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UNCERTAINTY ANALYSIS FOR THE PWR ROD EJECTION ACCIDENT USING THE RELAP-BARS CODE

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ABSTRACT

Uncertainty analysis has been carried out for the control rod ejection accident using the BARS 3-D pin-by-pin neutronic code coupled with the RELAP5/MOD3.2 thermal hydraulic code. It was modeled the central control rod ejection in the TMI-1 pressure water reactor (PWR) at hot zero power conditions. The analysis of uncertainties to a number of neutronic and thermal-hydraulic quantities was performed for the following parameters: local fuel enthalpy, maximum core power, and power pulse width. Calculated results showed that the uncertainty in key safety parameters would be determined to a great extent by the uncertainty in the control rod worth. The effect of initial core power on safety parameters was demonstrated on the basis of a calculation for the rod ejection accident starting from 33% of rated power.

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1. INTRODUCTION

The most important safety parameter of reactivity initiated accident is maximum local fuel pellet enthalpy. Now this parameter is used as an acceptance criterion for design-basis reactivity accidents in a light water reactor. 3-D best-estimate neutronics methods are available to calculate local fuel pellet enthalpy; but unlike 1-D or 2-D very conservative methodologies, these methods do not guarantee conservative estimation in key safety parameters during such an accident. Therefore, it is important to determine the uncertainty in fuel enthalpy calculated by a best-estimate code.

Recently a qualitative approach to an uncertainty analysis for the rod ejection accident (REA) was developed in Brookhaven National Laboratory (BNL) (USA) [1]. For the REA, the fact that the physics of the transient is relatively well-known allowed the authors to define a simplified methodology to estimate the uncertainty in fuel enthalpy. The approach is based on using point kinetics in determining the quantities, which determine the uncertainty in fuel enthalpy instead of a very complicated consideration of uncertainties in cross sections.

The approach was applied to the uncertainty analysis of a PWR REA at hot zero power conditions (HZP). The analysis took into account the point kinetics parameters, which were obtained from 3-D calculations and engineering judgement as to the uncertainty in those parameters. Sensitivity study related to them was carried out using the best-estimate code PARCS [2], which is based on an assembly-by-assembly neutron diffusion model. The results showed that the uncertainty in local fuel enthalpy would be determined primarily by the uncertainty in ejected rod worth and delayed neutron fraction. For an uncertainty in the former of 8% (one standard deviation) and the latter of 5%, the uncertainty in fuel enthalpy was 51% for control rod worth of 1.2β (β – delayed neutron precursor fraction). However, the authors considered only a few quantities of interest and their analysis was based on a conservative adiabatic assumption for fuel temperature calculation.

The objective of this study is to analyze the uncertainty in peak fuel enthalpy, core power, and power pulse width for a REA in the TMI-1 PWR at HZP conditions. Sensitivity of these parameters to a variety of neutronic and thermal-hydraulic quantities of the core was studied using the pin-by-pin neutronic model together with more realistic thermal-hydraulic model that are implemented in the RELAP-BARS code [3,4]. Another objective is to analyze the effect of initial core power on key parameters of the accident by a calculation of the TMI-1 REA starting from 33% of rated power.

Section 2 expounds the methodology of the uncertainty analysis for the PWR REA. In Section 3 the uncertainty in the peak fuel enthalpy, core power, and power pulse width is assessed to a number of neutronic and thermal-hydraulic quantities. The effect of the initial core power on parameters of the accident is analyzed in Section 4 by a comparative study of the TMI-1 REA starting from 33% of rated power. The conclusions are drawn concerning the results of the uncertainty analysis.

2. METHODOLOGY

The methodology of the uncertainty analysis is close to that developed in BNL. It is based on a sensitivity study to global quantities that are explicitly used in point kinetics equations or can be taken into account implicitly in point kinetics through thermal-hydraulic feedback. The approach does not require validity for the adiabatic assumption and is based on the non-adiabatic thermal-hydraulic model realized in the RELAP5/MOD3.2 code [3].

Assuming that a safety parameter (y) is a function of a number of above quantities (x) and the random error in each quantity is normally distributed, the square of the uncertainty in the parameter y can be written:

$$(\delta y/y)^2 = \Sigma(S_x)^2(\delta x/x)^2 \quad (1)$$

where $\delta x/x$ is the uncertainty in the quantity x, S_x is the sensitivity of the parameter y to the quantity x, and the summation is over all quantities of interest.

It was studied the uncertainty in the following safety parameters (y):

- peak local fuel enthalpy,
- maximum core power,
- power pulse width.

The sensitivities S_x to the quantities x were obtained from 3-D pin-by-pin calculations for different quantities x using the RELAP-BARS coupled code. The uncertainties in the neutronic and thermal-hydraulic quantities $\delta x/x$ were estimated by engineering judgement, using evidence from available references and validation results for the BARS code.

