RUSSIAN SAFETY APPROACH FOR NPP WITH SFR

Regulatory approach,

Self-immunity of SFRs, prevention of severe accidents and mitigation of there consequences

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CONTENT

RUSSIAN APPROACH TO FAST REACTOR SAFETY ANALYSIS

Russian approach to fast reactor safety analysis (1)

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)

- **International experience collected in the IAEA recommendations**

2. **Regulatory documents and reactor operation regulations are periodically updating on the base :**
   - **Russian and worldwide experience gained in preventing and mitigating of abnormalities and accidents**
   - **Russian and international experience in designing and safety analysis of advanced projects (GEN-4, INPRO)**

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)** ***.***

Russian approach to fast reactor safety analysis (2)

1. Regulatory Documents for NPP Safety Provision determine the set of **design and safety criteria.**
2. One of the Main document that determined common regulatory approach and common requirements **to safety analysis of fast reactors** in Russia is OPB-88/97.

It includes brief list of specific requirements regulating characteristics of various type reactors with regard to safety. It also requires that the SAFETY REPORT should be issued for each reactor. SAFETY REPORT should be developed in accordance with another regulatory document – "Special standard contents of safety analysis report".

3. Large number of **computer codes** are involved in the process of safety justification.

In accordance with Russian regulations those codes should be certified. The procedure of **certification** includes in itself verification of the code and its expertise by the team of independent experts (Team leader is normally the representative of "ROSTEHNADZOR") and expertise by Special Certification Board.

4. Finally it should be proved that project characteristics and reactor behavior under normal and accidental conditions satisfies **the set of design and safety criteria.**

Russian approach to fast reactor safety analysis (3)

Russian Regulatory Documents for NPP Safety Provision

- PBYa RU AS-89. Rules on Nuclear Safety of Nuclear Power Plants Reactor Units
- OPB-88/97. General Statements on Nuclear Power Plant Safety Provision
- NRB-99. Radiation Safety Codes
- OSPORB-99. Basic Sanitary Rules for Radiation Safety Provision
- SP AS-03. Sanitary Rules for Nuclear Power Plant Design and Operation
- NP-032-01. NPP Siting. Basic Criteria and Requirements for Safety Provision

Russian approach to fast reactor safety analysis (4)

Safety criteria applied in the safety analysis of Russian NPPs:

- Population Radiation Safety Criteria
- Personnel Radiation Safety Criteria
- Design Limits of Fuel Elements Damage
- Nuclear Safety Criteria
- Probabilistic Safety Criteria
- Safety Criteria for Radwaste Handling
- Defense-in-Depth Criteria
- Requirements for Safety Systems

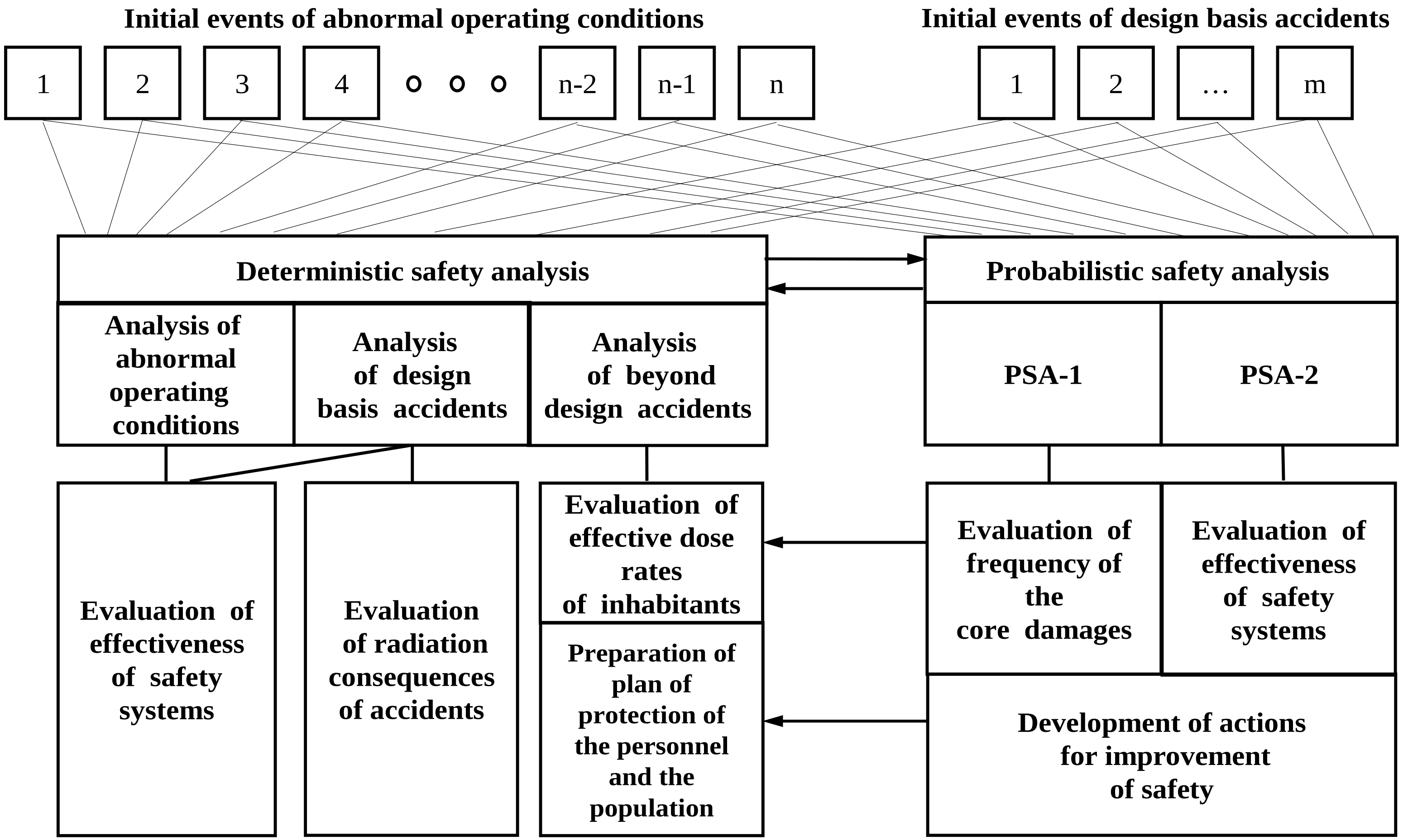
Design Limits of Fuel Elements Damage :

