

RELAP5 BASED SUBCHANNEL ANALYSIS FOR LICENSING OF ATUCHA2 NPP

N. Muellner¹, P. Camusso², O. Mazzantini² and F. D'Auria¹

¹GRNSPG, University of Pisa, Italy

²Nucleoeléctrica Argentina S.A., Buenos Aires, Argentina

nikolaus.muellner@univie.ac.at, ccamusso@na-sa.com.ar, mazzantini@na-sa.com.ar,
dauria@ing.unipi.it

Abstract

The departure from nucleate boiling ratio (DNBR) is one of the figures of merit which are typically presented in the chapter accident analyses of a final safety analysis report. In Argentina, like in other countries, the owner of a NPP has to demonstrate that for transients, which are expected to happen during the lifetime of a NPP, dryout or film boiling can be excluded with 95% probability for a 95% tolerance interval. A precise method to evaluate DNBR leads to a smaller tolerance interval, and helps to avoid unnecessary conservatism in the evaluation.

This paper presents DNBR evaluation for the Atucha 2 NPP (CNA2) using Relap5. CNA2 is a heavy water cooled, heavy water moderated pressurized water reactor with vertical coolant channels. It is shown that the results and uncertainties of a Relap5 subchannel analysis are comparable with those that are typically achieved by specialized subchannel codes.

Introduction

The Relap5 computer program is a flexible tool, which is usually applied for system thermal hydraulics. Nevertheless, [1] shows an application of Relap5 for subchannel analysis. This paper continues in the same direction, and presents an application of Relap5 for evaluation of the DNBR for accident analysis for the final safety analysis report (FSAR) for CNA2.

The licensing framework for CNA2 FSAR accident analysis made it necessary to evaluate the uncertainty of the calculated critical heat flux (CHF), such that that the thermal-hydraulic design of the reactor core can prevent the occurrences of dryout or film boiling with a probability of 95% for at least 95% of the fuel rods: “For operating conditions in which a critical boiling condition should be excluded, the minimum allowable margin to the critical boiling condition shall be specified in such a way that there is a 95% probability that at least 95% of the fuel rods concerned are protected from film boiling or dryout.” [2].

To evaluate the accuracy of the method code – experiment comparisons were necessary. In the 1980ies Columbia University (CU) performed a series of CHF experiments with a CNA2 fuel bundle simulator, with light and heavy water [3]. A series of code runs have been made to reproduce the experimental results, and a statistical method has been applied to evaluate the uncertainty of the predicted CHF values.

Once the uncertainty was known, a minimum value for DNBR, (DNBRMIN) has been determined such that any result above the value DNBRMIN guarantees that the probabilities mentioned above and required in [2] are fulfilled.

The paper presents a short description of the design of the NPP CNA2, a short description of the CU test rig and experiments and details on the modeling approach with Relap5 code. As main result of the experiment-code comparison the statistical limit DNBRMIN is characterized, and one example application for CNA2 accident analyses is presented. The paper finishes with conclusions.

1. Background

The section “Background” gives a description of CNA2 NPP, and a description of the CU fuel channel test rig and the experimental results.

1.1 Atucha 2 NPP

A description of CNA2 can be found in [4]. Atucha 2 is a two loop Siemens design pressurized water reactor, heavy water cooled and heavy water moderated. The channel-type design allows refueling the reactor during operation. The vertical fuel channels make the design unique in the world. CNA2 is utilizing heavy water as moderator and coolant; natural uranium may be used for the core. The moderator must be kept at a lower temperature than the primary coolant to be more effective. Like in CANDU reactors the coolant acts as neutron poison, leading to a positive void coefficient. The moderator system, in addition to its main purpose, serves as safety injection and residual heat removal system. Atucha 2 has two independent shut down systems - one system using absorber rods, which are inserted with an inclination from top side of the reactor pressure vessel (RPV) into the core, and a fast boron injection system. Main parameters of CNA2 are presented in Table 1, a schematic of the reactor coolant system (RCS) in Figure 1.

Table 1 Main parameters of Atucha 2 NPP [13].

