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INFLUENCE OF THE ACCIDENT BEHAVIOUR OF SPENT FUEL ELEMENTS ON CRITICALITY SAFETY OF TRANSPORT PACKAGES – SOME BASIC CONSIDERATIONS

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ABSTRACT

For each design of a package for the transport of spent nuclear fuel the criticality safety under routine, normal and accident conditions of transport has to be shown. Under accident conditions of transport the neutron multiplication is substantially affected by the condition of the fuel elements and the basket. On the other hand, the impact behaviour of spent fuel elements is not completely known and therefore conservative assumptions are taken for criticality safety assessment. A prudent choice of such assumptions based on a good knowledge of the relevant parameters significantly simplifies the proof of compliance.

The paper starts from recalling the main principles for maintaining subcriticality in a spent fuel transport package. Then these principles are examined systematically regarding possible changes in the arrangement of the fissile material (pin pitch expansion, pin slippage, release and recollection of fissile material). Based on a generic package design the parameters with the maximum influence are identified for each change. Furthermore, the paper gives criteria for determination of the relative importance of the different changes and directions for questions that should be answered in a safety demonstration.

INTRODUCTION

The criticality safety of a package for the transport of spent nuclear fuel largely depends on the condition of the basket and the fuel elements under accident conditions of transport. Since the structural performance of the basket is very design specific, this paper considers only the influence of the fuel elements.

The mechanical properties of fresh fuel elements are well known, but with increasing burn-up the uncertainties rise quickly. Experimental research on the mechanical properties of spent fuel is expensive and requires a lot of time for planning, preparation, execution and evaluation. There have been carried out several such experiments worldwide (see e.g. [1]), but the number of samples and the parameter space covered is still very limited. Due to the various designs, materials, operational and storage conditions it is not realistic to describe the behaviour of the spent fuel elements in detail. Therefore, the approach of starting from an exact knowledge of the behaviour of the spent fuel elements and assessing the criticality safety on this basis is not readily practical. Usually the missing knowledge has to be replaced by conservative assumptions, but too conservative assumptions can also prevent the proof of criticality safety.

In order to derive more information about the influence of the behaviour of spent fuel elements on the criticality safety of the package this paper starts from the neutron physical end of the safety demonstration, postponing the mechanical investigation for the time being. Recalling the main criticality safety principles possible influences from the rearrangement of the fissile material can be assessed. Arguments from the point of view of construction of the fuel elements and the packaging are included as a second step. Conclusions from this approach are demonstrated.

SAFETY CONCEPTS FOR MAINTAINING SUBCRITICALITY IN PACKAGES FOR TRANSPORT OF SPENT FUEL

Packages for the transport of spent nuclear fuel have thick walls and lids thus limiting the neutron interaction between packages in an array. Regarding criticality safety it is therefore usually sufficient to consider a single package consisting of the fuel elements, the basket and the cask walls and lids (mainly acting as a neutron reflector). For focusing on the behaviour of the fuel elements we consider further that the basket remains unchanged during the drop testing. According to the transport regulations, all void spaces in the cask are assumed to be completely flooded with water.

Then the main safety concepts that are used in combination for maintaining subcriticality in a package for the transport of spent fuel are

- Limited concentration of fissile nuclides: The concentration of fissile nuclides in the unirradiated fuel element is determined by the fuel design, and the reactivity decreases further by irradiation in a light water reactor.
- Built-in neutron absorbers: usually the basket retaining the fuel elements is completely or partially made of a material containing a strong neutron absorber, such as ^{10}B. These absorbers may need support from moderating material close to them, often implemented as a gap between opposite absorber plates.
- Limited moderation: Many fuel elements are nominally undermoderated.

Under accident conditions of transport the fuel elements can be shifted and damaged, possibly leading to weakening of the latter two safety measures. In particular, the effect of the built-in neutron absorbers decreases when some fissile material is moved to a region without absorber plates or into gaps between absorber plates (if there are any gaps). The moderation ratio can be increased by increasing the pin pitch, by removing some pins from the lattice and by collecting well moderated fuel released from the fuel elements.

GENERIC PACKAGE MODEL FOR DEMONSTRATING THE REACTIVITY EFFECT OF FUEL DAMAGE

A generic package model was created to assess various changes to the condition of the fuel in a spent fuel transport package. Some basic features for this model were taken from the OECD/NEA burnup credit criticality benchmark phase II-B [2].For better representation of the fuel rod lattice expansion the model had to be modified. Table 1 gives an overview of the most important parameters of the model used in this paper. Cross-sections of the model are presented in Figure 1. All void spaces of the package (other than the gap between the fuel and the cladding) are assumed to be filled with full density water. The fuel hardware is conservatively modelled as full density water as this is done in most criticality safety assessments. The fuel elements are modelled unirradiated to keep the discussion as simple and clear as possible.

