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OVERVIEW OF NEUTRONIC FUEL ASSEMBLY DESIGN AND IN-CORE FUEL MANAGEMENT

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Abstract

The civil and military utilization of nuclear power results in stockpiles of spent fuel and separated plutonium. Recycling of the recovered plutonium in Light Water Reactors (LWR) is currently practiced in Belgium, France, Germany, and Switzerland, in Japan it is in preparation. Modern MOX fuel, with its optimized irradiation and reprocessing behavior, was introduced in 198 1. Since then, about 1700 MOX fuel assemblies of different mechanical and neutronic design were irradiated in commercial LWRs and reached fuel assembly averaged exposures of up to 51 .OOO MWd/t HM. MOX fuel assemblies reloaded in PWR have an average fissile plutonium content of up to 4.8 w/o. For BWR, the average fissile plutonium content in actual reloads is 3.0 w/o. Targets for the MOX fuel assembly design are the compatibility to uranium fuel assemblies with respect to their mechanical fuel rod and fuel assembly design, they should have no impact on the flexibility of the reactor operation, and its reload should be economically feasible. In either cycle independent safety analyses or individually for each designed core it has to be demonstrated that recycling cores meet the same safety criteria as uranium cores. The safety criteria are determined for normal operation and for operational as well as design basis transients. Experience with realized MOX core loadings confirms the reliability of the applied modern design codes. Studies for reloads of advanced MOX assemblies in LWRs demonstrate the feasibility of a future development of the thermal plutonium recycling. New concepts for the utilization of plutonium are under consideration and reveal an attractive potential for further developments on the plutonium exploitation sector.

1. INTRODUCTION

In the early years of the civil utilization of nuclear power, plans for optimized fuel cycle scenarios using a combination of breeder and fission reactors were developed. Essential for that scenario was a commercial reprocessing capacity for spent fuel large enough to establish a closed fuel cycle. Several countries developed the reprocessing technology. After most countries decided to delay

their fast breeder reactor program, only a few countries further developed the reprocessing technology. Today, France and Great Britain operate commercial reprocessing facilities, in Japan construction of a plant is in progress.

The civil and military utilization of nuclear power results in stockpiles of spent fuel and separated plutonium. Since commercial breeder reactors are not available, either the final disposal in geological formations or reprocessing and recycling of its valuable components are possible choices. The decision of how to dispose of the spent fuel depends on the local political situation in each country. In Belgium, France, Germany, Switzerland, and Japan plutonium recycling in thermal reactors is reality or is in preparation.

The status of the plutonium fuel assembly development and its operational qualification is presented in two other sessions of this conference (Refs [1,2]). This paper gives a survey of neutronic fuel assembly and core design with MOX fuel for LWR. It describes the experience gained in almost 30 years and the status reached today.

2. STATUS OF PLUTONIUM RECYCLING

The experience with modem MOX fuel assembly designs in PWR and BWR is depicted in Table I.

2.1. Belgium

The first MOX rods to be irradiated in a power reactor were loaded in 1963 in the BR3 reactor, an 1 $1MW_e$ PWR. The number of MOX assemblies in that core was progressively increased and reached a ratio of 48 % in the last cycle in 1985. From the reprocessing of 530 t of spent fuel over a period of 10 years about 300 kg of fissile plutonium are recovered each year by the Belgian utility ELECTRABEL. Since 1995, MOX fuel is part of reloads for the PWRs Tihange 2 and Doel 3. Typically, 8 or 12 MOX fuel assemblies are reloaded to the core each cycle [3]. Both units are licensed for an operation with up to 37 MOX fuel assemblies in the core.

2.2. France

In 1985 it was decided to recycle plutonium in French pressurized water reactors. FRAGEMA submitted a generic safety report for plutonium recycling for the 900 MW, plant type in 1986. In 1987 a license was granted for a maximum of 30 % MOX fuel assemblies per reload and a total of one-third of the core loading. In the same year the first 16 MOX assemblies were loaded to the St. Laurent B1 core. Currently, 20 plants out of twenty eight 900 MW, PWRs are licensed for plutonium recycling and the target is to reload about 100 t heavy metal per year of MOX fuel corresponding to the reprocessing of 850 t heavy metal of spent fuel.

2.3. Germany

Experience with the plutonium recycling technology in Germany dates back to the insertion of a lead assembly at the Kahl nuclear power plant (VAK, BWR) in 1966. First recycling programs on an industrial level were conducted at the Obrigheim nuclear power station (KWO, PWR) in 1972 and at the first Gundremmingen plant (KRB-A, BWR) in 1974. Since then 10 PWRs and 2 BWRs were licensed for the use of MOX fuel assemblies. Furthermore, for 3 PWRs and 2 BWRs licensing is in preparation. Cores with up to 50 % MOX fuel assemblies are licensed and core designs for PWR with ≃ 35 % MOX ratio have been realized.

2.4. Japan

In the Long-term Program for Research, Development and Utilization of Nuclear Energy by the Japanese Atomic Energy Commission it is stated that recycling of nuclear fuel including the use of MOX fuel in LWRs is national policy for the future energy security, economy and resource saving [4]. A commercial reprocessing plant with a capacity of 800 t of spent fuel per year is under

construction at Rokkasho-mura to be commissioned in 2005. The Japanese LWR MOX fuel demonstration was started in 1986 at Tsuruga Unit-1 for BWR and in 1988 at Mihama Unit-1 for PWR [5]. The first MOX fuel licensing was started in 1998 for both PWR and BWR plants.

