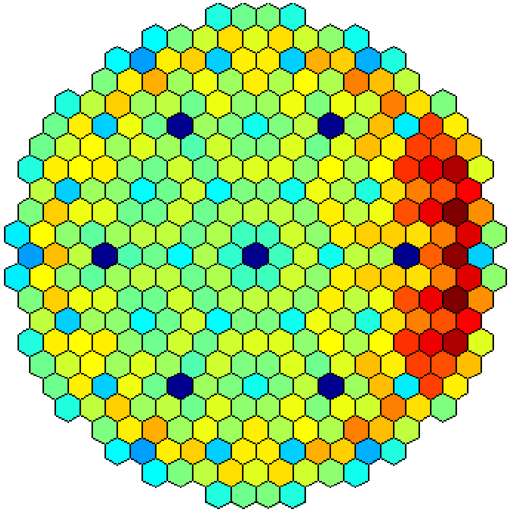

# Safety analyses and reactor dynamics

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16.5.2019
PHYS-E0562 - Nuclear Engineering, advanced course, Aalto University

# Outline

- Regulations
- Classification of transients and accidents

- VTT's reactor analysis code system
- Methods and assumptions for safety analyses
- Results of interest

- Description of transients and accidents

# Classification of transients and accidents

# Classification of transients and accidents

- Background
- Classification of events
- Requirements for various classes
- Difference to regulations used in other countries

Aim is to introduce how unwelcome events in NPP are classified and to give understanding what kind of requirements are set for safety analysis.

Lecture is based on Finnish regulations

# Background

**Government Decree 717/2013:**

*the safety of a nuclear power plant shall be assessed when*

*- applying for a construction license*

*- applying for an operating license*

*- in connection with plant modifications*

*- at regular intervals during the operation of the plant.*

*It shall be demonstrated that the nuclear power plant has been designed and implemented in a manner that meets the safety requirements. The safety assessment shall cover* ***all the nuclear power plant states.***

# Background

- New regulatory guides on nuclear safety (YVL Guides) compiled by STUK came into effect on December 2013.
- Guides are designed for light water reactors (normal size reactors)
- YVL Guide B.3 Deterministic safety analyses for a nuclear power plant

> Analysis shall cover the nuclear power plant's normal operational states, anticipated operational occurrences, postulated accidents, design extension conditions and severe reactor accidents.

- Normal operational states: reactor at full or partial power, hot standby, cold standby

# Classification of events

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- Transients and accidents are classified mainly according their expected frequency
- Acceptance criteria and assumptions used in deterministic safety analysis depend on class of event
  - More strict requirements for more probable events

**Regulations in Finland:**

| Class | Frequency |
|---|---|
| Anticipated operational occurence (DBC2) | > $10^{-2}$ /year |
| Postulated accidents, Class 1 (DBC3) | $10^{-3}$ /year < f < $10^{-2}$ year |
| Postulated accidents, Class 2 (DBC4) | < $10^{-3}$ /year |
| Design extension conditions (DEC) | |
| Severe accidents | < $10^{-5}$ /year |

Classification based on consequences

**U.S. NRC regulations:**

| Class | Description |
|---|---|
| Anticipated operational occurences | Conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit |
| Design basis accident | A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety. |
| Beyond design-basis accidents | Possible but were not fully considered in the design process because they were judged to be too unlikely. *New regulations also for this class* |

**Unofficial American Nuclear Society (ANS) standards**

| Class | Description |
|---|---|
| Condition I | Normal operation |
| Condition II | Incidents of moderate frequency |
| Condition III | Infrequent events |
| Condition IV | Limiting faults |

# General

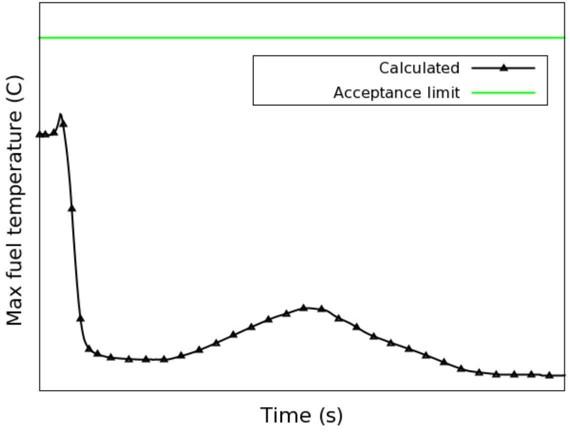
- Transients and accidents have not been explicitly listed in YVL guides
  - Reference to the IAEA reports were examples of the events to be analyzed are given
  - The scope of the analyzed events shall provide **a comprehensive assessment** of the nuclear power plant’s behavior during incidents and accidents as well as releases and doses due to incidents and accidents.
  - The inadvertent actuation of every system accomplishing a safety function shall be addressed as an initiating event.

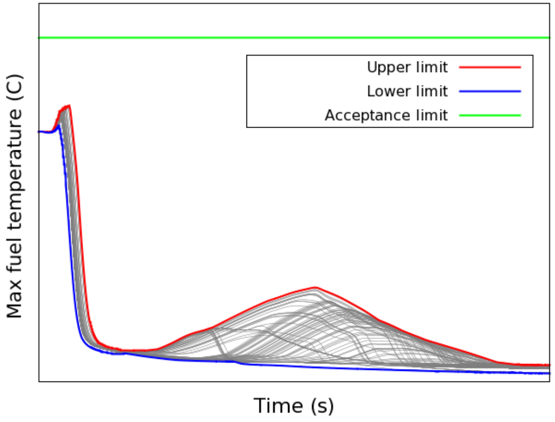
**“Limiting cases”**

Analyses shall cover anticipated operational occurrences and accidents **that determine or limit** the dimensioning of systems accomplishing safety functions.

# General requirements

- Acceptance criteria shown in following slides are for conservative analyses
  - Conservative analyses have to be supplemented with sensitivity analyses
- Also Best estimate + Uncertainty analysis is possible
  - result is acceptable if there is a 95% probability with 95% confidence that the examined parameter will not exceed the acceptance limit set for the conservative analysis method.



# General requirements

- For DBC2, DBC3, DBC4 and DEC events:

  It shall be shown that:
  - Reactor can be shut down
  - Reactor can be maintained in shutdown state
  - Plant can  be brought to controlled state and after that to a safe state
  - In the long term, plant can be brought to such a state that fuel can be removed from the reactor

# Safety classification

- YVL B.2:
  The nuclear facility’s systems, structures and components shall be grouped into the Safety Classes 1, 2, and 3 and Class EYT ( non-nuclear safety)

| Class 1 | Class 2 | Class 3 | EYT/STUK |
| --- | --- | --- | --- |
| •Structures and components whose failure would threat shutdown or coolability of reactor and requires immediate actuation of safety functions<br>•RPV<br>•Fuel<br>•Primary circuit | •Systems that are designed against postulated accidents<br>•main components and piping of the emergency core cooling system<br>•structures of the core support and reactor shutdown system<br>•primary circuit piping supports and brackets<br>•the reactor containment including structures relating to the containment isolation function<br>•fuel storage racks (risk of criticality accident) | •Systems that are designed<br>• to bring the facility into a safe state over a long period of time<br>•for severe reactor accident management<br>•to ensure the bringing of the facility into a controlled state in case of the failure of systems primarily taking care of a corresponding safety function<br>• to mitigate the consequences of AOOs unless they are assigned to a higher safety class for some other reason<br>•Main controllers of the NPP<br>•Systems that contribute to fuel handling or lifting of heavy loads and whose failure may damage structures important to safety or cause fuel failure<br>•Buildings and structures ensuring the operability and physical separation of Safety Class 2 systems | •Systems that protect safety functions against internal or external threats (e.g. fire protection systems)<br>•Systems that monitor the radiation, surface contamination or radioactivity of the plant, instruments, workers or the environment .<br>•e.g. Small pipes connected to class 3 systems and supports of Class 3 pipes<br>•Systems that are necessary for bringing the facility to a  controlled state in case of DEC B or DEC C |

## Odotettavissa oleva  käyttöhäiriö

# Anticipated operational occurrences AOO DBC2

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**Assumptions:**

Shall be analysed in two ways:

1. All plant systems operate according to design, except the initiating event and its consequences.
2. Conservative way
   - Actuation of non-safety classified systems
     - shall not be postulated as systems mitigating the consequences of the initiating event
     - shall be postulated if a system's designed operation could aggravate the consequences of the initiating event.
   - The most penalising failure shall be postulated in safety class 2 or 3 systems designed for AOOs or postulated accidents.
     - More about failure criteria in YVL guide B.1
   - Performance values for functioning components shall be chosen conforming to the acceptance limits in periodic tests.

# Anticipated operational occurrences DBC2

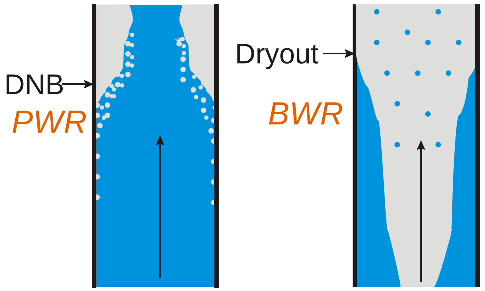
## Acceptance criteria:

- Must not require the initiation of safety systems designed for postulated accidents
- Pressure < design pressure
  - not a single safety valve opens

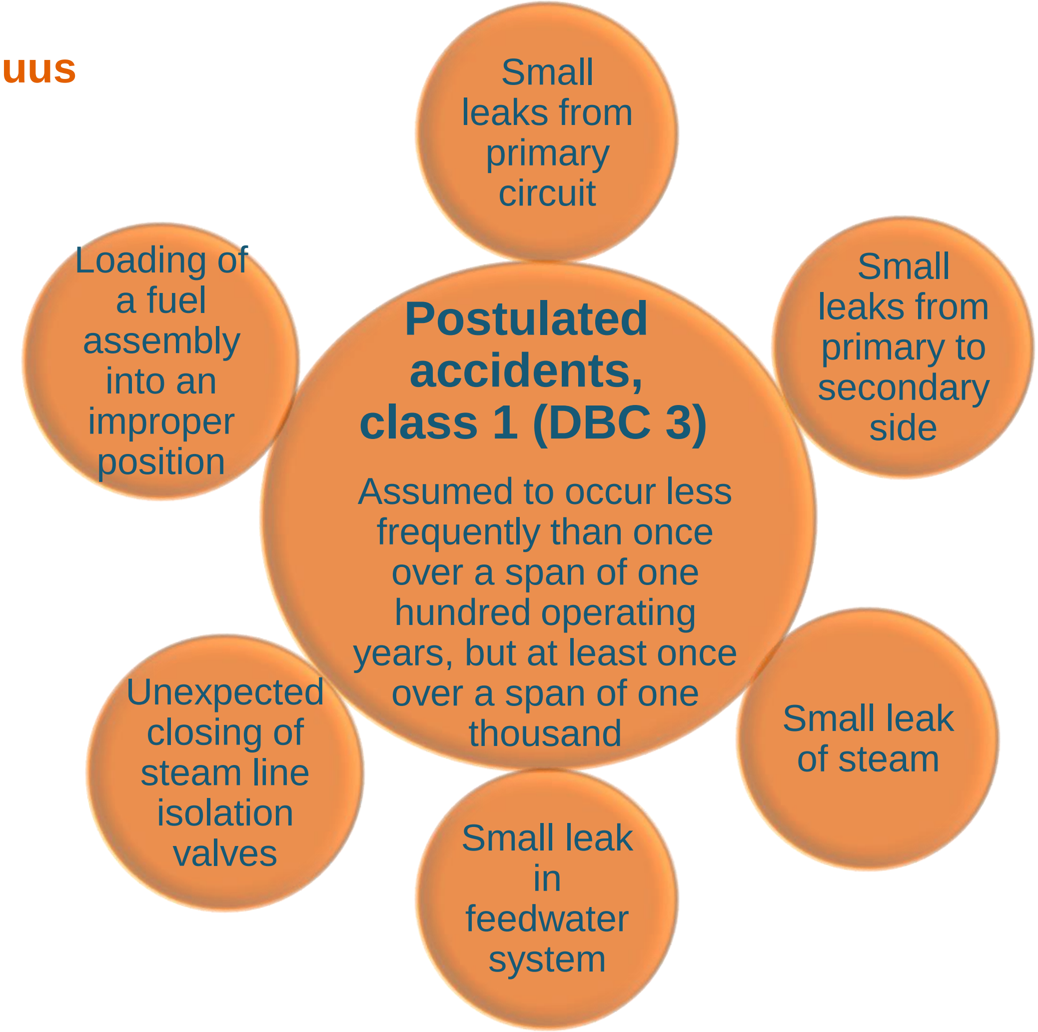
Designed for accidents

- Fuel
  - No melting in fuel pellet
  - Adequate cooling of the cladding shall be ensured
    - 95% probability at 95% confidence level that the hottest fuel rod does not reach heat transfer crisis
    - Or it may be demonstrated that the number of rods reaching heat transfer crisis does not exceed 0.1% of the total number of fuel rods in the reactor.
  - The probability of fuel failure caused by mechanical interaction between fuel and cladding (PCMI) shall be extremely low
- The dose (external+internal) of the individual of the population due to the event max. 0.1 mSv/year

*Heat transfer crisis:*



# Oletettu onnettomuus



# Postulated accidents, class 1 (DBC 3)

**Assumptions**:

- Safety-classified systems shall be assumed to operate at their minimum system performance
- Actuation of non-safety classified systems
  - shall not be postulated as systems mitigating the consequences of the accident
  - shall be postulated if a system's designed operation could aggravate the consequences of the initiating event.
- Loss of the external grid shall be combined with postulated accidents if it could aggravate the consequences of the initiating event.

# Postulated accidents, class 1 (DBC3)

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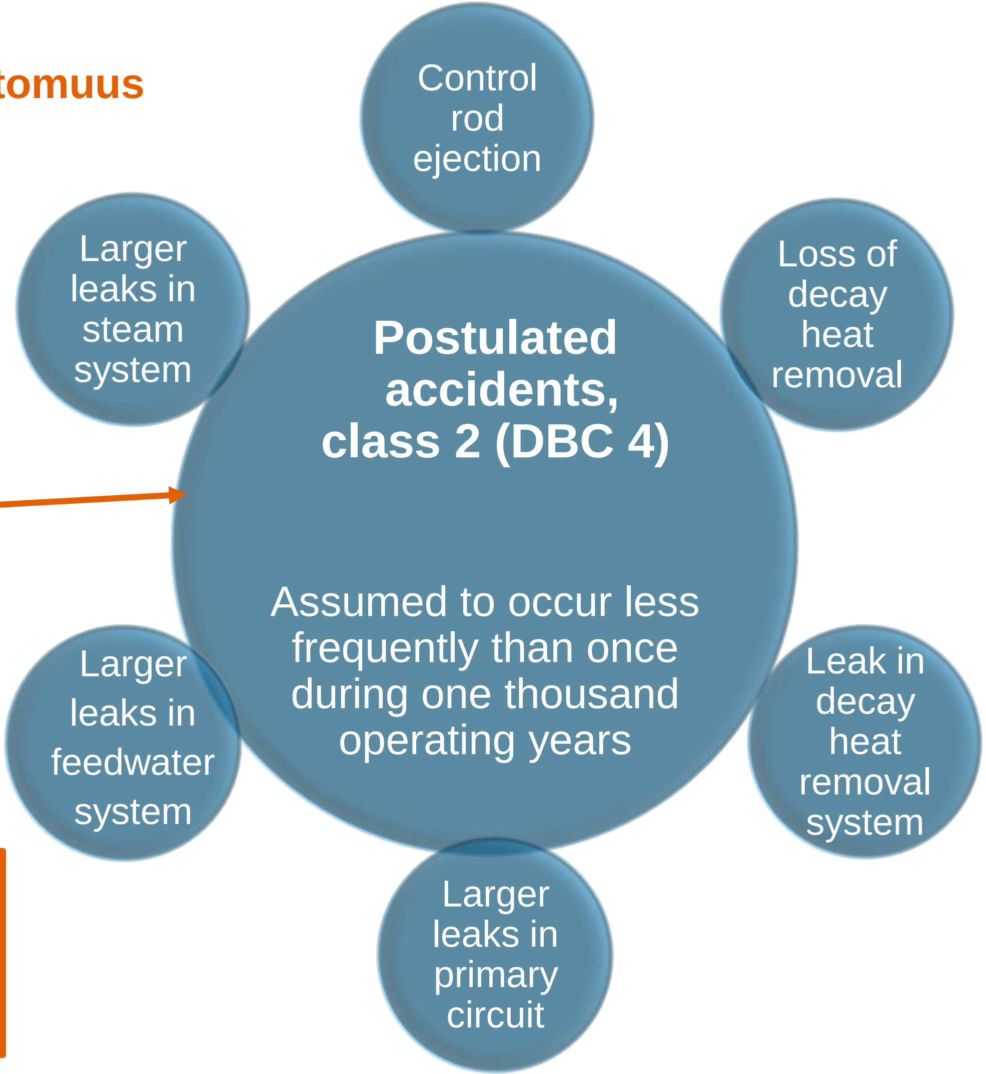
**Acceptance criteria:**

- Fuel
  - Max 1% of fuel rods reaches heat transfer crisis
  - Cladding temperature such that the integrity of the cladding is not endangered during an accident due to oxidation or changes in the cladding material properties
    - Max cladding temperature 650°C or separate justification
  - PCMI failure in < 1% of fuel rods
- Pressure < 1.1*design pressure
- 1.1*Containment pressure < design pressure
- Max. dose 1 mSv/year

# Oletettu onnettomuus




**Design of several safety systems is based on these DBC4 accidents**

# Postulated accidents, class 2 (DBC 4)

**Assumptions:**

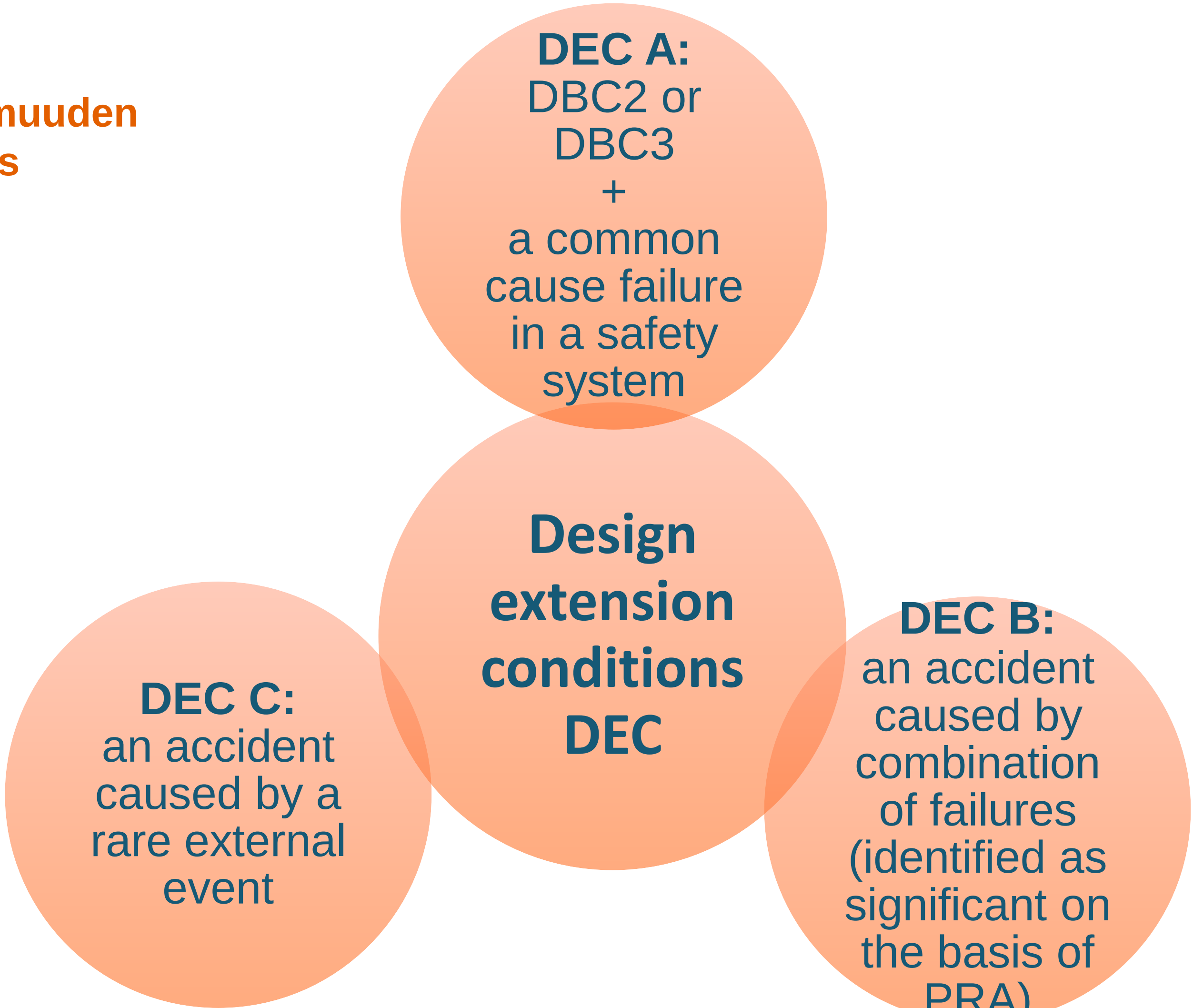
- Mainly as in DBC3 accidents
- Only safety class 2 systems may be assumed to be systems mitigating the accident from the initiating event to the controlled state.
- Operation of systems in lower safety classes shall be postulated if a system’s designed operation could aggravate the consequences of the initiating event.