Parameter	Unit	Value
PRZ pressure	MPa	11.5
SG pressure (Top)	MPa	5.6
Thermal power	MW	2160
Net electrical power	MW	692
Loops	#	2
Loop flow rate	kg/s	5350
Core inlet temperature	deg C	280
Core outlet temperature	deg C	315
Moderator tank inlet temperature	deg C	140
Moderator tank outlet temperature	deg C	200
Sub cooling margin UP	deg C	6
Core channels (one FA per channel)	#	451
Fuel rods per fuel assembly	#	37

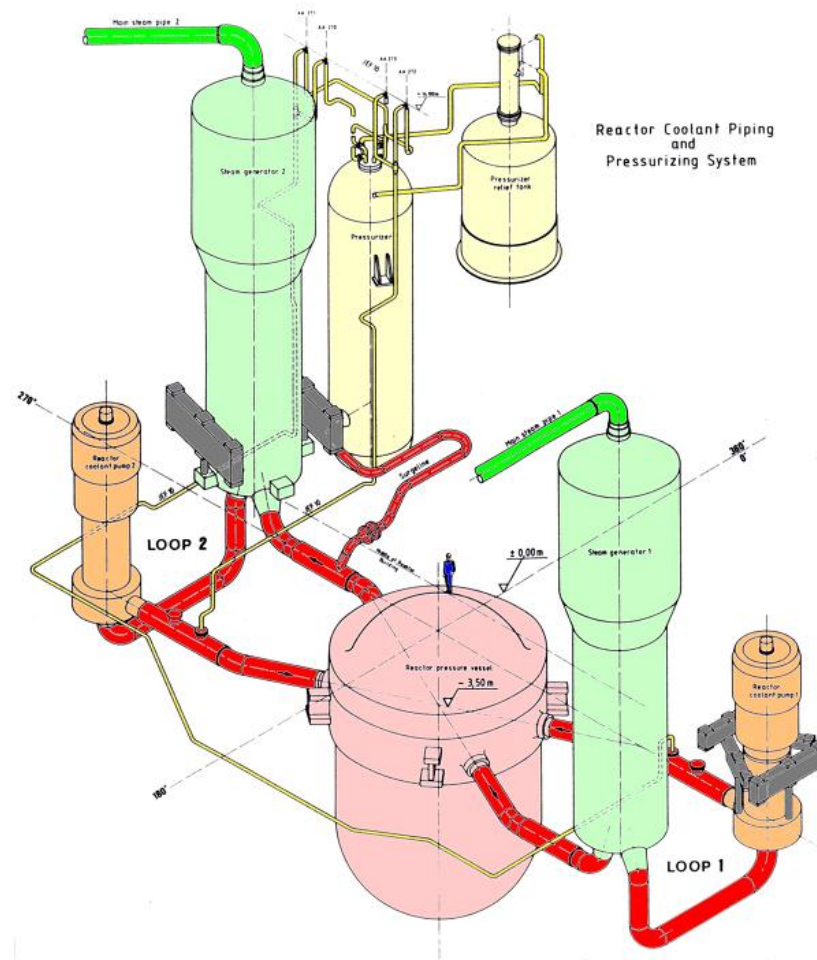


Figure 1 Reactor Coolant System of Atucha 2 NPP [13].

1.2 Atucha 2 fuel channel and Columbia University test rig

The test facility is a reproduction of a single CNA2 fuel channel. The grid spacers used in the facility are original grid spacers of CNA2. The spacing between the grid spacers is chosen as it is in CNA2. The only difference is that the CNA2 fuel channel is 5.3 m in length, while only the upper 3 m have been reproduced for the experiment. The test section was an electrically heated 37 rod bundle with triangular pitch (CNA2 FA). Six original CNA2 grid spacers were located at an axial spacing of 455 mm (beginning of a grid spacer to beginning of the next grid spacer). Total heated length was 3000 mm. Axial heat flux distribution was uniform, while radial three power zones simulate the power distribution of a representative fuel assembly (CNA2 fresh FA, see Figure 2 right).

Due to the specialties of the grid spacers (one spring shoe was holding the fuel assembly simulation in place, shifting it towards the opposite wall, see Figure 2 left), the axis of the assembly and the fuel channel were non concentric. The difference between the axes is 0.3 mm.

Test facility data

Test section circular housing inner diameter	108.2 ± 0.2 mm
Outer diameter of one heater rod	12.9 ± 0.05 mm
Number of heater rods	37
Resulting cross section flow area	4359 mm ²

Grid Spacers:

Number of grid spacers (in heated length section)	6
Height of one grid spacer	32mm
Cross section flow area with grid spacer	2980mm ²
Distance between grid spacers (center to center)	455 mm (*)

Eccentricity due to spring shoe of grid spacers	0.3mm
---	-------

Table 2 Details on CNA2 fuel assembly.

