SFR Inherent Safety Features and Criteria Analysis

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Abstract. This study presents the inherent safety performance of BN-type reactor 2800 MW of thermal power with MOX core during ATWS initiated by various accident initiating events (e.g. pump trip and pump seizure, reactivity insertion, secondary circuit failure etc.) with simultaneous failure of all shutdown systems in all cases under investigation. The BN-type reactor 2800 MW of thermal power with MOX core was under investigation.
The impact of various safety features on SFR inherent safety performance during ATWS was also analyzed. The decrease in hydraulic resistance of primary loop, increase in primary pump coastdown, the implementing of thermo-mechanical, leakage based and other self-actuated safety systems considered as additional natural feedbacks were considered. Performing analysis resulted in a set of recommendations to the characteristics of the features referred above for the purpose of enhancing the inherent safety performance of SFR under investigation.
In order to exclude the safety barrier rupture during ATWS the set of criteria defining the ATWS processes dynamics and requirements to them were recommended based on achieved results. These criteria are include the steady state natural circulation level (must exceed 7%. of nominal flow rate in most severe case), the coolant flow rate drop under the steady state natural circulation level (must be missing 1.5% of nominal flow rate in most severe case) and the time it takes for flow rate to reach the steady state natural circulation level (must be missing 101 s). The recommendations for the way to implement the self-actuated safety systems are also elaborated.
The analysis of admitted assumptions and obtained results revealed that in order to develop the refined requirements for the proposed criteria it is necessary to couple the SFR performance analysis for ATWS with uncertainty analysis. It is also necessary to take into account heat removal through passive heat removal systems even in a failure mode by heat-conductivity through the HX walls and to refine the acceptable temperatures of critical components of reactor (fuel, cladding, coolant and reactor tank) with respect to reactor inherent safety. The suitability of chosen acceptable temperatures values of critical components of reactor is discussed.
The results of the inherent safety analysis presented in this study are obtained by using the one-dimensional DYANA code for inherent safety analysis of fast liquid metal cooled reactors. Estimated sodium temperature and mass flow obtained from LOHS+LOF analysis via DYANA code were in reasonable agreement with those obtained from PHENIX benchmark end-of-life test.

Key Words: Inherent Safety, Sodium Fast Reactor, Natural Circulation, Accident Analysis.

1. Introduction

Nowadays the one of the most important advanced NPP requirements is the safety requirement. NPP safety is a result of safety management, legal and engineering measures. Engineering measures employed so far include those that enhancing inherent safety performance of reactor. According to [1] «inherent safety refers to the achievement of safety through the elimination or exclusion of inherent hazards through the fundamental conceptual design choices made for the nuclear plant». It should be noted that inherent safety doesn't exclude the implementation of the active and passive safety systems in reactor design, but it rather ensures the reliance and diversity of reactor safety performance. The aim of the implementation of inherent safety principles in a reactor design is to reach such a safety level that there would be no combinations of transient initiating events that could violate the integrity of NPP safety barriers. In this case the initiating events could be both internal (any NPP system failure) and external ones (resulted from specific to NPP site impacts of natural phenomenon or human activity) including terrorism. According to [2] «a set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences..». Thus a need arises to simulate such events and their consequences, to design additional engineer features and to provide their design-basis justification. That leads to increased timing budgets, physical resources and money supply required. The enhancing of inherent safety in advanced reactors designs makes possible to eliminate any beyond basis events on the early stage of conceptual design of reactor. Hence it would be not necessary to provide any additional safety management, to take any engineering measures with respect to beyond basis events on the stage of NPP operating.

2. ULOF Transient Analysis

The results of the inherent safety analysis presented in this chapter are obtained by using the one-dimensional DYANA code [3-6] for inherent safety analysis of fast liquid metal cooled reactors. To verify the code within the range of temperatures and flow rates to be taken at that study the comparative analysis of DYANA output data with PHENIX benchmark end-of-life test was carried out. Estimated sodium temperature and mass flow obtained from protected LOHS+LOF analysis via DYANA code [6] were in reasonable agreement with those obtained from PHENIX benchmark end-of-life tests [7].

