# DECOMMISSIONING LEVEL 2 PROBABILISTIC RISK ASSESSMENT METHODOLOGY FOR BOILING WATER REACTORS

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*This paper describes a methodology to evaluate the radiological risk of postulated accident scenarios during the decommissioning phase of commercial nuclear power plants. The fuel damage scenarios include those initiated while the reactor is permanently shut down and the spent fuel is initially located in the reactor pressure vessel and successively moved in the spent fuel storage pool. This paper focuses on the Level 2 PRA (post fuel damage) aspects of the analysis. The Level 2 decommissioning PRA uses event trees that assess the accident progression post spent fuel damage. Detailed deterministic severe accident analyses are performed to support the event tree development (e.g., success criteria and accident progression timings) and to provide source term information for the various release categories. Results are derived from the Level 2 decommissioning PRA event tree in terms of Large Release Frequency (LRF) and Large Early Release Frequency (LERF).*

## I. INTRODUCTION

PRAs are structured and logical analysis methods aimed at identifying and assessing risks in complex technological systems for the purpose of cost-effectively improving their safety and performance. In a PRA of Nuclear Power Plants (NPPs), accident scenarios are analyzed with event trees and/or fault trees. An event tree graphically represents the various accident scenarios that can occur after an initiating event. An event tree starts with an initiating event and develops scenarios, based on whether a plant system succeeds or fails in performing its function. The event tree then considers all of the related systems that could respond to an initiating event, i.e., the so-called functional events or top events, until the sequence ends in either a safe recovery or reactor core damage [1].

The PRA provides insights into the strengths and weaknesses of the design and operation of a nuclear power plant. PRA analysts typically estimate the risk developing the so-called Level 1, Level 2 PRA, and sometimes Level 3 PRA [1]. A typical approach is to end the analysis at the Level 2 PRA characterizing the accident consequences in terms of activity release rather than health consequences. The focus of this paper is to describe a methodology for the estimation of radioactive release during the decommissioning of the plant.

The technical approach used to evaluate the severe accident progressions for radiological releases to the environment involved a combination of deterministic and probabilistic models [2]. A baseline or 'best estimate' characterization of accident event chronology, mechanical loads on the containment, and/or reactor building and fission product release to the environment is generated using detailed, plant-specific calculations of severe accident behavior using a thermal-hydraulic code. The MELCOR severe accident progression code [3] was used for the deterministic portion of this methodology. Sensitivity calculations are also needed for selected sequences using the NPP MELCOR thermal-hydraulic model to measure the effects of major modeling uncertainties on the calculated results. Furthermore, the physical effects of some phenomena are not explicitly calculated by MELCOR (e.g., hydrogen detonation) or are represented by relatively simple models. Therefore, alternative sources of information are needed to make informed decisions on an appropriate modeling approach, and to supplement MELCOR calculations in areas that calculations are deficient.

A probabilistic model is used to specify the needed accident progression analyses and weigh the deterministic accident progression outcomes by applying event probabilities. The Level 2 model uses Accident Progression Event Trees (APETs), i.e., organized frameworks for delineating the various ways in which a fuel damage sequence (defined in the Level 1 PRA) could proceed in time through potential radioactive release to the environment. Similarly to low power-shutdown accidents [4], APETs developed for the Level 2 decommissioning PRA differs from APETs developed for full power sequences in some respects. First, the decommissioning APETs are considerably less complicated, due to the relatively simple nature of the accident sequences that can occur during decommissioning states. Second, accident scenarios can be initiated either in the

RPV, in Spent Fuel Pool (SFP), or in both. Therefore, accident progressions initiating in RPV, SFP, or both must be combined to obtain the radiological consequences from the results of the RPV and SFP fuel damage calculations. Consequently, a hybrid methodology needs to be developed to examine the fuel degradation progression in the RPV and the SFP separately. The results from the two calculations are combined into an integrated calculation using the thermal-hydraulic and fission product sources from the respective calculations.

This paper presents a methodology for the estimation of large (early) release during the decommissioning period of a General Electric Mark I boiling water reactor. The numerical results reported in this paper are provided to demonstrate the application of the proposed methodology and do not necessarily reflect calculated risk results of an actual nuclear power plant.

