Safety analyses and reactor dynamics

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

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Outline

Regulations

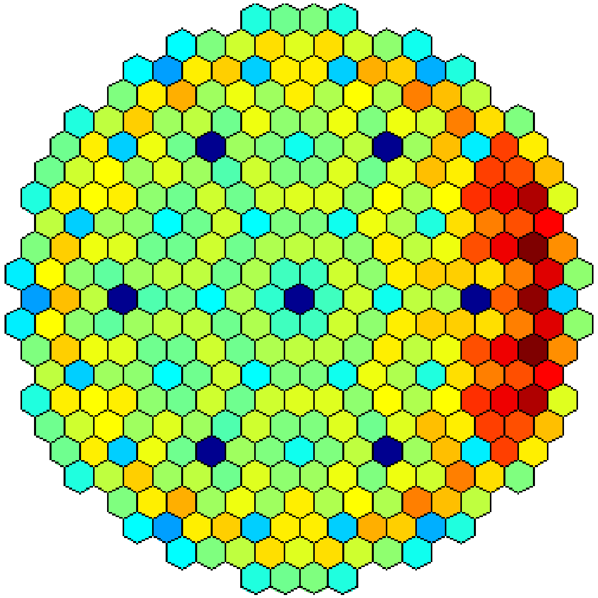
Classification of transients and accidents

Safety analyses codes

Methods and assumptions for safety analyses

Results of interest

Description of transients and accidents



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CLASSIFICATION OF TRANSIENTS AND ACCIDENTS

PUBLIC

Syrjälahti Elina

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.

Background

Nuclear energy act

7 d § (2008): The design of a nuclear facility shall provide for the possibility of **operational occurrences** and **accidents.**

7e § (2017) : Compliance with the requirements concerning the safety of a nuclear facility shall be reliably proven. The overall safety of a nuclear facility shall be assessed at least at 10-year intervals

Overall reform of the Nuclear Energy Act is in preparation

Nuclear energy degree

The applicant shall submit the

- preliminary safety analysis report (**PSAR**) when applying for a construction licence
- the final safety analysis report (**FSAR**) when applying for an operating licence
- (PSAR, FSAR) shall include
  … **a description of the behaviour of the facility during accidents…**

Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant

The safety of a nuclear facility shall be assessed when **applying for a construction license** and **operating license,** in connection with **plant modifications**, and at **Periodic Safety Reviews** during the operation of the plant.

It shall be demonstrated in connection with the safety assessment that the nuclear facility has been designed and implemented in a manner that meets the safety requirements.

The safety assessment shall cover the **operational states** and **accidents** of the plant.

Background

New regulatory guides on nuclear safety (**YVL Guides**) compiled by STUK came into effect on December 2013.

- Some updates after that

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
- severe reactor accidents.

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

Classification of events

- 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.

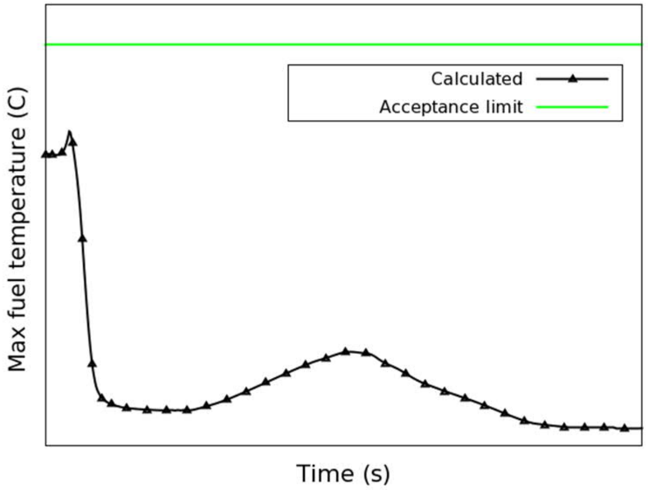
General

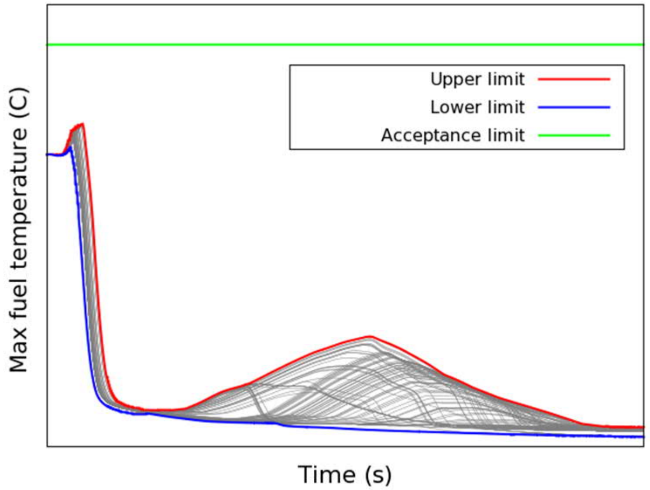
Acceptance criteria shown in following slides are for conservative analyses

- 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

Common 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

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)

Different requirements for different classes

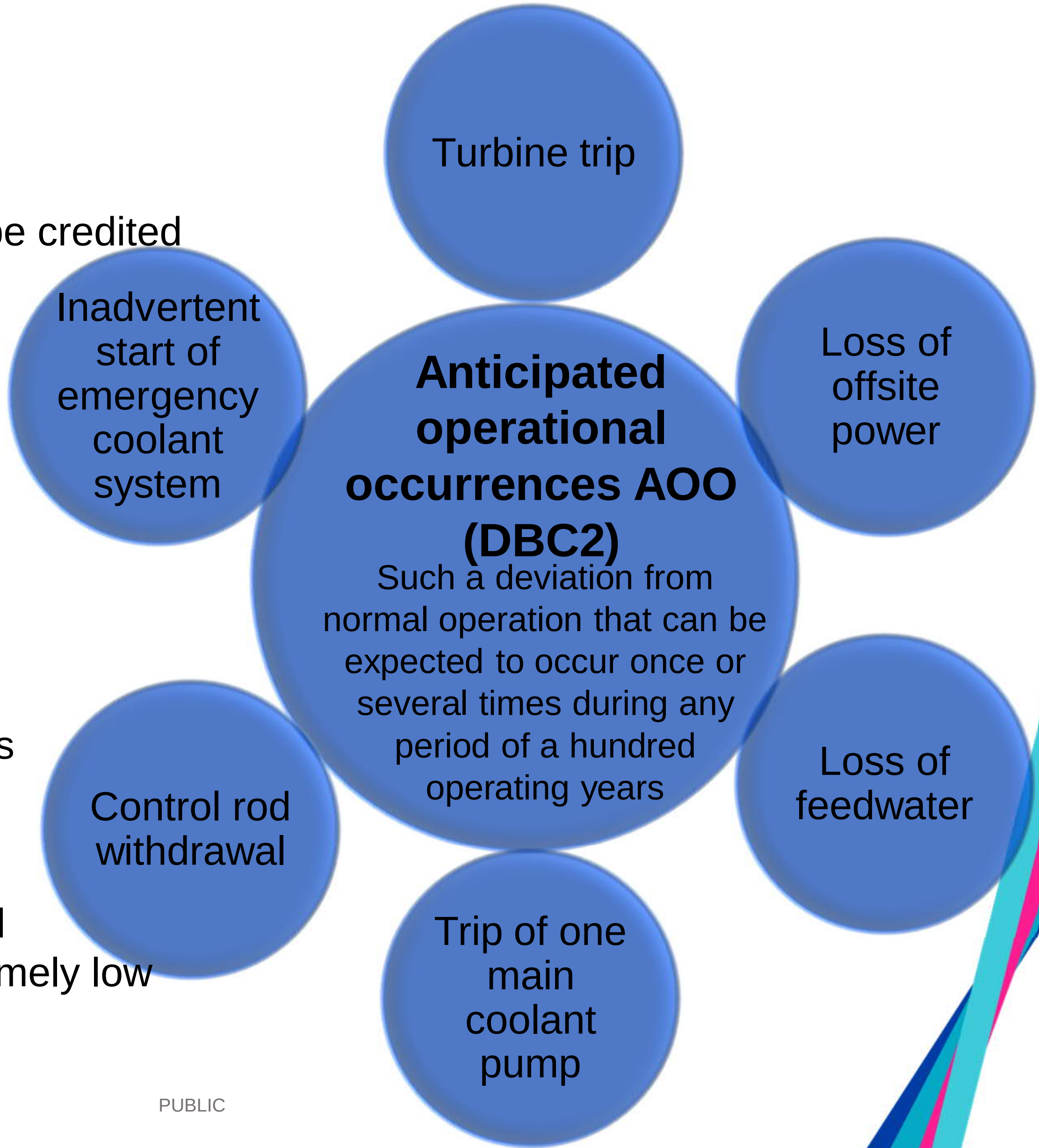
Odotettavissa oleva käyttöhäiriö

Analyses

- Conservative analyses with failure assumption
- Actuation of non-safety classified systems cannot be credited

Some acceptance criteria

- Must not require the initiation of safety systems designed for postulated accidents
- Pressure < design pressure
- 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 number of rods reaching heat transfer crisis <0.1 %
- The probability of fuel failure caused by mechanical interaction between fuel and cladding (PCMI) extremely low



Oletettu onnettomuus

Analyses

- Actuation of non-safety classified systems cannot be credited
- Loss of the external grid shall be combined with postulated accidents if it could aggravate the consequences of the initiating event.

Some acceptance criteria

- Pressure < 1.1 * design pressure; 1.1*Containment pressure < design pressure
- 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

Postulated accidents, class 1 (DBC 3)

Assumed to occur less frequently than once over a span of one hundred operating years, but at least once over a span of one thousand

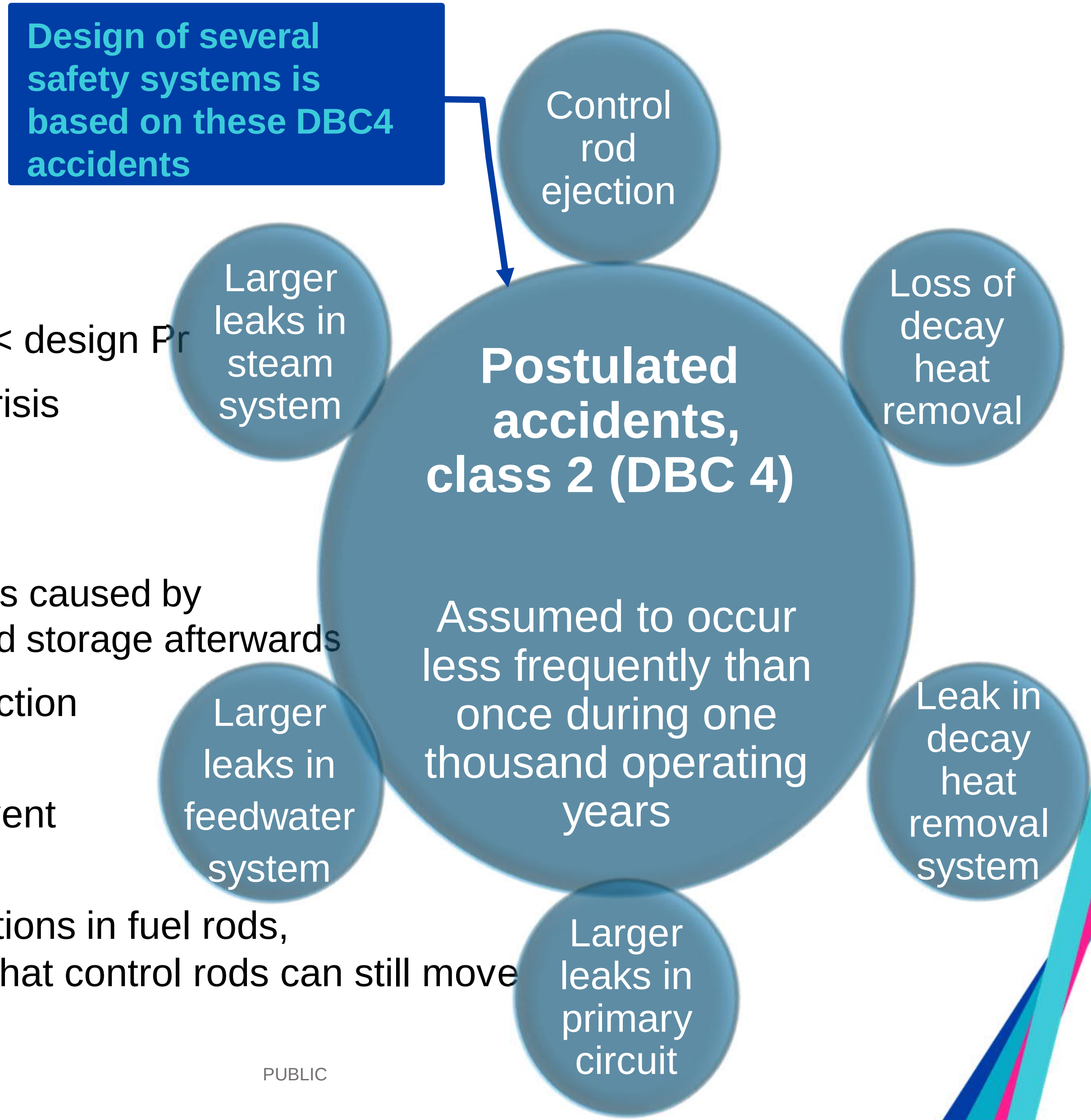
- Small leaks from primary circuit
- Small leaks from primary to secondary side
- Small leak of steam
- Small leak in feedwater system
- Unexpected closing of steam line isolation valves
- Loading of a fuel assembly into an improper position

Oletettu onnettomuus
Analyses

- Mainly as in DBC3
- Only safety class 2 systems can be credited

Some acceptance criteria

- Pressure < 1.2*design Pr; 1.1*Containm. Pr < design Pr
- 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 afterwards
- Hydrogen generation due to the chemical reaction between coolant and cladding < 1%
- Maximum enthalpy of fuel 963 J/gUO_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



Oletetun onnettomuuden laajennus

DEC A:
DBC2 or DBC3
+
a common cause failure in a safety system

Design extension conditions DEC

DEC C:
an accident caused by a rare external event

DEC B:
an accident caused by combination of failures (identified as significant on the basis of PRA)

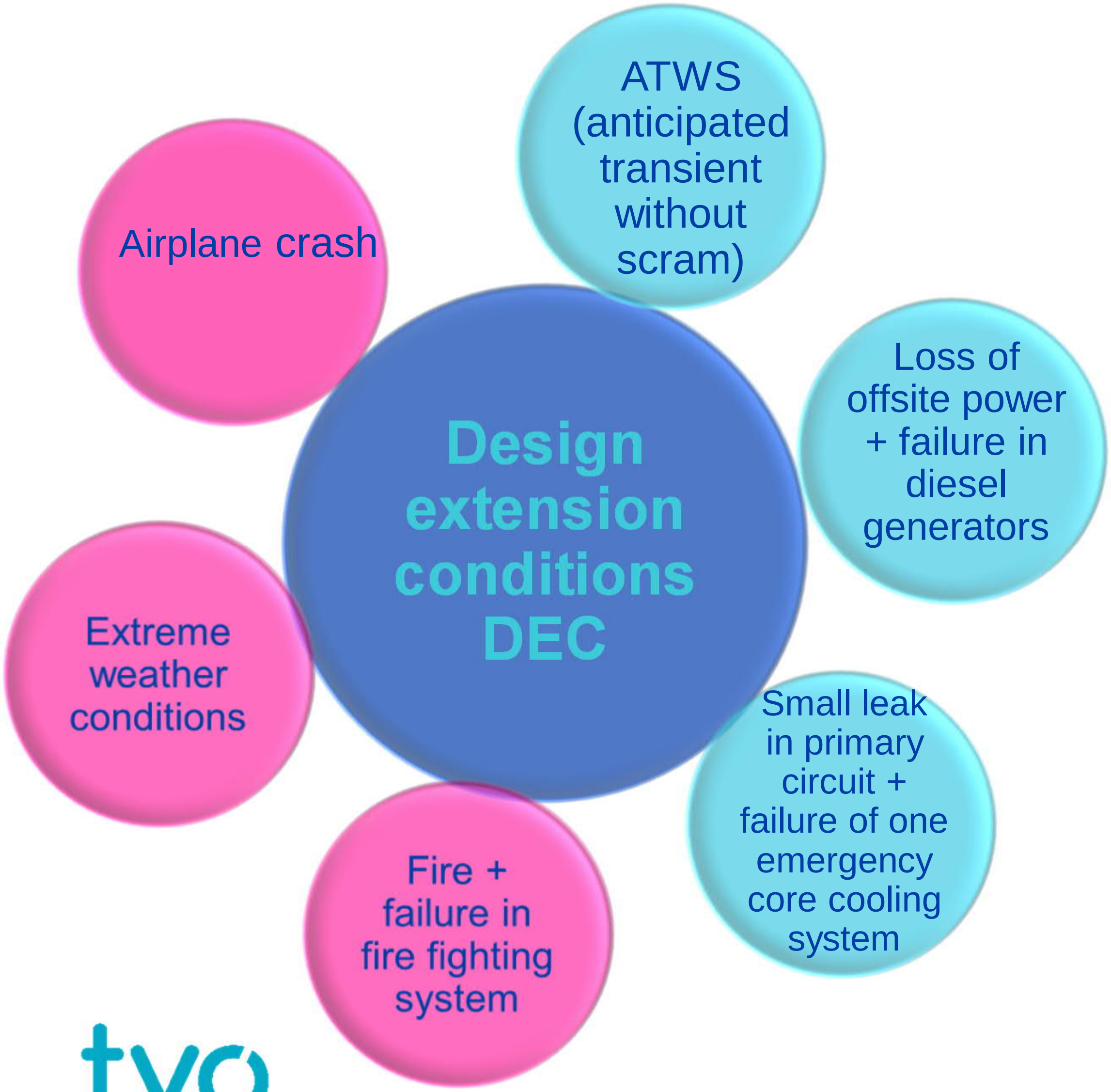
For DEC A accidents, the most penalizing single failure shall be assumed in one of the systems whose operation is required to accomplish a safety function in the event in question.

