PLUTONIUM DISPOSITION/RECYCLING IN PWR´s - GERMAN EXPERIENCE IN DESIGN, LICENSING AND VALIDATION

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ABSTRACT

The MOX introduction in LWRs (PWR and BWR) was started in Germany by Siemens with design beginning in the early 70´s. The process of commercial utilization of Pu recycling was based on these designs and initial MOX insertion at the Obrigheim-plant KWO (PWR) and the Gundremmingen-plant unit A KRB-A (BWR). Experiments with MOX fuel assemblies (FA) have been conducted under fully realistic power reactor conditions for neutronic/nuclear and fuel/technological aspects to fulfill the needs of testing and validation of applied methods.

The validated design procedures and experimental results have been used in the design and licensing process for all other German reactors. The licensing requirements to insert MOX are fully in compliance with licensing for other new FA-types.

Up to now the acquired know how was used for licensing 10 PWRs and 2 BWRs in Germany for commercial MOX use. The designs cover MOX-FAs of 14x14 to 18x18 type lattices for PWRs as well as for modern 9x9 and 10x10 type MOX-FAs for BWRs. The averaged contents of fissile plutonium (Pu_{fiss}) has been increased in several steps up to 5.85 w/o Pu_{fiss} at present to match the higher enrichments of U-FAs and the degradation of Pu isotopic composition versus burnup.

The needs for validation are discussed. An overview on the existing validation possibilities is given and some specific examples are treated explicitly.

Finally an outlook is given on the disposition of excess weapons grade Pu in existing VVER-1000 reactors in Russia.

1. INTRODUCTION

Germany has started the commercial recycling in PWRs in the early 70´s with the political intentions to

- Improve the resource utilization and waste disposal
- Avoid proliferation risks and
- Establish the closed fuel cycle, needed also for future fast breeders.

The existing reprocessing contracts of the German utilities deliver significant quantities of Pu also for future recycling. But under the present political situation some of the utilities intend now to dispose their MOX-FAs and to stop further reprocessing.

The experience acquired in design, preparing licensing documents and validation by experiments for the MOX insertion into Siemens built German LWRs can now help to establish the disposition of weapons grade Pu. Contributions in the trilateral cooperation of France, Germany and Russia for the disposition of Russian surplus Pu quantities and to the OECD/NEA "Task Force on Reactor-Based Plutonium Disposition" use the mentioned German experience.

This paper intends to give a survey on the experience on the fields of design, licensing and validation with preference to neutron physics aspects for PWRs. But it includes also fuel aspects and recycling in modern BWRs as far as it can be seen of relevance for the disposition of weapons grade Pu e.g. in Russian VVER-1000 reactors.

2. RECYCLING IN GERMANY

The technology of Pu recycling has been developed in Germany by SIEMENS/KWU since 1966, see reference 1. Mainly on the recycling programs conducted at the PWR plant at Obrigheim (KWO) and the first BWR plant at Gundremmingen (KRB-A) the needs of testing and validation of used methods have been fulfilled.

Any design of MOX fuel assemblies (FAs) and MOX containing cores has to obey the same safety requirements as for normal cores with U(oxide) fuel; Pu-bearing fuel rods and FAs have to meet the same thermal and mechanical limits as specified for U fuel.

Specific additional costs associated with the fabrication of MOX fuel give an incentive to select the Pu concentrations as high as possible under nuclear and thermal limitations. Pu should be concentrated in a minimum of fuel rods, but also in a minimum of MOX-FAs to minimize the additional costs for fabrication and transportation[2].

The neighboring U fuel rods and especially water gaps between assemblies and inner water areas in assemblies (as in BWR) influence the MOX-FA designs. The „all plutonium" assembly is an assembly comprising MOX fuel rods only. This design is clearly more appropriate for PWRs, where the effect of guide-thimbles on flux and power peaking are corrected by an adequate choice of enrichment mapping. The early proposal for KRB-A and actual MOX-FAs for large BWRs use this configuration with advantages, too.

The design of MOX-FAs is such that, in conjunction with the currently normal reload batches and reload strategies, they achieve burnups comparable to U-FAs and do not noticeably alter the length of the cycles.

The designs of MOX-FAs for KWO and KRB-A obey these design requirements. The verification of the design methods was foreseen by specific loading of the MOX-FAs in the reactors and by reactor measurements, usual core follow, post irradiation examinations and some special experiments.

The questions asked and the answers given can be explained on the MOX insertions at KWO in the years 1972-1978[1,3]. A prototype MOX-FA was loaded in 1972 at an outer core edge not sensitive to highest local power to have early access on partly burnt MOX fuel rods. Specific loadings were used to study the local power distribution on the basis of aero ball activation; MOX-FAs being neighboring or at aero ball positions with U-FAs at homologue core positions. From this we learned that we had not to fear unsymmetrical loadings! Further on in some cycles parts of the MOX-FAs were loaded at control rod positions to compare the reactivity worth to avoid restrictions in the MOX-FA positioning.

For an early qualification of the methods on influences by MOX loadings on

- critical boron concentration
- boron worth
- moderator temperature coefficient
- control rod worth
- local power distribution in the MOX-FA

a separate critical experiment at the KRITZ facility was conducted with the original MOX- and U-rods of the KWO reload foreseen for 1973/74[1]. An array of one central MOX- or U-FA was surrounded by the rods of about 6 U-FAs. No significant deviations were found between measurements and calculated results.

The lower reactivity worth of control rod insertion into fresh MOX-FAs - as predicted also by the theory - is without relevance to safety aspects. The shut down margin is at the minimum at the end of cycle, where the low moderator temperature coefficient causes a higher reactivity release at a postulated cool down during a main steam line break accident. This is a significant draw back for all critical experiments as long as they cannot be conducted with burnt fuel.

Control rod measurements at KWO have been conducted at the start up of cycle 5 (1973/74) and cycle 7 (1975/76) to close this gap[1]. By adapted loadings, the control rod worth at different coolant temperatures was measured for MOX- and U-FAs at homologous positions. The results are:

The loss of control rod worth in MOX-FAs

- declines with increasing burn-up
- declines with decreasing moderator temperature.

