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Insights and lessons learned from Level 2 PSA for Bohunice V2 plant

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Abstract
The paper provides a brief overview of the Level 2 PSA conducted for Bohunice V2 plant. The important features of the PSA model are described. Lessons learned from the study with regard to modelling approach are highlighted. Information is also provided on the results. Frequencies of the dominant scenarios leading to a radioactivity release are presented and plant vulnerabilities indicated. Insights from sensitivity analysis are provided with regard to modelling assumptions. The effectiveness of various measures affecting the progression of severe accidents and containment performance is discussed.

1 INTRODUCTION

1.1 Background

The Level 2 PSA for Bohunice V2 plant was one of the first systematic Level 2 PSA studies performed for WWER plants. In 1999, when the project was initiated, only few WWER plant have completed Level 2 studies. At that time such studies have been performed for most of the operating plants in Western countries. Insight from these PSAs proved to be useful in identifying plant vulnerabilities in the area of containment performance and accident management strategies as well as providing a basis for plant specific backfit analysis and evaluation of risk reduction options.

At the start of the project the Bohunice V2 plant had a comprehensive Level 1 PSA which covered full power states as well as shutdown and low power operating modes [1]. This provided a good starting point to extend the PSA study of Bohunice V2 to Level 2.

1.2 Objectives and scope of the analyses

The objectives of the Bohunice V2 Level 2 PSA project were to estimate the frequency of large radiological release and to identify sequences contributing to large radiological release.

The scope of work performed within the framework of this project fully complies with the current state-of-the-art practices in the area of Level 2 PSA [2]. All typical Level 2 PSA elements/tasks have been included.

Comprehensive analyses have been performed on containment isolation and damage. Supporting thermal hydraulic and structural analyses were an integral part of the study. Source Term analysis has been performed to provide quantitative data for radioactive releases. Comprehensive sensitivity analysis that addressed both the relevant modelling assumptions and potential plant improvements was also conducted.

1.3 Project organisation

The project was implemented by an integrated team consisting of staff of four firms.

ENCONET Consulting Ges.m.b.H. (Vienna, Austria) – the leader of the project team – was responsible for categorisation of plant damage states (PDS), preparation of confinement event trees (CET), and their quantification, and finally the analysis and interpretation of the main results. VUJE Trnava Inc. (Slovak Republic) provided supporting analyses related to identification of containment challenges and source term analysis. RELKO Ltd. (Bratislava, Slovak Republic) was responsible for major part of PDS modelling (preparation of PDS event tree/fault tree models and their quantification). Lenkei Consult Ltd. (Pécs, Hungary) conducted a structural analysis of the confinement building.

2 SELECTED METHODOLOGICAL ASPECTS

2.1 PDS definition and modelling

Grouping of core damage sequences into the plant damage states (PDS) was based on the characteristics of sequences which most influence the post-core damage accident progression and hence the releases.

The PDS grouping parameters selected for the Bohunice V2 level 2 PSA were chosen based on the review of other studies [3-5]. Due consideration was given to specific features of the plant. The PDS grouping parameters finally selected for Bohunice V2 level 2 PSA are provided in Table 1.