	Outer circle	Middle circle	Inner circle and central rod
Diameter of circle of rods	89.6 mm	60.2 mm	30.9 mm
Number of rods	18	12	7
Relative power	1.11	0.94	0.84

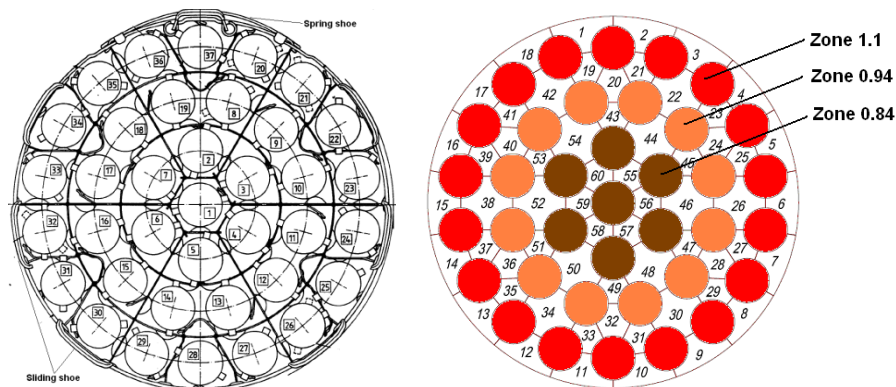


Figure 2 CNA2 grid spacer (left) and cross section of CNA2 fuel assembly with power zones (right).

Execution of experiments: One experiment, to achieve one CHF-point, was carried out by keeping pressure, flow rate and inlet temperature constant, while increasing the power. In an axially uniform heated channel occurrence of CHF can be expected at the channel outlet. Each heater rod in the channel outlet has been instrumented with thermocouples. When a significant rise in temperature has been detected, the power together with pressure, mass flux (flow), inlet and outlet temperatures were recorded and the experiment terminated. In total, 146 (124 with light water, 22 with heavy water) CHF experiments have been conducted, covering a broad range of pressures, inlet temperatures and mass fluxes. In addition, for experiments with low Reynolds numbers, the channel has been instrumented with a “thermocouple-cage”, measuring the fluid temperature at sixty positions at the channel outlet. For tests with mass fluxes above 5000 kg/(m² s) the thermocouple cage has been replaced with an additional grid spacer. Table 3 shows an overview on the experiments.

Table 3 Overview on available experimental data.

	TS17 H2O low Re	TS18 D2O low Re	TS25 H2O high Re	TS27 H2O high Re	Total
Pressure drop	18	5	12	6	41
Mixing	25	-	-	-	25
CHF	72	22	19	33	146
Total	115	27	31	39	212

2. Relap5 model of the facility, and results

The US-NRC version of Relap5 mod 3.3 (patch 3) has been used as basis for the analysis. The Relap5 option, which selects CHF evaluation based on Groeneveld lookup tables, has been selected. However, two modifications to the Relap5 sources have been made. First, the look-up tables which are implemented are based on [5]. The tables have been updated according to [6]. Second, Relap5 mod 3.3 patch3 does not apply fluid to fluid scaling, when heavy water is selected as working fluid – however, the lookup tables are based on light water. If fluid to fluid scaling is applied, the values for CHF are lower – which means, that neglecting fluid to fluid scaling moves the results in a not conservative direction.

In [5] it is suggested to apply a scaling law if the tables are used for non-aqueous fluids. Scaling works in three steps: first, pressure, quality and mass flux of the non-aqueous (NA) fluid are converted to their equivalent values for light water. For pressure, the ratio of liquid and steam phase of the NA-fluid are computed $(\rho_f/\rho_g)_{NA}$; the pressure, surface tension and viscosity corresponding to $(\rho_f/\rho_g)_{H_2O}$ is then looked up. For the thermodynamic quality $X_{NA} = X_{H_2O}$ is applied. The last quantity, the mass flux, can be scaled in different ways – [7] provides an overview on commonly adopted methods. [5] proposed Katto (1979), [8] to scale the mass flux. For the present work the Ahmad 1973 fluid to fluid scaling [9] has been chosen:

$$G \cdot D \cdot \mu_{(f/g)}^{0.25} \cdot \rho_{(f/g)}^{0.75} = G_{NA} \cdot D_{NA} \cdot \mu_{(f/g)_{NA}}^{0.25} \cdot \rho_{(f/g)_{NA}}^{0.75}$$

Where G indicates the mass flux, D the diameter of the tubes, $\mu_{(f/g)}$ liquid and vapor dynamic viscosity, $\rho_{(f/g)}$ the liquid and vapor density and σ the surface tension.