Table 1. Important parameters of the package model

Parameter	Value
Fuel element type	17x17-21
Fuel diameter	0.8192 cm
Fuel material	UO_2, 4.5% enrichment
Clad inner diameter	0.8357 cm
Clad outer diameter	0.9500 cm
Clad material	Zircaloy
Height of the active zone	370 cm
Modelled rod length	370 cm
Height of lower/upper fuel hardware	10 cm / 30 cm
Lower/upper fuel hardware modelled as	Full density water
Basket cell inner width	23 cm
Thickness of basket cell walls	1 cm
Material of basket cell walls in active zone	Stainless steel with 2% B
Material of basket cell walls not in active zone	Stainless steel
Gap between opposite basket cell walls	2 cm
Gap between basket and cask lid	10 cm
Cask body inner diameter	157 cm
Cask body inner height	420 cm
Cask body wall thickness	30 cm
Cask body bottom and lid thickness	30 cm
Cask body and lid material	Stainless steel

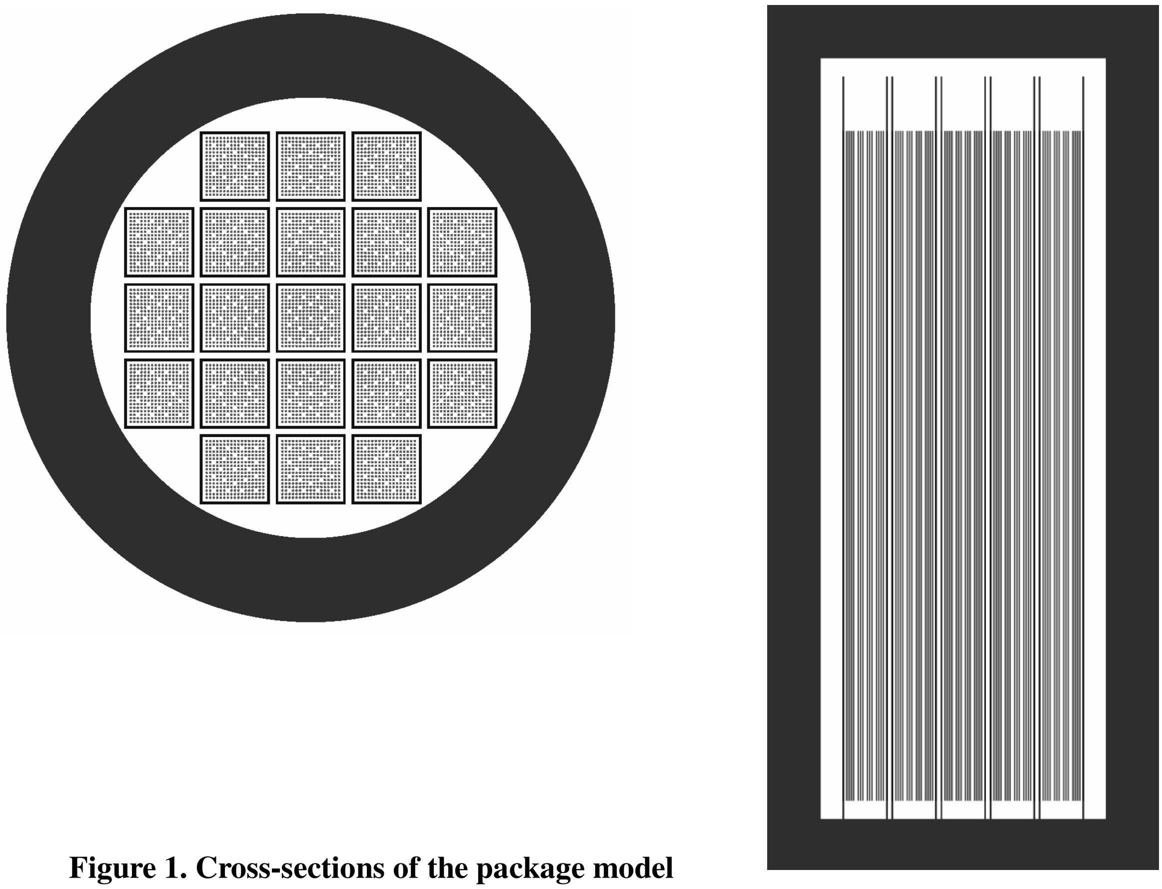

Figure 1. Cross-sections of the package model

All calculations were done with SCALE 5.1 using the sequence CSAS26 with CENTRM and the V6-238 library [3]. The statistical uncertainty of the calculated k_{eff} (1σ) was always below 10^{-3}.