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



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 onnettomuudet

In severe accidents:

Considerable part of the fuel in a reactor loses its original structure

Requirement:

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

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

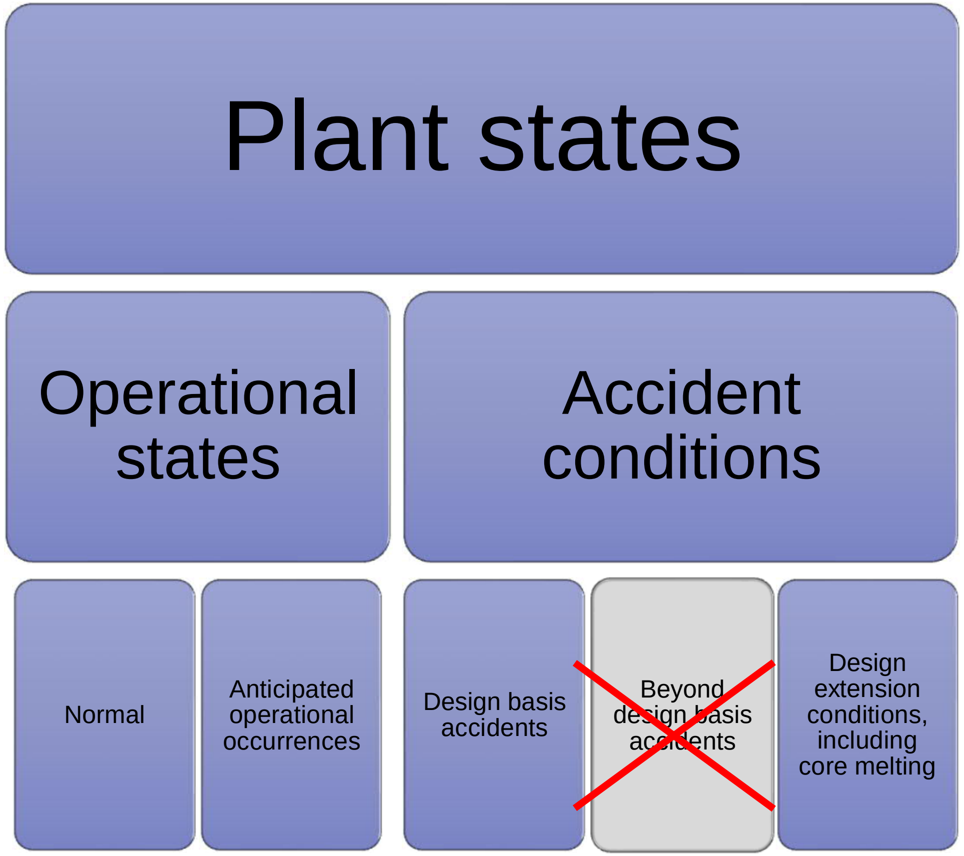
1.5*Containment pressure (including pressure increase due to hydrogen burn) < leaktightness limit

Summary of some acceptance criteria

Regulations in Finland:

Class	Frequency per year	Number of failed fuel rods DNB	Number of failed fuel rods 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



Safety guide SSG-2 Deterministic Safety Analysis for Nuclear Power Plants 2019

- 2009 version: 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
  - with fuel melting
  - without fuel melting

References

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

Reactor dynamics modelling

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?

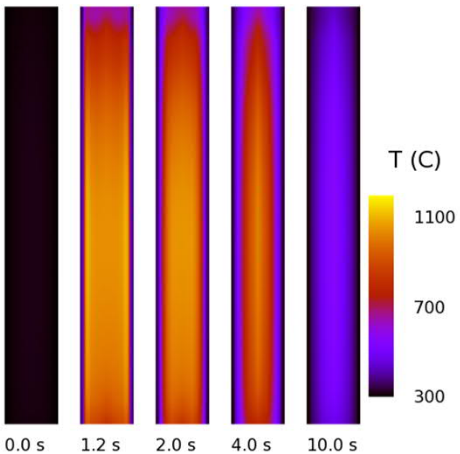
Maximum pressure in containment?

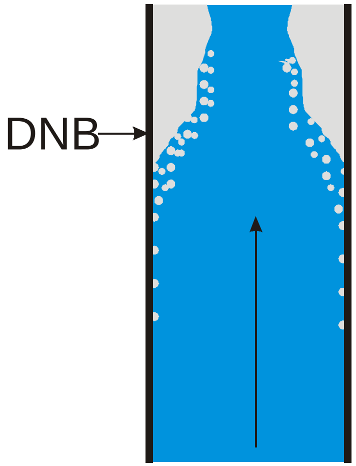
Possible dose ?

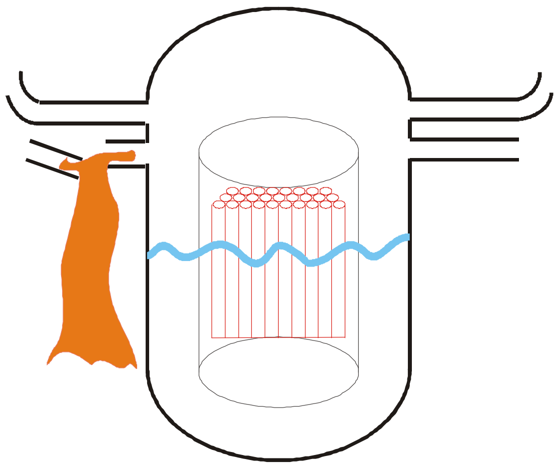
Background

Basically cooling of the fuel pins can be threatened from three different reasons:

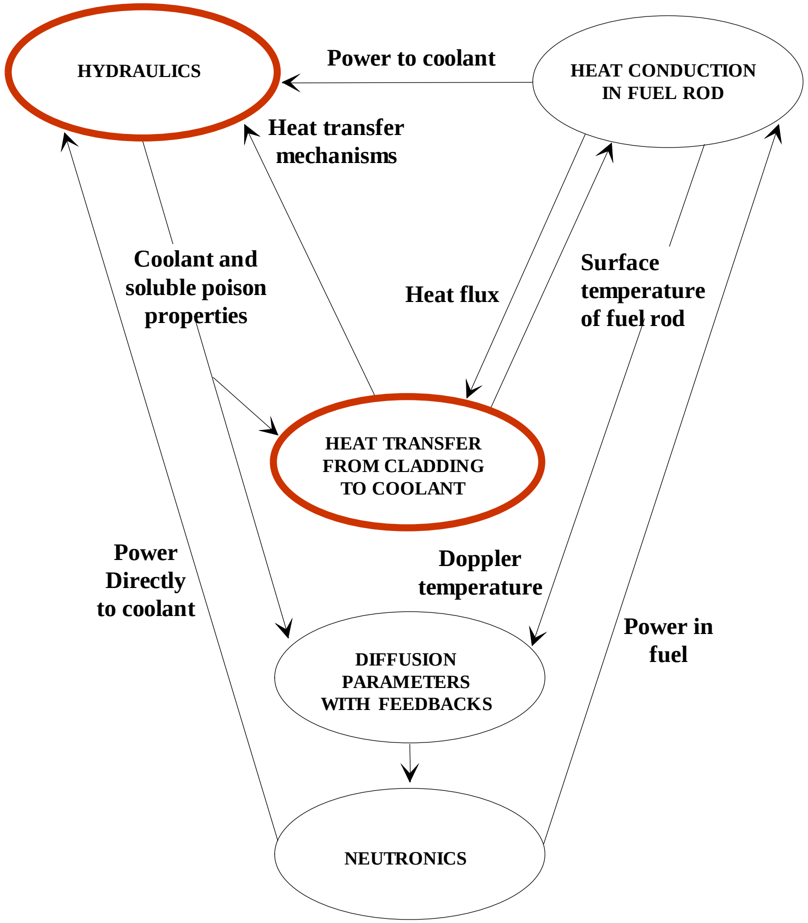
- Power increases so fast that heat cannot transfer from pellet to coolant
- Heat flux from a fuel rod to coolant is too high compared to coolant flow, and the surface of the fuel rod dries
- There’s not enough water in a reactor due to leaks







Reactor dynamics



Coupling of physical processes

in the core of a light water reactor

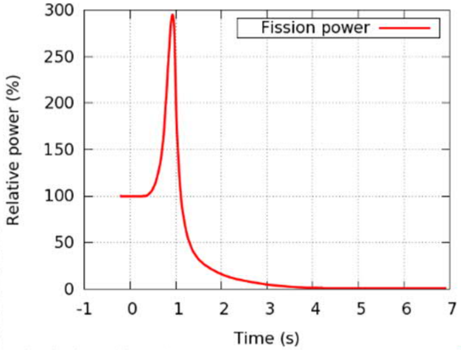
Reactor dynamics – transient analysis

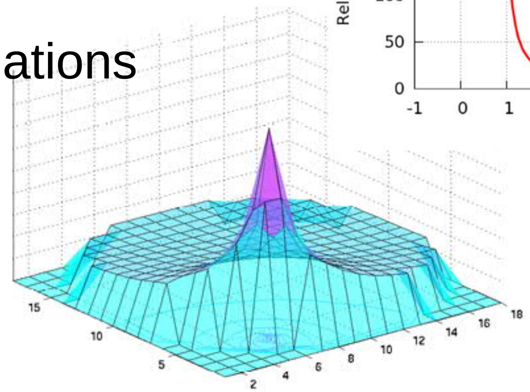
Events in which fission power development is important and its spatial distribution changes during the transient.

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

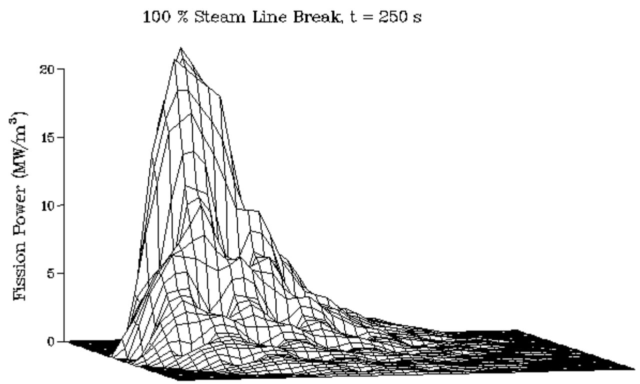
- Modelled period typically from couple of seconds to couple of hours
- Conservative reactivity properties in safety applications
- Best estimate in validation





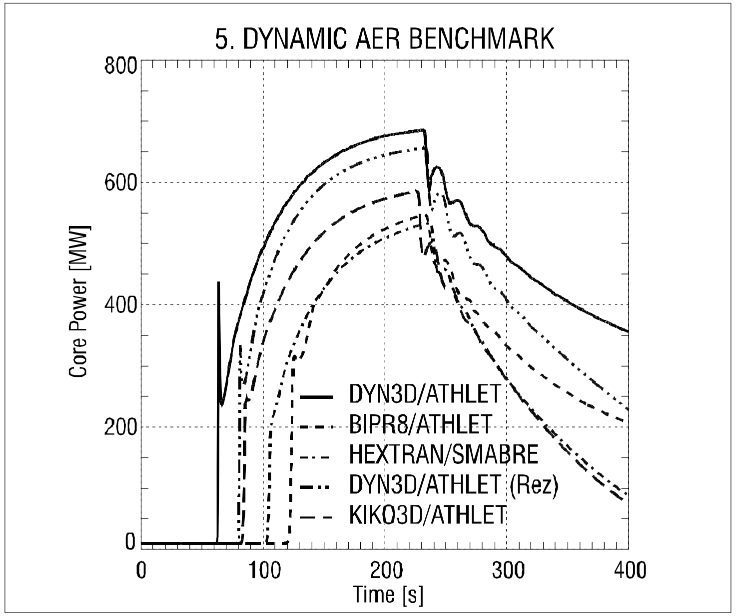
Reactor dynamics simulation – typical results

- 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



Examples of coupled neutronics/thermal-hydraulics codes

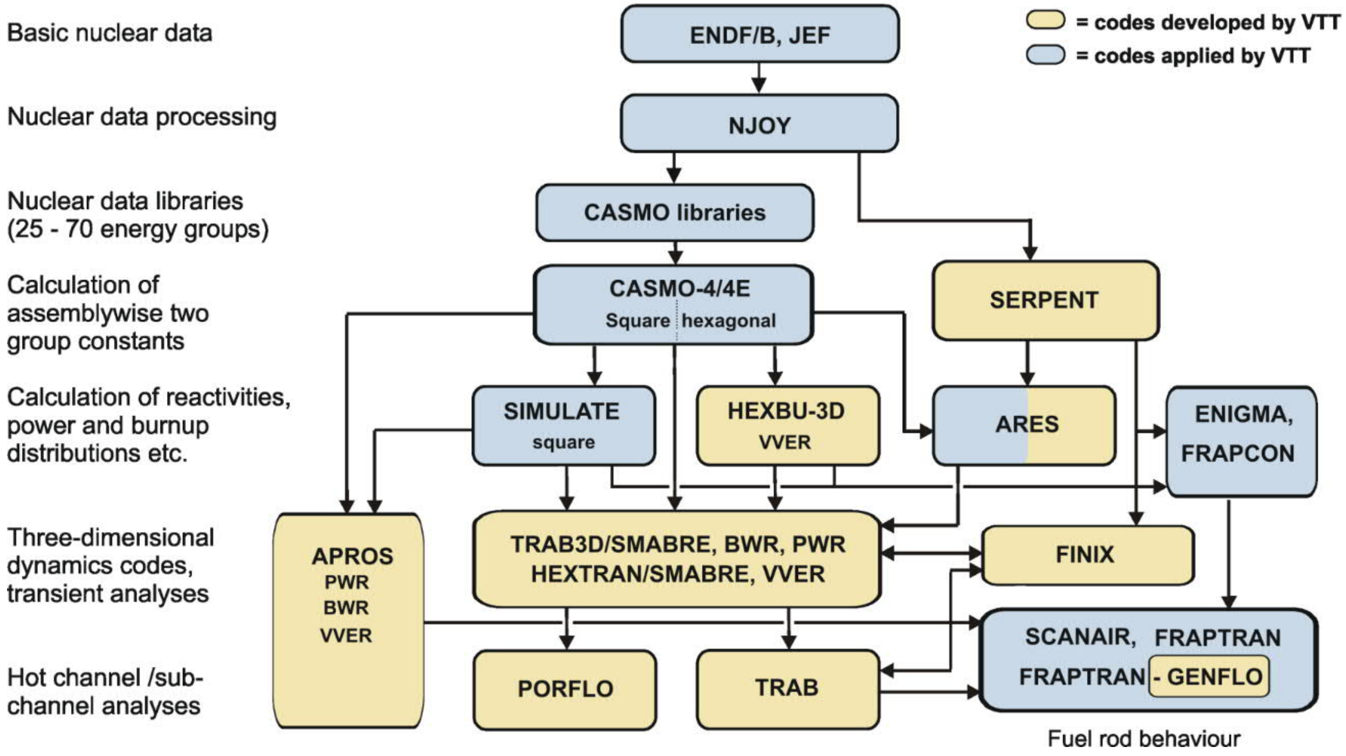
- **TRACE-PARCS**, U.S.NRC
- **SIMULATE-3K**, Studsvik
  - also couplings e.g. with RELAP5, TRACE, APROS
- **POLCA-T**, **BISON (1D)**, Westinghouse
- **ARCADIA**, Framatome
- **QUABOX-CUBBOX-ATHLET**, GRS, Germany
- **DYN3D-ATHLET**, HZDR, Germany
- **BIPR-ATHLET**, KI, Russia
- **KIKO3D-ATHLET**, HAS 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

VTT's Reactor Analysis Code system

Codes in the production use in 2019.
Renewal of the code system is in progress.



Steps of the reactor dynamical safety analyses

1. Group constants (Cross sections) for each material
2. Burnup and possible void/temperature/boron history for each neutronics node
3. Fuel properties from stationary fuel performance code
4. Conservativity tuning
5. Simulation of an event
6. DNB (hot channel calculations)
7. Uncertainty analysis
8. Fuel rod analysis with transient fuel performance code

Core modelling, VTT's codes as an example

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.

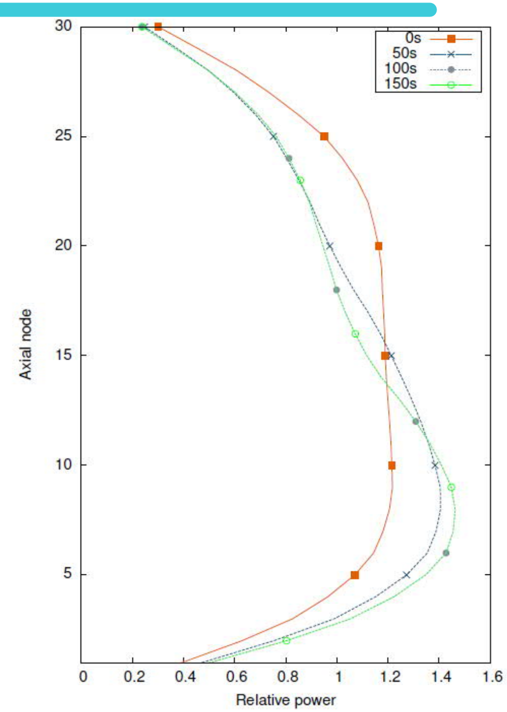
- Liquid mass, steam mass, mixture momentum, mixture energy

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

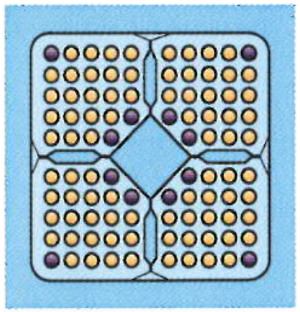
PWR/VVER circuit modelling with the system code SMABRE



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

Core modelling in reactor dynamics codes

- Each fuel assembly and channel is typically modelled separately
  - 163-500 channels, depending on a reactor type
- Axially from 20 to 30 nodes
  - This size (nodes ~ 10-20 cm) is optimal for widely used nodal method
- Modelling of modern fuel assemblies with
  - Water rods
  - Part length fuel rods
  - Axial discontinuities, e.g. Gd pellets, axially heterogeneous control rods



Westinghouse Optima-3

- Heat transfer is calculated with simplified models
  - Typically gas gap conductance with simple correlations, no models or only coarse models for dimensional changes etc.
  - Typically one average fuel rod in each assembly
  - Radially ~10 mesh points in a fuel pellet
- Nowadays also more detailed temperature calculation is possible
  - Pellet and cladding deformations, possibly also gas composition
    - S3K contains INTERPIN models, HEXTRAN & TRAB3D coupled with the FINIX fuel module
    - Pin-by-pin calculation of temperature based on pin-power-reconstruction, no feedback to neutronics

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

General solution of the two group diffusion equation is Linear combination of two characteristics equation (modes). Fundamental mode is polynomial function, transient mode exponential function

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

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

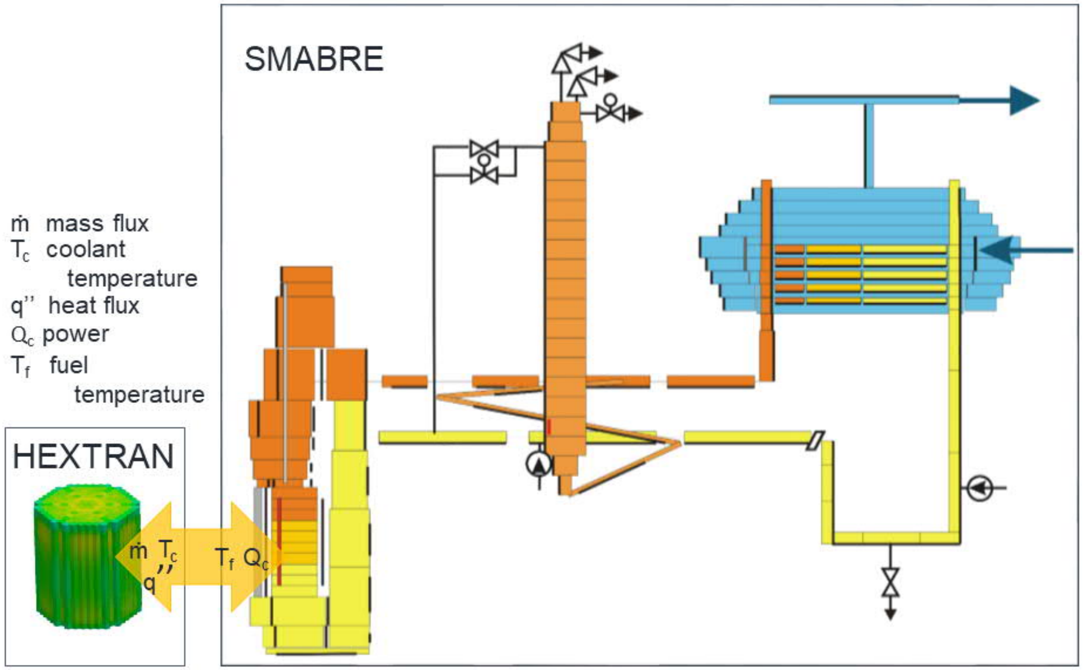
Coupling of neutronics and thermal hydraulic codes

