

Effect of Spatial Power Distribution during Reactivity Initiated Accidents in a PWR

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This study was undertaken to analyze spatial power effects on the course of reactivity initiated accidents in a PWR. Peak fuel enthalpy, considered as a key parameter in reactor safety analysis, depends on a spatial distribution of the energy deposition; thus, a comprehensive safety analysis should be performed using a 3D detailed dynamic calculation.

Analyses of spatial effects during the accidents were carried out with the BARS-RELAP dynamics code, based on a 3D pin-by-pin neutronics with assembly-by-assembly thermal-hydraulics and intended for modeling of a wide range of transients in various types of reactors. A PWR core model of the Three Mile Island Unit 1 was used to calculate a series of accidents with ejection of different control rods of the same worth and boron dilution events with asymmetrical reactivity insertion.

The BARS-RELAP calculations of the transients showed that local peak fuel enthalpy practically did not depend on power deformations in the core. To understand the role of spatial power effects on the major parameters of reactivity accidents, a simple approximation, based on the Nordheim-Fuchs model was used.

KEYWORDS: *spatial power effects, reactivity initiated accidents, PWR, local fuel enthalpy, power peaking factor, BARS-RELAP dynamics code, 3D pin-by-pin neutronics, heterogeneous method, Nordheim-Fuchs model*

1. Introduction

It is well-known that spatial effects play an important role in the consequences of reactivity initiated accidents (RIAs) in PWRs. From the point of view of reactor safety analysis, peak fuel enthalpy is a key parameter considered as the acceptance criterion for unacceptable fuel damage during RIAs. Obviously, its value depends on a spatial distribution of the core energy deposition and, thus, a comprehensive safety analysis should be performed using 3D detailed calculations.

However, a point kinetics (or 1D) model using core-wide coefficients having very large conservatism has been historically employed for safety analysis purposes. Besides, to take into account spatial effects, power peaking factors are used as additional coefficients in calculating peak fuel enthalpy. Despite the development of modern 3D detailed simulation codes, such highly conservative assumptions are still being applied in LWR licensing.

This study was carried out to understand the role of spatial effects on the consequences of PWR RIAs, especially for local peak fuel enthalpy, and to reduce unnecessary conservatism in further safety analysis. The work has focused on a 3D detailed simulation of the neutronics and thermal-hydraulics during rod ejection or boron dilution accidents in PWRs.

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It is well-established that spatial effects in the PWR core are revealed in a dual manner. On the one hand, the higher the core power peaking, the higher the local energy deposition. In some PWR RIAs, such as, for instance, a rod ejection accident (REA), energy deposition can be approximately related to fuel enthalpy under the assumption of an adiabatic process, (i.e. no heat transfer to the coolant); thus, the higher the core power peaking, the higher the local fuel enthalpy increase. On the other hand, the Doppler feedback is stronger in the hottest fuel regions and reduces the peak power and reactivity. Besides, local heating of the coolant also decreases the peak power and reactivity. Therefore, to provide a more detailed understanding and realistic modeling of PWR accident course it is necessary to use a 3D code with detailed pin power calculation capability.

In this study we used the BARS-RELAP code that allows a 3D pin-by-pin neutronics and assembly-by-assembly thermal-hydraulics simulation of a wide range of transients in various types of reactors. [1,2] This code used in Russian Research Centre “Kurchatov Institute” for analysis of LWR RIA consequences, includes the BARS neutron kinetics code, based on a 3D pin-by-pin heterogeneous method and validated against numerous LWR UO₂ and MOX fuel benchmarks. [2] The thermal-hydraulic code RELAP5/MOD3.2 (or /MOD3.3) is a “best-estimate” system transient code, which allows a multi-channel (assembly-by-assembly) model to describe the core thermal-hydraulics with direct representation of few “hot” fuel rods as individual thermal structures. [3] To enhance capabilities of the coupled BARS-RELAP code, a pin-by-pin fuel temperature reconstruction model has been developed and incorporated. [4]

In this work, two PWR models based on Three Mile Island Unit 1 (TMI-1) at end-of-cycle (BOC) and beginning-of-cycle (EOC) conditions was used. [5] Both models were used for a REA modeling when different control rods of the same worth of about 1.2 β were ejected. The latter was used to model the boron dilution event.

