PRESENT STATUS OF SODIUM VOID REACTIVITY PREDICTIONS IN CONVENTIONAL AND NONCONVENTIONAL FAST REACTOR CORE DESIGNS

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ABSTRACT

In this paper a review is given on the present status of the prediction of sodium void reactivity effects in conventional and nonconventional fast reactor core designs. First problems arising in the assessment of fast reactor safety are outlined and are discussed in relation to reactor physics aspects. Target accuracies for the sodium void effect are estimated. Nuclear data and calculational uncertainties are discussed, and the present accuracy achieved in various laboratories of the world is summarized. Some cases, for which the calculated void effects were rather far off the respective experimental results, were re-examined, resulting in significant improvements.

1. INTRODUCTION

Numerous experiments have been performed in critical assemblies throughout the world in assessing a reliable prediction of the reactivity effect due to sodium voiding in fast reactors. In large conventional breeders of about 1300 MWe the maximum Na-void reactivity is about 8 $, and a corresponding reactivity insertion in unprotected accident situations may lead to an energy release which may be difficult to contain within the system. In order to reduce the positive reactivity effect due to sodium voiding and, by the same means, to improve the breeding potential of fast reactors, internal zones of fertile material have been introduced; in addition, also the use of thorium in fast reactors has been discussed.

In this contribution, we concentrate on the discussion of the present status of sodium void reactivity predictions as obtained in various laboratories in the world; both conventional and nonconventional core designs are covered. Following items are discussed:

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Sensitivity of the void effect to the methods and data used, present accuracies in the prediction of conventional and heterogeneous experiments in zero power facilities, and the target accuracies in large fast power reactors. The relation to safety considerations is put into perspective. This review is partly based on the discussions[1] held at the 1976/77 meetings of the NEA-Committee on Reactor Physics.

2. REACTOR PHYSICS ASPECTS RELATED TO FAST REACTOR SAFETY

A brief review of present problems in the safety assessment of sodium cooled fast reactors is given to the extent necessary to judge the target accuracies of the associated physics parameters.

Local pin failures in a fast reactor core cannot be excluded during operation of the plant. They might be caused for instance by undetected material defects prior to assemblage of a subassembly or during assemblage. The important question is whether such a failure leads to a release of fuel into a coolant channel causing a local blockage and probably local sodium boiling. Failure to detect the blockage may lead to propagation between subassemblies producing a major accident condition.

In unprotected whole core accidents important objectives are: (1) More systematic knowledge on the location of pin failure and associated fuel movement in the central pin cavity towards the leak, (2) mode and amount of fuel expulsion into the coolant channel, (3) reliable description of fuel coolant interaction and fuel sweep out, (4) reliable assessment of the effects of fission gas release on the subsequent material displacements. The experimental evidence is not conclusive at present; large efforts are undertaken throughout the world.

With respect to fuel-coolant interactions, the analysis mainly is performed in a parametrized form, the parameters, as fragmentation- and mixing-time, or radius of fragmented particles, bearing a large uncertainty. The parametric models usually yield conservative answers with respect to the associated production in damaging energy. Fuel sweep-out with its associated negative reactivity feedback can produce a very important reduction in the severity of this accident sequence but only if it occurs rapidly enough to forestall the positive reactivity effects of voiding. It is therefore important to have a means of measuring the rate of fuel sweep out in experiments. The theoretical description of loss of flow (LOF) accidents is confronted with the problem that with a proper representation of intersubassembly noncoherence the accident sequence tends to proceed into gradual meltdown of the core instead of ending in a vigorous disassembly; the pressures generated may be too low to cause a massive dispersal of molten material. Therefore only a part of the core acts as a fluid, larger parts may even be completely intact. This "transition phase" is very difficult to model and deserves further attention.

This brief consideration of accident situations indicates that the current uncertainties in fast reactor safety analysis are dominated by the nonneutronic characteristics of the accident sequences, i.e. material dynamics, thermohydraulics and thermodynamics. These uncertainties force the reactor designers to use conservative estimates in order to assure the safety of the plant.

