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MOX Fuel Design Report

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
Rev. 0		Initial issue.
Rev. 1	all	Revised throughout to clarify, strengthen discussions, and remove unnecessary material.

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Nomenclature

<u>Acronym</u>	<u>Definition</u>
ADU	ammonium diuranate
APT	Average Power Test
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
AUC	ammonium uranyl carbonate
BOL	beginning of life
CHF	critical heat flux
DCS	Duke COGEMA Stone & Webster, LLC
DOE	Department of Energy
EDF	Electricité de France
EOL	end of life
FGR	fission gas release
FANP (Fr)	Framatome ANP, SSA (France)
FANP (Ger)	Framatome ANP, GmbH (Germany), formerly Siemens
FANP (US)	Framatome ANP, Inc. (U.S.)
GWd/MThm	gigawatt-days per metric ton of heavy metal
HFIR	high flux isotope reactor
k_{∞}	infinite multiplication factor
lbf	pounds-force
LEU	low enriched uranium
LHR	linear heat rate
LOCA	loss-of-coolant accident
LWR	light water reactor
MFFF	MOX fuel fabrication facility
MIMAS	micronized master blend
MOX	mixed oxide – uranium dioxide and plutonium dioxide
MThm	metric tons of heavy metal (plutonium plus uranium)
MWd/MThm	megawatt-days per metric ton of heavy metal
MWd/MTU	megawatt-days per metric ton uranium
NRC	U.S. Nuclear Regulatory Commission
OCOM	Optimized Co-milling
ORNL	Oak Ridge National Laboratory

Nomenclature (Continued)

<u>Acronym</u>	<u>Definition</u>
PCI	pellet-cladding interaction
PIE	post-irradiation examination
ppb	parts per billion (= ng/g)
ppm	parts per million (= µg/g)
PWR	pressurized water reactor
RCCA	rod cluster control assembly
RG	reactor-grade (plutonium)
RIA	reactivity insertion accident
SRP	Standard Review Plan
TD	theoretical density
WG	weapons-grade (plutonium)

1.0 Introduction and Summary

The U.S. Department of Energy (DOE) is implementing a program to dispose of a portion of the nation's surplus weapons-grade (WG) plutonium by reconstituting the plutonium into mixed-oxide (MOX) fuel pellets and using the fuel in commercial light water reactors (LWRs).

Accordingly, the DOE has contracted with Duke COGEMA Stone & Webster (DCS) to design and license the MOX fuel, fabricate lead assemblies, irradiate the lead assemblies, and qualify the design for batch irradiation.

This MOX Fuel Design Report presents the fuel design that will be used for the disposition of the WG plutonium. It demonstrates that the fuel rod design and the fuel assembly design will provide reliable, safe operation, comparable to that of equivalent low enriched uranium (LEU) designs, and demonstrates that the Mark-BW/MOX1 is acceptable for batch implementation up to a maximum fuel rod burnup of 50,000 MWd/MThm. In this topical report, approval is sought for:

- Using this fuel design in a confirmatory demonstration in which two or four MOX lead assemblies are irradiated for two cycles, conditional upon the receipt of required regulatory authorizations including modifications to the reactor operating license, and
- Batch implementation of the MOX fuel design with a fuel rod maximum burnup limit of 50,000 MWd/MThm, conditional upon successful operation of the lead assemblies for two irradiation cycles and upon the receipt of required regulatory authorizations including modifications to the reactor operating license.

In addition, approval is sought for:

- Extended irradiation of one or more of the lead assemblies for a third cycle. This irradiation would take the assemblies beyond the burnup limit (50,000 MWd/MThm maximum rod average) requested for batch implementation, but not beyond 60,000 MWd/MThm maximum rod average.

Results from the third cycle of irradiation are not necessary to support approval for batch implementation. Instead, they will be used to establish a database that will demonstrate successful operation beyond the approved burnup limit for MOX fuel. The results may potentially be used in future submittals to support an increased burnup limit for MOX fuel.

Throughout the report, additional background information is provided. An example of this is the process by which MOX pellets are manufactured. Approval of such information by the U.S. Nuclear Regulatory Commission (NRC) is not sought through this report.

Framatome ANP will perform plant-specific evaluations of the performance of the Mark-BW/MOX1 lead assemblies in the mission reactors; such evaluations will reference the approved version of this topical report. The plant-specific evaluations will not be submitted to the NRC for review and approval unless the licensee determines that NRC review and approval is required per the requirements of 10 CFR 50.59. The evaluation processes described in this report may also be used to justify small changes in the Mark-BW/MOX1 fuel assembly design without specific NRC review and approval. Examples of such small design changes would be changes in plutonium content or isotopics, small variations in stack length, pellet length, or dish and chamfer design, and changes to cladding length or plenum spring design. Other examples, discussed in Reference 1, would be changes in fuel rod or fuel assembly length to accommodate reactor-specific dimensions, modifications of spacer grids that do not affect the approved departure from nucleate boiling ratio correlation for the spacer grids, and substitution of components that have been separately approved by the NRC. Increases in maximum burnup are not considered to be small changes.

The fuel design to be used in the material disposition program is the Mark-BW/MOX1 design. The 17 × 17 Mark-BW/MOX1 fuel assembly is the Advanced Mark-BW (Reference 1) fuel assembly with the LEU fuel rods replaced by MOX fuel rods. The fuel rods contain MOX pellets based on the rod design and pellet specification used by Framatome ANP for European MOX fuel.

The MOX fuel in the Mark-BW/MOX1 is an intimate mixture of PuO₂ in a depleted uranium oxide matrix. Approximately 95% of the MOX material is composed of UO₂; thus, materials properties are similar to those of LEU fuels. The MOX materials properties have been determined from reactor-grade (RG) MOX fuel operating experience in Europe and are considered in the fuel rod analyses through the use of the COPERNIC fuel performance code (Reference 2).

MOX fuel is characterized in terms of plutonium isotopics as reactor-grade (RG) or weapons-grade (WG). The concentration of neutron-absorbing ²⁴⁰Pu is lower in WG plutonium than in RG plutonium, so MOX fuel made from WG plutonium will have a lower total plutonium concentration for the same total energy extraction. The neutronic performance of RG MOX fuel has been benchmarked to a wide range of operating and test data in Reference 3.

1.1 ***Other Supporting Topical Reports***

The implementation of MOX fuel is also supported by other topical reports. This section provides background information on how the various efforts are related.

This report addresses the fuel design to be used for MOX irradiation. Core design and safety analysis aspects of MOX fuel operation are addressed in other submittals. Duke Power will use CASMO-4 and SIMULATE-3 MOX for the nuclear design of cores that include MOX fuel. These two codes have been benchmarked against critical experiments encompassing fissile plutonium concentrations that bound the fissile plutonium concentrations the mission reactors will use. The applicability of CASMO-4 and SIMULATE-3 MOX to neutronically model LEU and partial MOX fueled cores is described in Reference 3.

The thermal-hydraulic effects of mixed cores containing Mark-BW/MOX1 fuel assemblies and LEU fuel assemblies will be evaluated using the approved statistical design methods described in Reference 4.

Duke Power has performed the necessary safety analysis evaluations for the irradiation of MOX fuel lead assemblies at McGuire or Catawba. The core response to limiting transients and accidents will not be significantly affected by the presence of up to four Mark-BW/MOX1 lead assemblies. Duke Power will document these safety analysis evaluations as part of the license amendment request for MOX fuel lead assemblies (Reference 5).

Duke Power will perform the non-loss-of-coolant accident safety analyses to support batch implementation of the Mark-BW/MOX1 in the mission reactors. Duke Power will submit a topical report describing these safety analyses prior to the submittal of a License Amendment Request for batch operation of MOX fuel.

Framatome ANP, Inc. {FANP (US)} will submit a generic topical report addressing the effects of batch implementation of MOX fuel on the loss-of-coolant accident (LOCA) evaluation model.

2.0 **MOX Design Considerations**

The MOX features that must be addressed in the design and manufacturing activities are presented in this section. The four areas considered are MOX fuel performance characteristics, plutonium content, MOX pellet homogeneity and microstructure, and operation in mixed cores. Sections 2.1 through 2.4, respectively, discuss these areas.

2.1 **MOX Performance Characteristics**

The Mark-BW/MOX1 design will use MOX fuel that is an intimate mixture of PuO₂ in a depleted uranium oxide matrix. The depleted uranium will be about 0.25% ²³⁵U. Approximately 95% of a MOX pellet is composed of UO₂; thus, the material properties will be similar to those of LEU fuels. However, the following physical characteristics are potentially affected by the addition of small amounts of PuO₂ powder:

- Thermal conductivity
- Thermal expansion
- Thermal creep
- Fission gas release
- In-reactor densification and swelling
- Helium gas accumulation and release
- Radial power profile
- Melting point

These physical characteristics of MOX fuel have been established through experimental and laboratory measurements. The Framatome ANP fuel performance code, COPERNIC (Reference 2), contains burnup-dependent physical properties for MOX fuel and treats each of the items listed above.

COPERNIC produces steady-state and transient extended-burnup fuel performance predictions and can be applied to LEU, UO₂-Gd₂O₃, and MOX fuel types. COPERNIC models specific to MOX fuel were developed for thermal conductivity, MOX material melting point, radial power profiles, and fission gas release. The other phenomena (thermal expansion, thermal creep, in-reactor densification and swelling, and helium gas accumulation and release) are common to LEU fuel, vary little from LEU fuel, or are conservatively described by the LEU fuel model. The

COPERNIC models (Reference 2) accurately predict the fission gas release of MOX fuel rods, including those subjected to transients.

2.2 **Plutonium Content**

With LEU fuel, the fissionable component is provided by ^{235}U . The ^{235}U concentration is specified by the fuel designer and controlled by the enrichment process. With MOX fuel, the ^{239}Pu and ^{241}Pu isotopes provide the fissionable component. The fuel designer again sets the fissile content, but for MOX both the total quantity of PuO_2 and the plutonium isotopic content are considered.

MOX fuel requires a neutronic model capable of calculating the effect of individual plutonium isotopes on the neutronic solution and the neutron transport between LEU and MOX fuel assemblies. A topical report prepared by Duke Power (Reference 3) discusses the neutronic effects of MOX and of mixing MOX and LEU fuel assemblies in a reactor core. Duke Power will use the methods described in the referenced report to account for the neutronic characteristics of these different fuel types to ensure safe operation of mixed LEU/MOX cores.

Plutonium is generated and subsequently burned in LEU fuel soon after irradiation begins, so LEU fuel and MOX fuel both contain significant quantities of plutonium. The uranium and plutonium contents of typical LEU and MOX fuels are shown in Table 2.1 at beginning-of-life (BOL) and at end-of-life (EOL). As can be seen, the heavy metal in both fuels (LEU as well as MOX) is primarily ^{238}U . At BOL, the LEU fuel has no plutonium; but shortly thereafter, the LEU fuel is producing a portion of its power from the plutonium that has been generated during operation. Thus, LEU fuel experience inherently includes irradiation of plutonium. European practice utilizes this experience by producing MOX fuel with RG plutonium. This is referred to as RG MOX.

As of the end of 1998, the European manufacturing facilities had produced a combined total of more than 435,000 RG MOX fuel rods, and these rods have been successfully irradiated in 35 commercial nuclear reactors. The European experience with RG MOX fuel in commercial reactors is described in more detail in Chapter 7. The differences between WG and RG MOX are addressed in Chapter 3. It is shown in that chapter that experience with LEU and RG MOX fuel is applicable to WG MOX in terms of total plutonium mass, fissile plutonium mass, and mass of individual plutonium isotopes.

2.3 ***Pellet Homogeneity and Microstructure***

LEU fuel is enriched in the ^{235}U isotope, an operation that occurs on a molecular scale. Homogeneity of the product is thus guaranteed on a very fine scale. For MOX fuel, homogeneity is provided by the manufacturing process. The Mark-BW/MOX1 will use MOX manufactured by the Micronized MASter blend (MIMAS) process, which involves blending and milling of UO_2 and PuO_2 powders (master mix) and then dilution of the master mix with more UO_2 to reach the final plutonium concentration. Microscopic examination of MOX pellets shows plutonium-rich agglomerates of the master mix, finely dispersed in a UO_2 matrix. The agglomerates, in turn, consist of very fine, individual particles of PuO_2 and UO_2 powder. The maximum plutonium concentration of the master mix agglomerates is determined by the ratio of UO_2 to PuO_2 in the master mix.

The maximum size and plutonium concentration of the agglomerates are determined by the specification and controlled by the manufacturing processes. This is done during production through a milling and sieving operation, followed by pellet pressing and sintering. Control of this process sequence is verified through metallographic examination and/or autoradiography of a representative number of samples from each batch of pellets. These examinations provide measurements of the plutonium-rich agglomerate size and the matrix grain size.

The fuel pellet specification provides limits on the average and maximum sizes of the plutonium-rich agglomerates. The Framatome ANP, SSA {FANP (Fr)} specification for European RG MOX limits the mean agglomerate size to less than 50 μm , with at least 95% of the agglomerates smaller than 100 μm . The maximum size for a pure PuO_2 particle is 400 μm . The FANP (US) specification from WG MOX fuel pellets has the same requirements. In addition, these requirements are consistent with the industry standard specification for MOX fuel pellets (Reference 6).

The limitation on the average size of the plutonium-rich agglomerates mitigates the effects on fission gas release. Because the agglomerates contain most of the fissile material, most of the fissions occur within the agglomerates. As a result, the agglomerates will function at a slightly higher elevated temperature, and most of the fission products will originate within the agglomerates. The higher temperatures tend to increase the fission gas release, and some fission products reside in the agglomerates, so MOX may have a higher fission gas release than LEU fuel at comparable burnups. The difference in fission gas release between LEU and MOX

fuel decreases with improved homogeneity. The COPERNIC MOX fission gas release database reflects fuels consistent with the current specification.

The maximum allowable PuO₂ particle size is based on the effect of large PuO₂ particles in reactivity insertion accidents (RIAs). Enriched UO₂ pellets with 550- μ m particles of pure PuO₂ were subjected to transient tests in SPERT (Reference 7). The results showed that the effect of the large particles was to reduce slightly the cladding failure threshold energy relative to that for LEU fuel. There was no indication of prompt fuel dispersal caused by the expulsion of the large particles into the surrounding water. Additional testing (Reference 8) was performed with MOX fuel. These tests used PuO₂ particles 400 and 1100 μ m in diameter that were embedded in the pellet surface. These newer tests with more relevant fuel pellet compositions showed that the large particles did not affect the failure threshold of MOX relative to LEU fuel. As stated above, the European specification for the maximum allowable size of 400 μ m for pure PuO₂ particles will be used to remain within the experience base for fuel performance.

2.4 Operation in Mixed Cores

Two types of mixed cores with MOX fuel may be distinguished. First, the Mark-BW/MOX1 assemblies may be mixed with assemblies of different design, such as those from other manufacturers. Second, there will always be a mixture of MOX and LEU fuel assemblies. Both types of mixed cores have been considered, and approved methods are available for handling them.

For any fuel design inserted on a reload basis, thermal-hydraulic compatibility of the introduced design must be established with the existing, or resident, fuel in the reactor core. The thermal-hydraulic compatibility of the Mark-BW/MOX1 is dependent only on the flow characteristics of the Mark-BW assembly design and is not dependent upon the MOX fuel inside the cladding. This compatibility must extend to protection for departure from nucleate boiling, hydraulic loads, and cross flow velocities (for flow induced fuel rod vibrations). For each of the compatibility requirements, the respective design criteria based on the Standard Review Plan (Reference 9) are used. Thermal-hydraulic effects of mixed core operation will be addressed for each reload core configuration using an approved mixed core methodology. For cores containing Mark-BW/MOX1 lead assemblies, the methodology for departure from nucleate boiling is provided in Reference 4. Hydraulic loads and cross flow effects are discussed in Sections 6.1.9 and 6.1.4, respectively.

A topical report (Reference 3) discusses the neutronic effects of mixing MOX and LEU fuel assemblies in a reactor core. Duke Power will use the methods described in the referenced report to account for the neutronic differences between the two fuel types and to design a core that can be operated safely.

2.5 **Conclusions**

This chapter has introduced the main differences between LEU and MOX fuel and has shown the methodologies that will be used to model those differences. MOX fuel has been safely used in European commercial reactors. Approved methods will be used to demonstrate that MOX fuel assemblies can also be safely used in the U.S. However, in the U.S., weapons-grade plutonium will be used rather than reactor-grade. Chapter 3 shows that the current experience with LEU and RG MOX fuel is applicable to WG MOX fuel.

Table 2.1 Comparison of Typical LEU and WG MOX Fuel Isotopics

Isotope	BOL		EOL	
	LEU Fuel	WG MOX Fuel	LEU Fuel (55,000 MWd/MTU)	WG MOX Fuel (45,000 MWd/MThm)
²³⁴ U	0.03	0.00	0.01	0.00
²³⁵ U	4.10	0.24	0.60	0.09
²³⁶ U	--	--	0.56	0.03
²³⁸ U	95.87	95.39	91.78	92.28
²³⁸ Pu	--	0.00	0.04	0.02
²³⁹ Pu	--	4.04	0.62	1.37
²⁴⁰ Pu	--	0.30	0.29	0.86
²⁴¹ Pu*	--	0.02	0.19	0.51
²⁴² Pu	--	0.00	0.10	0.16
²⁴¹ Am*	--	0.00	0.01	0.02

NOTE: Concentration (wt% of initial heavy metal) for the most abundant heavy metal isotopes in LEU and MOX fuels. Data in the table are given as an example and do not necessarily represent the isotopics of fuel that will be produced.

* Amount varies with decay time.