## II DEFINITION OF DECOMMISSIONING PHASES

Several configurations can be assumed to estimate the risk of the plant during decommissioning. In this paper, two main phases are proposed and analyzed: the first phase covers the permanent shutdown of the plant starting from full power to cold shutdown whereas the second phase covers the actual decommissioning phase when the fuel is transferred to the spent fuel pool before the final disposal.

The analysis of the permanent shutdown state of the plant cannot be addressed as a single state due to numerous changes in plant configuration. Multiple states must be defined, with each state being modeled explicitly using a fault tree/event tree methodology to quantify the fuel damage frequency. The identification of the shutdown phases is done considering two important parameters: i) the plant modes and ii) the decommissioning characteristics. The plant modes important for the decommissioning analysis are: cold shutdown, SFP movement, and actual decommissioning. Additionally, within each plant mode, the plant parameters and the status of some systems can change; therefore, the characteristics within each plant mode are important in order to identify additional phases. The main characteristics of interest are: primary system pressure, status of reactor vessel head, status of the reactor vessel well, location of fuel (RPV, SFP, or both), and availability of safety and support systems. Plant data, procedures, and working plan are good sources for the identification of the phases.

The number of phases can be large: yet, the finer the modeling, the more precise the estimation of the FDF. However, a compromise must be found between level of detail and practicability. In this paper, four phases are considered and are summarized in Table 1. For each decommissioning phase, the accident propagation will be analyzed using a linked event tree/fault tree approach.

TABLE 1. Summary of decommissioning phases.

| Phase | Description | Mode | Characteristics | | | | |
|---|---|---|---|---|---|---|---|
| | | | Primary Pressure | RPV Head | RPV Well | Fuel Location[1] | Available systems |
| P-1 | From start of cooling to RPV head removal | Cold SD | High to Low | ON | Dry | RPV | ALL |
| P-2 | From RPV head removal to fuel pool dam removal | Cold SD | Low | OFF | Dry to Flooded | RPV | ALL |
| P-3 | From fuel pool dam removal to end of offload | Fuel movement | Low | OFF | Flooded | RPV/SFP | Several systems unavailable |
| P-4 | From end of offload to fuel removal from SFP | Actual decommissioning | Low | OFF | Flooded | SFP | Several systems unavailable |

Note 1. This column identifies the location of the fuel from the most recent power operations prior to decommissioning. Most plants will also have fuel in the SFP from previous fuel cycles. The accident progression analyses include the older fuel in the SFP.

## III PLANT DAMAGE STATE DEFINITION AND INPUT TO DECOMMISSIONING LEVEL 2 PRA

The Level 1 decommissioning PRA identifies the combinations of system failures and/or human errors that can lead to damage of irradiated fuel in the RPV and/or the fuel storage pool. Each fuel damage sequence is assigned to a unique PDS based on the status of key plant systems (available or unavailable). The PDS development process integrates results of the Level 1 PRA for all contributions to the total decommissioning FDF including sequences initiated by internal events and external events. The attributes used to combine individual accident sequences from the Level 1 PRA to PDSs are at least: Sequence Category, i.e., internal events (IE), other internal hazards (IH), external hazards (EXT), seismic (SEIS); Accident phase, i.e., phases P-1 through P-4; Available or unavailable systems; Crane Status; Initial Fuel Status; SFP Status; Reactor Building Status; SFP Dam Status; SFP leakage rate; SFP leakage location. Note that for each PDS AC power is assumed not available and the RPV not pressurized.

Based on the Level 1 PRA decommissioning results, systems and AC power may be or may not be available in the Level 2 analysis. The crane status attribute asks whether the crane fails damaging the pool. The initial fuel status attribute accounts for the fact that the fuel can or cannot be mechanically damaged at the beginning of the accident progression. The SFP status, reactor building status, and the SFP dam status attributes ask for the integrity of the SFP, reactor building, and SFP dam respectively at the beginning of the accident progression. Each of these structures can either be intact or damaged (mostly due to seismic events). The SFP leakage rate and leakage elevation attributes account for any loss of coolant in the SFP: the amount of the leakage (large, medium, or small), and the location of the leakage (bottom of the pool or any other locations within the pool). The characteristics of the severe accident progression are specified by these attributes and are described in the following sections.

