ACCIDENT CONSEQUENCES AND ANALYSIS

IAEA Workshop on Severe Accident Management Guidelines

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Outline

- Results of a severe reactor accident
  - On-site consequences
  - Off-site consequence analysis
  - General considerations for source term calculations
  - Severe accident consequences codes
  - Emergency preparedness
- Accident analysis
  - Safety assessment and analysis
  - Types of accident analysis
  - Accident analysis methods
  - Computer codes for accident analysis
  - Quality of accident analysis

Appendix : Crosswalk code comparison exercise

Results of a severe reactor accident

Severe accidents can cause extensive damage to the reactor, up to and including total destruction of the reactor itself and of the surrounding civil structures;

- **If the containment is damaged, then large releases of radioactive materials may occur, causing contamination both on-site and off-site;**
- **This implies risk for public health and safety, and possible environmental damage;**
- **Releases may also cause societal disruption and have significant economic consequences.**

On-site consequences – intact containment

If the containment is still intact;

- **The radiation consequences on-site will be limited;**
- **It is relevant to monitor the radiation levels in working areas, such as the (control room, the room(s) for the Emergency Response Organisation, local areas where manual actions must be performed, locations where equipment must be restored or temporary equipment hooked on;**
- **Special precautions should be taken in case the containment needs to be vented;**
  - A containment filter will trap many aerosols, but all the noble gases escape, and capture of gaseous iodine may be limited (current filters are inefficient for organic iodides, research to improve capability in progress).

On-site consequences – containment failure

If the containment has failed;

- **Access to the site may be limited, thereby reducing the possibility for intervention and support;**
- **Evacuation of parts or all of the site and surrounding areas may be needed;**
- **Accident management is then only possible from protected rooms. Protective equipment (e.g., breathing apparatus) may be needed for operating personnel and workers;**
  - Changing shifts would be hampered by the radioactive contamination in the environment;
- **Staff may be exposed to radioactivity and may care about the radiation risk to their loved ones;**
  - This will result in elevated stress for staff at work and possibly reduced effectiveness.

Off-site consequences

A loss of containment will result in a release of airborne radioactivity in the form of noble gases and aerosol particles (eg. Xe, I and Cs), which will disperse from the site through the environment to the surrounding population, by expanding and downwind movement.

Measures in the environment are taken to protect people:

- **By sheltering (staying indoors);**
- **By distribution of iodine pills (to protect the thyroid from the absorption of radioactive iodine);**
- **By evacuation, dependent on the severity of the actual and/or the anticipated releases.**

Off-site consequence analysis (1/2)

The off-site consequences are often characterized by two phases:

- The emergency phase during and shortly after the accident;
- The long-term phase evaluating mainly radiological consequences;

Worldwide monitoring after a severe accident is now established;

The assessment of this source term can also be used…

- To assess the robustness of the containment features to retain fission products and gases;
- To develop or improve SAM systems for the mitigation of releases;
- To provide adequate protection for direct radiation on operating staff in the different reactor areas and control rooms (on-site consequences).

Off-site consequence analysis (2/2)

For the analysis of the consequences of a severe accident, knowledge is needed about the ‘source term’;

- This is the amount and isotopic composition of material released (or postulated to be released) from the reactor or spent fuel pool;
- The characterisation of the source term and its calculation as originating from a damaged containment is the input to off-site consequence analysis and environmental impact and protection to the public;

These calculations of atmospheric releases can then be used for:

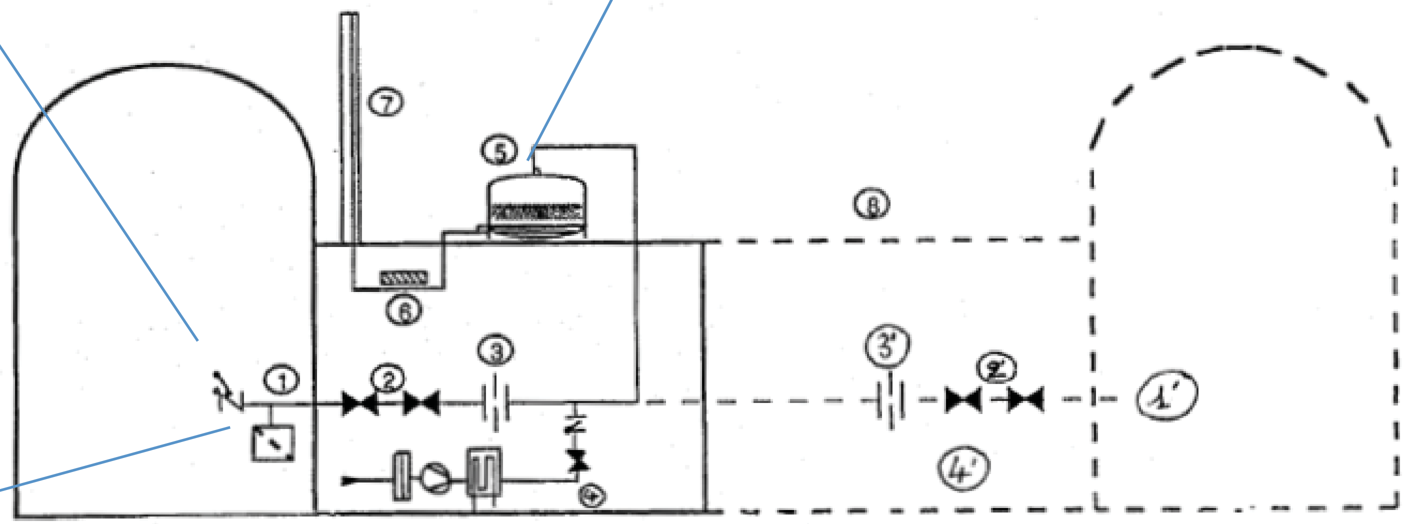
- Transport and dispersion modelling;
- Emergency response modelling;
- Estimation of health impact on the public.

Factors influencing the source term

The magnitude of the "source term" depends on the following factors:

- **The core inventory in fission products (the SFP inventory is different) and time since shutdown (decay time);**
- **The extent of fuel damage;**
- **The fraction of fission products released from the fuel;**
- **The retention of fission products in the RCS, retention and deposition in the containment and on the containment walls, chemical interactions, resuspension/revaporisation of fission products;**
- **The effects of containment spray;**
- **Filtered venting at containment, or containment leakage or break;**

Quantitative and qualitative understanding of the source term has been achieved through international and national research programmes.



