two main needs: 1) to adopt and to prove (to the regulatory authority) an adequate quality for the computational tools and 2) to account for the uncertainty. In relation with the first need, several computer codes and about two dozen nodalizations have been used, developed and, in a number of cases, interconnected among each other for the preparation of the FSAR BEPU-based Chapter 15 of CNA-II. The RELAP5-3D© code and nodalizations were the main computation tools adopted for the safety analysis and they represent the crucial conjunction point in the chain of codes and nodalizations for estimating the total effective dose equivalent starting from the derivation of the cross sections and passing through the reactor physics, the computational fluid dynamic, the fuel, the containment and the structural mechanical analyses. The need of qualification for the mentioned computational tools within the framework of the BEPU structure has been addressed together with the issue constituted by the qualification of "code-nodalization" user. In respect with the second need, the uncertainty associated with the BE calculations was quantified by the Code with the capability of Internal Assessment of Uncertainty (CIAU, [4]) and its extension CIAU-TN [5] following the procedures and criteria stated in the FSAR BEPU-based Ch. 15.0.

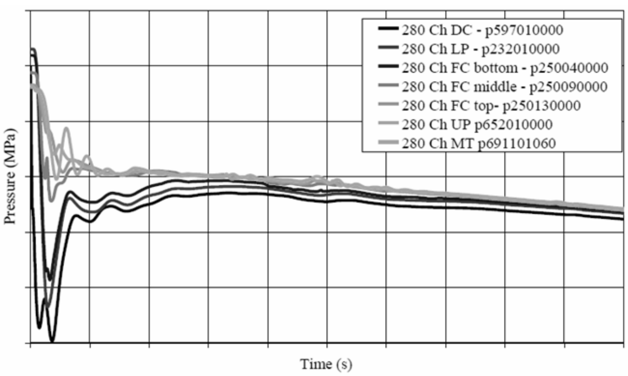
The execution of the overall analysis and the evaluation of results in relation to 83 Postulated Initiating Events (PIE) revealed the wide safety margins available for the concerned NPP that was designed in the 80's. Hereafter the attention is focused on the Double Ended Guillotine Break (DEGB) in cold leg as it provides, more than other scenarios, an idea of the activity performed which involved all the chain of codes from the cross sections generation to the estimation of the total effective dose equivalent.

The DEGB LOCA (or 2A LOCA) constitutes the 'historical' event for the design of ECCS in Water Cooled Reactors and is primarily adopted in vessel equipped NPP. It is assumed that the largest pipe connected with the Reactor Pressure Vessel can be broken: two end breaks are generated, typically the RPV side and the Main Coolant Pump (MCP) side. In the specific case of a PHWR like Atucha-II, the LBLOCA shall also be classified as a Reactivity Initiated Accident (RIA). Selected results are shown in Figure 2, in relation to the core and system safety analysis, and in Figure 3 for the fuel and radiological consequences analysis. The developed RELAP5-3D© nodalizations of CNA-II took benefit of the performed assessment activities [6].

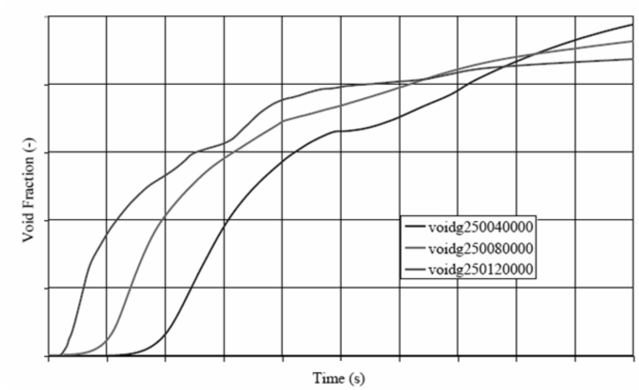
The break occurrence is at the origin of a depressurization wave that propagates throughout the primary circuit starting from both sides of the break. The amplitude of the pressure wave, RPV side, is smoothed primarily inside the lower plenum and at the connection between lower plenum and the fuel channels. On the MCP side, the pressure wave is smoothed primarily by the Steam Generator (SG) tubes. It can be seen that the moderator tank pressure is almost unaffected by the pressure wave propagation during the initial tenths of seconds of the transient and that a low amplitude pressure wave reaches the upper plenum (a). The local pressure in the Fuel Channel (FC) constitutes the main origin for the wide core void formation (b) which creates the conditions for fission power excursion that is limited and reduced by the fast actuation of the boron injection system in the moderator and, later on, by the Doppler effect. The boron injection in the moderator tank through the four lances (c) is predicted by a CFD calculation whose output is given to the RELAP5-3D© nodalization (with 280 equivalent core hydraulic core channels) to allow the prediction of the boron reactivity through the NESTLE code nodalization (simulating all the 451 fuel channel of CNA-II). The core void formation, from one side, and the boron reactivity, from the other, determine the behavior of the total core power (d) which can be characterized by the linear power profile in each individual FC (e) and by the three-dimensional power distribution (f) at any time instant. All the phenomena discussed above occur within the time interval of about 1 second.

