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Fuel Qualification Issues and Strategies for Reactor-Based Surplus Plutonium Disposition

B. S. Cowell
G. L. Copeland
D. L. Moses

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FUEL QUALIFICATION ISSUES AND STRATEGIES FOR REACTOR-BASED SURPLUS PLUTONIUM DISPOSITION

B. S. Cowell
G. L. Copeland
D. L. Moses

August 1997

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OAK RIDGE NATIONAL LABORATORY
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LIST OF ACRONYMS

AA	Asea-Atom
ABB–CE	Asea Brown Boveri–Combustion Engineering, Inc.
ABWR	advanced boiling-water reactor
ADU	ammonium di-uranate
AEA	Atomic Energy Act of 1954
AEC	U.S. Atomic Energy Commission
AECB	Atomic Energy Control Board
AECL	Atomic Energy of Canada, Limited
ARIANE	Actinides Research in a Nuclear Element (Belgonucléaire program)
ANL	Argonne National Laboratory
ARIES	Advanced Recovery and Integrated Extraction System
ATR	Advanced Thermal Reactor
AUC	ammonium uranyl carbonate
B&W	Babcock & Wilcox Co.
BN	Belgonucléaire S.A.
BNFL	British Nuclear Fuels, Limited
BRP	Big Rock Point
BWR	boiling-water reactor
CANDU	Canadian deuterium-uranium reactor
CEC	Commission of the European Communities
CFR	*Code of Federal Regulations*
CHF	critical heat flux
CLWR	commercial light-water reactor
CRL	Chalk River Laboratories
DOE	Department of Energy
DOE-MD	DOE Office of Fissile Materials Disposition
EBWR	Experimental Boiling-Water Reactor
EdF	Electricité de France
EEI	Edison Electric Institute
EPRI	Electric Power Research Institute
ETR	Engineering Test Reactor
FCF	Framatome Cogema Fuels
FFTF	Fast Flux Test Facility
FMDP	Fissile Materials Disposition Program
FMEF	Fuel and Material Examination Facility
Fra	Framatome
FSAR	final safety analysis report
GE	General Electric
GESMO	*Generic Environmental Statement on the Use of Mixed Oxide in Light Water Reactors*
HYDOX	hydride oxidation
IAEA	International Atomic Energy Agency
IFBA	Integral Fuel Burnable Absorber
LA	lead assembly
LANL	Los Alamos National Laboratory
LEU	low-enriched uranium
LLNL	Lawrence Livermore National Laboratory

LMR	liquid-metal fast breeder reactor
LTA	lead-test assembly
LUA	lead-use assembly
LWR	light-water reactor
MELOX	French MOX fabrication plant
MHI	Mitsubishi Heavy Industries
MIMAS	Micronized Master Mix (MOX fuel fabrication process)
MIR	Russian test reactor
MOX	mixed uranium-plutonium oxide
MT	metric ton
MTR	Materials Test Reactor
MTRIS	Minimal Technical Risk Implementation Strategy
NDE	nondestructive examination
NPD	Canadian power demonstration reactor
NRC	Nuclear Regulatory Commission
NRU	National Research Universal (Canadian test reactor)
NRX	National Research Experimental (Canadian test reactor)
NSSS	nuclear steam supply system
O/M	oxygen-to-metal
OH	Ontario Hydro
ORNL	Oak Ridge National Laboratory
PAAA	Price-Anderson Amendments Act of 1988
PDAC	pit disassembly and conversion
PDS	Plutonium Disposition Study
PHWR	pressurized-heavy-water reactor
PIE	postirradiation examination
PNC	Power Reactor and Nuclear Fuel Development Corporation
PNNL	Pacific Northwest National Laboratory
PRDP	Plutonium Recycle Demonstration Program
PRTR	Plutonium Recycle Test Reactor
PUP	Plutonium Utilization Program
PWR	pressurized-water reactor
QA	quality assurance
QC	quality control
R&D	research and development
RIS	Recommended Implementation Strategy
RBMK	Russian graphite-moderated, water-cooled reactor
RCC	rod control cluster
RG	reactor-grade
ROD	Record of Decision
S&S	safeguards and security
SBR	Short Binderless Route
SGR	self-generated recycle
SMP	Sallafield MOX Plant (BNFL)
SNM	special nuclear material
TIGR	Thermally Induced Gallium Removal
TRU	transuranic
VIP	Venus International Program (Belgonucléaire program)
VVER	Russian PWR

W	Westinghouse
WABA	Wet Annular Burnable Absorber
WG	weapons grade
ZED	Zero Energy Demonstration (Canadian critical facility)

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EXECUTIVE SUMMARY

The Department of Energy (DOE) has proposed irradiation of mixed-oxide (MOX) fuel in existing commercial reactors as a disposition method for surplus plutonium from the weapons program. The burning of MOX fuel in reactors is supported by an extensive technology base; however, the infrastructure required to implement reactor-based plutonium disposition does not exist domestically. This report identifies and examines the actions required to qualify and license weapons-grade (WG) plutonium-based MOX fuels for use in domestic commercial light-water reactors (LWRs).

Reactor-based plutonium disposition is herein described in terms of six processes: (1) pit disassembly, (2) plutonium metal-to-oxide conversion, (3) purification of oxide, (4) plutonia powder conditioning, (5) MOX fuel fabrication, and (6) MOX fuel irradiation. New technologies are being developed by DOE laboratories for the first four processes. Plutonium produced from these new technologies will be used in conjunction with an existing (or slightly modified) commercial MOX fuel fabrication process. However, the powder morphology and the impurity concentrations of this plutonia are outside the existing commercial MOX fuel experience base, and the acceptability of this plutonia must be demonstrated.

FUEL QUALIFICATION STRATEGIES

This report presents two implementation strategies: a Minimal Technical Risk Implementation Strategy (MTRIS) and a Recommended Implementation Strategy (RIS).

MTRIS removes the uncertainties associated with PuO_2 powder morphology and the impurity level issues to ensure that the feed material meets the existing feed specifications for commercial MOX fuel fabrication plants (other than isotopic composition). This is accomplished by aqueous purification of the plutonia (dissolution in nitric acid followed by solvent extraction or ion-exchange purification). Precipitation of the plutonium as the oxalate, followed by calcining, produces feed material that is physically identical to that normally used in the commercial process. Fabrication of lead assemblies (LAs) and mission fuel would be performed in one of the existing commercial fabrication plants in Europe. Fabrication of fuel for the entire mission in one of the existing European facilities might require added capacity and was therefore dismissed as an option as part of the DOE's Record of Decision.

The RIS replaces aqueous plutonium processing with the new dry processing technologies currently being developed. LAs would be fabricated in Europe, but mission fuel would be fabricated in a new dedicated domestic MOX fabrication facility. Plutonium of high purity would be selected for the initial fuel to afford more time for resolution of any impurity issues resulting from the use of dry plutonium processing technology. Aqueous purification would be utilized as required to obtain sufficient quantities of plutonium for the initial fuel. Fuel impurities outside the experience base would be introduced in a sequential fashion after mission startup via concurrent LA programs.

Both MTRIS and RIS would introduce MOX fuel into the reactors gradually through the use of small numbers of "UO_2 look-alike" assemblies, which can neutronically replace traditional low-enriched uranium (LEU) assemblies. The total number of MOX assemblies in the reactor would increase to about 30%, which is consistent with the European experience base. Higher plutonium throughput cores would be implemented later in the mission through concurrent LA programs, if desired.

GENERIC ISSUES

The following six questions are representative of the issues addressed in this report.

1. What circumstances could dictate the need for aqueous processing?

The dry plutonium processing methods currently under development have not been demonstrated to produce feed that is acceptable for the various commercial MOX fabrication processes, which were developed with oxalate-derived plutonia. It is also unclear whether the dry plutonium processing methods will yield plutonium oxide feed material with impurity concentrations entirely within the existing MOX fuel database. To this point, most of the DOE's Fissile Material Disposition Program emphasis on impurity removal research has concentrated on gallium. The pits that will be retired and used for the disposition campaign may contain varying concentrations of impurities; some of these may not be as readily removed as gallium by dry processing techniques. If this is the case, the technically preferred approach is simply to remove impurities via aqueous purification. However, the reintroduction of oxalate-derived plutonium into fuel fabrication processes that have been tailored for the use of dry-process-derived plutonium might generate the need for additional fuel fabrication and fuel irradiation testing while the disposition campaign is under way. Aqueous processing may also be desirable because of issues other than fuel performance. These issues are discussed in Chap. 5 (Sects. 5.2 and 5.4).

2. How much test fuel will be required and when is it needed?

If early mission fuel and LAs are fabricated in existing European facilities to specifications equivalent to those in commercial use, only lead use assemblies (LUAs) (four to eight) may be required one to two cycles ahead of the first mission fuel. If test fuel is produced in a domestic lead test assembly (LTA) fabrication facility, LTAs will likely be required 5 years ahead of the mission fuel to allow time for postirradiation evaluation and Nuclear Regulatory Commission (NRC) review and approval. Estimates of test fuel amounts in this case range from 128 kg MOX (about 6 kg plutonium) for 4 LTAs, containing 16 rods each, up to 8 full LTAs, containing 3200 kg MOX for pressurized-water reactors (PWRs) or 480 kg MOX for boiling-water reactors (BWRs). This range should cover either MTRIS or RIS because the final fuel specifications are within the experience base and only the processing will be different. These LTAs will need to be inserted in the reactor in the year 2003 to support loading the mission fuel from the MOX plant completed at the end of 2006.

3. What domestic options exist for obtaining test fuels?

The preferred option is to obtain the test fuel from the European fabrication plant that is the model for the domestic plant. If other considerations preclude shipment of plutonium to Europe, then a domestic source in advance of the MOX facility is required to avoid a major schedule penalty. A prototypic LTA fabrication facility built in an existing DOE facility is probably the most desirable method of achieving this capability. Facility access by fabricator, fuel vendor, and utility personnel will be necessary to ensure production of prototypic fuel. The fuel fabricator should be an integral partner in the design, construction, and operation of the facility so that the credibility of the fuel is unquestioned. This is discussed in Chap. 5 (Sect. 5.5.4).

4. What new fuel performance data, code benchmarking, and validation activities will be required?

Licensing and use of commercial MOX fuel in Europe has been accompanied and supported by the development of an extensive experimental database that includes the validation and benchmarking of a group of neutronics, fuel performance, fuel-cycle management, thermal-hydraulic, and accident analysis methods and computer codes. These data and codes are proprietary and thus have not been independently reviewed for their applicability in the United States for licensing and operation of surplus WG plutonium MOX fuel. Ultimately, the applicability and acceptability of these data and codes in the U.S. licensing process will be known only after the selected fuel vendor and the utility evaluate the data and present them to NRC for acceptance.

It is quite likely that the European database and methods will, with minor modifications, be suitable to support U.S. licensing of the surplus WG plutonium MOX fuel. However, it appears that few, if any, MOX fuel postirradiation melting (fission product release) tests have been performed. Though not strictly required by NRC because it was not required as part of the original licensing process, a limited number of these severe fuel damage tests may be desirable to confirm that the source terms resulting from MOX-fueled reactors are not substantially different than those from LEU-fueled reactors. This is not viewed (from the programmatic standpoint) to be a significant issue in scheduling and cost.

5. Do either BWRs or PWRs have an advantage over the other type of reactor?

Although the largest number of MOX rods irradiated to date are certainly of the PWR type, both types of reactors appear to be equally capable of being licensed to burn MOX fuel at approximately the same disposition rates. Vendors of both reactor types had made substantial progress toward receiving NRC permission for routine MOX usage when efforts were discontinued in the 1970s because of nonproliferation policy concerns. Both reactor types are currently licensed to burn MOX fuel in Europe. The technology base for the various reactor types is discussed in Sect. 5.6.1.

6. Will a license amendment be required before loading a lead assembly (LTA or LUA) in the reactor?

It appears that a license amendment **will** be required to change the fuel description in the Technical Specifications. Furthermore, an exception may be needed for the requirements listed in Title 10 *Code of Federal Regulations* (CFR) Part 50.46(a)(1)(i) because of the fuel description included there. Once these actions are accomplished, the actual loading of the LA could probably be accomplished through a reload according to 10 CFR 50.59. Precedence for this approach is provided by the MOX LTA campaigns of the 1970s. The key to the success of this approach may be to obtain early NRC approval of the fuel performance codes, calculation methods, and the database on which they are based. This topic is discussed in Chap. 4 (Sect. 4.3) and Appendix C.

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1. INTRODUCTION

The United States has declared 38.2 metric tons (MT) of weapons-grade (WG) plutonium as surplus to national security needs.[1] It is anticipated that additional inventories of plutonium will eventually be declared surplus, bringing the total quantity to ~50 MT.[2] The U.S. Department of Energy (DOE) Fissile Materials Disposition Program (FMDP) has been directed to implement a program to provide for disposition of this surplus weapons-usable plutonium. In a Record of Decision (ROD) issued on January 14, 1997, DOE announced that a dual-path disposition strategy would be pursued.[3] Both immobilization in glass or ceramic material and burning as mixed-oxide (MOX) fuel in domestic commercial light-water reactors (CLWRs) would result in final disposal in a geologic repository. This disposition strategy is in accordance with the "Preferred Alternative" in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement.*[4]

This report discusses the implementation options and strategies for the reactor-based disposition path. Reactor-based disposition may be subdivided into six processes: (1) pit disassembly, (2) plutonium metal-to-oxide conversion, (3) plutonium purification, (4) plutonia powder conditioning, (5) MOX fuel fabrication, and (6) MOX fuel irradiation. While proven technologies exist for all six processes, domestic capabilities do not include all six. Furthermore, advanced technologies for the first four processes are under development as part of this program. A more complete description of the reactor-based disposition option may be found in the four volumes of the *Fissile Materials Disposition Program Reactor Alternative Summary Report.*[5–8]

This document includes information related to the qualification of WG MOX fuel for use in domestic CLWRs. It should not be confused, however, with a fuel qualification plan in the commercial sense. Because most of the databases supporting both core physics and fuel performance codes and methods are proprietary and unavailable to FMDP, it is impossible to prepare a commercial fuel qualification plan that identifies the specific tests and experiments needed to complete the databases. This document contains, therefore, both more and less than a commercial fuel qualification plan. It includes brief technology descriptions that provide a context for the implementation strategies that form the main part of this document.

The utility(ies) eventually selected for the disposition program will, of course, have the ultimate responsibility of ensuring the acceptability of the MOX fuel for their reactors and of convincing the Nuclear Regulatory Commission (NRC) of this acceptability. The primary purpose of this report is to discuss some of the issues in advance of the selection of the consortium and the reactor type(s) so that interested parties may have access to the information. When utilizing this document, the reader must keep in mind the following caveat stated most eloquently by Westinghouse (W):

> During selection of a MOX fuel design for the PDR, it became obvious that the development of a fuel, including irradiation testing, is tied to the selection of a fuel fabrication process defined in process and product specifications. Data generated with fuel produced in a non-prototypical process may or may not be admissible as data for licensing. In addition, fuel performance codes are calibrated against fuel with specific physical attributes such as pore distribution, densification characteristics, and homogeneity, and with specific manufacturing processes such as sintering schedules and temperatures. Therefore, it is not useful to plan fuel tests unless a fuel specification has been selected as well as a fabrication process.[9]

Critique of this report has been sought from the DOE Office of Fissile Materials Disposition (DOE-MD), other DOE laboratories, and commercial entities. The authors have made extensive efforts to incorporate the comments received; however, the opinions expressed in this report are ultimately those of the authors and do not necessarily represent official positions of DOE.

The ROD reserved the option for using existing Canadian pressurized heavy-water reactors, known as Canadian deuterium-uranium (CANDU) reactors, if a multilateral agreement is reached among Russia, the United States, and Canada for joint disposition of a portion of the Russian and U.S. surplus plutonium. This report in no way is meant to bias opinion for or against this option. Many discussions of issues and options in the remainder of this document do not give equal weight to the domestic CLWRs and the CANDUs; however, in many cases, this lack of parity is due to a lack of applicability to the CANDU options. In other cases, issues and options that are equally applicable to both domestic CLWRs and CANDUs are not called out specifically as applying to the CANDUs. Finally, the ROD did not preclude the use of surplus plutonium as fuel for the Fast Flux Test Facility (FFTF) if FFTF is utilized for tritium production. However, this report does not address qualification of FFTF fuel.

The remainder of this document is divided into six chapters. Chapter 2 contains a brief summary of the current status of MOX fuel technology, both domestic and foreign. Chapter 3 contains a description of the fuel qualification requirements for WG MOX fuel. Qualification is described in terms of the stakeholders and the requirements for the fabrication facility, the fabrication process, process materials, and the finished product. Chapter 4 describes the licensing methodology to be applied to the pit disassembly and conversion (PDAC) facility, the MOX fuel fabrication facility, and the mission reactors. The technology base necessary to implement MOX fuel utilization is described in Chap. 5 in terms of the six processes listed above. Chapter 6 contains implementation strategies. A minimum technical risk strategy is first described. A recommended strategy, taking into consideration nontechnical constraints, is then described. Finally, a set of contingency strategies is provided. Conclusions are summarized in Chap. 7. Three appendixes are also included. Appendix A contains a brief history of MOX fuel use in the United States and abroad. Appendix B contains a more detailed discussion of the fuel qualification process as it applies to feed materials and the fabrication process. Finally, Appendix C summarizes the CLWR fuel reload licensing process, including a description of the license amendment process and the simpler 50.59 process.

2. CURRENT STATUS OF MOX FUEL TECHNOLOGY

As described in Appendix A, a development or pilot-scale MOX fuel infrastructure was developed domestically by the early 1970s. This infrastructure included facilities and expertise for plutonia supply, MOX fuel fabrication, in-pile and out-of-pile testing, and postirradiation examination (PIE). Most of the low-enriched uranium (LEU) fuel fabricators of the time were engaged in various MOX fuel research and development (R&D) projects, and many had their own pilot fabrication lines. Commercialization of the technology was limited by government approval of reprocessing and MOX fuel fabrication facilities, which was delayed pending completion of the *Generic Environmental Statement on the Use of Mixed Oxide in Light Water Reactors* (GESMO) proceedings. However, changes in the U.S. nuclear nonproliferation policy halted all MOX development. Since that policy was announced in 1977, most of the infrastructure and expertise has disappeared. All of the pilot fabrication facilities have either been decontaminated and converted to other purposes or decommissioned to greenfield. Many of the researchers have retired. Furthermore, uranium fuel technology has advanced significantly since the 1970s, and these advancements need to be incorporated into domestic MOX. Other than completion of ongoing studies, domestic MOX fuel work since the mid-1970s has been limited to fuel design conducted in support of foreign commercial customers.

At the time of the U.S. policy change, the domestic MOX fuel designers and fabricators operated at the forefront of technology. According to arguments made by the fuel vendors at the time, the lead test assemblies (LTAs) and limited reload experience, both domestic and foreign, provided a sufficient database for commercial implementation. The European designers and fabricators have continued to update their technology since that time. New facilities for fuel reprocessing have been brought on-line to provide the plutonia feedstock. Existing European MOX fuel fabrication facilities have been updated, and new ones have been constructed. MOX fuel is currently licensed for use in French, German, Belgian, and Swiss reactors.

Almost all of the described research, development, testing, and commercial utilization incorporated reactor-grade (RG) plutonium in the MOX fuel. RG plutonium differs from the surplus weapons plutonium in its isotopic makeup, its powder morphology, and its impurities. Current RG plutonium is typically derived from high-burnup, LEU fuel and thus contains significant quantities of the higher plutonium isotopes (^{240}Pu, ^{241}Pu, and ^{242}Pu). These higher isotopes change the neutronic behavior of the plutonium. Much of the early domestic development and testing utilized 90% fissile plutonium supplied by the U.S. Atomic Energy Commission (AEC). This 90% fissile plutonium more closely resembles WG plutonium than current RG plutonium, but nonetheless contains over 50% more ^{240}Pu than does the surplus WG plutonium.

Most of the MOX fuel that has been tested used plutonium separated through the PUREX process and precipitated as oxalate from nitrate solution. The calcined oxide product has a characteristic morphology unique to the process. The aqueous processes commonly used have been demonstrated to produce sinterable powder. The hydride process that is the baseline for the surplus plutonium disposition mission is not currently supported by an irradiation experience base. Both laboratory and in-reactor testing may be required to demonstrate that the powder characteristics are suitable. An additional effect of the use of the hydride process for metal-to-oxide conversion is lack of intrinsic impurity removal. Although the surplus WG plutonium metal was originally of high purity at the time of separation, subsequent treatment, alloying, and processing introduced impurities. One of the most discussed of these is gallium, which was added as an alloying agent at concentrations up to 1%. The effects of gallium and other impurities or additives on the conversion facility, the MOX fuel facility, and fuel rod performance must be determined to be acceptable.

Because a MOX fuel infrastructure does not currently exist in the United States, some new facilities will be required. Necessary capabilities include (1) pit disassembly, (2) plutonium metal-to-oxide conversion, (3) plutonium purification, (4) plutonia powder conditioning, (5) MOX fuel fabrication, and (6) reactors. A new PDAC facility will be required to implement the dry Advanced Recovery and Integrated Extraction System (ARIES) process for pit disassembly and possibly metal-to-oxide conversion. New or modified facilities may be utilized for plutonium purification, either in a stand-alone facility or as an integral part of the PDAC facility. Following plutonium purification, the pure plutonia product may require conditioning to make it suitable for fuel fabrication. Conditioning processes will also likely be included as an integral part of the PDAC facility. None of the domestic MOX fuel fabrication pilot lines remain operational. A new MOX fuel fabrication facility is planned to support the FMDP mission. It is expected that the facility design and processes will mimic those of one of the existing European MOX fuel fabrication facilities. All of the reactor disposition options currently under consideration use existing commercial reactors. All necessary transportation of plutonium and fresh MOX fuel will utilize the existing transportation systems.

Because such a long period has elapsed since the curtailment of domestic MOX fuel utilization, there is uncertainty concerning the applicability of the previous domestic experience base in terms of its age, its fidelity, and subsequent improvements to fuel technology. By obtaining a modern commercial European fuel fabrication process and its associated fuel performance database, these uncertainties could be avoided. The European experience base with MOX fuel utilization, measured in percentage of total MOX rods irradiated, comprises well over two-thirds of the worldwide experience. It is believed that reliance on this database may smooth the introduction of MOX fuel into domestic reactors by providing data necessary for qualification of the fuel to the various stakeholders. However, the degree to which utilities (and their fuel suppliers) will rely on foreign experience has not yet been determined.

3. FUEL QUALIFICATION REQUIREMENTS

This section describes the qualification requirements for all the involved parties or stakeholders for the insertion and operation of MOX fuel into a currently operating U.S. reactor under the DOE-MD sponsorship. As is evident from the following discussion, overlap exists in the stakeholder requirements in some cases. Following the stakeholder discussion, the methodologies for qualification of feed materials, facility and processes, and the fuel design are discussed.

3.1 STAKEHOLDER REQUIREMENTS

The term "fuel qualification" has a variety of meanings depending on context. Important stakeholders include the utility, the regulator, the fuel designer/vendor, the fabricator, and in the case of surplus WG plutonium MOX, the DOE. To each of these stakeholders, fuel qualification has a different meaning. Each organization has a set of requirements that must be met by the qualification process. The intent of the surplus WG plutonium MOX program is to maintain the traditional relationships among the parties as much as possible.

The terminology utilized throughout the remainder of this report is consistent with the authors' understanding of current practice in the European MOX market. Some important differences are apparent in comparison with the domestic urania market. The utility contracts with a fuel supplier, or fuel designer, who provides (as a minimum) a mechanical design for the assembly. The fuel designer can also provide additional services such as neutronic design for both individual assemblies and reloads, if the utility does not perform these activities in-house. The fabricator, unless otherwise specified, is the organization that actually makes the MOX fuel pellets, rods, and assemblies. The fabricator can be, but is not necessarily, the same as the fuel designer. This is in fact the case for most domestic LEU fabrication. The fuel designer often supplies the assembly hardware to be utilized by the fabricator in completing assemblies. The fabricator may produce urania powder on-site. However, more commonly among MOX fabricators, the urania is purchased from a separate supplier.

3.1.1 Utility

The utility, as the owner of the reactor and holder of the operating license, is responsible for the safe, economical operation of the reactor. Nuclear fuel must therefore be safe, reliable, economical, and licensable. Some of the qualification requirements that are imposed by the utility therefore include these assurances.

- Utilization of the fuel will not adversely impact the safety of the plant's operation.
- The fuel will operate to some negotiated burnup under specified conditions without failure.
- The fuel will contain the specified energy content.
- The fuel can operate within the required envelope of conditions such that operational flexibility is not reduced.
- The fuel is readily licensable.

For surplus plutonium MOX, modified and additional requirements may be anticipated. The utility may desire that surplus plutonium MOX fuel

- operate to the same burnup as current urania fuel;
- be as flexible as current urania fuel in terms of allowable power ramp rates, thermal margins, and interactions with co-resident urania fuel;
- be as reliable as current urania fuel (zero defects) with similar warranties; and
- accommodate any currently planned cycle extensions.

In many cases, the qualification requirements of the utility are not clearly elucidated. Some may be listed in the purchase agreements between the utility and the fuel supplier. Additional insight into the utility requirements may result from the procurement acquisition strategy workshop.

3.1.2 Regulator

The regulatory role in fuel qualification is based on the same premise as that for initial licensing. The regulator must assure the protection of the public, the plant workers, and the environment. The U.S. Nuclear Regulatory Commission (NRC) performs this duty for domestic LWRs, and the Canadian Atomic Energy Control Board (AECB) does it for CANDU reactors. The regulatory role includes implementation of the necessary regulations, review of the licensee submittals, and assurance of adherence to all applicable laws and regulations.

Many of the overall qualification requirements are imposed because of regulatory considerations, but most of these are only reported to the regulator indirectly through the utility. For example, the fuel fabrication facility may be licensed by the regulator for utilization of special nuclear material (SNM). However, the quality assurance (QA) requirements that are imposed upon the facility are done so because the utility, as holder of the reactor license, is required to show that all suppliers have implemented an acceptable QA program. The NRC nevertheless reviews, approves, and audits fuel vendor QA programs to ensure compliance with the requirements.

Simply speaking, the regulatory requirements are a subset of the utility requirements. However, NRC does not have a standard list of requirements that a new fuel must satisfy in order to be approved for use. Instead, an applicant must prove his case to the NRC's satisfaction. Some indication of the level of detail required could be obtained from previous licensing topical reports, but these reports are proprietary. Therefore, only high-level guidance is available. The utility applicant must demonstrate to the regulator that surplus WG plutonium MOX fuel

- utilization will not decrease safety margins to an unacceptable extent, and
- performance can be predicted with acceptable uncertainty.

