Joint KINS-IAEA-ANNuR/ANSN/FNRBA
BPTC Course on Nuclear Safety,
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Safety Analysis:

Deterministic safety analysis

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Work Experiences

38 years experiences in Safety Analysis and Engineering Evaluation

- **1984 - 1990: Korea Atomic Energy Research Institute (KAERI)**
  - Project manager for inspection of Kori units 3 and 4
  - Assignee for US NRC

- **1990 - 2009 Korea Institute of Nuclear Safety (KINS)**
  - Head of thermal-hydraulics R&D department
  - Safety Analysis for KSNP, APR-1400, RELAP/CANDU development

- **1993 - 1996 University of Ottawa (UO)**
  - Research Associate on thermal hydraulic R&D

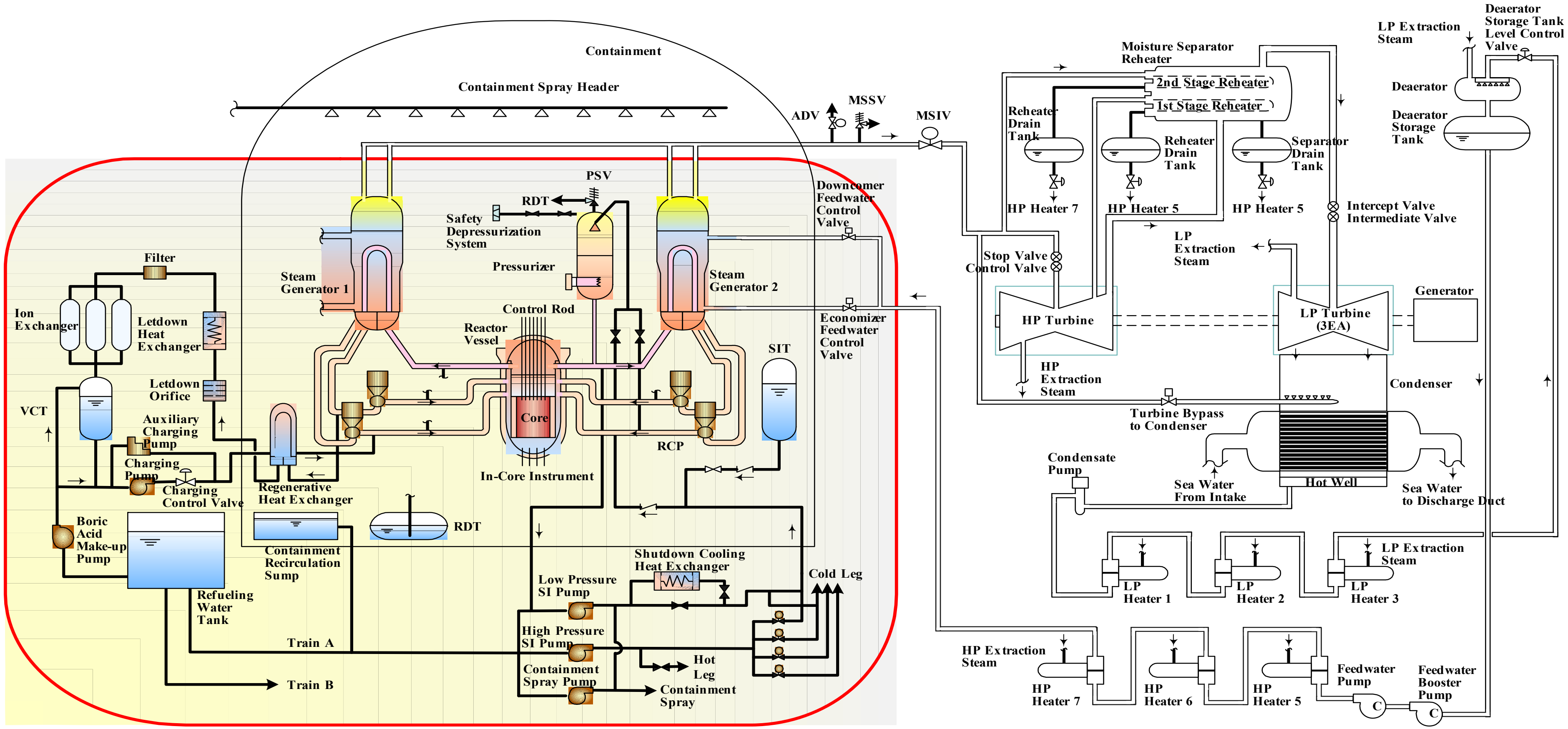
- **2009 - 2016: Safety officer of NSNI/SAS (IAEA)**
  - Safety Officer as a team leader of severe accident analysis and management
  - Safety standard development and Generic Reactor Safety Review Service

- **2016 - at present: Korea Institute of Nuclear Safety (KINS)**
  - Ex-Director of Safety R&D division and currently *Professor of INSS*
  - Vice President, *IAEA 8th CNS and 8th-9th Joint Review Meeting*
  - Senior Advisor, *KINS Technical support services to the NRRC of the KSA*

Presentation Outline

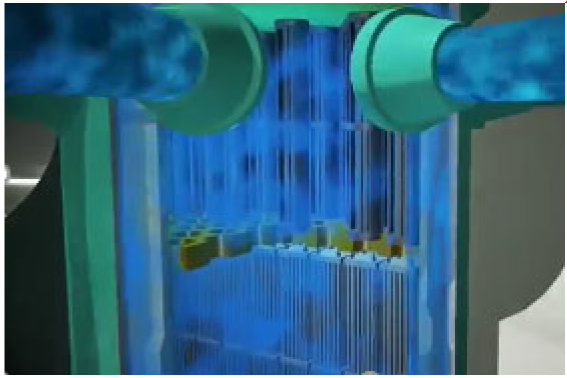
1. Needs for Safety Analysis

Design of Nuclear Power Plant



Design and Normal Operation of a NPP

Transient in the Reactor



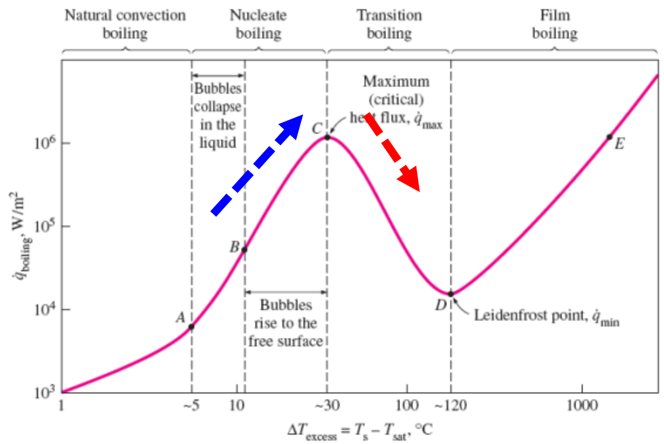


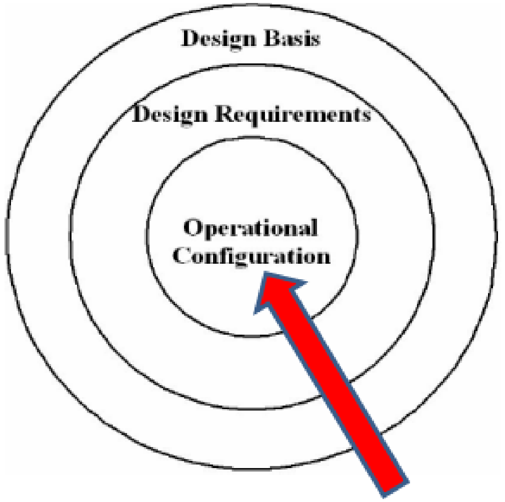
Figure 4 typical boiling curves for water at 1 atm pressure.

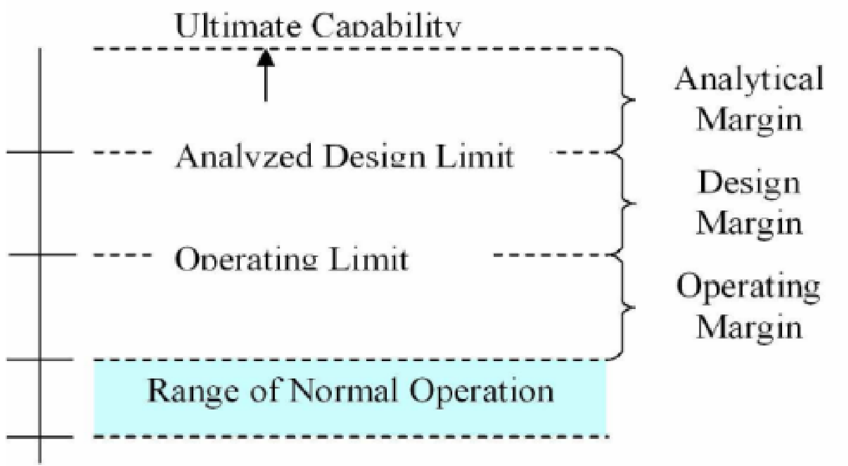
CHFR (or DNBR) = [CHF]calculated / [Actual Heat Flux]estimated

“How safe is safe enough?”

- **Provide a measure of sufficiency/adequacy of safety provisions** embedded **in the design** of a nuclear installation and its operational process

- For instance, the **limiting values** (e.g. CHFR/DNBR) likely safety limits are determined in the design or established for plant operation which shall **not be exceeded during normal operations including anticipated operational occurrences**.



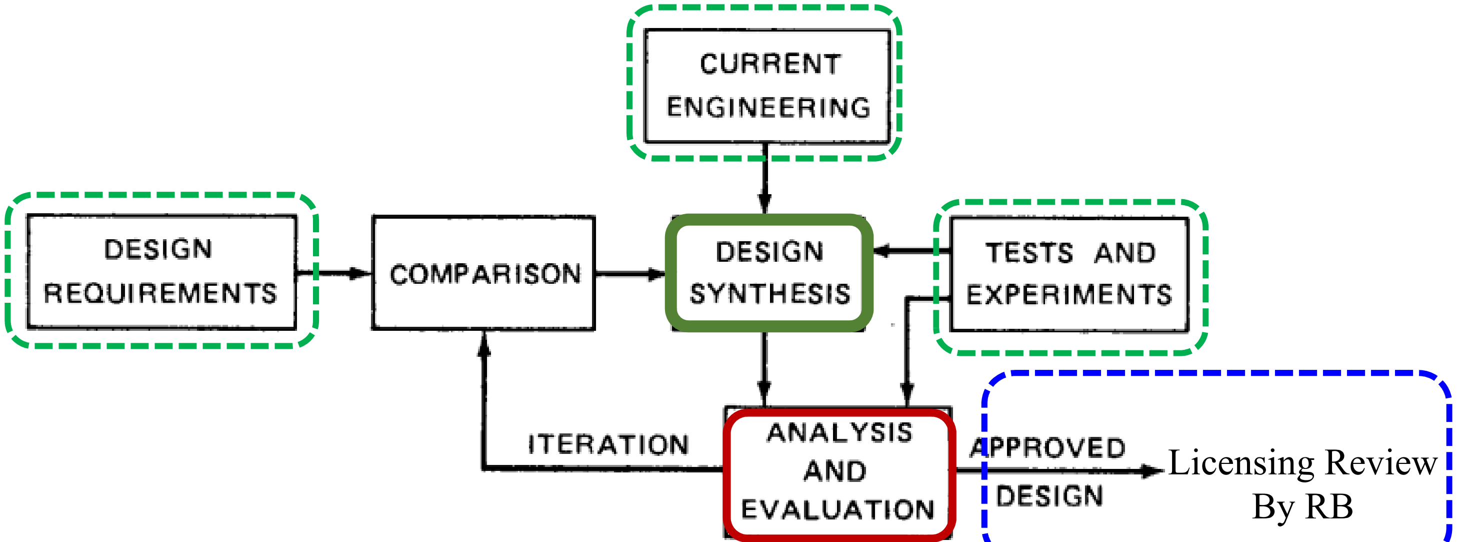


1. Needs for Safety Analysis

TID-26241 Nuclear Power Plant Design Analysis

- Two major importance to the designer must be considered **in the safety analysis** as well as the detailed **effects of various postulated accidents**.
  - **The power distribution** may vary with time as the fuel is depleted and also as the result of different fuel reloading strategies.
  - The **thermal consequences** of start-up, shutdown, and inadvertent operating situations

Table: Design feedback and iteration process

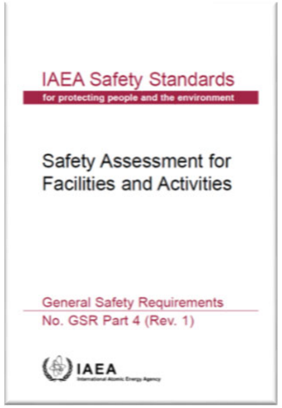


1. Needs for Safety Analysis (cont'd)

- **In the design process, safety analyses** are analytical studies aimed at demonstrating;
  - to confirm **the safety** of Nuclear Power Plants (NPP) through an quantitative analysis **for the postulated transients and accident conditions.**
  - to confirm **adequacy of limiting conditions for operation (LCO), limiting safety system settings**, and **design specifications for safety-related components** and **systems to protect public health and safety**.
  - to confirm **performance of reactor protection system(RPS), engineered safety features(ESF)**, and **adequacy of emergency operating procedure(EOP)**.

1. Needs for Safety Analysis (cont'd)

- **The results of the safety analysis** ensure that the plant designed to meet all the design acceptance criteria at commissioning and throughout the life of the plant.
- Therefore, safety analysis is **an essential element of a plant design as** well as **of the licensing process**.
- Such analyses are **an integral part** of any licensing process and are part of **the Final Safety Analysis Report (FSAR)** for every nuclear power plant.



Safety Assessment for Facilities

Safety Assessment

Safety Analysis

Two complementary methods

Deterministic Safety Analysis (DSA)

Predicts the response to postulated events with predetermined assumptions; checks fulfilment of acceptance criteria

Probabilistic Safety Analysis (PSA)

Combines the likelihood of initiating events, potential scenarios and their consequences into estimation of CDF, source term or overall risk

Safety Approach

- **Defence in depth**
- **Safety Margin**
- **Multiple barriers**
- **Engineering Factors**

Safety Features

- **Site characteristics**
- **Engineering aspects**
- **Safety function**
- **Human factors**
- **Long-term safety**
- **Radiation protection**

1. Needs for Safety Analysis (cont'd)

- Safety analysis support **the safe operation of the plant** by serving as an important tool in developing and **confirming;**
  - the plant's protection and operating specifications and limits (technical specifications, control system set points, control parameters),
  - Operability and integrity of system, structure and components (SSCs),
  - Maintenance and inspection requirements, and
  - Operating procedures,
    - Normal and abnormal operating procedures,
    - Emergency operating procedures (EOPs), and
    - Severe accident management guidelines (SAMGs).

