SEVERE ACCIDENT ANALYSES: A HISTORICAL REVIEW FROM THE VERY EARLY DAYS TO THE NEAR-TERM FUTURE

The high complexity inherent to reactor severe accidents has highly conditioned from the beginning of the nuclear age the capability to reliably model these scenarios. Nonetheless, noticeable accomplishments have been made and nowadays fully integrated computer codes are available. The last generation of these analytical codes is numerically robust, fairly extensive in the amount of phenomena presently considered and largely validated to the extent possible. This being said, there is still room for meaningful improvement and uncertainty quantification (UQ) might play a significant role in it. This paper briefly walks the reader along all these topics and discusses that time has come for BEPU (Best Estimate Plus Uncertainty) application in the severe accident domain.

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ANÁLISIS DE ACCIDENTES SEVEROS: REVISIÓN HISTÓRICA Y PERSPECTIVAS

La enorme complejidad inherente a los accidentes severos en reactor ha condicionado la capacidad para su simulación desde el comienzo de la era nuclear. A pesar de ello, se han logrado avances encomiables y actualmente se dispone de códigos integrados de gran capacidad. La última generación de estas herramientas analíticas es numéricamente robusta, contempla un conjunto muy amplio de distintos tipos de fenómenos y ha sido extensivamente validada. Ello no excluye que exista aún recorrido para mejoras significativas y la cuantificación de incertidumbres podría jugar un papel determinante en esta labor. El presente artículo resume brevemente estos tópicos y discute la oportunidad y los beneficios esperables de la aplicación de las técnicas de análisis de incertidumbres en el dominio de los accidentes severos.

INTRODUCTION

Severe accident codes can be seen as the capitalization of severe accidents knowledge. Their high complexity comes from the intrinsic nature of these events. Thermal, fluid, mechanical, chemical and physical phenomena take place at the same time and strongly interact with each other in a scenario in which boundary conditions vary over a broad range and often become extreme. Besides, these events extend for days and even weeks, like in the Fukushima accident, which turns out to be an outstanding challenge for codes that have to integrate over long periods in which very fast and slow phenomena may be occurring at the same time. Even further, severe accidents are not restricted to a limited region of the plant, they usually encompass the entire plant and all the physical scales in which phenomena develop, from the micro to the macro one; in other words, the only way reasonable accuracy can be achieved in the mathematical integration is by meshing the entire spatial domain. Lastly, safety systems can drastically affect the accident progression, so that their performance should be also accounted for in codes under all possible conditions (human action included). As a consequence, major hypotheses, assumptions and approximations are in the essence of severe accident codes, either in their approach and/or in the specific models.

This paper describes the evolution of methods used for severe accident analyses from the early days to present. A review of historical milestones in severe accident simulation is followed by an illustration of their current predictive capability by referring to major international benchmarking in reactor-like scenarios. Finally, uncertainty quantification is discussed as the next necessary step in severe accident analysis. The scope of the paper is focused on reactor scenario applications of integral tools, with a strong flavour of USA developments when describing the historical records of severe accident analyses.

A GLIMPSE INTO HISTORY

WASH-3 ("Summary report of reactor safeguards committee", 1950) was one of the first documents in which a serious accident was postulated [1]. By assuming fuel melting, failure of the primary pressure barrier and an uncontrolled release of radionuclides to the environment, an exclusion area was calculated through a simple expression just dependent on the thermal reactor power (P, kW_t):

$$R = 1.6 \cdot 10^{-2} \cdot p^{1/2}$$

Namely, a 3000 MW_t reactor would require an exclusion area of about 30 km.

The WASH-740 report (1957) on "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants" [2], was the first report on the consequences of a major accident at civilian reactors. It addressed the "worst-case" accident scenario (a core meltdown with a 50% release of the fission product inventory to the atmosphere with no credit to the containment) and estimated very adverse consequences (3400 early fatalities, 43000 acute injuries and $7B, for a 500 MW_t reactor).

