BWR-club PSA Benchmarking – Bottom LOCA during Outage, Reactor Level Measurement and Dominating Initiating Events

Anders Karlsson[*a], Maria Frisk[b], and Göran Hultqvist[c]
[a] Forsmarks Kraftgrupp AB, Östhammar, Sweden
[b] Risk Pilot AB, Stockholm, Sweden
[c] Havsbrus Consulting, Öregrund, Sweden

Abstract: Benchmarking is an important activity in order to eliminate unjustified differences between PSA models and enable harmonisation. It could also be used in order to understand plant differences. As part of the BWR-club PSA activities benchmarking of bottom LOCA during outage, reactor level measurement and dominating initiating events have been performed. Modelling of bottom LOCA during outage varies between the BWR-club members and work performed within the BWR-club aims at compiling and understanding these differences. When it comes to reactor level measurement modelling varies from a more detailed modelling to more of a "black box" approach. Information has also been collected from the BWR-club members regarding dominating initiating events in their PSA studies. The initiating event frequencies, scope of the PSA studies and risk importance of different initiating events vary between the BWR-club members and work has been performed compiling and understanding these differences. BWR-club reports have been issued for bottom LOCA during outage and reactor level measurement, while the benchmarking of dominating initiating events is yet to be finalised.

Keywords: PSA, Bottom LOCA during Outage, Reactor Level Measurement, Initiating Events

1. INTRODUCTION

This report deals with the benchmarking activities performed within the European network BWR-club. Twelve power plants (representing 17 BWR units) in Europe have filled in a questionnaire regarding bottom LOCA during outage in the PSAs of their boiling water reactors (BWRs). Ten power plants in Europe with 14 BWRs have filled out an extensive form considering reactor level measurements in PSA models. Work is on-going regarding dominating initiating events. The activities were presented at an earlier stage at the PSA 2013 conference in Columbia, South Carolina, USA in September 2013 [1].

The aim of this paper is to identify and highlight similarities and differences and to create a discussion that initiates efforts to harmonise and further develop PSA modelling in these three areas.

Anders Karlsson is working at Forsmark Nuclear Power Plant in Sweden, where Göran Hultqvist also worked before becoming a consultant. Maria Frisk is working at the consultant company Risk Pilot AB based in Stockholm, Sweden. This group has led this benchmarking project with the active participation of representatives from other BWR-club members.

1.1. BWR-club

BWR-club is a network consisting of European boiling water reactor (BWR) owners. The aim of the network is information and experience exchange between its members. This is performed through a yearly European conference, yearly European technical workshops (including one for PSA), benchmarking activities and the creation of BWR-club reports. BWR-club was initiated in 2010 by associated members leaving the BWROG associated program. BWR-club has developed contacts with Japanese utilities (JBOG) and maintains contacts with US utilities through BWROG.

[*] ask@forsmark.vattenfall.se

1.2. The Importance of Benchmarking in PSA

There are several reasons for performing PSA benchmarking. One reason is to understand differences in plant design and modelling. Another even more important reason is to identify how and why similar plant designs differ in modelling and to enable harmonisation. For events with no or little data and which at the same time may have large effect on PSA results, harmonisation is vital in order to gain PSA acceptance.

1.3. BWR-club PSA Benchmarking

This benchmarking project has been performed as a BWR-club PSA Workshop activity. The activities started in 2011 and have been on-going periodically since then. BWR-club reports on bottom LOCA during outage and reactor level measurement have been issued in the beginning of 2014 [2,3]. A BWR-club report on dominating initiating events is yet to be finalised and follow-up questions are to be sent out to the BWR-club members.

<u>Bottom LOCA during outage</u>

Modelling of bottom LOCA during outage varies significantly between the European BWRs. While it dominates the results for cold shutdown in some PSAs, and thereby also is major part of the total risk, it is screened out in other PSAs. Internationally some PSAs do not even consider the cold shutdown state at all. Bottom LOCAs during outage, in particular large ones, are typical examples of events with low frequency but with very serious consequences and weak barriers against core damage.

Is it justified to have such differences? What are the motives behind the differences: different plant designs, different PSA traditions, different requirements from regulators, etcetera?