Section 2 focuses on basic features of the heterogeneous method used in the BARS neutron kinetics code. A brief description of the BARS-RELAP coupled model for LWR RIA analysis and the TMI-1 PWR core model are given in Section 3. Next section presents the calculation results of the REAs with an approximate analysis based on the Nordheim-Fuchs adiabatic model. Section 5 presents the calculation results of the boron dilution events with asymmetric reactivity insertion.

2. Basic Features of the Heterogeneous Method

An advanced method of the heterogeneous reactor theory, implemented in the BARS code, is a unique tool for LWR comprehensive applications including reactor dynamics modeling. Historically well-known as the Galanin-Feinberg method, this theory was being developed and applied to calculate channel-type heavy water or graphite reactors. During past several years its capabilities have been greatly extended mainly for LWR analysis including MOX fuel cores. [6] Let us consider some basic features of the advanced heterogeneous method.

Instead of traditional neutron cross sections, heterogeneous equations use matrices of cell boundary conditions (so-called Λ -matrices), which establish the relationship between neutron fluxes and currents at cell boundaries. Besides, "time absorption" Λ -matrix components and activation matrices establishing the relationship between neutron flux and different reaction rates, are prepared. Any Λ -matrix is determined from a set of neutron transport calculations of a cell with varying neutron currents at its boundary. Unlike traditional whole fuel assembly calculations, this method allows for anisotropic neutron diffusion effects (i.e. the difference in radial and axial neutron migration) by using dipolar and axial components of Λ -matrix. Such a feature may be greatly important in some LWR transient analysis when the anisotropic effects become essential, for example, if local voiding occurs within a fuel assembly.

Another important aspect is a correct representation of the core-reflector interface where the diffusion approach is not valid and needs significant corrections. These corrections derived from a comparison between a diffusion and transport solutions of a sample steady state problem (as a rule, in 1D geometry), generally, do not guarantee accurate solution near the core-reflector interface. On the contrary, the heterogeneous method needs no corrections.

The heterogeneous theory uses an analytical representation of the neutron flux distribution as a superposition of Green's functions. In LWR applications this method allows for directly complicated core geometry by explicit representation of fuel pins, absorber rods, tubes, etc. Unlike standard diffusion approaches used the finite-difference or nodal approximation of the neutron flux, the heterogeneous theory uses a difference representation of Green's functions. Such an approach has higher accuracy compared with neutron flux approximations.

Axial representation of the neutron flux is described using Fourier series expansion, which gives faster convergence than the traditional finite difference approximation, especially for integral parameters, such as fuel assembly power. This axial representation allows accurate modeling of axial changes in the core, for example, control rod movement.

Another advantage of the heterogeneous theory deals with the possibility of using few energy groups. The BARS validation calculations of LWR systems including the cores with MOX fuel, showed that it is enough to use no more than 4-5 energy groups to provide rather high accuracy in pin-by-pin design calculations of LWRs.

The BARS neutron database (Λ -matrices) is prepared by the UNK code, which solves the multigroup neutron transport equation for various reactor cells using the collision probability method. [7] The code allows for detailed structure of resonant cross sections by using a fine energy mesh with optimal mesh condensing near the resonance. Totally about 10,000 groups can be used. The UNK neutron database is generated on the basis of ENDF/B-VI.

3. Calculation Methodology

3.1 BARS-RELAP5 Coupled Model

An important feature of the LWR pin-by-pin analysis with the BARS-RELAP code is a relatively large running time for a single calculation by BARS compared with RELAP5. The BARS code allows choosing time steps depending only on relative changes in neutronic and thermal-hydraulic (TH) parameters, but its absolute values can significantly change. Usually, the BARS time step much larger than the RELAP5 one.