Loss of significant quantities of reactor coolant inventory and low velocities of the two phase mixture (g) may prevent adequate cooling of the fuel bundles, which together with the fission power excursion lead to temperature excursions (d). Uncertainty evaluation by CIAU [4, 5] has been performed and the upper band has been proved to be below the acceptable limit (h). The fission power decrease associated with the boron scram creates the conditions for quenching caused by the liquid vaporization and the two-phase mixture crossing the core and coming from the upper plenum and the pressurizer. Containment pressurization (i), predicted by an appropriate RELAP5-3D© nodalization coupled with the primary system nodalization, occurs in this period achieving a peak value below the design limit.

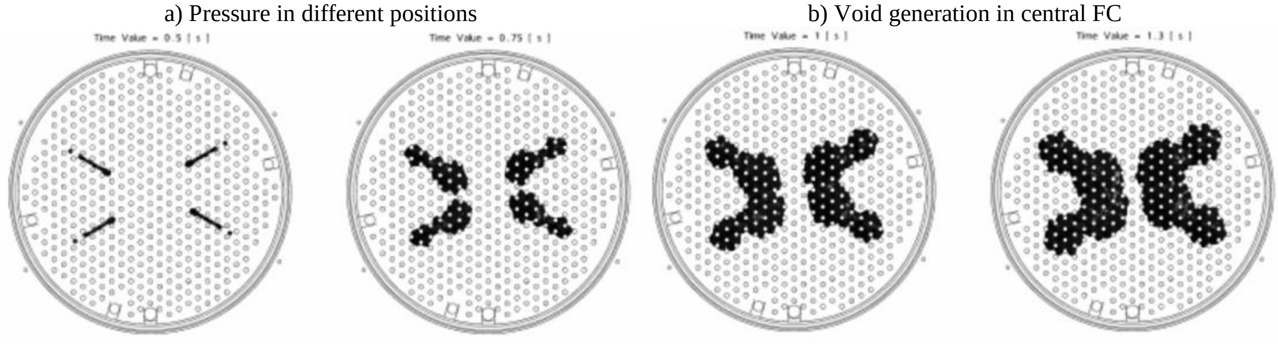
The phenomena above may occur at a time scale of 10 seconds and before the intervention of ECCS that are constituted by accumulators and by pump driven components injecting cold liquid in eight positions of the primary loop. The actuation of ECCS brings to the recovery of the coolant mass to