3.1.3 Fuel Designer/Vendor

The utility relies on a fuel designer/vendor for its fuel supply. While some of the larger utilities perform their own reload analyses and may even specify enrichments, enrichment zoning, and burnable absorber types and concentrations, all utilities rely on the fuel designer for mechanical design of the assembly. The fuel designer develops a fuel design envelope for an assembly within which the reload analysts can manipulate the assembly characteristics. The fuel designer usually assists the utility in obtaining regulatory approval of the fuel design, if this has not already been performed by the fuel designer generically. In the United States this is done through preparation of one or more licensing topical reports. The fuel designer is also responsible for obtaining regulatory approval for the design and analysis codes and methods utilized. The fuel designer/vendor qualification requirements can be summarized as follows. The surplus WG plutonium MOX fuel qualification process must demonstrate that

- the fuel meets the utility's requirements,
- the fuel performance is reliable such that a standard fuel warranty can be offered,
- the codes and methods are applicable to the fuel to regulatory satisfaction, and
- the fuel can be manufactured by available suppliers.

3.1.4 Fuel Fabricator

The fuel designer in many cases subcontracts fuel fabrication to one or more suppliers. These fuel fabricators do not warrant the performance of their fuel per se, but they guarantee that the supplied product meets the specifications accompanying the purchase agreements. Thus, the requirements of the fuel fabricator are not as extensive as those of the fuel designer or the utility. The primary requirement is that the fuel design be compatible with the existing fabrication process and facility (unless compensation is available to cover modifications). Because the surplus WG plutonium MOX fuel will be fabricated in a new dedicated domestic facility, a unique arrangement is likely to exist among the utility, fuel designer, and fabricator. However, in any case, one of the primary requirements of the fuel fabricator is to have a reliable and timely supply of uniform feed materials.

3.1.5 DOE

In the unique case of surplus WG plutonium MOX fuel, another stakeholder in the fuel qualification process is DOE. DOE requirements include maximizing the

- value of both the fuel design and qualification,
- timeliness of qualification activities, and
- safety of the fuel.

The requirements also include minimizing the

- licensing requirements of the fuel, and
- technical risk associated with the fuel design.

3.2 FUEL QUALIFICATION METHODOLOGY

The fuel qualification philosophy adopted for surplus WG plutonium MOX is adapted from commercial MOX programs. The philosophy guiding qualification of European RG MOX fuel is that the addition of small quantities of plutonium does not significantly change the properties and behavior of RG MOX fuel relative to urania fuel. The vast majority of RG MOX fuel is urania, the behavior of which does not change. Therefore, the purpose of qualification is not to reproduce the entire urania fuel database for RG MOX fuel but simply to demonstrate the applicability of this large and growing database to the MOX fuel as well.

This philosophy has been adopted and modified for the MD mission. The differences between surplus WG plutonium and commercial RG plutonium are not great. Furthermore, this plutonium makes up but a small fraction of the material in MOX fuel—the remainder consisting of UO_2. Surplus WG plutonium MOX fuel should, therefore, behave similarly to both commercial RG MOX and urania fuels. Some differences, notably neutronic behavior, exist, but these differences are well-understood. The purpose of fuel qualification will be to prove the applicability of the existing urania and RG MOX databases to the specific surplus WG plutonium MOX fuel. Most of the existing RG MOX and urania databases are expected to be applicable. Additional specific qualification data developed through surplus WG plutonium MOX fuel testing (including rod tests and/or LTAs) may be required to bridge any gaps in the data.

Qualification of a nuclear fuel entails three separate activities: qualification of the feed materials, qualification of the fabrication process and facility, and qualification of the final product. The feed materials include urania powder, plutonia powder, binder, lubricant, pore former, cladding, and fuel assembly hardware. Each of these must be shown to meet its respective specifications with a high degree of confidence. The fabrication process and facility must also be qualified or shown to be capable of producing both in-process and final products that meet the

specifications. Finally, the finished fuel product as defined in the specifications must be qualified to the fuel vendor, the utility, and the regulator.

The first two facets of qualification are not always labeled as part of fuel qualification. They entail proving that various products meet a specification provided by the customer. The producer of feed powder or final fuel is not concerned with the validity of the specification, but only with the ability of his process to meet the supplied specification. It is in the final step, qualification of the fuel product, in which the bulk of the qualification activities take place. The term "fuel qualification" is normally reserved for these activities. Qualification of the product may be defined as proving to the designer, the end user, and the regulator that the product as defined in the specifications is reliable and safe to operate. Qualification activities such as out-of-pile tests, irradiation tests, PIE, and other tests are used to verify that the specifications define a satisfactory product.

3.2.1 Qualification of Feed Materials

All of the feed materials that go into fabrication of the fuel product must be qualified. These feed materials include urania powder, plutonia powder, binder, lubricant, pore former, cladding, and fuel assembly hardware. Each of these must be shown to meet its respective specifications with a high degree of confidence. As defined for the purposes of this document, qualification of each of these materials entails only those actions necessary to ensure that the product meets the supplied specification repeatedly. It makes no attempt to test the validity of the specification itself. Validation of the feed and product specifications is done as part of qualification of the product.

Qualification of plutonia powder is described as an example. MOX fuel fabricators currently transfer a feed powder specification to their suppliers. In most cases, this specification was developed by the MOX fuel fabricator, in conjunction with the supplier(s) through iterative negotiation. The specification must define a product that satisfies the needs of the customer (fuel fabricator), but also a product that the powder fabricator can supply reliably.

For the MD mission, the chosen MOX fuel designer and/or fabricator will supply the plutonium processor with the existing commercial plutonia powder specification. The plutonium processor will have to develop processing capability to meet the existing specification and then implement a sampling plan that provides statistical proof of the continued compliance with the powder specification. Often, the sampling rate during "qualification" is much higher than that actually used during routine fabrication. The sampling rate may be reduced as confidence in the consistency of the product develops. A significant effort may be required to develop the processing capability to meet the specification, but this is a developmental activity and not part of qualification.

For important feed materials like the powder, the customer may include details about the production process itself in the specification. These may include not only acceptable processes but also acceptable operating parameters. For less important materials, a much less involved specification may suffice.

An additional requirement is that the PDAC facility operations and sampling must be performed under an approved QA program to maintain compliance with the regulations promulgated in Title 10, *Code of Federal Regulations*, Part 50 (10 CFR 50), Appendix B. While the utility, as the ultimate licensee, is theoretically responsible for quality control (QC) of its suppliers, utilities in general exert little influence on their suppliers' QA programs, other than through audits that ensure the vendor's work is performed in accordance with the approved program.

One notable feature of qualification of plutonia powder (and other feed materials) is that the plutonium processor is not responsible for the ultimate performance of the plutonia powder in MOX fuel, but only for meeting the powder specification. The validity of the specification is proven by the available database supported by the in-pile and out-of-pile testing performed as

part of qualification of the fuel product by the fuel vendor, not through qualification of the feed materials or fabrication process.

The facilities needed for qualification of feed materials are limited to the production facility itself and a supporting analytical capability. In most facilities supplying nuclear materials or components, the necessary analytical capabilities are available on-site. For any new facilities required for the MD mission, such analytical capability should be included in the scope of the new facility unless it already exists as part of the site's infrastructure.

3.2.2 Fabrication Process and Facility Qualification

Qualification of the fabrication process and facility is similar to qualification of the feed materials, with a change in roles. The customer for the feed materials is generally the supplier of the final fuel form to the fuel designer. The fuel fabricator is, therefore, only responsible for supplying a product that meets the fuel specification, but not responsible for the ultimate performance of fuel that meets the specification. One purpose of qualification of the fabrication process and facility is to set production process parameters. In combination with the accompanying sampling plan, the qualification process serves a production control and QA function to ensure compliance in a statistical sense with the requirements of the specification.

Qualification of the fabrication process, therefore, shares much similarity with qualification of the important feed materials. However, more detailed information is available on the subject of fabrication process and facility qualification. Appendix B contains a more detailed summary of this process, organized by the associated documentation hierarchy. While no generic description can encompass all qualification programs as implemented by the various fuel fabricators worldwide, the description in Appendix B was developed through interaction with a number of fuel fabricators and is an attempt to provide a generic description of the overall process.

3.2.3 Qualification of a New Fuel Product

The term "fuel qualification" is normally reserved for those activities that entail qualification of the finished fuel product. Qualification of the product may be defined as proving to the stakeholders (the utility, the regulator, the designer, and the fabricator) that the product as defined in the specifications is reliable and safe to operate. Qualification activities such as out-of-pile tests, irradiation tests, PIE, and other tests are used to verify that the specifications define a satisfactory product.

Figure 1 is a flow diagram of the traditional qualification process for a new fuel. As shown in Fig. 1, traditional fuel development and qualification begins with basic fabrication R&D. Based on this work, preliminary specifications are developed for feed materials, intermediate and final products, and individual process steps. Preliminary procedures are also developed. A QA/QC program is also established. Using this guidance, test fuel is fabricated and characterized through extensive inspection and destructive testing. These characterization tests may include measurement of basic physical properties if necessary and would include measurement of those important parameters specified in the pellet specification. If these tests indicate that the fuel meets the preliminary specifications, the fuel is inserted into a reactor for irradiation. Otherwise, additional fabrication research is performed.

Test fuel is irradiated in either a test reactor or incorporated into standard fuel bundles. W and General Electric (GE) tend to perform test rod irradiations in commercial reactors to the extent possible because of cost and technical advantages: W incorporates a limited number of test rods in a standard fuel assembly, and GE incorporates segmented test rods into specially modified fuel bundles. Combustion Engineering Inc. (now ABB–CE) uses test reactors for this type of

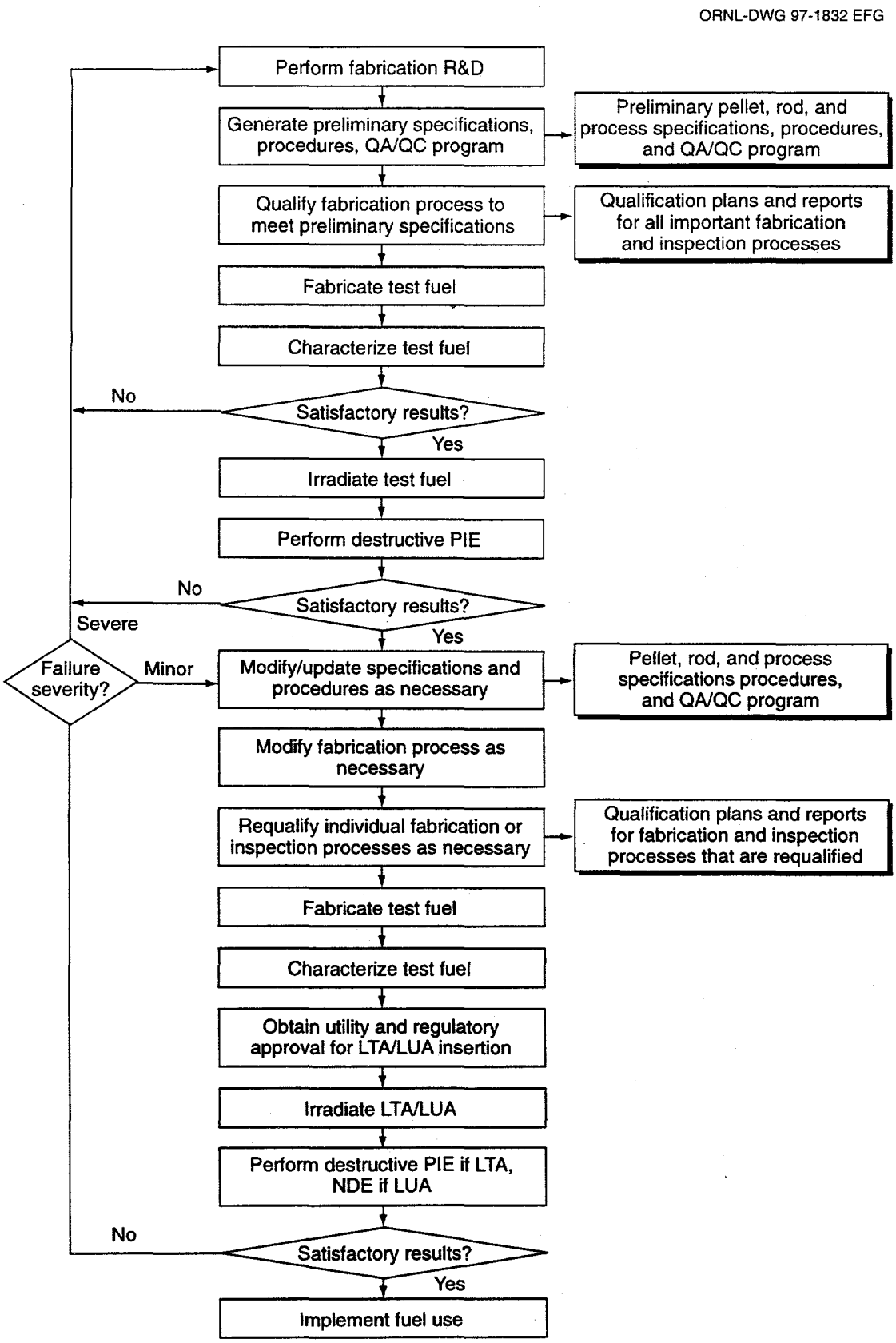

Fig. 1. Traditional fuel qualification process.

irradiation. The important aspect of this type of testing is that only a small quantity of fuel is tested.

Upon completion of the test fuel irradiation, destructive PIE is performed. In many cases, several variables are included in these preliminary irradiations. If none performs satisfactorily, additional fabrication R&D is performed. If one or more of these fuels behaves satisfactorily, the preliminary specifications and procedures are adopted or modified as necessary to reflect the information gained through irradiation. The fabrication process itself may also be modified to meet the updated specification.

All of the fabrication, inspection, and test operations that are modified are then requalified to fabricate fuel to the updated specification. An alternate way of qualifying the process is to produce a single "qualification" batch of fuel that is shown to meet the specifications. Under the more customary approach of qualifying individual processes, each of the important fabrication processes (i.e., powder blending, milling, granulation, pressing, sintering, grinding, cleaning, rod loading, welding, and bundle assembly) is shown to produce the specified intermediate or final product within specified tolerances. The sampling rate associated with the "qualification" fuel is generally much higher than that associated with normal production to identify any inconsistencies in the products. The sampling plan would be developed by the fuel fabricator and reviewed by either the utility or its representative.

For each of the processes to be qualified, destructive and nondestructive testing are normally performed on the product according to a qualification plan, with the results documented in a qualification report. If an intermediate or final product does not meet its specifications reliably, modifications to the process equipment or operational parameters are made until the product can be produced within tolerance with a stated statistical certainty.

Based on information in the qualification report, previous irradiation tests, out-of-pile tests, and historical data, approval is obtained from both the utility and the regulator for insertion of the LTA or lead use assembly (LUA) containing the test fuel.

The distinction between LTA and LUA testing is the scope of PIE planned. Destructive examination is planned for LTAs from their inception to obtain additional information for database enhancement. LUAs are usually subjected only to nondestructive examination (NDE). It is difficult to describe generic guidelines for determining when a LTA is utilized instead of a LUA. Much of the decision is based upon the vendor's and the utility's willingness to accept risk. The availability of irradiation data for a particular fuel assembly type might be sufficient to justify only having LUAs. Similarly, the distinction between the segmented or test rod irradiations performed in commercial reactors and LTA irradiations is not clear. If more than a few rods are included, it is more likely to be called a LTA than a test rod irradiation. The term lead assembly (LA), which refers to both LTAs and LUAs, is often used when details about the planned PIE are not available.

The LTA or LUA fuel is produced with a qualified process that meets the required fuel pellet and rod specifications. Because it is loaded in a commercial reactor, this fuel must meet the same quality requirements as standard fuel. However, it has historically been loaded into non-limiting positions within the reactor. The LTA/LUA fuel is usually examined visually at each refueling outage and sipped for leak detection. Fuel samples from LTAs, in the form of either entire rods or rod segments, may be removed and sent to hot cells for destructive examination after each cycle. Upon completion of the specified irradiation period, the LTA/LUA fuel is again subjected to NDE in the fuel pool. LTA fuel is then transferred to a hot-cell facility for destructive examination. LUA fuel may also be subjected to destructive hot-cell examination, but only to investigate any unforeseen difficulties with the fuel's performance or to increase the performance database.

The other significant difference between LTA and LUA irradiation is the timing of introduction of the corresponding fuel design. An LUA is utilized when the existing database is sufficient to obtain utility and regulatory approval for the fuel use. The LUAs are irradiated for confirmation purposes and to demonstrate to NRC the utility's ability to utilize the new fuel appropriately. Therefore, upon completion of one or two operating cycles, fuel based on an LUA design may be loaded into the reactor. Fuel based on an LTA, however, cannot be loaded until the required performance data are obtained from the destructive testing.

It is important to note that because of the historical placement of LAs in nonlimiting positions, successful operation of a LA does not necessarily guarantee successful operation of the corresponding reload assemblies that are operated in limiting positions. Recent difficulties with reload assemblies operating under more stringent conditions than the LAs upon which their approval is based may change the historical practice. Future LA programs may include assemblies operated in limiting positions.

3.2.4 Qualification of FMDP WG MOX Fuel

The WG MOX fuel to be utilized under FMDP is not a new fuel. It is an evolutionary modification of the well-characterized RG MOX fuel that is currently utilized in a number of countries. Furthermore, the RG MOX fuel is itself an evolution from the LEU fuel utilized in most of the CLWRs of the world. The qualification approach is, therefore, to enter the qualification process at an intermediate point based on the application of both RG MOX and LEU data. It is anticipated that the proprietary databases supporting both LEU and RG MOX fuel utilization will require only minimal additions to support WG MOX fuel qualification. Most of the information can be obtained through limited out-of-pile testing and a single LTA program.

Qualification requirements for the surplus WG plutonium MOX focus primarily on the differences between surplus WG plutonium MOX and RG MOX fuels that are related to the dry processing route and the gallium content (and perhaps other impurities) of the plutonium. These factors are being investigated in ongoing out-of-pile testing and in-reactor demonstration tests. Pending satisfactory resolution of these issues, it is felt that implementation of the mission in CLWRs can proceed initially through the introduction of LTAs as long as the proposed use is within the current experience base—that is, partial core loads containing no integral burnable absorber. If desired, introduction of full MOX cores, which may require burnable absorbers, can be implemented incrementally after the testing is completed.

3.2.5 Qualification Plan for CANDU WG MOX Fuel

The expected qualification requirements for the CANDU reactor option, when compared to the LWR options, are unique in two ways: (1) while having some test reactor experience with MOX fuel, they do not have the extensive experience base which exists for the LWRs, and (2) the licensing agency would be AECB. The Atomic Energy of Canada, Ltd., (AECL) 1996 optimization report outlined an extensive fuel qualification plan designed to provide assurance to the utility [Ontario Hydro (OH)] and AECB that the MOX fuel bundle can be successfully irradiated in the Bruce A reactors.[10] The plan described here is taken from that report.

The proposed implementation plan begins the mission with the standard 37-element bundle, which is physically identical to the current Bruce A bundle, then switches to an advanced fuel being used for the CANDU 6 reactors (a 43-element bundle designated CANFLEX). This allows a quicker start of the mission with two reactors using the 37-element bundle (plutonium throughput of 2.9 MT/year) and shortens the total mission by switching to the CANFLEX fuel in four reactors (plutonium throughput of 4.8 MT/year). Some factors considered in choosing this plan

include the lower risk qualification for the 37-element bundle, minimizing the required size of the MOX plant, and shortening the total mission time.

3.2.5.1 Fuel qualification plan for the 37-element CANDU WG MOX bundle

Because the physical features of this MOX bundle are identical to the bundle currently in use, this testing is concentrated on the fuel element performance and the nuclear physics aspects of the fuel. The tests consist of element performance tests in the National Research Universal (NRU) reactor at Chalk River, bundle tests in the NRU, nuclear physics measurements in the Zero Energy Demonstration (ZED-2) reactor, critical heat flux (CHF) tests in both water and Freon, and demonstration tests in one of the Bruce power reactors.

The element tests in NRU will be used to test all aspects of the fuel performance under the range of conditions in the Bruce reactors. The data generated will be used to modify the existing fuel performance codes that predict fuel performance. The Parallex test now under way is a part of these tests. When the design is finalized, additional elements will be tested in NRU under continuous high-power irradiation (beyond the Bruce conditions) and under power ramping conditions to verify the satisfactory performance of the fuel. Two to four demountable bundles and four to eight fixed bundles are envisioned for this testing phase.

Bundle testing includes prototype testing of one or two bundles in the NRU at continuous high-power and burnup conditions that envelope the expected Bruce conditions. In addition, the lattice pitch properties of the fuel will be measured in the ZED-2, including reactivity compared to the natural uranium fuel lattice, void reactivity, flux distributions across the bundle before and after coolant voiding, and coolant and fuel temperature reactivity effects. This testing will require 35 fixed bundles and 1 demountable bundle.

CHF testing with electrically heated element bundles is proposed in both water and Freon. The CHF characteristics of the MOX bundle need to be determined for the various bundle radial flux/energy profiles to support the safety and licensing process to obtain approval for loading the MOX bundles.

Finally, power reactor demonstration tests of 100 MOX bundles in a Bruce reactor have been suggested by OH. A small number of these would be selected for PIE in the hot cells. These demonstration tests serve the purpose of LAs in LWRs. Fabrication of 3700 elements for the 100 bundles in advance of the availability of the MOX fabrication plant would require an intermediate prototype facility or fabrication in an existing European facility as has been proposed for the LWR LAs.

The 37-element bundle qualification program is projected to take place from 1997 through 2002 at a total cost of $42.3M (1996 U.S. dollars).

3.2.5.2 Fuel qualification plan for the advanced 43-element CANDU WG MOX CANFLEX bundle

The fuel element testing for the 37-element qualification is expected to cover the conditions for the CANFLEX elements so that no additional testing of fuel element performance is needed. The CANFLEX qualification focuses on the bundle testing. In addition to the in-reactor (NRU and ZED-2) tests and the CHF tests that would be required, out-of-reactor tests are needed to address the design requirements for the heat transport system, fuel channel, and fuel handling systems. Finally, proof tests of CANFLEX bundles in a Bruce reactor are required to demonstrate that the CANFLEX bundles meet the design requirements established by OH.

Bundle testing for CANFLEX includes prototype testing of one or two bundles in the NRU at continuous high-power and burnup conditions that envelope the expected Bruce conditions. In addition, the lattice pitch properties of the fuel will be measured in the ZED-2, including

reactivity compared to the natural uranium fuel lattice, void reactivity, flux distributions across the bundle before and after coolant voiding, and coolant and fuel temperature reactivity effects. This testing will require 35 fixed bundles and 1 demountable bundle.

CHF testing with electrically heated element bundles is proposed in both water and Freon. The CHF characteristics of the MOX bundle need to be determined for the various bundle radial profiles to support the safety and licensing process to obtain approval for loading the MOX bundles.

Out-of-reactor testing will be required to demonstrate compliance with the design requirements for the 43-element bundle. Among these tests are bundle characterization measurements to ensure compatibility with the Bruce fuel channels, flow tests for endurance and pressure drop measurements, and fuel handling tests to ensure compatibility with the fuel loading and removal machines. Twelve natural uranium CANFLEX Bruce bundles will be needed for these tests.

Two demonstration tests have been suggested by OH for the 43-element CANFLEX design: (1) a 50-bundle natural uranium irradiation, and (2) a 100-bundle MOX irradiation. The natural uranium bundles may be made by one of the Canadian fuel manufacturers. Fabrication of the 100 MOX bundles would require either a prototype U.S. fabrication facility or fabrication in an existing European facility as above. The 43-element CANFLEX bundle qualification program is projected to take place from 1998 through 2005 at a total cost of $35.4M (1996 U.S. dollars).

4. LICENSING METHODOLOGY AND ISSUES

Depending on the outcome of a publicly announced, pending request by DOE for Congressional action to change the *Atomic Energy Act of 1954* (AEA), as amended, DOE is poised to enter a new period of transition from the self-regulation of DOE-owned nuclear facilities to external regulation of DOE nuclear facilities and activities by NRC. The PDAC facility is proposed to be a DOE-owned nuclear facility and is expected to be subject to the regulatory transition. DOE has indicated that it will seek NRC licensing for the MOX fuel fabrication facility, which will be owned by DOE and located at a DOE site but will be operated by a DOE contractor. The commercial reactors that are proposed to be used to convert surplus WG plutonium MOX fuel into spent fuel are owned by commercial power-generating companies and are already licensed and regulated by NRC.

4.1 PIT DISASSEMBLY AND CONVERSION FACILITY LICENSING

Under AEA Sects. 101 and 110, DOE nuclear facilities are currently exempt from licensing. Under Sect. 201 of the *Energy Reorganization Act of 1974*, NRC is responsible for all "licensing and related regulatory functions" authorized in AEA. DOE does not license its own nuclear facilities but implements procedures to authorize construction or operation based on the approval of the Facility Authorization Bases.

As indicated above, DOE is entering a period of transition with respect to oversight responsibility for assuring the nuclear safety of DOE nuclear facilities and activities. Currently, DOE has already been in a period of transition stemming from the passage of the *Price-Anderson Amendments Act of 1988* (PAAA). DOE has been replacing its existing contractually based directives system for authorizing the construction and operation of nuclear facilities with a statute-based, hierarchical system of regulations, safety guides, and technical standards for making such authorizations. However, even with the implementation of rulemaking, there are still contractually based issues and limitations stemming from contract law and the contractor indemnification provisions in AEA Sect. 170. Rulemaking to implement PAAA began with draft rules issued for comment in 1991, and although some rules have been finalized to codify some requirements previously stipulated only in contractually binding directives, the PAAA rulemaking activity is now in abeyance. DOE's only PAAA rules in place at this time are for procedures for managing violations of nuclear safety requirements (10 CFR Part 820), general rules and definitions of terms (10 CFR 830.1 through 830.7), QA program requirements (10 CFR 830.120), and radiation protection of workers (10 CFR Part 835).

4.1.1 Current DOE Guidance for Establishing Facility Authorization Bases with Respect to Nuclear Safety

At present, pending either the completion of PAAA rulemaking or the implementation of a transition plan for switching to NRC licensing and regulation under new legislation needed to amend AEA, the key applicable DOE directives and standards for authorizing the construction and operation of the PDAC facility with respect to nuclear safety requirements follow:

- DOE Order 421.3 (previously 5480.23), *Nuclear Safety Analysis Reports*, April 10, 1992.
- DOE Order 423.1 (previously 5480.22), *Technical Safety Requirements*, Change 1, September 15, 1992.
- DOE Order 420.1, *Facility Safety*, Change 2, October 24, 1996.
- DOE-STD-101-92, *Compilation of Nuclear Safety Criteria for Potential Application to DOE Nonreactor Nuclear Facilities*, March 1992.

- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, December 1992.
- DOE-STD-3009-94, *Preparation Guide for U.S. DOE Nonreactor Nuclear Facility Safety Analysis Reports*, July 1994.
- DOE-STD-3013-94, *Criteria For Safe Storage of Pu Metals and Oxides*, December 1994.

