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Analysis of Fuel Criticality during Severe Accidents

Interest in the analysis of beyond design basis accidents, involving a combination of several failures with fuel damage, has increased throughout the world after the Fukushima accident. Stress tests were performed at NPPs, and development of severe accident management guidelines was started. These activities necessitated calculations to analyze the probability of beyond design basis accidents and assess their initiating events and consequences. One of the aspects in analysis of beyond design basis accidents is to determine the potential for re-criticality during such accidents.

The paper provides results of some criticality safety calculations for VVER reactors performed, in particular, by ÚJV Řež and SSTC NRS experts. It is shown how criticality can occur in different severe accident phases.

K e y w o r d s: criticality, severe accident, VVER, corium.

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Аналіз критичності палива під час важких аварій

У всьому світі після аварії на АЕС Фукусіми збільшився інтерес до аналізу запроектних аварій з накладанням кількох відмов та з пошкодженням палива. На АЕС проведено стрес-тести, почали розроблятися керівництва з управління важкими аваріями. Внаслідок такої діяльності зросла потреба в проведенні розрахункових аналізів як імовірності виникнення запроектних аварій, так і дослідження умов виникнення аварій та їх наслідків. Одним з аспектів аналізу запроектної аварії є визначення можливості виникнення критичності протягом аварії.

У статті наведено результати деяких розрахункових аналізів безпеки критичності для реакторів ВВЕР, зроблених, зокрема спеціалістами ÚJV Řež та ДНТЦ ЯРБ. Показано можливість виникнення критичності на різних стадіях протікання важкої аварії.

К л ю ч о в і с л о в а: критичність, важка аварія, ВВЕР, розплав.

In the analyses of beyond design basis accidents with core meltdown, the issue of potential re-criticality does not generally arise since core meltdown can occur only as a result of major core exposure and thus without sufficient neutron moderation. In case of supply of emergency cooling water, quite high boron content is assumed in PWR, so that criticality can be excluded.

There have been numerous experiments on test facilities around the world since the early 1980s. In particular, they were performed by Japanese Atomic Energy Research Institute (JAERI), Forschungszentrum Karlsruhe (FZK), Korea Atomic Energy Research Institute (KAERI), Hungarian Atomic Energy Research Institute (AEKI), and many others [1]:

CORA, QUENCH (FZK, KIT);
CODEX (AEKI/KFKI);
ALPHA/MUSE (JAERI);
PHEBUS (IPSN);
PARAMETER (IBRAE).

There were also some computer approaches, primarily for BWR reactors: [2], [3] and [4].

The Fukushima accident renewed interest in severe accident analysis, especially in terms of computer modeling [5].

Important in-vessel phenomena

IAEA identifies processes occurring in the core during a severe accident from different viewpoints [6]:

Thermohydraulics:

Natural circulation of steam and non-condensable gases (to delay the overall heating of the core), ≤1500 K;

Reflooding of hot, damaged cores (oxidation, hydrogen production, fission product release and melting), >1500 K;

Oxidation of core materials:

Zircaloy oxidation in steam (protective oxide film), 1500 K — 3000 K;

Oxidation of B_4C in steam (hydrogen and methane production);

Loss of core geometry:

Swelling and rupture of the cladding (release of fission products), 1000–1500 K;

Liquefaction and relocation of control and structural materials (segregating from the fuel, increasing the potential for re-criticality, local blockages), 1500 and 1700 K;

Liquefaction and relocation of zircaloy cladding (reduction in hydrogen production and heat generation due to oxidation), 2000 — 2200 K;

Liquefaction and slumping of the fuel (formation of large blockages), < 2870 K;

Relocation of molten pool materials into the lower plenum

Fragmentation of embrittled core materials, < 1500 K, > 1500 K;

Heating and failure of the lower head (vessel failure);

Other factors:

Impact of alternative core/vessel designs, 1000–1700 K;

Re-criticality, 1500 — 1700 K;

Fission product release and transport.

Thus, while the initial and final states of fuel can be set in computer modeling, the condition of the core (fuel, internals and control rods) between these states is determined very approximately and strongly depends on the performance of safety systems and personnel actions to mitigate consequences of the accident. At various stages of the accident, the core will be cooled down by water. If there is adverse ratio of the parameters, this can lead to criticality and self-sustaining chain reaction.