After that the CHF_{H_2O} value is looked up using the generated data table. As last step, CHF_{NA} is computed from CHF_{H_2O} :

$$CHF_{NA} = CHF_{H_2O} \cdot \left(\frac{\mu_{(f/g)_{NA}}}{\mu_{(f/g)}} \right)^{0.25} \cdot \left(\frac{\rho_{(f/g)}}{\rho_{(f/g)_{NA}}} \right)^{0.75}$$

The nodalization for the CU facility follows the subdivision in 60 subchannels (coolant centered, Figure 2 right); each subchannel is simulated with 13 axial nodes (from bottom to top: length of the first node 0.27 m, then all other nodes 0.2275 m). Each heater rod is simulated with four to six heat structures, depending on how many adjacent subchannels there are for the rod (no heat transfer is simulated between heat structures belonging to the same heater rod). The subchannels are interconnected with multiple junctions at each level. Counter current flow limitation and choked flow model are disabled for the analysis.

Fixed form loss coefficients have been entered to simulate the grid spacers, and tuned to fit experimental results.

The area of the subchannels has been calculated considering the shift of 0.3 mm of the fuel assembly inside the channel. As a result, the areas of the subchannels 1-18 are not equal; Subchannel 1 and 2 have the largest area, 10 and 11 the smallest among the 18 subchannels.

The composition of the heater rods has been entered like nuclear fuel (UO_2 , gap, zircalloy), since information on the materials of the heater rods was unavailable. However, due to the very smooth increase of power the material properties do not significantly influence the results, as could be shown; sensitivity calculations assuming other materials have been made, to be able to quantify the distortion. Figure 3 shows the nodalization scheme. Full details are provided in [10].

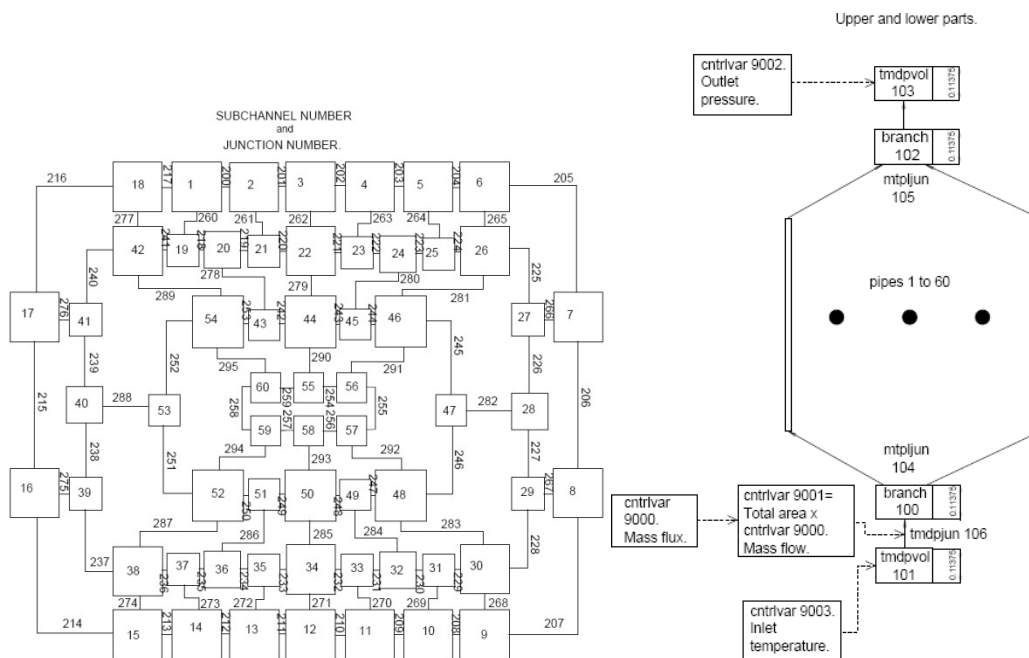


Figure 3 Nodalization scheme 60 channel nodalization, TH subchannels [10]