To provide the data exchange between RELAP5 and BARS, the interface code having its own simplified thermal-hydraulics model, is used. This provides RELAP5 with all necessary data obtained from BARS. Besides, this code considers additional TH parameters including pin-by-pin fuel temperature reconstruction capability.

3.2 TMI-1 PWR Core Model

The PWR core model was based on TMI-1 used as an international benchmark exercise. [5] The reference design for the core was derived from the reactor geometry and operational data. The reactor model includes 177 15×15 fuel assemblies and 64 reflector assemblies. Each fuel assembly contains fuel pins with the same isotopic composition varying along the core height (totally 24 axial layers). Thus, the core having one-eight symmetry contains 36,816 fuel pins and the total number of calculation nodes in the reactor model is about 1,000,000.

Two cores were used: at EOC and BOC conditions, respectively. The cores differ in burnup (40.7 and 18.2 GWd/t), boron concentration in the coolant (5 and 1700 ppm), and fuel isotope composition. Initial state is the same: a hot zero power (HZP) condition (10^{-6} of rated power, 551 K and 15.4 MPa), safety rod banks 1–4 (safety rods) are withdrawn, regulating banks 5–7

are inserted. Bank 8 contains axial power-shaping rods with a low worth (partially inserted at BOC or completely withdrawn at EOC). Figure 1 shows the arrangement of the control rod banks in one-eighth symmetrical part of the core where H8 is the central fuel assembly.

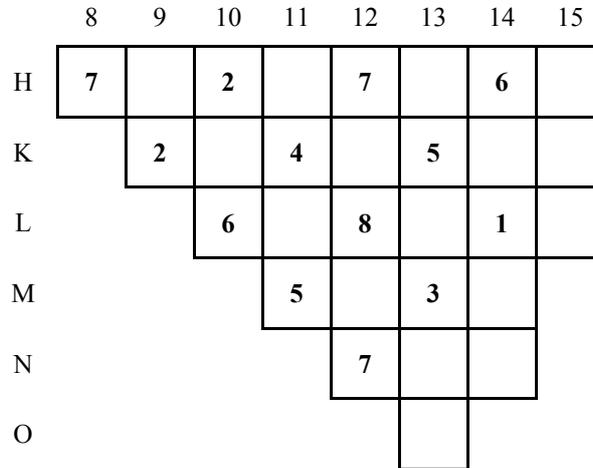


Fig.1 Radial arrangement of control rod banks in TMI-1 PWR core

The RELAP5 thermal-hydraulics model allows for one TH channel per assembly including an additional thermal structure for the “hottest” fuel pin with the same coolant TH parameters as those in an assembly where this pin is located. The bypass reflector assemblies are treated as a single channel. The RELAP5 model uses 24 axial TH nodes; besides, both the inlet and outlet plena and the axial reflectors are represented explicitly.

The value of the delayed neutron fraction was 521.10 pcm at EOC and 632.34 pcm at BOC.

4. Results of Rod Ejection Accident Calculations

A REA transient is initiated by withdrawal of a control rod assembly of Bank 7 either from the core center (H8) or periphery (N12 or H7) in 0.1 s at both BOC and EOC conditions. A 2.5-second transient simulation was considered; no reactor scram was assumed. Because a real worth of any control rod was less than \$1, to study REA transients above prompt critical, the worth of the ejected rod was artificially increased to reach approximately \$1.21.

The following parameters were studied: reactivity, power, energy deposition and peaking factors (pin-by-pin F_q , assembly-average F_q' and axial F_z) in the core; power, temperature and enthalpy including their local values in fuel assemblies. Table 1 gives the major parameters of the REAs. The core energy deposition, peak fuel temperature (T_{max}) and peak enthalpy rise (H_{max}) occur at the end of the transient. The power peaking factor is presented by two figures: the maximum value and the value at the end of the transient.