3.0 Weapons-Grade Plutonium

No data currently exist on the performance of WG MOX fuel in nuclear power reactor cores. However, there are extensive data on the performance of RG MOX fuel and LEU fuel in power reactors. This section highlights the subtle differences between RG MOX and WG MOX, and it demonstrates the applicability of the current experience with LEU fuel and RG MOX to WG MOX. The characteristics and behavior of LEU fuel and MOX fuel derived from RG plutonium provide a sound basis for understanding the impacts of MOX fuel derived from WG plutonium.

MOX fuel is characterized in terms of the source of the plutonium as either WG or RG. WG plutonium has a higher relative percentage of ^{239}Pu than RG plutonium, thus allowing WG MOX to achieve the same burnup with a lower overall plutonium concentration. Typical isotopic values are listed in Table 3.1. The source material for the WG plutonium, derived from actual weapons material, will also typically contain gallium as an alloying agent.

Pure RG and WG PuO_2 are chemically identical, and many of the physical properties outlined in Section 2.1 are also the same. The physical properties that could be affected by WG plutonium are addressed in this chapter. Since the isotopics for WG plutonium are different, the isotopic effects of WG plutonium relative to RG plutonium are addressed in Section 3.1. The impact of gallium as an impurity is addressed in Section 3.2. The effect of WG MOX on pellet microstructure is addressed in Section 3.3. The impact of WG MOX on mixed cores is also addressed in Section 3.1.

The following sections demonstrate the applicability of the current experience with LEU fuel and RG MOX to WG MOX.

3.1 *Isotopics*

RG plutonium is produced from reprocessed spent light water reactor LEU fuel that has been irradiated to commercial burnups, which are typically in the range of 30,000 to 50,000 MWd/MTU. The plutonium isotopes produced at these burnups, and extracted following irradiation, include not only ^{239}Pu but also ^{240}Pu , ^{241}Pu , and ^{242}Pu . In contrast, WG plutonium is created from irradiating ^{238}U to low burnups and separating the plutonium before substantial percentages of the heavier plutonium isotopes (^{240}Pu , ^{241}Pu , and ^{242}Pu) build up. Typical plutonium isotopic compositions of WG and RG material are shown in Table 3.1, and allowable ranges for WG material are shown in Table 3.2. Whereas the typical RG material has about

21% ^{240}Pu , the typical WG material has only about 7% ^{240}Pu . Such differences in isotopics can be addressed through neutronic modeling. The neutronic modeling explicitly treats each plutonium isotope so that any difference in isotopics is directly modeled. As LEU fuel is depleted, the isotopic production of plutonium will include the neutronic effects of WG plutonium (low burnups) and RG plutonium (high burnups). Therefore, if a neutronic code models LEU fuel with depletion with acceptable results, it is able to model all the major plutonium isotopes.

Table 3.3 shows representative isotopic masses of unirradiated LEU, WG MOX, and RG MOX fuel assemblies with the same fuel mechanical design. For the purposes of this calculation, the same total heavy metal loading was used for all three assemblies, and the initial uranium enrichments and plutonium concentrations were chosen to produce the same infinite multiplication factor (k_{∞}) at approximately 20,000 MWd/MThm burnup. The tables show that all three fuel types are predominantly uranium. All the measures of plutonium mass (total, fissile and individual isotope masses) for the WG MOX fuel assembly fall between those of the LEU and RG MOX fuel assemblies.

The use of WG plutonium reduces the PuO_2 content of MOX fuel relative to RG material. Table 3.1 gives typical concentrations of absorber isotopes (^{240}Pu plus ^{242}Pu , rounded to the nearest percent) as 7% for WG material but 24% for RG material. Variations in the content of fissile and absorber isotopes have implications for MOX fuel design. To achieve a given design burnup, an increase in the concentration of absorber isotopes would normally be offset by a larger fissile content, which means a larger total plutonium content. The WG MOX fuel described in Table 3.3 contains only about 4.4% plutonium but allows energy extraction equivalent to that of RG MOX with a plutonium content of about 7.2%.

The information in Table 3.1 and Table 3.3 was used in CASMO-4 infinite lattice neutronic calculations to compare the behaviors of LEU, WG MOX, and RG MOX fuel. The results of these calculations are shown in Figure 3.1 through Figure 3.5 and discussed below.

As nuclear fuel is used, the elemental and isotopic constituents of the fuel change. For LEU fuel, ^{235}U is depleted, plutonium is produced, and the isotopics of the plutonium evolve. The LEU fuel plutonium isotopics are initially similar to those of unirradiated WG MOX fuel, but they evolve toward those of unirradiated RG MOX fuel. For WG MOX fuel, plutonium is depleted, and the isotopics of the plutonium again evolve toward those of RG MOX. For RG MOX fuel, the plutonium is depleted, and the isotopics of the plutonium further degrade, that is, there is a

progressively larger percentage of ^{240}Pu . These characteristics are shown in Figure 3.1, Figure 3.2, and Figure 3.3.

As a result of the changes described above, the sources of fission change markedly with burnup for LEU fuel. However, both RG MOX and WG MOX fuel have little thermally fissionable uranium, so the fissions in both MOX fuel types are approximately 90% plutonium at any burnup. This effect is shown in Figure 3.4. Figure 3.1 through Figure 3.4 show that the characteristics of WG MOX fuel are generally between those of RG MOX and LEU fuel over a range of burnups.

The reactivity change of the fuel with burnup results from the change in elemental and isotopic composition. Depletion of ^{235}U and fissile plutonium (^{239}Pu and ^{241}Pu) reduces reactivity, as does buildup of fertile plutonium (^{240}Pu). Conversely, buildup of fissile plutonium and depletion of fertile plutonium increase reactivity. The net result of these factors on the fuel neutronic performance is illustrated in Figure 3.5, which shows the infinite multiplication factors (k_{∞}) of LEU, RG MOX, and WG MOX fuel assemblies as a function of burnup. LEU fuel reactivity decreases most steeply with burnup, while RG MOX fuel decreases the least. The reactivity behavior of WG MOX fuel lies between that of LEU fuel and that of RG MOX fuel over a range of burnups.

Several important points can be made relative to the calculations discussed above.

- LEU fuel, RG MOX fuel, and WG MOX fuel are fundamentally similar in that they produce energy from a mixture of uranium and plutonium fissions. From a neutronic perspective, they differ only in the relative amounts of various fissionable and fertile isotopes of uranium and plutonium.
- Significant plutonium fissions (up to about 40% of the total) occur in medium- and high-burnup LEU fuel.
- WG MOX fuel has a lower mass of each of the plutonium isotopes than RG MOX fuel. For the same reactivity at the middle of life (20,000 MWd/MThm), the amount of plutonium in WG MOX fuel is less than the amount of plutonium in RG MOX fuel.
- RG MOX fuel has a more complicated mix of plutonium isotopes than WG MOX fuel, particularly at low burnup.
- The reactivity behavior of WG MOX fuel as a function of burnup is between that of LEU fuel and that of RG MOX fuel.

Some important conclusions can be drawn from these points.

- The ability to predict the behavior of cores loaded initially with LEU fuel requires the capability to model plutonium fuel behavior.
- RG MOX fuel, with more plutonium and a more complicated mixture of plutonium isotopes, presents a greater challenge to neutronic modeling methods than does WG MOX fuel.
- Nuclear analysis methods that are demonstrated to model both all-LEU fuel cores and mixed LEU - RG MOX fuel cores with an acceptable accuracy will also be capable of modeling mixed LEU - WG MOX fuel cores with a similar level of accuracy.

Comparison to operating reactor core measurements in both all-LEU fuel cores and mixed LEU – RG MOX fuel cores is the approach that has been used by Duke Power to qualify the CASMO-4 and SIMULATE-3 MOX computer codes for application to WG MOX fuel analyses (Reference 3).

3.2 *Impurities*

Plutonium derived from weapons material will contain small amounts of gallium, but the manufacturing process is designed to reduce the gallium concentration in MOX to trace levels. The design impurity level for gallium for the WG MOX fuel is similar to current trace levels of gallium in LEU fuel. Gallium is a low-melting-point element and is liquid at slightly above room temperature. At high concentrations, it can cause embrittlement in metals and alloys (Reference 10) and is considered undesirable in both the processing and use of MOX fuel. In weapons material, however, gallium serves the purpose of stabilizing the δ phase of plutonium. Less than 1% by weight of gallium is sufficient to stabilize δ -plutonium at room temperature (Reference 11). Larger gallium concentrations are not expected since they would result in unnecessary dilution of the plutonium for weapons, but for the purposes of this report, the maximum gallium concentration in the source of WG plutonium is assumed to be 1.2%.

A concern has been expressed that gallium could cause degradation of the cladding (Reference 10). Also, the gallium could migrate to the cooler regions of the fuel rod, particularly the susceptible heat-affected weld zone, and cause embrittlement and subsequent fuel rod failure.

To eliminate the potential harmful effects of gallium, the DCS fabrication process will utilize an aqueous polishing step to remove gallium from the WG plutonium prior to conversion to the oxide form. The polishing step to be implemented at the MOX Fuel Fabrication Facility (MFFF) utilizes a solvent-extraction process to purify the feed material for conversion to PuO₂ powder.

Other processes, such as ion exchange, may be used for lead assembly fabrication, but the same specifications will apply.

Based on COGEMA experience and predictions, the use of a polishing process will allow production of MOX fuel pellets with gallium levels in the parts-per-billion (ppb) range. Gallium at these extremely low concentrations will not have a detrimental effect on processing equipment or cladding performance for the reasons discussed below.

3.2.1 Effectiveness of Polishing Process

The effectiveness of the polishing process for removing gallium has been evaluated through a series of laboratory tests conducted by Oak Ridge National Laboratory (ORNL) (Reference 12). The ORNL tests introduced gallium in known quantities prior to subjecting the material to the same chemical process planned for the production facility. To allow the measurement of the very small amounts of gallium remaining after the polishing process, the gallium was first activated in ORNL's High Flux Isotope Reactor (HFIR). The decontamination factor (DF) is used to characterize the effectiveness of the polishing operation. Specifically, DF is the ratio of the gallium concentration to plutonium concentration before polishing divided by the ratio after polishing (Reference 12). The ORNL tests confirmed that the DF for the process is greater than 10^5 . The initial gallium concentration is at most 1.2%, so polishing with a DF of greater than 10^5 yields a final gallium concentration less than 120 ppb. The specification for the powder to be processed at the MFFF is expected to impose a 120-ppb gallium limit on the finished PuO_2 .

A WG PuO_2 powder specification has been developed from the existing European RG PuO_2 specification, and, with one exception, the list of impurity elements is the same in both documents. The sole difference between the two lists of impurities is the addition of a gallium limit to the WG PuO_2 powder specification because of the known presence of gallium in the starting materials.

When polished feed PuO_2 powder with a gallium concentration less than 120 ppb is diluted with depleted UO_2 powder, the final gallium concentration in the finished MOX pellet is comparable to the concentration in current LEU fuel. The concentration of gallium in current LEU is described in Section 3.2.2.

3.2.2 Gallium Content of Current LEU Fuels and Components

Trace levels of gallium can be found in current LEU fuel and components. These fuels have operated successfully for decades, with no indication of gallium-related fuel failures.

Furthermore, gallium is produced during operation from the direct activation of zinc that is typically present as an impurity in cladding material and LEU pellets. Gallium is also present as an impurity in LEU fuel rod components (e.g., cladding and plenum springs).

Archive samples of fuel pellets and components have been analyzed at ORNL to determine the levels of gallium in LEU fuels that have operated successfully. The pellet samples analyzed at ORNL represent four batches of FANP (US) fuel fabricated over a five-year period from 1990 through 1994. Both Mark-B (15 × 15) and Mark-BW (17 × 17) fuel types were included, as were pellets from two pellet vendors. The results of these analyses are shown in Table 3.4.

As shown, the gallium level in the archive LEU fuel pellets is approximately 10 ppb. The batches of fuel represented by these archive samples operated successfully, with no indication of cladding degradation or failure. The remaining archive fuel components (the spring and cladding) were found to contain higher levels of gallium. The average gallium content of the fuel rod plenum spring samples was 38 parts per million (ppm), or 38,000 ppb. The presence of gallium in the plenum spring material is significant in that it illustrates the levels of gallium that have been present in fuel components for many years, but the presence of gallium was never known because measurements have not been performed previously at these extremely low levels.

The archive Zircaloy-4 cladding samples contained an average of 275 ppb gallium. This measured gallium level corresponds to the same total mass of gallium in the cladding as would be present in fuel pellets if those pellets had a 50 ppb concentration. (The different concentrations for equivalent gallium mass reflect the different masses of cladding and fuel.) The results of this evaluation are significant in that the mass of gallium introduced in the rod from the WG plutonium is much less than the mass of gallium already present in cladding and fuel pellets that have operated successfully.

The polishing process will reduce the gallium content in the feed plutonium to less than 120 ppb. The concentration of PuO₂ in WG MOX is only about 5%, so, when it is diluted with depleted UO₂, the polished plutonium contributes approximately 6 ppb or less to the gallium content of

the finished MOX pellets. Thus, the finished MOX pellets are expected to contain gallium at approximately 10 to 20 ppb. This level of gallium is consistent with the levels of gallium in MOX fuels that have operated successfully. Thus, the presence of gallium from the WG plutonium presents no additional risk of cladding failure from gallium.

3.2.3 Fuel Performance with Gallium

Testing to determine the effects of gallium on fuel performance, at significantly higher levels than expected in the mission reactor fuel, is currently underway in the Advanced Test Reactor (ATR) (Reference 13). The Average Power Test (APT) began irradiation in January 1998 with two types of MOX fuel:

1. The first fuel type was untreated relative to impurities and contained a gallium concentration of 3.0 ppm.
2. The second fuel type was thermally treated to reduce the impurities and contained gallium at the 1.3 ppm level.

Test rods have been examined after burnups of 8,000, 21,000, and 30,000 MWd/MThm, operating at heat rates of 5 to 10 kW/ft. The burnups are projected to reach 50,000 MWd/MThm during future irradiation cycles. The post-irradiation examinations (PIEs) are aimed at determining the effects of gallium on fuel rod performance, including the potential embrittlement of the Zircaloy-4 cladding. Results from the PIE at 30,000 MWd/MThm are summarized below:

- SEM/microprobe examination of the fuel and cladding revealed no abnormal behavior (Reference 13).
- Gas release does not exceed that in early European MOX (Reference 14).
- No gallium migration to the cladding has been detected. Analyses of unirradiated archive samples and irradiated cladding indicate no transfer of gallium to the cladding within the measurement uncertainty limit (Reference 13).

These tests will be followed and are expected to provide additional assurance that operation of MOX pellets with gallium concentrations as great as 3.0 ppm offers no concern for fuel rod performance.

3.3 ***Pellet Microstructure***

The use of WG plutonium for MOX fuel in place of RG plutonium has the potential to affect fuel performance with respect to:

- Thermal conductivity

- Fission gas release
- Fuel pellet swelling
- Pellet radial power distribution

The plutonium fissile content (^{239}Pu plus ^{241}Pu) of the WG MOX fuel is typically about 93%, whereas the RG MOX fuel is about 75% (see Table 3.1). Further, the RG material contains significantly higher concentrations of ^{240}Pu , which acts as an absorber, reducing the reactivity of the RG material relative to the WG material. As is shown in Table 3.3, the plutonium concentrations for MOX fuel from the WG material is reduced approximately 40% to maintain the same total reactivity as the MOX fuel made from RG material.

On a macroscopic scale, the thermal conductivity and the pellet radial power distribution could be affected by WG MOX. The thermal conductivity of the MOX fuel model defined by the RG experience is a function of the overall plutonium content. Thermal conductivity is a parameter that is determined by chemical and physical properties and is not dependent upon small changes in atomic mass from isotopic differences. Since the WG material has lower overall plutonium concentration, the thermal conductivity of the WG MOX fuel will be less affected than will RG MOX. Therefore, the impact of WG MOX on thermal conductivity is explicitly modeled using the overall plutonium content.

The fuel pellet radial power profile could affect fuel centerline temperatures and fission gas release. The impact of WG MOX on fuel pellet radial power profiles is addressed in Reference 2 as a part of the responses to Requests for Additional Information. In summary, it was shown that using the RG MOX power profile for WG MOX has an insignificant effect on fuel temperatures and fission gas release. Therefore, the macroscopic effects of WG MOX relative to RG MOX are either modeled explicitly or are insignificant.

On a microscopic scale (10-400 μm), almost all the power and the fission products produced by the fission process originate in the agglomerates. This “pseudo power density” of the agglomerate for WG MOX could affect the localized heating and fission product production if the fissile density of WG MOX were significantly different than that of RG MOX. The distribution of fissile material within the $\text{PuO}_2\text{-UO}_2$ matrix is controlled by the manufacturing process. In the MOX fuel fabrication process using RG material, the primary blending and micronization is performed with a UO_2/PuO_2 ratio of 70/30. This process step establishes the fissile content of the plutonium-rich agglomerates. The master mix is then diluted with UO_2 to reach the final

plutonium concentration. Thus, the microstructure of the pellet from RG material consists of a uniform UO_2 matrix with uniformly distributed PuO_2 - UO_2 agglomerates containing 30% PuO_2 . In the MOX fuel fabrication process using WG material, the primary blending and micronization will be performed with a nominal UO_2/PuO_2 ratio of 80/20.