Table 1. *PDS grouping parameters.*

PDS Parameter	Parameter values	Sequences following this branch /sequence specific features
Confinement status	Intact	Sequences with confinement isolation success & no bypass
	Not isolated	Core damage sequences in which the confinement fails to isolate
	Bypassed	Unisolated SGTR or interfacing system LOCA
Sequence type	Transient with FW loss	Transient initiators with the RCS intact and failure of all feedwater
	Reactivity transient	Transients in which reactivity control has failed, for example ATWS
	SLB inside confinement	In these sequences the baseline pressure inside the confinement will be higher and the steam concentration will be higher.
	SLBI/Transient with FW loss, Containment not isolated	In this case the baseline pressure and the steam concentration are not important.
	LOCA(7- 20mm) with FW loss	In these sequences induced ruptures are considered credible, LPSI cannot inject due to high pressure, and the integrity of the reactor cavity may be threatened after vessel failure.
	LOCA(20 - 40mm) with FW loss	In these sequences induced ruptures are not considered credible, LPSI cannot inject due to high pressure, and the integrity of the reactor cavity may be threatened after vessel failure.
	LOCA(7- 40mm) with FW available	Small LOCAs (7 – 40 mm) with feedwater available. Induced ruptures are not considered credible, and LPSI cannot inject due to high pressure. The integrity of the reactor cavity may be threatened after VF (low probability).
	LOCA(7- 40mm) with FW available + aggressive feed & bleed from SG AND LOCA(40 – 500mm)	RCS depressurised, meaning that LPSI can inject and the cavity is not threatened.
Safety injection	LPSI/HPSI successful	Success of at least 1LP or 1HP train in injection and recirculation
	Recovered early (prior vessel failure)	Success of hardware for at least 1LP or 1HP train in injection and recirculation. Injection not successful until ac power recovered.
	Recovered late (after vessel failure)	Success of at least 1LP or 1HP train in injection and recirculation phase. Injection is not successful until ac power is recovered.
	Failed	Failure of all injection, either in injection or recirculation phases.
Cavity water	Wet	Sufficient water is injected into confinement prior to vessel failure to overflow SG boxes and to flood the cavity (large LOCA case)
	Dry	Cases in which the overflow level is not reached in the SG boxes.
Confinement spray	CSS successful	Success of at least 1 train of the spray system
	Recovered early (prior RPV failure)	Success of hardware for at least 1 train, but spray cannot operate until ac power is restored.
	Recovered late (after RPV failure)	Success of hardware for at least 1 train, but spray cannot operate until ac power is restored.
	Failed	Failure of all trains of the spray system.

Assignment of core damage sequences to plant damage states was performed based on a grouping logic presented in the form of a decision tree. The PDS grouping parameters were used as headers in the decision tree. Accident sequences were assigned to PDS by following appropriate branches under each header. The appropriate branch to follow was decided based on the characteristics of the particular core damage sequence being considered.

The use of a grouping diagram of this type ensures that the process is performed in a systematic, repeatable manner. Another advantage is that the diagram explicitly identifies which combinations of grouping parameter values are possible and which are not.

The PDS grouping logic (decision tree) was develop for both full power and non-full power plant operational states (POSs). Several POS groups were introduced to facilitate the PDS grouping process:

- G0 - Full power operational states.
- G1 - Both the RCS and the confinement closed (POS 1, 11 and 12, similar to the full power states).
- G2 - RCS is closed but the confinement open (POS 2, 3, 7, 8, 9 and 10).

G3 - Both the RCS and confinement are open, fuel in the reactor vessel (POS 4, 5S and 6).
G4 - Fuel relocated to the refuelling pool (applicable to long refuelling i.e. POS 5L).

The Level 1 PSA was extended by adding of headers and branches that would describe the status of confinement isolation, confinement spray, and safety injection systems. The PDS grouping logic provided a basis for identifying the modifications needed. New event trees were developed for Loss of Offsite Power in order to allow different recovery periods to be modelled and the correct assignment of event tree sequences to PDS. Modifications were also made for ATWS sequences that were not properly addressed in the original Level 1 PSA study.

Modification of event trees required some changes in the related fault tree models. Some modifications were needed in shutdown PSA model, in which certain parts were common for POSs included in two different PDS groups (e.g. G1 and G2). To allow for treating separately the selected PDS groups the ET/FT model needed to be split. In some cases new fault trees were needed to model new ET heading with different configuration of safety injection and spray system trains.

2.2 Source Term categorization / analysis

Source term categories (STC) were defined on the basis of appropriate attributes that affect fission product (FP) releases and accident consequences. The attributes of the source term categories were established by considering the relevant initial and boundary conditions that affect the releases: the extent of fuel damage during the accident, the primary circuit status (LOCA vs. non-LOCA), the containment isolation status, and the corium status.

