SH2705 Simulation Course
Safety Assessment and Safety Analysis

Sean Roshan

Safety Assessment

Assessment of all aspects of a practice that are relevant to protection and safety; for an authorized facility, this includes siting, design and operation of the facility.

Analysis to predict the performance of an overall system and its impact, where the performance measure is the radiological impact or some other global measure of the impact on safety

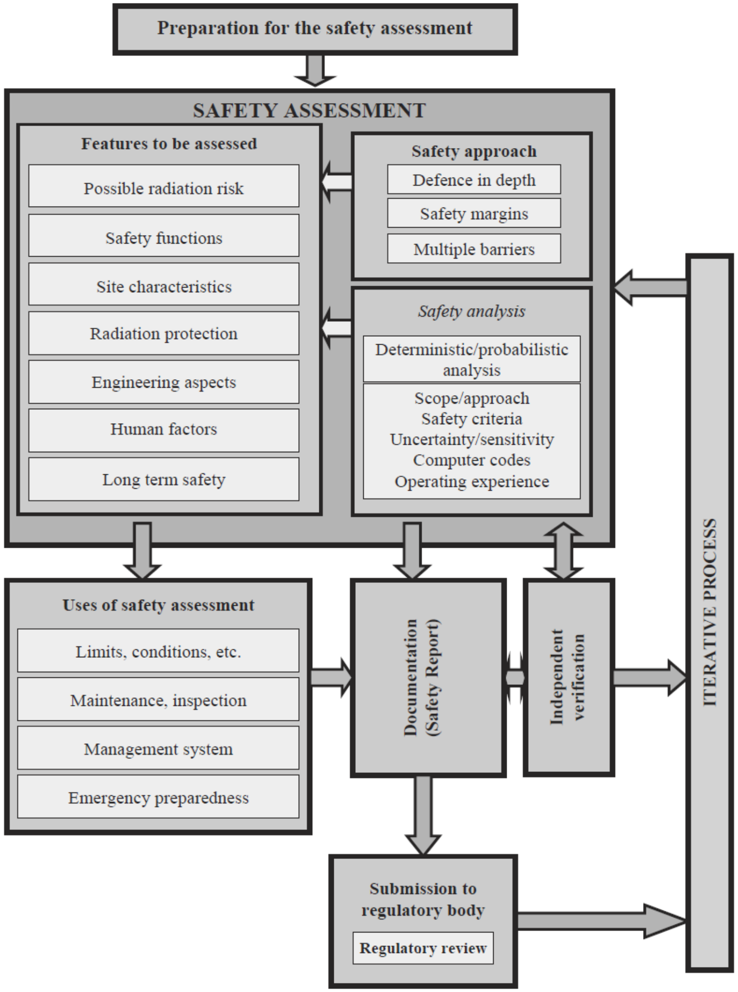
I.A.E.A

Safety Assessment (contd.)

- Safety assessment is undertaken as a means of evaluating compliance with safety requirements (and thereby the application of the fundamental safety principles) for an energy transformation plant (e.g. an NPP) to determine the measures that need to be taken to ensure safety.
- Its is the assessment of all aspects of the plant that are relevant to protection and safety. This includes siting, design and operation of the plant.
- Safety assessment is the systematic process that is carried out throughout the lifetime of the plant to ensure that all the relevant safety requirements are met by the proposed (or actual) design.
- Safety assessment includes, but is not limited to, the formal safety analysis.

I.A.E.A

Safety Assessment (contd.)



Overview of safety assessment process for an NPP

Needs for Safety Assessment

Safety assessment is used for:

- Licensing of a new plant and/or renewing the license of an operating plant,
- Safety review, periodic safety review,
- Plant modification, upgrading and modernization projects,
- Assessment of operational experience,
- Final safety analysis report, updating of safety analysis report,
- Determining and updating of operational limits and conditions,
- Updating the safety relevant programs and procedures

Safety Assessment is performed by:

- Designer safety assessment
- Licensee safety assessment
- Regulatory safety assessment and review

Safety Assessment during lifetime of a facility

Safety assessment is carried out during different stages of a facility's lifetime:

- Site evaluation for the facility or activity;
- Development of the design;
- Construction of the facility or implementation of the activity;
- Commissioning of the facility or of the activity;
- Commencement of operation of the facility or conduct of the activity;
- Normal operation of the facility or normal conduct of the activity;
- Modification of the design or operation;
- Periodic safety reviews;
- Life extension of the facility beyond its original design life;
- Changes in ownership or management of the facility;
- Decommissioning of a facility;
- Closure of a disposal facility for radioactive waste and the post-closure phase;
- Remediation of a site and release from regulatory control.

Insights and Practical Experience

- An NPP design involves millions of individual components design, decisions impacting safety.
- Safety Assessment of these millions of design elements without Codes and Standards implies millions of individual issues to be assessed.
- Use of Codes and Standards reduces critical safety related design decisions on NPP Design Margins to re-producible, "transparent", and mutually accepted approaches.

Insights and Practical Experience

Regulations, Codes, Standards

Hierarchy of Safety Requirements Documents used to address Design Margins:

- National Laws – Obligatory (Policy)
  - Issued by Governments or Parliaments with inputs from Regulatory Body
- Regulations – Obligatory (General)
  - Issued by Regulatory Body, may reference Codes & Standards
- Regulatory Guidance – Suggested (Detailed)
  - Issued by Regulatory Body, defines accepted option, may reference Codes & Standards.
- Codes & Standards – "Optional?" (Detailed)
  - Issued by Professional Groups, defines acceptable option.

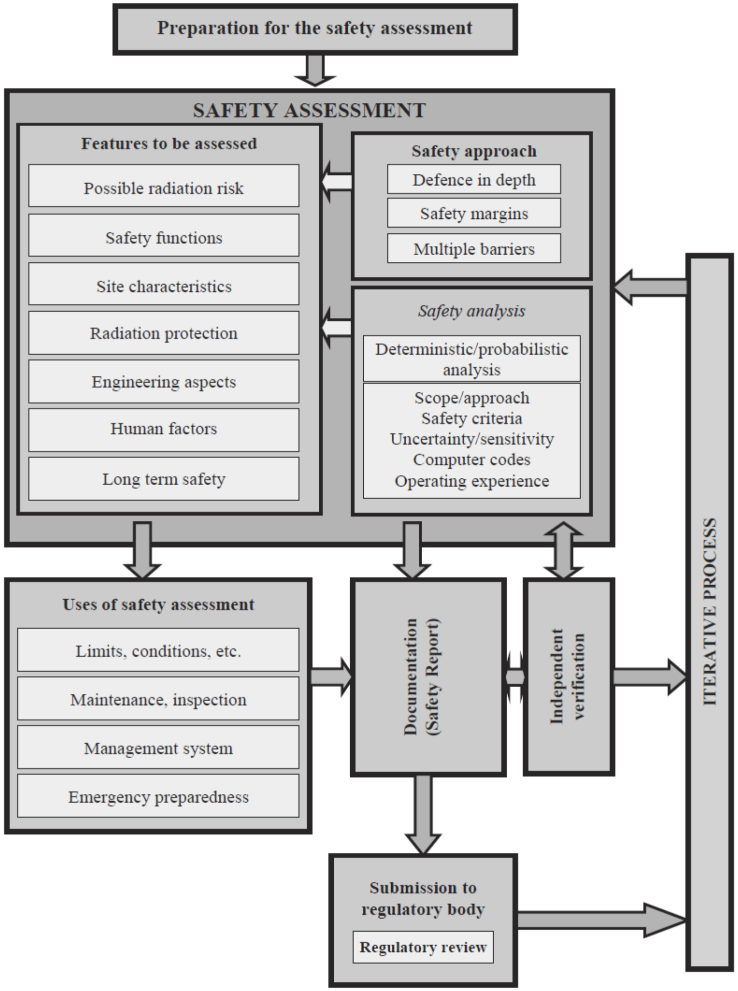
Insights and Practical Experience (contd.)