Under station blackout or loss of pump power supply conditions there is gradual reduction in primary sodium flow rate takes place due to pump trip. Pumps run-down time under all primary pumps trip conditions is sufficient to hold the temperature of the reactor components (fuel, cladding, primary sodium, reactor vessel, for LWR there are critical power ratio and steam-zirconium reaction margin) within acceptable limits and to remove reactor decay heat if reactor is scrammed. Besides, reactor design also includes auxiliary power supplies (batteries, diesel generators etc) providing primary pumps operating in the mode of decay heat removing (generally low flow rates about 5-6% of nominal value [8]). Natural circulation of primary sodium is developing if all power supplies failed. The most dangerous initiating event is pump seizure. Even single pump seizure may cause the excursion of acceptable temperature of fuel cladding.

ULOF transient dynamics depends not only on combination of initiating events but on the initial state conditions and parameters like initial power right before the accident, control rod position etc.

The results of ULOF transient analysis for fast sodium cooled reactor [9,10] are presented in this chapter. MOX-fueled reactor 2800 MW of thermal power is under investigation. Seizure of all primary pumps (case A) and total loss of primary pumps power supplies (case B) are considered separately. Both accidents are under «without scram» and PDHRS system failure conditions. All secondary systems are normally operating. Initial state provides 100% nominal power.

Firstly, it is necessary to explore the conditions leading to violating of integrity of NPP safety barriers. There are 5 physical safety barriers for reactor under investigation (fuel matrix, steel cladding, reactor vessel and primary devices vessels, secondary devices vessels, containment).

The safety barriers are termed through the acceptable temperatures (temperature criteria) of reactor critical components: fuel, cladding, primary sodium and reactor vessel. The values of all used temperature criteria are listed in TABLE I. The melting point is chosen for fuel temperature criteria because fuel melting causes increasing in gaseous fission products pressure under the cladding, besides the phase change gains the fuel swelling process. All these factors may cause the fuel cladding break. The value of cladding criteria is defined by the stress limits of fuel pin under the specific reactor conditions. The boiling temperature is set as the primary sodium criteria, because the sodium boiling can produce a high positive reactivity excursion due to the sodium void effect (if it is not negative). Finally the stress strain behavior of reactor vessel under the specific reactor conditions defines its temperature criteria.
FIG. 1 and *2* illustrate the core components temperature excursion under cases A and B conditions accordingly. Case A is observed to be the most dangerous one (*FIG. 1*) followed by cladding temperature unacceptable excursion on 8 s. and sodium boiling onset on 15 s. of transient. Case B also initiates unacceptable cladding temperature excursion on 9 s. and sodium boiling onset on 15 s. of transient. This is very specific for ULOF transient to produce exceeding of cladding temperature criteria.
Next consider how do the values of some important reactor parameters impact on the peak temperature of the reactor components in term of the inherent safety. The reducing of primary circuit pressure drop leads to the mitigation of ULOF transient. Temperature excursions are run down the acceptable values if primary circuit pressure drop reached $0{,}804\Delta P_0$ and $0{,}9\Delta P_0$ under cases A and B conditions accordingly. To maintain the integrity of NPP safety barriers in terms of inherent safety of the reactor it requires the peak temperatures of the reactor components to stay within acceptable safety limits even under the most severe and low-probability conditions. In this work the most severe accident scenario is under the case A conditions. As the primary circuit pressure drop is reduced to $0.804\Delta P_0$ the reactor meets the requirement of safety barrier integrity under both case A and B conditions.

TABLE I. TEMPERATURE CRITERIA (MAXIMUM ACCEPTABLE TEMPERATURES OF REACTOR CRITICAL COMPONENTS).

Component	**Temperature criteria, К**
Coolant	1153
Cladding	1073
Fuel	3023
Reactor vessel	1023

The safety barriers integrity conditions can be defined through the criteria characterizing the ULOF transient dynamics. The authors recommend such criteria as F_{NC} (steady state natural circulation level), ΔF (coolant flow rate drop under the steady state natural circulation level) and $\Delta\tau$ (the time it takes for flow rate to reach the steady state natural circulation level) (*FIG.* 2). Now to mitigate the ULOF transient the requirements to F_{NC}, ΔF and $\Delta\tau$ should be worked out.
The normalized primary sodium flow rate under case A conditions with reduced to $0{,}804\Delta P_0$ pressure drop is presented in *FIG.* 2. Referring to the *FIG.* 2 recommended criteria values are shown in TABLE II. The peak cladding temperature is within acceptable limits, primary sodium boiling margin is 90º if the requirements to the pressure drop and criteria are met.
The increasing of primary pump run-down time leads to the mitigation of ULOF transient under case B conditions. Temperature excursions are run down the acceptable values if run-

down time reached the value of 1,8τ_0. Only case B is considered because the increasing of primary pump run-down time under case A conditions obviously is not effective.

