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MOX Fuel Design Report

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Framatome ANP, Inc.

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
Note:		This is a new document.

Abstract

The U.S. Department of Energy (DOE) is implementing a program to dispose of a significant portion of the nation's surplus weapons-grade (WG) plutonium by reconstituting the plutonium into mixed-oxide (MOX) fuel rods 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 ultimately qualify the design for batch irradiation.

The fuel will be used in the McGuire and Catawba nuclear power reactors in a manner similar to the ongoing use of MOX fuel in Europe. Plans call for each MOX fuel assembly to be used for two cycles so it will clearly meet the National Academy of Science's Spent Fuel Standard for plutonium disposition upon discharge from the reactor.

The fuel design to be used in the material disposition (MD) program is the Mark-BW/MOX1 design. This 17x17 fuel assembly utilizes the Advanced Mark-BW (Reference 1) fuel assembly structure with a MOX fuel rod design. The fuel rod contains MOX pellets based on the proven rod design and pellet specification used by Framatome ANP for European MOX fuel.

MOX fuel is an intimate mixture of PuO_2 in a depleted or natural uranium oxide matrix. Approximately 95% of the MOX material is composed of UO_2; thus, the materials' properties closely match those of UO_2 fuels. These properties are well established from extensive MOX fuel operating experience in Europe and are considered in the fuel rod analyses through the use of the COPERNIC (Reference 3) fuel performance code.

The MOX fuel is characterized in terms of plutonium isotopics as reactor-grade (RG) or WG. The WG material has a much higher percentage of fissile material (^{239}Pu and ^{241}Pu) compared to the RG material, thus allowing lower plutonium concentrations with WG material to achieve the same total energy extraction. The neutronic performance of the MOX fuel has been benchmarked to a wide range of operating and test data.

WG plutonium, derived from actual weapons material, will also contain impurities, most notably gallium, present from the alloying process. The manufacturing process utilized by DCS for fabrication of MOX fuel from weapons-derived plutonium effectively eliminates the gallium from the feed plutonium prior to conversion to PuO_2.

This MOX Fuel Design Report confirms the safe and reliable operation of the fuel design that will be used for the disposition of the WG plutonium. It demonstrates that all aspects of the fuel rod design, fuel assembly design, and fuel fabrication process will provide reliable, safe operation, comparable to equivalent UO_2 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.

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Nomenclature

Acronym	Definition
ADU	ammonium diuranate
AOA	axial offset anomaly
APT	Average Power Test
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
AUC	ammonium uranyl carbonate
BOL	beginning of life
BPRA	burnable poison rod assembly
CHF	critical heat flux
DCS	Duke COGEMA Stone & Webster, LLC
DOE	Department of Energy
EDF	Electricitϑ de France
EOL	end of life
FGR	fission gas release
FRA-ANP (Fr)	Framatome ANP, SSA (France)
FRA-ANP (Ger)	Framatome ANP, GmbH (Germany), formerly Siemens
FRA-ANP (US)	Framatome ANP, Inc. (U.S.)
GWd/MThm	gigawatt-days per metric ton of heavy metal
HFIR	high flux isotope reactor
Hm	heavy metal – plutonium plus uranium isotopes
HTP	high thermal performance
IFM	intermediate flow mixing
INEEL	Idaho National Engineering and Environmental Laboratory
k_∞	infinite multiplication factor
lbf	pounds-force
LEU	low enriched uranium
LHR	linear heat rate
LOCA	loss-of-coolant accident
LWR	light water reactor
MD	material disposition (program)
MFFF	MOX fuel fabrication facility
MIMAS	micronized master blend
MOX	mixed oxide – uranium dioxide and plutonium dioxide

Nomenclature *(Continued)*

Acronym	Definition
MSMG	mid-span mixing grid
MThm	metric tons of heavy metal
MWd/MThm	megawatt-days per metric ton of heavy metal
MWd/MTU	megawatt-days per metric ton Uranium
MWd/t	megawatt-day per metric ton
MWt	megawatt-thermal
NRC	U. S. Nuclear Regulatory Commission
OCOM	Optimized Co-milling
ORNL	Oak Ridge National Laboratory
PCI	pellet-cladding interaction
PIE	post-irradiation examination
ppb	parts per billion
ppm	parts per million
PWR	pressurized water reactor
R&D	research and development
RCCA	rod cluster control assembly
RG	reactor-grade (plutonium)
RIA	reactivity insertion accident
SRP	Standard Review Plan
T	Tonne – 1000 kg
TD	theoretical density
WG	weapons grade (plutonium)

1.0 **Summary**

The U. S. Department of Energy (DOE) is implementing a program to dispose of a significant portion of the nation's surplus weapons-grade (WG) plutonium by reconstituting the plutonium into mixed-oxide (MOX) fuel rods 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 ultimately qualify the design for batch irradiation.

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MOX fuel is an intimate mixture of PuO_2 in a depleted or natural uranium oxide matrix. Approximately 95% of the MOX material is composed of UO_2; thus, the materials' properties closely match those of UO_2 fuels. These properties are well established from extensive MOX fuel operating experience in Europe and are considered in the fuel rod analyses through the use of the COPERNIC fuel performance code.

The MOX fuel is characterized in terms of plutonium isotopics as reactor-grade (RG) or WG. The WG material has a much higher percentage of fissile material (^{239}Pu and ^{241}Pu) compared to the RG material, thus allowing lower plutonium concentrations with WG material to achieve the same total energy extraction. The neutronic performance of the MOX fuel has been benchmarked to a wide range of operating and test data.

This report does not explicitly address core design and safety analysis aspects of MOX fuel. Duke will perform the core design using CASMO-4 and SIMULATE-3MOX. 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-3MOX to model low enriched uranium (LEU) and partial MOX fueled cores is described in the report, *Duke Power Company Nuclear Design Methodology Using CASMO-4/SIMULATE-3MOX* (Reference 2), which was submitted to the U. S. Nuclear Regulatory Commission (NRC) in August 2001 for review and approval.

Duke will perform the necessary safety analysis evaluations for the irradiation of MOX fuel lead assemblies at McGuire or Catawba. The core response to limiting transients will not be significantly affected by the presence of a limited number of Mark-BW/MOX1 lead assemblies. Duke will document these safety analysis evaluations as part of the license amendment request for MOX fuel lead assemblies.

Duke will perform the non-loss-of-coolant accident (LOCA) safety analyses to support batch implementation of the Mark-BW/MOX1 in the mission reactors. Duke will submit a topical report to the NRC describing these safety analyses.

For the LOCA evaluations for the lead assemblies at McGuire or Catawba, Duke will submit a License Amendment Request containing an evaluation of the impact of MOX fuel on the LOCA, provided by Framatome ANP, Inc. (US) {FRA-ANP (US)}.

For batch implementation, FRA-ANP (US) will submit a generic topical addressing the effects of MOX fuel on the LOCA Evaluation Model. The specific models to be evaluated for MOX application include the decay heat model and fuel rod model. The use of the existing UO_2 decay heat model will be justified for MOX fuel. The RELAP fuel pin gap conductivity model, currently based on the TACO code, will be modified to facilitate initialization with the MOX gap model used in COPERNIC. Also, the use of multiple MOX concentrations within the assembly and the differing types of fuel in the core necessitate that a core model be developed capable of analyzing the core with different fuel types.

2.0 MOX Design Considerations

The unique MOX features which must be addressed in the design and manufacturing activities are presented in this section. The three areas considered are MOX performance characteristics, MOX isotopics, and MOX pellet homogeneity and microstructure. The MOX performance characteristics are modeled in the COPERNIC (Reference 3) fuel performance code, which has been submitted to the NRC for review and approval. The uranium oxide and MOX fuels are quite similar due to the fact that both are primarily composed of ^{238}U, and both contain significant quantities of plutonium isotopes by end-of-Life (EOL). However, the different isotopics do affect performance, and these impacts have been successfully addressed through the European experience with MOX. The UO_2 fuel is enriched by the ^{235}U isotope, an operation which occurs on a molecular scale, while the MOX fuel is produced by the blending of UO_2 and PuO_2 powders which occurs on a macroscopic scale and results in islands of plutonium-rich particles. Specifications have been developed to assure that the islands of plutonium-rich particles are sufficiently small and distributed throughout the fuel pellet.

2.1 *MOX Performance Characteristics*

MOX fuel is an intimate mixture of PuO_2 in a depleted or natural uranium oxide matrix. Approximately 95% of the MOX material is composed of UO_2; thus, the material properties closely match those of UO_2 fuels. The following physical characteristics are potentially affected by the addition of small amounts of PuO_2 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 well established through extensive experimental and laboratory measurements. The FRA-ANP fuel performance code, COPERNIC (Reference 3), contains burnup-dependent physical properties for MOX fuel. The code has been submitted to the NRC for review and approval.

COPERNIC produces accurate steady-state and transient extended-burnup fuel performance predictions and can be applied to UO_2, UO_2-Gd_2O_3, 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 UO_2 fuel, vary little from UO_2 fuel, or are conservatively described by the UO_2 model. The thermal models in the COPERNIC code contain advanced gap conductance, gap closure, fuel thermal conductivity, radial power profile, and fuel rim models.

The predominant factors affecting fission gas release from UO_2 or MOX fuel are the power and temperature of the rod. The COPERNIC models have been shown to accurately predict measured gas release from MOX fuel rods, including those subjected to transients.

2.2 *Isotopics*

With UO_2 fuel, the fissionable component is provided by ^{235}U. The ^{235}U concentration is specified by the fuel designer and produced through the enrichment process. With MOX fuel, the ^{239}Pu isotope provides most of the fissionable component. This concentration is also determined by the fuel designer, but the quantity of PuO_2 added is dependent on the fissile content of the feed plutonium. For RG plutonium, the fissile content can vary over a wide range; to achieve the fissile content specified by the fuel designer, the fabrication process utilizes a mixture of available feed materials.

When inserted into the reactor, uranium-based fuel operates as MOX fuel soon after irradiation begins due to the generation and subsequent burning of plutonium. The uranium and plutonium contents of both fuel types are shown in Table 2.1 at beginning-of-life (BOL) and at EOL. As can be seen, both fuels (uranium-based as well as MOX) are primarily ^{238}U. At BOL, the uranium-based fuel has no plutonium; but by EOL, the uranium-based fuel is producing a significant portion (about 40%) of its power from the plutonium that has been generated during operation. Thus, uranium oxide and MOX fuels are quite similar. However, there are differences in isotopics and properties that affect performance; these differences have been successfully addressed, as evidenced by the extensive European experience with MOX fuel in commercial reactors.

2.3 *Pellet Homogeneity and Microstructure*

UO_2 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 since the enrichment operation is in the gaseous phase. Metallographic examinations of sintered UO_2 pellets will thus show very uniform appearances and grain sizes. By contrast, MOX manufactured by the MIcronized MASter blend (MIMAS) process 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 content. The products of this process are not as homogeneous as the UO_2 pellet on a micro-scale, although they are approximately the same on a macro-scale. Microscopic examination of MOX pellets shows plutonium finely dispersed in a UO_2 matrix and micron size islands of plutonium-rich particles. The particles are not pure PuO_2 particles but master mix particles with a maximum plutonium content determined by the ratio of UO_2 to PuO_2 in the master mix.

The maximum size and plutonium content of the particles are determined by the manufacturing processes. This is done during production through a milling and sieving operation, followed by a sintering process that induces diffusion of the PuO_2-bearing particles into the UO_2 lattice. Control of the process 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 effective particle size, the grain size, and the plutonium content.

The fuel pellet specification provides limits on the average and maximum sizes of the plutonium-rich agglomerates. The FRA-ANP specification for European RG MOX limits the mean particle size to less than 50 microns, with at least 95% of the particles less than 100 microns. The maximum particle size is 400 microns. The same specification has been adopted for the WG MOX.

The limitation on the average size of the plutonium-rich particles is based on the need to satisfy reprocessing requirements and to mitigate any possible effects on fission gas release. Early process fuels, while having good performance, were found to have poor solubility in pure nitric acid during a second reprocessing operation. This deficiency was a factor in the development of the current processes for MOX fuel fabrication.

In MIMAS fuel, the large agglomerates (>30 microns) retain the majority of the fission products. Approximately 25% of the total plutonium content of the pellet is present in these large agglomerates (Reference 4). Compared to uranium fuel, the difference in fission gas release between UO_2 and MOX fuel decreases with improved homogeneity. The COPERNIC MOX fission gas release database reflects fuels consistent with the current specification.

