SAFETY APPROACH OF BORAX TYPE ACCIDENTS IN FRENCH RESEARCH REACTORS

Y. CHEGRANI, F. GUPTA, S. PIGNET, V. TIBERI, L. HEULERS
Institut de Radioprotection et de sûreté Nucléaire,
Fontenay-aux-Roses,
France

Abstract

Most of pool type French research reactors are designed to withstand an explosive BORAX accident, defined as a pressure load on the pool walls. The purpose of this paper is to present the approach implemented at IRSN to analyse this accident by linking safety assessment and supporting studies. Examples of recent work on Jules Horowitz Reactor (JHR) and ORPHEE will be presented. Although all aspects of the accident are addressed, we will focus on the first two frames of the transient: the reactivity insertion and the consequences on the core. The first step of the BORAX analysis is to identify the most penalizing plausible reactivity insertion. This means characterising the sequences of events that can induce a reactivity surge and evaluate the worth of such variation. Neutronic computations are then required to quantify the reactivity increase. To comply with the geometrical specificities of research reactors, IRSN chose to use the homemade Monte Carlo code MORET5. The control rod worth calculations on the JHR were in good agreement with the operator results, whereas in ORPHEE, IRSN demonstrated that the beam channels reactivity worth was largely. In both cases the obtained results allowed an interesting dialogue with the operator and were used in the conclusions of the safety assessment. Following the accidental sequence of events, the second stage analysed by IRSN is the power transient occurring in the core and the consequences on the fuel. IRSN applied on JHR a homemade simplified model based on point kinetics and standard thermal balance equations to compute power evolution taking into account the temperatures of the fuel for feedback reactivity. As heat exchange coefficients between cladding and water for such fast transients are unknown, IRSN took the conservative hypothesis of adiabatic heating of the plates. The comparison the JHR power pulse calculation results against SPERT experimental measurements enabled IRSN to be optimistic about the possibility that a slow reactivity insertion would not lead to severe consequences on the core. It also highlighted a lack of knowledge about fast transient physical processes and the need of validated tools if a refined simulation is to be carried out.

1. INTRODUCTION

As technical support of the French Nuclear Safety Authority, the IRSN (Institut de Radioprotection et de Sûreté Nucléaire) performs the safety assessment of French nuclear facilities. All French running research reactors are designed to withstand the consequences of an explosive reactivity transient, which means that ultimately, the walls of the reactor pool would not be damaged by the shockwave, and that the water inventory will remain sufficient.

These last few years, a lot of activity surrounded research reactors in France; in 2007 the Preliminary safety report of the JHR was presented, in 2009 the second decennial safety review of ORPHEE was launched, and more recently in 2010, the JHR proposed its containment design and started the civil engineering work. The implications of the safety assessment work are different when carried out on a running facility compared to a future reactor. However, the actual safety analysis and design robustness assessment methodology are similar.

The Borax transient is the reference accident for the pressure load on the reactor pool. The transient can be subdivided in several consecutive events, but this paper will only describe the supporting studies carried out for the first two steps which are the reactivity insertion and the consequences on the core, since the fraction of molten fuel is a key factor in the steam explosion. Examples will be taken from the recent studies carried out for ORPHEE and JHR.

2. DESIGN AND SAFETY REQUIREMENTS OF FRENCH RESEARCH REACTORS

2.1 Explosive reactivity accident: BORAX transient

Several in-pile experiments have been dedicated to the study of explosive transients induced by a reactivity insertion. The first was the BORAX campaign, which consisted in a series of increasing power transients finally ending with a destructive test. SPERT-I and SPERT-IV have been designed to complete the experimental knowledge after the fatal SL-1 accidental explosive transient. All these cases delivered clues to understand the underlying processes.

Borax type transients are observed in aluminium plate fuel reactors, because of the low melting point of this material (around 660°C). The insertion of reactivity reaching prompt criticality generates a power pulse, with sometimes a power ratio at the peak of up to 10^3 the nominal power. When the power excursion is too brutal or too high, the energy is not properly dissipated by the fuel plates and the coolant, which results in a violent local melting of the aluminium into fine droplets, offering a large contact surface with the still liquid water. This thermo-dynamical interaction is a steam explosion, creating a vapour bubble, expanding through the core channels, bringing additional damage to the intact fuel sub-assemblies and eventually to the pool walls. In some cases the shockwave propels a vertical water jet, acting as a hammer on the roof of the containment hall.