Accident progression was the first parameter considered in the grouping process. Four main source term groups were selected depending on the sequence type: a large LOCA, transients or small LOCA, interfacing LOCA, and open reactor (or fuel pool) sequences. All other parameters were considered within each of these main groups.

A total of 74 STCs were defined for the 4 main groups: 32 for large LOCA, 32 for transients or small LOCA, 8 for interfacing LOCAs, and 2 for open reactor (or fuel pool) sequences.

Source term was estimated based on plant specific thermal hydraulic analyses for representative SA scenarios [6]. The analyses were performed using SA codes available at that time in Slovak TSOs. Modified versions of the STCP package modules were used, relevant to VVER units. Source term was quantified for 9 groups of fission products defined in accordance with their common chemical and physical characteristics (I, Cs, noble gasses, Te, Sr, Ru, La, Ce, and Ba).

2.3 Confinement Event Trees

Confinement event tree (CET) models have been developed for full power and shutdown PDS (G0 – G4 PDS groups). Several CETs structures were prepared to address accident sequence groups that differ with the initial status of the confinement. There were 7 groups (3 for full power and 4 for shutdown states) for which CET models were prepared.

This approach to organisation of CET models has been found practical from the modelling point of view. The basic rationale behind this approach was that the initial status of the confinement has a significant effect on the CET logic. It is easier to handle a large number of variations by using a different tree structure, rather than by customising the sub-models to address all individual cases.

The headings selected for the CET models are based on the results of the identification of confinement challenge phenomena. They are described in Table 2.

Table 2. *Headings used in the Containment Event Trees.*

CET Header	Relevant characteristics
Induced RCS rupture	Creep rupture of the hot leg or SG tubes (RCS depressurization, potential bypass).
Very early hydrogen burn	H_2 burn prior to the in-vessel phase of the accident (hydrogen depletion).
Very early containment failure	Confinement failure due to hydrogen combustion before vessel failure
Reactor vessel failure	Possibility of core cooled in-vessel (subsequent confinement challenges avoided).
Early containment failure	Confinement fails due to hydrogen combustion loads at the time of vessel failure
Debris cooled ex-vessel	Prevention of molten core concrete interaction
Late containment failure	CF due to the combustion of hydrogen / combustible gases during the ex-vessel phase
Long term containment failure	Overpressurization by non-condensable gas generation or basemat melt-through

The approach selected for development of CET branching probabilities was to develop sub-models, which decompose the physical process corresponding to each of the CET headings into a sub-event tree model. Such sub-models (referred to as decomposition event tree models) aim to take into account the uncertainty related to confinement phenomena by delineating the different physical possibilities for the process related to the CET header. The decomposition event trees (DETs) provides a graphical representation of the different possibilities, which have been considered under each CET header.

It should be noted that in this study, each CET header actually uses several different decomposition event tree models. For each heading the different possible boundary conditions are identified. These different boundary conditions are essentially the different combinations of PDS characteristics and accident sequence characteristics, which affect the physical process modelled by the header in question. The CET uses a separate sub-model for each different set of boundary conditions.

Each CET header uses several different DET models depending on different possible boundary conditions. These different boundary conditions are essentially the different combinations of PDS characteristics and accident sequence characteristics, which affect the physical process modelled by the header in question. The CET uses a separate sub-model for each different set of boundary conditions. Each CET heading is decomposed into several more detailed questions easier to answer or quantify, suitable DET is selected based on the answers.

The DET models provide graphical representation of different possibilities under each CET header and assign probabilities. Where possible and reasonable, standard sources for subjective probability data were used. In this respect, a useful study was the NUREG-1150 severe accident risk study, which expended considerable resources on a formal expert elicitation process to generate subjective probabilities.

End-states were defined by CET model as a string with the following information:

LOCA/TRANSIENT –	A, T
CORE DAMAGE EXTENT –	PART_CD, FULL_CD, CD+MCCI,
SPRAY STATUS –	SPRAYS_OK, NO_SPRAYS
CONFINEMENT STATUS -	VEARLYCF, NOT_ISOL, EARLY_CF, CAV_DOOR, etc.