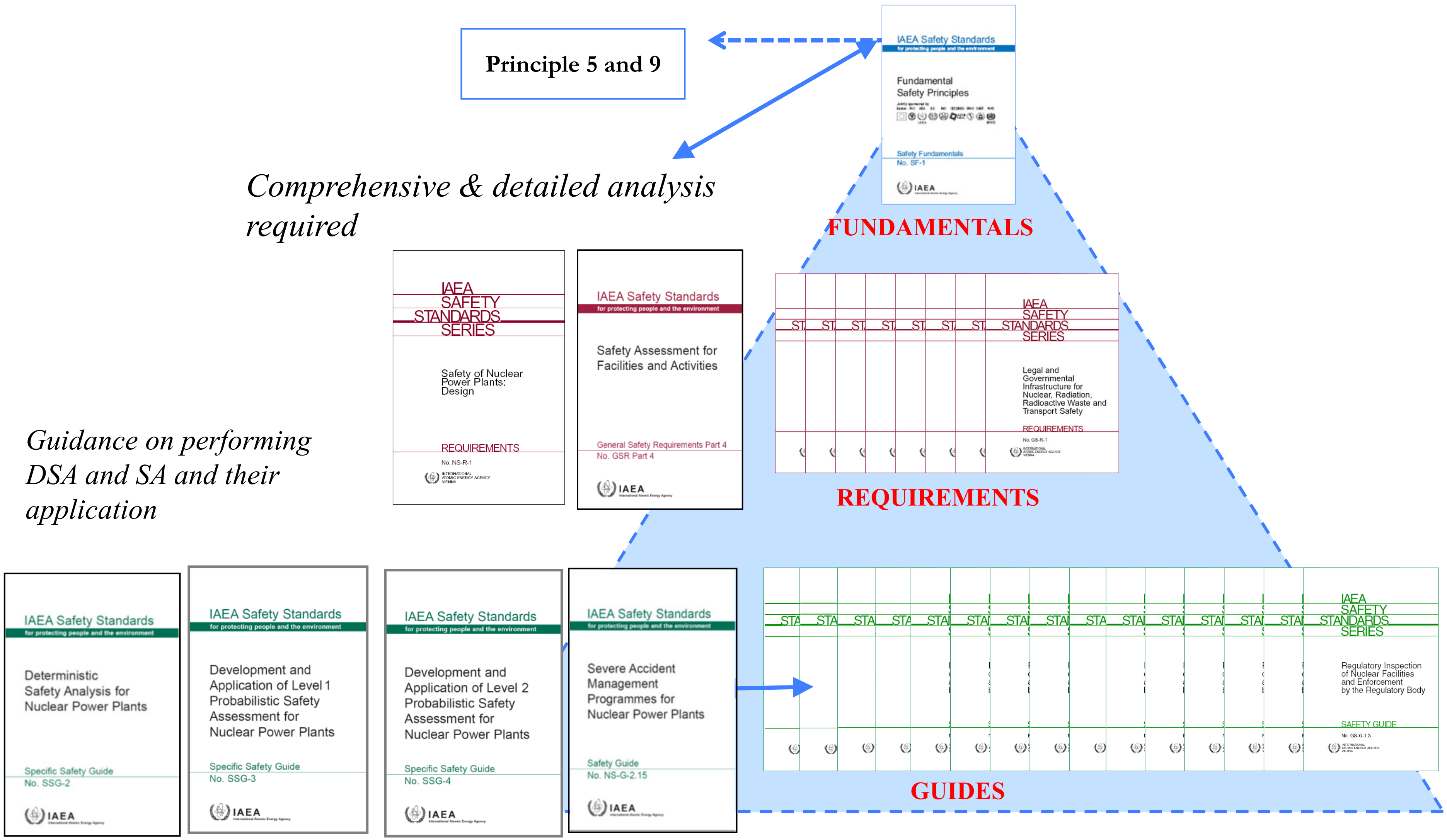
2. Introduction of Safety Analysis: DSA

1) Purpose
2) Safety Requirements
3) Analysis in the plant state

(2-1) Purpose of safety analyses

- Safety analyses are undertaken as a means of evaluating compliance with safety principles and safety requirements for all nuclear facilities for the protection of workers, the public and the environment from harmful effects of ionizing radiation.
- They are to be carried out and documented by the organization responsible for operating the facility, are to be independently verified and are to be submitted to the regulatory body as part of the licensing or authorization process.

(2-2) Safety Requirements



IAEA SAFETY STANDARDS

(2-2) Safety Requirements (cont'd)

Safety Standards

- Safety Assessment for Facilities and Activities, GSR Part 4
- Safety of Nuclear Power Plants: Design, SSR-2/1
- Deterministic Safety Analysis for Nuclear Power Plants, SSG-2
- Development and Application of Level 1 PSA for NPPs, SSG-3
- Development and Application of Level 2 PSA for NPPs, SSG-4

Safety Report Series (SRS)

- SRS No. 23 Accident Analysis for NPPs
- SRS No. 29 Accident Analysis for NPPs with Pressurized Heavy Water Reactors (PHWR)
- SRS No. 30 Accident Analysis for NPPs with Pressurized Water Reactors (PWR)
- SRS No. 43 Accident Analysis for NPPs with Graphite Moderated Boiling Water RBMK Reactors (RMBK)
- SRS No. 52 Best Estimate Safety Analysis for NPPs: Uncertainty Evaluation (BEPU)

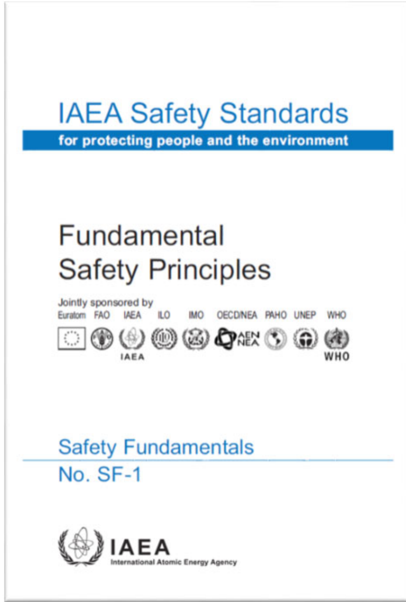
(2-2) Safety Requirements (cont'd)

TECDOCs

- IAEA TECDOC - 1351 Incorporation of Advanced Accident Analysis Methodology into Safety Analysis Reports
- IAEA TECDOC - 1352 Application of Simulation Techniques for Accident Management Training in NPPs
- IAEA TECDOC - 1379 Use of Computational Fluid Dynamics Codes for Safety Analysis of Nuclear Reactor Systems
- IAEA TECDOC - 1539 Use and Development of Coupled Computer Codes for the Analysis of Accidents at NPPs
- IAEA TECDOC - 1550 Deterministic Analysis of Operational Events in NPPs
- IAEA TECDOC - 1578 Computational Analysis of the Behaviour of Nuclear Fuel Under Steady State, Transient and Accident Conditions
- IAEA TECDOC - 1332 Safety Margins of Operating Reactors; Analysis of Uncertainties and Implications for Decision Making
- IAEA TECDOC - 1418 Implications of Power Uprates on Safety Margins of NPPs

SF1 Fundamental Safety Principles

- The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation.
- To ensure that facilities are operated and activities conducted so as to achieve the highest standards of safety that can reasonably be achieved, measures have to be taken:
  - To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;
  - To mitigate the consequences of such events if they were to occur;
  - To control the radiation exposure of people and the release of radioactive material to the environment.

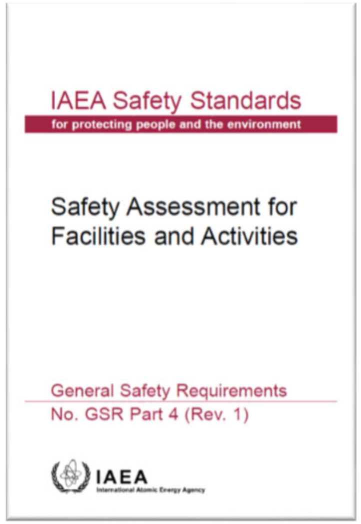


GSR Part 4 Safety Assessment for Facilities and Activities

Requirement 3: Responsibility for the safety assessment

The responsibility for carrying out the safety assessment shall rest with the responsible legal person or organization responsible for the facility or activity.

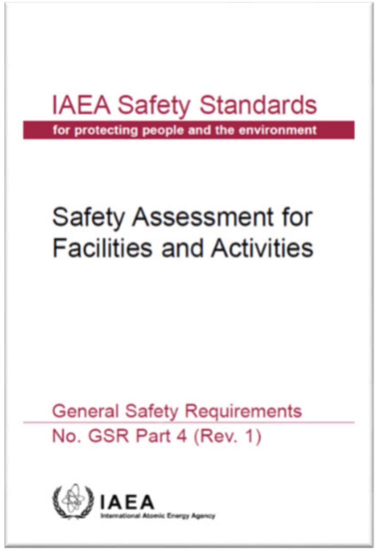
- Generally, the operating organization shall be responsible for the safety assessment.



(2-2) Safety Requirements (cont'd)

GSR Part 4 Requirement 4: Purpose of the safety assessment

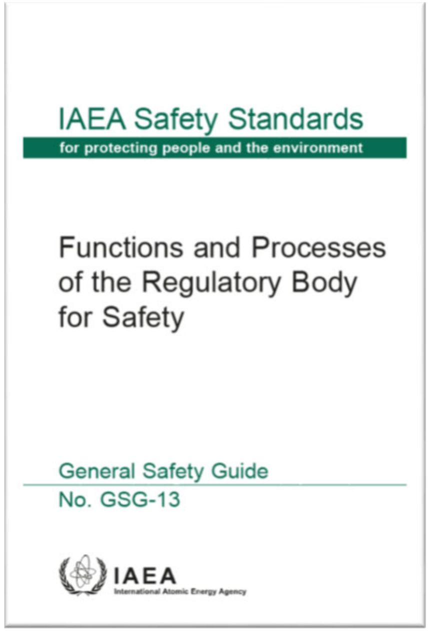
- The primary purposes of the safety assessment shall be:
  - to determine whether an adequate level of safety has been achieved for a facility or activity and
  - to determine whether the basic safety objectives and safety criteria established by the designer and the operating organization in compliance with the requirements for protection and safety



GSG-13 Functions and Processes of the Regulatory Body for Safety: recommendations

Verification of the safety analysis

- Examination of the submissions from the authorized party on its management arrangements and operational procedures and verification of the safety analysis……. In carrying out the review and assessment, the regulatory body may find it useful to perform its own analyses or research.
- The regulatory body should determine whether the authorized party has defined criteria that meet the safety objectives and requirements relating to:
  - Engineering design;
  - Operational and managerial aspects;
  - Normal operation, anticipated operational occurrences and accident conditions.



(2-2) Safety Requirements (cont'd)

SSR 2/1 Requirement 42: Safety analysis of the plant design

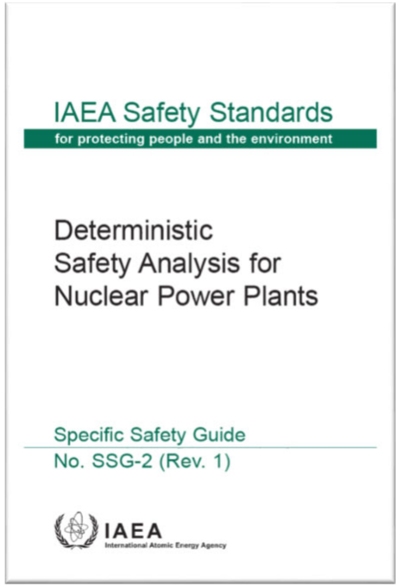
A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to demonstrate:

- the design basis for the items important to safety
- the overall plant design is capable of complying with authorized limits for radioactive releases and with the dose limits in all operational states
- the overall plant design is capable of meeting acceptable limits for accident conditions

(2-2) Safety Requirements (cont'd)

IAEA SSG-2 Deterministic Safety Analysis for NPPs

- The objective is to provide recommendations and guidance on performing deterministic safety analysis for designers, operators, regulators and technical support organizations. It also provides recommendations on the use of deterministic safety analysis for:
  - a) Assessing compliance with regulatory requirements;
  - b) Identifying possible enhancements of safety and reliability;



(2-3) Analysis in the Plant Sates

- **Operational states** include normal operation as well as anticipated operational occurrences(AOOs).
- **Accident conditions** include accidents that are within the design basis and design extension conditions.
- **Design extension conditions** include severe accident conditions, which are characterized as states with significant core degradation.

Table : Plant states.

Operational states		Accident conditions	
Normal operation	Anticipated operational occurrences	Design basis accidents	Design extension conditions

Grouping by Frequency of Occurrences (IAEA)

Occurrence (per RY)	Characteristics		Terminology	Acceptance Criteria
$10^{-2} \sim 1$ (Expected during plant life)	Expected	Anticipated operational occurrences	Anticipated transients; transients; frequent faults; incident of moderate freq.; upset/ abnormal cond.	No additional fuel damage
$10^{-4} \sim 10^{-2}$ (Chance greater than 1% over the plant life)	Possible	Design Basis Accidents (DBAs)	Infrequent incidents; infrequent faults; limiting faults; emergency conditions	No radiological impact at all or no radiological impact outside the exclusion area
$10^{-6} \sim 10^{-4}$ (Chance less than 1% over the plant life)	Unlikely	Beyond Design Basis Accidents (BDBAs)	Faulted conditions	Radiological consequences outside exclusion area within limits
$< 10^{-6}$ (Very unlikely to occur during plant life)	Remote	Severe Accidents	Faulted conditions	Emergency response needed

Grouping by Frequency of Occurrences (USA)

<table>
<tr><th colspan="3">ANS</th><th colspan="2">USNRC</th></tr>
<tr><th>Freq. per RY</th><th>ANSI/ANS-51.1 (1983)</th><th>ANS N18.2 (1973)</th><th>RG 1.70 (Rev. 2)</th><th>10 CFR</th></tr>
<tr><td>Normal Operation</td><td>Plant Condition 1 (PC-1)</td><td>Condition I</td><td>Normal operation & operational transients</td><td>Normal operation</td></tr>
<tr><td>$> 10^{-1}$</td><td>Plant Condition 2 (PC-2)</td><td>Condition II</td><td>Incidents of moderate frequency</td><td rowspan="2">Anticipated operational occurrences (AOOs)</td></tr>
<tr><td>$10^{-2} \sim 10^{-1}$</td><td>Plant Condition 3 (PC-3)</td><td>Condition III</td><td>Infrequent incidents</td></tr>
<tr><td>$10^{-4} \sim 10^{-2}$</td><td>Plant Condition 4 (PC-4)</td><td rowspan="3">Condition IV</td><td rowspan="3">Limiting faults</td><td rowspan="3">Accidents</td></tr>
<tr><td>$10^{-6} \sim 10^{-4}$</td><td>Plant Condition 5 (PC-5)</td></tr>
<tr><td>$< 10^{-6}$</td><td>Not considered</td></tr>
</table>

(2-3) Analysis in the Plant Sates (cont'd)

Normal operation is defined as operation within specified operational limits and conditions.

- The analysis is applied to normal operation with the aim of showing that normal operation can be carried out safety including;
  - acceptable doses to workers and the public
  - acceptable planned releases of radioactive material.
- The analysis should demonstrate that plant parameters remain within acceptable limits.

(2-3) Analysis in the Plant Sates (cont'd)

An **anticipated operational occurrence (AOO)** is an operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility.

- Because of appropriate design provisions, it does not cause any significant damage to items that are important to safety or lead to accident conditions.
  - Do not lead to any significant fuel damage, therefore, no offsite consequences.
- The analysis should demonstrate that plant parameters remain within acceptable limits.

(2-3) Analysis in the Plant Sates (cont'd)

Design basis accidents (DBAs) are accident conditions against which a facility is designed according to established design criteria.

- DBAs are not expected to occur in the life of the plant, but are of sufficiently high probability that they are reasonably considered as tests of the safety design of the plant.
- The analysis should demonstrate that the damage to the fuel and the release of radioactive material are kept within authorized limits.