Regulations, Codes, Standards

- Piping, Pressure Vessel, Containment, Seismic Structural support design limits (stress analyses) are historically performed based on conservative Industry Codes and Standards. (in most countries)
- Sizing of cabling, breakers, electrical components are historically based on Loads in conservative Industry Codes and Standards. (in most countries)
- Fuel Rod Critical Heat Flux (CHF) Design Margins typically rely on detailed analysis including uncertainties.
- DBA conseravative LOCA has been replaced by Best Estimate LOCA which relies on detailed consideration of uncertainties.

Overview of Safety Assessment

- Safety Assessment of NPPs has evolved to assess mitigated accident source terms based on assumption of working engineered safety systems.
- Safety Assessment of NPP safety systems involves assessment of both Reliability and Design Margins.
- Assuring Reliability is based on Single Failure Criteria, and On-line Testability.
- Assuring Safety Margins is based on either use of conservative Regulations, Codes, and Standards, or in a limited number of areas performing detailed margins analysis (Fuel Rod CHF limits, Best Estimate LOCA).



Safety Assessment Hierarchy

Safety Assessment

Safety Analysis

Deterministic Safety Analysis

- Predict the response to postulated events with predetermined assumptions
- Checks fulfilment of acceptance criteria

Probabilistic Safety Analysis

Combines the
- likelihood of initiating scenarios
- and their consequences

Into estimation of
- Core damage frequency
- Source term release
- Overall risk

Evaluation of engineering factors important to safety

- Proven engineering practices
- Defence in Depth
- Radiation protection
- Safety classification
- Protection against internal and external hazards
- Combination of loads
- Selection of materials
- Single failure criterion
- Redundancy, diversity
- Equipment qualification
- Ageing
- Man – machine interface

…

Safety Analysis

- Evaluation of the potential hazards associated with the operation of a facility or the conduct of an activity.
- The formal safety analysis is part of the overall safety assessment; that is, it is part of the systematic process that is carried out throughout the design process (and throughout the lifetime of the facility or the activity) to ensure that all the relevant safety requirements are met by the proposed (or actual) design.

Safety Analysis (contd.)

Applying methods of deterministic and probabilistic analysis, shall be provided which establishes and confirms the design basis for the items important to safety and demonstrate that overall plant design is capable of meeting the prescribed and acceptable limits for radiation doses and releases for each plant condition category and that defense- in-depth is achieved

Safety Analysis (contd.)

- Safety analysis is an essential element of a safety assessment. It is an analytical study used to demonstrate how safety requirements are met for a broad range of operating conditions and various initiating events.
- Safety analysis involves deterministic and probabilistic analyses in support of the siting, design, commissioning, operation or decommissioning of an NPP.

Safety Analysis (contd.)

Safety analysis is the study of how the reactor behave during abnormal conditions.

- as a step in the design process
- an essential part of the safety assessment in the licensing process

Deterministic safety analysis employs calculational models which describe the physical processes in reactor/plant systems to examine the plant's behavior after an assumed initial event or malfunction.

- "Deterministic" is because of one result (one sequence) at a time.

Probabilistic safety analysis studies the reliability of the safety systems and identifies event sequences which can lead to core melting (Level 1), containment failure (Level 2) and off-site consequences (Level 3).

Goal of Safety Analyses

Identify hidden failures or weakness in safety functions (backdoors in the system)

- Instability events
- Steam line blockage
- Pressure control failure
- Control rod insertion by electrical motors
- Fuel damage because of dryout
- …

Safety Landscape is *very complex*, appropriate methods for analysis should be developed to define the boundary of this landscape

Safety Analysis: Deterministic approach

Deterministic approach: analyze plant behavior under specific predetermined operational states and accident conditions, with specific set of rules in judging design adequacy (conservative approach is often used for design purposes)

A **deterministic** model does not include elements of randomness. Every time you run the model with the same initial conditions you will get the same results

Safety Analysis: Probabilistic approach

Probabilistic approach: determining all significant contributors to risk and evaluating the balance of the overall system configuration

A **probabilistic** model includes elements of randomness. Every time you run the model, you are likely to get different results, even with the same initial conditions. A probabilistic model is one which incorporates some aspect of random variation.

Deterministic and Probabilistic analysis are two complementary approaches to be used

Safety Analysis Report

- A Safety Analysis Report (SAR) is a document which enables the regulatory body to assess the safety of a nuclear power plant.
  - Preparing a preliminary SAR (PSAR) is part of the licensing process for new build of nuclear power plants.
  - Final SAR (FSAR) is part of for operating license.
  - Revision of a FSAR is usually part of the process of extending a license.

Deterministic Safety Analysis

- The mechanistic thinking era
- Determinism, everything has a cause and can be explained why it happened
- Knowing everything that can happen when dealing with a machine
- Analysis to know how things work

Deterministic Safety Analysis (contd.)

- DSA answers the question; if a reactor design is adequate and licensable?
- DSA is a tool in developing plant protection and control systems, set points and control parameters, and the technical Specifications of the plant.

Why "deterministic"? :

- Traditionally:
  - No randomness (probabilities) in the calculation (in principle)
  - Results are single numerical values with probability 1 (this is not true for "best-estimate" analyses)

Deterministic Safety Analysis (contd.)

DSA is used to:

- Demonstrates the effectiveness of the equipment incorporated to prevent escalation of AOOs and DBAs to severe accidents and to mitigate their effects.
- Demonstrates that the safety systems can:
  - Shutdown the reactor and maintain it in safe shutdown condition during and after DBA.
  - Remove residual heat from the core after reactor shutdown from all operational states and DBA conditions.
  - Ensure that radioactive releases during DBA are below acceptable limits.

Deterministic Safety Analysis (contd.)

DSA for normal operation of the plant:

- Ensures that normal operation is safe (with radiological doses and releases of radioactive materials within acceptable limits. Also checking if they are ALARA) and with plant parameters not exceeding operating limits.

- Should establish the conditions and limitations for safe operation, including:
  - Safety limits for reactor protection and control and other engineered safety systems.
  - Operational limits and reference settings for the control system.
  - Procedural constraints for operational control of processes.
  - Identification of allowable operating configurations.