The concentration of fissile plutonium in the agglomerates can be determined as follows. For a 70/30 master mix composition and the typical RG isotopics given in Table 3.1,

$$\begin{aligned} &\text{mass fraction of fissile plutonium, as fraction of heavy metal} = \\ &(\text{fraction of PuO}_2 \text{ in master mix}) \times \{\text{fraction of fissile Pu in RG PuO}_2\} = \\ &0.30 \times (0.674 + 0.076) = 0.225. \end{aligned}$$

Similarly, an upper bound on the concentration of fissile plutonium in an 80/20 master mix with WG plutonium can be calculated from the maximum concentrations of ^{239}Pu and ^{241}Pu given in Table 3.2:

$$\begin{aligned} &\text{mass fraction of fissile plutonium, as fraction of heavy metal} = \\ &(\text{fraction of PuO}_2 \text{ in master mix}) \times \{\text{fraction of fissile Pu in WG PuO}_2\} = \\ &0.20 \times (0.950 + 0.010) = 0.192. \end{aligned}$$

The mass fraction of fissile plutonium will be slightly smaller for typical isotopics. Thus, the mass fraction of fissile plutonium in an 80/20 master mix with WG plutonium is comparable to that in a 70/30 master mix with RG plutonium.

Using the same process as used with the RG material, this master mix with a UO_2/PuO_2 ratio of 80/20 is diluted with UO_2 to reach the final plutonium concentration. The WG material has a higher fissile content and less ^{240}Pu parasitic material, but the larger uranium content of the 80/20 master mix provides a compensating dilution of the fissile material. As a result, the plutonium-rich agglomerates from the WG material will have a fissile content, that is, the number of fissile nuclei per unit volume, that is less than or equal to that for the agglomerates in fuel produced from RG material using the 70/30 ratio. In addition, the limits on agglomerate size are the same for WG MOX and RG MOX. Since the fissile density for WG MOX is comparable to that of RG MOX, the pseudo power density of the agglomerates of the WG MOX is also comparable to that of RG MOX. Therefore, local heating in the agglomerates will be the same or less severe in the MOX pellets from WG plutonium than it is in MOX pellets made from RG material.

The nominal 80/20 mix for the WG material is within the COGEMA/BELGONUCLEAIRE experience base for the MIMAS fuels produced in Europe. The MIMAS process has been qualified in Europe for a range of UO_2/PuO_2 mixtures, including the 80/20 mix to be used for the WG material. Production quantities of MIMAS fuel using a plutonium primary blend of 20.5% to 25.9% were fabricated for the SENA reactor. This fuel used plutonium with a fissile content (^{239}Pu plus ^{241}Pu) of 75%. These fuels were irradiated in SENA for three cycles with no problems or issues (Reference 15).

The following conclusions can be drawn regarding microscopic effects:

- The UO_2 matrix that establishes the overall pellet microstructure is the same since the same process and the same type of feed UO_2 are used in both cases.
- The grain size, particle size, and particle distribution will be the same since the process is the same in terms of blender operation, size of sieves, pressing conditions, and sintering conditions.
- Local heating in the agglomerates will be the same (or possibly less severe in WG MOX) since the particle size and distribution are the same, and the master mix adjustment has maintained or reduced the density of fissile nuclei in the plutonium-rich agglomerates.

It can be concluded that WG MOX fuel will perform like RG MOX fuel for considerations involving pellet thermal-mechanical behavior (thermal conductivity and pellet radial power distributions), fission gas release, transient response, and swelling.

3.4 **Conclusions**

This chapter addresses the differences between WG MOX and RG MOX. The differences examined include isotopics, impurities, and pellet microstructure. In each case, it is shown that the characteristics and behavior of LEU fuel and MOX fuel derived from RG plutonium provide a sound basis for understanding the characteristics and behavior of MOX fuel derived from WG plutonium. Neutronic methods that are qualified for LEU and RG MOX applications can also be applied to WG MOX. The aqueous polishing process provides a means for reducing the gallium content of WG MOX to acceptable levels. Since WG MOX and RG MOX use the same processing and WG MOX has a smaller total concentration of plutonium, the microstructural properties derived for RG MOX are applicable to WG MOX. These results support the lead assembly and batch implementation of the Mark-BW/MOX1 fuel design.

Table 3.1 Sample Unirradiated Nuclear Fuel Isotopics

Isotope	Isotopic Fractions		
	LEU	RG MOX	WG MOX
²³⁵ U	4.1%	0.25%	0.25%
²³⁸ U	95.9%	99.75%	99.75%
²³⁹ Pu	0.0%	67.4%	92.5%
²⁴⁰ Pu	0.0%	21.1%	6.9%
²⁴¹ Pu	0.0%	7.6%	0.5%
²⁴² Pu	0.0%	3.0%	0.05%

NOTE: Any discrepancy in the total heavy metal loading is due to the presence of trace quantities of ²³⁸Pu. Information in this table is for purposes of illustration only.

**Table 3.2 Acceptable Isotopics
for WG Plutonium**

Plutonium Isotope	Acceptable Range (wt%)
^{238}Pu	< 0.05
^{239}Pu	90.0 - 95.0
^{240}Pu	5.0 - 9.0
$^{241}\text{Pu}^*$	< 1.0
^{242}Pu	< 0.1

NOTE: These isotopic ranges are not entirely consistent with a common definition of WG plutonium, i.e., less than 7% ^{240}Pu . However, the ranges are consistent with the 2000 U.S.-Russian Federation Agreement on Plutonium Disposition Agreement which defines WG plutonium as plutonium with an isotopic ratio ($^{240}\text{Pu}/^{239}\text{Pu}$) of no more than 0.10.

* Amount varies with decay time. ^{241}Am is included because it is produced from ^{241}Pu by beta decay.

Table 3.3 Sample Unirradiated Nuclear Fuel Composition

	Mass (kg)		
	LEU	WG MOX	RG MOX
Heavy Metal Loading	462.6	462.6	462.6
Total Uranium	462.6	442.4	428.9
²³⁵ U	19.0	1.1	1.1
²³⁸ U	443.5	441.3	427.8
Total Plutonium	0.0	20.2	33.2
Fissile Plutonium	0.0	18.8	24.9
²³⁹ Pu	0.0	18.7	22.4
²⁴⁰ Pu	0.0	1.4	7.0
²⁴¹ Pu	0.0	0.1	2.5
²⁴² Pu	0.0	0.0	1.0

NOTE: Any discrepancy in the total heavy metal loading is due to the presence of trace quantities of ²³⁴U, ²³⁸Pu, and ²⁴¹Am. Information in this table is for purposes of illustration only.

Table 3.4 Gallium in LEU Fuel and Components

Fuel Pellets

Unit	Fuel Type	Pellet Vendor	Nominal Enrichment (²³⁵ U)	Date of Manufacture	Pellet Gallium Content (Avg. 5 samples) (ppb)
Catawba Unit 1	Mark-BW (17 × 17)	General Electric	3.55%	October 1990	11.5
McGuire Unit 2	Mark-BW (17 × 17)	Siemens	3.65%	December 1992	8.7
Three Mile Island	Mark-B (15 × 15)	Siemens	4.75%	June 1993	9.0
Davis-Besse	Mark-B (15 × 15)	Siemens	3.79%	May 1994	10.8

NOTE: Average Pellet Gallium Content – 10.0 ppb ± 2.7 ppb.

Fuel Components

Component	Number of Samples	Average Gallium Content
Plenum Spring	9	38,000 ppb
Zircaloy-4 Cladding	6	275 ppb