Further results related to neutronic/nuclear properties were reached by the usual start up measurements and measurements during the MOX containing cycles for reactivity coefficients (e.g. boron worth, combined (moderator and fuel) temperature coefficients) and on the global

power distribution (via aero ball or other in-core detectors). To assess the local power distribution separately from critical experiments γ-scan measurements have been firstly conducted on an MOX-FA of KRB-A at a burn-up of about 6 MWd/kg.

By the extraction of single MOX-rods at different burn-up states, technological aspects were evaluated in addition to verifying nuclear/neutronic behavior by isotopic composition measurements.

The technological investigations are related to the fuel rod and to the fuel itself, and are to some extent dependent on the specifications and fabrication methods. It is firstly to be verified whether the results for U-fuel and U-rods are valid or to what extend deviations can be found for MOX-fuel and MOX-rods.

The measurements on fuel rods can be conducted in the fuel pool at the power plant and in hot cells, and cover mainly the rod dimensional behavior and e.g. the corrosion of the cladding during burn-up. Hot cell investigations are mainly used to calibrate the models of fission gas release and to study the structure of the fuel versus power history and burn-up.

The technique of segmented rods was used to get pre-irradiated MOX rodlets for ramping tests at a test reactor (HFR at Petten) to study the transient behavior.

Besides the just mentioned isotopic composition and burn-up measurements also solubility tests to fulfil the reprocessing requirements were performed on cut probes of irradiated MOX fuel. Relying on the published results from the German recycling programs (see e.g. references 1, 3-6) one can conclude that the validation on MOX covered the whole field which was and is investigated for U fuels, too. No significant deviations were found which could impair the design and the safety of modern LWRs by using MOX.

In figure 1 a survey over the core management experience of SIEMENS KWU Group with commercial MOX Insertion in PWRs and BWRs is presented.

In the first years of the MOX fabrication for use in LWRs several test phases with reprocessable and soluble granulated fuel powders were performed[7]. Until 1981 MOX fabrication started from sinterable UO_2 powder of good flowability. In most cases, PuO_2 was delivered as powder calcined from precipitated Oxalate with very fine particles or was prepared directly from Plutonium-Nitrate solutions. Mechanical blending of such powders gave homogeneity with PuO_2 particle sizes, which prevent hot spots in the MOX. Such fuel met the requirements of operation and exhibited excellent irradiation behavior. Results from reprocessing of such fuels, however, showed an unacceptable residual insolubility in pure nitric acid.

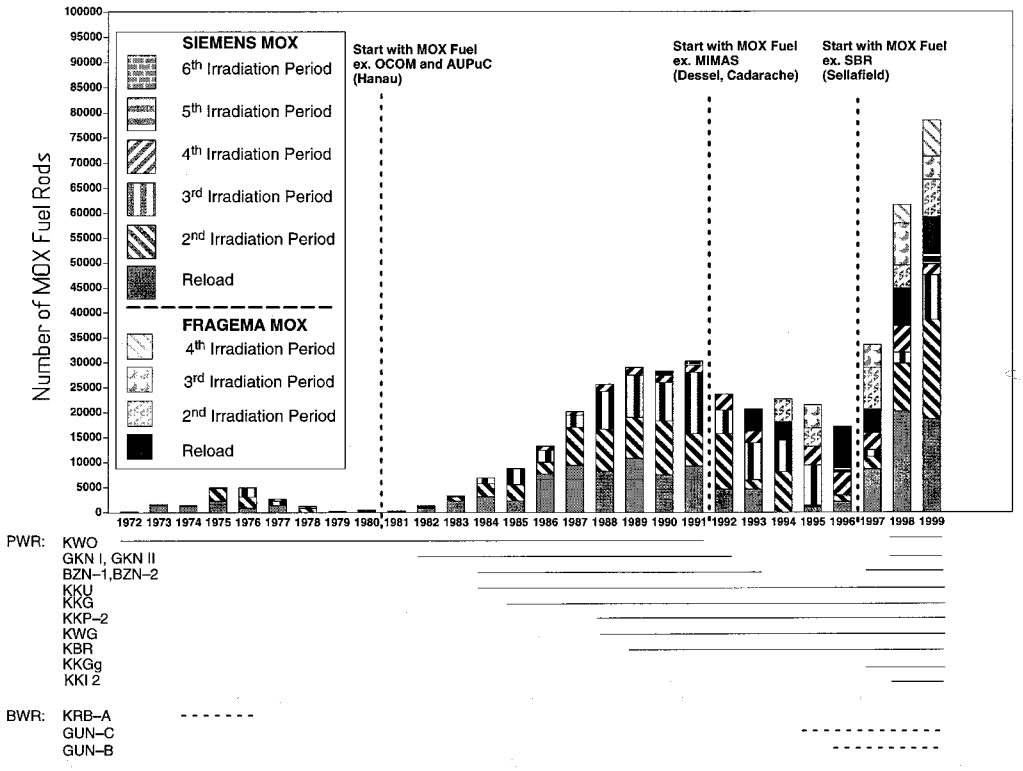

Figure1. Core Management Experience with Commercial MOX Insertion in BWR and PWR by SIEMENS KWU Group (Status: End of 1999)

Therefore new powder preparation routes[8] were developed to meet the required solubility specifications. The OCOM and the comparable MIMAS method avoid pure PuO_2 particles by optimized-co-milling or micronizing a master-mix of U with about 30% Pu, which is blended with pure UO_2 at the final Pu content to achieve complete homogenization of the mixture. The Short Binderless Route (SBR) uses high energy co-milling of PuO_2 and all UO_2 to achieve the complete homogenization of the mixture. The mentioned new MOX fuels fulfill the utility requirements and have been the precondition to start recycling in additional NPPs.

Based on the principles established for former MOX designs of FAs of the types 14x14, 15x15 and 16x16, a "standard" MOX FA was designed for use at five 1300 MW_e power plants (KKU, KKG, KWG, KKP-2, and KBR) which is shown in figure 2. This MOX FA type has been inserted since the mid of 80´s with good operation results.[4]

The average fissile plutonium content (Pu_{fiss}) is 2.91 w/o in three fuel rod types with different Pu_{fiss} contents using natural Uranium (U_{nat}) as carrier material. Four additional water rods (i. e. cladding tubes filled with water in connection with the primary circuit) at the center of the FA increase the moderation there in order to flatten the power distribution.