Requirements for deterministic methodology

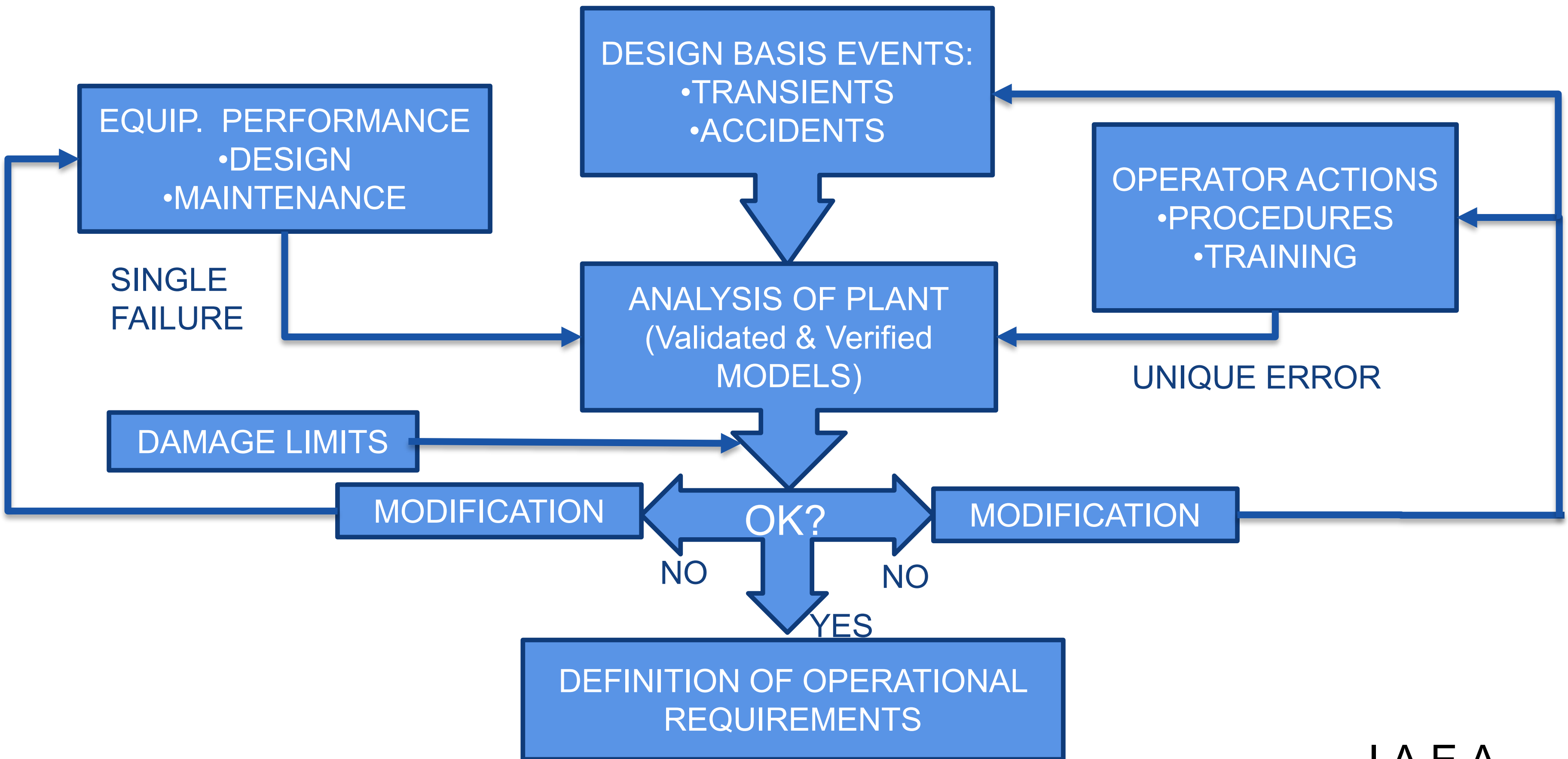
- Compliance of operational limits in the assumptions
- PIEs appropriate for the design and the site
- Analysis of the events sequences resulting from PIEs
- Compliance with acceptance criteria
- Assessment of the degree of conservatism
- Uncertainty method together with the best estimate methodology

The Deterministic Approach

We need to limit the scope and extent of the deterministic safety analysis:

- simplifying assumptions and bounding situations
- design basis events
- single failures
- actions taken to fulfill the assumptions made in the safety analysis
- unique error of the operator

The Deterministic Approach (contd.)



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The Objectives of DSA

- Confirm that the design of an NPP meets design and safety requirements
- Derive or confirm operational limits and conditions that are consistent with the design and safety requirements for the NPP
- Assist in establishing and validating accident management procedures and guidelines
- Assist in demonstrating that safety goals, which may be established to limit the risks posed by the NPP, are met

DSA Applications

- DSA establish and confirm the DB for items important to safety, ensuring that the plant design meets safety and radiological criteria (integrity of barriers).
- As part of the Safety Assessment with the aim to determine the effectiveness of "defense barriers".

DESIGN BASIS (DB): the range of conditions and events taken explicitly into account in the design of a plant, so that the plant can withstand them w/o exceeding authorized limits, by the planned operation or safety systems.

DSA Applications (contd.)

- **Design Applications**
  - Designer: as part of the design and construction process
  - Operating organization, to confirm the design

DSA must be parallel to the design process, with iteration between them.

- **Licensing Applications**
  - Calculations for Final Safety Analysis Report (FSAR)
  - Fuel reload analysis
  - Periodic SA of an operating plant
  - Safety justification of a design modification

The final SA must reflect the final plant design. DSA is also used for evaluating design changes, supporting decision-making processes, revealing new issues, etc.

- **Regulatory Applications**
  - Audit calculations
  - Evaluation of emergency operating procedures
  - Review of significant events and incidents
  - Evaluation of emergency operating procedures
  - Unresolved Safety Issues Evaluation

Types of DSA

DSA can be conservative or best- estimate:

- Conservative: use pessimistic or worst-case assumptions and models. Most of the analysis presented to regulatory bodies follow this approach.

- Best-estimate or realistic: most of assumptions and models are realistic (some conservatisms are maintained), include uncertainty analysis.

Conservatism

The concept of conservatism was introduced to the analyses, when:

- the capabilities (physical knowledge and modelling, experimental database and the computer capacity) were insufficient for the realistic calculations.
- to account for the statistical character of the plant data
- to account for equipment failures.

By using pessimistic assumptions and simplifications concerning the initial conditions, boundary conditions and the physical models, it was believed that the limiting results could be obtained for the chosen bounding cases.

One of the basic difficulties of that approach is that too many and too coarse assumptions may result in very unrealistic sequences, which in the worst cases may turn out not to be conservative, since the real progression might lead to more limiting results.

Best-estimate

- **Best-estimate or realistic DSA:**
  - Started to develop when the capabilities for simulating the phenomenology originated by accidents increased.
  - Try to unbiasedly reproduce the real plant behavior during an accident or transient.
  - Realistic models and assumptions.
  - Must include an uncertainty analysis for the important results, that must be given with an "error interval".

Best Estimate Approaches

The BE codes are best estimate in the sense that the physical modelling applied uses the best knowledge of the phenomena available. In such areas, as severe accident analysis, the BE methods have been striven for from the beginning.

The basic elements of successful application of the BE methods are:

- the codes are carefully validated against the existing database,
- the code users are well educated,
- the associated uncertainties can be quantified or are at least qualitatively understood and managed.

The code users should be experienced in performing complex system analysis.
The validation calculations against the experimental database from various facilities and successful simulation of the plant experience from the real transients and accidents are efficient means for user training

Best Estimate Approaches (contd.)

- **Uncertainty evaluation in DSA:**
  - In principle, being realistic is harder that being pessimistic. Conservative models can be simple.
  - Need for robust demonstration that there are large safety margins.
  - In both approaches you must know the accuracy of your models and assumptions. But in the BE approach you must quantify such accuracy (uncertainty study).
  - Given an accident scenario in a plant, a conservative analysis can make use of only one or some few computer code runs. But in a BE Plus Uncertainty (BEPU) analysis you need "many" computer runs, in order to carry out the uncertainty analysis.

