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Application of Best Estimate Thermalhydraulic Codes for the Safety Analysis of Research Reactors

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ABSTRACT

An established international expertise in relation to computational tools, procedures for their application including Best Estimate (BE) methods supported by uncertainty evaluation, and comprehensive experimental database exists within the safety technology of Nuclear Power Plant (NPP). The importance of transferring NPP safety technology tools and methods to RR safety technology has been noted in recent IAEA activities. However, the ranges of parameters of interest to RR are different from those for NPP: this is namely true for fuel composition, system pressure, adopted materials and overall system geometric configuration. The large variety of research reactors prevented so far the achievement of systematic and detailed lists of initiating events based upon qualified Probabilistic Safety Assessment (PSA) studies with results endorsed by the international community. However, bounding and generalized lists of events are available from IAEA documents and can be considered for deeper studies in the area [1].

In the area of acceptance criteria, established standards accepted by the international community are available. Therefore no major effort is needed, but an effort appears worthwhile to check that those standards are adopted and that the related thresholds are fulfilled. The importance of suitable experimental assessment is recognized. A large amount of data exists as the kinetic dynamic core behaviour form SPERT reactors tests [2]. However, not all data are accessible to all institutions and the relationship between the range of parameters of experiments and the range of parameters relevant to RR technology is not always established. However, code-assessment through relevant set of experimental data are recorded and properly stored.

An established technology exists for development, qualification and application of system thermal-hydraulics codes suitable to be adopted for accident analysis in research reactors. This derives from NPP technology. The applicability of system codes like RELAP5, COBRA and MARS to the research reactor needs has been confirmed from recent works [3]. Definitely, system codes are mature for application to transient analysis in research reactors. However, code limitations have been found in predicting pressure drops as a function of mass flux at low values of mass flux when nucleate boiling occurs. The importance of the Whittle & Forgan experiments shall be mentioned and therefore furthermore, code validation must be demonstrated for the range of parameters of interest to research reactors [4].

1 INTRODUCTION

The lack of full understanding of complex mechanisms connected to the interaction between thermal-hydraulic and neutronics still challenge the design and the operation of nuclear reactors. This lacking is generally overcome by adopting conservative safety limits. Nowadays, the availability of powerful computer and computational techniques together with the continuing increase in operational experience imposes the revisiting of those areas and eventually the identification of design/operation requirement that can be relaxed [5]. So far, almost all of the safety analyses of research reactors have been performed using conservative computational tools [6], [7], [8]. The application of BE method constitutes a real necessity in the safety and design analysis and allows getting more realistic simulation of the processes taking place during the steady state operation and transients. In comparison to the conservative approaches, the application of BE methods results in a precise prediction of the system behaviour leading to mitigation of constraining limits in design and operation. In this framework, an attempt is made to apply the BE technique to perform safety evaluation under research reactors operational conditions.

An attempt to perform standardized safety analyses for RR was proposed by the IAEA [9] in the framework of core conversion from the use of highly enriched uranium fuel to the use of low enriched uranium fuel. In this regard, the facility operator would be required to submit an amendment to, or a revision of, the Safety Report. For this purpose, a safety-related benchmark problem for an idealized generic 10 MW MTR light-water pool-type reactor was specified in order to compare calculational methods used in various research centers and institutions.

The related benchmark problem covers large steady state kinetic and thermal-hydraulic calculations and wide range of hypothetical typical RR transient conditions. However, almost all of the safety analyses have so far been performed using conservative computational tools [10]. In the current framework the IAEA 10 MW MTR research reactor problem [9] is re-considered through the use of a best estimate computational thermal-hydraulic code. The results obtained, considering a large spectra of RR typical transients, using RELAP5/Mod 3.2 code system [11] are hereafter presented and commented.