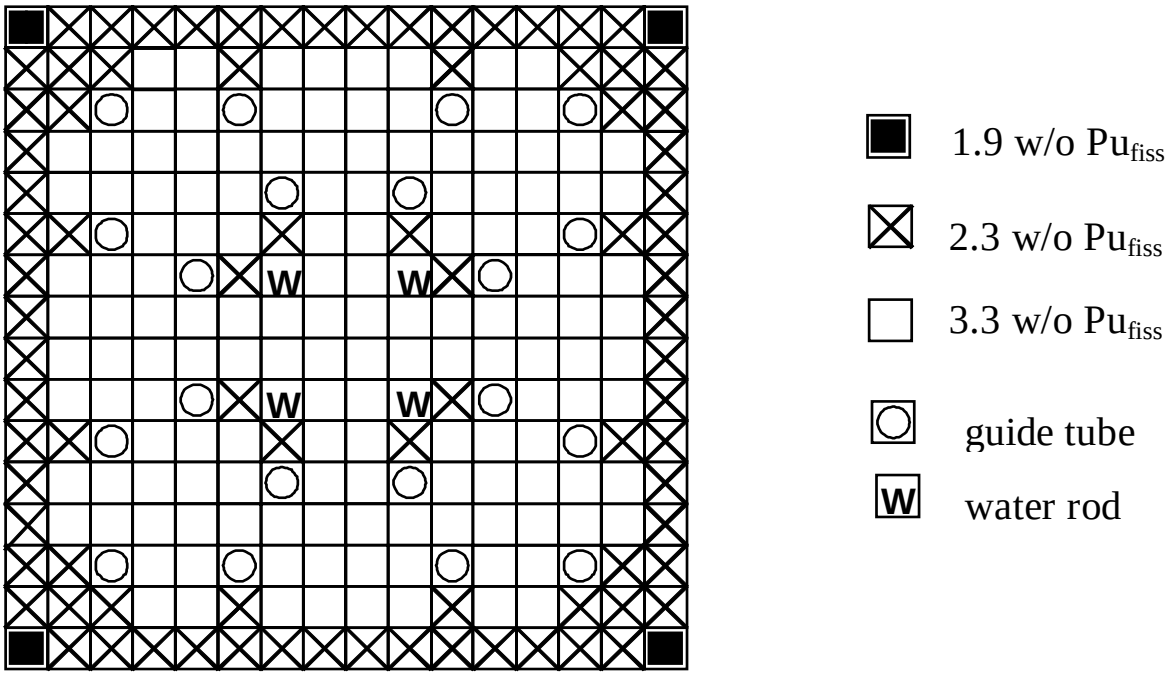

Figure 2. Standard MOX FA (16x16), Carrier Material U_{nat}, 2.91 w/o Average Pu_{fiss} Content

3. LICENSING REQUIREMENTS AND PROCEDURES

Any design of new types of fuel assemblies and their use in (PWR) cores has to obey given design requirements.

The relevant rules and regulations include for Germany to fulfill the requirements of the Atomic Energy Act

- The Radiological Protection Objective, equivalent to the IAEA Safety Objectives
    - General Nuclear Safety Objective
    - Radiation Protection Objective
    - Technical Safety Objective
- The Guidelines by the Federal Ministries
- The Guidelines and Recommendations of the Reactor Safety Commission (RSK)
- The Safety Standards of the Nuclear Safety Standards Commission (KTA)

An example is the KTA 3101 on “Design of Reactor Cores of Pressurized and Boiling Water Reactors”. It is valid for all types of new FAs. No specific rule or regulation was seen as necessary or was implemented for MOX fuel.

Studies and proofs have to be given to fulfill the requirements for licensing purposes on a matrix of topics shown in table I (see e.g. references 4,6). The field of requirements belong to the categories of normal operation and accidents/transients for the core, the fuel storage and for transients (including LOCA and external events). These areas of the analyses cover neutron

physics, thermal hydraulics, fuel rod and fuel assembly structure design, transient analysis and radiological aspects. The parameters which govern these fields of investigations are formulated as so-called Safety Related Boundary Conditions[9] and are a clearly structured and also a complete list of primary and derived parameters comparable to the so-called key safety parameters in use in other countries.

Table I. Safety Evaluations Related to MOX-FA Licenses

Area of Analysis	Categories of Requirements		
	Normal Operation		Accidents
	Reactor Core	Spent Fuel Pool and New Fuel Store	Transients, LOCA, External Events
Neutron Physics	MOX-FA-Design Core Characteristics	Sub-Criticality Decay Heat	Boron Worth Reactivity Coefficients Control Rod Worth
Thermal Hydraulics	unchanged	-	-
System Dynamics	Control Rod Worth	-	as above
Fuel Rod Design	Fission Gas Pressure Corrosion	-	Fuel Rod Failure Limit
FA Structure Design	unchanged	-	unchanged
LOCA Analysis	-	-	evaluated
Radiological Aspects	Activity Inventory	Activity Inventory Release Rates	Activity Releases

Following a presentation of C. Faber[10] of TÜV Bayern Sachsen at TOPFUEL ´95 on “Practice and Trends in Fuel Licensing” four parties are involved in the licensing procedure:

- The utility and
- The fuel designer/manufacturer on one side
- The licensing authority and
- The safety assessors/inspectors (TÜV), engaged by the authority on the other side.

The licensing process includes the following steps:

- Utility/designer inform the licensing authority on the licensing project
- The licensing authority decides, if the consequences of altered operation might affect public interests. Then open public discussion on the basis of a safety report on all safety-related aspects is required.
- Preparation of detailed safety analyses, test programs, etc. by utility/designer
- Examination by safety assessors (TÜV) including answers on additional requests established during examination. The final safety assessment may comprise some conditions/impositions
- Delivery of the license, possibly including some impositions (restrictions, measurements to be foreseen,...)

Central studies and reports to prepare assessment by the safety assessors and the licensing are related to

- Neutronic design of FAs and cores containing a new type of FAs
- Mechanical fuel rod design
- Criticality safety evaluation
- Radiological impact
- Evaluation of transients and accidents
- Thermohydraulics and fuel assembly structure (only in the case of deviations).