Based on the Level 1 PRA decommissioning accident sequence results, PDSs are identified for each phase of the decommissioning identified in Table 1. As an example, Table 2 shows an example of PDS attributes for phase P-4. Sequences mapped to specific PDSs are identified by analyzing cutsets and using flags to identify what systems are available and not available. The specific attributes assigned to each PDS are used to develop Level 2 APETs and MELCOR accident progression inputs.

TABLE 2. Details of PDS characteristics in phase P-4.

| Phase, Plant Damage State | Description | Sequence categories | Unavailable Systems | AC power | Crane status | Initial fuel status | SFP status | RB status | SFP dam status | Leakage rate | Leakage location |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **P-4, PDS-01** | Boil-off Scenarios | IE, IH, EXT, SEIS | ALL | NONE | OK | OK | OK | OK | OK | N/A | N/A |
| **P-4, PDS-02** | Dam Failure Scenarios | SEIS | ALL | NONE | OK | OK | OK | OK | FAILED | Large through dam, then boil-off | Only dam failure |
| **P-4, PDS-03** | SFP Failure Scenarios | SEIS | ALL | NONE | OK | OK | DAMAGED | OK | OK | Medium | Bottom of pool |
| **P-4, PDS-04** | Crane Failure Scenarios | SEIS | ALL | NONE | FAILED | DAMAGED | DAMAGED | OK | OK | Small and Medium | Bottom of the pool to 1/4 of active fuel |
| **P-4, PDS-05** | RB Failure Scenarios | EXT, SEIS | ALL | NONE | OK | DAMAGED | DAMAGED | FAILED | OK | Large | Bottom of pool |
| **P-4, PDS-06** | Non-Seismic Leakage Scenarios | IE, IH, EXT, SEIS | ALL | NONE | OK | OK | OK | OK | OK | Small | Bottom of pool |

## IV LEVEL 2 PRA – DECOMMISSIONING APET APPROACH

Severe accident progression and fission product source term calculations are performed for each PDS. The calculations generate information needed to characterize the chronology of key events as well as the magnitude of timing of radiological releases for each of the PDSs identified in the Level 1 PRA. These results are integrated within the APET framework that delineates various ways in which a fuel damage sequence could proceed in time (see Fig. 1). For a given accident sequence, plausible variations in severe accident progression arise from a combination of the systemic (or random) probabilities that available accident mitigation systems could fail to adequately perform their function and from limitations in our ability to confidently predict the evolution of severe accident phenomena with contemporary computer models. As a result, a single accident sequence identified in the Level 1 assessment of fuel damage frequency results in several plausible severe accident progressions in the Level 2 PRA. The relative probability of each of these accident progressions is evaluated in the APET.

The definition of the APET end state source terms are developed directly from the deterministic MELCOR results. Specific integrated RPV and SFP MELCOR source term calculations are performed to characterize the sequence source term. Consequently, each APET sequence is assigned to a specific integrated MELCOR calculation. The results utilized from the MELCOR calculations include (a) the timing of the start of radioactive release to the environment, (b) the total Cs-137 activity released the environment, (c) the total activity released to the environment, and (d) an indication whether the I-131 activity released to the environment exceeded the large early release criteria. Consequently, unlike other PRA methodologies that calculate the source term based on the accident progression characteristics using algorithms derived from deterministic calculations, the present approach directly uses the results from the deterministic calculations with the APET. The current approach does not allow for complex variations in attributes that may affect the source term. However, each result in the sequence source terms can be directly tied to a mechanistic evaluation of the accident progression. All complex interactions affecting the source term are calculated using the physics models within MELCOR.

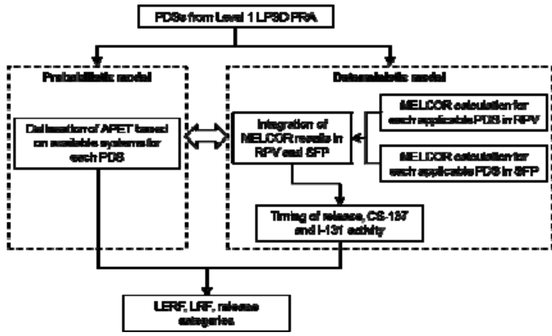


Fig. 1. Overview of probabilistic and deterministic (RPV and SFP) approach.

