Use of the deterministic and probabilistic approach in the Belgian regulatory context

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Abstract

The Belgian nuclear power plants (NPPs), as most other NPPs, have been designed on a deterministic basis.

Since many years, a program is underway to develop, in the framework of the periodic safety reviews, plant specific PSA models for all Belgian NPPs. The objective was mainly to confirm the robustness of the deterministic design, to identify design weaknesses (if any), and to address these weaknesses if necessary. This program is now well advanced and is shortly summarised. The findings of these PSA analyses and the applications of PSA confirm that the probabilistic approach can deliver valuable complementary insights to the deterministic approach.

First, the complementary aspects of the deterministic and probabilistic approaches from a methodological point of view are identified.

Further, this complementarity is illustrated by some examples taken from these analyses, their conclusions, and their applications.

A first example concerns the PSA results for low power and shutdown states. Whereas little explicit attention has been paid to the safety analysis for these operational modes in the deterministic approach, the results of the PSA analyses have indicated the need to improve safety in low power and shutdown states. Besides a discussion of typical measures taken to improve safety based on PSA results, a discussion is also held on how to develop further the deterministic approach for low power and shutdown states. In view of the latter, a specific subject has been defined within the upcoming periodic safety reviews of the NPPs.

As a second example, the introduction of a new PSA application called PSA based event analysis (PSAEA) will be described. Based on the PSA models resulting from of the periodic safety review, probabilistic precursor analysis has become a part of the AVN process of feedback of operating experience. Whereas in former times the operational feedback analysis was exclusively based on a deterministic analysis (root cause analysis), it is now complemented for a number of selected events by a probabilistic analysis, giving insights into the importance of the event, critical components and/or actions, etc. The development of the methodology for PSA based event analysis, the internal use within AVN and related international activities are described.

An outlook towards potential future applications in other safety evaluation activities is also given.

1. Introduction

The Belgian nuclear power plants (NPPs), as most other NPPs, have been designed on a deterministic basis. Since many years, a program is underway to develop, in the framework of the periodic safety reviews, plant specific PSAs for all Belgian NPPs.

This paper describes the status and the scope of the Belgian PSA programme, the approach followed for the review process of PSAs, and the current use of the PSAs for the safety evaluation of operating plants. Insights on the complementarity aspects of the deterministic and the probabilistic approach are described, and examples are given on applications.

2. Status and scope of the Belgian PSA programme

AVN, as regulatory organisation, is performing the review of the Probabilistic Safety Assessments (PSAs) of the Belgian nuclear power plants. The PSAs are performed by the Architect-Engineer (Tractebel Engineering) of these plants on behalf of the Utility (Electrabel).

The table below gives a synthesis of the status of the Belgian PSA programme.

Plant	PSA scope	Original PSA Date Completed	Revised PSA Date Completed
Doel 1 and 2	Level 1 + 2	expected 2003 or 2004	
Tihange 1	Level 1 + 2	expected 2003 or 2004	
Doel 3	Level 1 + 2	1993	2000
Tihange 2	Level 1 + 2	1993	2000
Doel 4	Level 1	expected 2003	
Tihange 3	Level 1	expected 2003	

In the level 1 part, internal initiators for power and non-power states are analysed. Internal hazards (fire and flooding) and external natural or man-induced hazards are not covered. Based on the results, actions of different types (hardware modifications, development of new or improvement of existing procedures, changes in operational conditions) have been implemented.

All level 2 analyses performed for the Belgian NPPs are limited to the analysis of the containment response, with the aim to investigate dominant containment failure modes. No source term analyses have been performed.

For the Doel 3 and Tihange 2 PSAs, the level 2 analysis was limited to a binning into Plant Damage States (PDS) and a deterministic analysis of the containment behaviour for some dominant core damage sequences with the STCP code. In this way, the phenomena (hydrogen burning, basemat melt-through, etc.) threatening the containment integrity could be identified for each of these typical accident scenarios, however without obtaining information on the probabilities of the failure modes. Mitigating measures (for instance autocatalytic hydrogen recombiners) have been installed.

For the Doel 1 and 2 and Tihange 1 PSAs, a probabilistic analysis using Containment Event Trees (CETs) is performed, using MELCOR for the analysis of the severe accident progression. The aim is to obtain better insights in the importance of the different containment failure modes.

3. Complementarity of the deterministic and probabilistic approach

3.1 Methodological aspects

Both approaches are well known and need not to be explained here in more detail.

Nevertheless, a short reminder of the methodological aspects of both approaches that are on the basis of this complementarity is worthwhile. They are summarised in the table below.

