Code package for analysis of fast reactor safety: tasks of its updating and development

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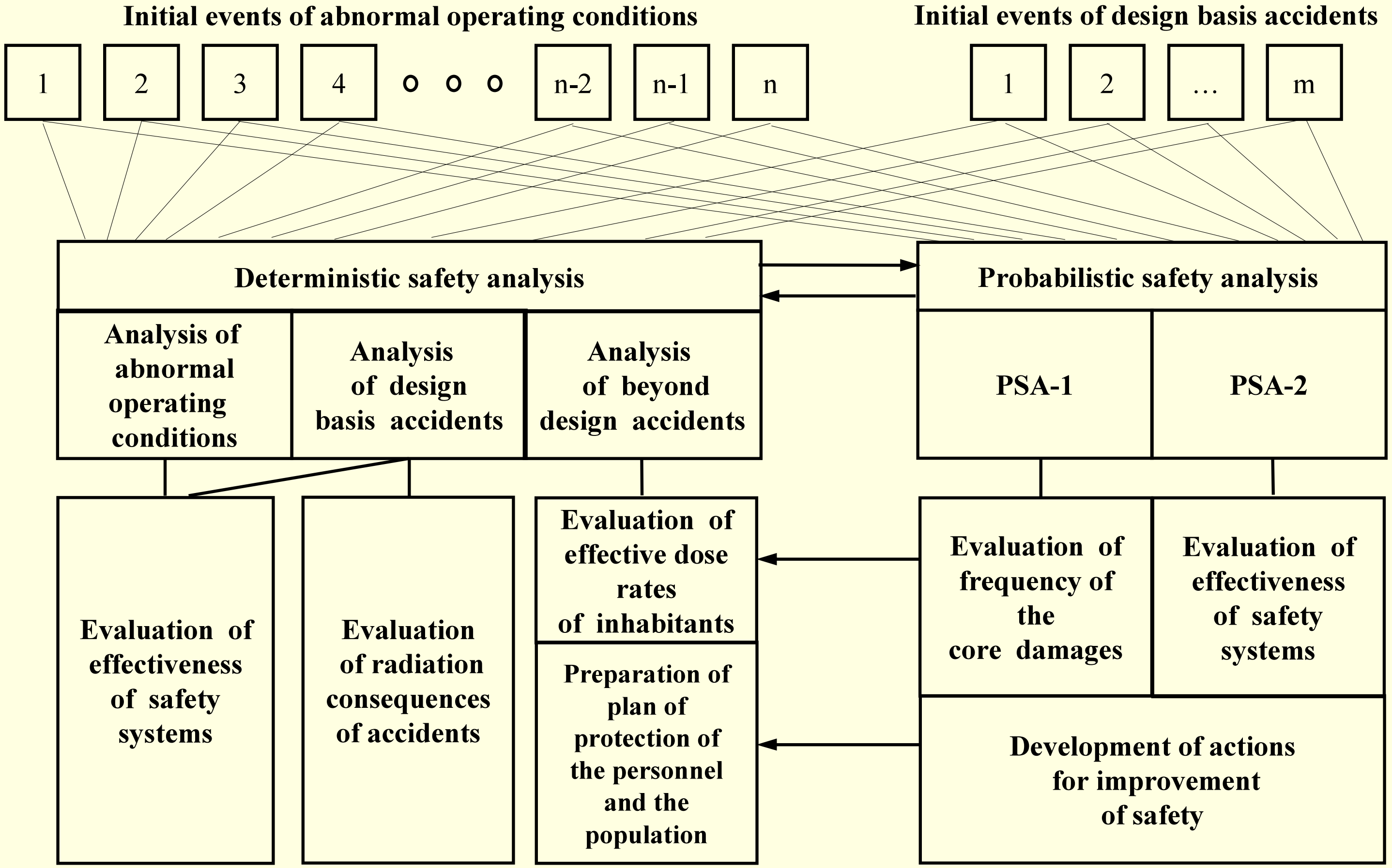
Introduction

1. **Russian approach to fast reactor safety analysis was formed on the basis of the large experience gained in designing and operating of nuclear reactors and, in particular, fast neutron reactors**.*(This experience included in itself large sodium leaks, leading to radioactive sodium releases from the primary circuit of the reactor, as well as failures of steam generator tubes causing water and steam penetration to the secondary sodium).*
2. **Experience gained in preventing and mitigating such abnormalities and accidents was used for updating normal operation and safety systems and improving reactor operation regulations and Regulatory documents**
3. **Main document that determined common regulatory approach and common requirements to safety analysis of fast reactors in Russia is OPB-88/97. It contents brief list of specific requirements regulating characteristics of various type reactors with regard to safety**
4. **One more important document - Special standard contents of safety analysis report.**

SAFETY ***is its capability of keeping radiation doses of personnel, inhabitants and environment within permissible limits under normal operating conditions, abnormal operating conditions and in case of accident*** **(OPB-88/97) .**

Grounds of NPP safety concepts and tasks related to safety analysis

RUSSIAN concepts of safety analysis of fast reactors that is reflected in the Russian regulatory documents corresponds to "defense-in-depth" principle developed by IAEA .