3. UNCERTAINTY ANALYSIS

The uncertainty analysis in safety parameters was carried out for the TMI-1 PWR with a high burnup core. The reactor of 2772 MW rated power, having one-eight symmetry, contains fuel assemblies with fuel burnup ranged from 23 up to 58 GWd/t (at the end of the cycle) [5] as shown in Figure 1. The REA was defined for the central control rod at HZP with an ejection time of 100 ms [6]. The reference (without scram) transient duration was 2.5 s. Figure 2 shows the core power and the reactivity as a function of time for the reference transient calculated with RELAP-BARS.

The following values of the key parameters were obtained as reference ones:

- ejected rod worth – 1.21β;
- peak power – 386% of rated power;
- time of peak – 338 ms;
- power pulse width – 62.6 ms;
- maximum fuel pellet enthalpy – 37.6 cal/g;
- maximum increase in fuel pellet enthalpy – 20.6 cal/g.

In the reference transient the local fuel enthalpy reaches its maximum value at the end of the transient (2.5 s). To estimate a real time when the fuel enthalpy reaches its maximum, an additional transient with scram was calculated. In this transient it was supposed that reactor scram occurs with 0.45 s delay at 35% of rated power. Control rods movement during the scram was modeled with a speed of 155.8 cm/s [5].

Figure 3 shows the fuel pellet enthalpy increment as a function of time for both reference and additional transients. The peak of 17.4 cal/g in the fuel enthalpy increment occurs at the time of 0.785 s for the transient with scram. The reference transient overestimates the fuel enthalpy by less than 0.5% at that moment.

	8	9	10	11	12	13	14	15
H	52.86 Bank 7	30.19	56.25	30.85	49.53 Bank 7	28.11	53.86 Bank 6	55.78
K		57.94	30.80	55.43	29.83	53.95 Bank 5	25.55	49.17
L			57.57 Bank 6	30.22	54.40	27.86	23.30	47.30
M				49.71 Bank 5	28.85	52.85	40.94	
N					48.75 Bank 7	23.86	41.45	
O						37.34		