Power reaches its maximum at about 0.3 s, and then due to negative temperature feedback (the Doppler effect) the power excursion is terminated. Reactivity and total peaking factor have a similar behavior: the maximum is reached just after the rod ejection, followed by a 0.15-second “plateau” with further decrease due to the core heat-up. Approximately 80% of fuel enthalpy rise is produced during 0.2-0.3 s after the power peak. A typical behavior of the core power and the power peaking factor, F_q , during the REA transient is shown in Figure 2.

As seen in Table 1, noticeable differences in the power peak, fuel temperature and enthalpy rise occur. In case with ejection of the peripheral rod the peak power and energy deposition are lower by 2-3 times compared with ejection of the central rod H8. On the other hand, the

same, but inverse relationship occurs for the power peaking factor. Thus, one can conclude that there is a correlation: the higher the power peaking factor, the lower the peak power and energy deposition in the core.

Actually, as the REA calculation results show, the effect of the power peaking factor is revealed in dual manner: a higher value of the peaking factor tends to decrease the total energy deposition and, on the other hand, to increase the local fuel enthalpy.

Table 1 Major core parameters during PWR REAs

	H8 / EOC	H12 / EOC	N12 / EOC	H8 / BOC	N12 / BOC
Rod worth (\$ / pcm)	1.211 / 631	1.211 / 631	1.211 / 631	1.211 / 766	1.211 / 766
Peak power (GW)	11.05	5.80	4.39	37.49	13.07
Pulse width (ms)	61.9	62.8	64.5	39.8	41.4
Energy deposition (GJ)	1.915	1.070	0.849	5.442	2.043
T _{max} in assembly (K)	812.0	776.1	785.6	898.9	898.0
T _{max} in fuel pin (K)	835.3	815.8	851.6	935.9	996.1
H _{max} in assembly (cal/g)	18.88	16.15	16.87	25.52	25.45
H _{max} in fuel pin (cal/g)	20.65	19.16	21.90	28.37	33.04
F _q (max / end)	8.04 / 6.51	14.0 / 11.6	20.1 / 16.2	4.11 / 3.09	12.9 / 9.52
F _q ' (max / end)	7.42 / 6.03	12.0 / 9.90	16.0 / 13.0	3.71 / 2.79	10.2 / 7.51
F _z (max / end)	2.76 / 2.45	2.69 / 2.41	2.71 / 2.44	1.50 / 1.25	1.47 / 1.29