<u>Reactor level measurement</u>

Reactor level measurement controls many systems of importance for plant safety and differences in modelling could therefore have a major impact on the overall PSA results. Different modelling of the reactor level measurement function could therefore lead to significantly different conclusions regarding the plant safety, even though the function itself often is more or less identical between different plants. This demonstrates the importance of benchmarking in the field of reactor level measurement.

<u>Dominating initiating events</u>

Benchmarking has also been initiated for dominating initiating events. The methodology for estimating initiating event frequencies is a much debated issue is Sweden, both when it comes to difference between the PSAs of different plants (Forsmark NPP, Oskarshamn NPP and Ringhals NPP) and when it comes to differences between probabilistic and deterministic analyses at the same plant (e.g. Forsmark NPP). This BWR-club benchmarking activity is partly connected a research and development project regarding initiating events performed within the Nordic PSA Group (NPSAG) called *Common Methodology for Analysis of Initiating Events* [4,5].

2. BOTTOM LOCA DURING OUTAGE

Bottom LOCA during outage is most probably caused be human error and the frequencies for those scenarios are estimated by using human reliability analysis (HRA). It could also occur as a mechanical failure or break in pipes or tubes which are the barrier for the reactor water. Such breaks may result in smaller break flows compared to the possible maximum break flows from human induced failures, which may lead to large open penetrations into the reactor pressure vessel (RPV).

There are a large number of HRA methods and the HRA results may also vary within an HRA method, even if the analysed manual actions are more of less similar [6,7].

The questionnaire sent out to the BWR-club members regarding modelling of bottom LOCA during outage was focusing on:

1. Main recirculation pumps
    a. External
    b. Internal
2. Control rod drives
3. Core instrumentation probes

Below follows a summary of the answers given to the questionnaire. The questionnaire was subsequently iterated in order to obtain as comparable answers as possible.

2.1. HRA Assessment of Maintenance Work

As failure data is scarce or non-existing for human induced failure causing bottom LOCA during outage, the failure data will have to be developed by methods used for assessing human errors.

Work with overhaul or replacement of:

- Valves in recirculation loop for external pump reactors
- Recirculation pumps in internal pump reactors
- Drive control rod drives
- Core instrumentation probes

is performed according to maintenance procedures. These procedures include critical steps that have to be performed correctly not to cause leakages of water out from the RPV. All critical steps include possible ways to stop minor leakages by going back in the procedures or by performing complementary actions. The assessment of risk for getting large openings in the RPV from such maintenance work include the risk for making mistakes and also the risk of not performing the countermeasures listed in the procedures. This means that if one mistake is leading to a small leakage and the countermeasures are successfully performed there is no bottom LOCA event. The initiating event frequencies for these bottom LOCA scenarios include the probability to get to a condition causing large leakages that cannot be stopped by following the maintenance procedures. The event trees do not include any recover actions to stop the flow as there are no such actions described in the manuals. The event trees include the responses from other plant functions and systems.

For units with internal main recirculation pumps there are technical solutions for avoiding leakages during dismounting of the pump e.g. sealing plugs (from above) and tight flanges (from below).

2.2. Main Recirculation Pumps

The function of main recirculation pumps is to cool the reactor core by recirculating the water inside the RPV and thereby maintaining an acceptable margin against dry out. Within certain limits the main recirculation pumps also controls the power of the reactor.

During outage maintenance work is performed on the main recirculation pumps, either with fuel still placed in the RPV or with the fuel removed from the RPV. Incorrect dismantling of main recirculation pumps may lead to very large bottom LOCAs. The main recirculation pumps may either be external, which means that they are connected to the RPV by piping, or internal, which means that they are attached directly to the RPV.

External pump reactors

The answers for units with external pumps are relatively similar regarding assumed flow rates (1895 – 2300 kg/s), but the frequencies vary significantly ($1.20 \cdot 10^{-7} - 1.87 \cdot 10^{-4}$ /year). It could also be noted that in the United States main recirculation pump LOCA during outage is normally not included in the PSAs.

One plant only assesses the risk for mechanical failures with probabilities in the range of 10^{-7} per year. The other plants are assessing the risk for human errors resulting in event frequencies between 10^{-4} down to 10^{-7} per year. The bases for these differences are not described in the responses. It indicates differences in HRA methodology. The break flows are large and the following scenarios will have small margins for avoiding core damage.