LEGEND:

1. Pre-filter;
Existing penetration, 300 mm diameter for 1300MWe plants, 250 mm diameter for 900 MWe plants.

2. Manual valves, operatd by reach rods from behind shelding.

3. Pressure letdown orifice.

4. Filtered dry air supply during normal operation / Preheating for "H2 risk" in case of SA

5. Sand filter.

6. Radiation monitor

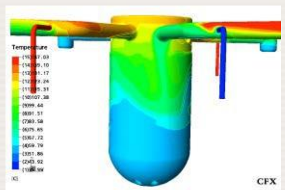
7. Plant stack, with small vent stack

8. Arrangement for twin units (900 MWe)

Schematic diagram of the containment venting system

Role of experiments in understanding severe accidents

SAFETY ANALYSIS



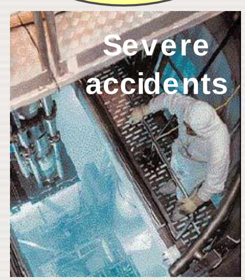
Perform studies

Complex problems
Confrontation with designer
Check pertinence

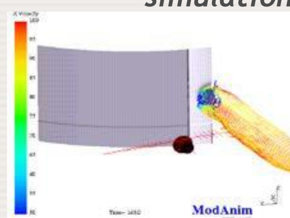
Thermalhydraulic simulations

RESEARCH EXPERIMENTS MODELLING

Provide knowledge
Prepare the future



Plane Crash studies



Building, commissioning

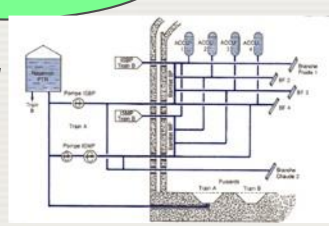
Shutdown

Design

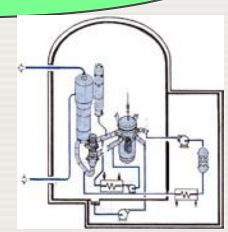
Perform analyses

Operation, modifications

Design and reliability of systems



Follow-up of equipments, operation feed-back, event analysis, periodic safety reviews



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Computer codes for source term calculations (1/4)

Two main types of computer codes are used for the analysis of severe accidents:

- **Integral codes. These are capable of simulating the whole event, from the start of core damage until the release of fission products. The codes use simplified models for the various physical phenomena, in order to being able to capture the whole event. Where lack of detail exists, sometimes user-specified values need to be provided;**
- **Mechanistic codes, using a mechanistic approach, i.e. trying to approach the physical phenomena from their physical basis. They usually focus on a single phenomenon, within known boundary conditions. An example is the distribution of hydrogen, upon the hydrogen source been provided by another code (e.g., an integral code).**

Practically speaking, severe accident progression is modelled by integral parameter code systems. Today, these integral codes are state-of-the-art tools for source term calculations and serve as reservoir of knowledge of severe accident phenomenology.

Computer codes for source term calculations (2/4)

Examples of codes calculating severe accident sequences (based on the older thermo-hydraulics codes RELAP5 (US) and ATHLET-CD & ICARE/CATHARE (EU):

- **Integral codes MELCOR (US), ASTEC (EU), Athlet-CD with extensive and detailed analysis of source term release and chemistry, also SCDAPSIM (US) developed from SCDAP/RELAP5;**
- **Also Japan develops IMPACT/SAMPSON code, while Russia develops its SOCRAT code system;**

Industry uses fast operating commercial integral codes in MAAP (US and EU improved in version 5);

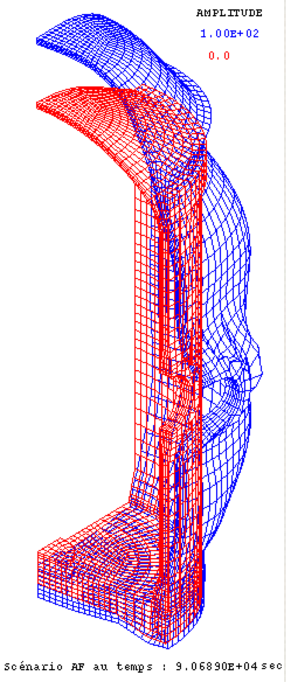
These integral codes are also used in sensitivity studies, uncertainty characterization and risk evaluation for probabilistic safety analysis (PSA) Level 2 studies, estimating the risks of PSA scenarios;

The codes also being used to investigate accident management strategies.

Computer codes for source term calculations (3/4)