The 10 CFR 100 ("Reactor Site Criteria") [3] (1962), was based on the design-basis loss of coolant accident (invariably initiated by the reactor-coolant system pipe break that would yield the highest containment pressure). It calculated the distances for the exclusion area, the low-population zone and to the nearest population center of about 25000 people, as a function of the reactor thermal power (Figure 1). These results adopted prescriptive assumptions [5]: a large fraction of core melted; 100% of the noble gas fission products, 50% of the halogen inventory (50% of which would remain available for release from containment), and 1% of the particulates, immediately released to the containment atmosphere following the pipe break; and a leaking containment (0.1%/day). These estimates were considered the balance between the "worst-case" approach and the "fail-proof ESFs"

In reviewing WASH-740, it was recommended to focus on enhancing primary system integrity and measures to ensure emergency core cooling (ECC) to reduce the LOCA probability. At the time the ECC criteria were derived (Appendix A & K of 10 CFR 50 [6,7]), computer codes had drawbacks for simulating the complex phenomena associated with large LOCAs, so that some conservatism (i.e., 20% higher decay heat rate, no nucleate boiling during blowdown, pessimistic assumptions regarding coolant delivery to the lower plenum, etc.) and prescriptive heat transfer correlations (the Baker-Just correlation for heating rate due to oxidation) were assumed.

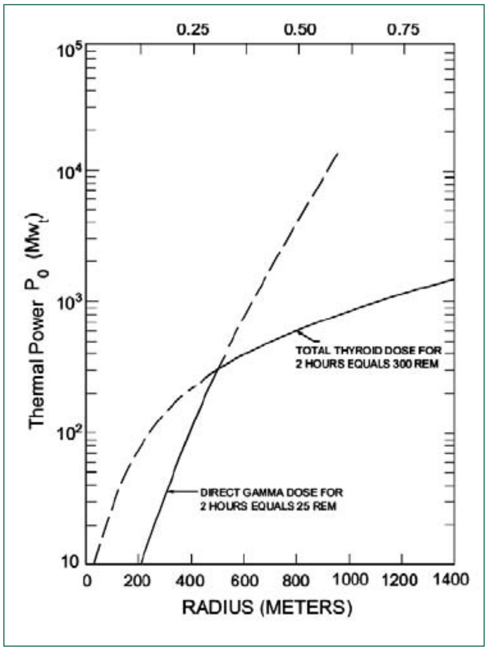

Figure 1. TID-14844 exclusion radii [4].

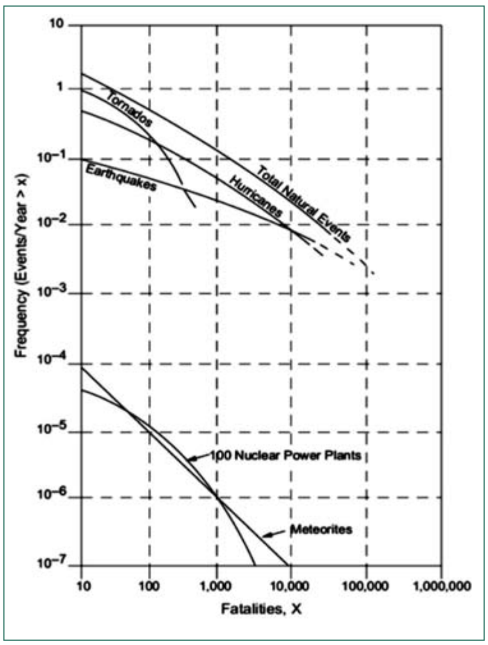

Figure 2. Frequency of natural events involving facilities [4].

Once the ECCS design criteria had been set, a comprehensive assessment of reactor safety was demanded and, then, attention was largely focused on severe accidents. The Reactor Safety Study (1975), known as WASH-1400 [8], intended to get realistic estimates of the potential effects of LWR accidents on public health and safety in a systematic way. This was the first attempt to adopt a probabilistic approach based on fault trees to estimate component and system failures frequencies and event trees to delineate possible sequences leading to core meltdown and radionuclide release. At the time, an integrated tool for the calculation of accident progression and source term did not exist. When data were available, a model for the phenomenon was included, but when data were lacking the phenomenon was just omitted. As a result, the source terms estimated reflected uncertainties and poor understanding of applicable phenomena.