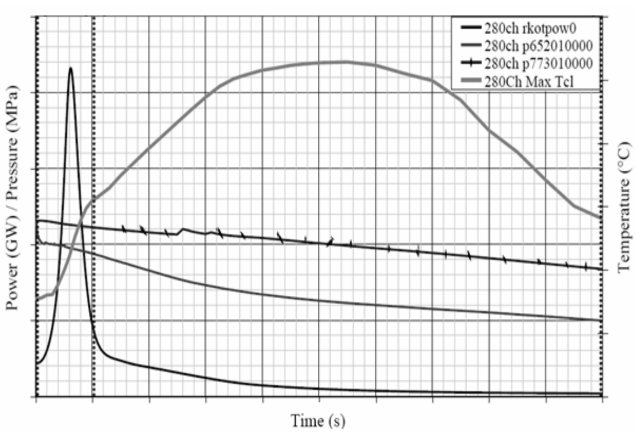
a) Pressure in different positions



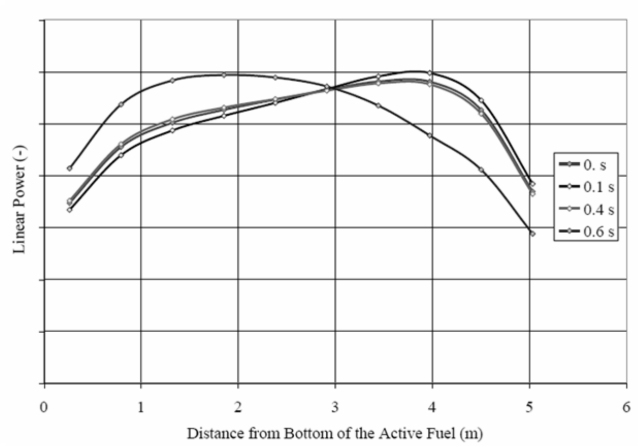
b) Void generation in central FC



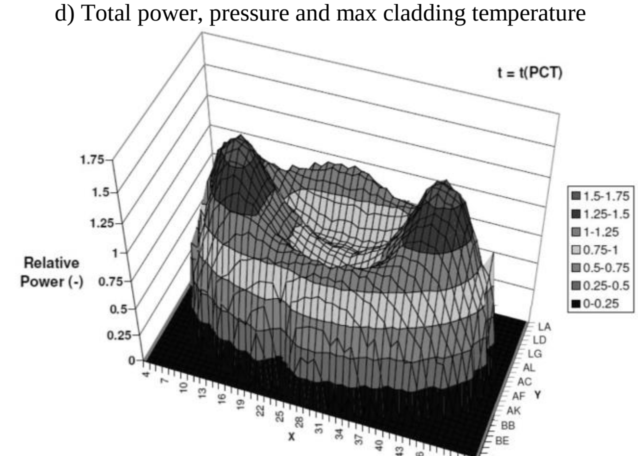
c) Contours of boron concentration on a vertical plane passing by one lance axis



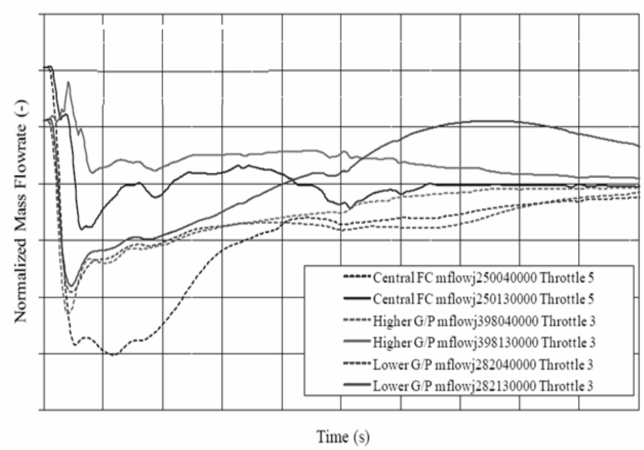
d) Total power, pressure and max cladding temperature



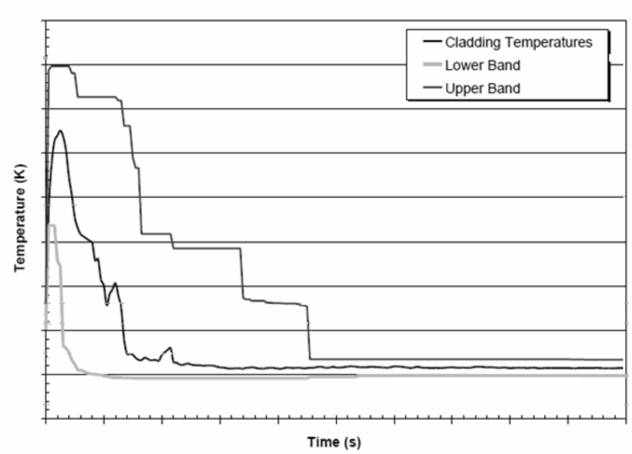
e) Non-dimensional linear power profile in central FC



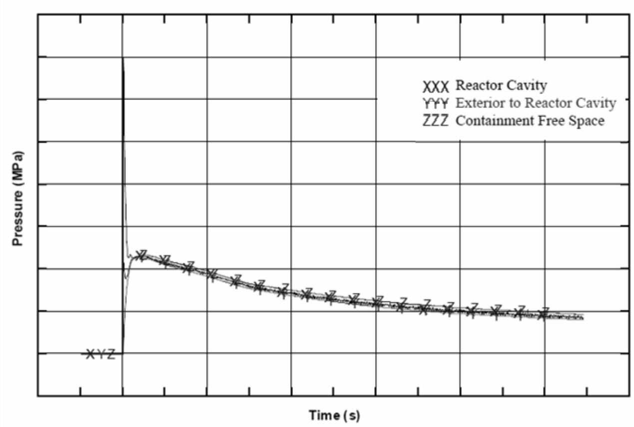
f) Three-dimensional power distribution in the core