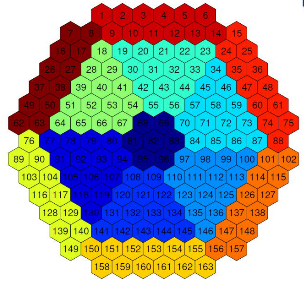
Internal coupling:
neutronics and thermal hydraulics in different codes

External coupling: different code for the core and loop

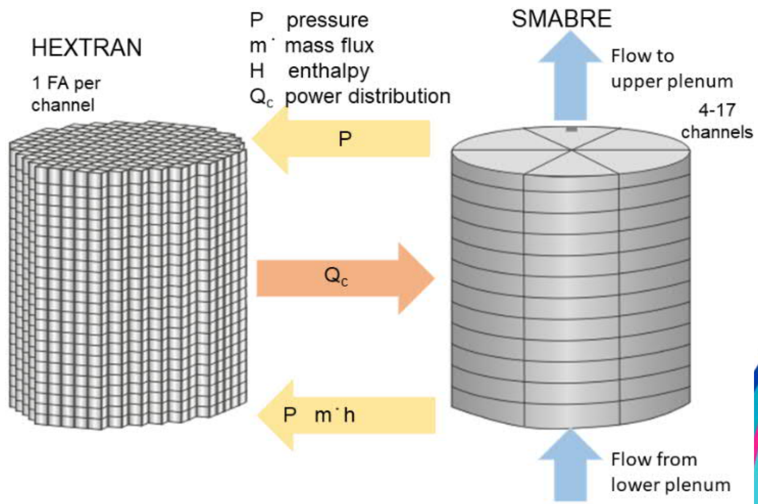
Parallel coupling:
Core thermal hydraulics is calculated with both codes
Totally independent codes coupled together

Internally coupled HEXTRAN-SMABRE





Parallelly coupled HEXTRAN-SMABRE



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

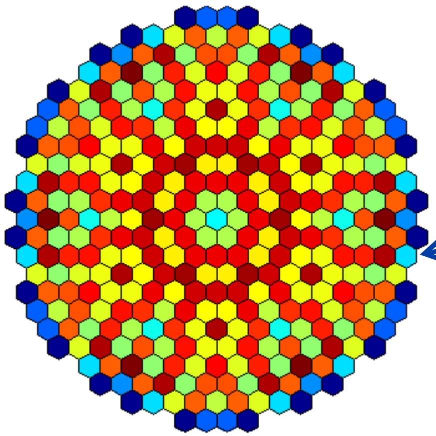
Core loading

- Burnup
- Cross sections

Fuel assemblies

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

Reactor core

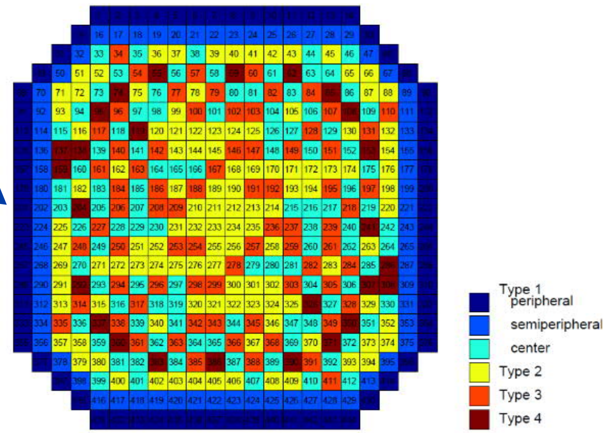


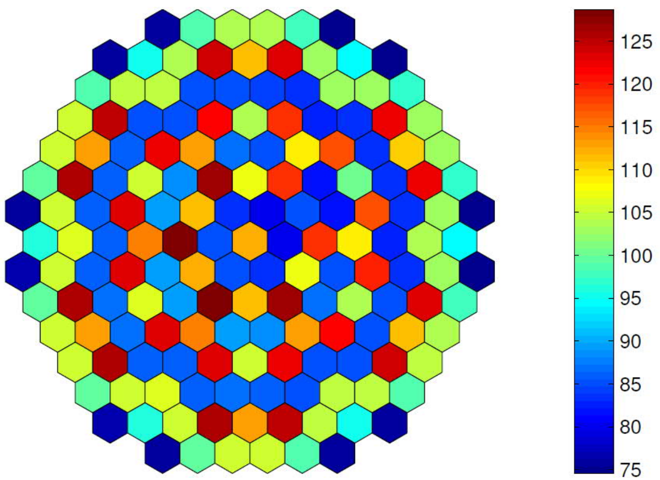
Power distribution
VVER-440 BOL

- 3 different enrichments

Fuel assemblies in BWR core

- 4 types of assemblies





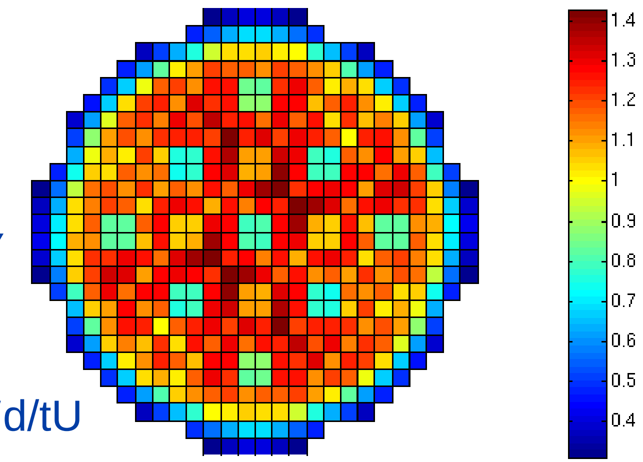
Relative power (%), t=0s

Power distribution
VVER-1000

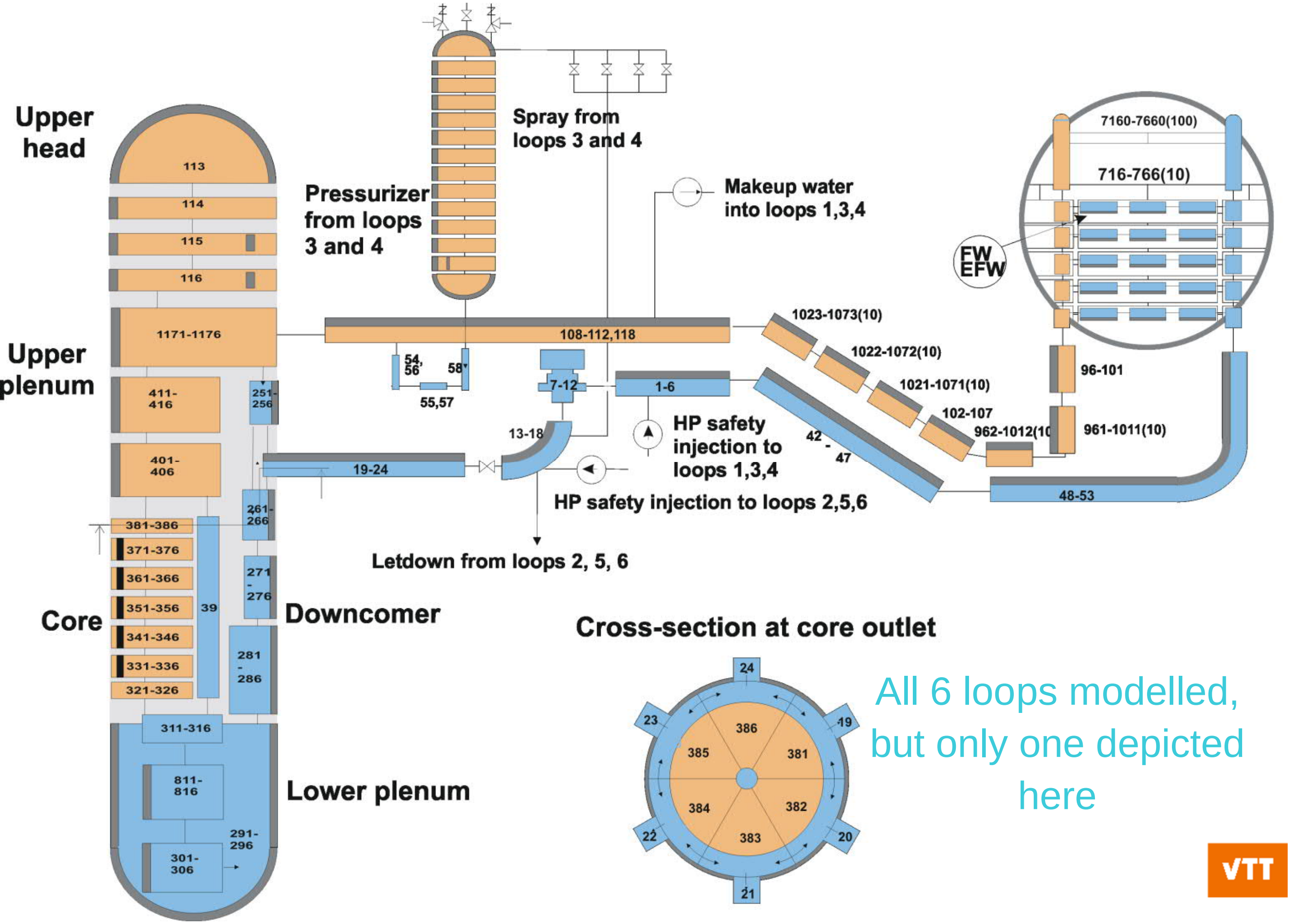
- One assembly has been replaced with different FA

Power distribution in BWR

- 2 types of assemblies
- Assembly burnup 0.14-32.9 GWd/tU



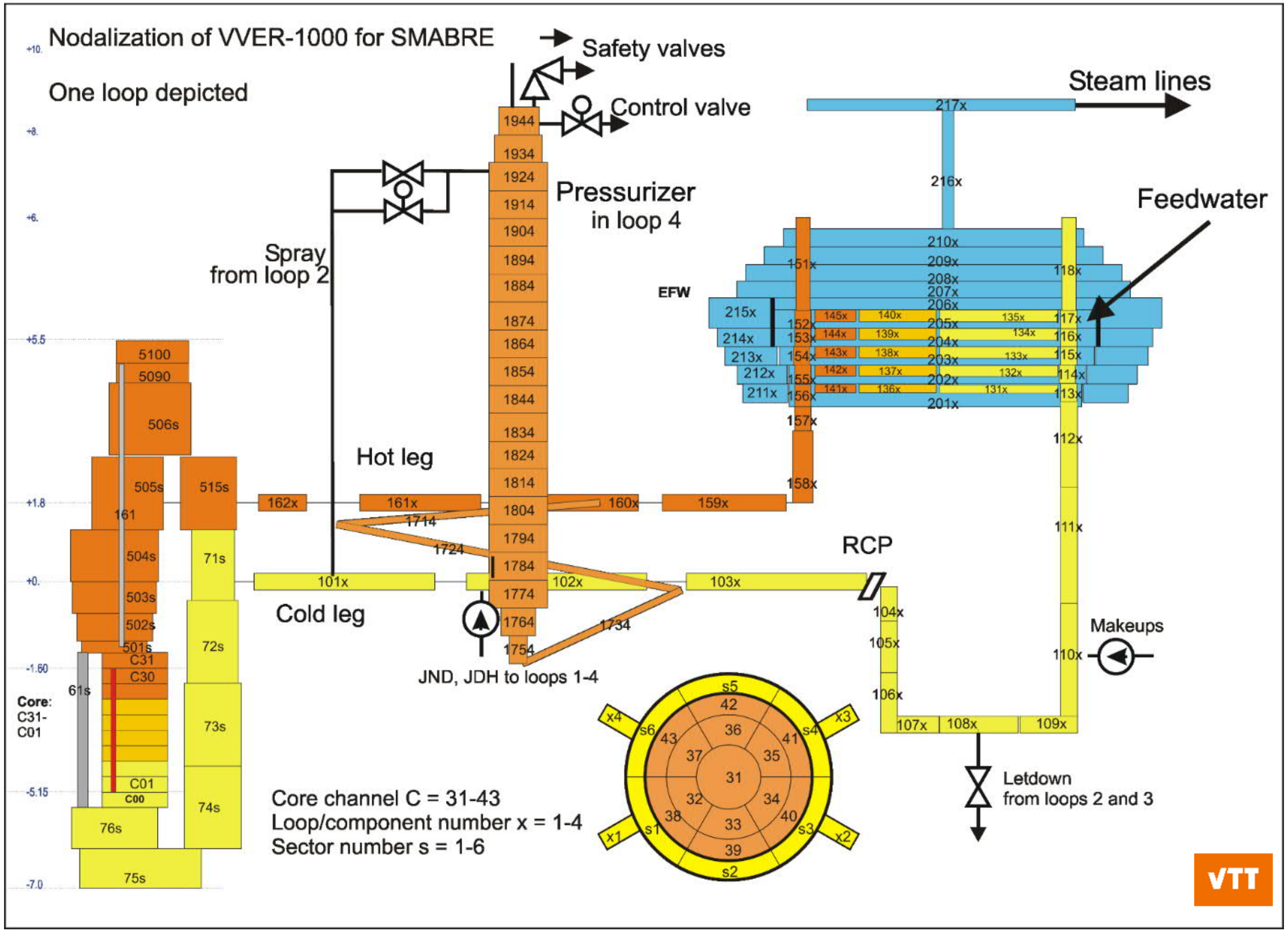
SMABRE:
Primary loop nodalization for Loviisa VVER-440



All 6 loops modelled, but only one depicted here

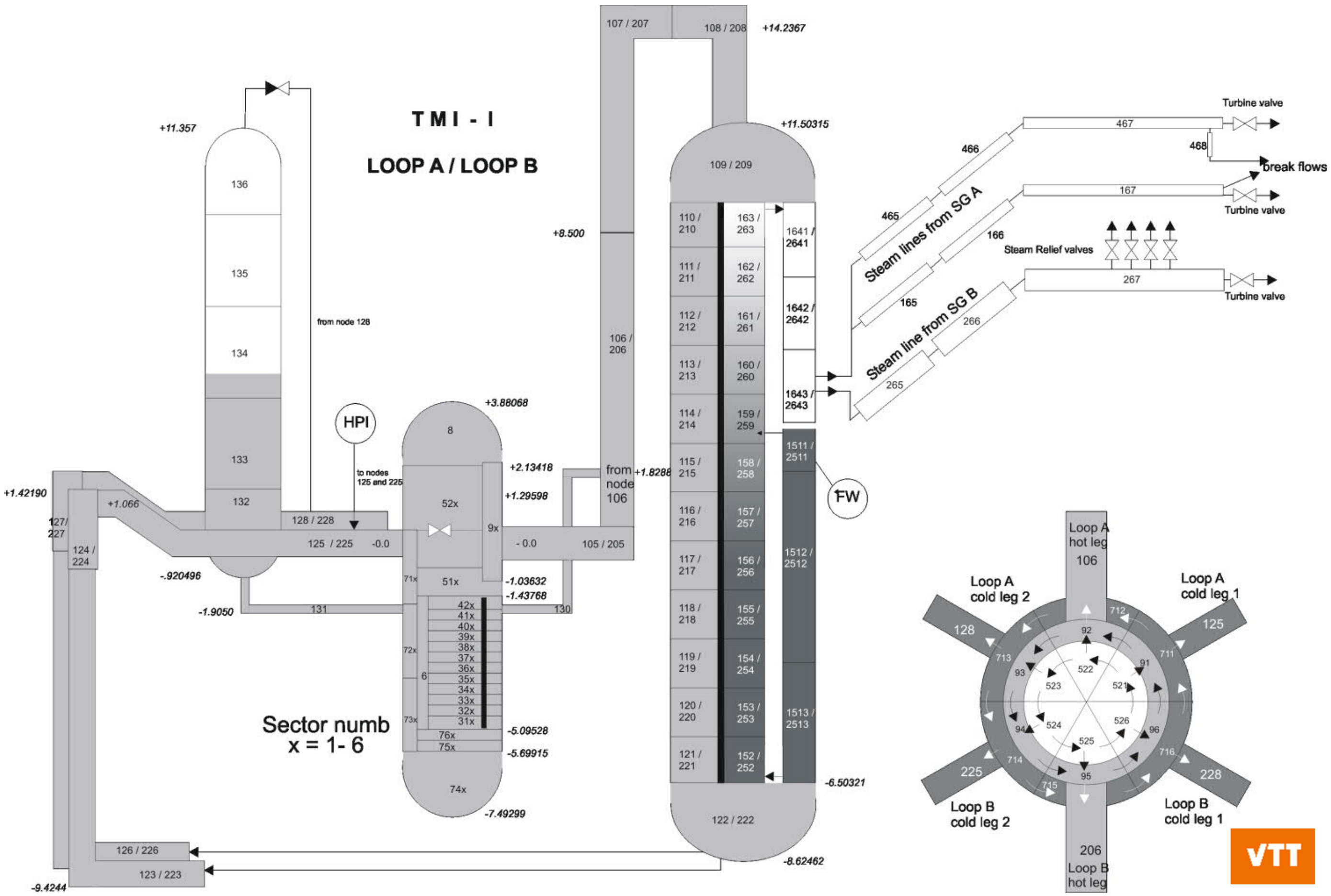
- Modelling of NPP for system code takes about 1 year
- 3 dimensional hydraulics (CFD) impossible for all the tanks, pressure vessel, pressurizer, steam generators for longer transients
- 1 dimensional hydraulics applied for 3-D phenomena with suitable nodalization
  - Sectors in RPV,
  - Dense nodalization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

Primary loop nodalization for VVER-1000



SMABRE model for VVER-1000 (Kalinin-3)