2 THE BENCHMARK PROBLEM

The IAEA Research Reactor as defined in reference [9] is a pool 10 MW MTR type. The core is cooled and moderated by downward forced circulation of light water. The reactor core consist of 5 x 6 grid core (Figure 1) containing 21 Standard MTR Fuel Elements (SFE) (Figure 2) and 4 Control Fuel Elements (CFE) (Figure 3). The core is reflected by graphite on two opposite faces and surrounded by water. The SFE contain 23 standard plates whereas the CFE contain 17 standard plates with special region to receive the 4 fork type absorber blades. No information was made available for the coolant loop in the benchmark specification volumes [9]. In the current framework, a standard nodalization of a typical MTR research reactor was considered [12], [13]. The reactor pool above the core zone is also modelled in order to adequately simulate the natural convection process (Figure 4). The benchmark problem consists in some protected transients in MTR Highly Enriched Uranium core (HEU) and Low Enriched Uranium (LEU) cores. These generic reactors are representative of medium power research reactors with high fissile loading and more demanding thermal-hydraulic requirements, and used only to give an indication of the reliability of the methods adopted.

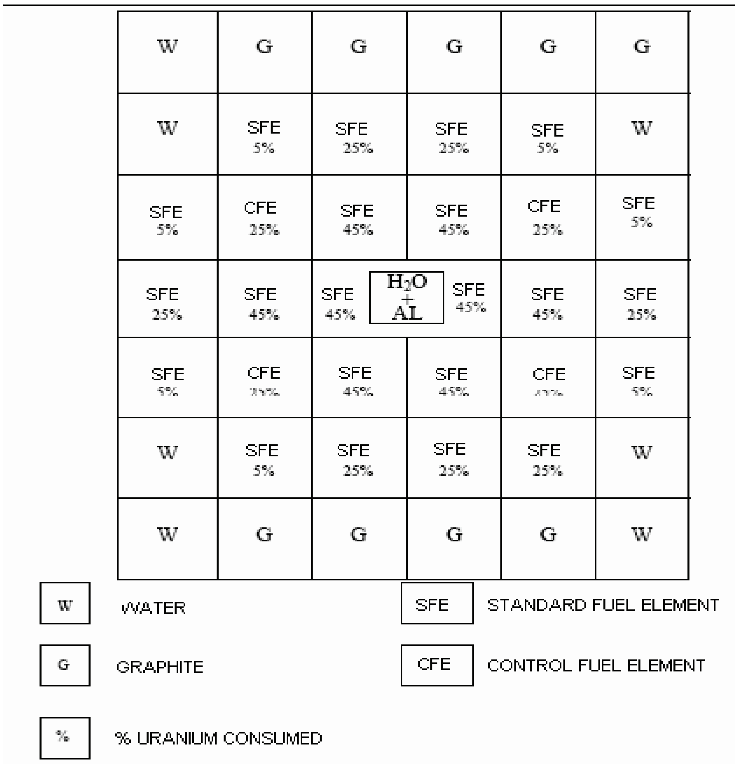

Figure 1: Core Configuration

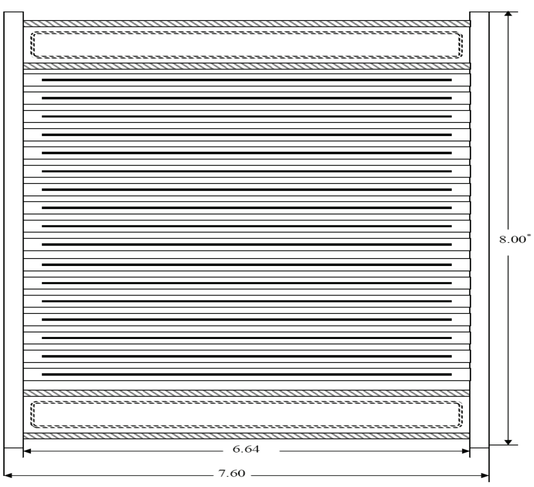

Figure 2: Standard Fuel Element

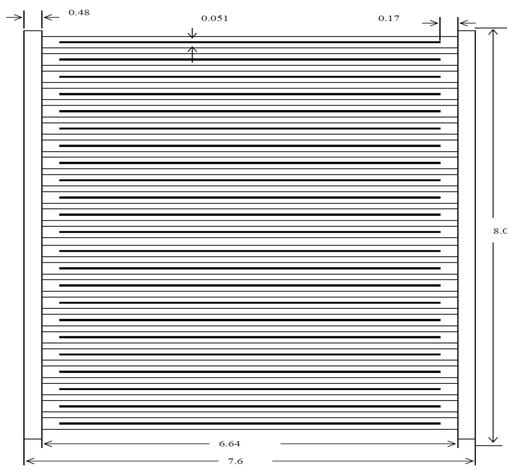

Figure 3: Control Fuel Element

2.1 Kinetic or overpower transients

The Fast RIA (FRIA) transients are initiated by a super prompt ramp positive reactivity addition of \$1.5/0.5 s in the HEU and LEU cores. The Slow RIA (SRIA) consists in a continuous insertion of 9¢ /s in the HEU core and 10 ¢/s in the LEU core. The reactor is assumed to be at an initial operating power of 1W and with full downward cooling flow (not as the benchmark specifications which consider initial upward flow). The safety system is activated when the core power exceeds 12 MW, by inducing a negative reactivity of -\$10 in 0.5 sec within a response delay time of 0.025 s.

2.2 Thermal-hydraulic transients

The flow decay is modelled as an exponential (exp(-t/T)) decrease with a period T equal to 1 sec and 25 s for the Fast LOFA (FLOFA) and the Slow LOFA (SLOFA) cases, respectively. The LOFA transients are initiated at a nominal core power of 12 MW and full core downward cooling flow conditions. The reactor scrams when the flow decay is reduced by 15%, with a response delay time of 0.2s. In the Benchmark specifications, the calculations are stopped when the flow reach 15% of its nominal value. However, in the present case the simulation goes beyond this limit. The Natural Convection Valve (NCV), as modelled in the RELAP5 nodalization, allows a flow reversal and the establishment of passive decay heat removal process by natural circulation flow [13].