A number of meaningful conclusions were settled: nuclear power plant (NPP) severe accidents have a negligible contribution to public risk, although NPP risks are mainly dominated by severe accidents; a huge number of accident sequences might lead to a core meltdown and the resulting core melt frequencies were notably higher than previously estimated (~$5{\cdot}10^{-5}$/r-y); and, containment integrity plays a key role of in determining severe accident consequences, although different containment types withstand severe accident loads differently. Figure 2 is a glimpse of the results of this first probabilistic analysis.

What definitively changed the perception of severe accidents as "incredible events" was the accident at Unit 2 of the Three Mile Island (TMI-2) NPP on March 28, 1979. It made safety research focus on severe accidents to better understand the physical phenomena and develop models capable of predicting their evolution, particularly in terms of radionuclide releases; no less important, attention was also paid to data sources to identify relevant severe accident sequences for various classes of reactors. The results of this vast effort were integrated years later (1990) in a major Probabilistic Risk Assessment (PRA): "An Assessment for Five U.S. Nuclear Power Plants" (NUREG-1150) [9]. At about the same time the Industry Degraded COre Rulemaking (IDCOR) Program was launched, whose main purpose was to develop models for assessing the risks of severe accidents.

Two major milestones occurred in between TMI-2 (1979) and the publication of NUREG-1150 (1990): the review of the assumptions, procedures and available data to

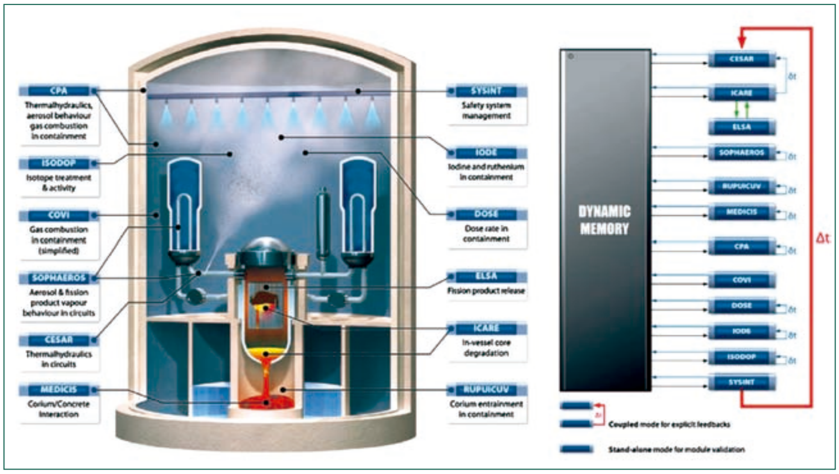

Figure 3. ASTECs scheme [12].

predict fission product behaviour (NUREG-0772 [10]); and the development of a coupled suite of separate effect codes that allowed calculating complete accident sequences (The Source Term Code Package, STCP [11]). NUREG-0772 concluded that CsI was the dominant form of iodine and stated its potential retention within the vessel or containment (vs. the one of I_2) and highlighted the potential of containment engineering safety features to effectively deplete a good fraction of airborne in-containment source term; however, the considerations of integral effects in accident sequences was rather limited. As for STCP, it was a key milestone in deterministic severe accident analysis, but it had significant shortcomings; particularly, the data transfer and feedback effects between different codes in the suite.

In NUREG-1150 the focus was put on four major variables: the frequencies of core damage accidents from internal and external events (for 2 of the plants); the performance of containment structures under severe accident loadings; the potential magnitude of radionuclide releases and offsite consequences of such accidents; and the overall risk (the product of accident frequencies and consequences). As for the methods used for estimating accident progression and radioactive material transport, an extensive assessment of the available severe accident experimental and calculational data bases was performed. Among the computer codes used were: the STCP, which was the most widely used at the time; and to a notably lesser extent CONTAIN, MELCOR and MELPROG. Beyond the use of computer codes, engineering judgement expert elicitation was often necessary.

The analytical tools cited above might be grouped into three families according to their scope and structure: system-specific codes, like CONTAIN and MELPROG, which track containment and in-vessel events, respectively; coupled codes (STCP), which articulate several modules in such a way that different aspects of the accident are calculated sequentially and the results of some codes become initial and boundary conditions for others, but with no or limited feedback; and integrated codes, in which all the related aspects of an accident are calculated simultaneously in the same time step (i.e., MELCOR or MAAP). Even though these codes were certainly useful to get a detailed understanding of accident phenomena and how different phenomena interact with each other, their use in NUREG-1150 was limited. The reasons for this were: their inability to analyze a very wide range of scenarios with diverse boundary conditions in a timely and cost-effective manner; their only partial validation against experiments; and their lack of some key models for the progression of the accident.