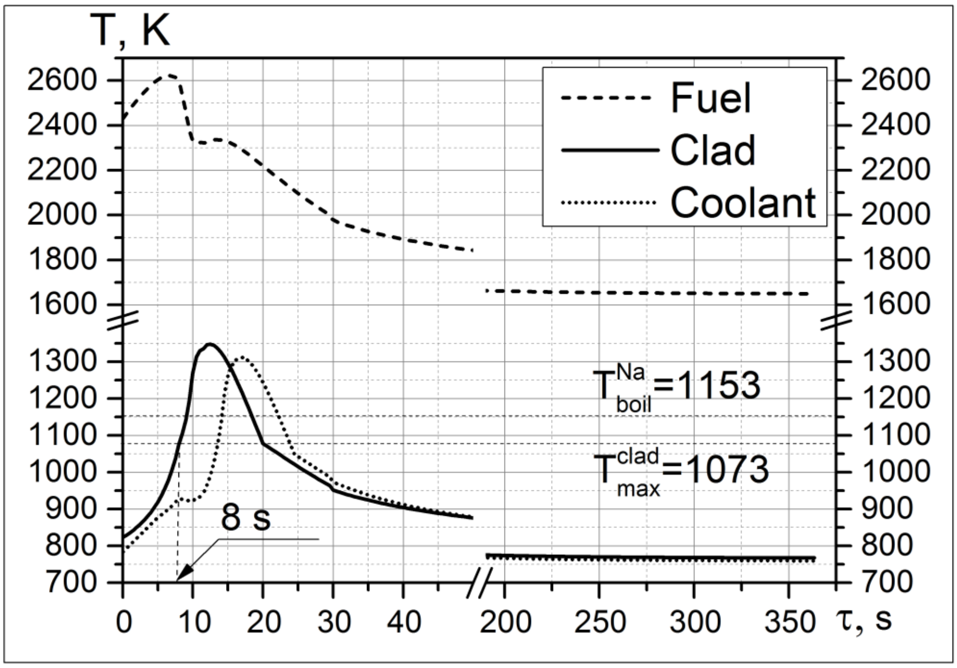

FIG. 1. Peak temperatures of fuel, cladding and primary sodium under case A conditions.

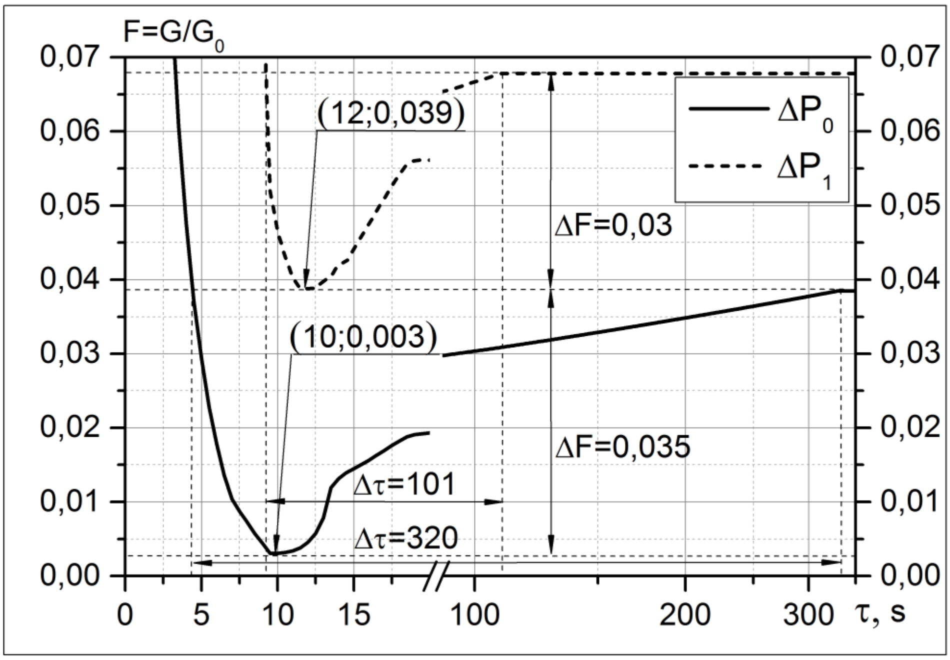

FIG. 2. Primary sodium flow rate under case A conditions with reduced primary circuit pressure drop.