**IV.A. Deterministic evaluation of Decommissioning sequences**

The technical foundation of the Level 2 decommissioning PRA will be developed from results of plant-specific calculations of severe accident progression using the MELCOR computer code. The MELCOR model has the ability to evaluate accident progression and fission product release from fuel located either in the RPV or in the SFP but not from both locations in a single calculation (i.e., a typical limitation in most severe accident codes). Consequently, a hybrid methodology is used to examine the fuel degradation progression in the RPV and the SFP separately. The results from the two calculations are combined into an integrated calculation using the thermal-hydraulic and fission product sources from the respective calculations. In this manner, the chronology of severe accident events and associated radionuclide release to the environment can be calculated for the full range of accidents identified in the Level 1 PRA.

*IV.A.1. RPV accident progression model*

The MELCOR calculations performed in support of the Level 2 decommissioning PRA are chosen to provide detailed information on accident progression and fission product source terms for the PDSs identified in the Level 1 PRA. MELCOR calculations for each and every PDS are impractical to be performed. Therefore, only calculations for a group of PDSs that could successfully represent the range of conditions associated with the accident initiator, expected severe accident progressions, and the scope of fission product source terms across all PDSs can be performed. Criteria used to select representative PDSs included the following: i) PDSs with significant frequency (in general, the PDSs that comprise at least the top 99% of the total), ii) PDSs expected to lead to large fission product source terms (e.g., sequences initiated with failure of the reactor building), iii) PDSs from each phase of the decommissioning phase (i.e., Phases P-1 through P-4), iv) PDSs expected to rapidly progress to fuel damage (e.g., unisolable LOCA initiators), and v) PDSs that characterize the response with fuel in various states of off-load.

After applying these selection criteria, a series of representative PDSs are identified for baseline MELCOR calculations. The RPV accident progression calculations in Phase P-1 occur at the start of the permanent shutdown. The reactor state and configuration varies from the end of full-power operations to cold shutdown. However, the decay power is relatively low because the reactor is shut down. Since the decay heat and the system fluid temperature and pressure are lower than full-power sequences, the response of the reactor system to accidents is slower than full power accidents from high system pressure.

RPV calculations in Phases P-2 and P-3 are complicated by the status of the hydraulic connections to the reactor well during the shutdown operations. In Phase P-2, the spent fuel pool is usually isolated from the reactor well by a dam. Unless the SFP dam fails due to a seismic event, the accident in the RPV occurs independentlydam of an accident in the spent fuel pool. The reactor model in these phases includes a configuration where the reactor and the drywell heads are removed. The thermal-hydraulic response of the spent fuel pool includes the appropriate decay heat load.

The decay and radioactive inventory is specified to correspond to the time of the decommissioning phase. Prior to the accident progression, steady state simulations might be run to establish the nominal conditions for each phase (i.e., decay heat power, water temperature, and water level) to produce the input parameters for the accident RPV accident progression calculation.

*IV.A.1. SFP accident progression model*

The SFP fuel degradation calculations are performed by using a specially developed models. Unique SFP models are developed for each phase of the accident to represent the hydraulic connections to the RPV well, the number of fuel assemblies in the SFP, and the appropriate decay heat and radioactive inventory. A matrix of calculations is performed that simulate the following conditions: i) two shutdown configurations: prior to offload with the spent fuel pool dam installed and during shutdown with the dam removed, ii) four scenarios: Loss-of-decay heat removal (LODHR), RPV LOCA that quickly drains the SFP to the top of the SFP dam, SFP LOCA which is conservatively modeled as a hole at the 25% active fuel elevation, iii) SFP dam failure, and iv) two offload configurations: prior to offload and 100% offload.

Additional spent fuel pool calculations are performed to examine initial mechanical damage to the fuel in the spent fuel pool due to collapsing structures in the reactor building dome (e.g., crane). These calculations characterize the impact of initial seismic damage on the spent fuel pool accident progression.

The SFP MELCOR model is also slightly modified to include hydraulic connections to the reactor well. The decay heat and radioactive inventory is specified to correspond to the time of the PDS. The thermal-hydraulic boundary conditions from the RPV are characterized by using a heat structure model of the fuel assemblies in the RPV. The heat structure accurately characterizes the thermal mass of the fuel in the RPV (i.e., if any in that phase).