The degree of uncertainty of reactor physics parameters is much less than the uncertainty in the non-neutronic characteristics, because the physics methods could be checked experimentally in great detail for many examples in zero power facilities. From the physics investigations in the SEFOR reactor, and since some years also from PHÉNIX and PFR, essential experience has been gained with respect to larger test and prototype fast reactors. The most important aspects of reactor physics work related to the safety assessment of fast sodium cooled reactors are the reliable prediction and verification of: Doppler reactivity effects in the predisassembly and disassembly phase of a nuclear excursion, sodium void reactivity distribution in a complex core situation, reactivity effects of fuel and clad motion, and control rod worth distribution.

3. TARGET ACCURACY FOR THE SODIUM VOID REACTIVITY EFFECT

In the light of the discussion in section 2, in transient overpower (TOP) and LOF accidents the reactivity associated with boiling and ejection of sodium out of the coolant channels acts as the most important positive driving force in the predisassembly phase. The course of the accident sequence is mainly determined by the voiding mechanism within each subassembly and the subsequent progression from subassembly to subassembly, the uncertainty of these processes being definitely larger (probably by about a factor of two) than those associated with the determination of the corresponding reactivity, which at present in the central core regions amounts to about ± 15-25 %. It is difficult to state a simple value for the desired target accuracy because this value depends on the particular design and on the maximum permissible void reactivity. At the present time it appears to be an open question whether one can place a definite value on the latter quantity.

Sensitivity studies in unprotected LOF and TOP-type excursions for conventional fast reactor cores show that an uncertainty of about 15 % to 20 % in the Na-void reactivity coefficient could be tolerated, keeping in mind that the non-neutronic uncertainties in the voiding initiation and also the process itself dominate the neutronic uncertainties. As a consequence for conventional fast reactor core designs the already obtained experience for the maximum positive Na-void effect seems to be almost sufficient for a meaningful prediction of safety characteristics of fast systems. But care has to be taken in the simple extrapolation of the results obtained in critical assemblies to the application in fast power reactor systems: Less certain information is obtained for operating conditions concerning for instance, the temperature, the different isotopic composition of plutonium and the influence of fission products as distributed neutron absorbers.

Non-conventional U/Pu cores with internal fertile zones can reduce the positive void effect appreciably, depending on the special design. As a consequence, the safety characteristics of large heterogeneous fast reactors with respect to sodium voiding are more favourable than the corresponding ones for conventional cores. But if the excursion brings the core to slumping and compaction phases, the resulting energy release finally is more or less determined by the latter phenomena, thereby making the benefit of a low sodium void to disappear. With respect to target accuracy of Na-void effect in such systems, it first appears as if one could relax to a value less than that quoted for conventional designs. For obvious reasons this truly holds only for those systems, where the maximum permissible void reactivity is limited to less than 1 $. For other designs at the present stage of the investigation, accuracy of about 20 % for the maximum void effect should be the goal of the physics analysis and this should hold also for the introduction of thorium into fast reactor systems (e.g. into the blanket only).

4. THE THEORETICAL ASSESSMENT OF THE SODIUM VOID REACTIVITY EFFECT

The theoretical interpretation of the sodium void effect has to be rather sophisticated because of the compensating effects of the positive spectrum and negative leakage contributions. In the frame of this paper we only can briefly summarize the present status of the art. The basic approach is rather similar in all laboratories throughout the world: a) use of diffusion theory (mainly two dimensions) with a well established group constant set for the determination of the reactivity effect either by computing directly the eigenvalue differences or by means of perturbation theory (mainly exact perturbation), b) correction for global transport effects, c) correction for cell heterogeneity including neutron streaming in the voided regions. The associated uncertainties in these calculations are related to the uncertainties in the basic nuclear data used and to the various approximations in establishing the correction factors.