TABLE II. REQUIREMENTS TO THE CRITERIA CHARACTERIZING THE ULOF TRANSIENT DYNAMICS UNDER CASE A AND B CONDITIONS.

Criteria	**Case B Restrictions**	**Case A Restrictions**
$\Delta\tau$	≤101 s.	≤ 101 s.
ΔF	≤ 1.5% nominal flow rates	≤ 3% nominal flow rates
F_{NC}	≥ 5.3 nominal flow rate	≥ 7% nominal flow rates

The normalized primary sodium flow rate under case B conditions with increased to $1,8\tau_0$ run-down time and other important illustrations are presented in [11]. The recommended criteria values are shown in TABLE II. The peak cladding temperature is within acceptable limits, primary sodium boiling margin is 65º if the requirements to the run-down time and criteria are met.
It should be noticed that integrity of NPP safety barriers is achieved under all investigated transient initiating events if the requirements to the criteria characterizing the ULOF transient dynamics are met.
To develop the refined requirements for the proposed criteria it is necessary to couple the SFR performance analysis with uncertainty analysis. It is also necessary to take into account the heat removal through DRACS even in a failure mode by heat-conductivity through the HX walls and to refine the acceptable temperatures of critical components of reactor with respect to reactor inherent safety.

3. ATWS Analysis

The inherent safety performance analysis of the same BN-type reactor during the sets of simultaneous or subsequent faults with differing velocity (TABLE III) results in a set of diagrams (*FIG*. 3, 4) that illustrate the value of reactivity required to be inserted to maintain the integrity of the safety barriers performance in all possible cases within the accepted range of conditions ($\Delta\rho_{ext}$, ΔT_{in}, $\delta(\frac{G}{G_0})$).

TABLE III. ATWS CASES INVESTIGATED FOR BN-TYPE REACTOR.

<table>
<tr><th rowspan="2">Case</th><th colspan="3">ULOF Perturbation</th><th colspan="3">ULOHS Perturbation</th><th colspan="3">UTOP Perturbation</th></tr>
<tr><th>Start, τ_0, s.</th><th>Rise, $\Delta\tau_0$, s.</th><th>$\delta(\frac{G}{G_0})$.</th><th>Start, τ_0, s.</th><th>Rise, $\Delta\tau_0$, s.</th><th>ΔT_{in} , deg.</th><th>Start, τ_0, s.</th><th>Rise, $\Delta\tau_0$, s.</th><th>$\Delta\rho_{ext}$, β_{eff}.</th></tr>
<tr><td>P1</td><td>0</td><td rowspan="12">10</td><td rowspan="12">[-1;0]</td><td>0</td><td>15</td><td rowspan="12">[0;200]</td><td rowspan="12">0</td><td rowspan="12">10</td><td rowspan="12">1,4</td></tr>
<tr><td>P2</td><td>0</td><td>0</td><td>10</td></tr>
<tr><td>P3</td><td>0</td><td>0</td><td>5</td></tr>
<tr><td>P4</td><td>0</td><td>5</td><td>10</td></tr>
<tr><td>P5</td><td>0</td><td>10</td><td>15</td></tr>
<tr><td>P6</td><td>0</td><td>10</td><td>10</td></tr>
<tr><td>P7</td><td>0</td><td>10</td><td>5</td></tr>
<tr><td>P8</td><td>0</td><td>15</td><td>10</td></tr>
<tr><td>P9</td><td>5</td><td>0</td><td>10</td></tr>
<tr><td>P10</td><td>10</td><td>0</td><td>10</td></tr>
<tr><td>P11</td><td>15</td><td>0</td><td>10</td></tr>
<tr><td>P12</td><td>5</td><td>10</td><td>10</td></tr>
</table>