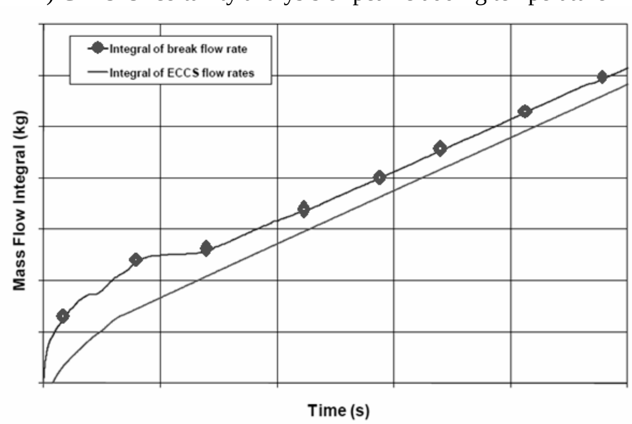
g) Mass flow-rates in different fuel channels



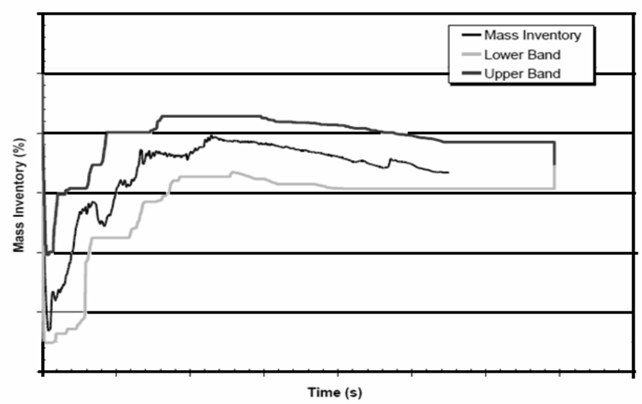
h) CIAU Uncertainty analysis of peak cladding temperature



i) Pressure in the reactor cavity



j) Integral mass of break flow rate and sump safety injection



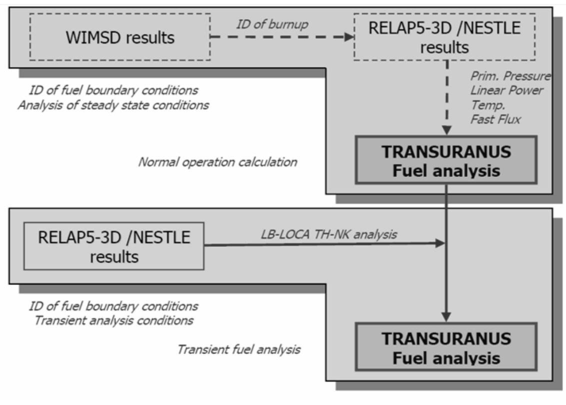
k) CIAU Uncertainty analysis of primary system mass inventory

Figure 2 2A LOCA in Atucha-II NPP (Core and System Analysis).

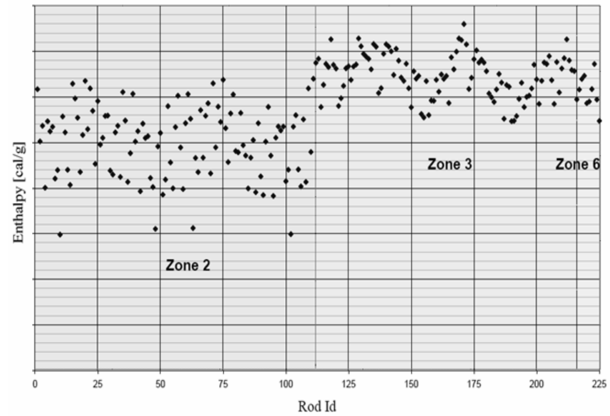
such an extent to bring the overall system in a stable circulating mode with ECCS pumps suctioning liquid from the containment sump and liquid cooling provided by proper heat exchangers (j). The CIAU uncertainty evaluation of the primary system mass inventory is provided in (k).

The Transuranus code nodalization is adopted together with the 280-ch RELAP5-3D©/NESTLE nodalization following the two-steps approach depicted in Figure 3 (a), i.e. i) the normal operation phase which is needed to achieve the initial conditions at which the LB LOCA occurs (this is the time period from the insertion of the fresh fuel rod into the core, up to the time at which the equilibrium burn-up of the reference core is reached); ii) the transient phase with TH conditions derived from the 280-ch RELAP5-3D©/NESTLE nodalization. Each step implies the execution of 451 Transuranus calculations.