*IV.A.3. Integrated deterministic calculations*

Integrated MELCOR calculations are performed to combine the radiological consequences from the results of the RPV and SFP fuel damage calculations respectively. The integrated calculation is specified by extracting the thermal-hydraulic and radionuclide results from the separate RPV and SFP calculations as boundary conditions for the integrated reactor building calculation. The RPV radionuclide and thermal-hydraulic sources are realistically specified at various locations including, (a) the exit from the RPV to the reactor well (i.e., if the vessel head is removed or via drywell leakage), (b) through the open personnel hatch or containment drywell failure location, (c) through a pipe break, and (d) and into the containment venting system. If molten core-concrete interactions (MCCIs) occurred in the containment, then the thermal-hydraulic sources also include hydrogen, carbon monoxide, and carbon dioxide and as well as any released radionuclides. In the same manner, the time-dependent of release of radionuclides and thermal-hydraulic sources (i.e., steam and hydrogen) from the spent fuel pool are specified for release into the RB dome.

### IV.B. Decommissioning Level 2 PRA framework

The Level 2 decommissioning PRA APET is constructed within the framework of commercial event tree software (see Fig. 2 and Fig. 3). In this paper CAFTA [5] has been used. The APET delineates major junctures in severe accident progression and links together as a single, chronological series of events, which can either occur or not occur. Uncertainty in the occurrence of these events is treated probabilistically to generate a spectrum of possible accident progression sequences that could develop from a particular PDS. Each combination of events is therefore characterized by a relative probability and an associated radionuclide source term.

The APET uses information on system status from the definition of each PDS as input and affects the manner in which the probability of some events in the APET is quantified. For example, the probability that ignition sources are available in the reactor building to ignite flammable mixtures of hydrogen gas in the reactor building (RB) depends on the availability of ac power. This dependency is accounted for the probability assigned to the APET event representing early hydrogen ignition in the RB.

The APET mainly considers uncertainties in the physical response of the reactor building due to the release and combustion of hydrogen. Hydrogen combustion is the dominant mechanism for generating pressures above capacity of the building. In contrast to the primary containment (during full power accident sequences), the slow release of steam and non-condensable gas into the RB atmosphere is not an important mechanism for reactor building over-pressure because of reactor building leakage. Pressure increases due to hydrogen combustion (both deflagrations and detonations) can occur on time scales faster than the time required by gases to leak. Therefore, the potential for hydrogen combustion dominates the probability of RB failure.

The evaluation of RB integrity analyzed in this paper involves four APET top events: MCCI, early ignition, H2 detonation, and H2 deflagration.

The APET is then solved considering the probability of each top event and generating the final probability distribution of each end state. Using Monte Carlo methods, the probabilities of the top events are randomly sampled across their respective probability distributions to determine the likelihood of each PDS end state. From the previous deterministic accident progression calculations, each end state has a specific consequence measures (e.g., LR, LER, and total activity release). Simultaneously, the relative frequency of each PDS is sampled and combined all other PDS. Consequently, the integrated result includes sampling of the PDS frequency from the Level 1 characterization and sampling of the top events in each PDS. Standard statistical techniques are used characterize the results including the mean and median of each consequence measure and its distribution with the 5$^{th}$ percentile, and the 95$^{th}$ percentile.

The distributions of top events are determined evaluations of available technical studies (i.e., for hydrogen detonation [6]) to develop parameter distributions (usually conservatively specified through broad normal or lognormal distributions). Sensitivity analyses are performed to assess the impact of assumptions in the top events probability on the overall risk.

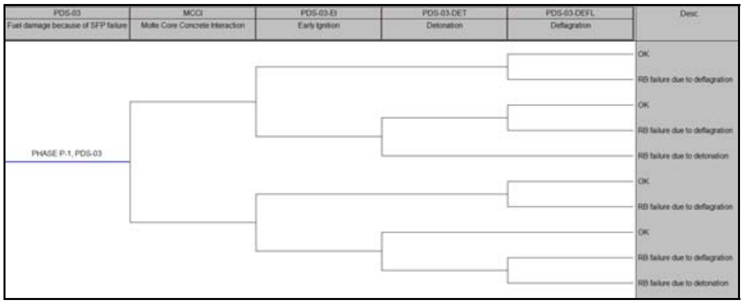


Fig. 2. Example APET (phase P-1 through P-3).

