PRESENT STATUS AND FUTURE DEVELOPMENTS OF THE IMPLEMENTATION OF BURNUP CREDIT IN SPENT FUEL MANAGEMENT SYSTEMS IN GERMANY

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Abstract

The paper describes the experience gained in Germany in applying burnup credit methodologies to wet storage and dry transport systems of spent LWR fuel. It gives a survey of the levels of burnup credit presently used or intended to be used, the regulatory status and future developments planned, the codes used for performing depletion and criticality calculations, the methods applied to verification of these codes, and the methods used to treat parameters specific of burnup credit. In particular it is shown that the effect of axial burnup profiles on wet PWR storage designs based on burnup credit varies from fuel type to fuel type. For wet BWR storage systems the method of estimating a loading curve is described which provides for a given BWR fuel assembly design the minimum required initial burnable absorber content as a function of the initial enrichment of the fuel.

1. INTRODUCTION

Implementation of burnup credit in spent fuel management systems operated in Germany is required for two reasons. It is intended to:

- Increase initial enrichments by considerable amounts without scrapping existing systems;
- Reduce the frequency of spent fuel shipments to the minimum unavoidable.

The spent fuel management systems concerned are:

- Wet storage of PWR and BWR fuel (and, possibly, of MOX fuel);
- Dry transport of PWR and BWR fuel (and, possibly, of MOX and WWER fuel);
- Disposal (final storage) of PWR and BWR fuel (and, possibly, of MOX and WWER fuel).

Burnup credit for disposal might be required if burnup credit is taken for dry transport, but at the present moment there is no thinking about applying burnup credit to final storage. Applying burnup credit for WWER fuel might be inefficient because of the number of irradiated WWER fuel assemblies presently stored in Germany and because of the fact that there might be serious difficulties in describing axial burnup distributions in criticality safety analysis of spent WWER fuel assemblies. Applying burnup credit for MOX fuel might be inefficient because of the physics of the plutonium isotopes:

- The reactivity of a dry and, therefore, fast MOX system is determined by the total plutonium content of the system;
- As illustrated in Figure 1, in a wet storage system the reactivity changes of MOX and UO_2 fuels with increasing burnup are significantly different.

The attention is, therefore, mainly focused on wet storage and dry transport of PWR and BWR fuel in the following.

2. REGULATORY STATUS AND ACTIVITIES

Higher initial enrichments and lower frequencies of spent fuel shipments are in compliance with the objectives of the relevant German laws and regulations establishing that spent fuel management systems have to be designed and operated in such a way that worker and public exposure as well as off-normal or accident risks are reduced to a minimum, and that the design and the safety evaluation of these systems have to be consistent with established developments in science and technology.

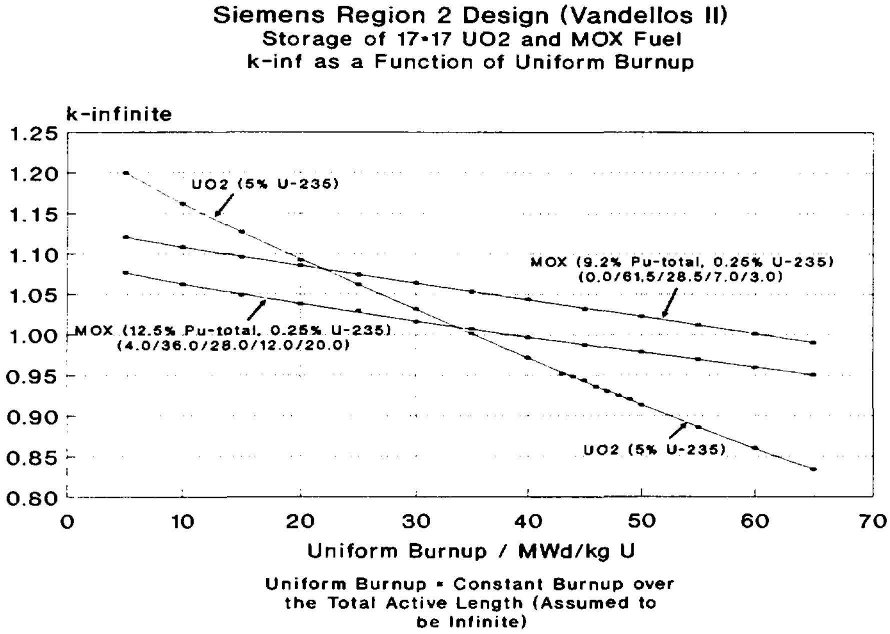

FIG. 1. Neutron multiplication of a wet storage system for loadings with spent PWR UO_2 fuel assemblies having an initial enrichment of 5 wt.% ^{235}U and loadings with spent PWR MOX fuel assemblies having different total plutonium contents and different isotopic vectors.