These documents provide the top-level requirements for defining and documenting the facility safety analysis and operational controls. These documents also prescribe design criteria, some of which may be invoked by reference to other DOE directives and policies; NRC regulatory and technical guidance; and other DOE, government, and nongovernment technical standards.

4.1.2 Transition to NRC Licensing of the PDAC Facility

The current DOE proposal as announced in the recent press release would have nondefense nuclear facilities transitioning to NRC regulation within 5 years of the passage of authorizing legislation and defense nuclear facilities doing so within 10 years. It is assumed that, for new facilities, NRC would begin licensing and regulatory oversight of the design and construction process as soon as the legislation is passed. In this case, the PDAC facility will be subject to licensing for the possession and use of SNM under 10 CFR Part 70. DOE and NRC will have to agree on whether DOE safety analysis and design criteria guidance documents, as described in Sect. 4.1.1, can be used in some cases in lieu of the analogous regulatory guidance in the NRC's Division 3, 5, and 8 regulatory guides. Otherwise only NRC guidance documents will be used. In this case, licensing of the PDAC facility will be approached in a manner similar to the planned licensing of the MOX fuel fabrication facility, which will be subject to licensing for the possession and use of both source material and SNM under 10 CFR Parts 40 and 70. This licensing approach is discussed in the next section.

4.2 MOX FUEL FABRICATION FACILITY LICENSING

The legal and regulatory aspects of the AEA-specified indemnification of the DOE contractor that obtains an NRC license for the construction and operation of a contractor-leased DOE-owned nuclear facility are beyond the scope of this presentation specifically because the procedural mechanisms are yet to be defined and may require amending the AEA to clarify. Currently AEA Sect. 170 provides for DOE indemnification of contractors operating DOE-owned facilities and NRC indemnification of licensees, but not both situations at once. Financial qualification requirements will have to be defined elsewhere, but the facility technical design and the technical qualifications of the facility operator will be judged by adherence to or compliance with NRC regulations and regulatory guidelines for materials licenses.

NRC is in the process of revising regulatory requirements for fuel cycle facility license applications per 10 CFR Part 70. A public meeting on proposed rule changes and guidance was held in May 1995, and the draft rules were released in April 1995 for public review before the public meeting. A revised *Regulatory Guide 3.52* (Rev. 2 draft) and a new standard review plan (draft NUREG-1520) were also provided for public comment at that time, but these documents will not be issued as final guidance for providing the input to the health and safety section of the license application until the rulemaking is finalized. The alternatives for the revision of 10 CFR Part 70 based on public input were sent by the NRC staff to the Commission in SECY-96-079, "Alternatives for Regulating Fuel Cycle Facilities," on May 6, 1996. Action by the Commissioners is expected during 1997. Assuming that these actions are finalized before the MOX fuel fabrication facility licensing process begins, the key guidance documents for the 10 CFR Part 70 license application follow:

- *NRC Regulatory Guide 3.39*, Revision 0, "Standard Format and Content for License Applications for Pu Processing and Fuel Fabrication Plants," January 1976.
- Draft *NRC Regulatory Guide 3.52*, Revision 2, "Standard Format and Content for the Health and Safety Sections of License Applications for Fuel Cycle Facilities," January 1995, NRC Public Document Room Accession No. 9504060249, April 4, 1995.
- Draft NUREG-1520, *Standard Review Plan for the Review of a License Application for Fuel Cycle Facility*, NRC Public Document Room Accession No. 9504060249, April 4, 1995.

These documents require that the license applicant demonstrate that the facility design and facility operating procedures adhere to regulatory guidance in several other NRC Division 3, 5, and 8 regulatory guides. These guidance documents will be used by the DOE contractors who design and operate both the PDAC facility and MOX fuel fabrication facility to satisfy NRC regulatory requirements for licensing.

4.3 REACTOR LICENSING

The licensing procedures for converting the fuel cycle of currently operating licensed commercial reactors to use surplus WG plutonium in MOX fuel are relatively straightforward and are documented in Appendix C. These procedures are based upon the regulations at 10 CFR 50.59 that address the licensee's responsibilities for assessing and reporting the safety significance of contemplated changes in the reactor design and for taking the appropriate actions in response to that assessment prior to implementing the design change. If the design change is determined not to be safety significant nor to require a change in the Technical Specifications in the license as judged per the regulatory criteria, the licensee may proceed with the change prior to the annual reporting of changes to NRC and the annual submission of updates to the final safety analysis report (FSAR) to reflect the design change as required under 10 CFR 50.71. If the design change is found by the licensee to be safety significant with respect to changes in the design bases documented in the FSAR or to require changes in the Technical Specifications of the facility, the licensee must submit a formal request for a license amendment under 10 CFR 50.90 prior to implementing the design change. In this case, before the design change may be implemented by the licensee, NRC must process and officially approve the request under 10 CFR 50.91 and 50.92. There is also the possibility that all design changes associated with the conversion to use weapon-origin MOX may be required by NRC to be included in a license amendment application notwithstanding the provisions of 10 CFR 50.59. Applicable regulatory and technical guidance to be used by the licensees in handling these licensing actions is discussed in Appendix C.

While there is no distinction between LA and reload fuel licensing requirements, additional consideration of MOX LAs is warranted. LEU LAs are typically loaded through the 10 CFR 50.59 process, under the provision for "conduct[ing] tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question." Insertion of MOX fuel LAs would require a change in the Technical Specifications due to the specific reference to LEU fuel in the core description of the "Design Features". Furthermore, a waiver may also be required for 10 CRF 50.46 due to the fuel description contained therein. Use of the 10 CFR 50.59 process for MOX LA insertion without NRC approval assumes prior approval of a license amendment request to change the core descriptions to include MOX fuel. Precedence for such an approach is provided by the MOX LTA campaigns undertaken domestically during the 1970s. Once the required changes to the Technical Specifications were approved, insertion of the MOX LAs could proceed through the 50.59 process. Depending on the details of the amendment request and on any restrictions that NRC might

impose, the license amendment for the LA irradiation could also authorize routine loading of MOX fuel.

4.4 REGULATION OF SAFEGUARDS UNDER THE AGREEMENT WITH THE INTERNATIONAL ATOMIC ENERGY AGENCY

The licensing requirements for the MOX fuel fabrication facility, according to 10 CFR 40.31(g) and 70.21(g), include responding to an NRC request for information related to materials accountability and safeguards as part of the construction permit application. The MOX fuel fabrication facility meets the definition of an installation subject to placement on the "United States eligible list" as defined by 10 CFR 75.2(a) and 75.4(k)(3) and (5). This means that the facility is subject to the U.S. safeguards agreement with the International Atomic Energy Agency (IAEA). The MOX fuel fabrication facility will contain formula quantities or Category I quantities of SNM as defined under Annex II, "Categorization of Nuclear Material," of The Convention on the Physical Protection of Nuclear Material, as documented in the IAEA Information Circular (INFCIRC/274/Rev.1), May 1980. The MOX fuel fabrication facility will also be subject to the physical protection requirements of 10 CFR Part 73 and the materials control and accountability requirements of 10 CFR Part 74.

The information to be provided to NRC for purposes of safeguards is defined in 10 CFR 75.11(b) and constitutes the basis upon which the NRC will make its determination of the imposition of appropriate safeguards requirements of the type described in 10 CFR 75.11(d). Such requirements will be imposed upon the licensee via either Facility Attachments to the license as given in 10 CFR 75/8(a) or license amendments as given in 10 CFR 75.8(b).

While there is no specific regulatory guidance issued by NRC for compliance with 10 CFR Part 75, guidance has been issued for compliance with related provisions of 10 CFR Part 70. Relevant NRC guidance includes:

- *Regulatory Guide 5.52*, Revision 2, "Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (other than Nuclear Power Plants)," July 1980.
- *Regulatory Guide 5.13*, "Conduct of Nuclear Material Physical Inventories," November 1973.
- *Regulatory Guide 5.58*, Revision 1, "Considerations for Establishing Traceability of Special Nuclear Material Accounting Estimates," February 1980.

5. TECHNOLOGY BASE FOR THE MD MISSION

An extensive technology base exists for supporting the MD mission. For discussion purposes in this document, the mission has been divided into six activities: (1) pit disassembly, (2) conversion of metal to oxide, (3) plutonium purification, (4) PuO_2 conditioning, (5) MOX fuel fabrication, and (6) reactor irradiation. Repository disposal of the resulting spent MOX fuel is not included in this discussion. Figure 2 is a processing flow sheet showing each of these activities in relation to the others with material flows identified. In the remainder of this section, the existing technology base supporting each of the six activities is described briefly, including an assessment of the applicability of this technology base to the MD mission. Ongoing R&D activities that support and/or extend the technology base are then summarized. Additional work that is currently envisioned to support implementation of the technology is described. Finally, other issues and implementation options are discussed.

5.1 PIT DISASSEMBLY

A significant portion of the surplus WG plutonium is in the form of intact weapons components or pits. To be utilized in MOX fuel, the plutonium metal must first be separated from the remainder of the pit. The technology and activities supporting pit disassembly are described below.

5.1.1 Technology Base

Several options exist to effect plutonium separation from the remainder of the pit: mechanical separation, aqueous dissolution, and ARIES under development by Los Alamos National Laboratory (LANL) and Lawrence Livermore National Laboratory (LLNL). ARIES has been designated by the MD program as the reference process for pit disassembly. The process consists of a mechanical bisection of the spherical pit into two hemishells. Each hemishell is exposed to hydrogen (diluted in an inert gas carrier) at room temperature. After an induction period, the plutonium metal reacts with the H_2 to form PuH_{2+x}. This hydride spalls from the surface of the bulk metal, exposing additional metal for reaction. Reasonable plutonium separation from the remainder of the pit may be accomplished using this technique, although some carryover of impurities has been experienced in the limited development testing performed to date. The hydride process is a dry process and is being designed to produce a small waste stream. The hydride process has been under development for several years to support plutonium recycle within the defense complex as a replacement for aqueous techniques that had been used previously and has been demonstrated recently at the laboratory scale.

ARIES consists of more than simple pit disassembly. It is being designed to separate the plutonium, convert it to a stable form (either metal or oxide), package it into storage cans, decontaminate and assay these product cans, and decontaminate and package the nonplutonium materials.

5.1.2 Ongoing R&D Activities

As part of the ARIES development project, hydride separation of plutonium from the rest of the pit is being demonstrated at the laboratory scale. An integrated demonstration is being constructed at LANL's TA-55 facility. When completed, this demonstration line will be utilized to disassemble and process up to 200 pits to demonstrate the concept feasibility at pilot scale. R&D being conducted as part of this effort is expected to help determine the relevant reaction rates and, therefore, the potential plutonium throughput rates.

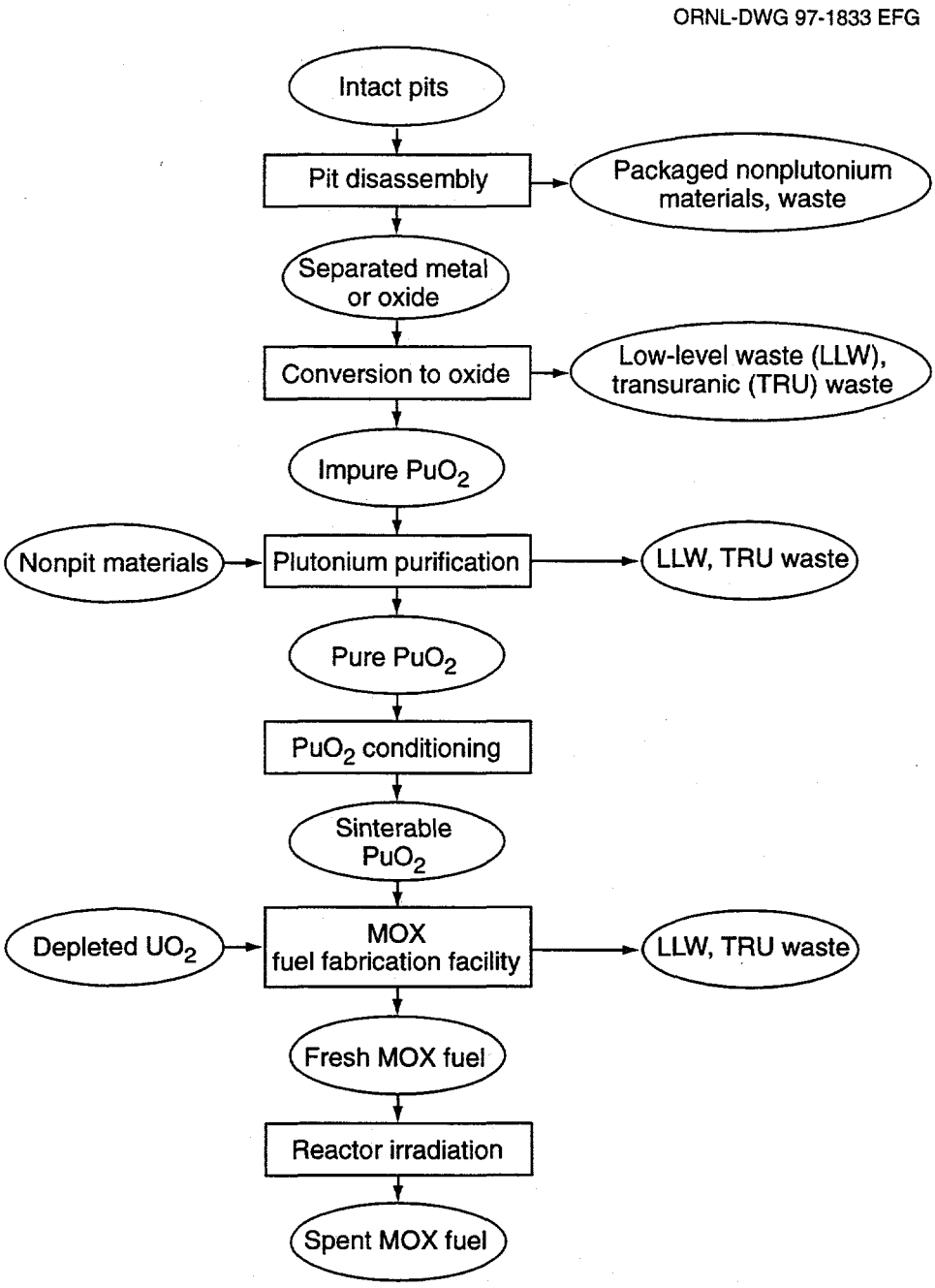

Fig. 2. FMDP processing flow sheet.

5.1.3 Additional R&D Needs

Following successful operation of the ARIES demonstration line, additional scale-up and modification will be necessary to modify the design for full-scale operation. This effort will require site-specific evaluation of the potential locations for the production ARIES facility.

5.1.4 Additional Issues and Options

The only additional issue concerning pit disassembly technology is suitability of all of the surplus WG plutonium for MOX fuel use without aqueous purification. It is anticipated that some of the pit types that are included in the surplus materials may be diverted to the immobilization alternative. However, according to the developers, ARIES is being designed to accommodate all of the surplus pit types.

5.2 CONVERSION TO OXIDE

The plutonium metal in surplus weapons components must be converted to oxide for utilization in MOX fuel. Conversion to oxide is a well-established technology, and several viable options exist.

5.2.1 Technology Base

Production of PuO_2 may be accomplished using one of several techniques. Six techniques are described—four based on conversion from nitrate solution and two based on conversion from plutonium metal. Most of the PuO_2 utilized to date for MOX fuel has been converted to oxide using the oxalate conversion process. Oxalic acid is added to Pu(IV) [or Pu(III)] nitrate solution to precipitate $Pu(C_2O_4)_2 \cdot 6H_2O$ [or $Pu_2(C_2O_4)_3 \cdot 9H_2O$] under closely controlled conditions. The precipitate is then filtered, dried, and calcined to the oxide. The product, known as oxalate-derived plutonia, has a distinct morphology unique to the process and is sinterable ceramic powder suitable for direct fabrication. The oxalate conversion process has been utilized in both defense facilities and commercial facilities around the world to convert the plutonium nitrate solution from PUREX or ion exchange purification.

The second technique for production of plutonia from plutonium nitrate solution is direct calcination of the nitrate. Direct calcination has not been widely used and can result in an inactive ceramic product as a result of partial sintering of the oxide during calcination.

The remaining techniques for conversion from nitrate solution to oxide are used for co-conversion of mixed nitrate solutions of uranium and plutonium. The MOX powder commonly referred to as coprecipitated material is produced by adding ammonia to a dilute mixed nitrate solution. The uranium is precipitated as $(NH_4)_2U_2O_7$ and the plutonium as $Pu(OH)_4$. The mixed precipitate is then filtered, dried, and calcined to oxide. A second conversion technique for mixed solutions is based on the carbonate process used for uranium conversion. In this AUPuC process, Pu(IV) is oxidized to Pu(VI) followed by the addition of ammonia and carbon dioxide. The mixed precipitate, $(NH_4)_4U/PuO_2(CO_3)_3$, is filtered, dried, and calcined to oxide.

For the reference ARIES process, the plutonium product will either be in the form of metal or hydride. The simplest technique for conversion of metal to oxide is direct oxidation.

An alternative technique that is under development as part of the ARIES project is the hydride oxidation (HYDOX) process. Two competing HYDOX processes are under development: a two-step process in which plutonium hydride is oxidized and a three-step process in which plutonium hydride is converted to nitride that is subsequently oxidized. Neither process has been utilized industrially to supply sinterable plutonia. However, preliminary evidence

suggests that a higher degree of homogeneity **may** be achievable in fuel fabricated with this hydride-derived powder because of its reduced tendency toward agglomeration.

5.2.2 Ongoing R&D Activities

R&D of plutonium oxidation processes is currently being conducted as part of the ARIES development project at LANL and LLNL. This work is expected to provide the data necessary to select the two-step or the three-step processes for metal-to-oxide conversion. Characterization of the product oxide is being conducted as part of this work. No work on aqueous processes for oxide production is being conducted.

5.2.3 Additional Work

Once a reference process has been identified, scale-up and industrialization will be required. This work will be performed as part of the overall industrialization of ARIES. Construction and operation of the ARIES demonstration line should provide the information necessary to design and construct the production facility.

5.2.4 Issues

Because of the early stage of development of the HYDOX process, compatibility of the product oxide with any of the commercial fuel fabrication processes has not been demonstrated. However, according to the developers, the ARIES process is being adjusted as necessary to meet the needs of the fuel fabricator. It is currently assumed that the powder converted through HYDOX can be conditioned to meet the fabricators' requirements. Because all current commercial MOX fuel fabrication processes include a milling step for the plutonia, it is probable that the morphology of the plutonia will not be important. This assumption must be proven through demonstration.

The second issue with respect to conversion is the location of the conversion process. If the plutonium is to be stored for long periods prior to fuel fabrication, metal may be the preferred form. However, if the PDAC facility containing ARIES is to be operated for only a short time and decommissioned, it may not be available for conversion of the metal storage ingots when they are needed by the fuel fabricator. The location of the plutonium purification operations and the processes to be utilized may also affect the location of the conversion facility.

5.3 PLUTONIUM PURIFICATION

The surplus plutonium may be categorized as (1) that arising from ARIES processing of pits and (2) the remainder that includes clean metals and oxides, contaminated metals and oxides, MOX fuels, and other forms. The purification needs of each category differ greatly.

The plutonium derived from ARIES processing of pits is expected to be relatively pure material. Contamination may be present as a result of carryover of nonplutonium components from the pit. Of the potential impurities that might be carried over, gallium has received the most attention thus far. However, other impurities may also be carried over. Because several different pit types are included in the surplus inventory, the types and quantities of impurities may vary. Additionally, contamination may result from the processing itself.

The remaining material (i.e., that not derived from ARIES processing of surplus pits) includes a wide array of potential impurities. Each category of material may require dedicated treatment to remove impurities specific to that category.

5.3.1 Technology Base

The current reference process for purification of pit-derived plutonium focuses on gallium removal. This process, coined the Thermally Induced Gallium Removal (TIGR) process, consists of a thermal treatment of PuO_2 powder under reducing conditions. A reducing atmosphere of Ar - 6% H_2 is passed through the powder at a temperature of ~800 to 1000°C, reducing Ga_2O_3 to Ga_2O. The more volatile Ga_2O is carried out by the flowing gas atmosphere. The TIGR process should be useful for removal of all species that are volatile under the specified conditions. However, nonvolatile species will not be removed.

The reference process for purification of all contaminated materials and the backup process for purification of pit-derived plutonium is aqueous processing. Aqueous purification consists of two steps—dissolution and separation. Plutonium-bearing materials would be dissolved in nitric acid, with HF additions as required to ensure complete dissolution. The resulting plutonium nitrate solution would be subjected to solvent extraction, ion exchange, or both depending on the impurities and the plutonium concentration. The separation inherent to the precipitation process may in fact be sufficient to achieve the needed purity without additional processing. Both solvent extraction and ion exchange are well-developed technologies that have been widely used for plutonium processing in both defense and commercial facilities around the world. These aqueous processes have historically been associated with large volumes of liquid radioactive waste such as those produced in various defense facilities throughout the world. *However, modern processing facilities that incorporate solvent, nitrate, and condensate recycle are designed to produce no liquid radioactive wastes.*[11] For plutonium processing, a small volume of evaporator solids (possibly vitrified into a glass) and other solid wastes would be disposed of as transuranic (TRU) waste. No fission product waste is anticipated because of the lack of fission products in the surplus materials.

5.3.2 Ongoing R&D Activities

The reference TIGR process for volatile impurity removal is under development at LANL. Additional planned experiments may help provide statistically significant data that can be used to quantify the removal of gallium and other volatile impurities.

5.3.3 Additional R&D Needs

Beyond the laboratory-scale experiments currently under way, additional work will be required to modify and scale up the TIGR process for implementation at the pilot and industrial scale. Repeated experiments may have to be performed to produce statistically significant results. Gallium trapping techniques may have to be developed to protect the furnace(s) in which the TIGR process is performed.

Beyond TIGR, which is being developed specifically for gallium removal, R&D may be required to develop additional dry processes for other impurities expected to be present in both pit-derived and other surplus materials. Blending of PuO_2 derived from various sources has been suggested as a method of obtaining a suitable product. However, blending may not resolve the issue of impurities present in significant quantities that may prove unacceptable.

The aqueous backup processes for plutonium purification are well-developed and require no additional R&D.

5.3.4 Issues

A detailed comparison of the purification options has not been performed. Furthermore, because of the limited data available on the dry treatment processes, meaningful comparisons of the waste generation between the options are difficult to make.

The TIGR process is being developed only for gallium removal. Processes for other impurity removal have not been proposed. Additional processes may require development and demonstration for removal of impurities other than gallium for either fuel performance or programmatic reasons.

Finally, the purification process can be sited with ARIES in the PDAC facility, in a separate facility, or integrated with the MOX fuel fabrication facility. The current planning is for inclusion in the PDAC facility. However, many advantages may accrue from collocation of an aqueous system with the MOX facility including uniformity of plutonium into the fabrication process, the ability to recycle contaminated scrap generated as part of MOX fuel fabrication, and reduced purity and powder requirements on the feed materials. If aqueous purification is not available at the MOX fuel fabrication facility, contaminated scrap could be diverted to the immobilization alternative.

5.4 PLUTONIA POWDER CONDITIONING

In the reference flow sheet, a plutonia powder conditioning step may be required following oxide production and/or TIGR. Powder conditioning is expected to consist of mechanical treatment (milling) and oxygen-to-metal (O/M) ratio adjustment.

5.4.1 Technology Base

The oxide powder produced early in the ARIES development work was coarse in comparison with the fine oxalate-derived powder more commonly utilized for MOX fuel fabrication. To increase the homogeneity of MOX fuel produced from this powder, the plutonia powder used to fabricate the test fuels thus far produced in TA-55 has been milled extensively as part of the fabrication process. Milling of plutonia powder with and without the addition of urania diluent are well-established technologies. Furthermore, the TIGR process results in partial sintering of the plutonia particles as evidenced by the reduction in surface area that has been measured in the limited experiments performed to date. Therefore, powders requiring TIGR processing may need to be milled following thermal treatment. The milling inherent to the commercial MOX fuel fabrication processes may be sufficient; but this has not yet been determined.

Limits on the O/M ratio are set in the plutonia feed specification. Because it is conducted in a reducing atmosphere, TIGR can drive the plutonia slightly substoichiometric; that is, O/M < 2.0. An adjustment to the O/M ratio is performed by heating with a controlled steam partial pressure.

5.4.2 Ongoing R&D Activities

Both the milling and O/M adjustment that are currently proposed as conditioning steps for the dry-process plutonia powder are being investigated and tested in ongoing experimental activities at LANL. However, statistically significant results are not yet available. Completion of the ongoing activities should provide some of the required data, such as powder characteristics before and after treatment (particle size, O/M, bulk density, and tap density) and acceptability to the fabricator (achievable density and homogeneity).

5.4.3 Additional R&D Needs

Additional experimentation will be required to demonstrate the acceptability of the plutonia powder conditioned through milling and O/M adjustment. Only limited fabrication testing has been performed to date. Statistically significant data should be developed to demonstrate the applicability of these processes to the spectrum of feed materials expected.

5.4.4 Issues

The powder morphology or crystal structure characteristic of oxalate-derived plutonia currently utilized in commercial practice cannot be reproduced through mechanical conditioning of hydride-derived material. Milling of the powder changes the particle size and, therefore, the surface area. However, these variables cannot be changed independently through milling. The HYDOX powder morphology has not been characterized sufficiently to assess its effects on fuel fabrication. The ease with which HYDOX powder can be utilized will be determined through dedicated testing after selection of a consortium and its associated fuel fabrication process.

An additional issue with respect to powder conditioning is the variety of sources of plutonia utilized in the fabrication facility. Nonpit materials and contaminated scrap may be processed through solvent extraction and/or ion exchange, resulting in an oxalate-derived product. This material must be incorporated into the fabrication process along with hydride-derived material. Again, this may prove to be a discriminator among the fabrication processes because some may be more robust in their ability to incorporate various plutonia powders without process modification.

5.5 MOX FUEL FABRICATION

All of the reactor types under consideration utilize sintered MOX fuel pellets clad in zirconium alloy cladding. Fabrication of this type of MOX fuel is well-established technology supported by a rapidly expanding experience base.

5.5.1 Technology Base

MOX fuel fabrication technology is established and commercially available. MOX fuel fabrication technology borrows heavily from LEU fuel fabrication and is modified only as necessary to accommodate the particulars of plutonium, including glove-box or remote operation. Laboratory, small-scale pilot, and/or commercial facilities exist or have existed in many countries, including Belgium, Canada, France, Germany, Italy, Japan, the United Kingdom, and the United States.