CRITICALITY SAFETY IMPORTANCE OF VARIOUS FORMS OF DAMAGE TO THE FUEL ELEMENTS UNDER ACCIDENT CONDITIONS OF TRANSPORT

Moderation Change due to Fuel Rod Lattice Expansion

Since the fuel elements nominally are undermoderated, a fuel rod lattice expansion increases the reactivity of the system. Such a fuel rod lattice expansion may basically occur due to the bending of the fuel rods during a drop of the package when the fuel element is in a vertical position. The reactivity increases with increasing fuel rod pitch and increasing axial length of the expansion. Since the exact bending curve of the fuel rods is not known the disturbed area is usually modelled by an expanded lattice of straight rods. Using this assumption, the influence of the increased rod pitch and the length of the expansion on k_{eff} for the package model defined above is shown in Figures 2 and 3. The effect can be important and should always be considered.

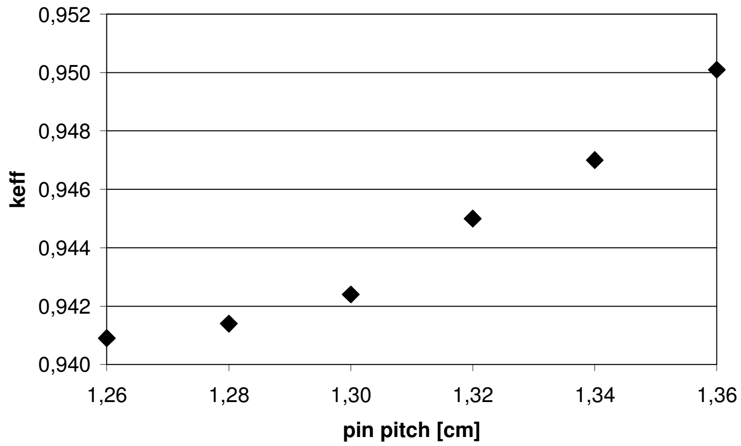

Figure 2. Influence of the lattice pitch in the expanded region on k_{eff} (lattice expansion over an axial length of 50 cm)

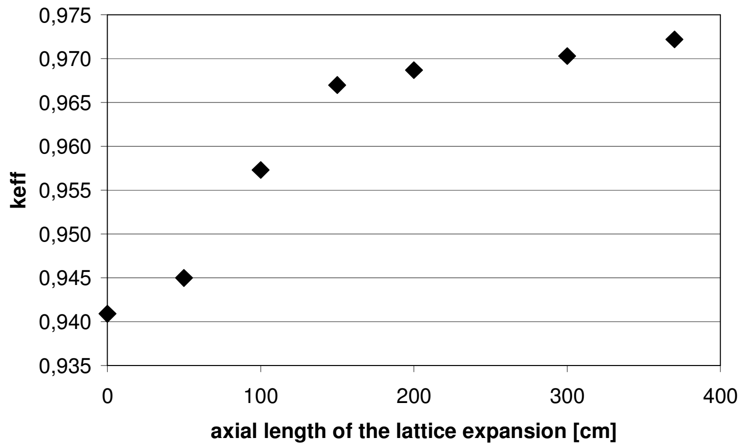

Figure 3. Influence of the axial length of the lattice expansion on k_{eff} (lattice pitch in the expanded part 1.32 cm)

Moderation Change due to Axial Shift of a Small Number of Fuel Rods

Since the fuel elements are undermoderated detaching some fuel rods from the lattice can enhance neutron multiplication by increasing the moderation ratio. Fuel rods can be removed from a part of the fuel element by axially shifting them out of this zone. In the case of the package model defined in this paper some rods could be shifted onto the lid end of the package, thus optimizing the moderation near the bottom of the active zone.

However, regarding the shift of fuel rods the model differs significantly from the real cask. A real fuel rod is longer than the active part of the fuel. To increase the moderation ratio in the active part of the fuel some fuel rods must be shifted by more than this fuel free length. Additionally, for a significant rise of the neutron multiplication factor the moderation must be enhanced at a length of at least 20 cm. This means that altogether some rods, which must be well spread over the fuel element section, must axially move more than about 30 cm into the zone of the fuel hardware with the other rods staying in place. This scenario seems very unlike.