The plants have also submitted information on what precautions shall be establish during the period when actual overhaul or replacement are performed in order to avoid leakage from main recirculation pumps or to mitigate the consequences of leakage from main recirculation pumps. For units with external main recirculation pumps unloaded core and isolation valve locked close are common precautions made.

Internal pump reactors

The answers for units with internal pumps differ significantly both for assumed flow rates (circa 300 – 1580 kg/s) and for frequencies ($4.40 \cdot 10^{-8} - 4.52 \cdot 10^{-6}$ /year). It could again be noted that in the United States main recirculation pump LOCA during outage is normally not included in the PSAs.

Some plants have screened out large leakages from the PRV during outage as an event based on the strong administrative barriers. Others have not screened it out and the probabilities are in the level of 10^{-6} to 10^{-7} per year.

The resulting flow rates in the case of human error failure are high and the following scenarios will have small margins for avoiding core damage.

For some of the plants that have screened out large bottom LOCA during outage based on pump dismantling, have instead included bottom LOCA during outage related to mechanical failure (break). It is not clear if others have included these minor breaks in their assessments.

The plants have also submitted information on what precautions shall be establish during the period when overhaul or replacement are performed in order to avoid leakage from main recirculation pumps or to mitigate the consequences of leakage from main recirculation pumps.

For internal main recirculation pump reactors demanding containment airlock(s) to be closed are the precaution normally being done. Some procedures recommend closing the containment airlock(s) only after the failure has occurred. In the latter cases the time available for manual actions in order not to lose too much reactor water outside of the containment is small.

2.3. Control Rod Drives

Control rod drives are typically used for inserting control rods hydraulically during a scram or mechanically with the help of electrical motors (or high pressured water) controlling the power of the reactor. Incorrect dismantling of control rod drives may lead to medium or large bottom LOCAs. Compared to the main recirculation pumps the design of the control rod drives is more similar between the units in the survey, which makes comparisons easier. The answers varied extensively regarding assumed flow rates (7.1 – 210 kg/s) and the frequencies also varied significantly ($1.80 \cdot 10^{-6} - 1.00 \cdot 10^{-2}$ /year). In the United States control rod drive LOCA during outage is normally not included in the PSAs.

Two of the plants have assessed control rod drive LOCAs caused by the mechanical failures that can cause smaller leakages from the vessel in the range of 10 kg/s or lower. These plants have screened out or not assessed the risk for human errors during control rod drive maintenance.

All other plants have assessed human errors during control rod drive maintenance. The frequency varies from 10^{-4} to 10^{-6}. The flow rates are high and seem to be based on similar assessments of the end state of the failure. Different values are depending on plant design. The flow rates are high enough to transfer the plant into a scenario with high demands on safety systems and functions.

The plants have also submitted information on what precautions have been made in order to avoid leakage from control rod drives or to mitigate the consequences of leakage from control rod drives. Closing of containment airlock(s) in case of an event is a precaution normally being done. Other solutions involve technical specifications requiring at least two emergency core cooling trains available and the use of sealing plugs.

2.4. Core Instrumentation Probes

Core instrumentation probes are used for monitoring the reactor core. Incorrect dismantling of core instrumentation probes may lead to significant but much smaller bottom LOCAs than what is assumed for main recirculation pumps and also normally smaller than what is assumed for control rod drives. Like for the control rod drives the designs are relatively similar regarding core instrumentation probes, which make comparisons easier. In the United States core instrumentation probe LOCA during outage is normally not included in the PSAs.

The answers vary somewhat regarding assumed flow rates (5 – 20 kg/s) and frequencies ($3.20 \cdot 10^{-4}$ – $3.40 \cdot 10^{-3}$ /year).

One of the plants has only assessed LOCAs caused by the mechanical failures. The failure rate is very low and this plant has screened out or not assessed the risk for human errors during core instrumentation probe maintenance.

Several plants have screened out the failure scenarios based on the long time needed for getting into a critical core cooling condition depending on the flow rates and the available water above the core during outages.