2.3 Flow blockage transient

In the following, an attempt to model the behavior of the FA under severe accidents is investigated by considering hypothetical initiating event inducing partial and total flow blockages. Such event constitutes a severe accident for this type of reactor since it may lead to local dryout and eventually to a loss of the FA integrity. For this purpose, each FA is simulated in order to perform a BE approach of the physical phenomenology involved during such transient. A standard fuel element (SFE) channel (Figure 5) was chosen to be subjected to such events.

3 RESULTS AND DISCUSSION

3.1 RIA transients

Typical power behavior following a positive reactivity insertion in the HEU and LEU cores is well predicted by RELAP5/Mod 3.2 calculations. As reported in [10], during the both considered FRIA cases, the power excursion is stopped by the scram shutdown. For illustration, the response of the HEU core is shown in Figure 6. As expected, the power response exhibits exponential rise during the earlier transient instant, which is slightly slowed by prompt Doppler feedback effect. The energy released during the transient did not alter significantly the power course since the coolant temperature rise occurs during the control rods insertion period. In case of SRIA, a beginning of self-limiting power behavior was observed (Figure 7) when the power trend begins to quench under the delayed feedback effect due to the coolant temperature rise before the reactor is scrammed. Differences between the channel codes and the system RELAP5/Mod3.2 code are more emphasized for the coolant temperature response as shown in Figure 8. The outlet temperature as predicted by RELAP5 exhibit a small delayed response due to the inertial effect of the whole cooling loop. This effect is not observed in the results of channel codes calculations where fixed core inlet flow is assumed. During the FRIA case, subcooled Nucleate Boiling (NB) regime occurs for a short period just before the peak power time occurrence. This two-phase flow regime takes place when the cladding temperature is higher than the onset of NB temperature (≈126°C).

The used channel code predict a certain amount of void while the void predicted by RELAP5/Mod 3.2 code remains null even though higher cladding temperature is obtained (up to 160°C). Similar result was obtained by newer RELAP5/Mod 3 version [14]. This may be due to inadequate RELAP5 model in predicting the amount of void under low pressure operating conditions [13].

The higher peak power predicted by RELAP5 could be explained by the absence of void feedback contribution during the power excursion course. For the SRIA, differences between REALP5 and channel codes results are practically inexistent for both power (Figure 7) and temperature responses. The core and the coolant loop interactions are weak since the dynamic of the transient is relatively slow.