Nodalization of TMI-I for OECD MSLB benchmark



VTT's SMABRE model for 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

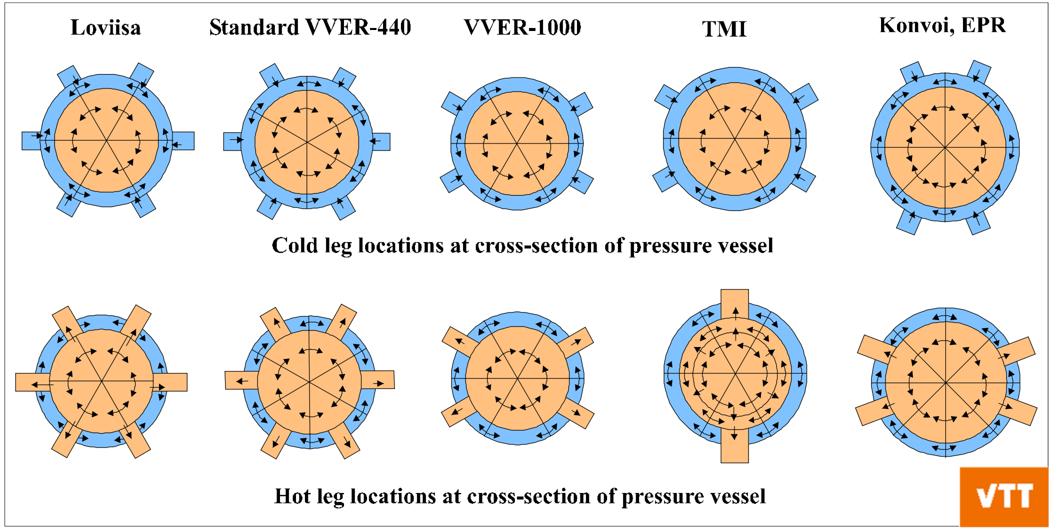
SMABRE: modelling of mixing

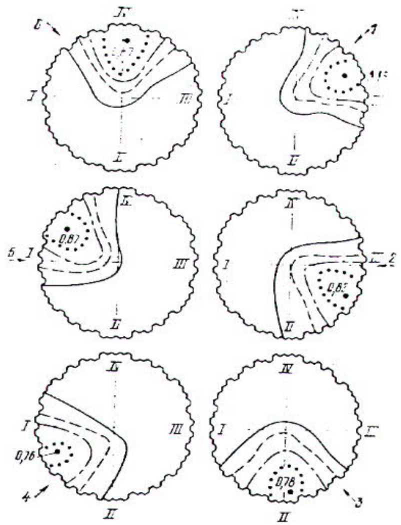
System codes have only limited capabilities to model mixing

- No 3D solution, coarse nodalization

In VTT's SMABRE code 3D effects are taken into account by means of reactor vessel nodalization (sectors)

- 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

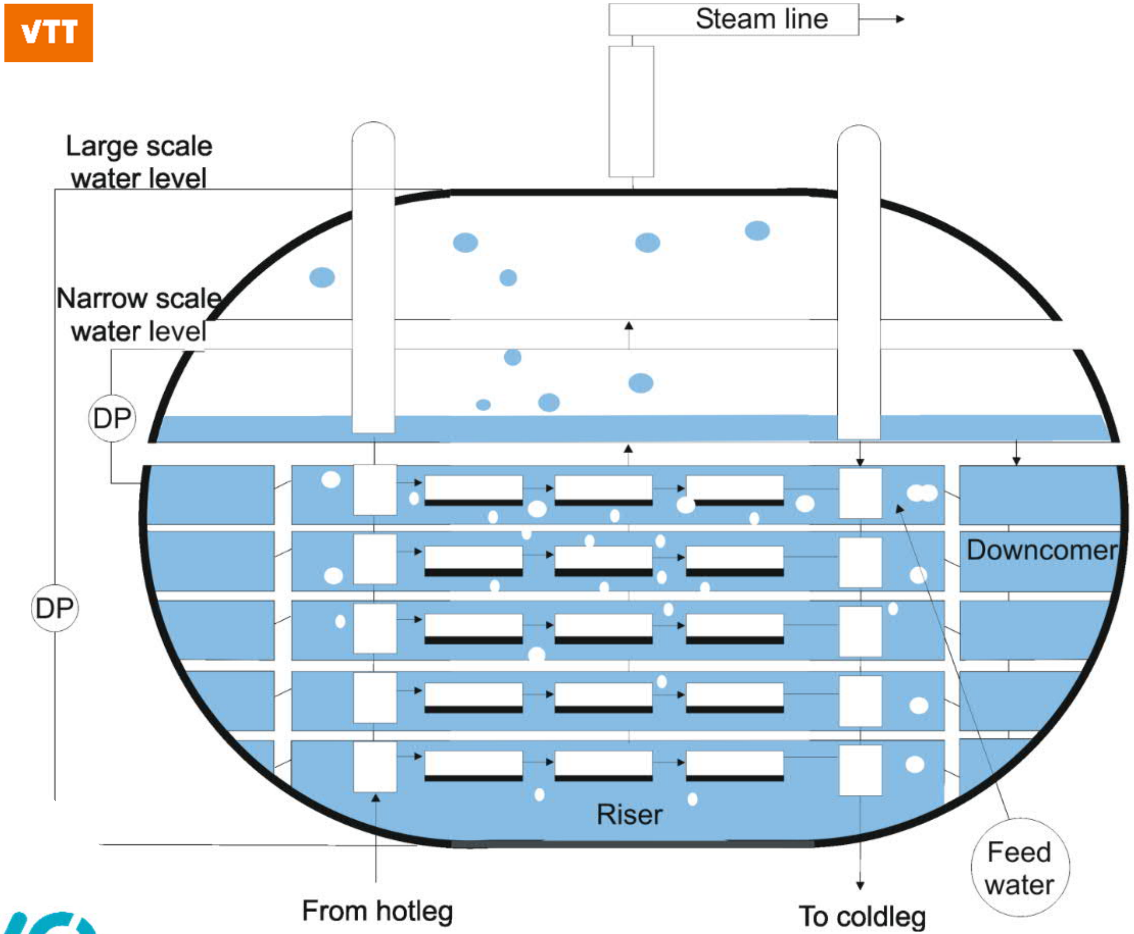




Temperature measurements in Loviisa VVER-440 at core inlet

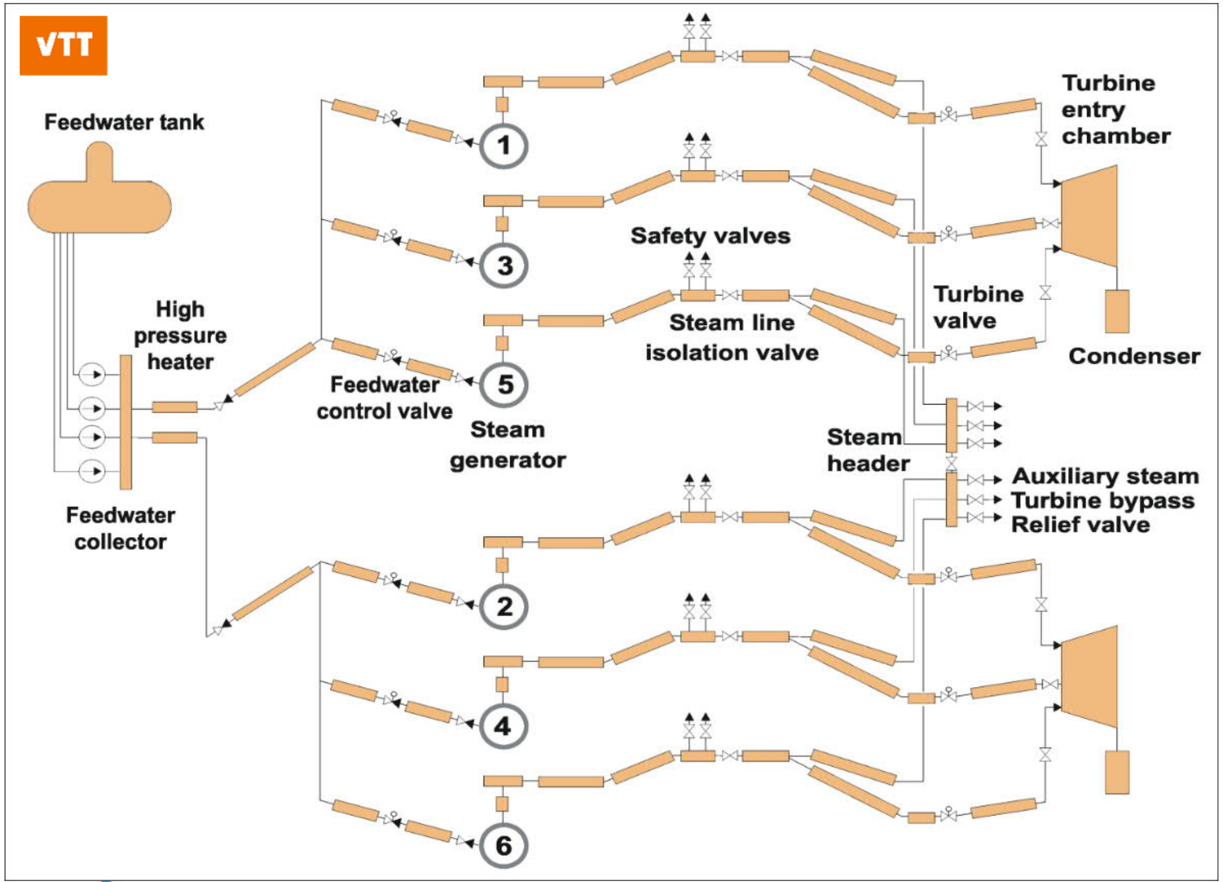
Strong flow spinning in Loviisa RPV

SMABRE (VTT): 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 (VTT):
Secondary side nodalization for Loviisa

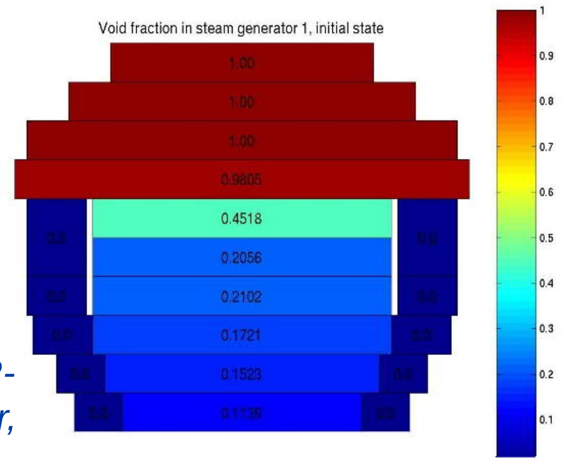


- Two turbines in VVER-440
- Only one steam header and feed water tank
- No turbine bleed
-> no feedwater (FW) pre-heaters
-> FW temperature according to feedwater tank enthalpy
- Check-valve features, volume of FW line e.g. in MSLB

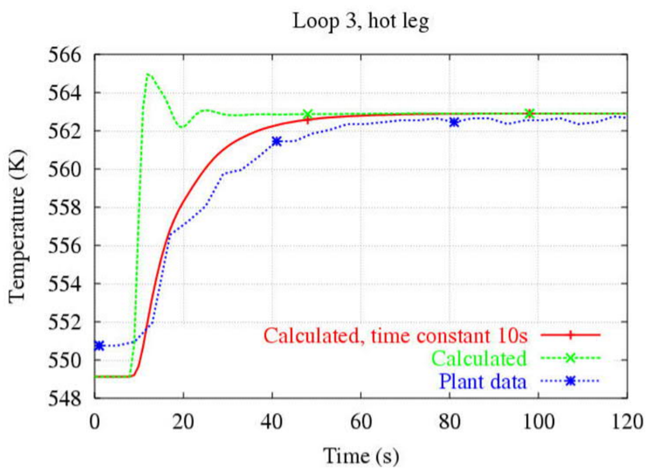
Validation against measurements

Simulation of real physical quantity
or simulation of measured value?

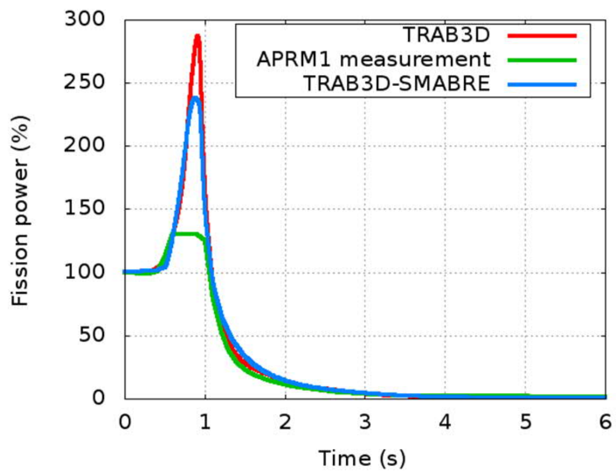
- Delays and time constants of measurement?
- Range of measurement?
- Measured quantity?
  - E.g. water level



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

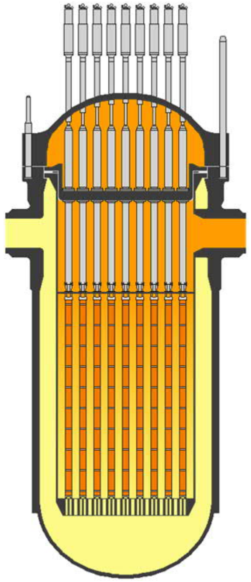
Reactor core produces heat also after reactor shutdown due to radioactive decay of the short-lived fission products (I-134, Cs-138, Cs-140…) and actinides (U-239, Np-239)

Proportional to power before shutdown

Immediately after shutdown ~6-7% of total power

After 1 hour ~1%

After 1 week ~0.5 %



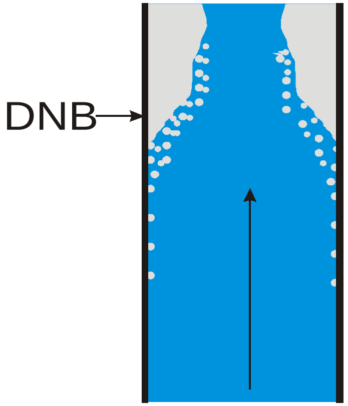
0,5% >20 000 kW

~ 5-10 kW

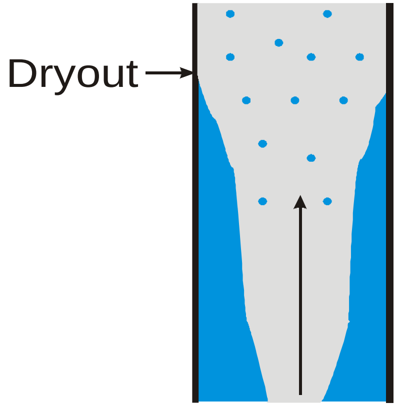
Heat transfer crisis

Two different mechanism:

PWR



BWR



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

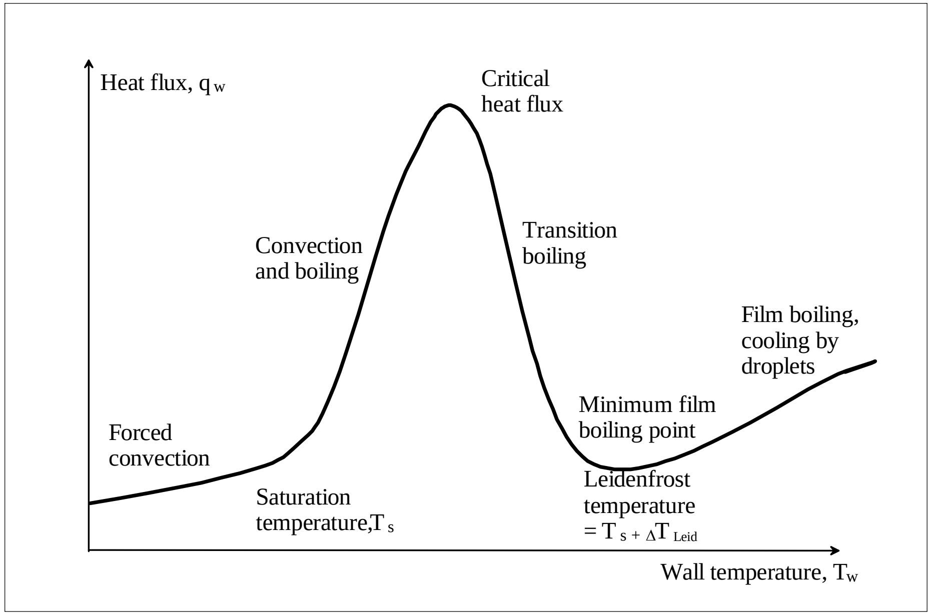
Heat transfer crisis

Critical heat flux CHF

- **Heat flux at which the boiling crisis will occur**
- Depends on a large number of factors
- 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
- In practice CHF is modelled with **empirical correlations**
  - Usually each fuel type has its own correlation
    - Based on measurements
    - Developed by fuel vendor
  - Also various general, public correlations are available

Heat transfer modes and flow patterns

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

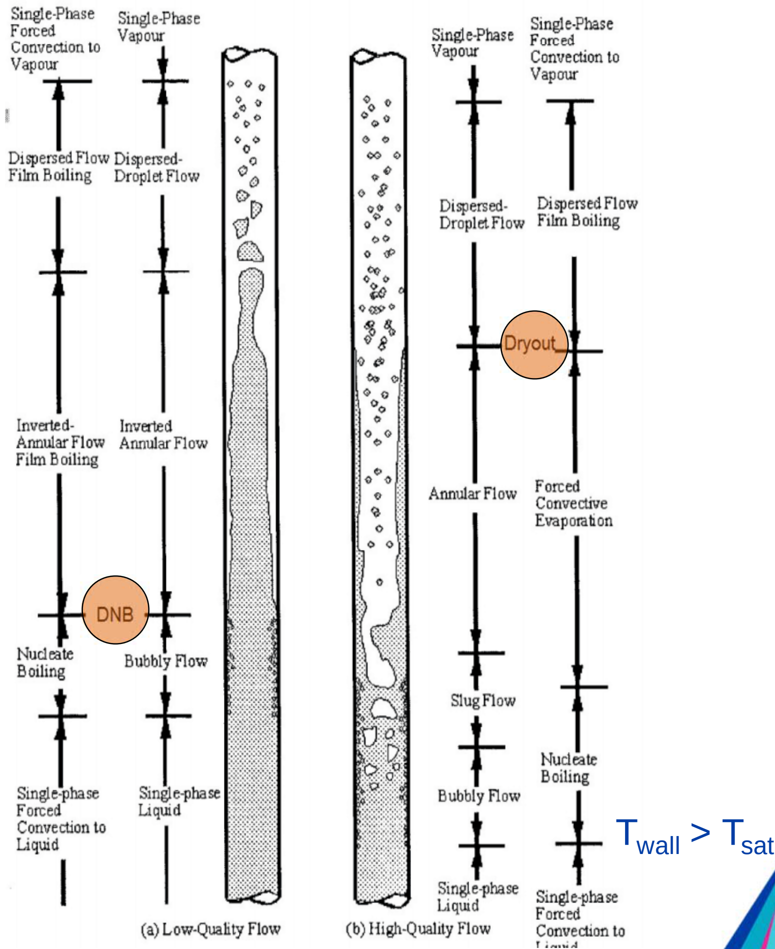
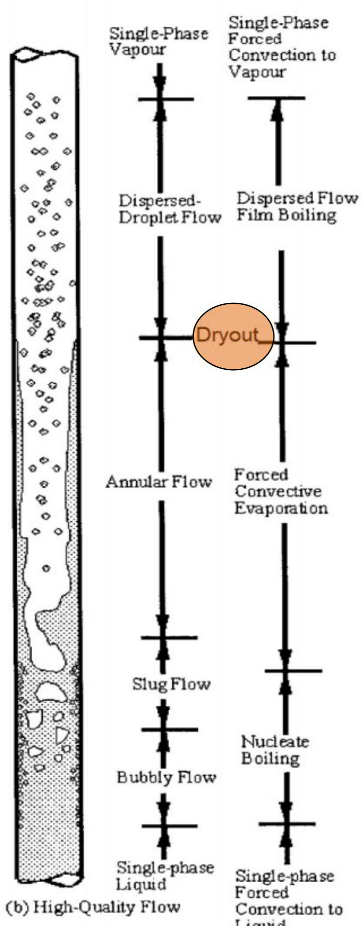
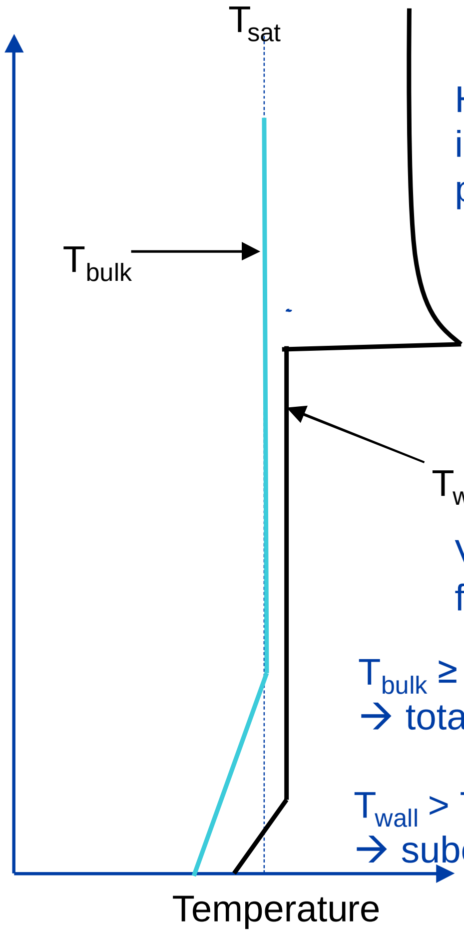


Figure: https://www.nuclear-power.net/

Boiling in BWR channel





High steam velocity improves heat transfer in post-CHF region

Void fraction increases → flow becomes unstable

$T_{bulk} \geq T_{sat}$
→ 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

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 or subchannel model with cross flows.