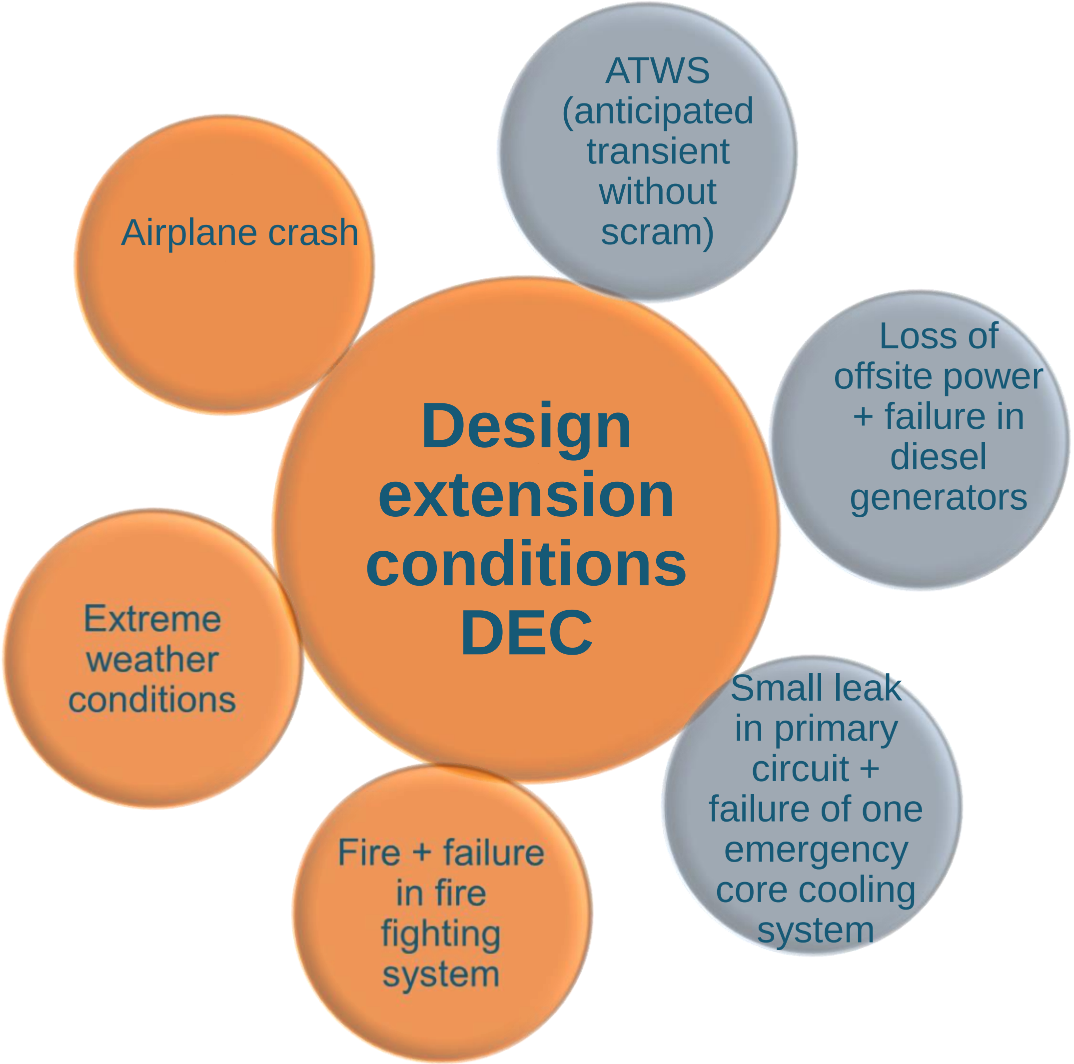
# Postulated accidents, Class 2 (DBC 4)

**Acceptance criteria:**

- Fuel
  - Max 10% of fuel rods reaches heat transfer crisis
  - No excessive embrittlement of the cladding
    - Cladding temperature <1200 °C
    - Oxidation such that fuel can withstand loads caused by accident and by the handling, transport and storage after an accident
  - Hydrogen generation due to the chemical reaction between coolant and cladding < 1%
  - Maximum enthalpy of fuel 963 J/g$UO_2$ to prevent fragmentation and melting of fuel pellets
  - No melting in control rods; structural deformations in fuel rods, control rods and other reactor internals such that control rods can still move
- Pressure < 1.2 design pressure
- 1.1*Containment pressure < design pressure
- Max. dose 5 mSv/year

## Oletetun onnettomuuden laajennus




Design extension conditions DEC
Airplane crash
ATWS (anticipated transient without scram)
Loss of offsite power + failure in diesel generators
Small leak in primary circuit + failure of one emergency core cooling system
Fire + failure in fire fighting system
Extreme weather conditions

# Design extension conditions DEC

**Assumptions:**

- For DEC A accidents, the most penalizing single failure shall be assumed in one of the systems whose operation is required to accomplish a safety function in the event in question.
- For DEC B and C accidents, a single failure need not be assumed
- Loss of the external grid assumed only if it is the likely consequence of an initiating event.
- Best estimate methods
  - for the plant's initial state
  - For the performance of operating subsystems
  - Statistical uncertainty analysis not needed

# Design extension conditions

**Acceptance criteria:**

- Fuel
  - As in DBC4
  - No limitation for number of failed fuel rods
- Pressure < 1.2 design pressure
- Max. dose 20 mSv/year

# Severe accidents
# Vakavat reaktorionnettomuudet

- Considerable part of the fuel in a reactor loses its original structure
- Frequency < $10^{-5}$ /year

Remark: Terms Severe accident and Design extension condition may have different meaning in different regulations. For example in some WENRA reports:

DEC A ≙ DEC A+B+C

DEC B ≙ Severe accidents

# Severe accidents

## Acceptance criteria:

- No such radioactive release that require for extensive civil defence operations
- No long-term restriction on use of land and water areas
  - $^{137}Cs$ release max. 100 TBq
- 1.5*Containment pressure (including pressure increase due to hydrogen burn) < leaktightness limit

# Severe accidents

**Acceptance criteria:**

- No such radioactive release that require for extensive civil defence operations
- No long-term restriction on use of land and water areas
  - $^{137}Cs$ release max. 100 TBq
- 1.5*Containment pressure (including pressure increase due to hydrogen burn)  < leaktightness limit

# Summary of some acceptance criteria

## Regulations in Finland:

| Class | Frequency per year | Number of failed fuel rods DNB | PCMI | Pressure * from design pressure | Dose |
|---|---|---|---|---|---|
| DBC2 | $> 10^{-2}$ | < 0.1% | Very improbable | < 100 % | < 0,1mSv |
| DBC3 | $10^{-3} < f < 10^{-2}$ | <1 % | <0.1 % | < 110 % | < 1 mSv |
| DBC4 | $< 10^{-3}$ | < 10 % | | <120 % | < 5 mSv |
| DEC | | - | | <120 % | <20 mSv |
| Severe | $< 10^{-5}$ /year | Considerable | | - | Limit based on consequences, no exact value |

# IAEA Requirements and guides

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- Plant states
  - Operational states
    - Normal
    - Anticipated operational occurrences
  - Accident conditions
    - Design basis accidents
    - ~~Beyond design basis accidents~~
    - Design extension conditions, including core melting

Safety guide SSG-2 Deterministic Safety Analysis for Nuclear Power Plants 2009

Beyond design basis accidents

Specific Safety Requirements SSR-2/1 (Rev. 1) 2016

“Criteria shall be assigned to each plant state”

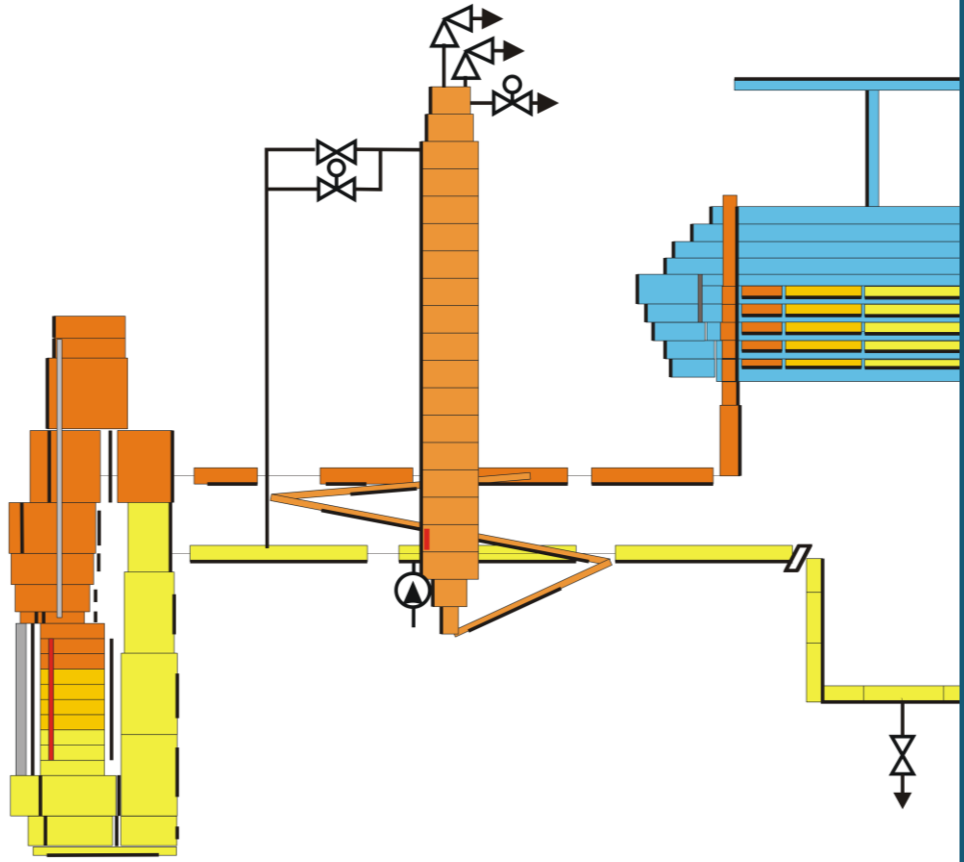
Requirements set also for design extension conditions (conditions with and without fuel melting).

Renewal process is going on.

# Further reading

- Finnish regulatory guides (YVL guides):
  http://plus.edilex.fi/stuklex/fi/lainsaadanto/luettelo/ydinvoimalaitosohjeet/
  in English:
  http://plus.edilex.fi/stuklex/en/lainsaadanto/luettelo/ydinvoimalaitosohjeet/

  B.3 Deterministic safety analyses for a nuclear power plan
  B.4 Nuclear fuel and reactor,
  B.5 Reactor coolant circuit of a nuclear power plant
  B.6 Containment of a nuclear power plant
- IAEA Safety Standards,
  - Specific Safety Requirements SSR-2/1 (Rev. 1) Safety of Nuclear Power Plants: Design
    https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1715web-46541668.pdf
  - Specific Safety Guide No. SSG-2, Deterministic Safety Analysis for Nuclear Power Plants
    http://www-pub.iaea.org/MTCD/publications/PDF/Pub1428_web.pdf



# Reactor dynamics modelling

# Overview

- Methods for safety analyses
- Tools for 3D transient and accident modelling
- Typical results
- Other steps of safety analyses
  - Hot channel modelling
  - Heat transfer crisis

# Background

Is cooling of fuel rods or the integrity of reactor threatened during transients and accidents?

Are fuel rods in or near heat transfer crisis?

Fuel enthalpy and temperature?

Cladding temperature?

Maximum pressure in primary circuit?

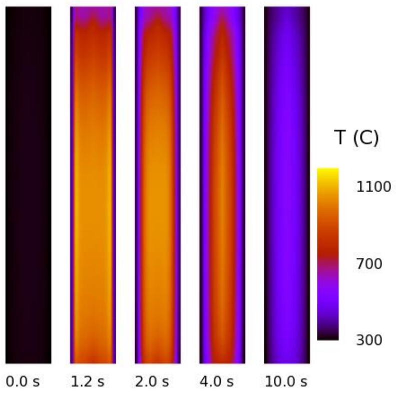
Maximum pressure in secondary circuit?

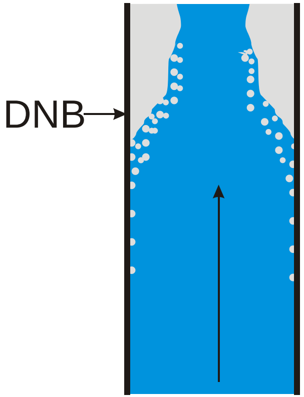
Maximum pressure in containment?

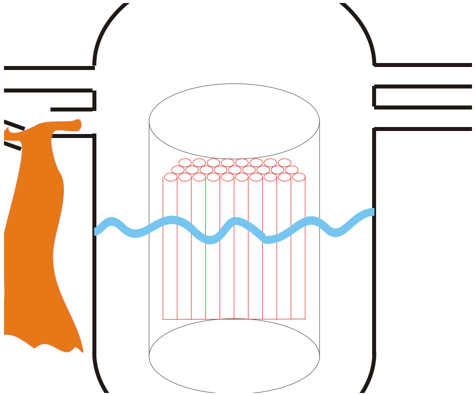
Possible dose ?

# Background

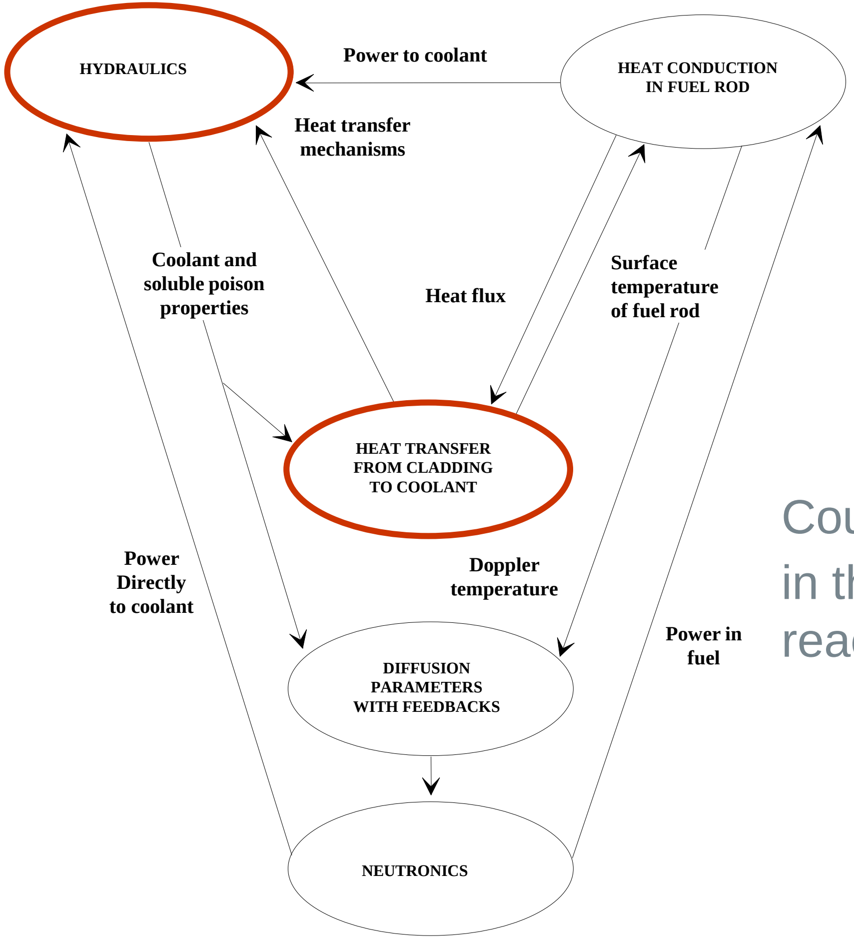
- Basically cooling of the fuel pins can be threatened from three different reasons:
  - Power increases so fast that heat cannot transfer from pellet to coolant
  - Heat flux from fuel rod to coolant is too high compared to coolant flow, and surface of the fuel rod dry
  - There’s not enough water in a reactor due to leaks



# Reactor dynamics




Coupling of physical processes in the core of a light water reactor

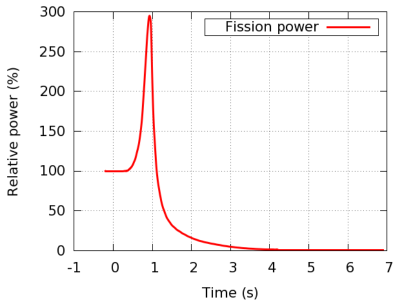
# Reactor dynamics – transient analysis

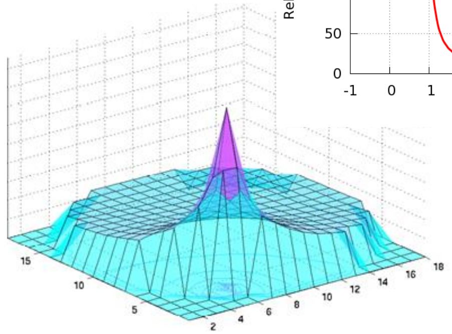
- Events in which fission power development is important and its spatial distribution changes during the transient

- Aim is to analyse the phenomena in a reactor core and also in primary and secondary circuit during transients
- Models for the phenomena in the core are tightly coupled and solved together in an iterative process

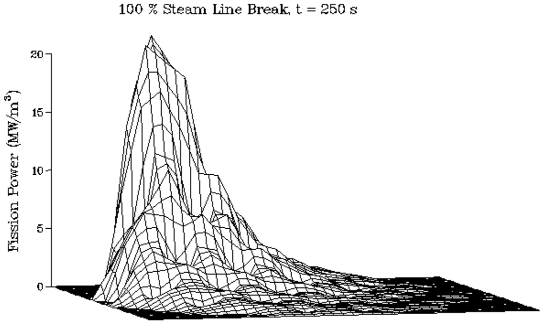
- Modelled period typically  from 10 seconds to 2 hours

- Conservative reactivity properties in safety applications
- Best estimate in validation