Deterministic safety analysis of NPP design

Analysis of abnormal operation

The main objective of analysis of reactor abnormal operating conditions is to justify design requirements to the speed of response, effectiveness and other characteristics of safety systems and confirm meeting safety criteria and requirements in NPP design.

Tentative list of initial events (IE) of abnormal operating conditions for FR (15 from 40):

- **reactor vessel failure (leakage);**
- **de-energizing of primary pump in different operation modes;**
- **closure of one check valve with all primary pumps in operation;**
- **incorrect opening of check valve in shut-down loop of heat removal system with all other loops in operation;**
- **unauthorized movement of control rod in various reactor states;**
- **unauthorized movement of shim rod in various reactor states;**
- **penetration of hydrogen containing materials to the core;**
- **emergence of gas bubbles in the core and their movement through SA;**
- **water leak into sodium;**
- **loss of feed water supply to one or all SG;**
- **loss of grid power supply (loss of power supply of auxiliaries);**
- **failures of ionization chambers including that caused by failure of their heat removal;**
- **failure of the main secondary pump;**
- **disconnection of turbo-generator from the grid;**
- **failure of the main steam pipeline, etc.**

IE list may change on the basis of analysis of specific reactor plant design, its operating modes and maintenance regulations

Deterministic safety analysis of NPP design

Analysis of design basis accidents (DBA)

In Standard Contents of Safety Analysis Report as Applied to NPP with FR there is the following tentative list of DBA IE:

- **partial or full blockage of one SA because of materials swelling, penetration of impurities in the coolant or foreign objects followed by destruction and meltdown of the fuel elements;**
- **failure of the primary piping in the section having no safety jacket;**
- **failure of the primary cover gas system;**
- **failure of spent fuel storage drum wall.**

GENERAL STATEMENT: It should be demonstrated that safe operation limits related to the fuel element failures, are not exceeded.

"Most important DBA" is a full blockage of FSA cross section. It should be proved that there is no "chain" propagation of destruction from one SA to another. This propagation if it take place it must be limited and total number of destoyed SAs is not more than 7.

Deterministic safety analysis of NPP design

Analysis of beyond design basis accidents (BDBA)

BDBA list (*approved by the RF GAN)***:**

- **loss of grid power supply with simultaneous failure of reactor safety system;**
- **total loss of grid and independent power supply;**
- **guillotine rupture of the primary sodium pipeline having no safety jacket;**
- **guillotine rupture of the secondary sodium pipeline;**
- **water-sodium interaction in the steam generator cell;**
- **total loss of grid and independent power supply with simultaneous failure of reactor safety system (combination of the first and the second accident scenarios) (ULOF);**
- **failure of the main and guard reactor vessels and fire in the reactor pit;**
- **penetration of considerable amount of hydrogen or carbon-containing materials**

(from lubrication system of the main primary pump) to the primary sodium;

- **fire causing damage of control and power supply systems.**

Most representative and probably most dangerous in the List is ULOF.

Based on the estimation of effective and equivalent radiation doses of personnel and inhabitants during one year after accident, conclusions are drawn on meeting Radiation Safety Standards and necessity of protection measures, in particular, evacuation of inhabitants

Computer codes used for safety analysis of NPP

Total number of codes normally involved in safety analysis is more then 20.
14 from 20 codes will be briefly presented below.

- **Brief description of the codes with the area of its application;**
- **Examples of codes validation against experiments for some of them;**
- **Examples of codes application for some of them.**

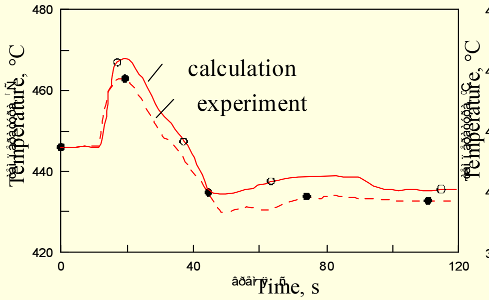
DINROS is system code intended for analysis of transient and accident processes in multi -circuit multi-loop reactor plants with detailed modeling of reactor design.

CODE MODEL

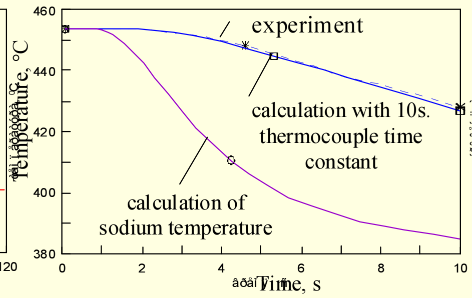
- **1D 1-phase thermal hydraulic models for primary, secondary and turbine systems**
- **simulation of reactivity feedbacks, control system and safety systems of reactor**
- **thermal-mechanical pin model**

CODE APPLICATION:

- **abnormal operating conditions;**
- **DBAs that do not lead to sodium boiling**



a)