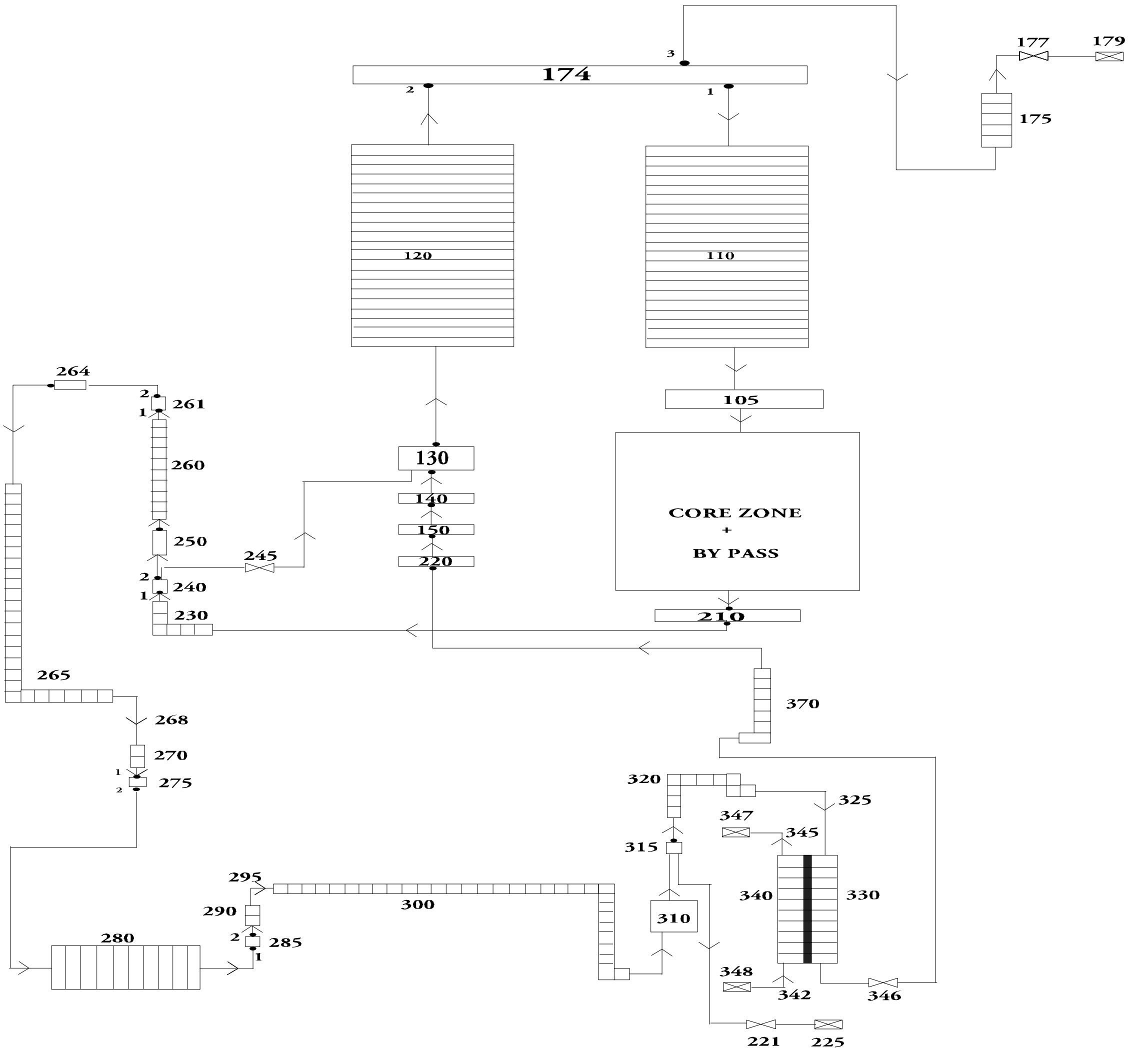

Figure 4: Plant Nodalization

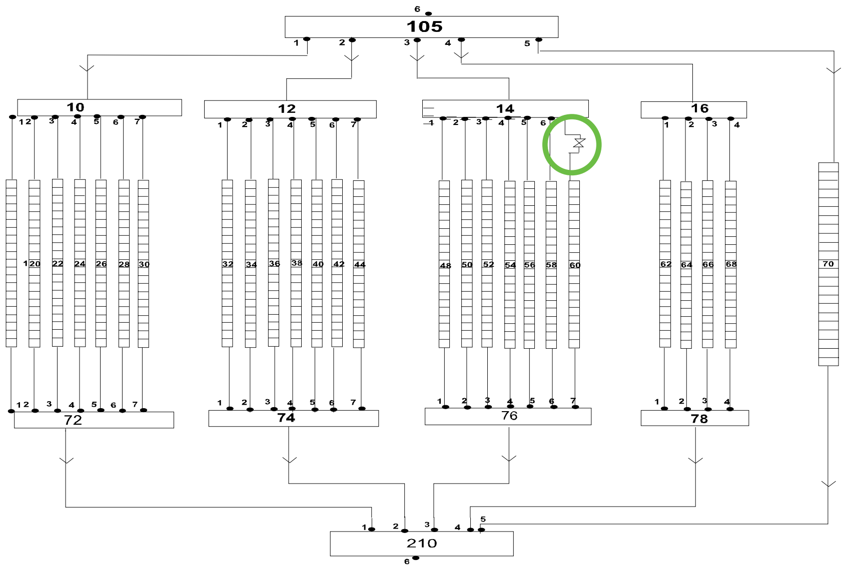

Figure 5: Core nodalization