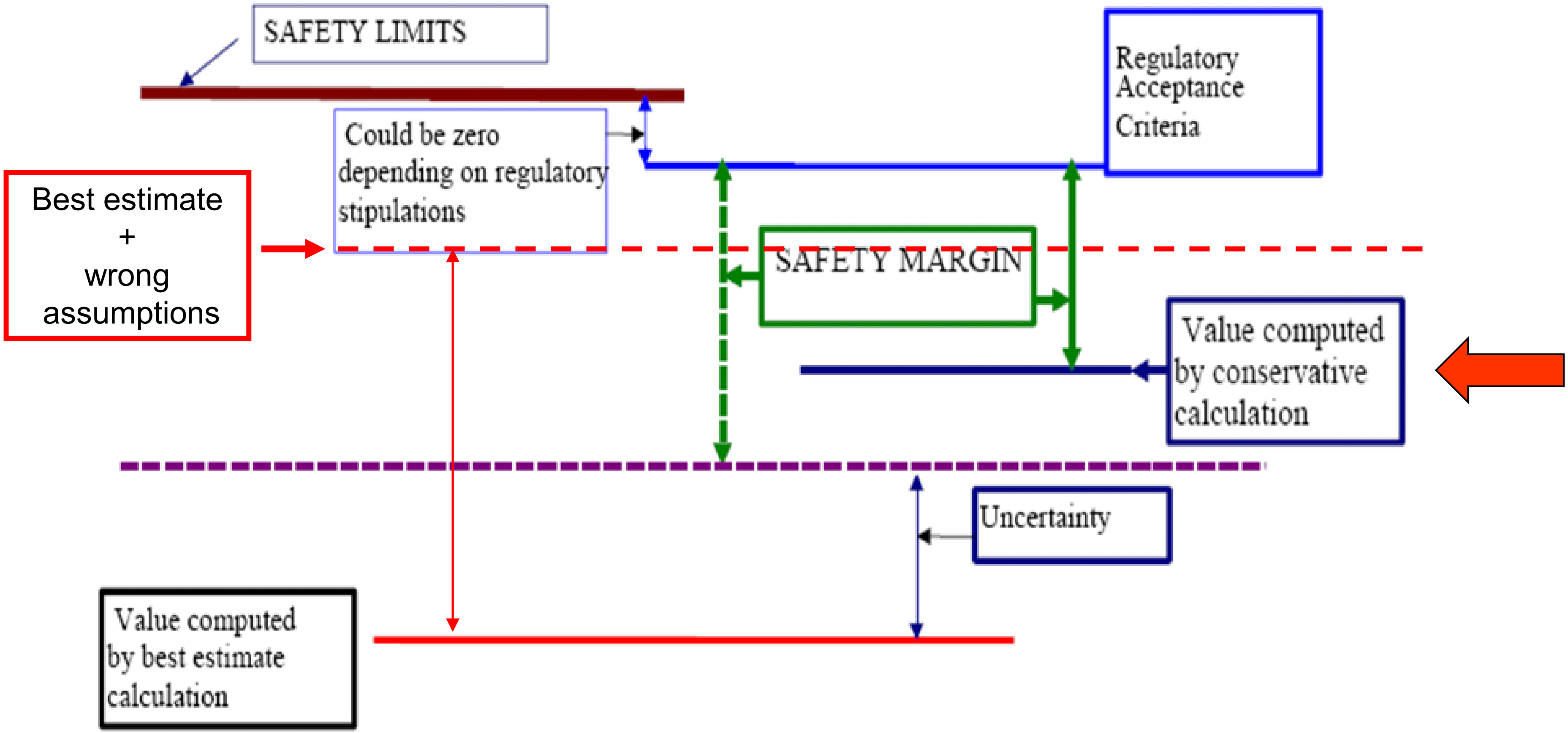
Evolution of DSA

Applied codes	Input & BIC (boundary and initial conditions)	Assumptions on systems availability	Approach
Conservative codes	Conservative input	Conservative assumptions	Deterministic*
Best estimate (realistic) codes	Conservative input	Conservative assumptions	Deterministic
Best estimate codes + Uncertainty	Realistic input + Uncertainty	Conservative assumptions	Deterministic
Best estimate codes + Uncertainty	Realistic input + Uncertainty	PSA-based assumptions	Deterministic + probabilistic

Best Estimate Approaches (contd.)

- **The advantages of a realistic DSA:**
  - You look for the "real" performance of your plant. Conservative methodologies use to be physically unrealistic (misleading sequences of events, unrealistic time scales, missing of physical phenomena). BE calculations can provide guidance in developing accident management plans.
  - Lower margins: real safety margins adopted for a plant with a conservative approach may be unnecessarily large. BEPU margins may permit augment reactor power.
  - You have a precise idea about the sensitivity of the calculations to variables and parameters.

Regulatory Acceptance Criteria



Conservative vs. Best-estimate approaches

- Deterministic Safety Analysis has been traditionally carried out with a conservative or pessimistic bias.
- Conservative DSA makes use of pessimistic assumptions everywhere, so that the results of the analyses are expected to be "worse" than realistic ones ("bounding"):
    - Conservative initial and boundary conditions.
    - Models in the computer codes are chosen as conservative.

Conservative vs. Best-estimate approaches

- Conservative DSA have been very popular, because it is relatively "easy" to perform. But the convenience of such an approach does not "excuse" the analyst from being unaware of the accuracy of the models and assumptions.
- A very characteristic example of conservative analysis: LOCA analysis for LWR according to section 46 and appendix K of the 10 CFR 50. The conservativeness imposed by the appendix K requirements is very large, because some parameters/models are given overwhelmingly pessimistic values.

CONSERVATIVE VS. BEST-ESTIMATE
LOCA Analysis

LOCA analysis for LWR section 46 and appendix K to 10 CFR 50

- Conservatisms imposed by the Appendix K to 10 CFR 50:
  - Stored energy: initial steady temperatures chosen so as to maximize the stored energy in the fuel.
  - Decay heat: heat generation rate from radioactive decay are 1.2 times the 1971 ANS Standard (this is an overestimation of about five standard deviations !!!).
  - Metal-water reaction: conservative Baker-Just model. If cladding ruptures, both inner and outer surfaces are assumed to react.

CONSERVATIVE VS. BEST-ESTIMATE
LOCA Analysis (contd.)

- Discharge from break: critical flow is based on the conservative Moody model multiplied by discharge coefficients (from 0.6 to 1.0) that lead to the worst results.
- ECCS bypass: during most of the blowdown period for a PWR cold leg break, the ECCS water is assumed to be ineffective in refilling the system.
- No return to nucleate or transition boiling: once Critical heat flux (CHF) has occurred in the blowdown period, no return to nucleate or transition boiling is allowed during blowdown; it must be postponed until the reflood period.

CONSERVATIVE VS. BEST-ESTIMATE
LOCA Analysis (contd.)

- Film boiling correlations, chosen to under predict data.
- Single failure: it is assumed that one of the ECCS components fails.
  - The failure leading to the highest damage is chosen.

CONSERVATIVE VS. BEST-ESTIMATE
LOCA Analysis (contd.)

Acceptance criteria for a LOCA Analysis (after 10 CFR 50.46)

- Peak cladding temperature (PCT) lower than 2200 °F.
- Maximum cladding oxidation lower than 0.17 times the total cladding thickness before oxidation. If cladding rupture is predicted, the inside surfaces will participate in the oxidation
- Maximum hydrogen generation resulting from the cladding oxidation: lower 0.01 times the amount that would be generated if all the cladding metal were to react.
- Core geometry will remain amenable to cooling.
- Long-term cooling.

CONSERVATIVE VS. BEST-ESTIMATE
LOCA Analysis (contd.)

Best-estimate LOCA analysis:

- Makes use of realistic assumptions and codes: TRACE, RELAP5, APROS, POLKA-T, SIMULATE, TRAC-P, TRAC-B,…that incorporate state-of-the-art models.
- Must include an uncertainty analysis.
- Drops out the Appendix K requirements.
- Regulatory door open:
  - SECY-83-472
  - 1988 revision of 10 CFR 50
  - Regulatory Guide 1.157 (1989)
  - CSAU Methodology (1989)

Contents of Safety Analysis Report

- **USNRC Regulatory Guide 1.70** *

1. Introduction and general description of plant
2. Site characteristics
3. Design of structures, components, equipment and systems
4. Reactor
5. Reactor coolant system and connected systems
6. Engineered safety features
7. Instrumentation and control
8. Electrical power
9. Auxiliary systems
10. Steam and power conversion system

* For new reactors see RG 1.206 issued in 2007

- ***IAEA Safety Standards, No. SSG-61***

1. *Introduction and general considerations*
2. *Site characteristics*
3. *Safety objectives and design rules for structures, systems and components*
4. *Reactor*
5. *Reactor coolant system and associated systems*
6. *Engineered safety features*
7. *Instrumentation and control*
8. *Electrical power*
9. *Auxiliary systems and civil structures,*
   - A. *Auxiliary systems,*
   - B. *Civil engineering works and structures*
10. *Steam and power conversion systems*

Contents of Safety Analysis Report cont.