To highlight the German situation related to licensing of MOX for PWRs one has to go back to 1972 when a first (lead test) MOX fuel assembly (FA) was inserted at NPP Obrigheim (KWO). Since then individual (different) licenses had been required by the different utilities and delivered by the licensing authorities of the different German states for 9 PWRs related to about 18 different MOX-FA designs (see table II).

Table II. MOX Licensing/Insertion for German PWRs

Station Name		FA-Type	Licenses of Year	Number of different MOX-FA designs inserted
Obrigheim	KWO	14x14-16	1972,1984,1987	3
Neckar I	GKN-I	15x15-20	1982/83	2
Unterweser	KKU	16x16-20	1984,1996	3
Grafenrheinfeld	KKG	16x16-20	1985,1993,1999	4
Philippsburg 2	KKP 2	16x16-20	1988,1997	2
Grohnde	KWG	16x16-20	1986	1
Brokdorf	KBR	16x16-20	1986	2
Isar 2	KKI 2	18x18-24	1988	1
Neckar II	GKN II	18x18-24	1988	1
Emsland	KKE	18x18-24	1988	1, not yet recycling

Some of the mentioned licenses had been updated in the meantime related to higher Pu content and different carrier material with qualified MOX fuel of different MOX fabrication facilities.

4. ACTUAL MOX-DESIGNS

4.1 CURRENT STATUS OF LICENSING IN GERMANY

The present status of MOX licensing for German nuclear power plants has given rise to certain differences caused by the different procedures adopted by utilities and state authorities. An overall-view of the current licensing status of MOX licenses, which are in use or have been granted for German LWRs is given in table III. The numbers of MOX-FAs per reload or the total numbers in the core given in the table are restricted only by the licenses and not by technological limitations.

Table III. Current Status of MOX Licensing for LWRS in Germany

Reactor type	Plant-Name	Status of License	Pu_{fiss}-Content in w/o	Number of MOX-FAs per Reload	MOX-FA-Content in the Core in %
PWR:	KWO	in use	3.8	8	26
	GKN I	in use	3.04	-	9
	GKN II	in use	3.8 [a]	-	37
	KKU	in use [e]	3.5 [d]	16	33
	KKG	in use [f]	3.07 [a]	16	33
	KKI 2	in use	Equivalent to 4.0 w/o ^{235}U	24	50
	KWG	in use [e]	3.2	16	33
	KBR	in use	Equivalent to 4.0 w/o ^{235}U	- [c]	- [c]
	KKP-2	in use	4.65	24	37 [b]
	KKE	granted [e]	Equivalent to 4.0 w/o ^{235}U	16	35
	KWB A	in preparation [e]	Equivalent to 3.5 w/o ^{235}U	24	42
	KWB B	in preparation [e]	Equivalent to 3.5 w/o ^{235}U	24	42
	KMK	in preparation	-	-	-
BWR:	GUN B/C	in use	3.3/4.04	68	38
	KKB	in preparation	-	-	-
	KKK	in preparation	-	-	-

[a] changes in the carrier material and/or Pu-quality can be compensated
[b] temporary restriction
[c] according to the amount of Pu-generation in the plant (up to 16 MOX FAs per reload of 48 FAs)
[d] max. nominal Pu_{fiss} content of a fuel rod
[e] modification or extension in preparation or in licensing procedure
[f] extension to 4.65 w/o Pu_{fiss} granted

As it has been common practice in Germany in the past to license U_{nat} as carrier material, planned changes to depleted uranium could not be compensated under some existing licenses by increasing the fissile plutonium content. In this case a burnup equivalent MOX-FA design could not be realized up to now. Here new licenses are necessary, which are in preparation, in licensing procedure or already granted. Lower Pu quality has to be treated in the same way, where burnup equivalence requires compensation for higher contents of the neutron-absorbing isotopes ^{240}Pu and ^{242}Pu by increasing the Pu_{fiss} content.

4.2 MOX FA DESIGN WITH RAISED Pu_{FISS} CONTENT

The above mentioned standard MOX-FA (see figure 2) had to be redesigned for several reasons:

- The enrichment of U-FAs loaded together with MOX-FAs is increasing. In order to obtain equivalent MOX-FAs, the content of fissionable plutonium nuclides has to be increased too. Usually, this requires a change in geometrical arrangement of the fuel rods with different Pu_{fiss} contents within the FA.
- Instead of U_{nat}, depleted uranium (U_{tails}) is proposed as carrier material. The lower content of the fissionable nuclide ^{235}U in the matrix material has to be compensated by increasing the Pu_{fiss} content. In this case the available plutonium can be concentrated in fewer FAs.
- Changes in the Pu quality are caused by higher burnup of the reprocessed U-FAs. This requires higher Pu_{fiss} contents to compensate for the effects of the neutron absorbing plutonium isotopes ^{240}Pu and ^{242}Pu, respectively.

An extensive survey of the different types of MOX-FA used in German LWRs is given in reference 11 and 12.

MOX-FA designs for PWRs

Designs for all boundary conditions have been established. Examples are given in the following chapters. In all cases of MOX-FAs for PWRs the main features developed by Siemens

- maximum three different fuel rod types (Pu_{fiss} contents)
- water rods in the central region of the FA

are established also international as a proven design.

A first design with raised plutonium content was made for a NPP with 14x14 rod lattice in 1987 to match the increase in enrichment of the reload U-FAs to 4.0 w/o ^{235}U. Eight MOX-FAs of that design were inserted in 1988. Based on the carrier material U_{nat}, an average content of fissile material of 3.8 w/o Pu_{fiss} is used. In this case there is no need for the use of water rods.

A further design was made for a NPP with a 16x16 rod lattice, triggered by the change in carrier material from U_{nat} to U_{tails} with compensation of the lower ^{235}U content by a higher Pu_{fiss} content at the same time. The enrichment of U-FAs has remained unchanged in this case. A FA design with an averaged Pu_{fiss} content of 3.48 w/o (carrier material U_{tails} with 0.25 w/o ^{235}U) was realized.