2.5. Switzerland

The first MOX fuel assemblies were loaded to the Beznau Unit 1 in 1978 and in 1984 to Unit 2 (KKB, PWR). In 1996, a maximum of 26 % of the fuel assemblies in the core were MOX assemblies. The Goesgen nuclear power plant (KKG, PWR) started its recycling program in 1997. In addition to the commercial recycling programs, R&D work was performed at the Paul-Scherrer-Institut (PSI), Wtirenlingen. The international PROTEUS experiment investigated mainly the effect of tighter lattices and increased plutonium concentrations on the PWR design.

TABLE I. EXPERIENCE WITH MOX RELOADS IN BWR AND PWR FROM 1981 TO 1998

COUNTRY/ REACTOR -/ FA-TYPE	NO. OF REACTORS	NO. OF MOX FA RELOADED	MAX. AV. PU_{FISS} IN W/O / CARRIER MATERIAL	MAX. FA – EXPOSURE AT EOC IN MWD/THM
Belgium				
PWR (17x17-24)	2	56	4.9 / U_{tails}	43900
France				
PWR (17x17-24)	17	992	4.5 / U_{tails}	44250
Germanv				
PWR (18x18-24)	2	4	4.6 / U_{tails}	8000
PWR (16x16-20)	5	364	4.2 / U_{tails}	44900
PWR (15x15-20)	1	32	3 .o / U_{nat}	42000
PWR (14x14-16)	1	41	3.8 / U_{nat}	37000
BWR (9x9-1)	2	116	3.0 / U_{tails}	32000
Switzerland				
PWR (15x15-20)	1	28	4.8 / U_{tails}	23000
PWR (14x14-17)	2	152	4.1 / U_{tails}	51000

2.6. Others

In the US experience with MOX fuel was gained mainly throughout the 1960s and 1970s. During that time MOX assemblies were part of reloads for PWRs and a BWR. After the political decision was made not to pursue the recycling of plutonium, work on that topic was suspended. Today, the reduction of stockpiles of weapons grade plutonium from the dismantlement of nuclear warheads is part of the strategic arms reduction treaty. Recycling of that material in LWRs is considered as one way to assure its peaceful use.

Investigations on MOX fuel were performed in the United Kingdom and countries of the former Soviet Union. It was concluded that AGRs and VVERs are suited for being loaded with a certain quantity of MOX fuel assemblies, but no commercial recycle activity in these reactors was reported. In the United Kingdom a commercial reprocessing facility and a MOX fuel fabrication plant are in operation.

3.1. Early Qualification Steps

Neutronic codes are qualified by comparison of the derived results with measurements and by theoretical benchmarks. For first approaches to the MOX technology, the experience gained with the neutronic properties of depleted uranium fuel was extrapolated. This was a reasonable approach, since more than 50 % of the energy in high exposure uranium fuel is produced by the plutonium isotopes ^{239}Pu and ^{241}Pu. The first explicit qualification steps for MOX fuel were performed in the 1960s. Safety and feasibility of reactor quantities of MOX fuel was demonstrated in the late 1960s and early 1970s in zero power critical facilities, like the Plutonium Recycle Critical Facility (PRCF) in Hanford, USA, the KRITZ facility in Studsvik, Sweden or the VENUS critical facility at CEN/SCK in Mol, Belgium.

These experiments were especially planned for the basic qualification of cross sections and codes. For handling and safeguards reasons, the measurements were performed on fresh fuel. Typical reload enrichments at that time were 3.0 to 3.2 w/o ^{235}U and consequently the matching average concentrations of fissile plutonium were in the range of 2 to 3 w/o, equivalent to plutonium concentrations of 3 to 4 w/o. The experiments were designed to represent the physical and neutronic conditions of PWRs and BWRs as closely as possible. Investigated were effects of varying plutonium compositions and the temperature dependence of the reactivity of the designed cores. The measurement of pinwise fission rate distributions in configurations with central MOX fuel assemblies surrounded by uranium fuel assemblies was another important aspect of the experiments. Confidence in the calculational methods was gained by the recalculation of those measurements and the demonstration of an adequate accuracy.

3.2. Actual Status of Design Method Qualification

Since then, measurements from operating power reactors contributed to the verification of the design codes and were the basis for required updates of the cross section data base. These data comprise results from routinely performed startup and core follow measurements as well as from especially defined programs for the verification of specific aspects, e.g. the comparison of the control rod worth in uranium and MOX fuel assemblies. Reference [6] gives a comprehensive overview on existing experimental data. For the determination of the accuracy of predicted local power density distributions, aside from critical experiments, γ-scan measurements have been conducted on irradiated PWR and BWR MOX fuel assemblies at the reactor site. More recently, additional

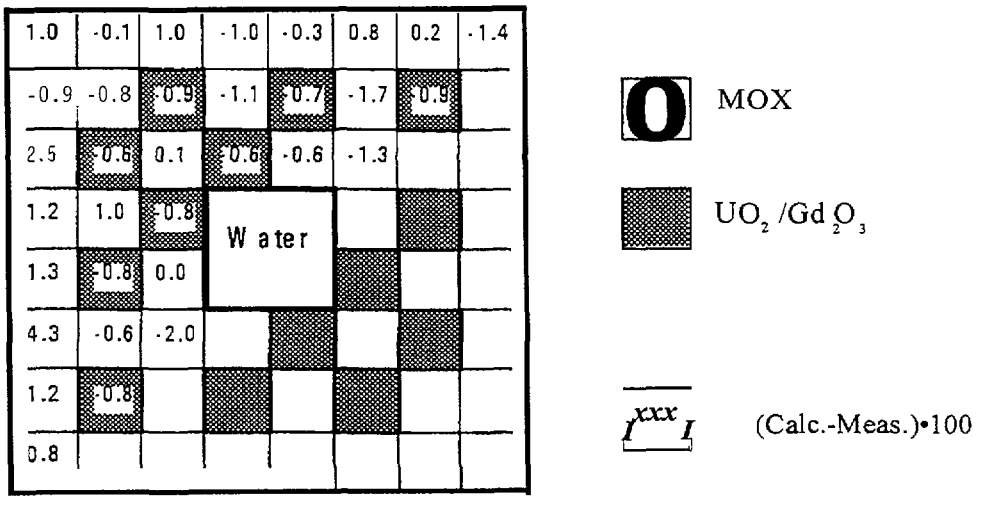

FIG. I VENUS BWR UO₂/MOX Mockup Core
Comparison of Measured and Calculated Fission Rates