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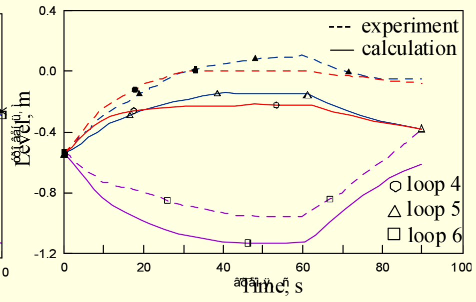


c)

Verification of DINROS code

(a) Comparison of calculated and measured temperature values at SA outlet in experiment with loop 5 shut-down. Sodium temperature
(b) temperature at the outlet of SA in experiment with drop of safety rod
(c) Sodium level in vessels of main pumps of BN-600 reactor with shut-down of 4th and 5th loops

Computer codes used for safety analysis of NPP

GRIF code

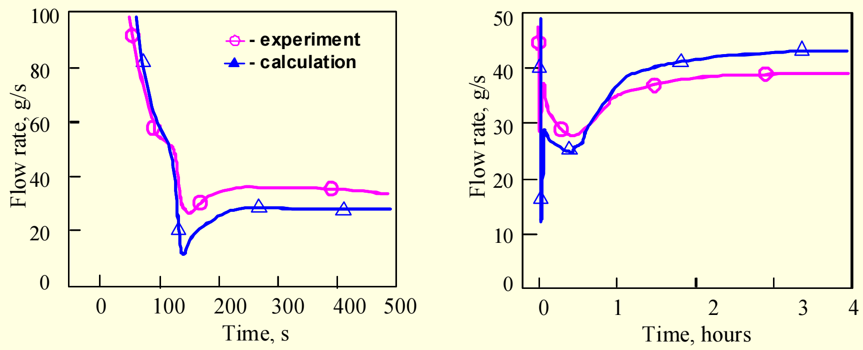
GRIF is also system code intended for analysis of transient and accident processes in sodium cooled FR but with the emphasis on more detailed description of spatial distributions of thermal-hydraulics parameters in primary circuit

CODE MODEL

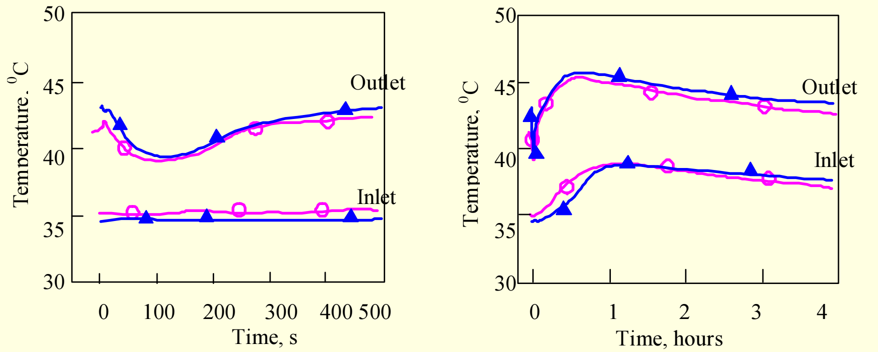
- **3D 1-phase thermal hydraulic models for reactor primary circuit including 3D model for inter wrapper space in the core;**
- **1D 1-phase thermal hydraulic models for secondary and auxiliary systems.**

CODE APPLICATION:

- **abnormal operating conditions and accidents where spatial distributions of parameters over reactor are important (decay heat removal in the case of LOF) ;**
- **DBAs that do not lead to sodium boiling**



a) Water flow rate in the core



b) Water temperature at the inlet and outlet of the core

Verification of GRIF code. RAMONA experiments.