Further MOX-FA designs have been drawn up for other PWRs on the basis of U-FA enrichments up to 4.3 w/o ^{235}U and are shown in figure 3. For the German 1300 MW_e NPPs with 16x16 rod lattice, a MOX-FA compatible with U-FAs with an enrichment of 4.0 w/o ^{235}U has been designed. With depleted Uranium (with 0.25 w/o ^{235}U content) as carrier material the averaged Pu_{fiss} content of the MOX FA is 4.2 w/o.

For the 1300 MW_e NPPs with an 18x18 rod lattice, MOX-FAs compatible to enrichments of U-FAs of 4.0 w/o ^{235}U have been designed. The neutronic design calculations lead to a burnup equivalent MOX-FA design with an averaged Pu_{fiss} content of 4.6 w/o with the carrier material U_{tails}(0.25 w/o ^{235}U).

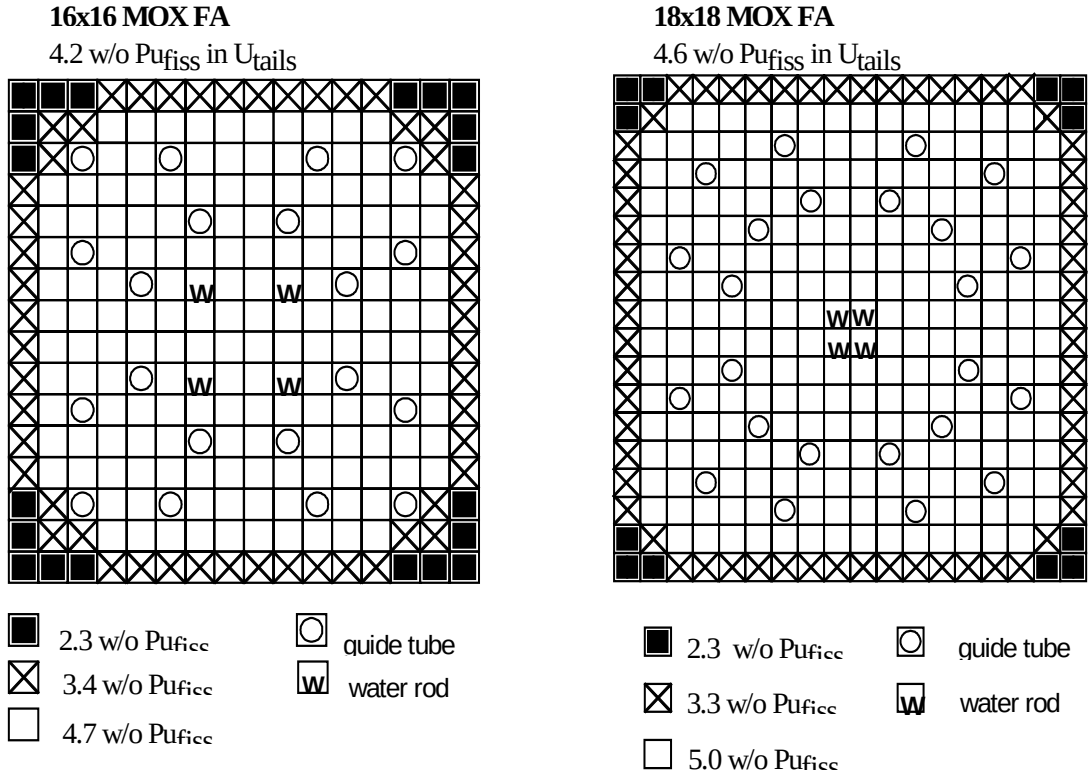

Figure 3. Different MOX-FA Designs for Higher Burnup for 16x16 and 18x18 FAs

For 14x14 rod lattices of Westinghouse type reactors, a newly designed MOX-FA using U_{tails} with 0.25 to 0.30 w/o ^{235}U as carrier material and an averaged Pu_{fiss} content of 4.75 w/o fulfils the required burnup equivalence to U-FAs with 4.25 w/o ^{235}U. Because of the instrumentation tube at the center of the FA it is not necessary to increase the moderation by adding water rods.

For the 15x15 rod lattice type a MOX-FA with the up to now highest content of Pu_{fiss} has been designed. The requirement of burnup equivalence to U-FAs with 4.3 w/o ^{235}U can be fulfilled with an average content of 4.8 w/o Pu_{fiss}, using U_{tails} with 0.25 w/o ^{235}U as carrier material. In the 15x15 rod lattice FA one central water rod is sufficient to flatten the radial power distribution. The MOX-FA designs mentioned above are shown in the figure 4.

MOX-FAs inserted together with U-FAs of an initial Uranium content exceeding 4.3 w/o ^{235}U can be designed, as was demonstrated in design studies for a FA with 14x14 rod lattice type. With an averaged content of fissile Plutonium (Pu_{fiss}) of 5.85 w/o with the carrier material depleted Uranium (U_{tails}), the compatibility to U-FAs with about 4.6 w/o ^{235}U is achieved. First drafts of a 17x17 MOX-FA for the European Pressurized Water Reactor (EPR) result in a comparable content of fissile Plutonium of about 6.5 w/o.