(2-3) Analysis in the Plant Sates (cont'd)

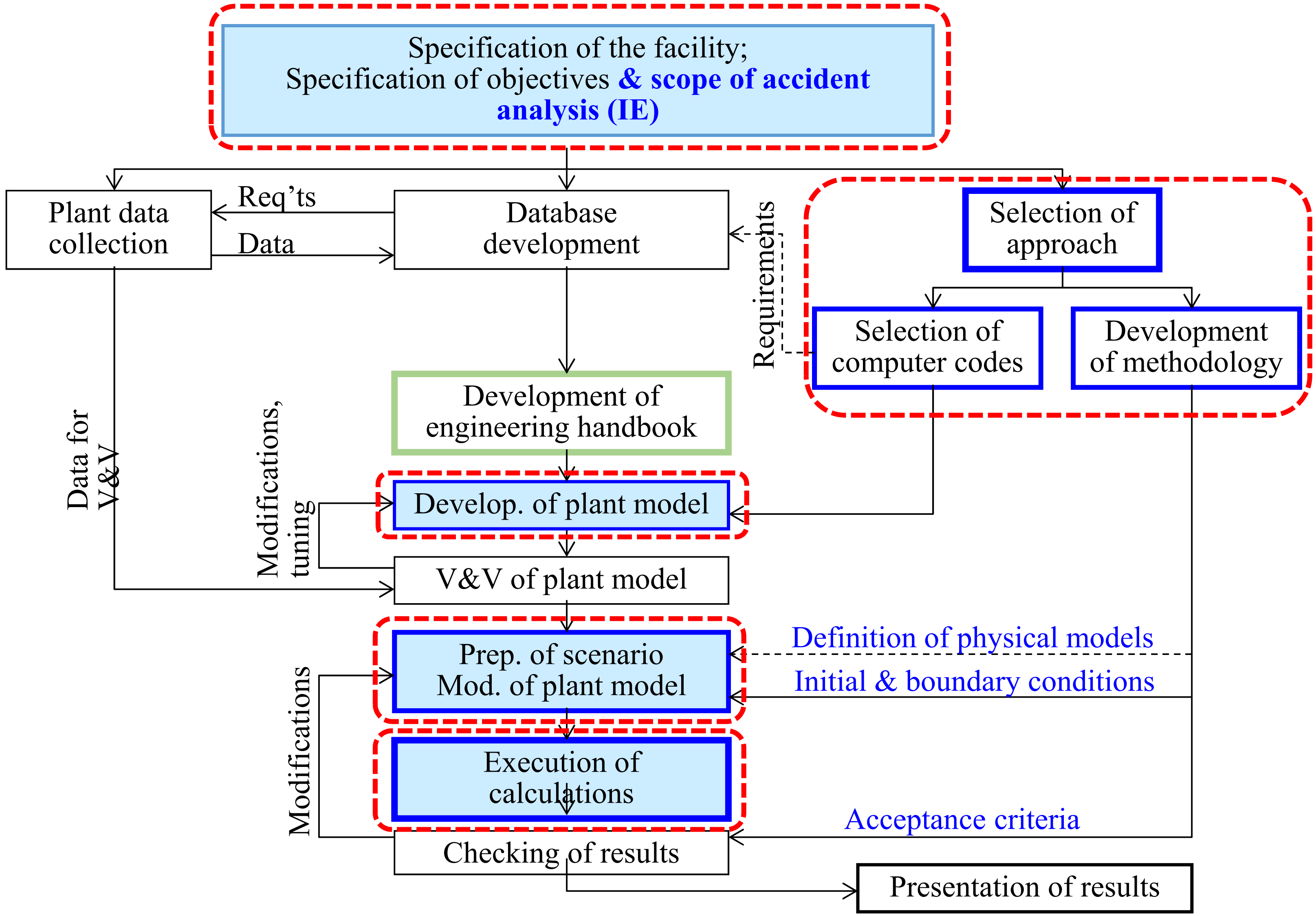
Design Extension Conditions(DECs) are accident conditions that are not considered for design basis accidents, but are considered in the design process of the facility to minimise or practically eliminate releases of radioactive material to protect members of the public outside the site.

- DECs are of extremely low frequency, so they have not historically been considered to be within the design basis.
- **The principal role** of the deterministic analysis of DECs is to define those scenarios that will progress to severe accidents.

3. Safety analysis approaches

1) Classification of Initiating Events
2) Overview of Deterministic Safety Analysis
3) Acceptance Criteria for DBAs
4) Verification and Validation of Codes

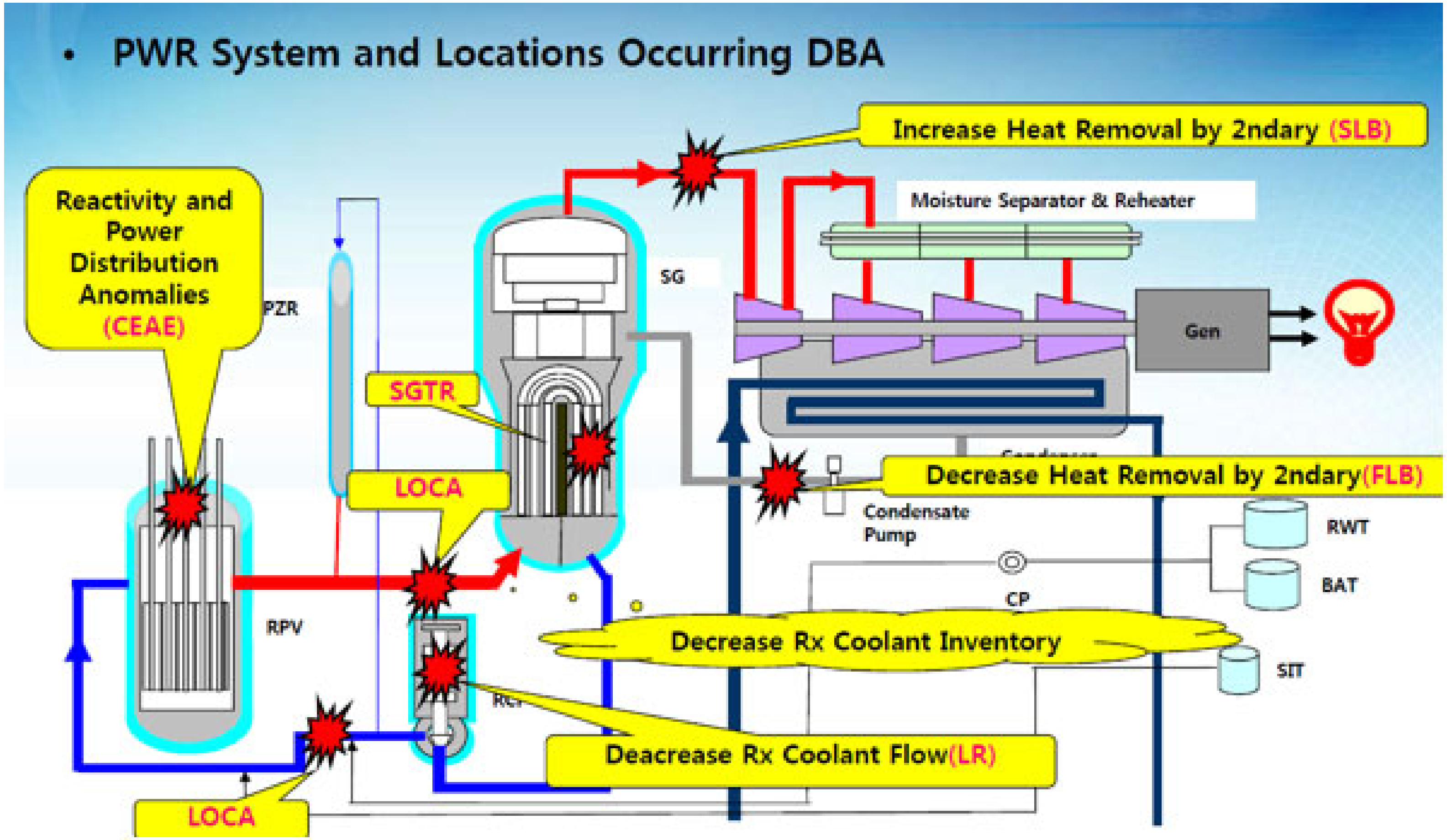
Basic Procedure of the Safety Analysis



(3-2) Overview of DSA (cont'd)

1. **Major Steps in Deterministic Analysis**
   1.1 Identification of Initiating Events
   1.2 Sequence of Events and Systems Operation
   1.3 Evaluation of consequences using computer codes

(3-1) Classification of Initiating Events (cont'd)



CEAE: Control Element Assembly Ejection
LOCA: Loss of Coolant Accident
SGTR: Steam Generator Tube Rupture
SLB: Steam Line Break
FLB: Feeder Line Break
LR: Reactor Coolant Pump Locked Roter

Initiating Events

① Reactivity and Power Distribution Anomalies

- Uncontrolled Control Element Assembly Ejection (CEAE) from a Subcritical or Low-Power Startup Condition
- Uncontrolled Control Element Assembly Ejection (CEAE) at Power
- Control Element Assembly Miss-operation
- Startup of an Inactive Reactor Coolant Pump (RCP)
- Inadvertent Decrease in Boron Concentration in the Reactor Coolant System
- Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
- Spectrum of CEA Ejection Accidents

② Decrease in Reactor Coolant Inventory

- Inadvertent opening of a pressurizer pressure relief valve
- Failure(leakage) of small lines carrying primary coolant outside the containment
- Steam generator tube failure
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB)

Examples of Initiating Events (cont'd)

③ **Increase in heat removal by the secondary side**

- Decrease in feedwater temperature
- Increase in feedwater flow
- Increase in steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure(break) inside and outside the containment

④ **Decrease in heat removal by the secondary side**

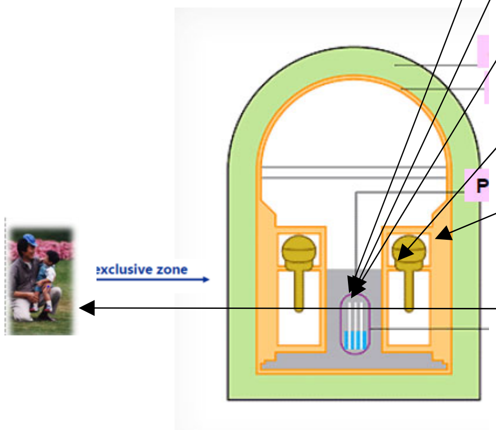
- Loss of external load
- Turbine trip
- Loss of condenser vacuum (LOCV)
- Closure of main steam isolation valve
- Loss of nonemergency ac power to the station auxiliaries
- Loss of normal feedwater flow
- Feedwater system pipe break inside and outside the containment

(3-2) Overview of Deterministic Analysis

Systematic Evaluation

Assumptions

Modeling



Analysis	Results
Fuel Behavior	Fuel Centerline Temp., Enthalpy, etc
Core Thermal-hydraulic Analysis	DNBR
Core Neutronic Analysis	MTC, DTC, Reactivity, Power(t), etc
System Thermal-hydraulic Analysis	Max Pressure, PCT, M/E Release, etc
Containment Analysis	Containment P, T
Radiological Consequence Analysis	Dose at EAB, LPZ, Control Room

(3-2) Overview of DSA (cont'd)

1.1 Identification of Initiating Events

- Limiting initiating events
- (e.g.) Increase in heat removal by the secondary side
  - Decrease in feedwater temperature,
  - Increase in feedwater flow
  - Increase in steam flow (Limiting)
  - Inadvertent opening of a steam generator relief or safety valve

(3-2) Overview of DSA (cont'd)

1.2 Sequence of Events and Systems Operation

- Step-by-step from initiation to finalized condition (e.g. occurrence, sensor trip, insertion of control rods, attainment of safety valve setpoint, opening/closing of safety valve, generation of containment isolation signal, containment isolation, operator action credited, etc.)
- Use normal operating plant I&C assumed and reactor protection system
- Use only safety-related system
- Credited operation of engineered safety systems

① Scenarios of event (example)

SINGLE FAILURES ASSUMED IN ACCIDENT ANALYSES

	EVENT	WORST FAILURE ASSUMED
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	
15.1.1	Feedwater system malfunctions causing a decrease in feedwater temperature	One protection train
15.1.2	Feedwater system malfunctions causing an increase in feedwater flow	One protection train
15.1.3	Excessive increase in secondary steam flow	No protection action required
15.1.4	Inadvertent opening of a steam generator relief or safety valve	One safety injection train
15.1.5	Steam system piping failure	One safety injection train
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	
15.2.1	Steam pressure regulator malfunction or failure causing a decrease in steam flow	N/A
15.2.2	Loss of external load	One protection train
15.2.3	Turbine trip	One protection train
15.2.4	Inadvertent closure of main steam isolation valves	One protection train
15.2.5	Loss of condenser vacuum and other events causing a turbine trip	One protection train
15.2.6	Loss of non-emergency AC power to the plant auxiliaries	One auxiliary feedwater pump
15.2.7	Loss of normal feedwater flow	One auxiliary feedwater pump
15.2.8	Feedwater system pipe break	One auxiliary feedwater pump

② System Operation

- Use only safety-related system
  - Reactor protection system
  - Safety injection system
  - Auxiliary feedwater system
  - Overpressure protection system
  - Main steam/feedwater isolation system
  - Emergency diesel generators
  - Reactor containment system

② System operation (cont'd)

- **Protective Actions and Safety Systems Actions**
  - Single failure criterion
  - Limiting delay time for protection safety system function used (calibration error, drift, instrumentation error, etc.)
- **Operator action**
  - Operator action can be credited mostly after 30 minutes after the initiation of event
  - To apply earlier action time, justification is required by analyzing operator responses
    - 15 minutes for boron dilution event(easy to recognize in MCR)