Computer codes used for safety analysis of NPP

HYDRON code

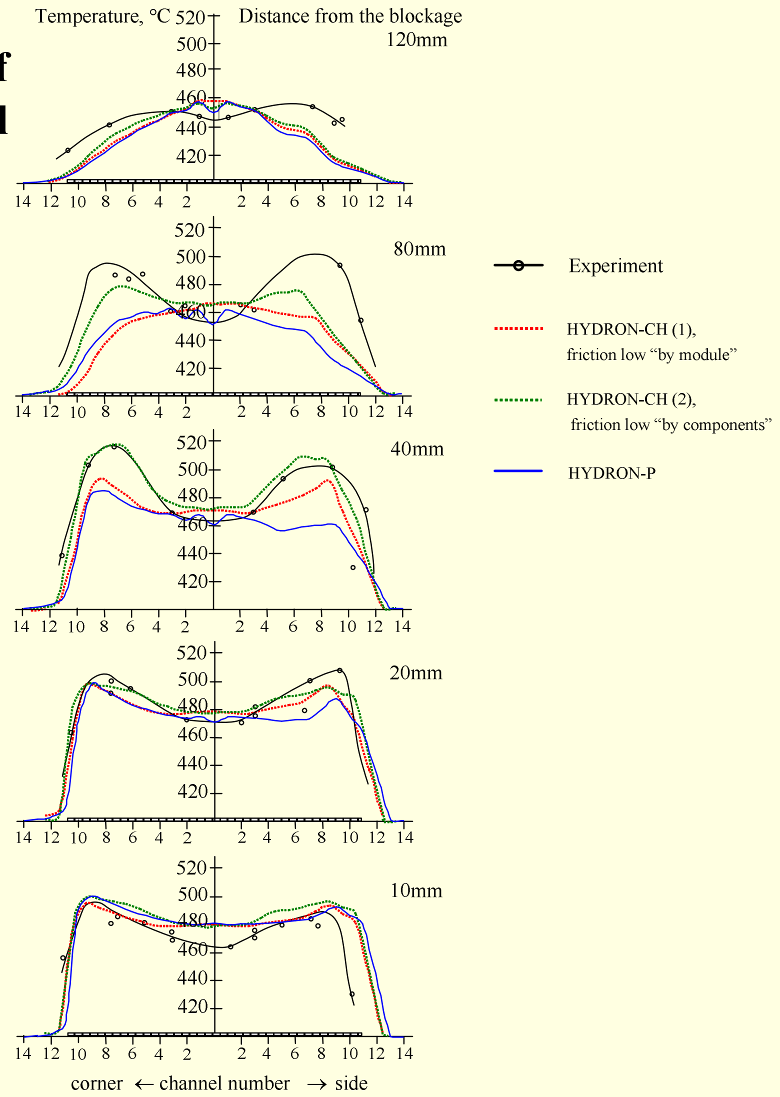
HYDRON - code is intended for analysis of transient spacial distributions of thermal hydraulic parameters in core subassemblies

CODE MODEL

- **3D 1-phase thermal hydraulic models for FSA;**
- **3D thermal model for each pin belonging to FSA**

CODE APPLICATION:

DBA – partial blockage of SA cross section



Verification of HYDRON code on FZK experiments .
Central blockage 49%. Radial temperature profiles in sodium.
(Run №1: V0=4 m/s; q=68 W/cm2; Tin=400°C) .

Computer codes used for safety analysis of NPP

SUBMELT code

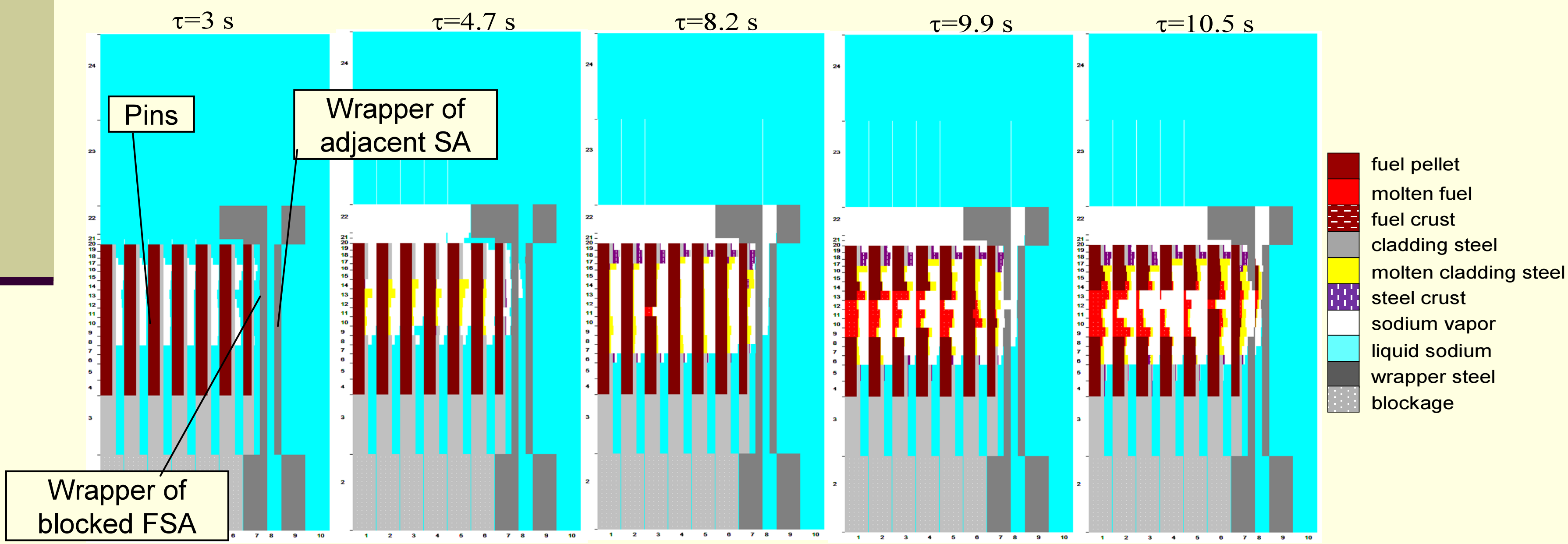
SUBMELT- the code was developed for analysis of process of FSA destruction under conditions of DBA caused by blockage

CODE MODEL

- **2D multi-component multi-speed thermally non-equilibrium model for of boiling , melting and freezing of the components**

