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TRACE CODE VALIDATION FOR BWR SPRAY COOLING INJECTION BASED ON GÖTA FACILITY EXPERIMENTS

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Abstract

Best estimate codes have been used in the past thirty years for the design, licensing and safety of NPP. Nevertheless, large efforts are necessary for the qualification and the assessment of such codes. The aim of this work is to study the main phenomena involved in the emergency spray cooling injection in a Swedish designed BWR. For this purpose, data from the Swedish separate effect test facility GÖTA have been simulated using TRACE version 5.0 Patch 2. Furthermore, uncertainty calculations have been performed with the propagation of input errors method and the identification of the input parameters that mostly influence the peak cladding temperature has been performed.

1. Introduction

The development of Thermal-Hydraulic System Codes (THSC) begun between the sixties and the seventies when conservative approaches have been used to demonstrate the safety margins necessary for the licensing and operation of a Nuclear Power Plant (NPP). During the years, the improvements in computer technology and in computational methods lead to a new generation of THSC that can provide more realistic results without the need for conservative assumptions.

Nevertheless, the results of these advanced system codes are still affected by several kinds of uncertainties derived from many sources [1]. Therefore, it is necessary to perform a series of procedures for the code validation, using data deriving from Integral Test Facility (ITF) or Separate Effect Tests Facility (SETF). Moreover, many methods for the evaluation of uncertainties have been developed in order to evaluate the reliability of any thermal-hydraulic code calculations, taking into account the possible sources of errors [1].

The aim of this work consists of validation of the U.S. NRC code TRACE for the Spray Cooling Injection in a BWR reactor. The data from the Swedish GÖTA separate effect test facility are used and compared with the results derived from the modelling of the facility with the code.

Furthermore, the Propagation of Input Errors (PIE) method is used to perform an uncertainty analysis on the code, and to identify which input parameters have the strongest influence on the figure of merit (peak cladding temperature).

2. Description of the facility

The GÖTA test facility was located in the Studsvik Thermal Engineering Laboratory during the 70's. From 1975 to 1977 this facility was used to investigate the thermal-hydraulics of BWR fuel bundles subjected to spray cooling from the top and reflooding from the bottom.

The layout of the test loop consists mainly of two pressure vessels: a test vessel and a pressurizer, which also form a water reservoir for the spray system in the loop. The pressure regulation system was not used for experiments at atmospheric pressure. In these tests, steam produced was vented directly to the atmosphere [2].

A main circulation pump drove the water from the pressurizer vessel to a water cooler and then to the spray injection systems in the test vessel. One of the two spray systems injected water through a nozzle into the fuel bundle, while a spray ring delivered water equally to all four sides of the square bypass channel.

The water collected in the lower plenum below the bundle and the water outside the bundle canister was drained back directly to the pressurizer by two separate lines that operated automatically and independently.

In this paper only the test bundle has been modelled and simulated with the TRACE code, no attempt has been made to model the pressurizer and the other components of the loop.

2.1 The Test bundle

The test bundle is located in the test vessel. It consists of 64 rods assembly positioned by spacers in a bundle channel (inner shroud or canister). The lower, non-heated part of the rods is extended out of the channel to a bottom flange where they are fixed. Forty rods have 5 thermocouples each, mounted on the inner side of the rod cladding.

The bundle channel is mounted "leak proof" in a flange in the pressure vessel. The flange separates the upper and the lower plenum; the water falling outside to the bundle is collected in the upper plenum, while the water moving down through the channel arrives in the lower plenum of the vessel.

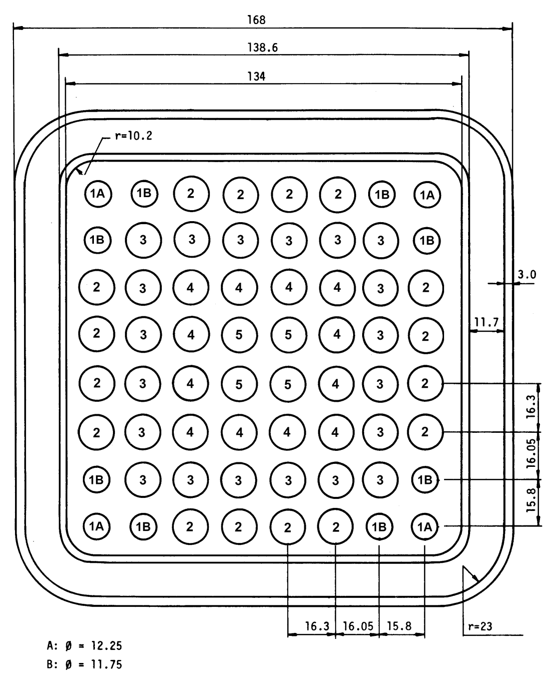
Outside the bundle, there is an outer channel that simulates the bypass region in the reactor. The outer channel is also mounted on the flange that divides the lower and the upper core, but there are openings that allow the passage of water from the gap between the inner and the outer shroud to the remaining part of the upper plenum.

