A Few Examples of ISPs Addressing Specific Reactor Safety Problems

Michel Réocreux

Institut de Radioprotection et de Sûreté Nucléaire
BP3 13115 Saint Paul Lez Durance Cedex

Abstract. Four International Standard Problems which were related to safety reactor problems are briefly discussed. ISP-20 (Steam Generator Tube Rupture in DOEL 2) is a unique ISP as it is based on a real incident which occurred in a commercial Power Plant. This ISP clearly illustrated the special problems of an ISP based on a real plant, namely limited access to precise plant data, some lack in the detailed knowledge of sensor behaviour, etc.

ISP-26 (ROSA IV-LSTF small break test) was an open ISP. A qualitatively good prediction of the measured events was obtained even if some modelling deficiencies were identified. ISP-27 (BETHSY Exp. 9.1 B) was a blind ISP. All important trends observed during the test were qualitatively calculated by most computer codes. However, some deficiencies in calculating some variables were evident. ISP-33 (PACTEL Natural Circulation) was an exercise with a test facility modelled on the basis of a Russian VVER 440 and with participations from Eastern and Western organisations. ISP-33 was a double-blind exercise. The simulation of some variables caused some problems although they were in principle not too complicated. Post-test calculations demonstrated significant improvements.

For all the four ISPs, the influence of the code user was evident and caused some scatter in the results. A specific study was performed in ISP-26 to clarify from where those user effects were coming.

The reactor safety problems related to those ISPs are detailed and the specific contribution of the ISPs to bring solutions is discussed.

1. INTRODUCTION

The ISP activity (International Standard problem) has been launched in the mid seventies and lasted up to now during almost 30 years. It consisted in an international comparison exercise of code predictions performed by different participants. Those predictions refer to some commonly agreed experiment performed and offered by the so called host country. The rules applied for such exercises are well defined [1]. In the beginning, due to the considerable difficulties encountered by the international community on thermalhydraulic modelling, the ISPs were oriented towards specific physics modelling questions (i.e. two phase flows resulting from depressurization, reflood,…). But very soon the choice went to more global tests [2] performed on integral experiments, which were designed for representing plant transients.

This paper will present and discuss a selection of those ISPs which are focussing particularly on specific reactor safety problems. As starting basis, we have used the paper presented by E. Hicken [3] during the THICKET 04 seminar. At that time three ISPs were selected and discussed. Besides this discussion Prof. Hicken - who was heading the CSNI Working Group 2 for about 10 years with many ISPs initiated and performed – gave his own prospect on the needs for additional ISPs. The author –who succeeded to Prof Hicken as chairman of the Principal Working Group 2 for the next 10 years- will provide additional views on what can be concluded from experience gained with those ISPs regarding safety needs. An additional ISP has been added to the selection in [3]. Presently the four ISPs selected for this paper are:

ISP20: Steam Generator Tube Rupture on Doel 2
ISP26: ROSA IV LSTF small break test
ISP27: BETHSY Experiment 9.1B
ISP33: PACTEL Natural Circulation test
Each ISP will be described in the next chapter. We will first refer to the plant transient which was intended to be simulated. Selection of results will be then presented in order to get a representative picture of the comparisons. The main conclusions obtained in the comparison report and agreed upon by the participants at the time of the ISP will be summarized.

In the last part of the paper, a general analysis of the outcome of those ISPs regarding safety problems they were intended to cover, will be performed and general conclusions will be drawn on the role of those ISPs to fulfil the plant safety needs.

2. SELECTED INTERNATIONAL STANDARD PROBLEMS: DESCRIPTION AND CONCLUSIONS

2.1. ISP20: Steam Generator Tube Rupture in DOEL 2

Doel 2 is a Westinghouse 2 loop PWR situated in Belgium with TRACTEBEL as architect/engineer. The plant was rated at 392 MWe and commissioned in 1975. The plant experienced on June 1979 a steam generator tube rupture incident. Belgium offered in 1985 this transient as the basis to an ISP. Several discussions subsequently took place. It was considered on one side, that the available plant data were very valuable because they were giving the behavior of a real plant which allows testing code models on full scale facilities, eliminating the scaling uncertainties in extrapolating from test facilities to real plant. However, on the other side, the quantity and the quality of the data on a real plant were not fitting the standards which were requested for ISPs [1] such as those obtained on well instrumented test facilities. Furthermore plant initial and boundary conditions had uncertainties which were not easy to cope with. A basic problem stood also with the proprietary character of the plant data. It had to be imperatively solved in order that the ISP could be organised. As the drawings of an operating plant could not be transmitted, it was found that a RELAP 5 data deck could be given to the participants and used as starting point for writing their own data deck. This way of managing such an ISP was not standard as well as the restriction on measured data, initial and boundary conditions. Despite of those unavoidable drawbacks, the steam generator tube rupture (SGTR) incident on Doel 2 was chosen as ISP because of its uniqueness as real plant transient and because of the uniqueness of the Belgium offer with regards of other plant transients which could have been candidate and which have neither been proposed.

2.1.1 Main events sequence of the plant transient

A more complete description of the plant sequence can be found in the ISP report [4]. The main events can be visualized on figure 1 where the main plant parameters are represented.

**Initial condition**
Just before the incident, the primary system was in a heat up phase after a 24 hours stop for repair work. The pressure had reached the rated value of 155 bars with a RCS temperature of about 255 °C. The water level in the pressurizer was controlled and kept constant by the charging flow. Primary pumps were running. Control rods were down. The total heating power was 11 MW.

**Initiating event**
At the time of SGTR (point A) a quick decrease of the level of pressurizer and a pressure decrease were observed. A quick increase of level in the Steam generator B (SGB) occurs. The SGTR was diagnosed by the operator when activity level was observed in the faulted SG (point D). The SGTR was a 7 cm long longitudinal crack located in the U bend, inducing a break flow rate around 15 kg/s.