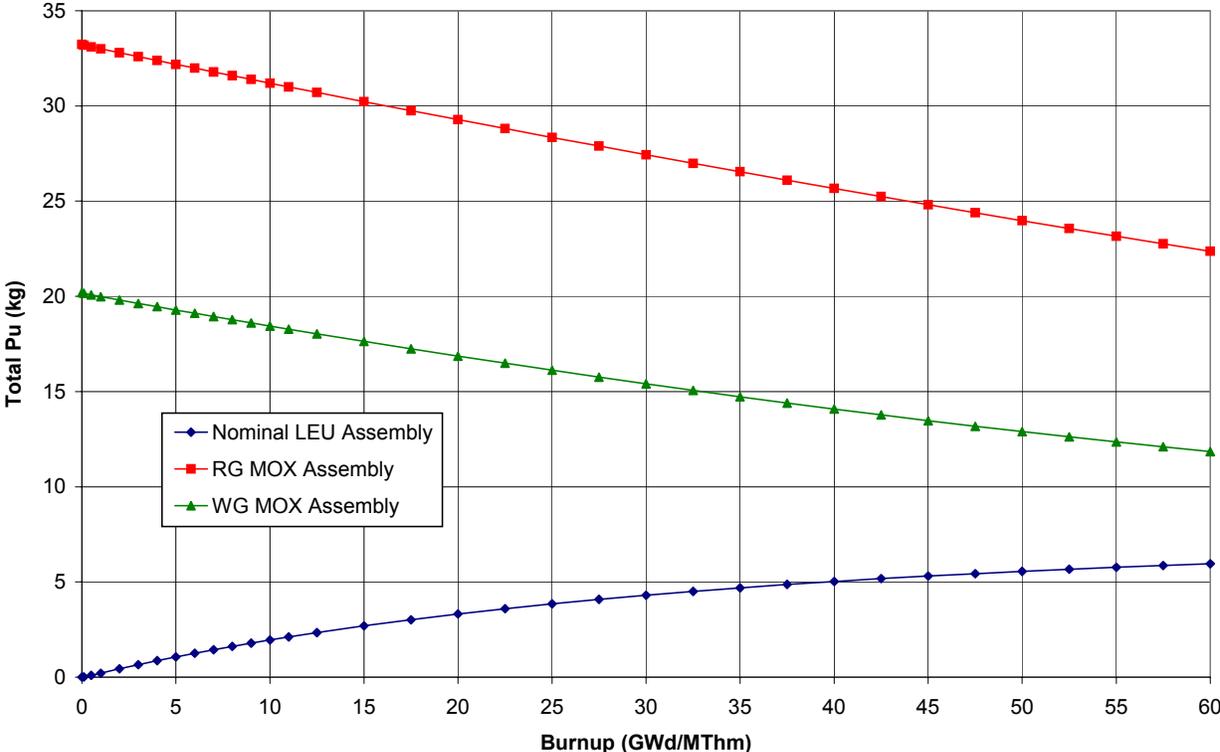


Figure 3.1 Total Plutonium Mass per Assembly

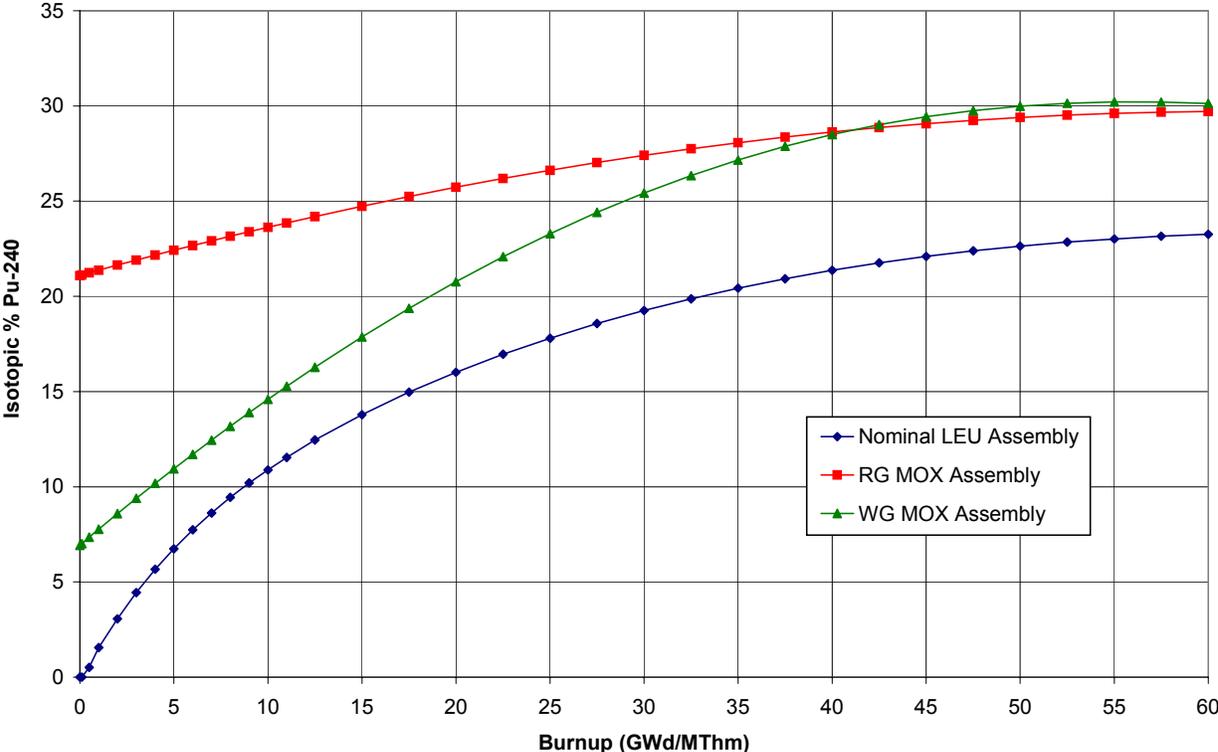


Figure 3.2 ²⁴⁰Pu Concentration as a Fraction of Total Plutonium

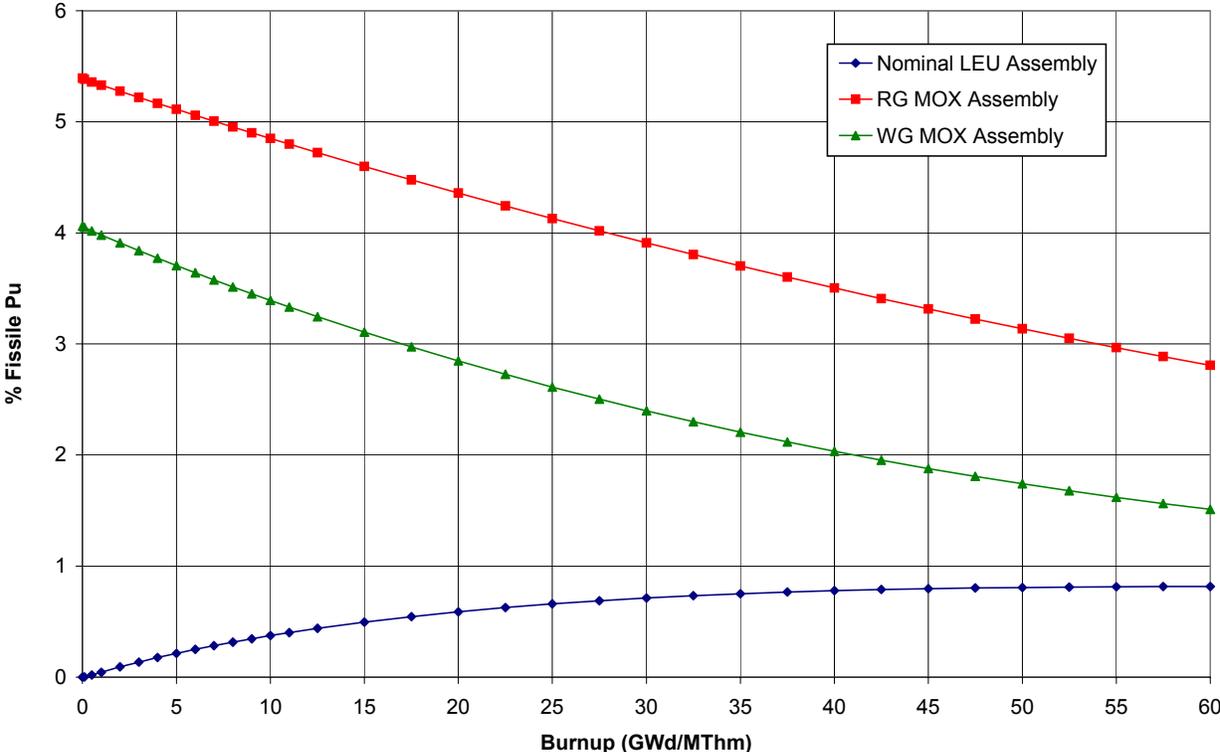


Figure 3.3 Fissile Plutonium as a Fraction of Total Heavy Metal

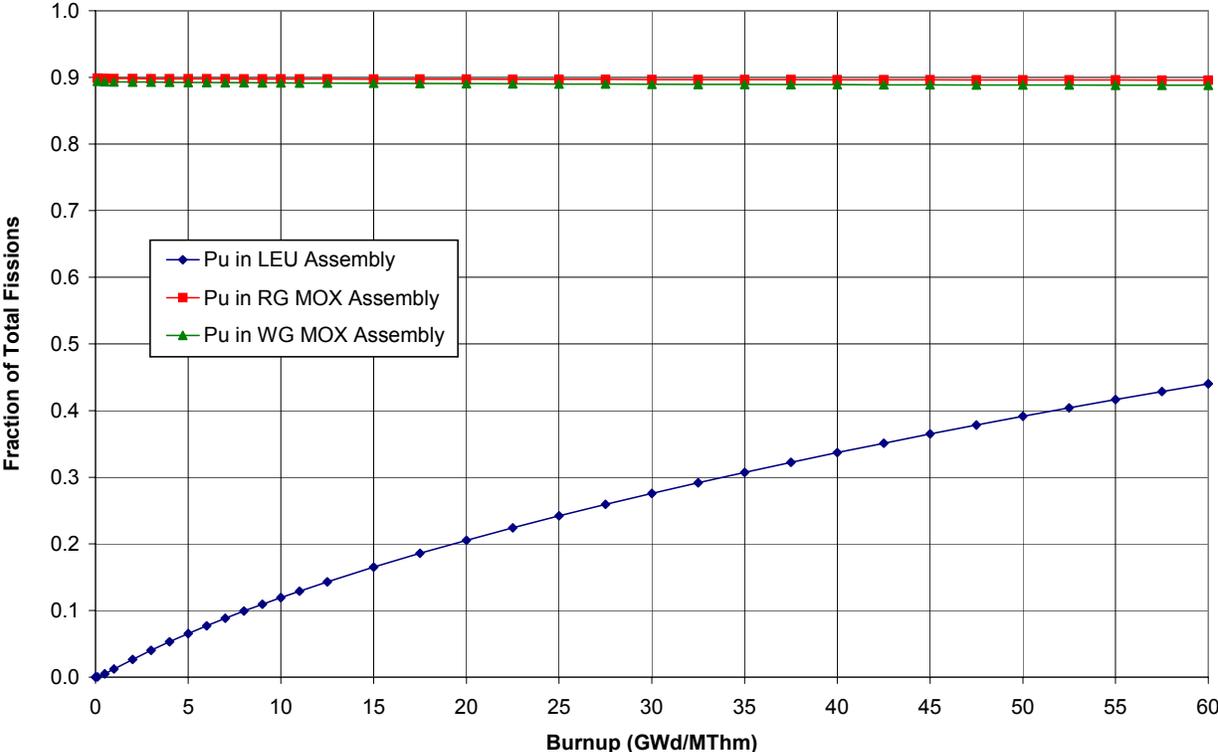


Figure 3.4 Plutonium Fissions as a Fraction of Total Fissions

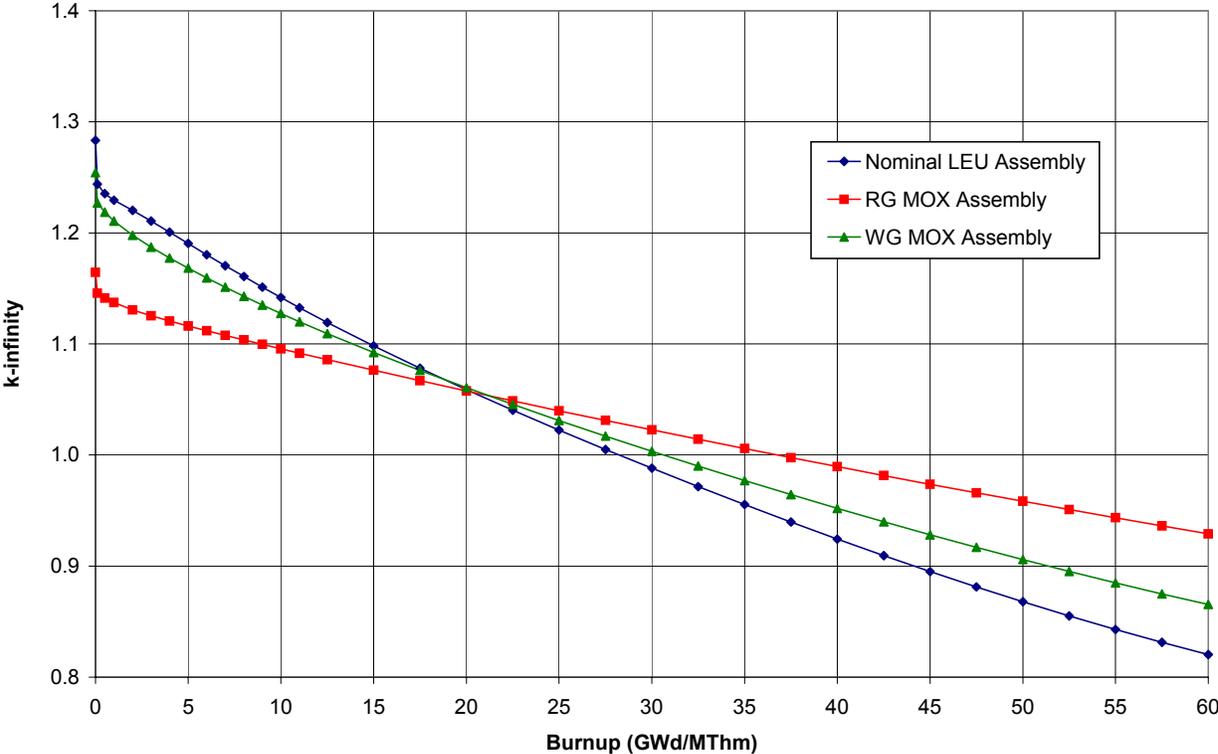


Figure 3.5 k_{∞} vs. Burnup

4.0 **Manufacturing Processes**

The MIMAS fabrication process developed by BELGONUCLEAIRE and COGEMA will be replicated in the U.S. facility for producing the MOX fuel. In the U.S. implementation, the standard MIMAS processing will be preceded by an aqueous polishing step to remove gallium. The use of polished plutonium ensures that the MOX fuel produced with the MIMAS process in the U.S. with WG plutonium will be free of any gallium effects and thus consistent with the MOX fuel produced and irradiated in Europe. This link to the European MOX fuel fabrication process provides assurance that the materials and operational data from Europe are applicable to the U.S. program. The link is strengthened by the use of a European MOX fabrication plant for lead assembly fabrication (as discussed in Section 8.0 of this report).

Licensing of the U.S. fabrication facility is not included in the scope of this report. The information provided here is intended to show that MIMAS is an established process for producing MOX fuel pellets.

4.1 **Process Description**

The MIMAS process for fabricating MOX fuel for LWRs was developed by BELGONUCLEAIRE and COGEMA to produce fuel pellets characterized by an intimate dispersion of plutonium in the fuel matrix. (See Figure 4.1 for the MIMAS process outline, combined with the aqueous polishing step unique to the processing of WG PuO₂.) The MIMAS process is currently in use at the BELGONUCLEAIRE P0 plant located at Dessel, the COGEMA Cadarache plant, and the COGEMA MELOX plant.

The process was developed in 1984 by BELGONUCLEAIRE to meet requirements for finely dispersing the plutonium while maintaining a pellet microstructure similar to that of a standard LEU fuel pellet. To achieve these objectives, the PuO₂ powder is micronized with UO₂ powder and sintered recycled scrap to form a master blend with a plutonium content typically in the range of 20% to 35% of the total mass. The successive blending and sieving steps deliver fine, plutonium-rich, master blend particles.

This master blend is force-sieved and then mechanically diluted and mixed with free-flowing UO₂ powder to obtain the specified plutonium content of the MOX fuel. This process maintains the physical characteristics of the UO₂ powder while reducing the heterogeneity of the plutonium distribution.

After final blending, the fuel is processed, as in LEU fuel fabrication, by pressing the final blend into green pellets, sintering, grinding, and inspecting the pellets before loading them into rods.

The fuel properties and performance of MIMAS-produced MOX pellets have been collected in a database that has been used for code benchmarking and verification. The database is discussed in Chapter 7; benchmarking and verification are discussed in Reference 2. Since the European MIMAS process will be used directly for lead assembly fabrication and replicated for MFFF production, this database remains applicable to the fuel produced for the WG plutonium disposition program. Applicability is assured by the use of product specifications, both for the incoming UO_2 and PuO_2 powder and for the MOX pellets, that are consistent with those used for FANP-designed MOX fuel produced in the three European MOX plants that use the MIMAS process.

4.2 **Conclusions**

MIMAS is an established process for producing MOX fuel pellets. The use of this process supports the applicability of European MOX experience to the irradiation of WG MOX fuel.

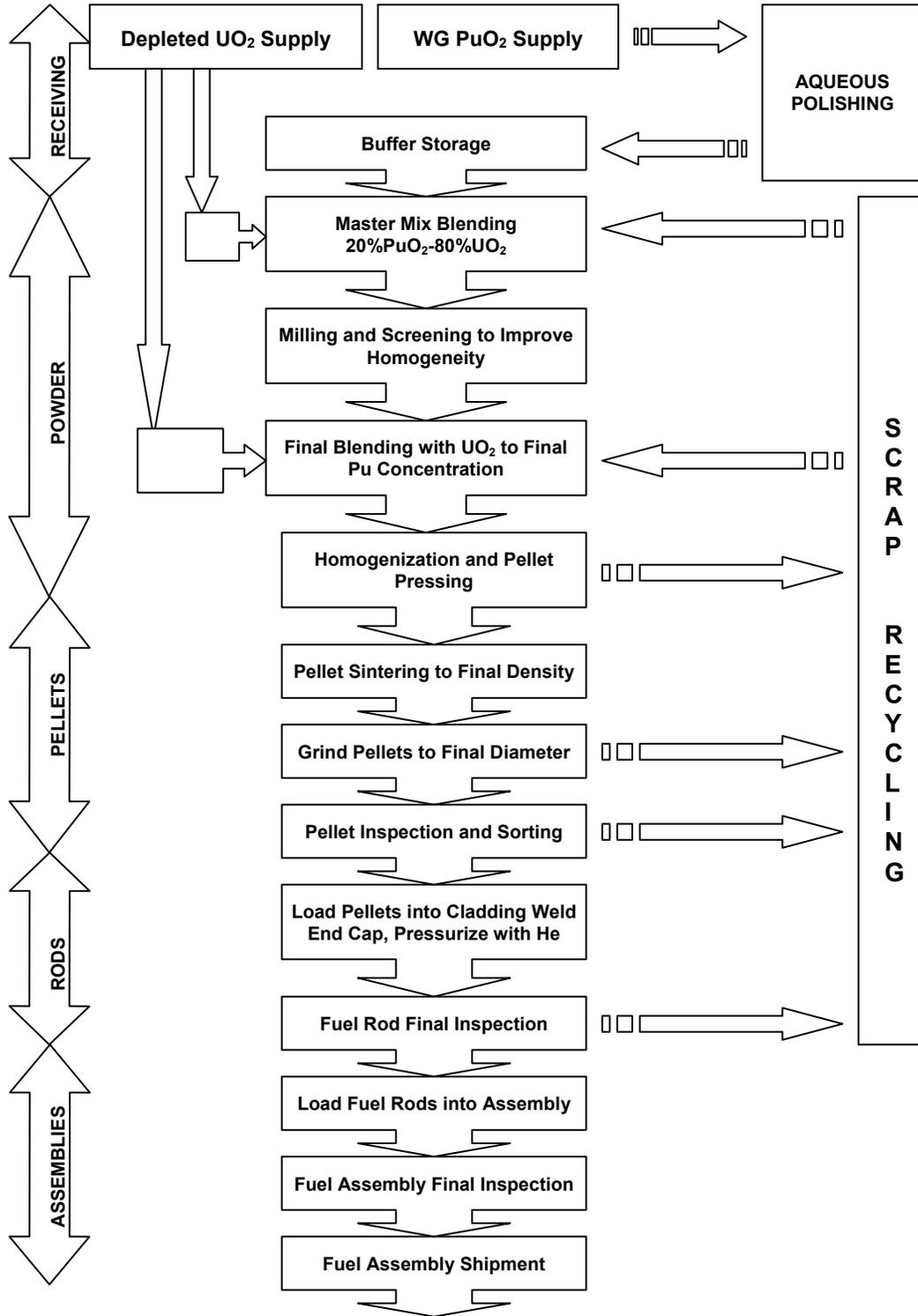


Figure 4.1 MIMAS Flow Diagram

5.0 **Mark-BW/MOX1 Fuel Assembly Description**

The Mark-BW/MOX1 fuel assembly design is the Advanced Mark-BW fuel assembly design with the LEU fuel rods replaced by MOX fuel rods. The Advanced Mark-BW fuel assembly (Reference 1) is a 17 × 17, standard lattice fuel assembly specifically designed for Westinghouse 17 × 17 reactors. The Mark-BW/MOX1 fuel assembly design includes the following base features of the Advanced Mark-BW fuel assembly:

- Seated fuel rods
- Floating intermediate spacer grids
- Removable top nozzle
- Mid-span mixing grids
- TRAPPER™* bottom nozzle
- M5®* alloy for fuel rod cladding, guide thimbles, and intermediate spacer grids

5.1 **MOX Fuel Rod Design**

The fuel rod design consists of UO₂-PuO₂ (MOX) pellets contained in a seamless M5® tube with M5® end caps welded at each end. The fuel stack is nominally 144 inches long, and the fuel pellets have a diameter of 0.3225 inch. The fuel rod cladding has a 0.374-inch outside diameter and a 0.0225-inch wall thickness. This configuration leaves a small clearance (approximately 0.003-inch radial clearance) between the inside diameter of the cladding and the outside diameter of the fuel pellets.

The fuel rod utilizes one stainless steel spring in the upper plenum to prevent the formation of fuel stack axial gaps during shipping and handling while also allowing for the expansion of the fuel stack during operation. The fuel stack rests on the lower end cap. The lower end cap is made from M5® and has a bullet-nose shape to provide a smooth flow transition in addition to facilitating reinsertion of the rods into the assembly if any rods are removed after the assemblies have been irradiated (e.g., during fuel examination programs). The upper end cap is also made of M5® and has a grippable shape that allows for the removal of the fuel rods from the fuel assembly, if necessary. The fuel rod is filled with helium gas prior to final sealing.

The fuel pellets are a sintered ceramic of high density UO₂-PuO₂. The fuel pellets are cylindrically shaped with a dish at each end. The top and bottom rims of the pellets have a

* TRAPPER and M5 are trademarks of Framatome ANP, Inc.

chamfer that eases the loading of the pellets into the cladding. The dish and chamfer geometry also reduces the tendency for the pellets to assume an hourglass shape during irradiation. The design density of the pellets is 95% of theoretical density (TD), with an expected maximum plutonium content of 6%.

The schematic diagram of Figure 5.1 shows an axial cross section of the MOX fuel rod for the Mark-BW/MOX1.

5.2 **MOX/LEU Design Comparison**

A comparison of typical fuel rod design details for the MOX and LEU fuel rod designs is summarized in Table 5.1. The MOX fuel rod design differs from the LEU fuel rod design only in the areas of fuel rod length, design density, and maximum fuel rod burnup.

- Fuel Rod Length – The additional fission gas release from the MOX fuel is accommodated by increasing the fuel rod length and, thus, the plenum volume. This increase in rod length can be incorporated in the MOX design while maintaining the required fuel rod shoulder gap due to the lower burnup limit for the MOX design. (See Sections 6.1.7 and 6.1.8.)
- Design Density – The design density for the MOX design is 95% TD whereas the LEU fuel rod design utilizes 96% TD pellets. The selection of 95% TD for the MOX was made to be consistent with European experience with RG MOX.
- Dish and Chamfer Design – The LEU Advanced Mark-BW uses a U.S. dish and chamfer design, whereas the Mark-BW/MOX1 uses a European design.
- Maximum Fuel Rod Burnup – The objectives of the fissile material disposition program, i.e., burning and degrading WG plutonium, can be accommodated with a maximum fuel rod burnup that is less than that currently being used for LEU fuels.

The differences in dish and chamfer volume and design density, along with the difference in theoretical density between UO_2 and PuO_2 result in a small difference in heavy metal loading.

5.3 **MOX Neutronic Design**

The Mark-BW/MOX1 neutronic design will use a three-zone plutonium distribution planned for batch implementation (with the average plutonium content adjusted as necessary). A sample zoned design is shown in Figure 5.2; the final zoned configuration will be optimized as part of the fuel cycle design and is not addressed in this report. The zoning design is intended to optimize the trade-off between core management and production efficiency for batch implementation. Radial zoning was used in some early MOX tests (Reference 16) and is routinely used in European MOX fuel. The use of radial zones within the fuel assembly is not

unique to MOX, having been implemented previously on FANP (US) Mark-B (LEU) fuel assemblies.

5.4 **Conclusions**

The Mark-BW/MOX1 fuel assembly design is the Advanced Mark-BW fuel assembly design with the LEU fuel rods replaced by MOX fuel rods. The minor design changes associated with the MOX fuel rods have been justified in light of the properties of MOX. Use of the Advanced Mark-BW design (Reference 1) in conjunction with the technical bases of the MOX fuel rod as defined in this report supports both the lead assembly program and batch implementation of the Mark-BW/MOX1 fuel assembly design.