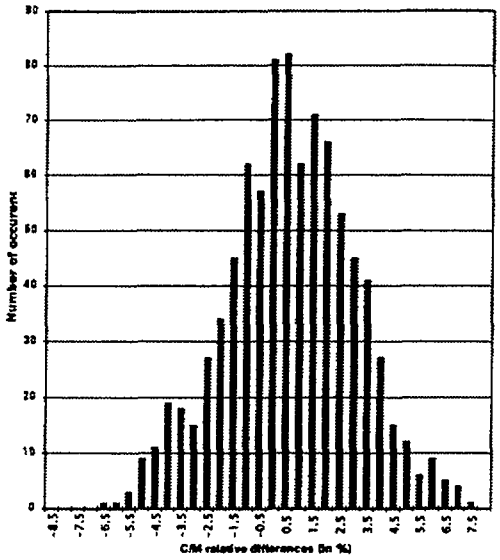

FIG. 2 PWR - Comparison of Calculated and Measured Detector Signals

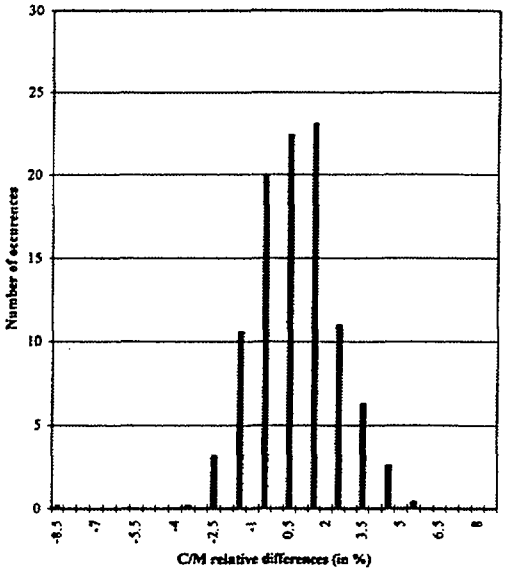

FIG. 3 BWR - Comparison of Calculated and Measured γ- Tip Signals

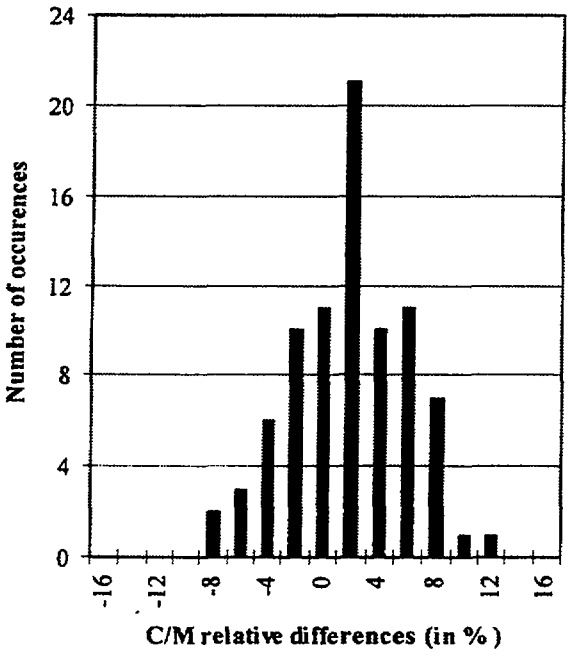

FIG. 4 PWR - Comparison of Calculated and Measured Control Rod Worths

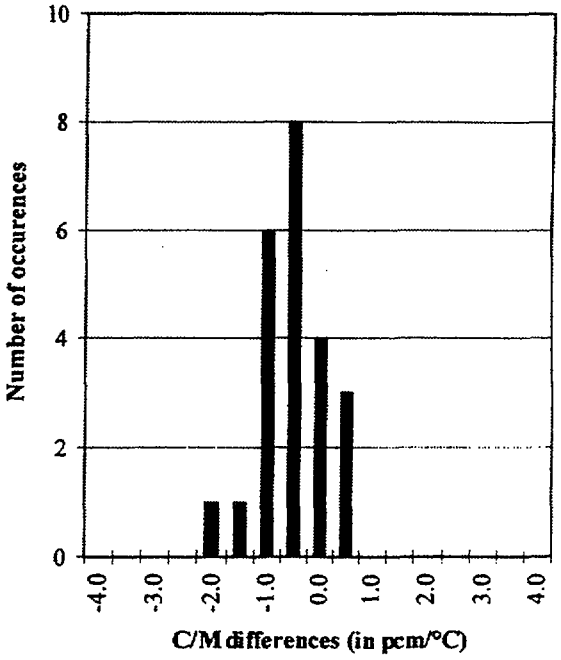

FIG. 5 PWR - Comparison of Calculated and Measured Moderator Temperature Coefficients

experiments and measurements (e.g. PROTEUS, VENUS, EPICURE, Refs 7 - 93) were performed to address the effect of design parameter changes like changed moderation ratio, increased exposures or higher plutonium contents. Some of those activities were conducted in international cooperation.