- **USNRC Regulatory Guide 1.70** *

11. Radioactive waste management
12. Radioactive protection
13. Conduct of operation
14. Initial test program and Inspections, ITAAC-design (Tests, Analyses, and Acceptance Criteria) certification
15. Accident analysis
16. Technical specifications
17. Quality assurance
18. Human factors engineering
19. Severe accidents

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- ***IAEA Safety Standards, No. SSG-61***

11. *Management of radioactive waste*
12. *Radiation protection*
13. *Conduct of operation*
14. *Plant construction and commissioning*
15. *Safety analysis*
16. *Operational limits and conditions for safe operation*
17. *Management for safety*
18. *Human factors engineering*
19. *Emergency preparedness and response*
20. *Environmental aspects*
21. *Decommissioning and end of life aspects*

U.S. NRC standard review plans (SRP)

- The U.S. NRC has several standard review plans (SRP) for staff use in reviewing proposed licensing actions. These actions may relate to
  - nuclear facility
    - constructing,
    - operating,
    - decommissioning.
  - nuclear materials or waste
    - possessing,
    - using,
    - storing,
    - transporting.
- The SRP establish criteria to use in evaluating applications to construct and operate nuclear power plants.
- The SRP is not a substitute for the NRC's regulations, and compliance with it is not required, However, the applicant must show that they are complying with the regulation U.S. NRC and full fill the acceptance criteria

Review of Safety Analysis Reports for Nuclear Power Plants : LWR Edition (NUREG-0800)

Chapter 1, Introduction and Interfaces
Chapter 2, Sites Characteristics and Site Parameters
Chapter 3, Design of Structures, Components, Equipment, and Systems
Chapter 4, Reactor
Chapter 5, Reactor Coolant System and Connected Systems
Chapter 6, Engineered Safety Features
Chapter 7, Instrumentation and Controls
Chapter 8, Electric Power
Chapter 9, Auxiliary Systems
Chapter 10, Steam and Power Conversion System
Chapter 11, Radioactive Waste Management
Chapter 12, Radiation Protection
Chapter 13, Conduct of Operations
Chapter 14, Initial Test Program and ITAAC-Design Certification
Chapter 15, Transient and Accident Analysis
Chapter 16, Technical Specifications
Chapter 17, Quality Assurance
Chapter 18, Human Factors Engineering
Chapter 19, Severe Accidents

NUREG-0800, Chapter 15, Transient and Accident Analysis

1. Steam System Piping Failures Inside and Outside of Containment (PWR)
2. Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve
3. Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR
4. Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)
5. Loss of Nonemergency AC Power to the Station Auxiliaries
6. Loss of Normal Feedwater Flow
7. Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
8. Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
9. Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
10. Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition
11. Uncontrolled Control Rod Assembly Withdrawal at Power

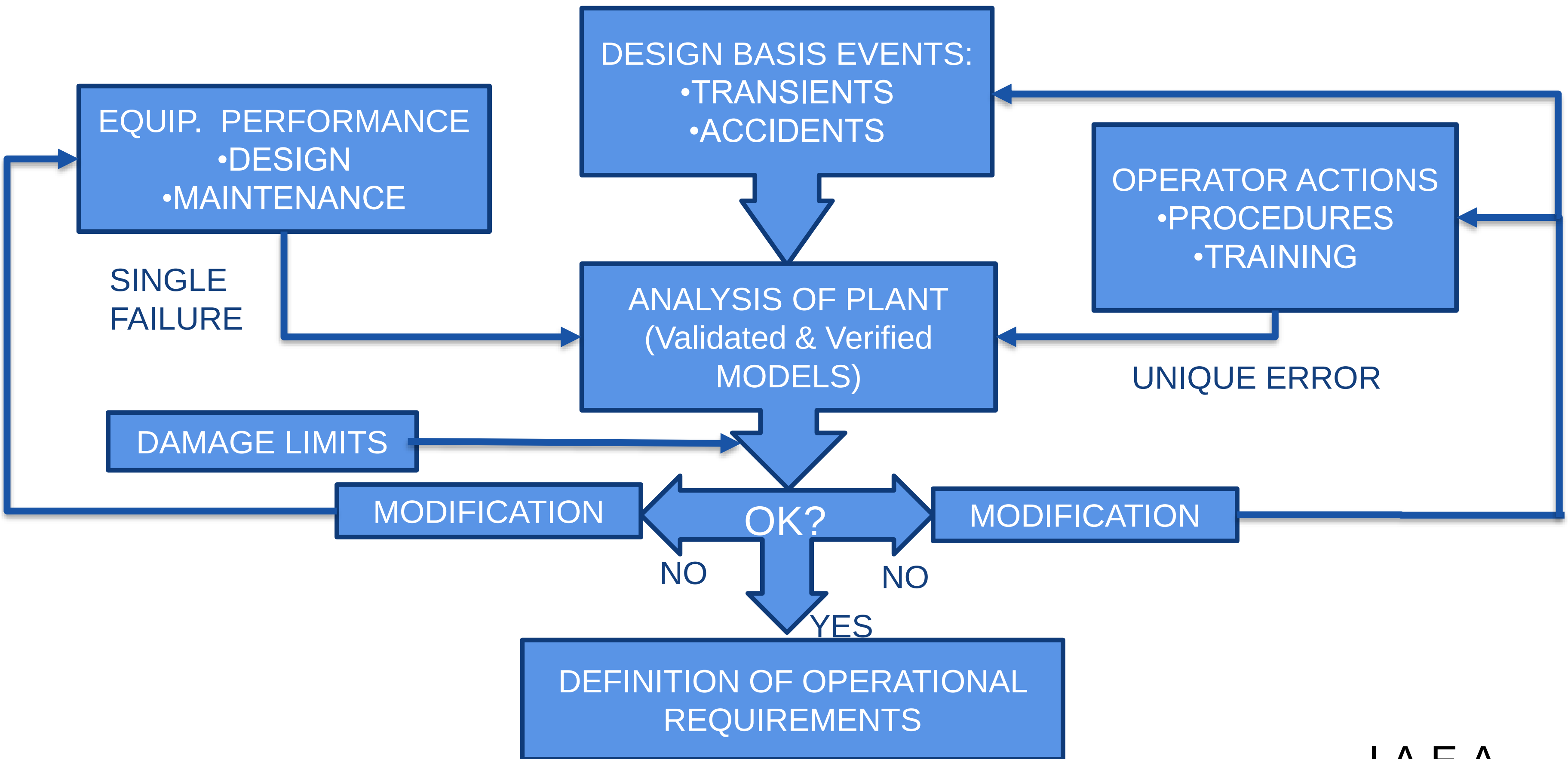
NUREG-0800, Chapter 15, Transient and Accident Analysis (contd)

12. Control Rod Misoperation (System Malfunction or Operator Error)
13. Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
14. Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)
15. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
16. Spectrum of Rod Ejection Accidents (PWR)
17. Radiological Consequences of a Control Rod Ejection Accident (PWR)
18. Spectrum of Rod Drop Accidents (BWR)
19. Radiological Consequences of Control Rod Drop Accident (BWR)
20. Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
21. Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve

SH2705 Simulation Course

DSA input guidelines – Example

THE DETERMINISTIC APPROACH
SAFETY ANALYSIS -DETERMINISTIC



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DSA APPLICATIONS

- **Design Applications**
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  - Operating organization, to confirm the design

DSA must be parallel to the design process, with iteration between them.