**Mitigation phase**
The faulted SG was isolated (D) except the discharge at the turbo pump. The set point of the discharge valve of the affected SG was increased at the maximum value in order to avoid steam release. Steam discharge at the intact SG was initiated by the operator which induced a faster depressurization and the decrease of RCS temperature.

The operator tripped the primary pump of the affected loop to reduce the RCS heat input. Safety injection signal was generated on low pressurizer level. All four HPIS pumps started at 107 bar (E) and stabilized the pressure (F).

Because of the steam discharge, the level of the intact SG dropped below a low set point (G) which opened both steam discharge lines to the turbine auxiliary feed water pump. This resulted in a quick pressure decrease in both SGs followed by a rapid increase (H) of the level in the intact SG. The steam discharge to the turbine pump was stopped 8 minutes later (I).

In order to reduce the leak rate, the operator tried to equalize the pressures between RCS and affected SG. With that purpose, he started the main pump in intact loop and opened fully the spray system in the pressurizer (J). The result
Figure 1: Plant Parameters Evolution.

was a rapid RCS pressure decrease. As HPSI was fully working, an increase of the water inventory occurred and induced the increase of the water level in the pressurizer which went off scale (K). The operator stopped then the PRZ spray which caused an increase of the RCS pressure leading to the cut off the HPSI. The pressure stabilized then to 107 bars (L).

Safety injection cancelling phase
In order to reduce the leak rate with the objective of avoiding the flooding of the affected SG and the opening of the safety valves, it was necessary to decrease the RCS pressure, the final goal being to start as soon as possible the shutdown cooling system (28 bar).

The operator first tried to cancel the safety injection signal in order to stop the HPSI pumps which obliged before to stop the PRZ spray. This took around 20 minutes due to the need of overriding protection systems.

Three HPSI pumps were tripped simultaneously (M) and soon after the last one after checking the subcooling margin in the RCS. The RCS pressure dropped to 65 bar (O) for which the charging system compensated the leak rate. Due to an air supply problem in conjunction of the isolation phase of the reactor building, it took 20 minutes to open the discharge of the letdown lines (P). After stopping a charging pump (Q) the pressure decrease to the point (R) that the residual heat removal system could be coupled.

2.1.2 ISP results
Eight organisations from five countries (Belgium, Finland, France, Italy, and Yougoslavia) participated to this ISP. Concerning the codes, mostly RELAP 5 was used. France (IPSN, EDF and Framatome) used CATHARE and Finland used both RELAP 5 and SMABRE. Those codes are still used but were here early versions (20 years before). The results obtained by the comparisons of the code predictions and the actual plant transient will not be detailed here and can be found in [4]. Some examples are given hereunder which give an overall impression about discrepancies and agreements. In fig. 2 to 5, the pressure in the hot leg, the collapsed water in the pressurizer, the break mass flow rate and the mass inventory in the RCS are shown as calculated and compared with the data.
2.1.3 Conclusions

Conclusions were discussed by the participants at the final workshop (December 1987).

This ISP was the first ISP performed on a real plant transient. Specific problems related to the use of a real power plant transient were identified such as:

- Limited access to precise plant data, e.g. local geometries, structural heat losses, etc.
- Lack of precise knowledge of all sources and sinks of mass and heat. Indeed many systems, although of minor importance at full power, may influence the plant behavior at low power.
- Signal interpretation due to some lack in a detailed knowledge of sensor behavior, calibration and localisation.

Most participants succeeded in reaching an acceptable simulation of the primary system parameters (pressure, temperatures, inventory), while for the steam generator parameters, although the main trends were acceptable, a large spread in the data was noticed between code predictions and transients and even between predictions with the same code. No participant succeeded however in obtaining a quantitative good overall agreement, and this was attributed more to the uncertainty in the boundary conditions and plant data precision, than to inability of the codes.

The largest differences between the three codes RELAP 5, CATHARE and SMABRE showed up in the treatment of the condensation phenomena both in the primary system and the steam generators. Although large differences were observed in the condensation and vapour generation rates between different codes, the resulting pressure differences in the primary system were small, while large pressure differences were found in the damaged steam generator.

2.2 ISP 26 ROSA-IV LSTF Cold Leg Small Break LOCA Experiment

ROSA-IV LSTF facility is a 1/48 volumetrically scaled model of a 4 loop PWR. Main scaling principles include conservation of elevation (for natural circulation), representation of the 4 plant loops by two identical loops. Hot and cold legs were scaled in order to conserve volume and the ratio of the length to the square root of pipe diameter in expectation that the flow regime transitions can be simulated appropriately. The maximum heating power is 10 MW i.e. 14% of the scaled down full plant power. Several other scaling measures have been taken in relation to the

![Figure 2: RCS pressure.](image)

![Figure 3: Collapsed water level in PRZ.](image)

![Figure 4: Break mass flow rate.](image)

![Figure 5: Mass inventory in RCS.](image)
reduced power in order to simulate as closely as possible the plant transients which cannot be simulated for the starting phase when power stands between full power and reduced power. Those scaling measures involved in particular specific initial conditions in order that some physical quantities for the particular transient are conserved between the experiment and the plant. Numerous instrumentations have been installed (2500 instruments) which allows a quite detailed knowledge of the physical phenomena occurring in the facility.