The other plants have assessed human errors during core instrumentation probe maintenance. The frequency varies from 10^{-3} to 10^{-4} with most in the $3 \cdot 10^{-4}$ to $4 \cdot 10^{-4}$ range. The flow rates are in the range of 10 – 20 kg/s and seem to be based on similar assessments of the end state of the failure, but with different values depending on plant design. The flow rates are such that safety system and functions should be able to handle the scenarios if these functions are working properly. With reduced function of safety system these scenarios will cause core damage.

The risk for core damages will be dependent on the frequencies for getting into the failure state during core instrumentation probe maintenance or if these failure modes are screened out.

The plants have also submitted information on what precautions have been made in order to avoid leakage from core instrumentation probes or to mitigate the consequences of leakage from core instrumentation probes. Closing of containment airlock(s) in case of an event is a precaution normally being made. Other solutions involve technical specifications requiring at least two emergency core cooling trains available.

2.5. Discussion

The lessons learnt from the responses are that the input data for bottom LOCA events during outage are developed based on quite different ways of assessing the risk for different failure modes. Some plants even screen out human induced failures for these category of event. It is unclear whether this is done based on detailed failure mode assessments or on an "engineering judgement" base. The plants that screen out human induced failures instead assess failures based on mechanical failures.

Plants that perform assessment of human induced failures seem to get data with large uncertainties. It is however not acceptable to end up with such large variation in the results. There is thus a need to develop a common methodology among BWR owners on estimating the frequencies for leakage from the RPV during outages.

Another issue to be addressed as a result of this benchmarking is what kinds of water flows that should be considered. Should it be the worst case scenario, the most probable or both? To obtain a deeper understanding it is also important to reflect on PSA and HRA differences regarding the estimation of frequencies. The HRA methodology is also of importance. Other forms of bottom LOCAs during outage are another area to be addressed. This could for example be LOCAs caused dropping heavy equipment or LOCAs affecting spent fuel pools.

For more detailed information see [2].

3. REACTOR LEVEL MEASUREMENT

Reactor level measurement in BWR-reactors is performed by measuring the weight of the water by assessing the pressure difference between the reference chamber and a position that is under water. See Figure 1.

During normal operation with steady state condition this measurement can with high accuracy (in the range of centimetres) measure the water level in the RPV. By assuring that the instrument pipes, sensing lines, connecting the pressure transmitter with the reactor vessel and the reference chamber is filled with degased water and that the difference pressure transmitters are calibrated the operators can rely on the output from these level measurements and the logics steered by the measurements. As the difference pressure is depending on the temperature of the water and on the temperature outside the instrument pipes most plants are equipped with measurement of these temperatures. The measured differential pressure is compensated or corrected by the outputs from these temperature measurements. If this compensation does not work or do not exist the output will not present the actual level correct when these parameter are changed. This happens when the pressure is dropped in the reactor vessel or a pipe-break occurs in the containment.

When the temperature in containment increases (or decrease) this will also affect the weight of the water in the sensing lines and the differential pressure changes. With very high temperature in containment it will also be possible to boil the water in the sensing lines.

Some plants have installed functions to cool the sensing line during scenarios when there is risk for high temperature in containment. Some plants have installed their reference chamber in such a way that degasing of the steam entering into the reference chamber is not transferred back to the reactor vessel. This can lead to an accumulation of non-condensable gases in the reference chamber and also into the water in the instrument lines. With gas in the water the weight of the water changes and effects the measurements.

During a rapid depressurisation the gas in the water will also result in flashing out water from the instrument lines. For plant with reference chamber lacking natural degasing it is of importance handle degasing by other means.

As parts of the reactor level measurement system is connected to the RPV and installed in the containment changes of parameters are transferred to all trains of the systems. If some of these changes introduce failures into one train it is very likely that this will also occur in the other trains and even in parallel measuring systems measuring on other levels.

The output from a level transmitter and its amplifier is assessed by limit switches to initiate safety systems if water level passes certain levels. The operators get information by the instrument in the control room. If the instrument fails it will not affect the limit switch function.

A reactor level measurement system needs power to its transmitters, amplifiers and compensators. The transmitters and amplifiers are at least calibrated once per year after the outage as well as the limit switches. Failure of the reactor level measurement system could result in too early or too late actuation of safety system. This will also happen in case of loss of power to the different sub-components. The worst case will be cases when a transmitter or amplifier is blocked on a certain level and this happens in several trains simultaneously.

