Methodology of Probabilistic Safety Assessment Level 2

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PSA Lvl 2

PSA L2 is used to determine the response of the containment during core damage to estimate the frequency and type of releases. At this level of PSA, we examine the behaviour of the containment for heating, hydrogen production, combustion and explosion, as well as the interaction of corium with concrete.

Physical phenomena's during major accidents

The physical phenomena's associated with the release of fission products during major accidents at NPPs are folliwing:

- Phenomena's related to the development of failures inside the reactor vessel (degradation and melting of the reactor core, flow of molten korium to the bottom of the reactor vessel and possible mechanisms of vessel damage and system),
- phenomena leading to rapid damage to the containment and early releases of radioactive fission products (direct heating of the containment, production and combustion of hydrogen and explosion of steam),
- physical phenomena that may lead to delayed damage to the containment, resulting from the interaction of the corium with concrete and water,
- phenomena related to the release of radioactive fission products from fuel, their transport within the reactor cooling system, release outside the reactor vessel and washout from the containment atmosphere.

Degradation of the core in the reactor vessel

During major accidents in nuclear power plants, when the reactor core remains uncovered for a sufficiently long time, local temperature increases in the fuel rods, which can lead to significant and irreversible degradation.

At a temperature of about 1300K, zirconium which is a part of the fuel cladding, is oxidised by water vapor

$$Zr + 2H_2O \xrightarrow{552{,}66\ kJ/mol} ZrO_2 + 2H_2.$$

At such a high temperature, the reactor's control rods are subject to an even faster degradation. The materials from which they are built melt and relocate to the lower parts of the core, which leads to a significant reduction in the possibility of receiving heat from those fuel rods that have not yet been damaged and a further increase in temperature. The increase in temperature and the formation of gases within the fuel pellets leads in turn to an increase in pressure inside the fuel rods (sometimes higher than the pressure in RCS).

Core degradation in the reactor vessel

Overpressure leads to swelling of the fuel, increase of its volume, additional tensions, and consequently to unsealing or even bursting of the fuel cladding. This phenomenon occurs at temperatures above 2300-2500 K.

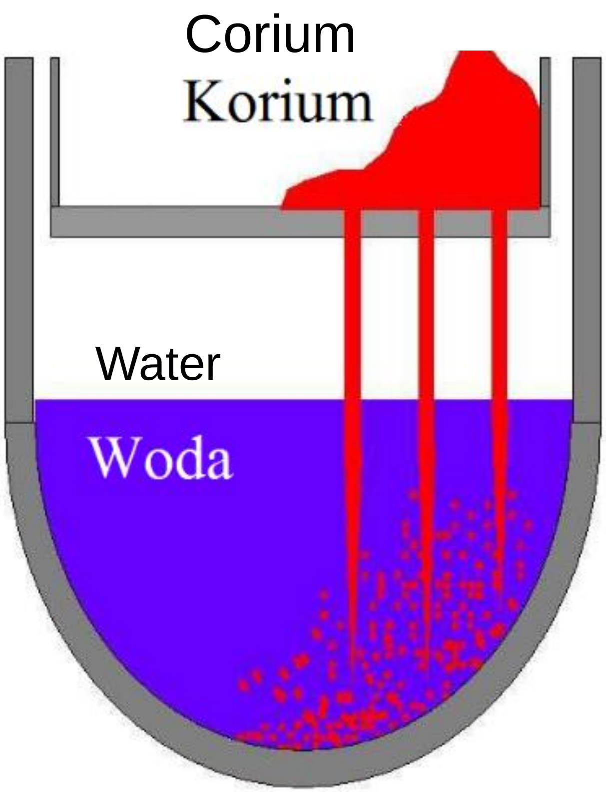
When this temperature is reached, the fuel is partially mixed with the molten metal, which leads to gradual changes in the geometry of the core, the local porosity and the flow area of the coolant. In the temperature range 2100-2900 K, the viscosity of the U-Zr-O mixture is a growing function of oxidation. Knowledge of the core material degree of oxidation is crucial to determine the further possible progression and relocation of the corium within the reactor vessel

Test results to check the reaction of fuel assembly materials to high temperatures.