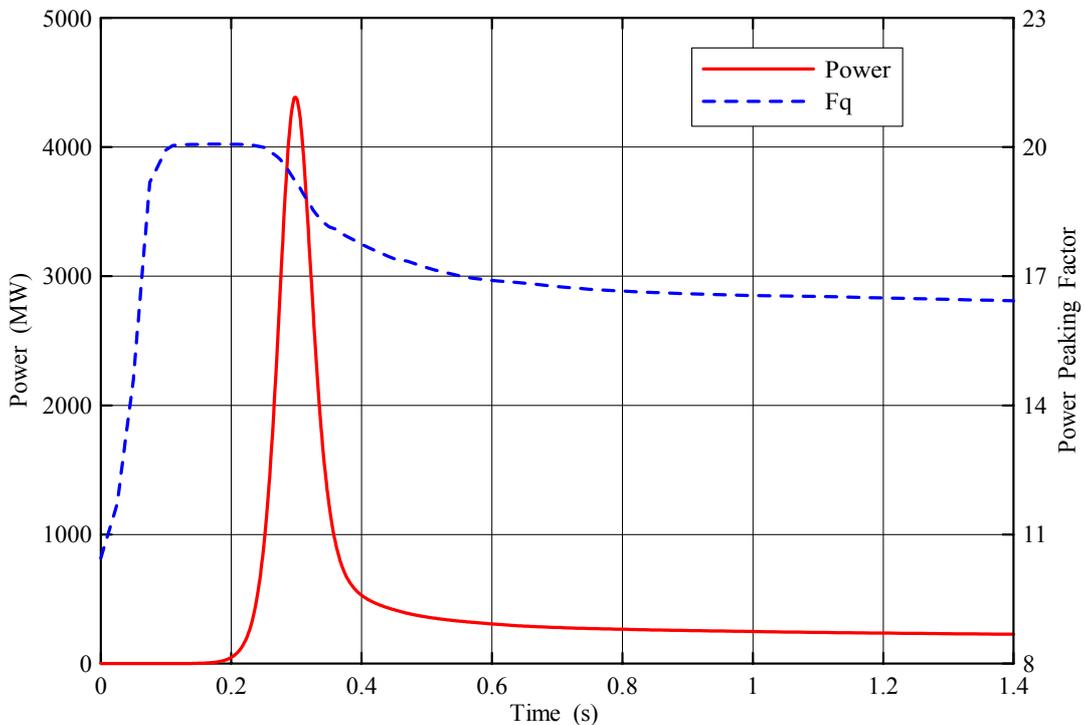


Fig.2 Core power and power peaking factor vs. time for Case N12 / EOC

4.1 Analysis with Nordheim-Fuchs Model

To study the influence of the power peaking factor on major REA parameters it is worth to consider a simple analytical approximation based on the Nordheim-Fuchs model widely used to estimate various parameters in pulsed reactors. [8] The model based on the point-kinetics model, is valid if the inserted reactivity, ρ_o , is above prompt critical (i.e. $\rho_o > \beta$). It is assumed that reactivity change due to negative feedback is proportional to the energy deposition, Q :

$$\rho(t) = \rho_o - \gamma Q(t) \quad (1)$$

According to this model, the maximum power, P_{\max} , and the total energy deposition, Q_{tot} , are given as:

$$P_{\max} = (\rho_o - \beta)^2 / (2\Lambda\gamma) \quad \text{and} \quad Q_{\text{tot}} = 2(\rho_o - \beta) / \gamma \quad (2)$$

Here β , γ , and Λ are the delayed neutron fraction, the reactivity feedback coefficient and the neutron lifetime. Despite simplicity of the model, it is useful for qualitative understanding of spatial effects on the REA parameters. It is obvious that β and Λ are practically independent on the ejected rod position at both EOC and BOC. Thus, the only parameter responsible for spatial effects is γ . Historically, this parameter is considered approximately, independently on any spatial effect due to neutron flux, power or temperature, as:

$$\gamma \approx \gamma_o = -\alpha_D / (C_p V), \quad (4)$$

where α_D , C_p , and V are the Doppler coefficient, the fuel heat capacity, and the fuel volume. Obviously, γ_o is approximately constant; thus, such an approach is not able to explain those revealed differences in P_{\max} and Q_{tot} at both EOC and BOC (see Table 1).

To study above effects, let us estimate the parameter γ from Eq.(1) as $\gamma = -\delta\rho/\delta Q$, where $\delta\rho$ is the reactivity change due to the energy deposition increase, δQ . An adiabatic assumption (i.e. no heat transfer to the coolant) implies that the only feedback is the Doppler effect. Using an approximate expression for both $\delta\rho$ and δQ in terms of the fuel temperature change, ΔT , and the fact that the peaking factors for ΔT and the power practically coincide, one can obtain an analytical estimate of the lower and upper limits of the feedback parameter γ :

$$\gamma_{\min} = \gamma_o \quad \text{and} \quad \gamma_{\max} = \gamma_o F_q \quad (6)$$

Taking into account that the lower limit of γ means the uniform change in fuel temperature in the core (i.e. $\Delta T = \text{const}$ and $F_q = 1$), therefore, one can expect that the linear dependence on F_q can hold out. Figure 3 shows the relative feedback parameter, γ/γ_o , depending on the pin-by-pin and assembly-averaged peaking factors, F_q and F_q' , respectively. At HZP conditions in TMI-1 PWR: $\gamma_o \approx 1.0 \times 10^{-6} \text{ MJ}^{-1}$. From the BARS-RELAP calculation results the values of γ , F_q , and F_q' were averaged over the pulse ‘‘adiabatic’’ region (0.2–0.4 s). As seen in the figure, both dependencies can be approximated by linear functions.