The estimate of radioactive material transport was a part of the risk analysis in which this limited use of computational tools was paradigmatic. Instead of mechanistic models, parametric models were developed which described the source terms as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a wide range of projected accidents and nine classes of radionuclides. Both the equations and the parameters were derived through detailed examination of the results of STCP analyses of selected accidents.

THE AGE OF THE INTEGRATED CODES

A vast experimental and analytical research has been accomplished in the last three decades on severe accidents. The result has been a continuous improvement of codes by enhancing models already implemented, by including new models of phenomena and/or plant components, by updating numerics so that model integration becomes more efficient and accurate, by extending code validation as much as feasible to make them more robust and physically sound and, finally, by extending the codes scope to designs not accounted for in the original versions.

Severe accident codes are fully integrated, engineering analytical tools developed to describe the progression of accidents in light water reactors. Most of them, though, are being extended in recent years to some non-reactor systems, like spent fuel pools and dry casks. It is important to highlight the integral nature of this code generation because the intense feedback among different types of phenomena makes it indispensable to properly describe accident progression. Examples of the most widely used severe accident codes family are ASTEC [12], MAAP [13] and MELCOR [14].

Severe accident codes usually consist of three major blocks: basic physic-chemical phenomena (i.e., heat and mass transfer, gas combustion, aerosol transport, etc.); reactor-specific phenomena (i.e., core degradation; engineering safety systems; ex-vessel phenomena; etc.); and in-code or external user utilities (i.e., material properties; specific systems performance; severe accident management actions; etc.). Their scope includes: thermal-hydraulics of the reactor coolant systems, containment, reactor cavity; core heat-up, fuel degradation, melting and relocation; core-concrete interaction; fission product and aerosol release, transport and deposition; hydrogen production, transport and combustion; and engineering safety features performance. Figure 3 illustrates both the scope and the integral nature of the codes based on the ASTEC modules scheme.

Models in severe accident codes are intended to be mechanistic (i.e., MELCOR and ASTEC); nonetheless, the large uncertainties existing in

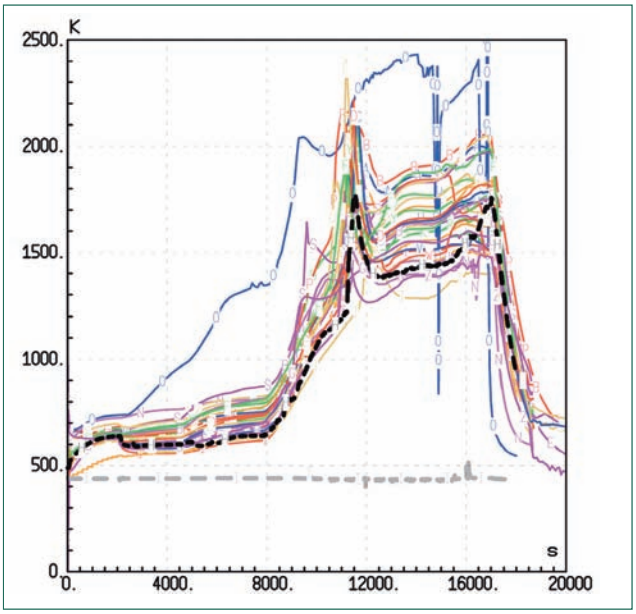

Figure 4. Outlet core temperature (FPT1 [15]).