Table 5.1 Mark-BW/MOX1 Preliminary Design Summary

Parameter	Value	
	Advanced Mark-BW	Mark-BW/MOX1
Pellets (Reference 1)		
Fuel Pellet Material	Enriched UO ₂	PuO ₂ and Depleted UO ₂
Fuel Pellet Diameter, in	0.3225	0.3225
Fuel Pellet Density, % TD	96	95
Fuel Pellet Volume Reduction Due to Chamfer and Dish, %	1.24	1.11
Rods (References 1 and 17)		
Fuel Rod Length, in	152.16	152.40
Fuel Rod Cladding Material	M5 [®]	M5 [®]
Fuel Rod Inside Diameter, in	0.329	0.329
Fuel Rod Outside Diameter, in	0.374	0.374
Active Fuel Stack Height, in	144	144
Maximum Fuel Rod Burnup, MWd/MThm	62,000	50,000
Assemblies (Reference 1)		
Fuel Assembly Length, in	159.85	159.85
Lattice Geometry	17 × 17	17 × 17
Fuel Rod Pitch, in	0.496	0.496
Number of Fuel Rods per Assembly	264	264
Heavy Metal Loading per Assembly, kg	466	463
Number of Grids		
Bottom End	1	1
Vaneless Intermediate	1	1
Vaned Intermediate	5	5
Mid-Span Mixing	3	3
Top End	1	1

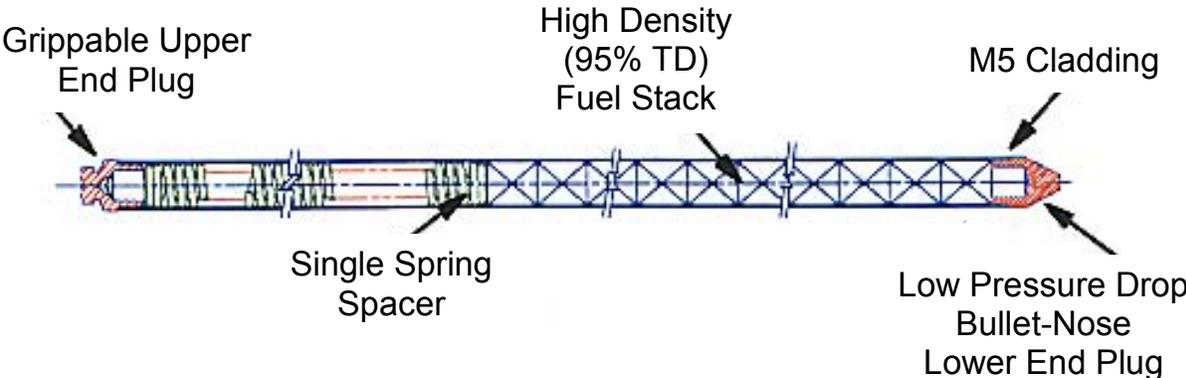


Figure 5.1 Mark-BW/MOX1 Fuel Rod Design

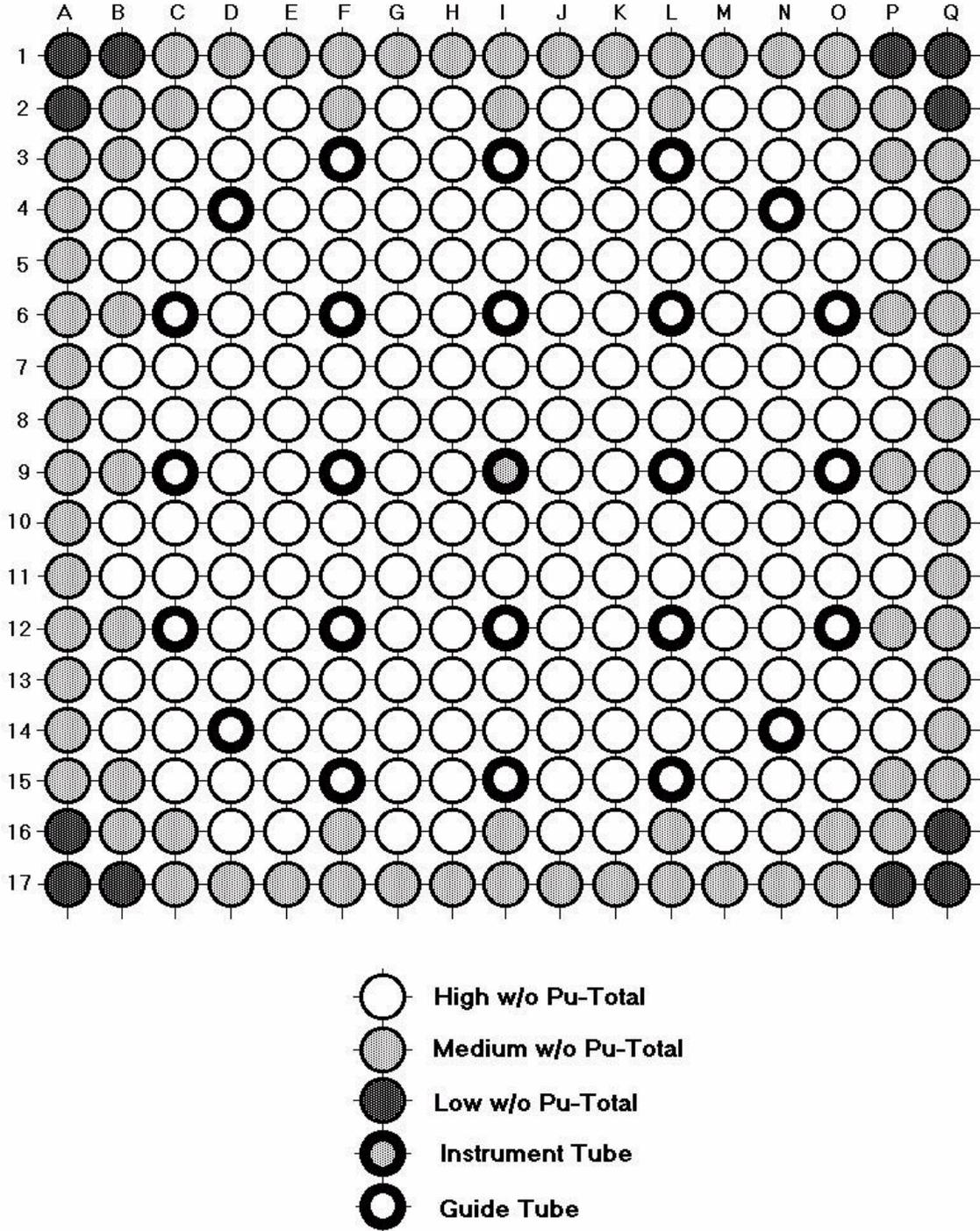


Figure 5.2 Mark-BW/MOX1 Fuel Assembly Design

NOTE: From Reference 5.

6.0 **Mark-BW/MOX1 Design Evaluation**

As discussed in Section 5.0 of this report, the Mark-BW/MOX1 fuel assembly is the Advanced Mark-BW fuel assembly design, with the LEU fuel rods replaced by MOX fuel rods. Therefore the fuel assembly mechanical criteria and methods are directly referenced to those provided for the Advanced Mark-BW design in Reference 1. An example design evaluation is presented to ensure that the Mark-BW/MOX1 fuel assembly design meets all applicable criteria to maintain safe plant operation. The mechanical analysis demonstrates that the fuel assembly satisfies the requirements outlined in Section 4.2 of the Standard Review Plan (SRP) (Reference 9). Future analysis updates to incorporate design modifications or input changes will use the same methods and criteria.

The design of the Mark-BW/MOX1 is such that it preserves the interface with resident fuel assemblies and all reactor internals and all equipment for normal handling. The Mark-BW/MOX1 is designed to preserve the original plant licensing bases for all reactor internal components.

Fuel rod analysis methods that are affected by the incorporation of MOX utilize the COPERNIC fuel performance code (Reference 2). Analysis methodology and MOX-specific models are provided in the COPERNIC topical report. COPERNIC is also used to provide pressures, oxide thicknesses, and strains for mechanical analyses that use approved methods from other sources. The mechanical and thermal analyses presented in the following paragraphs will be redone, if necessary, when final fuel cycle design information is available. If rod design changes are necessary, revisions will meet the same criteria presented herein. The fuel rod design used for these analyses is presented in Table 5.1, with a comparison to the Advanced Mark-BW (LEU) design.

6.1 ***Fuel System Damage***

6.1.1 Stress

6.1.1.1 Fuel Assembly Stress

The Mark-BW/MOX1 fuel assembly stress intensities were shown to be less than the stress limits based on the American Society of Mechanical Engineers (ASME) Code Section III criteria per Reference 1. The following fuel assembly components were evaluated:

- Guide thimble assembly

- Top and bottom nozzles
- Grids/grid restraint
- Quick disconnect
- Holddown spring assembly
- Instrument sheath
- Fuel rod cladding

Positive margins were determined for all fuel assembly structural components, showing that the Mark-BW/MOX1 fuel assembly is structurally acceptable for normal operating conditions.

6.1.1.2 Cladding Stress

The fuel rod cladding was analyzed according to the design criterion established in Reference 1 for the stresses induced during operation, using the approved methodology for M5[®] cladding (Reference 18). Conservative values were used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential temperature, and unirradiated cladding yield strength. The analyses of the fuel rod cladding stresses demonstrated positive margins for all operating conditions. The pressure on the cladding was also shown not to cause buckling. With the worst cases for geometry and pressures, the cladding was shown to have an acceptable margin to the pressure that would cause buckling.

6.1.2 Cladding Strain

The Mark-BW/MOX1 fuel rod was analyzed according to the design criterion established in Reference 1, which limits the fuel rod transient strain to 1% for Condition I and II events. The maximum local linear heat rate corresponding to the 1% transient strain criterion was determined, using the COPERNIC code and methodology (Reference 2). The transient strain limit uses cladding circumferential changes before and after a linear heat rate (LHR) transient to determine strain. [

] The calculated LHRs vary [

]. The transients that induce 1%

cladding strain are not limiting to the plant's operation (comparable to LEU fuel) and are much more severe than the maximum transient the fuel rod is expected to experience.

6.1.3 Cladding Fatigue

The Mark-BW/MOX1 fuel rod was analyzed according to the design criterion established in Reference 1, which limits the total fatigue usage factor to less than 0.9, while using an approved methodology (Reference 18) and the procedures outlined in the ASME Code. All possible Condition I and II events expected and one Condition III event were analyzed to determine the total fatigue usage factor experienced by the fuel rod. Conservative inputs in terms of cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, and differential temperature across the cladding were assumed.

The fatigue utilization factor for the fuel rod was shown to be [], which is well within the 0.9 limit. COPERNIC was used to predict the effects of operational transients on cladding temperatures, pellet diameter, and rod internal pressures in the fatigue calculations.

6.1.4 Fretting

The Mark-BW/MOX1 design was evaluated according to the design criteria established in Reference 1, which

- limits the span average cross-flow velocities to less than 2 feet per second, and
- requires that the fuel assembly design to be shown to provide sufficient support to limit fuel rod vibration and clad fretting wear.

The evaluation made for the Advanced Mark-BW fuel assembly per Reference 1 is applicable for the Mark-BW/MOX1 design. The differences between the MOX and LEU fuel rods, i.e., a small cladding length change and the substitution of MOX pellets for LEU pellets, have a negligible effect on the fuel assembly cross flow velocity and do not alter fuel rod or fuel assembly vibration characteristics. Mixed-core analyses considering resident fuel included those with and without mid-span mixing grids, and they demonstrated that the maximum cross flow velocity was less than []. The Mark-BW/MOX1 fuel rod fretting performance is based on that of the Advanced Mark-BW design presented in Reference 1, which includes the proven performance of the standard Mark-BW; the successful three-cycle operation of the lead assemblies in North Anna Unit 1; out-of-core life and wear and flow-induced vibration testing; and analytical benchmarks and evaluations.

6.1.5 Oxidation, Hydriding, and Crud Buildup

The Mark-BW/MOX1 design was evaluated according to the design criterion established in Reference 1, which limits the best-estimate corrosion of the fuel rod cladding to 100 μm , per Reference 19. Hydrogen pickup is controlled by the corrosion limit.

The Mark-BW/MOX1 fuel rod was analyzed for cladding corrosion using the COPERNIC code (Reference 2) using models approved for predicting M5[®] cladding oxide thickness. The results of the analysis are shown in Figure 6.1. The analysis demonstrates that the predicted oxide thickness for MOX fuel is [], which is well below the design limit of 100 μm . The hydrogen pick-up rate of the M5[®] cladding has been found to be approximately []. At the predicted corrosion level, the maximum hydrogen content of the M5[®] cladding at 50,000 MWd/MThm is approximately []. The upper limit for hydrogen pick-up is []. This level of corrosion and associated hydriding will not adversely affect the structural integrity of the fuel rod during its design lifetime. Also, M5[®] cladding (Reference 18) has been shown through in-reactor testing to exhibit acceptable performance in terms of oxide film thickness growth and hydrogen uptake to fuel rod burnups in excess of 62,000 MWd/MThm. That value is greater than the maximum MOX design fuel rod burnup, 50,000 MWd/MThm.

6.1.6 Fuel Rod Bow

The Mark-BW/MOX1 fuel rod bow was evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly, which is in accordance with Reference 1. The evaluation made for the Advanced Mark-BW fuel assembly per Reference 1 is applicable for the Mark-BW/MOX1 design. The differences between the MOX and LEU fuel rods, i.e., a small cladding length change and the substitution of MOX pellets for LEU pellets, have a negligible effect on the fuel rod bow. Thus, the Mark-BW/MOX1 fuel design will not have greater rod bow than Mark-BW Zircaloy-4 design. European post-irradiation examinations of MOX fuel have confirmed the applicability of the LEU rod bow experience to MOX fuel. The rod bow peaking penalty for the Mark-BW/MOX1 design will be based upon the amount of rod bow evaluated in the Advanced Mark-BW Mechanical Design Topical Report (Reference 1).

6.1.7 Axial Growth

The Mark-BW/MOX 1 fuel assembly was evaluated according to the design criteria established in Reference 1, which require that

- the fuel assembly-to-reactor internals gap allowance be designed to provide positive clearance during the assembly lifetime, and
- the fuel assembly top nozzle-to-fuel rod gap allowance be designed to provide positive clearance during the assembly lifetime.

The Mark-BW/MOX1 axial gap between the top nozzle and reactor internals was analyzed to show that sufficient margin exists to accommodate the fuel assembly growth for the design burnup (Reference 1). This analysis utilized a conservative maximum fuel assembly growth prediction, including the effects of the higher MOX fast neutron fluence. The minimum fuel assembly-reactor core plate gap at end of life for a 60,000 MWd/MThm maximum assembly burnup was determined to be [] under worst-case (cold) conditions.

The Mark-BW/MOX1 fuel rod shoulder gap, i.e., the axial gap between the top nozzle adapter plate and the fuel rods, was analyzed to show that sufficient margin exists at the design rod average burnup (50,000 MWd/MThm) to accommodate the fuel assembly growth and the fuel rod growth. The calculation included the effects of the MOX fast neutron fluence. The analysis modeled the shoulder gap directly, as opposed to applying separate axial growth predictions to the fuel assembly structure and fuel rod assembly. Utilizing the axial growth model for M5[®] cladding and guide thimbles, the conservative analysis predicted that an acceptable shoulder gap would exist at a burnup of 60,000 MWd/MThm under hot EOL conditions. The shoulder gap would be larger under cold EOL conditions or at smaller burnups, such as the MOX assembly design burnup. The calculation used a conservative approach, so the actual shoulder gap will be larger.

6.1.8 Fuel Rod EOL Pressure

The Mark-BW/MOX1 fuel rod was evaluated for internal pressure according to the approved methods and criteria presented in Reference 2 for MOX fuel rods. The analysis results indicate that the rod design presented in Table 5.1 meets the fuel rod internal pressure criterion (Reference 2). Some of the analysis inputs (in particular, the fuel rod power history), are preliminary and conservative in nature. As inputs are revised, the analyses will be redone using the approved methods and criteria of Reference 2.

6.1.9 Assembly Liftoff

The Mark-BW/MOX1 was evaluated according to the design criteria established in Reference 1. The design criteria require that

- the fuel assembly holddown springs must be capable of maintaining fuel assembly contact with the lower support plate during normal operating conditions and Condition I and II events, except for the pump overspeed transient
- the fuel assembly shall not compress the holddown spring to solid height for any Condition I or II event, and
- the fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all Condition I through IV events.

The assembly liftoff evaluation is plant-specific and depends on the plant flow rates, bypass flow, fourth-pump startup temperature, and resident fuel pressure drop. The differences between the MOX and LEU fuel rods, i.e., a small cladding length change and the substitution of MOX pellets for LEU pellets, have no effect on the Mark-BW/MOX1 pressure drop.

Conservative values for fast fluence on the holddown spring arising from the specific MOX fuel neutron spectrum are used to establish the holddown spring relaxation characteristics. The higher fast fluence increases holddown spring relaxation. The resulting holddown spring loads and corresponding fuel assembly liftoff margins through the life of the fuel assembly are conservatively calculated. The analysis showed that the Mark-BW/MOX1 fuel assembly will not lift off under any normal operating condition. The minimum margin-to-fuel assembly liftoff occurs at end of life at 85 °F. For the pump overspeed condition, the fuel assembly will experience some liftoff. The liftoff will be minimal, and the holddown spring deflection will be less than the worst-case normal operating cold-shutdown condition. The holddown spring does not go solid for any operating condition. In addition, the fuel assembly top and bottom nozzles were shown to maintain engagement with reactor internals for all operating conditions.

6.2 ***Fuel Rod Failure***

6.2.1 Internal Hydriding

The Mark-BW/MOX1 fuel rod was evaluated according to the design criterion established in Reference 1, which precludes internal hydriding by appropriate manufacturing controls.

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and the formation of hydride platelets. This failure mechanism is precluded in the Mark-BW/MOX1 fuel rods by tight controls on the moisture content of the fuel pellets and fill gas in the fuel rod. The fuel pellet specification limit for hydrogen content of the MOX pellets is 1.3 ppm (95/95 upper tolerance limit). That limit is consistent with the industry standard for MOX pellets (Reference 6) and is the same specification limit used for European MOX fuel

designed by FANP. The MOX fuel rod specification controls the total hydrogen in the pressurization gas, including H₂O, H₂, and hydrocarbons, to 15 ppm.

6.2.2 Creep Collapse

The Mark-BW/MOX1 fuel rod was evaluated according to the design criterion established in Reference 1, which requires that the predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life.

The fuel rod was analyzed for creep collapse using approved methods described in the CROV topical report (Reference 20). No cladding creep collapse was predicted to occur within a burnup of 60,000 MWd/MThm using the CROV code, which models the change in the cladding ovality over time. The analysis extended to a burnup that is greater than the maximum MOX fuel rod design burnup, 50,000 MWd/MThm. COPERNIC provided the cladding temperatures and rod internal pressures that were subsequently input into CROV. Worst-case cladding initial ovality and pellet axial gaps were assumed in the analysis.

6.2.3 Overheating of Cladding

The Mark-BW/MOX1 fuel rod was evaluated according to the design criterion established in Reference 1, which requires that departure from nucleate boiling not occur on a fuel rod during normal operation and anticipated operational occurrences for a 95% probability at a 95% confidence level. The requirements related to overheating and cladding are addressed in plant-specific transient analyses. NRC-approved methods are used to perform the transient analyses.