The bypass channel is cooled by a spray ring on the top of the vessel, and it is maintained in its position by 8 space supports mounted 100 mm below the bundle mid-plane. The spray nozzle for the injection in the bundle is mounted inside the inner shroud and it ends 170 mm above the top grid. The inner channel is made of Zr-4 and has 40 thermocouples mounted on its four sides, placed in pockets drilled from the outside.

The heaters, designed by Watlows Mfg Co, used an Inconel-600 clad and an inner coil (80 % Ni and 20 % Cr) embedded in a boron nitride matrix. Five thermocouples were drawn inside the cladding.

The experiments were run with time-dependent power in the bundle. The power decay curves were chosen to correspond to the power decay in BWR during the spray cooling injection following a postulated LB-LOCA. For a guillotine break on a recirculation line, this phase is calculated to occur approximately 15 seconds after the LOCA. The portion of the power generated by fission products has been calculated according to ANS 1971 decay heat standard plus 20%.

The power inside the channel is delivered in such a way that five different groups of rods can be distinguish. Each group presents a different power peaking factor with respect to the average value and a fixed axial power shape. The internal rod-to-rod power distribution is reported in Figure 1 (highlighting the various groups) together with the axial power profile and the location of the level where the experimental data are available.



Rod Group	Peak Factor
1A	1.2
1B	1.1
2	1.1
3	0.92
4	0.90
5	0.90

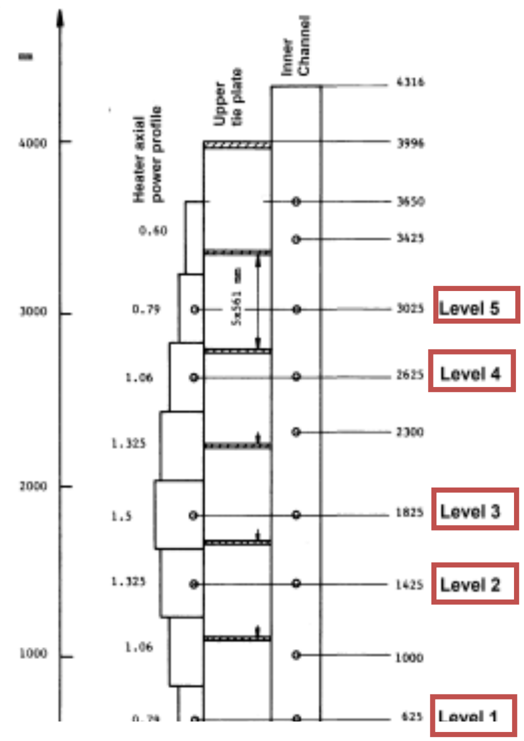


Figure 1 Internal rod to rod power distribution (right) ad axial power shape (left) [2].

2.2 The test number 78

In this paper, only the results of Test number 78 will be consider, and compared with the TRACE code calculations. The test was started heating the rods with a pre-defined power curve until the prescribed initial peak cladding temperature (PCT) at the nominal power was reached. The physical parameters at the beginning of the experiment are listed in the Table 1.

Table 1 Test number 78 initial parameters [2].

Parameter	Nominal	Actual
Initial Bundle Power [kW]	300	303
Spray flow rate in the bundle [kg/s]	0.12	0.12
Fraction of the spray flow rate in the bypass [kg/s]	20%	20%
Pressure [MPa]	0.1	0.11
Initial Peak Cladding Temperature [°C]	700	736
Spray Water Temperature [°C]	60	60
Steam Venting	Top	Top

The spray system was then activated, and the power was decreased according to the ANS 1971 decay heat standard plus 20%. The water collected in the lower plenum and in the lower part of the bypass channel was drained by two separate lines in short and frequent periods, whenever the void fraction in these volumes fell below fifty per cent [3].

The only experimental data extracted from test number 78 was the temperatures for each rod group at five different heights. For each rod group and level, a range of temperatures and an average value of all the rods in the group are reported. It is important to note that this is a significant limitation for the correct understanding of the experiment and for the code simulation as it is not possible to identify the temperature evolution of one single rod. Furthermore, at certain levels and for certain groups the gap between the maximum and minimum measured temperature may be more than six hundred Kelvin, even if the cross section of the assembly is symmetrical in the radial direction.