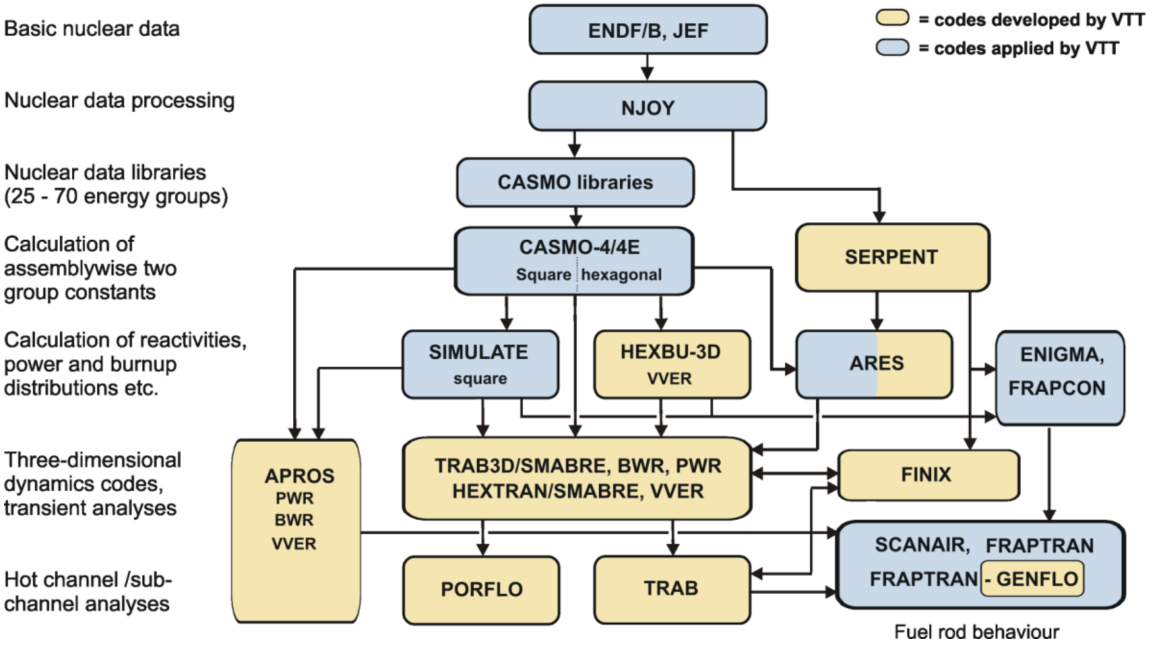



# Reactor dynamics simulation – typical results

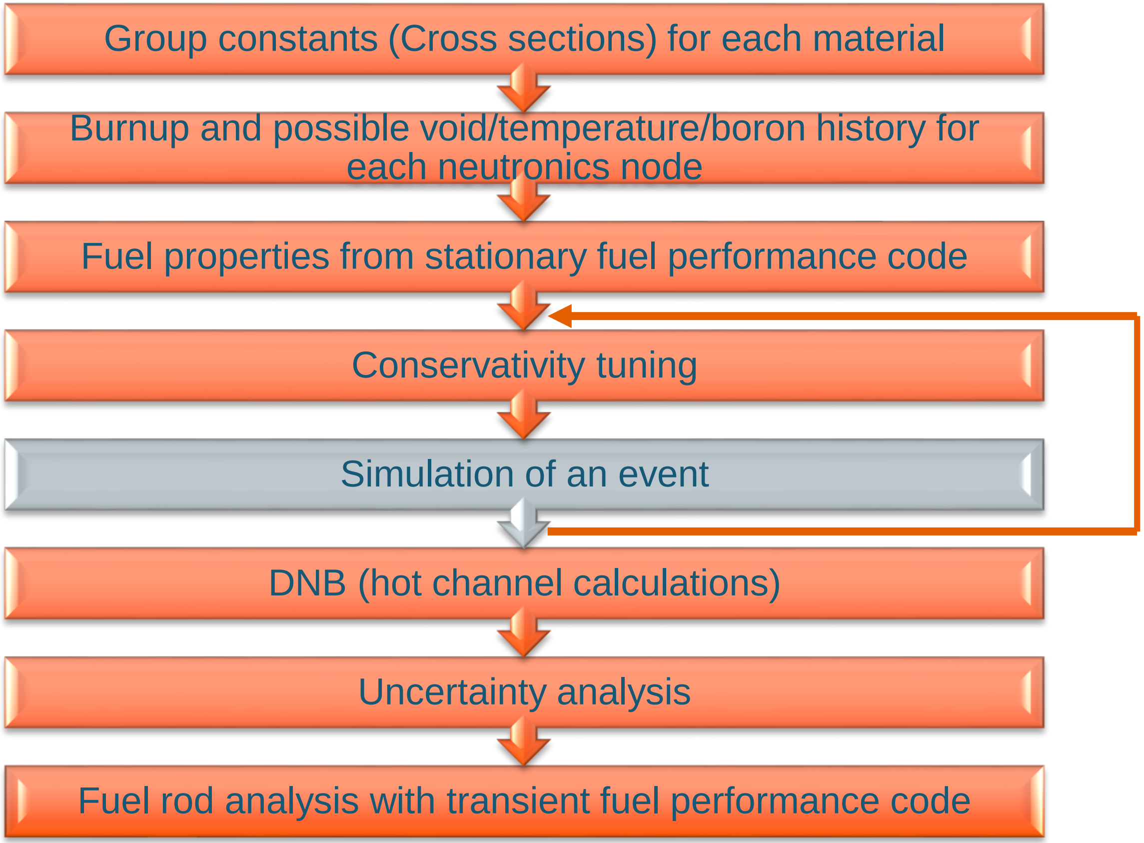



- Maximum fission power (global, local)
- Maximum pressure
- Possible activation of a protection system
- Possible recriticality, time of
- Boric acid concentration
- Maximum of linear power in a fuel rod
- Minimun critical heat flux (CHF) or dry-out margin (DNB)
- Maximum temperature in a fuel rod
- Maximum enthalpy in a fuel rod

# VTT's Reactor Analysis Code system



# Steps of the reactor dynamical safety analyses



# VTT's reactor dynamics code system: Core modelling

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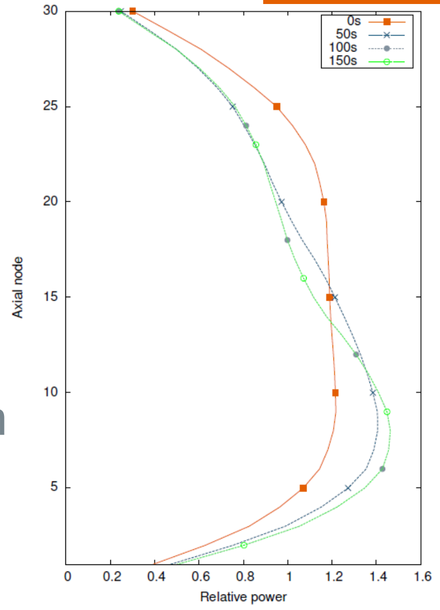
- Hexagonal HEXTRAN or rectangular TRAB3D
- 3D neutron kinetics with nodal two-group diffusion equations
- 1D thermal hydraulics of separated core channels with four conservation equations
  - Liquid mass, steam mass, mixture momentum, mixture energy
- One-dimensional cylindrical heat transfer in fuel rods, solution according to Fourier's law with finite element method
- Implicit time-discretization methods allow flexible time-step choices

- TRAB3D includes a model for a BWR circuit
- PWR/VVER circuit modelling with the system code SMABRE



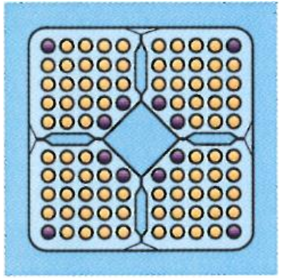

*Axial power distribution during VVER-1000 transient due to switching off one main coolant pump*

# VTT's reactor dynamic codes: core modelling

- Each fuel assembly and channel modelled separately
  - Typically 163-500 channels
- Heat transfer calculated for one average fuel rod in each assembly either with internal models or with FINIX fuel behavior module
- Axially from 20 to 30 nodes

This size (nodes ~ 10-20 cm) is optimal for current nodal method

- Radially up to 11 mesh points in a fuel pellet
- Modelling of modern fuel assemblies with
  - Axial discontinuities, e.g. Gd pellets, axially heterogeneous control rods
  - Water rods
  - Part length fuel rods



Part length rods

Westinghouse Atom brochure

# Simulation of one time step in HEXTRAN and TRAB3D

- Disturbances
- Delayed neutron and time discretization source calculations
- Delayed power calculation
- Prediction of new fission power and flux levels

**Outer iteration**

**Pressure balance iteration**

- Heat transfer in fuel
- hydraulics

- Diffusion parameters
- Coupling coefficients between nodes

Inner iteration of neutronics

- Assemblywise flux levels

Internal shape of the flux within a node is a slowly varying function of the average flux of the node and its neighbours → in inner iterations only the average values of the fundamental mode flux are solved

# Two-group diffusion equations in HEXTRAN

$$\begin{cases} \dfrac{1}{v_1(t)}\dfrac{\partial \Phi_1(\bar{r},t)}{\partial t} - D_1(t)\nabla^2\Phi_1(\bar{r},t) + [\Sigma_{a1}(t)+\Sigma_{12}(t)]\,\Phi_1(\bar{r},t) = [1-\beta]\,S_f(\bar{r},t)+S_d(\bar{r},t) \\[2ex] \dfrac{1}{v_2(t)}\dfrac{\partial \Phi_2(\bar{r},t)}{\partial t} - D_2(t)\nabla^2\Phi_2(\bar{r},t) + \Sigma_{a2}\Phi_2(\bar{r},t) = \Sigma_{12}\Phi_1(\bar{r},t) \end{cases}$$

Assumption 1:
The flux distribution can be separated in respect of the time and place:

$$\frac{\partial \Phi_i(\bar{r})}{\partial t} = \alpha_i \Phi_i(\bar{r})$$

where $i = 1$ or 2 for fast and thermal group, respectively
and $\alpha_1$ = constant at time $t_1$

$$\alpha_i = \frac{\bar{\Phi}_i^1 - \bar{\Phi}_i^0}{\Delta \bar{\Phi}_i^1}$$

where $\bar{\Phi}_i^1$ and $\bar{\Phi}_i^0$ are node average flux values at time $t_1$ and $t_0$.

Assumption 2:
The delayed neutron source has inside a node the same distribution as the prompt fission neutron source.

$$S_d(\bar{r}) = \frac{\bar{S}_d}{\bar{S}_f} S_f(\bar{r}) \qquad \text{at time } t_1$$

With these assumptions

$$\begin{cases} \nabla^2\Phi_1(\bar{r}) - \dfrac{1}{D_1}\left(\Sigma_{a1}+\Sigma_{12}+\dfrac{\alpha_1}{v_1} - \left(1-\beta+\dfrac{\bar{S}_d}{\bar{S}_f}\right)(\nu\Sigma_f)_1\right)\Phi_1(\bar{r}) + \dfrac{1}{D_1}\left(\left(1-\beta+\dfrac{\bar{S}_d}{\bar{S}_f}\right)(\nu\Sigma_f)_2\right)\Phi_2(\bar{r}) = 0 \\[2ex] \nabla^2\Phi_2(\bar{r} - \dfrac{1}{D_2}\left(\Sigma_{a2}+\dfrac{\alpha_2}{v_2}\right)\Phi_2(\bar{r}) + \dfrac{1}{D_2}\Sigma_{12}\Phi_1(\bar{r}) = 0 \end{cases}$$

# Two-group diffusion equations in HEXTRAN

VTT

General solution to these is a linear combination of two characteristic solutions or spatial modes: the fundamental mode $f_I(\bar{r})$ and the transient mode $f_{II}(\bar{r})$. The spatial shapes are determined by the Helmholz equations:

$$(\nabla^2 + B_I^2) f_I(\bar{r}) = 0$$

.

$$(\nabla^2 - B_{II}^2) f_{II}(\bar{r}) = 0$$

$B_I$ and $B_{II}$ are the characteristic bucklings of the modes and they can be determined by substitution of trial solutions of the form $\Phi_i(\bar{r}) = f(\bar{r}\phi_i$ to the equations on the previous slide. Using relation $\nabla^2 f(\bar{r}) = -B^2 f(\bar{r})$ the spatial modes can be eliminated and a matrix

equation for the constants $\phi_1$ and $\phi_2$ can be written. Characteristic bucklings can be solved by setting determinant of the matrix equaion to zero.

The fast and thermal fluxes are related to spatial mode in the following way

$$\Phi_1(\bar{r}) = f_I(\bar{r}) + R_{II} f_{II}(\bar{r})$$

$$\Phi_2(\bar{r}) = R_I f_I(\bar{r}) + f_{II}(\bar{r})$$

where

$$R_I = \frac{\Sigma_{12}}{\Sigma_{a2} + \frac{\alpha_2}{\nu_2}} \frac{1}{1 + B_I^2 L_2^2}$$

$$R_{II} = \frac{\Sigma_{a2} + \frac{\alpha_2}{\nu_2}}{\Sigma_{12}} (1 - B_{II}^2 L_2^2)$$

In HEXTRAN fundamental mode is a polynomial function and transient mode an exponential function.

# VTT's reactor dynamics codes: System code SMABRE

- Five-equation thermal hydraulics model with drift flux phase separation,
- Non-iterative solution of field equations
- Sparse matrix inversion is used for solving the pressure, void fraction and enthalpy distributions
- Fast running code - several simulator applications
- Supercritical water properties – used in HPLWR

# Coupling of neutronics and thermal hydraulic codes

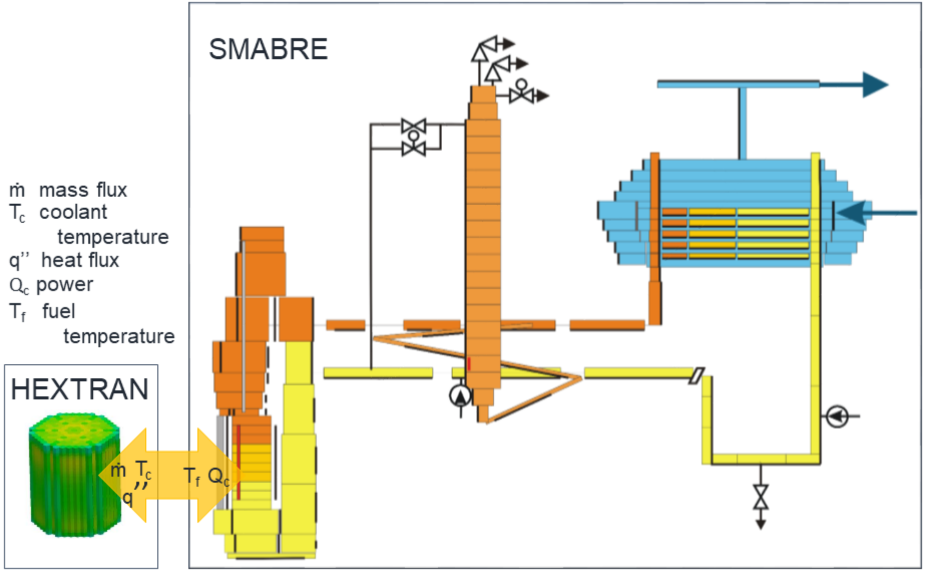
VTT

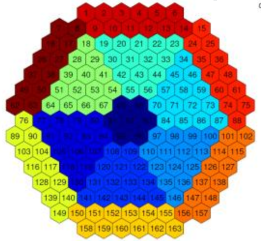
**Internal coupling**:
neutronics and thermal hydraulics in different codes

External coupling:
different code for the core and loop

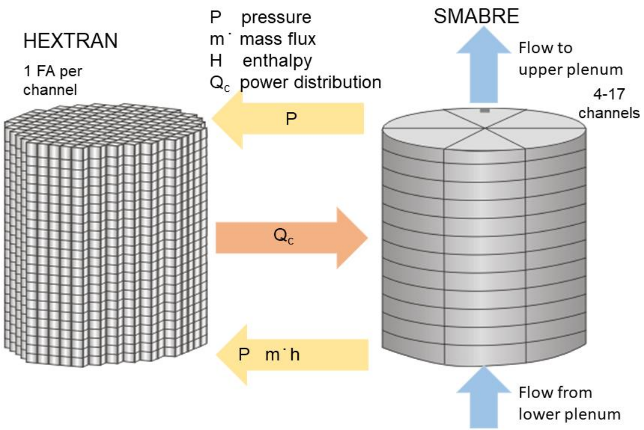
**Parallel coupling**:
Core thermal hydraulics is calculated with both codes
Totally independent codes coupled together

*Internally coupled HEXTRAN-SMABRE*



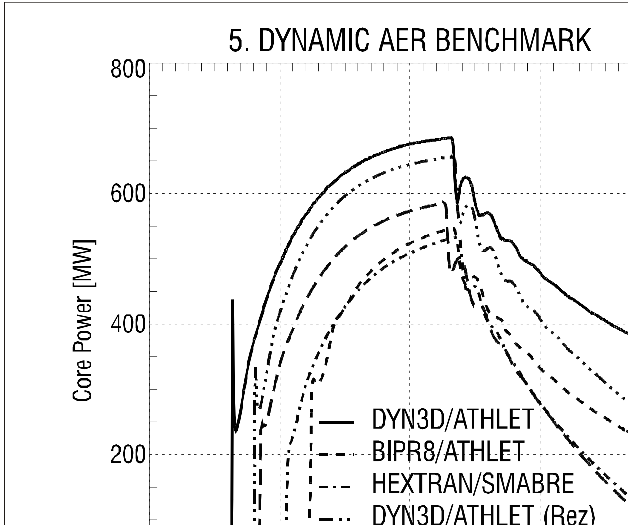



*Parallelly coupled HEXTRAN-SMABRE*



# Examples of coupled neutronics/thermal-hydraulics codes

- **TRACE-PARCS**, U.S.NRC
- **SIMULATE-3K**, Studsvik
- **POLCA-T**, Westinghouse
- **ARCADIA**, AREVA
- **QUABOX-CUBBOX-ATHLET**, GRS, Germany
- **DYN3D-ATHLET**, HZDR, Germany
- **BIPR-ATHLET**, KI, Russia
- **KIKO3D-ATHLET**, HAS Centre for Energy Research, HUNGARY
- **FLICA-OVAP**, CEA, France
- **HEXTRAN-SMABRE, TRAB3D-SMABRE, APROS**, VTT, Finland




AER benchmark 5
Total reactor power during
a main steam header break

# Models for reactor dynamical calculations - What is needed

Geometry and material properties of primary and secondary circuit

Properties of control and protection systems

- Pumps, valves, heaters, spray,…

Signals, measurements, delays,…

Operational conditions at initial state

Core loading

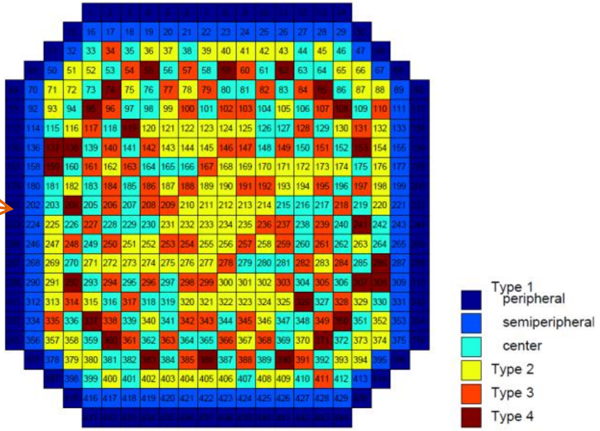
- Burnup
- Cross sections

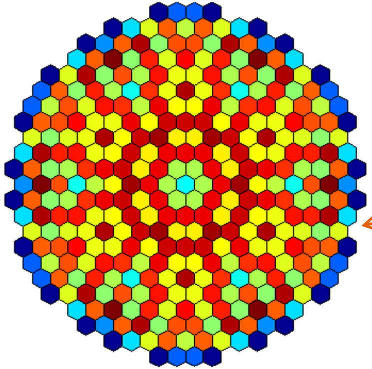
Fuel assemblies

- Geometry
- Material properties
- Local and distributed friction, spacers,
- Drift-flux, CHF, DNB, CPR etc. correlations