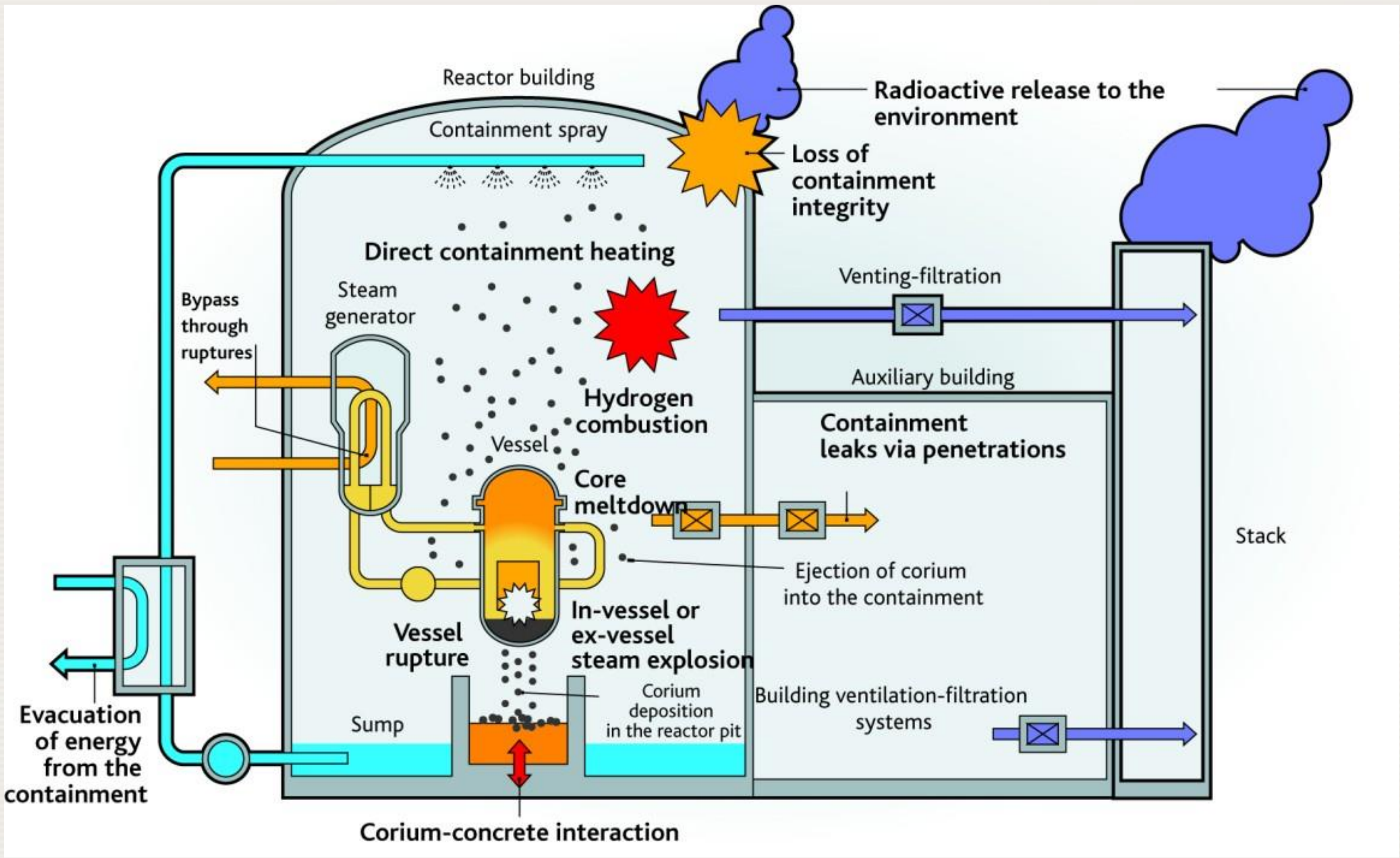
Special purpose codes can be used to investigate phenomena in more detail:

- **Structural mechanics codes using finite element methods (FEM) to evaluate structural response, e.g. rupture of the lower head, containment failure;**
- **Computational fluid dynamics (CFD) codes to calculate fluid behaviour in detail, e.g. natural circulation in the vessel;**
- **Example of containment failure calculation using FEM:**
  - Elastic and inelastic deformation of steel/concrete;
  - Pressure load time evolution;
  - Stress peaks, e.g. at fixed points of containment;
  - Constraints to deformation, e.g. concrete structures, components within reactor building annulus ;
    - Fragility curves;
    - Probability of containment failure dependent on pressure load.

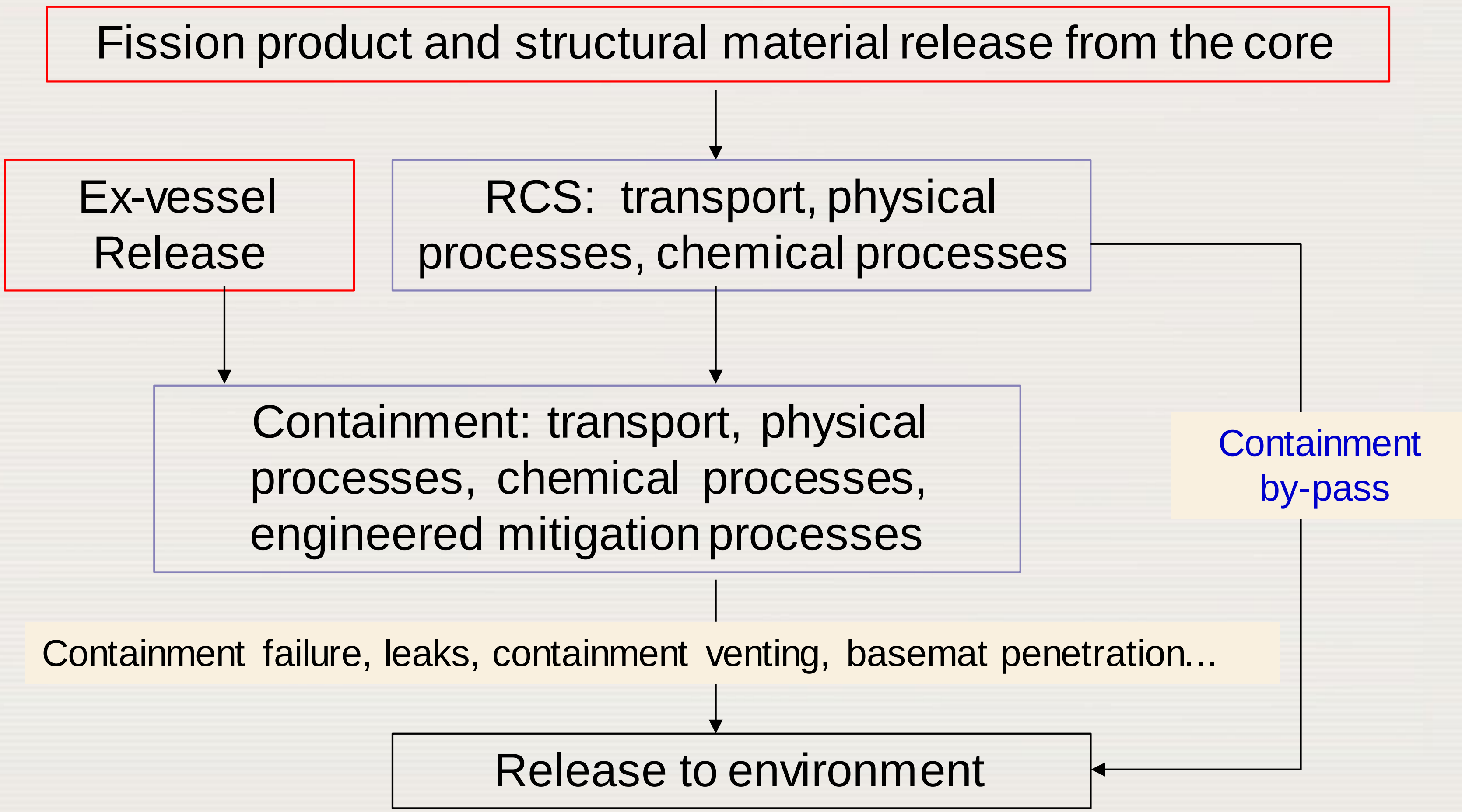


Computer codes for source term calculations (4/4)

Phenomena considered in integral codes



The Source Term Pathway



Mechanisms considered in source term calculations

Main factors influencing the source term are, amongst others:

- Inventory of fission products;
- Release of fission products, actinides and structural materials from the core, according to their volatility;
- Transport and deposition of these materials in the circuit in the case of a PWR, involving aerosol formation and behaviour (thermophoresis, diffusiophoresis, electrophoresis, sedimentation, reaction kinetics, revaporisation, and chemical combination) which can result in retention of fission products in the RCS;
- Phenomena in the containment, e.g. aerosol physics as in the circuit, chemical reactions, in particular important for iodine and ruthenium due to their volatile species (Ru volatile under oxidising conditions such as after air ingress);
- Leakages of buildings;
- Containment filtered venting (an accident management measure).