Radioactive materials are released into the pool, and the containment atmosphere. A proper design withstanding such accidents, to prevent radioactive material from leaking out with the water or through the air (because of the water hammer, or the uncovering of the core) appears to be a prime safety requirement.

2.2 Safety requirement and assessment process

In France, this type of accident is taken into account in the design of research reactors. Although it is considered as a “Beyond Design Basis Accident” (BDBA), which occurrences are prevented by robust defence lines, this accident is actually an extension of the “Design Basis Accident” (DBA) domain. Indeed, it is the reference loading for the design of important safety related equipments (containment buildings, pool walls, post-accidental heat removal systems, filtration circuits, etc.); accordingly, these important equipments have functional requirements in order to allow mitigating the Borax transient.

For French research reactors, the design of the pool must guarantee that the core is maintained covered in any situations, including situations following core-damaging accidents. Indeed, the open-air melting of the core is categorised as an excluded major accident; which means that no particular measure is planned to mitigate the consequences to the environment in case of such accident. That is why one of the main goals of BORAX studies is to demonstrate that no water leak on the pool will appear in case of BORAX.

To verify that the design complies with the requirement, IRSN details all the steps of the accidental process, untying the dependencies from the initiating event to the ultimate consequences on the core and the concrete structures. Each step and sub-step of the scenario and its consequences are analysed and the conservatism of the hypotheses is assessed. The steps are:

— The analysis of the reactivity insertion scenarios;
— The analysis of the consequences on core of the reactivity insertion before the steam explosion;
— The analysis of the consequences of the steam explosion (taking into account a state of core defined in the previous step), which leads to a pressure load on the pool walls;

— The analysis of the ability of the pool structure to withstand the pressure load defined previously.

When the demonstration is supported by complex numerical simulations, the analysis of the methods and premises can be carried out by homemade computations. In this paper, we describe examples of computations performed in the scope of the two first steps.

3. REACTIVITY INSERTION

3.1 Choice of the event taken into account for consequences evaluation

As previously mentioned, Borax type accidents are initiated by supercritical reactivity insertions. Several aspects of the reactivity insertion are analysed by IRSN; the proposed scenarios are assessed on each level. First, the reactivity transient itself if investigated, to verify that the reactivity source is correctly identified, and that it covers for all other reactivity events. If some events should lead to more severe reactivity injection, IRSN assess that robust lines of defence allow excluding the event. For example, on the JHR, the reference reactivity insertion is the ejection of one control rod at its fastest speed; IRSN concluded that this was correct after assessing the other sources of reactivity available in the core. But IRSN verified also that the design would exclude unforeseen events that could trigger the ejection of several control rods.

Once the envelope scenario is assessed, the amplitude and sometimes the timing of the reactivity insertion must also be verified. Concerning amplitude of reactivity insertions home made computations have been performed for ORPHEE and JHR reactors (see 3.2). Concerning duration of reactivity insertions, the hypothesis for ORPHEE was not questionable since a step wise insertion is taken into account, for JHR this point was analysed and discussed, computations described in 3.4 allow evaluating the influence of this duration on the consequences on core of the accident.

3.2 Quantification of the reactivity insertion

In order to assess the amount of reactivity taken into account for the evaluation of the pressure loading that the pool walls should withstand in case of BORAX, static core neutronics calculations in normal and abnormal states are performed in IRSN. Two examples of such calculations are described here. The first one has been done in the frame of ORPHEE reactor's second decennial safety review, the second one in the frame JHR safety demonstration process.

3.2.1 MORET code

Standard power reactor oriented deterministic codes cannot easily be applied to fully apprehend the specificities of research reactors, because of their large spectrum of designs. On the contrary, Monte-Carlo type codes present interesting advantages. This is why, to be able to perform such computations, the IRSN has started to extend the use of the homemade criticality Monte-Carlo code, MORET5 [1]. It is both a multi-group and continuous energy neutronic Monte-Carlo code. It mainly allows to estimate the neutron effective multiplication factor, fluxes, reaction rates and leakage for any three dimensional systems. Initially designed to meet the needs of criticality studies, it has been used in most studies of nuclear fuel cycle facilities and fissile material transport in France. Neutron simulation in MORET has been optimized to reach a good compromise between CPU resources and accuracy, making the code suitable for project studies.