- **Operation limit:** 0.05 % of fuel elements with gas leakage and 0.005 % of fuel elements with direct fuel- to-coolant contact
- **Safe operation limit:** 0.1 % of fuel elements with gas leakage and 0.01 % of fuel elements with direct fuel-to-coolant contact. Clad temperature for fuel pins shall not exceed 800 °C during a short-term period
- **Maximum design limit:** destruction of fuel elements in seven fuel assemblies
- **Design limits for design-basis accidents:** to be established in the design

For the reactor vessel:
- 590 °C for abnormal operation and design-basis accidents
- 700 °C for beyond design-basis accidents

Russian approach to fast reactor safety analysis (5)

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



Russian approach
to fast reactor safety analysis (6)

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 be changed on the basis of analysis of specific reactor plant design, its operating modes and maintenance regulations

Russian approach to fast reactor safety analysis (7)

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 destroyed SAs is not more than 7

Russian approach to fast reactor safety analysis (8)

Deterministic safety analysis of NPP design

(Analysis of beyond design basis accidents (BDBA))

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

- 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

Accidents selected for analysis should potentially be able to lead to severe core damage and release of radioactive products out of primary circuit

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 IMPLEMENTED OR TO-BE IMPLEMENTED FOR SAFETY ANALYSIS OF THE NPP WITH THE FAST REACTORS

Total number of thermal-hydraulics codes normally involved in safety analysis is more then 20. 15 codes are listed below.

Code application area	Code title
"System" codes implemented for analysis of the neutronics, thermo-hydraulic, and thermo-mechanical transient processes based on the 1-D distributed and lumped models	DINROS
	DINRAN
Codes implemented for 2-D and 3-D analysis of the reactor thermo-hydraulics, including the accidents that don't lead to sodium boiling	GRIF
	ND
	HYDRON-CH
	SACTA
Codes intended for analysis of the severe reactor accidents, that can simulate sodium boiling and molten material relocation in the course of core meltdown	GRIF-SM, BOS
	TWOCOM
	SUBMELT
	COREMELT, GVA
Codes implemented for analysis of possible cosequencies of severe accidents -mechanical impact on the reactor vessel due to thermal interaction of molten fuel with coolant and due to a vapor bubble expansion - power excursion due to formation of secondary critical configuration	ANPEX
	INTERACT
	DYNAMICA 3
Codes intended for an analysis of the processes related with confining the molten fuel in the reactor vessel and radioactive gas release in the case of severe accident	BRUT
	RGT

General results of safety analysis of Russian fast reactors

1. In-depth safety substantiation of unit №3 of BAES plant with BN600 reactor, performed in the frame of works on reactor life-time extension confirmed that all reactor characteristics meet to regulatory criteria.

2. Final safety substantiation of unit №4 of BAES plant with BN800 reactor is close to its completion now. Some additional improvements of core and reactor design (Near-zero SVRE, new system of decay heat removal, measures on decrease of probability of sodium leakages) were resulted in the increase of BN800 reactor safety refer to BN600.

3. BN1200 project is developing now in accordance with Federal Target Program as a reactor of GEN-4. Innovations and improvements directed to increase of economical efficiency and safety will be realized in that project.

STUDIES ON INFLUENCE OF SODIUM VOID REACTIVITY EFFECT ON THE CONCEPT OF THE CORE AND SAFETY OF ADVANCED FAST REACTOR

SVRE and core concept (1)

NEW requirement after Chernobyl accident
– zero integral sodium void reactivity effect (SVRE)

Possible concepts:

1. arrangement of sodium plenum above the core
   (improved safety feature – in the case of severe accident voiding of sodium plenum due sodium boiling leads to introduction of negative reactivity)
2. decrease of core height
3. annular or modular core and etc.
4. adding moderator to the reactor

Concept 1 was adopted - *for the BN-800 reactor design*
Combined concept (1+2) is proposed - *for BN-1200 type reactor*

Main advantage – improved safety feature
(in the case of severe accident voiding of sodium plenum due sodium boiling leads to introduction of negative reactivity)
Main disadvantage – deterioration of technical and economical characteristics
(decrease of breeding ratio, significant increase of dimensions of the core diagrid and rotating plug, decrease of control rods worth etc.)