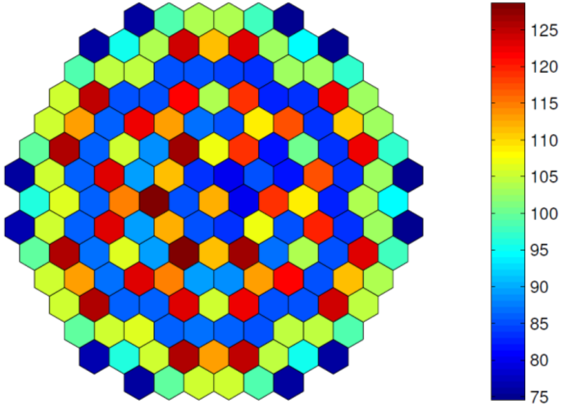
# Reactor core

Fuel assemblies in BWR core
- 4 types of assemblies



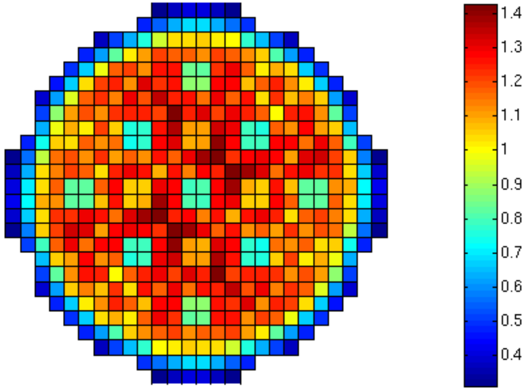



Power distribution
VVER-440 BOL
- 3 different enrichments




Relative power (%), t=0s

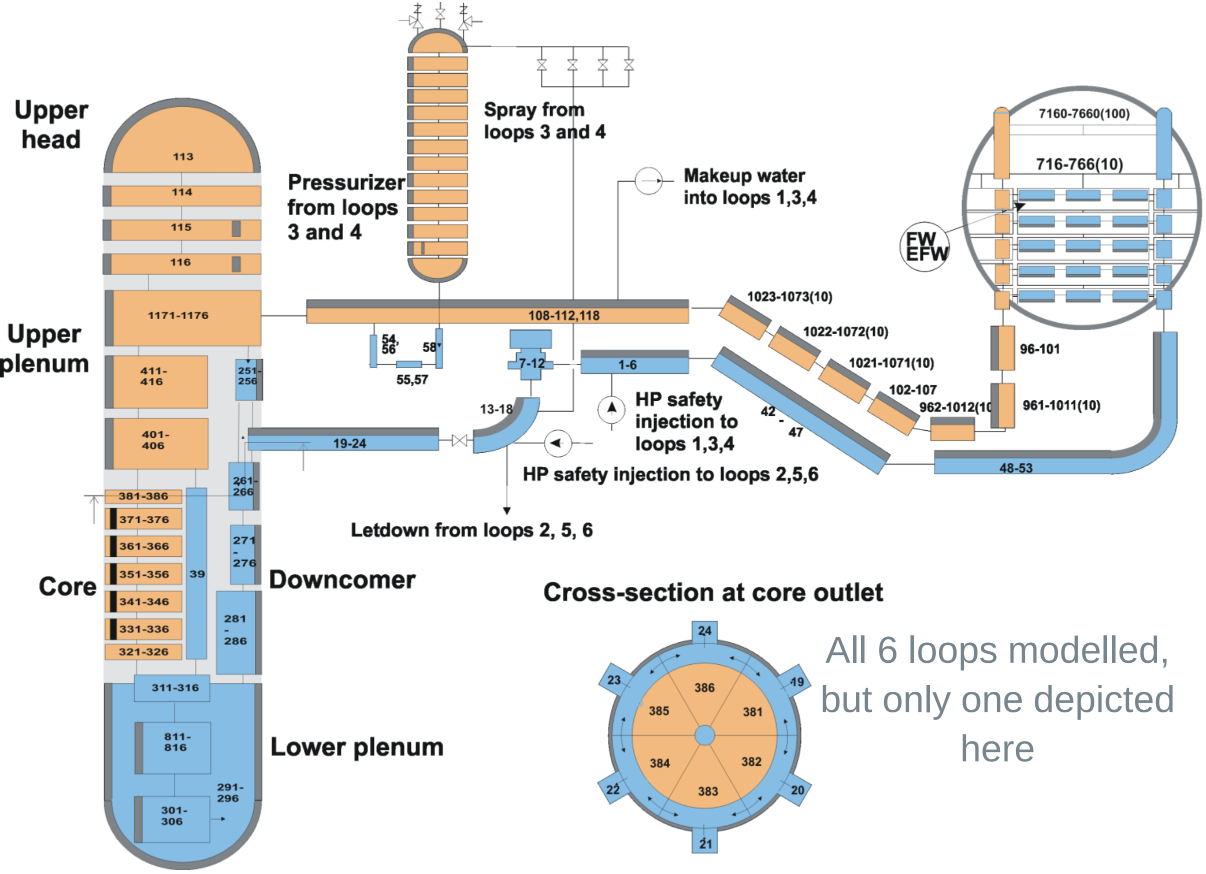
Power distribution
VVER-1000
One assembly has been replaced with different FA




Power distribution in BWR
- 2 types of assemblies
- Assembly burnup 0.14-32.9 GWd/tU

# SMABRE: Primary loop nodalization for Loviisa VVER-440

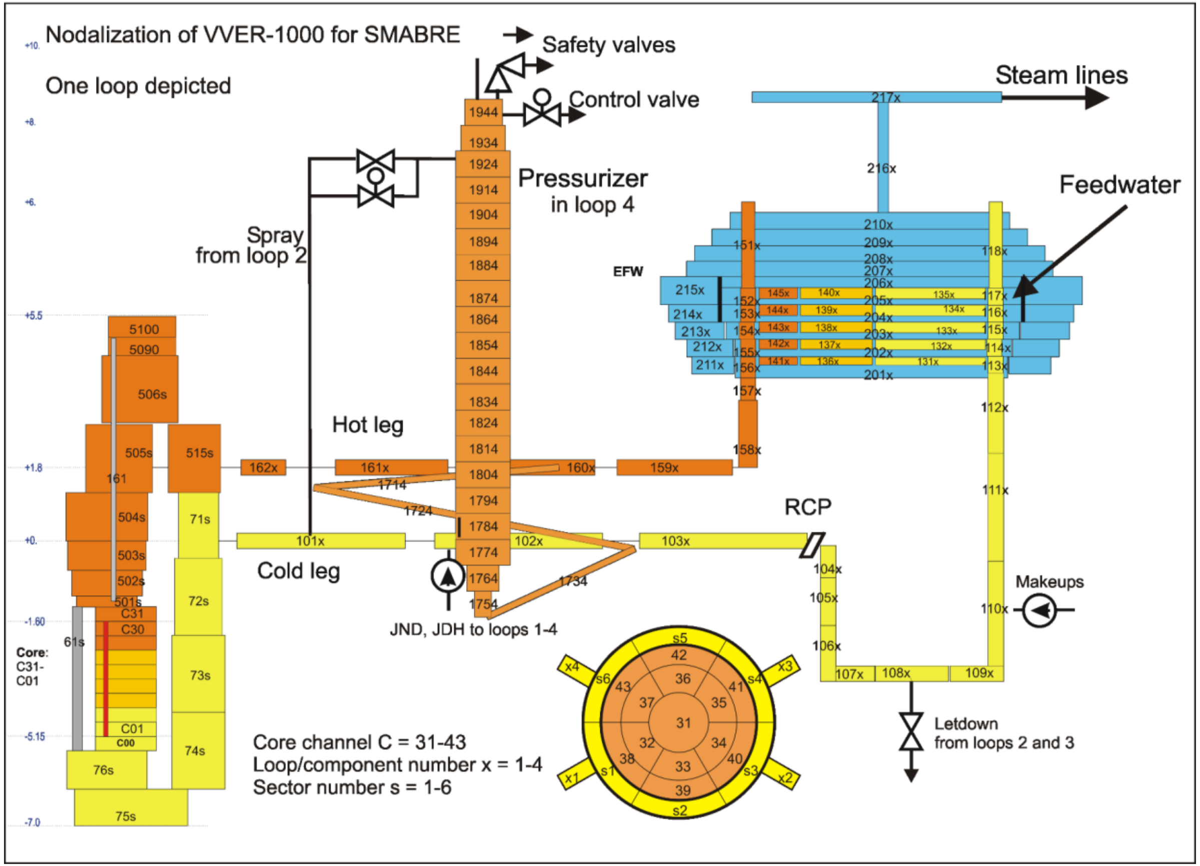





All 6 loops modelled, but only one depicted here

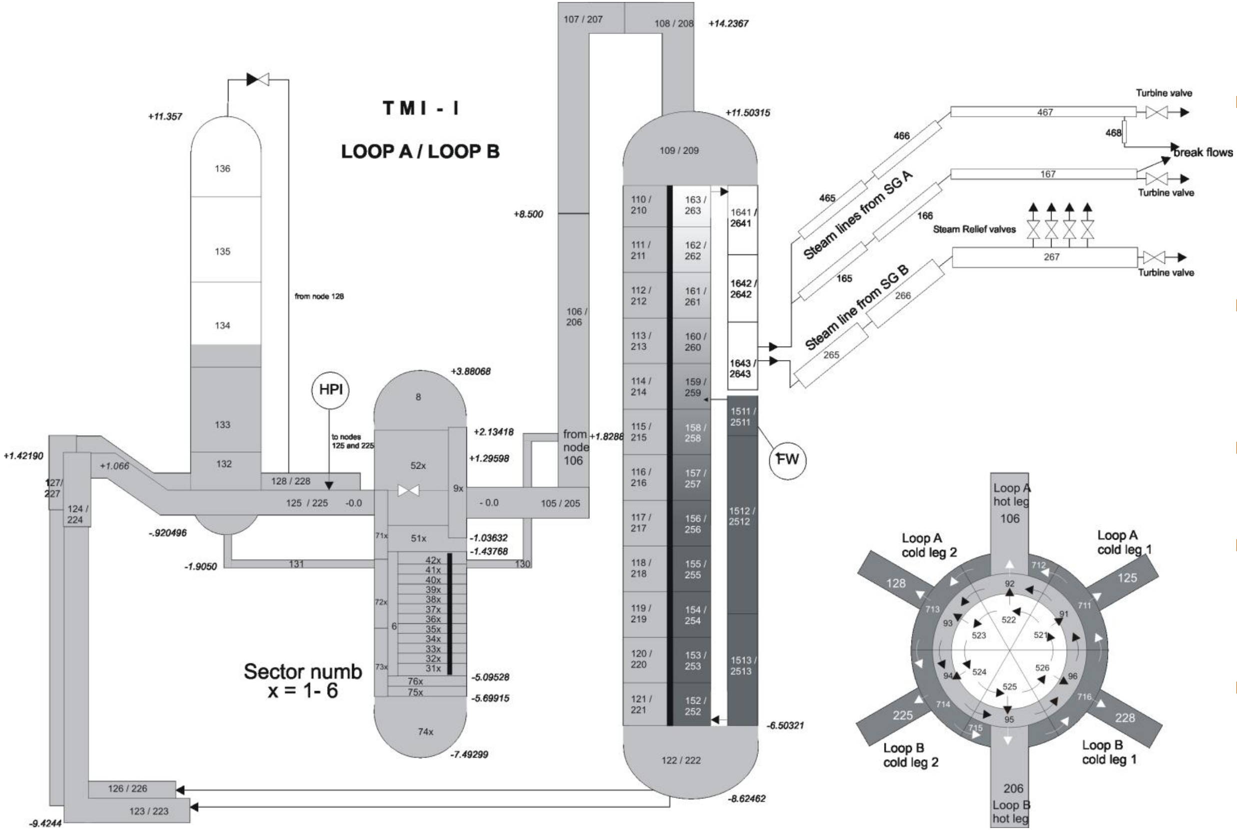
- Modelling of NPP for system code takes about 1 year
- 3 dimensional hydraulics (CFD) impossible for all the tanks, pressure vessel, pressurizer, steam generators for longer transients
- 1 dimensional hydraulics applied for 3-D phenomena with suitable nodalization
  - Sectors in RPV,
  - Dense nodalization in pressurizer
  - Several nodes in loop seals for phase separation
- Typically ~1000 nodes

# Primary loop nodalization for VVER-1000




VTT

# Nodalization of TMI-I for OECD MSLB BM



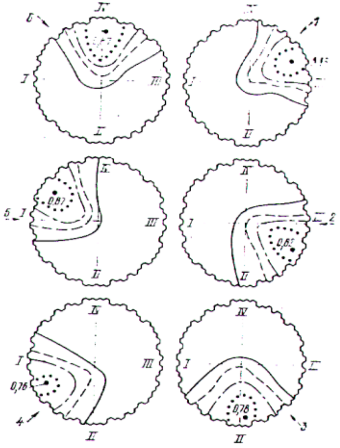

- Three Mile Island in Harrisburg, Babcock & Wilcox design
  - Two hot legs
  - Four cold legs
- Only two SGs - less water compared to other plants with vertical SGs
- Once through SG, superheated steam in SG outlet
- Bottom of SG is below the core elevation – effects to natural circulation
- Very fast response in accidents

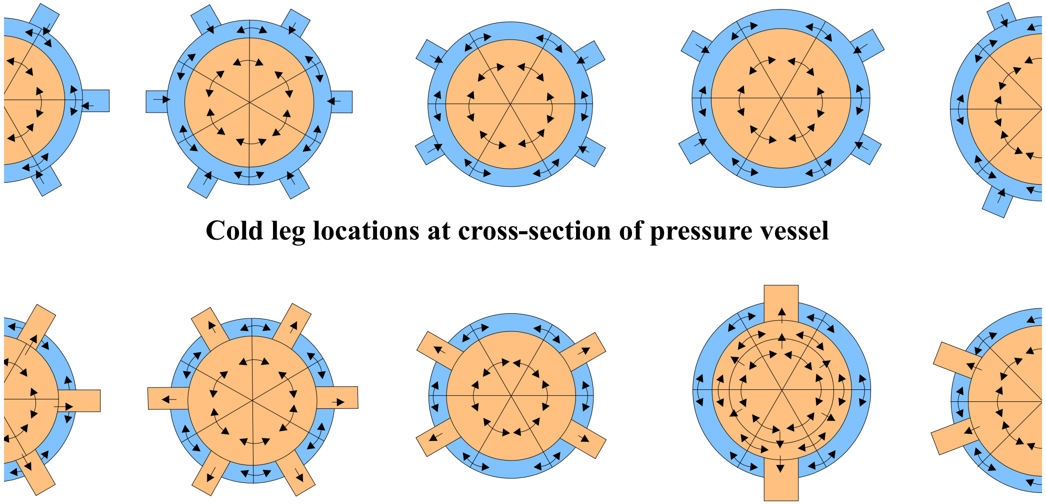
# SMABRE: modelling of mixing

Reactor pressure vessel nodalization: mixing in Reactor Pressure Vessel / sectors

Temperature measurements
in Loviisa VVER-440 at core inlet

Strong flow spinning in
Loviisa RPV



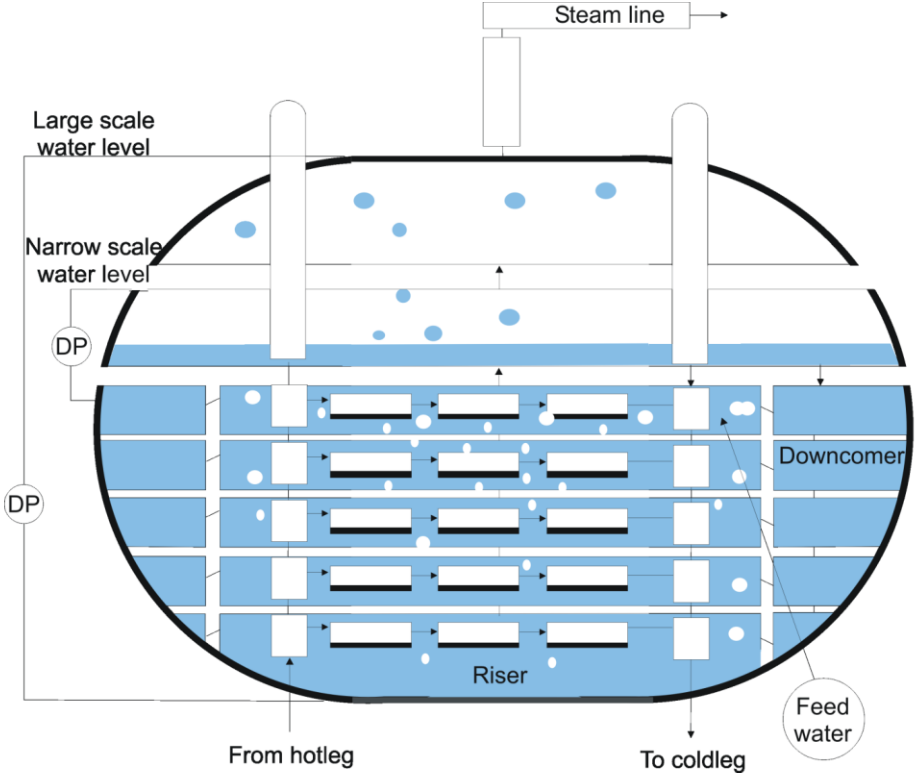


Cold leg locations at cross-section of pressure vessel

Simple turbulent mixing model of SMABRE:

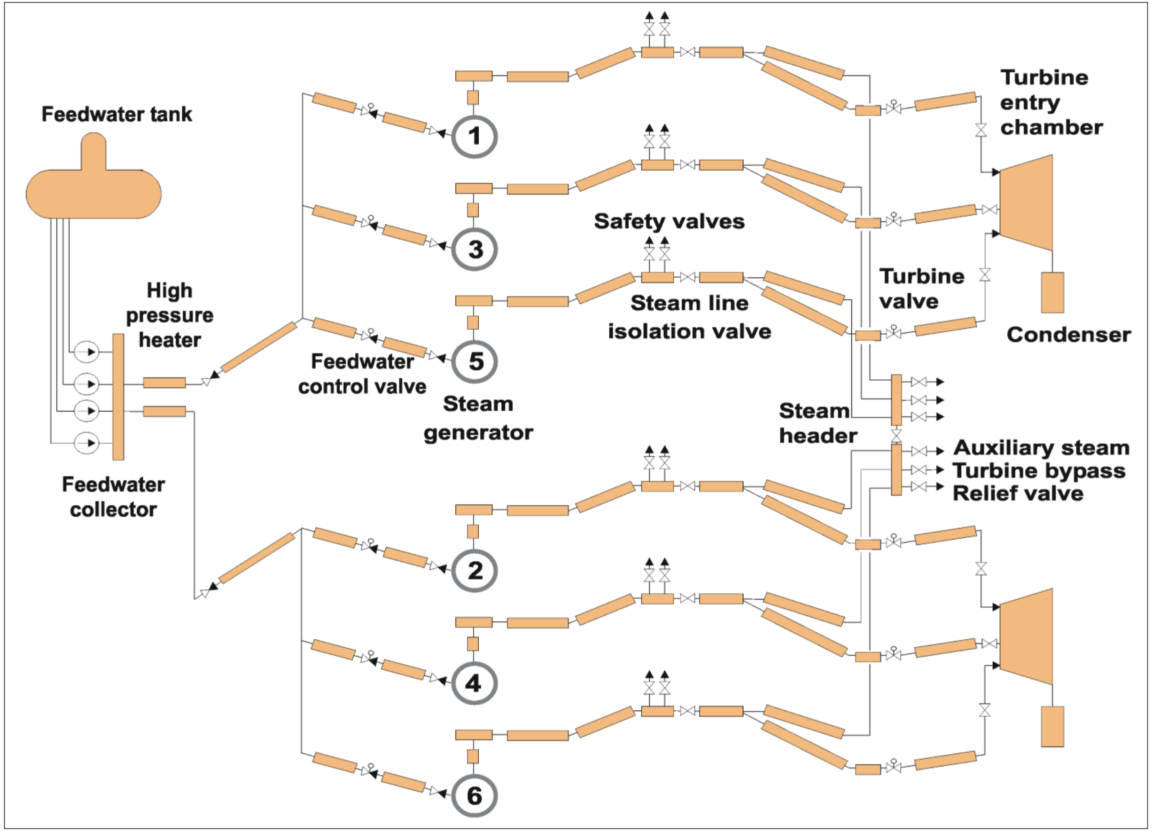
- Change of fluid enthalpy and content of boron acid between neighbouring sectors
- Final tuning with mixing factor
- Used also for vertical direction with mixing factor for upward and downward mixing