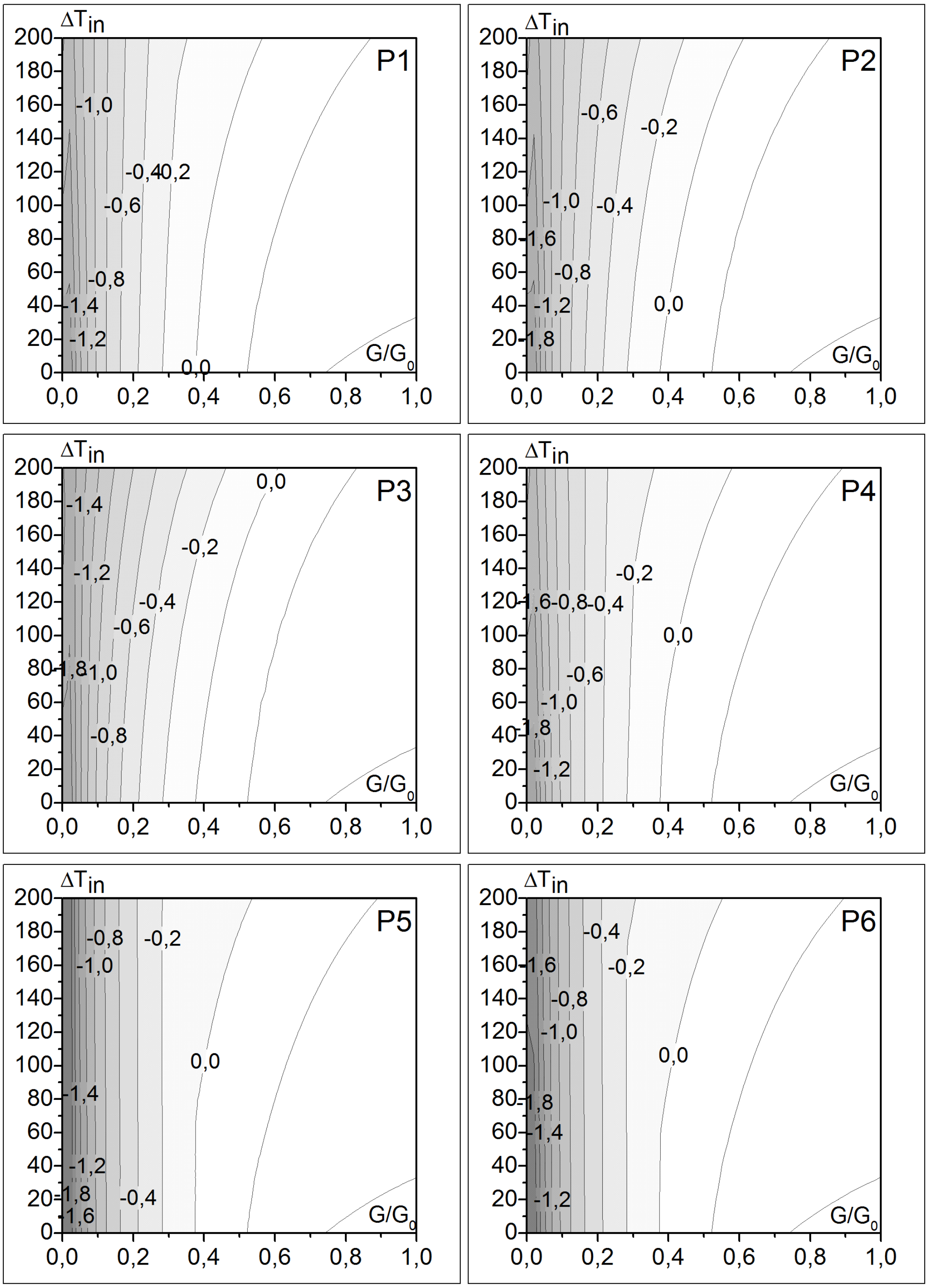

FIG. 3. ATWS P1-P6 (value of reactivity required to be inserted to maintain the integrity of the safety barriers).

LOF delayed start and moderation (cases P9-P12, *FIG. 4*) generally lead to increasing of the reactor inherent safety level what is being explained by the fact that positive reactivity is appeared to be partly compensated due to the reactivity feedbacks under sustainable core cooling when LOF and/or LOHS are still not initiated. LOHS delayed start and moderation (cases P1, P4-P8, *FIG. 3, 4*) also generally lead to increasing of the reactor inherent safety level though the reactor inherent safety performance gets worse under the high values of primary flow rate perturbations G/G_0~0.02 what is being explained by differing velocities of excess reactivity compensation and change of power due to the reactivity feedbacks. The most severe set of accidents is the case P2 – simultaneous initiating of LOF, TOP and LOHS accidents without scram (ΔT_{in}=0, total loss of flow and 1,4 β_{eff} reactivity insertion in particular, *FIG. 3*). The absence of LOHS (ΔT_{in}=0) means the zero-value perturbation of inlet sodium temperature hence moderates sodium temperature rise that in turn retains the response of thermo-mechanical feedbacks which make a major contribution to the negative reactivity component and depend on inlet sodium temperature (grid plate expansion etc).