Despite previously extensive domestic capabilities and experience, only a single laboratory facility in the United States (TA-55 at LANL) is capable of fabricating fuel containing plutonium. A second facility, the Fuels and Materials Examination Facility (FMEF) on the Hanford reservation, was constructed to fabricate fast-reactor plutonium fuel. FMEF has never operated and could be modified to produce LWR MOX fuel. Also a number of commercial facilities have been decontaminated, but these have not been assessed for their potential for restart.

5.5.2 Ongoing R&D Activities

Three activities related to fuel fabrication are currently under way. Two of these consist of the fabrication of test fuel for irradiation in test reactors. The third activity is generic fabrication studies to investigate the effects of feed variability.

Parallex is planned as a joint irradiation of U.S. and Russian CANDU MOX fuel in the NRU reactor. Approximately 4 kg of test fuel have been fabricated in TA-55 using depleted ex-ADU urania diluent supplied by AECL. One of the goals of the Parallex test is to help set a homogeneity specification for CANDU MOX fuel.

The second test fuel fabrication campaign focuses on generic LWR MOX fuel. Approximately 1 kg of MOX will be fabricated during 1997 to meet a generic LWR MOX fuel pellet specification. The Parallex fabrication process will be modified as necessary to meet the LWR specification.

Finally, a feed variability study is planned at LANL to assess the effects of variations in both urania and plutonia. Urania from various sources and conversion methods will be mixed with a single batch of plutonia. Similarly, plutonia from different sources and conversion methods will be mixed with a single batch of urania.

5.5.3 Additional R&D Needs

Additional test fuel above and beyond the requirements of the three ongoing activities may be required. Furthermore, fuel meeting commercial fuel specifications and quality will be required to implement an LTA or LUA project. Finally, when the mission fuel fabrication process is chosen, development may be necessary to modify the fabrication process to accept the HYDOX powder, to accommodate unique impurities, or to accommodate urania powder converted through a process different from that typically utilized. Prior to the completion of procurement activities, it is difficult to assess the R&D needs to implement the mission fabrication process. However, because the matrix material defines many of the MOX properties, a single urania source should be utilized to ensure comparability among the tests.

5.5.4 Issues

5.5.4.1 Fabrication process

The suitability of HYDOX powder, conditioned as necessary, likely depends on the fabrication process. Some indication of the suitability of the powder may be obtained from the ongoing feed variability studies. However, it is likely that the eventual mission fabricator will perform additional fabricability tests in prototypic equipment.

Two options are available for obtaining domestic MOX fuel fabrication capability: (1) license European technology from one of the existing commercial MOX suppliers or (2) update domestic technology developed in the 1970s. If European technology is to be utilized, some of the early fabrication may be performed in the corresponding European facility prior to the availability of a domestic facility.

5.5.4.2 Urania supply

Urania constitutes about 95% of MOX fuel. Several methods exist for urania powder production. Each production method results in different powder properties. Each of the existing commercial MOX fuel fabrication processes is associated with one or more of these urania powders. Because the urania dominates the overall characteristics of the MOX powder, its flow properties are important in both blending and pelleting.

The master mix fabrication processes rely on the free-flowing properties of ex-AUC urania (i.e., urania resulting from calcination of ammonium uranyl carbonate precipitate). Use of urania powder that is not free-flowing would require the addition of a slug/granulate operation of the final MOX powder blend prior to pelleting to achieve a free-flowing press feed.

MOX fabrication processes in which all of the MOX powder is milled, such as co-milling or the Short Binderless Route (SBR), may be less sensitive to the urania powder properties, because high-intensity milling partially erases the powder's ceramic history. This is evidenced by the use of dry conversion powder, produced through the Integrated Dry Route, in the SBR process.

Historically, domestic fabricators of both LEU and MOX fuels have utilized ex-ADU urania (i.e., urania resulting from calcination of ammonium di-uranate precipitate). Ex-ADU urania is not free-flowing, and thus a slug/granulate step has been included in the fabrication processes. Recently, because of environmental compliance issues related to the aqueous ADU powder production, the domestic producers have implemented dry conversion processes to supplement or replace ADU capacity. The adaptability of these dry conversion powders to the master mix MOX fabrication processes is unknown to the authors.

The bottom line with respect to urania supply is that each of the commercial MOX fabrication processes has been optimized for a particular urania powder. If an alternative urania powder is to be utilized (e.g., due to lack of domestic production capacity for a particular powder), fuel fabrication process development and modification may be required to adapt to the new powder.

5.5.4.3 MOX fabrication facility scrap recycle

Scrap recycle at the MOX fabrication facility is another issue for consideration. If the plutonium purification facility is located at the MOX plant and equipped for uranium/plutonium partitioning, contaminated scrap can be blended back into the process stream. However, if aqueous processing is used for scrap recovery and dry processing is used for the bulk of the PuO_2 feed powder, two distinct types of PuO_2 feed materials will have to be handled in the MOX fabrication process. This may not be a problem because of the small scrap volumes, but it may require some process development or demonstration. Furthermore, in the absence of aqueous purification of scrap, contaminated scrap could be diverted to the immobilization alternative.

5.5.4.4 Sources for test and LTA fuel

Fabrication of test fuel and LA fuel must be accomplished well in advance of completion of the domestic MOX fuel fabrication facility to avoid major delays in the implementation schedule. For example, LTAs are required at least 5 years in advance of the corresponding reload to allow sufficient time for irradiation, cooling, PIE, and NRC review and approval of the resulting data.

If shipment of plutonium overseas is not permitted, domestic sources of this fuel must be developed. Backfitting of a fabrication process into a building originally constructed for either a different purpose or a different fabrication process may prove difficult. Results from a site evaluation provide some information on this subject, but resolution will require additional site-specific work and the selection of the actual mission fabrication process.[12]

It is important in considering the fabrication facilities to distinguish between fabrication of (1) test fuel, (2) LA fuel, and (3) mission reload fuel. Fuel fabricated for irradiation in one or more test reactors need not be subject to commercial QA requirements. Furthermore, total quantities of test fuel need not be large (on the order of a few kilograms of total MOX). The existing TA-55 laboratory-scale facility is presently being utilized for fabrication of test fuel.

Conversely, fuel fabricated for irradiation in LTAs or LUAs must be produced in accordance with commercial quality requirements that as a minimum comply with 10 CFR 50 Appendix B. The total quantity of fuel required for an LTA program can range from a few kilograms of MOX up to a few metric tons of MOX. Smaller quantities are required to incorporate only a handful of test rods into one or more standard assemblies (a minimum of 128 kg MOX is estimated for 64 full-length rods). Larger quantities are required to fill complete assemblies with test fuel (a maximum of 3200 kg MOX is estimated for eight LTAs). Most importantly, the test fuel must be

prototypic of that eventually produced in the mission fabrication plant. The processes and materials must be representative of the final process. Thus, it is important to have the selected fuel fabricator involved from the start in designing, constructing, and operating the LTA fabrication facility. Domestic MOX LTA fabrication capacity does not currently exist. The following options exist for production of MOX LTAs.

1. Procure LTA fuel from the European fabrication facility upon which the domestic facility design is based.
2. Build an interim prototypic fuel fabrication line in an existing structure at a DOE site and operate it as an R&D facility independent of the domestic mission facility.
3. Build an interim prototypic fuel fabrication line in an existing structure on the DOE site chosen for the mission facility, and operate it as an R&D facility in a separate structure under DOE orders.
4. Build a high-priority fabrication line as a modular component of the mission fuel fabrication facility.

The authors believe that the first choice, fabrication in an existing European facility, is preferable on technical and economic grounds. Fuel resulting from the existing commercial line on which the domestic facility design will be based should, without question, be prototypic of the mission fuel. Schedule and cost may also favor this approach. However, due to the production goals of any commercial facility, experimentation with a variety of plutonia and urania feeds may be more difficult and expensive than it would be in a dedicated experimental line.

Political or programmatic considerations may rule out shipment of surplus WG plutonium overseas. Under these circumstances, construction of an interim R&D prototype line in an existing DOE structure (option 2 or 3) is recommended. If it proves desirable to maintain a development line for processing small batches of unique feeds or for conducting process development, then option 3 would be preferable because it would allow continued operation of the R&D facility as an extension of the mission facility.

The small quantity of LA fuel required presents the possibility that a safeguards and security (S&S) Category III facility (as defined per DOE Order 5633.3B) could be utilized for fabrication in conjunction with a Category I storage vault for storage of feed plutonium and completed rods/assemblies. The main advantage of a Category III facility rather than a Category I facility is ready access by fuel fabricator and utility personnel, including foreign nationals. The main disadvantage of a Category I facility is the smaller batch sizes and more complicated material flow sheet. The DOE sites are being evaluated for both Category I and III facilities to fulfill this need if and when it develops.

For any existing or modified domestic R&D facility, the imposition of a commercial QA/QC program is mandatory. Many of the issues to be addressed as part of the construction and operation of the mission facility will apply to the R&D facility. The degree to which the demands of the fuel fabricator will outweigh views of the site personnel must be negotiated. The fuel fabricator must be given sufficient freedom in the design, operation, and oversight of the facility to allow the fabricator to claim ownership of the product.

The last option listed, construction of a high-priority line as a modular part of the mission facility, may not prove practical because of the tight schedule and the logistics of starting up an actinide facility while construction proceeds in the remainder of the plant. The primary advantage of this option is that the line could continue to produce mission fuel as an integral part of the mission facility throughout the mission.

5.5.4.5 Fabrication of mission fuel

Mission fuel fabrication requirements are similar to those of LTA fabrication with the exception of throughput. One important difference is the stated goal that the mission facility be licensed under NRC regulations rather than DOE Orders. The options include

1. completion/modification of FMEF or another existing facility,
2. construction of a new facility, and
3. contracting for fabrication in one of the existing European facilities.

The primary issues with the first option include reconstitution of the licensing basis to the satisfaction of NRC and modification of the facility to accommodate one of the commercial MOX fabrication processes. Construction of a new facility eliminates both of these concerns but may increase both cost and implementation schedule.

Contracting for the entire mission in one of the existing European facilities appears to be the most desirable option from the technical standpoint. However, a detailed evaluation of this option has not been performed. Institutional and policy considerations complicate the picture. This option was ruled out as part of the ROD process in part because it might require additional capacity at the chosen European facility.

5.6 MOX FUEL IRRADIATION

Three reactor types are being considered for disposing of surplus WG plutonium as MOX fuel. These three thermal spectrum reactors are pressurized-water reactors (PWRs), boiling-water reactors (BWRs), and pressurized-heavy-water reactors (PHWRs). MOX fuel is usually understood to refer only to the mixed uranium/plutonium oxide fuel used in thermal spectrum reactors and is distinct from the uranium-plutonium oxide fuel utilized in fast spectrum reactors. A brief history of the development and utilization of MOX fuel in the world is contained in Appendix A.

5.6.1 Technology Base

This description of the MOX fuel irradiation technology base is organized by the reactor type. Five nuclear steam supply system (NSSS) designs representing three thermal reactor types are under consideration for disposal of surplus WG plutonium: W, ABB–CE, GE, Babcock & Wilcox Co. (B&W), and AECL. Each of the reactor types is discussed generically, followed by descriptions of the specific NSSS designs along with potential advantages and disadvantages of each. For each of the LWR vendors, several NSSS models may exist. For the disposition mission, because of the desire to utilize a small number of well-operated and well-maintained nuclear units that have sufficient licensed lifetime remaining without life extension, it is likely that only the latest models will be considered for the mission. In the following discussion, it is assumed that only the latest models will be considered for the MOX mission even though no official decision regarding particular units or NSSS models has yet been rendered.

All three of the reactor types under consideration for disposal of surplus WG plutonium utilize a thermal neutron spectrum. The fuels utilized in the three reactor types differ in geometry and materials but are similar in many respects. The technology base supporting one of these may be applied to the others with some limitations. The majority of the world's MOX fuel experience has been with PWRs, though BWRs are also supported by an extensive experience base. MOX fuel utilization in PHWRs is limited to testing and utilization in a small number of test/demonstration reactors. Each of these reactor types is described in more detail below.

The applicability of the experience base may be assessed relative to the operating envelope as determined by the linear heat rating, the peak pellet burnup, the assembly average burnup, and

the maximum centerline temperature. Within the last 20 years, both LEU and MOX fuel designs have evolved in an effort to reduce the demands on the fuel, increase the operating margins, and reduce fuel failure rates while increasing discharge burnup. One of the primary manifestations of this evolution is the continuous reduction in the fuel rod diameter. By reducing the fuel rod diameter, the designer readily lowers the linear heat rating and the maximum centerline temperature while maintaining a constant bundle power output. When assessing the MOX fuel experience base, note the changes in the fuel design envelope that have occurred over the last 20 years. Early data that bound the current design limits may not be fully applicable due to differences in fuel technology.

5.6.1.1 PWRs

PWRs are the prevalent reactor type throughout most of the world. Pressurized-water NSSSs have been supplied by many companies including W, ABB–CE, B&W, Framatome (Fra), KWU (now part of Siemens), Siemens, and Mitsubishi Heavy Industries (MHI). Because of similarity of design, fuel experience is widely applicable among the various reactor designs. Many of the companies compete for supply of LEU and MOX fuel assemblies to reactors of their own type as well as those of their competitors. The most common PWR fuel assembly type is the 17 × 17 assembly, which is utilized in certain NSSS models by W, Fra, and MHI. Fra technology originated with the W design, so applicability of W technology to Fra and vice versa may prove better than its applicability to other PWR designs, but this advantage is difficult to quantify.

MOX fuel has been tested and/or utilized in PWRs in the United States, Europe, and Japan. For the most part, the MOX fuel assembly designs have been similar to LEU designs. MOX core loadings have for the most part been limited to about one-third of the fuel assemblies, although limited experience exists to 50% MOX assemblies and higher.

Because most of the PWR designs utilize rod control cluster (RCC) type control assemblies and soluble boron for reactivity control, MOX fuel is usually loaded in all-MOX assemblies that are placed in non-RCC positions during their initial irradiation cycle. Graded enrichment can be utilized to minimize peaking caused by the high thermal neutron flux at the interface between MOX and adjacent LEU assemblies. Removable discrete burnable absorber rods may also be inserted into the RCC openings in MOX assemblies to provide additional reactivity control.

Most of the PWR designs in operation include sufficient RCC positions to provide adequate shutdown margin with about one-third MOX assemblies without major equipment modifications. The reduction in reactivity worth of the RCCs and soluble boron caused by increased MOX utilization levels may be accommodated by increasing the number of RCCs (a major backfit to existing reactors that in many cases would require replacement of the vessel head), increasing the use of discrete absorber rod clusters, and/or substituting enriched boron for the natural boron typically used for soluble boron systems.

W. The W four-loop 17 × 17 NSSS design is often considered to be the reference PWR, if not LWR, design. The same basic technology was adopted for the Fra 900-MW class of reactors, the Fra 1300-MW class, and the MHI NSSS designs. W has supplied MOX fuel fabricated in a former domestic facility as well as in one of several foreign facilities under contract. Most of the world's MOX fuel utilization, measured in terms of rod-years, has taken place in Fra 900-MW plants that share many similarities with domestic W units.

W was a major player in early MOX fuel development, fabrication, and utilization. However, following the decline of domestic MOX infrastructure, W corporate capabilities related to MOX fuel have been reduced to design and performance modeling, leaving the fabrication duties to overseas subcontractors including BN and British Nuclear Fuels, Ltd. (BNFL). W continues to support international customers in the areas of MOX fuel design and supply. This fact may prove to be an advantage for W in terms of licensing of domestic MOX fuel utilization. However, the

W codes and methods being utilized for overseas MOX fuel supply are not currently approved by NRC for domestic MOX fuel design and supply. Both the physics and the fuel performance codes will have to be approved by NRC prior to loading of reload quantities of MOX in a domestic nuclear unit.

Under contract to DOE, W has developed several core designs for disposition of surplus WG plutonium. W uses three types of burnable absorber for reactivity control: soluble boron (either natural or enriched), a zirconium diboride pellet coating known as Integral Fuel Burnable Absorber (IFBA), and discrete absorber rod clusters loaded in the guide thimble pin locations known as Wet Annular Burnable Absorber (WABA). W developed three MOX core designs relying on use of IFBA. Two of these utilized radial plutonium concentration zoning within assemblies to counteract peaking between MOX and adjacent UO_2 assemblies. Used as part of a partial MOX core, these designs were predicted to dispose of 0.389 MT and 0.439 MT of plutonium per reactor year. W also developed a full-MOX design that was predicted to dispose of 1.07 MT of plutonium per reactor year without radial enrichment zoning through incorporation of additional IFBA and use of WABA. W also developed a core design that did not rely on IFBA coating on the MOX rods. This design utilized only WABA and enriched soluble boron and was predicted to dispose of 1.13 MT of plutonium per reactor year. Fuel assembly designs utilizing radial plutonium concentration zoning in the MOX assemblies were also prepared for the transition from LEU to full MOX cores.

ABB–CE. While ABB–CE has never had MOX fuel fabrication capabilities, ABB–CE was involved in several MOX fuel development and testing projects during the 1970s. ABB–CE included the capability to fuel the entire core with MOX in its latest model built domestically, the System 80. The three Palo Verde units are the only System 80 units that were completed domestically. According to ABB–CE, these three units were not only designed but also constructed to utilize full MOX cores. Design features of the System 80 that were incorporated for MOX capability include (1) increased decay heat removal capability for core and spent-fuel pool cooling, (2) provision of extra control element assemblies (and additional vessel head penetrations that can accommodate additional control element assemblies), (3) modifications to the chemical and volume control system and the safety injection systems to accommodate increased boron concentration, and (4) increased vessel thickness to mitigate increased vessel fluence. Two earlier ABB–CE NSSS designs utilize the same basic assembly design but do not include the MOX fuel-specific design features of the System 80. Three units of the 3410 series are operational—San Onofre Units 2 and 3 and Waterford 3, and one unit of the 2815 series is operational—Arkansas Nuclear One Unit 2. These units could probably accommodate partial MOX core loadings, but would require equipment modifications to allow full-core MOX utilization.

ABB–CE fuel rods are similar to W/Fra/MHI 17 × 17 fuel rods in composition, physical dimensions, and operating conditions. Therefore, the existing PWR MOX fuel technology base is expected to be readily applicable to ABB–CE MOX fuel. ABB–CE has expressed a desire to purchase all of the necessary technology, data, and expertise from one or more of the European companies to support their supply of MOX fuel. ABB–CE currently has neither physics nor performance codes that are approved by NRC for MOX fuel. ABB–CE has not stated whether they would incorporate the European data into its existing (and approved) LEU codes or whether it would attempt to obtain NRC approval for the European codes and methods.

ABB–CE prepared a core design for WG plutonium disposition that included erbia as burnable absorber within the MOX fuel pellets. This core design was predicted to dispose of 1.587 MT of plutonium per reactor year in the System 80 plants at equilibrium. ABB–CE also produced core designs that rely on an erbia burnable absorber in LEU rods only. ABB–CE investigated four core designs differentiated by the number of MOX rods, LEU rods, and

LEU-erbia rods in the fuel assemblies. The plutonium disposition rates for these flexible cores were predicted to range between 0.91 and 0.98 MT per reactor year.

B&W. In 1971, B&W acquired ARCO-NUMEC as a wholly owned subsidiary. This purchase included the plutonium fuel fabrication facilities at Apollo, Pennsylvania. These facilities had been used previously for fabrication of MOX fuel for Saxton core II and for the GE plutonium subcritical reactor. No publications have been found that indicate commercial use of B&W MOX fuel. However, B&W did manufacture plutonium fuel for FFTF. Utilizing a similar process in the same facility, B&W manufactured test MOX fuel for inclusion in the Plutonia Fuel Study sponsored by the Electric Power Research Institute.[13] During the 1970s, B&W did investigate utilization of MOX fuel in its reactors. B&W's latest NSSS design, the Model 205, included the necessary design features to accommodate full-core MOX utilization. None of the units based on the Model 205 have been completed, but two units at Bellefonte remain under construction and may become available in the time frame of interest to this program.

B&W's fuel division has been purchased by a foreign consortium and is now known as Framatome Cogema Fuels (FCF). It is possible that FCF, through its corporate affiliates, has access to the French MOX fuel data held by Framatome and Cogema. This may provide FCF an advantage because no licensing of information or purchase of a database would be necessary to transfer the European data to a domestic fuel supplier. Because FCF has not been involved in the MD programs, knowledge about the status of its fuel design and performance codes is not available.

B&W fuel shares many similarities with other PWR fuel including the W/Fra/MHI 17 × 17 designs. Therefore, the existing PWR MOX fuel technology base is thought to be fully applicable to B&W fuel. Because B&W did not participate in either the Plutonium Disposition Study (PDS) or the follow-on MD program, no WG MOX fuel assembly or core designs have been developed by B&W. However, because of the similarities between the various PWR reactors and fuel designs, variations of the W and ABB–CE designs are expected to be appropriate for use in B&W reactors.

5.6.1.2 BWRs

BWRs are the second most prevalent reactor type in the world. BWRs share many similarities with PWRs and are therefore often lumped with PWRs into the generic LWR category. Important boiling-water NSSS suppliers include GE, Asea-Atom (AA), Toshiba/Hitachi, and KWU, which is now a part of Siemens Power Corporation. As with PWR fuel, BWR fuel suppliers compete to supply fuel bundles for their own and their competitors' reactor designs. BWR pellets are similar in most respects to PWR pellets, with the primary difference being the physical dimensions.

MOX fuel utilization has been widely demonstrated in BWRs. Because most BWR designs utilize large cruciform control blades that move in channels separate from the fuel assemblies, MOX fuel bundle designs for BWRs are typically different from those of PWRs. "Island designs," in which LEU rods surround an island of MOX fuel rods located in the center of the bundle, isolate the MOX rods from the control blades and preserve control blade worth. The lack of soluble boron in the BWR core removes one of the methods of compensating for lost control blade worth that is available to the PWR designer. However, the use of cruciform control blades separate from the fuel bundles compensates for this somewhat.

BWR MOX fuel utilization has also been limited to partial core loadings, although the means of achieving such loadings have differed. Through island designs, the entire core may be loaded with MOX assemblies, although the core-wide fraction of MOX rods may be as low as 10% to 20%. An additional advantage of the island design is that burnable absorber may be incorporated into the LEU rods surrounding the MOX rods to provide further reactivity control. All

MOX assemblies with graded plutonium concentration have been utilized in BWRs, but the plutonium concentrations allowable in the outer rows of the bundle must be kept low to limit power peaking.

GE. GE designed all of the domestic BWR NSSSs and continues to provide fuel to many of them. Siemens Power Corporation and ABB–CE also compete for domestic BWR fuel supply. GE investigated MOX fuel utilization in its domestic and foreign reactors during the 1960s and 1970s and has continued to participate in international MOX fuel activities to support its international customers. Most of the operating domestic BWRs are based on the three latest GE NSSS designs: the BWR 4, 5, and 6 share a common fuel design.

In addition to GE's own BWR MOX fuel technology base, much additional MOX fuel work has been done in BWRs in Germany, the Netherlands, and Sweden. All of this information should be readily applicable to fuel designs for GE BWRs. Furthermore, BWR fuel pellets are very similar to PWR fuel pellets, and thus much of the PWR technology base is applicable to GE reactors. The Japanese have announced a plutonium recycle policy that will utilize GE's Advanced BWR NSSS design rather than constructing more Advanced Thermal Reactors (ATRs).

GE, like the other fuel designers, would have to obtain NRC approval of its design and performance codes and methods prior to reloading MOX fuel domestically. Its international work indicates that the design and performance modeling capabilities still exist, but this has not been verified.

For the PDS, GE developed two conservative bundle designs: an island design and a full-MOX design utilizing gadolinia in the MOX pellets as an integral burnable absorber. Cores loaded with these fuel designs were predicted to dispose of 0.19 and 0.75 MT of plutonium per reactor year. As part of the follow-on work, GE further refined these earlier bundle designs into three distinct bundle/core designs: UO_2-alike, High-MOX, and Full-MOX. The UO_2-alike design optimizes the island design concept to achieve a predicted disposition rate of 0.43 MT of plutonium per reactor year. The High-MOX bundle design replaces all of the burnable absorber-free LEU rods with MOX to increase the predicted disposition rate to 0.83 MT of plutonium per reactor year. The Full-MOX design replaces these remaining LEU-gadolinia rods with MOX-gadolinia rods to increase the predicted disposition rate further to 1.5 MT of plutonium per reactor year.

5.6.1.3 PHWRs

PHWRs have never utilized MOX fuel on a commercial basis. However, extensive testing has been performed in test and/or demonstration reactors of this type. The prevalent PHWR design is the CANDU reactor designed by AECL. An alternative PHWR design that has been pursued in Japan since the early 1970s is the ATR designed jointly by Hitachi, MHI, Sumitomo Heavy Industries Ltd., and Fuji. The ATR was designed as a plutonium-burner. The only ATR is the Power Reactor & Nuclear Fuel Development Corporation's Fugen reactor, which has operated with MOX. The experimental Mehrzweckreaktor PHWR in Germany has also utilized MOX fuel successfully.

PHWRs share many similarities with PWRs. They utilize zirconium-alloy cladding to encapsulate pelletized oxide fuel. In that sense, much of the data for PWRs and BWRs apply to PHWRs and vice versa. However, the fuel duty in a PHWR is typically quite different from that experienced in an LWR. Because of their inherent neutron economy, PHWRs typically utilize low-enrichment (or low-plutonium content) fuel. CANDU reactors are operated with natural UO_2. The discharge burnup is typically much lower than that of LWR fuel, about 10,000 MWd/MT for natural uranium CANDU fuel. However, because of the use of larger rod diameters, the average linear heat generation rate in CANDU fuel is much higher than that experienced in BWRs and PWRs.

AECL. AECL and its utility partner, OH, have expressed an interest in disposing of surplus U.S. WG plutonium in the four Bruce A reactors located in Tiverton, Ontario. The four Bruce A units have been limited by AECB to 75% of their nominal capacity as a result of operating difficulties during 1993–1994. Unit 2 was mothballed in 1995 because of maintenance needs on the steam generators and pressure tubes, but the unit has not been permanently closed and could be restarted after refurbishment. Unit 1 is scheduled to be mothballed in 2000 because of degradation of its pressure tubes.[14] Four additional units on the same site, collectively known as Bruce B, have also been discussed for MOX burning.

Because MOX fuel has not been utilized commercially in CANDU reactors, the qualification of CANDU MOX may be more difficult than qualification of LWR MOX. However, AECL and OH have developed a MOX fuel qualification strategy in conjunction with AECB. This strategy, described in Sect. 3.2.5, outlines the additional tests and demonstrations necessary to fulfill remaining data needs.

AECL developed two MOX fuel bundle designs for WG plutonium disposition as part of the 1994 PDS—one based on the current reference 37-element bundle and a second based on the advanced 43-element CANFLEX design. The predicted plutonium disposition rates for these two designs are 1.05 and 0.98 MT of plutonium per reactor year, respectively. The advanced fuel design actually lowered the plutonium disposition rate per reactor year by increasing the discharge burnup achieved in the fuel. Two fuel designs developed for the current FMDP increased the plutonium content per bundle, increasing the projected plutonium disposition rates to 1.45 MT of plutonium per reactor year with 37-element bundles and 1.2 MT of plutonium per reactor year with the 43-element CANFLEX bundles.