2.2.1 Test sequence
At time 0 the break is opened in one of the cold leg. The size of the break is a 5% break referred to the area of the cold leg in the plant and scaled down to the LSTF facility. The reactor scram was sent at a pressurizer pressure of 129.7 bars about 10s after the break. A loss of offsite power was assumed simultaneously to the scram. This induced the non operation of the turbine bypass system, the trip of the reactor coolant pump which completely stopped at about 265s after the break. The safety injection signal occurred at 122.7 bars, 12 s after the break but the HPIS was not activated due to an assumption of failure. The secondary pressure increased after the closure of the turbine throttle valve. The SG relief valves operated and maintained the secondary pressure at 80 bars. A first core uncovering occurred between 120s and 155s inducing core heat up. The core level dropped concurrently with the level on the cross over leg downflow sides. At about 140s after the break, loop seal clearing occurred in both loop. The core liquid level recovered rapidly. The break experienced high quality flow. Depressurisation was accelerated. RCS pressure became lower than secondary pressure at 180s. RCS continued to empty and the core uncovered again at about 420s. Due to the continuing depressurisation, accumulators were actuated at 45.1 bars, 455s after the break. The core was covered again by two phase mixture.

2.2.2 ISP results
This ISP was an open ISP which means that the participants had the experimental results in hand. Nineteen calculations were submitted and seven computer codes were used. Eighteen participants came from fifteen countries. In order to make the comparisons readable, the host organisation gathered the calculations in two groups, the calculations using RELAP 5 and the calculations using other codes (ATHLET, CATHARE, TRAC, SMABRE, SATAN, NOTRUMP). The results of the comparisons will not be detailed here and can be found in [5]. Only examples restricted to RELAP5 are given hereunder in order to get a flavor of the discrepancies and agreements.

![Figure 6: Mass in the RCS.](image1)

![Figure 7: Break Flow Rate.](image2)
In order to get readable plots, the RELAP5 contributions have been split in two groups which are reported in separate plots. On figure 6 the mass inventory in the primary system is given. Even though this inventory is a very integral quantity, significant scatter is found. This could be explained by the differences obtained in the prediction of the break flow rates which are plotted on figure 7. On figure 8 the core differential pressures give a measure of the quantity of liquid in the core and hence of the core level. Those levels are of course depending of the overall mass inventory in the primary system but also of the liquid distribution in the circuit. Consequently it is quite difficult to find the causes of the observed scatter, error on mass inventory or error in mass distribution or both.

The rod temperatures which are the sensitive safety parameters are given on figure 9 at level 5 and on figure 10 at level 8. They are directly depending of the occurrence of the core uncovery. The discrepancies observed with measurements are both in timing and in amplitude.

2.2.3 Conclusions
Comparisons between calculations and experimental results [5] showed in general qualitatively good predictions namely for mass depletion, pressure evolution, loop seal clearing and core level drop. This was considered not to be a surprising result, as the ISP was an open problem. The conclusions changed quite completely when analysing quantitatively the results. In fact it appeared that all the calculations did not predict the responses of the important phenomena with satisfactory accuracy. This was the case, as mentioned earlier, for the core level drop and the subsequent rod temperature excursions.

It had been shown that one important cause common almost to all submitted calculations for those quantitative discrepancies was obviously the poor prediction of the break flow rate. Most codes failed in particular in the prediction of the dependence of the flow rate on the upstream conditions. This resulted in shifts of the timings of major events and made the comparisons difficult. But the break geometry was very particular to the facility. Consequently the observed discrepancies were judged as not really typical of the plant case. The host organization made some sensitivity studies substituting in the participants’ input decks the calculated break flow rate by the experimental one. The calculated results showed generally better agreement.

JAERI as host organization, made a particular effort in attempting to identify the sources of the differences among the submitted calculations. In particular they performed a lot of sensitivity studies using their own RELAP5 calculation or the RELAP5 input data decks that participants provide to them. Numerous sensitivity studies on
modelling and nodding were performed. They showed that the calculated results depend on user subjective choices and decisions. It was the first time that the so called "user effect" was investigated in such a detailed way in an ISP. Several sources of users' effect were identified [5]. Identification of all sources would have required much more effort which was not feasible by only one organisation.

Finally the ISP26 had demonstrated the capabilities and the limitations of the current advanced computer codes for complicated two phase flow situations such as the one simulated on LSTF. Investigation of the sensitivities to user choices had been performed and had shown that even with advanced codes the user effect was of the same order than with the first generation codes. Even though the interest of assessment on global experiment remained crucial, the host organization found that the complexity of phenomena made very difficult the analysis of the causes of the discrepancies. They recommended that "ISPs based on separate effect tests should deserve more emphasis than given so far in the thermalhydraulic ISPs". They recommended also that the high load of organizing such an ISP should be shared between several organisations.

2.3 ISP-27: BETHSY Test 9.1b - 2" Cold Leg Break without HPSI and Delayed Ultimate Procedure

The BETHSY integral test facility is a scaled down model of a 3 loop 900 MWe FRAMATOME PWR; the overall scaling factor applied to every volume, mass flow rate and power level is close to 1/100, while the elevations are 1/1 in order to preserve the gravitational heads. Similarly to LSTF, the core power has been limited to 10% of the nominal power. The facility was designed to operate the full range of pressures and to withstand high temperatures allowing procedures in severely impaired cooling conditions. Bethsy facility had 3 identical loops as in the plant with main coolant pump capable of delivering nominal flow rate. Every primary and secondary safety systems were simulated. The cold and hot legs were scaled down using Froude number in order to simulate two phase flow regimes. Specific attention was put to the heat losses control which may cause undesirable distortions. More than 1200 measuring instruments were installed for physical investigation. A computerized control system was installed in order to be able to simulate emergency operating procedures (EOPs) in an automatic mode.
2.3.1 Test sequence

The test 9.1b which was chosen for the ISP was a scaled down 2 inches cold leg break without HPSI and delayed ultimate procedure.