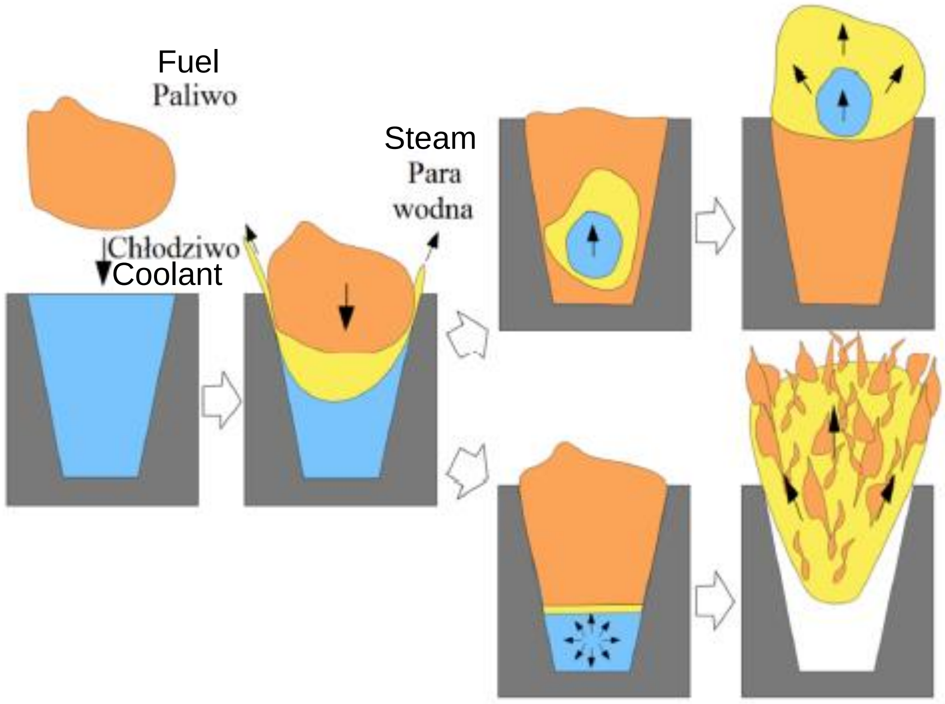
Flow of the corium to lower parts of the vessel

It is usually assumed that when the water flows into the lower part of the reactor vessel, it is filled with cooling water even in the event of LOCA. Interaction of the corium (temp> 2,500 K) with water leads to fragmentation into more or less regular fragments.

By fragmentation of the molten core, the surface of the interaction increases significantly, which affects the intensity of steam production. Producing a large amount of steam creates a risk of explosion and loss of RCS system integrity due to the rapid increase in pressure. In addition, the contact of the hot corium with the walls of the reactor vessel poses a risk of melting the entire body and releasing radioactive materials to inside the containment.



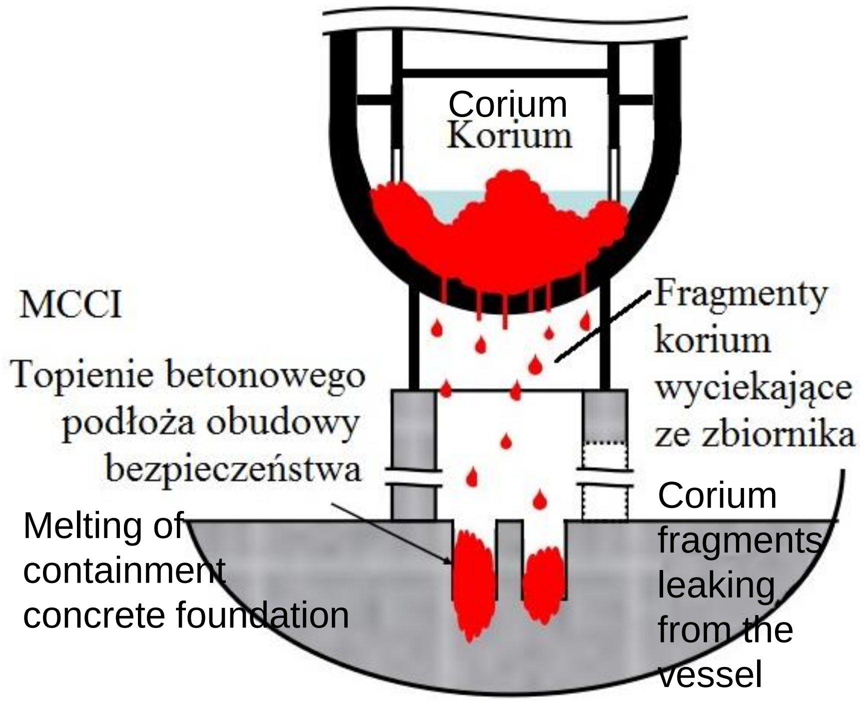
Vessel failure



The contact between the corium and the cooling water can lead to a very rapid production of water vapor, a rapid pressure increase in the RCS, and even an explosion leading to the destruction of the reactor vessel.