No neutronics calculation

Boundary conditions from a three-dimensional core model

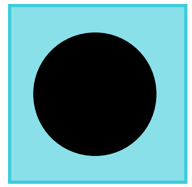
Hot assembly, hot rod conditions

Parameters which are not well known must be varied

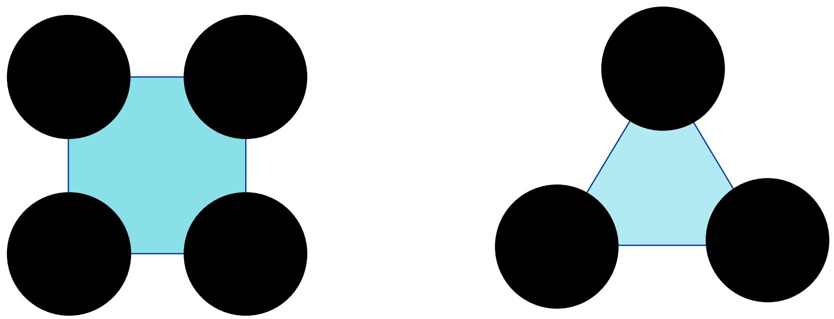
Models are not as detailed as in fuel behavior codes

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

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 at VTT

Traditionally TRAB-1D as a hot channel model

- Apart from neutronics, models similar as in 3D codes, but the program has long been used only for hot channel applications
- Includes some submodels specific to hot channels

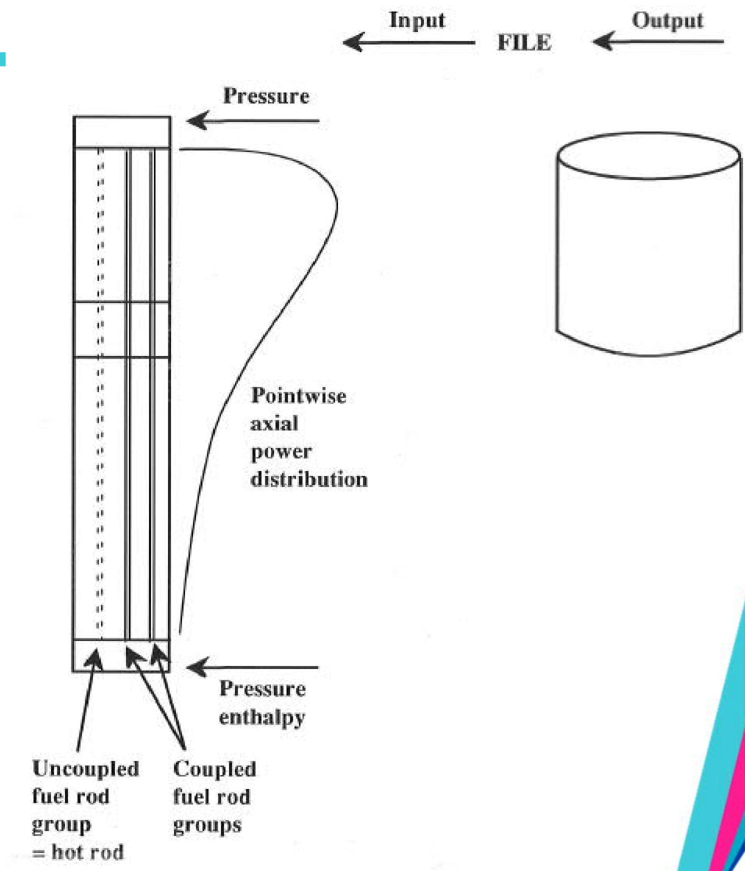
Includes only one isolated fuel channel

No neutronics calculation

Time-dependent boundary conditions from a full core calculation (TRAB3D/HEXTRAN)

- Axial power distributions
- Inlet enthalpy
- Inlet and outlet pressure

For example Westinghouse's BISON code have corresponding "slave" mode for hot channel analysis



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
  - E.g. in licensing analyses
  - 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

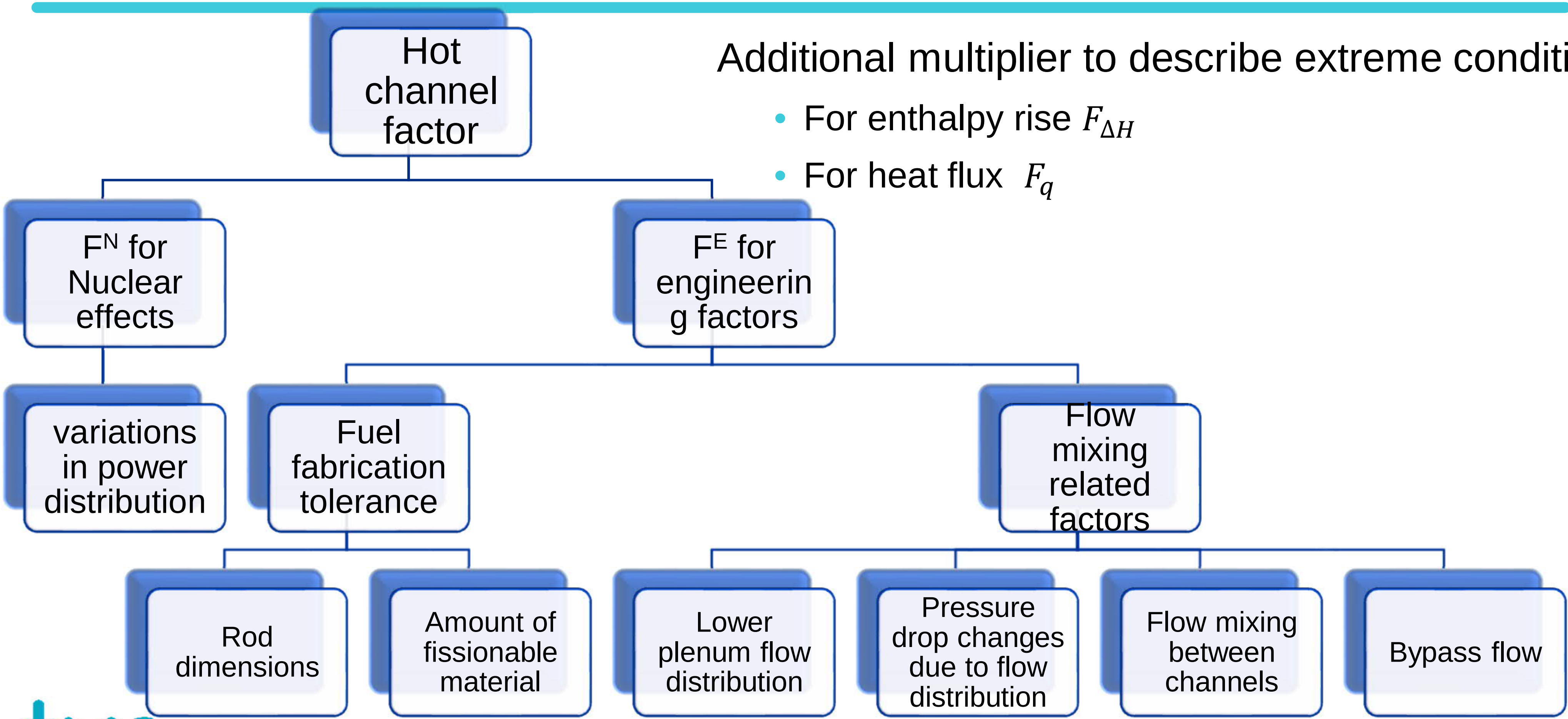
In addition to conservative hot channel analyses, DNB can be evaluated also during 3D transient simulation

- 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 rod analyses with FRAPTRAN-GENFLO

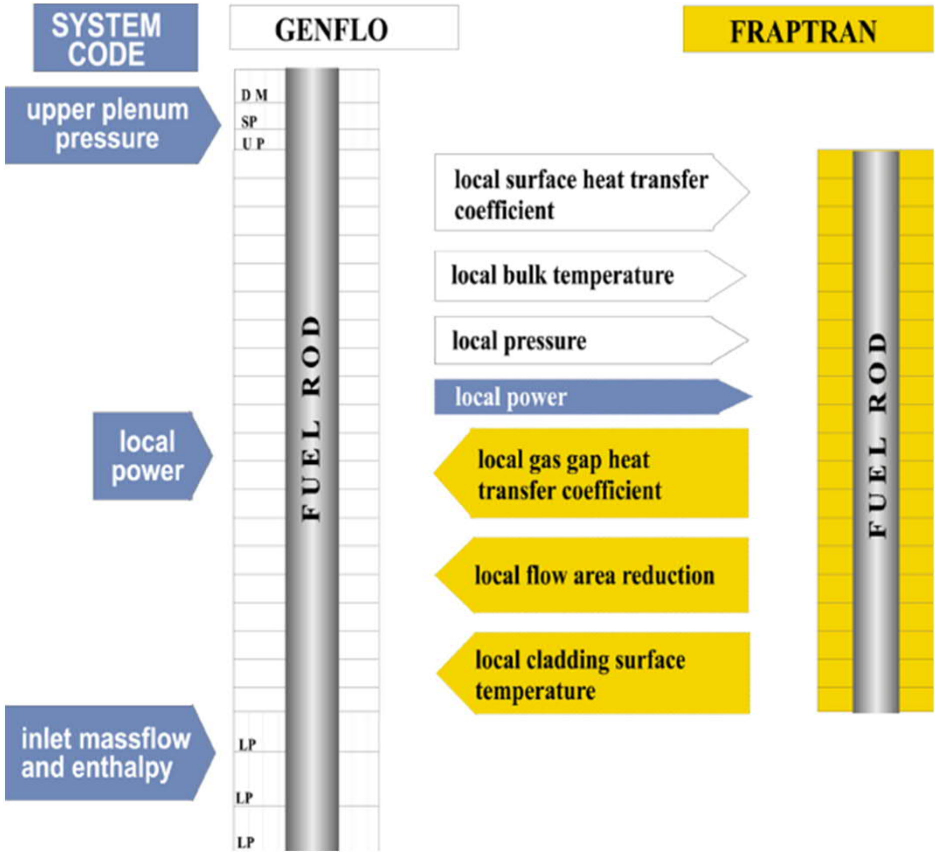
For more challenging conditions coupling of fuel performance code FRAPTRAN (U.S. NRC) and thermal hydraulics code GENFLO (VTT), coupling done by VTT

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)

Transient analyses for BWRs

At TVO, several transients are simulated during the core design of each cycle

- Fast pressure increase transients
- DBC2 & DBC3
- Transient cases selected on the base of extensive analyses performed during the license renewal
- The actual core loading of each cycle

Transient calculations with BISON

- plant model, I&C
- 1D-core (“average channel”)
- Initialisation of the transient calculations is performed with SIMULATE, initial state of the BISON is adjusted to the SIMULATE results

Several state points are analysed for each transient

- Approximately every 1000 EFPD (≈ 8 -10 different burnup points during cycle)
- Several (5 -10) flow – power combinations

→ Approximately 50-100 cases / transient

Transient analyses for BWRs

CPR evaluation is done with BISON SLAVE at TVO.

Hot channel simulation is performed for approximately 5 channels per fuel type in each BISON transient simulation case

- → Thousands of hot channel simulations

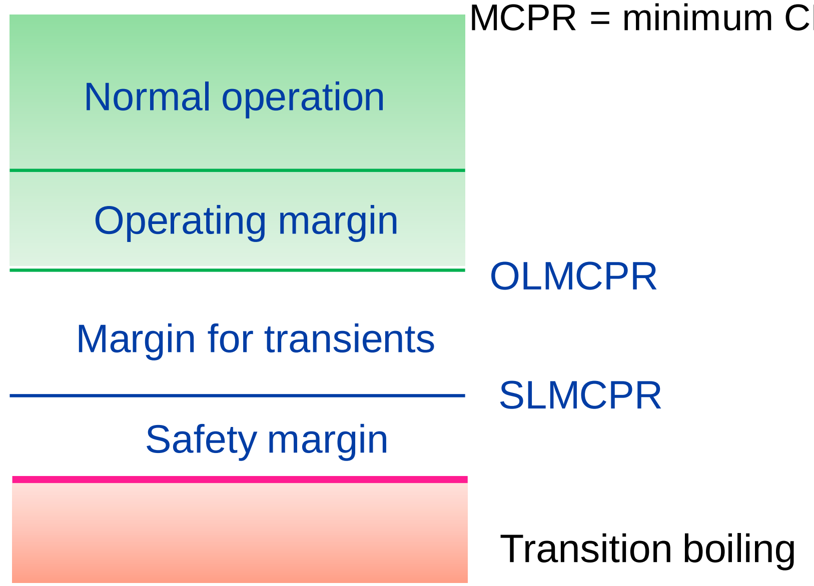
Hot channels are selected on the base of stationary SIMULATE-3 simulations

- power peaking; CPR

Cycle specific analyses: Analysis is performed for real core loading. There is no need to cover any other core loadings.

General CPR concept:

CPR =critical power ratio
MCPR = minimum CPR



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

OLMCPR= Operating limit minimum critical power ratio is
OLMCPR > SLMCPR + max ΔMCPR
Where ΔMCPR is change of MCPR during the limiting transient

Transient analyses for BWRs

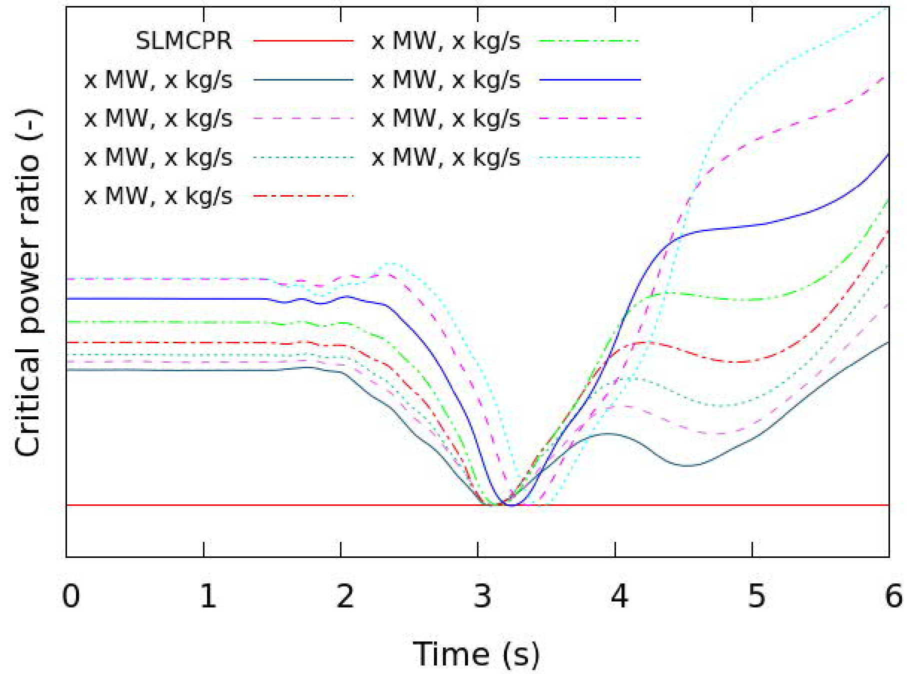
TVO case:
OLMCPRs are determined on the base of hot channel calculation results:

- For each simulated case (*transient+power+flow+burnup*), most limiting hot channel is channel in which smallest power change leads to SLMCPR
- Of all calculated burnup points and transients: OLMCPR is the highest initial CPR

OLMCPRs are given separately for each simulated main coolant flow

	Fuel type	4000 kg/s	b kg/s	c kg/s	d kg/s	e kg/s	7600 kg/s	8360 kg/s
Different OLMCPR for each fuel type	1	xx	xx	xx	xx	xx	xx	xx
	2	xx	xx	xx	xx	xx	xx	xx
	3	xx	xx	xx	xx	xx	xx	xx

CPR margins values are followed continuosly on the plant, OLMCPRs are part of the operational limits and conditions (TTKE=turvallisuustekniset käyttöehdot)