Modem core design with its higher enrichments of uranium and MOX fuel assemblies has become more demanding with respect to accuracy and reliability of the design codes. Qualified codes prove their capability of predicting safety parameters for a large variety of core designs every year. Current nuclear design methods for both PWR and BWR MOX fuel employ the standard calculational methods also applied for UO_2 fuel. Advanced features for MOX fuel description were introduced by some vendors, e.g. updated nuclear cross section data, corrections derived from colorset calculations, or extensions to the nodal core calculation methods.

The international standard for neutronic fuel assembly design is the use of 2-dimensional multigroup transport codes. These codes determine the flux and power density in either the homogeneously or heterogeneously represented pins and provide condensed and homogenized few group (typically 2 energy groups) cross sections for the reactor codes. The multigroup cross section data used in today's production codes for uranium and MOX design are generally based on data files, like ENDF/B [10], JENDL [11], or JEF [12]. The derived cross section libraries utilized in the spectral codes are qualified by the recalculation of a large variety of measurements and suited for the description of actual core designs. Three dimensional coarse mesh methods with pin power reconstruction techniques are applied to steady state and transient reactor calculations for PWRs and BWRs. These methods have proved to be adequate, accurate and efficient.

Fig. 1 demonstrates the level of accuracy reached with the recalculation of pinwise fission rate distributions measured in critical experiments. The VENUS BWR UO_2/MOX mock-up, selected as an example out of a series of similar experiments (Ref [13]), used an 8x8 fuel bundle geometry with a central water rod. The maximum plutonium concentration in the MOX rods in this example was 9.9 w/o Pu_{fiss}. The calculated data was derived from a 2-dimensional transport calculation. Benchmarking of the PWR design codes on UO_2/MOX mock-ups with high plutonium concentrations confirms the same level of accuracy (e.g. Ref [14]). Fig. 2 shows an example of the quality of prediction of axially integrated detector signals in MOX fuel assemblies for PWR, Fig. 3 an example for γ-tip-measurements in BWR. A comparison of measured and calculated control rod worths and moderator temperature coefficients in PWR, including cores with MOX assemblies, is depicted in Fig. 4 and 5.

The conclusion from those comparisons is that the prediction of key safety parameters for actual cores with MOX fuel assemblies is of the same quality as for uranium cores. However, taking into account a smaller data base with measurements for cores with MOX fuel and specified manufacturing tolerances for plutonium concentration and composition, specific tolerance factors may be applied to certain predicted safety parameters in MOX fuel assemblies (e.g. linear heat generation rate, DNBR).

Special consideration should be given to minor isotopes. These isotopes can be described as of minor important for today's neutronic MOX fuel assembly and core design but may have impact on future developments or on radioactivity aspects. One representative of that group of isotopes is ^{242}Pu, which will become more important with higher plutonium concentrations and the degradation of the plutonium quality. Differences in cross section data for curium and some americium isotopes are found by comparisons of different data files. An improved representation in future cross section data files will become important for high burnup MOX designs.

4. NEUTRONIC FUEL ASSEMBLY DESIGN

4.1. General Aspects and Design Targets

The nuclear characteristic of MOX fuel is a result of the neutronic properties of the different plutonium isotopes. The n-values for thermal neutrons of the fissile plutonium isotopes are higher

than that of ^{235}U, where both fission and capture cross sections are about a factor of 2 larger. The absorption cross sections of the plutonium isotopes ^{240}Pu and ^{242}Pu show strong resonance peaks in the near thermal region. As a result, the neutron spectrum in MOX fuel is hardened, i.e. at the same power level the thermal neutron flux is much lower than in uranium fuel (see Figure 6).

In the past, studies were performed to find technically and economically optimized MOX fuel assemblies. The issue of those studies was that despite of its different neutronic characteristics no changes in the mechanical and thermal hydraulic designs are required in comparison to the uranium assemblies loaded to the core at the same time. Only slightly different properties of the (U, Pu)O_2 fuel, e.g. melting temperature and thermal conductivity, do not require changes of the fuel rod design. With an unmodified mechanical fuel rod and assembly design the following targets were addressed in neutronic design studies:

(1) The discharge burnup of the MOX fuel assemblies should match closely that of the uranium fuel assemblies; or

(2) The average plutonium content was chosen as high as reasonable to reduce the number of required MOX assemblies.

These design criteria were complemented by more general demands. MOX fuel assemblies should be compatible to uranium assemblies with respect to loading strategies and the number of possible in-core cycles without additional restraints for their operation.

Neutronic fuel assembly and core design cannot be seen as independent processes. Core design calculations are used to determine the appropriate average plutonium concentration for the MOX fuel assemblies required to meet the design goals. The neutronic MOX fuel assembly design procedure starts after the average plutonium content is defined. Then again, the result is confirmed with core design calculations.

The neutronic MOX fuel assembly design for thermal reactors has to face large gradients of the thermal flux at the interface of MOX and uranium fuel assemblies. The fact of an increasing thermal flux in the direction of the adjacent uranium fuel assembly is addressed by a gradation of the