For the purpose of this work, it will be particularly important to verify the capability of the TRACE code to simulate the correct temperature behaviour in the mid-plane of each rod group (where the temperature reaches its maximum), in order to predict the peak cladding temperature of the entire assembly, and the position where this value is reached.

3. Description of the model

The simulation of the experiment has been carried out with the U.S. NRC code TRACE, version 5.0 Patch 2. Two different models with different components have been tested. The first one uses two PIPE components to simulate the fuel assembly and the bypass channel, and eight heat structures (HTSTR component) representing the six rod groups (shown in Figure 1), the inner and outer walls. The second model tested uses the CHAN component for the simulation of the fuel assembly (rods and canister wall) and a PIPE component for the by-pass channel. The CHAN component has been designed with the specific purpose of simulating BWR fuel assembly. The nodalization diagrams of both models are shown in Figure 2.

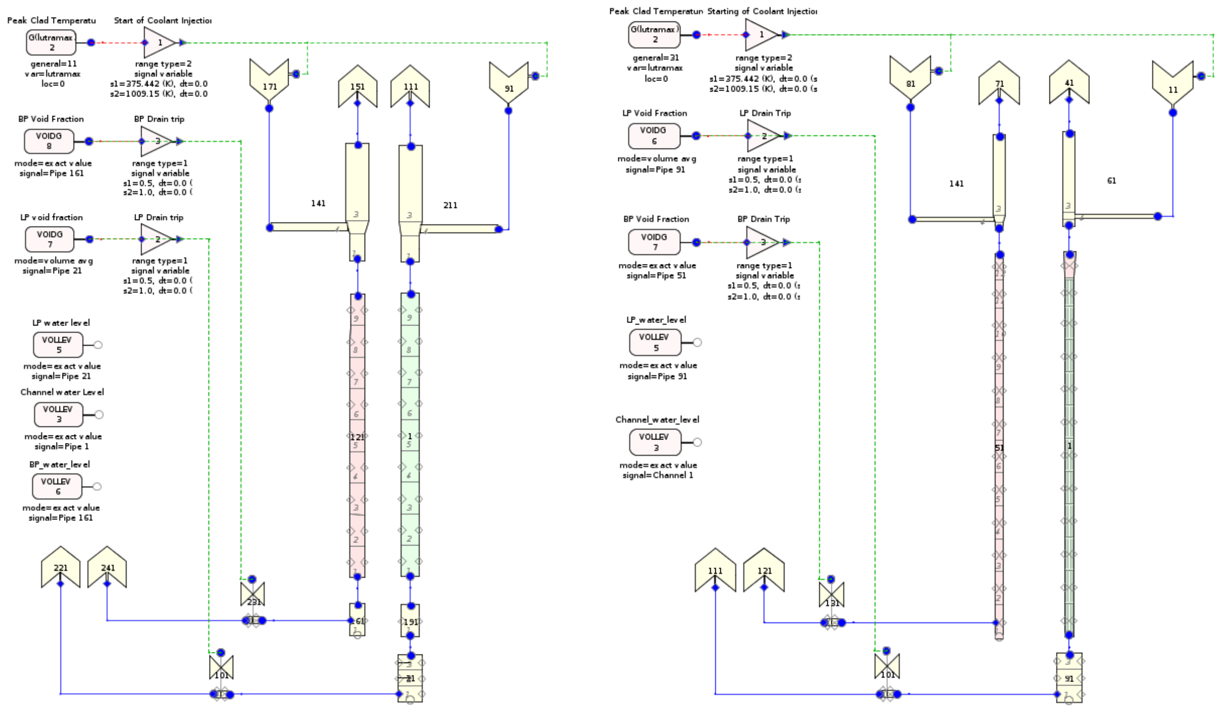


Figure 2 Nodalization diagrams for PIPE (left) and CHAN (right) models.

The other elements of the GÖTA facility have been modelled using the same components for both models. In particular, two FILL components controlled by TRIP and control blocks have been used to represent the coolant spray injection system. The upper part of the bundle has been modelled using two BREAK components for the steam vent (one per each channel), and two TEE components that simulate the steam line and the spray cooling line.

A PIPE component has been used to simulate the lower plenum of the bundle. The water draining from the bottom of the test vessel has been modelled using two BREAK components. Two VALVE components connect the lower plenum and the lower part of the bypass channel to the BREAK components, respectively. The opening and closing of the valves is controlled by two TRIP blocks.

3.1 Closure models and code options

TRACE code allows great flexibility in the way in which the simulations are performed, by allowing, including or excluding certain code features by the user.