# SMABRE: Steam generator nodalization for Loviisa VVER-440, 6 SGs




- Internal circulation in SG primary and secondary side
- Decrease of heat transfer area according to water level decrease in SG
- Simulation of large and narrow scale water level measurements in edge area
- Five SG outlet tubes between SG and Steam line

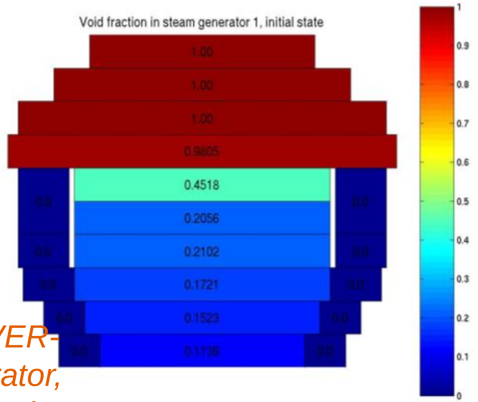
# SMABRE: Secondary side nodalization for Loviisa



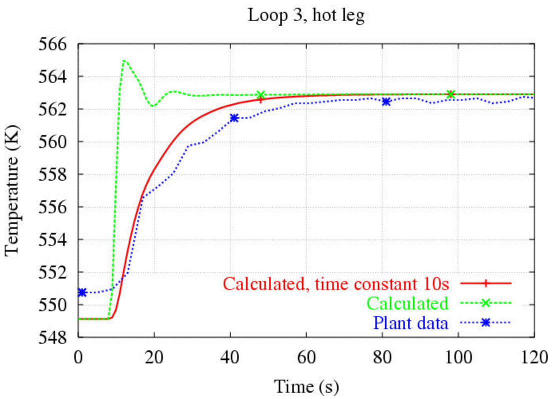

- Two turbines in VVER-440
- Only one steam header and feed water tank
- No turbine bleed -> no feedwater (FW) pre-heaters -> FW temperature according to feedwater tank enthalpy
- Check-valve features, volume of FW line e.g. in MSLB

# Validation against measurements

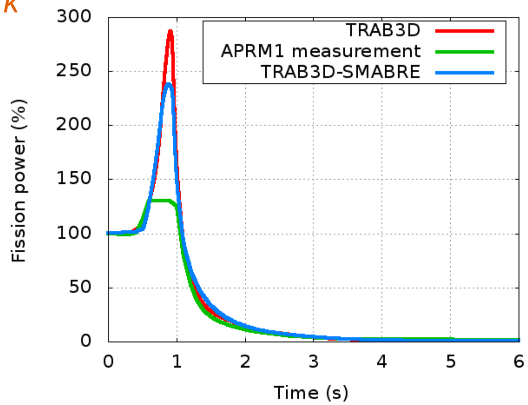
- Simulation of real physical quantity
  **or** simulation of measured value?
  - Delays and time constants of measurement?
  - Range of measurement?
  - Measured quantity?
    - E.g. water level




*Void fraction in VVER-1000 steam generator, V1000CT-benchmark*



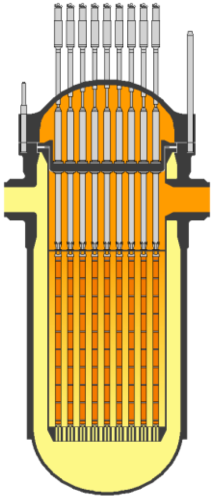

*Hot leg temperature after switching on 1 MCP in VVER-1000, V1000CT-benchmark*




*Power during BWR overpressurization transient*

# Decay heat

- Reactor core produces heat also after reactor shutdown due to radioactive decay of the short-lived fission products (I-134, Cs-138, Cs-140…) and actinides (U-239, Np-239)
- Proportional to power before shutdown
- Immediately after shutdown ~6-7% of total power
- After 1 hour ~1%
- After 1 week ~0.5 %

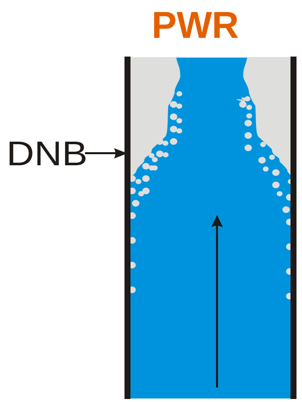


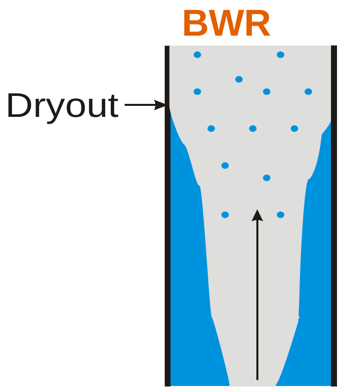
0,5% >20 000 kW

~ 5-10 kW

# Heat transfer crisis

**Two different mechanism:**




Nucleate boiling region

- **Low steam quality**
- If the heat flux is high enough, the vapour generation can establish a vapour film that isolates the coolant from the wall
  Departure from nucleate boiling (DNB)

Annular flow region

- **High steam quality**
- The liquid film dries out
  Dryout

# Heat transfer crisis

**Critical heat flux CHF**

Heat flux at which the boiling crisis will occur

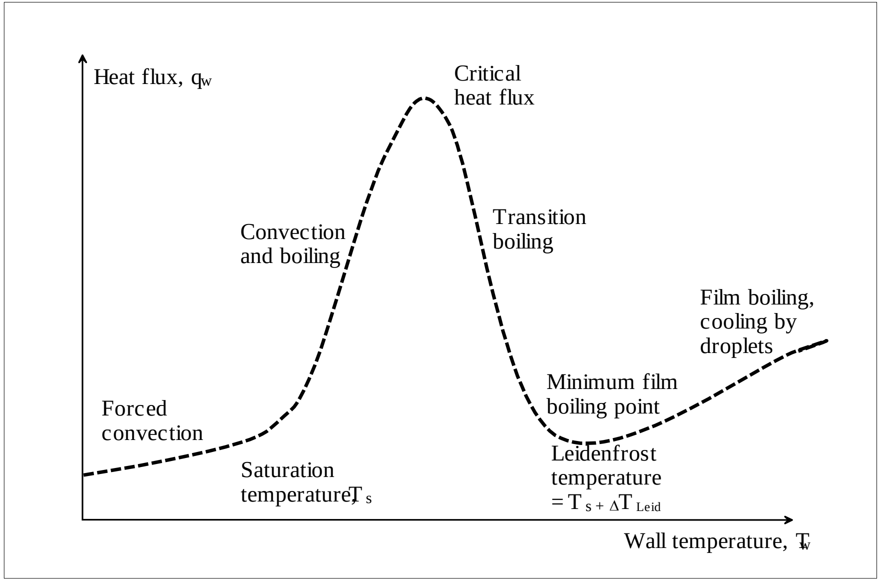
- Depends on a large number of factors
- Various empirical correlations are available

- Multitude of parallel terms
  - The occurence of CHF, burnout, dryout, boiling crisis, departure from nucleate boiling etc.
- Complicated phenomena, difficult to model
  - Even in subcooled and low quality region detailed mechanism causing CHF covers several phenomena concerning bubble and slug deformation

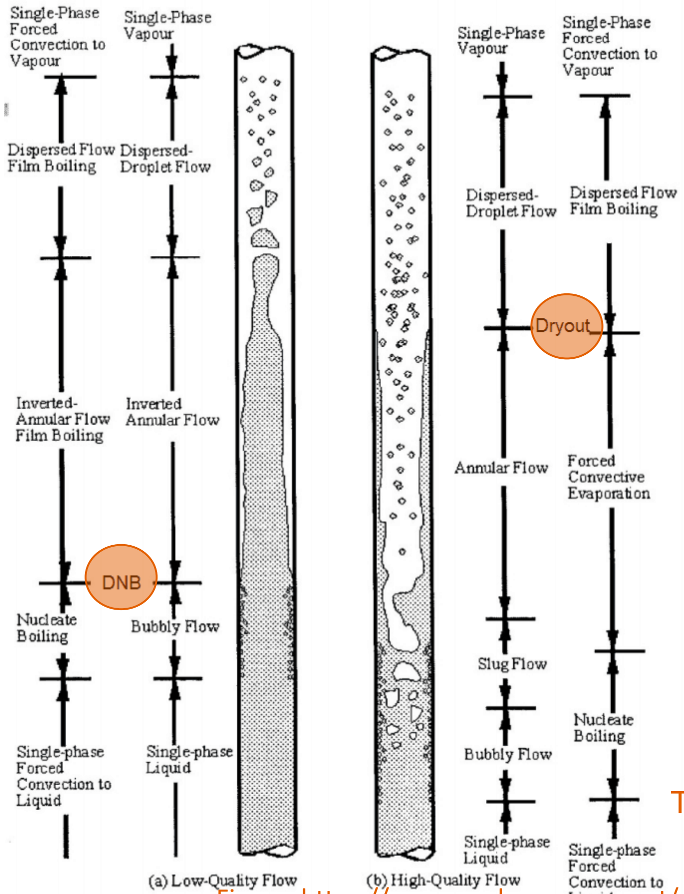
# Heat transfer modes and flow patterns

VTT

Boiling curve with heat transfer modes:
heat flux as a function of wall temperature



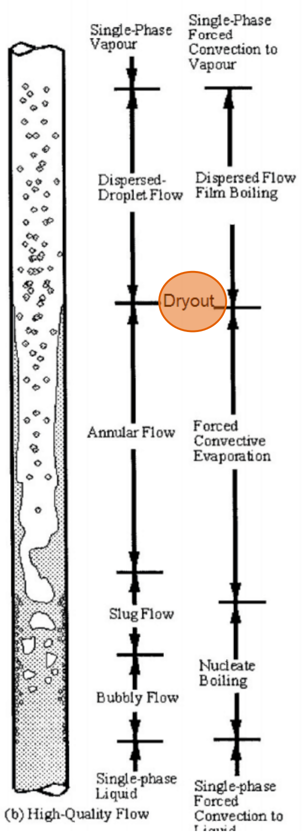

Critical heat flux: upper limit to the heat flux that is possible to transfer from fuel to the coolant in normal operation

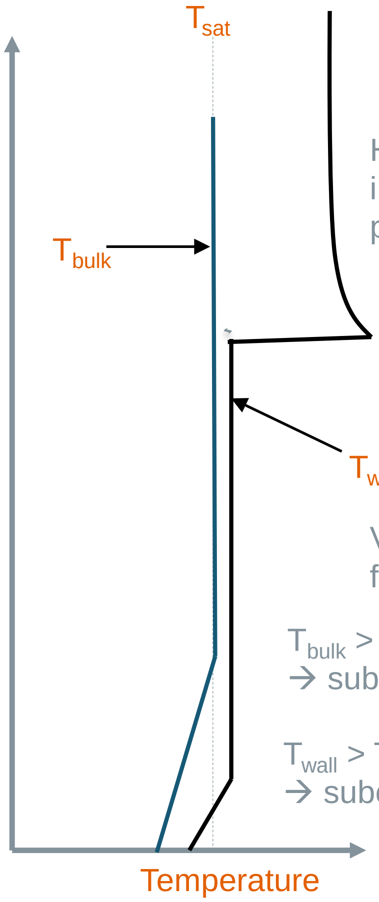



$T_{wall} > T_{sat}$

Figure: https://www.nuclear-power.net/

# Boiling in BWR channel




High steam velocity improves heat transfer in post-CHF region

Void fraction increases → flow becomes unstable

$T_{bulk} > T_{sat}$
→ subcooled total boiling starts

$T_{wall} > T_{sat}$
→ subcooled nucleate boiling starts

- Total power of FA affects more to dryout than in PWR where local heat flux is most relevant
- Dryout power and the outlet steam quality increases with tube length
- Dryout heat flux decreases with tube length
  → Boiling length $L_B$ used in correlations
  Distance of dryout height and height, in which steam quality begins to increase from 0

# BWR: the concept of Critical Power

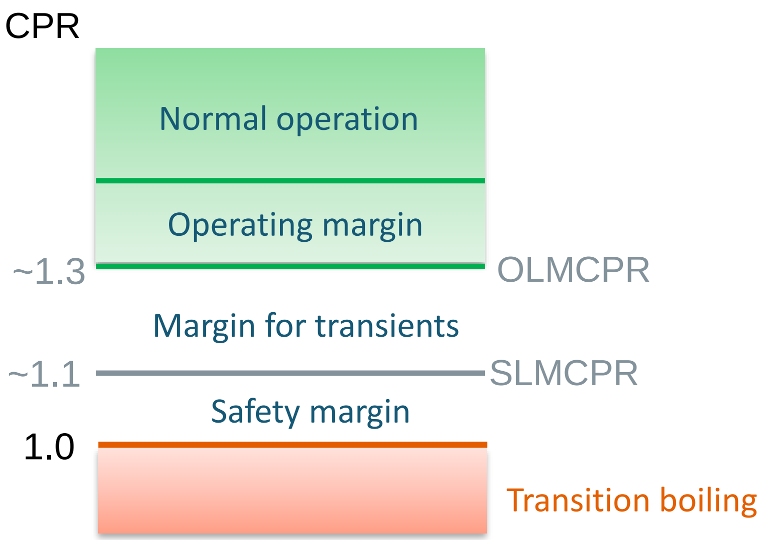
CPR = critical power ratio
MCPR= minimum CPR

SLMCPR= Safety limit minimum critical power ratio is the minimum CPR during the most limiting AOO transient so that fuel rods avoid boiling crisis

OLMCPR= Operating limit minimum critical power ratio is

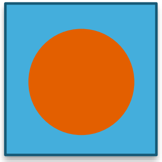
OLMCPR > SLMCPR + max ΔCPR

Where ΔCPR is change of CPR during the limiting transient

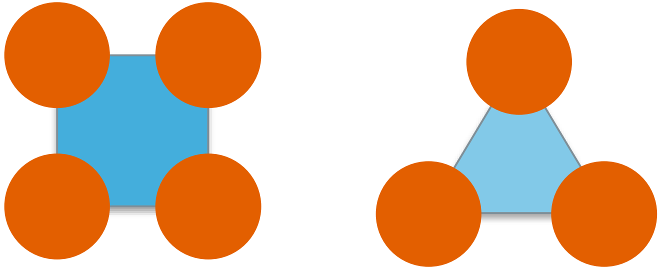


# Hot channel analyses and DNB evaluation

- Last step of safety analyses
- To analyse the most severe conditions for a fuel rod during a transient
- Isolated thermal hydraulic channel, no neutronics calculation
- Boundary conditions from a three-dimensional core model
- Hot assembly, hot rod conditions
- Parameters which are not well known must be varied
- Models are not as detailed as in fuel behavior codes
- Thermal margins within acceptable limits?
  - DNB / CHF / CPR
  - Linear power
  - Fuel enthalpy
  - Cladding oxidation
  - Number of failed rods



*Isolated hot channel*



*Subchannel configuration for square and triangular lattice*

# VTT's hot channel methodology

VTT

Transient analysis with a coupled neutronics-thermal hydraulics code with a 3D neutronics core model

→

Select assemblies (usually ~20) for hot channel analyses and rerun the case

- Maximum assembly power during a transient
- Maximum pin power during a transient
- Maximum power increase during transient
- Burnup of assembly

→

Transient analysis with a coupled neutronics-thermal hydraulics code with a 3D neutronics core model

Store boundary conditions for hot channel analyses

↓

Determine hot channel factors

→

Run a series of hot channel analyses for the selected assemblies

- Parameters with largest uncertainties are varied

→

- DNB / CHF / CPR
- Linear power
- Fuel enthalpy
- Cladding oxidation
- Number of failed rods

# Hot channel analyses

- Why separate hot channel analyses?
  - Easy to do many variations
  - Easier to handle conservative assumptions
  - Easy to vary correlations, as well as fuel & channel properties
  - Easier to cover different loadings
  - The results of a typical loadings can be used for all similar cycles
  - Other assemblies are covered by varying the hot channel factors
  - Full core transient simulation is not disturbed by artificially manipulated rods in extreme conditions

- In addition to conservative hot channel analyses, DNB can be evaluated also during 3D transient simulation
  - E.g. in some plants scram initiation due to online DNB-value

# Hot channel factors

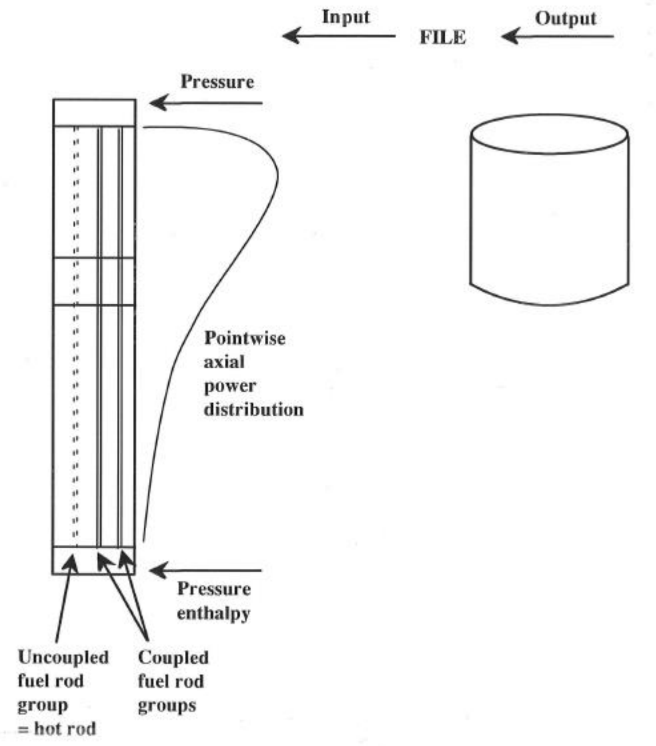
Additional multiplier to describe extreme conditions

- For enthalpy rise $F_{\Delta H}$
- For heat flux $F_q$

- Hot channel factor
  - $F^N$ for Nuclear effects
    - variations in power distribution
  - $F^E$ for engineering factors
    - Fuel fabrication tolerance
      - Rod dimensions
      - Amount of fissionable material
    - Flow mixing related factors
      - Lower plenum flow distribution
      - Pressure drop changes due to flow distribution
      - Flow mixing between channels
      - Bypass flow

# Hot channel analyses

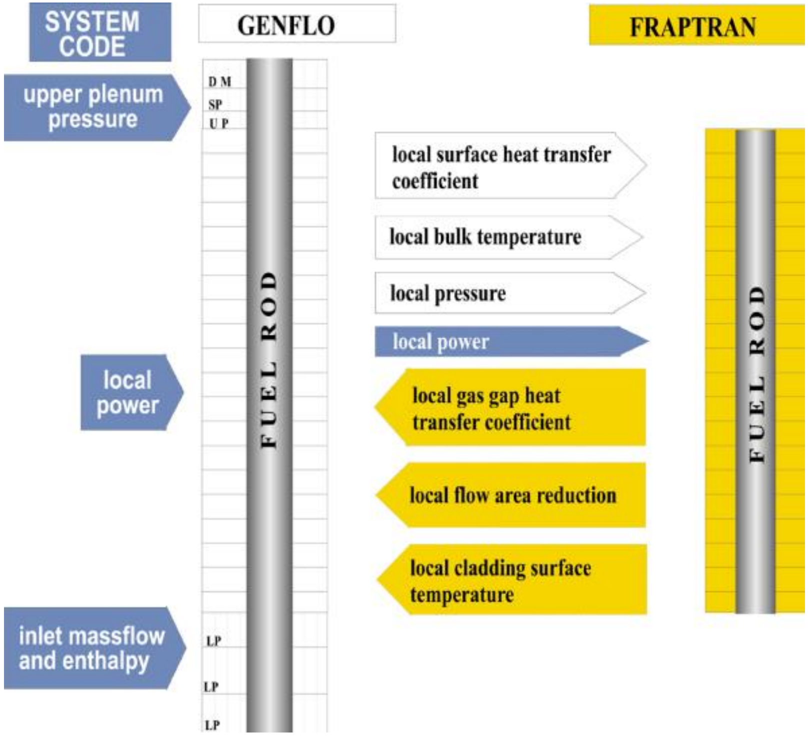
- TRAB-1D as a hot channel model
  - Apart from neutronics, models similar as in 3D codes, but nowadays used only for hot channel applications
  - Includes some submodels specific to hot channels
- Includes only one isolated fuel channel
- No neutronics calculation
- Time-dependent boundary conditions from a full core calculation (TRAB3D/HEXTRAN)
  - Axial power distributions
  - Inlet enthalpy
  - Inlet and outlet pressure