4.1 Sensitivity of Sodium Void Reactivity Effect to Nuclear Data

It is well known that the sensitivity of the void effect due to nuclear data changes is dependent on the system for which this sensitivity is investigated and on the way how criticality is being readjusted for the sodium-in case. Among others, McKnight et al.[2] have presented results for a 1000 MWe conventional fast reactor. One of the main conclusions from this study is that the basic cross sections have to be varied by a larger amount than the present basic data uncertainties for each isotope in order to achieve a 20 % difference in prediction of the void effect. The main influence is coming from α of Pu-239, σ_c of U-238, and to some extent σ_c of Pu-240. The elastic scattering is calculated well in a many group scheme (above/about 1000 for spectrum calculations). But the inelastic scattering in U-238 with its uncertainty in the total inelastic cross section and in the scattering matrix remains a further source of unreliable prediction for the maximum void effect of a conventional

reactor. Only an unfavourable combination of all these uncertainties may lead to larger discrepancies.

The NEA-Committee on Reactor Physics has proposed a large fast reactor benchmark[3] to be calculated by the interested laboratories. The predictions of the reactivity effect for voiding the inner core zone are within a band of ± 15 %, if the results from USA, UK, France, Japan, USSR and Germany are compared, the scatter being ± 10 %, if USSR is excluded from this comparison [4]. This favourable agreement is being disturbed, if larger voids encompassing the region of changing sign of the spatial void distribution are calculated (complete voiding of the core region); the prediction is extremely unreliable for the void effect in the outer core and axial blanket. For these small reactivity effects computational deficiencies are superimposed on the basic nuclear data uncertainties.

From the mentioned sensitivity studies and the benchmark intercomparison it is justified to conclude that the present data uncertainties are likely to result in an uncertainty of about/below 15 % in the prediction of the maximum void reactivity effect of a large conventional reactor. This conclusion is indirect and is based on the assumption that the corresponding data evaluations and group constant preparations are performed independently in various countries and that the prescribed standard calculational scheme (2-dimensional diffusion theory) does not yield larger deviations in the central core region; further, a common large systematic error in the differential cross section measurements is excluded.

4.2 Computational Errors in Whole Core Calculations

Referring to 2d diffusion theory as the standard calculational scheme errors in the prediction of the sodium void effect may arise from too broad energy groups, using the same macroscopic group constants for the sodium-in and sodium-out case (especially for large voids with strong spectrum changes), inappropriate mesh-size, numerical difficulties especially for small voids and the use of first order perturbation theory. These uncertainties are well understood [5] and the associated errors in the void effect prediction in principle can be avoided to a large extent. Due to cancelling effects of the components of the void effect, one has to investigate all these possible errors carefully for each new design, especially for the interpretation of an experiment.

This cumbersome investigation is not always performed; instead one uses frequently a standard group constant set, which may or may not be adjusted to k_{eff} of numerous critical assemblies. If the eigenspectrum of a newly considered experiment differs from that used as a weighting spectrum for the standard set, errors may arise which often are attributed to basic nuclear data uncertainties. Care has to be taken also to check the consistency in the integration schemes of the diffusion equation and of the subsequent perturbation calculation, which can be important if the magnitude of the void effect is small.

4.3 Transport Effects in Whole Core Calculations

It is self-explainatory that diffusion theory has to be corrected for transport effects especially for the determination of the void effect in the outer core and blanket regions. This correction for a central void is reported to be about 4 % in ZPPR-3 analysis [6]. In principle, if ray effects can be eliminated, the assessment of this transport effect in 2 dimensions adds to the computing time. Frequently this is accomplished in 1 dimension. Though for the maximum void effect of a large reactor this correction is of minor importance, in certain cases, such as the initial phase in a LOF accident situation, where boiling starts near the upper core/blanket interface, the proper magnitude of the initial reactivity effect is important.