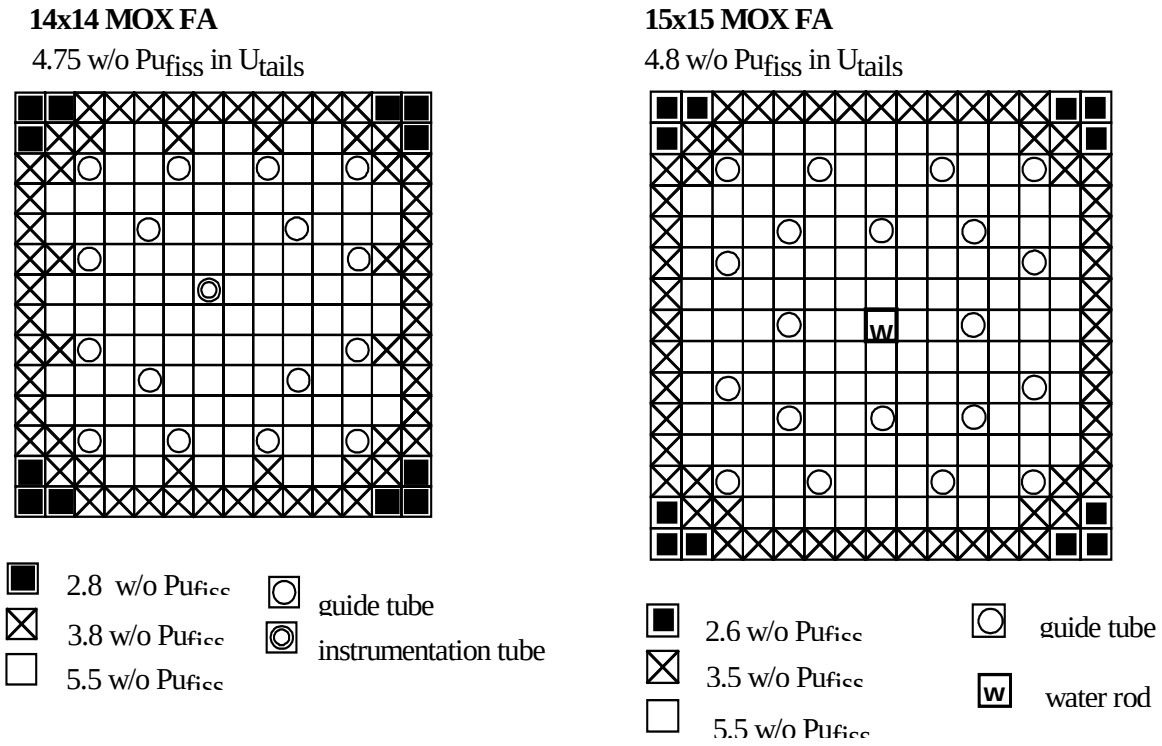

Figure 4. MOX-FA Designs for Higher Burnup for 14x14 and 15x15 MOX FAs

Core performance with MOX-FA in a PWR

An equilibrium core based on full low leakage loading with 88 MOX FAs (45 % of the core) of a 1300 MW_e NPP with 16x16 rod lattice type has been investigated in a licensing study. The equilibrium core has a reload batch of 20 MOX-FAs and 24 U-FAs (majority is Gd poisoned). Based on this study a license was granted for reload of MOX-FAs with 4.2 w/o Pu_{fiss} in U_{tails} of figure 3.

The important cycle characteristics are listed in table IV. The coolant or moderator temperature coefficient (MTC) tends to more negative values for increasing plutonium content in the core. The Doppler coefficient is hardly influenced by plutonium. This is of importance with respect to the shutdown margin.

The boron worth decreases with increasing number of MOX-FAs in the core. The boron control system must therefore handle larger concentration differences during reactor operation. The critical boron concentration must be raised during loading operations to keep the reactor at a required level of subcriticality. The use of boric acid with enriched ^{10}B can increase the effective boron capacity.

As regards the net control rod worth for stuck-rod configurations, the data depend more on the loading scheme than on the MOX-FA fraction in the core. Thus MOX fractions of up to 50% without need for increasing the number and/or the efficiency of control rod system were found possible.

Table IV. U-MOX Equilibrium Cycle (PWR with 1300 MW Electrical Power)

MOX FA loading number / % of the reload batch (core)	20 / 45	
number of reload MOX- and U-FAs	20 / 24	
MOX-FA type	16 x 16	
Pu_{fiss} content in w/o ^{235}U content of carrier material	4.2 0.25	
^{235}U content of U-FAs in w/o	4.0	
		Effect compared with Uranium core characteristics:
cycle length in efpd	315	**Same**
MOX-FA burnup in MWd/kgHM averaged discharge burnup maximum FA burnup	 54 60	**about same**
initial boron concentration (without Xenon) in ppm	1775	**Lower**
boron worth at BOC in pcm/ppm	- 5.5	**Lower** **(U core ~ -8.3)**
MTC at EOC in pcm/K	- 73	**Higher** **(U core ~ -55)**
Net control rod worth at EOC in % $\Delta\rho$	5.6	**lower or same**

The primary operating results include information on cycle length, power distribution, reactivity coefficients, and control rod worth of cores containing MOX-FAs. The reliability of the design methods is validated by measurements of these quantities. The neutron physics experience[7] is based on start-up measurements, in-service cycle monitoring and specific measurements required under licensing commitments.

MOX FA designs for BWRs

Early MOX insertion programs in BWRs started somewhat later than in PWRs and came to an end more or less in the late 70´s. In 1994 a license was granted to Gundremmingen B/C to insert MOX-FAs of the type 9-1 and in 1998 to insert ATRIUM 10^{TM} MOX-FAs.
Therefore for thermal recycling of plutonium in BWRs MOX-FAs of the 9x9 and 10x10 rod lattice type have been designed.