5.6.2 Ongoing R&D Activities

Because of the extensive experience base supporting RG MOX fuel utilization in thermal spectrum reactors, the ongoing experimental activities have been limited to investigating the differences between commercial RG MOX fuel and the WG MOX fuel as it is currently envisioned. These differences are primarily limited to isotopics, powder characteristics due to dry plutonium processing, and impurity content. A series of out-of-pile corrosion tests is under way to investigate gallium corrosion on prototypic cladding samples. A generic LWR irradiation test is being conducted to investigate the stability of residual gallium in MOX fuel pellets during irradiation. This materials irradiation is meant to extend the results of the out-of-pile tests to the in-pile conditions. A CANDU MOX fuel irradiation is scheduled for insertion in the NRU reactor. Finally, ORNL is participating in the ARIANE program, which is designed to benchmark and validate the ORIGEN computer code.

The out-of-pile corrosion experiments will subject prototypic cladding samples from the vendors to gallium and gallium-containing compounds in progressively more prototypic conditions, including sintered generic MOX fuel pellets. The tests will be conducted over a wide range of temperatures, encompassing all anticipated operating conditions and for extended exposure periods. Corrosion mechanisms will be investigated under both stressed and unstressed conditions. The tests will provide preliminary indication of the potential for gallium interactions with zirconium alloy cladding.

The LWR irradiation test has been developed as an integral part of the gallium investigation that is meant to extend the out-of-pile results to an in-pile situation. Gallium-containing MOX materials will be irradiated along with similar materials without gallium. The tests are designed to investigate the stability of residual gallium under irradiation and to investigate zirconium alloy corrosion due to the combined effects of irradiation, gallium, and fission products.

The Parallex CANDU MOX fuel irradiation will provide an additional benchmark for in-pile gallium behavior. Parallex will also provide a demonstration of the feasibility of WG MOX fuel

use in CANDU reactors. Through testing of two different plutonium homogeneities, it may also provide a basis for the homogeneity specification to be required on mission CANDU MOX fuel.

ARIANE is a cooperative, international program managed by Belgonucleaire. It is designed to benchmark and validate the ORIGEN computer code for calculating isotopic compositions of MOX fuel for a range of discharge burnups. Commercial LEU and MOX fuels irradiated in power reactors are being dissolved and analyzed for isotopic composition. Based on the recorded operational history of the fuel, calculated predictions of isotopic composition are being compared to the measured values.

5.6.3 Additional R&D Needs

Following successful completion of the out-of-pile and in-pile tests currently under way, additional testing in the form of LTAs will likely be required to demonstrate the acceptability of residual gallium under prototypic conditions prior to large-scale implementation. Additionally, other impurities in the surplus weapons materials may not be removed sufficiently through dry processing, resulting in the need for additional out-of-pile testing. Under the best-case scenario, all of the impurities would be addressed in an aggregate manner through a single LTA program.

Other than impurities, some additional issues need to be addressed before large-scale implementation. First, the codes and methods used for physics analyses and irradiation performance predictions must be validated, benchmarked, and approved by the NRC. Licensing and use of commercial MOX fuel in Europe has been accompanied and supported by the development of an extensive experimental database that includes the validation and benchmarking of a group of neutronics, fuel performance, fuel-cycle management, thermal-hydraulic, and accident analysis methods and computer codes. These data and codes are proprietary and thus have not been independently reviewed for their applicability in the United States for licensing of surplus WG plutonium MOX fuel. Ultimately, the applicability and acceptability of these data and codes in the U.S. licensing process will be known only after the selected fuel vendor and the utility evaluate the data and present them to NRC for acceptance.

It is quite likely that the European database and methods will, with minor modifications, be suitable to support U.S. licensing of surplus WG plutonium MOX fuel. However, it appears that few, if any, MOX fuel postirradiation melting (fission product release) tests have been performed to date. Even though not required by NRC because it was not required as part of the original licensing basis, it may be desirable to perform a limited number of these severe fuel damage tests to confirm that the source terms resulting from MOX-fueled reactors are not substantially different than those from LEU-fueled reactors. This is not viewed (from the programmatic standpoint) to be a significant schedule or cost issue.

5.6.4 Issues

One of the more difficult issues to resolve is determining which, if any, of the NSSS designs has an advantage in terms of ease of implementation. A minimum database is necessary to qualify fuel; additional information is useful to reduce uncertainty and increase confidence in the product but otherwise is unimportant. Until some access to the proprietary databases is obtained, a fair ranking of the NSSS vendors by ease with which MOX fuel can be implemented in their designs cannot be done.

A related issue is determination of how many fuel types should be made. Theoretically, if all of the fuel designers agreed to a single MOX fuel pellet specification such that their pellets differed only in physical dimensions, almost no penalty would be imposed by serving more than one reactor type. However, in reality, each fuel designer will propose a unique pellet specification.

While difficult to quantify, a clear advantage is obtained by designing, fabricating, and obtaining regulatory approval to load only a single fuel design.

As described in the vendor-specific discussions of Sect. 5.6.1, several fuel assembly designs have been prepared for WG MOX disposition. A number of these included burnable absorber in the MOX pins. Incorporation of burnable absorber into MOX fuel has only been demonstrated in a few test rod irradiations. Significant effort may be required to implement these burnable absorber fuel designs. However, burnable absorber MOX can be developed off the critical path through LTAs included in routine MOX fuel reloads. Assessment of the need for increased throughput should be made once the mission is started with lower throughput fuel designs. If sufficient justification for burnable absorber MOX exists, it can be developed at that time. Other methods of increasing the throughput, such as additional separate absorbers, are available and could also be implemented later in the mission if desired.

6. IMPLEMENTATION STRATEGY

The optimum FMDP reactor-based disposition implementation strategy would include an early mission start, minimum total project cost, low technical and regulatory risk, and a minimum of fuel cycle changes/disruptions to complete the mission in a timely and cost-effective manner. The timely initiation and completion of the mission identified as key criteria in the ROD of January 14, 1997, can be achieved by reducing the technology development requirements. Furthermore, the technical risk and total mission cost can be greatly reduced through reliance on proven technologies.

Two important differences between the PuO_2 produced from surplus weapons components by the ARIES-HYDOX process and the RG recycle PuO_2 used in commercial MOX may affect fuel fabrication or fuel performance:

1. The powder morphology is different. The authors do not believe that this difference, which may affect powder blending, pressing, sintering, and other fabrication parameters, has been addressed by the MOX fabricators. However, because all of the leading commercial MOX fabrication processes mill the plutonia, this difference may or may not prove to be significant.
2. Only minimal impurity removal takes place during ARIES processing. Thus, some impurities may remain in the PuO_2 at concentrations above the experience base. While gallium is the best known of these impurities, others may exist. A dry thermal process (TIGR) under development at LANL is expected to decrease the gallium concentration, but the final acceptable and reproducible concentrations of gallium and other impurities in sintered MOX pellets have yet to be determined.

The manner in which these differences are incorporated into the implementation path forms the basis for (1) a minimal technical risk implementation strategy (MTRIS) described in Sect. 6.1; (2) a preferred or recommended implementation strategy (RIS) that acknowledges certain constraints and ground rules imposed on the mission, as described in Sect. 6.2 ; and (3) alternative implementation paths in which variations to the constraints and ground rules are considered, as described in Sect. 6.3.

6.1 MTRIS

Technical risk is incurred by adopting an implementation strategy based on demonstrating negligible consequences of these differences between surplus WG plutonium MOX and commercial RG MOX. This technical risk can be eliminated by choosing a strategy that remains within the experience base—that is aqueous purification and oxide production through oxalate precipitation/calcination. Thus, MTRIS incorporates these options for purification and oxide production along with the minimum risk technology for the remainder of the processes described in Chap. 5. In particular, the MTRIS consists of the following processing options shown graphically in Fig. 3:

1. Pit disassembly via ARIES in a new domestic facility.
2. Plutonium purification via nitric acid dissolution, followed by solvent extraction and/or ion exchange.
3. Conversion to sinterable PuO_2 via oxalate precipitation/calcination.
4. Fabrication of both LTA/LUA and mission MOX fuel using proven European commercial technology in one of the existing European fuel fabrication facilities.
5. Irradiation of "LEU look-alike" assemblies at the self-generated recycle (SGR) level of ~30%.

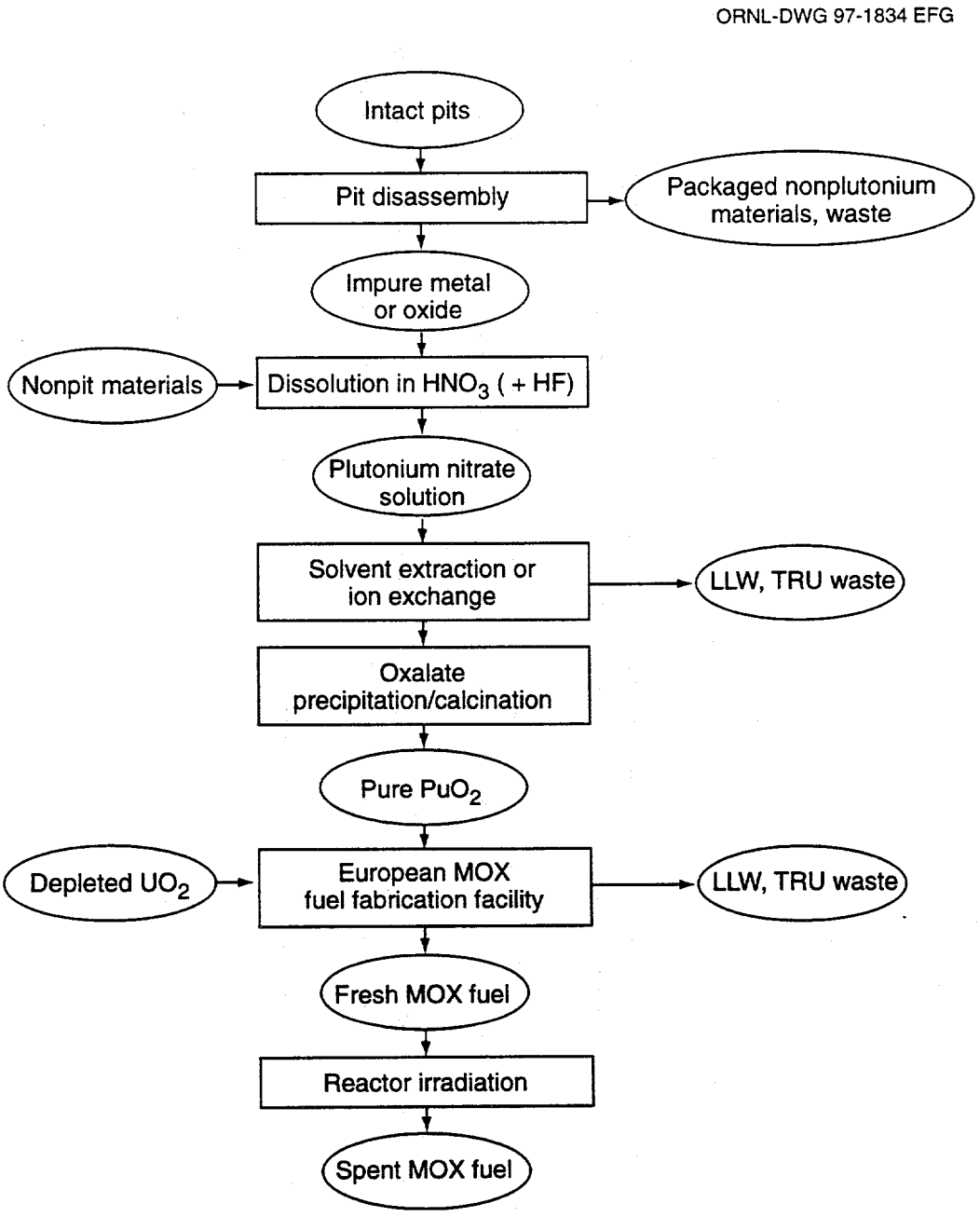

Fig. 3. MTRIS processing flow sheet.

By remaining within the experience base, uncertainties in process implementation and licensing are minimized. Delineation of this provides a basis for the technical risks and/or uncertainties associated with the recommended implementation path in Sect. 6.2. The details of MTRIS are provided for each of the processes.

Figure 4 contains the implementation schedule associated with MTRIS. Because all fabrication is performed in Europe using proven technology, no development activities are required. Remaining activities on the critical path schedule include (1) providing capacity for plutonium dissolution, purification, and oxalate precipitation/calcination; (2) licensing LUA and mission MOX fuel use based on European data; and (3) providing international transportation of PuO_2 and MOX LUAs. PuO_2 will be produced in existing and/or modified aqueous processing facilities within the DOE complex. Three years are allocated for licensing of the MOX LUAs; this is expected to be sufficient because of the reliance on proven European technology and data.

6.1.1 Pit Disassembly

ARIES has been selected by FMDP as the reference process for pit disassembly. It is therefore incorporated into this MTRIS for separation of the plutonium from the remainder of the weapons components for those pits to which it is applicable. However, under this MTRIS, no specification is attached to the plutonium product from the ARIES process with the exception of its solubility. Subsequent nitric acid dissolution, purification, and conversion to oxide will be used to obtain plutonia powder characteristics similar to those utilized in current commercial MOX fuel fabrication. An ARIES demonstration line is under construction at LANL. Operation of this facility is expected to provide a portion of the plutonium feed necessary for fabrication of LUAs and early mission fuel. This demonstration line may continue to disassemble pits until the PDAC facility is operational in 2006 or beyond.

The surplus plutonium currently exists in a number of forms other than ARIES-compatible pits. If the ARIES demonstration process line were unable to provide the quantity of plutonium product needed to implement LUA and initial reload irradiation, this alternative form material would be processed early in the mission implementation to provide additional plutonium product. The same facilities that would be utilized for aqueous purification and oxalate conversion to oxide would be used to process this material directly.

6.1.2 Conversion to Oxide

The most refined and demonstrated technique for conversion to sinterable oxide is the oxalate process described in Sect. 5.2. It will be used to convert all of the WG plutonium. Existing and/or modified facilities capable of aqueous purification can be used for this conversion or a line could be included with the PDAC facility. Aqueous processing lines in Europe could also be utilized for the purification of PuO_2 if desired.

6.1.3 Plutonium Purification

Aqueous processing has been demonstrated in both defense and commercial facilities around the world for purification of plutonium nitrate solutions. Solvent extraction, anion exchange, and cation exchange have been utilized successfully for plutonium purification. The plutonium product from ARIES as well as the other surplus plutonium materials will be dissolved in nitric acid, purified through one of these aqueous processes, and prepared for oxalate precipitation. Existing and/or modified facilities at LANL and SRS could be utilized for aqueous purification.

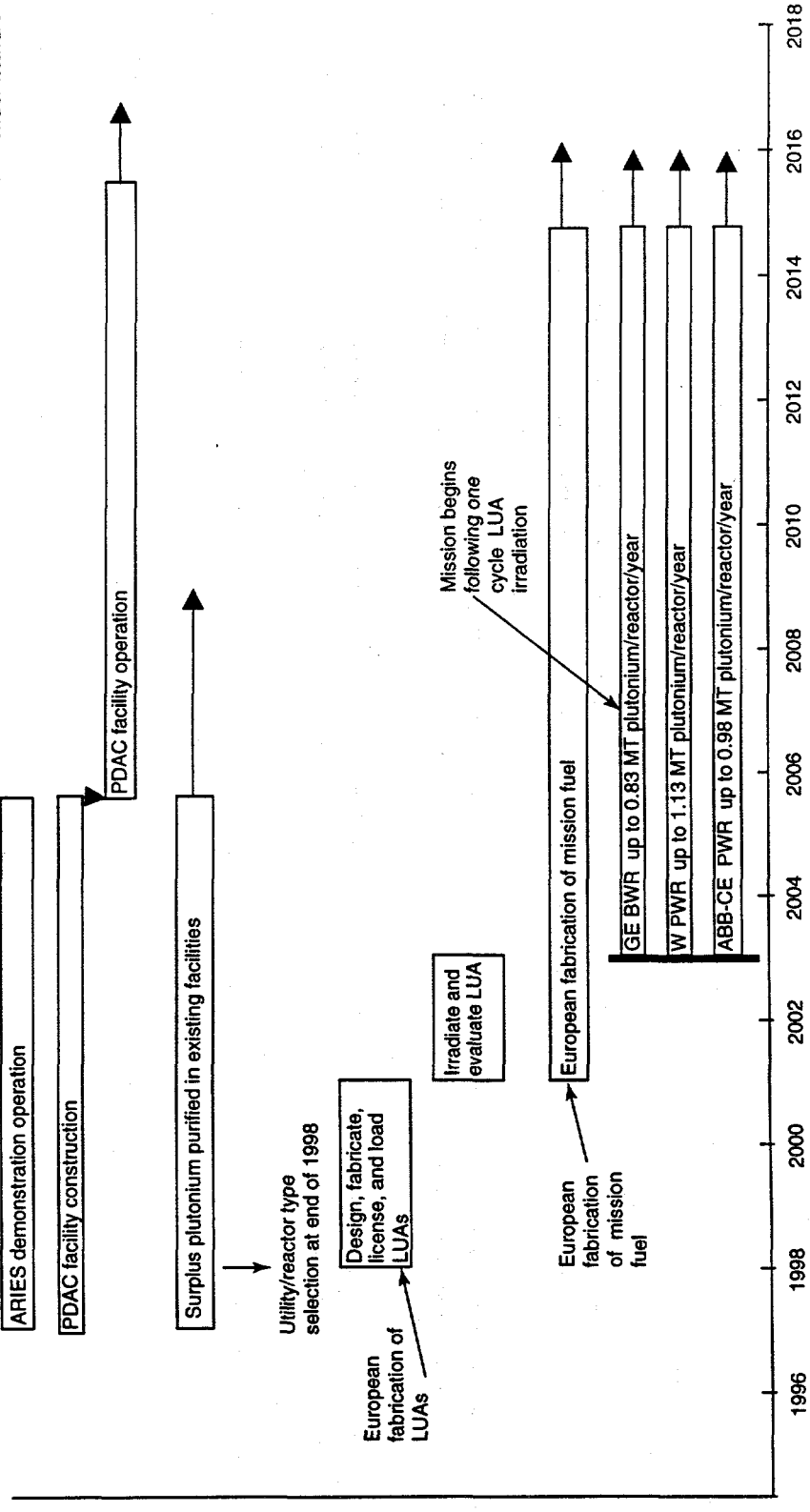

Fig. 4. MTRIS with European fabrication of LUAs and mission fuel.

6.1.4 Plutonia Powder Conditioning

Because oxalate conversion would be utilized to convert the purified plutonium nitrate solution to PuO_2 under this MTRIS, no powder conditioning would be required. The oxalate-derived PuO_2 would be utilized directly in the MOX fuel fabrication facility.

6.1.5 MOX Fuel Fabrication

To minimize the technical risks associated with the transition from LEU to MOX fuel, one of the existing commercial MOX fuel fabrication facilities in Europe (P0, Cadarache, MELOX, MDF, or SMP when completed) would be utilized to produce the LUAs for irradiation in one or more domestic CLWRs. Use of an existing MOX fuel fabrication facility would minimize the delay in mission implementation by removing construction of a new domestic MOX fuel fabrication facility from the critical path. According to recent estimates, additional MOX fuel fabrication capacity may be required to complete the mission with only European facilities.[15] Although use of a European fuel fabrication facility would require overseas transport of WG plutonium and return of MOX fuel assemblies, commercial technologies exist for completing such transport.

Under MTRIS, fuel fabrication of all of the mission fuel would continue in the facility utilized for LUA fabrication. Once the barriers to overseas transport and MOX fuel fabrication were overcome through the LUA program, there would be no *technical* incentive to transition to an alternate (domestic) MOX fuel fabrication facility. Fabrication of mission MOX fuel would begin immediately upon completion of the LUA fabrication campaign.

6.1.6 Reactor Irradiation

MTRIS remains within the experience base for MOX fuel utilization. Fuel assemblies externally equivalent to LEU assemblies would be utilized at the level of SGR, ~30% of the core. These LEU look-alike assemblies are designed to be roughly equivalent to the surrounding LEU assemblies in terms of energy content and reactivity swing during irradiation. Furthermore, these LEU look-alike assemblies include graded plutonium content to reduce power peaking at the interface with surrounding LEU assemblies. LUAs would be loaded one cycle in advance of the mission fuel loading to enhance regulatory and utility acceptance. The first mission fuel loading would consist of about one-third of the reload fuel such that the SGR level would be approached gradually. No differences in the LUAs and mission assemblies are anticipated because the designs would rely on proven technology. The specifics of the assembly design would, of course, depend on the type(s) of mission reactors eventually chosen through the procurement process.

To minimize the technical risk, fuel management would remain at the SGR level of plutonium throughput during the mission. More aggressive fuel management strategies would only be implemented if the plutonium disposition schedule demanded it or if clear economic or technical incentives existed. These higher plutonium throughput designs would be proven through one or more LA programs conducted as part of the mission reloads such that no schedule impact would be experienced.

6.1.7 Contingencies

The MTRIS has flexibility to adjust the implementation strategy for potential variations in the desired plutonium disposal rate. The use of an LEU-compatible fuel assembly design provides wide variation in the potential disposal rates below some as-yet-undefined maximum value. To reduce the plutonium disposal rate, fewer MOX assemblies would be loaded at each shutdown with the excess simply remaining in storage pending future needs. Because the LEU-compatible

fuel assembly can be exchanged on an equal basis with an LEU assembly, one must merely have the required number of LEU assemblies available for reload to adjust the plutonium disposal rate downward.

6.1.8 R&D Needs

The only new technology in MTRIS is the ARIES pit disassembly. The R&D to implement this process will be required; however, this strategy imposes no specific quality requirements on the ARIES product because it will be subsequently dissolved.

6.2 RIS

Any recommendation is necessarily dependent on the assumptions and ground rules in effect at the time of the recommendation. This RIS is no exception. Furthermore, the constraints imposed on FMDP are anticipated to change during mission implementation as a result of policy decisions, technical developments, and other factors. No attempt has been made during development of this RIS to forecast this evolution in constraints, although the effects of such changes are discussed in Sect. 6.3. The constraints considered in development of this RIS include the following:

1. Timeliness. The mission is to begin within 10 years of ROD (i.e., 1/2007) and be completed within 25 years of ROD (i.e., 1/2022).
2. Dry plutonium processing. ARIES has been selected as the reference technology for pit disassembly and will include HYDOX conversion to PuO_2. Plutonium purification will be accomplished through TIGR. Other processes may need to be developed to remove impurities other than gallium.
3. Backup aqueous purification. If the dry conversion and purification prove inadequate to meet the requirements of either the fuel fabricators or fuel designers, aqueous purification and oxalate conversion to oxide will be utilized.
4. Limited European fuel fabrication. The lead assemblies and a limited quantity of early mission fuel may be fabricated in one of the existing European facilities to expedite mission implementation.
5. Domestic fuel fabrication. The bulk of the mission fuel will be produced in a new facility dedicated to FMDP.

Based on these constraints, this RIS has been developed from the processing technology options described in Sect. 5. The RIS, depicted graphically in Fig. 5, includes the following process options:

1. Pit disassembly via ARIES, initially in a demonstration line and later in a new domestic facility.
2. Conversion to PuO_2 via the HYDOX process as part of ARIES.
3. Plutonium purification via dry processes including TIGR.
4. Mechanical powder conditioning to produce sinterable PuO_2.
5. Fabrication of LTA/LUA and early mission MOX fuel using proven European commercial technology in one of the existing European fuel fabrication facilities. Mission fuel will be fabricated in a new domestic facility once it becomes operational.
6. Irradiation of LEU look-alike assemblies at the SGR level of ~30% MOX core loading for the first few cycles. Transition to higher throughput core designs and potentially higher impurity levels will occur gradually through LA programs conducted as part of the mission.

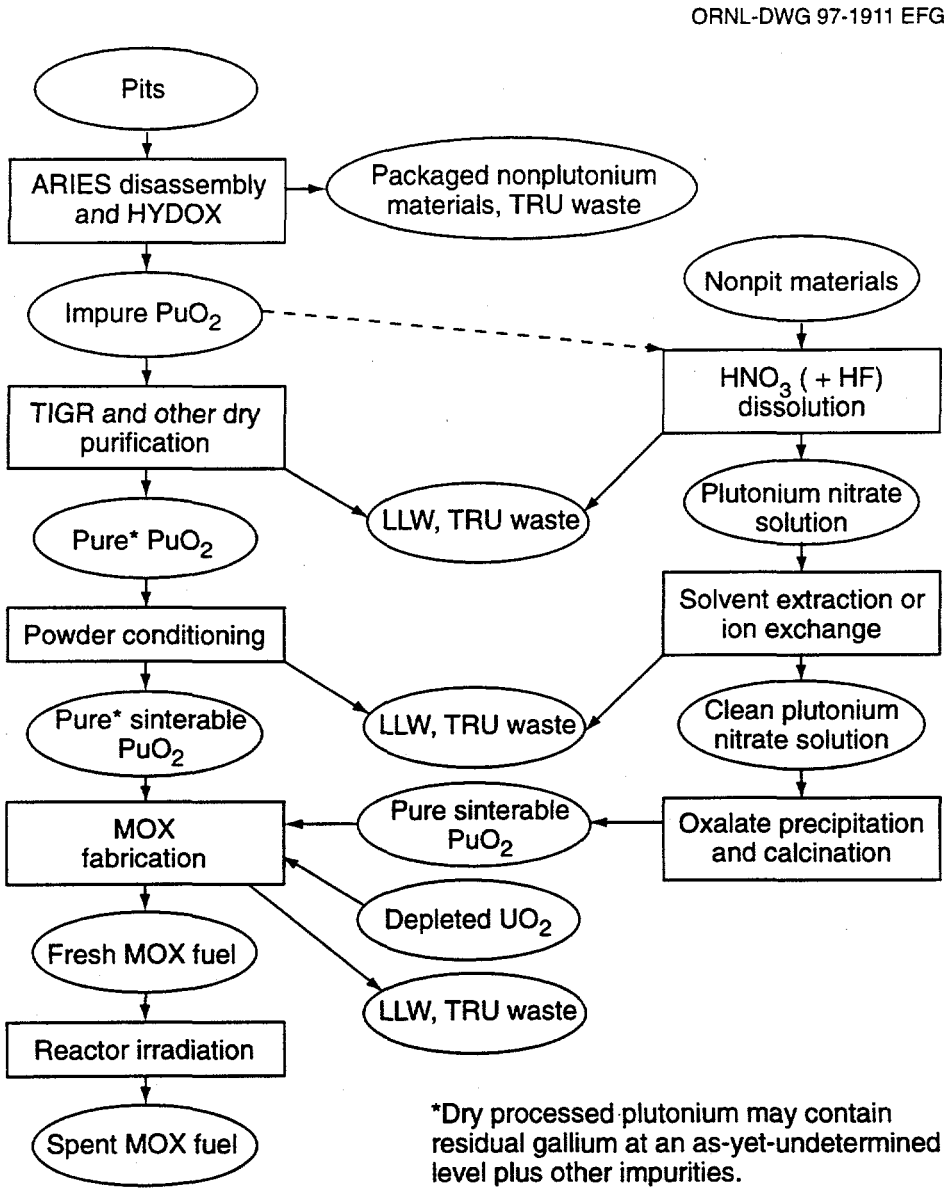

Fig. 5. RIS processing flow sheet.