1.3 Evaluation of consequence using computer codes

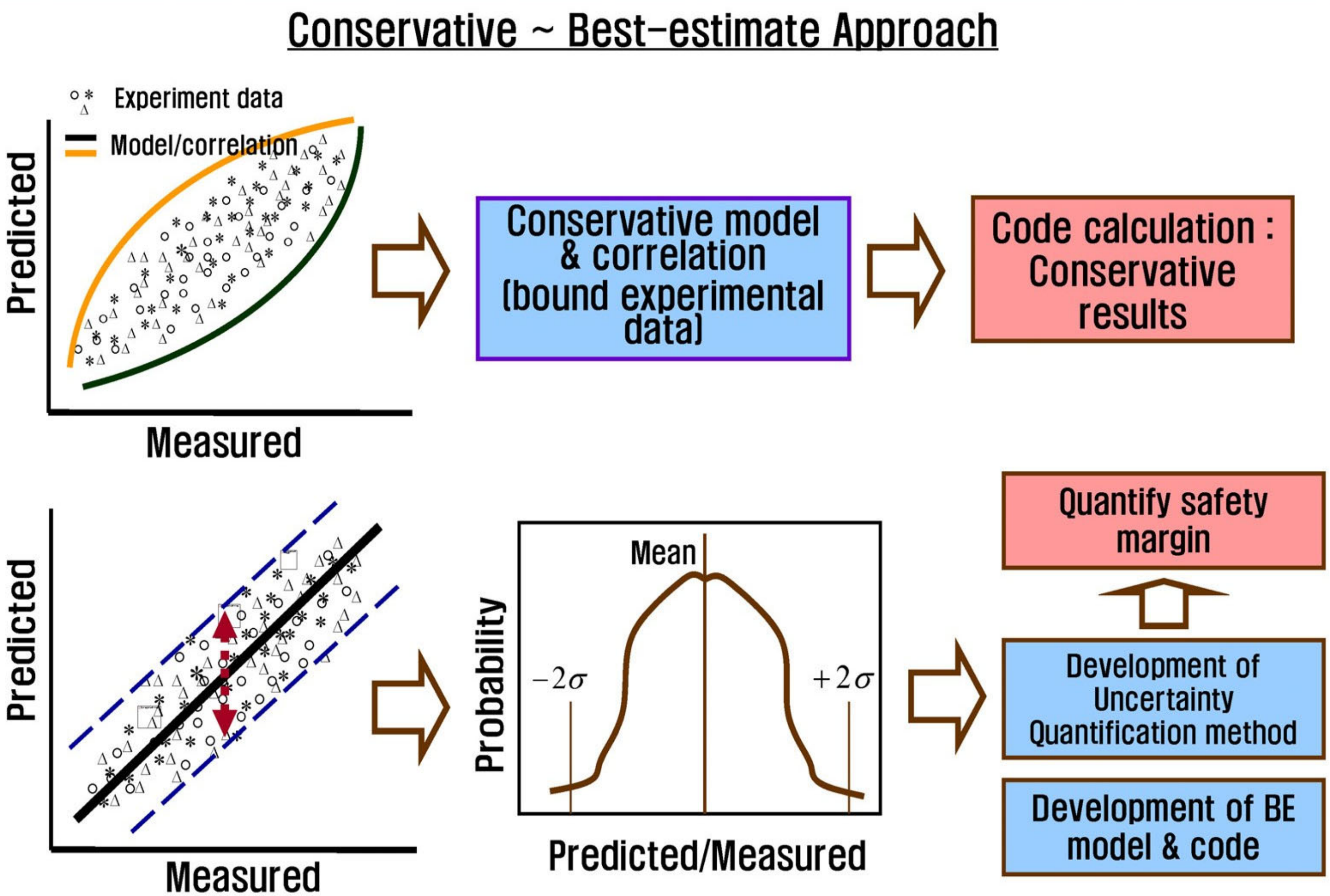
① Methods

- Conservative Analysis
  - Direct comparison of analysis results with acceptance criteria

    (eg.) PCTallowable > PCTconservative > PCTactual
- Best-Estimate analysis + Uncertainty
  - Comparison of analysis results plus uncertainty with acceptance criteria

    (eg.) $PCT_{allowable} > PCTBE + PCT_{uncert.} > PCT_{actual} > PCTBE - PCT_{uncert.}$

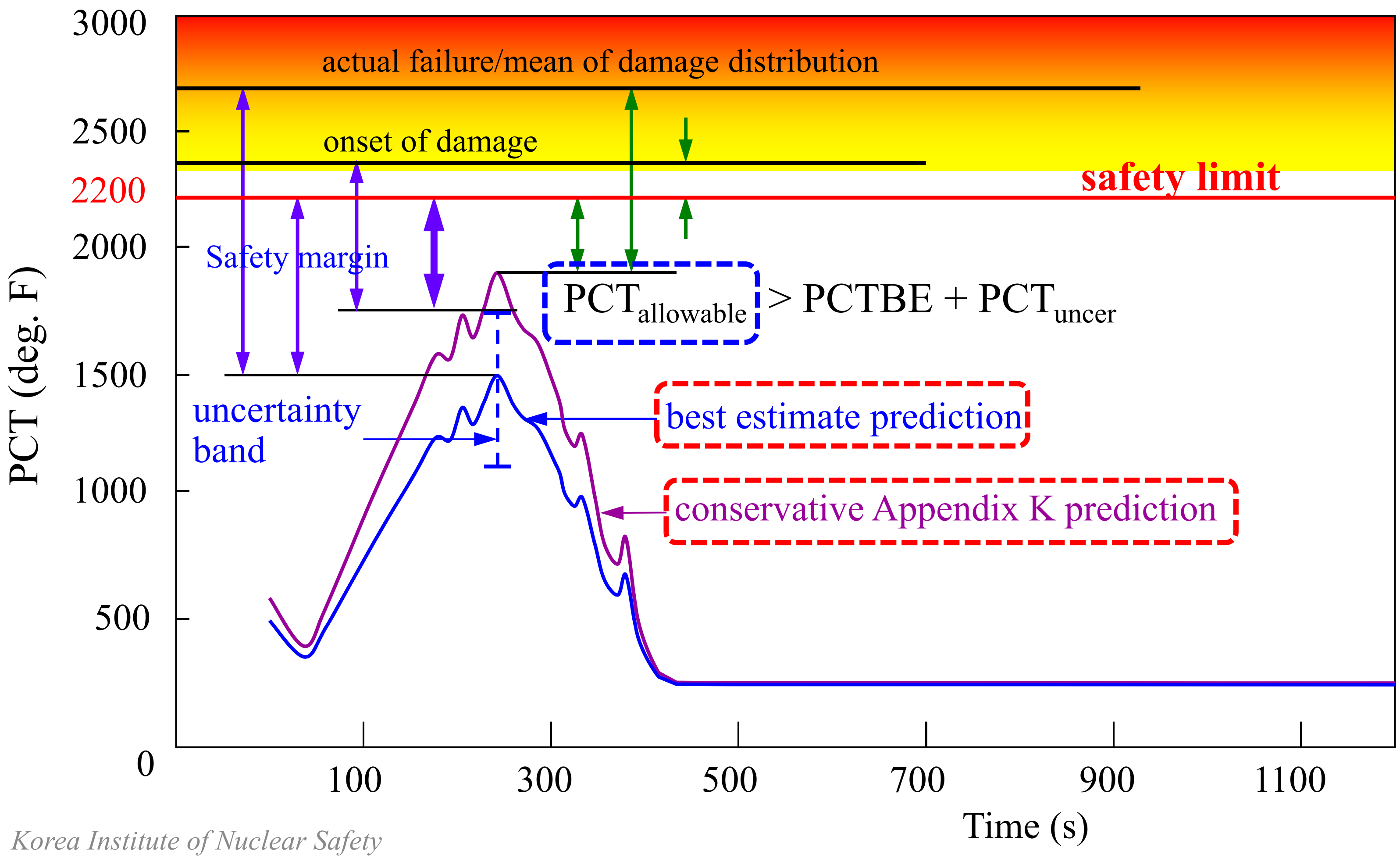
(3-2) Overview of DSA (cont'd)



Example: Safety Margin - Design Analysis

- The term originated in design analyses (e.g., fuel rods)

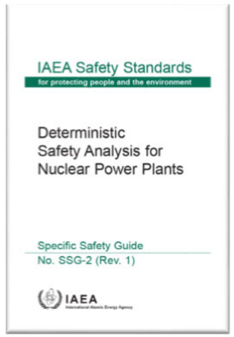
From USNRC/RES Dr. M. Gavrilas; SMAP Madrid, 10/19-20/2006



Example: Options for performing deterministic safety analysis

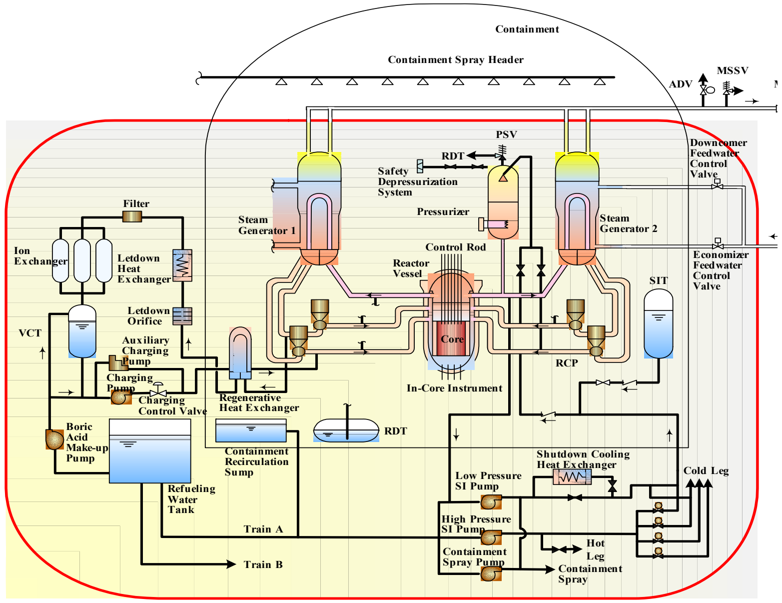
Option	Computer code type	Assumptions about systems availability	Type of initial and boundary conditions
1. Conservative	Conservative	Conservative	Conservative
2. Combined	Best estimate	Conservative	Conservative
3. Best estimate plus uncertainty	Best estimate	Conservative	Best estimate Partly most unfavourable conditions
4. Realistic*	Best estimate	Best estimate	Best estimate

* For simplicity, the terms 'realistic approach' or 'realistic analysis' are used in this Safety Guide to mean best estimate analysis without quantification of uncertainties.



(3-2) Overview of DSA (cont'd)

② Plant modelling



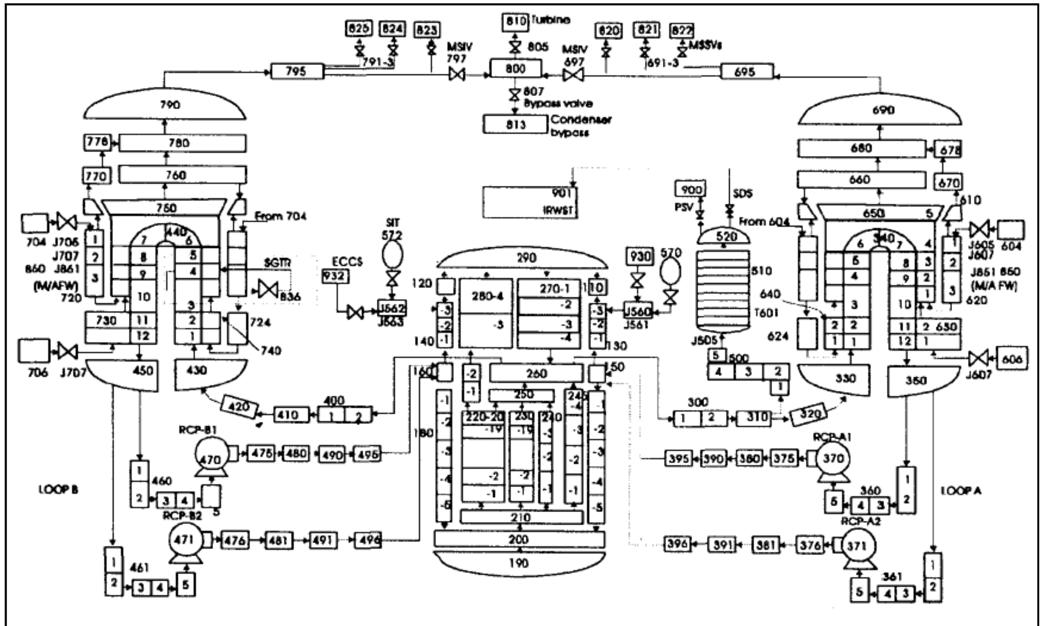


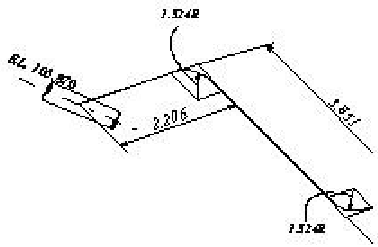
Fig. 1. Schematic Diagram of APR1400 Nodalization

Example : Engineering handbook

Design information for SSCs

Parameter Type : BRANCH SURGE1 Description: Pressurizer surge line volume	Card #: 0510101

Calculations, Comments and Figures:



$$L_{M1} = 2.206 - \Delta - R_{HL} \quad /1/$$

$$\Delta = r\left(tg\frac{\alpha_1}{2} - \frac{\alpha_2}{2}\right), \quad \alpha_1 = 118.5^\circ \wedge \alpha_2 = 61.5^\circ$$

$$R_{HL} = 0.368 \quad /2/$$

$L_{SL} = 1.750$	$D_i = 0.257$	/1/
$A_{SL} = 5.196 \cdot 10^{-2}$	$\Delta h_{SL} = 0.090$	/1/
$V_{SL} = 9.093 \cdot 10^{-2}$	$\varepsilon = 4.572 \cdot 10^{-5}$	/3/, page 3-56, Table 3.3.9

References (Title, page number and revision):
/1/ CLI ANALYSIS-RC-PRESS SURGE LINE drw # C-318-601, rev.0, sheet 2
/2/ REACTOR COOLANT PRIMARY LOOP drw # E-304-601
/3/ Marks' STANDARD HANDBOOK for MECHANICAL ENGINEERS, Eugene A. Avallone, Theodore Baumeister III, Ninth Edition

Parameter Type : CNTRLVAR Description : SG Downcomer NR Water Level	Card #: 20500420-20500425

Calculations, Comments and Figures:

NOTE: Control Variable 041 simulate calibrated measurement of the SG collapsed level. Current RELAP5 model calculates gravitational pressure head between measurement taps positioned inside SG. Acceleration pressure drops and conditions in the reference leg are neglected, /1/, pp.13-17. For SG2 corresponding control variable is 51. Calibrated conditions are taken as arithmetical middle between two SGs at the end of steady state.

Formula: SG LVL narrow (%) = $100(\Delta p_{meas} - \Delta p_{0\%}) / (\Delta p_{100\%} - \Delta p_{0\%})$

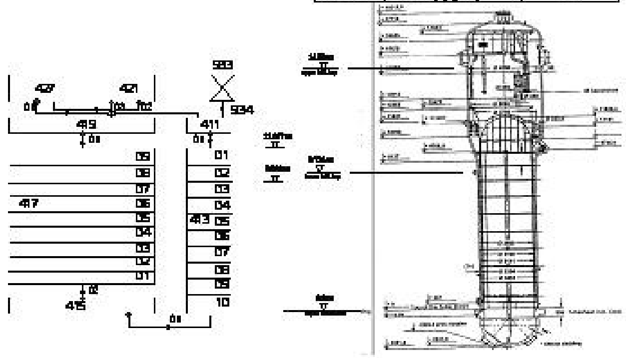
ρ_{Gcal} - density of saturated steam under calibration condition
ρ_{Lcal} - density of saturated liquid under calibration condition
ρ_{cv} - control volume density
$g = 9.80665\ m/s^2$
$H_{NR} = 5.155$ - NR (narrow range) span
$H' = 3.767$ - height between FW J-tubes outlet and upper NR tap
$H'' = 1.388$ - height between FW J-tubes outlet and lower NR tap
H_i - control volume height inside NR span
ρ_{DCcal} - density of liquid inside downcomer under calibration condition is taken from iterative RELAP5 runs for design steady state conditions (calibrated conditions) for SG model only.

Rearranged formula used:

RELAP5 CNTRLVAR model (BE conditions):

Calculated calibrated conditions

	BE Flow
pressure (MPa)	6.81
saturated liquid density (kg/m³)	748.33
steam density (kg/m³)	33.6
FW Temperature (K)	492.6
CR	3.7
downcomer liquid density (kg/m³)	777.6



References (Title, page number and revision):
/1/ Water Level Recommendations and Calibration Data for the Krško Replacement Steam Generators, KWU NDM2/97/E051, Rev. c, Erlangen, 1999.