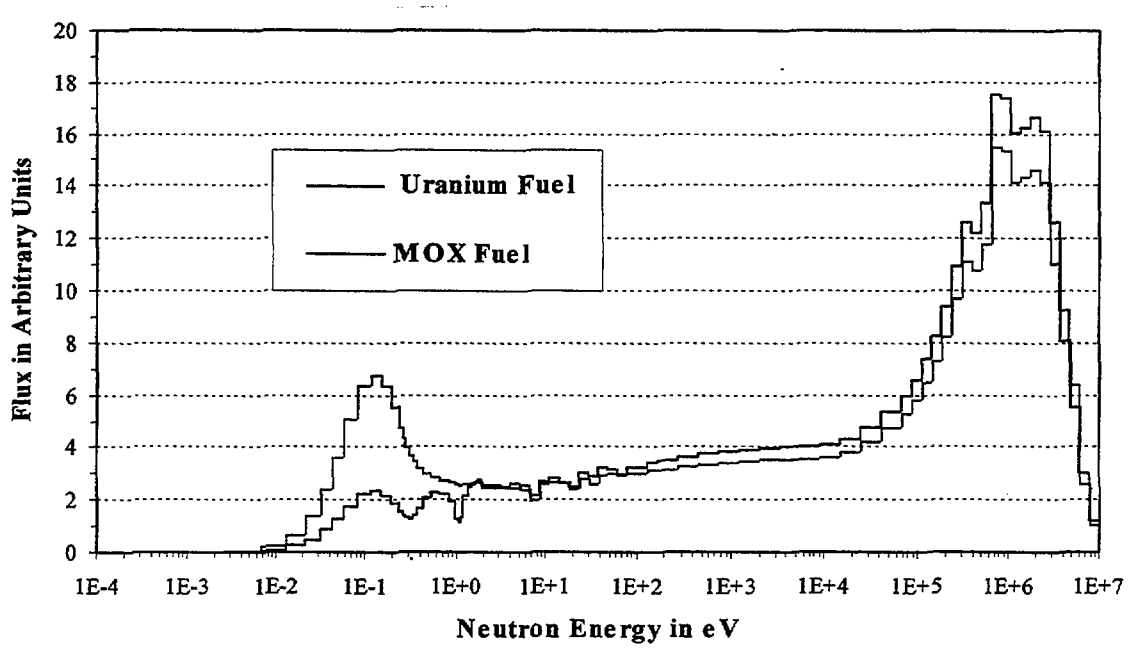

FIG. 6 Neutron Flux vs. Fission Neutron Energy in Uranium and MOX Fuel (PWR)

plutonium content in the fuel rods at the fuel assembly edge. Two to 3 rod types are typical for MOX fuel assemblies used in the last 25 years of plutonium recycling in PWR. Optimized BWR fuel assemblies are more heterogeneous. Larger water gaps and water structures within the fuel assembly result in MOX fuel assembly designs with 4 to 7 different rod types. The 'all plutonium' assembly, comprising of MOX fuel rods only, is appropriate for PWRs and BWRs. In comparison to the earlier 'island' type MOX fuel assembly design, with MOX rods in the center and uranium rods at the periphery of the assembly, it has clear economic advantages. The additional costs associated with fabrication and transport are then restricted to a minimized number of fuel assemblies.

For PWRs, MOX fuel assemblies are currently designed without burnable absorbers, whereas for BWRs neutron poisons are normally required in all fuel bundles. The exposure dependent reactivity behavior of the BWR fuel bundles is optimized by using UO_2/Gd_2O_3 rods. MOX fuel rods are generally unpoisoned. Optimized BWR MOX bundles are designed with a limited number of gadolinium rods and a minimum of different rod types to gain the maximum economic advantages. Those designs are still considered as of the 'all plutonium' type.

4.2. MOX Carrier Material and Plutonium Composition

The carrier material for the plutonium and the plutonium composition are important parameters for the MOX fuel assembly design. Both have impact on the reactivity characteristic of the MOX fuel.

4.2.1. The carrier material for the plutonium

Three options for carrier material were investigated for the commercial **MOX** recycling strategies - natural uranium (U_{nat}), tails uranium (U_{tails}) from the enrichment process and depleted uranium separated in the reprocessing process. First designs were based on natural uranium. Actual MOX fuel is mainly fabricated with tails uranium as carrier. For economic reasons, typical ^{235}U enrichments in the tails assay are 0.2 - 0.3 w/o. The somewhat lower ^{235}U enrichment compared to natural uranium is compensated for by an increase of the plutonium content of 0.3 - 0.4 w/o Pu_{fiss}. Depleted uranium was used in test assemblies only.

4.2.2. The plutonium composition

The isotopic composition of the plutonium is determined by the origin, the initial enrichment, the discharge exposure and the intermediate storage time of the reprocessed fuel. So far, the commercial plutonium used for the manufacturing of MOX fuel was mainly generated in uranium fuel assemblies. Piutonium from spent MOX assemblies was used for demonstration purposes only. The plutonium quality can be defined as ratio of the fissile isotopes in the isotopic composition to the total plutonium content. Table II shows examples for plutonium qualities employed in MOX fuel

TABLE II. EXAMPLES OF PLUTONIUM QUALITIES USED FOR MOX FUEL FABRICATION

Plutonium/ Pu-Quality	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
			in w/o		
Magnox-Pu/ *78.9*	0.3	74.3	19.9	4.6	0.9
LWR-Pu/ *68.9*	1.5	60.1	24.5	8.8	5.0
2nd Generation-Pu/ 58.0	1.3	43.8	34.3	14.2	6.4
Weapons grade Pu/ ~94*)	≈ 0	95	5	≈ 0	≈ 0

*) weapon grade plutonium, so far not used for the fabrication of commercial MOX fuel
Pu-Quality = $(^{239}Pu + ^{241}Pu)*100/Pu_{tot}$

fabrication. Variations of the plutonium quality affect the exposuredependent reactivity of the MOX fuel. The higher the plutonium quality the higher is the beginning of life reactivity and the larger is the slope of the reactivity vs. exposure. This has impact on the average plutonium concentration required for meeting the basic design goals. Fuel assembly design is not sensitive to reasonable changes of the plutonium quality. New optimization of the enrichment zones is only required if the recalculation of the fuel assembly design results in unacceptable power density distributions or if a change in the average plutonium concentration is required due to safety evaluations. Independent of neutronic aspects related to the core, variations in the plutonium quality have to be considered for handling and storage of the MOX fuel assemblies.