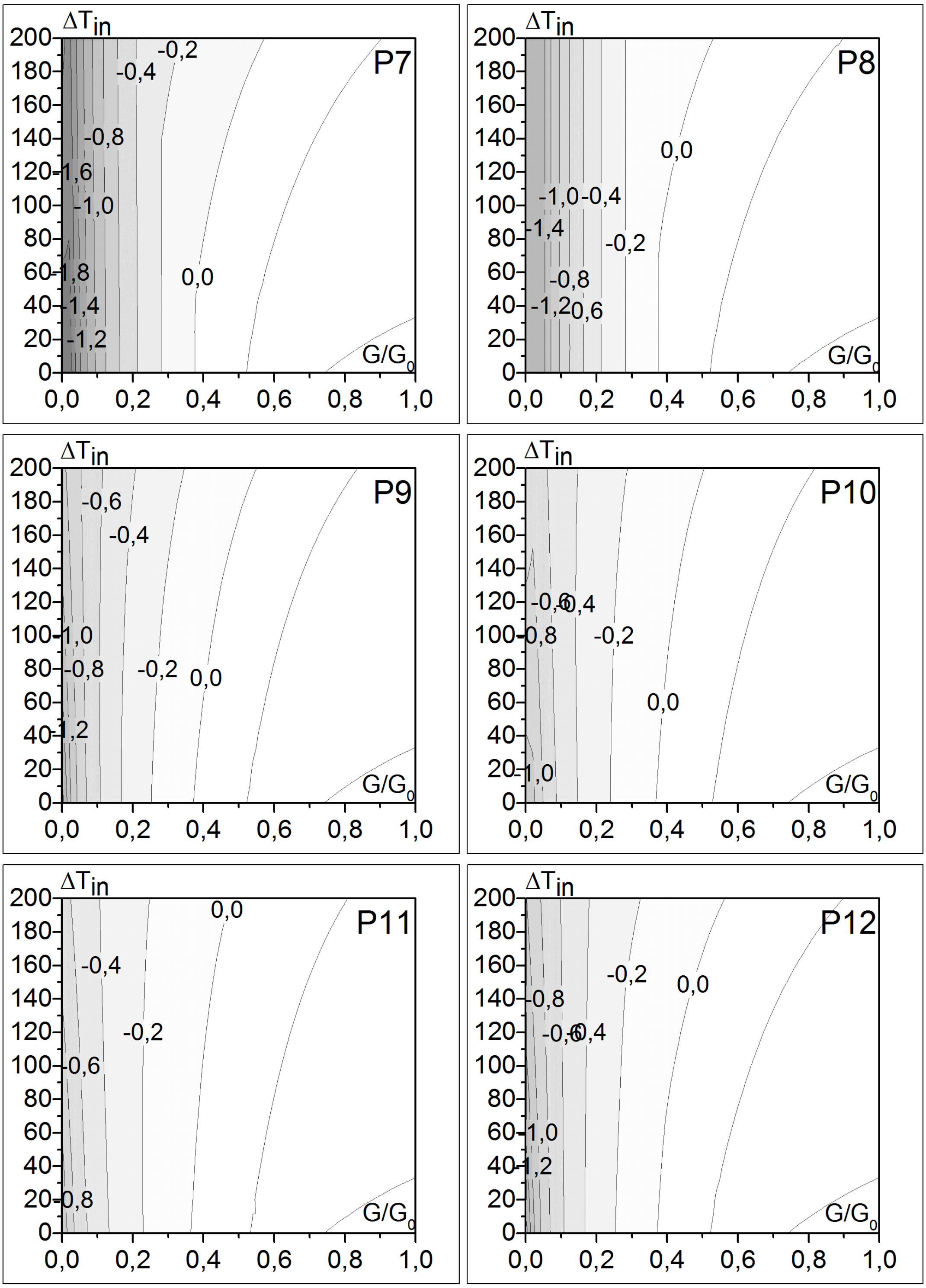

FIG. 4. ATWS P7-P12 (value of reactivity required to be inserted to maintain the integrity of the safety barriers).