52.86	- fuel burnup (GWd/t)
Bank 7	- No. of regulating Bank

Figure 1. One-Eight Core Layout

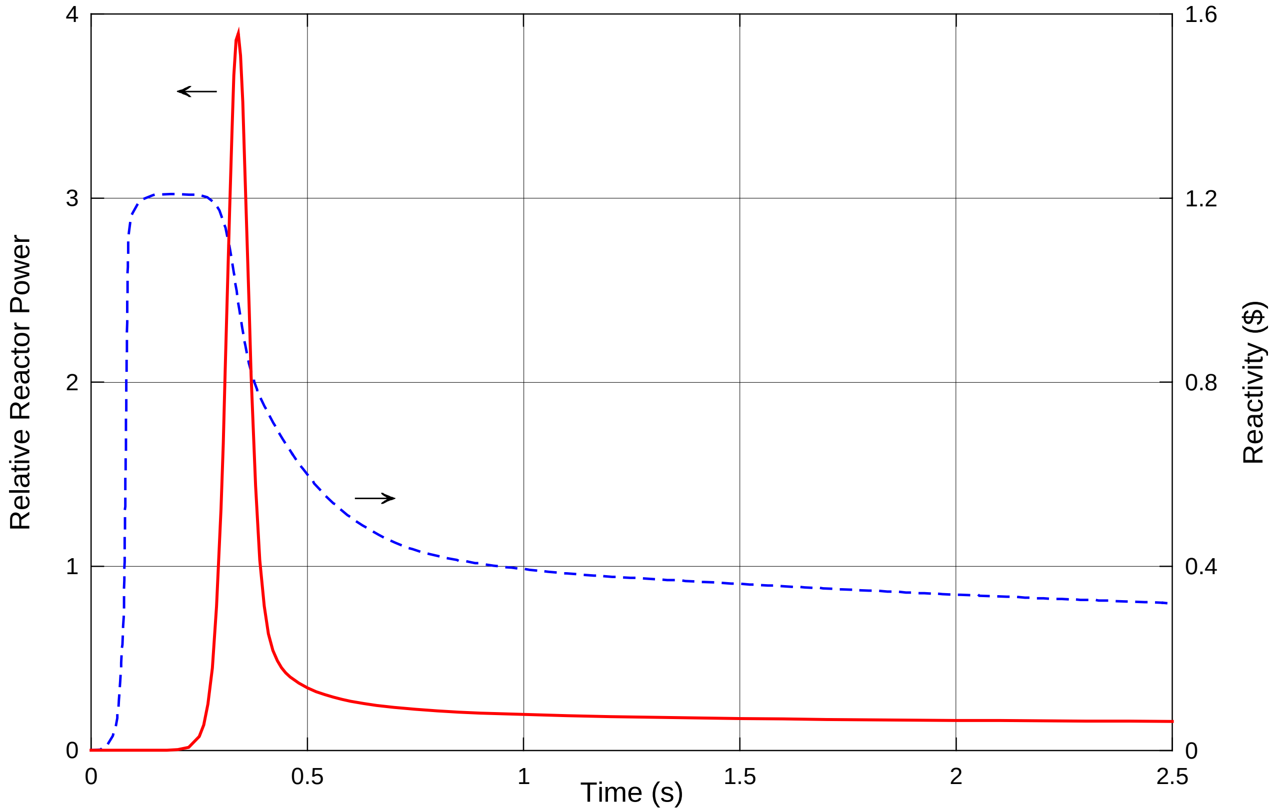

Figure 2. Core Power and Reactivity vs. Time

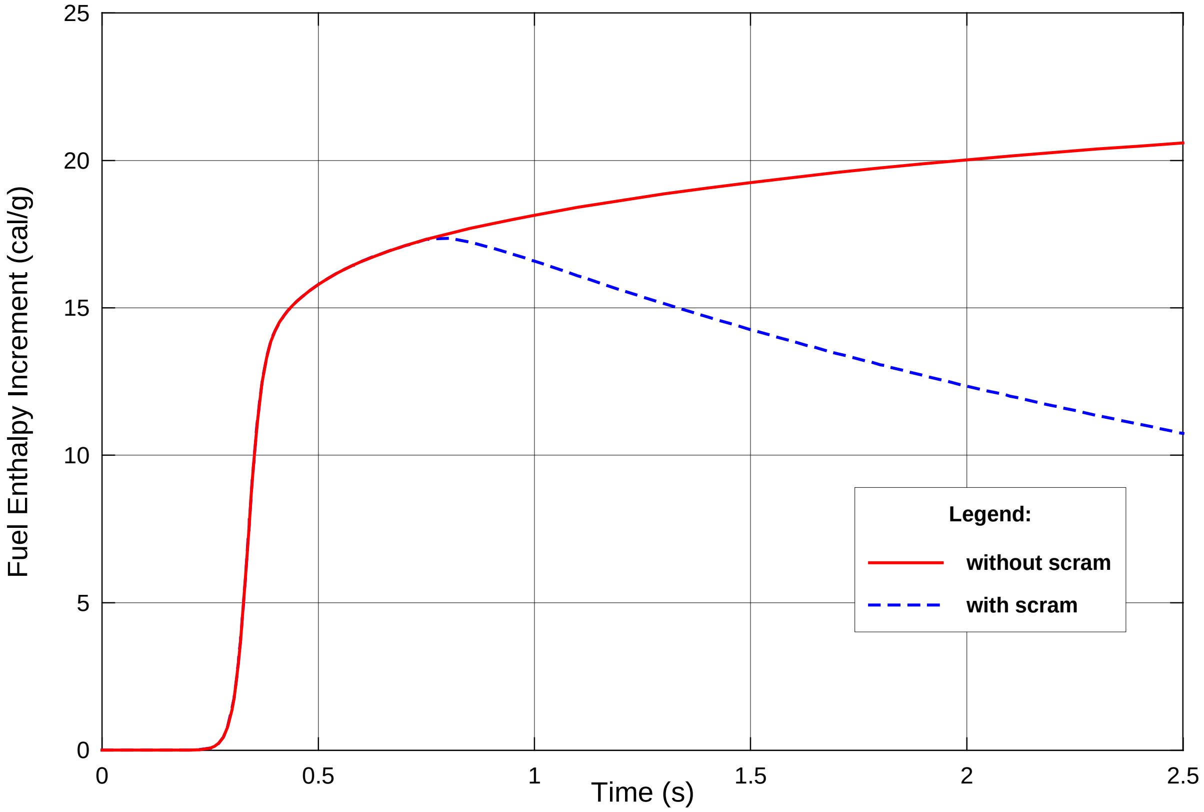

Figure 3. Fuel Pellet Enthalpy Increment vs. Time

The following neutronic and thermal-hydraulic quantities were taken into consideration during the uncertainty analysis:

- reactivity worth of the ejected rod (ρ),
- delayed neutron precursor fraction (β),
- fuel temperature (Doppler) reactivity coefficient (α_d),
- moderator density reactivity coefficient (α_m),
- pellet heat capacity (C_p),
- gap conductance (h_g),
- pellet conductivity (K_f),
- clad-moderator heat transfer coefficient (h_w),
- fraction of energy deposited directly in the moderator (γ),
- radial power peaking factor for the pellet (F_p).

Sensitivity S_x was obtained as a result of corresponding calculation of the transient with perturbed quantities. Different variations of these neutronic and thermal-hydraulic quantities from their reference values were used. The reference value of the delayed neutron precursor fraction was perturbed by –10%; the value of the Doppler reactivity coefficient was changed by –9%; the value of the moderator density reactivity coefficient was increased by 2.1 times. All table data for pellet heat capacity, gap conductance, pellet conductivity, and clad-moderator heat transfer coefficient were increased by 10%. The energy deposited directly in the moderator was not taken into account in the reference transient. To estimate the sensitivity to this quantity, the values of 2% and 5% were considered for the fraction of energy deposited in the moderator. In the reference calculation the radial power distribution in the pellet was assumed as uniform one. To estimate the sensitivity to this distribution, the parabolic power distribution was considered with the peaking factor of 1.05.