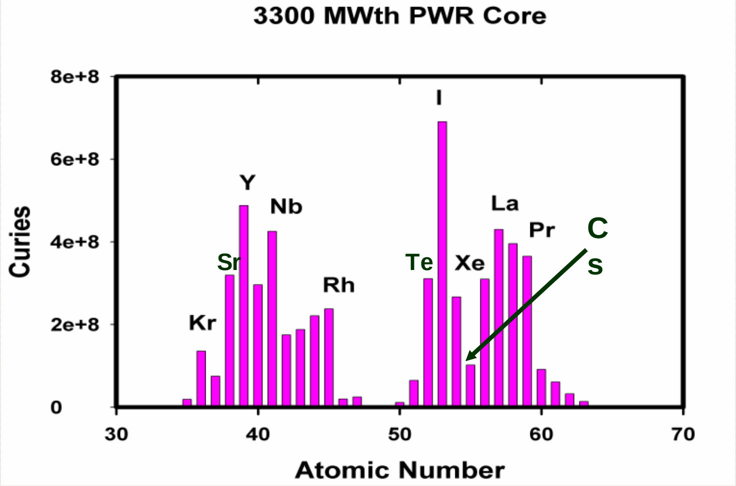
Inventory and role of fission products

Initial inventory of fission products (FP):

- **2000 kg in a French PWR 900 MWe (Xe 300 kg, Kr 22 kg, Cs 160 kg, I 13 kg, Mo 180 kg, Ru 140 kg, Zr 200 kg, Ba 80 kg etc.):**
- **Corresponds mainly to the mass of stable isotopes: example of total iodine mass of 13 kg, incl. 0,8 kg of radioactive iodine.**

Wide range of half-lives

- **^{133}Xe: 5 days, ^{85}Kr: 10 years, ^{137}Cs: 30 years, ^{131}I: 8 days, ^{129}I: 1.7x10^{7} years.**



Volatility of Fission Products

3300 MW_{th} LWR

Volatility	Elements	Inventory (Ci)
Noble Gases	Krypton (Kr) Xenon (Xe)	$1.7x10^8$ $2.2x10^8$
Very Volatile	Iodine (I) Cesium (Cs)	$7.5x10^8$ $2.3x10^7$
Moderately Volatile	Tellurium (Te) Strontium (Sr) Barium (Ba)	$1.8x10^8$ $3.5x10^8$ $3.4x10^8$
Less Volatile	Ruthenium (Ru) Lanthanum (La) Cerium (Ce)	$2.4x10^8$ $4.7x10^8$ $3.9x10^8$