For the maximum-allowable particle size, the specification is based on the effect of large agglomerates in reactivity insertion accidents (RIA). MOX rods with PuO_2 particle sizes of 550 microns were subjected to transient tests in SPERT (Reference 5). The results showed that the effect of the large particles was to reduce slightly the cladding failure threshold energy. Furthermore, there was no indication of prompt fuel dispersal caused by the expulsion of the large particles into the surrounding water. Additional testing (Reference 6) has supported these conclusions using particles of 400 and 1100 microns. The specification limit of 400 microns for current RG MOX fuel is supported by these tests.

Table 2.1 Comparison of Uranium Based on MOX Fuel (WG) Isotopics

	BOL		EOL	
Isotope	Uranium Fuel	MOX Fuel	Uranium Fuel	MOX Fuel
^{234}U	0.04	0.00	0.02	0.00
^{235}U	4.60	0.24	0.82	0.09
^{236}U	--	--	0.62	0.03
^{238}U	95.36	95.39	91.51	92.28
^{238}Pu	--	0.00	0.04	0.02
^{239}Pu	--	4.08	0.65	1.39
^{240}Pu	--	0.29	0.28	0.85
^{241}Pu*	--	0.00	0.19	0.50
^{242}Pu	--	0.00	0.09	0.16
^{241}Am	--	0.00	0.01	0.02

NOTE: Concentration (wt% of initial heavy metal) for the most abundant isotopes in uranium and MOX fuels.

* Amount varies with decay time.

3.0 **WG Plutonium**

The characteristics and behavior of MOX fuel derived from WG plutonium is bounded by the experience base with MOX fuel derived from RG plutonium. The MOX fuel is characterized in terms of plutonium isotopics as RG or WG. Typical plutonium isotopic concentrations for WG and RG plutonium are compared in Table 3.1. It can be seen that the WG material has a much higher percentage of fissile material (^{239}Pu and ^{241}Pu) compared to the RG material, thus allowing lower plutonium concentrations with WG material to achieve the same total energy extraction. WG plutonium, derived from actual weapons material, will also contain impurities, most notably gallium, present from the alloying process.

Three aspects of weapons-derived plutonium, isotopics, impurities, and pellet microstructure are addressed in this section. The fuel characteristics, as a function of burnup, of the MOX fuel derived from WG plutonium are bounded by the range of fuel characteristics of LEU fuel and of MOX fuel derived from RG plutonium. This is due to the lower concentration of ^{239}Pu in the MOX fuel derived from WG plutonium relative to the MOX fuel derived from RG plutonium. The manufacturing process used for the MOX fuel will effectively eliminate the additional impurities which are present in the WG plutonium feed material. The differences in pellet microstructure are such that the behavior of the RG plutonium-derived MOX fuel bounds that of the WG plutonium-derived MOX fuel.

3.1 ***Isotopics***

RG plutonium is produced from reprocessed spent LWR uranium-based fuel that has been irradiated to commercial burnups, typically in the range of 30,000 to 50,000 MWd/MTU. The plutonium isotopes produced at these burnups, and extracted following irradiation, include significant percentages of ^{240}Pu, ^{241}Pu, and ^{242}Pu. The WG plutonium is created from irradiating ^{238}U to very low burnups and separating the plutonium before substantial percentages of the heavier plutonium isotopes build up. Whereas the RG material typically has 24% ^{240}Pu, the WG material is limited to less than 7% ^{240}Pu. These differences in isotopics are readily addressed through the appropriate analytical model. See Table 3.1 for typical plutonium isotopic composition of WG and RG material.

The use of WG plutonium significantly reduces the PuO_2 content of MOX fuel relative to RG material. The WG material is about 95% fissile, whereas the RG material contains significant

amounts of absorber isotopes (^{240}Pu and ^{242}Pu). Thus, MOX fuel from RG material can require plutonium contents as high as 8% to 9%.

The use of WG plutonium significantly reduces the radioactivity of MOX pellets relative to RG material. As noted above, the WG material allows a reduction in the PuO_2 content. Furthermore, the WG material contains much lower levels of the main neutron emitters – ^{238}Pu, ^{241}Am, and ^{240}Pu – than the RG material. Thus the neutron dose from WG material is significantly reduced compared to the RG material. In a similar manner, the heating due to the alpha activity, primarily from ^{238}Pu and ^{241}Am, and the gamma dose rates from these two isotopes are significantly smaller for WG MOX pellets compared to the RG material.

In LWRs, LEU fuel, RG MOX fuel, and WG MOX fuel all produce power as a result of nuclear fissions induced by a neutron field. For all three fuel types, the fissions occur primarily due to capture of thermal neutrons by uranium and/or plutonium. Both conventional LEU fuel and WG MOX fuel can be thought of as clean fuels. When initially loaded, both fuels produce power primarily from the fission of one isotope (^{235}U for LEU fuel, ^{239}Pu for WG MOX fuel). Both fuels have relatively small amounts of heavy parasitic isotopes in their composition. In contrast, RG MOX fuel contains important quantities of poisoning isotopes that complicate calculations. Due to the presence of the parasitic fertile plutonium isotopes, a RG MOX fuel assembly will require significantly more plutonium than a WG MOX fuel assembly with the same reactivity.

Table 3.2 and Table 3.3 show representative characteristics of unirradiated LEU, WG MOX, and RG MOX fuel assemblies with the same fuel mechanical design. The initial uranium enrichments and plutonium concentrations were chosen to produce an equivalent reactivity at approximately 20,000 MWd/t burnup. The tables show that all three fuel types are predominantly uranium. The plutonium mass (for both total and individual isotopes) of the WG MOX fuel assembly falls between that of the LEU fuel assembly and that of the RG MOX fuel assembly.

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 unirradiated WG MOX fuel, but they rapidly evolve toward RG MOX fuel. For WG MOX fuel, plutonium is depleted, and the isotopics of the plutonium evolve toward unirradiated RG MOX. For RG MOX fuel, the plutonium is depleted,

and the isotopics of the plutonium further degrade (i.e., a progressively lower percentage of fissile plutonium). These characteristics are shown on Figure 3.1, Figure 3.2, and Figure 3.3.

As a result of the changes described above, the source of fissions changes 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 on Figure 3.4.

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. WG MOX fuel behavior lies between that of LEU fuel and RG MOX fuel.

Several important points can be made relative to the different fuel types discussed above.

- LEU fuel, RG MOX fuel, and WG MOX fuel are fundamentally similar and, from a neutronic perspective, differ due to the relative amounts of various fissionable and fertile isotopes of uranium and plutonium.
- Significant plutonium fissions occur in medium- and high-burnup LEU fuel.
- RG MOX fuel has higher initial concentrations of heavy plutonium isotopes than WG MOX fuel. For the same reactivity, the amount of plutonium in RG MOX fuel is significantly greater than the amount of plutonium in WG MOX fuel.
- 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 all-uranium fuel requires the capability to model plutonium fuel behavior.
- RG MOX fuel presents a greater challenge to neutronic modeling methods than WG MOX fuel.
- WG MOX fuel characteristics as a function of burnup are generally bounded by LEU fuel and RG MOX fuel.

Thus it can be concluded that nuclear analysis methods that are demonstrated to model LEU fuel and RG MOX fuel with an acceptable accuracy should also be capable of modeling WG MOX fuel with a similar level of accuracy. This is the approach that has been used by Duke to qualify the CASMO-4 and SIMULATE-3MOX computer codes for application to WG MOX fuel analyses (Reference 2).

3.2 *Impurities*

Plutonium derived from weapons material may contain small amounts of alloying elements as impurities. The most notable of these impurities is gallium. Gallium is a low-melting-point element and is liquid at slightly above room temperature. It can cause embrittlement in many metals and alloys and is considered undesirable in both the processing and use of MOX fuel.

There are two primary concerns with the presence of gallium in nuclear fuel. The first relates to fabrication of the fuel. The second relates to the operation of the fuel and, particularly, the potential for cladding attack, with subsequent fuel rod failure.

The percentage of gallium present in plutonium derived from weapons is on the order of 1% by weight (maximum of 1.2%). Depending on the quantity of plutonium being processed during fuel fabrication, this concentration could fail various furnace components used in the thermal processing (sintering) and result in extensive repairs or replacement of contaminated items. Since the mission reactors require tonnage quantities of fuel, the risk associated with furnace downtime and failures from gallium embrittlement could be high; therefore, it is required that the gallium be reduced to low levels prior to any sintering operations.

Regarding in-reactor performance, a concern has been expressed that gallium could cause degradation of the cladding. Also, the gallium could migrate to the cooler regions of the fuel rod, particularly the susceptible heat-affected weld zone, and cause embrittlement and fuel rod failure.

To eliminate the potential harmful effects of gallium, the DCS fabrication process will utilize an aqueous polishing step to effectively eliminate gallium and other impurities 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 produce an acceptably pure feed material for conversion to PuO_2 powder. Other processes, such as ion

exchange, may be used for lead assembly fabrication and are expected to produce equivalent feed material.

Based on COGEMA experience and predictions, the use of a polishing process is expected to allow production of MOX fuel pellets with gallium levels in the parts-per-billion (ppb) range. Gallium at these extremely low concentrations is not expected to have any 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 to remove gallium has been evaluated through a series of laboratory tests conducted by Oak Ridge National Laboratory (ORNL) (Reference 7). 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). These tests confirmed that the decontamination factor (DF) for the process is greater than 10^5. Such a DF produces a final gallium concentration less than 120 ppb in the feed PuO_2 powder, for plutonium containing a maximum initial content of 1.2% gallium. When this polished feed PuO_2 powder is then diluted with depleted UO_2 powder, the final gallium concentration in the finished MOX pellet is comparable to current LEU fuel.

3.2.2 Gallium Content of Current UO_2 Fuels and Components

The polishing process will reduce the gallium level of the feed powder to trace levels, consistent with the level of gallium found as an impurity in currently operating UO_2 fuel pellets. 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 from processing as an impurity in cladding material and UO_2 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 UO_2 fuels that have operated successfully. The pellet samples analyzed at ORNL represent four batches of FRA-ANP (US) fuel fabricated over a five-year period from 1990 through 1994. Both Mark-B (15x15) and Mark-BW (17x17) 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 UO_2 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 polishing process will reduce the gallium content in the feed plutonium to less than 120 ppb; following dilution with depleted UO_2, the polished plutonium contributes approximately 6 ppb or less to the gallium content of the finished MOX pellets. This polished PuO_2 is then diluted with depleted UO_2 that is expected to contain trace levels of gallium, at levels comparable to the enriched UO_2 samples inspected at ORNL. Thus, the finished MOX pellets are expected to contain gallium at approximately 10 to 20 ppb. This level of gallium in the MOX fuel is consistent with the levels of gallium that have operated successfully; therefore, gallium from the WG plutonium offers no concern for the MOX fuel.

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 ppm, or 38,000 ppb. This level of gallium present as an impurity in the spring material is consistent with the level of gallium found by ORNL in the plenum springs used in the Advanced Test Reactor (ATR) tests at Idaho National Engineering and Environmental Laboratory (INEEL). 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 236 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 (due to 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 current operating cladding and fuel pellets. 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 ATR (Reference 8). 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 clad revealed no abnormal behavior.
- Gas release is consistent with European MOX experience.
- Any gallium migration to the clad has been minimal. Analyses of unirradiated archive samples and irradiated cladding indicate no transfer of gallium to the cladding within the measurement uncertainty limit.

These tests will continue to 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
- Radial power distribution

The plutonium fissile content – ^{239}Pu plus ^{241}Pu – of the WG MOX fuel is typically 94%, whereas the RG MOX fuel is 70% (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. Thus, the plutonium concentrations for MOX fuel from the WG material must be reduced approximately 40% to maintain the same total reactivity as the MOX fuel made from RG material. This reduction in total plutonium concentration ensures that the macroscopic plutonium effects on fuel performance are bounded by the data from MOX fuel made from RG plutonium.

On a microscopic scale, the distribution of fissile material within the PuO_2-UO_2 matrix is controlled by the manufacturing process. In the MOX fuel fabrication process using RG material, a primary blend 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 MIMAS 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.

For the WG material, the primary blend will be performed with a UO_2/PuO_2 ratio of 80/20. Using the same process as used with the RG material, this master mix is diluted with UO_2 to reach the final plutonium concentration. However, since the WG material has a relative 35% higher fissile content and significantly less ^{240}Pu parasitic material, the 80/20 master mix will produce plutonium-rich agglomerates from the WG material that are equivalent in fissile content with the fuel produced from RG material using the 70/30 ratio. The resulting pellet microstructure for the MOX pellet from WG plutonium will be equivalent to the pellet microstructure of the MOX pellet made from RG material.

- The UO_2 matrix that establishes the overall pellet microstructure is the same since the same process and the same 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.
- The distribution of fissile material will be the same since the particle size and distribution are the same, and the master mix adjustment has maintained the same fissile content of the plutonium-rich agglomerates.