In order to fully extend the capabilities of this code to answer reactor safety specific issues, developments have been implemented in the 5.A version of the code. This version allows computing kinetic parameters that are useful for reactivity accident calculations, and is currently being tested and validated. Developments are also planned to output reactor parameters more easily by using predefined scores: power distribution, residual power, Shannon entropy indicator, Doppler Effect, mass balances…

Currently, MORET has an extensive experimental validation database for the multi-group version. It is made up of more than 2300 benchmarks models, mainly issued from the ICSBEP Handbook, covering a broad variety of criticality configurations in terms of fissile medium, moderator, reflector and neutron spectrum. The validation of MORET 5.A.1 is currently in progress to extend it to reactor applications and to its new continuous-energy capabilities, on the basis of the reactor benchmarks presented in IRPHEE handbook.

Both studies presented in this paper only required the standard core function of MORET, which is the k-eff calculation. Reactivity worth evaluations are indeed series of criticality calculations.

3.2.2 ORPHEE case

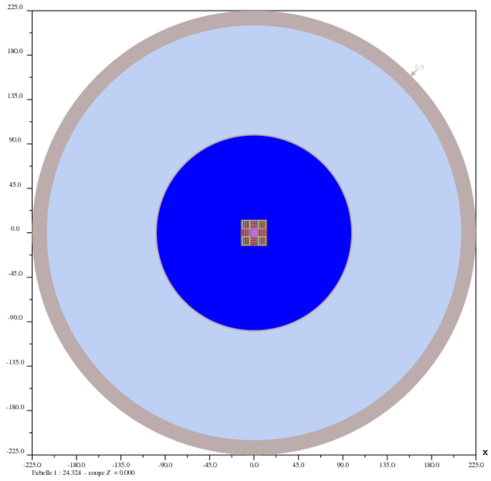
The first reactor-case neutronic calculation presented here is of the ORPHEE [2] reactor during its second decennial safety review. The pool of this reactor was designed to withstand a borax transient. ORPHEE is a neutron source type research reactor, where a heavy water reflector is equipped with 2 cold hydrogen sources and 9 neutron beam channels leading to experimental facilities. The safety report identified the brutal rupture of these channels as the most probable initiating event of a borax transient. Two scenarios were identified:

Instantaneous flooding of all 9 channels and hydrogen sources, called the “flooding effect” and considered as the reference case

Disappearance of all 9 channel structures, called the “structure effect”, considered as the conservative - or sensitivity - case.

The aluminium beam noses filled with helium are “neutron-transparent” structures placed near the maximum flux plane, enabling neutrons to leak out of the reactor into the beams. These hypothetical scenarios simulated a brutal insertion of reactivity by creating an excess of moderation and reflection. These two initiating events were identified and calculated during the design phase in the late 80’s. Flooding effect and structure effect were evaluated at 1.5$ and 1.2$ respectively.

The core and its reflector have been fully represented in MORET5 3D geometry (FIG. 1, 2). The calculated critical heights of the control rods in the nominal case were in very good agreement with the in-core measurements available in the safety report. The media used for the channels were then substituted to simulate the given borax initiating scenarios.



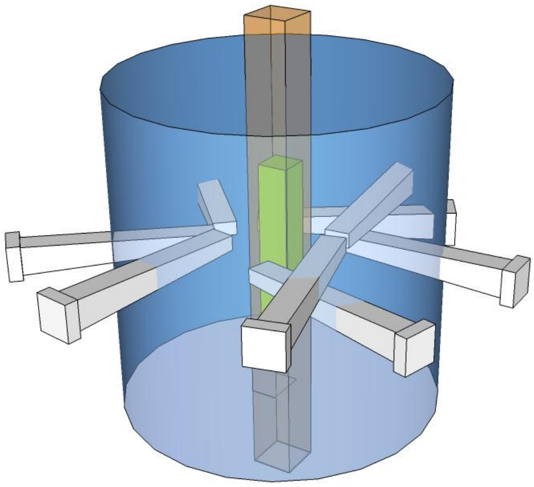


FIG.1: top view of ORPHEE as modelled in MORET5 without beam nozzles (left).

FIG.2: 3D view of the core and the reflector with the beam nozzles (right).