Results: To compare the code performance against a single experiment, the boundary conditions from the experiment (pressure at the outlet, fluid temperature at the inlet and mass flux) has been imposed on the RELAP5 nodalization. The power has been set to 75% of the reported CHF-power. The power has then been increased. The rate of increase was set in a way that the process behaves quasi-static, and that important phenomena and parameters, which are important during change in power, like heat capacity of involved materials, can be neglected. Once the actual calculated heat flux reached the calculated CHF in one of the heated structures, the calculation was stopped. The detected CHF value was recorded, and compared to the heat flux of the experiment in the rods of the maximum power zone. So for all tests, an experimental CHF_e and calculated CHF_c have been evaluated and a ratio between these two values has been calculated.

□□□□□□=□□□□□□□□

To derive DNBRMIN, a one-sided 95% confidence interval is used, taking into account that the sample is limited. The formula for a one sided tolerance interval can be found e.g. in [11]:

$$\hat{\sigma} = \sqrt{\frac{1}{n} \left(\sum_{i=1}^n \left(\frac{CHF_{predicted,i}}{CHF_{experimental,i}} - 1 \right)^2 \right)}$$

with

$$\hat{\sigma} = \sqrt{\frac{1}{n} \left(\sum_{i=1}^n \left(\frac{CHF_{predicted,i}}{CHF_{experimental,i}} - 1 \right)^2 \right)}$$

and

$$\hat{\sigma} = \sqrt{\frac{1}{n} \left(\sum_{i=1}^n \left(\frac{CHF_{predicted,i}}{CHF_{experimental,i}} - 1 \right)^2 \right)}$$

where p is the percentage of the population, γ is the desired confidence level, and z is the corresponding value of the z distribution. In our case, $p=\gamma = 0.95$, and the corresponding z -value is 1.645. The DNBRMIN is then evaluated to

$$\frac{CHF_{predicted}}{CHF_{experimental}} = \frac{CHF_{predicted}}{CHF_{experimental}} + \frac{CHF_{predicted}}{CHF_{experimental}}$$

The light water data (124 data points, Figure 4 left) has been used to evaluate the limit DNBRMIN, the heavy water data (22 data points, Figure 4 right) to validate this limit. The limit has been evaluated to 1.26. For light water runs ($n=124$) k was calculated to be 1.892, while m and σ have been evaluated to 1.033 and 0.119 respectively. The fewer ($n=22$) heavy water runs have a higher k of 2.333, while m and σ are 1.017 and 0.088 respectively.

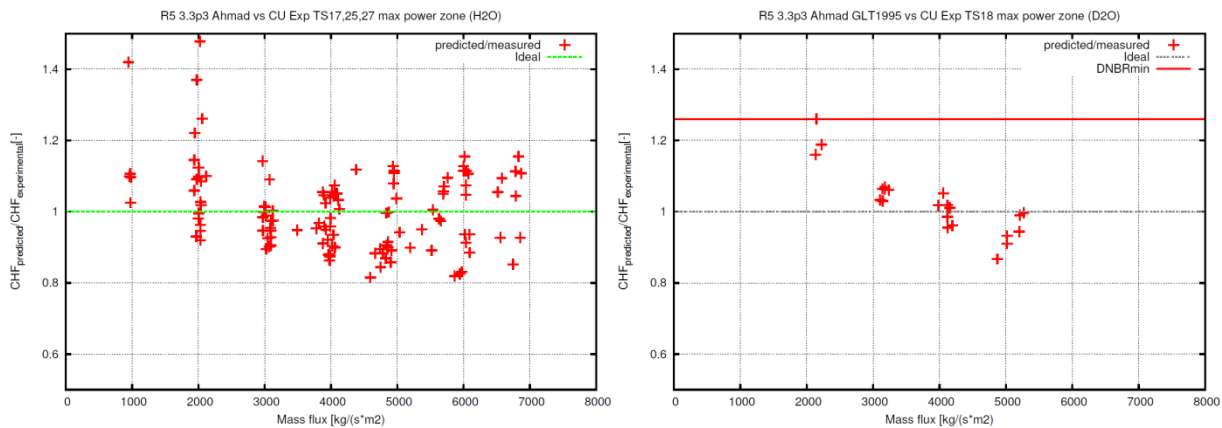


Figure 4 Results – Relap5 predicted CHF over experimental CHF, light water (left) and heavy water (right)

3. Relap5 model of the CNA2 fuel channel, and results

The nodalisation for the CNA2 fuel channel: A nodalization for the fuel channel, following the nodalization strategy of the 60-channel Columbia University, has been developed. However, a couple of changes were necessary to model adequately CNA2 fuel channel conditions.