EXAMPLE of APPLICATION: **Rapid blockage of SA inlet cross section for BN-600 reactor**

Main objective – **to prove non-propagation of the accident to the neighboring SA**



Fields of components concentrations in τ time points in failed SA

Auxilary codes for study of acident consequencies

TWOCOM - for analytical study of the consequencies of accidents related to gas injection into sodium flow

CODE MODEL

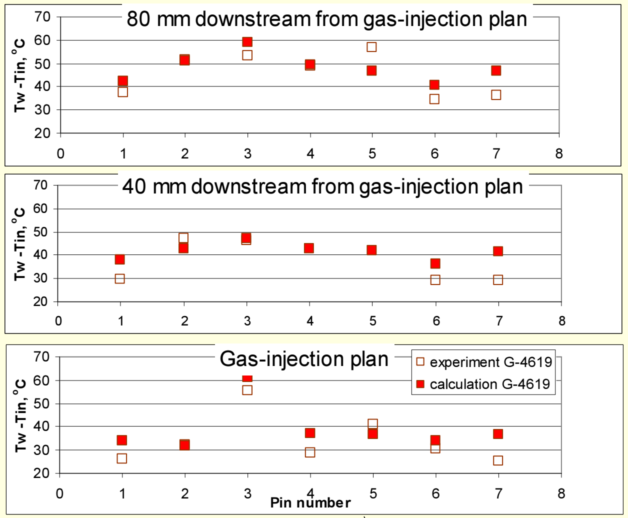
•3D model of subassembly;

•two-component two-speed thermally non-equilibrium model of thermal-hydraulics

CODE APPLICATION:

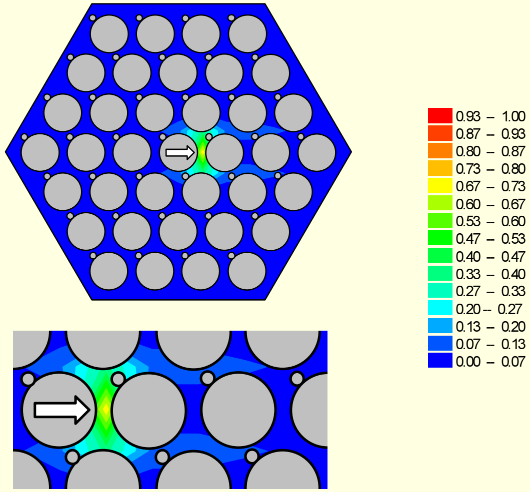
analysis of abnormal operating conditions and DBA leading to fuel element failure and release of fission gas to sodium *(Blockages)*

Experiment G-4619



a)

Calculation G-4619



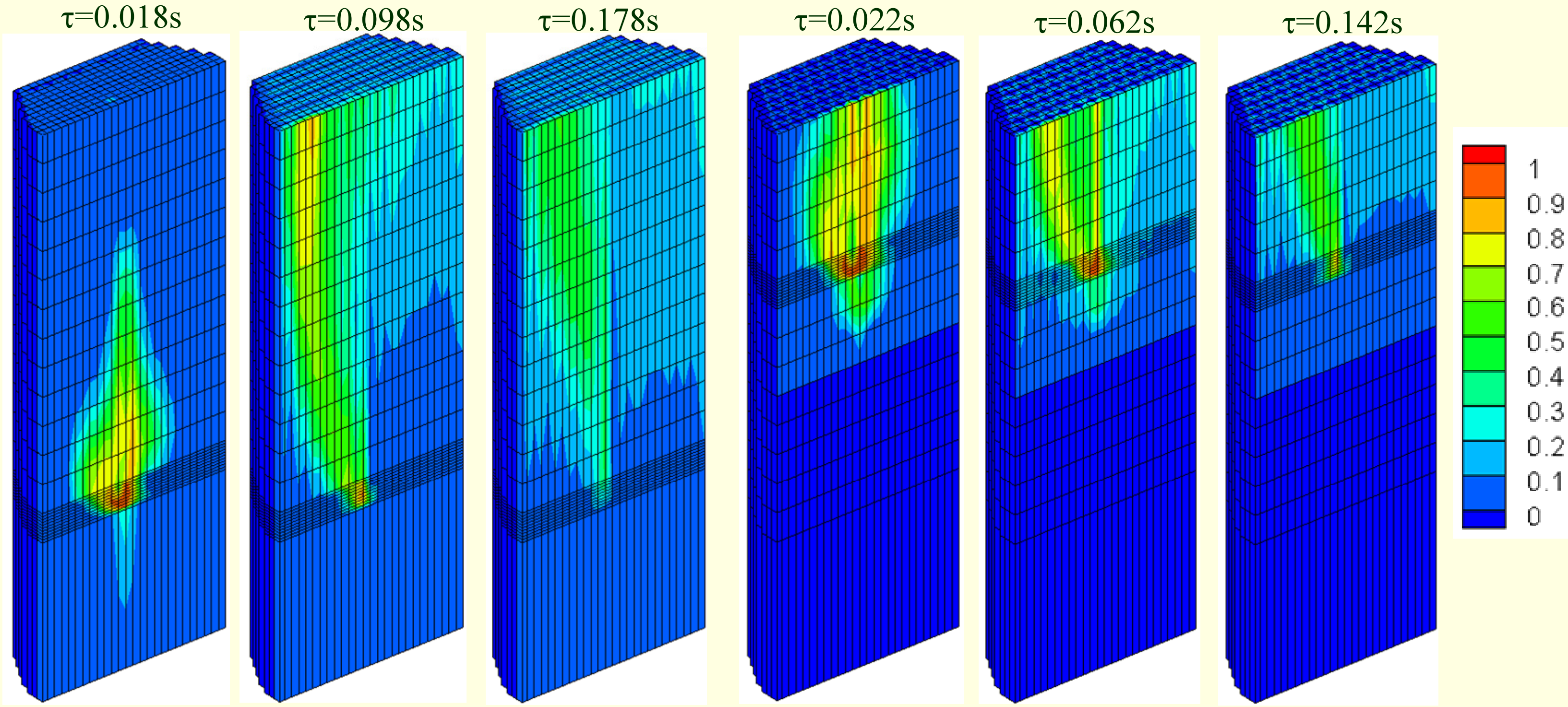
b)