The studies on influence of sodium void reactivity effect (SVRE) on safety characteristics of BN-1200 type reactors carried out for ULOF accident on BN-1200 type reactor

SVRE and core concept (2)

Procedure of complex analysis

Core designs that were chosen for comparative safety analysis of BN-1200 type reactor:

1. Core with sodium plenum designed in the stage of technical proposal *(reference option)*
2. Core with upper axial fertile blanket *(Traditional design)*
3. Core of increased height (100cm refer to 85cm) and with sodium plenum *(Combined option)*

Comparative analysis includes in itself:

- Analytical analysis of neutronic characteristics
  (Reactivity effects, heat release distributions and etc.)
- Reactor safety analysis for three core designs and under ULOF accident affected by SVRE
- Option of most favourable core design within the framework of specified scope of the analysis

SVRE and core concept (3)

Input data

Table 1 Main technical characteristics of the reactor plant

1	Reactor thermal power, MW	2800
2	Range of power control, % N_o	25-100
3	Reactor refueling interval, eff. days	330
4	Parameters of the primary circuit: sodium temperature at the core inlet, ^oC sodium temperature at the IHX inlet, ^oC sodium flow rate in IHX, kg/s primary sodium pump pressure head, mlc	 410 550 15784 58
5	Parameters of the secondary circuit: sodium temperature at the SG inlet, ^oC sodium temperature at the SG outlet, ^oC sodium flow rate in one loop, kg/s secondary sodium pump pressure head, mlc	 527 355 3193 47
6	Number of primary and secondary loops	4

Comment: The same reactor parameters were chosen for all considered cases

SVRE and core concept (4)

Input data

Table 2 ***Main characteristics of core designs under study the reactor plant***

	CASE 1	CASE 2	CASE 3
Core height, cm	85	85	100
Upper blanket	Sodium plenum	Fertile blanket	Sodium plenum
Number of SAs	432	432	366
Fuel enrichment, %.	17.96	17.62	17.11
Max fuel burn-up, % h.a	16.44	16.38	18.51
Average fuel burn-up in spent SA, % h.a.	10.76	10.56	10.79
SVRE, % $\Delta k/k$	+0,5	+1.9	+1.31
Excess reactivity for the fuel burn-up, % k/k	1.5	1.3	1.29
BR	1.25	1.34	1.25
Core BR	0.88	0.88	0.93
Pu inventory, kg	7531	7390	7168
Max power density, W/cm3	378	374	425
Max linear power, kW/m	41.9	41.4	47.1
Max SA power, MW	8.24	8.42	10.64

Comments:

- **Number of Sas is decreased for case 3**
- **SRVE is minimal for case 1**
- **Additional excess of reactivity for burn-up is necessary for case 1**
- **Breeding ratio is the same for cases 1 and 3**
- **Power density is higher for case 3**

SVRE and core concept (5)

Reactivity effects and coefficients

Table 3 Reactivity effects and coefficients due to core voiding

	Drying-out areas	1st 1/3 radial section of the core	2nd 1/3 radial section of the core	3rd 1/3 radial section of the core	All SAs of the core
1	BA + SP+1/3 section of the core height	-0.00452 0.00167 -0.00247	-0.00360 0.00128 -0.00197	-0.00232 -0.00004 -0.00156	- - -
2	BA + SP+2/3 section of the core height	0.00021 0.00702 0.00425	0.00018 0.00519 0.00114	-0.00156 0.00075 -0.00053	-0.00007 0.01262 0.00571
3	BA + SP + whole core height	0.00434 0.01102 0.00930	0.00278 0.00760 0.00308	-0.00212 0.00104 -0.00005	0.005 0.01877 0.01310

Comment:

- **Integral SVRE is positive for all cases but for reference option it is minimal**
- **For cases 1 and 3 voiding of upper core part gives negative effect**

__ Case 1 (Hcore=85, with Na plenum)
__ Case 2 (Hcore=85, with upper axial fertile blanket)
__ Case 3 (Hcore=100, with Na plenum)

SVRE and core concept (6)

COREMELT code was used for ULOF study

CODE PURPOSE is analytical studies on BDBA in the reactor accompanied by

- **sodium boiling**
- **fuel and steel melt-down**
- **molten fuel and steel relocation and refreezing in the core**

CODE MODEL

- **2D multi-component, multi-speed thermally non- equilibrium thermal hydraulic model, based on “Porous body” approach**
- **“Point kinetics”**

CODE 2D CALCULATIVE DOMAIN covers whole reactor including most important components of the primary circuit (such as heat exchangers and pumps

SVRE and core concept (7)

Behavior of reactor parameters ULOF accident

Comments:

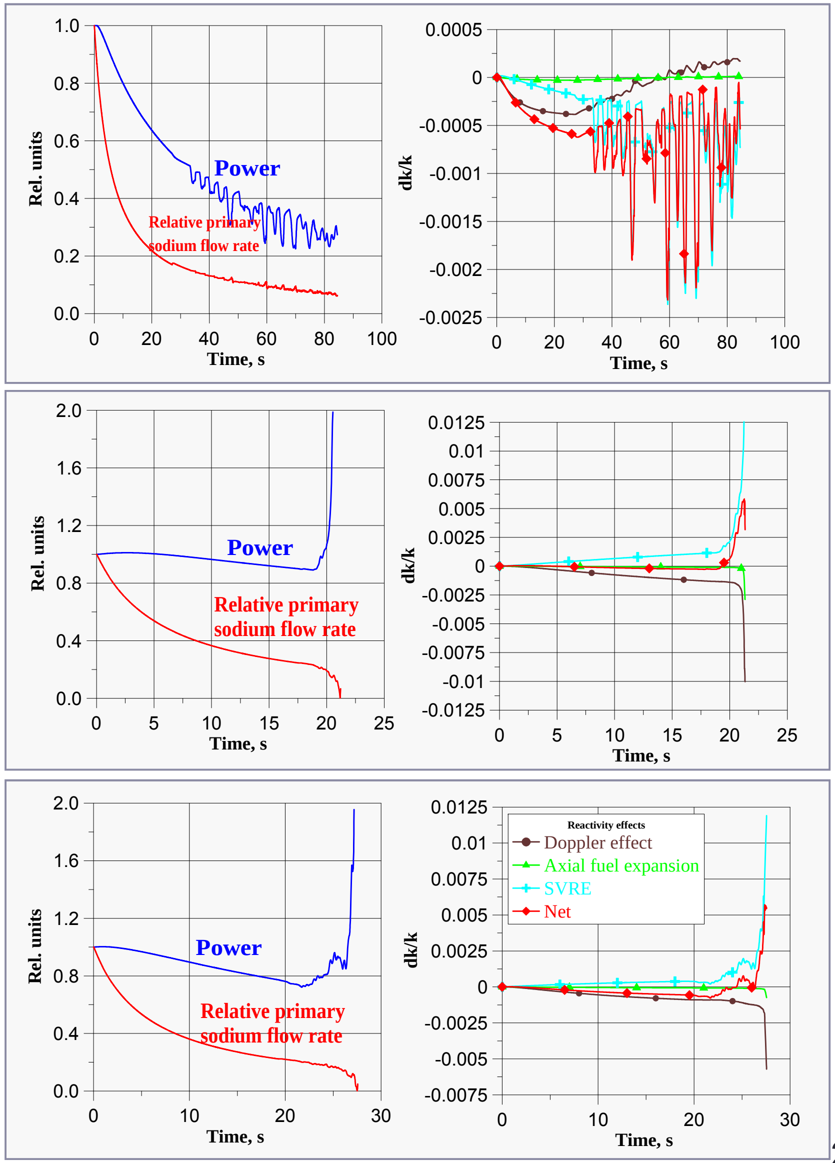
Case 1 - Power gradually decreases due to negative Net reactivity, core is heating up and after half a minute sodium boiling starts in the core, voiding of upper core part gives additional negative contribution in Net reactivity and power continue to go down. Reactor self-protection is provided

Case 2 and 3 - Sodium boiling results in positive contribution to Net reactivity, reactor runaway occurs leading to the core disruption after 20-28 seconds Reactor self-protection is NOT provided

Case 1

Case 2

Case 3



SVRE and core concept (8)

Conclusion

- Reference **Case 1** has advantages of two other options from the standpoint of reactor safety. It assures reactor self-protection (in contrast to the other options) under conditions of ULOF BDBA. Unless potential losses that may emerge in case of NPP accident are taken into account, reference option loses out to case 2 and 3 in energy cost value
- **Case 2** *(with the upper fertile blanket)* has an advantage of reference option from the standpoint of breeding ratio and this point may become of crucial importance under certain conditions
- **Case 3** *(core height increased up to 100 cm)* is most preferable from economical standpoint. Its advantages are caused by
  - reduction of the number of fuel SA in the core,
  - decrease of fuel inventory
  - of excess reactivity for the fuel burn-up
- It is difficult to make quantitative estimates of potential losses caused by the accidents, however it is obvious that more or less severe accident in any NPP would cause a chain of considerable economical losses in the whole nuclear energy industry. Therefore **final preference is given** to **Case 1 design assuring self-protection of the reactor** in spite its additional costs and even under conditions of incredible BDBA .

COMPARISON OF SELF-PROTECTION FEATURES OF ADVANCED FAST REACTOR WITH MOX AND NITRIDE FUEL

Comparison of self-protection features of advanced fast reactor with MOX and Nitride fuel

The Comparative analysis of self-protection features was carried out for ULOF accident for BN1200-type reactor

Three cases were studied:

Case 1 – ULOF on the reactor with MOX-fueled core;

Case 2 – ULOF on the reactor with Nitride-fueled core;

Case 3 – ULOF on the reactor with Nitride-fueled core with additional insertion of external positive reactivity

Self-protection features of Core with MOX and Nitride fuel (1)

COREMELT code was used for ULOF study

Reactivity effects that were taken into account:

- Doppler effect
- Sodium temperature and voiding effect
- Axial expansion of fuel pins

Calculative domain of COREMELT code simulating the design of advanced fast reactor