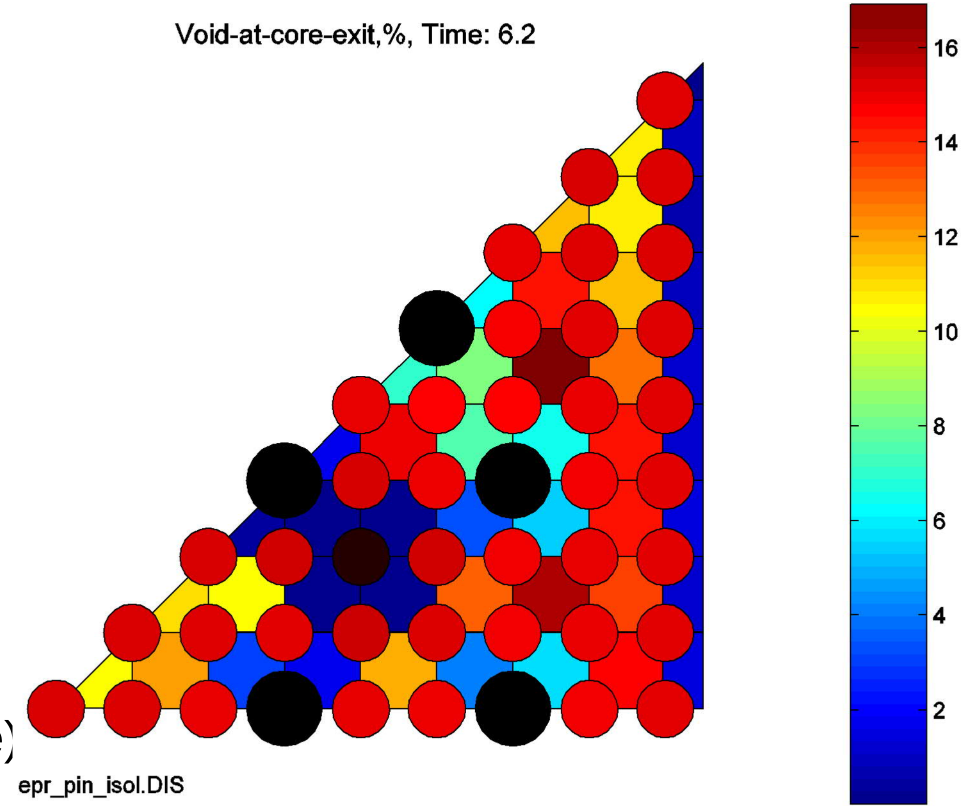


CPR of one fuel type during a transient with each power – flow combination at limiting burnup points during the cycle

Subchannel analyses

COBRA is well-known subchannel code

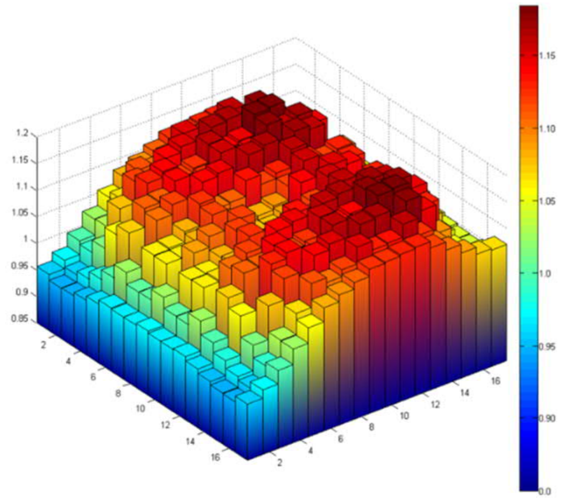
- Developed in USA
- Several versions
  - COBRA-3C/MIT
  - COBRA-IV
  - COBRA-EN
    - available in NEA data bank
    - 3- or 4-equation model
  - COBRA-TF
    - Conservation equations for vapor, continuous liquid and entrained liquid droplets = 9 equations
  - Several commercial codes are based on COBRA (VIPRE, COBRA-FLX…)
- Other subchannel codes: FLICA (France), KANAL-K (Russia),….
- Power as a boundary condition



Fuel rod modelling

Pinwise power distributions are needed for more accurate and new types of analyses

- Hot channel analyses
- Coupling with 3D thermal hydraulics
- Fuel rod analyses



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

More accurate modelling of power distribution, e.g. by means of pin power reconstruction

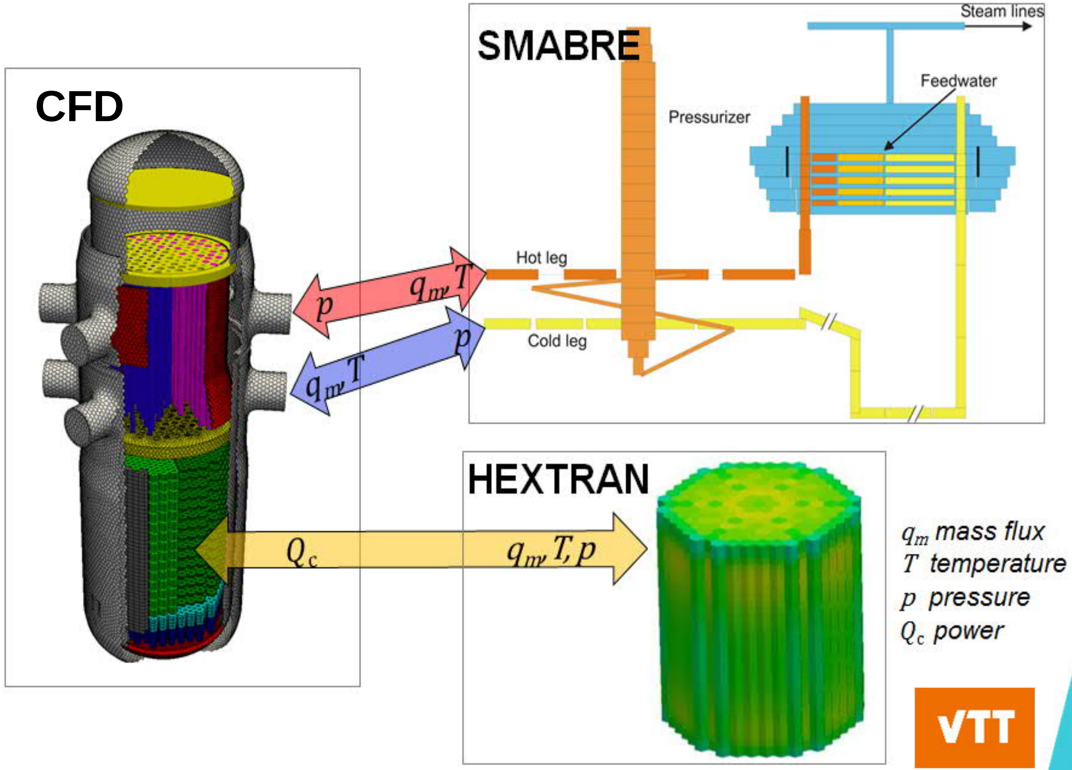
Recent development of reactor dynamics modelling - Multiphysics

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 solvers 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

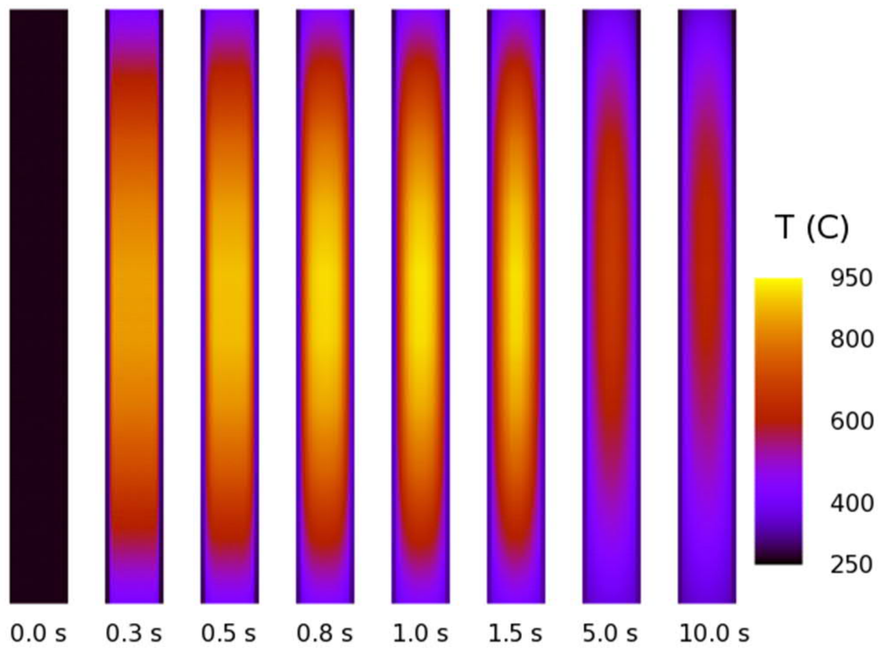


Recent development of reactor dynamics modelling - Multiphysics

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
  - Available from NEA databank



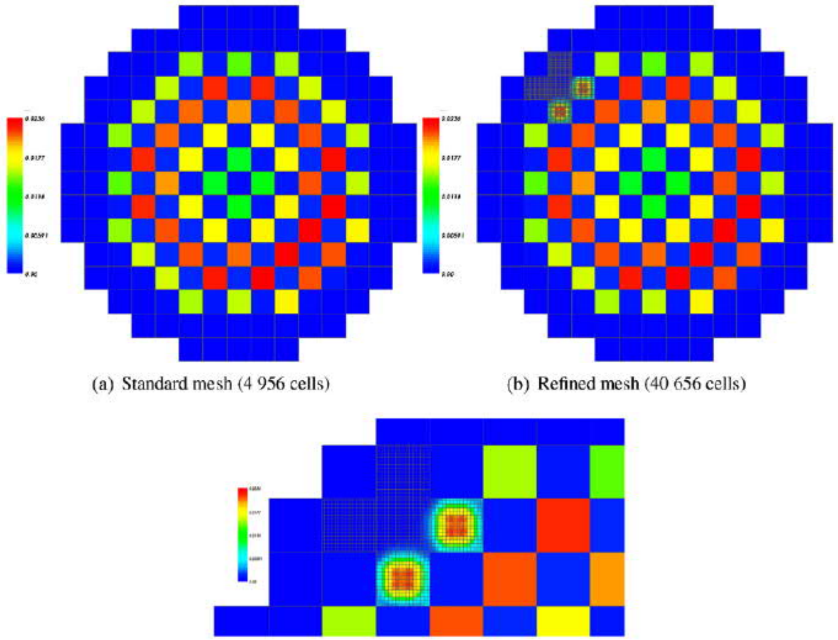
Example: temperature distribution in a fuel rod during CRE in VVER-440

Recent 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



(a) Standard mesh (4 956 cells) (b) Refined mesh (40 656 cells)

(c) Zoom on refined zone

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

Description of transients and accidents

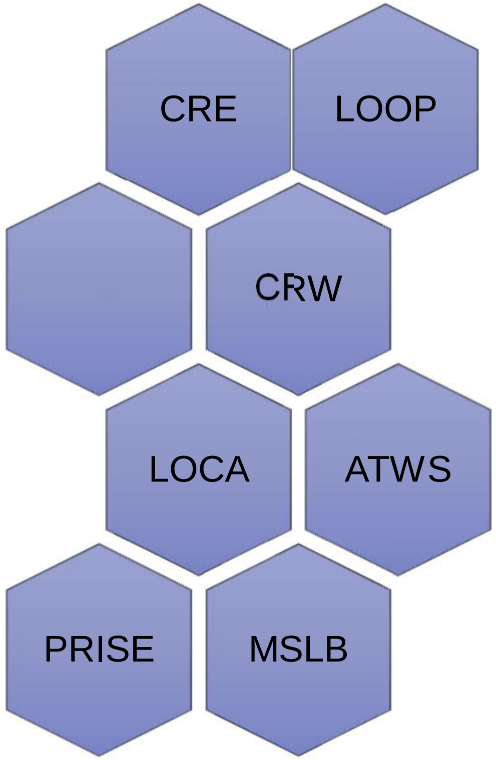
Background

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

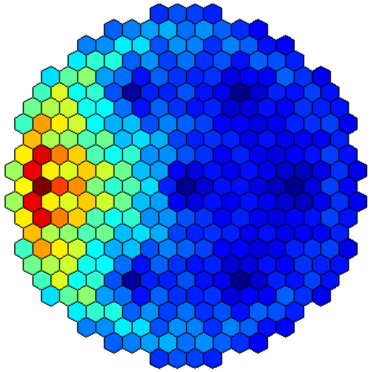
- e.g. high initial primary pressure is conservative for pressure analysis, but for fuel rod cooling low initial pressure is conservative

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

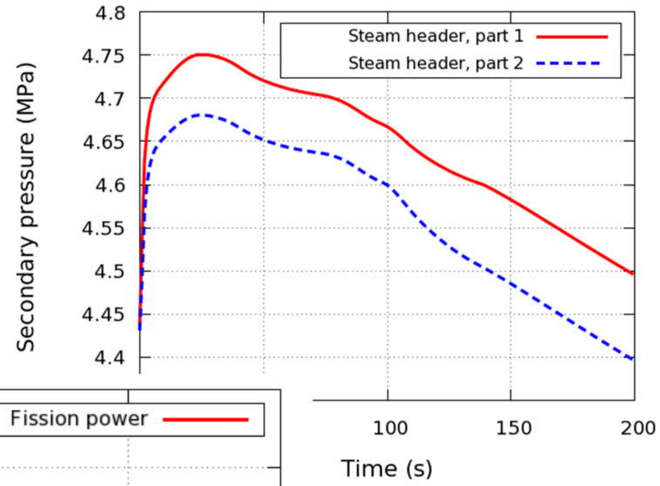
Turbine trip

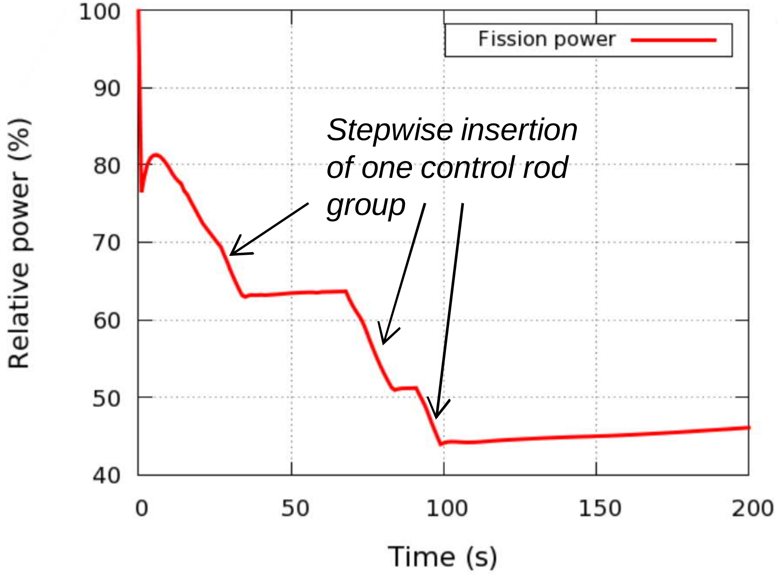
Often assumed that turbine trip causes also loss of offsite power

Example: Turbine trip in NPP with 2 turbines, no loss of offsite power

- 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

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





Turbine trip tests at OL3 from different power levels 3/2022-5/2022

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

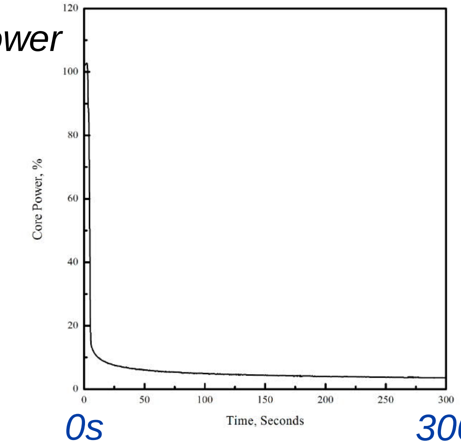
Start-up of diesels after delay

LOOP test (30% NP) was done at OL3 24.3.2022

DNBR?
Maximum pressure?
Does the pressure remain within the permitted limits without opening of valves designed for accidents (e.g. pressurizer safety valves)

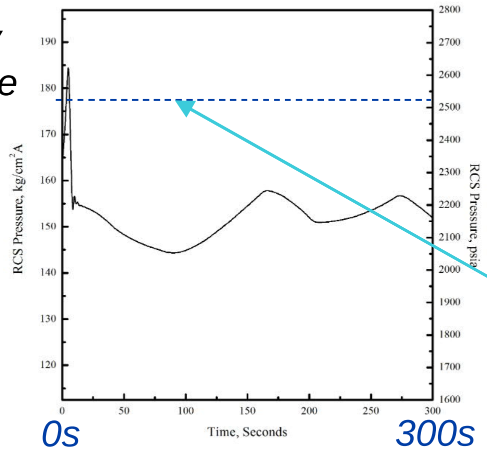
Loss of offsite power LOOP

Core power



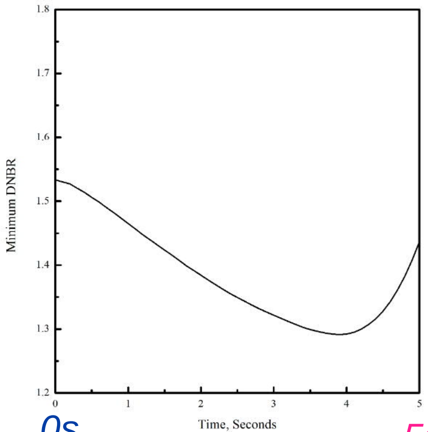
0s *300s*

Primary pressure



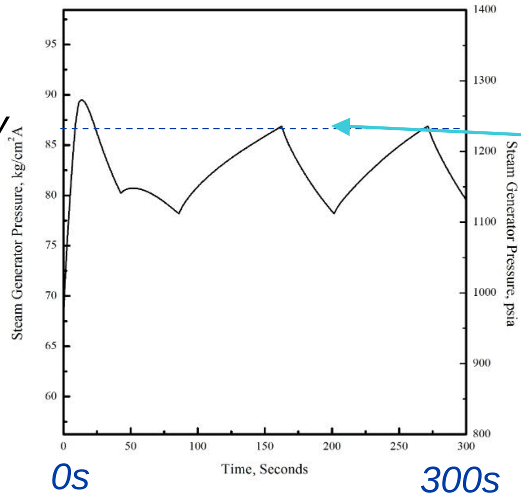
0s *300s*

Minimum DNBR



0s *5s*

Secondary pressure



0s *300s*

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

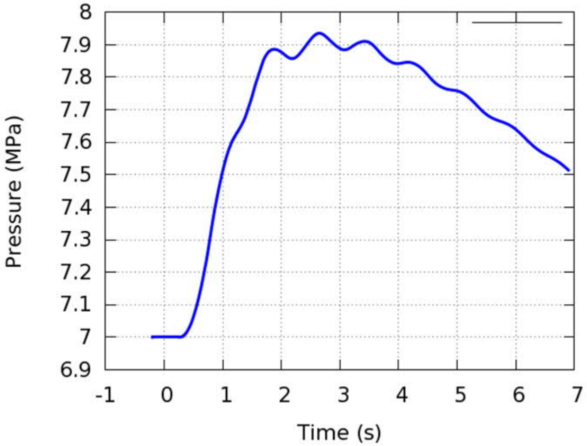
BWR Pressure transient

Initiating event sudden closure of steam line valves or failure of pressure controller

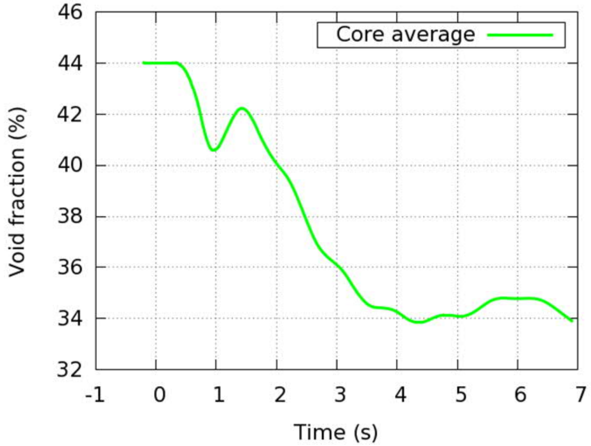
Happened in Olkiluoto 1985

Steam lines were suddenly closed in I-isolation at OL2 in 2020. No remarkable pressure and power increase.