Since calculations of fuel criticality during severe accidents have become of interest only recently, there are few data on the results obtained.

Criticality under severe accidents

One of the available analyses is calculation of the neutron multiplication factor in failure of the VVER-1000 core reported in [7]. As indicated in the report, all calculations of core K_{eff} for various stages of core damage were performed with the MCU-REA/1 code (Figure 1).

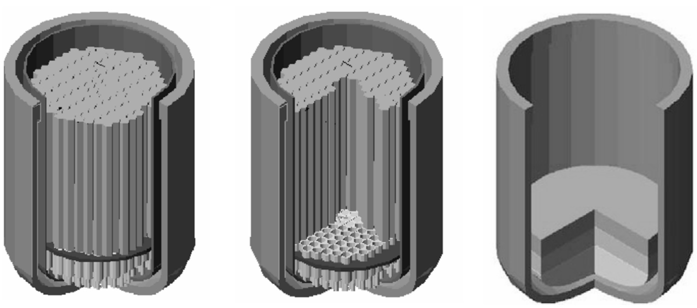


Figure 1 — various stages of VVER-1000 reactor core failure. Computer model

The parameters of the heterogeneous water-uranium lattice correspond to its maximum multiplying properties. The results of K_{eff} calculations for the core that preserves its structure are presented in Table 1.

Table 1 — K_{eff} of the core for different types of fuel cladding degradation

Cladding	K_{eff}
Normal cladding	1.216
Swelled cladding with mass conservation	1.215
Swelled ZrO_2 cladding with area conservation	1.227
Swelled ZrH_2 cladding with area conservation	1.264

Then a heterogeneous water-uranium lattice is considered with parameters corresponding to its maximum multiplying properties. Two cases of fuel location are considered: fuel remains in the form of corium particles instead of damaged fuel elements and fuel moves to the lower plenum (Table 2).

Table 2 – K_{eff} for two cases of fuel location

Number of damaged fuel assemblies	Fuel remains in the form of corium particles	Fuel in the form of particles moves to lower plenum
0	1.0831	1.0831
19	1.2369	—
61	1.2643	1.0788
91	—	1.0631
127	1.2860	0.9436

Figure 2 shows the results of K_{eff} calculation for VVER molten core located on the reactor pressure vessel bottom depending on the weight of zirconium and iron dissolved in the corium. The calculations assumed that the melt located on the bottom of the reactor pressure vessel is a complex multi-component porous system containing UO_2, Zr, and Fe. The weight of UO_2 is taken to be 80 tons.

As seen from the results presented in [7], the use of pure (non-borated) water for cooling may cause criticality at all stages of a beyond design basis accident with core melt.

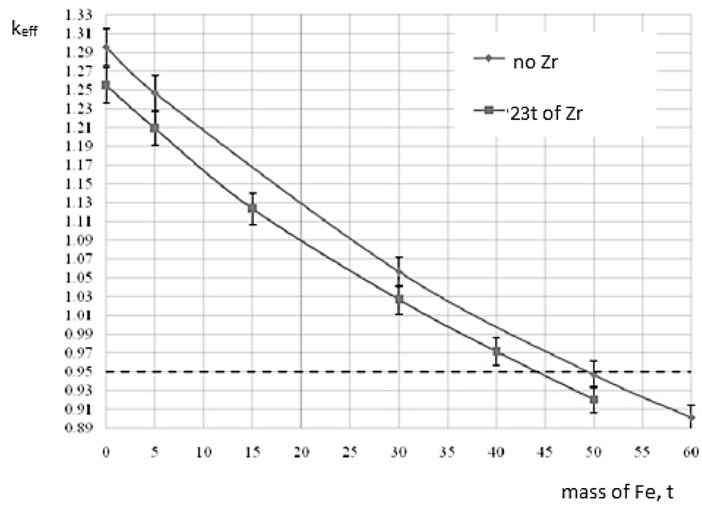


Figure 2 — K_{eff} of VVER molten core located on the bottom of the reactor pressure vessel depending on the weight of zirconium and iron dissolved in the corium

Another analysis presented in [8] and continued in [9] concerns BWR reactor fuel.

For conservative results, the spherical arrangement of the corium was chosen so as to achieve the lowest leakage of neutrons and critical mass. The holes within the corium were assumed to be completely filled with pure water (density = 1.0 g/cm^3) to promote critical moderation conditions. The geometrical model of the holes was set to have the body-centered cubic (BCC) structure to reproduce irregular placement at the beginning of the accident.