The most important quantity is the reactivity worth of ejected control rod. The sensitivity of the maximum fuel enthalpy to the control rod worth strongly depends on both reference and perturbed values of the rod worth. To obtain the conservative estimation for this sensitivity it is necessary to use the perturbation determined by engineering judgement of the uncertainty in the rod worth instead of arbitrary small perturbation. Unfortunately, the BARS pin-by-pin neutronic model does not allow increasing the rod worth by more than 4% against its reference value (1.21β). To consider larger perturbation in the control rod worth the following formula can be applied to the sensitivity of the fuel enthalpy to the control rod worth:

$$S_\rho = S_{\rho 1}(1-\beta/\rho)/(1-\beta/\rho_1) \quad (2)$$

where S_ρ is the sensitivity for the required perturbed value of the rod worth (ρ) and $S_{\rho 1}$ is the sensitivity for the perturbed value ρ_1 ($\rho > \rho_1$). $S_{\rho 1}$ was obtained from RELAP-BARS calculation with the rod worth increased by 3.7% in comparison with the reference value (ρ_0 = 1.21β). This formula was derived using a simple expression for the energy deposition obtained in a frame of the Nordheim-Fuchs approximation [1]. The formula was checked using another value for ρ_1 obtained by increasing the reference value ρ_0 by 2.4%. Both results are very close.

The following values were obtained for the uncertainties in the neutronic and thermal-hydraulic quantities:

- reactivity worth of the ejected rod – 15%;
- delayed neutron precursor fraction – 5%;
- fuel temperature reactivity coefficient – 15%;
- moderator density reactivity coefficient – 5%;
- pellet heat capacity – 8%;
- gap conductance – 110%;
- pellet conductivity – 25%;
- clad-moderator heat transfer coefficient – 10%;
- fraction of energy deposited in the moderator – 20%;
- radial power peaking factor for the pellet – 5%.

The uncertainty in the calculated rod worth was estimated as equivalent to two standard deviations based on the data presented in [1]. The uncertainties in the delayed neutron precursor fraction, the Doppler coefficient, and pellet heat capacity were taken from [1]. The maximum uncertainty in the Doppler coefficient, obtained from the BARS validation results [7] does not exceed the uncertainty estimated in [1] using engineering judgement. The uncertainty in the moderator density reactivity coefficient was obtained from the BARS validation results (as a result of comparisons with precise Monte Carlo calculations) [7]. The uncertainty in gap conductance was estimated taking into account that the gap closure could take place. It was obtained by a calculation of the transient with the closed gap. The uncertainty in pellet conductivity was estimated using the data from handbook [8], and the uncertainty in the clad-moderator heat transfer coefficient was taken from [9]. The uncertainties in the fraction of energy deposited directly in the moderator and in the radial power peaking factor for the pellet were estimated using engineering judgement.

It should be noted that unlike the maximum core power and power pulse width, the maximum fuel pellet enthalpy is a local parameter. So, the uncertainty in the local power should be taken into account to estimate the uncertainty in the fuel enthalpy together with above-mentioned quantities. A perturbation in local power is not calculated by RELAP-BARS directly without perturbations other quantities. Therefore, the approach proposed in [1] was used to estimate contribution of the uncertainty in local power to the uncertainty in fuel pellet enthalpy. If one assumes that the increase in fuel pellet enthalpy is proportional to the local power form factor F , then the contribution of the uncertainty in local power to the uncertainty in fuel enthalpy can be considered as the addition of $(\delta F\ /\ F)^2$ to the formula (1). F is defined as the fuel pellet power divided by the total core power. Based on the analysis carried out in [1], the uncertainty of 8% was taken for F .

Table 1 presents the calculational results for the sensitivity of fuel pellet enthalpy.

Table 1. Sensitivity of Fuel Pellet Enthalpy

Quantity (x)	PARCS	BARS (at = 0.785 s)		BARS (at = 2.5 s)	
	S_x	S_x	$(S_x)^2(\delta x/x)^2$	S_x	$(S_x)^2(\delta x/x)^2$
ρ	5.5	7.14	1.147	4.36	0.428
β	-4.0	-3.05	0.023	-1.34	0.004
α_d	-1.0	-0.90	0.018	-0.77	0.013
α_m	-	-0.18	$< 10^{-3}$	-0.39	$< 10^{-3}$
C_p	0.9	0.04	$< 10^{-3}$	0.15	$< 10^{-3}$
h_g	-	-0.09	0.010	-0.12	0.017
K_f	-	-0.04	$< 10^{-3}$	-0.20	0.003
h_w	-	-0.07	$< 10^{-3}$	-0.08	$< 10^{-3}$
γ	-	-0.03	$< 10^{-3}$	-0.02	$< 10^{-3}$
F_p	-	-0.20	$< 10^{-3}$	-0.23	$< 10^{-3}$