Sodium cavity

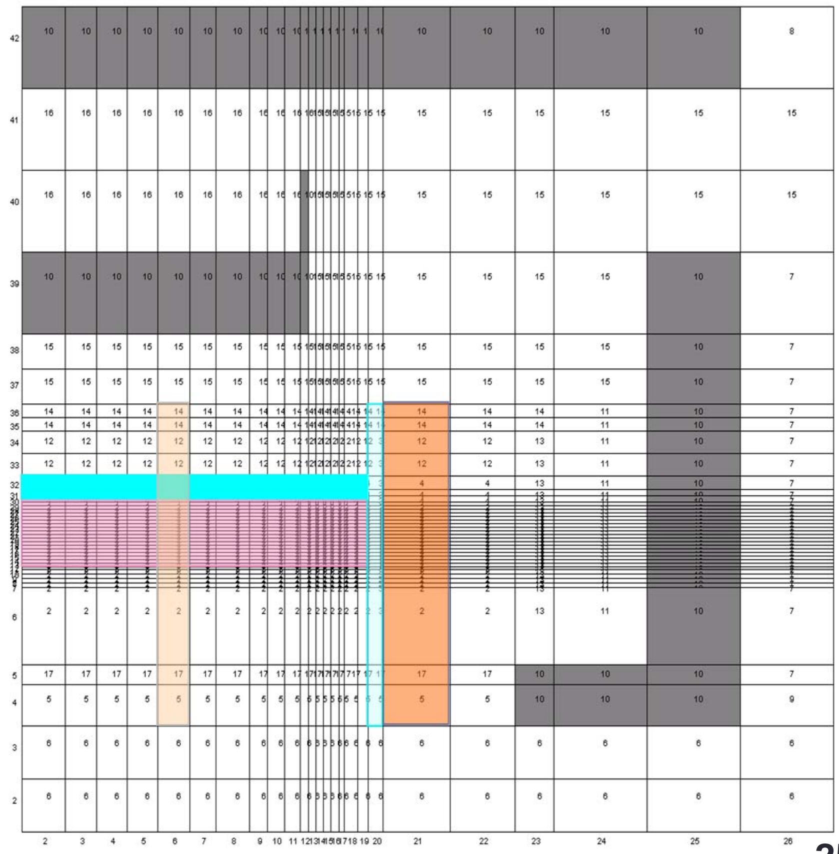
Core

“Inter-wrapper” sodium

Control rods

Radial blanket

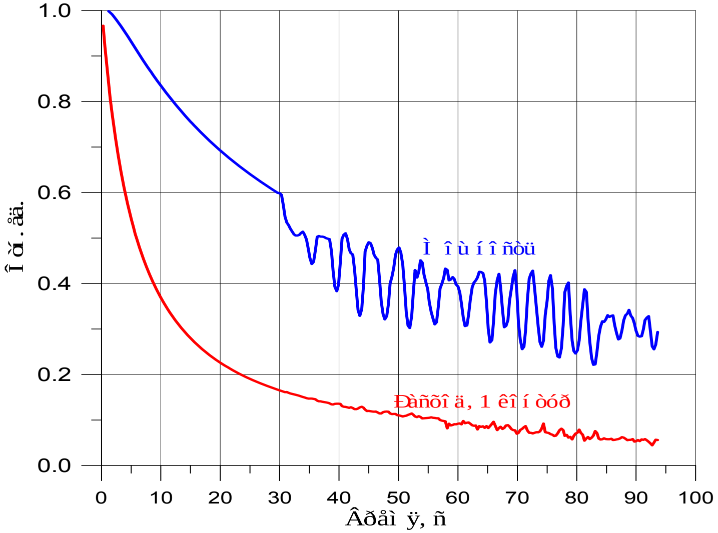
Structures



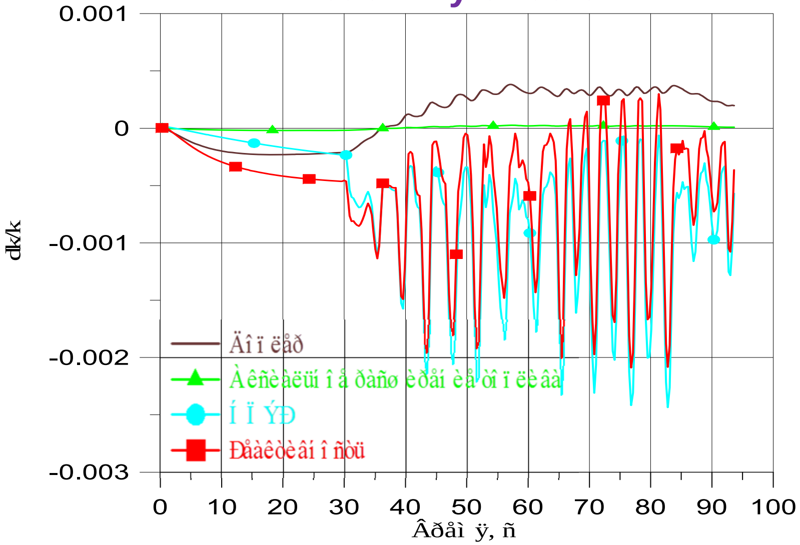
Self-protection features of Core with MOX and Nitride fuel (2)

CASE 1. Core with MOX fuel

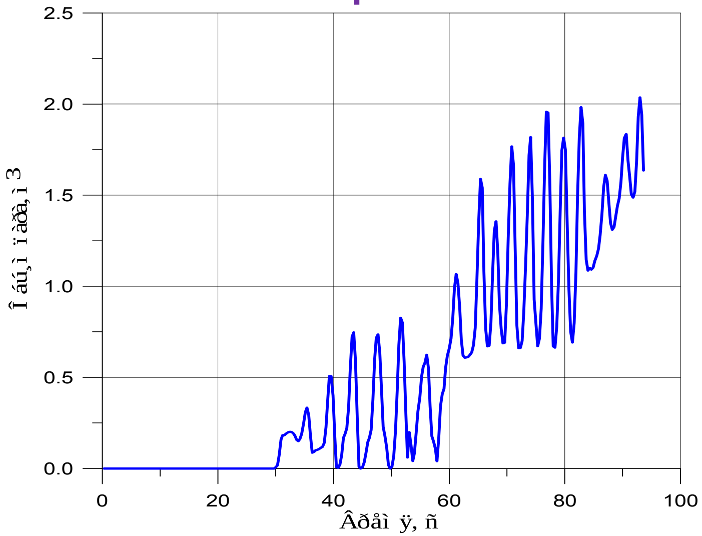
Reactor power and primary flow rate



Reactivity effects



Sodium vapor volume



Comment:

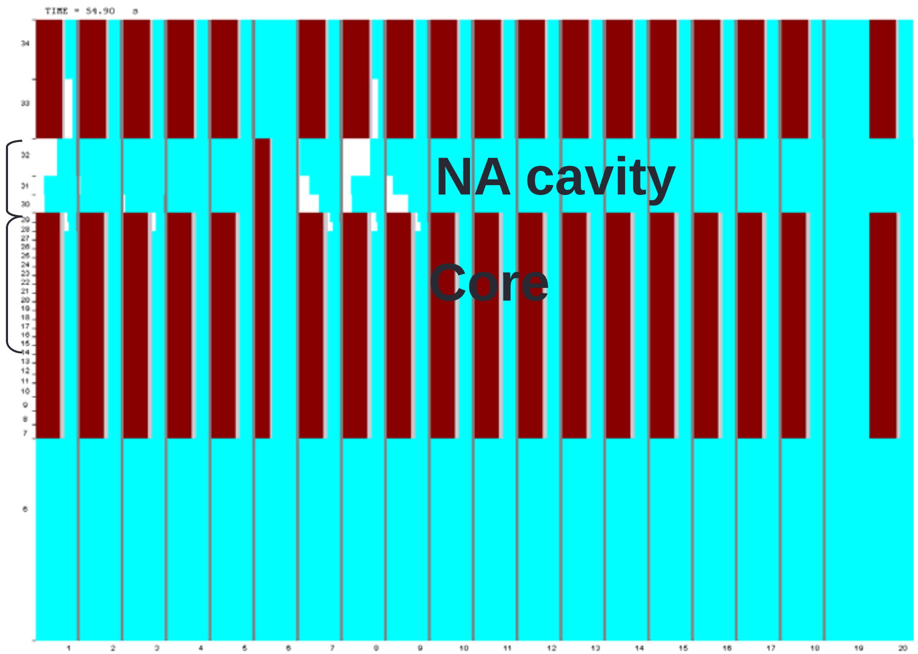
Power gradually decreases due to negative Net reactivity, core is heating up and after half a minute sodium boiling starts in the core, voiding of upper core part gives additional negative contribution in Net reactivity and power continue to goes down. Reactor self-protection is provided but the safety margin is not very large.

As it was demonstrated above the increase of core height from 85 to 100 sm results in core destruction

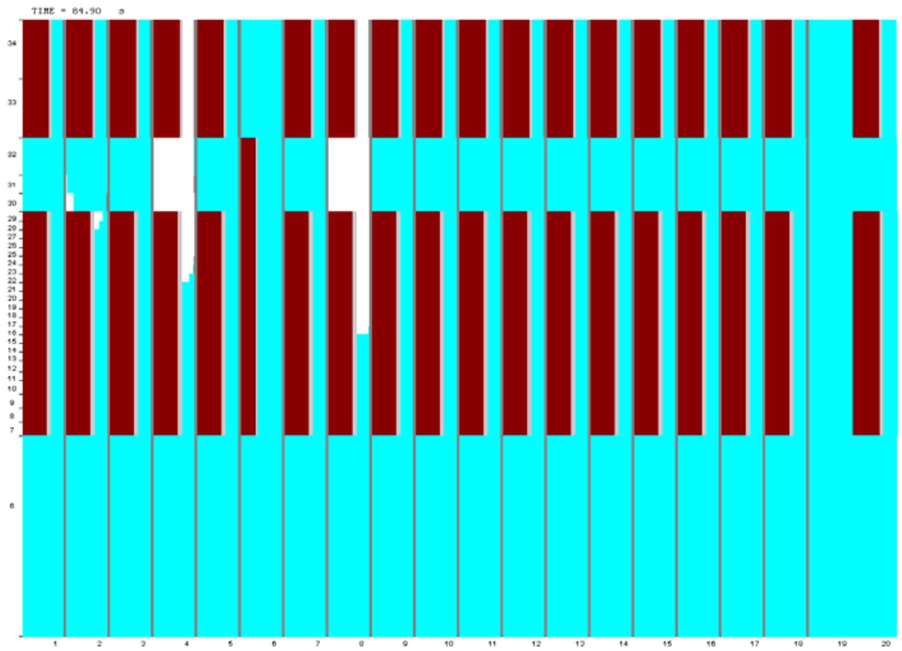
Self-protection features of Core with MOX and Nitride fuel (3)

CASE 1. Core with MOX fuel

V_{vapor}=0.4 m^3 V_{vapor}=1.2 m^3



τ = 55 s



τ = 85 s

Na	Fuel	Pin cladding	Na vapor	FSA hexcans

Sodium vapor distribution

Comment: Boiling is observed in a few selected channals

Self-protection features of Core with MOX and Nitride fuel (4)