4.4 The Treatment of Control Rods

An uncertainty in the B-10 absorption cross section of about 10 % exists above 200 keV. Besides this, neglecting the flux depression in the rods results in a significant overprediction of the void effect for a r-z control rod ring. A good reduction is achieved by using 3d calculational models (either direct diffusion or synthesis). The inadequacy of r-z model for the interpretation of void experiments has been studied extensively by McFarlane et al. [7] in ZPPR-3 and by Kaiser et al. [8] in ZPPR-5 especially for voiding in zones where off-center control rods are present. Only the 3d calculations gave satisfactory results in comparison to experiment: C/E lies in the range 0.8 to 1.0. Yoshida [9] has confirmed this independently. Thus again, in principle the associated errors in the modelling of the control rods can be reduced. Streaming in the follower region however poses difficulties [1].

4.5 Cell Heterogeneity and Neutron Streaming

The heterogeneous array of the materials in fast reactors has two aspects: a) it gives rise to a neutron flux fine-structure, which is influenced by spatial resonance self-shielding, and b) anisotropic diffusion of neutrons along sodium-filled or voided channels with associated reactivity effects takes place. The energy dependence of these inter-dependent effects is rather complex and cannot be discussed here [10]. Powerful methods have been developed in the past like e.g. the Karlsruhe KAPER-code [11], the UK-MURAL-system [12] and the French code HETAIRE [13]. They are based on collision probability methods, the KAPER code at Karlsruhe including the streaming effects using Benoist's approximation automatically. For sodium removal experiments in plate-type critical assemblies it is well known that the Benoist approximation yields infinite value for the diffusion coefficient in a true vacuum slot. Fortunately the smeared amount of structural material present in the channel is sufficient to apply the Benoist formalism without too large errors. This has been studied in detail by Gelbard et al. [14].

In order to check the validity of the theoretical approach, integral measurements have been performed at various laboratories to determine a) the flux fine structure in the unit cell and b) the reactivity change due to anisotropic diffusion by changing the orientation of the platelets in the cell. At Karlsruhe, with a slightly modified unit cell which allowed to orient the platelets both vertically and horizontally with respect to the core axis, and experiment [10] was performed. In this experiment, for each orientation of the platelets, the void zones were enlarged stepwise in the four central elements from a small central void zone to the full core height. The calculations were performed using one dimensional perturbation calculations in slab geometry. The radial bucklings came from two dimensional calculations. The anisotropy of diffusion entered into the calculations through the cross sections prepared using the cell program KAPER [11]. While the calculations slightly overestimated the diffusion term, it was found that the difference in the sodium void effect for different platelet orientations was well interpreted by the calculations. Measurements of this type have also been performed recently using the sector region of FCA Assembly VII-1, a mockup for the prototype Fast Breeder Reactor "MONJU". Shirakta and Iijima [15] report that the parallel to perpendicular ratio of diffusion coefficient $\tilde{D}_{\parallel}/\tilde{D}_{\perp}$ was determined by measuring the reactivity change in the core due to 90° turning of plate direction. Their calculations, using Benoist's formula gave good agreements for the ratio $\tilde{D}_{\parallel}/\tilde{D}_{\perp}$ with the measurements for both the cases of normal and voided cells. By these experiments a rather good confidence is gained in the methods used. As noted by Beck et al. [16] the diffusion coefficients themselves do not change by more than a few percent whereas the change in the diffusion coefficients can vary by as much as 60 % depending on the model used for the calculation of the diffusion coefficients. There is doubt [16] whether the streaming corrections can be obtained with an accuracy better than 10 % as this requires the individual diffusion coefficients of the normal and the voided cores to higher than 1 % accuracy. A 20 % uncertainty in the prediction of the overall heterogeneity effect (including streaming) appears to exist, this giving an uncertainty of about 5 % in the total void reactivity effect.

In addition to the plate/pin-cell heterogeneity there exists another heterogeneity effect in power reactors due to smearing out the subassembly wrappers. With streaming being neglected, Grimstone and Butland [17] quote about 10 - 15 % increase in total sodium void effect if the wrapper is treated heterogeneously. This effect will be reduced by the inclusion of the streaming effects. As a general statement, the modelling of the unit cells in the normal and voided configurations have to be done very carefully because of the sensitivity of the corrections cited. Slumped core configurations and other material redistributions, which may arise in accident situations, have been studied by Kaiser et al. in ZPPR-5 [8]. Based on diffusion theory with transport- and heterogeneity-corrections the experiments could be interpreted satisfactorily.