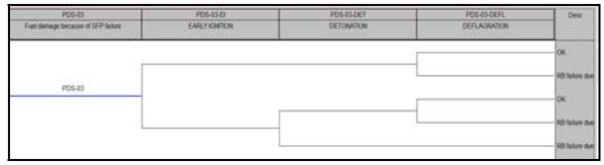


Fig. 3. Example APET (phase P-4).

**IV.C. Decommissioning APET development**

APET top events model the likelihood of MCCI, early ignition, detonation, or deflagration of hydrogen. APET top events are described in detail in the next sections. If MCCI occurs, then fission products can be released from the containment, which increases the amount of radioactive material that can be potentially released into the environment. An early ignition is defined as an ignition before the hydrogen concentration reaches a point where detonation is possible and therefore only deflagration is considered possible. If an early ignition does not occur, then there is a potential of detonation as well as deflagration. The effect of a detonation and deflagration can be the reactor building failure.

*IV.C.1. MCCI*

MCCI can occur in following two locations; (a) in the reactor cavity below the RPV and (b) in the SFP. As discussed previously, the sample calculations for this example did not result in MCCI in the SFP. Hence, only Figure 2 considers MCCI in the reactor cavity below the RPV. The MCCI node accounts for the likelihood that a severe reactor accident may eventually lead to the point where molten core debris fails the primary vessel lower head and falls into the reactor cavity. The interaction of molten core debris with structural concrete is known to produce vast amounts of aerosols and cause release of fission products. The probability of this event determined based on the decay heat and amount of fuel in the RPV for the specific PDS. It varies from relatively very certain at the beginning of phase P1 to not applicable in phase P-4. This parameter can be variated in the sensitivity analysis to assess the impact on the overall risk results.

*IV.C.2. Early Ignition*

Early ignition denotes a situation in which hydrogen combustion occurs near the flammability limit with a well-defined ignition source (i.e., versus random sources such as static electricity) that does not progress to a detonation. Well-defined ignition sources include hot debris, hot fuel in SFP, and hot aerosols leaving the pool. If early ignition occurs, then a deflation occurs rather than a detonation.

For early ignition PDSs without AC power available, a 0.5 mean probability of early ignition is used in the model. The probability accounts for the increased likelihood that hydrogen builds to high concentrations before any ignition. The uncertainty distribution is assumed to be a normal distribution with a mean of 0.5 and standard deviation of 0.05. Sensitivity analyses can be performed to assess the impact of this parameter on the overall risk.

*IC.C.3. Detonation*

The hydrogen detonation probability is a function of the maximum hydrogen concentration in the secondary containment (max[H2]) that is determined in MELCOR runs without combustion. The mean probability of hydrogen detonation (px) depends on the maximum concentration of hydrogen:

- $p_x = 0.01$ if max$[H_2] = 14\%$
- $p_x = 1$ if max$[H_2] \geq 28\%$, i.e., stoichiometric detonation is very likely
- $p_x$ = linear interpolation (in a semi logarithmic scale) if $14\% <$ max$[H_2] < 28\%$

Fig. 4 depicts the probability of detonation (px) as a function of max[H2]. The uncertainty distribution is assumed to be a normal distribution with mean px and standard deviation 0.02.

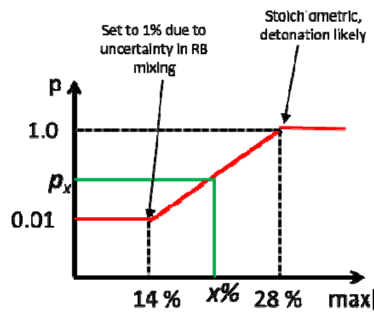


Fig. 4. Level 2 decommissioning detonation model.

*IV.C.4. Deflagration Given Early Ignition*

This APET node assesses the probability of RB overpressurization failure given deflagration from early ignition of the hydrogen-oxygen mixture in the secondary containment. In order to estimate the RB failure probability given deflagration, the probability distribution of the pressure generated by the deflagration (load) is compared with the probability distribution of the RB pressure resistance.

The median overpressure capacity of the weakest location $P_R$ $P_R$ is assumed to be 0.6 Bar with a lognormal standard deviation of $\sigma_R = 0.03$. This resistance lognormal distribution is compared with the load distribution to assess the probability of failure of the RB given deflagration, i.e., the overlap between the load and resistance curves. Fig. 5 depicts the load-resistance concept: the green area in Fig. 5 is the probability that the load is larger than the resistance, i.e., the probability of failure of the RB due to deflagration.