- **Licensing Applications**
  - Calculations for Final Safety Analysis Report (FSAR)
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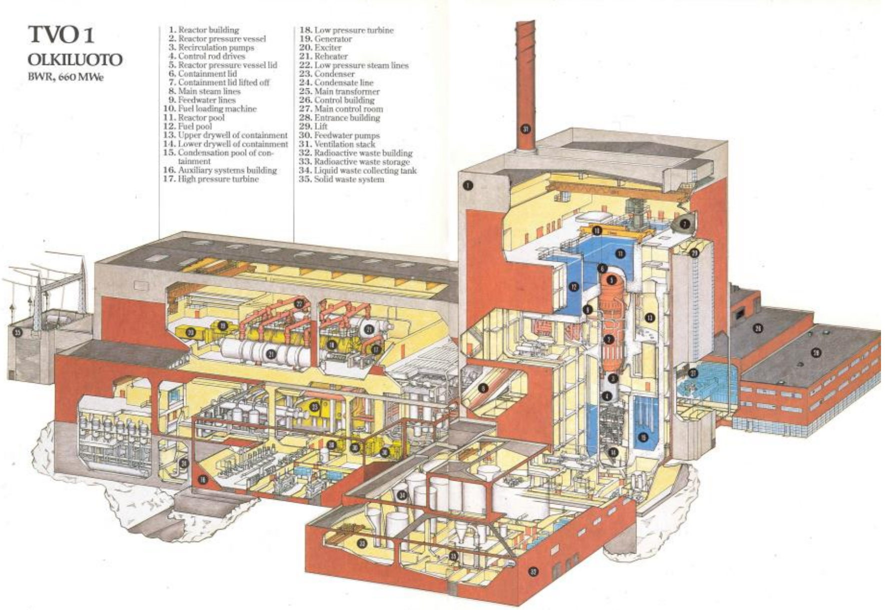
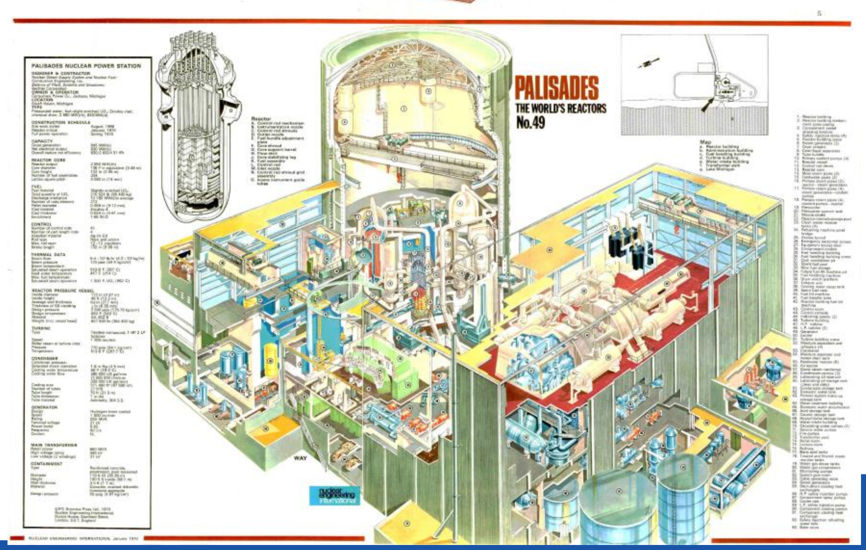
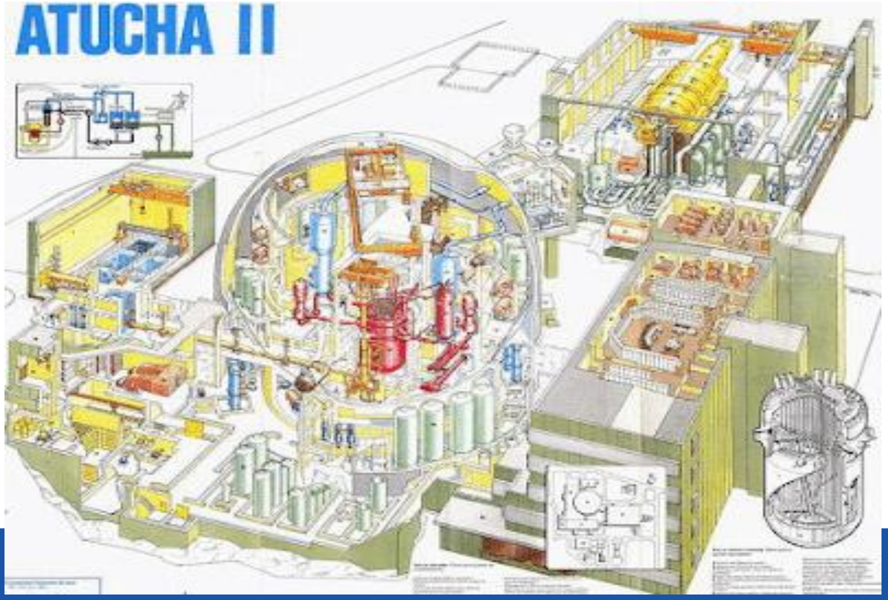
- **Regulatory Applications**
  - Audit calculations
  - Evaluation of emergency operating procedures
  - Review of significant events and incidents
  - Evaluation of emergency operating procedures
  - Unresolved Safety Issues Evaluation

COMPLEXITY OF THE SYSTEM







Methodology of DSA

A complex analysis of a power plant requires often a set of inputs and model. These models are called Evaluation models.

- EVALUATION MODELS of a nuclear power plant is developed in order to perform Deterministic Safety Analysis.
- An Evaluation Model (EM) is the calculation framework for evaluating the behaviour of a plant during a postulated transient or Design Basis Accident (DBA)

Methodology of DSA

An EM may include one or more computer codes, other calculational aids (analytical tools, calculational procedures), special models, and all other information necessary for application of the calculational framework to a specific event, such as:

- Procedures for treating the input and output information.
- Specification of those portions of the analysis not included in the computer codes for which alternative approaches are used.

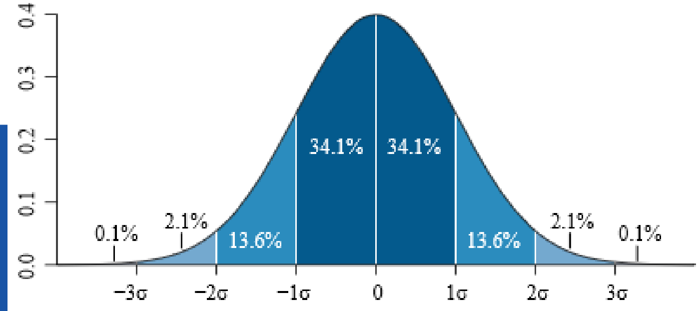
Input Data Preparation

- The construction of the input data to perform Safety Analysis must be subject of an adequate Quality Assurance program.
- All sources of data must be referenced and documented.
- The whole process must be recorded and archived to allow independent checking.

Input Data Preparation (conservative)

- Input data to a conservative DSA:
  - Conservative initial values of the plant variables.
  - Conservative boundary conditions through the transient (e.g., systems and operator performances).
  - Conservative physical models in the code.
- Different degrees of conservatism:
  - Most variables are set to "high" values (taking account of their probability distribution functions). E.g.,: average value plus "two sigma", or 95.4 percentile…
  - Some variables can be set to extremely high values. E.g.: values established in Appendix K to 10 CFR 50, for LOCA analysis.