The RELAP-BARS results for the sensitivity of fuel enthalpy S_x and for contributions of each quantity x to the uncertainty in fuel enthalpy $(S_x)^2(\delta x/x)^2$ are presented at two time moments: the time of the maximum fuel enthalpy in the transient with scram (t=0.785 s) and for the end of the transient (t=2.5 s). For comparison, corresponding results for the sensitivity of fuel enthalpy obtained using the PARCS code [1] are presented in Table1 too. The results demonstrate that the sensitivity of fuel enthalpy to the most of quantities strongly depends on time because of non-adiabatic nature of the transient. Maximum contribution to the uncertainty in fuel enthalpy is due to the uncertainty in rod worth. Qualitatively, the RELAP-BARS results are agreed with the PARCS ones. However, the PARCS results give conservative estimation for the sensitivity to the most of the quantities in comparison with the RELAP-BARS results, because the adiabatic approximation was used in the PARCS calculations. Table 1 shows larger value of the sensitivity to rod worth obtained using BARS at t = 0.785 s, because a perturbation of +15% in rod worth was considered using formula (2) instead of a very small perturbation used in PARCS.

Note that the BARS gave the value of 5.14 for the sensitivity for a perturbation in rod worth of +3.7% and the value of 4.93 for a perturbation of +2.4%.

The resulting uncertainty in the maximum fuel pellet enthalpy (the transient with scram) obtained using the BARS calculations is 110% for a rod worth of 1.2β. The uncertainty in fuel pellet enthalpy is 69% at the end of the reference transient.

The RELAP-BARS calculational results for the sensitivity of the maximum core power and power pulse width are given in Table 2.

The most contribution to the total uncertainty in the maximum core power and power pulse width is due to the uncertainty in rod worth as well as in case with fuel pellet enthalpy. Contributions of other quantities to the resulting uncertainties are very small. The resulting uncertainties were estimated as 216% in the maximum reactor power and 76% in power pulse width.

Table 2. Sensitivity of Maximum Core Power and Power Pulse Width

Quantity (x)	Maximum Core Power		Pulse Width	
	S_x	$(S_x)^2(\delta x/x)^2$	S_x	$(S_x)^2(\delta x/x)^2$
ρ	13.87	4.328	-4.98	0.558
β	-10.77	0.290	3.09	0.024
α_d	-0.97	0.021	< 0.01	$< 10^{-3}$
α_m	-0.01	$< 10^{-3}$	-0.02	$< 10^{-3}$
C_p	1.05	0.007	0.01	$< 10^{-3}$
h_g	< 0.01	$< 10^{-3}$	< 0.01	$< 10^{-3}$
K_f	< 0.01	$< 10^{-3}$	< 0.01	$< 10^{-3}$
h_w	-0.01	$< 10^{-3}$	-0.01	$< 10^{-3}$
γ	-0.03	$< 10^{-3}$	-0.01	$< 10^{-3}$
F_p	< 0.01	$< 10^{-3}$	< 0.01	$< 10^{-3}$

4. EFFECT OF INITIAL CORE POWER

To analyze the effect of initial core power on key safety parameters, a calculation of the ejection of the central rod in the TMI-1 PWR starting from 33% of rated power was carried out using RELAP-BARS. To simplify a comparison of the results for HZP and non-zero power conditions, the same maximum value of 1.2β for the reactivity was considered in REA from 33% of rated power. The same RELAP input deck was used in the RELAP-BARS calculation as in the HZP case. To reach a criticality for the initial steady state, an additional withdrawal of control rods at the core periphery was done in the BARS input deck. No other changes were done in the HZP input deck.

Figures 4 and 5 show the reactor power and reactivity as a function of time for both HZP and 33% of rated power cases. In the last case the core power reaches its maximum value of approximately 14 of rated power at about 0.13 seconds. This peak value by more than 3.5 times exceeds corresponding value for the zero power case under the same maximum value of reactivity.

The reasons for this very large difference are the following.

- The decrease of 80% in the Doppler reactivity coefficient for the non-zero power case in comparison with the HZP one. This produces the factor of 1.8 in the resulting difference.
- The increase of 3% in the “net” (without feedback) reactivity inserted by the control rod for the non-zero power case in comparison with the HZP one. Note that in the HZP case the “net” reactivity inserted by the control rod and the maximum reactivity are the same. This increase produces the factor of 1.55.
- The “net” effect of initial power on peak power. This produces the factor of 1.15 in the resulting difference.
- The increase of 10% in pellet heat capacity for the non-zero power case in comparison with the HZP one. This produces the factor of 1.1.