Important differences between this RIS and MTRIS include the reliance on dry plutonium conversion/purification and the construction/operation of a new domestic MOX fuel fabrication facility. The schedule impacts of these changes are shown in Fig. 6. Only a slight schedule penalty is incurred relative to MTRIS because *it is assumed that the dry powder is determined by the European fabricators to be acceptable in terms of impurity concentrations and powder morphology.* Additional schedule penalty could be incurred if this assumption is not valid. If sufficient plutonium inventories are not available to produce MOX that meets impurity levels within the commercial MOX experience base, then aqueous purification could be used to avoid impurity issues during the initial stages of the mission.

The backup for the ARIES process, aqueous dissolution, is proven technology and requires no R&D.

6.2.1 Pit Disassembly

The reference process for pit disassembly, ARIES, is included in this RIS. Initial disassembly will focus on the higher purity pits in the surplus inventory to minimize concerns about impurities in the initial fuel cycles. As the dry purification processes are developed sufficiently, a transition to pits containing higher impurity concentrations will occur. The ARIES demonstration line at TA-55 will be utilized to process pits until the PDAC facility becomes operational.

6.2.2 Conversion to Oxide

As part of ARIES, the separated plutonium will be converted to oxide through the HYDOX process. HYDOX capability will be included in both the demonstration line and the PDAC facility such that a uniform product can be obtained from both facilities.

A portion of the nonpit surplus plutonium exists as "clean metal." This material will also be converted to oxide via the HYDOX process either in the demonstration line or in the PDAC facility.

6.2.3 Plutonium Purification

It is assumed that dry processes can be developed to remove all the potential impurities, including gallium, to an acceptable level. The dry purification processes will be included as part of the demonstration line and the PDAC facility.

If satisfactory product cannot be obtained with the dry purification processes, aqueous purification will be utilized. Furthermore, some of the nonpit surplus material is known to contain impurities at concentrations above those currently allowed in feed for MOX fuel fabrication. These impure materials would have to be purified through solvent extraction or ion exchange if they were made into MOX fuel. Existing and/or modified facilities exist in the DOE complex for aqueous plutonium purification. If the entire inventory were processed, these facilities could be used to avoid difficulties with the dry processes.

6.2.4 Plutonia Powder Conditioning

Mechanical conditioning is included in this RIS as part of the ARIES process in both the demonstration line and the mission PDAC facility. This conditioning is required to obtain a satisfactory particle size and surface area in the hydride-derived PuO_2. Currently, the only conditioning process that has been identified is powder milling. However, blending of various feed streams may also be included to eliminate differences in the feed streams. A supply of "clean oxide" exists in the surplus inventory. Assuming that this material does not require purification

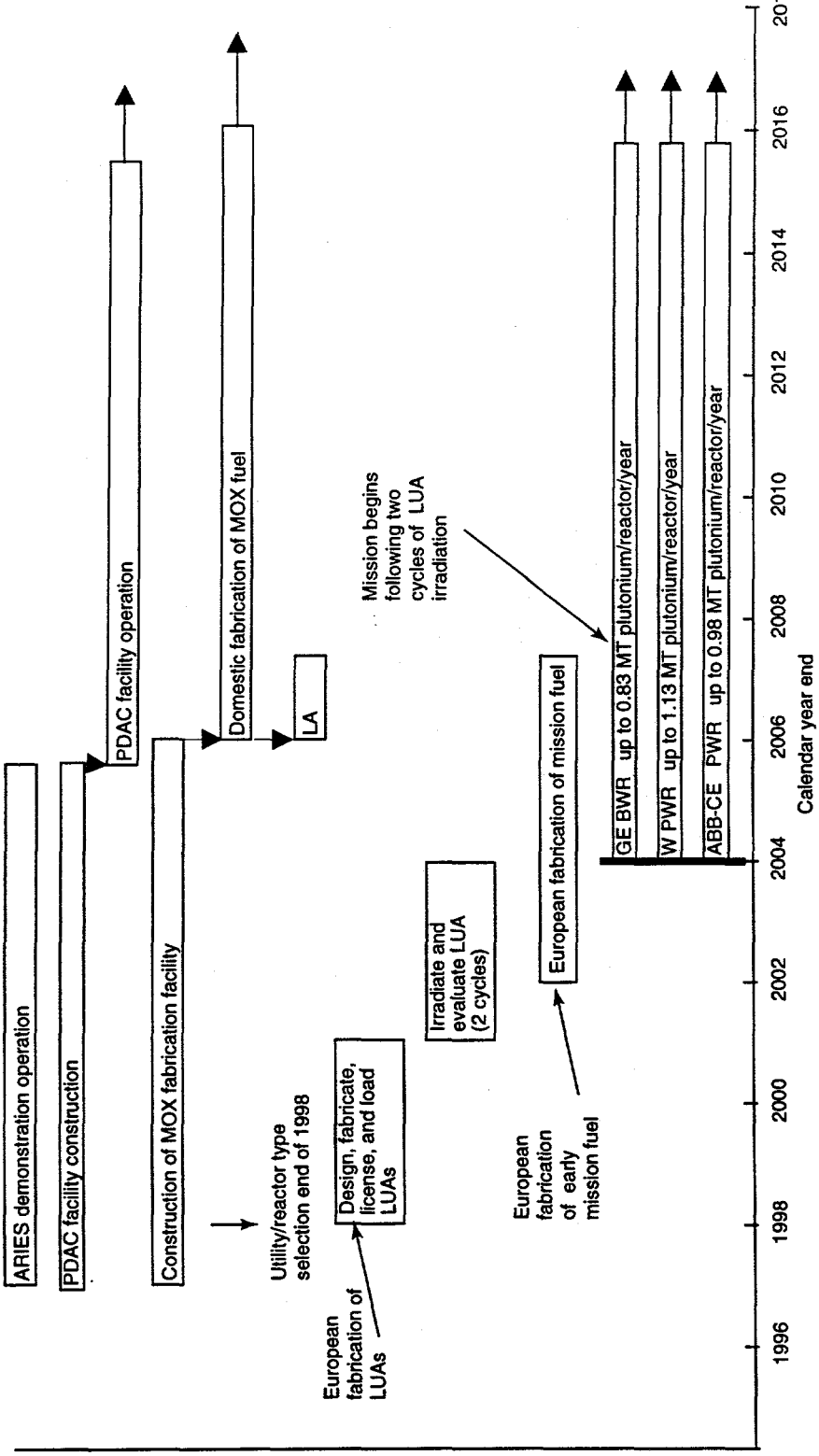

Fig. 6. RIS with European fabrication of LUAs and early mission fuel.

before fuel fabrication, it will be conditioned as necessary along with the hydride-derived material. No mechanical conditioning would be required for oxalate-derived material.

6.2.5 Fuel Fabrication

The LUAs and some early mission fuel would be fabricated in the existing European facility of the MOX fabricator in the selected consortium using the proven fabrication process. Process development and modification may be required to adjust to the unique powder morphology of the hydride-derived PuO_2. If the hydride-derived PuO_2 cannot be shown to be compatible with the chosen fabricator's process, aqueous purification and oxalate conversion would be used to implement the mission. Furthermore, if the United States were to insist on supplying its own depleted uranium rather than allowing the fabricator to use its normal suppliers, additional fabrication development would almost certainly be required.

The majority of the mission fuel would be fabricated in the new U.S. facility. The design of this domestic facility would be based on the same European process and technology utilized in the MOX fuel fabrication facility that produces the LUAs.

6.2.6 Reactor Fuel Management Strategy

As in MTRIS, the RIS would begin the mission with LEU equivalent assemblies loaded incrementally up to the SGR level of about 30% of the core. The number of reactors loaded and the plutonium throughput per reactor would be increased gradually as necessary. All changes in the implementation would be conducted gradually through introduction of appropriate LTAs/LUAs as part of the MOX fuel reloads such that no schedule impact is incurred. These changes may include (1) increasing the allowable impurity concentrations in the MOX fuel to reduce the purification requirements, (2) introducing fuel fabricated from hydride-derived powder if oxalate-derived powder is required by the Europeans, and (3) increasing the plutonium content in each MOX assembly and/or increasing the number of MOX assemblies charged.

An additional aspect of RIS is the gradual increase in average discharge burnup as confidence in the fuel is developed through irradiation experience. The initial MOX fuel assemblies would be irradiated only to an assembly average discharge burnup of 20 to 30 GWd/MT. As confidence in the fuel increases, this discharge burnup would be increased to the 40- to 50-GWd/MT assembly average commonly achieved in LEU assemblies, if this proves to be desirable. While increasing the number of assemblies in the core would accelerate the disposition schedule, increasing the discharge burnup would primarily affect fuel cycle economics while stretching the disposition schedule. However, the utility may require additional compensation if discharge burnup is not consistent with current LEU practice because increased operating costs are typically incurred with shorter cycles.

6.2.7 Contingencies

Two assumptions are implicit in the development of this RIS: (1) a sufficient quantity of higher purity plutonium is available to supply the LUA and initial core reload requirements with MOX fuel that has impurity levels within the experience base and (2) the fabrication process utilized by the fabricator chosen through the procurement process is adaptable to the dry-process PuO_2 feed material. If either of these assumptions is not met, aqueous processing could be utilized to produce a high-purity, oxalate-derived powder that would be acceptable. However, such a development would require utilization of additional facilities.

The main justification for basing the design of the new domestic MOX fuel fabrication facility on a proven European technology is the acceleration of the implementation schedule that is

possible by relying on the experience bases for fabrication and irradiation developed in Europe. Obtaining LAs and early mission fuel from the European facility also removes completion of the domestic fuel fabrication facility from the critical path for mission implementation. An important implication of reliance on European technology is that the proven processing parameters and fuel specifications from the European technology would be adopted for the domestic facility, at least initially. A second implication is that the European fuel performance codes and/or the supporting data bases would also be obtained to expedite licensing domestically. The adequacy and sufficiency of the fabrication and fuel performance data bases for licensing cannot be evaluated at this time because the data bases are almost exclusively proprietary and unavailable to the FMDP. Because the new domestic MOX fuel fabrication facility would be qualified to meet the same fuel pellet, rod, and assembly specifications applied to the European facility, the initial fuel assemblies produced in the new domestic MOX fuel fabrication facility could be incorporated directly into a reload batch. For licensing purposes, the fuel design is the approved item, and the fabrication facility must only be qualified to meet the appropriate specification. However, it is probable that a utility would require an LUA irradiation prior to large-scale use of fuel from a new facility.

6.2.8 Facility Needs

Because this RIS includes some unproved technologies, it would be prudent to maintain some development and testing capabilities in case difficulties are encountered during implementation. Most of the facilities would be available as part of the implementation. The operational period for others would be extended. In particular, it is recommended that processing capability be maintained for laboratory-scale pit disassembly, aqueous plutonium purification, test fuel fabrication, test reactor irradiation, and hot-cell irradiated fuel examination.

Pit disassembly. The TA-55 ARIES demonstration line for pit disassembly, conversion to oxide, dry plutonium purification, and oxide conditioning should be held in standby or operated continuously. This would provide continued ability to process unusual pit types and nonpit materials. Furthermore, it would provide some backup processing capacity in the event that difficulties were encountered with startup or operation of the PDAC facility. Finally, this demonstration line could be used to test new dry process advancements.

Aqueous purification. The dry conversion and purification processes are still under development and as yet unproved. Aqueous purification and oxalate conversion may therefore be required if the dry processes cannot be shown to produce a satisfactory product. Thus, retention of existing aqueous processing capability and/or construction of new capacity should be considered. The facilities at TA-55 and/or SRS should be maintained and possibly modified as necessary for insurance against potential schedule delays if difficulties with the dry processing are encountered during implementation. This capacity could also be utilized for purification of some of the impure materials to provide additional plutonium feed for the early fabrication campaigns as necessary. If an aqueous purification capacity were to be included in the new domestic MOX fuel fabrication facility or PDAC facility, retention of existing capacity would only be needed until the startup date of this new facility.

MOX fabrication. Because changes in feed materials are envisioned during the implementation, it is recommended that an R&D fabrication capability be developed/maintained as part of the mission fuel fabrication facility.

Reactor test facilities. Although no additional test reactor irradiations beyond the current demonstrations are proposed as part of this RIS, they may be required by one of the stakeholders, especially if unanticipated difficulties are experienced with the LTAs/LUAs. Test reactors are available worldwide for fuel irradiation. However, only two suitable facilities exist domestically: the High Flux Isotope Reactor at ORNL and the Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory. If overseas irradiations are disallowed by policy, one

of these two domestic reactors should be maintained until the mission has progressed through several reloads.

For CANDU fuel testing, the NRU reactor at Chalk River Laboratories (CRL) is optimum. However, this reactor is scheduled to be replaced in the 2005 time frame. Any replacement would be expected to contain equivalent testing capability.

Hot-cell PIE facilities. NRC qualification of PIE facilities might be necessary because none are currently qualified to generate licensing data for the NRC on LWR fuel. Suitable domestic hot-cell facilities currently exist at Argonne National Laboratory (ANL), ORNL, and Pacific Northwest National Laboratory (PNNL). If domestic PIE is required, capability should be maintained at one of these facilities.

For CANDU fuels, existing hot cells at CRL would be utilized. These facilities could also be used for examination of LWR fuel if necessary. Furthermore, suitable hot-cell facilities exist at a number of foreign research centers in Europe and Japan.

6.2.9 R&D Needs

The RIS is based on technologies described in detail in Chap. 5. R&D will be required for the following processes:

1. the ARIES pit disassembly process,
2. HYDOX conversion to oxide,
3. dry process PuO_2 purification including TIGR,
4. mechanical conditioning of hydride-derived powder, and
5. MOX fuel fabrication to accommodate the unique powder morphology of hydride-derived PuO_2.

6.3 ISSUES AND ALTERNATE OR BACKUP IMPLEMENTATION PATHS

The January 14, 1997, ROD concludes that the principal uncertainty with respect to using excess weapons plutonium as MOX fuel in CLWRs relates to the potential difficulty of gaining political and regulatory approval for the various operations and facilities. However, it also states that (1) significant schedule uncertainties exist, relating to both engineering and institutional factors and (2) opportunities for compressing or expanding schedules exist.

The RIS outlined in Sect. 6.2 was developed in accordance with certain constraints that are likely to change during mission implementation. While the evolution in constraints is impossible to predict, certain changes can be anticipated and analyzed for their potential impacts. It is therefore prudent to identify alternate/backup strategies for mission implementation that correspond to changes in the constraints. The remainder of this section discusses individual changes to the constraints imposed on the mission and the effects these changes have on the recommendations.

6.3.1 Introduction of Impurities in Initial Fuel

The impurity concentrations in dry-processed PuO_2 may fall outside the experience base. The RIS avoids this issue by processing higher purity pits early in the mission such that the potential impurities and dry purification techniques can be addressed off the critical implementation path. However, it would be possible to address the impurities issue at the start of mission implementation such that no additional development would be required later. Under this scenario, the LTAs and early mission fuel would be fabricated from PuO_2 derived from pits containing the maximum concentration of gallium and other potential impurities. Thus, the worst-case impurity concentrations would be addressed during mission implementation rather than later in the mission.

The only difference between this implementation strategy and RIS is that LTAs would be subjected to greater scrutiny prior to the insertion of the first MOX reload. The schedule penalty of about 1-1/2 years beyond RIS associated with this scrutiny is shown in Fig. 7. The processing flow sheet would be similar to that for RIS with the deletion of aqueous processing capability as shown in Fig. 8. It is assumed that the LTA program would include rod removals at the end of each of four cycles such that performance can be demonstrated to higher discharge burnup than expected for the mission fuel. The withdrawn rods would be subjected to destructive examination following a cooling period. PIE data from the first-cycle rods would be used along with the pool-side evaluations after two and three cycles to support the first mission reload. Hot-cell data from the higher discharge burnup LTA rods would be used to support NRC approval of additional irradiation cycles for the mission fuel.

Advantages. An advantage of this implementation strategy is the elimination of continued development after mission start. LTAs that would be introduced along with routine MOX fuel reloads under RIS would be eliminated under this approach. This would potentially reduce the overall mission cost, but sufficiently detailed cost estimates are not available to allow a quantification of any potential savings.

Disadvantages. This implementation strategy includes more technical risk than RIS. The maximum impurity concentrations may be unacceptable from a fuel fabrication and/or fuel performance standpoint. If difficulties were encountered with the LTAs containing maximum impurities, potentially large schedule penalties would be incurred. Furthermore, because mission fuel would be prefabricated before completion of the LTA campaign under this strategy, the potential exists to have a substantial portion of fuel fabricated that is unsuitable for reactor use.

Even if satisfactory results are obtained from the LTA tests with maximum impurities, institutional risk exists. The fuel fabricator may not accept the higher impurity levels in existing facilities for fear of contaminating the equipment. The fuel designer, fabricator, utility, or regulator may require additional testing beyond the one-cycle LTA data that are assumed to be available prior to the loading of the first MOX reload. Delays would be incurred awaiting hot-cell results for higher burnup LTA fuel.

6.3.2 Foreign MOX Fuel Fabrication

As described in Sect. 5.5, there is currently no commercial MOX fabrication capability in the United States. Therefore, RIS utilizes available European fabrication capacity to accelerate the implementation schedule, allowing LA irradiation and limited burning of mission fuel prior to the availability of the domestic MOX fabrication plant (end of 2006). This reliance on European fabrication capacity may become untenable for a number of reasons: adoption of a DOE policy prohibiting international transportation of the surplus WG plutonium, disappearance of the surplus capacity in European facilities due to increased commercial MOX utilization, political or force majeure situations that eliminate existing capacity, or inability to negotiate a suitable contractual arrangement. If overseas MOX fuel fabrication capacity cannot be utilized, one of the domestic options for fabrication of LTA and early mission fuel described in Sect. 5.5.4.4 must be pursued.

If the LAs could not be fabricated until the entire domestic MOX fuel fabrication facility becomes operational, implementation with an optimistic LTA program schedule would postpone mission start until about 2012 as shown in Fig. 9, assuming that PIE data would be required prior to mission start. However, if the domestic MOX plant design were based on proven European technology, perhaps only LUAs (leading the mission fuel by one or two cycles) would be required. In either case, the implementation strategy would be similar to the RIS with the exception of the MOX fuel fabrication site.

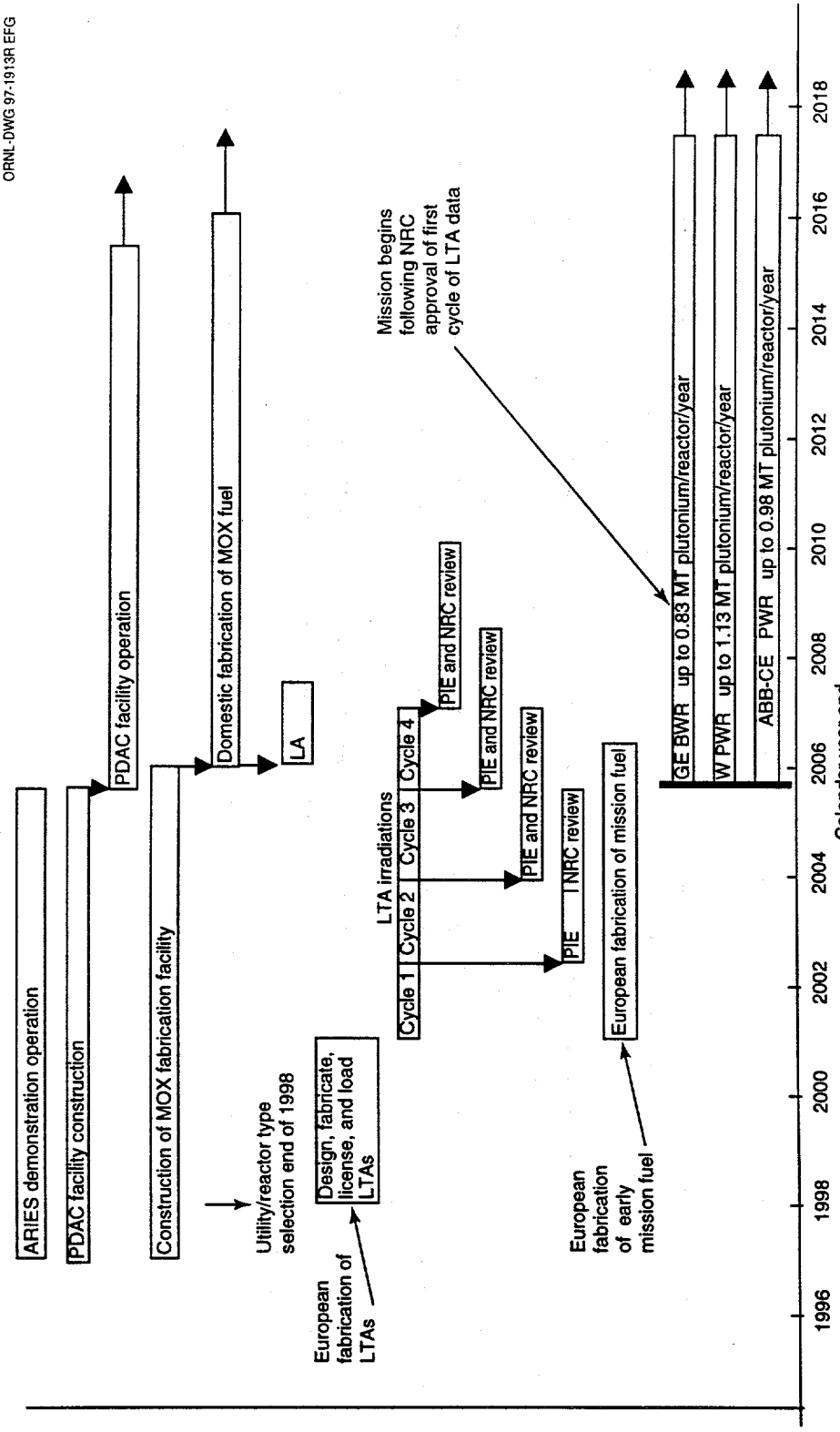

Fig. 7. LTA program with impurities and dry processing adding only 1-1/2 years to RIS mission start date (assuming success).

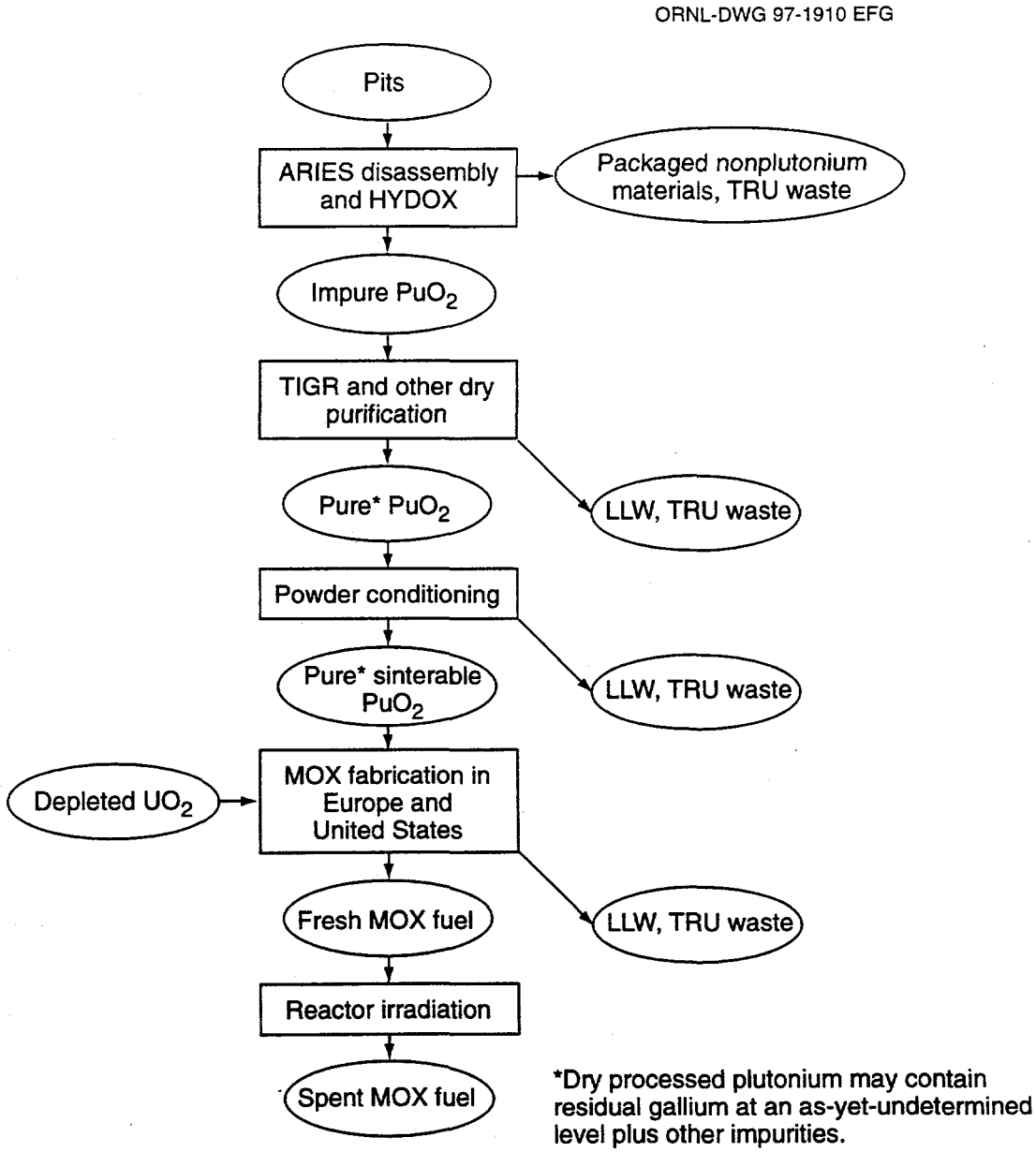

Fig. 8. Processing flow sheet assuming maximum impurity levels introduced with LTAs prior to mission start.