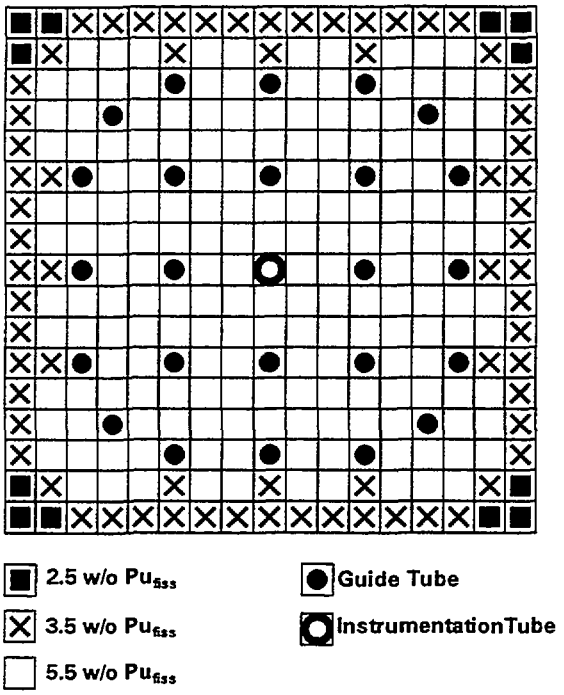

FIG. 7 MOX Fuel Assembly for the Reload in Belgian 900MW$_e$ PWRs

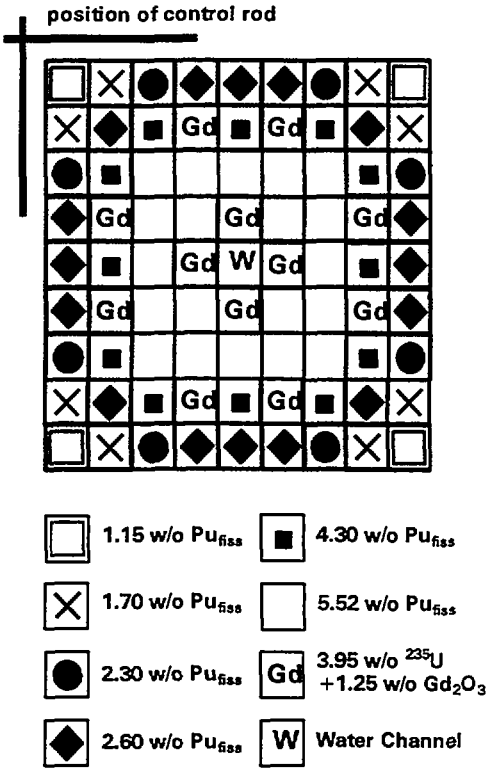

FIG. 8 MOX Fuel Assembly for the Reload in German 1300MW$_e$ BWRs

4.3. Examples of MOX Fuel Assembly Designs for PWR and BWR

Figures 7 and 8 show examples of MOX fuel assemblies designed for actual reloads for PWR and BWR in Belgium and Germany.

Modern MOX fuel assembly designs utilize central water structures like water channels for BWR or additional 'water rods' for PWR. This additional moderator increases the reactivity of the under-moderated MOX rods and flattens the power density distribution in the fuel assembly without introducing new rod types with increased plutonium concentrations. This design feature became particularly important with increasing plutonium concentrations and exposures. A reasonably flat power density distribution in a uranium fuel assembly environment limits the maximum exposure values and is a prerequisite for meeting the mechanical design criteria and for enough operational flexibility.

In BWR bundles the reduced reactivity worth of the gadolinia rods is to a certain extent offset by placing them near regions with high moderation, i.e. adjacent to the central water structures [15].

5. NEUTRONIC CORE DESIGN

Under actual licensing conditions MOX fuel assemblies are loaded into the core together with uranium fuel assemblies. Design criteria for these MOX assemblies have to be defined in harmony with those for the uranium assemblies. For safe and economic reactor operation it is essential to meet those criteria and to provide enough operational margins.

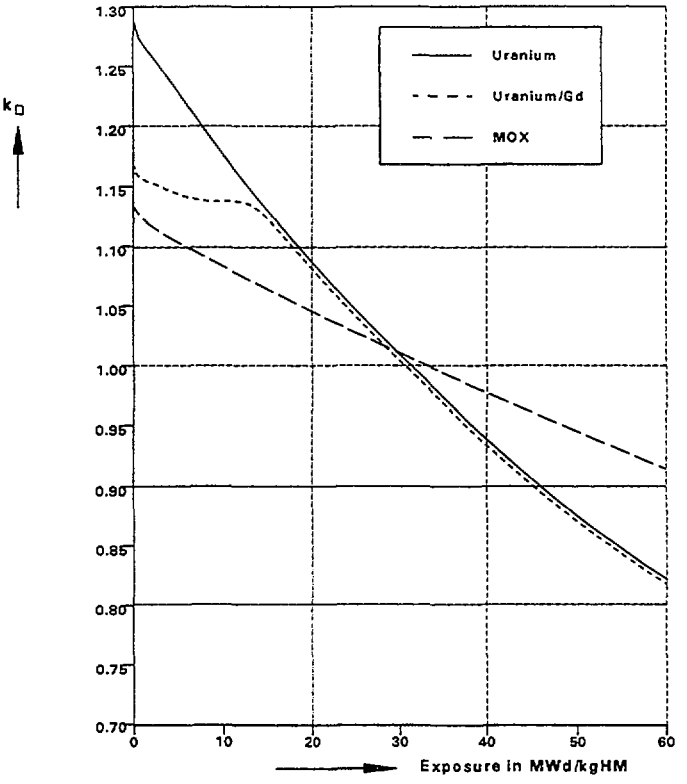

FIG. 9 k-infinite vs. Exposure for U, U/Gd and MOX Fuel (PWR)

The historical development of neutronic core design is characterized by an increase of the reload enrichments and, consequently, an increase of the discharge burnup. In the 1970s typical reload enrichments were 3.1 - 3.5 w/o ^{235}U. In the meantime many reactors worldwide are licensed for enrichments of 4.0 w/o ^{235}U and higher. In conjunction with low leakage loadings utilizing burnable absorbers, batch averaged exposures of 46 MWd/kg in PWRs and 43 MWd/kg in BWRs have been realized. MOX fuel assemblies had to follow that trend. An increasing content of fissile plutonium had to be used in order to match the increased uranium enrichments.