The equally hard set of accidents is case P6 - simultaneous initiating of LOF, TOP accidents with delayed LOHS (*FIG. 3*). In particular the partial LOF (G/G_0~0.02, *FIG. 3, 4*), requires a special attention since it may cause more severe consequences than the total LOF what is possible e.g. if the single MCP stayed on low speed operation level or if the pony-motor was enabled. The low head of operating pump destroys the natural circulation head which could be higher if it would be a total LOF. Thereby LOHS delayed start makes the consequences harder then under the case P2 for the forgoing reasons.
Based on draw conclusions the transient analysis of the certain accidents (TABLE IV) was carried out. The choice of those particular accidents is dictated by observing the ratable values of the excess reactivity$\sim 2\beta_{эфф}$ and by the essentially differing dynamics.

TABLE IV. CERTAIN ARWS FOR BN-TYPE REACTOR INHERENT SAFETY ANALYSIS (ALL UNPROTECTED).

<table>
<tr><th rowspan="2">Case</th><th colspan="3">ULOF</th><th colspan="3">ULOHS</th><th colspan="3">UTOP</th></tr>
<tr><th>Start time, τ_0, s.</th><th>Initiating event</th><th>Perturbation value, $\delta(\frac{G}{G_0})$, rel.un.</th><th>Start time, τ_0, s.</th><th>Perturbation rise time, $\Delta\tau_0$, s.</th><th>Perturbation value, ΔT_{in}, deg.</th><th>Start time, τ_0, s.</th><th>Perturbation rise time, $\Delta\tau_0$, s.</th><th>Perturbation value, $\Delta\rho_{ext}$, β_{eff}.</th></tr>
<tr><td>A1</td><td>0</td><td rowspan="3">Coast-down</td><td>-1</td><td colspan="3">Secondary and watersteam circuits are under normal operation, $\Delta T_{in} = 0$</td><td rowspan="3">0</td><td rowspan="3">10</td><td rowspan="3">1.4</td></tr>
<tr><td>A2</td><td>0</td><td>-0.98</td><td>0</td><td>5</td><td>+90</td></tr>
<tr><td>A3</td><td>0</td><td>-1</td><td>10</td><td>15</td><td>+100</td></tr>
</table>

4. Transient Analysis of the Most Severe Cases

The results (TABLE V, *FIG. 5*) of A1-A3 transient analysis show that the peak cladding temperature exceeds its safety criteria (followed by consequent fuel, coolant and reactor tank temperature criteria excess) very rapidly (3.7 s. since the accident onset). Moreover the accident perturbations didn't even come up to its anticipated maximum values. Since the excess of any criteria was considered unallowable the further accident evolution is not considered. This assumption makes considered accidents almost equal in early dynamics (first 4 s.). Hence it can be concluded that the LOHS delaying is equal LOHS absence (A1). Even simultaneous LOHS and LOF doesn't effect on consequences (A2) because the thermo-mechanical feedbacks is being late to compensate the excess. Thus all the A1-A3 cases could be narrowed down single A1 case. But the MCP stacking may cause an essential differing in dynamics as against MCP trip so it must be included as a sub case of A1 in further research.
It is also necessary to consider in depth the temperature excursions of the reactor critical components and the flow rate reduction under complete ULOF followed by forced circulation inset on the level of 2% of nominal flow rate (TABLE VI).

TABLE V. RESULTS OF THE INHERENT SAFETY ANALYSIS OF THE CERTAIN ATWS.