Unlike the HZP case, in the non-zero power case the effect of feedback appears during the ejection of the control rod. This provides power pulse behavior as faster and sharper compared with the HZP case.

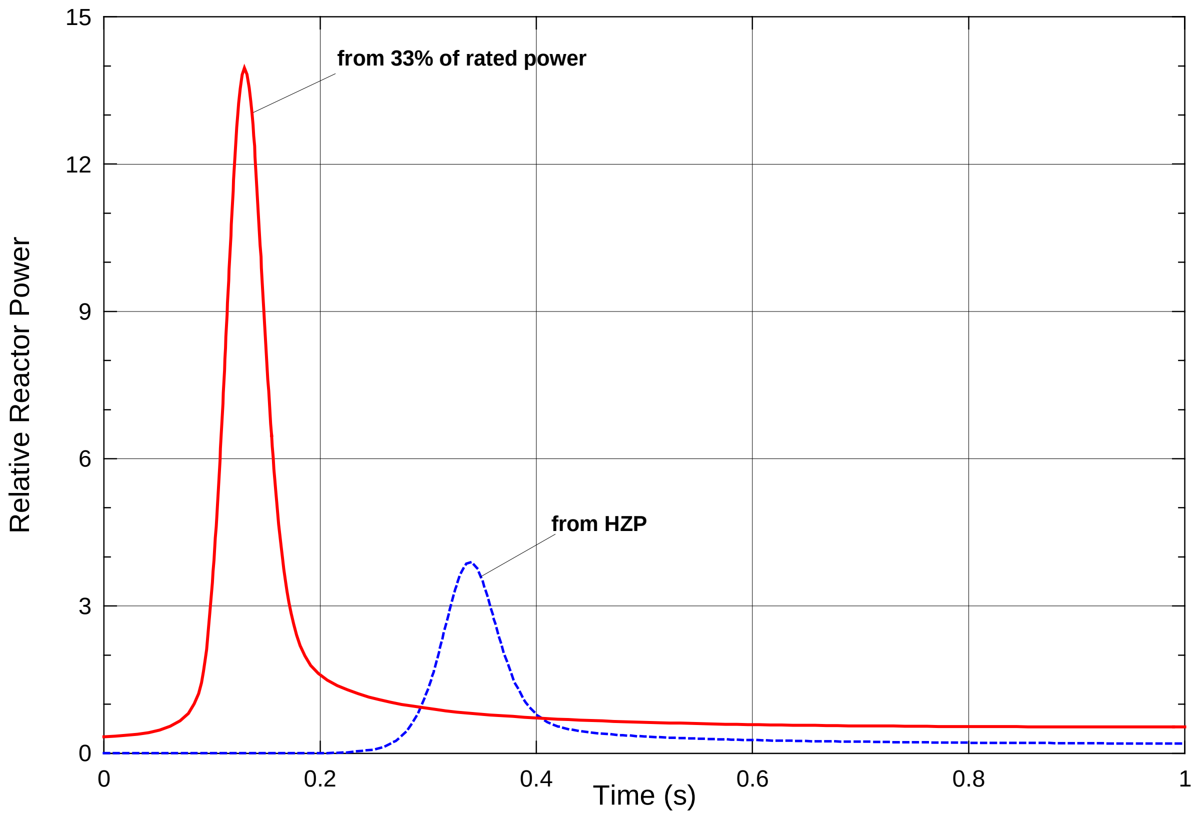

Figure 4. Reactor Power vs. Time

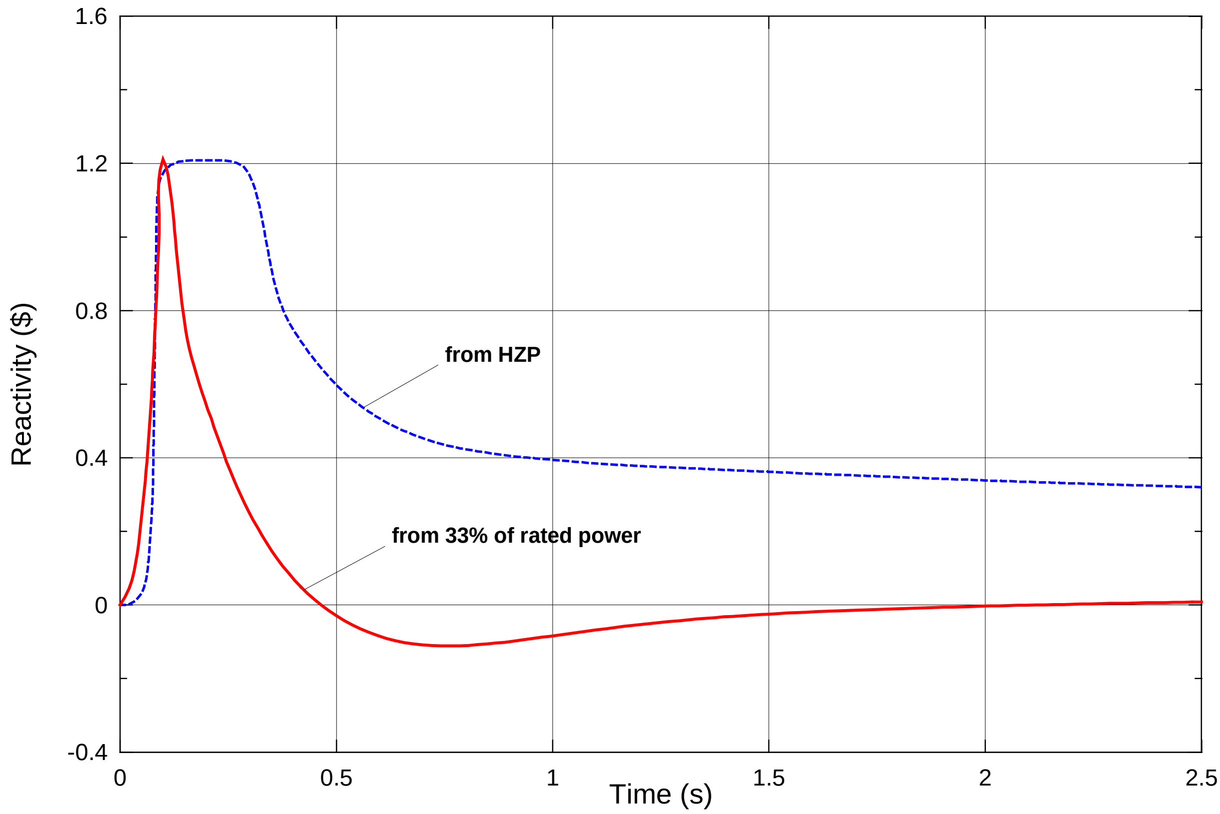

Figure 5. Reactivity vs. Time

Figure 6 shows assembly averaged radial power at three states of the core: at the initial steady state (0 s), at the time of the peak power (0.13 s), and at the end of the transient (2.5 s). Unlike the HZP case, significant deformations in radial power distribution take place after the rod ejection in the non-zero power case. Assembly powers differ up to 14.6% at the time of the peak power and at the end of the transient.

Figure 7 shows fuel enthalpy in the hottest fuel pellet as a function of time for two assemblies: K10 and H9. Up to the time of 1.6 s the maximum value of enthalpy occurs in the pellet located in assembly K10, but at the end of the transient assembly H9 contains the pellet with the maximum enthalpy. Change in the hottest pin location takes place due to power redistribution during the transient. This phenomenon can lead to some difficulties in a prediction of the hottest pin using assembly-by-assembly approach together with the pin reconstruction procedure to calculate fuel pellet enthalpy.