	Deterministic approach	Probabilistic approach
Initiating events	- limited to DBAs - beyond design events are not considered (although some analyses can be performed for these events)	All potentially important events are included
Systems	single failure criterion is used	multiple failures and CCFs are also considered
Operator behaviour	- for t < T: no action is postulated (T=10 to 30 min); - for t > T: no operator errors are postulated	errors in diagnosis and errors of execution are considered throughout the accident sequences
Analysis	conservative assumptions	as realistic as possible

3.2 Discussion

3.2.1 Complementarity versus "lack of coherence"

The above methodological aspects of both approaches, and in particular the way they are treated in both approaches, is in our opinion the basis of the complementarity of the deterministic and probabilistic approach. The application of both approaches according to this methodology has in our opinion (see § 4) given rise to valuable insights on how to improve the safety of the NPPs. We are therefore in favour of a continued combined application of both approaches while maintaining their methodological characteristics.

It seems to us that a recent publication [1] has another viewpoint on this matter. In Section 4.8 of [1] a discussion is held on the "lack of coherence" between the deterministic and probabilistic safety approaches. Some of the aspects mentioned in the table above (difference in accident sequences considered; single failure versus multiple and common cause failures) are mentioned as examples of this lack of coherence.

In [1] the single failure criterion is considered "a particularly archaic relic that should be re-examined". We recognise that in fact the systematic postulation of a single failure (and only a single failure) in the deterministic analysis of accident sequences is overruled by operating experience (where multiple and CCF failures have caused serious incidents and accidents) and is hence not always a conservative assumption (one of the often quoted characteristics of the deterministic approach). Although this fact is known since a long time, nobody ever succeeded to define an acceptable alternative. In this respect it is worthwhile to notice that documents (e.g. [2]) related to future reactor designs continue to consider the deterministic and probabilistic approaches in a complementary way ([2], Chap. 2.0) and that the single failure criterion is not called into question ([2], Chap. 2.1).

According to [1], an activity seems ongoing within the USNRC to address this lack of coherence. While this might lead to interesting considerations on both approaches, it is our opinion that formulating "some suggestions as to what could be done to achieve greater coherence" [1], should be accompanied with great care in order not to undermine what we called before the complementarity of both approaches. A feedback of probabilistic insights into the deterministic design basis approach can in our opinion be taken into consideration; the present discussions [6] on the definition of large break LOCA (double ended guillotine break or smaller?) are an example of this. However, for the events analysed within both approaches, we argue in favour of maintaining the methodology unchanged. The combined use of both approaches, each with their specific methodology, seems to be the best way to "compensate" (i.e. to provide a "safety net") for each other potential weaknesses, for instance:

- the consideration of multiple failures and common cause failures in the probabilistic approach compensates for the limitation of considering only a single failure in the deterministic analyses;
- the conservative approach on data and hypotheses in the deterministic analyses is providing margins in the design that may compensate for potential optimistic data and hypotheses in the "as realistic as possible" approach of the probabilistic safety analysis.

3.2.2 Uncertainties

When discussing pros and cons of the deterministic and probabilistic approaches, the "large" uncertainties are often mentioned as a potential drawback in the use of probabilistic results in decision-making. It is our feeling that this is an unbalanced view on both approaches. The deterministic approach is submitted as well to (large) uncertainties, mainly stemming from modelling assumptions and the extensive use of engineering judgment.

Reference [1] (see page 15) gives an interesting discussion on this matter on the basis of the Davis-Besse incident and concludes that "Unfortunately, deterministic safety evaluations do not seem to be held to the same level of accountability to address uncertainties as is the case with PRA."

Another example illustrating this issue are, in our opinion, the recent findings about the sump clogging in the recirculation mode of ECCS (amongst others based on [3]). If the probabilistic approach has maybe strongly underestimated the probability of sump clogging, this is mainly related to modelling assumptions that affect the deterministic analyses as well. In the context of this, it should be remarked that the probabilistic approach provides at least the opportunity to perform (very instructive) sensitivity analyses on the importance of this sump clogging issue.

4. Examples of experiences with complementarity in the Belgian context

4.1 Safety in non-power and shutdown conditions

Up to now, the main application of the PSAs for the Belgian NPPs concerns design evaluation. Indeed, the primary objective is to use the PSA, in the framework of the periodic safety review, as a complementary tool to the deterministic safety analysis. It should mainly provide valuable insights in the balance of the design, identify important contributions to the core melt frequency and constitute a useful tool to evaluate the effectiveness of proposed plant modifications.

The PSAs have been used to propose hardware modifications and improvements to the procedures (as well improvements to existing procedures as the development of new procedures).

As in many other countries, one of the major risk insights obtained from the level 1 part of the PSAs for the Belgian plants is the non-negligible contribution to risk in some non-power states (typically of the order of 25 to 35 % of the total core melt frequency). Taking into account the short time spent in some of these operating states, one observes, for instance for the mid-loop operational mode, that the estimated core melt probability per hour rises to about a hundred times the core melt probability per hour in the power state.