Verification of TWOCOM code.
Experiments with gas release to sodium loop (O-Arai, Japan). Pin-surface temperature rise (a) and gas component concentration field at the hole level (b), hole in the fuel element is shown by arrow.

Auxilary codes for study of acident consequencies

TWOCOM code

EXAMPLE of TWOCOM **APPLICATION to** BN-600 safety analysis:



hole in the core bottom level (k=5) *hole in the core mid plane (k=13)*

Gas concentration in vertical plane (central cross section) in various time points

Gas flowing out from central fuel element through 2mm dia. hole in the bottom and in the mid plane of the core.

CONCLUSION from the study: There is no pin-to-pin propagation of damage inside subassembly

Auxilary codes for study of acident consequencies

ND code - analysis of transportation of delayed neutron precursors released from failed fuel elements
in the reactor and determination of changes of delayed neutron sensor signals

CODE MODEL

- **1F 3D model reactor thrmal hydrailics;**
- **Lagrangian and Eulerian approaches for modeling of transportation of delayed neutron precursors along primary circuit.**

CODE APPLICATION:

"ND" code is used for estimation of effectiveness of fuel element failure detection system

Example of application of "ND" code is presented in POSTER 03-15P

Computer codes used for safety analysis of NPP

BOS code

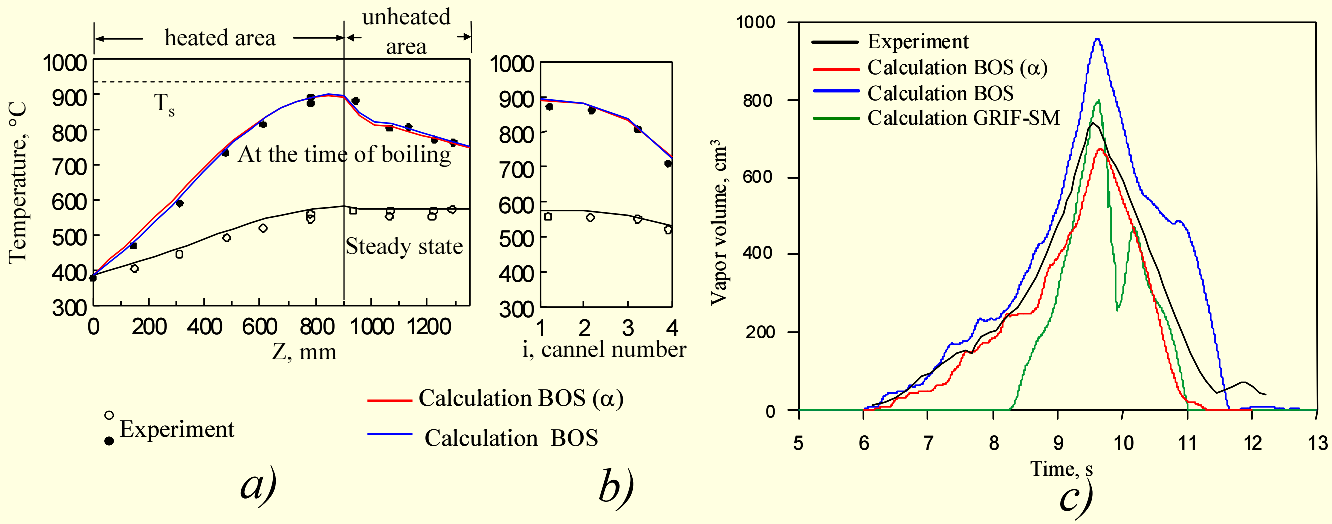
BOS is a code for analytical studies on reactor accidents accompanied by sodium boiling in the core

CODE MODEL

- **two-component thermally non-equilibrium model of two-phase flow**
- **2D core model for core as whole combined with 2D simulation of thermal hydravlic parameters inside FSA**
- **Only core region is modeled**