The total amount of corium was determined by changing the interval (D) between the poles, i.e. related to packing ratio, and the radius (R) of the whole corium shape.

Figure 3 shows a conceptual geometric model for MCNP criticality calculation.

These results are presented in the paper because this computer model may be applied to any reactor type; only the fuel type (uranium or MOX) and enrichment are relevant.

It was confirmed that the reactor corium resulting from 548 General Electric fuel assemblies has hardly the potential for re-criticality (Table 3). This is because Gd_2O_3 serves as an efficient and very strong neutron absorber in the fuel, even though pure water was filled in the containment vessel as a neutron moderator.

In case Gd_2O_3 is not included in the corium, the results showed the supercritical condition in the range from 1.04671±0.00195 to 1.37803±0.00147 despite a little amount of the corium (Table 4). It was also found out that criticality varied regardless of increase in the total amount of the corium changing with the packing ratio while the radius 'R' was constant.

Therefore, it is assumed that re-criticality of the reactor corium would be possible if there is no neutron absorber.

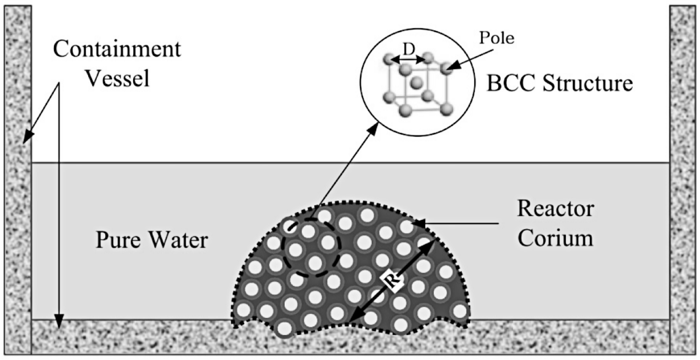


Figure 3 — conceptual model for criticality calculation from [8]

Table 3 — K_{eff} for reactor corium including Gd_2O_3

Packing ratio of corium [%]	Radius of whole corium shape (R)		
	50 cm	100 cm	150 cm
10	0.23462	0.25749	0.26275
30	0.34651	0.38308	0.39488
50	0.45089	0.51087	0.52625
60	0.49758	0.57373	0.59486

Table 4 — K_{eff} for reactor corium not including Gd_2O_3

Packing ratio of corium [%]	Radius of whole corium shape (R)		
	50 cm	100 cm	150 cm
10	1.05990	1.17743	1.20648
30	1.20503	1.34165	1.37803
50	1.11952	1.25780	1.29668
60	1.04671	1.18205	1.22068

In [9] a slightly different computer model is used. Molten rods, elements of fuel assemblies and internals were considered (Figure 4). The criticality coefficient K_{eff} depending on the radius of the molten fuel is given in Figure 5.

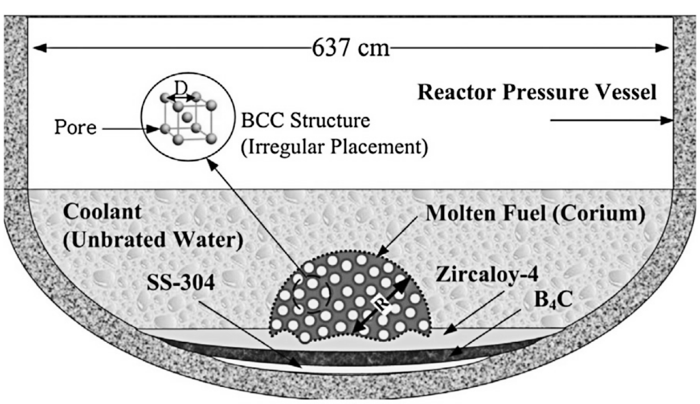


Figure 4 — conceptual model for criticality calculation from [9]

However, the critical mass was achieved with UO_2 of about 80 kg. The minimum amount exceeding the critical value of K_{eff} = 0.95 was approximately 60 kg. The total amount of uranium loaded in the core should be considered. The mass of uranium per GE 7×7 fuel assembly is slightly higher than 200 kg and 1/3 molten fuel can sufficiently result in re-criticality. In addition, the real reactor core is loaded with 500–800 fuel assemblies. The critical mass is attributable to K_{eff} = 1.11619 ± 0.00148 despite fuel degradation of 0.2%. Therefore, a small amount of corium, such as localized or lumped-mass corium, has a high potential for re-criticality and should be significantly addressed in the core criticality analysis for severe accidents.