The fragility curve used in the Level 2 PSA model for containment failure was derived on the basis of deterministic analysis (the best estimate confinement failure pressure) and expert judgement (quantitative estimates of uncertainties). Probability density function was assumed to be log-normal distribution, which is a usual assumption. The final parameters of the distributions (log-normal) used in the Level 2 PSA for Bohunice V2 are shown in Table 3.

***Table 3**. Parameters of confinement failure probability distribution used for Bohunice V2.*

Location	Median failure pressure (abs), MPa	Logarithmic standard deviation*	Pressure for 5% failure probability, MPa	Pressure for 95% failure probability, MPa
Bubbler tower corner	0.426	0.148	0.356	0.516
Cavity door	0.594	0.148	0.488	0.731

Note: The uncertainty distributions are applied to the overpressure, not the absolute pressure.

The uncertainty estimated in the plant specific analysis, which includes uncertainties in the material strength and uncertainties related to the methodological approach, is +/- 11.3% (standard deviation). This value was compared with NUREG-1150 expert judgement data [3] and the results from mock-up experiments [7]. Finally, the value 16% based on the NUREG study (converted from non-symmetrical distribution to log-normal) was applied for the standard deviation.

2.4 Quantification

The PDS model quantification was carried out using the RiskSpectrum code [8]. This code provides features for managing the PSA database and fault tree and event tree models. It also provides a fast minimal cutset generator. However, the code does not provide very good cutset editing features. It was necessary to use an external program for processing of shutdown operational state cutsets. The post-processing was performed using the program WINPROCESS.EXE - the Windows version of program PROCESS.EXE [9].

Quantification was performed by calculating all PDS sequences separately. It was not possible to generate minimal cutsets for individual PDS consequences by Riskspectrum feature, because the Riskspectrum code does not take into account success branch of event trees. A cutoff of 1E-12/yr was used. This cut-off value meets the accuracy requirements.

The resulting frequency of individual PDS consequence was obtained as a sum of sequence analyses results for the same consequence. The summation of single sequence frequencies was performed in MS Excel code after export of Riskspectrum results.

Then the CET was quantified using the code in which they were developed. Quantification was conducted for each PDS using sub-model (DET) which corresponds to that PDS (based on logical rules that are provided to select DET sub-model). End-states were assigned automatically to each CET sequence, textual identifiers served to provide link with RCs. Further grouping of sequences was conducted using a short script (written in Phyton).

3 MODELLING APPROACH – LESSONS LEARNED

The modelling approach selected for the project has been found effective and easy to apply. The analysis was manageable, repeatable, and easy to displayed/document. Relatively small number of PDS attributes was found sufficient to capture the relevant features of accident sequences. "Sequence type" found to be useful PDS parameter. The use of decision trees to for PDS grouping had major advantages (the process systematic and repeatable, combinations of parameters that are not possible eliminated).

Separate set of PDS for each POS group has been found convenient. In this way dependencies and boundary conditions specific for the PDS group were taken into account significantly reducing the number of PDS groups. It was also possible to identify risk contributors for full power and for the major phases of outage separately.

Some extension of the existing Level 1 PSA model and its re-quantification is always needed to obtain additional information on the accident sequences, mainly related to factors which could affect the progression of the accident after core damage.

Use of containment performance model with two levels (CET + DET) has been found advantageous. In this approach the model was traceable and easier to manage. Subjective judgement involved in assigning the probabilities to CET branches was also easier to document.

Definition of end-states has been found sufficiently detailed. It also allowed for flexible grouping with regard to source term characteristics.

Step-wise process of CET quantification has been found effective and at the same time easily traceable. Transfer of data from one tool to another specific to this process was straightforward.

4 OVERVIEW OF RESULTS

A brief overview of PDS frequency results is given in Fig.1. The PDS frequency contribution and the dominant PDSs (two highest contributors) are displayed for each of the PDS groups (G0-G4). It can be noted that the contribution of shutdown PDS groups (in particular G3) is significant as compared to full power.