# Hot rod analyses with FRAPTRAN-GENFLO

VTT

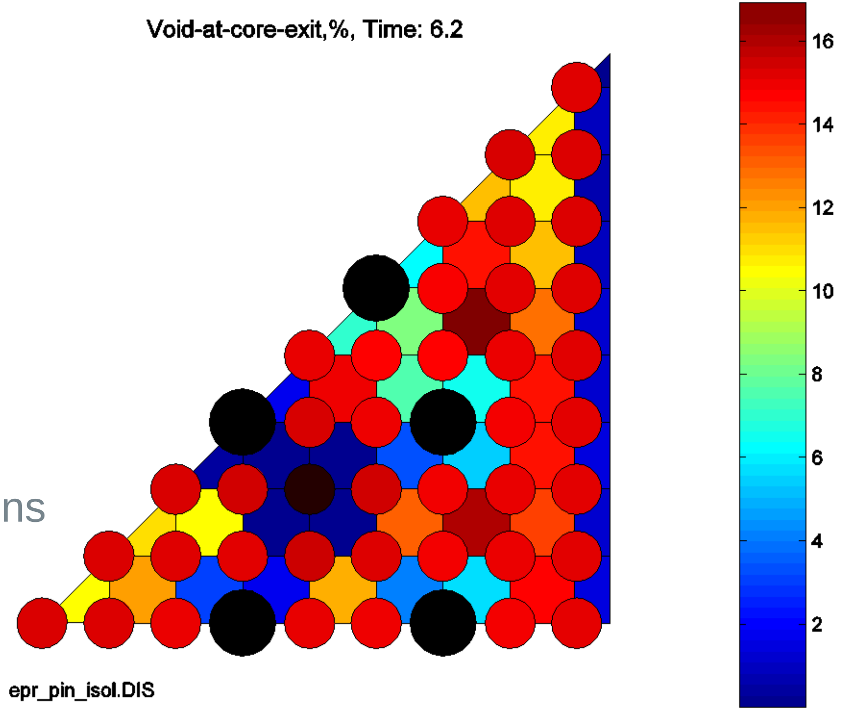
- For more challenging conditions coupling of fuel performance code FRAPTRAN (U.S. NRC) and thermal hydraulics code GENFLO (VTT)
- FRAPTRAN calculates behaviour inside a fuel rod
- GENFLO calculates overall thermal-hydraulic behaviour and surface heat transfer coefficients
- Recently added option to model several fuel rods in a subchannel
- Used also for statistical evaluation of fuel rod failures




Data exchange between the system code GENFLO and the transient fuel behavior code FRAPTRAN (various options)

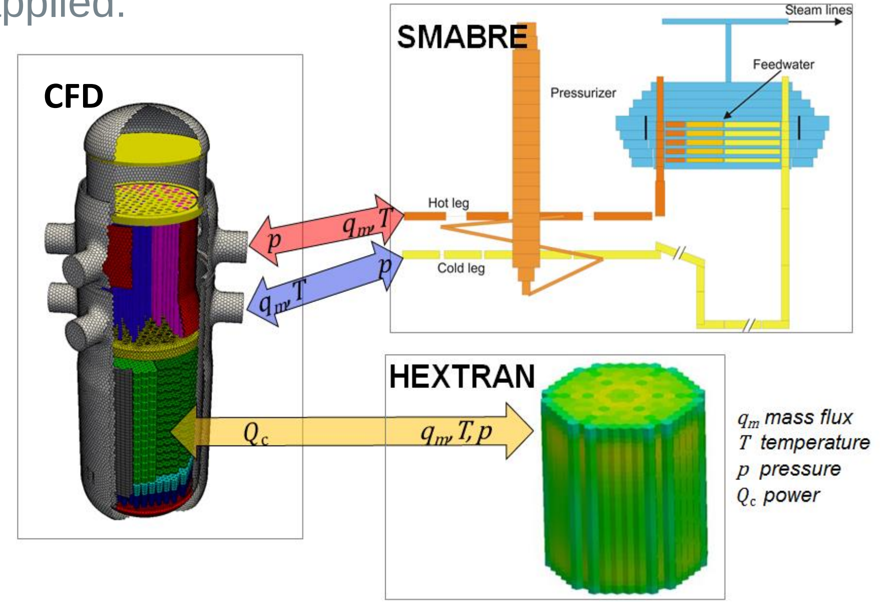
# Subchannel analyses

- COBRA is well-known subchannel code
  - Developed in USA
  - Several versions
    - COBRA-3C/MIT
    - COBRA-IV
    - COBRA-EN
      - available in NEA data bank
      - 3- or 4-equation model
    - COBRA-TF
      - Conservation equations for vapor, continuous liquid and entrained liquid droplets = 9 equations
    - Several commercial codes are based on COBRA
  - Power as a boundary condition



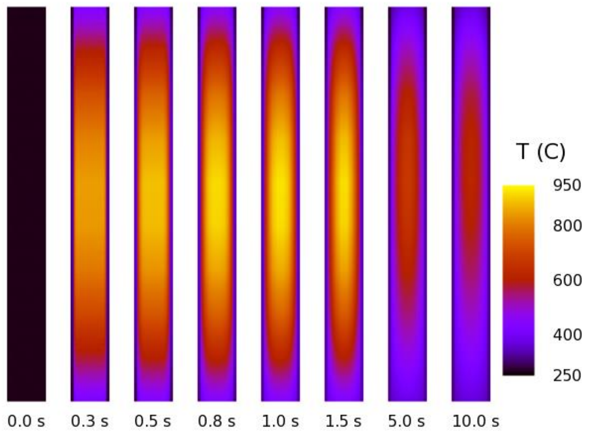
# Topical development of reactor dynamics modelling - Multiphysics

- Trend towards more accurate modelling
- Nowadays in reactor dynamics 1D hydraulics is applied for 3D phenomena
  - suitable nodalization, mixing coefficients
- At the moment modelling of the transients using 3D hydraulics (CFD) for all the tanks, pressure vessel, pressurizer, steam generators etc. is impossible
- In a limited area, 3D hydraulics can already be applied:
  - e.g. Reactor pressure vessel with Porous CFD-style thermal hydraulics solver PORFLO or OpenFOAM
    - 0.5-1 million cells
    - 2-phase, 3D thermal hydraulics
    - 6 equation model
    - Areas where accurate CFD modelling is unnecessary complex (e.g. reactor core) can be modelled with porosities
- Coupling with HEXTRAN neutronics and SMABRE system code model



# Topical development of reactor dynamics modelling - Multiphysics

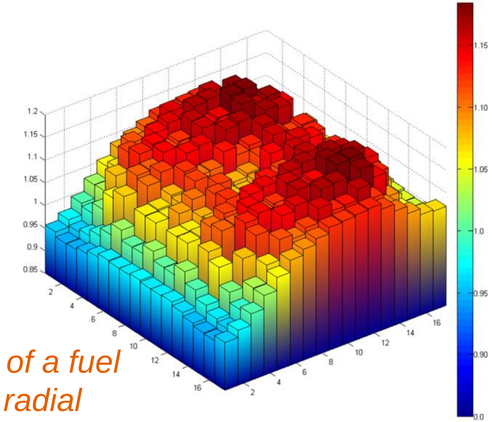
- More accurate modelling of fuel
- FINIX is a Fuel behaviour model and interface for multiphysics applications, developed at VTT
  - A lightweight fuel performance code that is primarily designed to be integrated as a subprogram into a larger simulation code at source code level
  - Aimed for multiphysics simulations involving reactor physics and thermal hydraulics, where fuel behaviour is often modelled with simple correlations and thermal elements
  - Coupled e.g. with HEXTRAN, TRAB3D and reactor physics code SERPENT 2




*Example: temperature distribution in a fuel rod during CRE in VVER-440*

# Current development of reactor dynamics modelling - Multiphysics

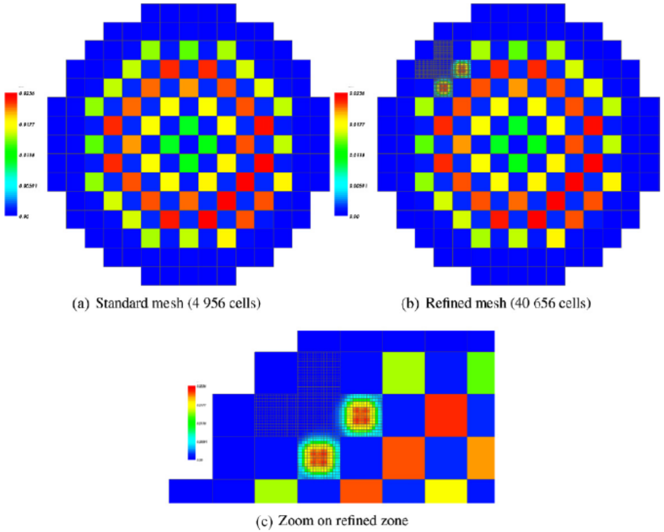
- More accurate modelling of power distribution
- TRAB3D has been supplemented with a pin power reconstruction model
- Pinwise power distributions are needed for more accurate safety analyses
  - Hot channel analyses
  - Coupling with 3D thermal hydraulics



*Relative power distribution of a top of a fuel assembly at the second outermost radial layer of an EPR core, reconstructed by TRAB3D.*

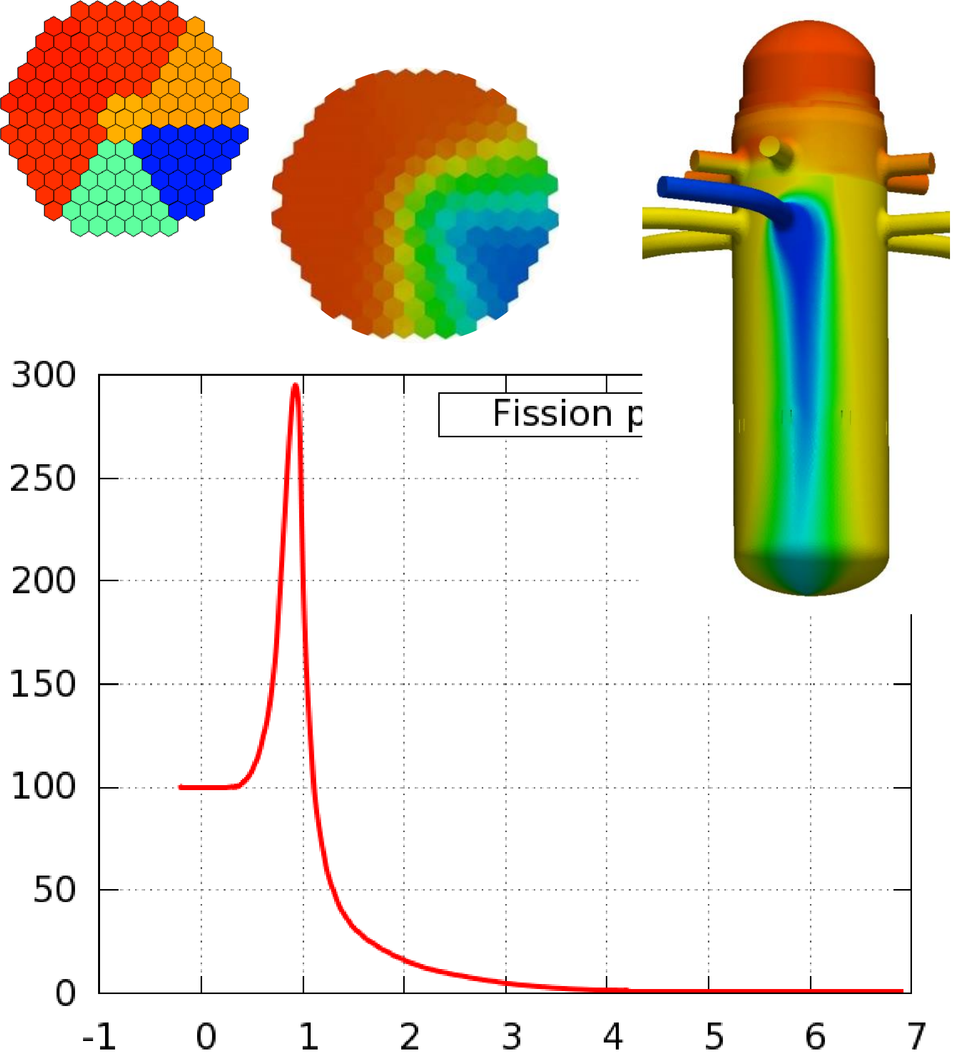
# Topical development of reactor dynamics modelling - Multiphysics

- Example:
  FLICA-OVAP / CEA, refinement of calculational mesh
  - Refinement of geometries
  - Neutronics with pin power reconstruction
  - 3D thermal hydraulics




*OECD NEA MSLB benchmark with MSLB benchmark with refinement at the subchannel scale for the hot channel*
*assembly and its neighbours. Quantity shown is initial void fraction at level z=3.5 m.*

*Fillion, P, Chanoine, A, Dellacherie, S, Kumbaro, A, FLICA-OVAP: A new platform for core thermal-hydraulic studies, Nuclear Engineering and Design 241 (2011) 4348-4358*

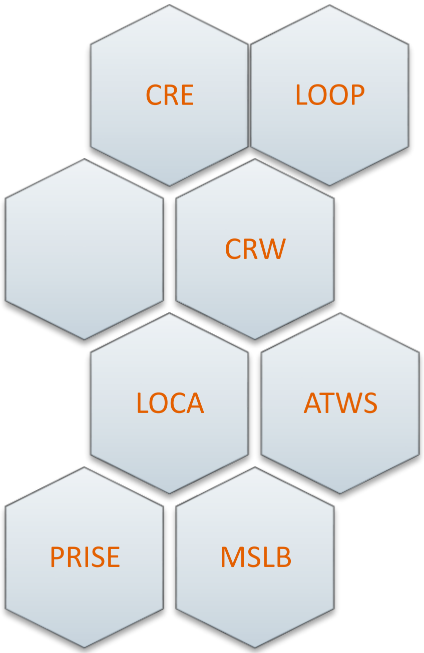



# Description of transients and accidents

# Outline

- Background
- Anticipated transients (Condition II)
- Postulated accidents (Condition III-IV)
- Design extension conditions

  - Initiating events
  - Phenomena during transients and accident



# General remarks

- Usually **limiting cases** are selected for detailed analysis and it is assumed that it covers also some other transients effected by same phenomena
  - For example:
    - MSLB covers also inadvertent turbine valve openings
    - ATWS can occur either due to mechanical blockage of control rods or due to signal failure
  - However, the worst-case is not necessary self-evident
    - In VVER-440 MSLB largest leak size is 263% in an elbow of steam line
    - Due to signals and actuation limits of protection systems some smaller leak size may lead to worse consequences
- Several coupled phenomena, operator actions and automatic operation of control and protection systems affect propagation of transients and accidents

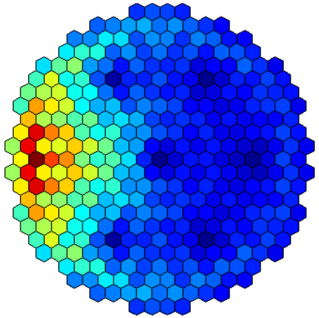
# Background

- Different kind of time scales
- Different kind of initial states, not only full power
  - Initial power: full, partial, zero
  - Beginning of cycle, end of cycle
- Often several variations of a transient are needed, because one conservative assumption is not necessary conservative for another criteria
  - e.g. high initial primary pressure is conservative for pressure analysis, but for fuel rod cooling low initial pressure is conservative

**Aim is to introduce different kind of events, transients and accidents, and to describe phenomena that affect the propagation of transients and accidents.**

# Background

- Types of transients and accidents:
  - Reactivity and power distribution anomalies
  - Increase or decrease of reactor coolant inventory
  - Decrease of reactor coolant flow rate
  - Increase or decrease in heat removal by the secondary side
  - Radioactive release from a subsystem or component
- Several initiating events can lead to these situations
  - Analysis of limiting cases
- Several coupled phenomena as well as operation of control and protection systems affect the progression of transients



# Abnormal operation and anticipated transients

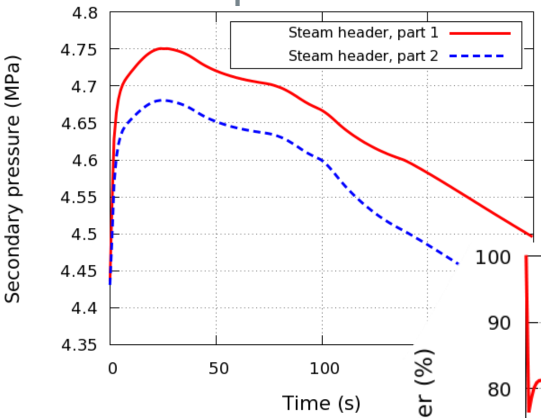
- Examples of initiating events that can lead to changes in flow conditions and reactor power:
  - Pump trip or other increase or decrease in pump speed
  - Inadvertent valve closures or openings
  - Turbine trip
  - Loss of offsite power
  - Control rod withdrawal
  - Boron dilution
  - Inadvertent actuation of the ECCS (Emergency core cooling system)
  - Malfunction of CVCS (Chemical & Volume Control System)
  - ….