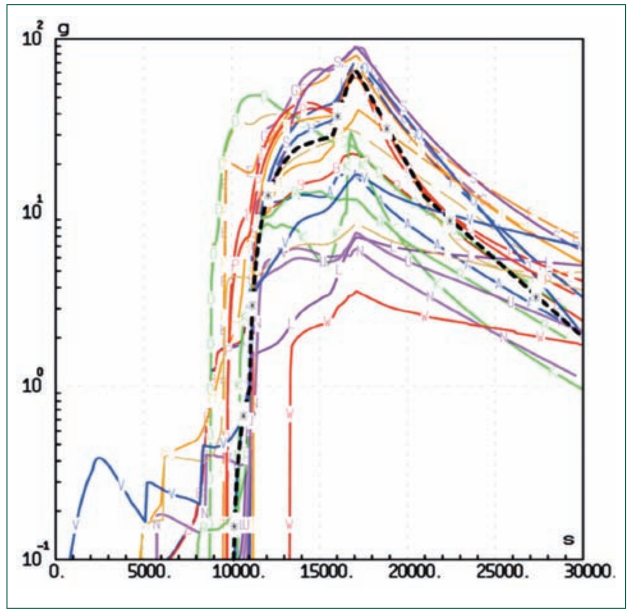

Figure 5. Airborne aerosol concentration (FPT1 test [15]).

some areas make it impossible to always follow such a mechanistic approach. This circumstance makes it highly recommended to conduct uncertainty and sensitivity analysis, for which some phenomena models are coded with default parameter values and sensitivity coefficients that are, in the end, optional user-adjustable parameters.

An aspect of outstanding relevance inherent to the variability of boundary conditions at different locations during a severe accident is plant nodalization, which is required by the lumped-parameter nature (i.e., a set of discrete nodes represent a complex spatial system) of these analytical tools. This is a major approximation in that the momentum equation is not solved within individual nodes and, as such, it should be submitted to sensitivity/uncertainty assessments. This being said, an extraordinary enhancement has been achieved since the first attempts of mechanistic estimates of fission product behaviour during LWR accidents [10], in which a 3-loop PWR RCS was represented by 4 spatial regions compared to the 100 ones that were used in the SOARCA study (section 4.1). This dramatic increase provides a much higher fidelity in the physical representation of a nuclear power plant and, hence, a better resolution of the driving forces governing fission product transport.

In following sections two examples of severe accident code evolution are given to illustrate the outstanding progress made through international benchmark exercises based on two experiments from the PHEBUS-FP project. The dispersion among the predictions may be interpreted as an indicator of uncertainties embedded in individual calculations and codes; additionally, the so-called "user effect" was clearly observable.

The ISP-46 benchmark

More than a decade ago, the ISP-46 turned out to be an outstanding opportunity to assess how codes had been improved and how close their predictions were to the data recorded during one of the most representative tests of a severe accident at the time, the PHEBUS-FPT1 experiment [15]. PHEBUS-FPT1 was a scaled-down simulation of a Small Break Loss Of Coolant Accident (SB-LOCA) in the cold leg of a 900 MWe PWR (Ag-In-Cd control rod) with low pressure in the primary system and it was selected for the

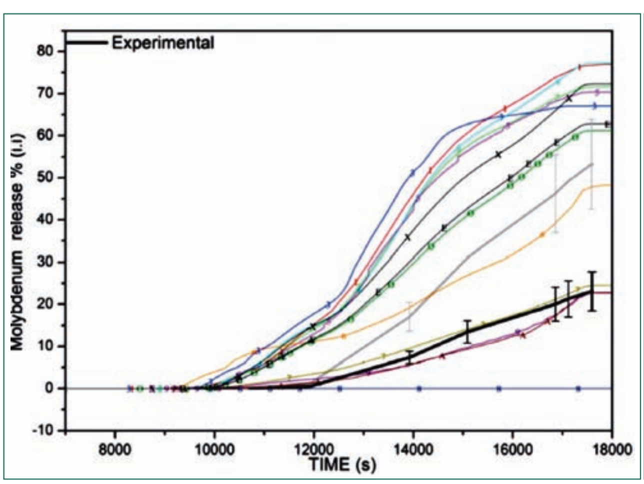

Figure 6. Molybdenum release (FPT3 [16])