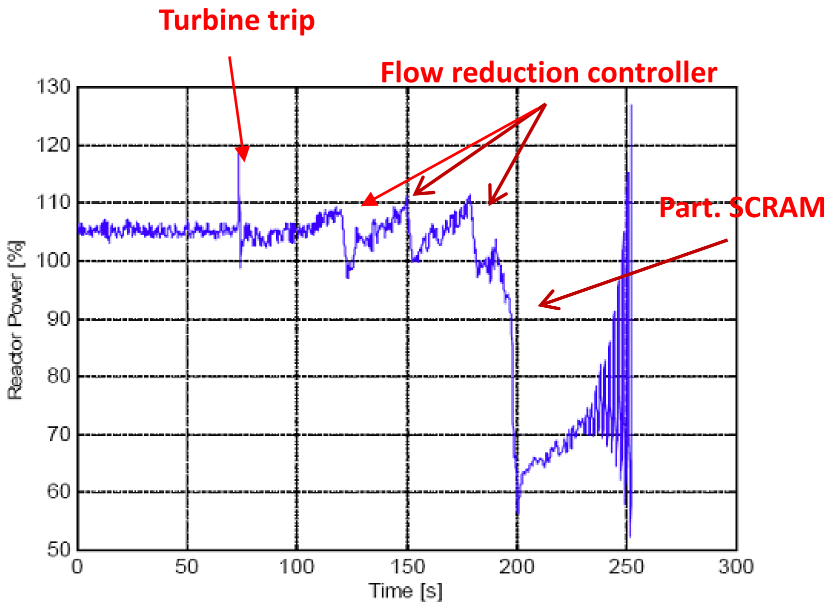
Normal distribution where each band has a width of 1 standard deviation



Conservative input Data Preparation (DB)

Conservative assumptions made for DB analysis:

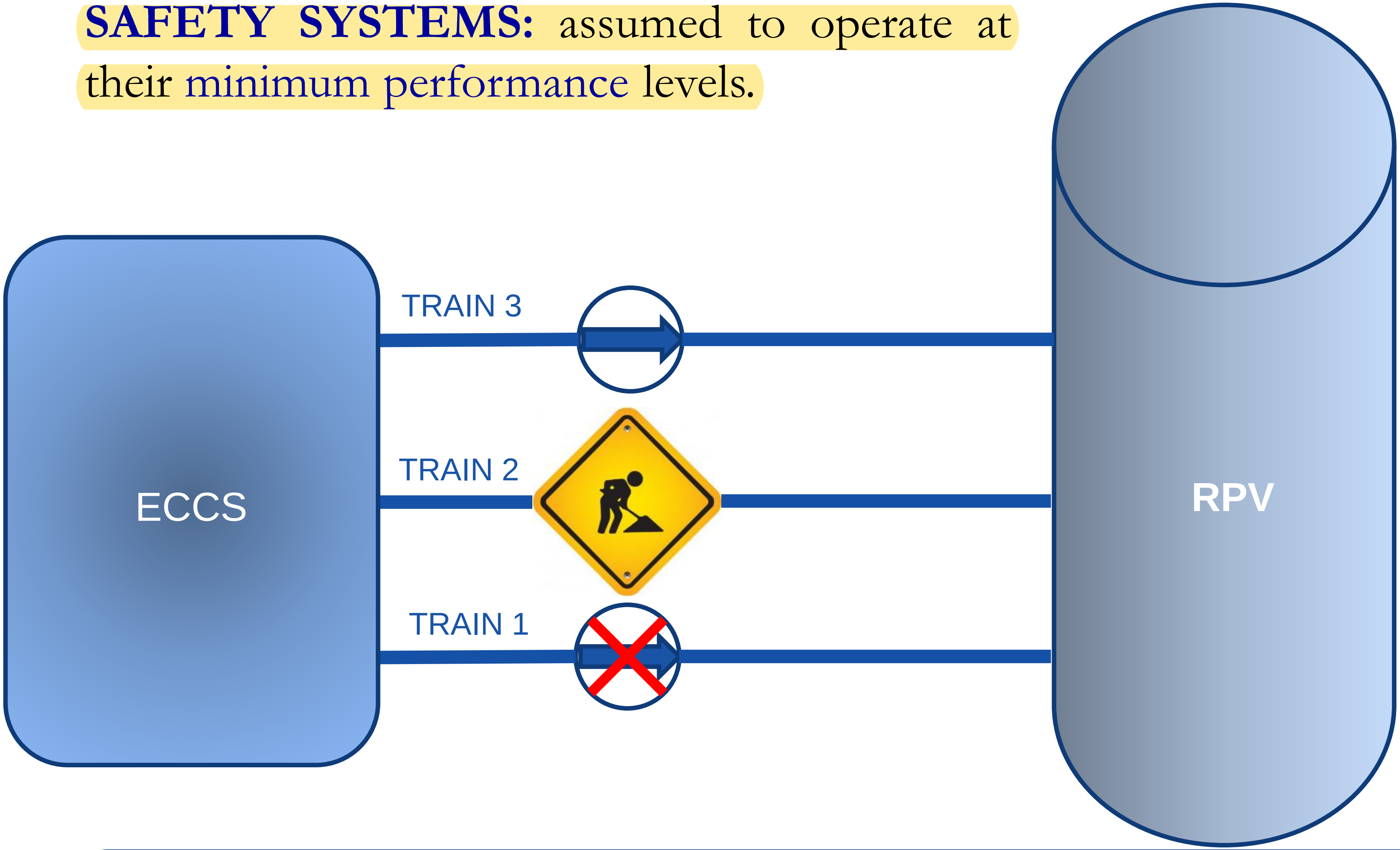
- **TIME:** Initiating event occurs at an unfavourable time.
- **BEHAIVOUR OF THE CONTROL SYSTEM:** It operates only if their functioning would aggravate the effects of the initiating event. No credit for mitigation.



- **SAFETY SYSTEMS:** All plant systems and equipment not designed as safety class, should be assumed to fail causing the most severe effects for the PIE **<u>if their operation does not have a aggravating the effects of the PIE</u>**
- **SINGLE FAILURE:** Worst single failure assumed in the operation of the safety groups required for the initiating event. For redundant systems it is often assumed running of minimum number of trains.

Conservative input Data Preparation (DB)

SAFETY SYSTEMS: assumed to operate at their minimum performance levels.



Conservative input Data Preparation (DB)

- **AVAILABELITY of SYTEMS**: Structures, systems or components that do not have PROVEN full operability during the accident should be assumed UNAVAILABLE.
- **HUMAN BEHAIVOUR:** Actions of the plant staff to prevent or mitigate the accident are only modelled when it is shown that there is sufficient time to perform them, and that procedures and training are adequate or have a negative effect on the outcome of the accident.

Conservative input Data Preparation (DB) - chain of the events

DB analysis should include any failures which could occur as a consequence of the IE, including:

- If the IE is part of an electrical distribution system, all the equipment powered from that part will be unavailable.
- If the IE is an "energetic event" (failure of pressurised system), failure of the equipment that could be affected.
- Fire, floods or external events: failure of the equipment neither designed nor protected against the effects.

Conservative input Data Preparation (AOO)

AOOs: operational processes deviating from normal operation that have the potential to challenge the safety of the reactor. According to design provisions, AOOs do not cause any significant damage to items important to safety, nor lead to accident conditions.

- Loss of feedwater flow,
- turbine trip,
- loss of off-site power

Conservative input Data Preparation (AOO)

- For AOOs, the deterministic SA should include many of the conservative assumption of the DBA analysis, especially those related to the systems for maintaining critical safety functions.

- *But it's not necessary to assume unavailability of all non-safety systems and equipment or no credit to mitigation by control systems, unless the PIE impose it.*

Input Data Preparation (best-estimate)

- Input data to a best-estimate DSA:
  - Plant and model parameters and variables that will participate in the uncertainty analysis: set to realistic values. But the input is not a single value, rather a probability density function (pdf).
  - Variables and parameters that will not intervene in the uncertainty analysis will be set to conservative values.
- Both conservative and BE analysis need to know the probability distribution of the uncertain variables and parameters. But the knowledge must be finer for the BE approach, coarser for the conservative one.

Methodology of DSA

A Methodology includes several types of calculations using different tools in evaluation models :

- Thermo-hydraulic, simulating the behavior of the coolant in the plant.
- Reactor dynamic, simulating the fission processes in the reactor core.
- Structural, simulating the behavior of structures against the loads, stresses,…
- Radiological.