2.1. Wet storage of LWR fuel

The criticality safety requirements for wet storage of LWR fuel are laid down in the standard KTA 3602 [1]. This code gives permission to take credit for the initial presence of burnable poisons within the fuel, prohibits to consider the presence of soluble boron in PWR storage pools in the evaluation of the normal operation conditions, does not prohibit to take credit for burnup, but requires to give reasons for deviating from the fresh fuel assumption if burnup credit is employed.

To provide obligatory guidelines for justifying the use of burnup credit the Criticality Safety Committee of the German society of standardization - *Deutsches Institut für Normung (DIN)* - is presently working out a criticality safety code for burnup credit in wet storage. This code will probably establish:

- criticality safety criteria being applicable to burnup credit;
- requirements for evaluating parameters characteristic of burnup credit (e.g., axial profiles, irradiation histories);
- methods acceptable for verification of fuel selection and loading.

2.2. Dry transport of LWR fuel

Licensing evaluations of dry transport systems are based on the application of the IAEA regulations Safety Series No. 6.

3. CURRENT AND INTENDED LEVELS OF BURNUP CREDIT

3.1. Wet Storage of PWR Fuel

PWR spent fuel storage racks developed by *Siemens KWU* on the basis of burnup credit were - or will be - delivered to foreign countries, namely to Spain (all PWR plants), Republic of Korea (Kori 3), South Africa (Koeberg 1+2) and Brazil (Angra 2). The design of all of these storage racks is based on the use of net fissile content plus actinide absorbers plus fission products (cf. Table I). This level of burnup credit is required for PWR storage racks for economic reasons.

TABLE I: LEVEL OF BURNUP CREDIT IN WET STORAGE OF PWR FUEL (SEE REF. [2] FOR COMPARISON)

Actinides	Fission Products		
^{235}U	^{95}Mo	^{144}Ce	^{149}Sm
^{236}U	^{99}Tc	^{143}Nd	^{150}Sm
^{238}U	^{103}Rh	^{144}Nd	^{151}Sm
^{237}Np	^{113}Cd	^{145}Nd	^{152}Sm
^{239}Pu	^{131}Xe	^{146}Nd	^{154}Sm
^{240}Pu	^{133}Xe	^{148}Nd	^{153}Eu
^{241}Pu	^{133}Cs	^{150}Nd	^{154}Eu
^{242}Pu	^{134}Cs	^{147}Pm	^{155}Eu
^{243}Am	^{135}Cs	^{155}Gd	
	^{148m}Pm	^{156}Gd	
	^{148}Pm	^{157}Gd	
	^{149}Pm		

3.2. Wet storage of BWR fuel

The spent BWR fuel storage racks developed by *Siemens KWU* for German BWR plants and for the Spanish plant Santa Maria de Garoña are based on a reactivity equivalence concept which provides for the maximum reactivity point in the fuel assembly's lifetime the minimum initial burnable absorber content required for a given fuel assembly design at given average initial enrichment, cf. Figure 2.

To get the reactivity equivalence curve (loading curve) shown in Figure 2 the neutron multiplication of the storage design has to be calculated - for the maximum reactivity point in the fuel assembly's lifetime at given average initial enrichment - as a function of the initial burnable absorber content (cf. Figure 3), and to obtain the maximum reactivity point at given average initial enrichment and given initial burnable absorber content the neutron multiplication of the storage design has to be calculated as a function of fuel burnup (cf. Figure 4). In all of these calculations one has to take account of the parameters (e.g., void history) affecting the maximum reactivity point.