Melting of containment concrete foundation

In the event of damage of the reactor vessel, the corium consisting of the melted core material and the structure of the fuel elements will escape to the containment concrete base. The interaction between the corium and concrete leads to gradual degradation of the concrete, which may result in the production of new gases and aerosols, the loss of the integrity of the containment and the release of radioactive substances. Breach of the reactor's lateral walls leads additionally to the contact of the corium with cooling water and pressure increase in the containment.



Methodology of PSA analysis from the results of Lvl 1 to LRF/LERF

Probabilistic Safety Assessmnet Level 2 – L2PSA is integrated process that includes the following tasks:

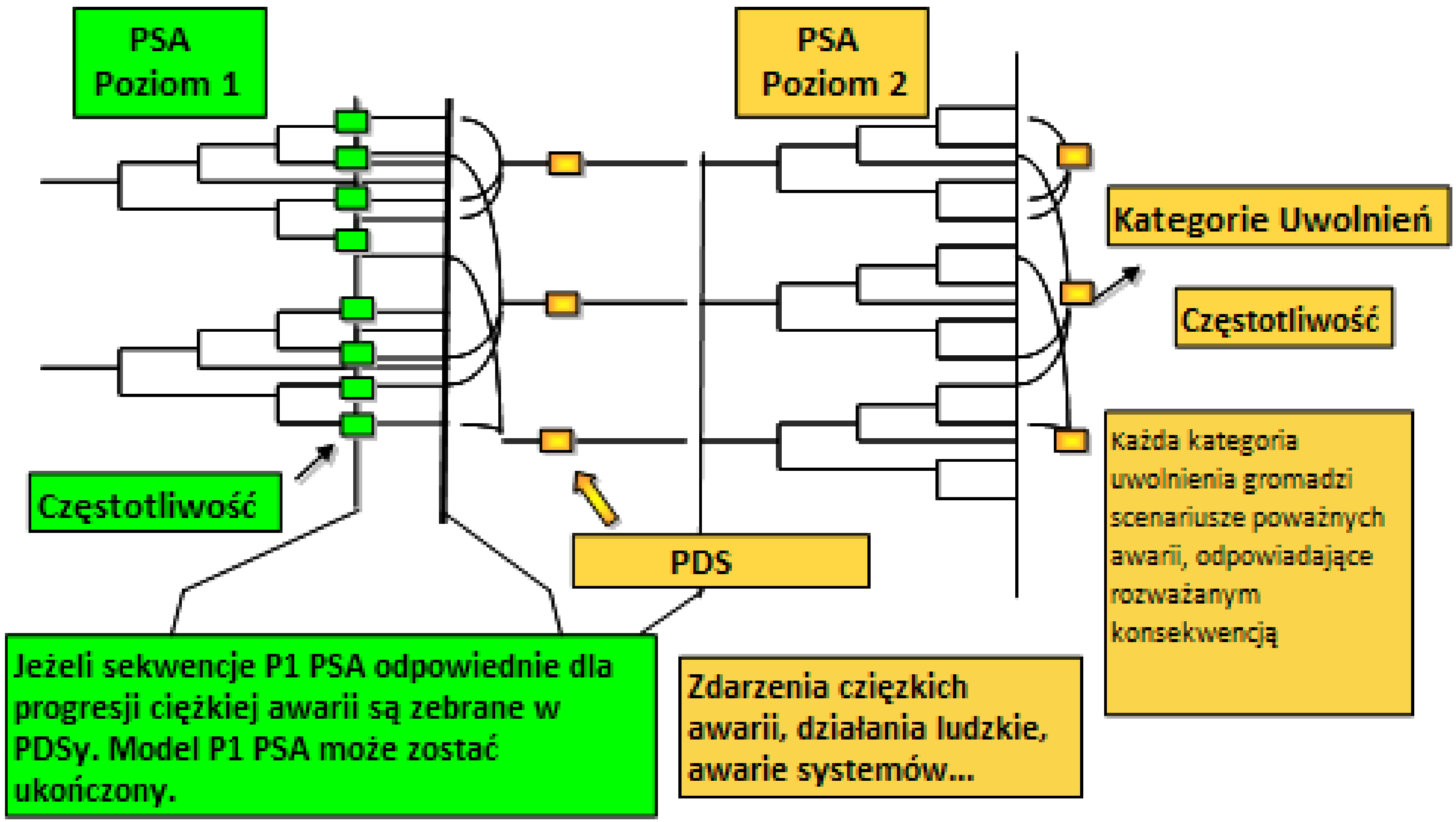
- Definition of Plant Damage States (PDS) – forming an interface with level 1 PSA,
- Definition of Release Categories (RC) – for considered major accident scenarious,
- development of major failure scenarios with the occurrence of core damage, where plant failure conditions can be seen as initiating events and end states are a set of specific release categories,
- frequency quantification for different accident scenarios and release categories.