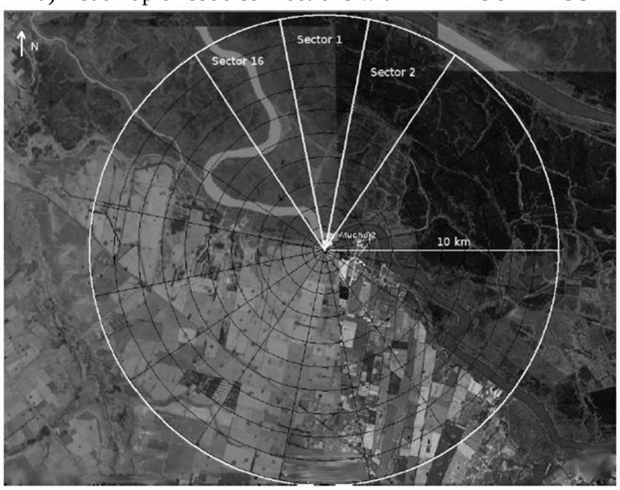
The 451 fuel assemblies of Atucha-II NPP are modeled one by one in the 3D-NK nodalization whereas the thermal hydraulic model represents 280 out of 451 channels. Linear heat rate is provided by the 3D-NK model assuming the 37 rods of each fuel assembly contribute with the same power, whereas the temperature and the rod-external pressure values are those corresponding to the hydraulic channel.



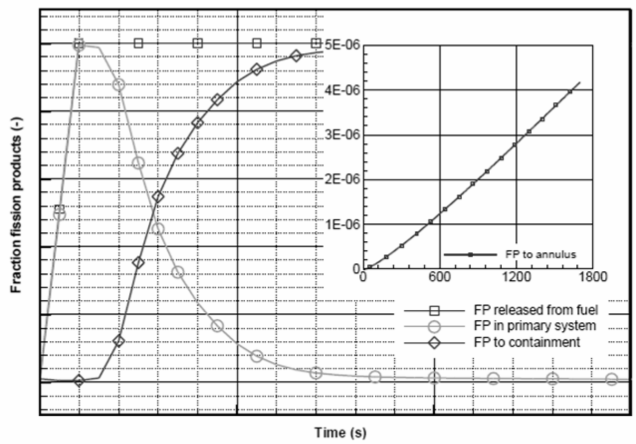
a) Roadmap of code connections with TRANSURANUS



b) Enthalpy released to the fuel evaluated by TRANSURANUS



c) MACCS2 grid applied to Atucha II surrounding area up to 10 km (EM/RCA)



d) Fraction of fission products released from fuel rods to primary system, to the containment and to the annulus

Figure 3 2A LOCA in Atucha-II NPP (Fuel and Radiological Consequences Analysis).

The LBLOCA transient starts at the time when the equilibrium burn-up conditions are reached for each rod (shuffling is also modeled in the calculations). The energy released to the fuel during the power peak is calculated as the difference between the value of the enthalpy of each fuel pin (representing the average pin of each fuel assembly) at the end of power excursion and the value at the beginning of the transient. The maximum is selected for each fuel pin as the calculated radial average energy density at any axial location. Figure 3 (b) reports some of the 451 values of the energy released to the fuel grouped according with the different zones.