To reduce the risk for getting the level signal to freeze in a specific level in many trains at the same time some plants have installed automatic system that continuously (within some seconds) compare the outputs from measurements in different trains. If the outputs deviate in one train from the other the operators are informed and corrective actions can be initiated. This function reduces the risk for common cause failure (CCF) in the reactor level measurement system.

3.1. Analysis

In general the European power plants model the reactor level measurement with high degree of detail. In the United States on the other hand reactor level measurement is modelled as a “black box” with only a few parameters as input. Europe and the United States seem to have different opinions about how reactor level measurement should be modelled in PSA studies.

Beneath the PSA modelling in Europe is assessed by looking into the different parts of the modelling and its data.

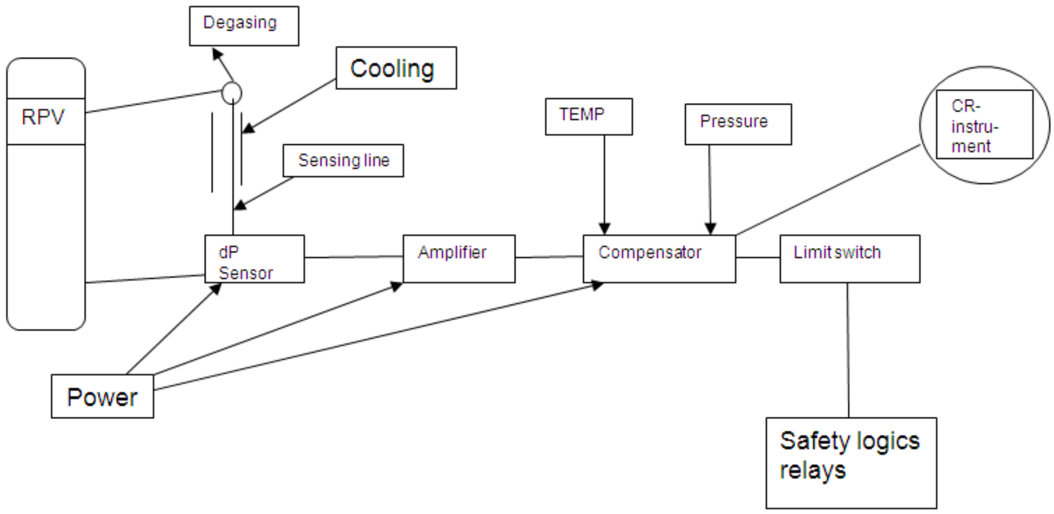
<u>Failure modes</u>

When it comes to failure modes the PSAs of the BWR-club members have many similarities, the following failure modes were modelled for a majority of the plants:

- Loss of function of dP (differential pressure) sensor
- No movement in limit switch
- Spurious movements in limit switch
- Power supply for components (modelled as fault tree or basic event)
- Incorrect calibration of dP sensor (HRA)
- Loss of function input relay in safety logics

Figure 1 illustrates a typical reactor level measurement function of a BWR.

Figure 1: Reactor Level Measurement Function



The reactor level measurement function is relatively well covered in the analysed PSAs. There are however a few failure modes that are generally not modelled.

- Loss of function sensing line[†]
- Loss of function amplifier
- Loss of function compensator

There are clearly more similarities than differences regarding reactor level measurement failure modes. The level of detail in modelling of power supply for components varies however.

Some differences can of course be explained by differences in design. But in case of very similar power plants the PSA-studies should be comparable and have similar parameters as input.

It is assumed that most plants calibrate their dP sensors during outage one time per year. The responses however indicate that at least one plant does this every three months.

At one plant the reactor level measurement system is viewed as a monitored system. This effect the failure rate of:

- Loss of function dP sensor
- Spurious movements in limit switch

This view is based on the fact that the system is used in the daily operation of the plant and failures will be detected by the operators without doing any tests.

Repair time for any failure on those components is set to eight hours in the modelling of this plant. This also affects the view of modelling CCF, where this plant does not include CCF on these components.