Thus, the fission density and fission product inventory will be the same in both WG and RG MOX fuels. Since the two fuels are equivalent in fissile content and distribution of the fissile material, it can also be concluded that WG MOX fuel will behave the same as RG MOX fuel for considerations involving pellet thermal-mechanical behavior – fission gas release, transient response, and swelling.

The thermal conductivity of the WG MOX fuel will be lower than that of UO_2 fuel but bounded by that of the RG MOX fuel. Since the two materials have equivalent distributions of fissile material, and the WG material has lower overall plutonium concentrations, the thermal conductivity of the WG MOX fuel will be less affected.

The fuel pellet radial power profile for WG MOX fuel will likewise be bounded by the RG MOX fuel performance. The distribution of fissile material is equivalent for the two materials, while the total plutonium concentrations are reduced for the WG MOX fuel.

The 80/20 mix being used 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.

Table 3.1 Typical Plutonium Isotopics for the Most Abundant Isotopes

Plutonium Isotope	WG (wt%)	RG (wt%)
^{238}Pu	0.0	1.0
^{239}Pu	93.6	59.0
^{240}Pu	5.9	24.0
^{241}Pu*	0.4	10.0
^{242}Pu	0.1	5.0
^{241}Am*	0.0	1.0

* Amount varies with decay time.

Table 3.2 Sample Unirradiated Nuclear Fuel Composition

	Mass (kg)		
	LEU	RG MOX	WG MOX
Heavy Metal Loading	458.0	458.0	458.0
Total Uranium	458.0	424.6	438.0
^{235}U	18.3	1.1	1.1
^{238}U	439.5	423.5	436.9
Total Plutonium	0.0	33.0	20.0
^{239}Pu	0.0	22.2	18.7
^{240}Pu	0.0	6.9	1.3
^{241}Pu	0.0	2.6	0.0
^{242}Pu	0.0	1.0	0.0

NOTE: Any discrepancy in the total heavy metal loading is due to the presence of trace quantities of ^{234}U and ^{238}Pu.

Table 3.3 Sample Unirradiated Nuclear Fuel Isotopics

	Isotopic Fractions		
Isotope	LEU	RG MOX	WG MOX
^{235}U	4.0%	0.25%	0.25%
^{238}U	96.0%	99.75%	99.75%
^{239}Pu	0.0%	67.3%	93.3%
^{240}Pu	0.0%	21.0%	6.5%
^{241}Pu	0.0%	7.8%	0.1%
^{242}Pu	0.0%	3.0%	0.1%

Table 3.4 Gallium in UO_2 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 (17x17)	General Electric	3.55%	October 1990	9.8
McGuire Unit 2	Mark-BW (17x17)	Siemens	3.65%	December 1992	11.5
Three-Mile Island	Mark-B (15x15)	Siemens	4.75%	June 1993	9.0
Davis Besse	Mark-B (15x15)	Siemens	3.79%	May 1994	10.8

NOTE: Average Pellet Gallium Content – 10.3 ppb +/- 2.5 ppb

Fuel Components

Component	Number of Samples	Average Gallium Content
Plenum Spring	9	38,200 ppb
Zircaloy-4 Cladding	6	275 ppb