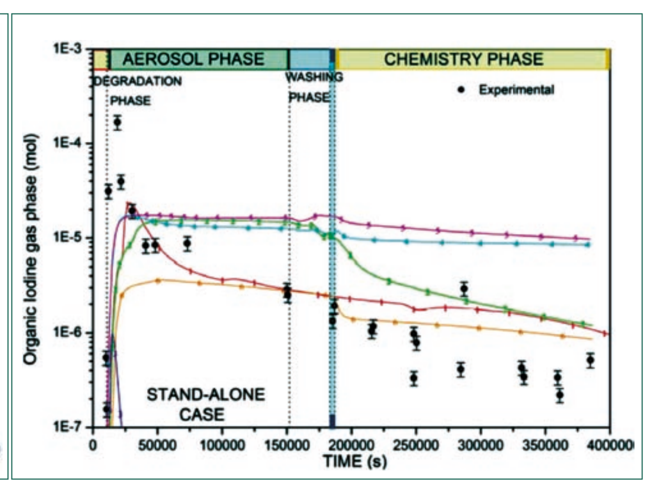

Figure 7. Gas Organic iodine concentration (FPT3 [16]).

International Standard Problem 46 (ISP46) conducted in collaboration between OECD/CSNI and European Commission (THENPHEBISP Project). As a summary, trends were generally followed both in thermal terms in the reactor (Figure 4) and as for airborne particles in containment (Figure 5). Nonetheless, the predictions scattering was remarkable in these and other variables. Much more poorly predicted was iodine chemistry, particularly the gaseous fraction in the primary system and the amount of organic iodides in the containment gas phase.

Improvements were proposed for codes and models, mainly concerning: release of structural material (especially control rod elements) as well as semi- and low-volatile fission products; iodine transport through the primary system (particularly the remaining gas fraction); and the optimum definition of parameters for in-containment iodine chemistry codes.

The SARNET FPT3 benchmark

A decade after ISP-46, the Severe Accident Research NETwork (SARNET) project of the 7th Framework Program of the European Commission coordinated a code-to-data benchmark based on the FPT3 test of the PHEBUS-FP Project [16]. An important aspect of the test was the presence of boron carbide as control rod material to study its potential impact on fuel degradation and FP speciation and transport in steam poor conditions. As above, temperature history of fuel bundle and containment thermal-hydraulics were well captured, but no code was able to reproduce accurately the final bundle state. The major deviations observed in which models need improvement are the release of semi-volatile and/or low-volatile fission products (Figure 6); but, beyond any doubt, the biggest modelling challenge was iodine chemistry in the containment (a high fraction of iodine observed in gaseous form), even when chemistry stand-alone codes were used (Figure 7).

CURRENT PRACTICE IN ACCIDENT PREDICTION

Two key milestones may help defining the current severe accident code predictability: the SOARCA study (State Of the Art Reactor Consequence Analyses) and the BSAF (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station) project.

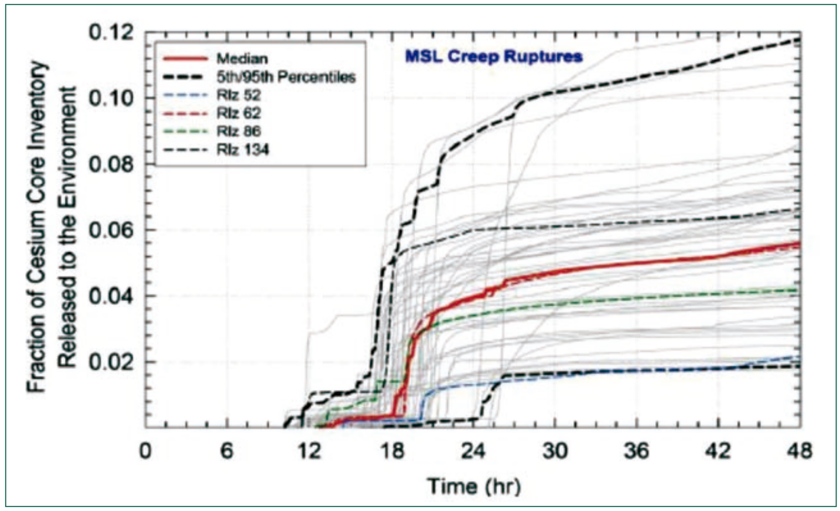

Figure 8. Preliminary uncertainty analysis results for a BWR-SBO (unmitigated) [18].