The quantification of the confinement event tree models provides the conditional probabilities for each confinement event tree end-point. By combining these probabilities with the input PDS frequencies, the frequencies of the confinement event tree endpoints were obtained. The endpoints are grouped according to their source term and confinement status characteristics and the corresponding endpoint frequencies are summed in order to generate the integrated results. The results obtained for PDS groups G0 – G4 are presented in Table 4.

Comparison of risk in terms of releases for different PDS groups (G0 – G4) is provided in Fig. 2. It shows contribution of large release vs. small / delayed release end-states. Large release end-states are defined here as those with confinement failure or open at or before vessel failure or those with confinement bypassed.

5 MAIN INSIGHTS FROM RESULTS

5.1 Plant risk profile

The following observations can be made based on the results obtained with regard to the risk profile:

- For power operational states the main concerns are hydrogen burn during in-vessel phase and cavity door failure due to loads at vessel failure. Both of these confinement failure modes are important risk contributors. A key feature of the core damage frequency profile is that high pressure PDS are dominant for early confinement failure.
- For shutdown operational states dominant risk contributors are vessel open / confinement open states. This is mainly due to high core damage frequency for the mid-outage states and disabled barriers during these states (both the confinement and reactor vessel are open to conduct outage).

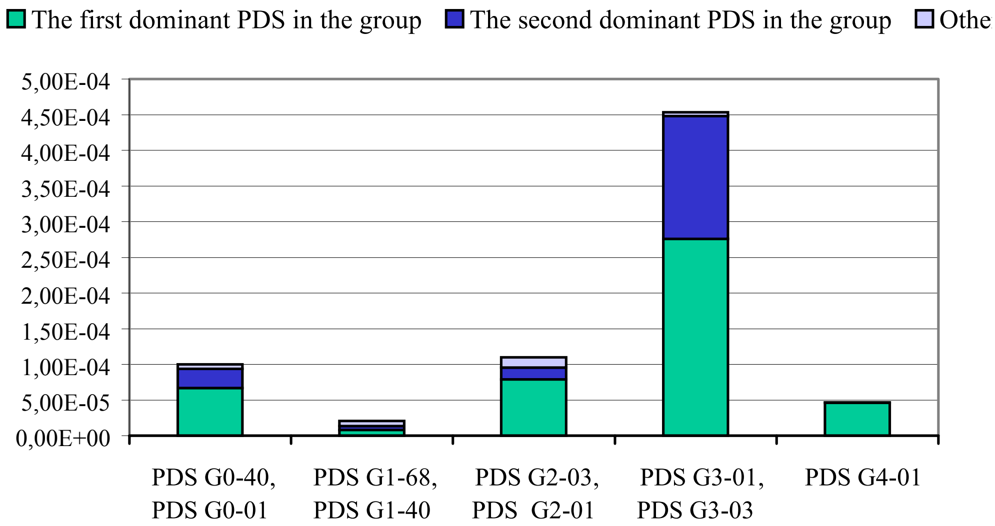

FIG. 1. *Summary of PDS frequency results for PDS groups G0 – G4 (PDS numbers provided below the bars are those of the two most dominant PDS in each group).*

PDSs with the highest frequencies:
- PDS G3-01 – man induced LOCA in POS6 as a dominant IE, (SI not initiated due to HE)
- PDS G3-03 – man induced LOCA in POS6 as a dominant IE, (SI not available)
- PDS G2-03 – cold overpressurisation in POS7 as a dominant IE,
- PDS G0-40 – medium LOCA in full power as a dominant IE,
- PDS G4-01 – loss of non-vital operational 6 kV bus in POS5L as a dominant IE.