After the break opening, the emptying of the RCS occurred at 50s. The depressurisation speeded up and the primary pressure stabilized slightly above the secondary pressure. Scram signal occurred at 41s and safety injection signal at 54s but failure of HPSI was assumed. Core power decay started, main feedwater was off and turbine bypass occurred. Auxiliary feedwater was on. Main pump coast down started at 356s and ended at 971s. The upper plenum started to drain. The mixture level reached the elevation of the hot legs at 400s. The downflow and upflow sides of SG tubes drained and when emptied, the pressure vessel restarted depleting simultaneously with the level drop in the downflow side of the loop seals. Loop seal clearing occurred in loop 2 only, despite a symmetric level drop in the 3 loop seals. The core uncovery (on about 2m) was stopped and the level in the pressure vessel rose. Due to unavailability of HPIS the mass inventory continued to decrease. Levels depressed again and a second core uncovery occurred with a significant rod temperature increase.

The Ultimate procedure was initiated (UPI) when core wall maximum temperature was reached. UPI comprised full opening of the 3 SG atmospheric dump valves. It gave rise to strong condensation in the primary side of the SG which induced liquid fall back to the core. The primary pressure followed the fast decrease of the secondary pressure and accumulator injection occurred at 2962s. The subsequent inflow into the core provoked the turnaround of the rod temperatures with a minimum core mixture level at 0.6m. The mixture level reached the top of the core 300s after accumulator initiation. After accumulator isolation, the RCS mass inventory remained constant but pressure continued to decrease. The LPIS pressure was reached and LPIS was started inducing a fast increase of RCS mass inventory. Stable conditions for starting RHRS were reached 8200s after the break and ended the test.

2.3.2 ISP results

This ISP was offered by France in 1989 as blind ISP. The preliminary comparison report for blind submissions was issued in November 1991. A workshop was held in January 1992 with blind and post test presentations. Open submissions could be submitted up to May 1992 and the final comparison report was issued in July 1992 [6]. Forty eight (48) calculations were received. The first 28 were blind calculations. The following 20 calculations were either post test calculations (18) or open calculations (2). As a whole, 25 organisations from 19 countries have sent submissions and 9 different computer codes have been used.

On following figures 11 to 21 are given some examples of the comparison results. The calculations for being readable have been gathered in three groups, the calculations with RELAP5 mod2, the calculations with RELAP5 mod3 and the calculations with the other codes including Relap 4 mod6, Athlet, Cathare 2, Tech-M4, Dynamika 5, Notrump.

On figures 11 to 14, are given the results for the integrated break mass flow. Figures 11 to 13 refer to the blind calculations obtained respectively with the 3 codes Relap 5 mod2, Relap 5 mod3 and the calculations with the other codes including Relap 4 mod6, Athlet, Cathare 2, Tech-M4, Dynamika 5, Notrump. Figure 14 refers to the post test calculation with Relap 5 mod3 (to be compared with figure 12). The other examples which have been chosen are the results of maximum core temperatures especially during the last core uncovery (see figures 15 to 21).

Figure 11: Integrated Break Flow (RELAP5/2) Blind.
Figure 12: Integrated Break Flow (RELAP5/3) Blind.

Figure 13: Integrated Break Flow (other codes) Blind.

Figure 14: Integrated Break Flow (RELAP5/3) Post Test.
Figure 15: Maximum Core Clad Temperature (RELAP5/2) Blind.

Figure 16: Maximum Core Clad Temperature (RELAP5/3) Blind.

Figure 17: Maximum Core Clad Temperature (other codes) Blind.
Figure 18: Maximum Core Clad Temperature shift at accumulator(other codes) Blind.

Figure 19: Maximum Core Clad Temperature (RELAP5/2) Post Test.

Figure 20: Maximum Core Clad Temperature (RELAP5/3) Post Test.
On figures 15 to 18 are given the results obtained in the blind calculation part of the ISP with Relap5 mod2 (fig. 15), with Relap5 mod3 (fig. 16) and with the other codes (fig. 17). There are two factors in the scatter of the calculations. The first one consists in the general shifts in the event timing prediction which makes the core uncovery taking place at different times in the transient. The second one consists in the difference of flow and heat transfer modelling when core uncovery occurs. In order to try to dissociate the effect of these two factors, the host organisation has reported the results by shifting them on the time at which accumulator is predicted. The results obtained by this shifting are given on figure 18 for the calculations performed with the other codes. Comparison with figure 17 shows that the shifted results look less scattered. The remaining dispersion is effectively linked to the flow and heat transfer models. As those models are quite different from one code to the other, the scatter remains still significant. The same types of presentation with Relap5 mod2 or Relap5 mod3 are exhibiting closer results. The remaining scatter in that case looks very similar to the one observed in the post test calculations which are given on figure 19 to 21. Comparisons between figures 15/19, 16/20 and 17/21 don't need large explanations as they are self demonstrating the effect between blind and open predictions.

2.3.3 Conclusions
All important trends [6] observed during the test were qualitatively calculated by most computer codes. Some deficiencies are evident and will be discussed below:

The break flow plays an important role in this transient as a number of key events are dependent on primary mass inventory. The prediction of the break flow depends of the break device modelling and of the phase separation from the cold leg to the break device. In order to avoid discrepancies coming from the break device modelling which were specific to the facility and which were untypical of what occurs in the plant, separate effect tests were performed to characterize the break device and were provided to the participants (also for blind calculations). Even with this information some break model specially the break model of the RELAP-5 code showed deficiencies. Additional difficulties were encountered because in some cases an early onset of two-phase discharge at the break occurred which probably meant that upstream conditions were not adequately predicted.

The depressurisation of the primary system was usually adequately calculated. Almost all codes did not predict properly the heat transfer in the core in case of partial uncovery. Obviously some deficiencies compensated each other because the calculated clad temperatures were close or even larger than the experimental values.

Phenomena associated with accumulator injection appeared to be well handled. Most codes experienced difficulty to shift the system to RHRS operating conditions because a low pressure stratified two-phase flow associated with thermal stratification during the final part of the transient was difficult to model – especially with 1D modelling.