The SOARCA study

The SOARCA study [17] analyzed severe reactor accidents at full-power conditions from internal and external events for a pressurized and a boiling water reactor, from the onset of core degradation to the assessment of consequences. The code used for the simulation of the accident progression within the plant was MELCOR. The SOARCA study included a number of sensitivity studies to examine issues associated with accident progression, mitigation, and offsite consequences for the accident scenarios of interest. As a follow-on study, an analysis of uncertainties impacting consequences was conducted [18]. As an example, once decided a specific scenario (i.e., unmitigated SBO in a BWR), a reduced set of uncertainty parameters were chosen and, once propagated, yielded results like those shown in Figure 8.

The OECD BSAF Project

Beyond applications to postulated plant sequences, the Fukushima event has resulted in the most challenging scenarios ever faced by severe accident codes. Under the OECD/NEA auspices, a CSNI project (BSAF) consisting of two phases was launched in 2011[19]. Modelers participating in the BSAF faced with scenarios in which boundary conditions were widely unknown, so that their aim has been to use severe accident codes in a forensic way to find out "defendable/plausible" scenarios consistent with the limited data recorded and which are physically feasible.

The results indicated that when codes assume the same boundary conditions and the core geometry is still intact, the big picture of the thermal response predicted by codes is comparable (Figure 9) [20]. However, once fuel and material start relocating, differences show up quite visibly (Figure 10) [20]. A deep examination pointed out that differences came from modeling uncertainties in RCS failure at high core temperatures, debris surface area, debris motion paths towards RPV lower head, failure/leak of containment and others; in particular, the strong impact of assumptions made on in-vessel core debris morphology has been proved to be highly influential when comparing MAAP5 and MELCOR-2.1[21]. Once in the stabilization phase of the accidents (i.e., venting and external water injection) lack of knowledge of the actual boundary conditions led to substantial differences in the code responses.

Even greater simulation challenges were encountered when the calculations scope was extended to (BSAF-Phase 2): 3-week duration, modelling of fission products and including the reactor building in the plant model. Although results are still preliminary, code dispersion bands of up to a factor of three in in-vessel H2 production or even orders of magnitude when predicting dose rates in suppression chambers are being observed.

Discrepancies found between code estimates and recorded data on timing and magnitude of key events have provided insights into potential improvements (i.e., relocation models; plant equipment) and have highlighted some model

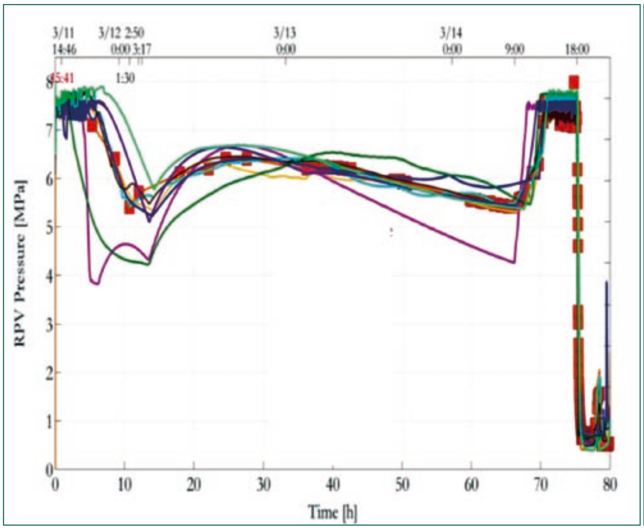

Figure 9. RPV pressure in 1F2 [20].

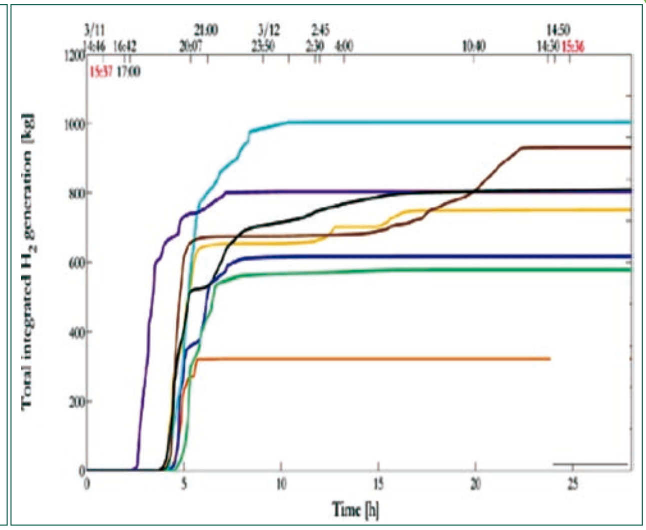

Figure 10. Integrated H_2 produced in 1F1 [20].

needs (i.e., fission product remobilization). More importantly, accident boundary conditions, equipment performance and management actions have been identified as major sources of uncertainties, which are indispensable to consider for a proper scenario analysis.