CODE APPLICATION:

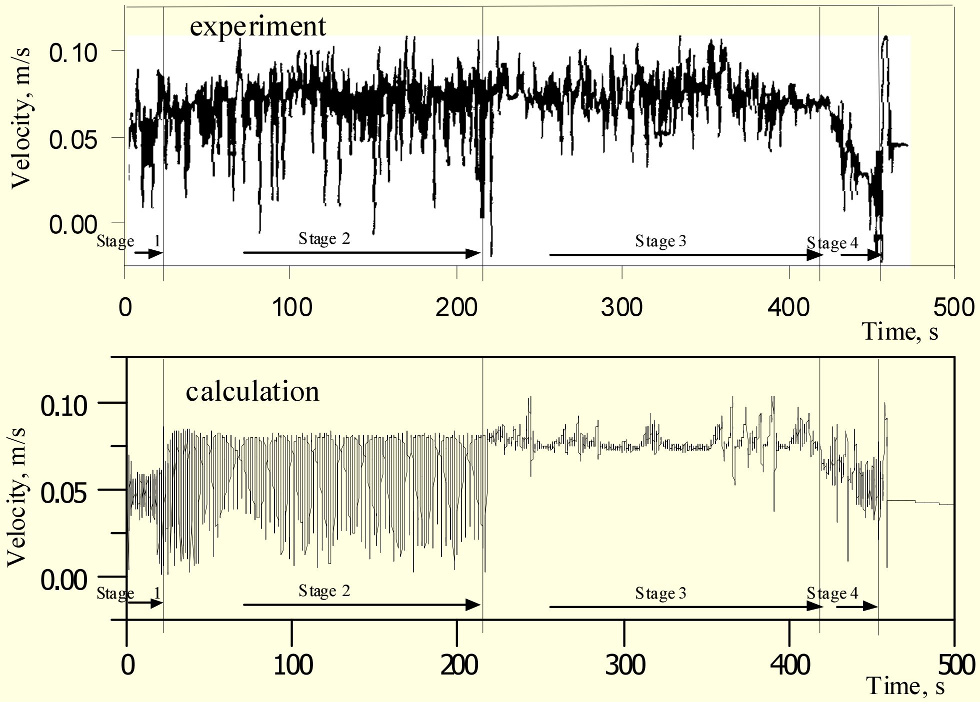
ULOF, UTOP accidents in the frame of safety justification



Verification of BOS code aganst FZK experiment on sodium flow rundown through 37-pin SA
(a) Axial distribution of sodium temperature in the central channel
(b) Radial distribution of sodium temperature for axial coordinate z=775 mm
(c) Total volume of sodium vapor in SA

Computer codes used for safety analysis of NPP

BOS code (2)



Verification of BOS code on sodium boiling in the sodium loop under natural circulation conditions Sodium velocity at the inlet of subassembly (FZK experiment N25). Disturbance of boiling process in sodium natural flow in case of linear power increase up to 40 W/cm