3.1.1 Fine Mesh Reflood and Axial Heat Conduction

The Fine Mesh Reflood option has to be turned on whenever the calculation results in a significant dry-out and heat up of the fuel rods. When this option is active, the HTSTR component has the capability to dynamically add or remove axial nodes during the TRACE calculation. This option produces more accurate prediction of the axial gradient at a quench front moving through a fuel assembly or a fuel rod. Since this option increases the temperature gradients in the axial direction, the user should consider the activation of the “Axial Heat Conduction” option.

We have confirmed that these two options significantly improve the temperature prediction at most of all the axial levels. Turning off these options results in an excessive increase in the temperatures and absence of quench phenomena. Nevertheless, a very important issue has been found. In fact, a very fast quenching in the lower part of the channel (Level 1), which is not confirmed by the experimental data, has been observed.

3.1.2 Radiation Model

The simulation of radiation heat transfer is very important for the correct representation of the GÖTA experiments. TRACE offers different possibility for the construction of the radiation model, and there are also differences depending on the model (PIPE or CHAN) used.

The CHAN component offers the option to evaluate the view factors and beam lengths between all the rods in the channel automatically. The advantage of this option is the high accuracy in the calculation of all the parameters within the channel. The disadvantage is that this option prevents the user to define a radiation model (RADENC component) that includes the outer surface of the canister wall and the inner face of the bypass channel wall. This limitation affects the results in a non-negligible way. The previous investigators found that 20%-25% of the heat radiated from the heater rods to the canister's inner side was re-radiated from the canister's outer side to the shroud [3]. The lack of canister wall radiation in the TRACE CHAN Model results in a higher temperature of the canister wall than measured experimentally.

On the other hand, no view factors or beam length calculation is provided for the PIPE component, and the user has to define these parameters manually. The radiation model is then implemented by acting directly on the RADENC component. Thus the heat exchange between the outer side of the canister wall and the bypass channel wall can be modeled. On the other hand, it must be noticed that the view factors are input as a matrix, with row index surface connected to column index surface. TRACE requires zero values on the diagonal (which connect the surface to itself) which means that a heat structure cannot “see” itself.

Regarding the canister wall, the same issue has been found in the PIPE model as in the CHAN model. Wall emissivity larger than 0.3 prevents the wall to quench, resulting in a very high temperature.

3.1.3 CCFL Model

The counter-current flow limitation consists in the limiting of the liquid flow from the upper part of the channel to the lower region due to the steam flow in the opposite (counter-current) direction. Any increase in the vapour flux leads to a reduction in the liquid counter-flow, until eventually it becomes zero.

In TRACE code, it is possible to evaluate CCFL through three types of correlations: Wallis [4], Kutateladze [5] and Bankoff [6].

These models have been tested, in particular Bankoff model for the upper tie grid (above the channel) and Wallis and Kutateladze models for the fuel channel. The implementation of the Bankoff model showed no-influence on the prediction of the PCT, so it has not been considered for the final calculations. The use of the Kutateladze correlation resulted in a slight over-prediction of the PCT, which is not found if the Wallis correlation is used.

Previous studies [7] have confirmed that the Wallis correlation is preferable if the hydraulic diameter of the pipe is relatively small (2.5 mm to 50 mm), so this correlation has been selected to model the CCFL in the GÖTA test bundle.

4. Results

In this section the results from the TRACE simulation are reported and compared with the temperatures measured at different levels in the channel. In order to simplify the visualization, all the experimental temperatures in each level are shown in one plot, and only the maximum and the minimum value are reported together with all the computed temperatures for that axial level. The comparison between the calculated and experimental data is shown on Figure 3 for the PIPE model and Figure 4 for the CHAN model.