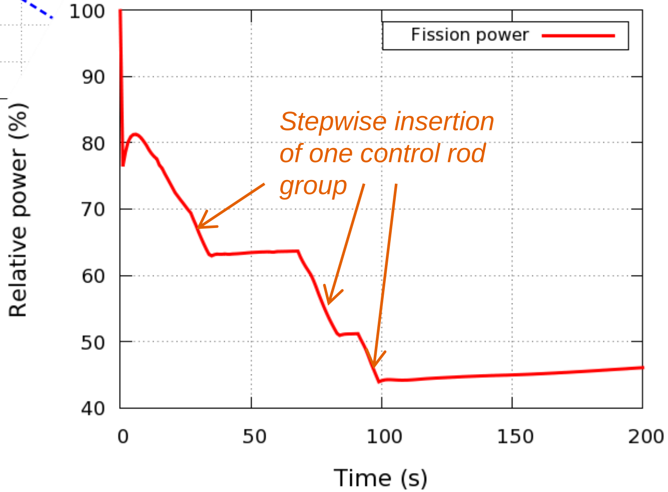
# Abnormal operation and anticipated transients

- Plant has to be designed and operated so, that in these kind of situations normal operation and protection systems are sufficient.
- Comprehensive analysis of these kind of events is part of licensing process.
- Conservative analysis
- Single failure assumptions
  - In redundant systems one in service, one fails

# Turbine trip

- Often assumed that turbine trip causes also loss of offsite power
- Example: Turbine trip in NPP with 2 turbines, no loss of offsite power




*HEXTRAN-SMABRE simulation of the load-drop experiment at Loviisa-1 NPP*

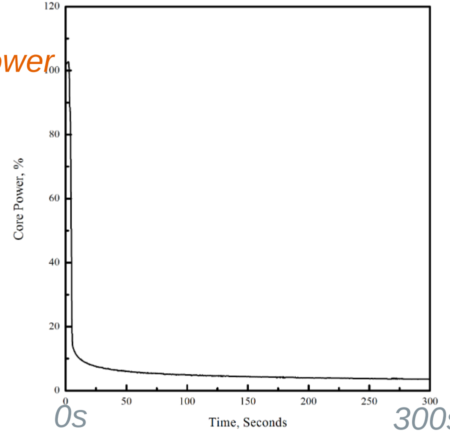
- Only one turbine trips → partial power
- Increase of secondary pressure → heat transfer from primary to secondary side decreases → Increase of primary pressure Normal pressure control only, Pressurizer (PRZ) safety valve opening not allowed in anticipated transients
- Turbine bypass valves open soon after turbine trip

# Loss of offsite power LOOP

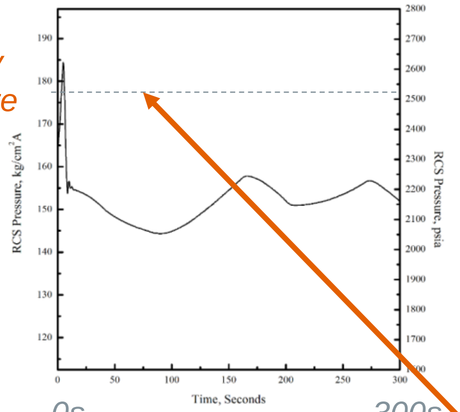
- Initiating event or consequence of turbine trip
- LOOP→ Trip of all RCPs and feedwater pumps, Normal PRZ heaters and spray not available
- Reactor trip soon after LOOP due low RCP speed or low flow
- Primary pressure and temperature increases due to weakened heat transfer to secondary side and non-availability of normal primary pressure control devices
- Heat transfer from fuel rods decreases due to diminished coolant flow
- Loss of primary heat sink due to turbine trip
  → decay heat removal has to be ensured
- Start-up of diesels after delay
- DNBR? Maximum pressure? Does the pressure remain within the permitted limits without opening of valves designed for accidents (e.g. pressurizer safety valves)

# Loss of offsite power LOOP

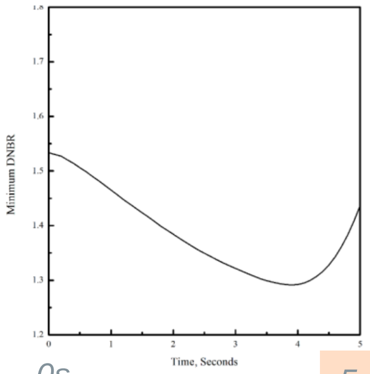
Core power



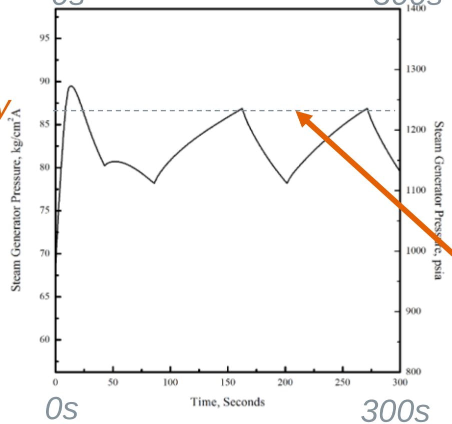

Primary pressure




Minimum DNBR




Secondary pressure




Minimum DNBR occurs typically few seconds after LOOP, but pressure may start to rise again several minutes after reactor trip

*Opening pressure for POSRV (Pilot operated safety and relief valve)*

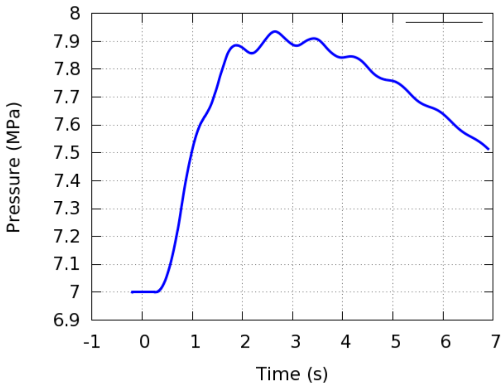
*Opening pressure for MSSV (Main steam safety valve)*

*Figures from https://www.nrc.gov AP-1000 FSAR*

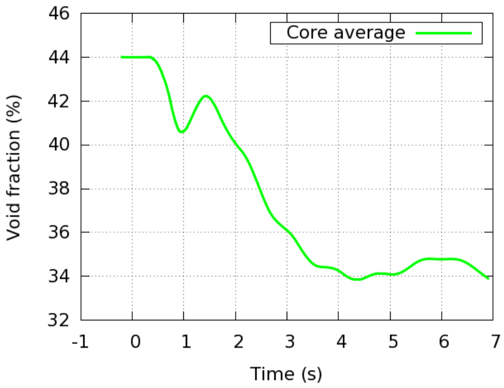
# BWR Pressure transient

- Initiating event sudden closure of steam line valves or failure of pressure controller
- Happened in Olkiluoto 1985

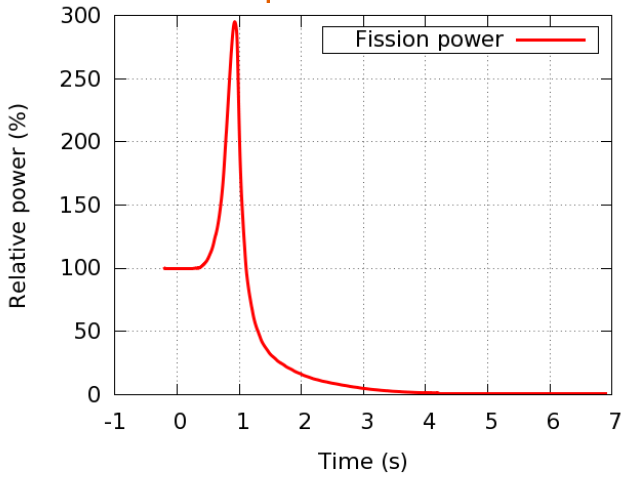
→ Pressure increase




→ Voids collapse




→ Fast power increase




*Figures from TRAB3D-simulation of the event.*
*Results match well with measured data.*

# BWR load rejection

- Example Olkiluoto 1 load rejection test June 16, 1998
  - From full power to 30 % power
  - Turbine valves close, dump valves open
  - Main circulation pumps and feedwater pumps stop
  - Asymmetric partial scram: one hydraulic scram group and one motor driven scram group
  - Local measurements during test (5 Local Power Range Monitors, LPRM, 4 at 4 heights, one at two heights, 4 Average Power Range Monitors, APRM, based on 28 LPRMs each)

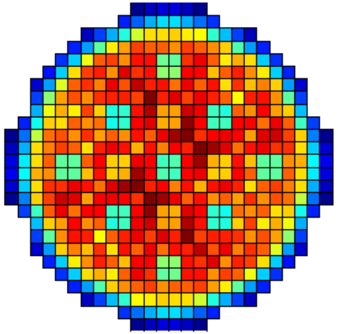
# Load rejection, BWR
# Olkiluoto 1 load rejection test June 16, 1998

TRAB-3D model

- Full core geometry with circuit, 500 channels and 25 axial nodes in core
- Mixed core with two different fuel types
- Part length fuel rods

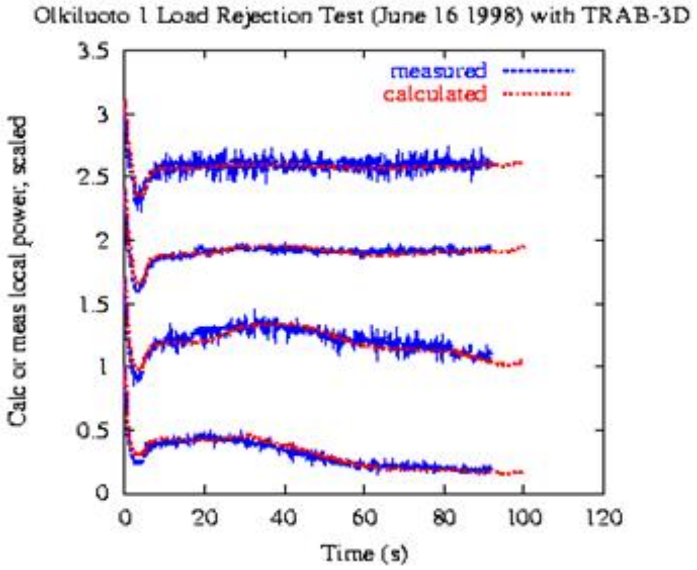
Main interest in validation of three-dimensional effects

- Transient boundary conditions from test results: turbine and dump valves, recirculation pump speed, feed water flow and temperature
- Direct comparison with measured values
- Simple model for measurements of local power (LPRM and APRM)
- Calculation to end of transient: 400 s

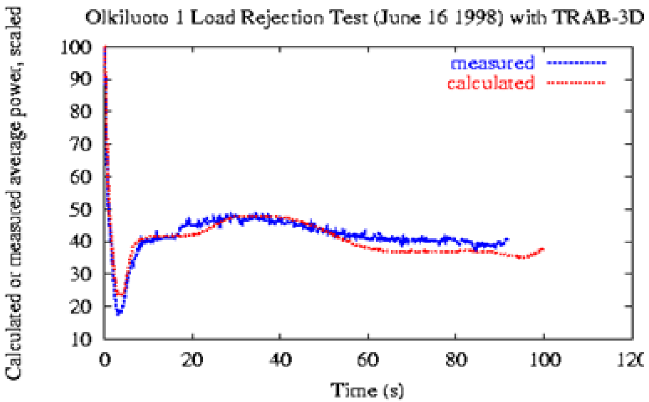


*Power distribution at initial state*

# BWR load rejection
# Olkiluoto 1 load rejection test June 16, 1998



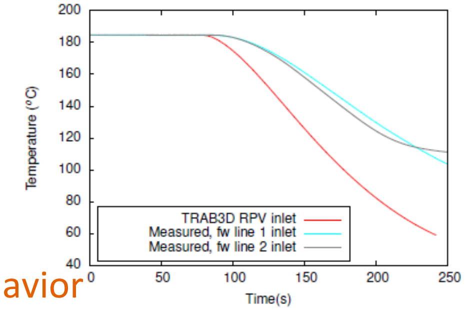

LPRM at different axial heights

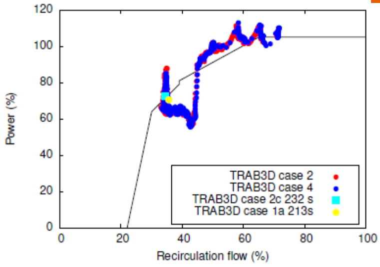


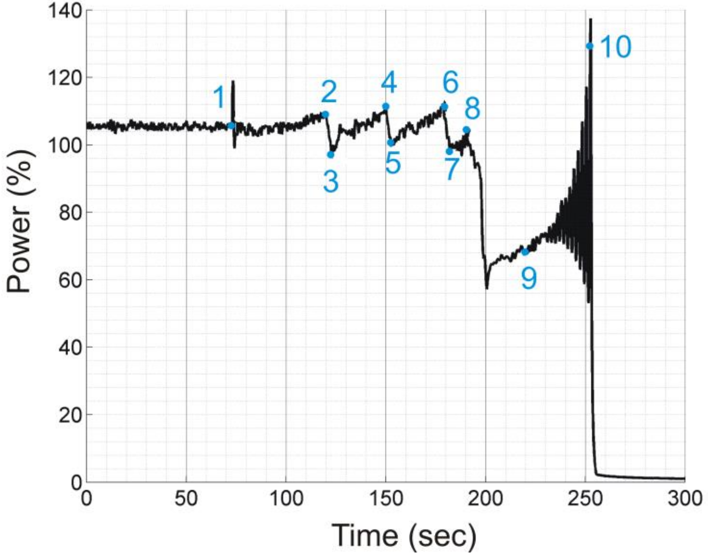

APRM based on 28 LPRM values

# BWR instability

- Wrong combination of power, flow and coolant temperature may lead to unstable behavior
- Several times in BWR's, also in Finland



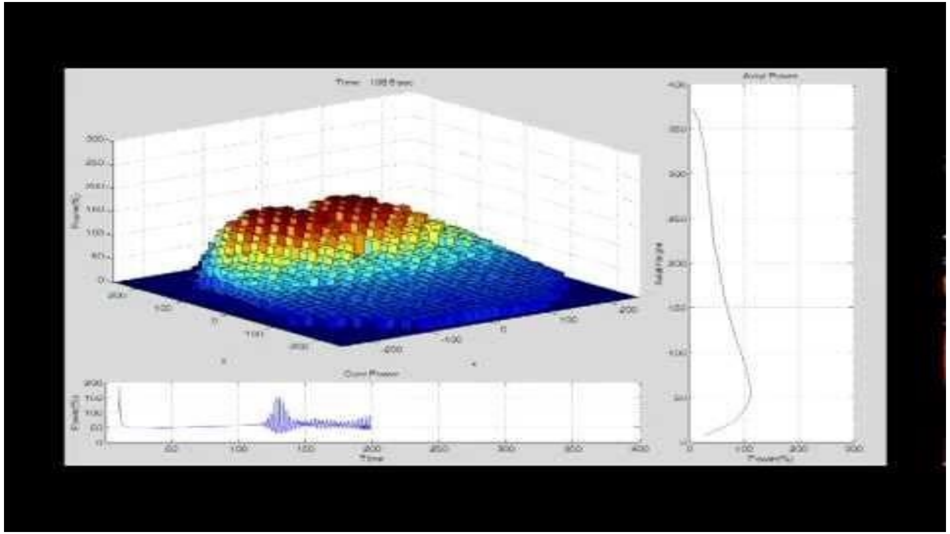

*Measured power during instability event*

- E.g. instability event in Sweden:
  **A short loss of external power** (1)
  → load rejection
  Wrong combination of signals
  → loss of feedwater heating, but no pump trip or partial scram

→ power increase due to cold water
→ A pump controller reduced the main recirculation flow 3 times (2-7)
Finally the operators partially scrammed (8) reactor. Due to cold feedwater power still increased and entered the unstable region of power-flow map (9)→ power oscillations
Finally, event was terminated by reactor scram (10)

# BWR instability

- Different type of instabilities can occur:
  - Global: whole core oscillates in phase
  - Regional: one part of the core oscillates out-of-phase in relation to another
- Reactors should be designed and operated in such a way, that instabilities are eliminated



https://youtu.be/3TZcZDVIvZk video by Peter Yarsky, NRC

# Boron dilution

- Inadvertent decrease of the boron concentration in the primary coolant
- Inhomogeneous dilution: a slug of water with low boron concentration is formed in the primary loop.
  → risk of rapid power increase
- Two types of heterogeneous dilution events:

## External inhomogeneous dilution

Diluted or pure water slug is created by injection from the outside

- e.g. malfunction of CVCS system

May occur during all conditions

- Power operation
- Shutdown conditions
- During accidents

## Inherent dilution

Dilution takes place through an inherent phenomena

- Boiling-condensing heat transfer mode inside the primary system
- Backflow from the secondary side in case of primary-secondary leakage

During accidents

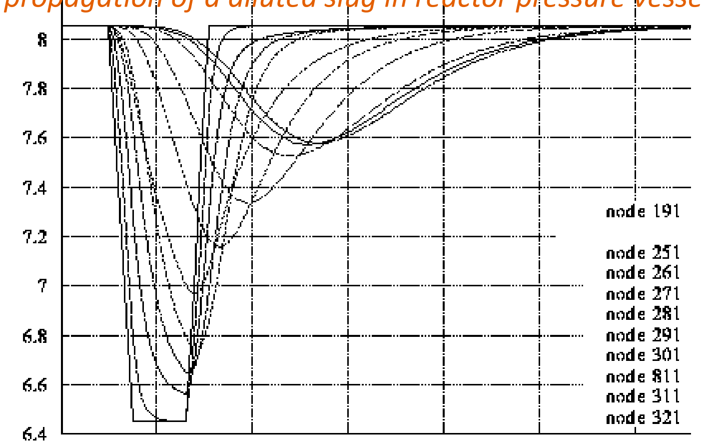
# Boron dilution

VTT

- In Loviisa VVER-440 several plant modifications have been done in 90s to prevent external dilution
- Propagation of the boron slug is challenging to model due to numerical diffusion
  - Often core response is modelled by giving slug properties directly at the inlet of a core
  - Numerical model that maintains sharp shape of the slug is needed
  - Reliability of CFD codes & mixing models; experiments are needed

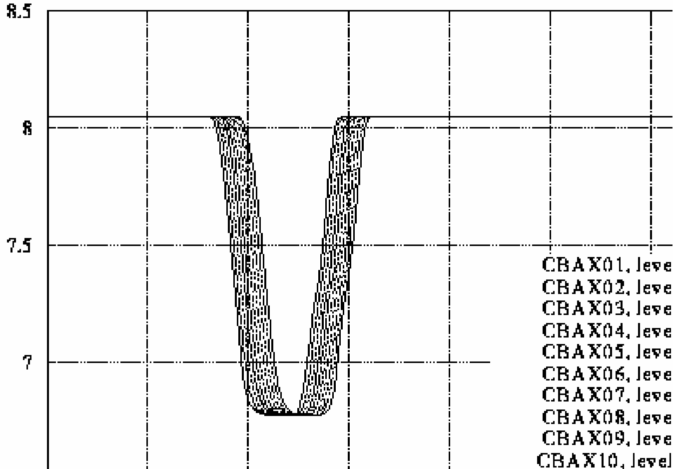
*Typical thermal-hydraulics code*

*Effect of numerical diffusion and mixing on the propagation of a diluted slug in reactor pressure vessel*




*HEXTRAN, APROS*

*Propagation of diluted slug in a core with HEXTRAN*



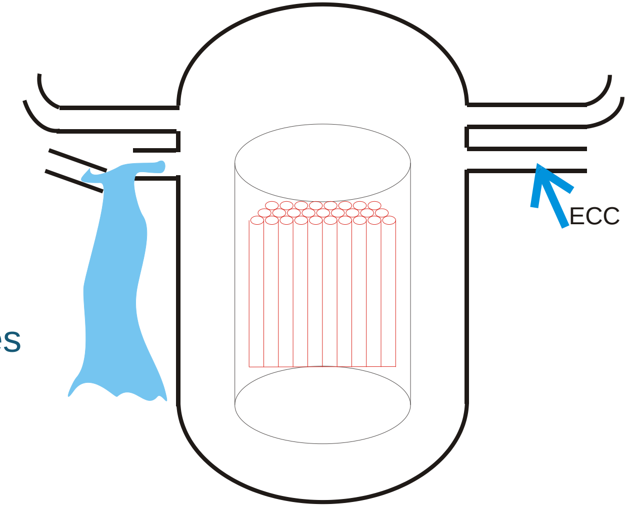
# Class 1 & 2 accidents

- These accidents are assumed much more unlikely than events in previous slides
- However, plant has to be designed so that also these accidents can be controlled and do not lead to severe consequences
- Plants have e.g. emergency core cooling systems and pressure limitation systems that has been designed for these accidents and are not needed for anticipated transients.
- Analysis of these accidents is an essential part of licensing process

# Loss of coolant accident LOCA

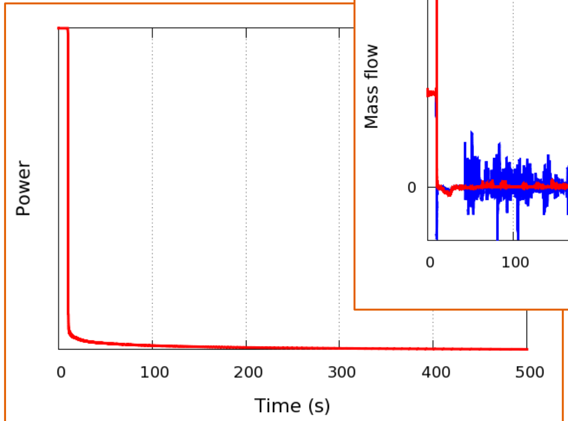
- Leak in a primary circuit
- Classification and consequences depend on leak size and location
- Design on many safety systems based on Large break LOCA (DBC4 / Design basis accident /condition IV)

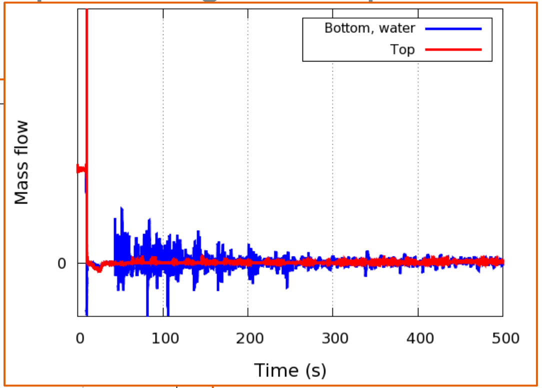
  - Double-ended break in a loop, flow area of the break 200% of cross section area of the pipe
  - Pressure decrease very fast
  - Scram & No coolant → fission power shuts down
  - **Decay heat** & no coolant →overheating of fuel assemblies