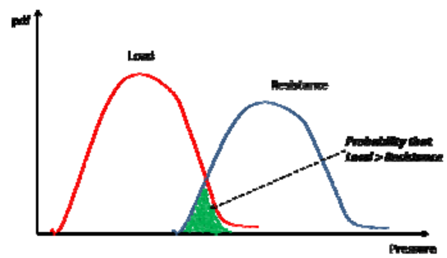


Fig. 5. Load-resistance curves to estimate the RB failure probability.

The load probability distribution (PDFL) has been assumed lognormal with mean $\ln(P_L)$ $\ln(P_L)$ and standard deviation $\sigma_L$:

$$\mathrm{PDF_L} = Lognorm[\ln(P_L), \sigma_L] \tag{1}$$

Where:

$$\mathrm{P_L} = \begin{cases} p_{10} & if\ \max[H_2] \geq 10\% \\ p_{\max[H_2]} & if\ \max[H_2] < 10\% \end{cases} \quad \text{and} \quad \sigma_\mathrm{L} = \sqrt{\frac{(p^* - p)^2}{2}} \tag{2}$$

and:

- $p_{10}$ = maximum pressure with 10% hydrogen deflagration
- $p_{\max[H2]}$ = $P_{\max[H_2]}$ = maximum pressure with maximum hydrogen deflagration
- $p^* = p_{14}$ if $\max[H_2] \geq 14\%$ or $p^* = p_{\max[H2\}}$ if $\max[H_2] < 14\%$
- $p_{14}$ = maximum pressure with 14% hydrogen deflagration

In order to estimate the overlap between the pressure load curve and resistance curve, a Monte Carlo approach is performed: two random values are sampled from a normalized uniform distribution; the first random value is used to estimate the resistance through the inverse lognormal distribution of the first sample and parameter distribution of the resistance (median 0.6 Bar and standard deviation 0.03); the second random value is used to estimate the load through the inverse lognormal distribution of the second sample and parameter distribution of the load (mean p and standard deviation σ). The calculated pressures are then compared and a “1” is collected if the load is larger than the resistance and a “0” vice versa. The amount of 1’s over the number of samples is the probability of failure of the RB due to the load. The standard deviation of the difference between the two distributions $\sigma_{delta}$ is:

$$\sigma_{delta} = \sqrt{\sigma_L^2 + \sigma_R^2} \tag{3}$$

*IV.C.5 Deflagration without Early Ignition*

The deflagration without early ignition node assesses the probability of RB overpressurization failure given no deflagration. Without the occurrence of early ignition there is a potential of detonation as well as deflagration. The same approach described in the previous section is followed; however, in this case, the parameters of the load distribution are slightly different and skewed towards higher hydrogen concentrations since no early ignition has occurred. In this node, the probability of RB given deflagration is larger than the previous node with early ignition. The same equation (1) is used for the deflagration without early ignition node, but:

$$\mathrm{P_L} = \begin{Bmatrix} p_{14} & \textit{if} \max[H_2] \geq 14\% \\ p_{\max[H_2]} & \textit{if} \max[H_2] < 14\% \end{Bmatrix} \quad \text{and} \quad \sigma_{\mathrm{L}} = \sqrt{\frac{(p^* - p)^2}{2}} \tag{4}$$

and:

- $p_{14}$ = maximum pressure with 14% hydrogen deflagration
- $p_{\max[H2]}$ = $P_{\max[H_2]}$ ■ maximum pressure with maximum hydrogen deflagration
- $p^* = p_{\max[H2\}}$ if $\max[H_2] \geq 14\%$ or $p^* = p_{14}$ if $\max[H_2] < 14\%$

## V. INTEGRATION OF PROBABILISTIC AND DETERMINISTIC RESULTS AND SOURCE TERMS ESTIMATION

The decommissioning APET model generates end states each with its own distribution. The source term definition builds on this structure. For each APET sequence, source term results from a single representative MELCOR calculation are specified. For all the probabilistic evaluations, the source term for each sequence does not vary. However, the variations in the uncertainties are propagated through the top events. In this manner, the conditional frequency of each end state, or APET sequence, varies based on particular values of top event split fractions sampled in the Monte Carlo simulation. This methodology allows identifiable propagation of uncertainties through the top events while retaining direct reference to the underlying technical basis for the source term.