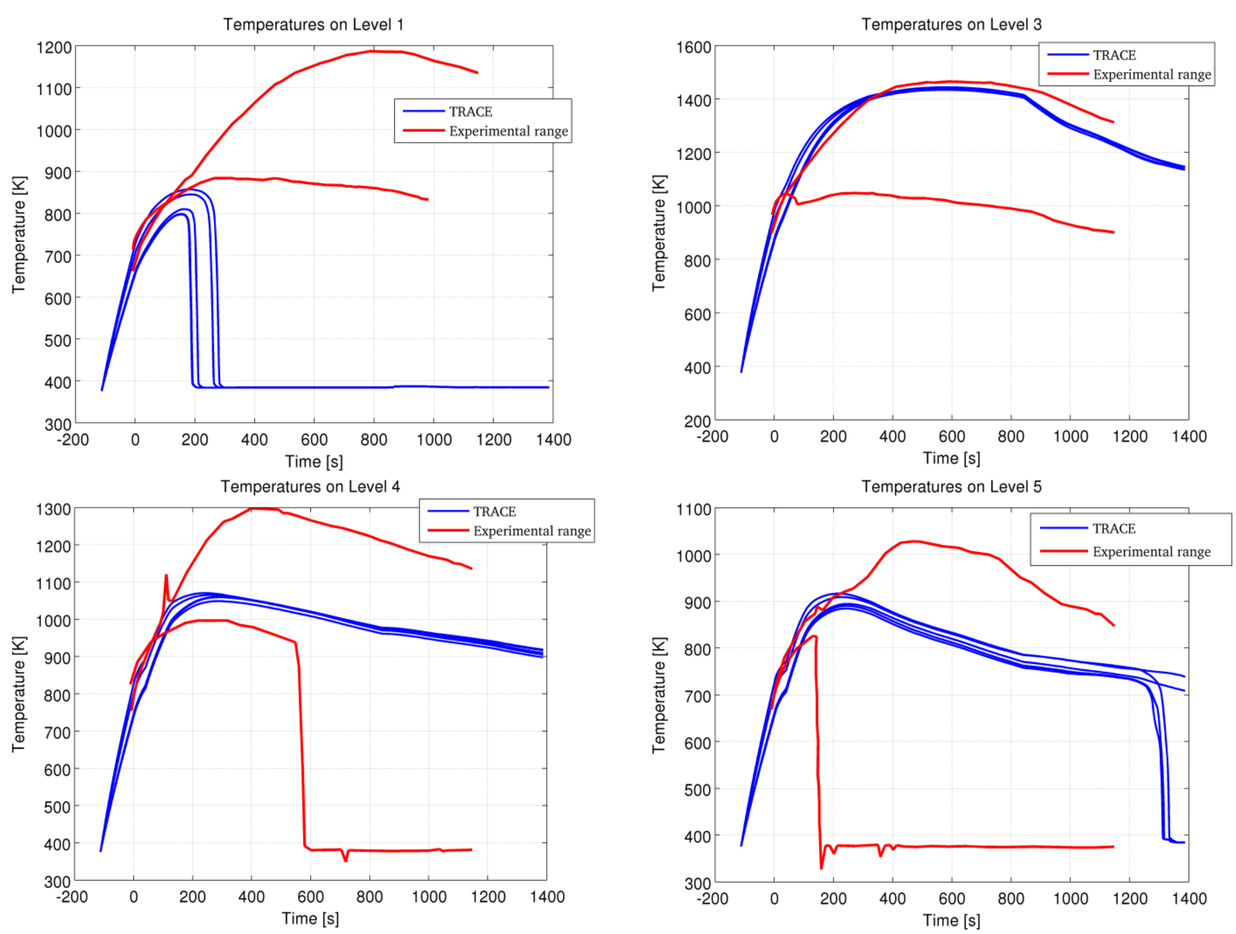


Figure 3 Results of the PIPE Model.