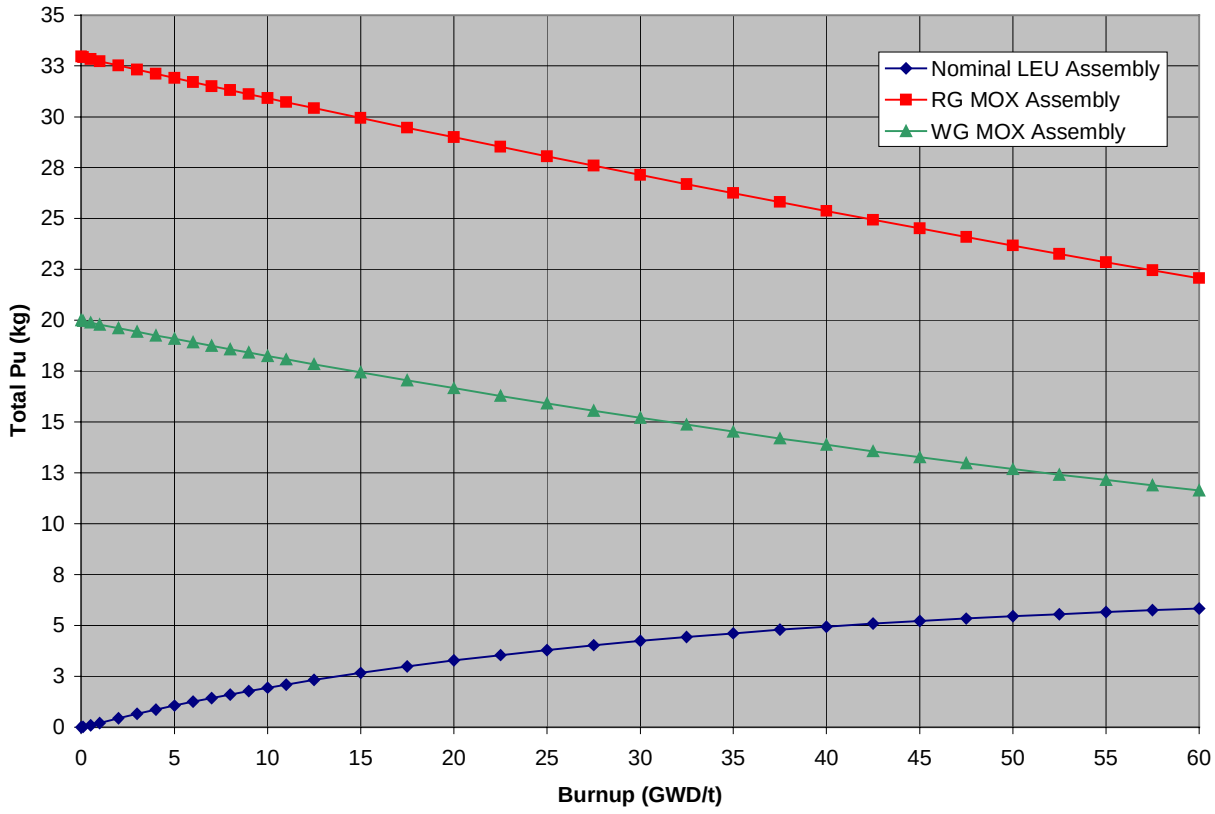

Figure 3.1 Total Plutonium Mass

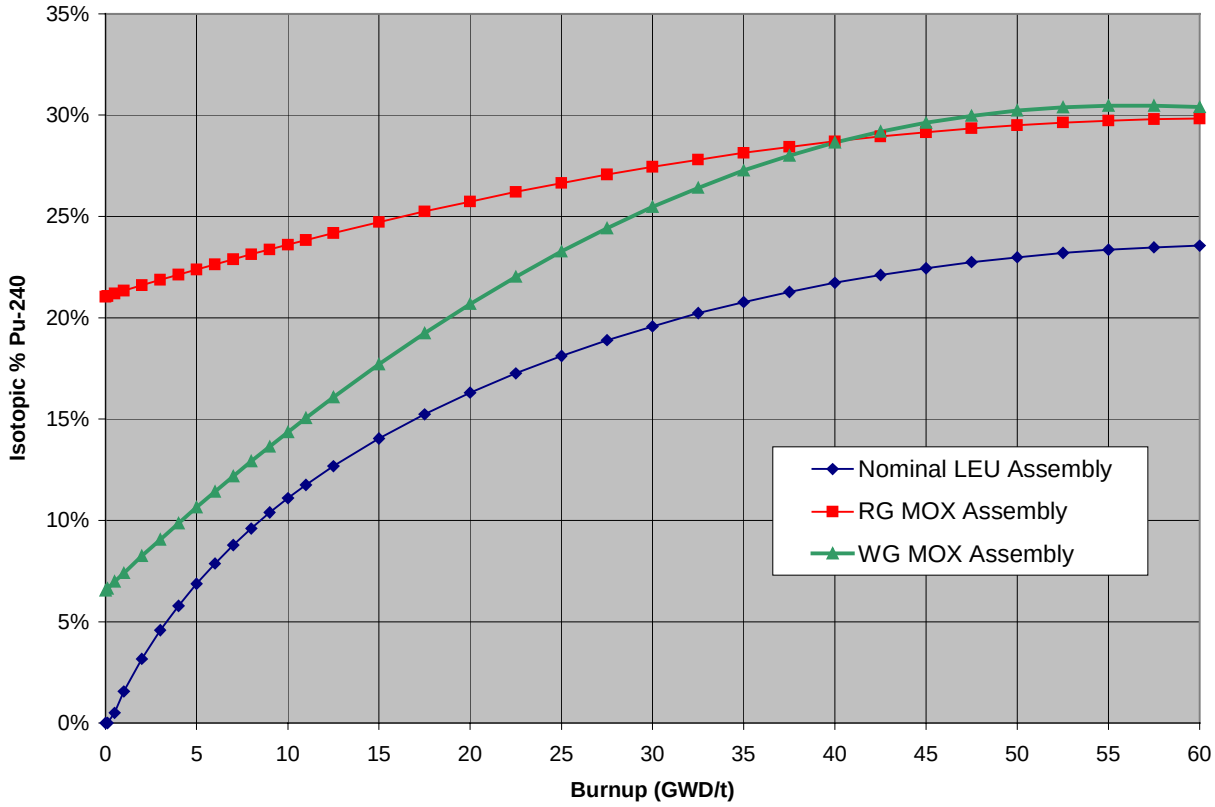

Figure 3.2 ^{240}Pu Concentration

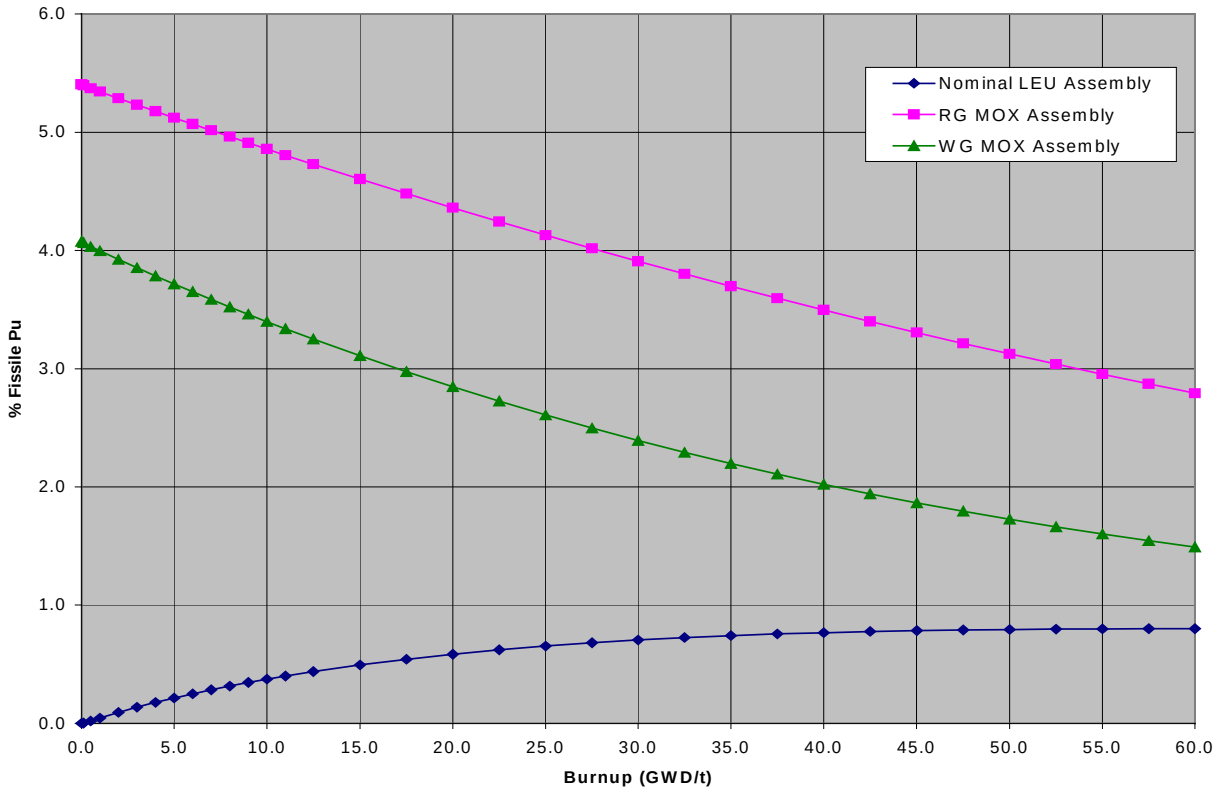

Figure 3.3 Fissile Plutonium

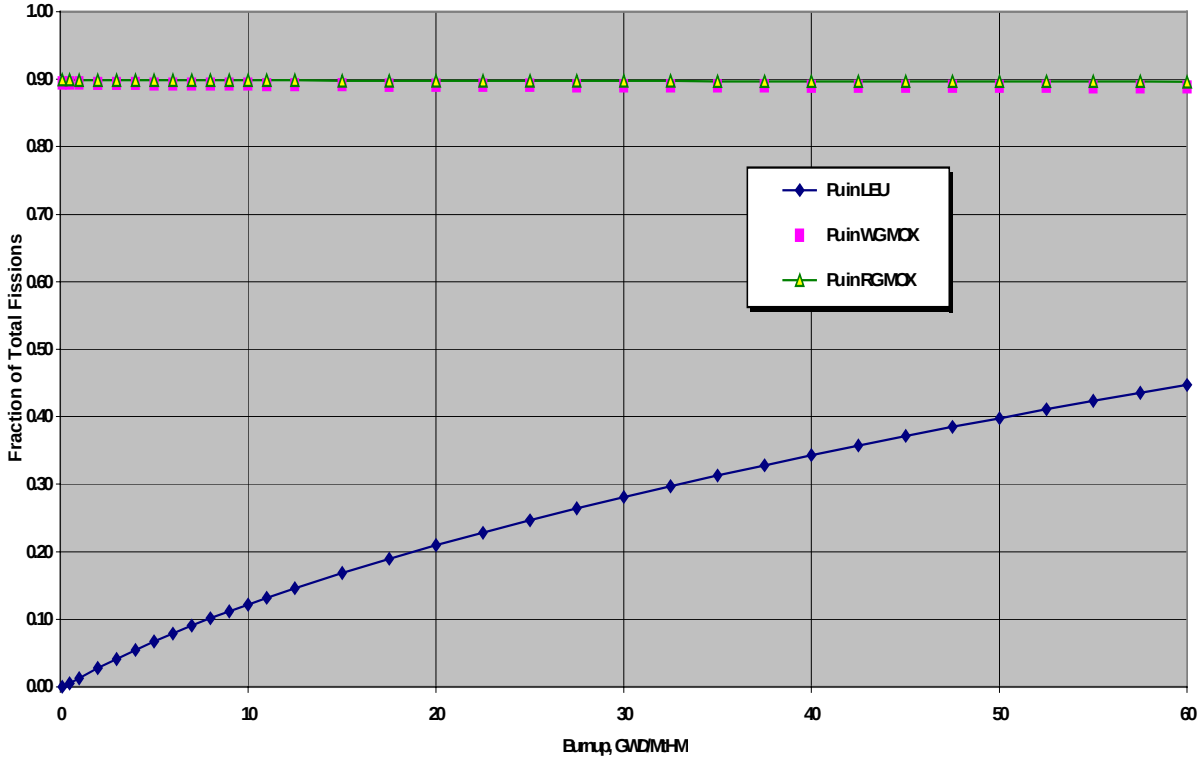

Figure 3.4 Plutonium Fissions – Fraction of Total Fissions

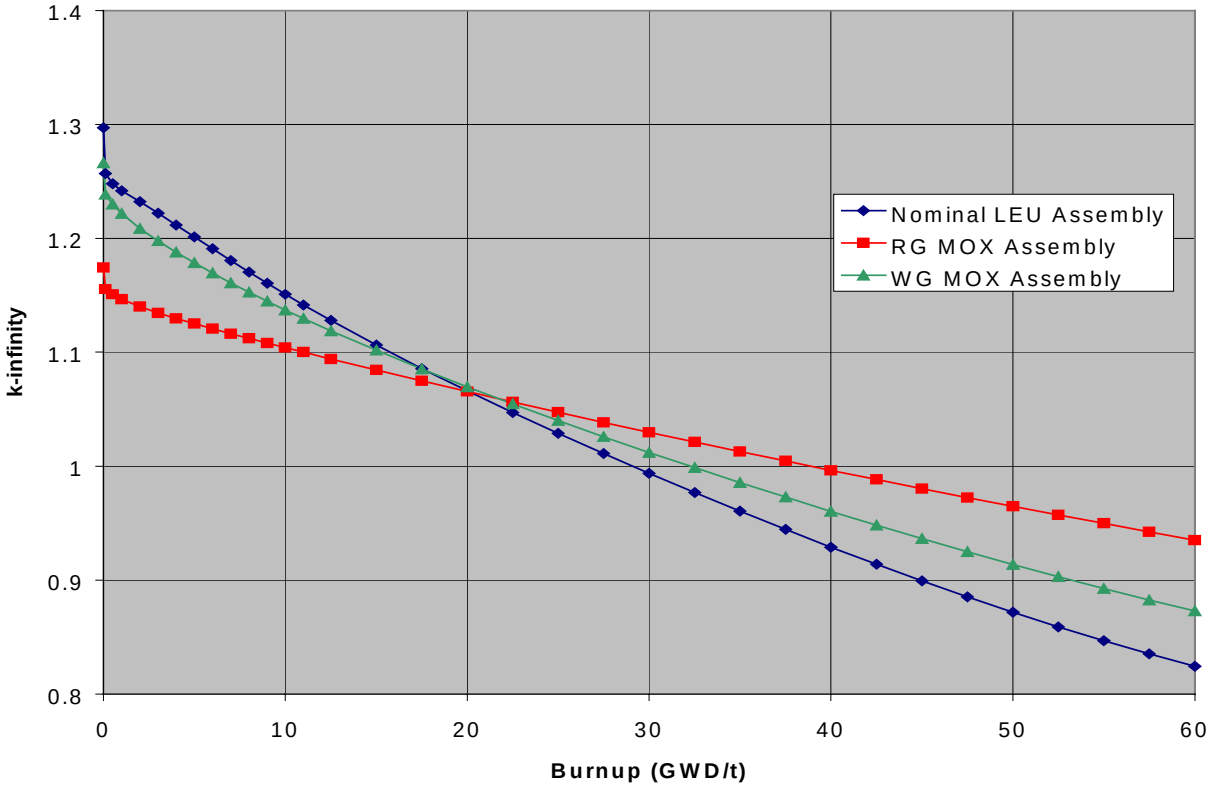

Figure 3.5 k_{oo} vs. Burnup

4.0 **Manufacturing Processes**

Fabrication processes developed by COGEMA/ BELGONUCLEAIRE will be replicated in the U.S. facility for producing the MOX fuel. Use of this proven MIMAS process for producing the MOX fuel pellets ensures that the performance of the U.S.-produced MOX fuel is consistent with the European database.

The fabrication process for the WG material includes an aqueous polishing step to remove impurities, most notably gallium. The use of polished plutonium ensures that the MOX fuel produced with the MIMAS process in the U.S. with WG plutonium is consistent with the MOX fuel produced and irradiated in Europe. This direct link to the European MOX fuel ensures the materials and operational data from Europe are applicable to the U.S. program.

The MIMAS process for fabricating MOX fuel for LWRs is the most recent evolution of the fabrication processes 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, with the aqueous polishing step unique to the processing of WG PuO_2). The MIMAS process is currently in use at the BELGONUCLEAIRE P0 plant located at Dessel, the COGEMA Cadarache plant and the COGEMA MELOX plant.

This process was developed in 1984 by BELGONUCLEAIRE to meet the requirements for high plutonium solubility while maintaining a pellet microstructure closer to the UO_2 pellet than the MOX fuel pellets initially produced by other processes. This process also has the benefit of allowing more recycling of scrap. To achieve these objectives, the PuO_2 powder is micronized with UO_2 powder and sintered recycled scraps to form a master blend with plutonium content in the range of 20% to 35% of the total mass. The successive blending and sieving steps deliver very small plutonium-rich particles whose plutonium content never exceeds the plutonium content of the primary blend.

This primary blend is force-sieved and then mechanically diluted and mixed with free-flowing UO_2 powder to obtain the specified plutonium content of the MOX fuel. The advantage of this process is to maintain the characteristics associated with the use of the UO_2 powder while significantly reducing the heterogeneous character of the plutonium distribution, which was observed in previous types of MOX fuel.

After final blending, the fuel is processed the same as in UO_2 fuel fabrication by pressing the final blend into green pellets, sintering, grinding, and inspecting the pellets before loading them into rods.

The main advantages of the MIMAS process regarding fabrication quality, flexibility, and throughput are:

- The micronization step, which concerns only about 20% of the powder, leads to reduced plutonium milling time and reduced plutonium dust production.
- The adequate dilution of primary blend in a flowable UO_2 powder avoids the use of any granulation after micronization.
- High flexibility, due to the capability for intermediate storage of the master blend and the ease of cross blending of powders for isotopic homogenization.
- The process allows for a high percentage of scrap recycling, qualified and used on a routine basis.
- The types and limited numbers of equipment used provide for minimal powder retention.
- The fine dispersion of primary blend in UO_2 is easily obtained by using efficient industrially proven mixers, which do not affect the morphology of the UO_2 powders.

The early differences that existed between UO_2 and MOX fuels have been dramatically reduced with the introduction of the MIMAS process. However, small differences still exist with regard to performance in reactor. The fuel properties and performance for MIMAS-produced MOX fuel are well established from an extensive database that has been used for code benchmarking and verification. By replicating the MIMAS process for the lead assembly fabrication and MFFF production, this database remains valid for the WG plutonium disposition program.

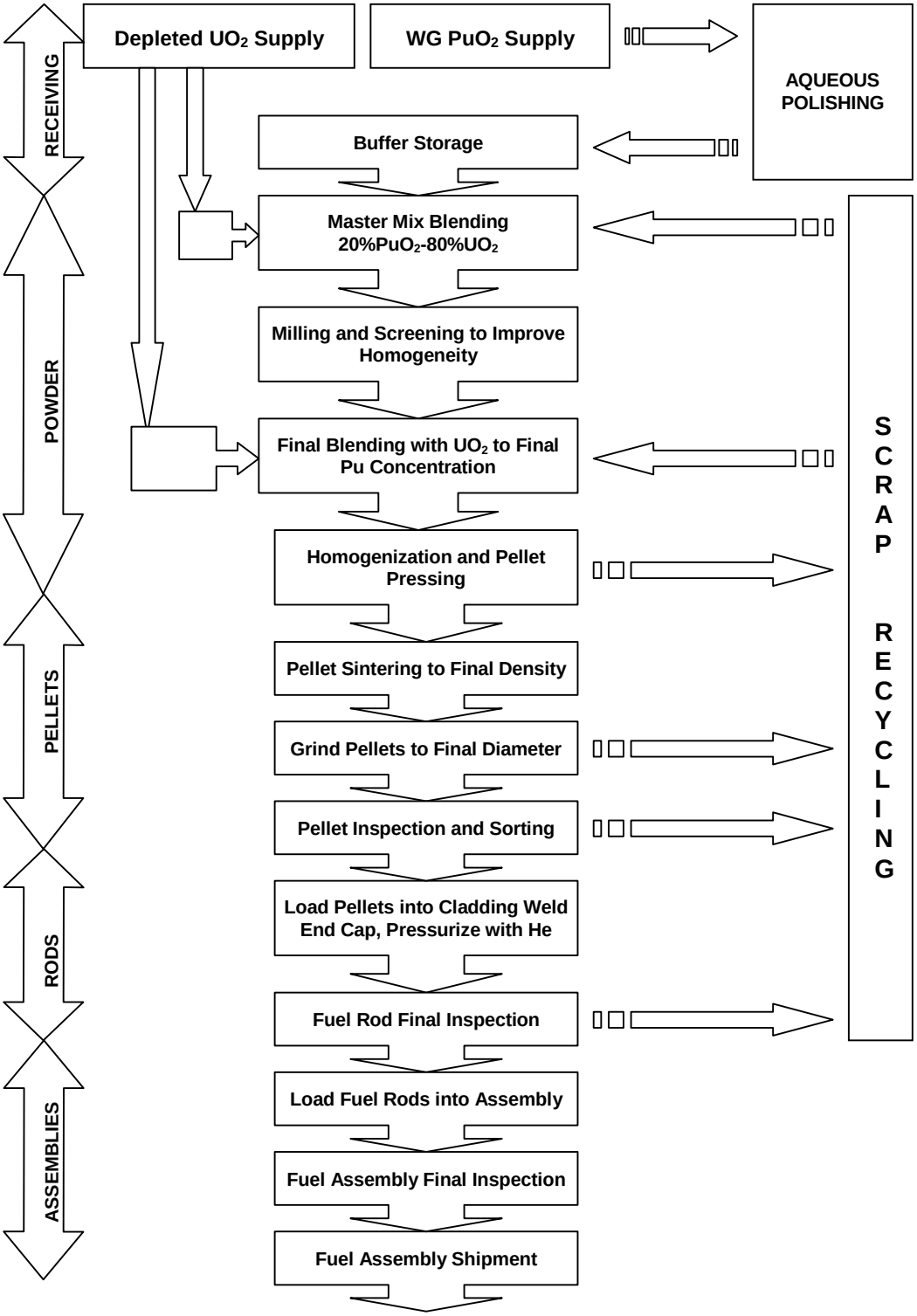

Figure 4.1 MIMAS Flow Diagram

5.0 Mark-BW/MOX1 Fuel Assembly Description

The Advanced Mark-BW fuel assembly (Reference 1) is a 17x17, standard lattice fuel assembly specifically designed for Westinghouse-designed reactors. All four mission reactors utilize the 17x17 product. The Advanced Mark-BW adaptation for MOX application, the Mark-BW/MOX1, is dimensionally and structurally identical to the Advanced Mark-BW, with the only change appearing in the fuel rod design. The Advanced Mark-BW and Mark-BW/MOX1 include the following base features:

- Seated fuel rods
- Floating intermediate spacer grids
- Removable top nozzle
- High thermal performance (HTP) spacer 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 UO_2-PuO_2 (MOX) pellets contained in a seamless M5 tube with M5 end caps welded at each end. The design typically utilizes a 144.0-inch fuel stack length. 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 top-hat shape that allows for the removal of the fuel rods from the fuel assembly, if necessary. The upper end cap has a hole to permit evacuation and back-filling of the fuel rod with helium gas prior to re-sealing.