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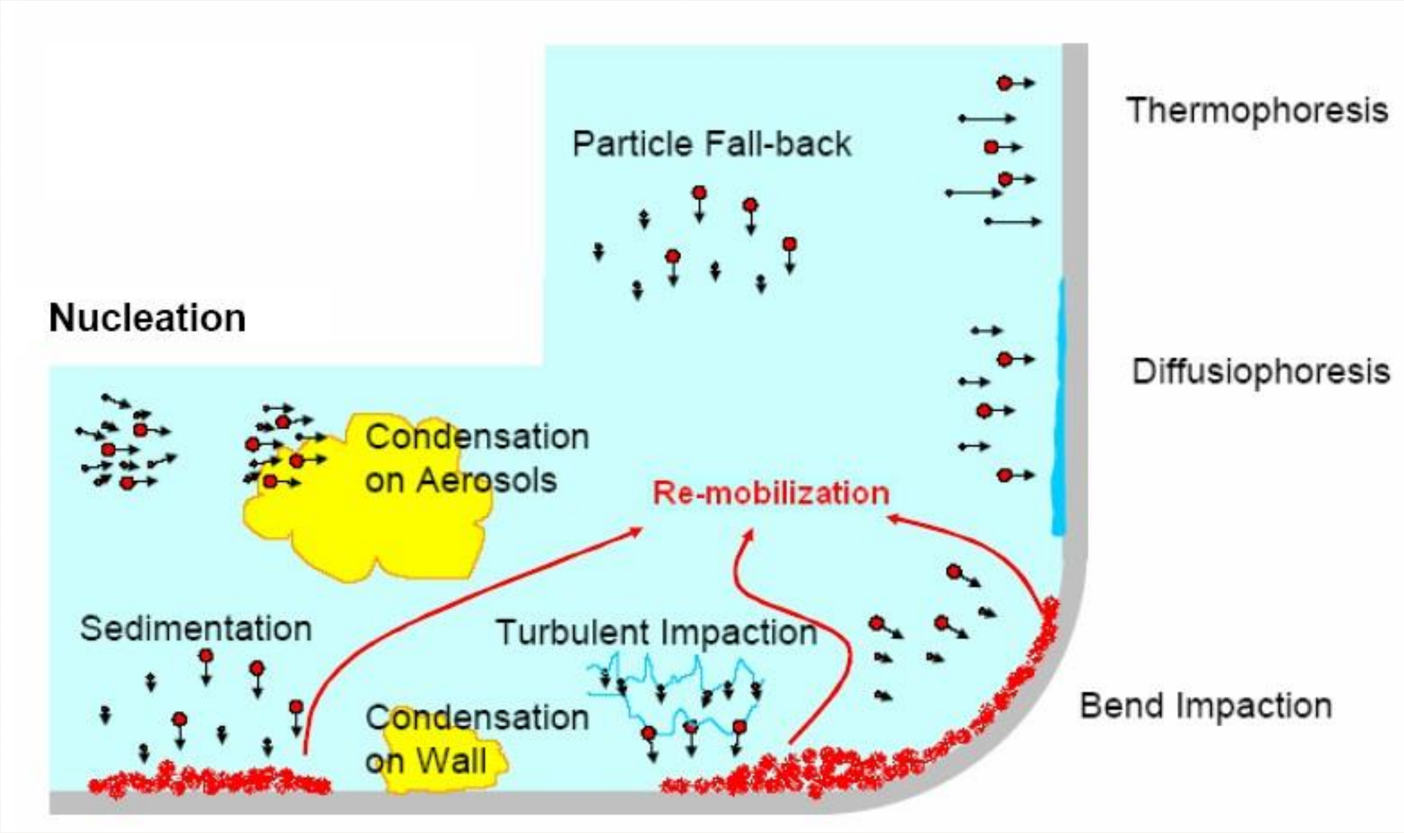
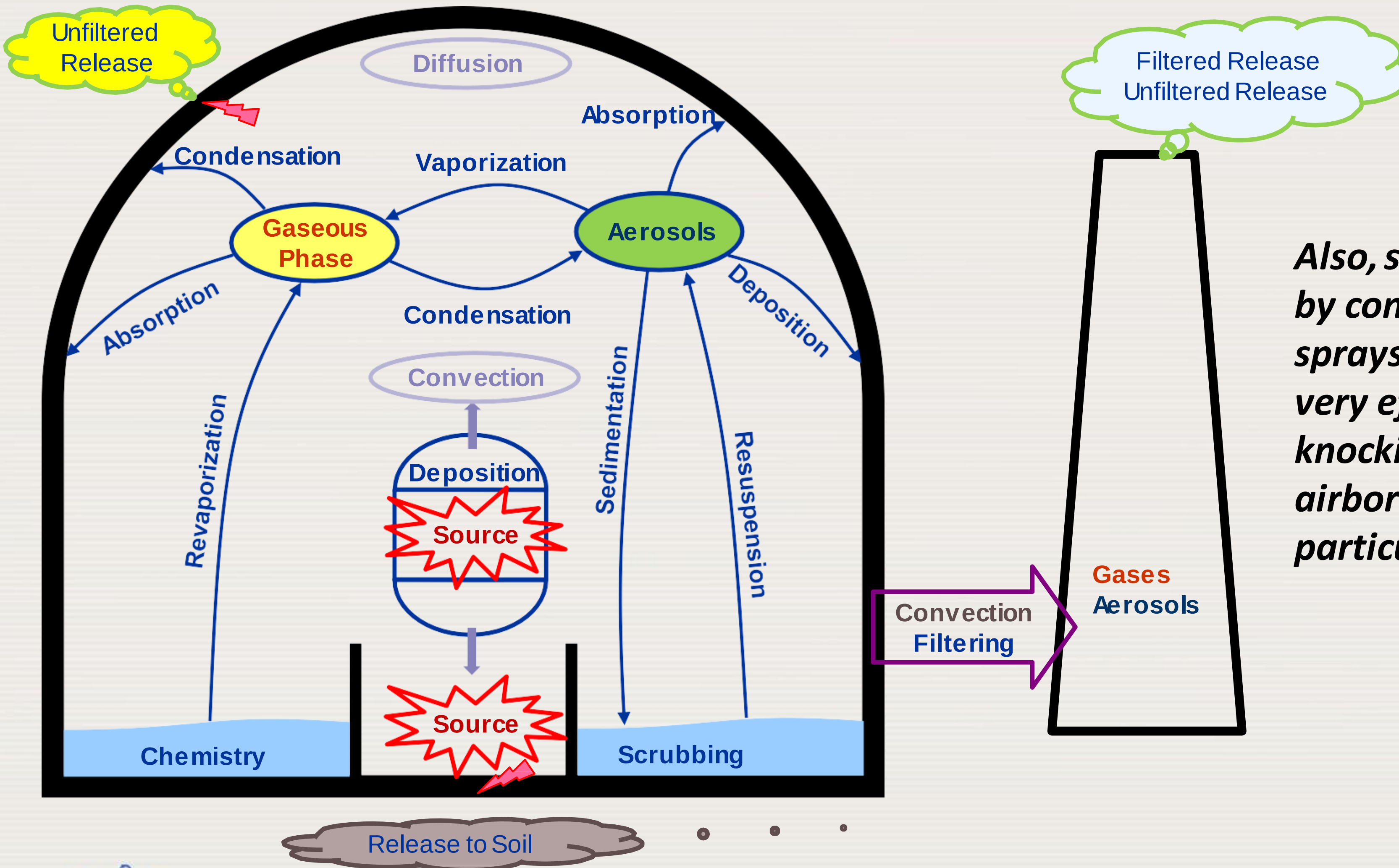
Source Term Phenomena in the Circuit (PWR)

...recall, airborne radioactivity is in the form of noble gases, volatile vapors and aerosol particles...



these deposition and retention processes are significant and must be considered in estimation of source term to the containment

Source Term Phenomena in the Containment



Also, scrubbing by containment sprays can be very effective in knocking down airborne particulate

Iodine Chemistry Phenomena in the Containment

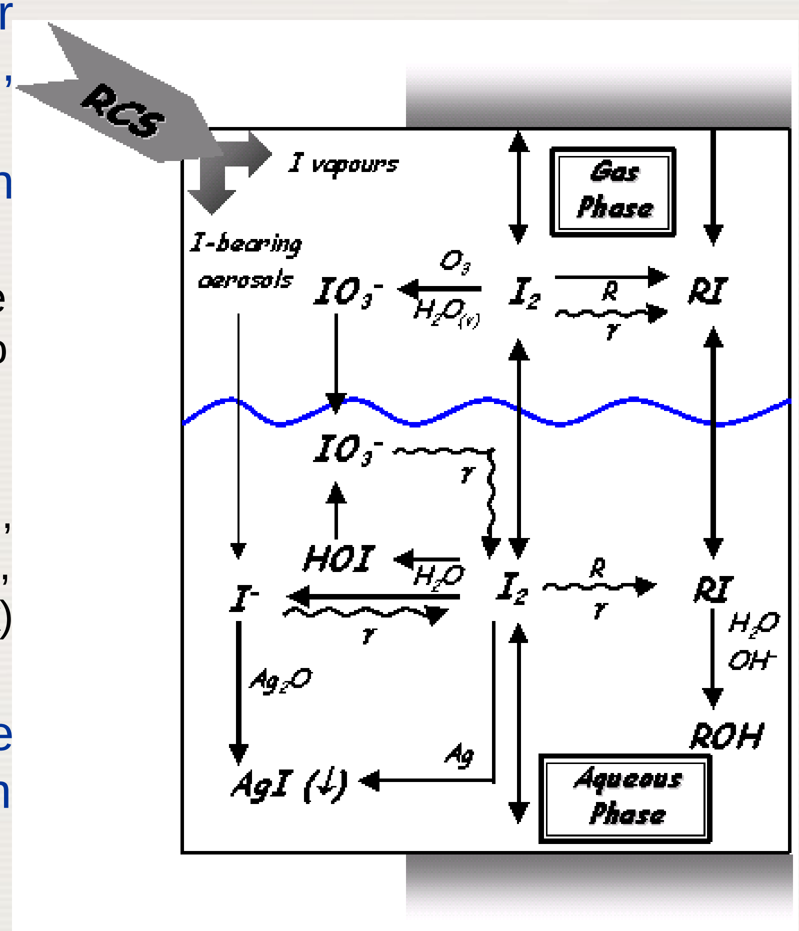
Simplified sketch of iodine chemistry in containment