Much higher reactivities were calculated with MORET5, in total disagreement with the design values for void and structure effect. Technical discussions ensued, and were concluded by 3D Monte Carlo calculations carried out by the CEA, providing reactivity values very similar to MORET5 results. The table (TABLE 1) below summarises and compares design diffusion calculation and up-to-date Monte-Carlo calculations.

TABLE 1: Comparison of calculated reactivity worth of various equipments in ORPHEE

Equipment	Safety report calculations (80's) in $	Up-to-date calculations in $
Flooding of Cold Source 1	0.21	0.25
Flooding of Cold Source 2	0.19	0.15
Vaporisation of H2 in Cold Source 1	0.17	0.12
Vaporisation of H2 in Cold Source 2	0.11	0.11
Flooding of 9 channels	0.45	1.65
Total of flooding and vaporisation effects	**1.46**	**2.66**
Structure disappearance of 9 channels	1.22	1.62
Total of flooding and structure effects	**2.9**	**4.3**

The impact of IRSN calculations with MORET5 on safety were measurable, since the periodic examinations and replacement schedule of the neutron beam noses will be modified and tightened to reduce the risk of failure. More over, the reference reactivity insertion considered for a borax accident will be set to 3$ with no sensitivity case, in ORPHEE's safety report.

3.2.3 JHR case

The second reactor case concerns neutronic studies for design of the JHR [3] pool. The considered accident for the dimensioning is the BORAX accident whose initiating event, in this case, is a control rod ejection. The study aimed at evaluating the conservatism of the reactivity insertion caused by a control rod ejection considered by the CEA (1990 pcm) in the frame of the borax accident analysis for this reactor. This analysis has also been carried out with the Monte-Carlo code MORET5 (FIG. 3). In the first step of the study, the capability of the code to properly describe the JHR has been tested by comparison against the results of reactivity calculations performed with the reference codes APOLLO2 and TRIPOLI4. The second step consisted in calculating the control rod worth for different rods in the core for a configuration provided by the CEA. Finally several sensitivity studies have been carried out to evaluate the conservatism of this configuration.

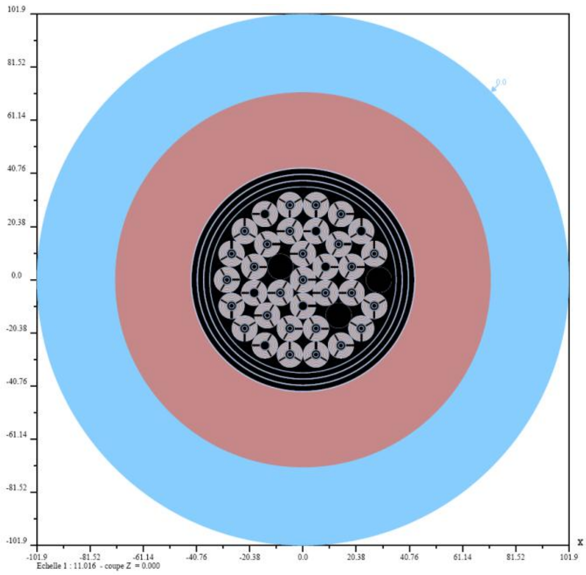


FIG. 3. Top view of JHR reactor modelled in MORET5

The comparison between the obtained results and CEA results showed few discrepancies and raised some questions in particular about nuclear data. However, even if an exhaustive demonstration of the conservatism of the value considered by the CEA is not possible, the MORET computations allow considering the order of magnitude of 2000 pcm as acceptable for the design studies for the BORAX accident.

4. EVALUATION OF THE CONSEQUENCES ON THE CORE

4.1 Energy source term for steam explosion

After the characterisation of the reactivity insertion, the next step is to evaluate the state of the core before the steam explosion, in order to calculate afterwards the pressure load on pools walls. For that purpose a de-coupling value is usually chosen which represents the consequences on the core of the reactivity insertion and which is used to compute the pressure wave induced by the steam explosion. This value is the energy stored in the fuel. The US experimental runs set the energy of an explosive borax transient at 135MJ. Historically, this same value has been used, with some adjustments, to design the pools of the French research reactor. However, a new realistic approach has been presented by the CEA for the design of its next reactor, the JHR. This new method is based on realistic scenarios and best-estimated transient results.