Thermal-hydraulic analyses, including critical heat flux (CHF) performance and CHF correlations (References 21 and 22), are not affected by the MOX fuel. Thus, no modifications to analytical tools are required in the fuel assembly mechanical analysis and thermal-hydraulic areas to accommodate MOX fuel pellets and the small increase in rod length compared to the LEU fuel rod.

For cores containing the lead assemblies, and for future batch applications, thermal-hydraulic analysis will use approved methods described in Reference 4.

6.2.4 Overheating of Fuel Pellets

The Mark-BW/MOX1 fuel rod was evaluated according to the design criterion established in Reference 1, which requires that fuel pellet centerline melting not occur for normal operation and anticipated operating occurrences for a 95% probability at a 95% confidence level.

Fuel rod thermal analysis methods, models, and criteria are presented in Reference 2 for LEU and MOX fuel. Analysis with COPERNIC showed that the LHR for centerline fuel melt of MOX fuel varies from [] to [] kW/ft at the maximum design burnup of 50,000 MWd/MThm.

6.2.5 Pellet/Cladding Interaction

Per Section 4.2 of the SRP, there are no generally applicable criteria for pellet-cladding interaction (PCI) failure. Cladding strain and fuel melt criteria are used to ensure that the fuel rod design is acceptable, per Reference 2.

6.2.6 Cladding Rupture

Cladding rupture is a LOCA-associated phenomenon. Its modeling within a LOCA analysis is governed by NRC-approved evaluation models. LOCA analysis for MOX batch applications will be provided separately; for MOX lead assemblies refer to Reference 5.

6.3 ***Fuel Coolability***

6.3.1 Cladding Embrittlement

Cladding embrittlement is a LOCA-associated phenomenon. Its modeling within a LOCA analysis is governed by NRC-approved evaluation models. LOCA analysis for MOX batch applications will be provided separately; for MOX lead assemblies refer to Reference 5.

6.3.2 Violent Expulsion of Fuel

The requirements on violent expulsion of fuel during a reactivity accident will be addressed in the plant-specific safety analyses supporting batch use of MOX fuel. For MOX fuel lead assemblies, control rod ejection simulations were performed with a representative MOX fuel lead assembly core. The calculated energy deposition in MOX fuel was well below the current regulatory acceptance criterion for LEU fuel, and well below values at which cladding failure was observed in CABRI reactivity insertion tests involving MOX fuel (Reference 5).

6.3.3 Fuel Rod Ballooning

Fuel rod ballooning is a LOCA-associated phenomenon. Its modeling within a LOCA analysis is governed by NRC-approved evaluation models. LOCA analysis for MOX batch applications will be provided separately; for MOX lead assemblies refer to Reference 5.

6.3.4 Fuel Assembly Structural Damage from External Forces

The Mark-BW/MOX 1 fuel assembly was evaluated according to the design criteria established in Reference 1, which require the following:

- Operating basis earthquake: Allow continued safe operation of the fuel assembly following an event by ensuring the fuel assembly components do not violate their dimensional requirements.
- Safe shutdown earthquake: Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies, control rod insertability, and a coolable geometry within the deformation limits consistent with the emergency core cooling system and safety analysis.
- LOCA or safe shutdown earthquake plus LOCA: Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the emergency core cooling system and safety analysis.

The differences noted between the MOX and LEU fuel rods, i.e., a small cladding length change and the substitution of MOX pellets for LEU pellets, have a negligible effect on the fuel assembly stiffness and frequency. Thus the sample faulted evaluation presented in Reference 1 for the Advanced Mark-BW fuel assembly is also representative for the Mark-BW/MOX1. Actual horizontal and axial faulted condition inputs are plant-specific, and the fuel assembly is evaluated accordingly.

6.4 **Conclusions**

The performance of the Mark-BW/MOX1 fuel assembly design has been evaluated against the criteria established in Reference 1, which address the requirements of Section 4.2 of the SRP. Three criteria (cladding rupture, cladding embrittlement, and fuel rod ballooning) were deferred to separate submittals. For every other criterion, the fuel assembly design is shown to satisfy the requirements. These results support both the lead assembly program and the batch implementation of the Mark-BW/MOX1 fuel design with a maximum fuel rod burnup of up to 50,000 MWd/MThm.

[]

Figure 6.1 Predicted Peak Cladding Oxide Thickness for MOX Rod

NOTE: From Reference 2.

7.0 Experience Base

7.1 Domestic Experience

7.1.1 MOX Experience

Prior to the U.S. policy decision in 1977 to defer indefinitely the commercial reprocessing and recycling of plutonium, there were a number of developmental programs completed that demonstrated the technical feasibility of MOX fuel. One of these, the Saxton program, had much in common with the current WG MOX program. The Saxton program (Reference 16) was supported by the U.S. Atomic Energy Commission and involved the design, fabrication, and irradiation of nine complete MOX fuel assemblies containing plutonium with a high ^{239}Pu content (Reference 23). Seven of the nine assemblies contained pelletized fuel, and the two remaining assemblies contained vibration-packed powder fuel.

For the seven assemblies with pelletized fuel, the MOX pellets were produced at a nominal 6.6 weight percent PuO_2 and with a ^{239}Pu content of 90.5%. Pellets were $94\pm 2\%$ of theoretical density, so the ^{239}Pu and total PuO_2 loadings in the Saxton study were higher than those proposed for the current lead assembly program. The Saxton pellet analyses show that the pellets would have met the current WG MOX pellet specifications for impurities with the possible exception of a higher hydrogen (moisture) content in some of the Saxton batches.

The Saxton fuel reached a maximum rod burnup of 51,000 MWd/MThm, which is greater than the burnup limit being requested in this topical report. A few rods failed during the program, but there was no indication that such failures were due to the use of MOX pellets. Failures were attributed to heavy corrosion or to thick hydride regions. Neither of these is related to the use of MOX fuel pellets.

To summarize, the U.S. Atomic Energy Commission supported a MOX irradiation program that was similar in scope to the current WG MOX program. None of the fuel was considered to have failed as a result of the use of MOX.

7.1.2 LEU Experience

FANP (US) has over 30 years of successful design and fabrication experience of nuclear fuel for pressurized water reactors (PWRs). Nuclear fuel assemblies were first delivered to Duke

Power's Oconee Nuclear Station in 1971; to date FANP (US) has supplied over 10,000 fuel assemblies of the Mark-B and Mark-BW design for PWRs.

Of particular significance, FANP (US) fuel has operated in all four of the mission reactors. For the mission reactor design (Westinghouse-designed reactors), FANP (US) began delivery of Mark-BW fuel assemblies in 1987 to Duke Power's McGuire Nuclear Station. Since that time, FANP (US) Mark-BW fuel has operated in the U.S. in eight Westinghouse-designed 17 × 17 reactors: Duke Power's Catawba Units 1 and 2, McGuire Units 1 and 2, Virginia Power's North Anna Unit 1 (lead test assemblies), TVA's Sequoyah Units 1 and 2, and Portland General Electric's Trojan Plant. As of February 2002, FANP (US) had supplied over 2,500 Mark-BW fuel assemblies to the 17 × 17 reactors, most of which were supplied to the mission reactors (McGuire and Catawba). Combined with the fuel experience of FANP (US)'s parent companies, a total of over 70,000 fuel assemblies have been designed, licensed, and successfully operated in reactors similar to the mission reactors around the world. The burnup experience of the FANP (US) Mark-BW fuel design is shown in Figure 7.1 to bound the expected MOX fuel burnups.

FANP (US) will provide the fuel design experience for the mission reactor fuel; FANP (US) has an established fuel assembly, fuel rod, and fuel component design experience base that will be applied to the MOX fuel. This experience ranges from the evolutionary revisions of long-established fuel designs to the establishment of new fuel designs.

7.1.3 Fuel Reliability

Fuel reliability of the Mark-BW/MOX1 design is expected to be consistent with the current Mark-BW reliability. The Mark-BW design has experienced a failure rate of less than one per 100,000 rods, from all manufacturing-related causes, since its inception in 1987.

7.2 ***European MOX Experience***

Fabrication and irradiation of MOX fuel in Europe provide a database on the use of MOX fuel. Fabrication and operation of MOX fuel in the U.S. will benefit from the experience of COGEMA, FANP (Fr), Electricité de France (EDF), and BELGONUCLEAIRE. This section and its subsections describe European experience with MOX.

7.2.1 European Qualification Experience

European experience that is applicable to the qualification of MOX fuel for the mission reactor irradiation includes a MOX fuel development and qualification program that has been in progress in Europe for 35 years. The first MOX fuel rods were loaded in the PWR test reactor BR3 by BELGONUCLEAIRE in 1963. FANP (Fr), COGEMA, and EDF have carried out a MOX fuel irradiation program in France since 1974 (Reference 24). The major elements of the French MOX qualification program are shown in Table 7.1.

7.2.2 European Fabrication Experience

In France, the first fuel rods using Zircaloy cladding with MOX fuel produced by the MIMAS process were introduced in the St. Laurent B1 core in 1987 (Reference 24). By mid-2000, MOX fuel was operating in 20 EDF commercial reactors (Reference 25).

The fabrication of MOX fuel in the U.S. will utilize the same MIMAS process used in Europe. This process is discussed in Section 4.0. Through the use of the aqueous polishing process, impurities in the WG plutonium will be eliminated. This process allows the production of PuO₂ powder consistent with (i.e., meeting essentially the same specification as) that from European MOX plants, and it ensures that the European experience base is applicable to the MOX fuel produced in the U.S. from WG plutonium.

The production of MOX fuel has been qualified in the MELOX, Cadarache, and BELGONUCLEAIRE P0 (Dessel) manufacturing plants. As of the end of 1998, these three facilities had produced a combined total of more than 435,000 MOX fuel rods, and the rods had been irradiated in 35 commercial nuclear reactors. These reactors include all 33 of the PWRs irradiating MOX fuel in Europe (Reference 26). The MIMAS process is in use at all three of these facilities. A listing of the European plants using MOX fuel from the MIMAS process is provided in Table 7.2.

7.2.3 European Operational Experience

The European operational experience is important to and will be used as a part of the fuel qualification effort, because it helps to benchmark the appropriate core physics analysis tools (Reference 3), it is the source of much of the data used to demonstrate the application of the COPERNIC fuel performance code to MOX (Reference 2), and it demonstrates the maturity of the MOX technology. This experience includes MOX fuel assemblies that have been irradiated

by EDF and other European utilities under a variety of fuel management schemes and operating conditions.

Operating schemes have included 1/3 MOX fuel core, 1/4 MOX fuel core, hybrid refueling (where LEU assemblies are used for four annual cycles while MOX assemblies are used for three); annual cycles; and extended cycle designs (References 27 and 28). The MOX fuel assemblies have been discharged with assembly average burnups as high as 58,000 MWd/MThm (Reference 25). Average linear power for these plants ranged from 5.43 to 6.28 kW/ft, with core exit temperatures from 610 °F to 619 °F.

The European experience also includes load-follow operation. Since 1991, two EDF reactors using MOX fuel have been operating under load-follow and frequency control conditions (Reference 24). Based on this successful experience, all of the EDF reactors using MOX fuel have been authorized, since 1995, to operate under load-follow conditions (Reference 29).

In the EDF 900 MW (157 fuel assembly core) plants, up to 16 MOX assemblies are loaded in an equilibrium batch using one-third core reload management. The replacement of LEU assemblies by MOX fuel assemblies is done without any penalty on core operating conditions. An extended rod burnup goal of 60,000 MWd/MThm (52,000 MWd/MThm assembly burnup) has been set for 2004 as part of the MOX Parity project. That goal comes in advance of the required mission reactor initial core loading. Furthermore, programs are underway in France to develop MOX designs capable of reaching assembly burnups in excess of 60,000 MWd/MThm over the next ten years (Reference 30).

In Belgian reactors, two schemes of fuel management are followed:

- Doel Unit 3 uses annual cycles with 1/4 core reloads.
- Tihange Unit 2 uses extended cycles with 1/3 core reloads, similar to the practice at the mission reactors. By the end of September 2002, a total of 104 MOX fuel assemblies had completed one to four cycles of operation, with a maximum fuel assembly discharge burnup of 46,600 MWd/MThm.

The current rod design burnup in France is 48,000 MWd/MThm (43,000 MWd/MThm assembly burnup) (Reference 25). In Belgium, the average assembly discharge burnup is about 44,000 MWd/MThm at Tihange 2 and 46,500 MWd/MThm at Doel 3 (Reference 31). Design assembly burnups as high as 55,000 MWd/MThm are currently proposed in Germany (Reference 32). Since a maximum fuel rod burnup of 50,000 MWd/MThm corresponds to a fuel

assembly burnup of about 45,000 MWd/MThm, the MOX exposure experience in Europe envelops the design burnup for the mission fuel. Table 7.3 shows the maximum discharge burnup for the European plants using MOX fuel produced by FANP (Fr)/COGEMA and by Framatome ANP, GmbH {FANP (Ger), formerly Siemens} with the same process to be used on the lead assemblies and mission reactor fuel. The MIMAS process was used by FANP (Fr)/COGEMA. FANP (Ger) has used both MIMAS and the similar Optimized Co-milling (OCOM) process, though their recent fuel has all been produced by the MIMAS process.

In Germany, the use of MOX fuel with M5[®] cladding is proceeding in advance of the U.S. application in the mission reactors (Reference 33). The most heavily irradiated MOX fuel assemblies with M5[®] cladding will reach an assembly average burnup as high as 64,000 MWd/MThm.

- The German reactor Philippsburg 2 loaded 16 MOX fuel assemblies with M5[®] cladding in 1998, 16 in 1999, and []].
- The German reactor Neckar 2 loaded 16 MOX fuel assemblies with M5[®] cladding in 2000 []].
- The German reactor Grafenrheinfeld has received 32 MOX fuel assemblies with M5[®] cladding, and loaded 16 in 2001 []].
- []]

Two fuel assemblies with some M5[®]-clad MOX fuel rods were loaded into EDF's Chinon 3 reactor in 2001; the target burnup for this fuel is greater than 55,000 MWd/MThm (Reference 25).

7.2.4 Fuel Reliability Experience

A comparison of the reliability of European MIMAS-produced MOX fuel with that of LEU shows similar operating experience. Reload quantities of MIMAS-produced MOX fuel rods have been irradiated in commercial reactors over the past 15 years. As of March 31, 2003, []]

]. None of the failures has been attributed to the use of MOX fuel. Similar failure rates have been observed in LEU fuel assemblies (see Section 7.1.3).

The fuel reliability experience with MOX fuel in Europe is expected to be applicable to the U.S. because of the similarity of the fuels. The use of aqueous polishing for preparing the WG plutonium will ensure that there are no effects due to gallium (see Section 3.2). Furthermore, the base fuel design to utilize the MOX pellets (Mark-BW) has been extremely reliable (see Section 7.1.3). Thus, the reliability of the MOX fuel with WG plutonium is expected to be very high.

7.2.5 European Experimental Data

Performance data for fuel and materials have been obtained from poolside and hot cell examinations. The examinations have concluded that there have been no differences in MOX fuel assembly operational characteristics relative to LEU fuel. MOX fuel has been examined at poolside after one to five cycles of irradiation. In addition, over 60 irradiated MOX fuel rods have been examined in French hot cells. Data on MOX fuel performance are also available from an out-of-core and in-core analytical test program.

Following are details of specific examinations supporting the overall qualification effort.

7.2.5.1 Hot Cell Examination of the Current MOX Fuel

Fuel rods from the first MOX fuel batch in the St. Laurent B1 reactor were characterized and withdrawn after each of three irradiation cycles. These data included rod burnups up to approximately 43,000 MWd/MThm and three different plutonium concentrations. Fuel rods irradiated for three cycles at St. Laurent B2, including load-following operation in the last cycle, were also examined. These examinations showed that the MOX fuel rods behaved similarly to LEU fuel for both waterside corrosion and rod dimensional effects. Furthermore, the rods operated under load-follow conditions at St. Laurent B2 were compared to reference MOX rods operated under base load conditions at St. Laurent B1, and it was found that the two sets of rods behaved similarly (Reference 25).

The equivalence of waterside corrosion rates for MOX and LEU fuel was confirmed recently by measurements on Zircaloy-4 cladding in high temperature reactors in Germany for a rod average burnup of 49,000 MWd/MThm. The maximum oxide thickness for the MOX fuel was within the range of oxide thicknesses for LEU fuel, confirming that MOX fuel performs the same as LEU fuel relative to Zircaloy cladding corrosion. Confirmation of the same equivalence for the advanced cladding (M5[®]) to be used on the Mission Reactor fuel has been obtained in Germany where M5[®] rods containing MOX fuel achieved a burnup of 55,000 MWd/MThm in 2002. Poolside measurements carried out []].

7.2.5.2 High-Burnup Poolside and Hot Cell Examination

To provide verification of performance and benchmarking data to support higher burnups, MOX assemblies have been irradiated for four and five cycles in the Dampierre 2 and Gravelines 4 reactors. The fuel reached maximum rod average burnups of 53,000 and 60,000 MWd/MThm, respectively. Poolside and hot cell examination showed that the dimensional changes in both the fuel assemblies and the fuel rods were similar to those of LEU fuel. The rod puncture data did not show any enhancement of fission gas release due to the burnup effect.

7.2.5.3 Analytical Experiments

Out-of-pile and in-pile experimental tests have been conducted to promote an improved understanding of MOX fuel behavior (Reference 34). These research and development programs conducted by the French partners in international programs, most notably the Halden Reactor Project, have addressed normal and off-normal conditions. The primary areas of research have concerned thermal, fission gas release, and mechanical properties.

These data have been used for the development and benchmarking of the models implemented in the COPERNIC thermal/mechanical code.