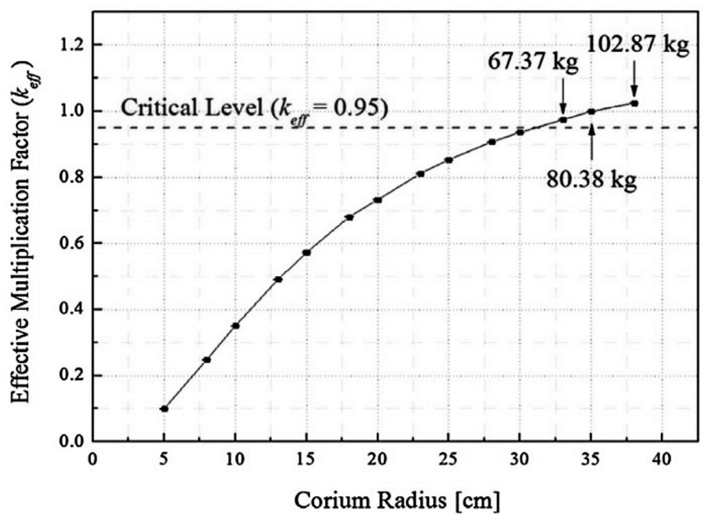


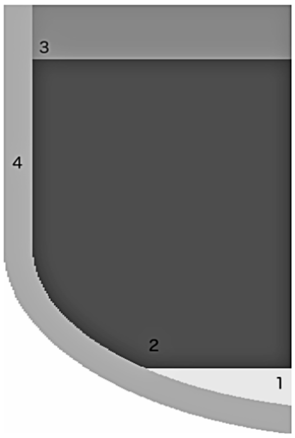
Figure 5 — variation of effective multiplication factor (K_{eff}) with increase in corium radius

The corium under the accident scenarios represented very high K_{eff} values (up to 1.4). Hence, supply of non-borated water into the layer-separated configuration can lead to relatively high chain reactions through enhanced moderation. Excessive borated water also results in positive reactivity. In addition, the prior verification of the boron quantity needed for criticality control is important to secure the supply in advance and to minimize the dilution time of boron.

The ÚJV approach [10] is based on the assumption that water in IO and all ECCs (active and passive) is borated. Moreover, regardless of the initiating event, water boiling results in increase in boric acid concentration in the remaining water and the formation of crystals on fuel rod surface. The behavior of boric acid crystals is of key importance, but has not been well investigated for severe accident conditions. However, the behavior of boric acid crystals and/or oxides will determine the remaining content of boron in debris or corium.

An example of analysis is shown below. This scenario assumed complete melting of all absorbing elements and damage of all fuel assemblies. The melt gets to the reactor pressure vessel bottom, and debris of the damaged fuel assemblies accumulate on top of the melt. Uranium has 4.6 % enrichment.

The melt and debris are surrounded by walls of the reactor pressure vessel (14.9 cm in thickness). There is a water layer (25 cm) above (Figure 6). There is vacuum at the ultimate boundary. The calculation was conducted using the MCNP code (Table 5).



1 — molten homogeneous mixture of stainless steel and boron stainless steel (1.8 wt.%);
2 — debris from fuel assemblies and shrouds;
3 — water;
4 — RPV wall

Figure 6 — conceptual model for criticality calculation

Table 5 — Values of K_{eff} versus density factor (df)

df	K_{eff}	σ(%)	df	K_{eff}	σ(%)
1.0	0.66505	0.02	2.0	1.40669	0.01
1.1	0.85588	0.01	2.1	1.41232	0.01
1.18	0.98054	0.01	2.15	1.41346	0.01
1.19	0.99389	0.01	2.2	1.41368	0.01
1.194	0.99980	0.01	2.25	1.41350	0.01
1.195	1.00066	0.02	2.3	1.41190	0.01
1.2	1.00761	0.01	2.4	1.40691	0.01
1.3	1.12306	0.01	2.5	1.39951	0.01
1.4	1.20808	0.01	3.0	1.33443	0.01
1.5	1.27201	0.01	—	—	—

The density factor characterizes the content of water in debris. As is seen from the previous table, the maximum K_{eff} corresponds to the density factor of ~2.2 and is equal to ≈1.4.