Today, MOX fuel assemblies with an average fissile plutonium content of up to 4.8 w/o (7 w/o Pu) are part of reloads for PWRs in Belgium and Switzerland. For German BWR, the average fissile plutonium content in actual reloads is 3.0 w/o (4.3 w/o Pu) with a maximum of 5.5 w/o in the highest enriched rods. BWR MOX bundles with an average fissile plutonium content of 3.6 w/o to be reloaded in 1999 are in fabrication.

Currently, for unchanged lattices the plutonium content in a fuel assembly should not exceed 13 w/o. Benchmarking calculations for infinite MOX lattices indicated that for plutonium concentrations greater than 13 w/o, k-infinite for the voided case may become higher than for moderated conditions, corresponding to a positive void reactivity coefficient. Further investigations are needed for qualification of nuclear data and codes in that point [6].

5.1. Reload Strategies

Parallel to the increase of the reload enrichments, out-in core loading strategies, where fresh fuel assemblies were positioned at the core periphery, were replaced by low-leakage loadings with its significantly increased fuel economy. The low-leakage ratio can be defined as number of reload assemblies not facing the core periphery divided by the total number of reload fuel assemblies. The higher the low-leakage ratio the more fuel assemblies need to be limited in its initial reactivity. This can be achieved either by a split of the reload enrichment, or far more common, by the use of burnable absorbers. For PWR the exposure dependent reactivity characteristic of MOX fuel (see Figure 9) allows to substitute fresh uranium fuel assemblies with burnable absorbers by MOX assemblies. Higher reactivity values and therefore higher power densities at higher exposures in comparison to uranium assemblies are unfavorable for waterside cladding corrosion and fission gas pressure buildup. These phenomena have to be considered for optimized core loadings. In BWR, with an adequate fuel bundle design, recycling cores don't require significant adjustments of the reload strategy in comparison to all uranium core designs. [16, 17]

Figure 10 shows an example of a core design realized in a Belgian 900 MW, PWR. The prediction standard for local power is demonstrated by the comparison of axially integrated measured and calculated detector signals, depicted in the same figure. The MOX fuel assemblies of that core were designed in compliance with the safety considerations of the licensing authority and the economics considerations of the utility. The content of fissile plutonium in those MOX fuel assemblies is 4.8 w/o with U_{tails} as carrier material.

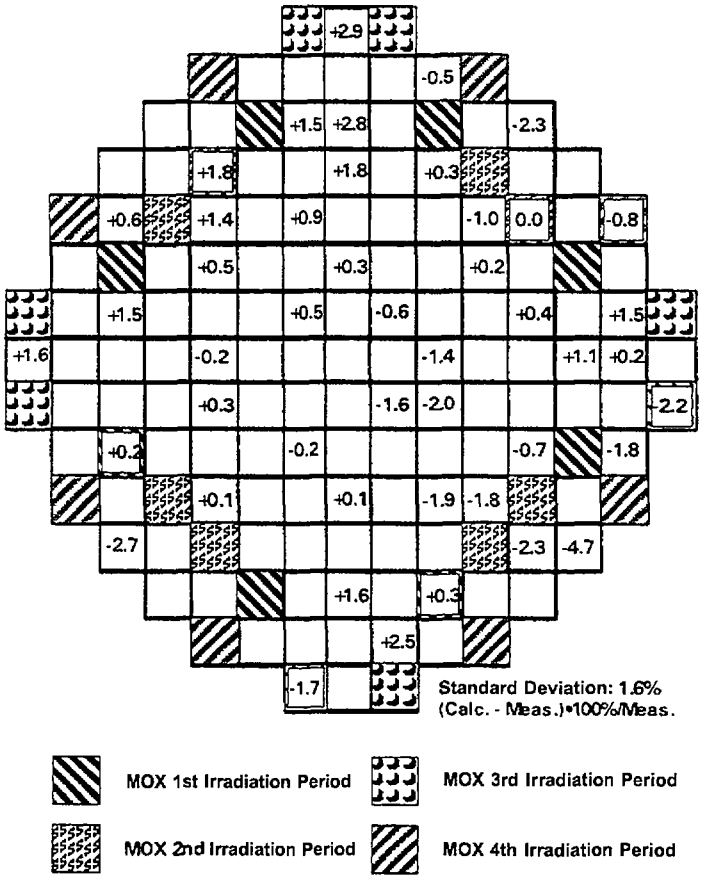

FIG. IO Comparison of Calculated and Measured Detector Signals in a MOX Loaded 900 MW_e PWR

Those recycling cores have to meet the same safetqfcriteria as uranium cores. This has to be demonstrated either for each designed core or in cycle independent safety analyses [18, 19]. The safety criteria are determined for normal operation and operational or design basis transients (e.g. main steam line break accident, LOCA, ATWS or external events). Some of the neutronic aspects affected by plutonium recycling are discussed in the following sections.

5.2. MOX Impact on Normal Operation

5.2.1. PWR

The control rod worth in MOX fuel assemblies is smaller than in uranium fuel. This effect is most pronounced in fresh fuel. It declines with decreasing moderator temperature and increasing exposure. Core design has to account for that. For the actually licensed ratio of MOX assemblies in the core, results from studies indicated that the influence on the net control rod worth (stuck rod configuration) caused by changes in the core loading pattern is larger than the MOX influence. A tentatively smaller bank worth is often accompanied by a reduced stuck rod worth and results then in an almost unchanged net control rod worth.