As a whole a large scattering of results was apparent even with the same code, which indicates a large user effect. Only strongly experienced participants have obtained good overall results.

A large number of laboratories participated to this ISP which meant that besides the technical benefits, an important interest in international cooperation was demonstrated. This outcome was made possible through the host organisation which supported the very high load due to the large number of participants.
2.4 ISP-33: PACTEL Natural Circulation Experiment with Stepwise Coolant Inventory Reduction

PACTEL facility is a volumetrically scaled (1:305) out of pile model of a Russian design VVER-440 PWR used in Finland.

The maximum operating pressure on the primary and secondary sides were 8 MPa and 4.6 Mpa (the VVER 440 has 12.3 MPa and 4.6 MPa). Three loops with double capacity steam generators were used on the model to represent the six loops of the reactor. The loops were scaled down near the Froude similitude in order to simulate the flow stratification phenomena on the reactor. Component heights and relative elevations corresponded to those of the full-scale reactor. The core consisted of 144 full-height, electrically heated fuel rods. The power was 1MW i.e. 22% of the scaled down full power of the reactor. Reduced pressure and reduced core power required appropriate scaled down initial conditions for the tests. All components such as pumps, steam generators, pressurizer, ECCS, etc…were active. The SGs were horizontal SGs like in the plant but with a reduced height which provoked a reduced natural circulation driving head. Instrumentation included 400 channels.

2.4.1 Test sequence

The experiment investigated natural circulation with different mass inventory in the RCS. The test started with full pressure at a power level of 155 kW (165 kW for the open calculations). After a period of stabilization of 1200 s, 60 kg of water were drained in 180 s from the lower plenum. The draining was followed by a period of 900s for the new natural circulation regime to establish and to stabilize. Then every 900 s, 60 kg water were drained from the lower plenum. The secondary pressure was reduced after the core began to heat-up for the first time. By this way the two phase flow natural circulation was investigated with decreasing amount of liquid and increasing void fractions. The test was terminated after the core temperatures began to rise again.

Similar tests have been performed on BETHSY, LSTF; PKL. The investigation of the successive steady state natural circulation flow regimes have proven to be a rather good method for checking the capabilities of predicting natural circulation during long lasting and slowly evolving small break or transient scenarios.

2.4.2 ISP results

The ISP was proposed as double blind ISP [7]. That means that it was the first time that a test was run on the facility. Codes could not be matched and adapted in advance on other tests of the same facility.

Double blind part of the ISP was followed by an open part [7]. The participants from 15 organisations in 12 countries submitted 21 double-blind calculations. Eight different codes were used for the double blind part (RELAP4 mod6, ATHLET, CATHARE 2, RELAP5 mod2, RELAP5 mod2.5, RELAP5 mod3, Tech-M4, Dynamika 5). Twenty open (post-test) calculations were performed by 14 organisations in 11 countries; five different codes were used.

Examples of results are shown in the following figures 22 to 29. Figures 22 and 23 give respectively the mass flow drained from the RCS and the time variation of the primary mass inventory showing the different steps of the test. On figure 24 are given the comparisons for the pressurizer of the experimental pressure and the different blind predictions with RELAP5 mod2/2.5. On figure 25 same comparisons are given but for the levels in the primary system. Figures 26 and 27 provide the levels in the hot legs respectively in loop 1 and in loop 2. Figures 28 and 29 are providing the primary pressure variation for pre-test calculations and post test calculations. This gives a picture of the scatter variations between pre and post test calculations.

2.4.3 Conclusions

Full conclusions can be found in the final report [7]. It came out that the overall transient was reasonably well calculated by all codes except by RELAP4/MOD6 indicating the limitations of the old conservative codes.

However, advanced best-estimate codes showed also some shortcomings. The main discrepancies were noticed in the prediction of the flow stagnations and the three pressure spikes after the second draining, and the time of the core heat-up at the end of the test. The heat-up was calculated to occur more or less too late. These problems were mainly due to the inability to calculate accurately the behavior of the hot leg loop seals and the accumulation of water in the horizontal steam generator tubes. The 2-phase natural circulation flow rate and the refilling rate of the pressurizer at the end of the test were generally over predicted. For blind calculations, the scatter within the RELAP5/MOD2 results was larger than the scatter within other code results.

Compared to the double-blind calculations the results in the open exercise were improved. This time the scatter within the RELAP5/MOD3 results was larger than the scatter within other code results. The remaining problems had to do with the accurate calculation of the loop seal behavior and the accumulation of water inside the steam
generators leading to a too late core heat-up. Also the codes tend to over predict 2-phase natural circulation flow rate in case of small gravitational driving heads. As expected, the accuracy of the calculated results decreased clearly with time as the deviations from reality accumulate. In summary, despite of some inaccuracies in the original specifications, ISP-33 proved to be a successful and valuable exercise.

3. CONTRIBUTION OF THE SELECTED ISPs TO SOLVING SPECIFIC REACTOR SAFETY PROBLEMS

The experiments, used for the selected ISPs in this paper, are integral experiments. By nature, the main objective of these experiments is to simulate globally all the needed plant components and to reproduce plant transients where specific safety problems are expected. Each of the selected ISP corresponds then to some reactor safety problem and in the following chapter, we will discuss how the selected ISPs are contributing to the solution of those corresponding reactor safety problems.