SSTC experts, as part of their trial calculations, also carried out analysis of fuel melting during a severe accident.

One such case was criticality analysis of molten core for the research reactor VVR-M, consisting of 210 fuel assemblies VVR-M2 with an enrichment of 19.7%. The computer model (Figure 7) is a cylinder composed of 210 fuel assemblies and surrounded by non-borated water with normal density (1 g/cm^3). The temperature of the system was 20 °C. The boundary conditions were set to vacuum. The multiplying properties of the system with changing ratio between the cylinder radius and height calculated with the SCALE code package are shown in Figure 8.

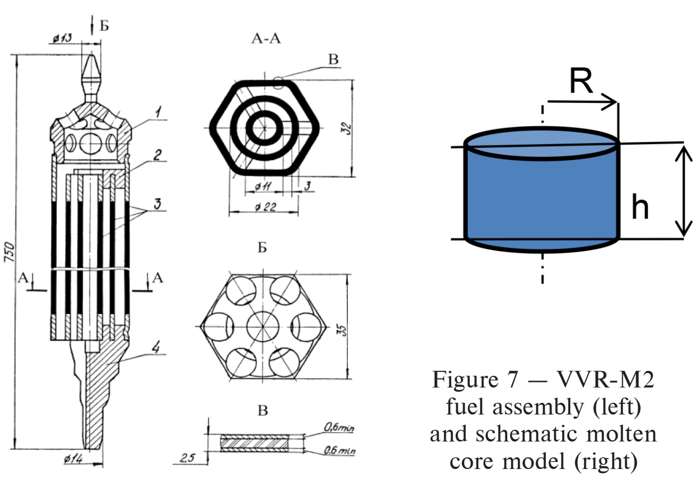


Figure 7 — VVR-M2 fuel assembly (left) and schematic molten core model (right)

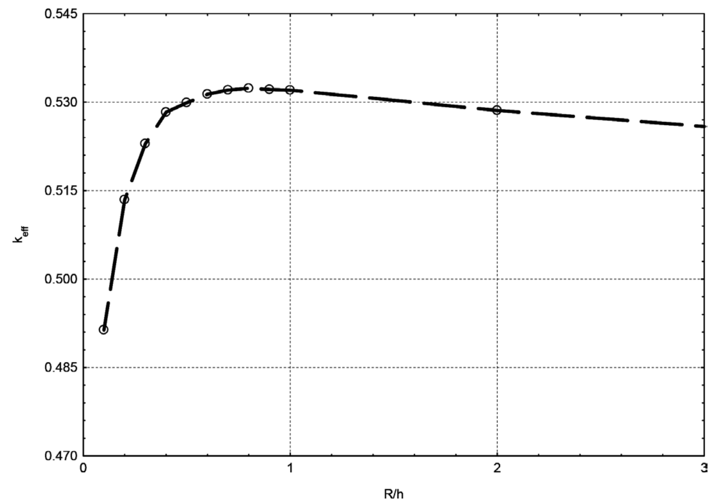


Figure 8 — criticality of molten core, for 210 VVR-M2 fuel assemblies

As seen from the results, the system having the molten core and surrounded by water is deeply subcritical.

Calculations for the case of fuel leakage from the fuel assemblies of TVSA type (4.45%/3.6% excluding burnable absorber) to the bottom of the reactor pool were also performed. Below are results of calculations performed to determine the critical weight and the size of the fuel-containing mass.

For this purpose, the fuel mass was assumed to be in the shape characterized by the lowest leakage of neutrons: a sphere surrounded by non-borated water with normal density (Figure 9). For the simplest determination of the critical dimensions of the molten fuel, such a model is acceptable. The water content in the fuel sphere ranged from 0 to 90 vol.% to examine the influence of fuel mass porosity and water ingress on multiplication properties (Figure 10).

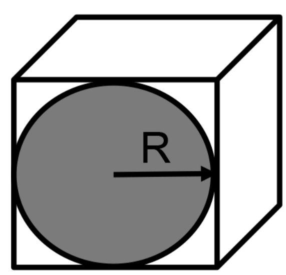


Figure 9 — model for criticality calculation

As shown by the calculations, the minimum weight of uranium that can lead to criticality in the total absence of any absorber and with optimum water-uranium ratio is 61 kg in the case of 4.4% enriched UO_2 fuel.