Example : Plant parameters

Operating information for SSCs

Nuclear design

- control rod worth, rod insertion time, shutdown margin
- control temperature feedback coefficients (fuel, moderator)
- power distribution (radial, axial)
- decay heat
- fission product inventory
- delayed neutron fraction

Fuel

- thermal conductivity (pellet, gap, cladding)
- gap fraction of fission product
- fuel and cladding dimension

Core thermal-hydraulics

- fuel rod heat flux
- heat transfer coefficient between cladding and coolant
- coolant flow rate
- core bypass flow rate

Example : Plant parameters (cont'd)

- **RCS**
  - coolant pressure/temperature
  - coolant inventory (Pressurizer level, charging flow, letdown flow)
  - pressurizer safety valve open/close setpoints
  - RCP coastdown curve
  - ESF actuation delay time
- **Main steam system**
  - coolant inventory (SG water level, feedwater flow rate)
  - steam pressure/temperature
  - main steam safety valve open/close setpoints
- **Instrumentation and control system**
  - process time including delay in instrumentation and actuation

Plant modelling for code calculation

Preparation of Input deck

100 new transnt

101 run

102 si si

110 nitrogen

115 1.0

*

* End.Time Min.Time Max.Time

201 200. 1.0e-8 0.0001 3 1000 250000 250000

* Variable Trips

510 time 0 gt null 0 0.0 l -1.0

401 time 0 gt timeof 510 10.0 l -1.0

402 acvliq 570 lt null 0 27.06 n -1.0

403 vlvarea 578 gt null 0 0.13 n -1.0

404 acvliq 570 lt null 0 1.630835 n -1.0

405 acvliq 570 gt null 0 1.630835 n -1.0

* Logical Trips

704 401 and -402 n -1.0

705 403 or -402 n -1.0

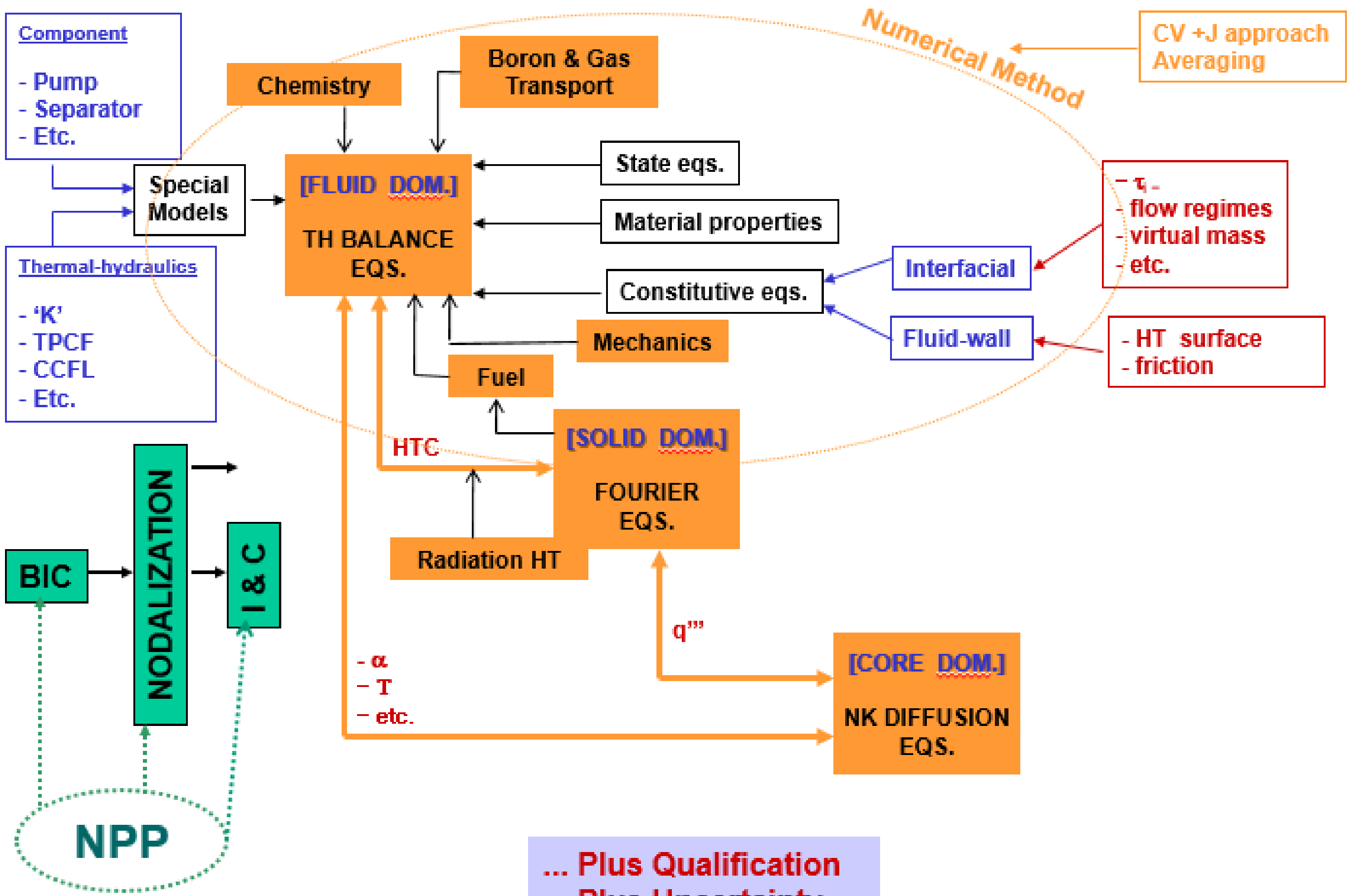
706 705 and -704 n -1.0

(3-2) Overview of DSA (cont'd)

③ **Computer codes for DBA Analysis**

- Conservative vs. Best-Estimate Codes
  - Conservative code
    - conservative models & assumptions based on Evaluation Models (e.g. Appendix K of 10 CFR 50)
  - BE code
    - realistic & detailed modelling, uncertainty quantification

The structure of a TH SYS code



... Plus Qualification
... Plus Uncertainty

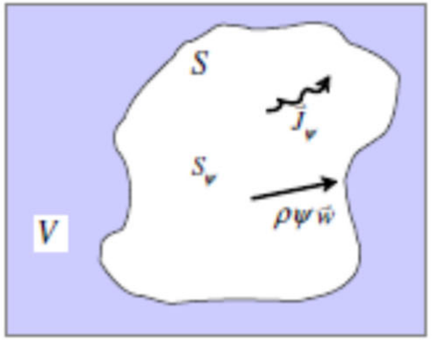
(3-2) Overview of DSA (cont'd)

③ **Computer codes for DBA Analysis**

- Characteristics of Best-Estimate T/H System Codes
  - Mixed hyperbolic-elliptic system of 6 conservation equations (mass, energy and momentum for the vapor & liquid phases)
  - Constitutive laws to describe the needed boundary conditions for each of the phases, e.g. friction between the phases and the wall
  - Typically 1-D modelling; partial implementation of 3-D modeling
  - Code validation with SET and IET data bases

Example: Code structure

- To discuss about RELAP5 point of view in balances, let's take that way: let us make a macroscopic (integral) balance on a volume V



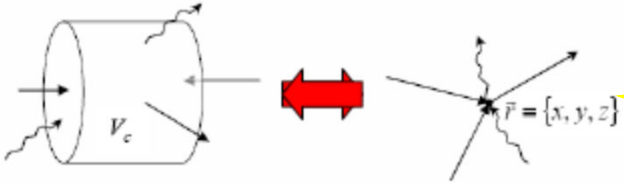
$$\underbrace{\frac{d}{dt}\int_V \rho\psi\, dV}_{\substack{\text{rate of change of }\Psi\\ \text{in volume V}}} = -\underbrace{\int_S \rho\,\psi\,\vec{w}\cdot\vec{n}\, dS}_{\substack{\text{flow of }\Psi\\ \text{due to fluid motion}\\ \text{(advection)}}} - \underbrace{\int_S \vec{J}_\psi\cdot\vec{n}\, dS}_{\substack{\text{surface}\\ \text{source of }\Psi\\ \text{(diffusion)}}} + \underbrace{\int_V \rho S_\psi dV}_{\substack{\text{volumetric}\\ \text{source of }\Psi}}$$

where ψ is the specific value per unit mass of Ψ .

- Some efforts are then spent to obtain a partial differential equation from this macroscopic balance

Example: Code structure

- Numerical schemes are therefore called upon, which often revert back the painful process made from the finite volume to the point



CV +J

- This is the very moment when students start becoming angry with you:

"why should we make efforts for getting PDEs
if we must finally revert back to finite volume balances?"

- This question admits multifold answers, but unfortunately none applies so much to RELAP5 code, in which finite control volumes are adopted

- In the case of two-phase flow, even more complicating aspects come into play, owing to the fact that at each spatial location either phase may be present at a given time
- This is customarily accounted for by functions like

Time & Space Avg

$$\alpha_k(\vec{r},t) = \begin{cases} 1, \text{ if the } k-th \text{ phase is present in } \vec{r} \text{ at time } t \\ 0, \text{ if the } k-th \text{ phase is absent in } \vec{r} \text{ at time } t \end{cases}$$

- Time and/or space averaging is then required to define suitable values of phasic volume fractions making some sense on engineering duration and/or length scales

Table : Thermal-Hydraulic System Codes

Name	Developer	Governing Eq.	Numerical Methods	T/H Dimension
TRAC-PF1	USNRC	2C, 2M, 2E(*)	SETs	1D, 2D, 3D Catesian, Cylinder
TRAC-M	USNRC	2C, 2M, 2E	SETs, Semi-implicit	1D, 2D, 3D Catesian, Cylinder
RELAP5/MOD3	USNRC	2C, 2M, 2E	Semi-impicit	1D
RELAP5-3D	USDOE, INEEL	2C, 2M, 2E	Semi-implicit Two-step nearly implicit	1D, 2D, 3D Catesian, Cylinder
COBRA-TF	PNL, USA	3C, 3M, 2E	Semi-implicit	3D Component Subchannel
RETRAN-03	EPRI, USA	2C, 1M, 2E	fully implicit	1D
CATHARE	CEA, France	2C, 2M, 2E	fully implicit(0D,1D) semi-implicit(3D)	0D,1D,2D,3D
ATHLET	GRS, Germany	2C, 1M, 2E 2M for DC	fully implicit semi-implicit	1D, 2D, 3D (FLUBOX)
MARS	KAERI	2C, 2M, 2E	fully implicit semi-implicit	1D, 2D, 3D

Table : Containment Analysis Codes

Code	Country	Type	T/H Dimension
CONTAIN	USA	Lumped parameter	Thermal hydraulics, Hydrogen burning, Aerosol models
COCOSYS	Germany	Lumped parameter	Thermal hydraulics, Hydrogen burning, Aerosol models
GOTHIC	USA/ Germany	Lumped parameter	Thermal hydraulics, Hydrogen distribution & reduction
WAVCO	Germany	Lumped parameter & 3D CFD versions	Thermal hydraulics, Pressure differences
CONTEMPT-LT	USA	Lumped parameter	Thermal hydraulics

3) Selection of initial and boundary conditions

Parameter	Conservative direction	
	Core cooling	System pressure
Reactor power	Max.	Max.
Reactor residual heat	Max.	Max.
Reactor coolant flow	Min.	Min.
Reactor core bypass	Max.	Min.
Reactor coolant temperature	Max.	Min.
Reactor coolant pressure	Min.[a]	Max.
Steam generator level	Min.	Min.
Steam pressure	Max.	Max.
Feedwater flow	N/A (consistent with power)	N/A (consistent with power)
Pressurizer level	Min.	Max.
Power peaking factor	Max.	Max.
CR worth available for reactor scram	Min.	Min.

[a] For LOCA analysis, a maximum value should be selected. For ATWS and SA analysis, best estimate plant initial conditions are typically acceptable, even for design and licensing type analyses.

Example : Typical Initial Conditions for plant accident analysis

TABLE II. CONSERVATIVE SELECTION OF NEUTRONIC PARAMETERS LEADING TO OVERESTIMATION OF REACTOR POWER

Parameter change	Reactivity feedback: Fuel temperature coefficient (FTC)	Reactivity feedback: MTC[a] + void	Reactivity feedback: Boron concentration coefficient (BCC)	Fraction of delayed neutrons	Prompt neutron lifetime
Increase of coolant temperature	Strong	Weak	Weak	Max.	Max.
Decrease of coolant temperature	Weak	Strong	Weak	Min.	Min.
Reactivity increase by CRs	Weak	Weak	Weak	Min.	Min.
Reactivity decrease by CRs	Strong	Strong	Weak	Max.	Max.
Void fraction in the core during LOCA	Strong	Weak	Strong	Max.	Max.
Boron dilution	Weak	Weak	Strong	Min.	Min.

[a] MTC: moderator temperature coefficient.

‘weak’ means minimum absolute value of a feedback coefficient
‘strong’ means maximum absolute value of a feedback coefficient.

Example : Typical Initial Conditions for plant accident analysis

TABLE III. INITIAL CONDITIONS FOR CONTAINMENT PRESSURIZATION ANALYSIS

Parameter	Conservative selection
Containment initial pressure	Max.
Containment initial temperature	Min.
Spray water temperature	Max.
Containment leak rate	Min.

Example : Typical Initial Conditions for plant accident analysis

- Trip points and time delays to trip assumed in accident analyses
- RTS and ESFAS actuation should be assessed in the light of the set points value and associated delay

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delays (c)
Power Range Neutron Flux – -- High setpoint	118%	0.5
	109% (b)	0 (b)
Power Range Neutron Flux – -- Low setpoint	35%	0.5
Overtemperature Delta T	Variable see Figure 15.0.3-1	6.0 (a)
	Nominal Technical Specifications Setpoints (b)	0 (b)
Overpower Delta T	Variable see Figure 15.0.3-1	6.0 (a)
	Nominal Technical Specifications Setpoints (b)	0 (b)
Pressurizer Pressure – -- High	2420 psia (16.7 MPa)	2.0
Pressurizer Pressure – -- Low	1835 psia (12.65 MPa)	2.0
	1935 psia (13.34 MPa) (b)	0 (b)
Pressurizer Water Level – -- High	100% of pressurizer level span	2.0
Reactor Coolant Flow – -- Low (from loop flow detectors)	87% loop flow	1.0
Undervoltage trip	N/A	1.5 (c)
Turbine trip	N/A	1.0
Steam Generator Water Level – -- Low-Low	0% of narrow range level span	2.0
Steam Generator Water Level – -- High Trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	91% of narrow range level span	2.5

(a) Total time delay (including Resistance Temperature Detector (RTD) bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

(b) For SGTR analysis.

(c) This time delay includes not only the trip breaker opening delay (.15s) and the RCCA release delay (.15s) but also the EMF decay delay assumed to be equal to .25s.