4.6 Temperature Effects

In accident situations due to sodium voiding the fuel temperature is drastically increased. The temperature rise causes the sodium void reactivity effect to increase due to reduced resonance selfshielding mainly in the fertile isotope U-238, thus increasing the capture rate. Uncertainties with regard to a reliable prediction of this effect have its origen in uncertain resonance parameters, especially in the unresolved resonance range. The change in resonance selfshielding is usually taken into account by the well established Doppler broadening procedure for heavy isotopes and has been checked carefully in Doppler sample measurements and especially in the SEFOR tests. The interpretation resulted in a ± 20 % uncertainty for the Doppler reactivity effect. The temperature range was restricted to 2000°K. It is not expected that the further increase in temperature to 3000°K or even larger might disturb severely the satisfactory agreement. For the sodium void reactivity effect the change from operating fuel temperatures (∿ 1800°K) in the normal condition to about 3000°K or more in parts of the core before disassembly results in an increase of about 20 %. If we assume an uncertainty of about 20 % for this effect, the additional uncertainty in the net sodium void effect due to temperature rise is below about 5 %.

4.7 Burn-up Effects

The extrapolation of the knowledge of sodium void reactivity gained in the fast critical assemblies to conditions existing in operating power reactors has an uncertainty due to the progressive replacement of control rod absorber by fission product absorption through a burn-up cycle. The experiments performed and reported so far do not lead to a firm conclusion on the influence of accumulating fission products on the sodium void effect. A major difficulty in these experiments is the availability of an appropriate material which can effectively simulate the fission products accumulation in a fast power reactor core. According to Koyama et al. [18] the experimental results obtained in the FCI VII-2 Assembly have shown that the central sodium void worth was affected up to 15 % by accumulation of fission products while those of the other materials were not sensitive to the burn-up. The main uncertainty in the prediction of the sodium void effect in high burn-up cores comes from two sources: (1) in representing the many isotopes by pseudo-fission product groups and (2) by the inaccurately known absorption data. In the investigations done in the Netherlands at the STEK-facility [19] average cross sections were adjusted and there is now more confidence in their reactivity effect, a standard deviation of about 5 to 10 % in the average group absorption cross section for the main fission products being quoted. This is in agreement with the results quoted by the French team [1]. At present the related uncertainty on the net sodium void effect has been discussed to be below about 10 % (1σ) for the effects of fission products [1].

The effect of higher Pu isotopes on the sodium void effect is well known [5]. The related uncertainty due to Pu-240 will be taken up below

in a special reinvestigation of an experiment in SNEAK.

5. EXPERIMENTAL UNCERTAINTIES IN THE DETERMINATION OF THE VOID EFFECT

Only a very brief summary will be given here. The sodium void reactivity is experimentally determined by observing the difference in the positions of the calibrated control rods in the unvoided and the voided states of the reactor. For those voided steps for which the reactor is subcritical, different methods such as subcritical source multiplication technique, polarity coherence technique, or inverse kinetics analysis are used to obtain the reactivity worth of the voids. The experimental error associated with the measurements of the sodium void effect arise due to temperature drifts, Pu-241 decay, gap closure corrections in some assemblies and those arising from the variation of the loaded plate thickness in the voided case due to the compressive load and slight geometrical dislocations of the platelet arrays. For very small central voids, the correction due to temperature drifts appears to have the same magnitude as the void effect itself, thereby making the interpretation very difficult. The experimental uncertainty is quoted to be nearly independent of void size and is of the order of $\pm$ 0.3 to $\pm$ 0.5 ¢ [1].

6. PRESENT STATUS OF THE PREDICTION OF THE VOID-REACTIVITY EFFECT

For conventional fast reactor cores the discussion in section 4 showed that about ± 15 % uncertainty in the prediction of the net sodium void effect might be dedicated to presently existing basic nuclear data uncertainties, as far as voiding in inner core zones is considered.