Case	**Exceeded temperature criteria**	**Time to exceeding, $\Delta\tau_{ex}$, c.**	**Uncompensated reactivity, $\Delta\rho_{ext}$, $\beta_{эфф}$**	**Normalized flow rate at $\Delta\tau_{ex}$, $\frac{G}{G_0}$, rel.un.**	**Inlet sodium temperature, T_{in}, K**

A1	Cladding	3.7	$0.29\beta_{eff}$	0.63	683
A2		3.4	$0.28\beta_{eff}$	0.66	746
A3		3.7	$0.29\beta_{eff}$	0.63	683

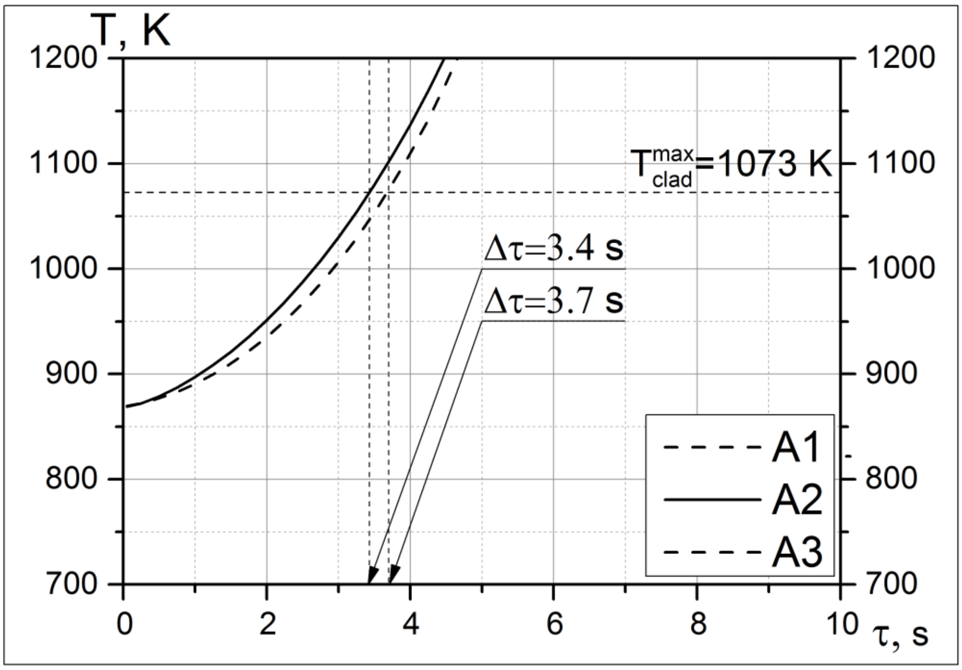

FIG. 5. Peak cladding temperatures under cases A1-A3.

TABLE VI. PARTIAL LOSS OF FLOW ACCIDENTS WITHOUT SCRAM FOR THE TRANSIENT PERFORMANCE ANALYSIS OF BN-TYPE REACTOR.

Case	**Time MCP coastdown begins, s.**	**Time forced ($\frac{G}{G_0} = 0.02$) circulation is initiated, s.**	**Notes**
A4a	0	2	$\Delta T_{in} = 0$; $\Delta \rho_{ext} = 0$.
A4b		5	
A4c		10	
A4d		-	

5. Conclusions

Transient performance analysis for pool-type MOX-fueled sodium fast reactor 2800 MW of thermal power is carried out. The impact of various features on SFR inherent safety performance for ULOF events was also analyzed. The decrease in primary pressure drop and increase in primary pump run-down time were investigated. Performing analysis resulted in a set of recommendations to varying parameters in terms of enhancing the inherent safety performance of SFR under investigation. In order to exclude the safety barrier rupture for ULOF events the set of thermal hydraulic criteria characterizing the ULOF transient dynamics and requirements to them were recommended based on achieved results: F_{NC} (steady state natural circulation level), ΔF (coolant flow rate drop under the steady state natural circulation level) and Δτ (the time it takes for flow rate to reach the steady state natural circulation level).

The inherent safety performance analysis of the same reactor during the sets of simultaneous or subsequent faults with differing velocity was carried out. The most severe sets of accidents were the ones followed by simultaneous initiating of LOF, TOP and simultaneous or delayed LOHS accidents, all unprotected. In particular the unprotected accidents initiated by ΔT_{in}=0 (or hardly delayed), total or partial (G/G_0~0.02) loss of flow and 1,4 β_{eff} reactivity insertion were considered in depth. To develop the refined requirements for the proposed criteria it is necessary to couple SFR performance analysis with uncertainty analysis. It is also necessary to take into account the heat removal through DRACS even in a failure mode by heat-conductivity through the HX walls and to refine the acceptable temperatures of critical components of reactor with respect to reactor inherent safety. It is also necessary to consider in depth the transient performance of reactor under complete ULOF followed by forced circulation inset on the level of 2% of nominal flow rate.

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