Example : Typical Initial Conditions for plant accident analysis

- Determination of maximum overpower trip point – power range neutron flux channel - based on nominal setpoint considering instrumentation errors

Nominal setpoint (% of rated power)		109	
Variable	Accuracy of Measurement of Variable (% Error)	Estimated Error (% of rated power)	Assumed Error (% of rated power)
Feedwater temperature	± 0.5	± 0.3	
Feedwater pressure (small correction on enthalpy)	± 0.5	± 0.3	
Steam pressure (small correction on enthalpy)	± 2.0	± 0.3	
Feedwater flow	± 1.25	± 1.25	
Calorimetric errors in the measurements of secondary system thermal power			± 2 (a)
Axial power distribution effects on total ion chamber current		± 3	± 5 (b)
Instrumentation channel drift and setpoint reproducibility		± 1	± 2 (c)
Total assumed error in setpoint (a) + (b) + (c)			± 9
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction (% of rated power)			118

Example : Typical Initial Conditions for plant accident analysis

- Initial radioactive inventory in the core for evaluation of radioactivity releases consequences

CORE AND GAP ACTIVITIES AT END OF EQUILIBRIUM CYCLE

Fuel and Rod Gap Inventories (Curies) - Core = 121 Assemblies (End of Equilibrium Cycle)			
Isotope	Core Fuel Inventory	Gap Fraction	Gap Inventory
Kr-83m	7.262 E+06	0.10	7.262 E+05
Kr-85	5.207 E+05	0.30	1.562 E+05
Kr-85m	1.576 E+07	0.10	1.576 E+06
Kr-87	3.064 E+07	0.10	3.064 E+06
Kr-88	4.320 E+07	0.10	4.320 E+06
Kr-89	5.322 E+07	0.10	5.322 E+06
Xe-131m	6.017 E+05	0.10	6.017 E+04
Xe-133	1.098 E+08	0.10	1.098 E+07
Xe-133m	3.467 E+06	0.10	3.467 E+05
Xe-135	2.752 E+07	0.10	2.752 E+06
Xe-135m	2.143 E+07	0.10	2.143 E+06
Xe-138	9.482 E+07	0.10	9.482 E+06
I-131	5.375 E+07	0.12	6.450 E+06
I-132	7.810 E+07	0.10	7.810 E+06
I-133	1.122 E+08	0.10	1.122 E+07
I-134	1.240 E+08	0.10	1.240 E+07
I-135	1.049 E+08	0.10	1.049 E+07

(3-3) Acceptance Criteria of DSA (cont'd)

- Limits and conditions set by a regulatory body to achieve an adequate level of safety for the entire range of operational states and accident conditions.
  - Acceptance criteria should be set in terms of the variable or variables that directly govern the integrity of a barrier such as PCT, DNBR, Pellet Enthalpy Rise, etc.
- Acceptance criteria may be related to the frequency of the event.

Example : Plant Conditions & Acceptance Criteria (USA)

Category	Condition I	Condition II	Condition III	Condition IV
Name	Normal operation & operational transients	Incidents of moderate frequency	Infrequent incidents	Limiting faults
Expected Frequency	Expected	Once per reactor year	Less than once during plant life	Not expected during plant life
Typical Acceptance Criteria	● Prevention of fuel failure (by avoiding CHF ● $P_{max} < 1.1\ P_{design}$		● Prevention of severe core damage ● Continuous cooling ● Radioactive release < 10% of 10CFR100 ● $P_{max} < 1.1\ P_{design}$	● Continuous cooling ● Radioactive release < 10CFR100 ● Separate criteria for LOCA
Example for PWRs	• Normal power operation • Start-up • Shutdown • Refueling	• Decrease in feedwater flow • Loss of offsite power • Turbine trip • Partial loss of coolant flow	• Total loss of coolant flow • Very small loss of coolant • Small break in steam line	• LOCA • MSLB • MFLB

(3-4) Verification and Validation of Codes

1) Code verification

- To ensure that the code design is appropriately implemented in accordance with the design requirements
  - the numerical methods
  - the equations into a numerical scheme
  - user options
- To include a review of;
  - the design concept,
  - basic logic and flow diagrams,
  - numerical methods and algorithms, and
  - computational environment.

(3-4) Verification and Validation of Codes (cont'd)

Functional Requirements of the Code

Code Verification Plan
- Objectives
- Approach
- Schedule
- Plan for testing

Code Verification Activities

Numerical algorithm
- Verification matrix
- Tests for
  - Conservation
  - Flow transitions
  - Convergence
- Robustness
- Versatility

Numerical solution
- Verification matrix
- Comparison with
  - Manufactured solutions
  - Analytical solution
  - Highly accurate numerical solution
  - Experiment

Source code
(Software quality engineering)
- Configuration management
- Software quality analysis and testing

(3-3) Verification and Validation of Codes (cont'd)

2) Code Validation

- To provide confidence in the code ability to predict safety parameter
  - quantify the code accuracy
- To be performed in two steps;
  - development phase: by the code developer
  - independent assessment phase: independent of the developer
- User should simulate validation tests without having any prior knowledge of the experimental results
- The results of a validation to be used to determine the uncertainty of the code

(3-4) Verification and Validation of Codes (cont'd)

A validation matrix usually includes for types of test:

- Basic tests (or fundamental tests)
- Separate effect test (SET)
- Integral effect tests (IET)
- Plant data

No	Type	Concerned NPP	Concerned phenomenon or DBA	Notes
1	Basic	-	Bottle emptying	To test code features
2			U-tube manometer	To test code features and dependency of results upon boundary conditions
3			Pressure drops in two phase flow	
4	SET		TPCF	Key phenomenon for DBA Analysis
5		PWR	Transient CHF	
6	ITF		SBLOCA	
7				Counterpart Test, to address the scaling issue
8	NPP			To perform Kv-scaled calculation

4. ECCS and LOCA Analysis

4. ECCS and LOCA Analysis

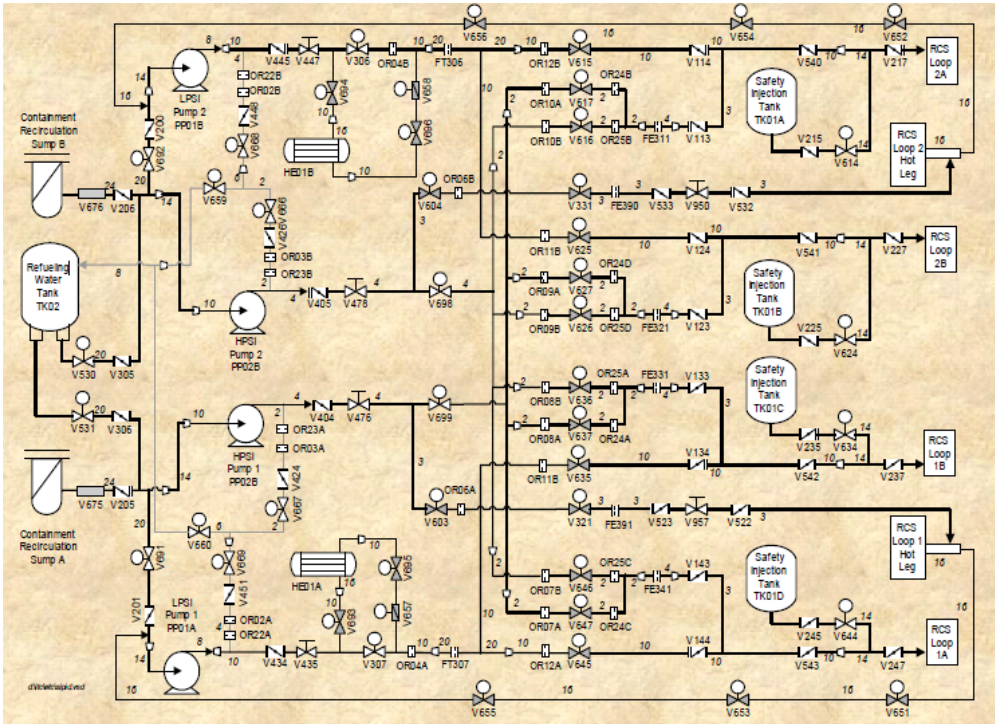
ECCS (Emergency Core Cooling Systems)

Functions

- Provide emergency core cooling water into the reactor following postulated accidents, e.g., LOCA, Steam Line Break, etc.

Major Components

- Active portion: High and low pressure safety injection and associated valves
- Passive portion: Safety Injection Tank (SIT), piping and instrumentation



ECCS Regulations

	USA	Korea
ECCS Design Criteria	10 CFR 50 App. A General Design Criteria 35-37	Nuclear Safety Act Regulations on Technical Standards for Nuclear Reactor Facilities, etc. Article 30
ECCS Acceptance Criteria	10 CFR 50.46	NSSC Notice 2017-23
ECCS EM	10 CFR 50 App. K	KINS Technical Guidance, KINS/GT-N007-1
ECCS BE	US NRC Regulatory Guide 1.157	KINS Technical Guidance, KINS/GT-N007-2

ECCS Regulations

- **ECCS Acceptance Criteria: 10 CFR 50.46**
  - Calculated maximum fuel element cladding temperature (PCT) < 2200 ^{o}F (Peak Cladding Temperature (PCT)
  - Calculated total oxidation of the cladding < 0.17 times the total cladding thickness (maximum cladding oxidation)
  - Calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam < 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders were to react (maximum hydrogen generation)
  - Calculated changes in core geometry ~ the core remains amenable to cooling (coolable geometry)
  - Calculated core temperature shall should be maintained at an acceptably low value for the extended period of time (long-term cooling)

ECCS Regulations

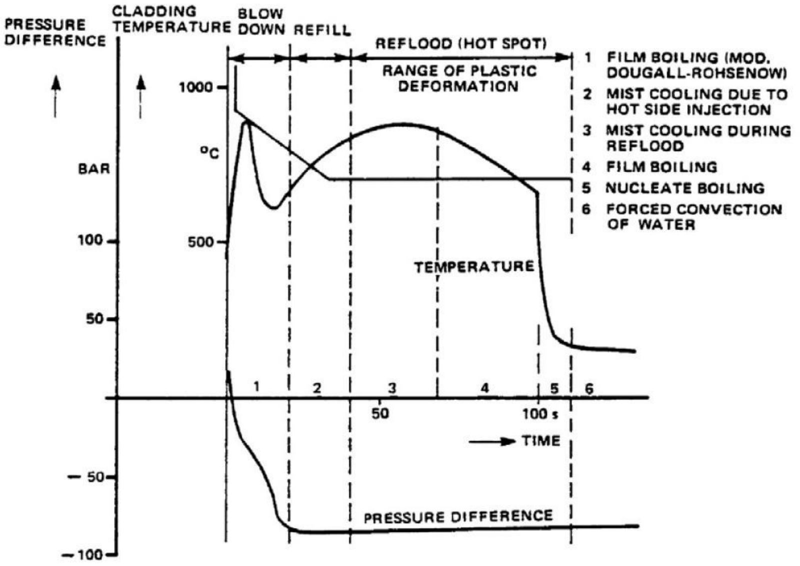
- Radiation dose to the public (10 CFR 100.11)
  - At exclusion area boundary (EAB)
    - 300 rem for thyroid and 25 rem for whole body for two hours
  - At control room
    - 300 rem for thyroid, 5 rem for whole body, 50 rem for skin for 30days
  - At low population zone outer boundary (LPZ)
    - 300 rem for thyroid and 25 rem for whole body for whole duration of the accident

4. ECCS and LOCA Analysis (cont'd)

LB LOCA Sequence of Event

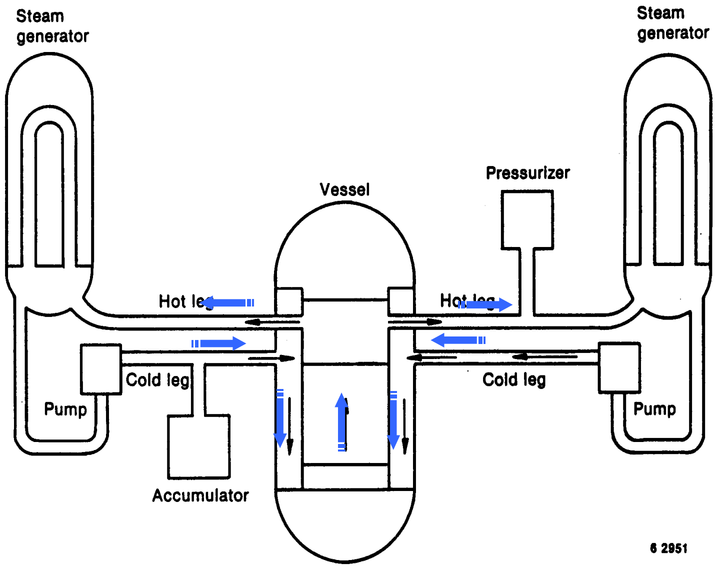
- **Blowdown Phase**
  - Discharge of Coolant through Break (0 ~ 25 ~ 30sec)
- **Refill Phase**
  - From End of Blowdown (EOB) to the time the bottom of the core of reactor vessel core filled with ECCS water (EOB ~ EOB+7 ~8 sec)
- **Reflood Phase**
  - From the core bottom flooding to complete core quenching (End of Refill to ~ Quenching time)
- **Long-Term Cooling Phase**
  - After complete Quenching to secure state

Double ended cold leg break, pressure difference across the cladding and cladding temperature at the hot spot



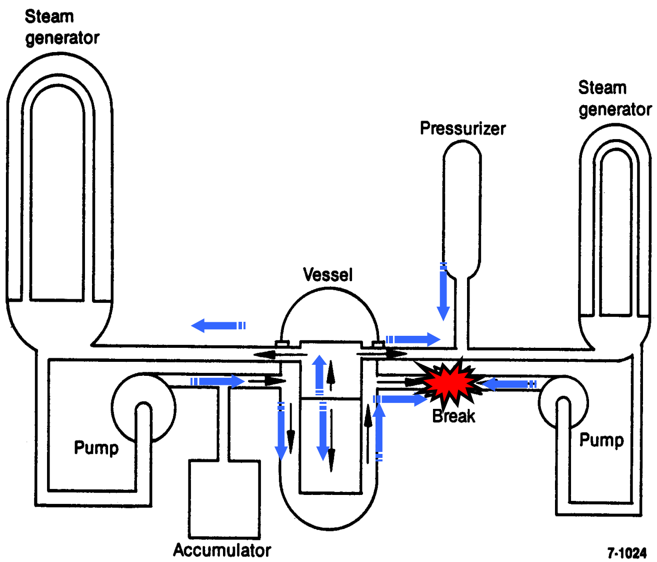
4. ECCS and LOCA Analysis (cont'd)