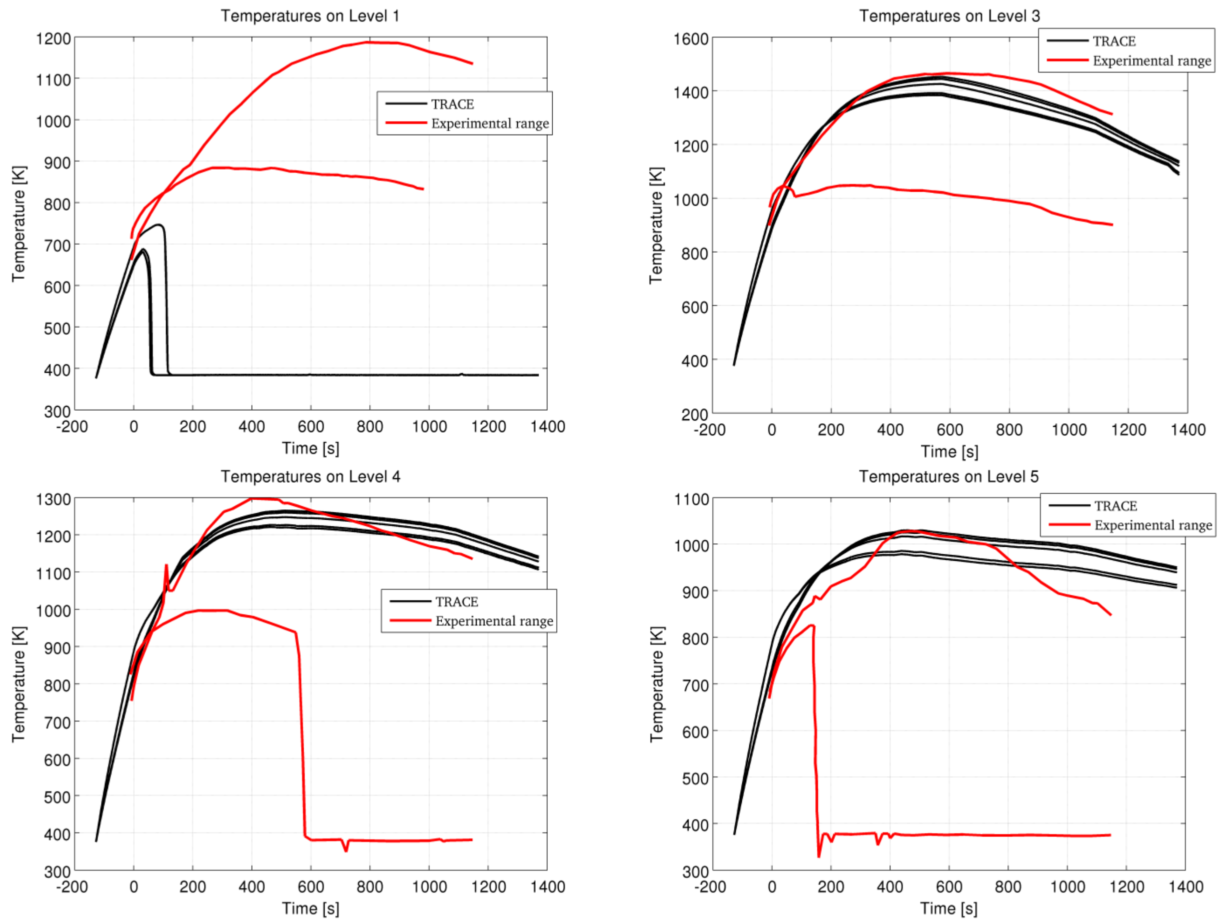


Figure 4 Results of the CHAN Model.

The results reported in Figure 3 show that the computed temperatures are, in most cases, within the range of the experimental data. In the upper part of the bundle (Level 4 and 5) the TRACE results follow an average temperature between the maximum and the minimum in the level. It is important to note that in the level where the Peak Cladding Temperature is located (Level 3) the TRACE code appears to predict the evolution of the maximum temperature very well. This is very important from the point of view of the reactor safety, because it proves that it is possible to evaluate the Peak Cladding Temperature with a certain precision.

The results of the CHAN Model (Figure 4) show some differences in the temperature behaviour relative to the PIPE Model. The temperatures in the upper part of the bundle (Level 4 and 5) are slightly over-predicted in the CHAN Model. Even if the Level 4 and 5 maximum cladding temperature is well predicted, the general temperature evolution shows that the quench of the surfaces occurs with a certain delay with respect to the experiment. On the other hand, the results in the mid-plane (Level 3) show that the quenching is predicted earlier than in the experiment, even if the PCT is higher than in the PIPE Model.

Moreover, the weak points found in the radiation model, present also in the CHAN component, results in an unjustified increase in the temperature of the canister wall mid-plane, which is not present in the PIPE Model. As has been explained in section 3.1.2, the code prevents the user from modeling the heat exchange by radiation from the outer side of the canister wall to the bypass channel, which results in an increase of the wall temperature and a delay in the quench.

It is now important to understand the reason for the large difference between the distribution of the calculated and experimental data. During the tests in the GÖTA facility it has been observed that the water injected from the spray in the top of the channel flows downward mainly in the region closer to the canister wall, while the central part of the channel is mainly occupied by the vapour that moves upward. This phenomenon causes a large temperature difference between the rods, because the rods in the peripheral region are cooled significantly more than the inner rods. This effect, together with the fact that the outer rods are facing the canister wall which is at a lower temperature, leads to a large difference between the minimum and the maximum temperature in one level. This is demonstrated by the fact that, even if the maximum radial power peaking factor is located in the corner rods (see Figure 1), the PCT is located in the rod group number 5 (inner rods).

It is not possible to simulate this multi-dimensional phenomenon in PIPE and CHAN component, because of the 1-D approximation used to solve the conservation equations in these components. This inherent limitation of the TRACE code explains the very low temperature differences in the radial direction: the difference between the various rod groups is only due to the radiation model.