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

The fuel pellets are a sintered ceramic of high density UO_2-PuO_2. The fuel pellets are cylindrically shaped with a dish at each end. The corners of the pellets have a 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 operation. The design density of the pellets is 95% Theoretical Density (TD), with a 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/UO_2 Design Comparison*

A comparison of typical fuel rod design details for the MOX and UO_2 fuel rod designs is summarized in Table 5.1. The MOX fuel rod design differs from the UO_2 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 shoulder gap due to the lower burnup limit for the MOX design.
- Design Density – The design density for the MOX design is 95% TD whereas the UO_2 fuel rod design utilizes 96% TD pellets. The selection of 95% TD for the MOX was made to be consistent with previous European experience with RG MOX. Future increases in the design density will be evaluated and adopted for the MD Program as the European MOX designs evolve to higher densities.
- Maximum Fuel Rod Burnup – The objectives of the MD Program can be accommodated with a maximum fuel rod burnup that is less than that currently being used for UO_2 fuels. However, for the most efficient use of the weapons plutonium, the design burnup may be increased as European experience at higher burnups provides the operational experience and data to justify the increase.

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 by Duke. This scheme optimizes the trade-off between core management and production efficiency for batch implementation, and is the same approach utilized in France by FRA-ANP (Fr), COGEMA, and Electricitϑ de France (EDF). Calculations of the lead assembly neutronics will model these assemblies explicitly using reactor physics codes to provide accurate power

predictions during each cycle of operation (Reference 2). The use of radial zones within the fuel assembly is not unique to MOX, having been implemented previously for improved neutron efficiency on the UO_2 versions of the Mark-B and Mark-BW fuel assemblies.

Table 5.1 Mark-BW/MOX1 Preliminary Design Summary

Parameter	Value	
	Advanced Mark-BW	Mark-BW/MOX1
Pellets		
Fuel Pellet Material	Enriched UO_2	PuO_2 and Depleted UO_2
Fuel Pellet Diameter, in	0.3225	0.3225
Fuel Pellet TD, %TD	96	95
Fuel Pellet Volume Reduction Due to Chamfer and Dish, %	1.24	1.11
Rods		
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	60,000	50,000
Assemblies		
Fuel Assembly Length, in	159.8	159.8
Lattice Geometry	17x17	17x17
Fuel Rod Pitch, in	0.496	0.496
Number of Fuel Rods per Assembly	264	264
Heavy Metal Loading per Assembly, kg	466.1	462.8
Number of Grids		
Bottom End	1	1
Vaneless Intermediate	1	1
Vaned Intermediate	5	5
Mid-Span Mixing	3	3
Top End	1	1

Table 5.2 Typical Plutonium Isotopics for WG Material, with Acceptable Ranges

Plutonium Isotope	WG (wt%)	Acceptable Range (wt%)
^{238}Pu	0.0	0.05
^{239}Pu	93.6	90.0 - 95.0
^{240}Pu	5.9	5.0 - 9.0
^{241}Pu	0.4	#1.0
^{242}Pu	0.1	#0.1

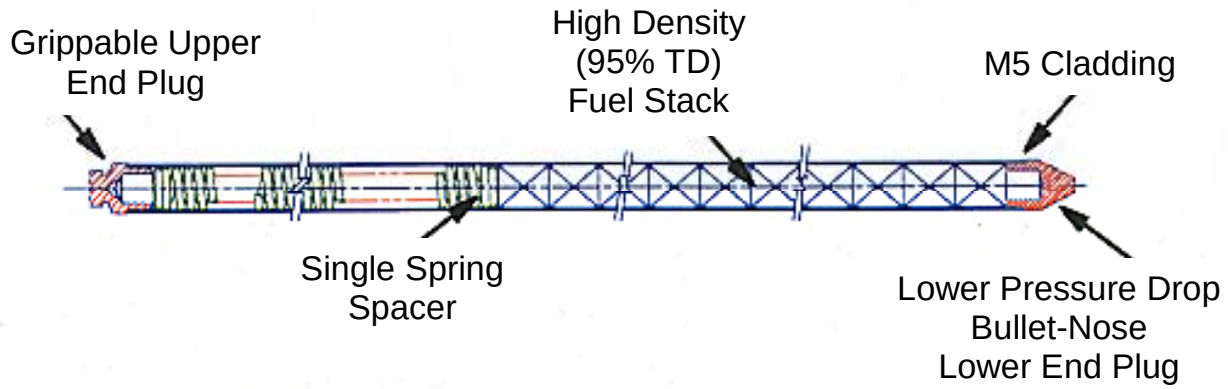

Figure 5.1 Mark-BW/MOX1 Fuel Rod Design

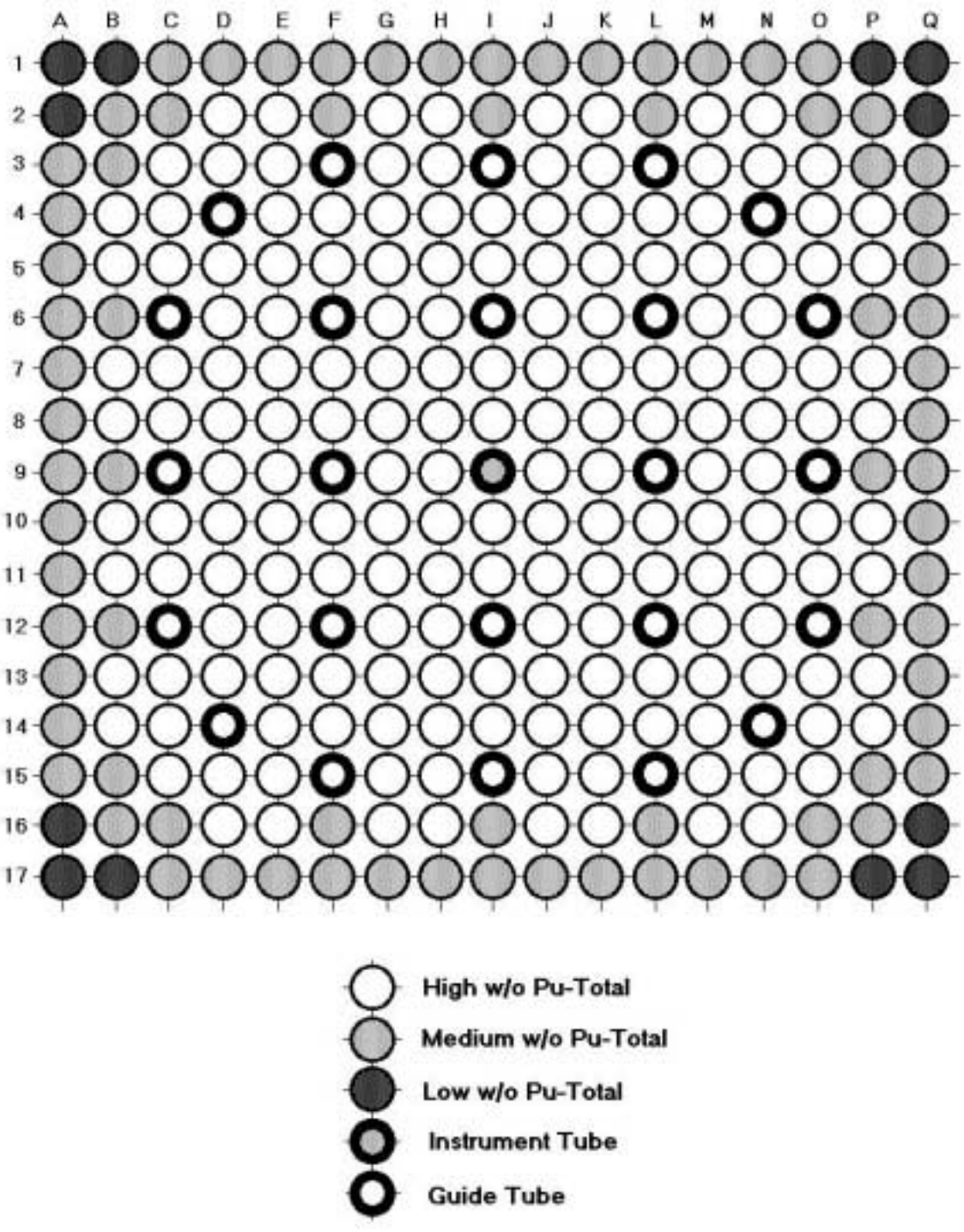

Figure 5.2 Mark-BW/MOX1 Fuel Assembly Design

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

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 in the Standard Review Plan (SRP), NUREG-0800.

The design of the Mark-BW/MOX1 fuel assembly 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.

The analyses performed in Reference 3 are applicable to the Mark-BW/MOX1 fuel assembly, except for those evaluations impacted by pellet characteristics. This section is arranged to follow the Section 4.2 criteria in the SRP, NUREG-0800. The fuel rod analyses follow the previously approved methods (see Reference 1), except that the fuel performance code, COPERNIC (Reference 3), is used with MOX-specific models and is currently under review by the NRC; COPERNIC has been used to provide pressures, oxide thickness and strains used in mechanical analyses.

Mechanical and thermal analyses on the fuel rod design have been completed using preliminary fuel cycle information provided by Duke. Methods used are as specified in the COPERNIC fuel performance code topical report. Analyses 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. This preliminary design is presented in Table 5.1, with a comparison to the Advanced Mark-BW (UO_2 design).

6.1 ***Fuel System Damage***

6.1.1 Stress

6.1.1.1 Fuel Assembly Stress

The Mark-BW/MOX1 design is dimensionally identical to the Advanced Mark-BW fuel assembly, with the substitution of MOX pellets for UO_2 pellets and a 0.25-inch increase in cladding length being the only changes. Therefore, the analyses of fuel assembly stress documented in the

Advanced Mark-BW Mechanical Design Topical Report (Reference 3) remain valid for guide thimble buckling, top and bottom nozzles, connections, and spacer grids.

6.1.1.2 Cladding Stress

The fuel rod cladding was analyzed for the stresses induced during operation. The American Society of Mechanical Engineers (ASME) pressure vessel stress intensity limits were used as guidelines, along with the NRC-approved methodology for M5 cladding (Reference 9). Conservative values are used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential temperature, and unirradiated cladding yield strength. Acceptable margins, comparable to UO_2 fuel, were found for cladding stress between the maximum-predicted stress and the allowable stress that is based on two-thirds of the minimum unirradiated yield strength.

6.1.2 Cladding Strain

The Mark-BW/MOX1 fuel rod was analyzed to determine the maximum transient the fuel rod cladding could experience before the transient strain limit of 1% is exceeded. The transient strain limit uses cladding circumferential changes before and after a linear heat rate (LHR) transient to determine strain. Cladding transient strain was also predicted using COPERNIC. Transient axial flux shapes were imposed at 10,000 MWd/MThm intervals starting at 20,000 MWd/MThm. The LHR that caused the pellet to swell and strain the cladding 1% established the LHR limit in each case. The calculated LHRs for transients that induce 1% cladding strain are not limiting to the plant's operation (comparable to UO_2 fuel) and are much greater than the maximum transient the fuel rod is expected to experience.