Table 4. *Final end-states with their corresponding frequencies for PDS groups G0-G4.*

PDS Group	Confinement state	Frequency	Contribution
G0	Cavity door fails due to loads at vessel failure	3.4E-5 /yr	34%
	Confinement fails due to hydrogen burn during in-vessel phase (VF)	3.2E-5 /yr	32%
	Long term basemat melt-through	1.8E-5 /yr	18%
	Confinement failure due to hydrogen burn at vessel failure	9.4E-6 /yr	9%
	Accidents with confinement isolation system failure (VF)	2.7E-6 /yr	3%
	Accidents no confinement failure (VF)	1.6E-6 /yr	2%
	Confinement bypassed due to SGTR or IFSL	1.5E-6 /yr	1%
	No confinement failure (NO VF)	4.0E-7/yr	<1%
	Confinement failures with vessel intact (NO VF)	4.0E-7/yr	<1%
	Late and long term confinement failures	2.8E-7 /yr	<1%
G1	Confinement bypassed due to SGTR or IFSL	8.3E-6/yr	40%
	Confinement fails due to hydrogen burn during in-vessel phase (VF)	3.3E-6/yr	16%
	Cavity door fails due to loads at vessel failure	3.2E-6/yr	15%
	Long term basemat melt-through	2.0E-6/yr	10%
	Accidents no confinement failure (NO VF)	1.1E-6/yr	5%
	Confinement failure due to hydrogen burn at vessel failure	9.6E-7/yr	5%
	Confinement fails due to H2 burn during in-vessel phase (NO VF)	7.9E-7/yr	4%
	Accidents with confinement isolation system failure (VF)	3.6E-7/yr	2%
	Late confinement failure (NO VF)	2.1E-7/yr	1%
	No confinement failure (VF)	1.8E-7/yr	<1%

PDS Group	Confinement state	Frequency	Contribution
G2	Confinement open, vessel fails / full CD	1.0E-4/yr	99%
	Bypass event (SGTR / IFSL)	7.1E-6/yr	<1%
	Confinement open, no vessel failure/ restricted CD	8.6E-7/yr	<1%
G3	Confinement open, full core damage	3.3E-4/yr	73%
	Confinement open, restricted core damage	1.2E-4/yr	27%
G4	Confinement open, core damage in fuel pool	4.6E-5/yr	100%

Note: Accidents with no confinement failure, confinement failure due to hydrogen burn in the pre-vessel failure period late confinement failures and accidents with confinement isolation failure can occur with or without vessel failure. This is indicated above by (VF), in cases where vessel failure occurs, or (NO VF) in cases where vessel failure does not occur.

5.2 Sensitivity to model assumptions

Several observations have been made from sensitivity analysis with regard to modelling assumptions. Some important features of the model, which appear to be influential on the results, are discussed below.

- The probability of cavity overpressure is sensitive to the probability of hydrogen burn in the cavity (due to small volume of the cavity). In most sequences, hydrogen burning is expected prior to vessel failure, which on the one hand, is a challenge to confinement integrity, but on the other hand, depletes hydrogen which is not available to burn later.
- The model is not sensitive to assumptions related to long term confinement overpressure or the detailed modelling of ex-vessel cooling in the cavity. The latter effect appears to be due to the PDS profile (basemat melt-through contribution of PDS with no injection available, for which ex-vessel cooling is impossible).
- Induced hot leg rupture in non-LOCA scenarios is unlikely. For these scenarios there are uncertainties in relation to the induced hot leg rupture due to uncertainties in creep rupture properties of steel. It should be noted that the situation is different for very small LOCAs.