The boron system is not only used for continuous reactivity control under normal operating conditions but is also utilized for the long term shut down of the plants. The reduced boron reactivity worth can be compensated for, depending on the number and design of the loaded MOX fuel assemblies, by an increase of the boron concentration in the storage tanks or by an increase of the ^{10}B-enrichment in the borated water bearing systems.

The limits for local power, $F\Delta H$, or DNBR remain unchanged. Incore or excore systems are set to the same limiting values as for uranium cores.

5.2.2. BWR

The impact of MOX fuel assemblies on the control rod system is small. The thermal flux recovers in the large water gaps between the bundles and results in almost unchanged control rod worths. Hence, cores are normally designed with a scattered distribution of the MOX assemblies with only one or two MOX assemblies assigned to each control rod.

5.3. MOX Impact on Transients

5.3.1. Reactor Kinetics

The effect of a reduced delayed neutron fraction by introducing MOX fuel impacts fast power transients and reactivity initiated accidents (RIA). The core power responds more rapidly to perturbations of the reactivity. Part of that effect is compensated for by a slightly more negative Doppler coefficient. Safety analyses are generally performed by assuming a conservatively low fraction of delayed neutrons.

5.3.2. Decay Heat Power

Decay heat has impact on large break loss of coolant accidents. Directly after shut down, decay heat power is smaller for MOX fuel than for uranium fuel. With time, the decay heat is more and more determined by the actinides and, after the decay of the short living fission products, becomes therefore higher in MOX fuel. The differences in the decay heat are of only little importance for the maximum cladding temperature reached during large break LOCA accidents.

5.3.3. PWR

More negative moderator temperature coefficients have to be considered in cold shutdown reactivity balances and transients with moderator cool down in conjunction with the tendency to a reduced bank worth and a smaller boron worth.

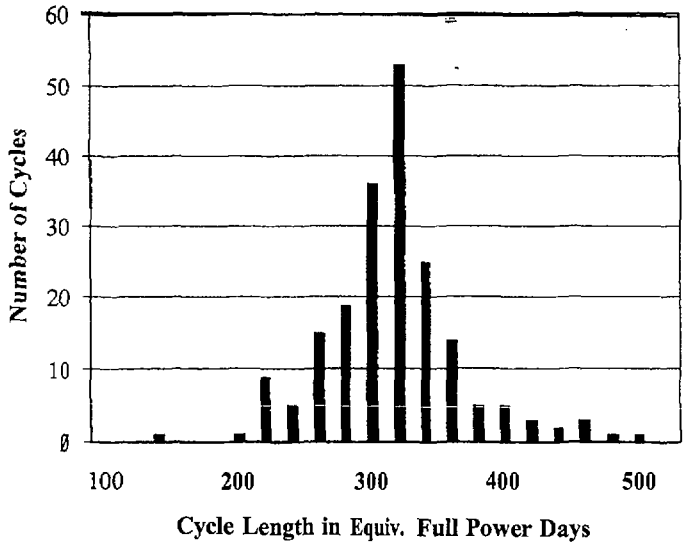

FIG. 11 Cycle Length Variation in German PWRs (Status: 1998)

5.3.4. BWR

The more negative moderator temperature and void coefficients affect the cold shutdown reactivity balances, transients with moderator cool down and transients with coolant pressure increase. MOX fuel with very high concentrations of plutonium may cause a shift to less negative values of the void coefficient [6].

5.4. Operational Flexibility

The operation of nuclear power plants is determined by the demands of the electricity suppliers. Load follow operation, variations in cycle length and reload batch size, but also unplanned outages are challenges for the flexibility of fuel assembly and core design. MOX fuel assemblies are providing the required flexibility under those boundary conditions. Fig. 11 gives an overview of cycle lengths realized with German PWRs [20].

6. SAFEGUARDS ASPECTS

The IAEA safeguards criteria for fresh MOX fuel assemblies are more restrictive than for normal uranium fuel assemblies. MOX fuel assemblies are transported in specially developed, sealed containers with safety trucks. After arriving at the plant site the seal is removed by an IAEA representative. Radioactivity and heat production make it advisable to store the MOX assemblies directly in the storage pool. Continuous video surveillance and a monthly inspection by an IAEA inspector make sure that no unauthorized manipulations occur. At some plants fresh MOX assemblies are stored in the dry storage facility. Then, each assembly is separately sealed and regularly inspected. Core loading is performed under surveillance of an IAEA inspector. No additional measures are required for the transportation of discharged MOX fuel assemblies relative to discharged uranium fuel assemblies.

7. CONCLUSION

In-core fuel management with MOX assemblies is common practice in a large number of nuclear power plants in several countries. A ratio of about 30 % (50 % in some plants) of MOX fuel assemblies in the core is possible while meeting applicable safety criteria. Experience with such core

loadings confirms the reliability of the applied modem design codes. The applicability of methods used for neutronic fuel assembly and core design has been proven for actual plutonium concentrations and compositions used.

Plans for a further increase of plutonium concentrations and the use of degraded or weapons grade plutonium qualities should be accompanied by

- a continuous verification of the utilized nuclear data and design methods.

- post irradiation examinations, i.e. isotopic measurements on highly depleted MOX fuel and, if possible, critical experiments with depleted fuel.

Conceptual studies for reloads of advanced MOX assemblies in LWRs demonstrate the feasibility of a future development of the thermal plutonium recycling. Hence, new concepts for the utilization of plutonium are under consideration or investigation and reveal an attractive potential for further developments on the plutonium exploitation sector.

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