Accordingly, the BWR storage racks based on the reactivity equivalence concept Figure 2 are designed to accommodate BWR fuel at its maximum reactivity point. In other words, taking credit for burnup of BWR fuel is actually taking credit for the initial presence of burnable absorber. This concept is based on the burnup credit level given in Table II.

Due to the fact that the problem of describing BWR axial burnup shapes in criticality safety analysis is unsolved at the present (under economic aspects), it is not intended to take credit for burnups higher than the one which refers to the maximum reactivity point.

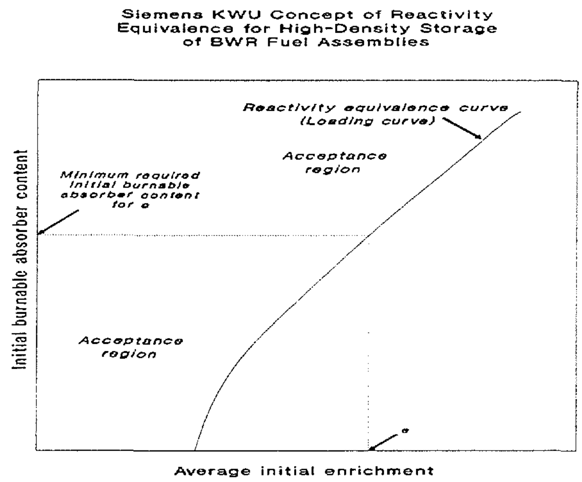

FIG. 2. Reactivity equivalence concept applied to wet storage of BWR fuel assemblies having attained their maximum reactivity.

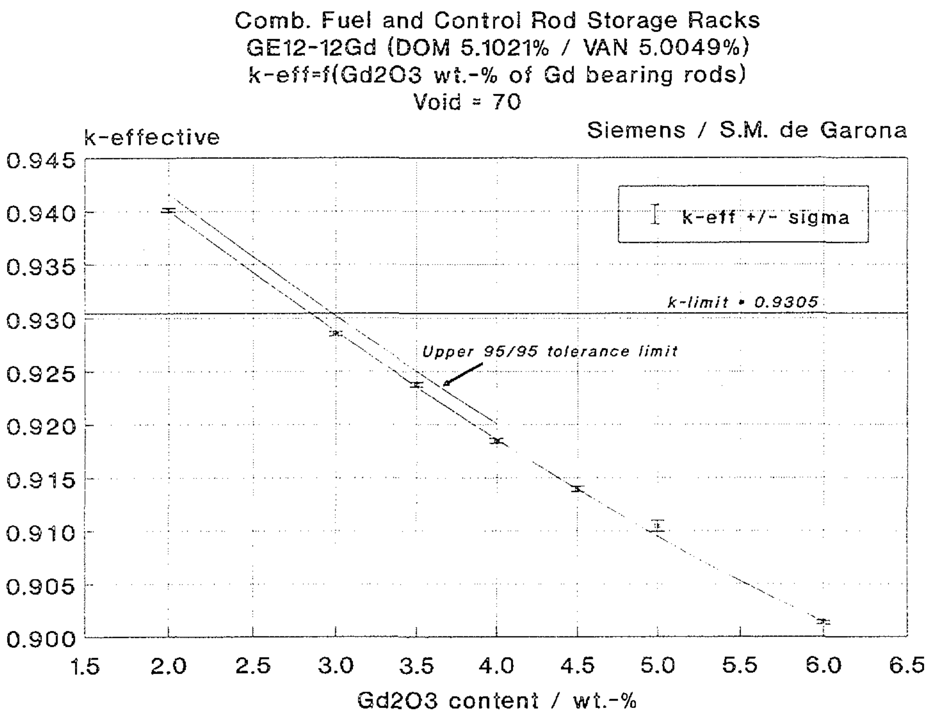

FIG. 3. Illustration of estimating the loading curve shown in Figure 2: The intersection of the upper 95%/95% tolerance limit of the neutron multiplication as a function of the initial burnable absorber content with the maximum permissible neutron multiplication k_{limit} provides one point of the loading curve Fig. 2.