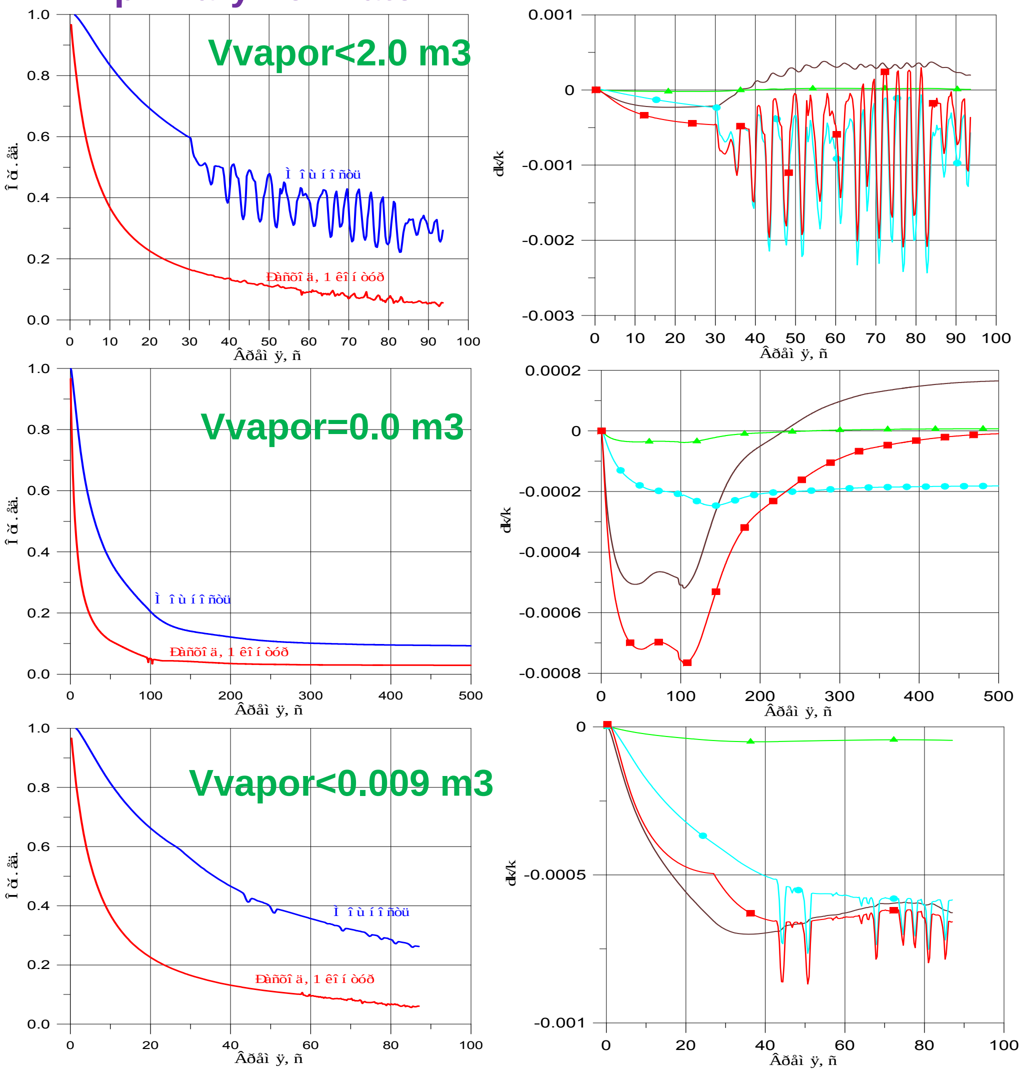
Comparison of different cases

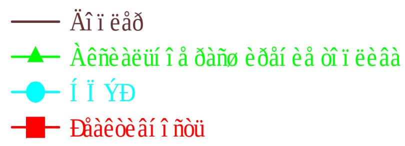
Reactor power and primary flow rate | **Reactivity effects**

Case 1 *MOX*

Case 2 *Nitride*

Case 3 *Nitride+Pos. react.*





Comments:

Case 2 – Due to more pronounced Doppler effect net reactivity and Power rapidly go down. As a result sodium boiling does not occurs.

Case 3 – Insertion of external positive reactivity - 0.6E-3 - causes sporadical flashes of sodium boiling in the upper part of the core where SVRE is negative

Positive reactivity ≈ 0.6E-3 is inserted during 30 seconds

Self-protection features of Core with MOX and Nitride fuel (5)

Conclusion

- Reactor with MOX fuel is close to its self-protection boundary in terms of SVRE value
- The level of self-protection of the core with nitride fuel is much higher if to compare with traditional MOX-core

Self-protection margin– condition that ensure reactor cooling down without core melting in the case of ULOF accident

BN-1200.
REALIZATION OF “DEFENSE IN DEPTH” PRINCIPLE and
TECHNICAL PROPOSALS AIMED AT NPP SAFETY IMPROVEMENT

BN1200 reactor is positioned as a reactor of 4-th generation
Therefore its design have to meet two well-known international initiatives GEN-IV and INPRO

Realization of “Defense in depth” principle in BN-1200 (1)

Level of defense-in-depth	Increase of defense-in-depth level proposed in **GEN IV** methodology	Increase of defense-in-depth level proposed in **INPRO** methodology
Level 1	In order to prevent abnormal operation and accidental conditions the set of operation modes considered in basic design should be extended	To assure high accent on inherent safety and passive safety properties
Implementation of measures in BN-1200 reactor	The set of operating modes is extended. Reactor inherent safety is provided by reactor neutronics and thermal hydraulics (zero SVRE, high BR of the core and low specific power rate of fuel)	

Realization of "Defense in depth" principle in BN-1200 (2)

Level of defense-in-depth	Increase of defense-in-depth level proposed in **GEN IV** methodology	Increase of defense-in-depth level proposed in **INPRO** methodology
Level 2	In order to control abnormal operating conditions and detect damage diagnostic systems should be improved and "man-machine" interface should be updated	To give higher priority to updated control and monitoring systems with the improvement of reliability, intellectuality and limiting indices
Implementation of measures in BN-1200 reactor	Updated system of reactor automatic control was proposed. Use of well-developed systems for informational support of reactor operation is stipulated	

Realization of “Defense in depth” principle in BN-1200 (3)