Results of similar calculations, but taking into account uniformly mixed claddings with fuel, are shown in Figure 11. The minimum weight obtained in this case was 70 kg of uranium.

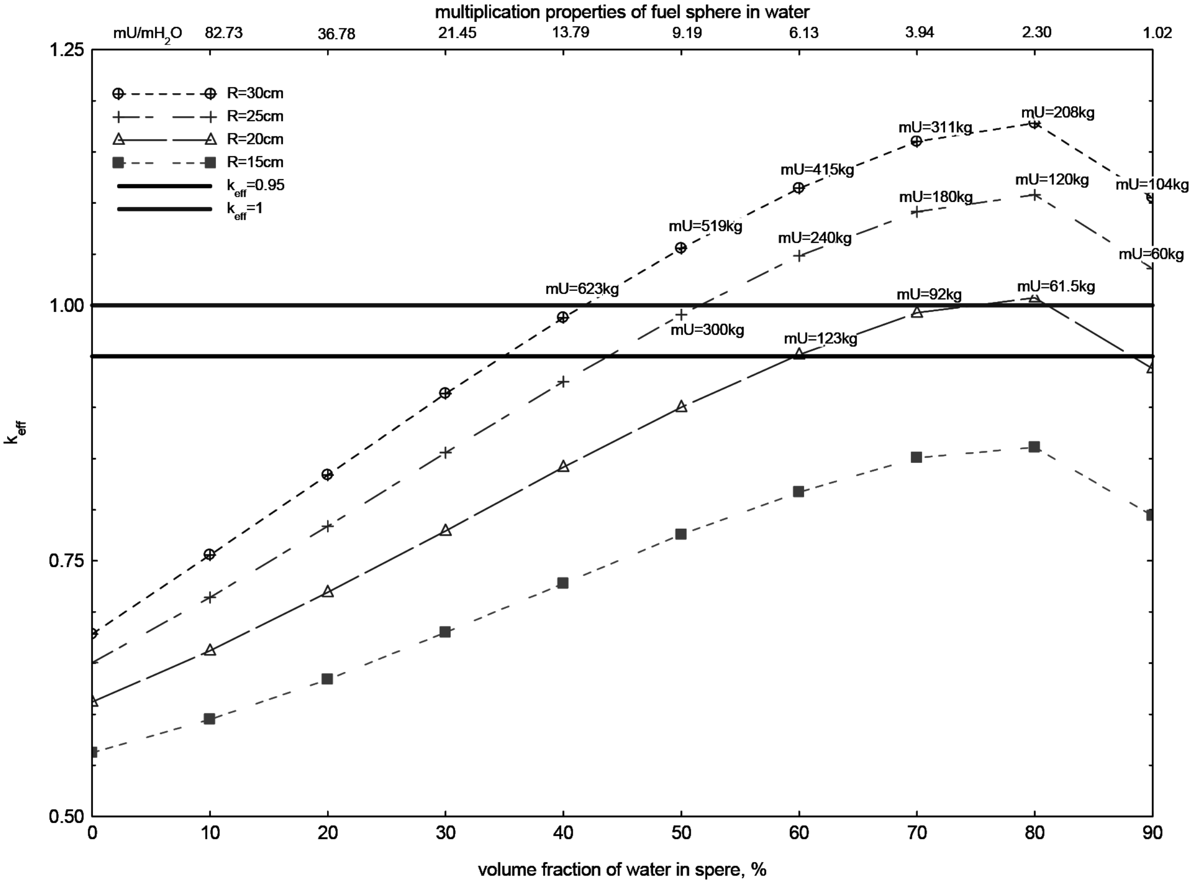


Figure 10 – criticality of fuel in water

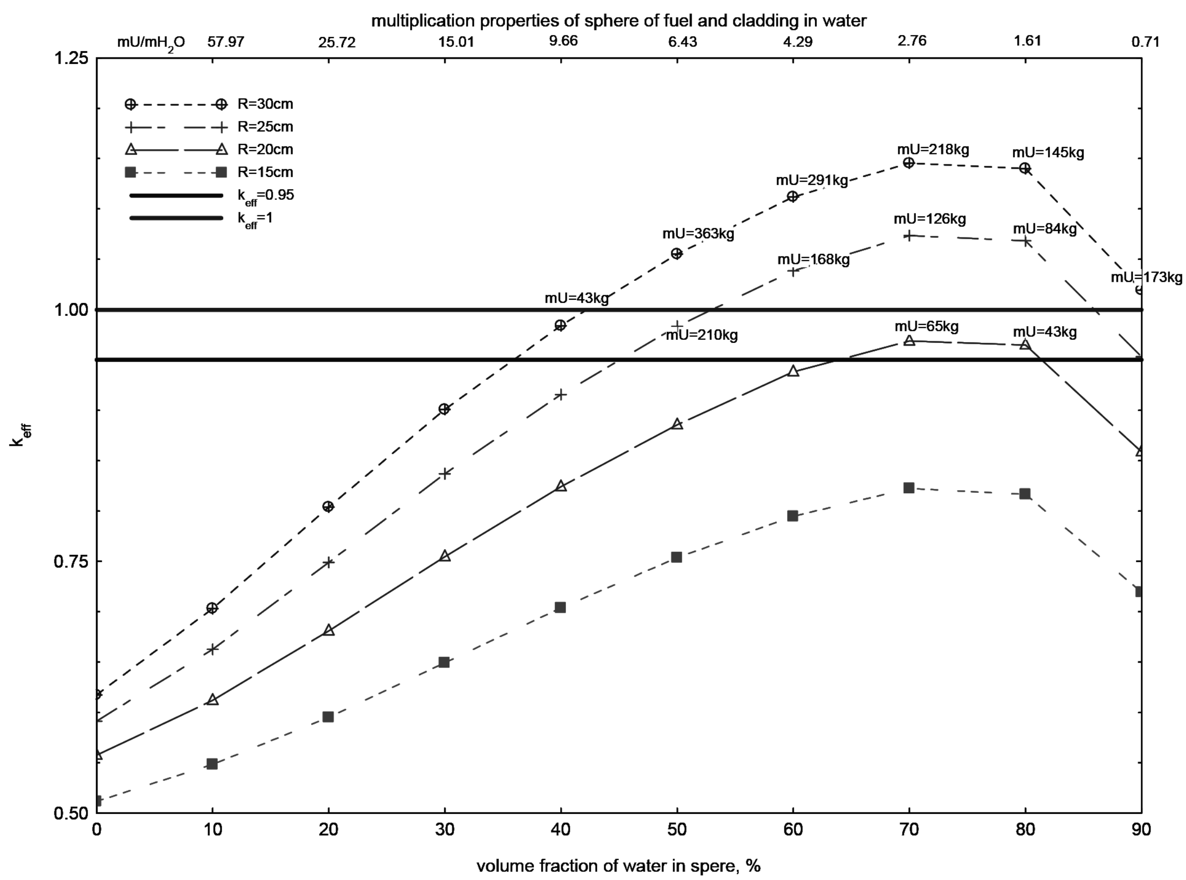


Figure 11 — criticality of fuel and claddings in water

Conclusions

As shown in the paper, the critical mass for typical VVER fuel (enriched to 4.4%) is not so great, about 60-70 kg of uranium, corresponding to 70-80 kg of UO_2 fuel.

Hence, it is theoretically possible that fuel-containing masses can reach critical condition in destruction of the reactor core. However, the main condition for achieving the criticality is the presence of water in and around the fuel-containing mass, which can be supplied by safety systems or personnel actions to retain the melt inside the reactor vessel and ensure its cooling.

Therefore, in the criticality analysis for severe accidents, it is important to take into account the following factors:

1. Boiling process
2. Boric acid behavior
3. Core degradation progress, relocation of fuel mass
4. Water injection conditions.

In the above regard, it is currently difficult to analyze the course of a severe accident in terms of the criticality analysis. The degree of conservatism and representativeness of any computer model will depend on various factors and accident progress.

A severe accident initiated in the spent fuel pool has some risk of criticality due to possible injection of non-borated water and further analysis is needed. Nevertheless, it proceeds much more slowly than in the reactor. Moreover, in most spent fuel pools, there are compact storage racks made of borated steel. All this reduces the possibility of reaching the criticality, but leads to other problems such as generation of hydrogen and other gases due to the interaction of the fuel assembly elements and storage rack structures with water at elevated temperatures.

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