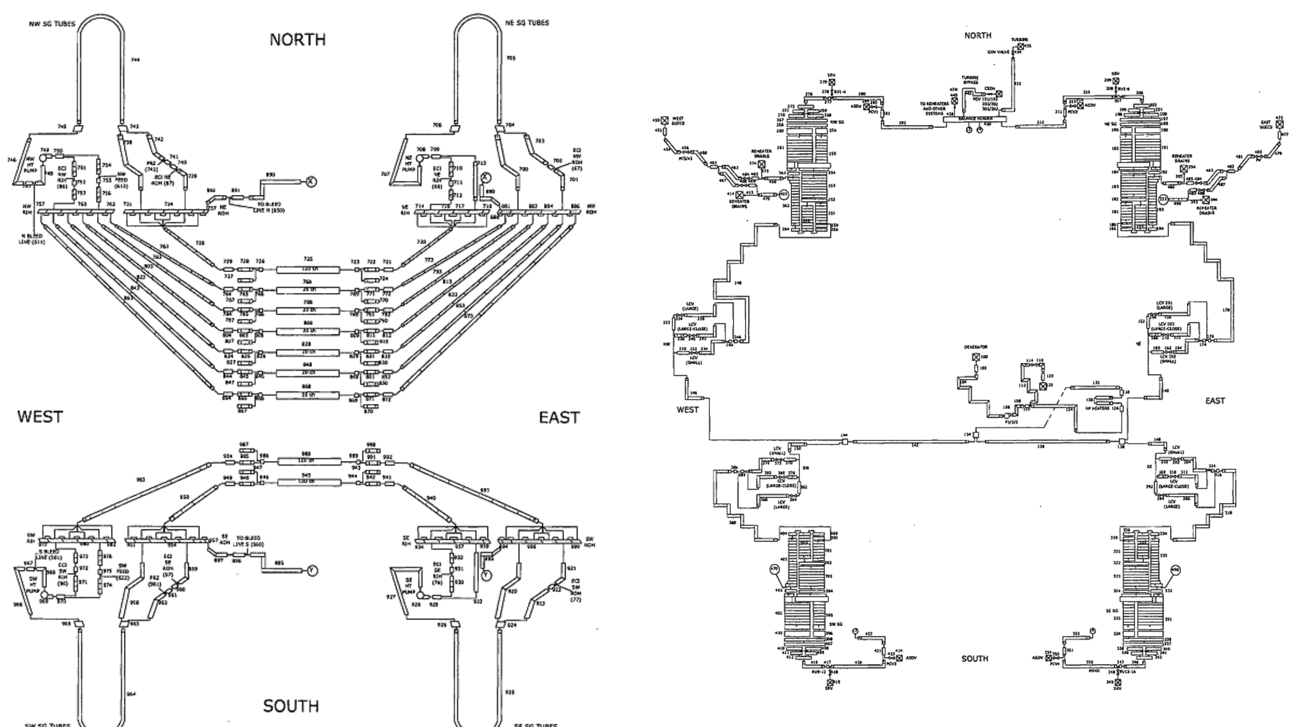
In relation to the evaluation of radiological consequences the following steps are foreseen: 1) the evaluation of the activity inventory of the reactor core which has been carried-out by MCNP-Origen, 2) the transport of the fission products released from the identified failed fuel rods in the primary system and following release to the containment (performed by Relap5-3D© fission product transport model), 3) the transport of fission products into the containment and release to the annulus building and to the environment which has been evaluated using the RELAP5-3D© Containment nodalization. Finally the Total Effective Dose Equivalent has been calculated via the MACCS2 code and model whose grid is applied to Atucha-II surrounding area up to 10 km as from Figure 3 c). A synthesis view of the radioactivity transport from the core to the environment can be drawn from the Figure 3 d) where the integral of radioactivity release is reported.

4. Analysis of LOF Event in DARLINGTON NGS

A detailed RELAP5/MOD3.3 model of a Darlington Nuclear Generating Station (DNGS) has been developed [11]. The model includes the primary Heat Transport System (HTS), the steam generators, the feedwater and the main steam supply systems, the pressure and inventory control system, and the emergency coolant injection system. The modeling of Darlington required efforts to describe the functionality of the process control systems and emergency control systems with sufficient accuracy.

The RELAP5/MOD3.3 nodalization of the Darlington HTS and secondary side system is shown in Figure 4. The HTS model (a) consists of the tube side of all four steam generators, the four HTS pumps, the four Reactor Inlet Header (RIH) and four Reactor Outlet Header (ROH), the feeders, the channels, the end-fittings and the associated connecting piping. Modeling of the channels in each core pass was done by parallel averaging of the channel flow paths in a given core pass. Therefore, each core pass is represented by one or more aggregate average channel. For three of the four core passes, one aggregate channel is used per pass representing the 120 channels comprising each pass.

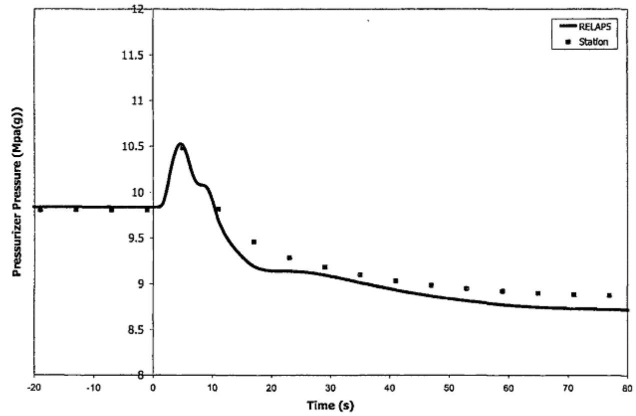
For the core pass originating from the North West (NW) RIH, the channels have been subdivided into six groups, each containing 20 channels. The core pass originating from the NW RIH therefore consists of six aggregate channels. The RELAP5/MOD3.3 model includes a detailed nodalization of the fuel channel end-fittings. Each channel in the core has two end-fittings, connecting the individual feeders from the headers to the channel. The secondary side RELAP5/MOD3.3 model (b) models the following main components: i) the steam generator feed water system from the deaerator