The activity of Cs-137 released to the environment is used as the primary basis for defining a large (early) release in this paper. Cs-137 has been shown to be an important contributor to long-term health effects from radiological releases associated with postulated severe accidents. Therefore, the release fraction for Cs-137 provides a first-order characterization of accident severity. Cs-137 has a long half-life. Finally, cesium is among the most volatile fission product species in the fuel (with the single exception of noble gases), and therefore is released earlier and in larger quantity than other radionuclides.

As an example, the source term specification for PDS-03 in phase P-4 is illustrated in Table 3. Table 3 represents the source term evaluation of the APET shown in Fig. 3. For each sequence, the total and Cesium-137 activity released to the environment, the timing of the release and the large early release flags are obtained from MELCOR calculations. In this example, all sequences PDS-03-02 and PDS-02-05 have LER flags, signifying a Cs-137 release > Cs-137 threshold within a few hours after the start of core damage, as evaluated in the deterministic MELCOR source term calculations.

TABLE 3. Example of decommissioning APET source term for PDS-03 in Phase P-4.

| Sequence | End State | Cs Activity (Bq) | Total Activity (Bq) | LR | LER | Note |
|---|---|---|---|---|---|---|
| PDS-03-01 | RB OK | 5.0E+15 | 9.0E+16 | X | -- | Release via RB leakage. |
| PDS-03-02 | RB Failure | 2.0E+17 | 7.0E+17 | X | X | |
| PDS-03-03 | RB OK | 8.0E+15 | 8.0E+16 | X | -- | Release via RB leakage. |
| PDS-03-04 | RB Failure | 9.0E+16 | 7.0E+17 | X | -- | |
| PDS-03-05 | RB Failure | 5.0E+16 | 5.0E+17 | X | X | |

For each PDS a process like the one presented in Table 5 is performed in order to estimate the final risk profile in terms of LRF, LERF and RC. All the information from each PDS is terms of contribution to LRF and LERF are used to estimate the total risk of the plant up to a radioactive release to the environment.

## VI. RESULTS

This section provides an overview of the integrated Level 1 – Level 2 decommissioning PRA. The American standard [7] defines the large early release as large early release the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects. The large early release frequency (LERF) is the expected number of large early releases per unit of time.

Deterministic thermal-hydraulic MELCOR calculations are used to determine the amount of release as well as timing of release. APET sequences are therefore mapped to LERF based on the amount and timing of release as presented in Table 5 for PDS-03 in phase P-4 as an example.

Table 4 presents an overview of the Level 2 PRA results. The Level 2 decommission PRA LERF is 3E-07.

TABLE 4: Level 2 decommissioning PRA results.

| | LRF | LERF | LERF 5th | LERF 50th | LERF 95th |
|---|---|---|---|---|---|
| **P-1** | 3E-06 | 3E-07 | - | - | - |
| **P-2** | 2E-06 | 2E-07 | - | - | - |
| **P-3** | 2E-06 | 3E-07 | - | - | - |
| **P-4** | 1E-06 | 5E-07 | - | - | - |
| | | | | | |
| **Total** | 8E-06 | 3E-06 | 1E-06 | 3E-06 | 5E-06 |

## VII. CONCLUSIONS

This paper presents a methodology for the Level 2 PRA during the decommissioning of the plant. The methodology uses results from the Level 1 decommissioning PRA and bins the output sequences into PDSs characterized by specific attributes. Each PDS will be then fed to a specific APET that considers success or failure of events for reactor building response the associated radionuclide release to the environment. The APETs used in the Level 2 PRA are supported by MELCOR calculations that consider potential severe accidents in the RPV, in the SFP, or in both simultaneously. An integrated approach that considers accidents in both RPV and SFP has been presented in this paper that combined with the probabilistic analysis produces the Level 2 PRA results in terms of LERF and LRF.

This methodology has been applied to several American and European BWRs. In this paper an example of application to generic General Electric BWR Mark I is provided. Numerical results are provided for demonstration purposes and do not reflect any actual LERF or LRF of a specific plant.

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