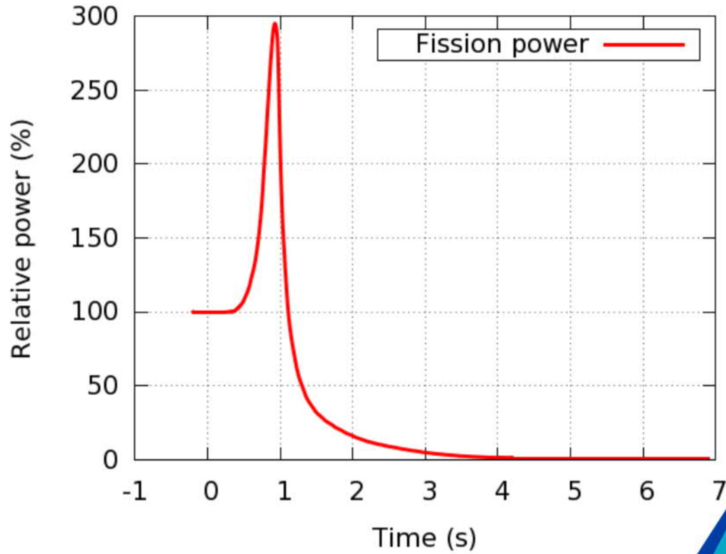
→ Pressure increase



→ 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

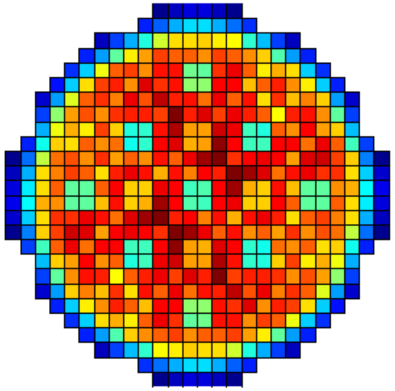
- From full power to 30 % power
- Turbine valves close, dump valves open
- Main circulation pumps and feedwater pumps stop
- Asymmetric partial scram: one hydraulic scram group and one motor driven scram group
- Local measurements during test (5 Local Power Range Monitors, LPRM, 4 at 4 heights, one at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

Utilized e.g. in the code validation at VTT

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

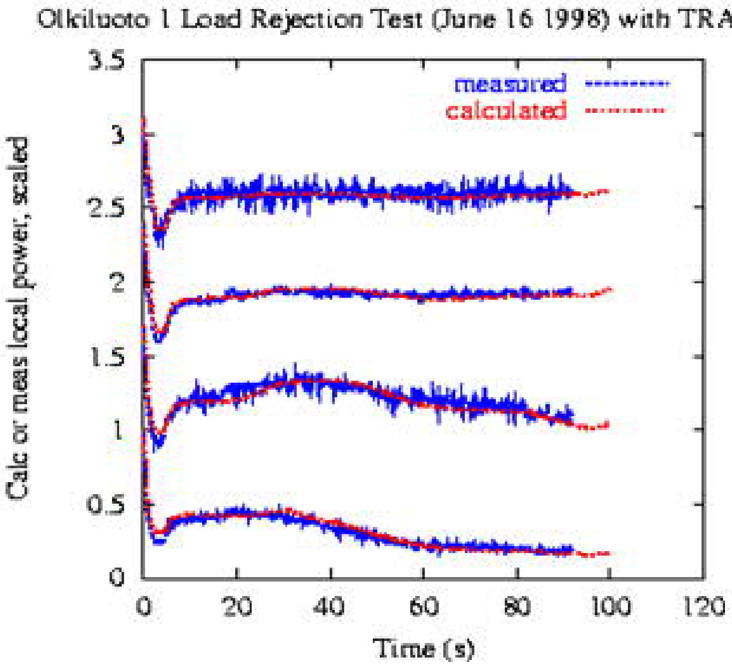


Power distribution at initial state

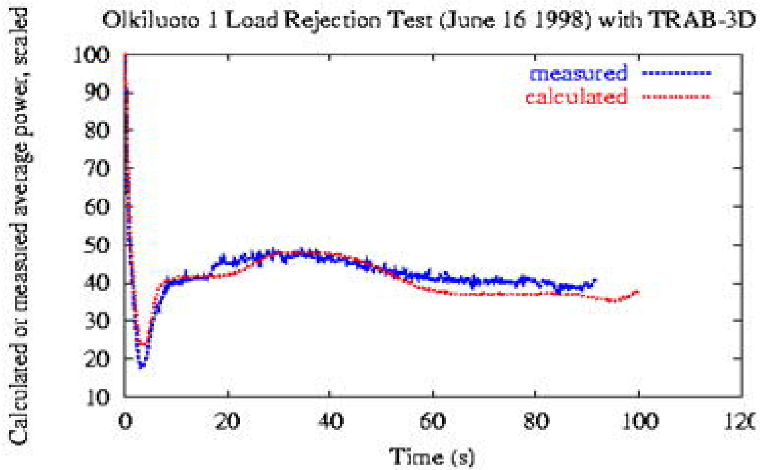
Main interest in validation of three-dimensional effects

- 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

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



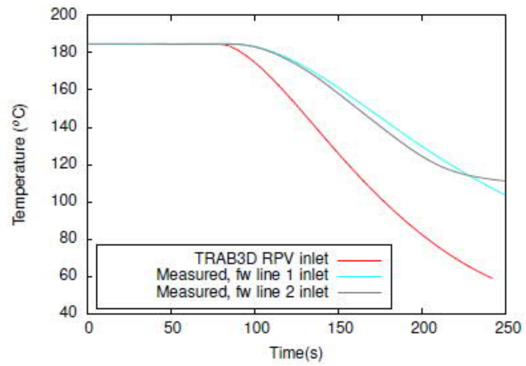
LPRM at different axial heights

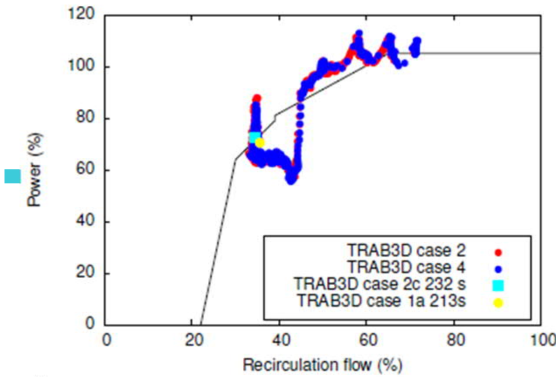


APRM based on 28 LPRM values

BWR instability

- Wrong combination of power, flow and coolant temperature may lead to unstable behavior
- Several times in BWR's, also in Finland in 1987

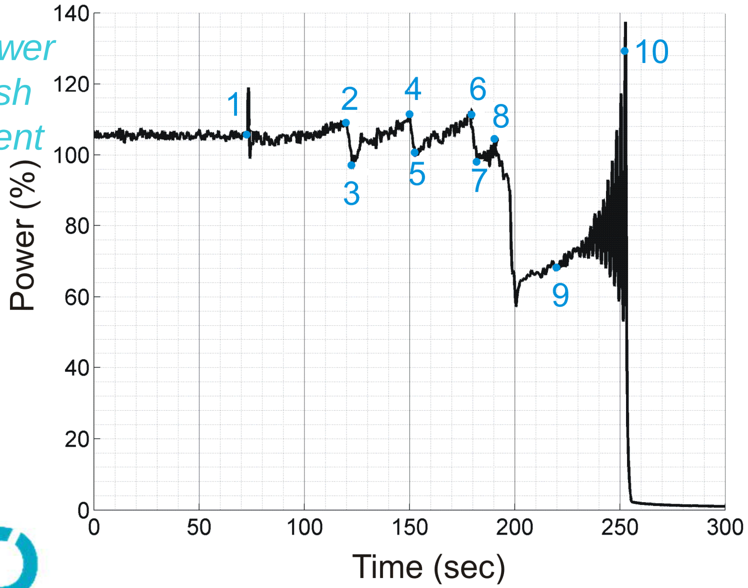




E.g. instability event in Sweden:

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)

Measured power during Swedish instability event



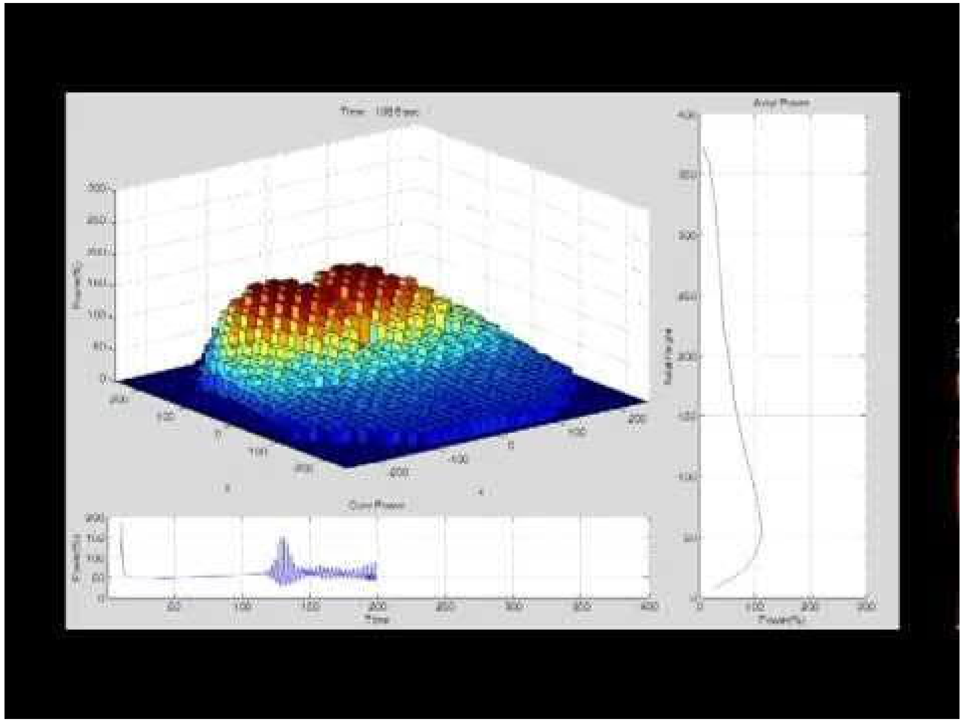
PUBLIC

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
- Local (single channels)

Reactors should be designed and operated in such a way, that instabilities are eliminated

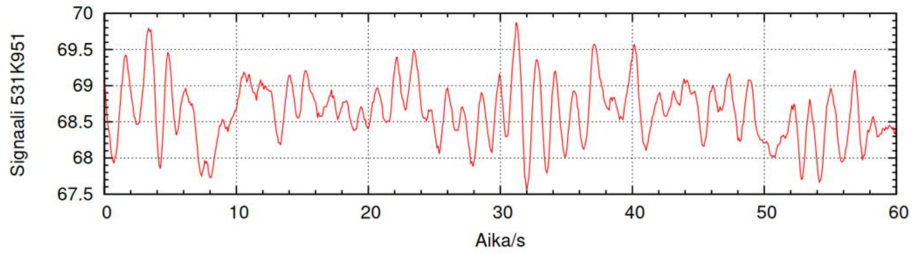


https://youtu.be/3TZcZDVIvZk video by Peter Yarsky, NRC

BWR stability

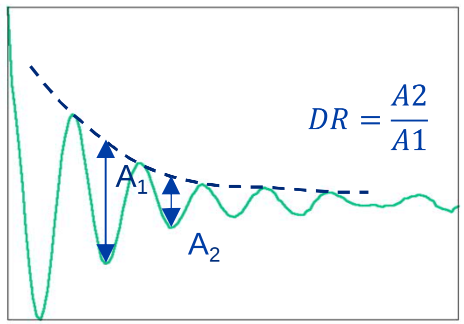
At OL1/OL2, decay ratio (DR) is

- Followed continuously at online reactor supervision system
- Taken into account in the core and fuel design
- Measured typically from 2 to 6 times during a cycle
    - Reduced flow and power
    - Measure APRM and LPRM signals (Average/Local Power Range Monitors)
    - DR can be calculated from APRM/LPRM by different methods



Fourier transformation

Autocorrelation function



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
- Homogeneous dilution slow phenomena, is easily detected
- 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

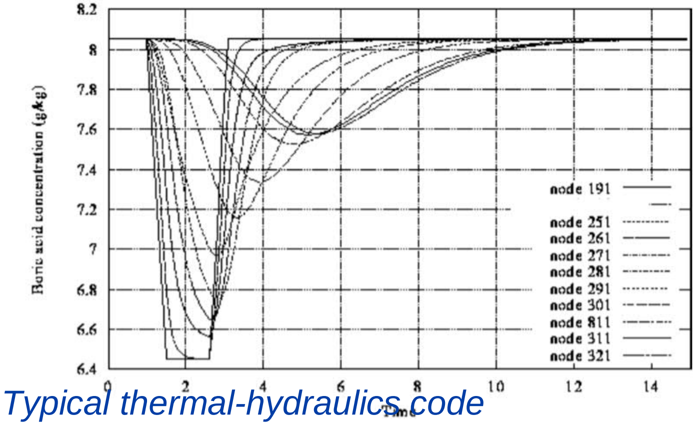
During accidents

Boron dilution

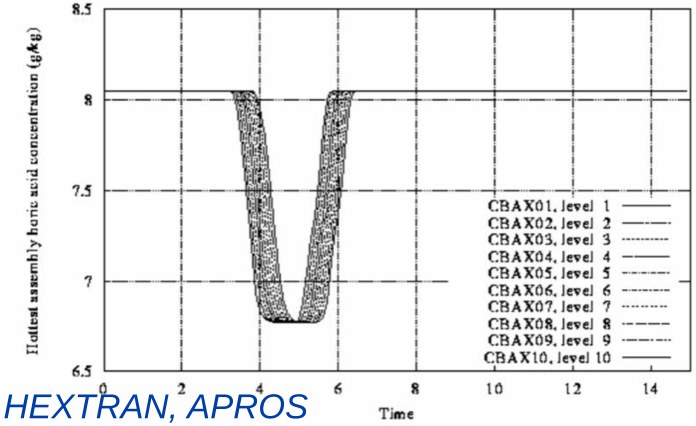
In Loviisa VVER-440 several plant modifications have been done in 90s to prevent external dilution.

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.

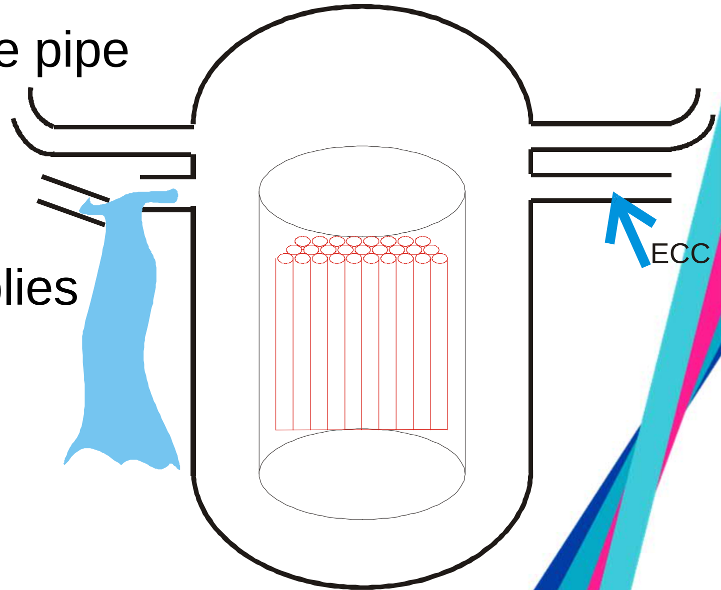
Loss of coolant accident LOCA

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).