To assess this part of the BORAX demonstration, IRSN identified sensitive phenomena for which R&D efforts are necessary before they can be included and simulated in precise calculations in the frame of a safety demonstration. This is the case of cladding-to-water heat exchanges during power pulse. Considering this lack of knowledge, IRSN uses simplified models to link the reactivity insertion inside JHR core, leading to a power pulse to probable consequences on fuel thanks to SPERT experimental results.

4.2 Models and main assumptions

The tool that has been used is an updated version of an older program, developed for the first RIA calculations on the JHR core, in 2007. For easier further use, we will refer to this point kinetic model as MIRAX. This simplified model is used to gather data on the second step of the Borax transient. Several simplifications and assumptions have been considered in the implementation of MIRAX.

In MIRAX, a core can basically be modelled as a layered flat surface composed of two materials -the fuel meat and the aluminium based cladding- in contact with a given volume of water. The plate is geometrically defined by the thickness of each layer, and the total heat exchange surface. Each material is defined by basic properties, such as conductivity, heat capacity, melting temperature, etc. Three physical models are implemented, the point kinetics neutronic, the heat conduction in the plates and the heat exchanges with the water.

4.2.1 Point kinetics

Regarding the neutronics, MIRAX is a fairly classic point kinetics model, with 6 groups of delayed neutrons. The use of point kinetics for the simulation of power transients in research reactors is permitted by the fact that the considered cores are of very small size, with negligible spatial effects. Doppler feed-back coefficient is provided to be used in a standard temperature dependant function. Water void coefficients can also be provided, but require the use of a questionable heat transfer model, this issue will be discussed further. The main result of this model is the power evolution.

4.2.2 Heat conduction

The second implemented model is the heat conduction through the plates. This model converts the generated power into a temperature variation, transmitted through the meshed material. The main result of this model is the temperature of the fuel meat for the Doppler reactivity calculation, and the wall temperature of the cladding, for the heat exchange model.

4.2.3 Clad-to-coolant heat exchange

This third model has been implemented to compute the moderator reactivity feedback. The heat flux transmitted from the cladding surface to the water is tabulated as a function of the temperature gradient between the surface and the saturation temperature of the water. The heat stored by the water bulk increases its temperature, which is used to calculate the water density evolution, which is the major contributor to the moderator or void feed back reactivity.

However, this model is implemented with very coarse simplifications. The main one is the absence of flow; the water is modelled as a bulk volume, storing energy. During a power transient calculated in MIRAX, the water is heated until evaporation. This is considered acceptable because of the kinetics of the transient itself - the power surge usually lasts less than 100 ms – in pool type reactors with low flow rates: the effective volume of water impacted by the transient is the actual volume in the water channel.

The other issue regarding this model originates in the physics of the phenomenon itself. Actually, knowledge about heat transfer coefficients in fast transients is scarce, and the transition boiling processes (nucleate boiling, critical flux and film boiling) are yet to be understood. The few measurements found in the literature do not cover the full range of flow rates, initial conditions, and geometries of the studied research reactors. For example, the JHR reactor is pressurised to 15 bars, with a flow rate of around $1 m^3 \cdot s^{-1}$ along aluminium curved fuel plates, whereas most experiments were carried either on flat type plates in pool reactors with natural convection or in analytical experiments in quasi static regime on zircalloy tubes.

4.3 JHR power excursion calculations

4.3.1 Description of the JHR model

The JHR core has been modelled in MIRAX using the most up-to-date data regarding the geometry. Neutronic data, such as kinetic parameters, feedback coefficients and reactivity insertion dynamics were extracted from the reactor's preliminary safety report and attached technical documents.

Regarding clad-to-water heat transfers, the previously mentioned lack of JHR-relevant data was resolved by excluding any heat exchange. The fuel plate was heated adiabatically. The conservatism of such a strong assumption had to be verified, because JHR fuel enrichment is quite low, and the Doppler Effect could be favoured by the excessive heating of the fuel. That is why comparison of "adiabatic" results with results obtained using various heat transfer functions have been performed. Power peaks obtained without heat transfer were larger than the others, demonstrating that the "adiabatic" hypothesis leads to conservative results.