At another plant the reactor level measurement system is complemented with an automatic function that continuously compare the output of from all level transmitters in all trains. Alarm is given if any transmitter has an output that has a minor deviation from the others. This has been used in the modelling to indicate a test frequency of one per week, even though the comparisons are made every second. This affects the modelling of the dP sensors. These are important differences in the modelling assumptions.

[†] If modelled it is the cooling of the sensing line that is considered.

The responses indicate that the modelling assume that when there is a failure the output will change to a maximum or minimum output. Such a failure mode will in most plant result in actuation of safety logics. No plant have modelled that there should be a failure in the dP sensor that freeze the output to a specific value and that any changes in the water level are not recorded and no safety logics are activated.

Common cause failure (CCF) groups

The answers to the questionnaire show that CCF groups are modelled relatively similar among the BWR-club members, even though there are also differences, mainly regarding the level of ambition to include CCF in the modelling of reactor level measurement.

The following CCF groups are the most common:

- No movement of limit switch – modelled at seven plants
- No signal from dP sensor (loss of function) – modelled at six plants
- Incorrect calibration of dP sensor (HRA) – modelled at three plants

As indicated above one plant does not model any CCF regarding reactor level measurement, as the system is monitored during normal operation.

Plant design regarding number of trains and the diversification of trains varies between the plants. Often the reactor level measurement is divided into coarse and narrow range.

Should all trains (coarse range as well as narrow range) be seen as one large CCF group or should coarse range and narrow range be separate CCF groups? The responses indicate that all members have separate CCF groups. Work is however on-going at one plant to assess a large CCF covering all measuring ranges.

Data and modelling

When it comes to data and modelling there are differences. Parameter values vary since different parameter sources are used. In the Nordic countries the T-book [8] is used, while other failure data sources are used in other countries. The test intervals also vary between the different plants.

The different data sources give a difference in failure data in the level of one decade, see Table 1. For these failure modes with almost similar components the failure rates should be within one third of a decade to be able to develop PSAs with low uncertainties. This implies that a work needs to be done regarding harmonisation of failure data.

Table 1: Reactor Level Measurement Failure Rates

Failure mode	**Highest failure rate**	**Lowest failure rate**
Loss of function sensing line	$2.4 \cdot 10^{-6}$	$6.00 \cdot 10^{-7}$
Loss of function dP sensor	$1.00 \cdot 10^{-6}$	$1.21 \cdot 10^{-7}$
No movement in limit switch	$1.94 \cdot 10^{-7}$	$4.91 \cdot 10^{-8}$
Spurious movements of limit switch	$3.80 \cdot 10^{-7}$	$4.6 \cdot 10^{-7}$
Loss of function input relay in safety logics	$1.76 \cdot 10^{-8}$	$8.40 \cdot 10^{-8}$

The test intervals are relevant for the following data is presented in Table 2.

Table 2: Reactor Level Measurement Test Intervals

Test	Typical test interval	Variation
Function dP sensor	1 year	1 month for one plant
Movement in limit switch	3 months	Between once a 1 month to 1 year
Function input relay in safety logics	3 months	No variation between the plants

Modelling of power supply for components of the reactor level measurement function varies. It is mainly a difference on level of detail. Power supply for level measurement components is often modelled in detail with the use of failure trees, considering several failure modes and considering CCF. The power supply could however also be modelled as a "black box" using only a few basic events. The power used for components in the reactor level measurement function is all DC power systems with battery back-up. Events with failures of the DC power systems will affect different parts of the measuring system. With a failure tree modelling it will be possible to catch the connections between loss of power and loss of measurement. The importance of this cannot be evaluated as no data have been delivered. When power fails to power level measurement functions it will give alarm or safety actuations therefore the safety risk will be low.

Total failure rates of a train

In table 3 data of unavailability of a reactor level measurement system is presented.

Table 3: Reactor Level Measurement – Total Failure Rates of a Train

Plant	Description	Result
1	One train	$4.38 \cdot 10^{-2}$
2	RPS, one train	$1.4 \cdot 10^{-5} – 8.003 \cdot 10^{-4}$
3	RPS, one train	$5.091 \cdot 10^{-4}$
4	RPS, one train	$7.6 \cdot 10^{-4}$

Failure of the reactor level measurement function and failure to actuate safety logics is depending on the ways the reactor level measurement is built into different functions. The data above indicate different outputs from existing studies.