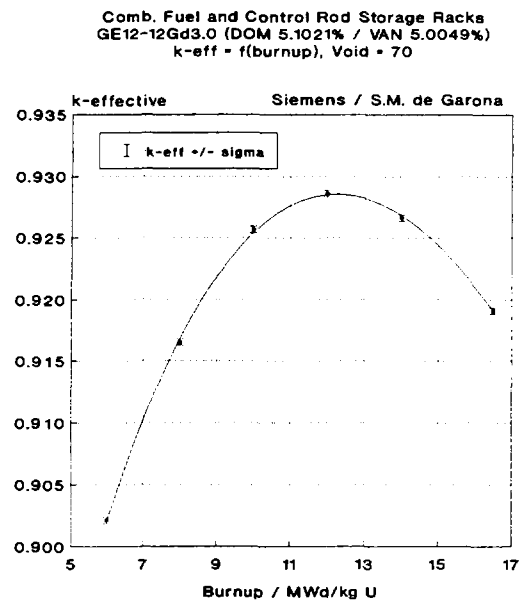

FIG. 4. Illustration of estimating the neutron multiplication as a function of the initial burnable absorber content: The maximum of the curve representing the neutron multiplication as a function of burnup provides one of the $k_{eff} \pm \sigma$ *bars shown in Fig. 3 .*

TABLE II. LEVEL OF BURNUP CREDIT IN WET STORAGE OF BWR FUEL

Actinides	Fission Products	Initial Burnable Absorber
^{234}U	^{133}Cs	^{154}Gd
^{235}U	^{135}Cs	^{155}Gd
^{236}U	^{143}Nd	^{156}Gd
^{238}U	^{145}Nd	^{157}Gd
^{237}Np	^{149}Sm	^{158}Gd
^{238}Pu	^{150}Sm	
^{239}Pu	^{152}Sm	
^{240}Pu	^{153}Eu	
^{241}Pu		
^{242}Pu		
^{241}Am		
^{242}Am		
^{243}Am		

3.3. Dry transport of LWR fuel

The standard casks used for shipping spent LWR fuel are the CASTOR casks developed by the *Gesellschaft für Nuklear-Behälter (GNB)*, Essen. The cask CASTOR V/52 is licensed to accommodate spent BWR fuel with average initial enrichments up to 4.6 wt.% ^{235}U. The licensing evaluation of this cask is based on:

- the fresh fuel approach for initial enrichments up to 4.2 wt.% ^{235}U;
- "uranium plus plutonium isotopes only" burnup credit for initial enrichments greater than 4.2 wt.% ^{235}U.

If the initial enrichment is greater than 4.2 wt.% ^{235}U it has to be ensured that:

- the fuel to be loaded is irradiated (this is ensured by checking the cesium γ dose); and that
- this fuel has a minimum average discharge burnup of 5 MW·d/kg U (this is ensured through the analysis of each fuel assembly's exposure history).

GNB intends to apply this "uranium plus plutonium isotopes only" burnup credit concept to other casks (e.g., the CASTOR V/19 cask used for spent PWR shipping) in order to:

- achieve firstly an increase of the respective maximum permissible initial enrichments by about 0.5 wt.% ^{235}U; and to
- get finally the license to consider the actual discharge burnups of the fuel to be loaded.

4. CALCULATION CODES

4.1. Depletion codes

The standards to be applied to depletion codes and to verifications of such codes are laid down in the safety code KTA 3101.2 [3].

4.1.1. PWR UO2 and MOX fuels

Depletion calculations for PWR UO2 and MOX fuels are performed with the aid of the *Siemens KWU* standard core design procedure SAV90 [4]. This procedure is used for spectrum and nodal reactor calculations as well as for pinwise reactor analysis. Among other data and features this procedure provides:

- the isotopic inventory as a function of burnup;
- axial power and burnup profiles.

The SAV90 procedure is based on broad empirical verification and validation. The experience with this procedure has been accumulated to about 200 first core and reload designs including KWU, Westinghouse and Framatome PWRs. The quality of prediction relies on statistics on the differences between measurement and calculation. To obtain these statistics the following sources of experimental information were exploited:

- Observation and evaluation of normal power operation:
  - Stationary and non-stationary activation rate distributions
  - excess reactivity as a function of burnup measured in terms of soluble boron concentration or control rod position
- Special measurement programmes:
  - Reactivity coefficients and equivalents describing the dynamic behavior of the reactor
  - Short-term (e.g. rod drop) and long-term (e.g. xenon) transients

  In addition to these physics measurements conducted at nuclear power reactors, the nuclide densities of irradiated fuel were analyzed:
- Isotopic inventory of spent fuel.