3.1 Reactor Safety Problems corresponding to the Selected ISPs

The first ISP (ISP20) as based on a real plant incident is directly linked with safety problems on the plants. Moreover this kind of incident, the rupture of a steam generator tube, has occurred in many other plants than Doel. The tubes of steam generators are components which are experiencing several problems such as corrosion; vibrations, leaks which may degenerate in tube "ruptures". The main concern for safety of such accidents is related to the fact that radioactivity may be released in the atmosphere during the course of the accident and that it has a relatively non negligible occurrence probability. In order to mitigate such incidents/accidents, procedures have been elaborated in order to limit the activity releases in the environment and in order that operators stop the reactor and put it back to a safe state. After TMI, the case of SG tube rupture has been a kind of a reference case for elaborating procedures and a large number of studies had been then performed. The interest of recalculating such incident is evident.
The second ISP on LSTF facility (ISP26) simulated a small break. It reproduced a typical physical situation which may be encountered on plants. The size of the break and the scenario (for instance the assumption of loss of offsite power) have been chosen so that the test investigated the most important safety problems encountered in the small break category of accidents, i.e. the RCS mass depletion connection with the break flow regimes, the core uncovery and core recovery.

The third ISP on BETHSY facility (ISP27) was based a 2'' cold leg break test combined with a High Pressure Injection System (HPIS) failure. In that case, the state oriented approach requires operators to start an Ultimate Procedure, which consists in fully opening the Steam Generator (SG) atmospheric dumps as soon as they are informed of the unavailability of the HPIS. Such full opening of the SG atmospheric dumps is a very common and important action for safety. The studied scenario assumed in addition a delayed application of this procedure, which is started only when the core outlet temperature rises and reaches values significantly higher than saturation temperature. This allowed investigating the core uncovery phenomena and the efficiency of the ultimate procedure in degraded conditions which are both significant safety problems.

The interest, from a safety point of view, of the fourth ISP on PACTEL (ISP33) was that it investigated, in the small break domain, a Russian reactor design with horizontal steam generators which was not extensively investigated before. Natural circulation was considered as a particularly suitable phenomenon to focus on the first VVER related ISP due to its fundamental importance in most accidents and transients, and due to the expected different behavior compared to Western PWRs. The difference between VVER and Western PWRs consists mainly in the effect of the horizontal steam generators and of the hot leg loop seals.

The second feature of interest for safety, was that this ISP was a double blind exercise. The double blind character of an ISP puts in fact the participant in a position very similar to the position where the safety analyst stands for analysing the plant accidents. Consequently double blind ISP can give valuable indication on how the plant safety problems can be solved and can provide information on the difficulties that may be encountered in the safety analysis.

3.2. ISPs: an Experimental Answer to Reactor Safety Problems?
The experimental facilities used for the selected ISPs are simulating the whole plant system. They are designed in order to represent as closely as possible what may occur on the reference plant. The tests run on those facilities are defined in order to reproduce transients taking place on the plant. The plant transients which are chosen to be simulated experimentally are often those that correspond to some safety problems. Hence, the experimental results obtained for those transients should provide some information on the safety problems themselves. Accordingly it is logical to try to find which direct solution the experimental results can give to the safety problem.

As the experimental facility has been designed to reproduce what occurs in the plant, the temptation is very high to consider that the obtained experimental results are evidences which provide "The" answer to the related safety questions. For example if the efficiency of an operating procedure is questioned, and if this operating procedure has been simulated on an integral experiment, it can be considered that the efficiency of the procedure on the plant is demonstrated by the success of the procedure on the experiment.

Such demonstration is comfortable because it does not imply sophisticated and sometimes hardly understandable analysis. The answer can be "touched" with a physical proof. It has to be recognized that every concerned body (safety authority, vendor, utility), when some safety question is put forward, is feeling comfortable if he can get some experimental proof. This is a general tendency which is going much further than simply the thermalhydraulic domain.

Indeed, the rationale under such use of the experimental results is facing the scaling question. Inherently to the scaling process, the direct use of experimental results to the plant is in fact invalid. In the scaling down, all the scaling laws cannot be satisfied at the same time. Consequently the integral experiment cannot be proven to be an exact simulation of what happens on the plant. Of course the scaling down compromises are chosen so that the main phenomena occurring on the plant should be experimentally reproduced at least qualitatively. But distortions remain which should prevent any direct transposition of the experiment behavior to the plant behavior. Of course, the facility as designed with the appropriate scaling compromises should give a response similar to the plant. Consequently, when the facility response provides an answer going in one direction, this answer can be considered as a good presumption but not as a demonstration for the plant. To obtain the status of demonstration, it should be established that there is no distortion introduced by the facility.

It is clear that quite often the presumption is assimilated or not far to be assimilated to the demonstration. This is of course encouraged when the answer drawn from the facility is favourable. When the answer is unfavourable, most of the time people remind that the facility response is not the reactor response and good reasons for distortions are generally found in order not to use directly the unfavourable response coming from the experimental facility.

This assimilation of the integral facility response to the reactor response has been quite often done when the decisions to build integral facilities were taken. The managers would like, with the money they were investing, to get "demonstration tests" that answer directly their safety concerns. It happened quite often in the end of 70's and beginning of 80's with the large break LOCA tests which were claimed to demonstrate that ECCS was working well. It was often declared that, based on the experimental results obtained on those facilities, safety problems with large Break LOCA were adequately handled on plants. Similarly in the end of 80's, beginning of 90's, the new facilities designed for small break and transient simulation were sometimes claimed to be able to show that the operator procedures on the plant were well designed and working.

ISPs do not stay away from this tendency. In order to maximize the output of ISPs or because some people are urging to close issues, some would like that ISPs will give answers directly useable for solving the related safety problem. This is obviously a basic error: ISPs can provide presumptions, but not demonstrations. It is often quite difficult to convince people not to fall in this confusion. The author encourages the experts when they will launch in the future new ISPs, not to forget this possible misunderstanding.

The best mean that we have now for transposing the behavior of integral facilities at the plant scale consists in the use of computer codes. More precisely it is the physical modelling which is embedded in the computer codes that allows such transposition. For that purpose, the computer codes and the physical modelling must be assessed both on separate effect tests and on integral facilities, particularly on tests related to specific safety problems. It is those assessed computer codes which will give the response of the plant that can be used to answer the safety problems. In this assessment process, ISPs play obviously an important role that we are going to discuss next.