A spread of unavailability in the range of 10^{-2} to 10^{-5}, with a lot of data at 10^{-5}, indicates a too large uncertainty of data for such an important function in a BWR plant.

Perhaps more importantly the modelling of power supply for components of the reactor level measurement function varies. It is mainly a difference regarding level of detail. Power supply for level measurement components is often modelled in detail with the use of fault trees, considering several failure modes and considering CCF. The power supply could however also be modelled as a "black box" using only a few basic events.

3.2. Discussion

Failure modes and the use of CCF are relatively similar between the plants, while parameter data vary to a larger extent, which at least to some extent is inevitable since different parameter sources are used. Harmonisation in the field of failure data is outside the scope of this paper. Such work has however been performed [9] but focused on harmonisation of methodologies not harmonisation of data.

The most important finding in this investigating of reactor level measurement are the differences in modelling of power supply for components of the reactor level measurement function varies. Should it be modelled in detail (as most plants do) or is "black box" modelling also acceptable?

For more detailed information see [3].

4. DOMINATING INITIATING EVENTS

Benchmarking regarding dominating initiating events is on-going. A questionnaire was sent out and representatives from the BWR-club members submitted information from their PSAs. This information has then been compiled. Some follow-up questions will however be sent out during spring 2014.

This benchmarking deals with power operation as well as shutdown and the plants have been asked to give information about their top 20 events with the highest initiating event frequency (all events as well as external events excluded) as well as the initiating events that are part of the 20 most dominating core damage frequency cut sets in their PSAs (all events as well as external events excluded).

One early conclusion is that the scope and the impact of external events vary significantly between the plants. Seismic events might for example dominate in some PSAs, even in areas with low seismicity, while it is screened out in other PSAs. The importance of common cause initiators (CCIs) also vary significantly. It is clear that there are different plant designs, but also different "PSA traditions" and different requirements from regulators, all leading up to different risk profiles in the PSAs.

The work will continue during 2014 and lead to a BWR-club report.

5. CONCLUSIONS

Different methodologies may lead to unjustified differences affecting safety prioritisations at the plants, even if the plant designs are similar. One obvious conclusion of this benchmarking activity is thus that methodologies, regarding for example failure modes, should be harmonised when plant designs are similar.

The lessons learnt from the responses are that the input data for bottom LOCA events during outage are developed based on quite different ways of assessing the risk for different failure modes. Some plants even screen out human induced failures for this category of event. It is unclear whether this is done based on detailed failure mode assessments or on an "engineering judgement" base. The plants that screen out human induced failures instead assess failures based on mechanical failures.

Plants that perform assessment of human induced failures seem to get data with large uncertainties. It is however not acceptable to end up with such large variation in the results. There is thus a need to develop a common methodology among BWR owners on estimating the frequencies for leakage from the RPV during outages.

Another interesting outcome of the benchmarking activities is what kinds of water flows that should be considered for bottom LOCAs during outage. Should it be the worst case scenario, the most probable or both? To obtain a deeper understanding it is also important to reflect on PSA and HRA differences regarding the estimation of frequencies. The HRA methodology is also of importance.

For reactor level measurement failure modes and the use of CCF are relatively similar between the plants, while parameter data vary to a larger extent, which at least to some extent is inevitable since different parameter sources are used.

An important finding is the differences in modelling of power supply for components of the reactor level measurement function. It is mainly a difference regarding level of detail. Power supply for level measurement components is often modelled in detail with the use of fault trees, considering several failure modes and considering CCF. The power supply could however also be modelled as a "black box" using only a few basic events. Should it be modelled in detail or is "black box" modelling also acceptable?

It is in general of interest to investigate on which bases events are screened out in some PSAs, which they dominate the results in other PSAs. Seismic events and other external events as well as bottom LOCA during outage are examples of that. It illustrates the problem of analysing events which have low frequencies and high degrees of uncertainty, which at the same time have very serious consequences and weak barriers against core damage.

It is clear that different plant designs, but also different "PSA traditions" and different requirements from regulators, lead up to different risk profiles in the PSAs. There is in other words a large need for harmonisation. To obtain harmonisation is however a long process and work must be performed in close cooperation with more resources than what is possible in this benchmarking activity.

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