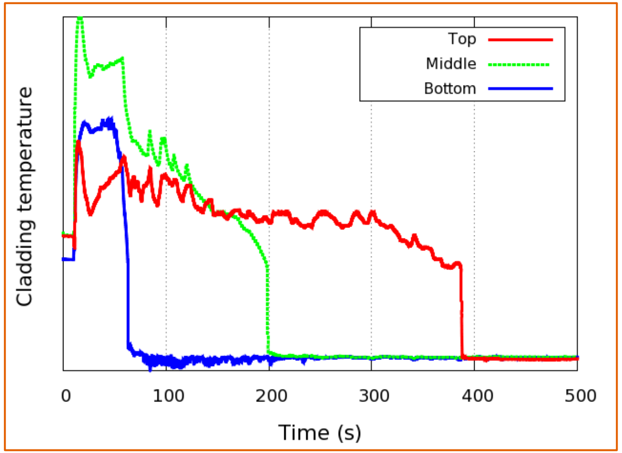


# Loss of coolant accident LOCA

- Nuclear power plant have to be designed so that:
  - core **can be filled** and it can be done fast enough.
  - reactor pressure vessel and steam generators can withstand loads
  - Containment can withstand pressure increase due to vaporized coolant
  - Local power (linear power) is not too high, because **decay heat** is proportional to preceding fission power




*FRAPTRAN-GENFLO simulation of LBLOCA*

# Control rod ejection CRE

- Inserted control rod flies away from a core very rapidly, typically in 0.1 seconds
  - Strong local power increase
  - Local temperature increase and boiling
    Power increase is cut of by Doppler phenomena
    before scram is activated or any other safety systems are able to react
- Not happened.
  - 2002 severe corrosion damages were found in RPV head of Davis-Besse. It has been assumed that ejection of several control rods would have been possible
- In BWR control rods are inserted from bottom
  → control rod drop corresponding RIA (reactivity initiated accident)

*Video: Relative fission power during CRE. 0.3 s*

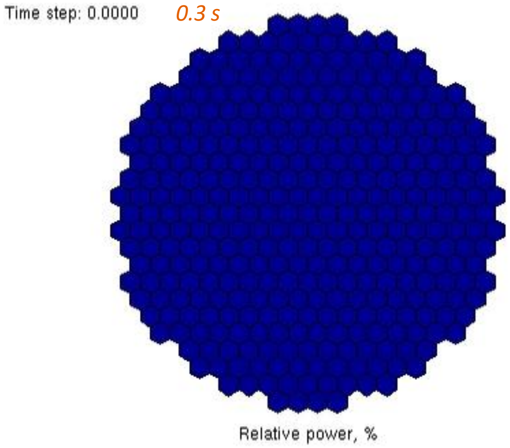
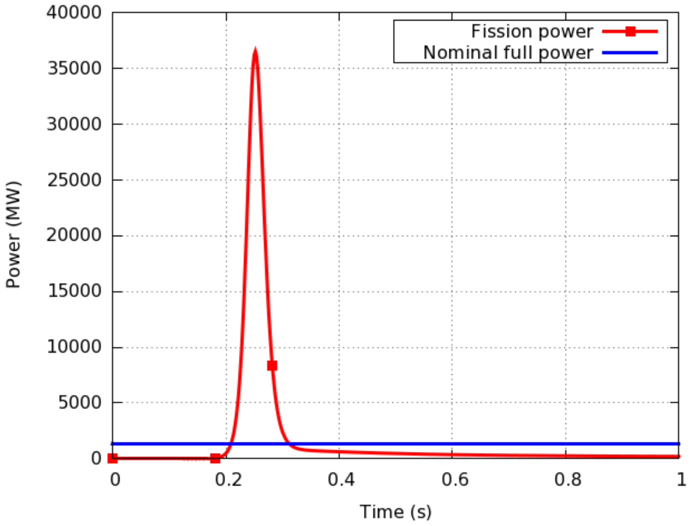
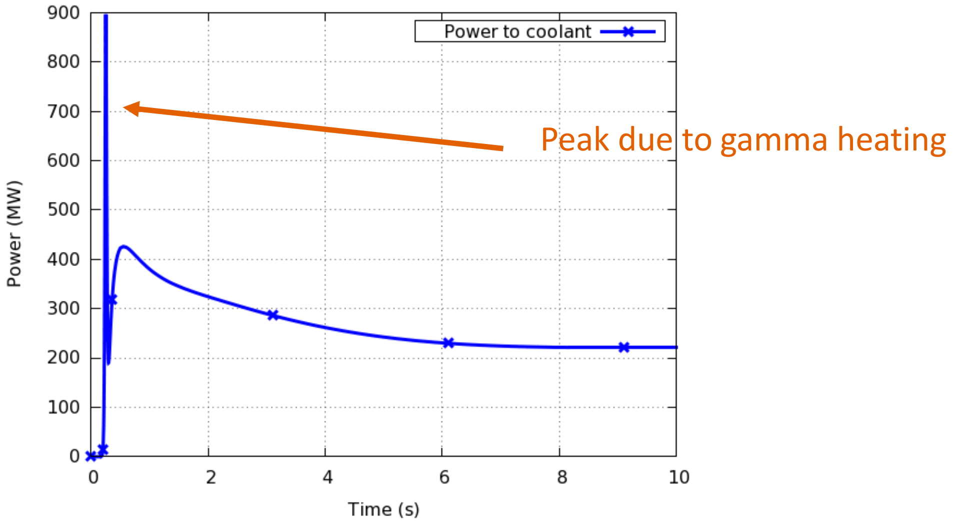
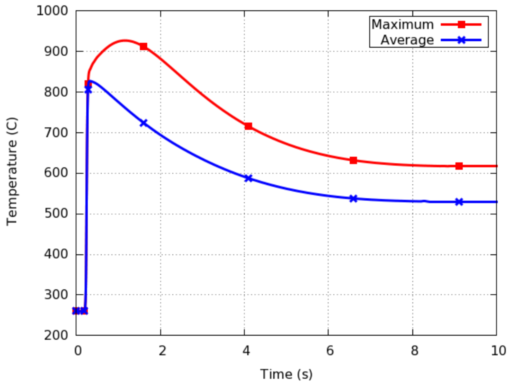



*Figure:en.wikipedia.org*

# Control rod ejection CRE




CRE is one of the accidents that may be more challenging at HZP (Hot Zero Power) state or at lower power levels than at full power

- Depends e.g. on fuel loading

*HEXTRAN simulation of the hypothetical VVER-440 CRE benchmark*

# Main steam line break MSLB

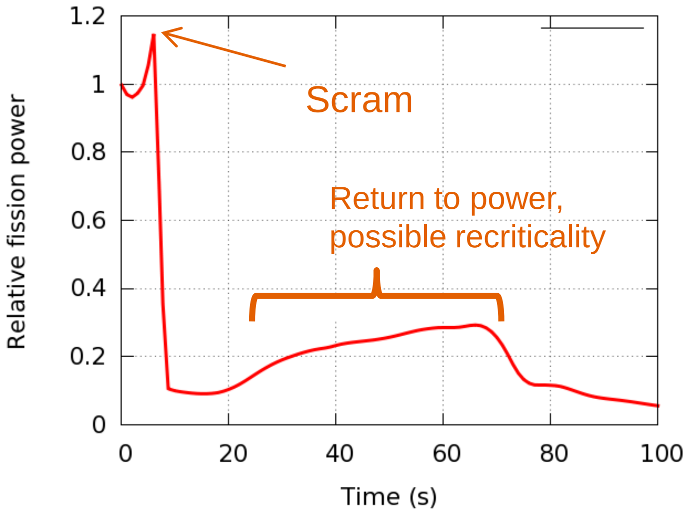
- Steam line is broken and steam flows to the environment
- Heat transfer from the primary to the secondary side continues

  -> secondary pressure drops

  -> primary water cools in steam generator

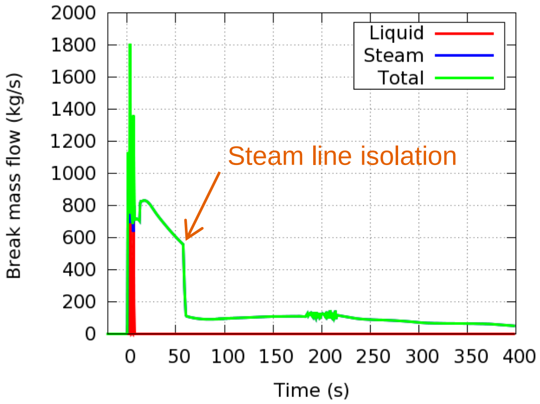
  -> water temperature at core inlet asymmetric and locally very cold

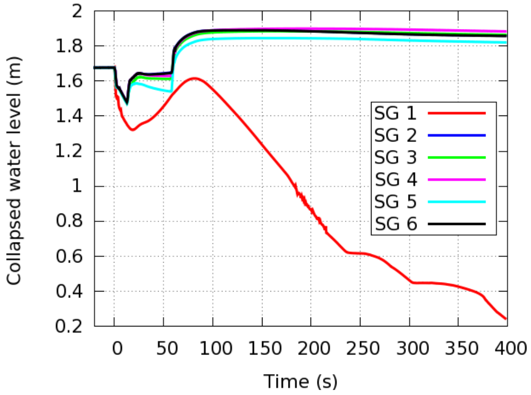
  -> possible recriticality and power increase **even if control rods have been inserted**. Boron injections are needed to ensure subcriticality.

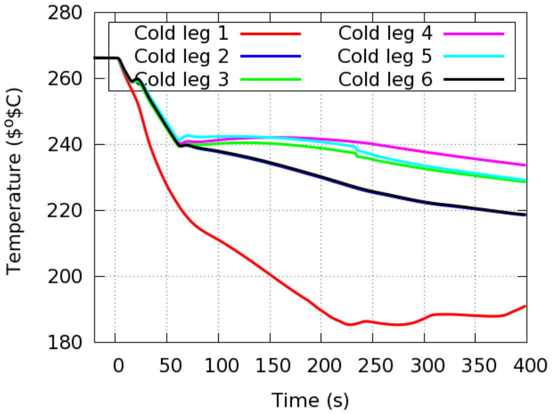


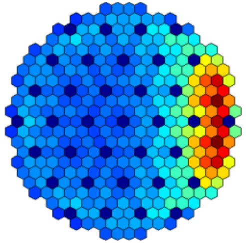

*TRAB3D simulation of the hypothetical TMI MSLB benchmark*

# Main steam line break MSLB



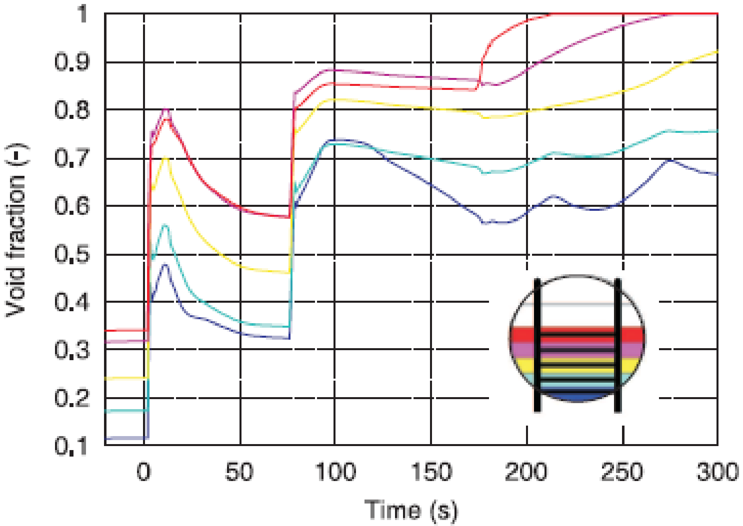



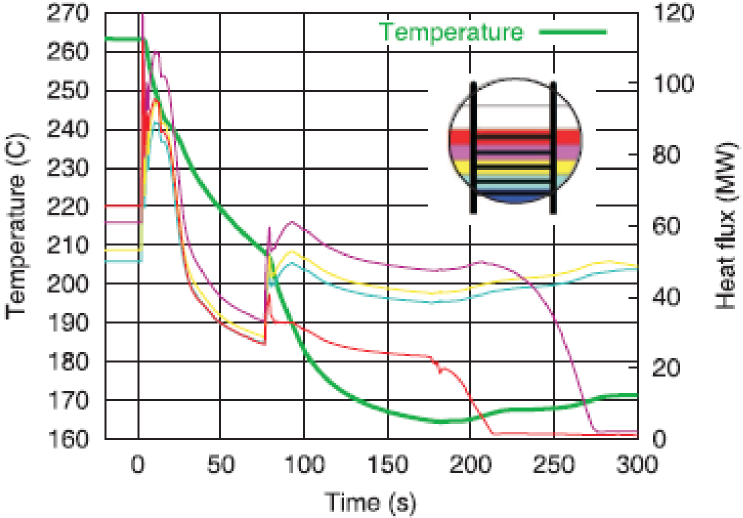
*Power distribution in a reactor core*

*HEXTRAN simulation of the hypothetical VVER-440 MSLB benchmark*

# Main steam line break MSLB




*Void fraction at riser side of steam generator.*




*Water temperature at cold leg 2 of primary circuit and heat flux from walls of the heat transfer tubes to secondary side.*

*HEXTRAN simulation of the hypothetical VVER-440 MSLB benchmark, break size 132%*

# PRISE Leak from primary to secondary circuit

- Failures in steam generator tubes lead to the leak of radioactivity to secondary side
- Risk of radioactive leak
  - Primary pressure > Secondary pressure
    → Flow from primary to secondary side
  - Important that secondary side valves do not open
  - Secondary side activity is continuously measured and thus leaking steam generator tubes can be detected soon and blocked.
  - Radiation doses analyzed assuming leaking SG tubes and open valves at the secondary side

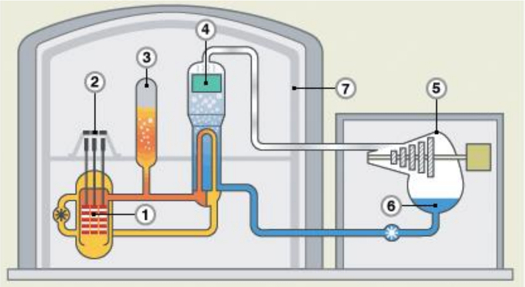



*Figure: BBC*

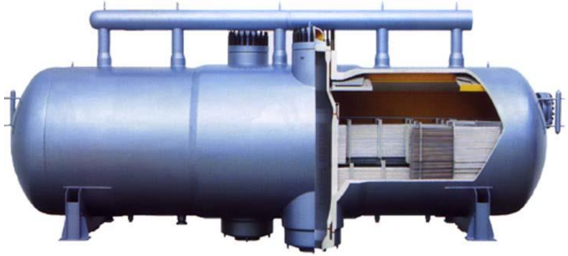
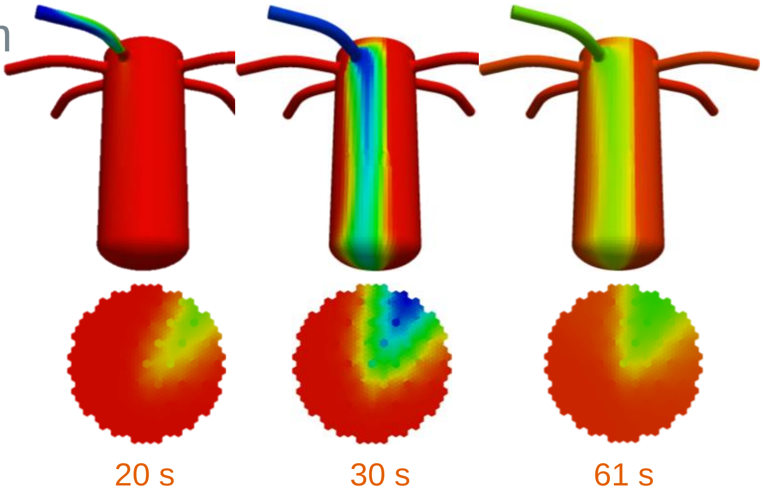


*Figure from Nucleartourist.com (original figure CEZ)*

# Other examples of AOOs and accidents

- Decrease in heat removal by the secondary system
  - Feedwater line break
- Reactivity and power distribution anomalies
  - Incorrect connection of an isolated reactor coolant system loop
- Increase in reactor coolant inventory
  - Inadvertent operation of emergency core cooling system or extra borating system
  - Malfunction of chemical and volume control system
- Decrease in reactor coolant inventory
  - Inadvertent opening of a pressurizer safety relief valve



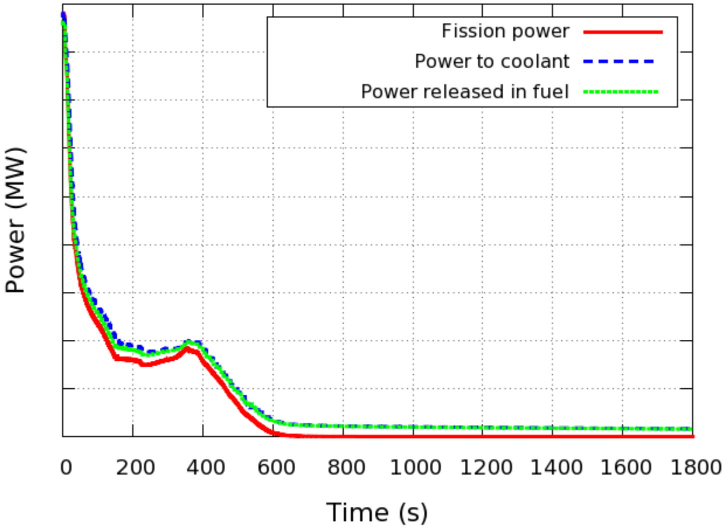

*Coolant temperature in VVER-440 RPV downcomer and at core inlet after connection of an cold, isolated loop*

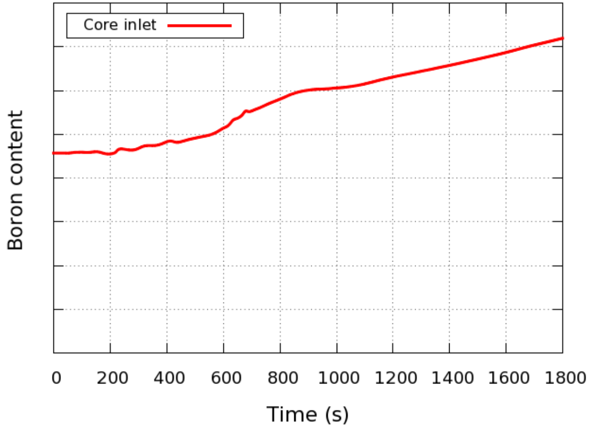
# ATWS

- Anticipated transient + failure of scram → DEC (Design extension condition)
- Scram may failure for several reasons, e.g
  - Faulty signal
  - Scram signal comes properly, but control rods do not move
- Acceptance criteria and initial assumptions for safety analyses differ from those used for corresponding events with scram
  - Not so strict failure assumption as in DBC2/DBC3/DBC4 (Condition II- Condition IV)
  - Higher maximum pressure  acceptable
  - No limitation for number of failed fuel rods
  - Higher radiation dose
  - Engineered safety systems as pressurizer safety valves and emergency boration system can be actuated
- Requirements for DEC cases are different in different countries, in many countries requirements have changed during last few years (after Fukushima)

# ATWS

- Example: loss of offsite power
- Power decreases
  - At first due to decrease of coolant flow (all RCPs trip)
  - Later due to boron injection




*TRAB3D-SMABRE simulation of LOOP+ATWS*

# Summary

- This lecture covered
  - Regulations for deterministic safety analyses
  - Safety analyses codes
  - Methods for safety analyses
    - Coupled 3D simulations, hot channel and hot rod analyses
  - Different type of transients and accidents

## Questions?