Results of the study of the effect of initial power on parameters of the accident are summarized in Table 3.

Comparison between the HZP case and the REA from 33% of rated power shows that the difference in the maximum fuel enthalpy increment is about 40% and the maximum fuel pellet enthalpies differ by about 29 cal/g (37.6 versus 66.7 cal/g).

Table 3. REA Parameters in Comparison with the HZP Case

Parameter	From 33% of rated power	From HZP
Maximum inserted reactivity (β)	1.22	1.21
Peak power of the core (GW)	38.7	10.7
Time of peak power (ms)	130	338
Power pulse width (ms)	44	63
Position of the hottest assemblies	K10 and H9	H9
Peak power of the fuel pin (MW)	2.82	0.835
Maximum fuel pellet enthalpy (cal/g)	66.7	37.6
Maximum fuel enthalpy increment (cal/g)	28.7	20.6
Minimum of coolant outlet density (g/cc)	0.691	0.755

	8	9	10	11	12	13	14	15
H	0.213 1.936 1.690	1.169 2.514 2.227	1.298 1.826 1.654	1.458 1.687 1.580	0.647 0.641 0.636	0.974 0.830 0.878	0.747 0.589 0.646	0.391 0.299 0.333
K		1.139 1.936 1.726	1.613 2.133 1.949	1.242 1.383 1.305	1.247 1.202 1.202	0.664 0.561 0.595	1.141 0.893 0.984	0.471 0.361 0.402
L			1.278 1.502 1.398	1.445 1.495 1.443	1.157 1.051 1.071	1.355 1.112 1.191	1.189 0.927 1.021	0.438 0.334 0.372
M				0.704 0.665 0.669	1.217 1.040 1.091	1.021 0.817 0.883	0.785 0.608 0.671	
N					0.664 0.537 0.579	1.062 0.826 0.908	0.499 0.383 0.424	
O			at t = 0 s → at t = 0.13 s → at t = 2.5 s →			0.607 0.464 0.514		