3.3 Code Assessment and Safety Problems. Role of selected ISPs
As discussed above, an assessed code is needed in order to answer safety questions. Obviously ISPs are providing an important contribution to this assessment. In particular the variety of codes used, the intercomparisons between various codes and the variety of users of the same code offer a basis of information on the code assessment much larger than when the assessment is performed only by one organisation. ISPs contribution is then unique and code assessment has to be considered as a major if not the first outcome from the ISPs.

However this major contribution must be moderated to its proper role. Some people, because they would like to emphasize too much the ISP role, or for some others, because they aimed at closing assessment issues with reduced efforts, would like to consider that running an ISP is self proving that the code is really assessed in the domain covered by the ISP. This is certainly a mistake: for giving the best possible answer to the scaling question, assessment must be based on several tests and facilities, separate effect tests as well as integral tests. It cannot be restricted only to ISPs. Even if the chosen ISP test is certainly one of the most significant in its domain, it cannot prove alone that the code is completely assessed and, there is an obvious need for complementary assessment (see for example ISP26 conclusions where a need for more assessment on separate effect tests was indicated by the host country itself).

In the ISP selection of this paper, ISP20 plays a specific role for the code assessment. As ISP 20 is a real plant transient the obtained assessment is a direct assessment without any transposition question. In that case a direct answer to the related safety question could be envisaged as there is no transposition to perform. However if one wants to extensively answer the SG tube rupture safety question, other ranges of parameters should be investigated. Those ranges cannot be generally simulated on the plant and sensitivity studies are required to cover them. An assessed code is again required for that purpose. But it is clear that being able to assess the code on one real case is certainly an advantage because it allows to by pass the transposition problem. Such kind of assessment should be strongly encouraged every time it is possible. However uncertainties in the knowledge of boundary conditions on the plant and weaknesses of the measurements in quality and quantity made the assessment on plant transient quite difficult. More assessment on real plant transient would be certainly more beneficial if better measurements and better knowledge of boundary conditions were available. The question of the measurements available on the plants is certainly crucial and should be investigated on the principle itself. As example of such instrumentation problem, it has to be noted that often codes cannot be assessed on the transients occurring on the plants, but are used as a mean to compensate the weaknesses of instrumentation in trying to understand more precisely what may have occurred during the transient.

For the other ISPs of the selection, they are based on integral experiments which are simulating different reference plants 3-loop PWRs, 4-loop PWRs, VVER440. The comparisons between the different code calculations and with the experimental results have shown for each of those ISPs that the codes were qualitatively able to predict small break transients. This means that their use for solving safety problems on reactors in this area can be envisaged positively. From those comparisons deficiencies have been also identified. This identification points out the limitations that should be taken into account in the code predictions at the plant scale for this category of transients. Even though limitations are negative for the capabilities of the codes themselves, their identification gives, for solving the reactor safety problems, very positive indications on how far the obtained plant predictions can be trusted. Of course one must evaluate what the limitations are becoming in the transposition. The knowledge of those limitations can be crucial in the application.

It has been shown in several of the selected ISPs that the break mass flow was one of the most influencing parameters and that it induced significant discrepancies between predicted and measured values. For some experts those difficulties are not safety significant because in the "real plant" case the break would hardly be a circular break area (most probably a crack with sharp edges) and because the area should be varied in a large range. Consequently the effect of discrepancies in the break flow should be more considered in the test prediction as a kind of experimental uncertainty that has to be reduced. In fact as it has been well analysed in ISP27, the discrepancies in the break flow prediction are due both to the difficulties in predicting the flow in the break device and to the difficulties in predicting the phase separation from the cold leg to the break device. In ISP27 separate effect tests allowed getting rid of the uncertainties for the flow calculation in the break device. Those separate effect tests allowed adapting the modelling of the break device which is not typical of a break in the plant. The observed remaining discrepancies in the break flow prediction could then be attributed mostly to the prediction of phase separation from the hot leg to the break device. As a result, the phase separation appears as a significant cause of the overall discrepancies. In the "real plant" case, the break area will have anyway to be varied in the studies. It will modify the "amplitude" of the break mass flow. However the phase separation from the leg to the break will affect the changes in the upstream conditions to the break and will have consequently a significant effect on the timing of break flow regimes changes and therefore on the timing of the safety sensitive events occurring during the transients.
Deficiencies have also been identified on heat transfer modelling. As a consequence the clad temperature predictions displayed significant deviations from the physical values. But calculations for many reactors have shown that the calculated maximum clad temperatures are well below the limit temperatures as specified in licensing requirements. Therefore, a deviation of the calculated temperature from the experimental result might not be relevant in safety assessments of reactors. This statement of some experts is partially true because there are cases where additional failures or delays in starting procedures may lead to higher temperatures which may become near the licensing limits. Moreover, calculations with best-estimate codes are also used for developing and validating Accident Management Measures and for developing training simulators. For those applications it is important that the calculated results do not mislead operators nor result in a non optimised Accident Management Measures. In that cases, the deficiencies in heat transfer modelling could be safety significant at the plant scale and should be coped adequately.

3.4. The Lessons Learned which are Directly Applicable to Reactor Safety Problems

In all ISPs, two observations can be made:

1. There is an evident user effect.
2. Post-test-calculations or open calculations show always a better agreement with experiments as compared with pre-test-calculations or blind calculations.

The user effect observed in ISP26 was quite disappointing for the host organisation. This ISP was one of the first performed with the brand new advanced codes. It was then foreseen that those modern codes would present less user effect than the old first generation codes. In fact it came out that the user effect was almost of the same order. User effect was not at all eradicated. In the following ISPs same observations were made which demonstrated some permanence of the user effect. In the ISP26 analysis, a detailed study had been initiated by the host organisation in order to find what was producing the user effect. It was the first time that such a study was performed. Specific report have been written next by the Principal Working Group 2 on the identified sources of user effect but even with this comprehensive report it has to be recognized that no way has been found to suppress this effect.