6.1.3 Cladding Fatigue

The fuel rod was analyzed for the total fatigue usage factor using the NRC-approved methodology (Reference 10) and the procedures outlined in the ASME Code. A maximum fatigue usage factor of 0.9 is allowed. All possible Conditions 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 predicted fatigue utilization factor for the fuel rod was less than the 0.9 limit for a lifetime that will exceed the fuel rod's design life. 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 is dimensionally identical to the Advanced Mark-BW fuel assembly, with the substitution of MOX pellets for UO_2 pellets and a 0.25-inch increase in cladding length the only changes. Therefore, the analyses documented in the Advanced Mark-BW Mechanical Design Topical Report (Reference 3) remain valid with respect to fuel rod fretting.

6.1.5 Oxidation, Hydriding, and Crud Buildup

The Mark-BW/MOX1 design has been analyzed for cladding corrosion using the COPERNIC code (Reference 3) using models approved for predicting cladding oxide thickness. The analysis demonstrates that the predicted oxide thickness for MOX fuel is significantly below the FRA-ANP licensed limit of 100 microns.

The MOX fuel rod uses M5 cladding (Reference 9) which has been shown through in-reactor testing to exhibit acceptable performance in terms of oxide film thickness growth and hydrogen uptake to burnups in excess of 62 GWd/MThm. This cladding material has demonstrated superior ductility at extended burnups to meet current NRC licensing criteria, and it has been extensively tested with in-core and out-of-core tests to show its resistance to spallation and pitting.

6.1.6 Fuel Rod Bow

The Mark-BW/MOX1 design is dimensionally identical to the Advanced Mark-BW fuel assembly, with the substitution of MOX pellets for UO_2 pellets and a 0.25-inch increase in cladding length being the only changes. Therefore, the analyses documented in the Advanced Mark-BW Mechanical Design Topical Report (Reference 3) remain valid for fuel rod bow.

6.1.7 Axial Growth

The axial gap between the top nozzle and reactor internals was conservatively analyzed to show that sufficient margin exists to accommodate the fuel assembly growth for the design

burnup. This analysis utilized a conservative maximum fuel assembly growth prediction, including the effects of the higher MOX fast neutron fluence.

The axial gaps between the top nozzle adapter plate and the fuel rods were conservatively analyzed to the design burnup to show that sufficient margin exists to accommodate the fuel assembly growth and the fuel rod growth, including the effects of the MOX fast neutron fluence. The analysis models 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 analysis demonstrates that a positive shoulder gap exists at the MOX assembly design burnup under both hot and cold EOL conditions.

6.1.8 Fuel Rod EOL Pressure

For the evaluation of internal rod pressure, a power history envelope was developed based on Duke's projected MOX fuel lead assembly peak pin power history and is expected to be representative of bounding envelopes for future partial-MOX fuel cycle designs. These analyses indicate that the rod design presented in Table 5.1 meets the fuel rod internal pressure criterion.

6.1.9 Assembly Liftoff

The Mark-BW/MOX1 design is dimensionally identical to the Advanced Mark-BW fuel assembly, with the substitution of MOX pellets for UO_2 pellets and a 0.25-inch increase in cladding length being the only changes. Therefore, the analyses documented in the Advanced Mark-BW Mechanical Design Topical Report (Reference 3) remain valid for assembly liftoff, using conservative values for fast fluence on the holddown spring arising from the specific MOX fuel neurton spectrum.

6.2 ***Fuel Rod Failure***

6.2.1 Internal Hydriding

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). The MOX fuel rod specification controls the total hydrogen, including H_2O, H_2 and hydrocarbons, to 15 ppm.

6.2.2 Creep Collapse

The fuel rod was analyzed for creep collapse using approved methods outlined in the CROV topical report (Reference 10). The acceptance criterion is that the predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life. 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. COPERNIC provides cladding temperatures and rod internal pressures that are 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 design contains no changes to the fuel rod outside diameter, fuel assembly structure, spacer grids, guide thimble, upper nozzle, lower nozzle, or any component or material other than the fuel rod internals. Thermal-hydraulic analyses, including critical heat flux (CHF) performance and CHF correlations (References 11 and 12), are not affected by the change to the rod internals. Thus, no modifications to analytical tools are required in the fuel assembly mechanical analysis and thermal-hydraulic areas to accommodate MOX fuel pellets.

Duke will perform thermal analyses for cores containing the lead assemblies with the VIPRE code, which has been approved by the NRC. The Mark-BW/MOX1 fuel assembly is designed to be hydraulically compatible with the resident fuel that will be in core when the lead assemblies are introduced. Mid-span mixing grids (MSMGs) are used in the lead assemblies and mission reactor fuel to closely match the thermal-hydraulic performance of the resident 17x17 fuel that utilizes intermediate flow mixing (IFM) grids, a similar component.

6.2.4 Overheating of Fuel Pellets

A small projected increase in fuel temperatures related to a reduction in thermal conductivity is calculated by COPERNIC. Fuel temperature predictions used for core safety analyses will directly include the effects of the MOX fuel influence on thermal conductivity. The LHRs that would result in centerline fuel melt for the MOX fuel are comparable to the fuel melt limit for UO_2 fuel.

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.

6.2.6 Cladding Rupture

The LOCA analyses will be performed and reported in a separate report. The impact of the MOX fuel will be specifically evaluated with respect to cladding rupture.

6.3 *Fuel Coolability*

6.3.1 Cladding Embrittlement

The LOCA analyses will be performed and reported in a separate report. The impact of the MOX fuel will be specifically evaluated.

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 to be submitted by Duke.

6.3.3 Fuel Rod Ballooning

The LOCA analyses will be performed and reported in a separate report. The impact of the MOX fuel will be specifically evaluated with respect to fuel rod ballooning.

6.3.4 Fuel Assembly Structural Damage from External Forces

The Mark-BW/MOX1 design is dimensionally identical to the Advanced Mark-BW fuel assembly, with the substitution of MOX pellets for UO_2 pellets and a 0.25-inch increase in cladding length being the only changes. Therefore, the analyses documented in the Advanced Mark-BW Mechanical Design Topical Report (Reference 3) remain valid for fuel assembly structural damage from external forces.

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. However, only minimal pressurized water rector (PWR) demonstration irradiations were completed, and no batch experience was obtained. Thus, the U.S. experience with MOX fuel is limited relative to the data available from Europe.

7.1.2 UO_2 Experience

FRA-ANP (US) has over 30 years of successful design and fabrication experience of nuclear fuel for PWRs. Nuclear fuel assemblies were first delivered to Duke's Oconee Nuclear Station in 1971; to date FRA-ANP (US) has supplied over 10,000 fuel assemblies of the Mark-B and Mark-BW design for PWRs.

Of particular significance, FRA-ANP (US) fuel has operated in all four of the mission reactors. For the mission reactor design (Westinghouse-designed reactors), FRA-ANP (US) began delivery of Mark-BW fuel assemblies in 1987 to Duke's McGuire Nuclear Station. Currently, FRA-ANP (US) Mark-BW fuel is operating in the U.S. in seven Westinghouse-designed 17x17 reactors: Duke's Catawba Units 1 and 2; McGuire Units 1 and 2; Virginia Power's North Anna Unit 1 (lead test assemblies); and TVA's Sequoyah Units 1 and 2. An eighth plant, Portland General Electric's Trojan Plant, also operated with FRA-ANP (US) Mark-BW fuel. As of February 2002, FRA-ANP (US) has supplied over 2,500 Mark-BW fuel assemblies to the 17x17 reactors, most of which were supplied to the mission reactors (McGuire and Catawba). Combined with the fuel experience of FRA-ANP (US)'s parent companies, a total of over 70,000 fuel assemblies have been successfully designed, licensed, and operated in reactors similar to the mission reactors around the world. The burnup experience of the FRA-ANP (US) Mark-BW fuel design is shown in Figure 7.1 to envelope the expected MOX fuel burnups.

FRA-ANP (US) will provide the fuel design experience for the mission reactor fuel; FRA-ANP (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, equal to the best in the industry. 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. 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.

7.2 ***European MOX Experience***

Fabrication and irradiation of MOX fuel in Europe represents the largest database for MOX fuel in the world. Fabrication and operation of MOX fuel in the U.S. will directly benefit from the experience of COGEMA, FRA-ANP (Fr), EDF, and BELGONUCLEAIRE. This experience will provide the data to support benchmarking, verification, and licensing of computer codes, as well as the processes for fabrication of the MOX fuel.

7.2.1 European Qualification Experience

The European experience directly 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. FRA-ANP (Fr), COGEMA, and EDF have carried out a MOX fuel qualification program in France since 1974. The major elements of this French MOX qualification program are shown in Table 7.1.

7.2.2 European Fabrication Experience

The first MOX fuel rods using zircaloy cladding with MOX fuel produced utilizing the MIMAS process were introduced in the St. Laurent B1 core in 1987. By mid-2000, MOX fuel was operating in 20 EDF commercial reactors.

The fabrication of MOX fuel in the U.S. will utilize the same MIMAS process used in Europe. Details of the process are provided in Section 4.0. Through the use of the aqueous polishing process, the impurities introduced to the WG MOX will be effectively eliminated, thereby

ensuring that the European experience 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 manufacturing plants. These three facilities have produced a combined total of more than 435,000 MOX fuel rods for 33 of the 35 commercial nuclear reactor units irradiating MOX fuel in Europe. In addition, the various production runs in these plants led to the development of the MIMAS process, which is currently in use at all three of these facilities. A complete listing of all of the European plants using MOX fuel from the MIMAS process is provided in Table 7.2.

7.2.3 European Operational Experience

The extensive European operational experience will be used in the fuel qualification effort to benchmark the appropriate core physics analysis tools and as an overall demonstration of 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.

The operating schemes include 1/3 MOX fuel core, 1/4 MOX fuel core, hybrid refueling (where UO_2 assemblies are used for four annual cycles while MOX assemblies are used for three); annual cycles; and extended cycle designs. The MOX fuel assemblies have been discharged with assembly average burnups as high as 55,000 MWd/MThm. 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, a more challenging fuel duty than the U.S. plant operational mode. Since 1991, two EDF reactors using MOX fuel have been operating under load-follow and frequency control conditions. Based on this successful experience, all of the EDF reactors using MOX fuel have been authorized, since 1995, to operate under load-follow conditions.

In the EDF 900 MWt (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 UO_2 assemblies by MOX fuel assemblies is done without any penalty on core operating conditions. An extended rod burnup goal of 61,000 MWd/MThm (52,000 MWd/MThm assembly burnup) has been set for 2004 as part of the MOX Parity project, well in advance of the required mission

reactor initial core loading in 2008. 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.

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 2001, a total of 104 MOX fuel assemblies had completed 1 to 3 cycles of operation, with a maximum fuel assembly discharge burnup of 46,500 MWd/MThm.

The current rod design burnup in France is 48,000 MWd/MThm (43,000 MWd/MThm assembly burnup). In Belgium, the average assembly discharge burnup is about 44,000 MWd/MThm at Tihange 2 and 46,500 MWd/MThm at Doel 3. Design assembly burnups as high as 55,000 MWd/MThm are currently proposed in Germany. Thus, the MOX exposure experience in Europe clearly envelops the projected typical maximum assembly burnup for the mission fuel of 45,000 MWd/MThm. Table 7.3 shows the maximum discharge burnup for the European plants using MOX fuel produced by FRA-ANP (Fr)/COGEMA and by Siemens with the same process to be used on the lead assemblies and mission reactor fuel {MIMAS for FRA-ANP (Fr)/COGEMA and Optimized Co-milling (OCOM) for Siemens}.

Use of MOX fuel with M5 cladding is proceeding in advance of the U.S. application of MOX with M5 in the mission reactors. The German reactor KKP-2 loaded 16 MOX fuel assemblies with M5 cladding in 1998; an additional 16 MOX fuel assemblies with M5 were loaded in 1999. Currently, 15 MOX assemblies are in their 4^{th} cycle of operation. The German reactor GKN-2 loaded 16 MOX fuel assemblies with M5 cladding in 2000 and 8 in 2001. An additional 18 assemblies have been supplied for future loading. The German reactor KKG has received 32 MOX fuel assemblies with M5 cladding, and loaded 16 in 2001. Current plans for use of M5 cladding with MOX fuel include 32 fuel assemblies to be delivered to the German reactor KKP and 20 fuel assemblies to KBR in 2002 and 2003.

Two fuel assemblies with some M5 cladding 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.

7.2.4 Fuel Reliability Experience

A comparison of the reliability of European MIMAS-produced MOX fuel with that of UO_2 shows very similar operating experience. During the 13 years that reload quantities of MIMAS-produced MOX fuel rods have been irradiated in commercial reactors, representing over 435,000 operating fuel rods, only 6 failed rods have been seen in MOX fuel assemblies through the end of 2000. None of the failures has been attributed to the use of MOX fuel. Five of the failures are known to be due to debris fretting; one is believed to be due to the same mechanism. Similar failures have been observed in UO_2 fuel assemblies.