Finally, the temperature calculated in the bottom of the channel (Level 1) show how the Fine Mesh Reflood and Axial Heat Conduction options affect the behaviour of the temperature at lower levels. A solution that fits better the experimental data can be obtained by de-activating those options, but this would cause worse prediction of temperatures on the mid-plane. Since it is more important to predict correctly the Peak Cladding Temperature, we have decided to maintain these options, accepting the error deriving from the code calculation.

5. Sensitivity and Uncertainty Analysis

The last part of the study consists of the application of a Sensitivity and Uncertainty analysis. Several methods have been developed for the uncertainty evaluation of system codes. In this work only Propagation of Input Errors (PIE) is considered [8].

5.1 Propagation of Input Errors Evaluation

The PIE approach consists of the statistical variation of the input parameters together with their uncertainties, in order to reveal the propagation of errors through the code. Instead of using the input parameters as discrete values, they are varied according to a Probability Density Function (PDF) and a certain number of calculations are performed in order to evaluate distribution and uncertainty of the output parameters.

The PDFs for the various input parameters are derived from the literature and from experts judgment, while the required number of calculations is given by the Wilks formula [9].The formula for the two-sided statistical tolerance intervals is given as:

$$1-\square 100\square-\square\ \ 1-\square\ 100\square 100\square-1\geq\square 100, \qquad (1)$$

which means that N calculations are required to be ▯ % confident that at least ▯ % of the combined influence of all the characterized uncertainties is below the tolerance limit. The U.S. NRC accept as a reasonable level of coverage/confidence 95%/95%which means that 93 calculations are required.

The PIE method has been applied for both the PIPE and the CHAN Models. The output parameter chosen for the PIE evaluation is the PCT, and the distribution of this parameter obtained from the PIE calculations is reported for both models in Figure 5.

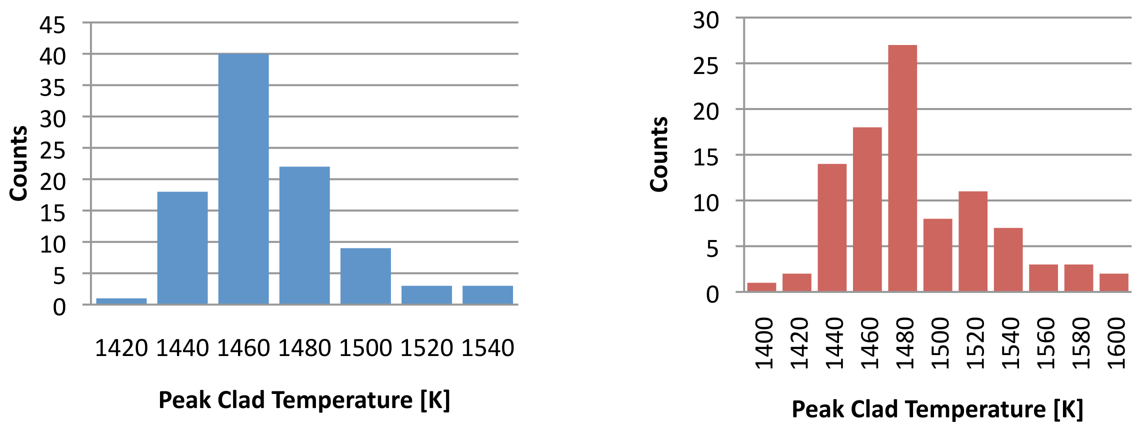


Figure 5 PCT distribution for the PIPE (right) and the CHAN Model (left).

The results of the uncertainty evaluation of the PIPE model show that the spread of the output parameter (PCT) can be approximated with the normal distribution, and the peak of the distribution is located on the interval 1440-1460 K, which is in a good agreement with the experimental data.

The distribution of the CHAN Model output parameter (PCT) differs in the location of the peak and the distribution. The peak of the PCT distribution is located on the interval 1460-1480 K, it has larger spread (standard deviation) and it does not fit the normal distribution. This implies that the CHAN Model results are more uncertain than the PIPE Model results.

5.2 Sensitivity Analysis

A sensitivity study has been performed on all the uncertain input parameters, in order to define the most influential input parameter on the figure of merit (PCT).

The Spearman Correlation is used to obtain the correlation coefficient between the input parameter and the output. This coefficient provides a quantitative measure of effect the input parameter has on the output figure of merit (PCT) [8]. In Figure 6 the absolute value of the Spearman correlation coefficient is reported for each parameter for PIPE and CHAN Model, respectively.