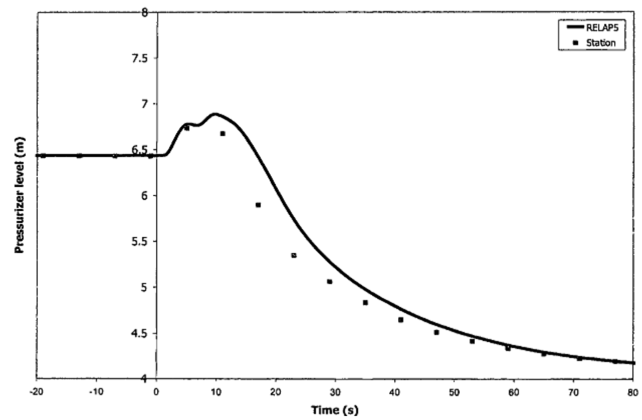
a) Heat Transport System

b) Secondary Side System

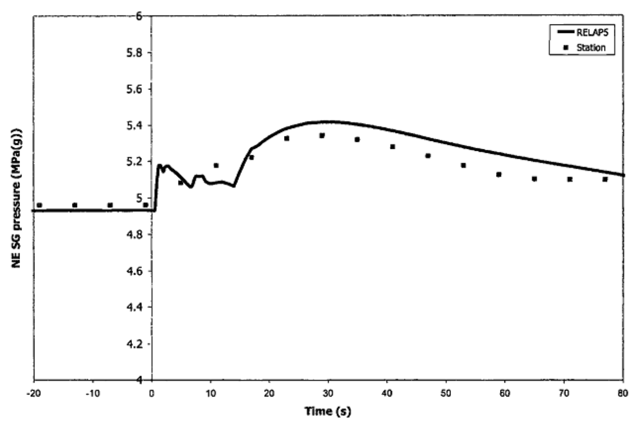
Figure 4 RELAP5/MOD3.3 Nodalization of DNGS for the LOF Event.



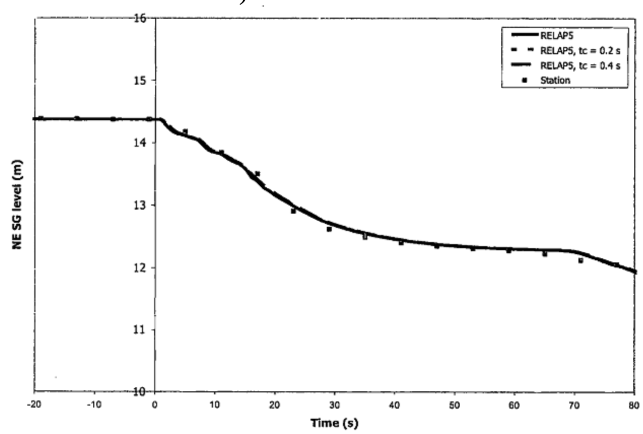
a) Pressurizer pressure



b) Pressurizer level



c) NE steam generator pressure



d) NE steam generator level

Figure 5 DNGS LOF Event, Comparison between Calculated and Plant Data.

up the shell side of the steam, ii) the shell side of the steam generators and ii) the steam supply system from the shell side of the steam generators up to the turbine. The starting point, i.e., the deaerator, and the ending point, i.e., the turbine, are both modeled as boundary conditions through time-dependent volume components. The developed Darlington nodalization includes also the Pressure and Inventory Control System and the Emergency Core Cooling System (though the latter were not called into operation in the LOF event).

The Loss-Of-Flow (LOF) event occurred at Darlington on 25 November 1993, was simulated using the developed RELAP5/MOD3.3 model. Figure 5 provides with a comparison between the station data and the calculated results both for primary and secondary side. The excellent agreement between the two sets of data demonstrates both the capabilities of the RELAP5/MOD3.3 code in dealing with CANDU design simulations and the goodness of the developed model [11].

5. Development and Qualification of RELAP5-3D Nodalization of the Core of OPAL RR

The Open Pool Australian Lightwater (OPAL) reactor is a 20 MW multipurpose Research Reactor (RR) for radioisotope production, irradiation services and neutron beam research. The OPAL RR uses Low Enriched Uranium (LEU) fuel in a compact core, cooled by light water and moderated by heavy water. It is Australia's only operating research reactor and is located at the Australian Nuclear Science and Technology Organisation (ANSTO) Lucas Heights site in Sydney. In 2006 the Australian Radiation Protection And Nuclear Safety Agency (ARPANSA) issued a licence to operate OPAL RR.

The objective of present work is to evaluate the variations of coolant, cladding and fuel temperatures between two stationary conditions corresponding at the nominal operating power of 20 MW and at power of 28 MW which corresponds to a possible increased analytical limit for the Reactor Protection System overpower trip.