The base model was globally validated against a calculation carried out by the CEA in 2007, with a very similar model. The "adiabatic" hypothesis was used as well at the time, for the same reasons; the only differences, although minor, lie in the thermal data of the fuel materials, and neutronic data. Results obtained with equivalent models are in very good agreement.

4.3.2 Influence of control rod ejection speed on power excursion

During the safety review process, the CEA studied the possibility to decrease the speed of the control rod ejection, in order to lower the amplitude of the power peak and the energy deposit, and consequently the fuel molten fraction, to finally reduce the risk of fuel-coolant interaction. The first control rods were designed to be ejected in no less than 90 ms, the proposed design would slow the motion down to 300 ms for a complete ejection. This modification was presented as an improvement and a strong prevention measure against borax type transients.

A slow down of the reactivity insertion kinetics will indeed lead to a milder transient because the balance of reactivity (inserted – feedback) will be lower. Yet, it was still unclear, considering all the uncertainties, how low the consequences would be. As demonstrated in the figure below, the volumetric power of the "300 ms" scenario is indeed of lesser amplitude than the "90 ms" scenario.

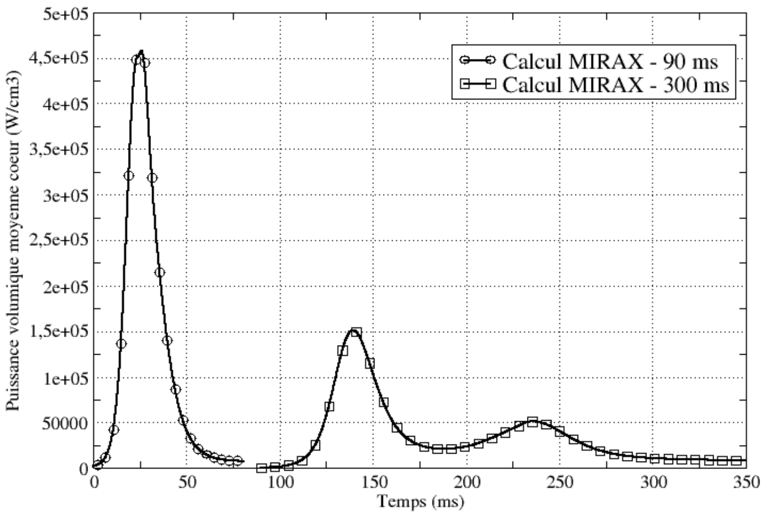


FIG. 4: JHR power transients calculated with MIRAX for two scenarios: 90 ms and 300 ms.

4.4 Discussion on consequences on core degradation thanks to SPERT experiments

The inherent limitations of the calculation tools used in the safety assessment do not allow for further conclusions about the state of core degradation in any of theses cases. This is why IRSN chose to include the knowledge gathered in past experiments, and most of all SPERT campaigns. A few cases were selected and analysed, to highlight a link between power transient and actual fuel degradation.

4.4.1 Presentation of relevant SPERT transients

Tests were selected from the first and last campaigns: SPERT-I [4] and SPERT-IV [5]. Theses two reactors were very similar in shape, layout, and fuel parameters. They are relevant to our safety assessment because SPERT-I and -IV are both pool type reactor, light water moderated and cooled, build with high enriched aluminium based plate type fuel. The difference between both reactors is that the first was used in explosive borax tests, whereas the second was used for self limiting power excursion.

The measurement devices that equipped the cores were limited. Available reference results mostly contain a core power trace and a temperature evolution in one point of the central subassembly.

In each campaign, we selected the most penalising tests, in terms of inserted reactivity and power peak amplitude. SPERT-I -54 was the ultimate destructive transient. This is considered as the worst case reference scenario. In the last campaign, we selected SPERT-IV-16. Although it registered the higher power peak, no fuel or clad melting was observed. All SPERT-IV transients were limited only by the moderator density effect, appearing with dilatation or vaporisation of the water. This feedback reactivity effect is primordial to the safety of these reactors that have no Doppler feedback effect due to the high enrichment of the fuel (93%).

4.4.2 Interpretation of JHR calculations

A simple comparison between MIRAX results and SPERT measurements demonstrates that in the “90 ms” scenario, the JHR behaves similarly to the explosive SPERT-I-54, whereas in the “300 ms” case the JHR seems to mimic the self limiting SPERT-IV-16 power excursion (see FIG. 5).