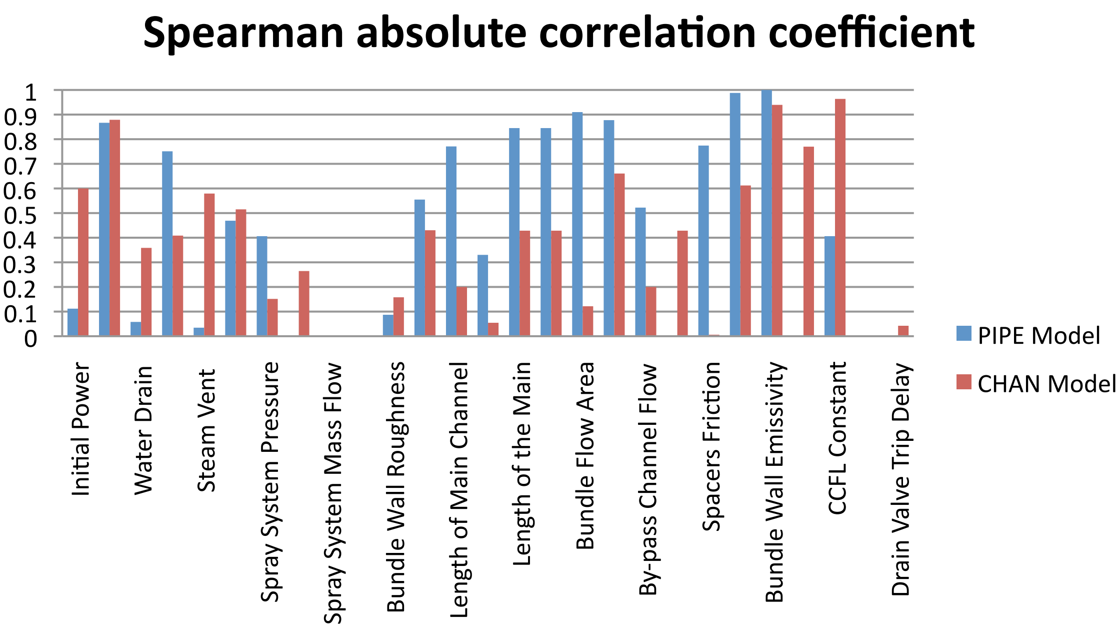


Figure 6 Absolute Spearman correlation coefficient for the PCT.

The results from the PIPE Model show that the two most influential parameters are the rod and the wall emissivity. This confirms that the radiation heat transfer plays a very important role in the cooling of the bundle, especially the interactions between the rods and the canister wall. Other important parameters are the one related to the geometry of the channel, i.e. the Flow Area and the Hydraulic Diameter. These parameters influence the mass flow rate of water and steam through the channel which determines the “coolability” of the rods.

The results from the CHAN Model show that the most influential parameter is the CCFL constant. The CHAN Model is much more sensitive to the CCFL phenomenon than the PIPE Model. The radiation model plays a secondary, although still important, role in the evolution of the transient.

6. Conclusion

In this paper, the results derived from two different TRACE models of the GÖTA test facility have been analysed and compared. An uncertainty study has been performed and the most influential parameters have been determined.

First, the results clearly show that the inability of the PIPE and CHAN components to model 2D phenomena does not prevent the code from correct prediction of the PCT and its evolution in time, which is very important from the safety point of view.

It was found that for the considered experiment, the PIPE Model provides a slightly better prediction of the PCT than the CHAN Model. However, both models predict an incorrect behaviour in the lower part of the channel (Level 1), where the activation of the axial heat conduction option results in a strong underestimation of the temperatures for all the rod groups.

These results may suggest that for the simulation of such kind of facility, it is better to use a more flexible component that allows the user to set more input parameters. For instance, the PIPE component has a possibility to create the fuel heat structures separately, and to set different value of emissivity in each one of them. This is not possible with the CHAN component, since only one value of emissivity is used for all the rods. Moreover, the possibility to set own value of the view factors represents additional degree of freedom for the PIPE component.

Finally, the results of the Uncertainty calculations show that the PIPE Model has a smaller uncertainty in the figure of merit (PCT). In fact, Figure 5 (left) shows how most of the computed PCTs are in the same interval as the experimental data (1440-1460 K). On the other hand, it can be noticed that the CHAN Model overestimates the PCT.

The most influential parameters for the PIPE Model are found to be the wall and the rod emissivity. These results highlight the importance of radiation heat transfer in a fuel assembly during the ECC phase, and generally in a Light Water Reactor. These results also emphasize the need for improvement of the radiation model in TRACE that currently presents a limitation for both the CHAN and the PIPE Model, as explained in Section 3.1.2.

On the other hand, the most influential parameter for the CHAN Models is the CCFL correlation constant. This explains the over-prediction of the temperature on the top of the bundle, which is due to an over-estimation in the CCFL phenomenon.

7. References

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