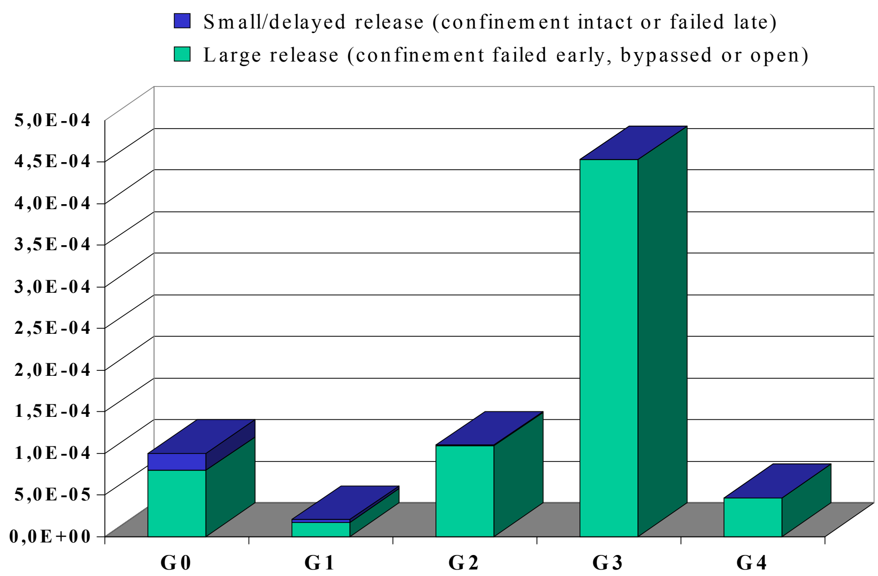

***FIG. 2**. Contribution of large release vs. small/delayed release endstates for different PDS groups*

5.3 Impact of plant improvement measures

A series of improvement options (Severe Accident Management measures) were investigated as compared to the state of plant analysed in the base case. These included the RCS depressurisation, cavity flooding, and hydrogen control. The modification options assessed in sensitivity studies all appeared to be beneficial. The following conclusions can be made based on this analysis:

- Implementation of RCS depressurization (EOP Fr.C-1) reduces vulnerability to cavity door failure (by a factor of 2). The frequency of low release end-states (intact confinement and melt-through confinement failure) increases.
- Implementation of cavity flooding with independent system reduces the vulnerability to cavity door failure, by reducing the number of sequences with vessel failure. An automatic actuation is more beneficial than manual.
- Implementation of an effective system for hydrogen control would be beneficial in reducing contribution of very early containment failure. This measure should be combined with one of the others providing protection against cavity overpressure.

6 CONCLUSIONS

The PSA study provided information on the frequency of large radiological release and identified sequences contributing to such releases. Plant vulnerabilities in the area of containment performance have been identified.

The results of the study provided a basis for evaluation of risk reduction options and plant specific accident management strategies.

It is worth noting that the radiological risk at shutdown operational states is significant. It cannot be reduced without changing the CDF profile. In this context preventive SAM measures (at EOP level) are of high importance.

All the SAM measures investigated within the study for full power states (RCS depressurisation, cavity flooding, and hydrogen control) have been found beneficial. It can be noted that these measurs are being considered in SAMGs currently under development for V2 plant.

7 REFERENCES

[1] RELKO Ltd., “Integrated Level 1 PSA Model for Unit 3 of J. Bohunice V2 NPP, Updating of the model”, Rep. 1R1099 (1999).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, “Procedures for Conducting Probabilistic Safety Assessment of Nuclear Power Plants (Level 2)”, Safety Series No. 50-P-8, IAEA, Vienna (1995).

[3] US NUCLEAR REGULATORY COMMISSION, “Severe accident risks: an assessment for five U.S. Nuclear Power Plants”, NUREG-1150, USNRC, June 1989.

[4] PRIOR, R., “Review and categorisation of accident sequences”, (PHARE 93, 4.2.7a, Tasks 1 and 2), Rep. WENX-96-18, June 1996.

[5] BAKKER, P.J.T., “Task 11 – Confinement bypass risk analysis”, PH2.09/95-11R-DN01, February 1999.

[6] PRIOR R., CVAN M., “Confinement Bypass Risk Analysis, Source Term Report”, PH2.09/95-11R-WE02, Rep. WENX-99-14, February 1999.

[7] EUROPEAN COMMISSION, “Containment evaluation under severe accidents (CESA) – synthesis of the predictive calculations and analysis of the first experimental results obtained on the Civuax mock-up”, FISA 99, Conclusion symposium on shared cost and concerted actions, Rep. EUR 19532, pp 510-519.

[8] RELCON TEKNIK AB, “RiskSpetrum Users Manual”, Version 2.1, April 1994.

[9] ENCONET CONSULTING, “Quantification of the Shutdown PSA model”, Rep. PH2.09/95-6FR-EN01 (1998).