In voiding outer core zones and blanket regions, the discrepancies between various predictions become larger, even in the case of prescribed standard methods. Part of this discrepancy can be attributed again to data uncertainties, but part of it also to calculational procedures (e.g. use of zone-independent group cross sections, insufficient convergence of flux and adjoints in these regions, different schemes in solving the difference equations, different tratment of anisotropic scattering of fast neutrons, etc.). Due to the fact that the void reactivity effect results from partially compensating contributions, and due to the fact that the magnitude of the void effect is small in outer core and blanket regions, the relative errors can easily reach large values. The safety assessment is dominated by the reactivity changes (besides the non-neutronic properties) in the inner core zone mainly.

As fas as the interpretation of sodium removal experiments in critical assemblies is concerned, additional uncertainties arise by the methods used to describe the corrections to the standard procedure (e.g. improvement over diffusion theory, cell heterogeneity including streaming) and by modelling the various void configurations. Again the related uncertainty depends frequently on the sophistication, which the

investigator is willing to apply in the interpretation of the experiment under investigation, sometimes being also limited by available computer time and related budget. In using the appropriate set of methods available it seems that a related uncertainty of ± 10 % is not unreasonable; for the modelling uncertainty to the void effect we put here a somewhat arbitrary figure of about 10 %. Combining the various uncertainties, we arrive at an uncertainty in the prediction of the net void effect in inner core zones of fast reactors to slightly higher about ± 20 % (1σ) at present.

It is interesting to compare this figure with the presently achieved accuracy in predicting the void effect in critical assemblies. The experience in the U.S.A. [1,16], UK [17], USSR [20], Japan [1,21], France [1,22] and Germany [23]: There is good agreement for the prediction of central void effects including the maximum value, the accuracy being about ± 15 to 25 %. In off-center areas, where the void reactivity contributions are smaller and become negative, the uncertainty may be larger (sometimes up to about ±30 - 40 %), depending on the complexity of the considered system (e.g. partially inserted control rods etc.) and the sophistication in the methods used.

As fas as power reactors are concerned, some of the corrections mentioned are of minor importance, some have to be dealt with additionally and are dependent on assumptions in the accident sequences. If we attribute about ± 5 % uncertainty in the void effect due to a temperature rise to above about 3000°K, and about ± 10 % for burn-up effects including fission products, we arrive at an overall uncertainty in the void effect prediction of about ± 25 % (1σ) for voiding inner core zones under the assumption that appropriate methods are applied.

For nonconventional fast reactor cores with internal fertile zones or subassembly islands the achieved confidence on the prediction of the void effect (lower in magnitude) is not so high at present. Measurements have been performed in the US [24], in France [25] and are underway in the UK in a common British/German experiment [26]. The US results, obtained in ZPPR-7, show satisfactory agreement between theory and experiment. The C/E value lies between 1.05 and 1.21, not taking into account the proper treatment of neutron streaming in the voided case. The consideration of neutron streaming will reduce the theoretical values by less about 15 %, so that indeed for this type of clean physics heterogeneous configuration no major discrepancy is left. Quite on the contrary first French results, obtained in the Pre-Racine assembly of MASURCA, show an overestimation of the central void reactivity effect by about a factor of two. Adjusted cross sections in the CARNAVAL IV System have been used. This discrepancy will be discussed in the next section.

In principle, the theoretical prediction of the void effects in nonconventional cores should not show up any surprises. Difficulties arise because the net void effect is smaller and the partial contributions have to be calculated rather carefully; in addition, its strong spatial variation in fertile and fissile regions plays a more important role; furtner, the effect of the build-up of fissile material in the fertile regions on the void effect requires attention from the theoretical side.

The existence of a large misprediction of the void effect, outside the uncertainty range quoted here, either for conventional or nonconventional "clean" experiments, most probably reveals the incompleteness of the analysis performed. This is demonstrated for two examples in the next section.