Iodine chemistry forms volatile species either molecular iodine (I_2) or organic iodide (ICH_3,..), the latter being more difficult to trap.

Competition between formation/destruction processes:

- ***Formation*** : ICH_3 production from iodine interactions with paints, iodide oxidation into molecular iodine, desorption of iodine adsorbed onto walls …
- ***Destruction*** : Adsorption of I_2 on painted walls, transfer towards sump and silver iodide formation, oxidation of I_2 and ICH_3 into iodine oxides (IOx) particles by air radiolysis products …

The noble gases and the volatile iodine can be released outside the containment through direct or filtered vents (e.g. sand filters).



Source Term Mechanisms – circuit and containment

Explanation of terms

- *Thermophoresis*: motion of suspended particles following a temperature gradient near a surface;
- *Diffusiophoresis*: spontaneous motion of dispersed particles in a fluid induced by a diffusion gradient (also called 'concentration gradient') of molecular substances that are dissolved in the fluid;
- *Electrophoresis*: motion of dispersed particles (having an electric charge) in an electric field;
- *Reaction kinetics*: speed of chemical reactions between substances, which depend on their reactive surfaces, temperatures, etc.;
- *Revaporisation/resuspension*: volatile fission products that have been deposited becoming volatile again
  - CsI deposits in steam generators can be significant and can later re-vaporize to produce a late stage release

Severe accident consequences codes

Examples of computer programmes for the assessment of severe accident consequences:

- **RASCAL makes dose projections after and accidental release of nuclides (available from USNRC RAMP program). Used in real time accident response for emergency response decisions**
- **MACCS (MELCOR Accident Consequence Code System) has been developed in the US (USNRC/SNL) to evaluate the impacts of severe accidents at nuclear power plants and surrounding public. The most popular MACCS2 is the latest package enhanced for more flexibility, extended library of nuclides and a semi-dynamic food-chain model. This code determines health consequences of a severe accident both in terms of Early Fatality Risk as on Latent Cancer Fatality Risk.**

Emergency Preparedness

To reduce the consequences of a radiological event, it is required to demonstrate reasonable assurance that adequate protective measures are taken against this radiological emergency:

- **e.g. evacuation, sheltering, respiratory protection, relocation, KI blockage, decontamination of people, decontamination of land and buildings, food chain protection, medical treatments;**

The International Commission on Radiological Protection gives recommendations on radiological protection (1-2 mSv/yr livable, 20 mSv/yr limit for attack, etc.);

An important feature is periodic exercises of emergency response capabilities, providing & maintaining adequate facilities and equipment, established procedures to notify the local response organisations and emergency personnel. Notice on the amount and description of the radiological signature has to be given nationally and internationally;

The European Commission RODOS system for nuclear emergency planning can also be mentioned here as a real-time online Decision Support system for nuclear emergency management.

Nuclear Safety Assessment (1/2)

Safety assessment of an NPP should demonstrate that there is no undue risk caused by plant operation. Safety assessment is a systematic process that is carried out throughout the lifetime of the facility or activity to ensure that all the relevant safety requirements are met by the proposed (or actual) design, including:

- **Showing that the plant has experience and safety research sufficient defence in depth, accounting for the operating experience and safety research;**
- **Plant equipment requirements (equipment qualification and consideration of the ageing and reliability of systems through redundancy and diversity);**
- **Plant systems design requirements (e.g. specific requirements on the reactor core, reactor coolant system, containment and engineered safety features).**

Nuclear Safety Assessment (2/2)

Safety assessment includes, but is not limited to, the formal safety analysis;

More generally, safety assessment can cover all aspects regarding siting, design, construction, operation and decommissioning of an NPP that are relevant to safety.

Types of Safety Assessment Include

- Design Basis Analyses
- Beyond Design Basis Analyses (Severe Accidents)
- Probabilistic Safety Analyses (Level 1, 2 and 3)

Safety Analysis

By the term safety analysis an analytical study is meant by which it is demonstrated how safety requirements, such as ensuring the integrity of barriers against radioactive releases and various other requirements, are met for initiating events (both internal and external) occurring in a broad range of operating conditions, and in other circumstances, such as varying availability of the plant systems,

Two balanced complementary methods of safety analysis, *deterministic* and *probabilistic*, are used jointly in evaluating the safety of an NPP.

Tools for Accident Analysis

Accident analysis is performed with a number of computer codes, as summarised in the section on accident consequences;

It is important that the computer codes used to perform accident analysis are verified and validated for the accident scenarios of interest, and contain models for the appropriate phenomena;

A number of codes which are widely accepted as well validated for different accident scenarios have been developed;

Phenomenological uncertainty and accident variability must be appreciated

Safety analysts should have a deep knowledge of both the code used in the analysis and the physics involved in the accident sequence which is simulated.