All these types of calculations are "deterministic"(no probability involved, conservative assumptions not for best-estimate methodologies).

Methodology of DSA

- Thermo-hydraulic calculations:
  - Performed with fluid-dynamic based codes
  - Simulating coolant behaviour in primary and secondary systems, the containment…
- Reactor dynamic calculations:
  - Performed with reactor dynamic codes
  - Simulating fission process in the core
- Structural calculations
- Radiological calculations

Methodology of DSA

EM DEVELOPMENT AND ASSESSMENT PROCESS:

EMDAP basic principles according to REGULATORY GUIDE 1.203, "Transient and Accident Analysis Methods of USNRC:

- Determine requirements for the EM: i.e., Mathematical models, components, phenomena, physical processes, etc., needed to evaluate the event behavior relative to adequate figures-of-merit (FOM).
- Develop an assessment base consistent with the above-mentioned requirements: experimental data. Sometimes performance of new experiments is required.
- Develop the EM: the calculational tools are selected or developed. For a particular plant and event, it is necessary to select proper code options, boundary conditions and the temporal and spatial relationship among the component devices.

Methodology of DSA

- Assess the adequacy of the EM: by comparing requirements and capabilities. Some of this assessment is best made during the early phase of code development, to minimize posterior corrective actions. It is important to assure that the calculational devices are used within the range of their assessment.
- Follow an appropriate Quality Assurance protocol during the EMDAP.
- Provide comprehensive, accurate, up-to-date documentation.

Requirements of DSA - General

To achieve the appropriate level of confidence, the safety analysis shall:

- Be performed by qualified analysts in accordance with an approved QA process;
- Apply a systematic analysis method;
- Use verified data;
- Use justified assumptions;
- Use verified and validated models and computer codes;
- Build in a degree of conservatism; and
- Be subjected to a review process

Requirements of DSA - Analysis Method

The analysis method shall include the following elements:

- Identifying the scenarios to be analysed as required to attain the analysis objectives;
- Identifying the applicable acceptance criteria, safety requirements, and limits;
- Identifying the important phenomena of the analysed event;
- Selecting the computational methods or computer, models, and correlations that have been validated for the intended codes applications;

Requirements of DSA - Analysis Method (contd.)

- Defining boundary and initial conditions;
- Conducting calculations, including sensitivity cases, to predict the event transient, starting from the initial steady state up to the pre-defined end-state;
- Accounting for uncertainties in the analysis data and models;
- Verifying calculation results for physical and logical consistency;
- Processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria.

Requirements of DSA - Analysis Data

The safety analysis shall be based on complete and accurate design and operational information.

The boundary and initial conditions used as the analysis input data shall:

- Reflect accurately the NPP configuration;
- Account for the effects of aging of systems, structures and components;
- Account for various permissible operating modes;
- Be supported by experimental data, where operational data is not available.

Significant uncertainties in analysis data, including those associated with nuclear power plant performance, operational measurements, and modelling parameters, shall be identified.

Requirements of DSA - Analysis Assumptions

Assumptions made to simplify the analysis, as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified.

The analysis of AOO and DBA shall:

- Apply the single-failure criterion to all safety systems and their support systems;
- Account for consequential failures that may occur as a result of the initiating event;
- Credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analysed accident;

Requirements of DSA - Analysis Assumptions (contd.)

- Account for the possibility of the equipment being taken out of service for maintenance; and
- Credit operator actions only when there are
  - unambiguous indications of the need for such actions,
  - adequate procedures and sufficient time to perform the required actions,
  - environmental conditions that do not prohibit such actions.

For the analysis of BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions which reflect the likely plant configuration, and the expected response of plant systems and operators in the analysed accident.

Requirements of DSA - Computer Codes

- Computer codes used in the safety analysis shall be developed, validated, and used in accordance with a quality assurance program that meets the requirements

BEPU - DSA example

Requirements for LOCA Analysis

Input Data Preparation (best-estimate)

- Input data to a best-estimate DSA:
  - Identify the plant and model parameters and variables that will participate in the uncertainty analysis:
    - set to realistic values. (The input is not a single value, rather a probability density function (pdf).
  - Variables and parameters that will not intervene in the uncertainty analysis (UA) will be set to conservative values.
- In BE analysis, we need to know the probability distribution of the uncertain variables and parameters in detail.

LWR LOCA BE Analysis Requirements

The goal is to obtain a best estimate prediction of the vital system variables such as:

- Pressure in the reactor vessel
- Pressure in the containment,
- Maximum fuel temperature
- Maximum cladding temperature

The first step in any LOCA analysis involves establishing the initial T/H conditions.

LWR LOCA BE Analysis Requirements

The BE modeling requirements for initial steady-state calculations in both BWRs and PWRs are:

- Complete geometrical simulation of the reactor system with realistic modeling of all the important flow paths, material masses, and system components.
- Steady-state or unperturbed transient modeling of mass, momentum and energy distribution for the coolant, including flow velocities and temperature, for single-phase and two-phase flow in all reactor components.
- Single-phase pressure drop in pipes, bends, fuel bundles, area changes, and in all special reactor components.
- Single-phase heat convection for water and steam and boiling heat transfer.

LWR LOCA BE Analysis Requirements

- Two-phase density (or void fraction) and velocity distribution in the boiling channels, including subcooled boiling voids.
- Two-phase pressure drop in boiling channels and in other reactor components such as pipes, separators, etc.
- Steady-state heat conduction and temperature distribution in solids.
- Heat conduction in the gap between fuel and cladding.
- Realistic modeling of the characteristics of specific system components such as pumps, steam separators, jet pumps, etc.
- Reliable approximation of the thermodynamic and transport properties of the reactor materials such as fuel, cladding, vessel, piping and the coolant (liquid and vapor).

BWR LOCA BE Modelling Requirements

- Time dependent distributions of mass, momentum and energy for the coolant material in all system components.
- Time dependent velocities and local flow densities in all one-dimensional and Multidimensional components including the following items:
    - One-dimensional flow through fuel channels (BWR), pipes, valves, pumps, etc.
    - Multidimensional flow through downcomer, lower plenum, upper plenum, bypass, and steam dome.
    - Flow through pumps in forward and reverse directions, with proper pressure loss coefficients.
    - Flow through steam separators and reverse directions, with proper loss coefficients. And dryers in forward
- Critical flow calculation at the break and at any internal junction that may experience very steep pressure gradients.

BWR LOCA BE Modelling Requirements

- Interfacial exchanges of mass, momentum and energy between vapor and liquid, including the effects of various two-phase flow patterns.
- Circulation pump characteristics.
- Safety and relief valve component modeling capability.
- Transient heat conduction and temperature distribution for fuel cladding, and other solid structures.
- Single-phase heat transfer to vapor and liquid in different flow geometries.
- Gap heat conductance.
- Boiling heat transfer including nucleate, transition, and film boiling at all pressures.

BWR LOCA BE Modelling Requirements

- Critical heat flux (or dryout) prediction relevant to BWR fuel geometries.
- Non-equilibrium temperature distribution between vapor and liquid with individual heat transfer between either phase and the channel walls.
- Liquid entrainment in vapor and de-entrainment.
- Countercurrent flow and CCFL effects at the side entry orifice and at the upper tie plate geometries.
- Radiation heat transfer between any fuel rod and other rods, surrounding steam and droplets, and the channel walls.
- Minimum film boiling temperature and rewet heat transfer.

BWR LOCA BE Modelling Requirements

- Power calculation with time-dependent neutron kinetics model, including at lease six groups of delayed neutrons.
- Decay heat of fission products with contribution from transuranic elements.
- Realistic trips with appropriate delay actions and parameter dependencies.
- Control system models with universal simulation capabilities.
- Containment simulation capability, including dry and wet wells, heat transfer, pool boiling, and condensation on the walls.