To summarize the above analysis, it is safe to conclude that the feedback parameter, γ , is approximately proportional to the power peaking factor, F_q . In its turn, according to Eq.(2), both the peak power, P_{\max} , and the total energy deposition, Q_{tot} , is inversely proportional to γ , and, therefore, to F_q . Under the adiabatic assumption the fuel enthalpy rise is proportional to Q_{tot} ; on the other hand the peak fuel enthalpy rise is proportional to F_q . Therefore, these two opposite effects are trying to compensate each other to eliminate dependence of the local fuel enthalpy rise on power deformations. This conclusion is confirmed by the REA calculations.

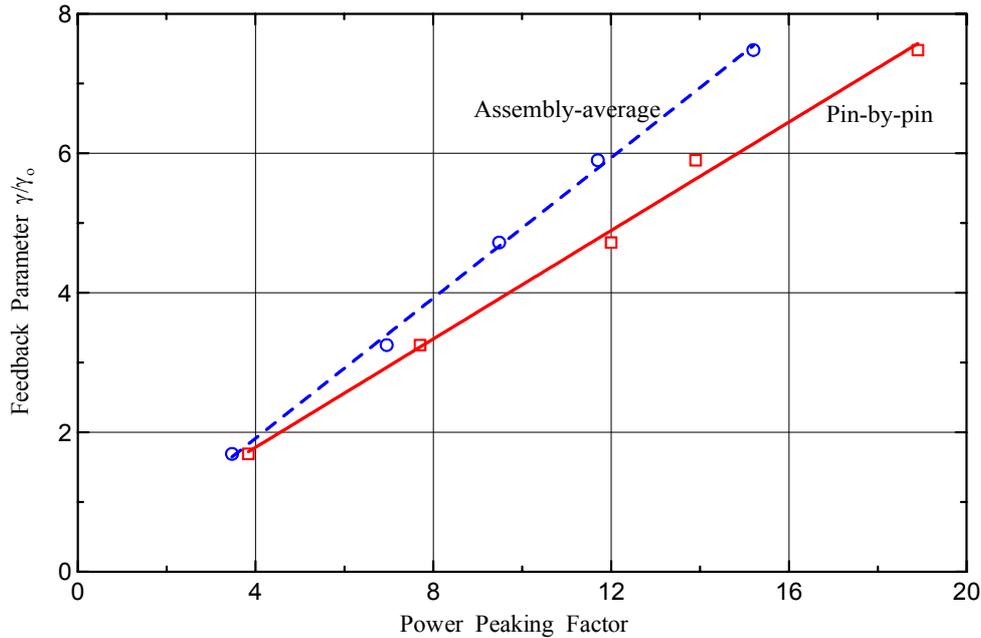


Fig.3 Feedback parameter vs. power peaking factor

5. Results of Boron Dilution Event Calculations

The boron dilution event was studied for the TMI-1 PWR at BOC. This accident follows a small break LOCA when water natural circulation is disrupted and reflux condensation allows deborate to build up in the cold leg. A conservative plenum inlet boron concentration obtained from a simulation performed by Framatome Technologies is used. [9]

A series of the parametric studies were performed with BARS-RELAP varying the spatial distribution of the inlet boron concentration to reach an asymmetrical reactivity insertion. The basic transient model had the uniform radial boron distribution. The core power versus time is shown in Figure 4. The power peaking factor, F_q , reaches its peak value of 10.2 at the time of the peak power; whereas, the local fuel enthalpy reaches its maximum of 74 cal/g ten seconds later just after the second power peak. The minimum value of reactivity was -16.9 before the boron dilution and the peak value reached 1.13 .