Computer codes used for safety analysis of NPP

COREMELT code

COREMELT - code for study of core desruption

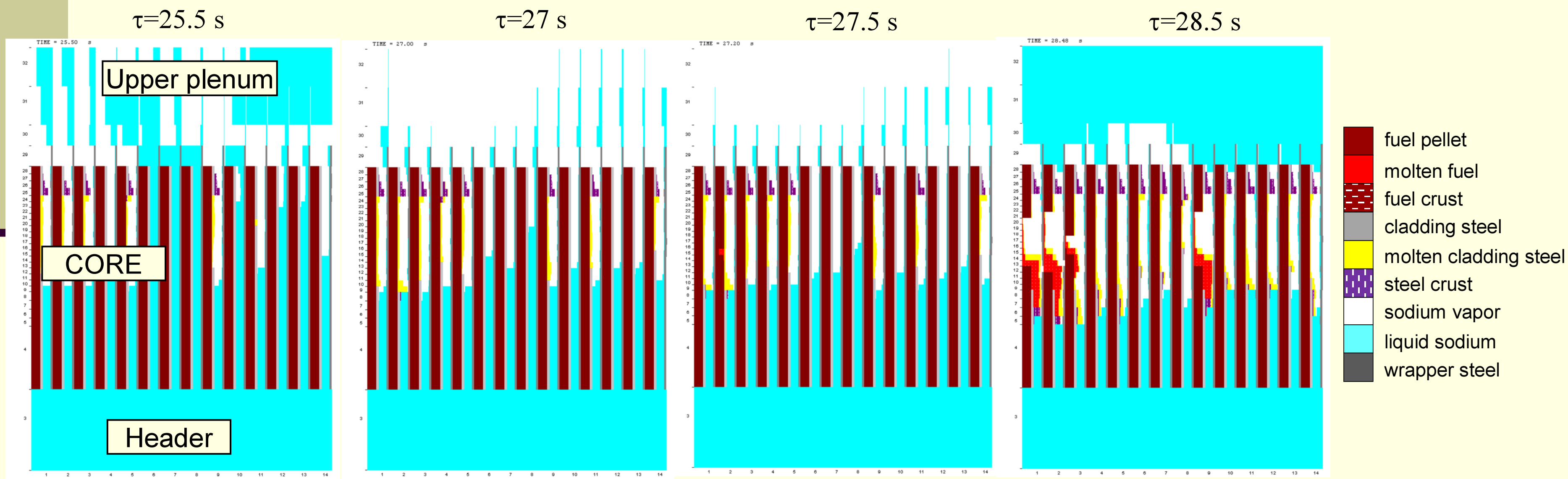
CODE MODEL

- **multi-component multi-speed thermally non-equilibrium model for of boiling , melting and freezing**
- **2D modeling of whole primary reactor circuit**

CODE APPLICATION area:

ULOF and UTOP BDBAs

EXAMPLE of APPLICATION: **Analysis of initial phase of ULOF accident for BN-600**



Fields of components concentrations in τ time points

Computer codes used for safety analysis of NPP

SACTA code + ACME code (Auxilary codes)

SACTA - 3D 1F thermal hydraulic code for detailed analysis of temperature in the core including 3D temperature in SA wrappers for each core subassembly.

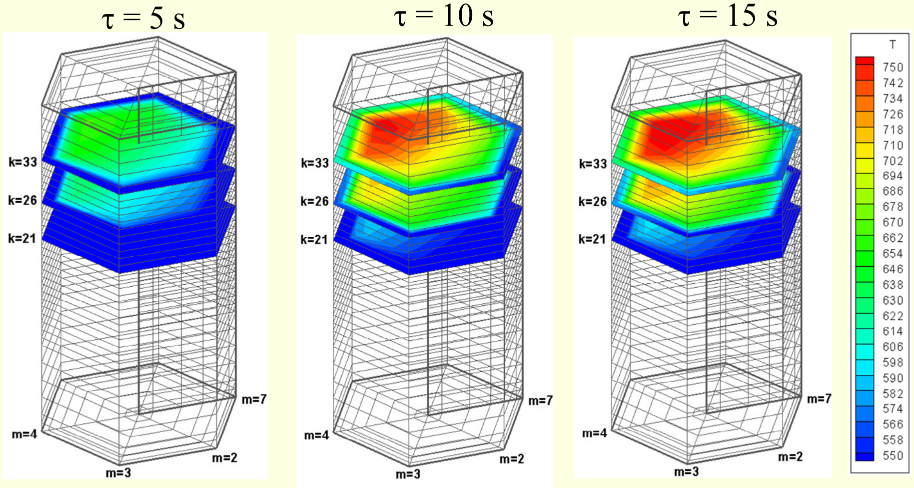
ACME - Analysis of strains and bending of the core and blankets SAs caused by materials swelling, thermal expansion, thermal and radiation creep and heterogeneous temperature

CODEs APPLICATION: evaluation *(in combination with neutronic code)* **of reactivity effects for accidents related with core radial and axial thermal expansion (*Initial stage of ULOF and atc.*)**

SACTA - → ACME → TRIGEX

SA wrapper temperature → Core deformation → Reactivity effect

Example of SACTA calculation →



Temperature field in peripheral core SA under accident conditions

Azimuthal non-uniformity of temperature in horisontal SA cross section leads to SA bending and this results in proper reactivity effects

Example of application of "SACTA" code is presented in POSTER 06-29P

Computer codes for study of FINAL stages of BDBAs of NPP

INTERACT, DINAMIKA-3, ANPEX, BRUT, RGT – the set of codes intended for analysis of late stages of BDBAs

INTERACT was developed for modeling of fuel-coolant thermal interaction in the upper plenum in order to determine pressure loading of reactor structures (vessel. Code is used for analysis of consequences of BDBAs. *(Post disassembly expansion stage of ULOF and UTOP)*

DINAMIKA-3 – Code purpose is evaluation of stresses and strains in reactor vessel and in-vessel structures on the basis of 3D finite-element technique.

ANPEX code was created for 2D analysis of recriticality event.

BRUT code. His application area is 2D analysis of transient processes of molten core material relocation in order to make a conclusion on confinement or non-confiment of molten materials within reactor vessel.

RGT – is special thermal hydraulic code for 3D analysis of radioactive materials transportation in cover gas system of the reactor in order to determine the release of these radioactive materials through ventilation stack to the atmosphere.

Plans for future

All above mentioned codes should be validated and proper sertificate should be issued.

At the moment Codes are in different stages of verification process. For some of them validation is almost finished and most important document – Verification report - is issued. For others verification is still under way.

Task "number one" for nearest future – to comlete validation and to attest codes used for safety justification before the end of 2012.

MORE GLOBAL PLANS for future:

- **development of advanced codes for support of design work on reactors and NPP as a whole on the basis of modern technologies;**
- **development of codes package for optimization of design approaches as applied to the NPP from the standpoint of safety;**
- **development of software for informational support of operation, diagnostics of failures, estimation of residual lifetime of reactor core elements and NPP components.**