Additionally, the sliding of some fuel rods changes the moderation ratio in about the same way as the lattice expansion. Therefore the two cases can be compared with each other. If , depending on the construction of the package, there is enough space for lattice expansion to result in optimum moderation rod sliding will clearly not lead to higher k_{eff}.

Reduced Absorber Effect due to Axial Shift of Fuel Elements into a Basket Region without Absorber Plates

If whole fuel elements or the majority of their rods can slide into a basket region without neutron absorber plates, the reactivity can rise significantly. If in the package model defined above all fuel rods are shifted 30 cm to the lid into the region where the basket cell walls are made of unborated stainless steel the effective multiplication factor increases from 0.9409 to 0.9823.

Despite this large influence on the neutron multiplication, this case should not be important in reality. At first, this case is not realistic, because the area near the lid where the fuel rods in the model move to is partially occupied by the fuel hardware. The presence of the fuel hardware in this region at least limits the moderation leading to a much lower k_{eff}. Additionally, the movement of fuel rods into a basket region without built-in absorbers can be prevented by an adequate construction of the basket where poisoning extends over the whole axial length where the active zone can be located under accident conditions.

Change in Moderation and Absorber Effect due to Accumulation of Fissile Material Released from the Fuel Elements

Not only the fuel elements can control the neutron multiplication but also a collection of fissile material released from the fuel elements and accumulated in an appropriate place. In this case the reactivity is mainly influenced by the position, mass and moderation of the released fissile material.

If the fissile material accumulates inside the fuel rod lattice (where it is released) k_{eff} of the system does not increase and may even decrease because the additional fuel replaces moderator in the undermoderated fuel element.

The reactivity of the system could rise if a sufficient amount of released fissile material were collected in a free space inside the package. This requires a large void volume: for 5% enriched uranium the smallest (spherical) critical volume is about 22 litres, requiring further reflection. If large geometrically unfavourable void spaces can be avoided during the construction of the cask and the cask is fully occupied by fuel elements the possible collection of released fuel should not introduce problems. In the cask model created in this paper a collection of released fuel in the 10-cm gap between the upper fuel hardware and the lid is investigated. Assuming that an unlimited mass of released fissile material is available, the maximum k_{eff} obtained is 0.9525.

There are other spaces in the model cask where an accumulation of an unlimited mass of fissile material compromises subcriticality. This could be avoided by a better construction of the packaging. Otherwise a permissible upper limit for the release of fuel could be deduced from the criticality calculations, combined with a proof that this release limit bounds the released mass obtained from investigating the mechanical behaviour (including all uncertainties).
If the criticality safety of the package relies on flux traps (as in the cask model defined above) the released fuel could replace some water from the gap inside the flux traps rendering the flux traps less effective. On the other hand it is unlikely that a sufficient amount of fuel released from the fuel rods gets into this gap in such an optimized way that the reactivity is significantly changed. In any case this scenario is very design specific and will not be discussed here in more detail.
If the released fissile material cannot form a (near) critical configuration on its own a collection of the released fissile material at the end of the active zone could be considered – in this case it could interact with the fissile material in the fuel element. With the package model created in this paper such a case could be studied by adding uranium oxide to the water above the active zone (leading to a supercritical system at optimum accumulated fuel mass and moderation). However, in reality this region is occupied by the fuel hardware, therefore the moderation is limited leading to much less neutron multiplication.

CONCLUSIONS

The criticality safety of a spent fuel transport package can partly be assessed without exact knowledge about the accident behaviour of the spent fuel elements. Starting from a consideration which principles and main parameters criticality safety depends on and taking into account a suitable construction of the packaging it is possible to find out what information about the spent fuel elements is necessary.
If enhanced moderation due to most unfavourable expansion of the fuel rod lattice compromises subcriticality, the deformation behaviour of irradiated fuel rods and spacers should be investigated in more detail. On the other hand, if the construction of the basket allows to show criticality safety assuming the worst deformation possible, no mechanical information about deformations is necessary.
If there is no appropriate place in the package for collecting fissile material with parameters leading to an unacceptable neutron multiplication, the exact amount of fuel possibly released from broken fuel rods need not be investigated. In other cases the permissible uncertainty in knowing the mass of fuel that can be released can be deduced from neutron physical considerations.
Criticality safety assessments of packages for the transport of spent nuclear fuel therefore should explain clearly to which extent they depend on information on the behaviour of the irradiated fuel elements in the package under accident conditions of transport.

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