The fuel reliability experience with MOX fuel in Europe is expected to be applicable to the U.S. The use of the aqueous polishing process for preparing the WG plutonium will ensure that there are no effects due to contaminants, such as gallium. Furthermore, the base fuel design to utilize the MOX pellets (Mark-BW) has reliability as high as any fuel design in operation in the U.S. 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 UO_2 fuel. MOX fuel has been examined poolside after one to four cycles of irradiation. In addition, 55 irradiated MOX fuel rods have been examined in hot cells. The data from these examinations, combined with a comprehensive out-of-core and in-core analytical test program on the current fuel products, are being used to confirm and upgrade the design models and codes necessary for the continuing improvement of the MOX product.

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 UO_2 fuel for both waterside corrosion and rod dimensional effects. Furthermore, the rods

operating under load-follow conditions behaved similarly to the reference rods operated under base load conditions. Moreover, prototypical MELOX fuel rods (MIMAS process) have been examined after one, two, and three irradiation cycles. Four-cycle fuel rods were hot-cell examined in 2001. Fractional fission gas release of the three-cycle fuel rods lies in the lower range of the MIMAS database.

The waterside corrosion result was also confirmed more recently on optimized Zircaloy-4 cladding in high temperature reactors in Germany for a rod average burnup of 49,000 MWd/MThm. For both MOX fuel and UO_2 fuel, the maximum oxide thickness was on the order of 80 microns at this burnup, confirming that MOX fuel performs the same as UO_2 fuel relative to zircaloy cladding corrosion. Confirmation of the same equivalence for the advanced cladding (M5) to be used on the Mission Reactor fuel will be obtained in Germany where M5 rods containing MOX fuel will achieve a burnup of 55,000 MWd/MThm in 2002. Poolside measurements carried out after two cycles of irradiation in the KKP-2 reactor (rod burnup of 37,500 MWd/MThm) indicated oxide thickness of 16 microns. Results from measurements after three cycles performed in 2001 are not yet available.

7.2.5.2 High-Burnup Hot Cell Examination

To provide verification of performance and benchmarking data to support higher burnup needs, four-cycle MOX fuel rods with burnups up to 53,000 MWd/MThm have been examined in hot cells. The data did not show any fission gas release enhancement due to the burnup effect. One assembly has completed a fifth irradiation cycle in the Gravelines-4 reactor. Fuel rods up to burnups of 61,000 MWd/MThm have been shipped to the hot cell for rod puncture and gas analysis.

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. These research and development (R&D) programs conducted by the French partners, as part of 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 UO_2 fuel. Transient fission gas release from the MOX rods was equivalent to that of UO_2 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 13).

These ramp test rods also produced information on transient fission gas release (since the rod 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 UO_2 contained 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 good behavior of MOX fuel. The ramp test programs carried out in the BR2 reactor are describe in the paper of M. Lippens at the Vienna Symposium on MOX Fuel Cycle Technologies (Reference 14) and references cited herein. The analytical test programs (testing of two-cycle and four-cycle MOX fuel rods from EDF/Framatome) at Halden are made or are being made in the framework of the Joint Program (Reference 15).

7.2.5.5 Reactivity Insertion Testing

Reactivity insertion tests have been used to determine the enthalpy addition criterion for UO_2 and MOX fuel. Three test series for reactivity insertion impact on UO_2 and MOX fuel were performed in the SPERT test program in Idaho, the RIA test program in the Nuclear Safety Research Reactor in Japan and, most recently, the RIA test series in the Cabri loop in France. The seven LEU and four MOX fuel tests at Cabri included two uranium fuel failures (tests NA-1 and NA-8) and one MOX fuel failure (test NA-7). The Cabri data are still being evaluated, and no definitive conclusions have been drawn about any differences between MOX fuel and LEU fuel behavior during RIA.

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 (SLBI reactor) 43,000 MWd/MTHM rod burnup	Qualification of product and performance modeling
1987-1991	Irradiation + PIE	Irradiation of MOX fuel rods in the small CAP 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 manage-ment 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 manage-ment licensing
1996	Analytical experiment DEFORMOX	Instrumented experimental irradiation of UO_2 and MOX fuel; online measurement of clad 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 (UO_2/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	X	X
3		Dampierre 1	X	X	X
4		Dampierre 2	X	X	X
5		Dampierre 3	X		
6		Dampierre 4	X		
7		Tricastin 1	X		
8		Tricastin 2	X	X	
9		Tricastin 3	X	X	
10		Tricastin 4	X		
11		St. Laurent 1	X	X	X
12		St. Laurent 2	X	X	X
13		Gravelines 1	X		
14		Gravelines 2	X		
15		Gravelines 3	X	X	X
16		Gravelines 4	X	X	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	X
24		Grafenrheinfeld			X
25		Phillipsburg 2		X	X
26		Brokdorf			X
27		Gundremmingen B			X
28		Gundremmingen C			X
29		Grohnde		X	
30		Isar 2		X	
31		Obrigheim		X	
32		Neckarwestheim 2		X	
33	Switzerland	Beznau 1			X
34		Beznau 2			X
35		Gosgen			X

Table 7.3 European MOX Burnup Experience

Country	Reactor		Maximum Discharge Burnups (MWd/MThm) of Assemblies Having Completed		
	Number	Type	3 Cycles	4 Cycles	5 Cycles
FRA-ANP (Fr) Deliveries					
France	20	17 x 17	40,500	46,000	55,000 (61,000 - Rod)
Belgium	2	17 x 17	44,000	46,500	
Germany	2	16 x 16 18 x 18	43,000	52,000	
FRA-ANP (Ger) Deliveries					
Germany	9	14 x 14 to 18 x 18		49,000	
Switzerland	3	14 x 14 and 15 x 15		54,000 (65,000 - Rod)	

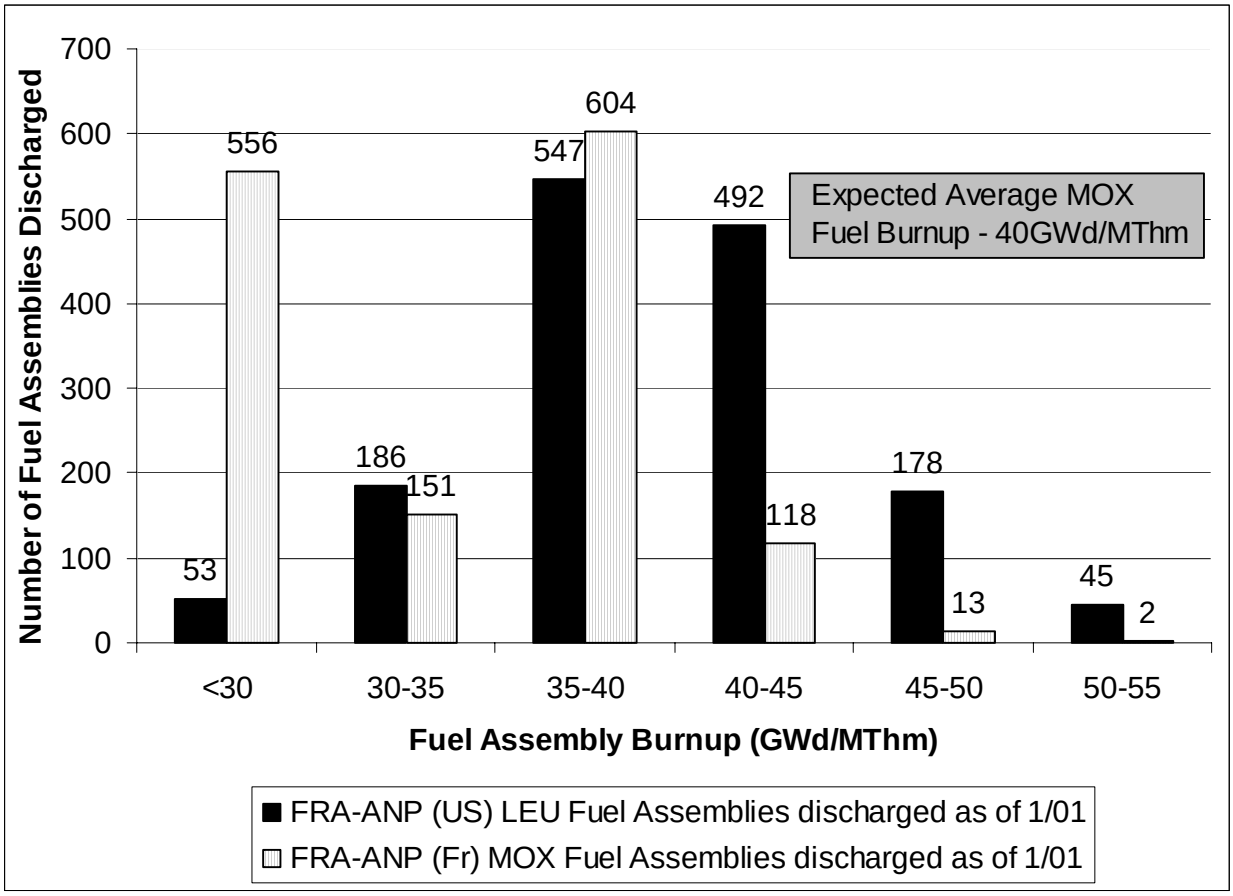

Figure 7.1 Framatome ANP Mark-BW and MOX Burnup Experience

8.0 **Lead Assembly Program**

Lead assemblies will be fabricated, irradiated, and examined as final confirmation of the design and fabrication processes.

Planning for the Lead Assembly Program is based on the fabrication of four (4) lead assemblies. However, considering possible limitations on feed WG plutonium for this program, two (2) lead assemblies are also considered acceptable. The use of two or four lead assemblies is consistent with previous lead assembly programs for confirmation of new designs prior to batch deployment. The operation of the lead assemblies is confirmatory; there are no data requirements from the lead assemblies to qualify any analytical tools or modify fuel performance models. The four mission reactors are of the same Westinghouse design and utilize the same UO_2 fuel for the resident core. Therefore, operation in any one of the mission reactors will be representative of operation in all of the mission reactors. Furthermore, the transition to batch implementation will be accomplished in phases. Following the second cycle of lead assembly irradiation, the first production batch is scheduled for operation starting in October 2008 using a partial MOX core loading of only 15%. The core fraction will be increased in the second MOX batch, with the maximum core fraction (approximately 40%) not reached until the insertion of the third MOX batch.

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. Irradiation is tentatively planned for Duke's McGuire Unit 1, Cycle 17, starting in April 2004. The 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 after each irradiation cycle. After two cycles, an accumulated burnup greater than 42,000 MWd/MThm is projected. Two cycles of irradiation are sufficient to demonstrate MOX fuel performance to the burnup limits requested in this report.

A third cycle of irradiation will be performed to gain information to support higher burnup operation. A hot cell examination on selected rods from the lead assemblies will be performed at a DOE facility following this third cycle.

8.1 *Purpose*

The primary purpose of the Lead Assembly Program is to confirm the acceptability of the MOX fuel design for certification of the mission reactor fuel for batch implementation. 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 Lead Assembly Program will demonstrate the successful application of the MIMAS process to the WG plutonium and the application of the aqueous polishing process to reduce impurities in WG plutonium to trace levels.

- Trace Levels of Impurities

  The Lead Assembly Program will help confirm that the presence of trace levels of gallium (<< 1 ppm) does not adversely affect fuel rod cladding integrity.

- Fuel Assembly Hardware

  The 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 LHR, coolant chemistry, 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 the existing movable incore detector system in order to validate the ability to predict and measure accurately 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. One fuel assembly design will be used for all four mission reactors, as described in Section 6.1. 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.

The lead assemblies, as well as the mission reactor fuel, will utilize Burnable Poison Rod Assemblies (BPRAs). The BPRAs will be supplied by FRA-ANP (US) based on the specification (boron concentration and number of active pins/assembly) provided by the utility.

The Advanced Mark-BW design used as the basis for the Mark-BW/MOX1 design is fully qualified (Reference 1). The only changes required are those associated with the fuel rod design to accommodate the MOX pellets. The MOX pellets will be fabricated to substantially the same specifications and with the same processes as the MOX pellets used in Europe, ensuring the applicability of the extensive European database. The master mix ratio of UO_2 to PuO_2 powder will be revised from 70/30 to 80/20 to account for the higher fissile content of WG plutonium compared to RG plutonium. This change to the processing will allow the fissile content of the plutonium-rich particles to remain the same as that of the RG product.