Types of Accident Analysis (1/3)

The results of safety analysis can be used in different areas:

- **Design Analysis;**
  - Design analysis is used in the design of a new plant or in modifications to the design of an existing plant, so that the designer can confirm that the design meets the relevant design and safety requirements;
- **Licensing Analysis;**
  - Licensing analysis is used in the design of a new plant, or in modification of the design of an existing plant, to provide evidence to the regulatory body that the design is safe. Regulatory bodies may require new calculations when new evidence arises from research, both theoretical and experimental, or from operational experience at the plant or similar plants;
  - Licensing analyses may include conservatisms to ensure margins of safety

Types of Accident Analysis (2/3)

- **Validation of EOPs and Plant Simulators;**
  - Emergency operating procedures (EOP) define the operator actions during anticipated transients and in accident conditions. Owing to the very limited possibility of using real plant transients for validation of EOPs, analyses by sophisticated computer codes are used to support the development and validation of EOPs. Where possible, use should be made of plant simulators;
- **Analysis of operational events;**
  - Accident analysis is frequently used as a tool for a full understanding of events occurring during the operation of NPPs, as part of the feedback of operational experience;
- **Regulatory audit analysis;**
  - Audit analysis is generally used by regulatory bodies to perform an independent verification of DBAs within the framework of licensing processes, to supplement the task of reviewing and assessing the design and operation of NPPs or to check the completeness and consistency of accident analyses submitted for licensing purposes;

Types of Accident Analysis (3/3)

- **Support for Accident Management and Emergency Planning;**
  - Analysis of accidents for supporting accident management describes the plant behaviour in conditions for DECs. Operator actions are normally accounted for in the assessment of DECs. The results from analyses of DECs are used to develop operator strategy, the main goals being to prevent severe core damage and to mitigate the consequences of an accident in the event of core damage. Analysis is needed to develop threshold values to initiate SAMG actions, and to develop scenarios for the validation of the SAMG and training for plant staff;
- **Probabilistic Safety Analysis (PSA)**
  - PSA is often used to verify compliance with safety goals or criteria, which are usually formulated in terms of quantitative estimates of core damage frequency, frequencies of radioactive releases of different types and societal risks. In ***Level 1 PSA***, the design and operation of the plant are analysed in order to identify the sequences of events that can lead to core damage and the core damage frequency is estimated. ***Level 2 PSA*** estimates the frequency, magnitude and other relevant characteristics of the release of radioactive material to the environment for the core damage sequences identified in Level 1. In ***Level 3 PSA***, public health and other societal consequences are estimated, such as the contamination of land or food from the accident sequences that lead to a release of radioactivity to the environment.

Accident Analysis Methods (1/2)

Accident analysis can be performed according to a conservative approach, a best estimate approach or a combination of the two:

- **Conservative Analysis;**
  - In the conservative approach, the result of the analysis bounds the plant's actual response. A conservative analysis does not give any indication of the margins between the plant's actual response and the conservatively estimated response;
  - Conservatism can be introduced in the code or in the plant data or both. A conservative code implements a combination of all the models necessary to provide a pessimistic bound to the processes relating to specified acceptance criteria. Conservative plant data are chosen in such a way that plant parameters, initial plant conditions and assumptions about availability of equipment give a pessimistic result, when used in a safety analysis code, in relation to specified acceptance criteria;
  - Excessive conservatism can be unrealistic and produce too stringent implications on costs

Accident Analysis Methods (2/2)

- **Best Estimate Analysis;**
  - A best estimate approach ensures that the predicted plant behaviour with given uncertainty includes the actual plant value. Best estimate analyses provide a good view of the existing margins or limits on NPP operation in relation to safety analyses;
  - The use of a best estimate code is essential for a best estimate analysis;
  - Such codes do not include models that are intentionally designed to be conservative;
  - A best estimate code includes a combination of the best estimate models necessary to provide a realistic estimate of the overall response of the plant during an accident;
- **Sensitivity and Uncertainty;**
  - Sensitivity analyses include systematic variations in code input variables or modelling parameters to determine the influence of important phenomena or models on the overall results of the analysis, particularly the key parameters for an individual event;
  - Uncertainty analyses include the estimation of uncertainties in individual modelling or in the overall code, uncertainties in representation and uncertainties in plant data for the analysis of an individual event;
- **See USNRC SOARCA analyses for PWR and BWR plants**

Summary of PSA (1/2)

PSA: 3 levels

PSA-1

Frequency of core damage sequences

→ Plant Damaged States →

PSA-2

Frequency, containment failure mode, release kinetics and amplitude

→ Release Categories →

PSA-3

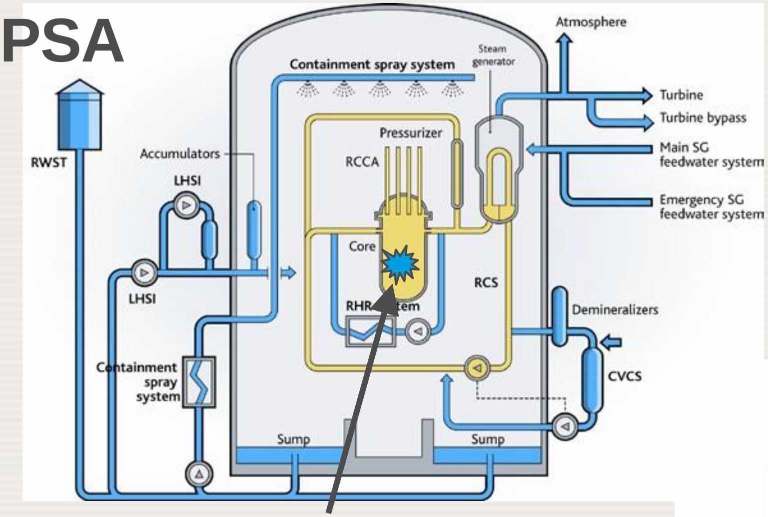
Frequency

Health, economical, environment consequences

PSA-1	PSA-2	PSA-3
Initiating events System reliability Thermal-hydraulics, Human reliability	Severe accident phenomena Systems behavior Human reliability	Atmospheric dispersion Weather, Population Economy
Operating procedures	Severe accident management guidelines	Counter-measures for the protection of population

Summary of PSA (2/2)