The improvements between pre-test and post test calculations or between blind and open calculations is also a long lasting observation and not only in ISPs. It would have been beneficial if, within the framework of those ISPs, a detailed study would have been performed and would have indicated what has been done by the analysts to receive the improvements – improvements in modelling, in nodalisation or even “tuning”.

The user effect which is almost exclusively observed in ISPs or benchmarks, together with the improvement effect between pre and post test prediction, should obviously be taken into account in all plant analyses based on code predictions. They are then certainly lessons which are directly applicable for solving the safety problems.

When a safety analyst is in the position to perform a plant prediction, he is always in a situation of pre test (blind) prediction as most of the time post test (open) do not exist for reactor. There is also absolutely no reason that, the safety analyst, as single user, even if he is well and extensively trained, does not pick up some user effect in his prediction as in any other reactor prediction. It has to be noted that generally the safety analysts consider that the user effect is a matter for the other analysts and not for them, but ISPs results show that those effects cannot be denied. Nevertheless we have to recognize that user effect is never taken into account explicitly in the reactor predictions. The question is then how this could be achieved.

Normally in the best estimate approach for reactor predictions, the user and the "pre-test" effects should contribute to the uncertainties of the reactor predictions. One has to recognize that most of the reactor uncertainties analyses do not take into account those contributions. One way could be to take those contributions as a whole. This is never done. Nevertheless this should be difficult because it should raise a lot of questions namely because of the obvious dependency between contributions and analyses. A global evaluation of contributions will then be considered as a judgement on the analyst skill and this is not obviously viable. The other more logical way would be to analyze and evaluate the uncertainties generated by all sources that have been identified as causes of the user effect. For pre test effect this could not be achieved as the sources have never been identified. For the user effect this uncertainty evaluation has also almost never been done because the related sources are precisely points which are not considered in the uncertainty analysis, such as effect of non converged calculation (special noding for compensating physical model weaknesses), effect of wrong models due to epistemic ignorance (model options), tricky personal modelling of phenomena not represented in the code; effect of use of a tricky tuned 1D model instead of a 3D model, errors in the input data deck, … Ignoring those factors is certainly a major drawback with a non negligible effect as can be seen on the picture of the scatter exhibited by the ISPs.
Finally one lesson from the ISPs activity, directly applicable to solving reactor problems with code calculations will be to take into account in the uncertainty evaluation all the sources related to user and pre test effect. This would certainly be a great progress and would largely improve the confidence we can get in the best estimate predictions.

4. FUTURE TRENDS

We will restrict here the discussion to thermalhydraulic ISPs in relation with reactor safety problems.

The first direction which is characterizing the future trends is most probably the use of best estimate codes for solving reactor safety problems. This activity which is underway will last several years. The use of BE codes requires that one proceeds to uncertainty evaluations. Those uncertainty evaluations on plants should interact strongly with the code assessment. In that context, ISPs or ISPs-like activities such as benchmarks oriented towards uncertainty evaluation may be of interest. The BEMUSE activity which is underway is entering in this category. In BEMUSE an uncertainty evaluation comparison has been performed on the basis of an old ISP (ISP 13 -LOFT). This comparison is presently followed by a benchmark exercise on a plant accident calculation. BEMUSE will certainly contribute to the improvement of uncertainty evaluation, but we can expect that some problems will remain and that some follow up ISP or ISP-like activities will be initiated in the future.

The second direction for the next years is the use and the development of 3D codes. Related experiments and validation of 3D codes in particular single phase or two phase CFD codes, which are necessary for a 3-D-analysis, are underway or will start. It seems highly probable that those activities will be accompanied at the international level by an ISP activity as it was done for the second generation thermalhydraulic codes.

But the future of the thermalhydraulic ISPs in relation with reactor safety problems cannot be envisaged without considering the general trends in the nuclear business. Till a recent period, the trends were oriented towards a generalized reduction:

- Decrease of the budgets for experimental and analytical work
- Decrease of the number of small and large facilities
- Decrease of the number of experienced experts
- Decrease of the number of students and of the number of chairs at universities in the nuclear field

With the "nuclear renaissance" those trends are generally reversing but not all in the same way:

- The budgets may increase but they may be more dedicated to operational problems. Besides keeping a high safety level, the main aims for industry are lifetime extension, power increase, higher burn-up and higher availability. Except for power increase, thermalhydraulic studies are not the first concern. Consequently, there may not be a need for running thermalhydraulics ISPs in relation with those plant problems.

- The tendency in the decrease of the number of small and large facilities will certainly stop but it is improbable that this tendency will reverse except for facilities related to 3D codes development and assessment. It seems clear that there will be a need to optimize at an international level the use of the few remaining facilities. Continuation of the facilities has been already permitted through some OECD projects. Sharing ISPs may be an additional and very efficient mean to contribute to the programme optimization for the remaining facilities. This will require that a minimum opening of the projects to outside countries will be effectively obtained so that some of the tests performed on those facilities can be offered as ISP.

- In relation with the nuclear renaissance, the number of required experts will increase drastically. For those new people there will be an enormous need to get experience and competence. The series of old ISPs could be a mean for internal training (with perhaps provision of some additional documentation). One can also imagine that some old ISPs could be repeated internationally in order to serve as learning process. This could be performed within some kind of junior CSNI working groups and of course with adequate adjustments for organizing such exercise.

ISP5s have been one of the most fruitful activity in CSNI. It is important for the coming generation to learn about the outcomes it has provided and it is hoped that this paper will have contributed to this learning process. With this knowledge, it will be up to the coming generation to decide in which area and how they would like to continue.
REFERENCES