The objectives and scope of L2PSA analyzes have huge importance for determining the level of details needed in defining of PDS and RC as well as in modeling the course of accident. The Level 2 analysis is not only used for the containment, but also for example, the operator's activities, etc. that are part of the development of the failure.

Probabilistic analysis is dependant on:

- Accessibility of information regarding individual physical phenomena occurring during major accidents;
- Structural analysis of the containment;
- Analysis of possible operator actions and errors;
- Analysis of safety systems
- Evaluation of the source terms

The process of implementing level 2 PSA analyzes



Source term

Analysis of the source elements necessary for L2PSA provides information on the characteristics of release categories in terms of composition and time. The process of a given analysis includes the following stages:

- 1. Selection of representative major accidents sequences within each category;
- 2. Identification of the source term assessment needs due to L2PSA objectives and criteria;;
- 3. Calculation of the source terms for representative sequences of major failures ;

Computation codes such as MELCOR, MAAP, ASTEC are used to model the release and spread of different groups of fission products. The use of such codes is considered as a minimum requirement for the assessment of release to the environment in a modern PSA. However, there is a spectrum of approaches even within integrated codes, some of them use simple "focused parameter" models and others use a more comprehensive approach to modeling. Even in a single integral code, both solutions can be used in different sub-models.

Methodology for L2PSA model creation

Event trees in L2PSA are used to determine the sequence of events and phenomenas of major accidents after the occurrence of core damage that question further barriers to determine the release of radioactive material. They provide a structured approach to the systematic assessment of the ability of a nuclear power plant to cope with core failure.

The Accident Progression Event Tree (APET) enables description of severe accidents by L2PSA sequences, from the Plant Damage States (PDS), which consists of grouping the L1PSA sequences up to the final installation state after a failure, including possible damage to the containment. The L2PSA sequences are grouped into release categories, which are the most characteristic L2PSA result.

PDS

Modeling level 2 part requires that Plant Damage States (PDS) are defined in terms of features that could affect the way in which the failure associated with the loss of containment integrity and the release of radioactive materials into the environment develops. These features provide enough information to apply as initiating events to level 2 event trees and support both the development of failure scenarios and ultimately the use of an appropriate set of release categories.

In addition to the features of the containment, the PDS characterization must take into account the instalation failures in the L1PSA i that could affect the housing or the release of radioactive material:

- the type of initiating event may, for example, affect to the rate of discharge of fluid into the containment, the course of the core melting, the production of hydrogen and for time when the release of radioactive materials occurs,
- failure mode of the core cooling function affecting the core melting time,
- degree of fuel damage,
- pressure in the system at the moment of core failure and condition of safety valves and other elements that could change RPV pressure before failure of the bottom RPV head,
- Determining of the reaction of the containment,

Release categories

The final goal of the majority of L2PSA is to provide information on the release of radionuclides into the environment. In principle, each individual APET sequence leads to a specific type of release. However, even if it was possible to perform a detailed analysis, it is better to group sequences that have similar releases. Sequence groups that have similar sequences are called release categories. There are many factors that influence the way the release categories are defined, mainly related to the purpose and scope of the PSA.

The release category defines the grouping of APET endpoints where the source term is expected to be released into the environment, due to all of the following conditions:

- Identical or similar, initial events and installation failures occurred ,
- Identical or similar, initial events and installation failures occurred,
- There are identical or similar designed features to alleviate the release of radioactive substances into the environment
- An identical or similar hcontainment response is anticipated.

Example of release category groups

RC-0	Wyciek projektowy
RC-1	Filtrowane odpowietrzniki
RC-2	Powolne zwiększanie ciśnienia, awaria obudowy (CF)
RC-3	Przenikanie przez płytę fundamentową
RC-4	Poźne zjawiska poza zbiornikiem reaktora, CF
RC-5	CF podczas awarii RPV, niskie ciśnienie RPV.
RC-6	CF podczas awarii RPV, wysokie ciśnienie RPV
Itp.	Przeciek obudowy, omijanie, V-LOCA, powrót do normalnego stanu, zraszanie obudowy

Evaluation of results

Evaluation of L2PSA analysis results according to IAEA-TECDOC-1229 should include:

- Familiarization with data and systems analyzed under PSA
- Verification of connections between L1 and L2 of PSA analyzes
- Familiarization with the failure development models used in PSA
- Verification of strength/endurance analysis of reactor containment
- An overview of the structure of probabilistic safety analysis
- Verification of quantification of events included in PSA analyzes
- Verification of the source terms analysis (possible releases)
- Verification of the uncertainty analysis carried out
- Verification of the results of the integrated risk analysis,
- Possible use of Level 2 PSA results.

Thank you for attention

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