Normal Operation



Core → Hot Leg → SG → Cold Leg → Downcomer→ Core

After LOCA



Core → Downcomer/Hot Leg → Cold Leg/Hot Leg → Break

4. ECCS and LOCA Analysis (cont'd)

Example: Preparation of Input Deck

```
free format
*
*
**************
* main data *
**************
*
*        numtcr        ieos        inopt        nmat        id2o
              1           0            1          24           0
ATLAS-SS
*
*
******************
* namelist data *
******************
*
 &inopts
  dtstrt=-1.0,
  icdelt=1,
  ithd=0,
  tpowr=0.0,
  usesjc=3,
  numgentbl=8,
  npower=1,
  nhtstr=14
 &end
*
****************
* Model Flags *
****************
*
*       dstep        timet
            0          0.0
*      stdyst       transi       ncomp        njun        ipak
            1            0          95         100           1
*        epso         epss
      1.0E-3       1.0E-3
*      oitmax       sitmax      isolut      ncontr       nccfl
           10           10           0           0           0
*        ntsv         ntcb        ntcf        ntrp        ntcp
           10            3           0           2           0
*
*************************
* component-number data *
*************************
*
*  Component input order (IORDER)
*-- type --- num ------------ name --------------- *   jun1   jun2   jun3
* VESSEL    *      2 s *                                *
* POWER     *     11 s * rod heat                        *
* PIPE      *    118 s * touhdcb2                      *   74761  74763
* PIPE      *    148 s * touhdca1                      *   74760  74762
* PIPE      *    150 s * ongby1                        *   74756  74759
* PIPE      *    160 s * ongby2                        *   74758  74757
* PIPE      *    241 s * CGT1                          *   74740  74744
* PIPE      *    242 s * CGT2                          *   74742  74746
* PIPE      *    243 s * CGT3                          *   74743  74747
* PIPE      *    244 s * CGT4                          *   74741  74745
* PIPE      *    261 s * CEA1                          *   74748  74752
* PIPE      *    262 s * CEA2                          *   74750  74754
* PIPE      *    263 s * CEA3                          *   74751  74755
* PIPE      *    264 s * CEA4                          *   74749  74753
* PIPE      *    300 s * hota-r                        *   74677  52112
* PIPE      *    310 s * hota-p                        *   52112  52111
* PIPE      *    320 s * hota-s                        *   52111  52110
* PIPE      *    330 s * sga-inp                       *   52110      1
* PIPE      *    340 s * u-tube-a                      *       1  52109
* PIPE      *    350 s * sga-oup                       *   52109  52106
* PIPE      *    360 s * sleg-a1                       *   52105  74674
* PIPE      *    361 s * sleg-a2                       *   52106  74676
* PUMP      *    370 s * rcp-a1                        *   74674  74673
* PUMP      *    371 s * rcp-a2                        *   74676  74675
* PIPE      *    375 s * cla1-d                        *   74673  52103
* PIPE      *    376 s * cla2-p                        *   52107  74675
* PIPE      *    380 s * cla1-s                        *   52103  52104
* PIPE      *    381 s * cla2-s                        *   52108  52107
* PIPE      *    390 s * cla1-p                        *   52104  74631
* PIPE      *    391 s * cla2-d                        *   74632  52108
* PIPE      *    395 s * clb1-r                        *   74631  74679
* PIPE      *    396 s * clb1-r                        *   74632  74680
* PIPE      *    400 s * hotb-r                        *   74678  52122
* PIPE      *    410 s * hotb-p                        *   52122  52123
* PIPE      *    420 s * hotb-s                        *   52123  52124
* PIPE      *    430 s * sgb-inp                       *   52124  52125
* PIPE      *    440 s * u-tube-b                      *   52125  52126
* PIPE      *    450 s * sgb-oup                       *   52126  52127
* PIPE      *    460 s * sleg-b1                       *   52128  74670
* PIPE      *    461 s * sleg-a2                       *   52127  74672
* PUMP      *    470 s * rcp-b1                        *   74670  74669
* PUMP      *    471 s * rcp-b2                        *   74672  74671
* PIPE      *    475 s * clb1-d                        *   74669  52118
* PIPE      *    476 s * clb2-d                        *   74671  52121
* PIPE      *    480 s * clb1-s                        *   52118  52117
* PIPE      *    481 s * cla2-s                        *   52121  52120
* PIPE      *    490 s * clb1-p                        *   52117  52116
* PIPE      *    491 s * clb2-p                        *   52120  52119
* PIPE      *    495 s * clb1-r                        *   52116  74682
* PIPE      *    496 s * cla1-r                        *   52119  74681
* PIPE      *    500 s * surgel                        *   52113  52114
* PIPE      *    510 s * pzr                           *   52114  52115
* BREAK     *    520 s * pzr-top                       *   52115
* FILL      *    605 s * sga_rej                       *   74781
```

4. ECCS and LOCA Analysis (cont'd)

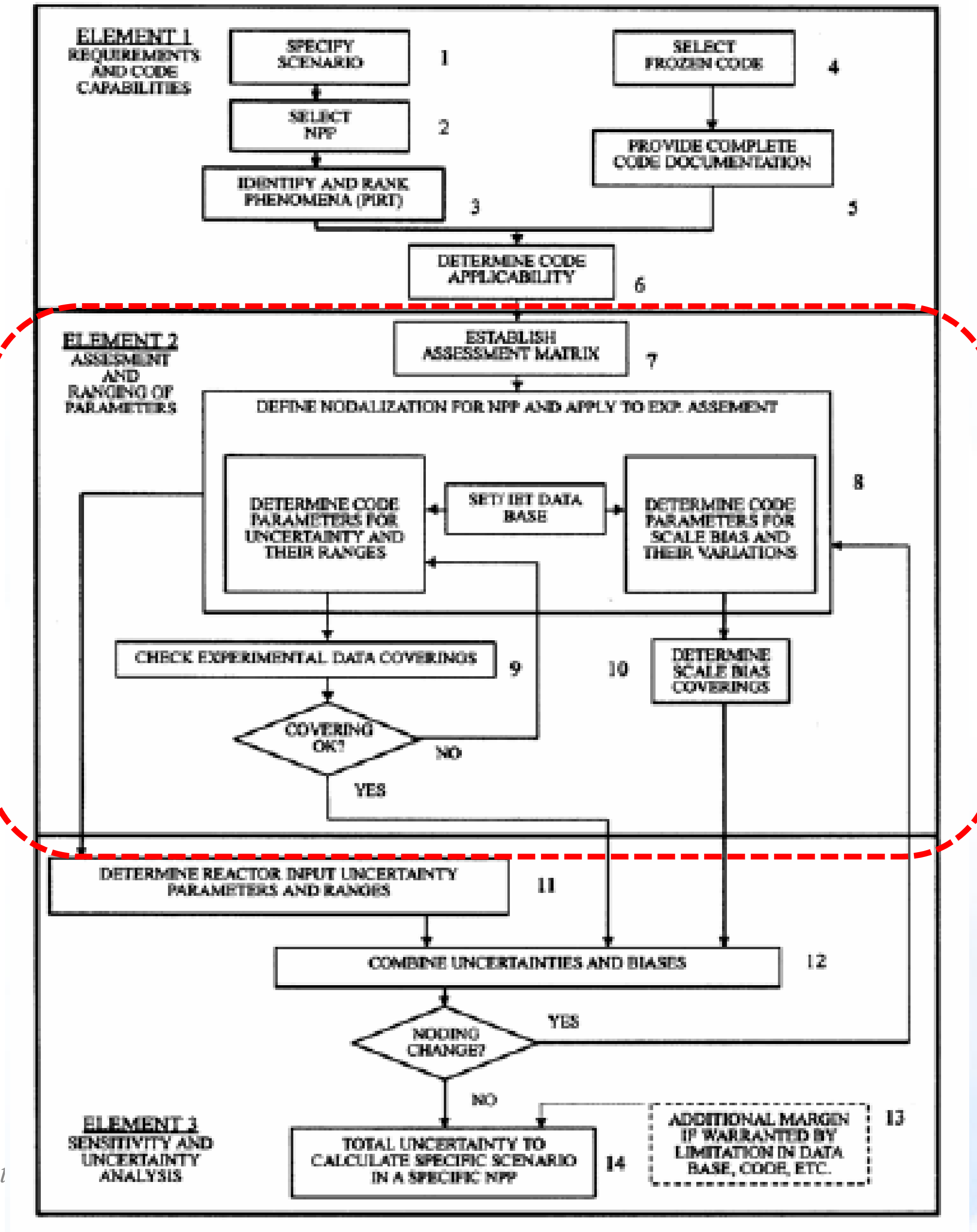
Evaluation Model (Conservative EM)

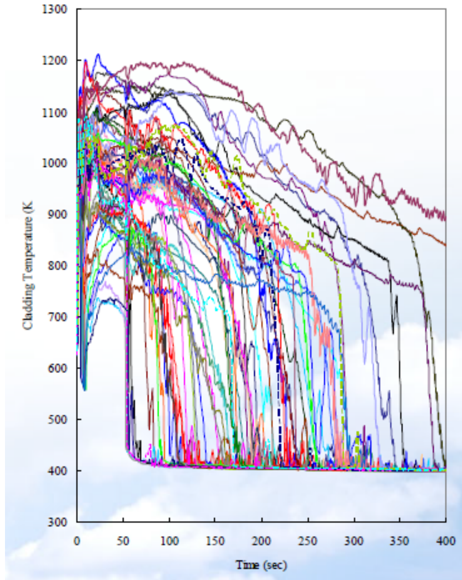
- Conservative analysis assumptions
- Required and acceptable features
  - Sources of heat during the LOCA (102% power, initial stored energy, fission heat, metal-water reaction rate)
  - Swelling and rupture of the cladding and fuel rod thermal parameters (Thermal/Elastic/Plastic Strain, Rod Burst)
  - Blowdown phenomena (Break characteristics, CHF HTC, Post-CHF HTC)
  - Post-blowdown phenomena (Single Failure, containment P, steamwater Reaction)

Evaluation Model (Conservative EM)

- KREM (KEPRI Realistic Evaluation Methodology)
  - Realistic evaluation method for LBLOCA of a PWR in accordance with KINS Technical Guidance
    - Developed following the philosophy of USNRC's CSAU (Code Scaling, Applicability, and Uncertainty)
    - Composed of 3-elements and 14 steps
    - Use RELAP5/MOD3.1K and CONTEMPT4/MOD5
  - Adopt the non-parametric statistical method to quantify the overall uncertainty of a LBLOCA at 95% probability and 95% confidence level
    - Add the Experimental Data Covering (EDC) to confirm the validity of code uncertainty parameters
  - Approved for a 3-loop Westinghouse plant in Korea initially and extended for UPI plant (Kori 1), DVI plant (APR1400) and OPR1000

Flow Chart of KREM & PCT





Thermal-Hydraulic Behaviour

① Blowdown Phase

- Period from break initiation until start of ECC injection (25~ 30 sec. after break)

② Refill Phase

- Period starting with ECC liquid injection to refill the bottom line of core

③ Reflood Phase

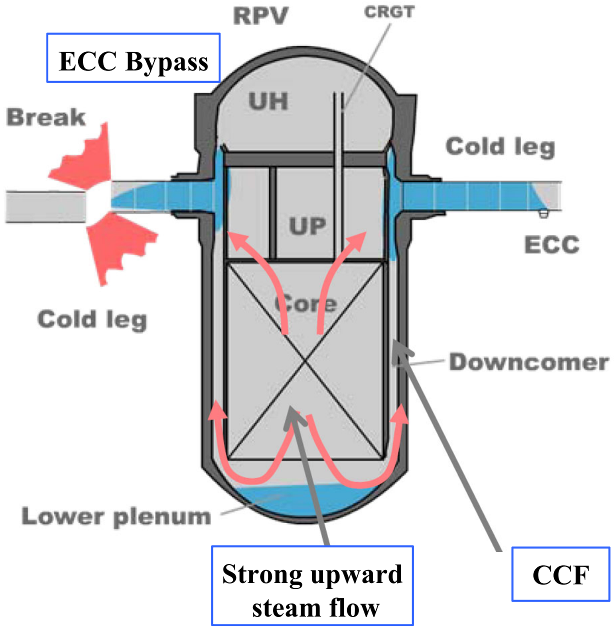
- Period beginning to refill after the downcomer and lower plenum have filled

④ Long-Term Cooling Phase

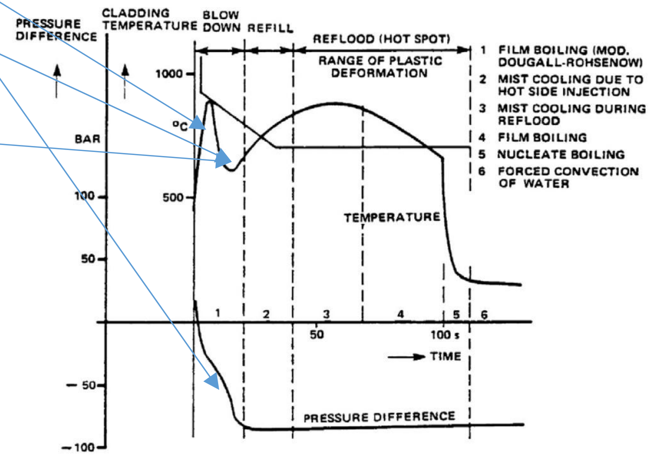
- After core quenching

① Phenomena during Blowdown Phase

- ECCS Bypass to Break (Large portion of SIT water bypassed)
  - High break flowrates of subcooled liquid from cold-leg (Choked flow)
- Nucleate boiling and flashing due to rapid depressurization
- Critical heat flux (CHF) occurring despite core voiding and reduced power
- Rapid cladding temperature increase due to initial stored energy and core uncovery (Blowdown PCT)
- Reactor trip resulting from low pressurizer pressure (generation of safety injection signal), RCS pump trip and accumulator injection to mitigate the consequence of LOCA



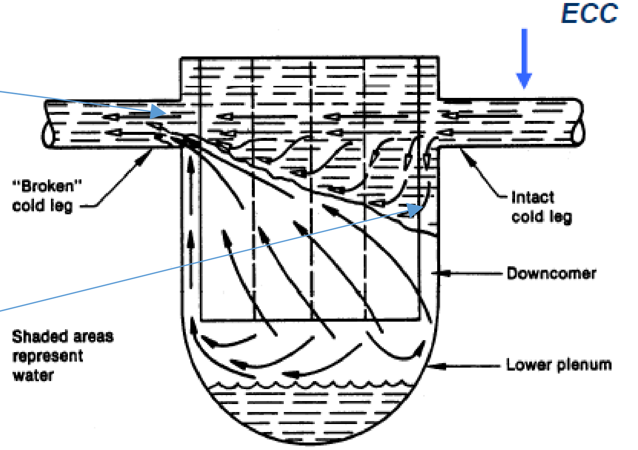
Double ended cold leg break, pressure difference across the cladding and cladding temperature at the hot spot



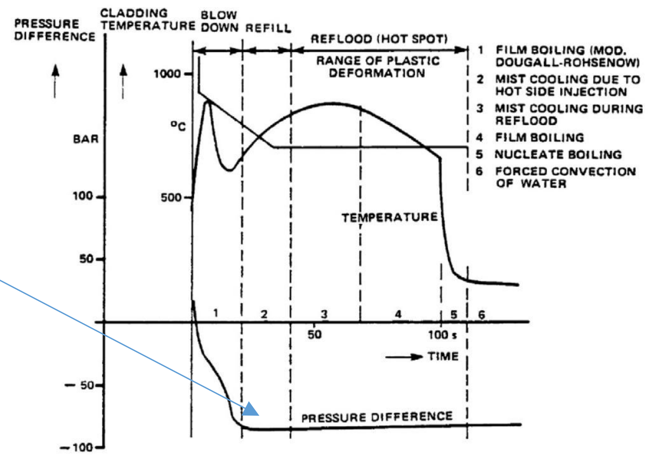
② Important Phenomena during Refill Phase

- ECC water mixing with steam in cold leg
- ECC water flowing into the downcomer bypasses through the break by escaping upward steam flow
- ECC liquid has penetrated into the lower plenum despite the sweep-out
- Decreased depressurization as the difference between vessel and containment pressures decreases
- The ECC refill period ends when the liquid level in the lower plenum reaches the bottom of the core barrel (BOCREC)