Giving the project's objective, a Best Estimate (BE) evaluation model of the core of OPAL RR has been developed using the RELAP5-3D code. The relevant assumptions are connected with the boundary conditions of the developed model (limited to the core) and in particular with the core inlet flow distribution and the outlet core pressure. The developed BE RELAP5-3D nodalization constitutes a very detailed model of the core of OPAL RR: totally 875 hydraulic volumes, 184 heat structure components and 2035 axial meshes have been used to model the hydraulics and the heat structures parts of the core of OPAL RR (see Figure 6). Giving the features of the developed model, it shall be noted that so far it can be used only to predict stationary behavior inside the core and it is not yet been qualified at transient level.

Based on the results of the performed stationary analysis, it can be concluded that the maximum fuel plate centerline and surface temperatures and the coolant temperatures slightly increase passing from the power uprate from 20 MW to 28 MW. Additional activities involving the development of the whole system and the transient analysis is currently on going.

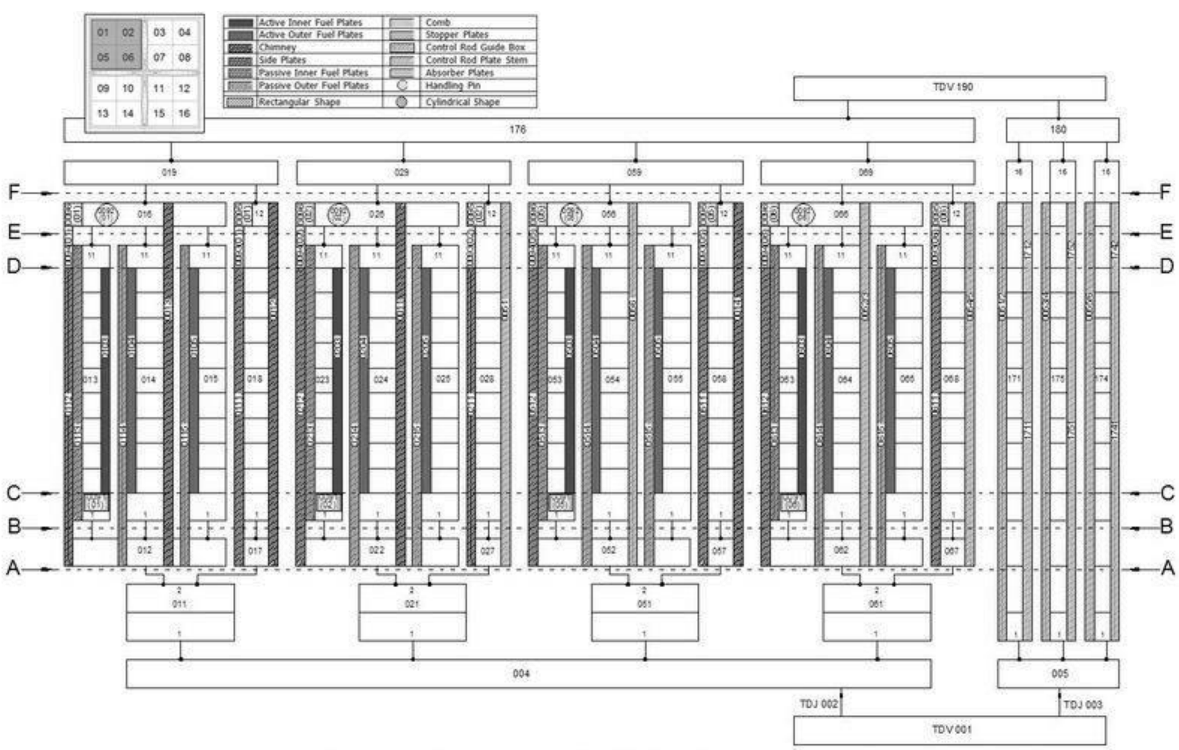


Figure 6 Axial View of the Nodalization of the First Quadrant of the Core of OPAL RR.

6. Conclusion

An exhaustive and detailed analysis of the developed RELAP5 simulations is beyond the possibility to be summarized in a single paper, rather spot results are included selecting them among the more relevant applications in heavy water reactor technology. The use of the code is made more robust thanks to the systematic application of developed methodologies and techniques regarding with the nodalization development and qualification.

The experience gained through the RELAP5 simulations of ITF/SETF experiments and NPP events for which measured data were available, has allowed the consolidation of the procedures for the nodalization development and the qualification of calculated results and their application to NPP cases with the target to perform safety analyses in a licensing framework. In this context, it is worthy to mention the application of RELAP5-3D© for the preparation of the Chapter 15 of the Final Safety Analysis Report of Atucha-II NPP (Argentina) which was licensed by the Autoridad Regulatoria Nuclear on 29 May 2014 and it is now connected to the Argentinean national grid.

7. References

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