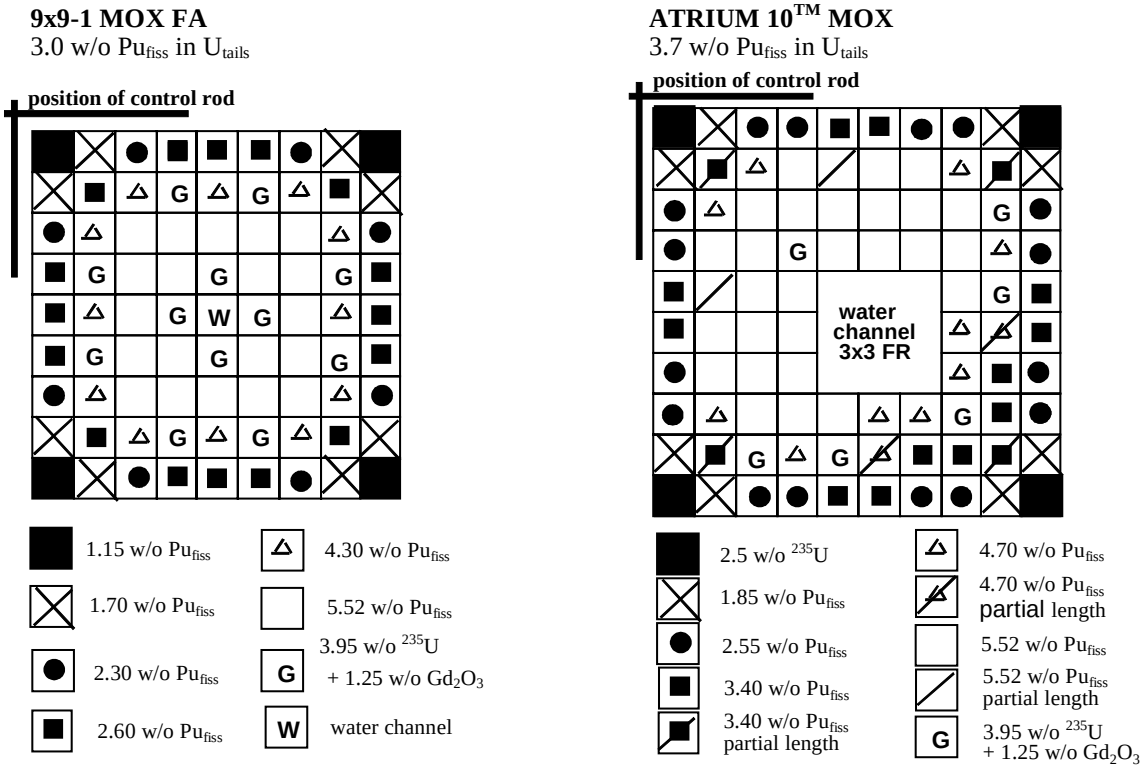

Figure 5. Different MOX-FA Designs for BWRs (9x9 and 10x10 Rod Lattice Type)

The MOX-FA for the insertion in a BWR is in general much more complicated because of the much higher heterogeneity in comparison to PWR. The 9x9–1 BWR MOX-FA contains 6 different MOX fuel rod types and 1 additional Gd poisoned U fuel rod type to avoid power peaks around the water channel and to reduce the initial reactivity. The averaged Pu$_{fiss}$ content is about 3 w/o with carrier material U$_{tails}$ (0.25 w/o ^{235}U). As an example for the progress in design of BWR MOX-FAs a design for a 10x10 rod lattice type (ATRIUM 10^{TM}) has been performed. The 9x9 MOX-FA and the ATRIUM 10™ MOX-FA are shown in the figure 5.

5. VALIDATION

The first MOX design and license had to rely on limited experience gained from national and international know how from experimental programs. So the evaluation of possible differences to usual U fuel with respect to neutronic and fuel aspects was used to show that no significant risk existed for the insertion of a first MOX-FA and immediately following first MOX batches. As no PWR test reactor is available in Germany all neutronic measurements and fuel testing was initiated by the MOX reloads of 1972 and 1973 at KWO[1,3]. Verification and validation possibilities are summarized in the following table V.

Table V. Safety Evaluation Matrix for MOX Fuel and Possibilities for Verification

Evaluation domain	Issues of special concern	Possible verification
Neutronic design of MOX-FA/ MOX containing Cores	Power distribution in the MOX-FA, reactivity values (boron, control rods), reactivity coefficients (esp. MTC)	• (critical experiments) • γ-scan on LTAs • startup and core follow benchmark cases • measurements
Mechanical fuel rod design	Fission gas pressure, corrosion, dimensional behavior, PCI	• benchmark cases • test irradiations/ramp tests • LTA examination
Evaluation of transients and accidents	• steam line rupture • control rod ejection • LOCA • RIA	• benchmark comparisons • RIA experiments
Criticality safety evaluation	Sufficient subcriticality at fabrication, transport and storage	• benchmark activities • critical experiments
Radiological impact	Activity inventory, activity releases	• existing codes (e.g. ORIGEN) • comparison to UOX cases • isotopic composition measurements
Thermohydraulics and FA structure	no, as identical to U-FAs	--

The validation process is continuously ongoing by

- Neutronic start up measurements and core follow of MOX containing cores
- Specific neutronic measurement programs (besides the mentioned early critical experiment at KRITZ e.g. measurements related to shut down margin and local power distribution by γ-scan)
- Fuel pool and hot cell investigations on irradiated MOX fuels
- Specific measurement programs for fuel/cladding behavior (especially ramp testing).

The acquired know how[4-5] was applied for all succeeding design and licensing cases and can be used for the program to utilize weapons grade Pu in Russian VVER-1000 PWRs, too[13-16].

Instead of giving the designer/manufacturers view of MOX specific safety evaluation one can rely on recent publications by safety assessors of Germany[17] and of Switzerland[18]. Both comment on licensing documents prepared by Siemens/KWU for German BWR and PWRs and for Swiss PWRs on all topics of the safety evaluation matrix (table I).

Discussing the differences of MOX compared to U, both publications of safety assessors conclude that no effects were identified during the numerous licensing procedures which would indicate that an operation with MOX fuel were less safe, would afford an alteration in safety systems or even different rules and regulations than an operation with U fuel only.

Commenting on the verification possibilities mentioned in table V one has to mention at first the benchmark comparisons performed by the OECD/NEA "Working Party on the Physics of Plutonium Recycling and Innovative Fuel Cycles – WPPR". Investigations on PWR MOX-cells[19] and for multiple recycling[20] had been presented at GLOBAL conferences. The results of a BWR benchmark for a modern 10x10-9-type MOX-FA will be published soon by the WPPR. At this conference, results related to weapons grade Pu will also be published, e.g. for VVER-1000 type hexagonal fuel assemblies[21]. To conclude on all these benchmarks one can state that the spread of results is sufficiently small but wishes for further improvements are addressed. Only limited possibilities are seen to reach this by additional critical experiments.

To validate the neutron physics as well as fuel aspects versus burnup one can rely on the irradiation of lead test assemblies (LTA) and on initial MOX-FA reloads with a certain extrapolation from existing know how being justified.