Figure 6. Assembly Averaged Power at Three States of the Core

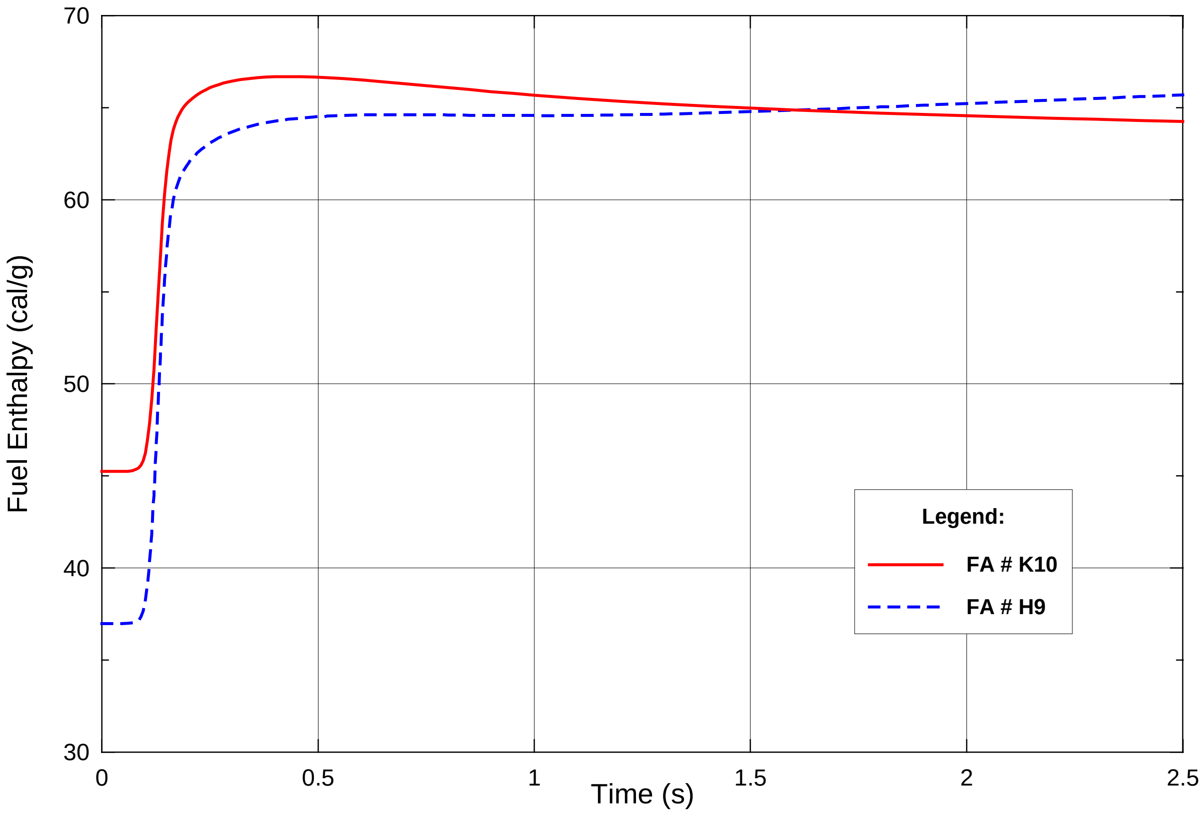

Figure 7. Fuel Enthalpy in the Hottest Pellet vs. Time

5. CONCLUSIONS

The uncertainty analysis for the PWR central rod ejection accident starting from the HZP conditions carried out with the RALAP-BARS code showed that the uncertainties in the key parameters of the accident would be determined to a great extent by the uncertainty in ejected rod worth. For a rod worth of 1.2β with the uncertainty of 15% (corresponding to two standard deviations), the uncertainty in local fuel enthalpy was estimated as 110%, the uncertainty in the maximum core power – as 216%, and the uncertainty in power pulse width – as 76%.

The results demonstrated non-adiabatic nature of the transient and showed that the sensitivity of fuel enthalpy to the most of neutronic and thermal-hydraulic quantities strongly depends on time. Qualitatively, the RELAP-BARS results are agreed with the PARCS ones. However, the PARCS results gave conservative estimation for the sensitivity of fuel enthalpy in comparison with the RELAP-BARS results because of the adiabatic approximation used in the PARCS calculations.

The comparative study of the accident starting from 33% of rated power showed strong dependence of a number of the REA parameters on initial core power. Under the same rod worth of 1.2β, the peak power in the transient from 33% of rated power was by 3.6 times greater than that in the transient from HZP. Unlike the HZP case, a change in the hottest fuel pin location takes place due to power redistribution during the transient from the non-zero power. This phenomenon can lead to some difficulties to predict the hottest pin using assembly-by-assembly approach together with the pin reconstruction procedure to calculate fuel pellet enthalpy. In comparison with the HZP case, the REA from 33% of rated power leads to the increase in the maximum fuel pellet enthalpy up to 66.7 cal/g.

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