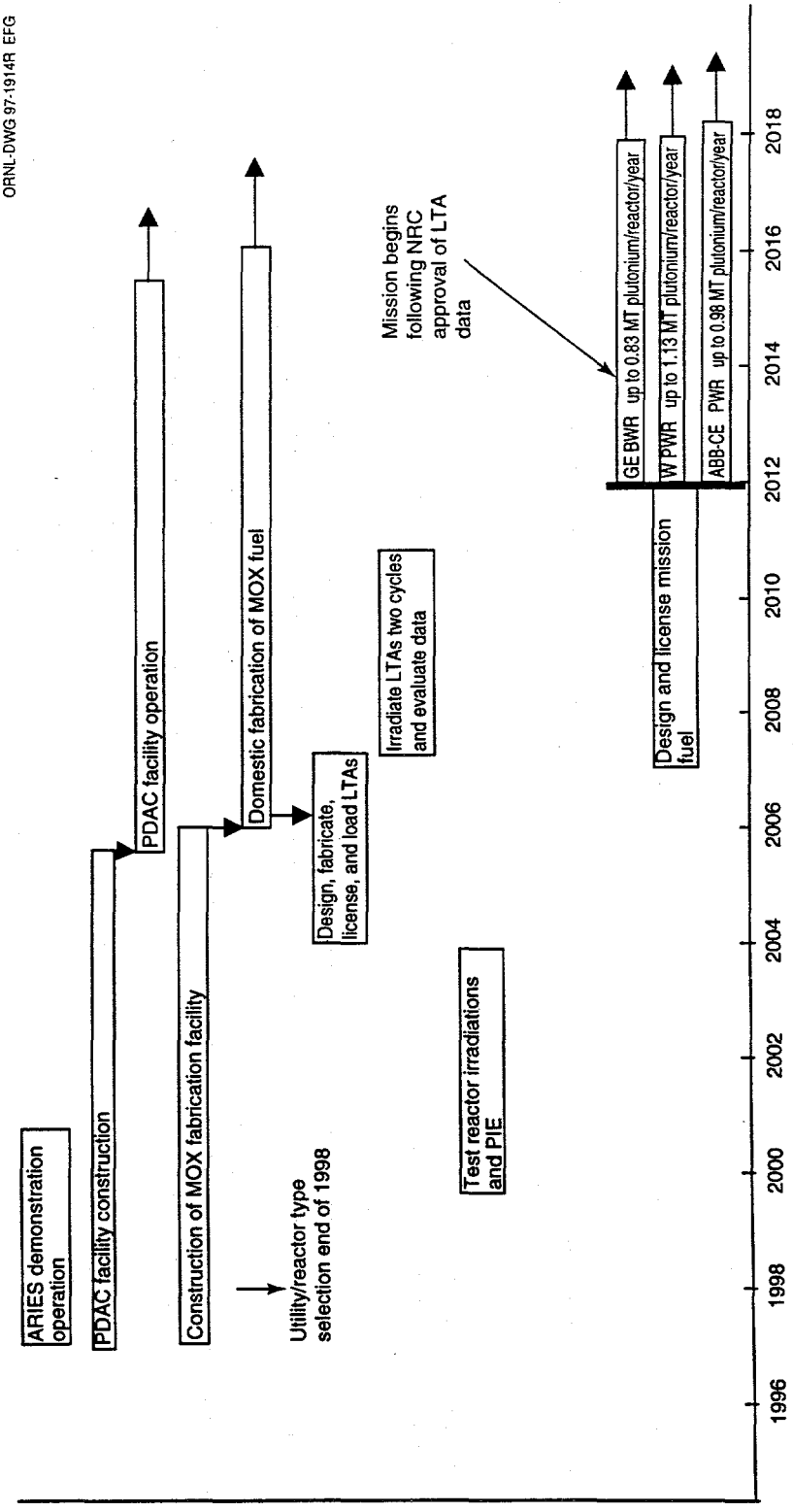

Fig. 9. Dependence on domestic MOX fabrication plant for LAs delaying mission start until 2012.

Because of the severe schedule penalty that would result if LA fabrication must await completion of the domestic MOX fuel fabrication facility, it is more likely that one of the other domestic LTA fabrication options would be pursued if European fabrication capacity could not be utilized. Pursuit of one of the domestic options for LA fabrication could preclude at least a portion of the schedule slippage as shown in Fig. 10. The flow sheet for either of these options would not differ appreciably from that of RIS, shown previously in Fig. 5.

Advantages. Provision of domestic LTA fabrication capacity would allow DOE to avoid the complications of international plutonium transportation without incurring the large schedule penalty associated with LTA fabrication in a completed domestic MOX fuel fabrication facility.

Disadvantages. As long as European fabrication remains a viable option for LTA fabrication, it is expected to offer the lowest technical risk, least cost, shortest implementation schedule, and highest institutional acceptance. Domestic fabrication of LTA and early mission fuel is a viable alternative, but such an implementation strategy would potentially increase technical risk, cost, and schedule while decreasing the institutional acceptance.

6.3.3 Delayed Mission Start or Completion Deadline

RIS was developed consistent with a timeliness criterion—disposition should begin by January 2007 and should be completed by January 2022. This criterion may be modified during the mission to account for policy changes, treaty agreements, or other developments. For example, if Russian implementation is delayed, a decision may be made to reduce the disposition rate and thereby stretch the disposition schedule. RIS outlines the shortest implementation schedule that could be pursued without incurring undue technical risk as a result of schedule considerations. Relaxation of the start date requirement would therefore not affect RIS, and the implementation schedule would closely resemble that shown in Fig. 6.

Conversely, the completion date requirement does have an influence on RIS. Depending on the number of reactors chosen for the mission and the total quantity of plutonium delivered to the reactor alternative for disposition, higher plutonium throughput core designs may be required to complete the mission within the allocated time frame. A relaxation of the completion date requirement would therefore impact RIS.

Advantages. Relaxation of the completion date requirement would allow the mission to use only low-plutonium content LEU look-alike assemblies for the entire mission. This would allow increased flexibility for downward adjustment of plutonium throughput. Relaxation of the completion date requirement would also allow a smaller number of reactors to complete the mission. This might also reduce the overall cost of the reactor option by reducing the incentive fee and by allowing a smaller MOX fuel fabrication facility to operate over a longer period such that the amortization is spread over a longer period. Additional economic analyses would be required to quantify the potential economic benefits.

Disadvantages. If the mission completion date were extended beyond the current constraint of January 2022, the choice of operating reactors would be reduced because of the currently scheduled shutdown dates for many of the U.S. nuclear plants. Only a few of the existing nuclear plants would be operating past 2022 without life extension.

6.3.4 Higher Plutonium Throughput Options

Both MTRIS and RIS start with low-plutonium throughput fuel assembly and core designs that are fully compatible with the resident LEU assemblies. RIS includes provision for increased plutonium throughput through less conservative assembly designs. Additional plutonium throughput capacity could be obtained by introducing an integral burnable absorber (boron, erbium, or gadolinium) to the MOX fuel. An integral burnable absorber is commonly used in

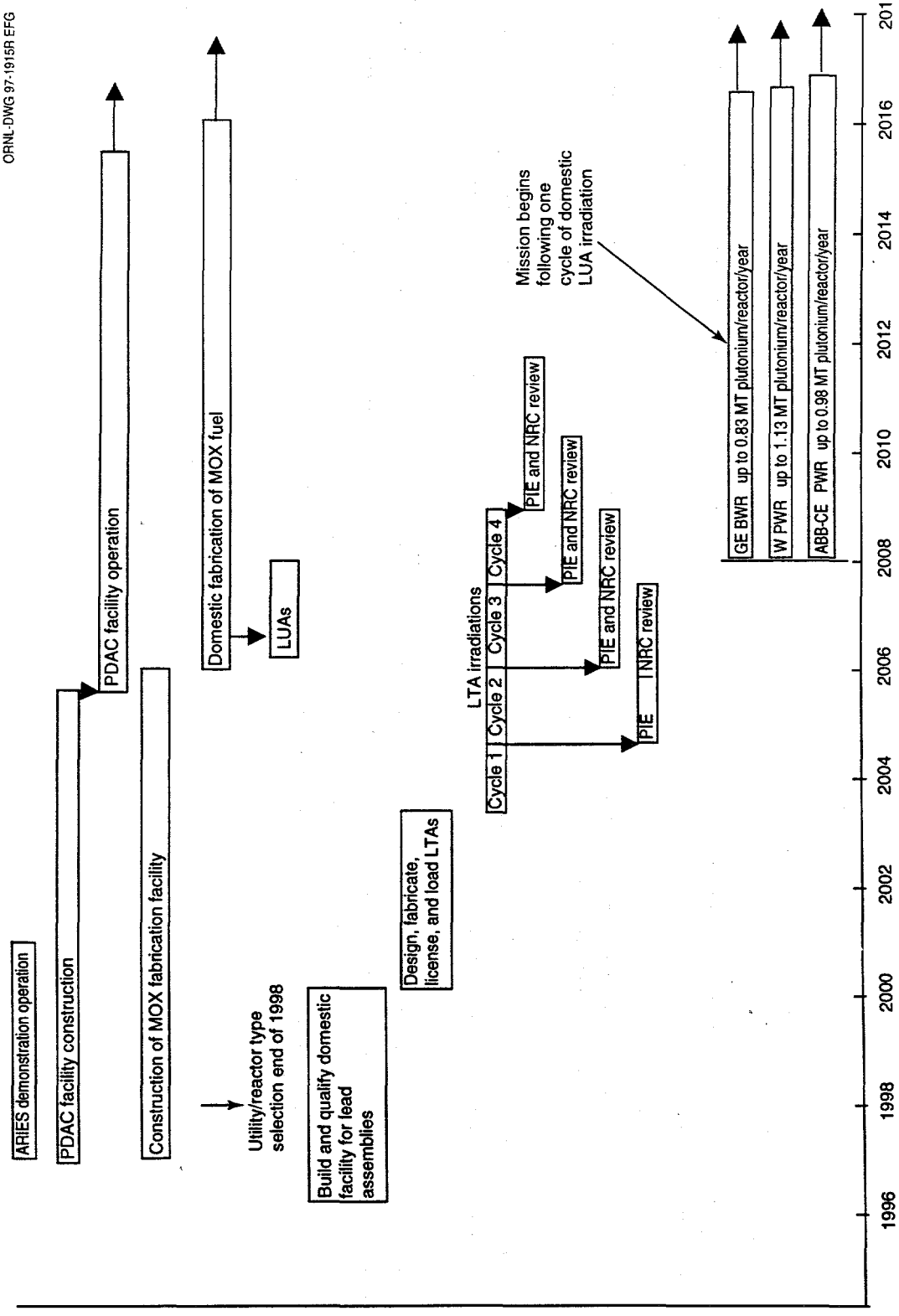

Fig. 10. Domestic LTA fabrication facility advancing mission start in absence of European fabrication.

LEU fuel, but it has only been tested to a limited extent in MOX fuel. The compatibility of trivalent burnable absorber atoms with MOX fuel has not been demonstrated. These trivalent atoms could lead to the formation of a perovskite-type compound during sintering, which could trap hydrogen in the fuel matrix.

The fuel assembly designs prepared by the reactor vendors for the 1994 PDS and the 1996 optimization studies provide an indication of the potential increase in plutonium throughput that could be obtained with an integral burnable absorber: GE indicated that a throughput increase from 0.83 to 1.5 MT plutonium/reactor/year could be obtained by adding gadolinia; ABB–CE indicated that throughput could be increased from 0.98 to 1.22 MT plutonium/reactor/year by adding erbia. Because one of the ground rules for the 1996 optimization study was elimination of an integral burnable absorber, W did not develop an optimized design using its ZrB_2 IFBA pellet coating, but increased plutonium throughput would probably result from addition of this burnable absorber also.

The initiation of the mission would be similar to RIS under this scenario. Furthermore, as shown in Fig. 11, decisions on whether to develop integral burnable absorber technology can be delayed until after the mission start. If the burnable absorber technology was shown to be desirable at that point, it could be developed in parallel to the mission so that no schedule impacts would occur.

Advantages. Development of a MOX fuel integral burnable absorber would allow a smaller number of reactors to complete the disposition mission in a shorter mission time. The cost impacts of this schedule reduction are not clear however; the fuel development costs and MOX fuel fabrication facility amortization costs must be considered.

Disadvantages. A disadvantage of introducing MOX fuel containing an integral burnable absorber is the potential cost increase associated with the fuel development and testing. However, if the fuel testing were performed through LTA testing as part of the mission reloads, the overall development costs would be minimized. Introduction of high-plutonium content assemblies and higher throughput core designs in general, including those containing this burnable absorber, might reduce downward flexibility in the disposition rate by reducing compatibility between the MOX assemblies and co-resident LEU assemblies.

The CANDU option is a special case with respect to a burnable absorber. None of the CANDU fuel designs include a burnable absorber in the MOX fuel. LEU or deuterium-uranium (DU) rods containing dysprosium would be included in some or all of the fuel bundles, but no burnable absorber would be included in the MOX fuel itself. However, a higher throughput scenario in AECL's 1996 optimization study involves a switch from the standard 37-element bundle to a newer CANFLEX design containing 43 elements of higher fissile loading.[10] The proposed discharge burnup of CANFLEX bundles is higher than that of the 37-element bundles, which actually lowers the plutonium throughput per reactor year. However, the MOX fabrication plant would be able to supply four reactors with CANFLEX bundles as compared to the two that can be supplied with 37-element bundles so that the total plutonium throughput would increase from 3.0 to 4.8 MT plutonium/year. As with the LWR schedule, decisions on whether to proceed with qualification of higher throughput fuel designs can be postponed until after mission start as shown in Fig. 12.

6.3.5 Flexibility for Lower Throughput

Downward flexibility in the plutonium disposition rate may be desirable to accommodate treaty or policy changes or to match Russian disposition rates. Downward flexibility is inherent in the reactors under consideration because they are currently operating with no plutonium feed and the required reactor hardware modifications, if any, will not change this capability. The issue is the rate at which plutonium feed can be adjusted. While it would be possible to shut down the

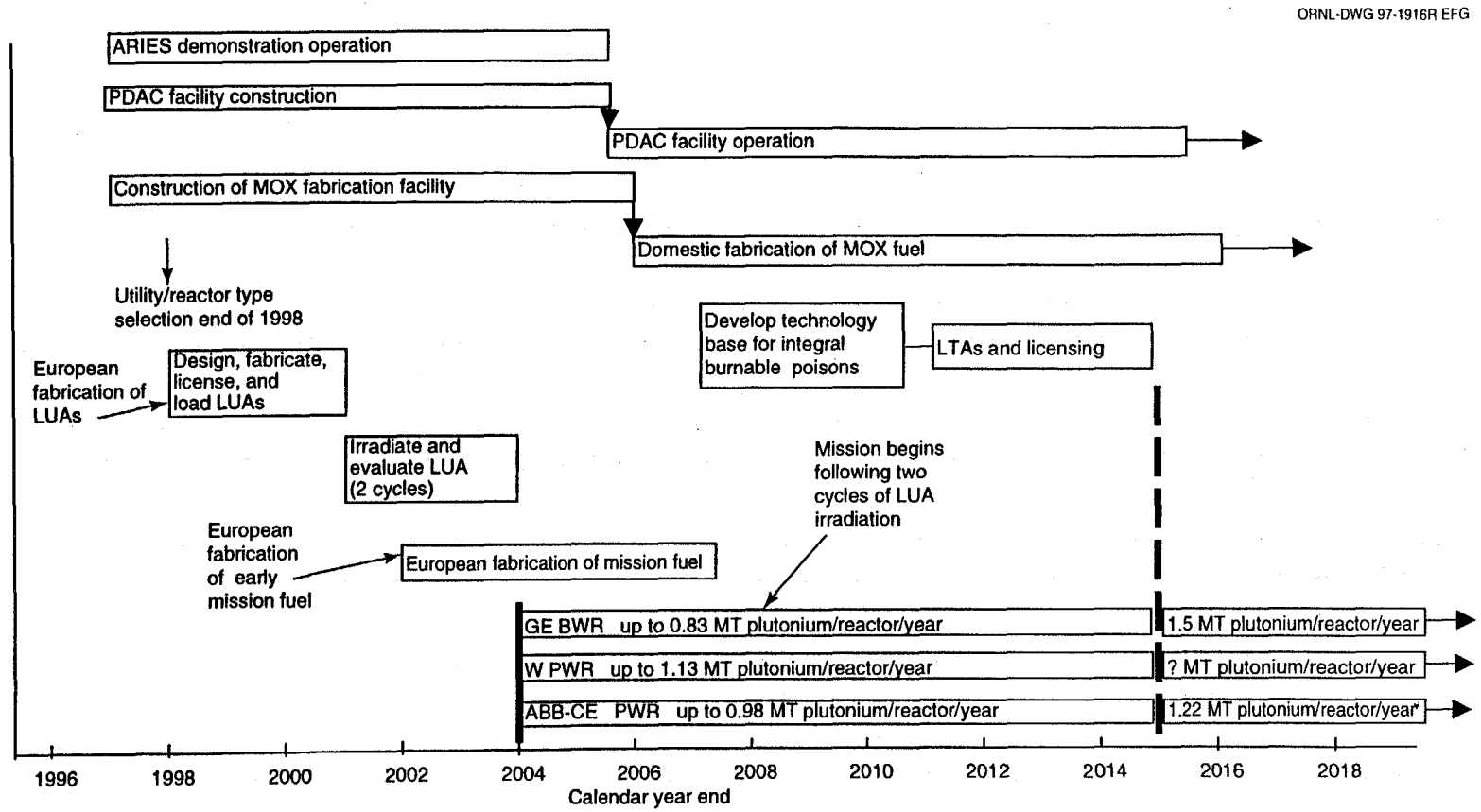

Fig. 11. Development of technology base for integral burnable absorbers in MOX that can increase plutonium throughput per reactor.

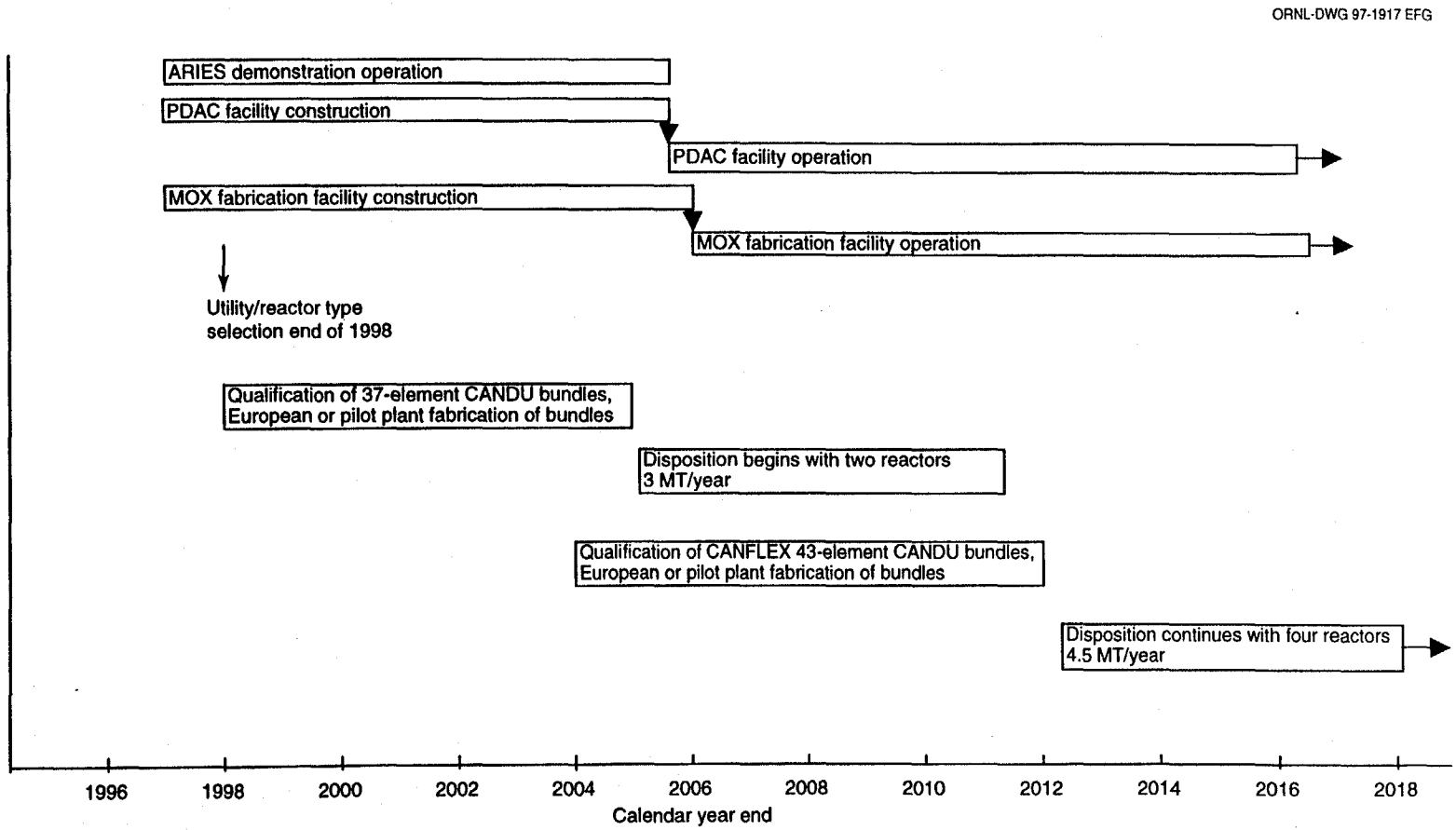

Fig. 12. CANFLEX fuel offering higher total plutonium disposition rate for CANDU option for fixed MOX plant capacity.

reactor and replace all the MOX assemblies with LEU assemblies, the more likely approach would be to increase the number of fresh LEU assemblies loaded at the next scheduled refueling outage. Once the transition to a high-throughput full-MOX core has been accomplished, downward adjustment of the plutonium feed rate would require a series of transition cycles. The interface between assemblies designed for full-MOX operation and LEU assemblies may be difficult to accommodate in the reload design. It is proposed that downward flexibility would be accomplished by utilizing only LEU-compatible assemblies for the mission duration. With LEU-compatible assemblies, reduced plutonium disposition rates can be readily accomplished by replacing MOX bundles with LEU bundles. Another option for reduced throughput is to replace a portion of the MOX rods within a single bundle with LEU rods. This should not impact the front end of the mission schedule as outlined in Fig. 6.

Advantages. This implementation strategy remains entirely within the known experience base in terms of reactor core loading strategy. This may reduce concerns with increased plutonium throughput. Furthermore, it would reduce the potential for reactor modifications. It would allow rapid downward adjustment of the plutonium throughput rate without impacting the reactor operation.

Disadvantages. The primary disadvantage of this option is that it extends the mission schedule or requires the inclusion of more reactors in the program. This may cause difficulties in selection of potential reactors because of limited remaining operating lifetime without life extension.

6.3.6 All European Fabrication Allowed for Mission

ROD specifically ruled out the use of European fabrication for the mission fuel and called for construction of a dedicated domestic plant that would be decommissioned upon completion of the WG MOX mission. If this policy were changed and sufficient capacity were contracted, fabrication of mission fuel in one of the existing European MOX fuel fabrication facilities might be very cost-effective. Because the domestic MOX fuel fabrication facility would be amortized over a limited lifetime, the high capital cost of a new facility would impact the overall alternative cost. Use of an existing facility would allow reduction in this capital charge by sharing it with other MOX fuel customers. This is one of the major considerations in the definition of MTRIS. This option was considered and eliminated as part of the ROD process.

Advantages. Although detailed economic analyses have not been conducted, the authors believe that the ultimate unit cost of MOX fuel, including international plutonium and MOX transport, obtained from an existing European facility would be substantially less than that obtained from a new dedicated domestic facility. The costs of siting, licensing, constructing, operating, and decommissioning a new facility would have to be recovered over a limited operating lifetime. Furthermore, the currently discussed mission duration is much shorter than the typical design lifetime of a MOX fuel fabrication facility. By utilizing an existing commercial facility, these costs would be shared with the other users. Furthermore, because the newer European facilities are much larger than the proposed domestic facility, additional benefit would be accrued from economy of scale. Both technical and schedule risk would also be reduced by utilizing an existing MOX fuel fabrication facility.

Disadvantages. Implementation of this scenario would require continued international transport of plutonium and MOX fuel rods or assemblies. However, no technical challenge is involved in this transport. Another related disadvantage is that use of an existing commercial MOX fuel fabrication facility might appear to be an endorsement of commercial MOX fuel use in conflict with the stated nonproliferation policy. This may increase resistance to implementation of the reactor option, especially if additional capacity is required in Europe to fulfill the mission requirements.

7. CONCLUSIONS

DOE has proposed irradiation of MOX fuel in existing reactors as a disposition method for surplus WG plutonium. The reactor alternative is supported by an extensive technology base; however, the required infrastructure does not exist domestically. MOX fuel technology was developed to the point of commercialization in the United States by the mid-1970s, but domestic work was discontinued when the policy decision to forego reprocessing was announced. MOX fuel development continued in Europe, and MOX fuel is now in commercial use there. The procurement of technology and experience data from Europe is recommended to expedite the implementation of the U.S. disposition mission.

The proposed WG MOX differs from commercial RG MOX in three ways: (1) the isotopic composition—WG plutonium has higher fissile content and lower ^{240}Pu content, (2) the morphology of the PuO_2 feed material—the reference dry conversion and purification processes produce a powder morphology that is distinct from that associated with the oxalate-derived PuO_2 commonly obtained from aqueous processing, and (3) impurities—dry conversion and purification is expected to leave certain impurities in the PuO_2 at levels above the experience base. The proposed MTRIS removes the last two uncertainties through aqueous purification and oxide production for all feed materials. RIS recognizes that dry conversion and purification have been chosen by DOE as the reference processes and that the fabrication questions with this dry-processed material should be answered early in the program. However, the impurity questions are avoided early in the mission by choosing higher purity material for the LAs and early mission fuel such that the impurity concentrations are within the experience base. The higher impurity levels would be introduced later in the mission once sufficient data were available to support their use. The required development and testing would be performed in parallel with the mission off the critical schedule. Both implementation strategies include a low-risk approach to core loading under which LEU look-alike assemblies would be placed in approximately one-third of the core. Under MTRIS, this core loading approach would be used for the entire duration of the mission. Under RIS, the plutonium loading would be increased incrementally as justified later in the mission.

The domestic MOX fuel fabrication plant is not expected to be operational until the end of the year 2006. This schedule is predicated on the procurement of at least some European technology to minimize development and design uncertainties. The procurement of European fabrication for lead assemblies and early mission fuel could allow insertion of MOX fuel before the domestic plant is available. Of more value than the fabrication technology are the related fuel performance and reactor physics data bases that would be required to obtain NRC approval for insertion of MOX fuel. Because these data bases are for the most part proprietary, they were not assessed for their adequacy and sufficiency for use by the utility and the NRC for licensing in the United States.