For transient analysis the most limiting axial power profile has been imposed. The validation has been performed using a uniform axial distribution function (ADF). Recent experimental studies [12] have shown that a non uniform ADF may introduce a difference in CHF at same local conditions. To account for possible uncertainties, the maximum value, which has been found experimentally, has been subtracted from the DNBR during transient calculations.

The length of the channel has been extended to 5.3 m, but the node length has been kept the same. The k-loss (energy loss) coefficients have been slightly adjusted to match the design value for the reactor core pressure drop.

DNBR is a relevant acceptance criterion for anticipated operational occurrences (AOO). One of the most challenging AOO regarding DNBR is the main coolant pump trip, as loss of flow in Atucha2 leads to an increase in power. A fast response of the reactor protection system is required to ensure suitable margins during this transient. In Figure 6 the DNBR development after the trip of one (out of two) main coolant pumps under conservative boundary conditions (assumption of single failure and repair case) is presented. Pressure, mass flow and temperature for the maximum power channel have been derived from a system analysis, and have been imposed on the Relap5 subchannel nodalization as boundary conditions. Two effects are opposing each other: the reduction in core flow rate (which decreases DNBR) and the reduction in power following a reactor trip (which increases DNBR). As one can see, DNBR is reducing in the first phase, but increasing on long term. The minimum DNBR during the transient stays well above the limit DNBRMIN. Occurrence of dryout or film boiling can be excluded with the required probabilities. Units have been removed from the Figure, since the data are property of Na-Sa.

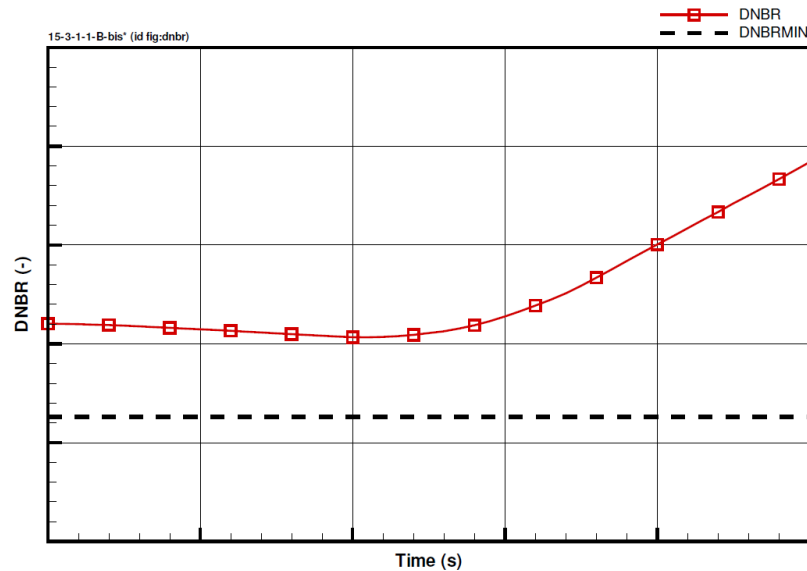


Figure 6 Results – Relap5 DNBR following a single main coolant pump trip

Effects, whose uncertainties are not quantified, have been treated by a bounding approach. Regarding radial power profile, an actual case of the possible power profiles at the NPP has been selected for the experiment (three power zones, see Fig. 2. During the experiments quasi steady state conditions prevail, which are known to lead to a lower DNBR than rapidly changing transient condition (at the same local conditions). The axial power profile, as mentioned in the first paragraph of this section, is flat in the experiment, while for the NPP the most challenging possible axial power profile has been considered. It has been shown experimentally that this may introduce a non conservative distortion (difference of CHF at same local conditions in a non-conservative direction). Therefore, a constant value, the maximum known distortion, has been added to the uncertainty of the results. A further difference between experiment and NPP coolant channel is the different heated length. However, with 3m length the facility accommodates five grid spacers, which are of CNA2 design. The length between two grid spacers corresponds to the upper part of the CNA2 fuel channel (where the distance between the grid spacers is largest). Therefore the different length of facility and plant is not taken as additional source of uncertainty (considering also that the power profile of the facility is flat, that a longer test section with a flat profile would not have any effect on the results, and that the distortion for the flat profile has already been accounted for). The node length of the Relap5 subchannel model is larger than the node length typically used in subchannel codes and models. But since the Relap5 model adopts the same length for both, experimental facility and NPP fuel channel, the calculated uncertainty corresponds to the node length. A lower uncertainty could be achieved using a larger number of axial cells – however, the margin to DNB during transient analysis was demonstrated to be sufficient, even with the, in comparison to typical subchannel approaches, larger node length.