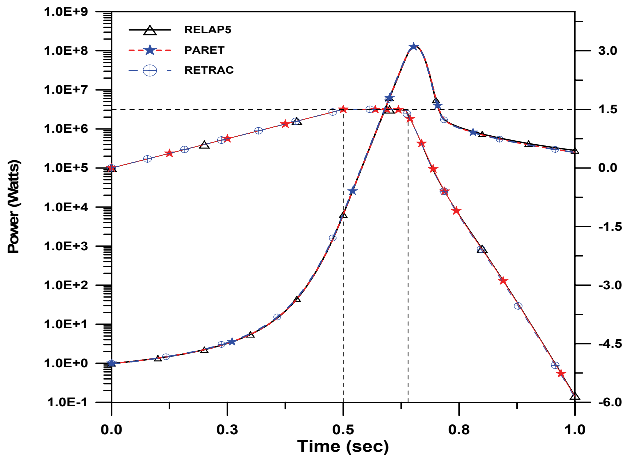

Figure 6: Core power course during FRIA

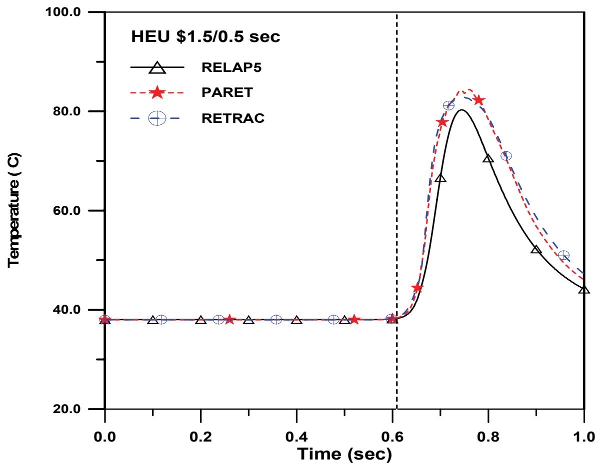

Figure 7: Core power evolutions during SRIA transients.