PSA

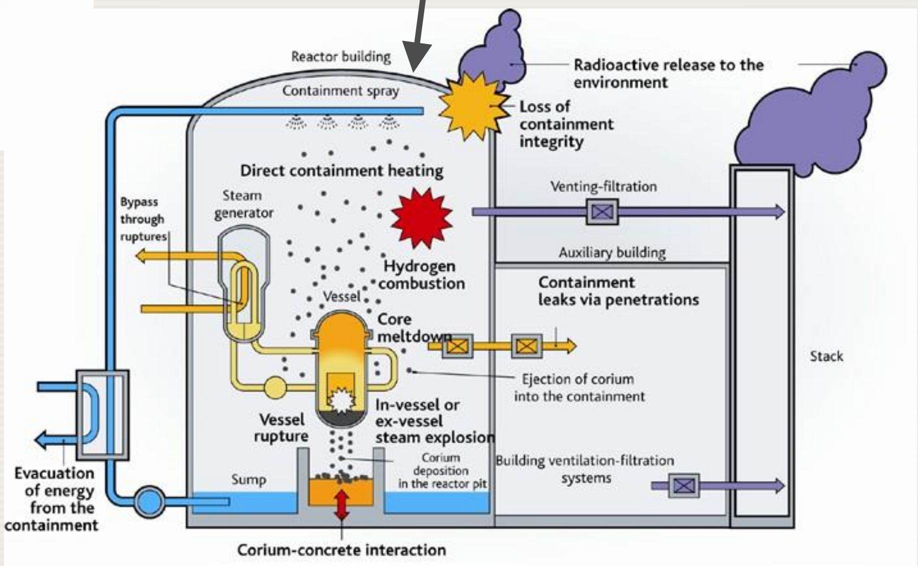


L1 PSA

Core degradation initiated by

- A lack of core cooling (e.g TMI-2)
- An overpower (reactivity initiated event: e.g Tchernobyl)

L2 PSA : Severe accident progression, containment failure mode, amplitude of release, kinetics of release.



Nuclear Safety Evolution

Design Safety Requirements have increased:

- **Separation of redundant trains;**
  - e.g. Fire confinement;
- **Passive safety systems;**
  - e.g. Physical processes instead of powered (active) technical components;
- **Protection against hazards;**
  - Natural hazards, e.g. seismic;
  - Man-made hazards, e.g. plane crash.

Classes of Severe Accident Code (recap 1/2)

Severe accident codes can be classified into three classes: fast running integral codes, detailed/mechanistic codes (usually slow running), and special (dedicated) codes:

- **Fast running integral codes:**
  - Their models are less mechanistically based but more of a parametric character, i.e. model parameters allow the user to investigate the consequences of uncertainties on key results;
  - These kinds of codes may also have been used for the design and validation of severe accident prevention and mitigation systems;
  - However, to obtain realistic results, a deep knowledge of the involved physical phenomena as well as user experience in performing severe accident analysis are required. Some examples of fast running integral codes are MAAP (US), MELCOR (US) and ASTEC (EU).

Classes of Severe Accident Code (recap 2/2)

- **Detailed codes:**
  - These model as far as possible all relevant phenomena in detail by mechanistic models;
  - Basic requirements for detailed codes are that the modelling uncertainties are comparable with (i.e. not higher than) the uncertainties in the experimental data used to validate the code and that user-defined parameters are only necessary for phenomena which are not well understood due to insufficient experimental data;
  - ATHLET-CD (EU), ICARE/CATHARE (EU), SCDAP/RELAP5 (SCADPSIM) (US), COCOSYS (EU) and CONTAIN (US) are examples of such detailed codes. In addition, ASTEC and MELCOR can be considered detailed codes, if the calculation is based on extensive nodalisation and detailed model options;
  - The drawback of detailed codes is the long computational times required in the analysis. Furthermore, most phenomena which become relevant in the simulation after core damage are not completely understood yet, which precludes the possibility of a detailed analysis of this phase.
- **Special (dedicated) codes deal with single phenomena:**
  - Examples are MC3-D for steam explosions and ADINA-F for molten pool behaviour.

Quality of Accident Analysis – Quality Assurance (1/3)

Quality Assurance (QA)

- **Accident analysis needs to be the subject of a comprehensive quality assurance programme applied to all activities affecting the quality of the final results;**
- **The quality assurance programme needs to define the quality assurance standards to be applied in accordance with national requirements and internationally recognized good practices;**
- **Such a programme would consider following general principles. Formalized quality assurance procedures and/or instructions need to be developed and reviewed for the whole accident analysis process, including:**
  - Collection and verification of plant data;
  - Verification of the computer input deck developed and documentation of detected errors;
  - Validation of plant models.

Quality of Accident Analysis – Input data (1/2)

The *preparation* of input data takes place in four phases:

- ***Collection*** **of plant data: from technical specification, documentation of plant design, operational data;**
- ***Development*** **of an engineering handbook and input deck: the engineering handbook details all the calculations and assumptions which have been used to develop the input deck from the plant data;**
- ***Verification*** **of the data: the input deck is checked for formal correctness. i.e. that no erroneous data have been introduced into it and that all formal and functional requirements are fulfilled accurately and therefore will permit its successful use;**
- ***Validation*** **of input data: the purpose of validating input data is to demonstrate that the model adequately represents the functions of the modelled systems.**

Quality of Accident Analysis – Input data (2/2)

The *validation* of input data for severe accident analysis takes place in six phases, a check should be made for:

- **Steady state response;**
- **Mass and energy balances;**
- **Time step convergence (sensitivity calculations with variation of the time step size) and spatial convergence (sensitivity calculations with variation of the core/primary system/containment meshing);**
- **Behaviour and function of system components;**
- **Timing of events (i.e. cladding rupture, onset of zirconium oxidation, beginning of fuel melting, relocation of fuel to the lower plenum, vessel failure);**
- **Timing of some key events and key parameters (integral hydrogen generation, fission product release fractions, peak temperatures and pressure response, cavity ablation, etc.).**

Quality of Accident Analysis – Overall check

The results need to be *checked for overall behaviour* ("reality check");

- **The predicted plant behaviour should be consistent with the expected plant behaviour;**
- **The timing of events in the accident sequence and key parameters, such as the hydrogen generation and peak temperatures, should be checked by engineering judgement, taking into account the experience from integral experiments as well as the results of other available severe accident analyses;**
- **This requires a detailed knowledge about the phenomena occurring during a severe accident.**
- **Severe accident phenomenological uncertainties and accident sequence variability are significant – uncertainty quantification is recommended (See USNRC SOARCA studies)**

Conclusions

A summary of the accident consequences and analysis has been presented, in particular:

- Results of a severe reactor accident:
  - On-site consequences;
  - Off-site consequence analysis;
  - General considerations for source term calculations;
  - Severe accident consequences codes;
  - Emergency preparedness;
- Accident analysis:
  - Safety assessment and analysis;
  - Types of accident analysis;
  - Accident analysis methods;
  - Computer codes for accident analysis;
  - Quality of accident analysis.