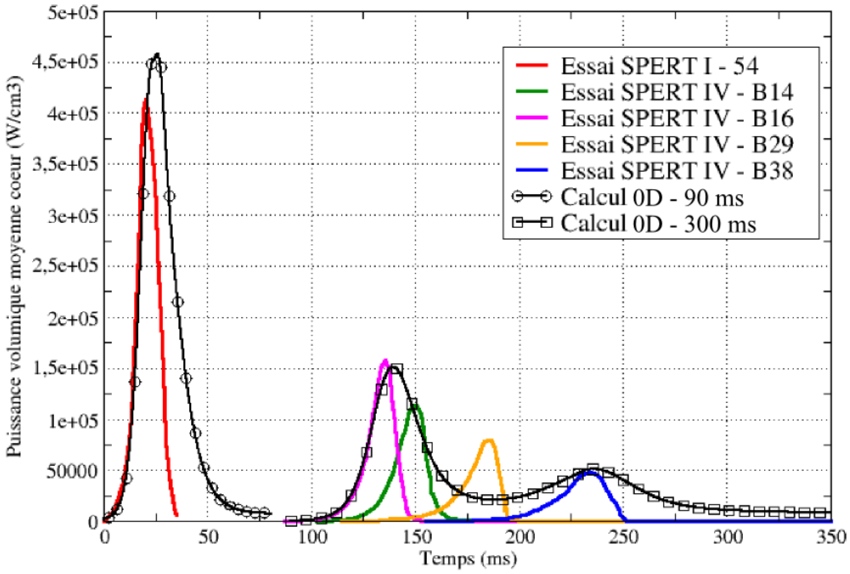


FIG. 5: comparison of calculated JHR transients and measured SPERT power traces.

It means that supposing the JHR core is consistent with SPERT ones, the pulse of 90 ms should lead to extensive degradation of core, whereas for a 300 ms type pulse fusion of fuel would be very limited.

All uncertainties aside, and considering the fact that MIRAX results are conservative – mainly because the void reactivity is not taken into account – this shows that the slow down of the control rod ejection from 90 ms to 300 ms is indeed a efficient preventive action against explosive borax transients.

5. CONCLUSIONS

Borax transients analysed are a key stage in the safety assessment of French research reactors. They define the reference pressure load on the walls of the reactor pool, which must withstand the transient. Safety assessment is based on a methodical expertise of the safety documents, cross-checking with all relevant available feed-back, evaluation the robustness and coherence of the safety demonstration. Supporting technical studies have been performed in order to investigate the calculation methods used by the operator, using home-made tools to test several hypotheses and assess the robustness of the method. Monte Carlo calculations on ORPHEE reactor showed inconsistencies in the safety report, which were used in the assessment to enhance the need of preventive measures. In the frame of the JHR safety assessment process, MORET calculations allowed confirming the evaluations of the control rods worth performed by the operator. A second type of calculation was illustrated in the JHR assessment of the power transient amplitude. Various hypotheses were tested in a point kinetics model to try and estimate the consequences on the core of a control rod ejection. Results showed that the slow down of the reactivity insertion guaranteed by design would benefit the safety of the JHR. On the other hand, this study highlighted to lack of progress in the knowledge of the very particular physical processes composing the borax transient.

Whether the conclusions of the safety assessment are in agreement with the operator, or raise doubts, the support calculations help IRSN to deeply understand the phenomena and engage rich and efficient technical discussions with the utilities. This is a step forward in the improvement of the safety process.

REFERENCES

[1] *First validation of the new continuous energy version of the MORET5 Monte Carlo code*, J.

Miss et al., Physor 2008, Interlaken, Switzerland, September 14-19, 2008.
[2] *Reactivity insertions for the Borax accident in ORPHEE research reactor*. Y. Chegrani et al., (IRSN, France), IGORR 2010. http://www.igorr.com/scripts/home/publigen/content/templates/Show.asp?P=834&L=EN
[3] *The Jules Horowitz Reactor, a new Material Testing Reactor in Europe*, D. Iracane (CEA, France) (IGORR 2005)
[4] IDO/16883: "Report of the SPERT-I destructive test program on an aluminium, plate-type, water-moderated reactor", R. W. Miller
[5] IAEA CRP: "SPERT-IV-D/12/25 Benchmark Specification", S.E. Day