7.2.5.4 Power Ramp Testing

Ramp testing has established that the performance of MOX fuel rods relative to PCI is equivalent to or better than that of LEU fuel. Transient fission gas release from the MOX rods was equivalent to that of LEU fuel.

Power ramp tests were performed in the Studsvik experimental reactor in a PWR environment in terms of temperature, power, and neutron flux. Short fuel rods were fabricated from segments of irradiated MOX fuel rods from St. Laurent B1. The rods were ramped from typical operational power levels to terminal levels up to 14.6 kW/ft without cladding failure, demonstrating the excellent performance of MOX fuel for PCI considerations (Reference 35).

These ramp test rods also produced information on transient fission gas release (since the rods did not fail and the gas inventory was retained). The measured fractional release rates of the five tested MOX fuel rods are consistent with the burnup and power and did not show any unexpected behavior. The current transient fission gas release model for LEU fuel in the COPERNIC code gives good agreement with the MOX transient gas release data. Other programs with ramp tests in BR2, OSIRIS, and Halden after irradiation in PWR reactors have also confirmed the fission gas release of MOX fuel. The ramp test programs carried out in the BR2 reactor are described in the paper of M. Lippens at the Vienna Symposium on MOX Fuel Cycle Technologies (Reference 36) and references cited herein. The analytical test programs (testing of two-cycle and four-cycle MOX fuel rods from EDF/Framatome) at Halden were made or are being made in the framework of the Joint Program (Reference 37).

7.2.5.5 Reactivity Insertion Testing

Tests have been used to investigate phenomena associated with rapid power increases in both LEU and MOX fuel. Test series examining reactivity insertion impacts on LEU and MOX fuel were performed in the SPERT test program in Idaho (Reference 7), the RIA test program in the Nuclear Safety Research Reactor in Japan (Reference 8), and the RIA test series in the CABRI loop in France (Reference 38).

Test results help substantiate acceptance criteria that are applied to licensing basis analyses of reactivity insertion accidents. These analyses and acceptance criteria will be discussed in reactor safety analysis submittals for the mission reactors.

7.3 **Conclusions**

There is successful domestic experience with material similar to WG MOX fuel. This experience extends to burnups, ²³⁹Pu loadings, and total plutonium loadings greater than those proposed for the current program. There is also successful domestic experience with the Mark-BW design, which is similar to the Mark-BW/MOX1 design. European experience includes

successful qualification, fabrication, and operation of a variety of MOX fuel assemblies. European MOX fuel assemblies have had good fuel reliability at fuel rod burnups up to 60,000 MWd/MThm. The proven MIMAS-produced MOX reliability, combined with the proven Mark-BW reliability, provides the basis for the expectation that the performance of the Mark-BW/MOX1 will continue at this high level. Understanding of the performance of MOX fuel is supported by hot cell examinations, analytical experiments, ramp tests, and reactivity insertion tests. Since similar assembly designs and similar fuels have been successfully used, past experience supports both the lead assembly program and the batch implementation of the Mark-BW/MOX1 fuel design with a maximum fuel rod burnup of up to 50,000 MWd/MThm.

Table 7.1 French MOX Qualification Program

Time Period	Item	Description	Purpose
1974-1986	Irradiation + PIE EURATOM PROGRAM	Investigation of MOX fuel performance – 10 contracts, 48,000 MWd/MThm rod burnup	Demonstration/ fuel performance modeling
1987-1991	Surveillance program + PIE	15 fuel rods examined after 1, 2, and 3 cycles of first MOX reload (St. Laurent B1 reactor) 43,000 MWd/MThm rod burnup	Qualification of product and performance modeling
1987-1991	Irradiation + PIE	Irradiation of MOX fuel rods in the Chaudière Avancée Prototype PWR under load-follow condition – rod burnup = 20,000 MWd/MThm	Fuel performance/ modeling
1989-1990	Analytical experiment (EDITH MOX)	Irradiation of a leaking MOX fuel rod in an experimental loop	Fission product behavior - EDF reload policy basis
1989-1992	Surveillance + PIE	Fuel rods examined after three cycles, irradiated under load-follow during third cycle – rod burnup = 43,000 MWd/MThm	MOX fuel performance under load-follow condition for qualification
1993-1994	Ramp testing + PIE	Ramp testing of two and three cycle fuel rodlets at Studsvik and OSIRIS	PCI data for load-follow qualification
1991-1994	Analytical experiment	Out-of-pile measurements of physical properties of current MOX product	Material properties modeling
1992-1993	Analytical experiment DENSIMOX	Experimental irradiation to get densification kinetics data	Material properties modeling
1993-1995	Analytical experiment GRIMOX	Instrumented experimental irradiation for fuel temperature and fission gas release (FGR) kinetics – 0 to 4,500 MWd/MThm burnup	Fuel performance at high burnup, for 1/4 core management licensing
1990-1994	Surveillance + PIE (4 Lead assemblies)	Fourth irradiation cycle at core periphery – 7 rods examined (3 and 4 cycles) – rod burnup = 52,000 MWd/MThm	Material properties modeling
1996-1998	Surveillance + PIE (1 Lead assembly)	Fourth irradiation cycle at core center – 4 rods examined – rod burnup = 53,000 MWd/MThm	Fuel performance at high burnup, for 1/4 core management licensing
1996	Analytical experiment DEFORMOX	Instrumented experimental irradiation of LEU and MOX fuel; online measurement of cladding deformation	Modeling

Table 7.1 French MOX Qualification Program (Continued)

Time Period	Item	Description	Purpose
1997-	Surveillance + PIE	First reload of second generation fuel design (MELOX fuel)	High-burnup surveillance - six cycles expected
1998-2000	Surveillance + PIE	Fifth cycle irradiation of one assembly at core center – rod burnup expected = 61,000 MWd/MThm	Fuel performance at high burnup for 1/4 core management licensing (LEU/MOX parity)
1987-1993	International program PRIMO	Examination of 15 rods irradiated at BR3 + ramp test – rod burnup = 55,000 MWd/MThm	Modeling for global rod behavior
1993-1998	International program FIGARO	Instrumented irradiation (central temperature + internal pressure) of rodlets pre-irradiated at Beznau – rod burnup = 48,000 MWd/MThm	Modeling for fuel temperature and FGR kinetics

Table 7.2 European Plants Using MOX from MIMAS Process

No.	Country	Reactor	MELOX	Cadarache	Dessel
1	France	Blayais 1	X		
2		Blayais 2	X	*	*
3		Dampierre 1	X	*	*
4		Dampierre 2	X	*	*
5		Dampierre 3	X		
6		Dampierre 4	X		
7		Tricastin 1	X		
8		Tricastin 2	X	*	
9		Tricastin 3	X	*	
10		Tricastin 4	X		
11		St. Laurent 1	X	*	*
12		St. Laurent 2	X	*	*
13		Gravelines 1	X		
14		Gravelines 2	X		
15		Gravelines 3	X	*	*
16		Gravelines 4	X	*	*
17		Chinon 1	X		
18		Chinon 2	X		
19		Chinon 3	X		
20		Chinon 4	X		
21	Belgium	Tihange 2			X
22		Doel 3			X
23	Germany	Unterweser		X	*
24		Grafenrheinfeld		X	*
25		Philippsburg 2		X	*
26		Brokdorf		X	*
27		Gundremmingen B			X
28		Gundremmingen C			X
29		Grohnde		X	
30		Isar 2		X	*
31		Obrigheim		X	
32		Neckar 2		X	
33	Switzerland	Beznau 1			*
34		Beznau 2		X	*
35		Gösgen			X

NOTE: X = current supplier. * = past supplier.

Table 7.3 European MOX Burnup Experience

Country	Number of Reactors	Assembly Type	Maximum Discharge Burnups (MWd/MThm) of Assemblies Having Completed		
			3 Cycles	4 Cycles	5 Cycles
FANP (Fr) Deliveries (as of December 2002)					
France	20	17 x 17	41,500	49,500	54,000 (60,000 - Rod)
Belgium	2	17 x 17	46,500	46,500	
Germany	2	16 x 16 18 x 18	48,500	51,500	(in progress)
FANP (Ger) Deliveries (as of April 2001)					
Germany	9	14 x 14 to 18 x 18		51,000	
Switzerland	3	14 x 14 and 15 x 15		56,000 (65,000 - Rod)	

NOTE: Burnups are assembly averages except as noted.

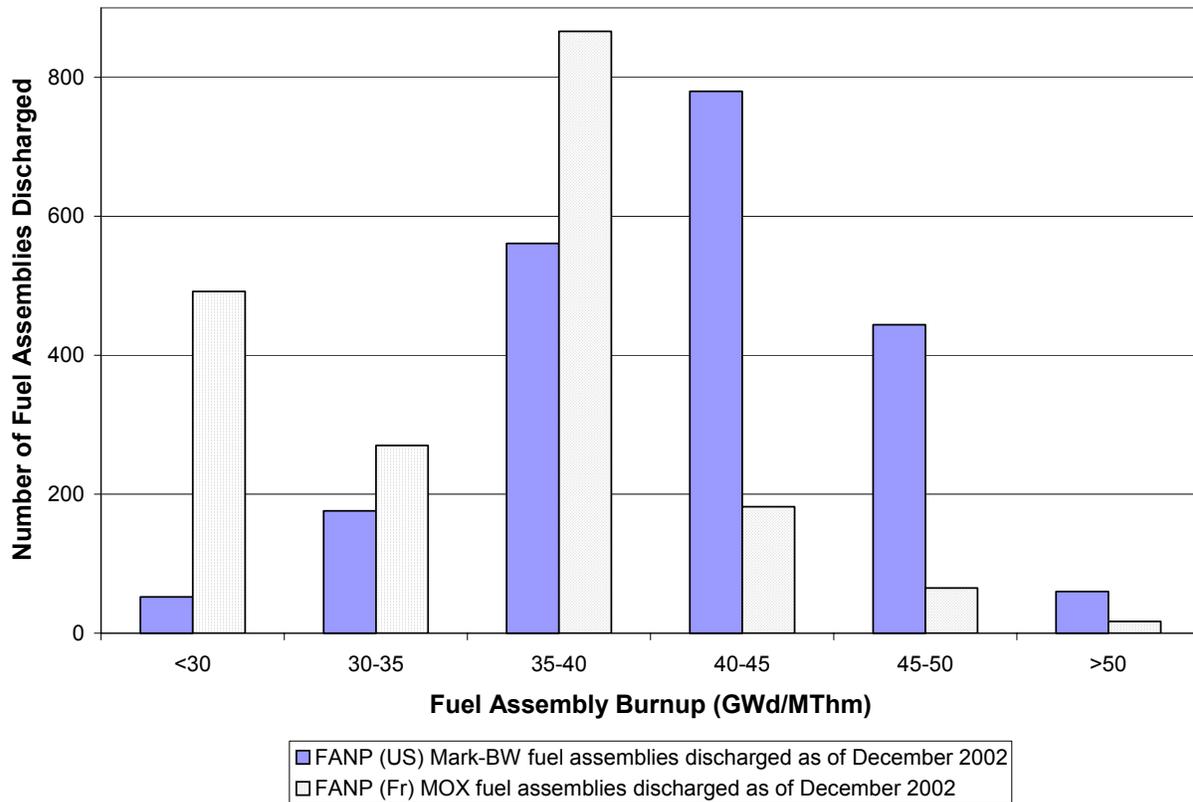


Figure 7.1 Framatome ANP Mark-BW and MOX Burnup Experience

8.0 Lead Assembly Program

Mark-BW/MOX1 lead assemblies will be fabricated and irradiated in a commercial PWR to demonstrate WG MOX as an alternative fuel for PWR systems. One or more of the lead assemblies will be nondestructively examined at the end of each operational cycle to confirm the operational predictions.

Planning for the lead assembly program is based on the fabrication of four (4) lead assemblies. However, if enough WG plutonium dioxide powder is not available on the required schedule, it may be necessary to limit the program to two (2) lead assemblies. FANP (US) has previously used either two or four lead assemblies to confirm new fuel designs. Irradiation of the lead assemblies is to confirm the design predictions; no data are required from the lead assemblies to qualify analytical tools or modify fuel performance models. All four of the mission PWRs are of the same Westinghouse design and use the same LEU fuel design for the resident core. Therefore, operation in one mission reactor is representative of operation in any of them. Furthermore, it is planned for the transition to batch implementation to be accomplished in phases. The first production batch of WG MOX fuel is scheduled for operation after the second cycle of lead assembly irradiation is completed and the results of nondestructive examinations of the lead assemblies are evaluated. Based on current plans, the first production MOX batch at each of the mission reactors will result in a MOX core loading of about 15%. The core fraction will be increased in the second and subsequent cycles of MOX operation, with the maximum core fraction (approximately 40%) being achieved with the insertion of the third batch at each reactor.

The Mark-BW/MOX1 lead assemblies will be fabricated with the same materials and processes, and using the same design as the mission reactor fuel. The assemblies will be irradiated in one of Duke Power's McGuire or Catawba reactors. The MOX lead assemblies will operate in high-power, non-limiting core locations representative of the batch operating conditions. At least one of the lead assemblies will be placed in an instrumented location. Poolside PIEs will be performed on at least one of the assemblies after each irradiation cycle (as discussed in Section 8.5). Accumulated assembly average burnups of greater than 42,000 MWd/MThm and maximum rod average burnups of about 47,000 MWd/MThm are projected at the end of two cycles. The Mark-BW/MOX1 assembly is based on the Advanced Mark-BW assembly, which is designed for a maximum rod average burnup of 62,000 MWd/MTU. Successful completion of

the second cycle of lead assembly irradiation will be considered sufficient to justify full batch implementation.

A third cycle of irradiation is proposed to confirm MOX fuel performance to fuel rod burnups approaching 60,000 MWd/MThm. A destructive hot cell examination on selected rods from the lead assemblies will be performed following this third cycle.

8.1 **Purpose**

The primary purpose of the lead assembly program is to provide operational experience to demonstrate the acceptability of the MOX fuel design. In achieving this purpose, the lead assembly program will address several issues:

- WG Plutonium vs. RG Plutonium

The fuel qualification effort relies heavily on European experience that is exclusively with RG plutonium. The lead assembly program will help to confirm that irradiation of MOX fuel from WG plutonium presents no unique challenges to the analytical methodologies that were developed for MOX fuel from RG plutonium.

- Manufacturing Processes

The Mark-BW/MOX1 lead assembly program will demonstrate both the application of the aqueous polishing process to reduce the impurity concentration in WG plutonium to trace levels and the application of the MIMAS process to the production of WG MOX fuel pellets. This will be accomplished by fabrication of the MOX pellets to specifications consistent with those applied to RG MOX fuel in Europe.

- Trace Levels of Impurities

The lead assembly program will help confirm that the presence of trace levels of gallium (< 1 ppm) do not adversely affect fuel rod cladding integrity. This will be done by demonstrating two cycles of successful operation of the Mark-BW/MOX1 lead assemblies.

- Fuel Assembly Hardware

Acceptable performance of the Mark-BW/MOX1 fuel design will be demonstrated.

- Fuel Irradiation History and Burnup

The lead assembly program will demonstrate acceptable MOX fuel performance under LHRs, coolant chemistries, and burnup conditions that are characteristic of U.S. PWRs operating on 18-month fuel cycles.

- MOX Fuel Assembly Neutronic Response

The lead assembly program will provide an opportunity to measure the WG MOX fuel assembly power using an existing movable incore detector system in order to validate the ability to predict and measure the core power distribution in a mixed core.

8.2 *Design Description*

The lead assembly design will be the Mark-BW/MOX1, which is intended to be the production fuel design to be used in the mission reactors. Three plutonium concentrations will be used within the assemblies, as shown in Figure 5.2. This three-zone design is consistent with the approach used in the EDF reactors and will be used in the mission fuel design. As noted in Chapter 5, the final zoning configuration may differ slightly from the design shown in this report.

The lead assemblies, as well as the mission reactor fuel, will utilize burnable poison rod assemblies for reactivity control. The burnable poison rod assemblies will be supplied by FANP (US) based on the specification (boron concentration and number of active pins/assembly) provided by the utility.

The Advanced Mark-BW design (Reference 1) is being used as the basis for the Mark-BW/MOX1 design qualification. The only changes required are those associated with the fuel rod design to accommodate the use of MOX. The MOX pellets will be fabricated by processes and to specifications that are consistent with those used by FANP (Fr) in Europe. As a result, the European database will be applicable. The primary difference between the U.S. and European specifications is the inclusion of a gallium limit on the WG PuO₂ powder. In addition, the UO₂ to PuO₂ powder ratio in the MIMAS master mix will be increased from approximately 70/30 to 80/20 to maintain the fissile content of the plutonium-rich agglomerates at approximately the same level for the WG MOX as for RG MOX (as discussed in Section 3.3).

8.3 *Fabrication*

The lead assembly pellets and fuel rods will be fabricated on the same equipment and with the same processes used for the production of RG MOX pellets and rods. Polished PuO₂ powder will be supplied by DOE for the lead assemblies. This powder will be produced to specification requirements provided by DCS and developed from specifications for the powder that will be produced in the MFFF for the mission reactors. The PuO₂ powder specification for the lead assemblies and mission fuel is developed from and consistent with that for RG PuO₂ powder, so that the chemical and physical properties of this powder will be within the database for powders

routinely used in Europe. This ensures consistency with the European product and permits the use of the European performance database. It is intended that fabrication of the lead assemblies will be performed in a facility that currently manufactures RG MOX fuel; this practice will minimize risk in the fabrication area. The facility will be certified by FANP (US) as meeting 10 CFR 50 Appendix B requirements prior to fabrication.