Two radially asymmetrical transients were calculated, both with a half-core change in the inlet boron concentration of 10% and 20% (increased at one half and decreased at another). The calculations show a significant pin-by-pin power deformation within the core (Figure 5). The power peaking factor increased by more than two times compared with the basic transient. The asymmetrical slug of deborated water flowing through the core leads to a large increase in the maximum reactivity: by 28% and 43%, respectively. Whereas, the peak fuel enthalpy increased by 25% and 44%, respectively, i.e. practically proportional to the reactivity change. Therefore, the reason for the increase in the peak fuel enthalpy is the reactivity rise, rather than spatial power deformations due to an asymmetrical boron distribution over the core.

Here we observe the same effect as in the REA analysis where the difference in the peak fuel enthalpy rise at BOC and EOC can be related to the difference in the absolute values of the inserted reactivity. Thus, one can generalize a conclusion from abovementioned facts, based on the REA and boron dilution analyses: the peak fuel enthalpy does not practically depend on spatial power effects in the reactor core during an accident.

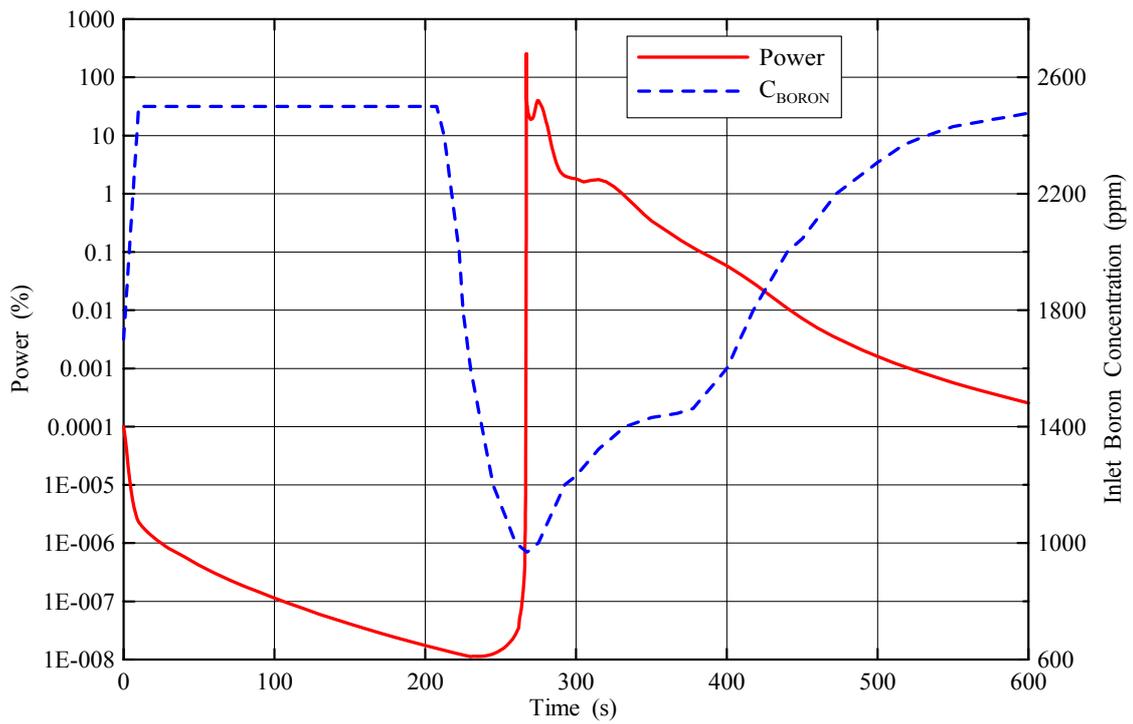


Fig.4 Power history for boron dilution event

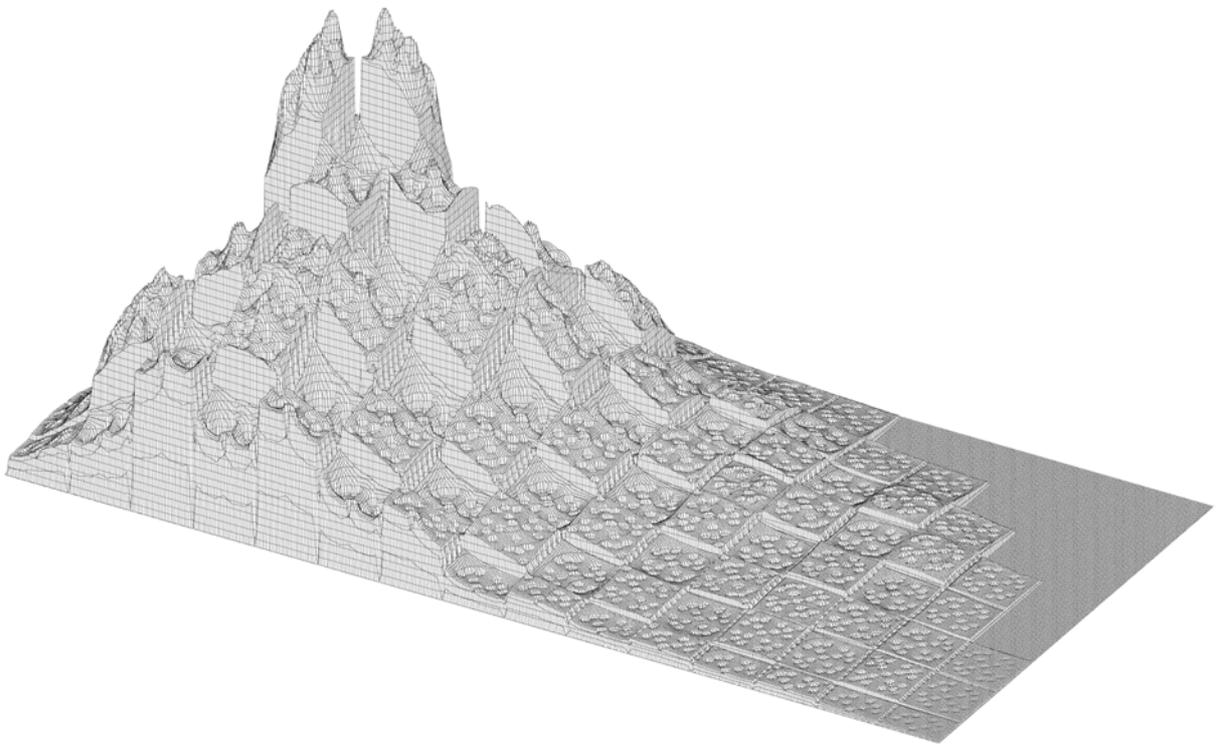


Fig.5 Pin-by-pin power distribution for asymmetric boron dilution event

6. Conclusion