To test the power distribution in modern Siemens MOX-FAs of high Pu content γ-scan measurements had been performed on a 9x9-1-type MOX-FA at one of the two Gundremmingen BWRs under international cooperation and on a PWR 15x15-25-type MOX-FA by Siemens with the active support by the Swiss utility Kernkraftwerk Goesgen-Daeniken AG.
The PWR measurements were conducted during the shut down period of the Goesgen plant in 1998 at rather high γ-background. The MOX-FA had an initial Pu_{fiss}-content of 4.8 w/o at 68% Pu_{fiss}/Pu and used depleted U as carrier material. The MOX-FA (see figure 4, right) was inserted at an inner core position nearly without power gradients and had reached a precalculated burnup of 19.1 MWd/kg. The measurements were conducted at 5 planes on

- 3 rods of low Pu content (2.6 w/o Pu_{fiss})
- 10 rods of medium Pu content (3.5 w/o Pu_{fiss})
- 24 rods of high Pu content (5.5 w/o Pu_{fiss})

The individual relative local power was measured via the ^{140}La-line at 1596 keV at a statistical error of 1.5%. The reproducibility was controlled for one of the rods, which was measured also over the full length. As the (theoretical) power shape does not significantly deviate for the different planes of the measurement all planes could be integrated. The experiments are in a reasonable good agreement with the theoretical calculations with the Siemens code system CASCADE-3D[22].

6. WEAPONS GRADE PLUTONIUM IN VVER-1000 REACTORS

Siemens' experience with fabrication and operation of MOX covers different Pu compositions and qualities (see table VI) ranging from MAGNOX type Pu with high ^{239}Pu content of up to 76% to Pu from reprocessing of LWR fuel with a composition of about ^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu = 2/58/25/9/6%. The least fissile composition tested came from the first MOX reprocessing. In all cases a good agreement between pre-calculations and measurements was reached. This should be valid also for weapons grade Pu (W-Pu).

Table VI : Examples of Typical Pu Isotopic Compositions

	Relative Pu isotopic mass fraction in w/o					Pu-quality Pu_{fiss}/Pu in %
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	
MAGNOX-Pu (early recycling)	0.3	74.3	19.9	4.6	0.9	78.9
LWR-Pu (actual MOX)	1.5	60.1	24.5	8.8	5.0	68.9
LWR-Pu in future	2.6	54.0	24.2	11.8	7.4	65.8
2nd generation recycling (test at KWO 1980)	1.3	43.8	34.3	14.2	6.4	57.0
Weapons grade Pu	~0	~94	~5	~1	~0	~95

The properties of W-Pu can be compared and assessed in comparison to commercial or civil Pu (C-Pu) but also to normal U fuel. Related to neutronic design one has especially to take care of the reactivity behavior versus burn-up (see figure 6) looking comparable to U fuel and to the low boron reactivity worth, comparable for all MOX fuels. Use of a reasonable low bounding value for β_{eff} has shown no consequences on acceptable transient and accident behavior.

Based on the German experience initial designs of MOX-FAs and cores were investigated under bi- and trilateral contracts with Russia for VVER-1000 reactors[15-16]. Compared with MOX-FAs based on C-Pu, the MOX-FAs containing W-Pu would have significantly higher initial reactivity and energy release at the beginning of their use at a very high boron concentration in the coolant. This would not allow full low-leakage loading patterns, which are advantageous for the shut down reactivity balance. Acceptable values for maximum fuel rod power and for shut down reactivity can be reached by using U/Gd fuel rods inside the MOX-FAs. Such designs of fuel assemblies are presented and investigated in the mentioned VVER-benchmarks[21].

The lower radioactivity of fresh W-Pu to normal MOX is of positive relevance to fabrication and handling but has no relevance to design aspects. Radioactivity and decay heat of irradiated W-Pu fuel can be expected to be between U and C-Pu fuel.
The mechanical behavior of W-Pu MOX is governed by the Pu content of the fuel. As no higher Pu contents result from neutronic design the properties (as e.g. heat conductivity and melting point) are all in a narrow band defined by U and C-Pu fuel. Until specific measurements on the fission gas release are available the rod design could be done with conservative assumptions for the needed fission gas plenum to fulfil licensing requirements.

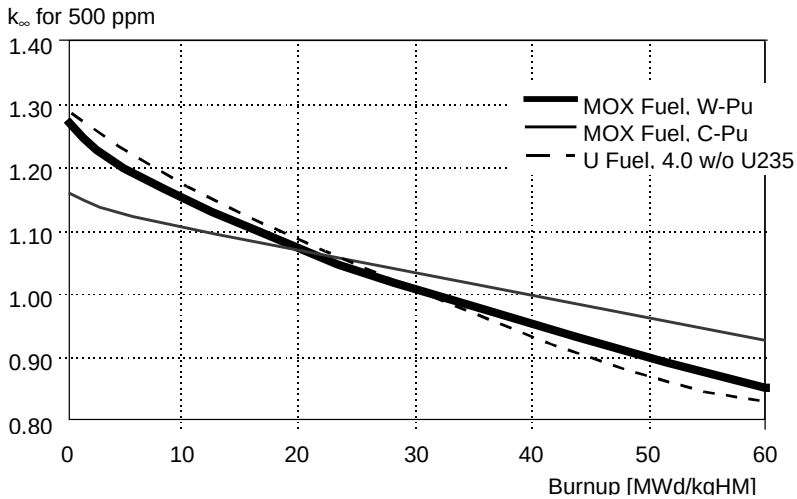

Figure 6. Comparison of Reactivity-Behavior for PWR-FAs

As the Russian side intends to use hollow pellets – as for their U-FAs – the rod design has to combine the European know how available for MOX pellets without central hole with the Russian know how for their U fuel with central hole to establish a verification basis following irradiation testing and LTA insertion.

7. CONCLUSIONS

Experience in Pu recycling in Germany allows MOX-FAs to be designed which result in core characteristics that assure safe normal operation and accident behavior of LWRs. This can be used as a sound basis to prepare other licenses.

A licensing needed to start reactor-based disposition of surplus weapons grade Pu (e.g. in Russian VVER-1000 reactors) could take profit from the following statements deduced from European MOX licensing experience:

- No specific MOX rules and regulations are needed
- Existing (and published) experience can be the basis for licensing by assessment of differences
- The results of Russian test irradiations and of the insertion of LTAs can be used later on to come to final licenses avoiding restrictions, which could be required for the insertion of initial MOX batches.

A precondition for starting reactor-based disposition of Pu is to establish a representative MOX fabrication with proven technology in the United States and in the Russian Federation. The existing international cooperations can insure proper designs and help in licensing.

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