8.3.1 Feed Material Requirements

8.3.1.1 Plutonium Feed

The WG plutonium oxide feed powder to be used in the fabrication of the lead assembly MOX pellets will have the same chemical and physical properties as the oxide powder routinely used in the fabrication of European MOX fuel. In both cases the oxide is derived from plutonium nitrate through the oxalate precipitation process. This process provides better control of the PuO₂ particle size, shape, and size distribution compared to product obtained by dry processing (e.g., burning plutonium metal to the oxide). Following precipitation and calcination at temperatures up to 650 °C, the PuO₂ powder will be homogenized through blending and then characterized. The chemical and physical properties of the PuO₂ powder will be measured and compared to the specification requirements that DCS has provided to be consistent with the database for powders produced in Europe. Thus, the European experience base will be applicable to the lead assembly product.

8.3.1.2 Plutonium Polishing

The WG plutonium feed contains gallium, which will be removed by an aqueous polishing process that is described in Section 3.2. The 120 ppb limit on the PuO₂ powder for batch assemblies is intended to ensure that the finished pellets will have a gallium concentration that is as low as can practically be achieved. A review of literature on interaction between gallium and zirconium alloys indicated that larger gallium concentrations would be acceptable. On this basis, the gallium acceptance level for the PuO₂ powder for the lead assemblies is 300 ppb. Selected increases in other impurity limits may also be considered, provided that appropriate technical justification can be made.

8.3.1.3 Uranium Feed

The majority of the European MOX irradiation experience is based on the use of depleted (or natural) UO₂ powder prepared by the ammonium diuranate (ADU) wet process, or by the

ammonium uranyl carbonate (AUC) wet process. Current production European MOX fuel uses ADU UO_2 powder produced in the COGEMA TU2 plant. A sufficient quantity of this UO_2 powder is being made available to DCS for the lead assembly program. This approach ensures that the UO_2 powder used in this lead assembly program is identical to that used in European MOX experience, so any effects due to differences in uranium feed characteristics are precluded.

Equivalent feed material will be used for the UO_2 supply for batch implementation in the mission reactors. The UO_2 powder for the MFFF is expected to come from a qualified U.S. facility that will fabricate ADU powder. Use of UO_2 powder from any other source or produced by a different process will be qualified in Europe with RG MOX before consideration for use in the mission reactors.

8.4 ***Irradiation Plan***

The lead assemblies will be irradiated in a McGuire or Catawba reactor. At least one of the assemblies will undergo three irradiation cycles. During each of the first two irradiation cycles, at least one of the lead assemblies will be located in an instrumented location to verify predicted operational neutronic performance. Neutronic data will be compared to similar data obtained from instrumented LEU assemblies to verify core predictions.

The lead assemblies will be located in relatively high-power but non-limiting positions for the first two cycles to ensure representative operating parameters for batch implementation. Figure 8.1, which is based on preliminary lead assembly and batch core designs, presents typical bounding power history envelopes for the MOX lead assemblies (three cycles) as well as five representative MOX fuel assemblies from batch use of MOX fuel (two cycles). Each curve is a composite of all of the fuel rods in one assembly and depicts the maximum power of any rod versus the maximum burnup of any rod in that assembly. As can be seen, after two cycles of irradiation, maximum rod burnups for lead and batch assemblies are comparable. The lead assemblies are projected to reach a maximum fuel rod burnup in excess of 47,000 MWd/MThm in two cycles, consistent with the proposed fuel rod burnup limit of 50,000 MWd/MThm.

While fuel qualification activities will be completed after the second cycle of lead assembly irradiation, a third irradiation cycle of one or more of the lead assemblies is planned. The purpose of the additional irradiation is to obtain data at higher burnup, thereby confirming performance, verifying margin predictions, and benchmarking fuel performance models at

extended burnups. The maximum fuel rod burnup is expected to exceed 57,000 MWd/MThm in this third cycle. This burnup experience would exceed the requested fuel rod burnup limit of 50,000 MWd/MThm, and the data may eventually be used to justify operation at extended burnups.

8.5 ***Fuel Examinations***

The PIEs provide performance data to confirm the assumptions and models used for design and analysis of the WG MOX lead assemblies. The evaluation of the performance depends on several tasks. These tasks are:

- Characterization of the as-built condition of the fuel
- Poolside PIEs
- Rod extraction and hot cell examinations
- Detailed operational history
- Data reduction and benchmarking to models and other data sources

The following sections describe these tasks in detail.

8.5.1 Characterization of the As-Built Condition of the Fuel

All of the major components of the lead assembly and fuel rods will be characterized prior to irradiation. The measured characteristics of lead assembly fuel pellets will be placed in a database for use in PIE comparisons. Representative sample pellets will be measured for grain size and microstructural features, including PuO₂ particle size, homogeneity of PuO₂ dispersion, resinter test performance, diameter, length, porosity distribution, and complete chemical impurity content. Archive samples will be retained from each MOX pellet lot.

For characterization of the lead assembly rods, a number of non-routine inspections will also be included in the lead assembly inspection steps. The length of each MOX rod, the pellet active length, and the plenum length will be measured and recorded by serial number. Samples of in-process end plug welds and seal welds will be retained. The weight of as-loaded pellets will be identifiable to each rod serial number. The rods will be uniquely marked so that the specific plutonium loading of each rod can be determined.

Consistent with standard nuclear practice, archive samples of the product will be retained for the MOX fuel program. A minimum of one full archive rod of each of the three plutonium loadings

will be retained. The purpose of the archive rods is to provide a baseline for root-cause analysis studies in the event of unexpected MOX fuel behavior and for comparison of the irradiated and unirradiated conditions during hot cell examinations.

Following standard nuclear identification procedures, each lead assembly will be identified with a unique serial number. The location of each fuel rod within each lead assembly will be recorded by serial number, and the location of the different plutonium loadings will be verified and documented for each assembly. Actual overall assembly dimensions will be recorded. Water channel spacing measurements will be taken at mid-span elevations.

All of the characterization data will be issued in a final report that documents all relevant data of the lead assembly pellets, rods, and assemblies. This information will be used as the pre-irradiation baseline data for the PIEs.

8.5.2 Poolside PIE

The lead assemblies are scheduled to be irradiated in one of the McGuire or Catawba plants. After two cycles of irradiation, the lead assemblies will reach an assembly average burnup of approximately 42,000 MWd/MThm, with a maximum rod average burnup of approximately 47,000 MWd/MThm. After each cycle, the assemblies are scheduled to be examined at poolside to verify acceptable performance and provide data for later evaluation. The planned scope of the poolside examinations includes the items listed in Table 8.1. This table includes the purpose of each inspection and the expected result, relative to LEU assembly performance.

After the first irradiation cycle, a basic poolside PIE of the lead assemblies will be performed during the refueling outage. The lead assemblies will then be reinserted for a second cycle of irradiation. Another basic PIE will be performed during the refueling outage after the second cycle. If the assemblies are found to be operating normally, one or more of the assemblies will be reinserted for a third cycle. An extended PIE will then be performed on the assemblies that are not reinserted for the third cycle. The extended PIE may occur after reactor restart.

A third cycle irradiation on one or more of the lead assemblies is planned to demonstrate that MOX assemblies can be irradiated to burnups approaching those of LEU fuel and to supplement the FANP (US) database on this fuel assembly design. After discharge, another PIE, including both basic and extended inspections of the assemblies, will be performed. Like the second-cycle extended PIE, this PIE may be performed after reactor restart.

The extent of the examinations listed in Table 8.1 is based on the assumption that four lead assemblies will be produced, and two of these will be irradiated for a third cycle. If only two assemblies are produced or if only one assembly is irradiated for a third cycle, the extent will necessarily be reduced.

8.5.3 Rod Extraction and Hot Cell Examinations

DCS will extract fuel rods from the lead assemblies after the third cycle of operation. The rods will then be shipped to a DOE host laboratory using a DCS contracted rod-shipping cask vendor. The scope of work to be performed in the hot cell is expected to include:

- Fission gas release
- Fuel cladding metallography
- Fuel pellet ceramography
- Burnup analysis
- Burnup distribution

After the hot-cell exam, hot-cell results will be compared to specific predictions, the overall LEU fuel database, and the overall MOX database.

8.5.4 Operational History

Detailed operational data will be obtained and recorded in a database to aid in the evaluation of the lead assemblies. At least one of the lead assemblies will be placed in an instrumented location to verify predicted operational neutronic performance during the first two irradiation cycles. Also, overall plant performance parameters, such as power levels, temperatures, transient conditions, and reactor coolant system chemistry, will be recorded in detail. Detailed core power distributions will be generated at various points in the fuel cycle to allow comparisons of predicted and measured performance. The detailed operational data will be provided in an appendix in the PIE report issued after each cycle.

8.5.5 Acceptance Criteria

After each fuel cycle, the lead assembly operational conditions and the PIE measurements will be compared to specific predictions and to the overall LEU fuel database. The acceptance criteria listed in Table 8.2 will be applied after the second cycle of irradiation. If the fuel meets these criteria, it will be concluded that the fuel is performing acceptably and can be certified for batch implementation.

More specific criteria for fuel assembly growth and shoulder gap will be developed to implement the requirements of Table 8.2. The bases for these criteria may include inputs from previous experience with LEU fuel and the results of the first-cycle PIE. The criteria will be formulated to provide confidence that the fuel will continue to perform acceptably to the maximum rod burnup of 50,000 MWd/MThm.

It is expected that a third cycle of irradiation will achieve maximum rod burnups greater than 50,000 MWd/MThm in the lead assemblies. Accordingly, additional margin is needed to provide confidence that the fuel will continue to perform acceptably through the third cycle. The acceptance criteria given in Table 8.3 will be used to determine whether such margin exists. If the fuel meets these criteria, it will be concluded that the fuel has sufficient margin for the lead assemblies to be safely irradiated for a third cycle.

8.6 **Conclusions**

A program of post-irradiation examinations has been planned. The results of these examinations will be used to verify that the Mark-BW/MOX1 fuel assemblies are performing as predicted. If the acceptance criteria in Table 8.2 are met, it will be concluded that the design is suitable for batch implementation. If the acceptance criteria in Table 8.3 are met, it will be concluded that the fuel has sufficient margin for the lead assemblies to be safely irradiated for a third cycle beyond the burnup limit requested for batch implementation, but not beyond 60,000 MWd/MThm maximum rod burnup.

Table 8.1 Lead Assembly Poolside PIE

Inspection	Extent	Purpose	Expected Result	Cycle		
				1	2	3
Fuel assembly visual	All assemblies	Verify no abnormal appearance compared to LEU.	Same as LEU with M5 [®] -clad fuel rods and guide thimbles	X	X	+
Fuel rod visual	Peripheral rods on all assemblies	Verify no abnormal appearance compared to LEU.	Same as LEU with M5 [®] -clad fuel rods	X	X	+
Fuel assembly growth	All assemblies	Confirm predictions and equivalence with LEU assembly.	Same as LEU with M5 [®] -clad fuel rods and guide thimbles	X	X	+
Fuel rod growth (shoulder gap closure)	Two assemblies, two longest peripheral rods in each assembly	Confirm acceptable margin for fuel rod operation. Verify shoulder gap.	Same as LEU with M5 [®] -clad fuel rods and guide thimbles	X	X	+
Fuel assembly bow and distortion	Two assemblies, two faces per assembly	Address incomplete rod cluster control assembly (RCCA) insertion issue. Verify fuel assembly growth models.	Same as LEU with M5 [®] -clad fuel rods and guide thimbles	X	X	+
Grid width	Two assemblies, two adjacent sides, four grids per assembly	Confirm grid growth predictions, equivalence to LEU fuel assembly.	Same as LEU with M5 [®] grids		+	+
Fuel rod oxide thickness	Two assemblies, four faces per assembly, four rods per face, plus two assemblies, one face on an internal row	Confirm equivalence to LEU rods. Compare to corrosion predictions.	Same as LEU with M5 [®] -clad fuel rods		+	+
Grid oxide thickness	Two assemblies, four grids per assembly, two adjacent sides, duplicate measurements	Confirm grid strength margins.	Same as LEU with M5 [®] grids		+	+

Table 8.1 Lead Assembly Poolside PIE (Continued)

Inspection	Extent	Purpose	Expected Result	Cycle		
				1	2	3
Fuel assembly RCCA drag force	Two assemblies, but only if drop time increases	Address incomplete RCCA insertion issue.	Same as LEU with M5 [®] guide thimbles		+	+
Guide thimble plug gauge	Two assemblies, four guide thimbles	Address incomplete RCCA insertion issue. Verify distortion-free operation.	Same as LEU with M5 [®] guide thimbles; all gauges pass all grid spans		+	+
Water channels (fuel rod bowing)	One assembly, two faces, seven spans in each direction	Determine rod bow equivalence to LEU rod and fuel assembly envelope.	Same as LEU with M5 [®] -clad fuel rods and guide thimbles		+	+

NOTE: X = performed during outage. + = may be performed on discharged fuel during normal reactor operation.

**Table 8.2 Lead Assembly Performance Criteria
for Batch Certification**

Measurement	Criteria
Fuel assembly growth	Cold measurements of fuel assembly growth shall be consistent with a positive clearance between the fuel assembly and reactor internals under both hot and cold conditions at a maximum rod burnup of 50,000 MWd/MThm.
Shoulder gap	Cold measurements of shoulder gap shall be consistent with a positive clearance between the fuel rods and top nozzle under both hot and cold conditions at a maximum rod burnup of 50,000 MWd/MThm.
Fuel assembly RCCA drag force	Drag force shall not exceed: <ul style="list-style-type: none"> • 100 lbf in dashpot and • 60 lbf above dashpot.
Fuel rod integrity	No fuel rods in the lead assemblies shall fail from MOX fuel-related causes.
Fuel rod oxide thickness	Peak oxide thickness (using moving average over 1 inch) shall not exceed 50 μm .

**Table 8.3 Lead Assembly Performance Criteria
for Third Cycle of Irradiation**

Measurement	Criteria
Fuel assembly growth	Cold measurements of fuel assembly growth shall be consistent with a positive clearance between the fuel assembly and reactor internals under both hot and cold conditions at the maximum rod burnup allowed in the fuel cycle design.
Shoulder gap	Cold measurements of shoulder gap shall be consistent with a positive clearance between the fuel rods and top nozzle under both hot and cold conditions at the maximum rod burnup allowed in the fuel cycle design.

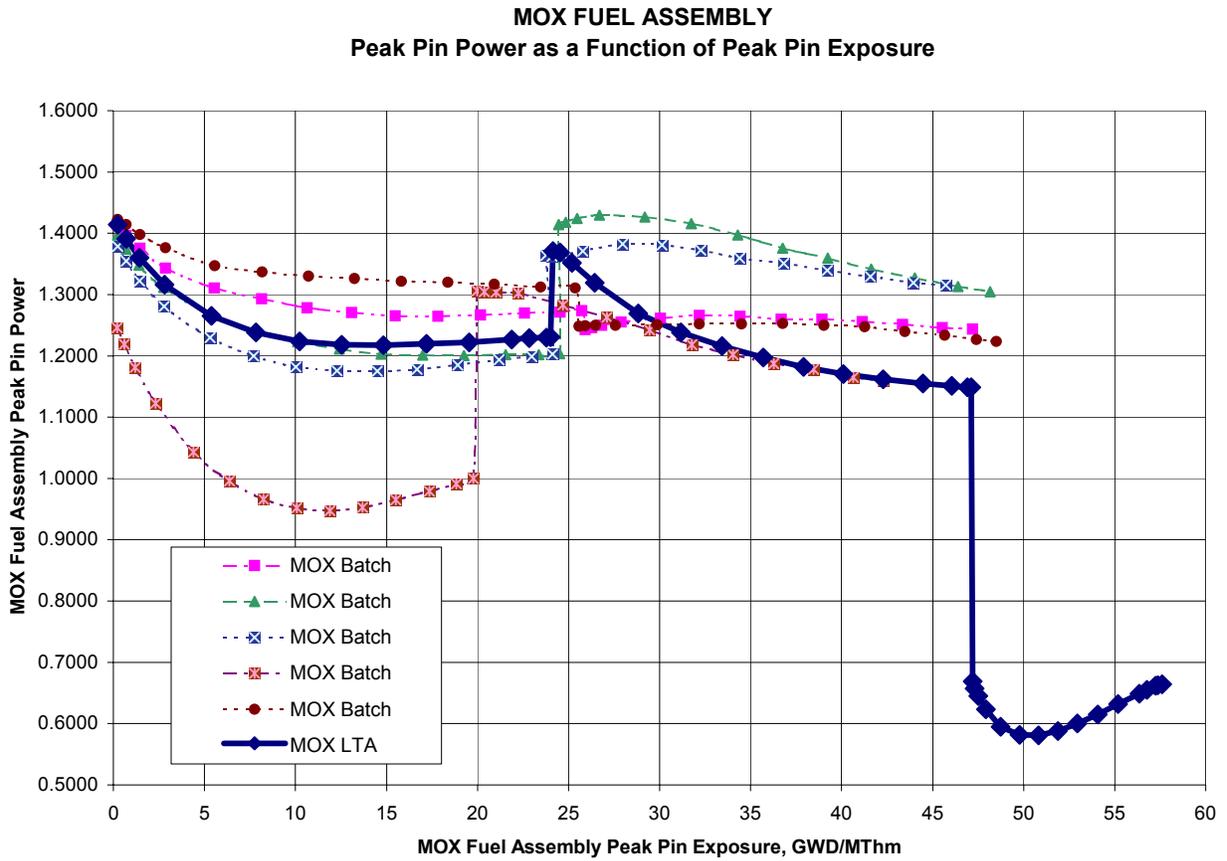


Figure 8.1 MOX Fuel Consolidated Rod Power Histories

NOTE: Data in the figure are given as an example and do not necessarily represent the peak rod power of fuel that will be produced.

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