In our opinion, this reflects clearly that the defence-in-depth during the non-power states is not so well implemented as in the power states. Looking back to the deterministic safety analyses performed for the licensing of the Belgian plants (ten to twenty years ago, if not more, and based on the USNRC rules and regulations), this is not such a surprise: it is clear that in these analyses much more attention has been paid to safety in the power states.

Hence, a major insight on defence-in-depth, gained from our level 1 analyses, is that safety in non-power states should be analysed in detail by a deterministic approach, applying the same defence-in-depth ideas as the one used in the past for the power states.

This has led to the definition of a particular subject in the upcoming periodic safety reviews of the Belgian plants, aiming to dispose of a complete and well-documented deterministic analysis of safety in non-power states.

4.2 Operating experience feedback analysis

The PSAEA (PSA based event analysis) methodology was established [5] in the framework of an international project on behalf of — and involving — the nuclear regulatory bodies from six countries. As a part of the project, first feasibility tests have been performed with the support of a utility.

The PSAEA procedure aims at the best-estimate assessment of the safety significance of an operational event using the available PSA model. It elaborates details for understanding the event, modelling the event, quantification, "what if" analysis, analysis & interpretation of results, and conclusions & reporting. As to the quantification, it explains the different characteristics and calculation treatment employed for real — or potential — initiating events and condition events, in order to obtain the probability of core damage conditional to the occurrence of the event.

Early pilot studies on real cases have proven the interest and the feasibility of the method. The objectives of the current AVN precursor program are mainly focused on (1) the determination of the quantitative importance of a few well-selected operational events per year, and – if sufficiently significant – on (2) the subsequent identification of potential safety issues for improvement (based on the real best-estimate case as well as on relevant what-if questions). AVN considers the identification of potential safety issues for improvement to be among the most important outcomes of the study, because they have the potential to be implemented and to make a real difference. Secondary objectives in this stage of early PSA experiences include the enhancement of the awareness of typical risk figures associated with both exceptional and more common events, and the resulting feedback on the PSA model itself.

Today, the PSAEA process is integrated in the larger process of follow-up of operating experience at AVN, and involves the following phases: screening and selection of events, analysis of events, internal review by PSA specialists as well as plant inspectors and staff members involved in experience feedback, presentation of the analyses to the utility for comment and for further consideration, and follow-up of identified safety issues for improvement (if appropriate).

In order to ensure a close contact with other experiences and developments in this field, AVN has taken the initiative to organise an annual meeting on probabilistic precursor analysis in the nuclear industry. The interaction with a wide audience of other practitioners and stakeholders has not only advanced the understanding of many technical issues, but it has also contributed to the evolution of a broader view on the process itself of precursor analysis by a nuclear regulatory organisation.

5. Future challenges in the Belgian context

Some time ago, discussions were started between the utility, the architect-engineer and AVN on how to transpose R.G. 1.174 [4] in the Belgian regulatory context. The implementation of a document like R.G. 1.174 in the Belgian regulatory context seems indeed a requisite in view of potential future uses of PSA, especially in view of relaxations on safety requirements.

The document was considered to be very valuable and to a large extent directly transposable to the Belgian context, except for some aspects. Mainly the acceptance criteria (diagrams of ΔCDF versus CDF and ΔLERF versus LERF) were felt to require adaptation. This discussion has not been finalised so far.

Another aspect that remains to be tackled is the impact of a partial scope of the PSA (fire, flooding, earthquake, ...not covered) with respect to the definition of the acceptance criteria.

A difficult point in the application of the R.G. 1.174 seems further the recommendation that risk-informed insights should not undermine the defence-in-depth and the existing safety margins. No specific guidelines on how to treat this in practice seem to be provided by the R.G. (as also pointed out in [1], page 42).

6. Conclusions

The use of PSA in the framework of the periodic safety reassessments has proven the usefulness of the probabilistic approach, as a complement to the deterministic one, the latter being the basis for the design of the Belgian NPPs. Some examples are discussed in the paper.

In AVN's opinion, both approaches should be used further in a complementary way. Cross-fertilisation of both approaches should be performed with great care in order not to undermine the benefits of the complementary character of both approaches.

7. References

[1] Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making; NUREG/CR-6813 (April 2003)

[2] European Utility Requirements for LWR Nuclear Power Plants. Revision C April 2001.

[3] NUREG/CR-6762 (Vol. 1); "GSI-191 Technical Assessment: Parametric Evaluation for Pressurized Water Reactor Recirculation Sump Performance" (and related Volumes); Prepared by Los Alamos National Laboratory for USNRC (2002).

[4] "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"; R.G. 1.174 (USNRC, July 1998)

[5] Enconet Consulting, A Framework for the PSA-Based Analysis of Operational Events, Final report, April 1997.

[6] Joint CSNI/CNRA Workshop on "Redefining the Large Break LOCA: Technical basis and its implications" (Zurich; June 23-24, 2003).