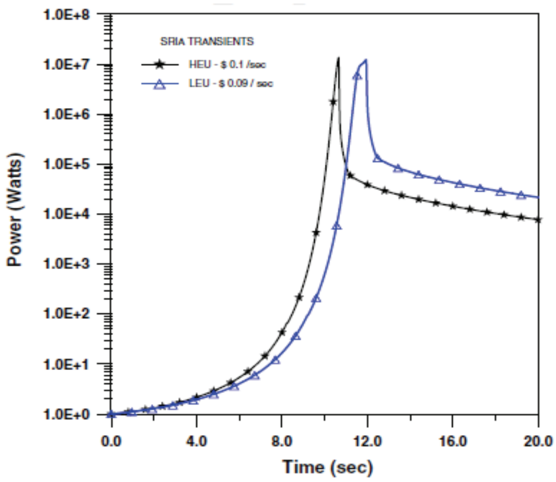

Figure 8 : Outlet coolant temperature FRIA

3.2 LOFA transients

During LOFA, the buoyancy forces, due to the coolant heat up by the decay heat generation, become important at a certain moment of the transient course, in comparison with the decaying pump active forces. A mixed convection flow establishes followed by a flow reversal and natural circulation regime. As specified in [14], the LOFA calculations are terminated when the flow decay reaches 15% of its nominal value. This threshold was adopted due to the inadequacy of almost of the channel computational tools in modeling such phenomena. The results of RELAP5 and two used channels codes for temperature behaviors are reported in Figure 9 and Figure 10. The fuel as well as the coolant temperature exhibits a steady rise due to the degradation of the core cooling as consequence of the flow decay. After the scram of the reactor power, a sharp decrease of core temperature is observed. However, due to the combined effect of constant decay heat and continuous reduction of the core flow rate, the core temperatures exhibit a second rise. The increase is further sustained as the flow regime passes though a laminar flow regime and further to a mixed flow when the NCV is opened. The core temperatures begin to decrease only when the natural circulation flow is fully established. In general, differences between the RELAP5 and channel codes predictions for the LOFA transients are mainly related to the thermal-hydraulic interactions between the core and the coolant loop. These mechanisms could not, in any case, be adequately simulated without considering the effect of the different components of the whole cooling loop.

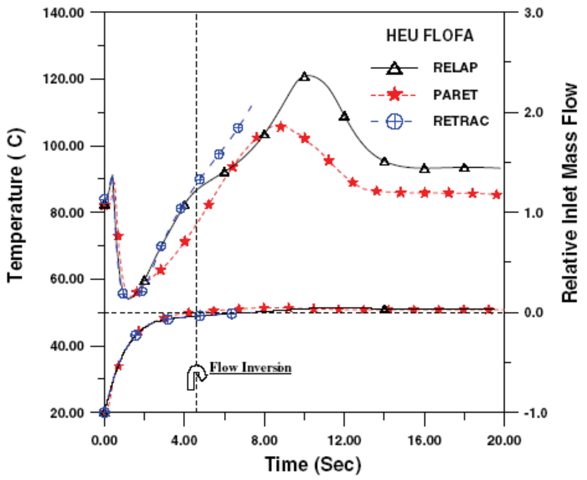

Figure 9: Clad surface temperature and relative mass flow rate during SLOFA transient

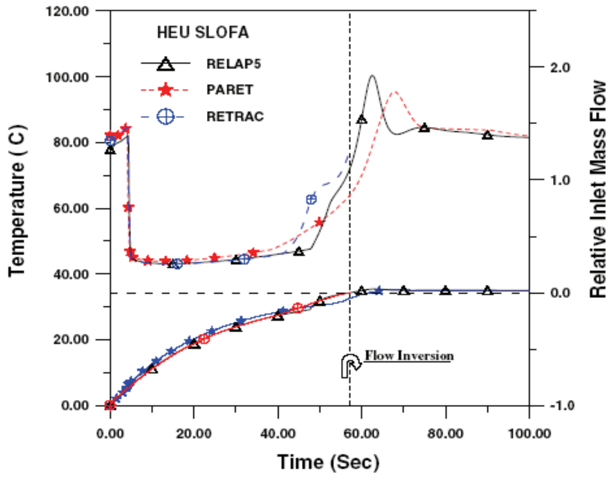

Figure 10: Clad surface temperature and relative inlet mass flow rate during FLOFA transient

3.3 Flow Blockage transients

The analyzed flow blockage cases are divided in two categories. The first one is a partial blockage of an SFE channel of 95% of nominal flow area obstruction. The second one is an extreme scenario where a total blockage of the cooling channel is considered.

3.3.1 Partial blockage

During the first period of the transient, a redistribution of the mass flow in each channel occurs as shown in Figure 11. Accordingly, the fuel and the coolant temperature in the obstructed channel experiences a sharp increase (Figure 12), and a beginning void production (Figure 13). As consequence, due to the strong contribution of feedback effects, the reactor power, as shown in Figure 14, exhibits a self-shutdown behaviour. Later, after 200 s the reactor power becomes enough low and the mass flow rate in the obstructed channel still sufficient to cool down the FA and to stop the void generation. The reactor power continues to reduce with dumped oscillation caused by the moderator feedback effect until end of calculations at 1000s. No FA damage has been observed during this transient.