4.1.2. BWR UO2 and MOX fuels

The isotopic inventories of irradiated BWR UO2 and MOX fuels are calculated with the aid of the code system MICBURN/CASMO [5-6]. The quality of prediction based on the published benchmarks can be supplemented by statistics derived from:

- comparisons to other BWR spectrum depletion codes (e.g. TGBLA, [7]); and from
- experience gained with the aid of off-line core simulation code systems such as CASMO/MICROBURN used by *Siemens KWU* and with the aid of on-line core simulator code systems such as the *Siemens KWU* code FNR-K:
  Comparisons of measured and calculated tip-signals result in an uncertainty of the calculated burnup. This uncertainty can be used to correct the calculated isotopic densities in an enveloping manner.

4.1.3. WWER fuels

Depletion calculations for WWER fuels can be performed with the aid of a code system developed by the *Kraftwerks- und Anlagenbau (K.A.B.) AG*, Berlin-Marzahn. This code system consisting of the modules NESSEL-4, NUKO, PYTHIA/TRAPEZ, DERAB is used for spectrum and nodal reactor calculations as well as for pinwise reactor analysis. Among other data and features this code system provides:

- the isotopic inventory as a function of burnup;
- axial power and burnup profiles.

The verification of the code system is mainly based on:

- Analysis of normal power operation measurement data which have been accumulated to 80 reactor years of German and foreign WWER plants;
- Analysis of test track measurement data;
- Analysis of actinide densities in spent WWER-440 fuel.

4.2. Criticality codes

The standards to be applied to criticality codes and to verifications of such codes are laid down in the safety standard DIN 25478 [8]. The criticality codes mainly employed in Germany are:

- the criticality portion of the SCALE package [9]; and
- the MCNP code [10].

Germany is represented in the Criticality Safety Benchmark Group at OECD NEA by the following institutions:

- Institut für Kernenergetik und Energiesysteme (IKE), University of Stuttgart;
- Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Garching,; and
- Bundesamt für Strahlenschutz (BfS), Salzgitter.

These institutions participate in the analytic activities of the benchmark group. It should be noted that the developers of the SCALE package have made extensive contributions to the burnup credit validation of the SCALE package by analyzing PWR reactor critical configurations [2].

5. PWR AXIAL BURNUP PROFILES

The *Siemens KWU* method used in the recent years for modeling PWR axial burnup shapes is illustrated in Figure 5. The real distribution is modeled by a step distribution. The number of steps is a free parameter because neighboring steps with differences smaller than a given threshold are combined to larger steps.