8.3 ***Fabrication***

The Lead Assembly Program will demonstrate the manufacturing processes that will be used for the disposition of the WG plutonium. These processes will replicate the processes used in Europe for fabrication of MOX pellets (A-MIMAS). Polished PuO_2 powder will be supplied by DOE for the lead assemblies and will be prototypical of the powder that will be produced in the MFFF for the mission reactors. The chemical and physical properties of this powder will be within the database of powders routinely used in Europe, thereby ensuring consistency with the European product and applicability of the European performance database. The PuO_2 powder specification developed for batch MOX fuel will be applied to the lead assemblies.

There will be four (4) complete assemblies fabricated, prototypical of batch production design and material, to demonstrate that the changes associated with implementation of MOX fuel do not adversely impact the operability of the fuel and core. The use of four fuel assemblies provides symmetry and adequate operational exposure, while supporting the mission schedule.

8.3.1 Feed Material Requirements

8.3.1.1 Plutonium Feed

The plutonium oxide feed powder 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 the nitrate through the oxalate precipitation process. This process provides significantly better control of the PuO_2

particle size, shape, and distribution compared to product obtained by dry processing (e.g., burning plutonium metal to the oxide). Close control of particle size and size distribution is essential in powder production both from a manufacturing perspective and fuel performance. Following precipitation and calcination in the temperature range of 600°C to 650°C, the PuO_2 powder will be homogenized and thoroughly characterized. The chemical and physical properties of such PuO_2 must be repeatable and within the PuO_2 powder specification that DCS will provide in order to be fully consistent with the database of powders produced in Europe. Thus, this experience base will be applicable to the lead assembly product.

8.3.1.2 Plutonium Polishing

WG plutonium may have a gallium content up to 1.2%. This gallium has the potential for causing manufacturing and operational problems and thus must be removed by polishing down to the ppb range in the finished MOX pellet. The specification for the PuO_2 powder will limit the gallium levels to less than 120 ppb following polishing. This limit will ensure that the finished pellets, after mixing with UO_2 powder, will contain only trace levels of gallium, comparable with gallium levels in current UO_2 fuel.

8.3.1.3 Uranium Feed

The majority of the European MOX irradiation experience is based on the use of depleted (and some natural) UO_2 prepared by the ammonium diuranate (ADU) wet route process, or by the ammonium uranyl carbonate (AUC) wet route process. The MELOX production and most of the European MOX fuel are based on ADU powder produced in the COGEMA TU2 plant. A sufficient quantity of this UO_2 powder will be made available by DCS for the Lead Assembly Program. This approach ensures complete similarity, from the UO_2 standpoint, between the lead assembly and the European MOX experience, thereby avoiding any possible effects due to differences in uranium feed characteristics.

For the UO_2 supply for batch implementation at the mission reactors, equivalent feed material will be used (i.e., the powder will be obtained by the same process as the TU2 process and with the same specifications and controls). The UO_2 powder for the MFFF will come from a U.S. facility qualified for the fabrication of ADU powder with the TU2 specification and controls. Use of UO_2 powder from any other source will be qualified in Europe with RG MOX before potential use in the mission reactors.

8.4 *Irradiation Plan*

The lead assemblies will be irradiated in a McGuire or Catawba reactor, with three cycles of irradiation planned. One of the lead assemblies will be located in an instrumented location to verify predicted operational neutronic performance during the irradiation cycles. Neutronic data will be compared to similar data obtained from instrumented UO_2 assemblies to verify core predictions.

The lead assemblies will be located in relatively high-power but non-limiting positions to ensure representative operating parameters for batch implementation. Figure 8.1 presents bounding power history envelopes from the MOX fuel lead assemblies (three cycles) as well as five representative MOX fuel assemblies from batch use of MOX fuel (two cycles). The figure is based on preliminary lead assembly and batch core designs. Each curve is a composite of all of the fuel rods in one assembly and depicts the maximum power of any pin versus the maximum burnup of any pin in that assembly. As can be seen, after two cycles of irradiation, maximum pin burnups for lead and batch assemblies are comparable. The lead assemblies are projected to reach a maximum fuel pin burnup in excess of 47,000 MWd/MThm in two cycles, consistent with the proposed fuel pin 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 will be performed to obtain data at higher burnup to confirm performance, verify margin predictions, and benchmark fuel performance models. The maximum fuel pin burnup is expected to exceed 57,000 MWd/MThm in this third cycle. This burnup exceeds the proposed fuel pin burnup limit of 50,000 MWd/MThm. However, these data may eventually be used to justify extended burnup operation and a future increase in the burnup limit for the MOX fuel.

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 licensing and PIE comparisons. The pellets will be measured for grain size and micro-structure features, including PuO_2 particle size, homogeneity of PuO_2 dispersion, resinter test performance, diameter, length, porosity distribution, and complete chemical impurity content. A statistically valid sample of pellets will be examined to completely quantify the MOX pellet attributes. 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. As a minimum, 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. A unique marking that will identify the rod to the specific plutonium loading will be used.

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 and one rod representative of each batch of MOX fuel produced (approximately ten rods total) 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 condition with the unirradiated base case during hot cell examinations.

Following standard nuclear identification procedures, each lead assembly will be specially identified with unique serial numbers. 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 every mid-span elevation.

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 McGuire or Catawba starting in 2004. After two cycles of irradiation, the lead assemblies will reach a burnup of approximately 42,000 MWd/MThm, with a maximum projected rod burnup of 47,000 MWd/MThm. After each cycle, the assemblies are scheduled to be examined poolside to verify acceptable performance and provide data for later evaluation. The poolside examinations will employ proven non-destructive techniques typically used in the examination of irradiated UO_2 fuel assemblies. The scope of the poolside examinations is expected to include the items listed in Table 8.1. This table includes the purpose of each inspection and the expected result, relative to UO_2 assembly performance.

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 (as a minimum):

- Fission gas release
- Fuel clad metallography
- Fuel pellet ceramography
- PCI
- Burnup analysis
- Burnup distribution

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 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 fuel rod power

histories will be generated following the completion of the fuel cycle to allow for better accuracy in comparing predicted-to-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 UO_2 fuel database. The measurements performed after the first and second cycles will provide the basis for final Certification that the Fuel Qualification Plan has been completed and the fuel is ready for batch implementation.

Lead Assembly Performance Criteria for Batch Operation

Measurement	Criteria
Fuel assembly growth	Fuel assembly growth shall not be greater than 0.41% dl/l at 44,000 MWd/MThm
Fuel rod growth	Fuel rod growth shall not be greater than 0.7% dl/l at 44,000 MWd/MThm
Fuel assembly rod cluster control assembly (RCCA) drag force	Drag force shall not exceed: • 100 lbf in dashpot • 60 lbf above dashpot
Fuel rod integrity	No failed fuel rods in the lead assemblies from MOX fuel-related causes
Fuel rod oxide thickness	Peak oxide thickness (using moving average over 1 inch) shall not exceed 50 microns

Later, after the third-cycle hot-cell exam, a second comparison will be performed to compare hot-cell results to specific predictions, the overall UO_2 fuel database, and both specific MOX results and the overall MOX database. In addition the hot cell results will be compared to poolside measurements to verify poolside measurement techniques.

Table 8.1 Lead Assembly Poolside PIE

Inspection	Purpose	Expected Result
Fuel assembly visual	Overcheck to provide confirmation of acceptable performance.	Same as UO_2 with M5 clad fuel rods and guide thimbles
Fuel rod visual	Overcheck to provide confirmation of acceptable performance.	Same as UO_2 with M5 clad fuel rods
Fuel rod crud measurements	Confirm equivalency to UO_2 fuel rod. Address axial offset anomaly (AOA) issues.	Same as UO_2 fuel – light crud deposits
Fuel rod growth (shoulder gap closure)	Confirm acceptable margin for fuel rod operation. Verify shoulder gap.	Same as UO_2 with M5 clad fuel rods and guide thimbles
Fuel assembly growth	Confirm predictions and equivalency with UO_2 assembly	Same as UO_2 with M5 clad fuel rods and guide thimbles
Fuel assembly RCCA drag force	Address incomplete RCCA insertion issue.	Same as UO_2 with M5 guide thimbles
Fuel rod oxide thickness	Confirm equivalency to UO_2 rod. Compare to corrosion predictions.	Same as UO_2 with M5 clad fuel rods
Fuel rod fission gas release	Confirm predictions.	Slightly higher than UO_2 rods due to power, thermal conductivity, microstructure differences
Water gaps (fuel rod bowing)	Determine rod bow equivalence to UO_2 rod and fuel assembly envelope	Same as UO_2 with M5 clad fuel rods and guide thimbles
Grid width	Confirm grid growth predictions, equivalency to UO_2 fuel assembly.	Same as UO_2 with M5 grids
Grid oxide thickness	Confirm grid strength margins.	Same as UO_2 with M5 spacer grids
Guide thimble plug gauge	Address incomplete RCCA insertion issue. Verify distortion-free operation.	Same as UO_2 with M5 guide thimbles, all gauges pass all grid spans

Table 8.1 Lead Assembly Poolside *(Continued)*

Inspection	Purpose	Expected Result
Guide thimble oxide	Verify guide thimble corrosion margins.	Same as UO_2 with M5 guide thimbles
Fuel assembly bow and distortion	Address incomplete RCCA insertion issue. Verify fuel assembly growth models.	Same as UO_2 with M5 clad fuel rods and M5 guide thimbles

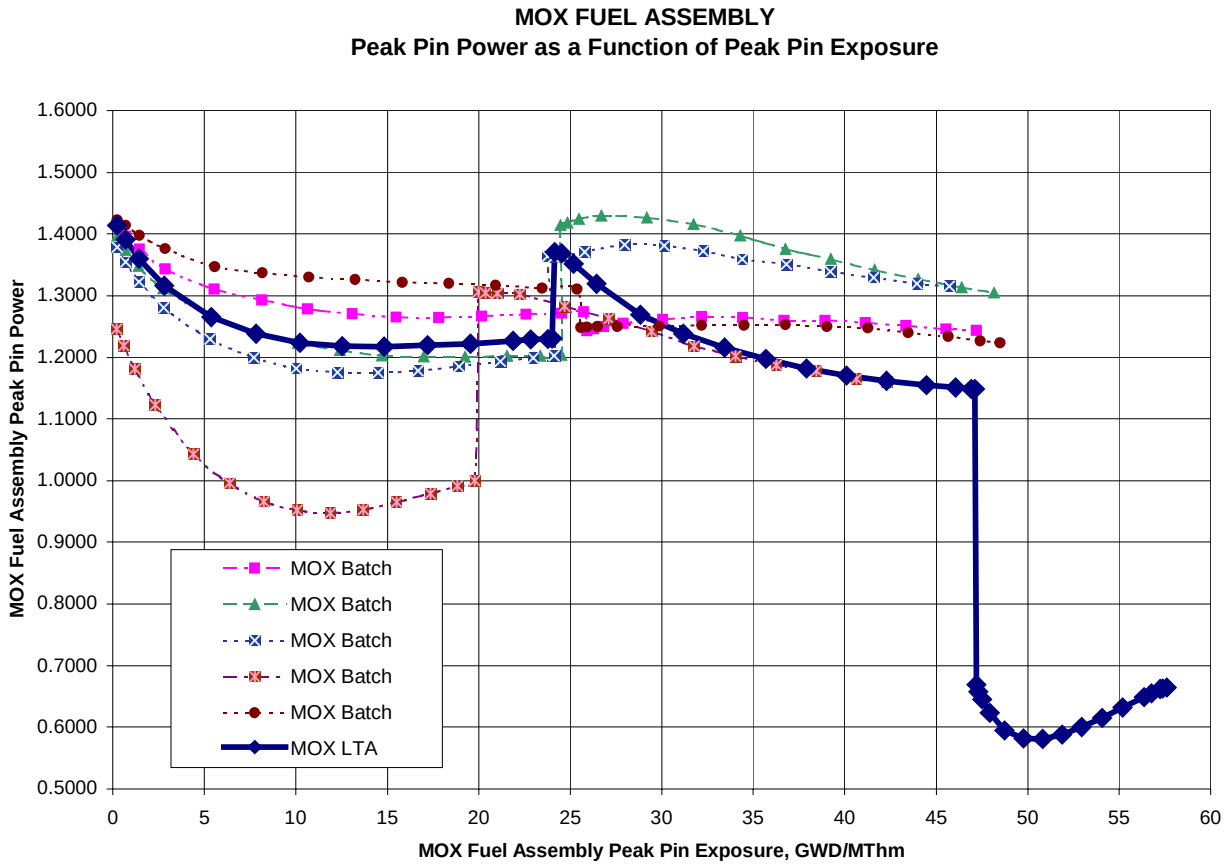

Figure 8.1 MOX Fuel Consolidated Rod Power Histories

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