THE NEXT STEP: UNCERTAINTY QUANTIFICATION

No questioning the fact that further work on model enhancement and code validation should be continued, maturity of severe accident codes (i.e., level of integration, large scope of phenomena addressed and extensive validation) and the computational resources presently available, seem to indicate that the time has come to foster BEPU (Best Estimate Plus Uncertainties) application in the severe accident domain. By doing so, severe accident predictions would become more valuable since by knowing the associated uncertainty band of any best estimate target variable, more reliable safety margins will be calculated and conservative assumptions in models might be removed. Additionally, sensitivity analyses and distribution variance will allow identifying those magnitudes heavily influencing the uncertainty band, so that further investigation might be better focused on reducing their associated uncertainty.

There were few pioneering studies [22]. Nonetheless, it has been probabilistic analyses of uncertainty quantification once in the "integral-code age", the ones which highlighted the need and the feasibility of such an extension in severe accident analyses. About a decade ago a study with MELCOR 1.8.5 [23] focused on the hydrogen production during a SBO in an Ice-Condenser plant. The results showed narrower distributions compared to those derived by expert elicitation in NUREG-1150 and pointed the Zr-melt breakout temperature and the particulate debris hydraulic diameter as the variables dominating the distribution variance (Figure 11). More recently [24], the SOARCA uncertainty analysis on a PWR SBO consequence analysis with the last releases of MELCOR and MAACS has been reported; although the results corroborate previous conclusions, new insights have been gained, like the unlikely early containment failure from H_2 combustion even with igniters not being functional or the delayed radioactive releases with respect to earlier studies.

Nevertheless, there are still key aspects open on how to apply BEPU methodologies into the severe accident arena. Examples are expansion of the uncertain parameters to consider as many as possible, consideration of still missing models in codes when doing the uncertainty analysis or innovative methods to rank severe accident phenomena relevant for the figures of merit [25]. These and others will be investigated in the MUSA project (Management and Uncertainties in Severe Accidents), which will be submitted to the next H2020-2018 call of the European Commission, with a broad international participation.

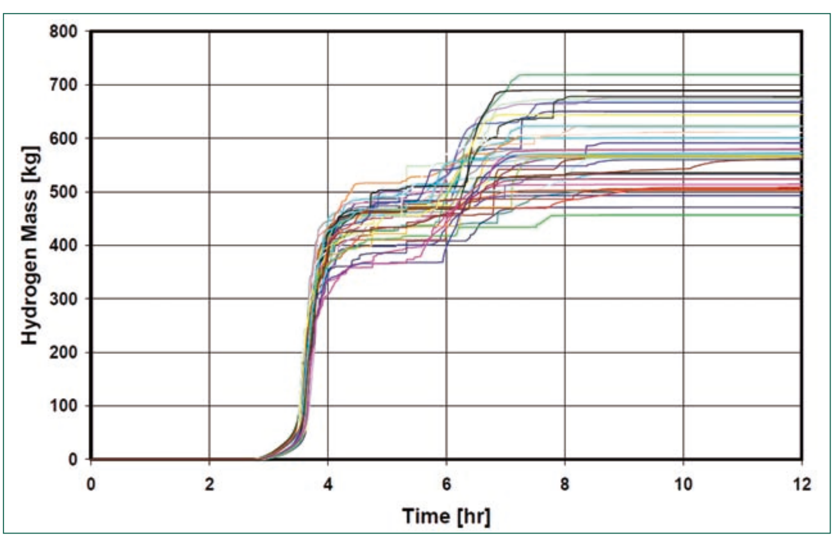

Figure 11. Hydrogen generation- MELCOR 1.8.5 (SBO-Ice Condenser plant) [23].

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