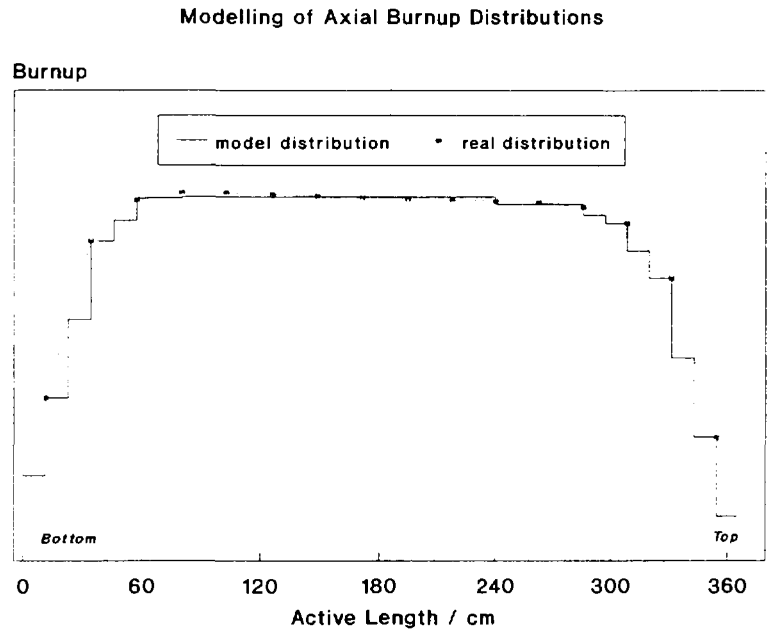

FIG. 5. Typical PWR axial burnup distribution

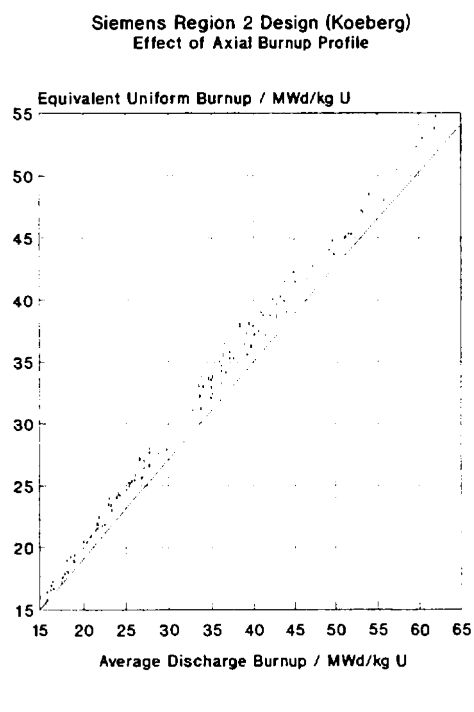

*FIG. 6. Wet storage of spent 17*17 PWR fuel assemblies: Effect of axial burnup shapes on burnup credit.*

Results obtained with the aid of this step distribution model for a PWR wet storage case are shown in Figure 6. Each of the small bars shown in that figure represent an analyzed axial shape based on measured data delivered from the nuclear power plant under examination. "Uniform burnup" means constant burnup over the full active length of the fuel assemblies. The "equivalent uniform burnup" is the uniform burnup which has the same neutron multiplication as the analyzed axial shape characterized in Fig. 6 by its average discharge burnup. If the equivalent uniform burnup obtained for an axial shape is less than the average discharge burnup referring to this shape, then this shape has a neutron multiplication higher than that one which would be obtained with a uniform distribution of the average discharge burnup. So, in the case shown in Figure 6 the effect of axial burnup shapes on the burnup credit is significant.

However, due to different reactor steering strategies and core geometries, this effect might differ from plant to plant. Figure 7 shows an other PWR wet storage case. In that case the axial shapes have no effect on the burnup credit. Therefore, the only conclusion which can be drawn is that it is essential to analyze axial burnup shapes specific of the plant under examination.

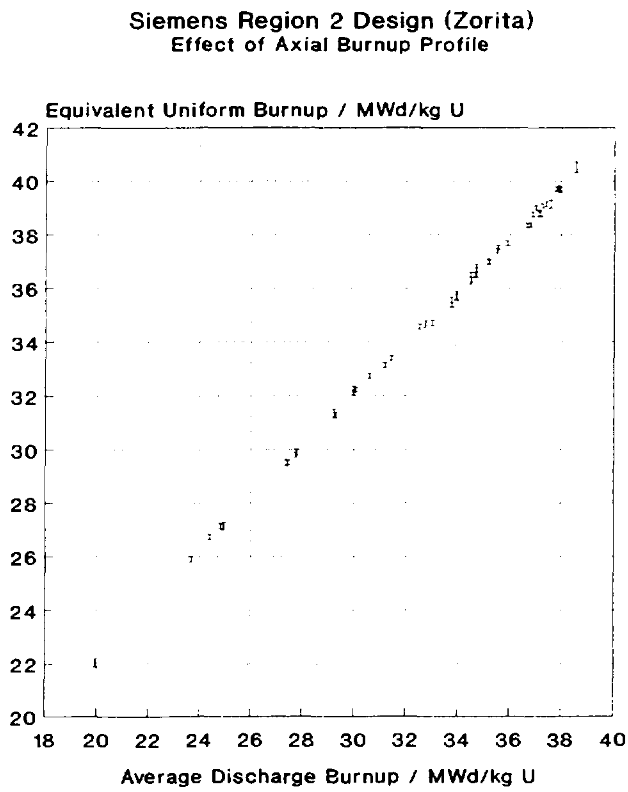

*FIG. 7. Wet storage of spent 14*14 PWR fuel assemblies of the José Cabrera (Zorita) type: In this case the axial burnup shapes analyzed show no effect on burnup credit.*