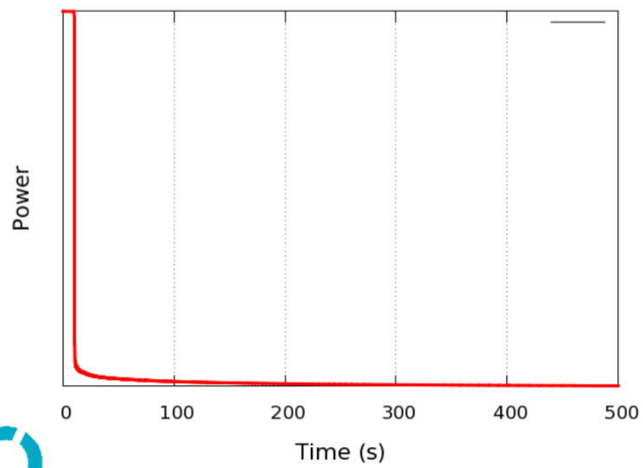
- Double-ended break in a loop,
  flow area of the break 200% of cross section area of the pipe
- Pressure decrease very fast
- Scram & No coolant -> fission power shuts down
- Decay heat & no coolant -> overheating of fuel assemblies

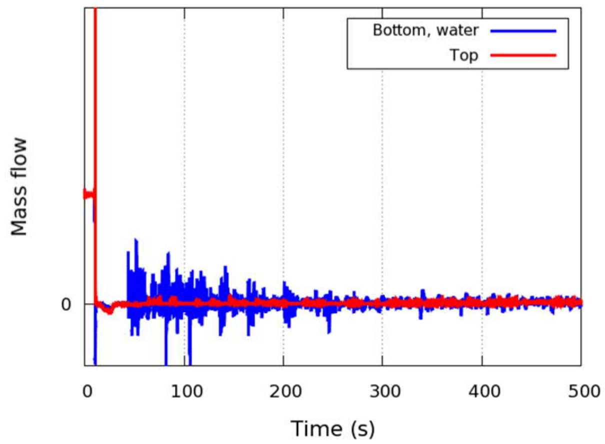


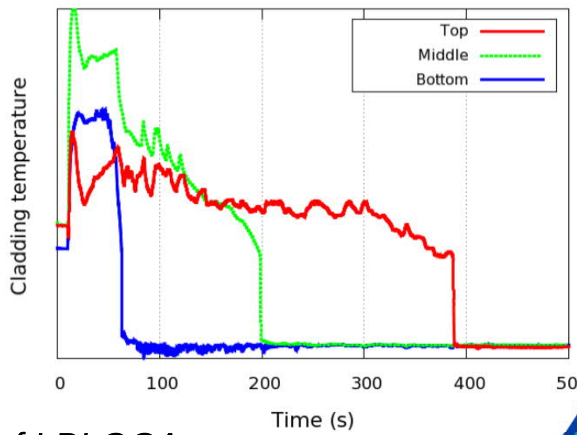
Loss of coolant accident LOCA

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

Never 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

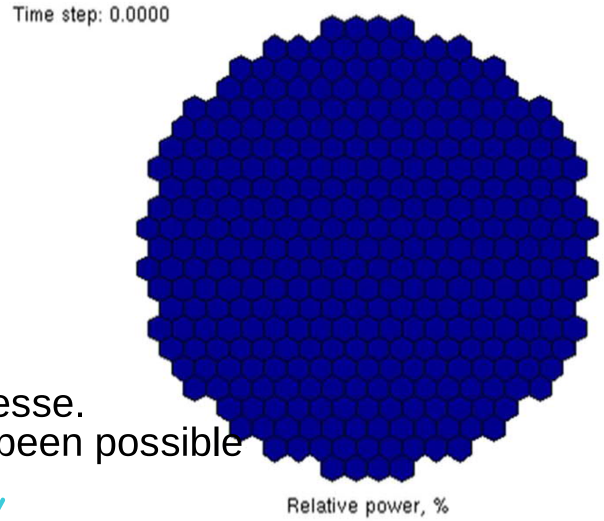
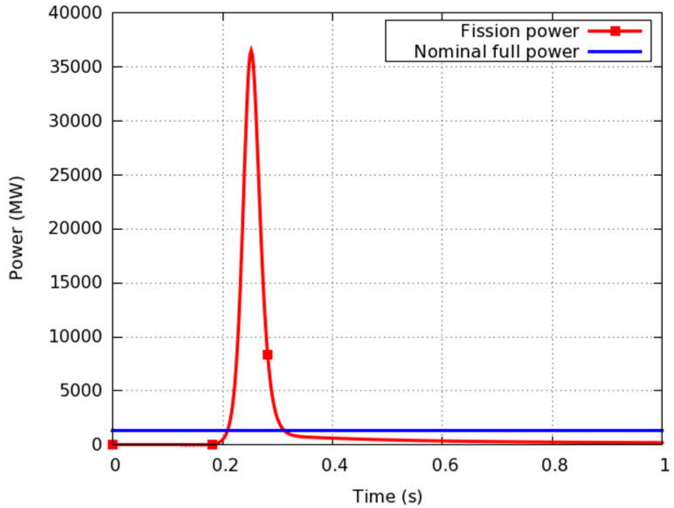
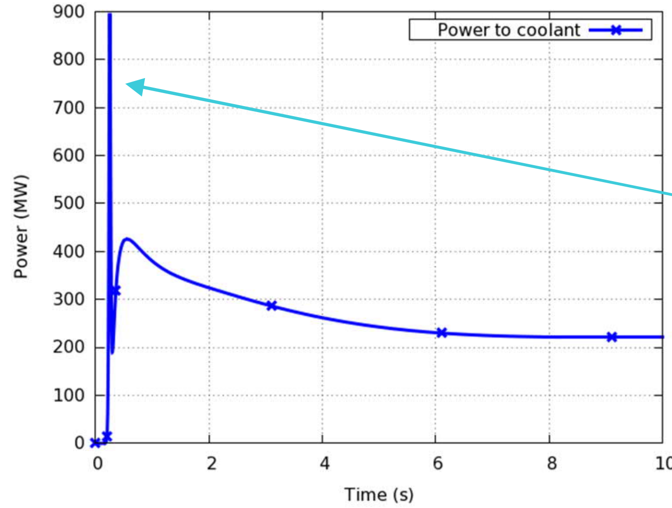


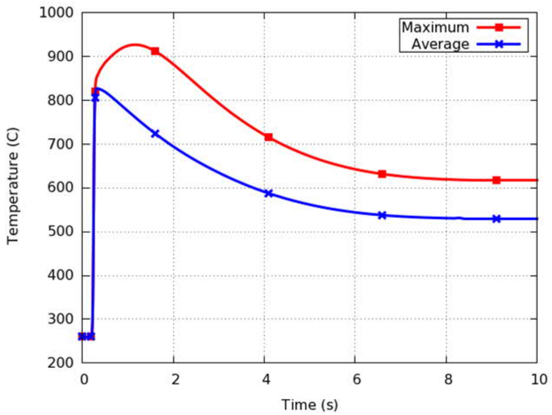
Figure:en.wikipedia.org

Control rod ejection CRE





Peak due to gamma heating



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

Figs: HEXTRAN simulation of the hypothetical VVER-440 CRE benchmark

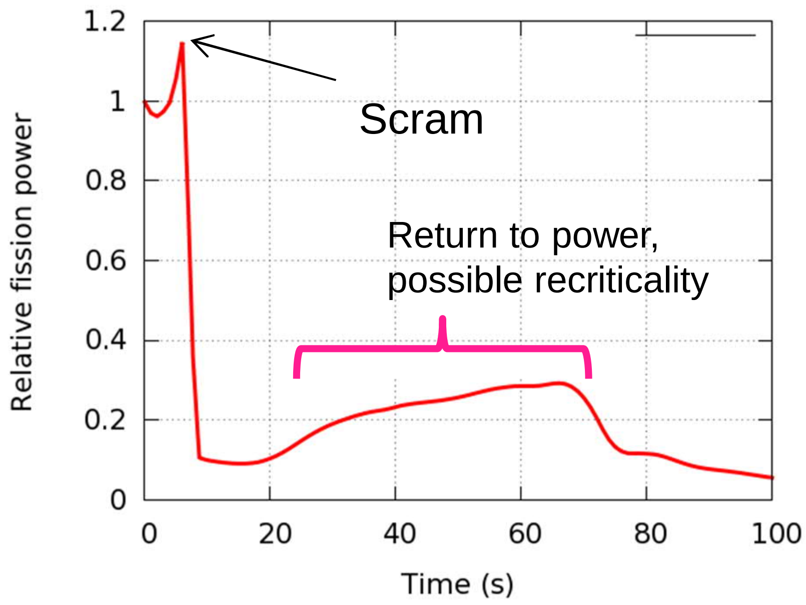
Main steam line break MSLB

Steam line is broken and steam flows to the environment

Heat transfer from the primary to the secondary side continues

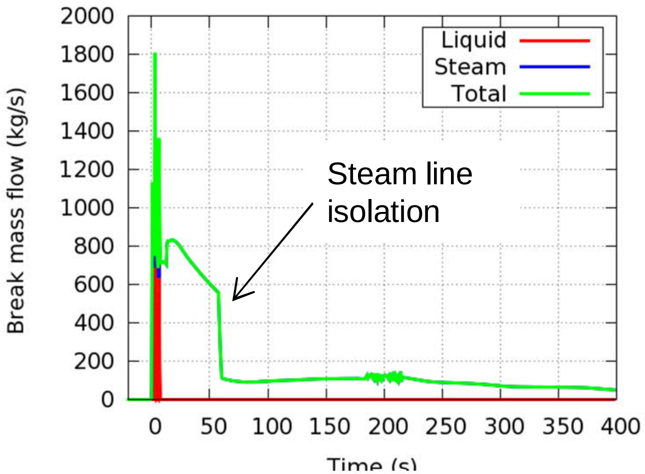
-> secondary pressure drops

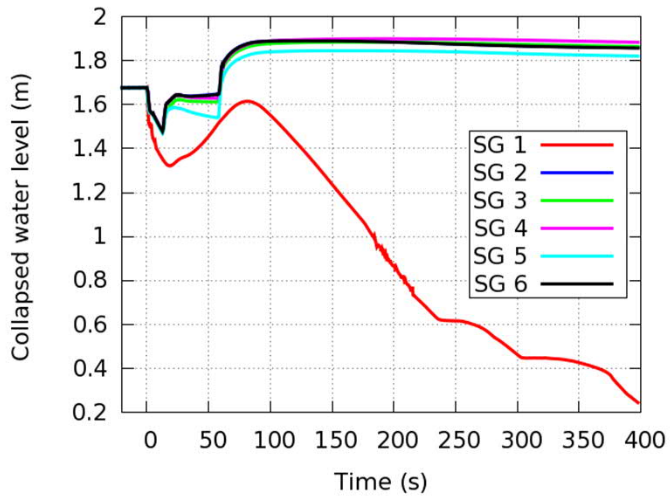
-> primary water cools in steam generator
-> water temperature at core inlet asymmetric and locally very cold
-> possible recriticality and power increase **even if control rods have been inserted**. Boron injections are needed to ensure subcriticality.

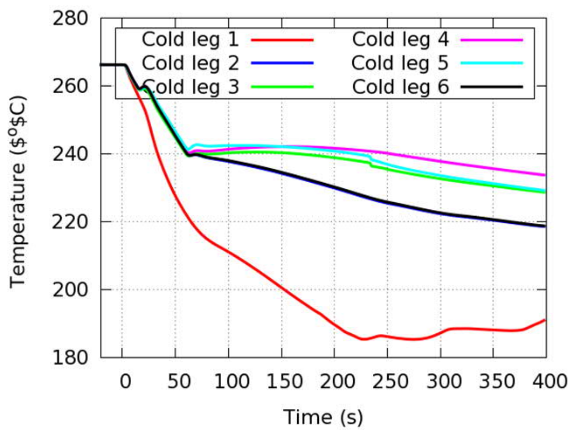


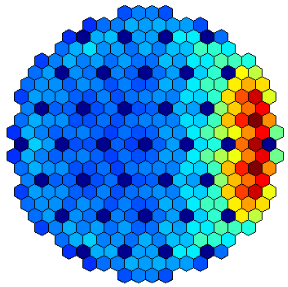
TRAB3D simulation of the hypothetical TMI MSLB benchmark

Main steam line break MSLB





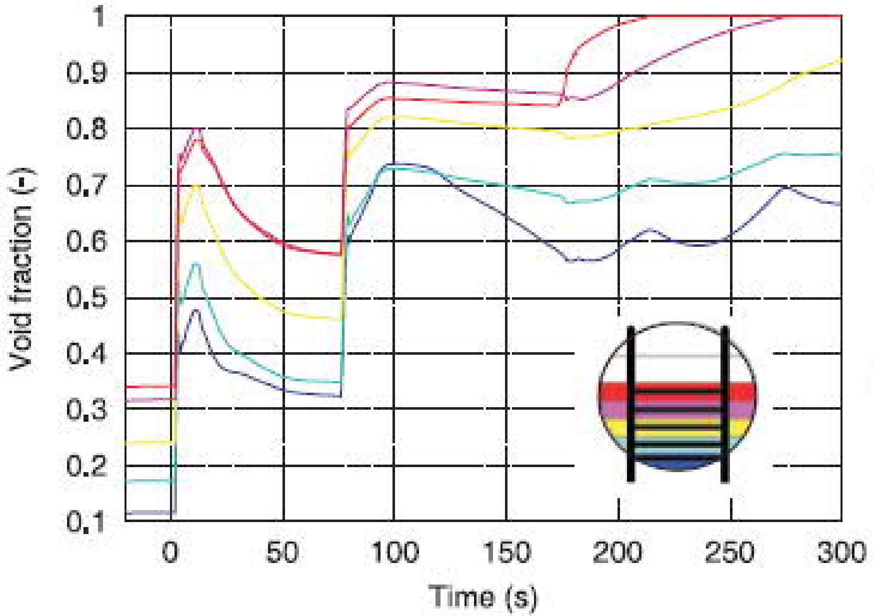




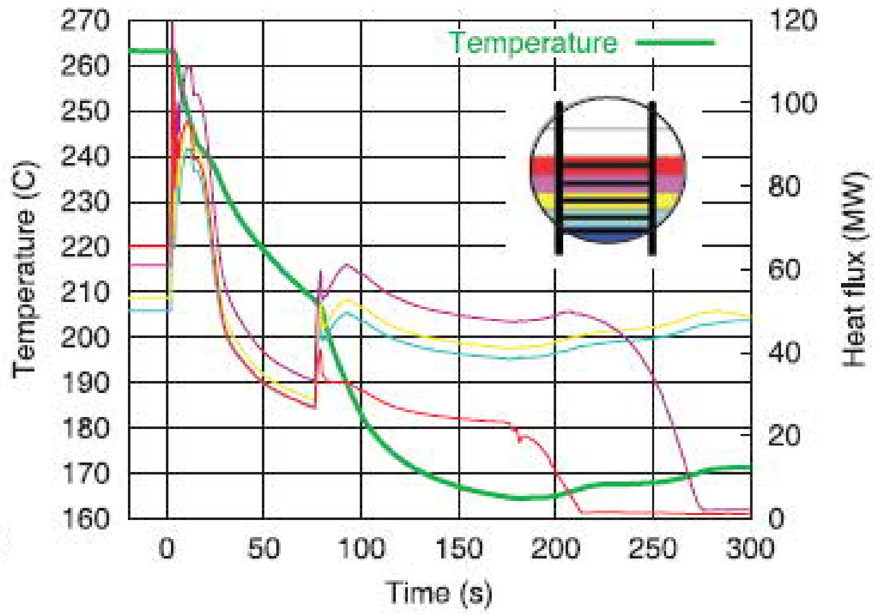
Power distribution in a reactor core

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

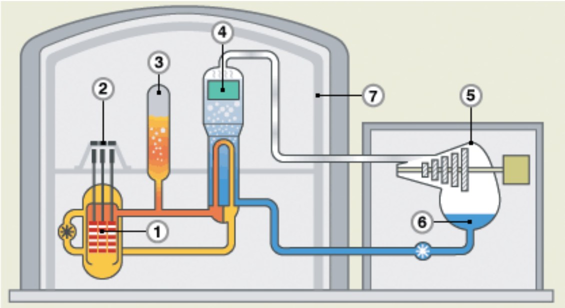


Figure: BBC

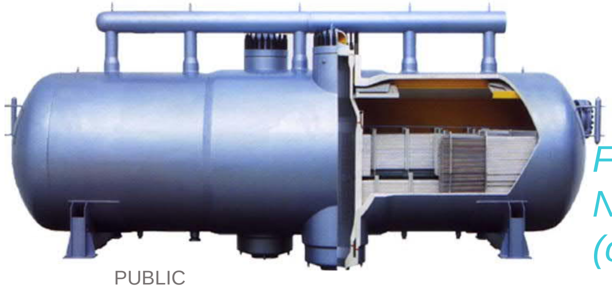


Figure from Nucleartourist.com (original figure CEZ)

Other examples of AOOs and accidents

Decrease in heat removal by the secondary system

- Feedwater line break

Reactivity and power distribution anomalies

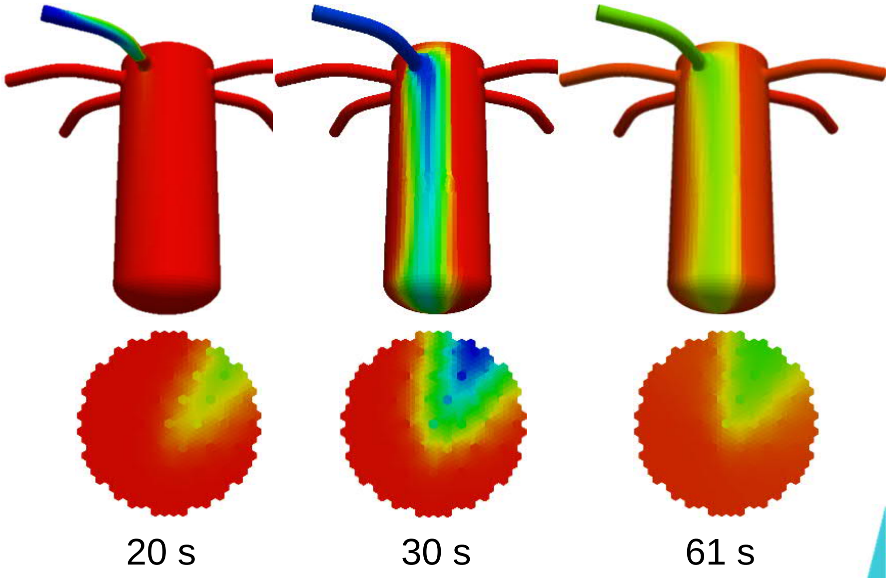
- Incorrect connection of an isolated reactor coolant system loop

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

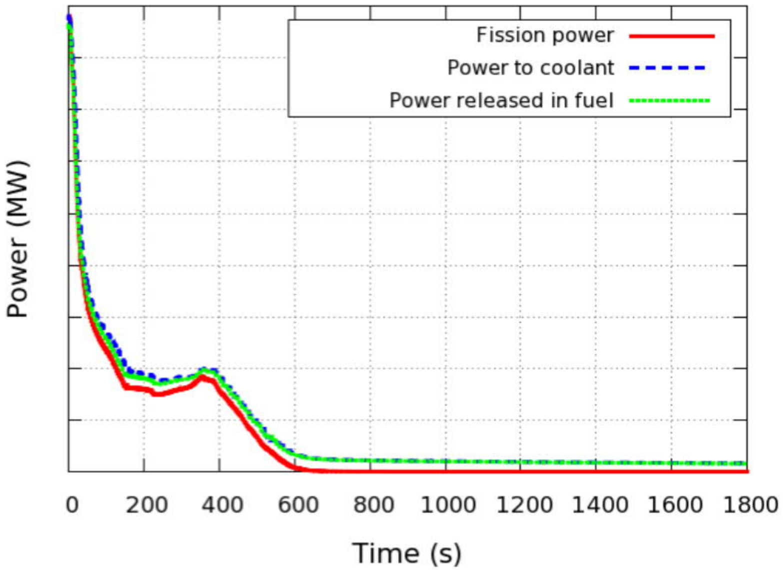
Requirements for DEC cases are different in different countries,
in many countries requirements have changed during last few years (after Fukushima)

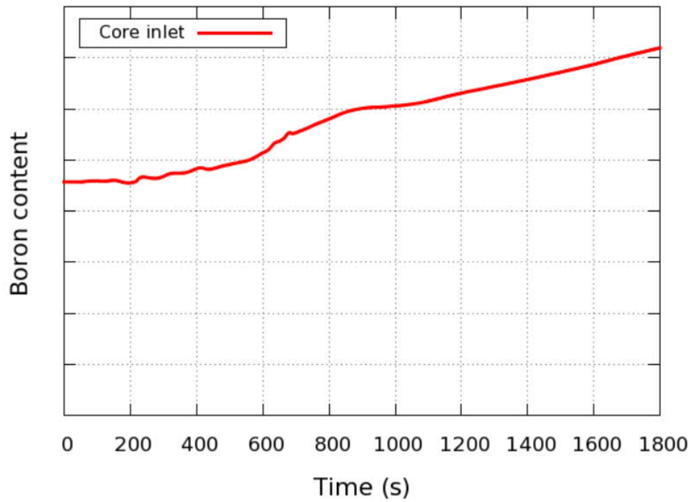
ATWS

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?

Thank you!