Level of defense-in-depth	Increase of defense-in-depth level proposed in **GEN IV** methodology	Increase of defense-in-depth level proposed in **INPRO** methodology
Level 3	In order to control design basis accidents, larger set of initial events and accidents is considered including some conditions triggered by multiple failures. Their analysis is made using “best estimate” methods and data. Probabilistic estimates and other analytical methods would make their contribution to accident analysis. Special attention should be paid to low-probable accident sequences	To achieve fundamental safety characteristics by optimization and combination of active and passive properties of equipment; to limit nuclear fuel damage, to increase permissible postponing intervention by several hours
Implementation of measures in BN-1200 reactor	Increase of safety level under design basis accident conditions is provided by the fuel power rate decrease, by in-vessel control systems. Hazardous development of accidents is eliminated even in case of safety system failure. Continuous improvement and systematic certification of computer codes used for safety analysis increases reliability of their predictions	

Realization of “Defense in depth” principle in BN-1200 (4)

Level of defense-in-depth	Increase of defense-in-depth level proposed in **GEN IV** methodology	Increase of defense-in-depth level proposed in **INPRO** methodology
Level 4	For the purpose of prevention of hazardous development of accidents, the wide range of preventive strategies of accident control is considered, including means of control of accidents accompanied by severe damage of the core. These include appropriate instruments for maintaining containment functions, such as containment building capability of withstanding to hydrogen burning or preventing melt-through	To increase reliability of the systems controlling complex of sequential events, to decrease the probability of severe core damage at least by an order of magnitude or even more in case, if the system site is located within the boundaries of urban area
Implementation of measures in BN-1200 reactor	Prevention of core damage under severe beyond design accident conditions is based on both inherent safety and two types of passive safety systems. Probability of the core damage decreases by at least one order of magnitude as compared to regulatory requirements – down to 10^{-6} 1/reactor·year	

Realization of “Defense in depth” principle in BN-1200 (5)

Level of defense-in-depth	Increase of defense-in-depth level proposed in **GEN IV** methodology	Increase of defense-in-depth level proposed in **INPRO** methodology
Level 5	Radiological consequences or radioactivity release should be decreased by appropriate modifications on previous levels and, particularly, by appropriate limitation of release source	No necessity for evacuation of groups of inhabitants or resettlement outside facility site should arise for innovative nuclear power systems
Implementation of measures in BN-1200 reactor	Releases of radioactive products into the atmosphere are prevented by limitation of the extent of the core damage, as well as by the special device for confinement of radioactive gas and aerosol release from the reactor. The above measures eliminate necessity of inhabitants evacuation to the NPP surroundings in case of accident	

Technical proposals aimed at NPP safety improvement (1)

1 Reactor core

1.1 Minimization of sodium void reactivity effect (SVRE)

1.2 Decrease of excess reactivity during cycle by increasing core BR

1.3 Decrease of specific power rate of fuel

1.4 Increase of stability of the core power profile during its lifetime

1.5 Decrease of coolant temperature rise in the core as compared to that in existing reactors

1.6 Decrease of operating limits of fuel element damage by means of use of the new structural materials and improvement of their operating conditions

1.7 Limitation of max design value of the fuel element damage by the scale of one SA owing to SA characteristics and use of advanced in-reactor control systems

1.8 Use of safety system devices based on passive principles

Technical proposals aimed at NPP safety improvement (2)

2 Reactor facility

2.1 Designing components, systems and structures of reactor facility for external effects of intensity higher than that in the existing analogs.

At least 7 magnitude is assumed for maximum credible earthquake and 6 magnitude is assumed for design earthquake according to MSK–64 scale.

Airplane parameters are assumed not lower than those recommended in item 6.8 of NP-068-05 and IAEA №50-SG-S5 (weight 5.7 tons, velocity 360 km/hour).

External shock wave; frontal pressure is assumed equal to 30 kPa with compression phase duration up to 1 s.

2.2 Use of safety jackets for all pipelines of the secondary circuit, this practically excluding non-radioactive sodium into reactor building rooms

2.3 Use of main sodium pumps having longer coast-down in case of power loss and rpm control within 15 - 100% range

2.4 Use of additional mobile power sources for the main pumps and decay heat removal system under extreme conditions

Technical proposals aimed at NPP safety improvement (3)

3 Monitoring, control, safety and diagnostics systems

3.1 Use of developed devices of in-vessel control. These devices should timely detect abnormal operation and damage of any core elements including fuel elements and SA

3.2 Adoption of safety system based on passive principles would assure NPP safety improvement without deterioration of partial power operation modes

3.3 Use of advanced decay heat removal system with emergency heat exchangers incorporated into the reactor vessel

3.4 Use of intellectual system of personnel information support under steady state, transient and accidental conditions

3.5 Development of complex systems for diagnostics of damage and evaluation of residual life time of the core structural elements and reactor components

3.6 Use of device for gas aerosols release confinement in case of functioning of reactor vessel safety valve with their further discharge through filters to decrease radioactivity release to the environment. This allows decreasing effective radiation dose of the inhabitants and, hence, decreasing protection measures planning area down to the NPP site and, at the extreme, eliminating any need for special measures for inhabitants protection

Technical proposals aimed at NPP safety improvement (4)

4 Updating reactor control under normal operation, transient and accident

4.1 Maintaining constant steam conditions within the wide range of reactor power using updated control system

4.2 Assurance of reactor operation in variable load mode, in particular, the possibility of automatic power decrease in case of accident down to auxiliaries power supply level if NPP turbo-generator is disconnected from the grid|

4.3 Assurance of reactor operating limits in case of abnormal operation by automatic control system without safety system actuation

THANK YOU FOR YOUR ATTENTION