7. RE-EXAMINATION OF EXPERIMENTS SHOWING LARGE DEVIATIONS FROM THEORY

As a first example we re-investigated the sodium void experiments in an environment of high Pu-240 content, performed at Karlsruhe in SNEAK-9C [27]. The experimental arrangement was as follows: (1) In the test zone POZ ZEBRA Pu-metal plates with 19 % Pu-240 in the 50 central core elements were used. The reactivity effect of axial voids of increasing height in 4 elements was studied. (2) In a separate experiment in the test zone C, 50 core elements simulated a carbide fuel composition using the ZEBRA Pu-metal plates together with graphite. Axial voids of increasing heights in 4 elements were studied in this core also. In our theoretical reinvestigation of these experiments, we included: the correct simulation of the platelet arrangement in the unit cell and a proper treatment of the numerical procedures for calculating the low signal void effect. Further, the increase of the capture group constants for Pu-240 by 30 % due to more recent and confirmed differential measurements resulted in a correction factor of 1.05. These improvements resulted in a C/E value of 0.75 for the oxide core. We also noticed that the use of the proper spectrum of POC core obtained by a space dependent fine group calculation instead of the standard KFKINR group set resulted in a correction factor of 1.4 for the carbide core. This factor is derived as follows. For a certain void configuration, the KFKINR set gave identical void effect for both oxide and carbide cores. (Also for other void sizes the differences were not significant.) We carried out a detailed fine group calculation both for the normal and the voided states of the carbide and the oxide core. This revealed an increase of 40 % for the void effect in the carbide core relative to that in the oxide core for the identical void configuration. This correction was not applied for the oxide core in view of the satisfactory experience [23] in using KFKINR set for the analysis of void effects in oxide cores. The results are summarized in the following table.

Table 1. A Re-investigation of the Maximum Measured Sodium Void Effect in High ^{240}Pu cores at Karlsruhe (Axial Voids in Four Elements)

Assembly	Core Height	Maximum for Void Height	Measured Effect ¢	Earlier Analysis $\frac{C}{E}$ (Ref. 31)	Present Results $\frac{C}{E}$
9C-2/POZ	60.2	39.6	3.45	0.5	0.75
9C-2/POC	58.56	37.5	5.97	0.39	0.51 0.72[a]

a - Includes a correction factor for spectral effects (see text).

It can be seen that the large discrepancy in the earlier analysis is practically removed. If one would use the proper eigenspectrum for the oxide core, the agreement very probably could be improved further and would be worthwile doing, if, on the other side, the experimental uncertainty could be confirmed.

In the second example we tried to find out the reason for the over-prediction of the void effect in the central fertile zone in the Pre-Racine experiment by about a factor of 2. In their analysis [25], the central fertile zone was homogenized. Homogenising the fertile zone leads to an underestimation of the leakage component without significantly affecting the positive spectral term. This leads to an over-estimation of the net sodium void effect. To demonstrate this, we carried out a simulation calculation as follows: We took the SNEAK-9C-POZ core and inserted in the centre a fertile zone of 6.14 cm radius. The size of the outer fissile zone has to be slightly increased to obtain a critical heterogeneous configuration. We voided the central fertile zone fully and also partly in our simulation calculations. The calculations of sodium void effect taking heterogeneity of unit cell in the fertile zone and the calculation using completely homogenized fertile medium are given in Table 2.

Table 2. Calculation of Na Void Effect in the Fertile Zone of a Heterogeneous Configuration of SNEAK-9C-POZ Core

Size of the Void	Use of Heterogeneous Diffusion coefficient	Quasi Homogeneous calculation
R, Radius = 3.07 cms H, Axial Height = 39.5 cms	0.394×10^{-4}	0.668×10^{-4}
R = 6.14 cms H = 50.89 cms (covers the Full Fertile Zone)	-0.339×10^{-3}	$+0.395 \times 10^{-3}$

As seen from Table 2, an overestimation of the net void effect by about a factor of two results from neglecting the proper heterogeneity effect.

These two examples confirm that in principle the predictional capability of the present tools available should not lead to rather large deviations from experiment for the sodium void effect.

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