This study was undertaken to analyze effects of spatial power distribution on consequences of PWR RIAs, in particular, the peak local fuel enthalpy. This parameter is still being used in LWR licensing with an unnecessary conservatism historically based on using simple models (1D or point kinetics) and additional coefficients “taking into account” local effects, such as, for instance, power peaking factors. Needless to say that such an approach leads to significant conservative assumptions that can be reduced only using modern 3D detailed dynamic codes with pin-by-pin power calculation capability.

Fuel enthalpy directly depends on a spatial distribution of the energy deposition in the core and this dependence is revealed in two ways. The higher the core power peaking, the higher the local energy deposition and, therefore, the local fuel enthalpy increase. On the other hand, high power peaking tries to reduce the peak power and, as a result, the core energy deposition (the Doppler feedback is stronger in hot fuel regions and due to the local coolant heat-up).

In this work, the BARS-RELAP code was used for a 3D detailed simulation of rod ejection and boron dilution accidents in the TMI-1 PWR. It was found that the peak fuel enthalpy rise practically did not depend on power deformations over the core. To understand these results, the simple approximation, based on the Nordheim-Fuchs model, was derived. Under some assumptions both the peak power and the energy deposition are inversely proportional to the power peaking factor, F_q . In its turn, the fuel enthalpy rise is proportional to F_q . Thus, the total effect due to the spatial power deformations becomes negligible.

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