4. Conclusion

The paper presented an application of a Relap5 based code for subchannel analysis for licensing of CNA2. Evaluation of CHF is one important step for subchannel applications. Relap5 mod 3.3 patch 3 has been used as base, the Groeneveld lookup table (GLT) based method has been chosen. Two modifications have been made on the level of the source code: first, a newer version of the GLT has been implemented. Second, Ahmad fluid to fluid scaling laws have been applied.

Since the source of a qualified code has been modified, on the one hand, and because the licensing framework required that the uncertainties of the DNBR calculations should be quantified, it was necessary to validate the method against experimental data. Experimental data of 124 light water CHF experiments have been used to estimate the uncertainty of the method. The number that has been found has been put to practice with 22 heavy water experiments. Uncertainties other than the statistical uncertainties, which arise from moving from the test facility to the fuel channel of the plant, have been conservatively added to the overall uncertainty (by reducing the calculated value of DNBR by a constant).

The application to CNA2 showed that the margins at the plant are such that the method is precise enough for the safety demonstration during transients.

5. References

- [1] Kovtonyuk, A.. et al. "RELAP5-3D Analysis of OECD-NEA/NRC BFBT Benchmark", Proceedings of ICAPP '08 Anaheim, CA USA, June 8-12, 2008
- [2] Safety Standards of the Nuclear Safety Standards Commission (KTA) KTA 3101.1 "Design of Reactor Cores of Pressurized Water and Boiling Water Reactors, Part 1: Principles of Thermohydraulic Design", February 1980
- [3] Lencina, L. et al. "Recopilación de datos experimentales obtenidos de los ensayos termohidráulicos realizados con el Elemento Combustible de Atucha II en los Circuitos Experimentales de la CNEA (CEBP y CEAP) y en la Universidad de Columbia.", Na-Sa Work Report 2008
- [4] Muellner, N. et al. "Optimizing The Initial Pressure Of Accumulators For The Atucha2 Npp Using An Optimization Method", Proceedings of ICONE17, Brussels, Belgium, 2009
- [5] Groeneveld, D.C. et al. "1986 AECL-UO Critical Heat Flux Look-up Table" Heat Transfer Engineering Vol. 7, nos 1-2, 1986
- [6] Groeneveld, D.C. et al. "The 1995 look-up table for critical heat flux in tubes", Nuclear Engineering and Design, Vol 163, 1996
- [7] Yang, S.K. "Assessment Of Fluid-To-Fluid Modelling Of Critical Heat Flux In Horizontal 37-Element Bundle Flows", Proceeding of the 27th Annual CNS Conference & 30th CNS/CNA Student Conference, June 11-14 2006, Toronto, Canada

- [8] Katto, Y. "A Generalized Correlation of Critical Heat Flux for the Forced Convective Boiling in Vertical Uniformly Heated Round Tubes", Int. J. Heat Mass Transfer, Vol. 21, pp. 1527-1542, 1978
- [9] Ahmad, S.Y. "Fluid to fluid modeling of Critical Heat Flux: a Compensated Distortion model", International Journal of Heat Transfer, Vol. 16, pp. 641-662, 1973
- [10] Camusso, C.P. "Subchannel nodalization of the Columbia test.", Na-Sa Internal Work Report, 2009
- [11] NIST/SEMATECH e-Handbook of Statistical Methods,
<http://www.itl.nist.gov/div898/handbook/>, 2009
- [12] Yang, J. et. al. "An experimental and analytical study of the effect of axial power profile on CHF", Volume 236, Issue 13, Pages 1384-1395 Nuclear Engineering and Design 2006
- [13] Mazzantini, O. et. al. "Description of Central Nuclear Atucha 2, complete", Na-Sa Internal Work Report, 2008