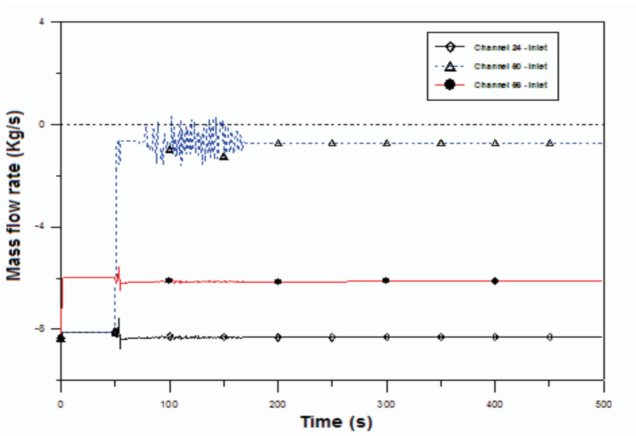

Figure 11: Mass flow in different core channels

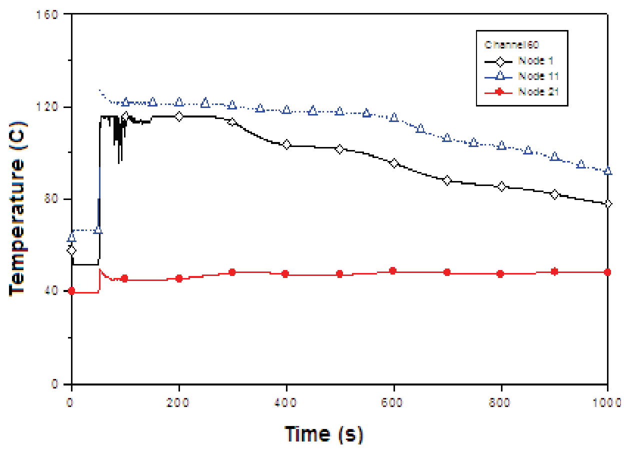

Figure 12: Coolant temperature

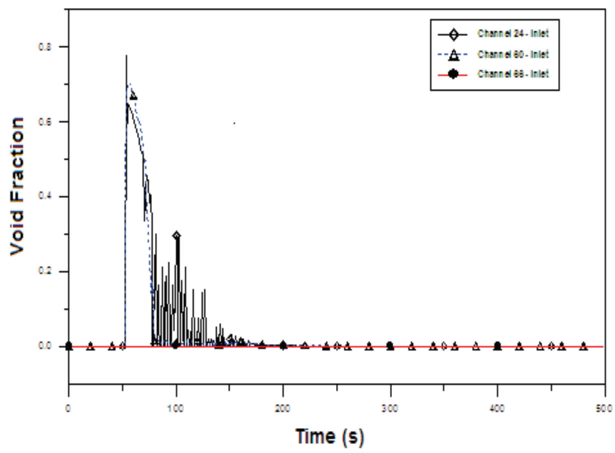

Figure 13: Void fraction in the obstructed channel

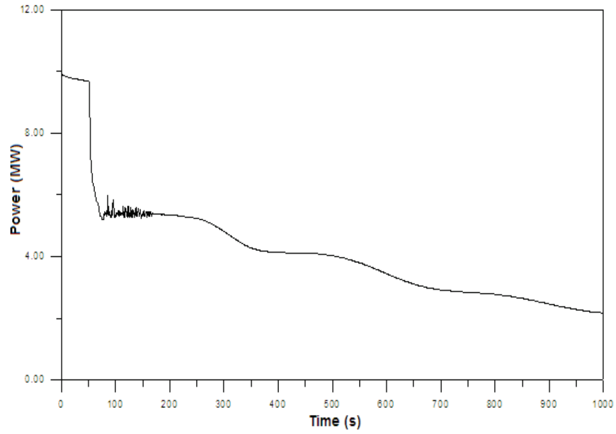

Figure 14: Reactor Power

3.3.2 Total blockage

In this extreme case, the loss of flow occurring in the obstructed channel leads to sharp two phase flow conditions with rapid increase of coolant and fuel temperatures (Figure 15). In these conditions a large vapour production occurs leading to local dryout of the fuel plates. In this case, even though large feedback is involved the reactor power as calculated by the point kinetic model Figure 16) remains enough high to make the cladding temperature reaching the fusion threshold. Calculations are stopped when the cladding temperature reaches its melting point, after only 8 s of the transient start. This result seems to be unrealistic. In fact, the point kinetic model used by RELAP5 code did not consider separately the individual kinetic behavior of each FA. In one hand, the power of the obstructed channel should exhibit self-shutdown behavior varying axially according to the void distribution. On the other hand, the intact FA channels the power should increase due to the decrease of the coolant and fuel temperatures as a consequence of the relative rise of the individually coolant mass flow rate. This constitutes the limit of the point kinetic approaches since the whole reactor core is seen as 0D (zero dimension). Therefore, a BE simulation of such kind of transients require the use of 3D kinetic calculations. This could be done using the current Coupled Codes computational tool technique.