6. CRITICALITY SAFETY CRITERION

As already told in Section 2.1, a criticality safety standard for burnup credit in wet storage is presently worked out in Germany. This standard will probably establish the following safety criterion:

$$k + \lambda\sigma \leq (1 - \Delta kS) - \Delta kI - \Delta kB - \Delta kM - \Delta kT$$

$k + \lambda\sigma$	:=	upper 95%/95% tolerance limit of the evaluated neutron multiplication
ΔkS	:=	margin of subcriticality:
ΔkS	$\geq$	0.02 for accident cases which are radiological not relevant and which have very small probabilities of occurrence, otherwise $\Delta kS = 0.05$.
ΔkI	:=	uncertainty related to the depletion calculations applied
ΔkB	:=	bias of the criticality code applied
ΔkM	:=	uncertainty related to the effect of axial burnup profiles
ΔkT	:=	uncertainties arising from manufacturing tolerances of the fuel management system.

All uncertainties have to be expressed at the 95%/95% tolerance limit because they are statistics, i.e. random variables defined by probability distributions.

Let's take the term ΔkI for example. It was already stated (cp. Section 4.1.1, e.g.) that the quality of the depletion calculation predictions is based on statistics on differences between measurement and calculation. Figure 8 shows as an example one of the SAV90 statistics on critical boron concentrations. (1 ppm corresponds to about 10-4 in k_∞. The amount of the lower 95%/95% tolerance limit is about 40 ppm).

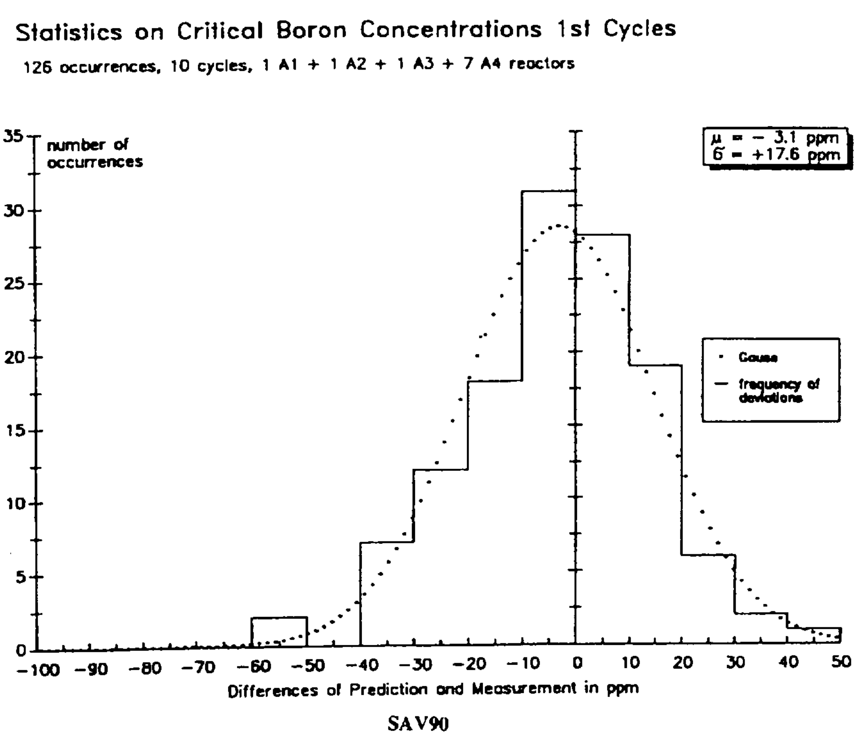

FIG. 8. SAV90 statistics on critical boron concentrations.

7. VERIFICATION OF FUEL SELECTION AND LOADING

According to the above-mentioned draft of the criticality safety standard for burnup credit in wet storage the following methods will be available:

- Measurement of each fuel assembly's reactivity and comparison of the results to the fuel assembly's reactivity referring to the minimum required burnup;
- Measurement of each fuel assembly's burnup or other correlative parameters;
- Analysis of each fuel assembly's exposure history or other correlative parameters to determine its burnup.

All uncertainties inherent to these methods have to be taken into account by deriving respective adequate decision criteria based on a 5% significance level.

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