ECC Bypass

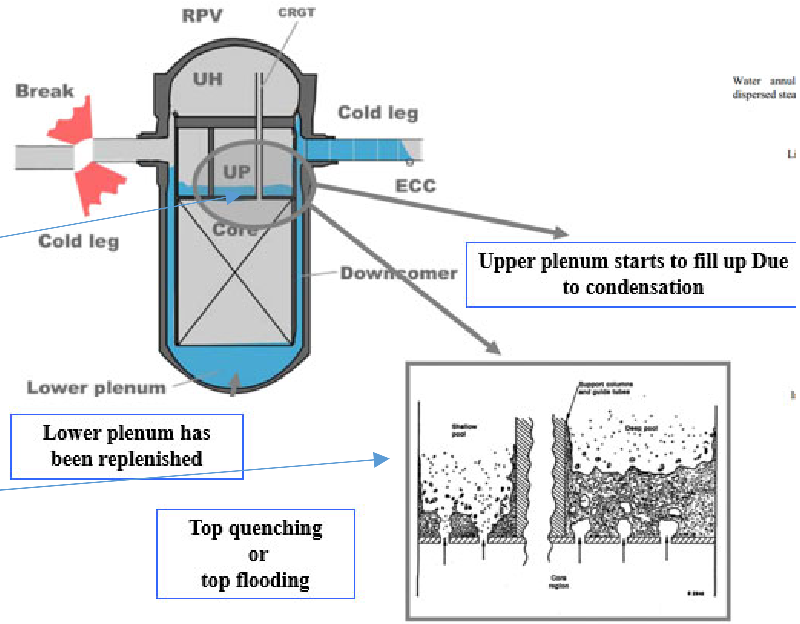


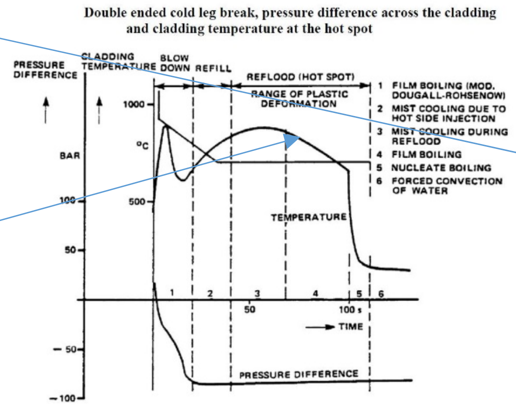
Double ended cold leg break, pressure difference across the cladding and cladding temperature at the hot spot

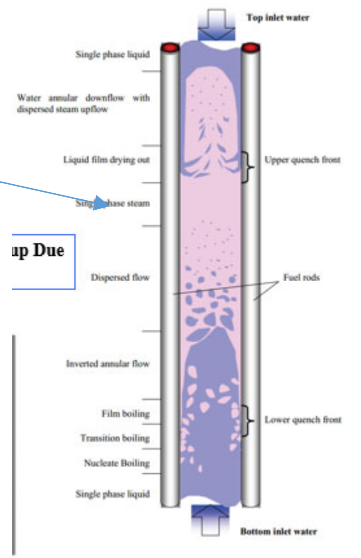


③ Important Phenomena during Reflood Phase

- Forming two-phase flow regimes in the core due to high temperature of fuel rods
- Core refill is rapidly due to the ECC injection
- Top-down quenching due to droplet de-entrainment at the tie plate and grid spacers, and liquid entrainment in the central core due to high temp. fuel rods
- Forming a two-phase pool in upper plenum by some of de-entrained liquid
- As the bottom quench progresses upward, liquid carried over by vapor. It causes an increase of pressure in SG and upper plenum, reducing the reflood rate (steam binding effect)
- Decreased PCT due to the continued safety injection by pump and core cooling



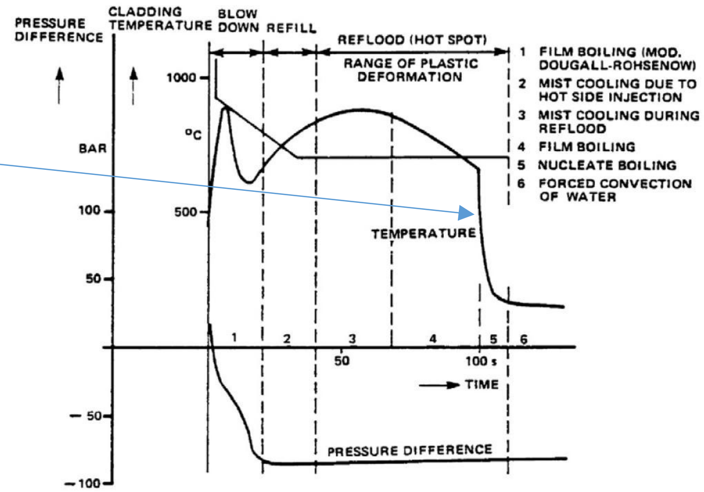




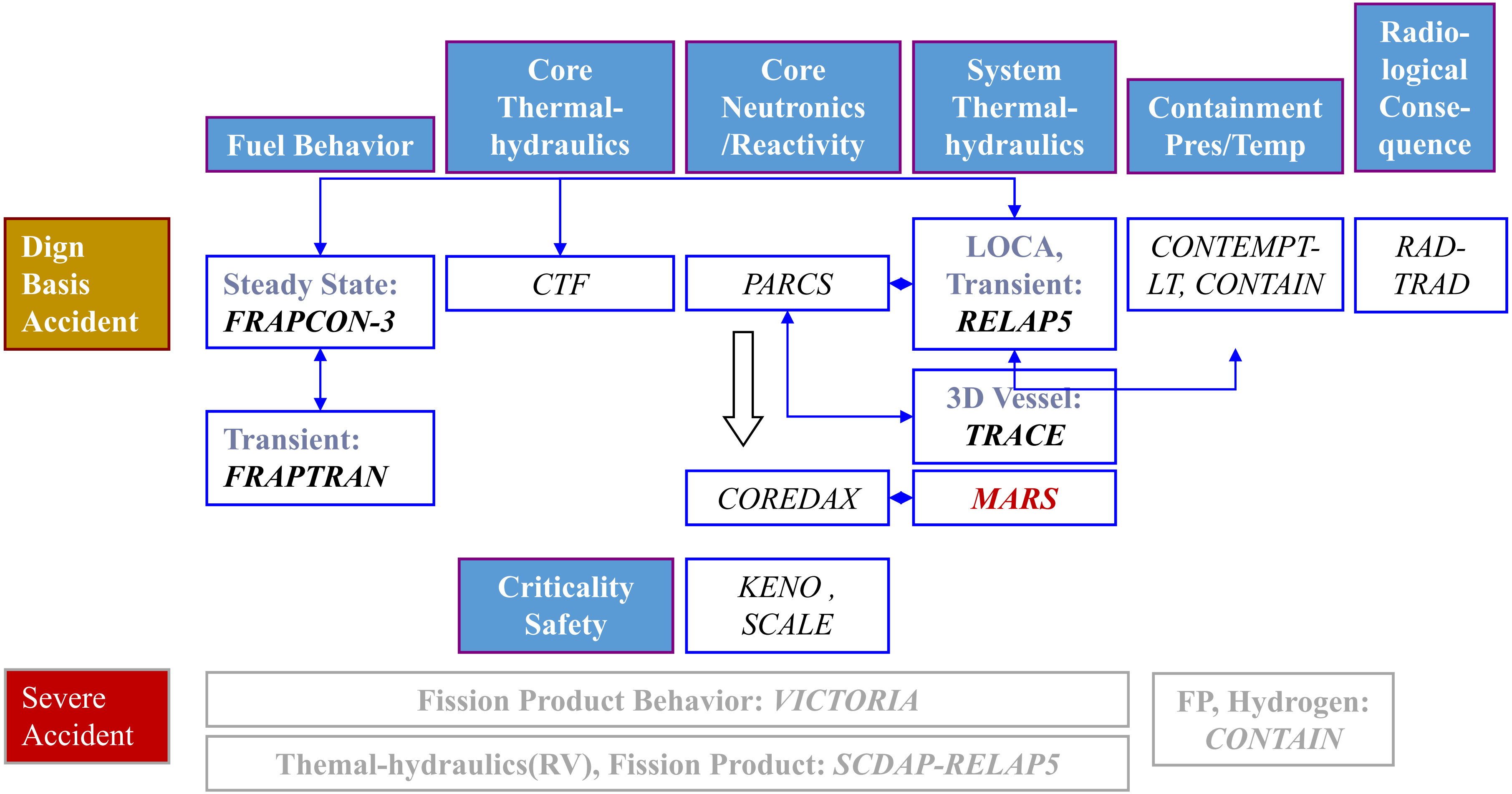
④ Long-Term Cooling Phase

- After Core Quenching, to remove the core decay heat and maintain the core at low temperature, water continuously provided by SIP
- Switchover of water source from RWT to Containment recirculation sump
- Long term cooling via Heat Exchangers of Shutdown Cooling System or Containment Spray System

Double ended cold leg break, pressure difference across the cladding and cladding temperature at the hot spot



(3-7) Regulatory Auditing Calculation



4. Application of deterministic safety analysis

Areas of application

- **Deterministic safety analyses** should be carried out for the **following areas**:
  - Design of nuclear power plants.
    - Such analyses require either a conservative approach or a best estimate analysis together with an evaluation of uncertainties.
  - Production of new or revised safety analysis reports for licensing purposes, including obtaining the approval of the regulatory body for modifications to a plant and to plant operation.
    - For such applications, in many countries, but not all, conservative approaches and best estimate plus uncertainty methods may be used.
  - The assessment by the regulatory body of safety analysis reports.
    - For such applications, in many countries, but not all, conservative approaches and best estimate plus uncertainty methods may be used.

Areas of application

- The analysis of incidents that have occurred or of combinations of such incidents with other hypothetical faults.
    - Such analyses would normally require best estimate methods, in particular for complex occurrences that require a realistic simulation.
- The development and maintenance of emergency operating procedures and accident management procedures.
    - Best estimate codes together with realistic assumptions should be used in these cases.
- The refinement of previous safety analyses in the context of a periodic safety review to provide assurance that the original assessments and conclusions are still valid.
    - As for the original analyses, both, conservative approaches and best estimate plus uncertainty methods may be used.
- By the Regulatory Body to provide independent oversight of licensee activities.

(4-1) The design of nuclear power plants

- The **design basis** for items that are important to safety is required to be established and confirmed by means of a comprehensive safety assessment.
- The design basis comprises the **design requirements** for structures, systems and components that must be met for the safe operation of a nuclear power plant, and for preventing or mitigating the consequences of events that could jeopardise safety.
- For example, deterministic analyses are carried out to determine what pressure and temperature the components of the primary coolant system must be able to withstand.

(4-2) The licensing of nuclear power plants

- The use of deterministic safety analyses to develop the design, and to **license a nuclear power plant,** are closely related.
- The plant must be designed so that it complies with **all the applicable regulations and standards** and this must be demonstrated in safety analysis reports in order to obtain licenses to construct and operate the plant.
- The analyses that are presented in the safety analyses reports should represent the current state of the design and should be presented in a way that demonstrates to the **regulatory body** that its **requirements** have been **met.**

(4-3) The assessment of safety analysis reports

- The **operating organisation** shall ensure that an **independent verification of the safety assessment** is performed by individuals or groups separate from those carrying out the design, before the design is submitted to the regulatory body.
- **Additional independent analyses** of selected aspects may also be carried out by or on behalf of the **regulatory body.**

(4-4) Application in plant modifications

- The **modification** of **existing** nuclear power **plants** is normally undertaken
  - to counteract the ageing of the plant,
  - to justify its continued operation,
  - to take advantage of developments in technology or
  - to comply with changes to the applicable rules and regulations.

(4-4) Application in plant modifications

- To comply with the **regulatory requirements**, a **revision of the safety analysis** of the plant design should be made
  - when major modifications or modernization programmes are implemented,
  - when advances in technical knowledge and understanding of physical phenomena are made,
  - when changes in the described plant configuration are implemented or
  - when changes are made in operating procedures owing to operational experience.

(4-4) Application in plant modifications

- Other **important applications** of deterministic safety analysis are aimed at the more **economical utilization** of the reactor and the nuclear fuel.
- Such applications encompass
  - up-rating of the reactor power,
  - the use of improved types of fuel and
  - the use of innovative methods for core reloads.
- Such applications often imply that the **safety margins** to operating limits are **reduced** and **special care** should be taken to ensure that the limits are not exceeded.

(4-5) Analysis of operational events

- The analysis of **actual events** that have occurred on operating nuclear power plants are a very important way of establishing the extent to which the deterministic analysis that has been performed **accurately represents** the **behaviour of the plant.**
- Such analyses should form an integral part of the **feedback** from operational experience.

(4-5) Application to the analysis of operational events

- **Operational events** may be analysed with the **following objectives:**
  - To check the adequacy of the selection of postulated initiating events;
  - To determine whether the transients that have been analysed in the safety analysis report bound the event;
  - To provide additional information on the time dependence of the values of parameters that are not directly observable using the plant instrumentation;
  - To check whether the plant operators and plant systems performed as intended;

(4-5) Analysis of operational events

- **Operational events** may also be analysed with the **following objectives:**
  - To check and review emergency operating procedures;
  - To identify any new safety issues and questions arising from the analyses;
  - To support the resolution of potential safety issues that are identified in the analysis of an event;
  - To analyse the severity of possible consequences in the event of additional failures (such as severe accident precursors);
  - To validate and adjust the models in the computer codes that are used for analyses and in training simulators.
- The analysis of operational events requires the use of a **best estimate approach**. **Actual plant data** should be used. If there is a lack of detailed information on the plant status, **sensitivity studies**, with the variation of certain parameters, should be performed.

(4-6) Development and validation of emergency operating procedures (EOPs)

- Best estimate deterministic safety analyses should be performed to confirm the **strategies** that have been developed to restore normal operational conditions at the plant following transients due to **anticipated operational occurrences and design basis accidents.**
- These strategies are reflected in the **emergency operating procedures** that define the actions that should be taken during such events.
- After the emergency operating procedures have been developed, a **validation analysis** should be performed.
- This analysis is usually performed by using a **qualified simulator**.

(4-6) Development of severe accidents management guidelines (SAMGs)

- Deterministic safety analyses should also be performed to assist the development of the **strategy** that an operator should follow if the emergency operating procedures fail to prevent a **severe accident** from occurring.
- The analyses should be carried out by using one or more of the **specialized computer codes** that are available to model relevant physical phenomena.

(4-6) Development of severe accidents management guidelines (SAMGs)

- For **light water reactors**, these include
  - thermo-hydraulic effects,
  - heating and melting of the reactor core,
  - the retention of the molten core in the lower plenum,
  - molten-core–concrete interactions,
  - steam explosions,
  - hydrogen generation and combustion, and
  - fission product behaviour.

(4-7) Periodic safety reviews

- New deterministic analyses may be required to refine previous safety analyses in the context of a **periodic safety review** to provide assurance that the original assessments and conclusions are still valid.
- In such analyses, account should be taken of any **margins** that may have become **reduced** and continue to be reduced owing to ageing over the period under consideration.
- **Best estimate analyses** together with an **evaluation of the uncertainties** may be appropriate to demonstrate that the remaining margins are adequate.

5. Summary

Summary (Recapping)

- Needs for Safety Analysis
  1. Introduction of Safety Analysis: DSA
     1) Purpose
     2) Safety Requirements
     3) Analysis in the Plate Sates
  2. Safety analysis approaches
     4) Classification of Initiating Events
     5) Overview of Deterministic Safety Analysis
     6) Acceptance Criteria for DBAs
     7) Verification and Validation of Codes
  3. ECCS and LOCA Analysis
- Current Trend of Safety Analysis
  - **Best estimate analysis** with uncertainty quantification (Mainly on LOCA and also Non-LOCA)
  - **Code model** to incorporate **the advanced design** (e.g. passive system and component)
  - **Computational Fluid Dynamic (CFD)** for technical issues (Mixing, CHF, Thermal Stratification in Pipe, Containment Flow Field, etc.)

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- IAEA, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, No. NS-G-2.2, Vienna 2000