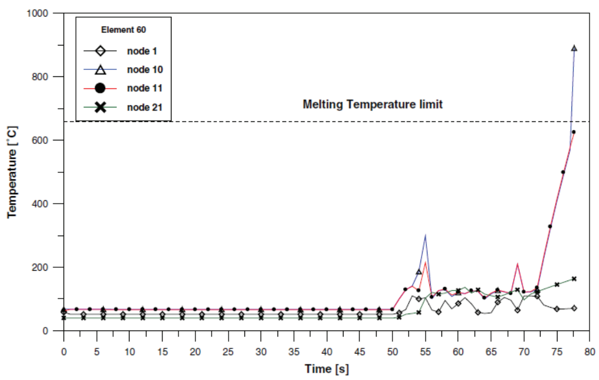

Figure 15: Fuel temperature in the obstructed channel.

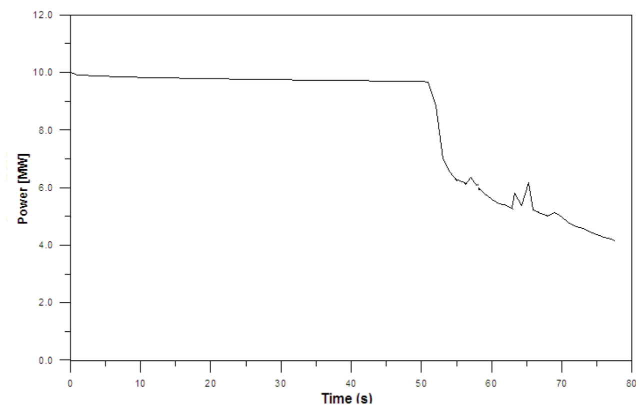

Figure 16: Reactor power

4 CONCLUSION

Increased consideration to safety issues for research reactors has emerged as a consequence of their enlarged commercial exploitation. So far, conservative computational tools were used to perform safety analyses for the design and development of such reactors. Nowadays it becomes necessary to review such limiting tools by using BE calculation methods. The current work constitutes an attempt to apply this technique to the Research Reactors operating conditions. For these purpose the well known IAEA Research Reactors Benchmark problem is considered. In general, for all the considered transients, the obtained results show similar trends with some specified channel codes results. However, as emphasized by the comparative study, the RELAP5 simulation seems to be more realistic since it take into account the interaction between the coolant loop and the core dynamic, especially, during fast power excursion and loss of flow transients.

The demonstration of applicability of qualified BE system codes to RR accident analysis constitutes the key message from this paper: a proper accident analysis technology should be developed for RR that could benefit of the experience available from NPP, considering that the risk level and the cost associated with RR are orders of magnitude lower. Recommendations are provided hereafter distinguishing between potential RR system thermal-hydraulic code users and decision makers in the area.

Recommendations to the users of computational tools are:

- To consider experimental data and to perform code-to-experiment comparison before any code application to prediction relevant to the RR design or safety analysis.
- To demonstrate that any code adopted for design and safety is qualified.
- To consider that any BE code, even though supported by the use of the optimised procedures, produces results that are affected by an unknown error, i.e. uncertainty.

ACRONYMS

0D	Zero Dimension
3D	Three Dimensional
BE	Best Estimate
CFE	Control Fuel Element
FA	Fuel Assembly
HEU	Highly Enriched Uranium
IAEA	International Atomic Energy Agency
LEU	Low Enriched Uranium
LOFA	LOss of Flow Accident
NB	Nucleate Boiling
NCV	Natural Convection Valve
NPP	Nuclear Power Plant
PSA	Probabilistic Safety Assessment
RIA	Reactivity Insertion Accident
RR	Research Reactor
SFE	Standard Fuel Element
F-	Fast
S-	Slow

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