If European fabrication of LAs is not possible, a domestic LTA fabrication line would be highly desirable to avoid severe schedule penalties on mission implementation. The alternatives for development of domestic MOX LTA fabrication capacity include building an interim prototypic fuel fabrication line in an existing DOE structure at either the site of the mission facility or another DOE site and completing a high-priority fabrication line as a modular component of the mission fuel fabrication facility.

Activities are currently under way to address the outstanding issues facing the reactor-based disposition alternative. The fabrication development process is being conducted to address the disassembly of weapons components and the subsequent fabrication of MOX fuel from the resulting PuO_2. This effort is focused on production of MOX fuel with the required density, pore

distribution and stability, homogeneity, and impurity levels using the dry process feed powder. The effects of gallium, the most commonly discussed impurity, are being addressed through both laboratory and in-reactor tests. These tests are scheduled to be complete by the year 2001, prior to the insertion of WG MOX fuel in a commercial reactor. However, it is virtually certain that lead assemblies will be required to support the WG MOX fuel qualification. If the WG MOX fuel remains within the experience base in terms of impurities and core loading, these may be LUAs because no additional fuel performance data would be required. However, to address the potential impacts of impurity levels outside the experience base, LTAs would likely be required. Additionally, lead assemblies may be required from the domestic MOX fabrication plant when it becomes available.

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APPENDIX A

HISTORY OF MOX FUEL USE IN LWRs

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APPENDIX A. HISTORY OF MOX FUEL USE IN LWRs

Light-water reactor (LWR) mixed uranium-plutonium oxide (MOX) fuel research, development, and testing have been ongoing since the late 1950s. During the early years, plutonium utilization was viewed as a solution to the perceived problem of limited uranium resources. Plutonium utilization was also viewed in some energy-resource-poor countries as a path to energy independence. Much of the early research was supported by government financing. The United States led MOX fuel development during this early period, but substantial programs also existed in several foreign countries. Analytical work, critical measurements, and test irradiations provided sufficient justification for implementation of test rod and lead test assembly (LTA) irradiations during the 1960s. By the early 1970s, MOX fuel development had progressed to the point of commercial utilization. Since that time, MOX fuel technology has continued to progress. MOX fuel is utilized commercially in a number of foreign countries, and its use is expected to expand over the next several years. The experience base is described more completely in the three subsections below. The domestic MOX fuel experience base is described, followed by a summary of the foreign experience base. Finally, a prediction of the current outlook for MOX utilization worldwide is given.

A.1 U.S. MOX FUEL EXPERIENCE

The earliest U.S. MOX fuel research began under U.S. Atomic Energy Commission (AEC) sponsorship in 1956. This Plutonium Utilization Program (PUP) was led by Pacific Northwest National Laboratory (PNNL). The purpose of the PUP was to develop the technology that would be necessary to implement plutonium recycle in thermal reactors. Activities carried out under the PUP included analytical activities and test reactor irradiations of fuel samples and test rods. Irradiations were conducted in the Materials Test Reactor (MTR), the Engineering Test Reactor (ETR), the Experimental Boiling-Water Reactor (EBWR), and the Plutonium Recycle Test Reactor (PRTR). The PRTR was a heavy-water moderated and cooled, pressure tube reactor built on the Hanford Reservation specifically for development of plutonium fuels, including MOX. Based on the early success of the PUP, the AEC subcontracted with Westinghouse (W) for implementation of the Saxton program. W designed nine LTAs for insertion into the second core of the Saxton pressurized-water research reactor (most of the fuel was actually fabricated by ARCO-NUMEC under contract). The AEC simultaneously sponsored boiling-water reactor (BWR) MOX fuel research that included irradiation of fuel samples and test rods in the Vallecitos BWR, the PRTR, and Dresden 1.

Following completion of the PUP and the Saxton programs, the AEC decided that additional government sponsorship of MOX fuel development was unwarranted. Development therefore shifted to the private sector. Based on experience gained through leadership of the Saxton program, W embarked upon the Plutonium Recycle Demonstration Program (PRDP) with the Edison Electric Institute (EEI), a consortium of utilities. The PRDP consisted of an initial analytical phase and a follow-on demonstration phase. Four all-MOX LTAs were fabricated by W and inserted into San Onofre 1, a W pressurized-water reactor (PWR). In addition to providing a demonstration of the viability of MOX fuel utilization, this irradiation provided a comparison of the relative performance of coprecipitated vs mechanically mixed MOX fuel. In what appears to be a continuation of the EEI-W PRDP, W fabricated four all-MOX LTAs for irradiation in the R. E. Ginna PWR. These assemblies were eventually inserted into the reactor in 1980, 6 years after they were originally scheduled for insertion. These four assemblies represent the most recent U.S. commercial MOX fuel irradiations.

A parallel program between the EEI and General Electric, (GE) the Plutonium Utilization in Boiling-Water Reactors Program focused on MOX fuel utilization in BWRs. Eight MOX rods

were loaded into reconstituted bundles in Big Rock Point (BRP), an early model GE BWR. Later, 32 MOX rods were loaded into BRP in a total of 16 fresh bundles. Following successful irradiation for one cycle, 3 all-MOX LTAs containing 204 MOX rods were inserted. Most of these rods were manufactured with AEC-supplied plutonium that was ~90% fissile (87% ^{239}Pu). The neutronic characteristics of this AEC plutonium are more similar to those of the weapons-grade (WG) plutonium than those of current reactor-grade (RG) plutonium. GE also fabricated five island-type MOX bundles for irradiation in Quad Cities 1. A portion of these rods also contained the 90% fissile AEC plutonium.

Several of the other U.S. fuel fabricators and nuclear steam supply system (NSSS) designers investigated MOX fuel utilization in the 1960s and 1970s. Of these, Exxon Nuclear Company, Inc. (now part of Siemens) had the largest MOX fuel development program. Exxon fabricated a total of 30 island-type MOX fuel LTAs containing a total of 720 MOX fuel rods for irradiation in the BRP BWR. Gulf United Nuclear designed 11 island-type MOX fuel LTAs containing 99 MOX rods, which Alkem supplied, for irradiation in Dresden 1. Nuclear Fuel Services also fabricated 4 all-MOX LTAs, containing 292 MOX fuel rods for irradiation in BRP.

One additional MOX fuel program worth mentioning is the Electric Power Research Institute (EPRI) Plutonia Fuel Study. The study was sponsored by most of the stakeholders in the U.S. MOX fuel industry: Babcock and Wilcox (B&W), British Nuclear Fuels Limited, the Central Research Institute of Electric Power Industry (Japan), Combustion Engineering (ABB–CE), Exxon Nuclear, GE, and W. The purpose of the study was to examine the thermal and irradiation-induced densification of MOX fuels for comparison against similar data already produced for urania fuels. The Plutonia Fuel Study subjected MOX fuels fabricated by many of the participants to both sintering and irradiation to measure the densification and compare the performance.

As a result of all of this testing and demonstration, MOX fuel technology in the United States was ready for commercialization by the early 1970s. At least two of the NSSS vendors (ABB–CE and B&W) included MOX utilization in their latest designs such that full-MOX cores could be utilized. At this time, the AEC initiated the environmental review required under Sect. 102(2)(C) of the National Environmental Policy Act of 1969. This review, entitled the "Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in LWRs," or GESMO, was prepared by the AEC and later the Nuclear Regulatory Commission (NRC) up through the public hearing stage. However, following President Carter's 1977 announcement of a nuclear nonproliferation policy that included an indefinite deferral of domestic commercial reprocessing and recycling of plutonium, the NRC terminated the GESMO proceedings prior to completion and terminated proceedings on pending or future plutonium recycle-related license applications, except those involving only small quantities of MOX fuel for experimental purposes.

No appreciable MOX fuel development or demonstration activities have occurred in the United States since this decision was announced in 1977. Some of the ongoing irradiations were carried through to completion, while others were terminated prematurely. Only one example of the advanced MOX-burner reactor designs was completed—the three-unit Palo Verde plant utilizing ABB–CE's System 80 design. A second, the two-unit Bellefonte plant utilizing B&W's Model 205 design, is partially complete and retains an active construction permit. Two of the domestic vendors, GE and W, have continued international MOX fuel work to support their international customers. This international work is almost entirely proprietary. Most of the international work consists of design and licensing support for supply of MOX fuel assemblies that are fabricated by one of the European companies under contract.

In summary, MOX fuel development in the United States progressed from early feasibility studies to commercialization by the mid-1970s. However, because of changing U.S. nuclear nonproliferation policy, almost no work has been performed since then. Improvements to urania

fuel fabrication techniques and overall fuel quality have therefore not been incorporated into domestic MOX fuel designs. Much work would be required to recover and update the domestic experience base to bring domestic MOX fuel technology to the same level of development as that of modern urania fuel.

A.2 WORLDWIDE MOX FUEL EXPERIENCE BASE

MOX fuel development in Europe followed a path similar to that of domestic development. Early domestic programs in several European countries focused on analytical work and test irradiations. The Belgians irradiated the world's first MOX PWR fuel assembly in the small BR3 in 1963. Major national programs were also undertaken in France, Germany, Italy, and the United Kingdom. In December 1974, the Council of Ministers of the European Communities attempted to consolidate these various independent programs under the auspices of the Research and Development Pluriannual Programme on Plutonium Recycling in Light Water Reactors. Under this program's Commission of the European Communities (CEC) sponsorship, MOX fuel irradiations were carried out in BR3, the High Flux Reactor at Petten, the Dodewaard BWR, the Lingen BWR, the Garigliano BWR, and the Chooz A PWR.

During the early 1970s, European MOX fuel fabrication capacity was limited to a number of research-scale facilities in Belgium, France, Germany, Italy, and the United Kingdom. The majority of the European fuels that were irradiated through the 1970s in Europe were produced by either Belgonucléaire S.A. or Alkem (now part of Siemens). In addition, some of the fuel rods and/or assemblies were provided by W, GE, and Exxon. Most of the commercial irradiations consisted of LTA projects. A notable exception is the Garigliano irradiations. Following successful irradiation of 16 LTAs, a complete quarter-core reload of island-design MOX bundles was inserted in 1975. This marked the first commercial-scale MOX fuel utilization in Europe. At about the same time, commercial-scale recycle was initiated in a German PWR and a BWR. It should be noted that these developments occurred as nuclear nonproliferation concerns developed in the United States. Announcement of the U.S. deferral of reprocessing and plutonium recycle apparently did not greatly affect the European projects.

By 1980, when the CEC program came up for renewal, changes in the nuclear industry had reduced the political support for MOX fuel research. Uranium supplies were found to be much more plentiful, reducing the potential shortfall in uranium supply. The reprocessing industry had not scaled up as readily as expected, so plutonium excesses were not available. Finally, advances in liquid-metal fast breeder reactor (LMR) technology indicated that rapid implementation of a LMR fleet was imminent so that no plutonium would be available for use in LWRs. For these reasons, the French and the British failed to support continuation of the CEC program, and it was not extended.

French, British, and Italian resources were refocused on LMR fuel development. Only the Belgian and German MOX programs continued in earnest through the early 1980s. Fuel development continued at both Belgonucléaire and Alkem. The 1970s vintage fuel had satisfactory in-pile performance, but was relatively insoluble in nitric acid without some HF addition. A requirement was imposed by the reprocessors that MOX fuel must be able to be reprocessed in the large-scale commercial plants being built without HF additions. Advanced fuel fabrication techniques, such as Belgonucléaire's MIMAS and Alkem's OCOM and AUPuC processes, were developed to generate a more homogeneous fuel, which has a higher solubility in nitric acid. These advanced fuels were proven through commercial irradiations during the early 1980s.

By the middle to late 1980s, it was apparent that the assumptions that had halted the earlier CEC MOX fuel program were inaccurate. Successes in the reprocessing industry and delays in LMR deployment had created a surplus of separated plutonium in several countries, most notably France. The French utility Electricité de France (EdF) therefore announced in 1985 its decision to

recycle separated plutonium through MOX fuel utilization in its fleet of 900-MW(e) class PWRs. This announcement, more than any other single event, pushed European MOX fuel utilization to the advanced state of commercial development in which it exists today. At the time, only two pilot-scale MOX fuel production facilities existed in the world—Belgonucléaire's 35-MT/year P0 plant, and Alkem's 35-MT/year Hanau plant. All the rest had either been decommissioned or converted to fast reactor plutonium fuel fabrication. Within a few years of the EdF announcement, many facilities other than P0 and Hanau had either been converted back to LWR MOX fabrication, were newly constructed, or were planned. The COGEMA facility at Cadarache was converted back to MOX fuel fabrication, and construction on the new MELOX commercial facility began. BNFL opened the 8-MT/year pilot MOX Demonstration Facility at Sellafield and began design of the large follow-on 120-MT/year Sellafield MOX Plant. The Germans constructed a 120-MT commercial plant at Hanau. Belgonucleaire prepared designs and license submittals for P1, a 60-MT/year extension to their existing facility at Dessel.

Since the late 1980s, the MOX fuel industry has matured in Europe. Hundreds of MOX fuel assemblies have been supplied to utilities in Germany, France, Switzerland, and Belgium. Most of this experience is with fuel fabricated via the MIMAS process. European MOX fuel utilization is currently limited by fabrication capacity because a separated plutonium surplus remains in several of the relevant countries. As the new MELOX and SMP facilities reach nominal capacity, this situation may improve. Longer term expansions to both MELOX, P0, and SMP may further increase fuel fabrication capacity.

One of the notable MOX fuel development efforts outside Europe may be found in Japan. Japanese nuclear utilities, research organizations, and commercial vendors have been researching plutonium fuels since the 1960s. However, much of this early research was focused on either LMR or heavy-water reactor development. The Power Reactor and Nuclear Fuel Development Corporation (PNC) built the Plutonium Fuel Development Facility, which began operation in 1965. A follow-on facility, the Plutonium Fuel Fabrication Facility, began operation in 1971. Fuels were fabricated in these facilities and by other fabricators in the United States and Europe to support the Joyo breeder reactor and the Fugen Advanced Thermal Reactor (ATR) programs. Irradiation testing of both fast and thermal-spectrum plutonium-based fuels was performed in a number of test reactors including the Saxton PWR, the Halden BWR, the GE Test Reactor, JRR-2, JMTR, Rapsodie, and Dounreay. Several thousand MOX fuel rods were fabricated for critical experiments in the Deuterium Critical Assembly built to support the ATR program. Since its initial operation in 1979, Fugen has burned over 600 MOX fuel assemblies containing more than 18,000 MOX fuel rods.

With the recent curtailment in the ATR program and the delays in the LMR programs, the Japanese have announced their plans to recycle their surplus plutonium in existing LWRs and new GE advanced BWRs (ABWRs) that are being constructed. However, the only reported Japanese LWR MOX fuel experience consists of LTA irradiations. Two island-type MOX fuel bundles were inserted in the Tsuruga-1 BWR in 1986. After three cycles of irradiation, they were removed and subjected to postirradiation examination (PIE). Four all-MOX LTAs designed by W were inserted into the Mihama-1 PWR in 1988. These four LTAs were also removed after three cycles of irradiation and subjected to PIE. In addition to these LTA projects, Japanese companies and organizations are heavily involved in the international programs researching MOX fuel use, including the VIP and ARIANE programs being coordinated by Belgonucléaire.

Russian experience with MOX fuel has been very limited. Of the three commercial reactor types found in the former Soviet Union, only the LMRs were considered for plutonium burning prior to Peristroika. Both the RBMKs and the VVERs were operated exclusively on urania fuel cycles. Furthermore, some questions existed concerning the safety of MOX fuel utilization in the RBMKs and VVERs. All plutonium recovered from the recycling plants was to be recycled in the

existing BN-600 and planned BN-800 fast reactors. Only in the past few years have any efforts been undertaken to implement MOX fuel utilization in the Russian VVER PWRs. Currently, the Russian MOX experience base is limited to a handful of test rod irradiations in the pressurized loops of the MIR test reactor. A MOX fuel LTA program is currently being developed with international cooperation for insertion of three LTAs into the Balakova-4 VVER within the next 5 years.

Canadian MOX fuel experience has been focused exclusively on utilization in CANDU heavy-water reactors. In many respects, this fuel is similar to that of LWRs, and so it is described here. Early CANDU-MOX fuel testing was performed in a number of test reactors. Early CANDU-MOX pellets were obtained by Atomic Energy of Canada, Ltd., (AECL) from Belgonucléaire and Aktiebolaget Atomenergi of Sweden. Later, pellets were fabricated in the Recycle Fuel Fabrication Laboratory at Chalk River. These fuels have been irradiated in the NPD, NRX, and NRU reactors. No commercial use of CANDU-MOX is reported. However, the NRU in particular is able to reproduce commercial CANDU operating conditions closely so that bundle irradiations in the NRU loops are essentially equivalent to LTAs for LWRs. A total of 12 bundles have reportedly been irradiated under CANDU-prototypic conditions. Other test reactor irradiations are suspected from other reports.

Based on 40 years of research, development, testing, and utilization, MOX fuel production and utilization is in many respects a mature, international industry. Most of the MOX fuel expertise resides in several European companies. The remainder is shared between the United States and Japan. MOX fuel utilization, which currently is on the order of 150 MT/year or less, is limited only by the available fabrication capacity. As MELOX and SMP reach their nominal capacities, this utilization rate can easily be expected to double. Furthermore, expansions to P0, MELOX, and SMP are in the planning stages. These expansions could bring the total production rate to more than three times its current level.

The irradiation performance of MOX fuel has been satisfactory. Perhaps due to the increased scrutiny placed on it during fabrication and irradiation, MOX fuel has, in fact, experienced fewer failures attributable to the fuel itself than urania fuel. MOX fuel utilization has been demonstrated in both PWRs and BWRs at least to the level of self-generated recycle (which corresponds to approximately one-third MOX rods at equilibrium). Impediments to its expanded use are not technical, but rather are political and economic.

A.3 PREDICTED NEAR-TERM WORLDWIDE MOX FUEL UTILIZATION WITHOUT SURPLUS WG PLUTONIUM MOX

MOX utilization is being driven by different factors in the countries of interest. In France, MOX fuel use is part of the national energy strategy of independence. While currently not economically competitive with urania fuel, MOX fuel is expected to become competitive in the next few years as economies of scale are realized in the MOX industry and as the allowable discharge burnup for MOX fuel is increased to the urania fuel limit. In Japan, as in France, MOX fuel use is being driven by the national policy toward energy independence, the excess of separated plutonium, and the availability of MOX technology. Japanese MOX fuel utilization is expected to increase dramatically over the next decade. All of the required fuel may initially be fabricated in Europe, but eventually much of it will likely be fabricated in a new Japanese facility. In Germany, where individual utilities are responsible for disposal of their own spent fuel, reprocessing and plutonium recycle have been the default solution for more than a decade. However, with the recent political setbacks to the German nuclear industry, some of the German utilities are considering long-term storage rather than reprocessing for their future spent fuel. PNNL appears to be interested in the MOX fuel industry as a means to assist its international reprocessing

customers, as there are no reported plans to implement MOX fuel use in Sizewell B, the only LWR in the United Kingdom.

The implementation of a surplus WG plutonium MOX fuel program at the scale currently envisioned (60–80 MT/year) would not have a great impact on the direction of the global MOX fuel industry. If no new facilities were added beyond those already in operation or under construction, surplus WG plutonium MOX would represent a maximum of 20% of the 300-MT/year capacity that will be operational when the program is implemented. If all of the planned expansions were completed, surplus WG plutonium MOX would represent only about 10% of the 500-MT/year capacity.

Despite the favorable outlook for plutonium utilization worldwide, no U.S. MOX fuel utilization is foreseen in the absence of the MD program because of the long-standing U.S. nuclear nonproliferation policy. All of the domestic commercial reprocessing facilities have been shut down or abandoned prior to completion. Furthermore, inexpensive uranium fuel supplies provide an economic disincentive toward commercial recycle in the United States.

APPENDIX B

FUEL FABRICATION PROCESS QUALIFICATION

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APPENDIX B. FUEL FABRICATION PROCESS QUALIFICATION

The following discussion is based to a great extent on input received from and discussions with D. Sunderland of the S. M. Stoller Corporation.

Qualification of the fuel fabrication process and facility can best be described through a listing and description of the associated documentation. While the specific nomenclature of the documents and the specific content of individual documents vary from vendor to vendor, the sum total of the information is roughly equivalent. The documentation for fabrication and for quality assurance (QA) are closely related and entwined.

The documentation is summarized in Fig. B.1 for the pellet production process. Similar documentation is required for rod and bundle production. In Fig. B.1, the documentation for fabrication and related QA is shown in dashed-line boxes. The documentation for qualification of the process is shown in solid-line boxes. A few of the documents have shared functions. For qualification, the primary document is the product specification. For the purposes of discussion, this is taken to be the pellet specification. Similar documents exist for finished rods and complete assemblies or bundles. The product specification is usually dictated to the fabricator by the fuel designer and/or utility customer. Iteration between the fabricator and the customer may be required to generate a specification to define a product that the fabricator can produce and a product that satisfies the customer's needs. The product specification describes the physical characteristics of acceptable pellets. Examples of the characteristics of importance for LWR MOX fuel include impurities, grain size, pore size and type, plutonium particle size, plutonium homogeneity, pore distribution, sintered density, densification limits, cracks, inclusions, chips, and surface finish. Dimensional information is commonly specified on a pellet drawing included in the specification. A fuel pellet specification may also include other information, such as references to other documents; the processes recommended to produce the product; feed materials and their specifications; and special operational controls such as cleanliness, QA requirements, and handling requirements.

Based on previous experience, fabrication development activities, and recommendations from the customer, the fabricator develops a process outline for fabrication of a product meeting the specification. The process outline may also be known as a production route specification and generally contains a flow sheet detailing the processes to be used. It also includes a description of the feed materials to be used, the important processes to be used, the inspection, and materials accountability. The process outline includes all fabrication, inspection, and test operations to be employed. It also identifies those processes that are to be qualified, and it is these processes for which process specifications are developed.

Each process specification defines the manufacturing process and possibly equipment selected to produce an intermediate and/or final product that meets the design requirements. In addition to a general process description, this document may include references to other documents, in-process and product inspection methods and acceptance criteria, process procedures and controls, feed material descriptions and specifications, deviations from normal operation, allowances for rework and repair, and process qualification requirements.

For each process to be qualified, a qualification plan is developed to meet the requirements laid out in the respective process specification. The qualification plan describes the process to be qualified and the scope of the qualification. It then outlines the test matrix to be used to qualify the particular product. It specifies the range of process parameters to be qualified, the number of tests to be carried out, and the acceptance criteria for the tests and the product. It includes a discussion of nonconformances and what actions are to be taken in such an event. It may also define the conditions under which requalification is required.

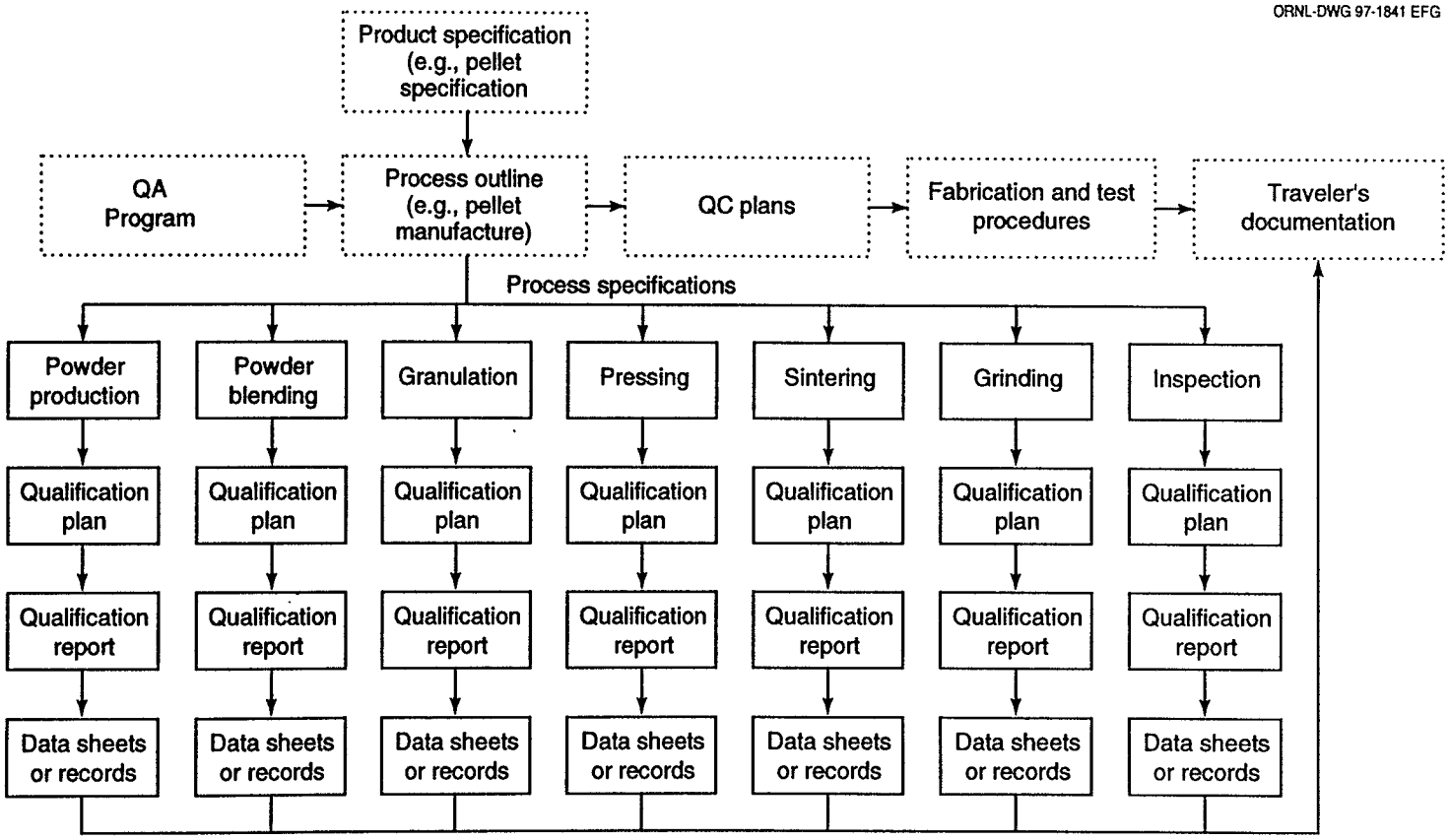

Fig. B.1. Fabrication, qualification, and QA documentation for pelleting.

Upon completion of the qualification tests as defined in the qualification plan, a qualification report is prepared. This document includes a description of the equipment and/or process being qualified, the date of the testing, the particular qualification plan defining the testing, the product being qualified, the tests and methods used for qualification, the results of the tests, acceptance criteria for the tests and product, the qualified parameters and possibly their acceptable ranges, nonconformances and their resolutions, and the specific steps to be taken to implement the qualified parameters into production.

In many facilities, the actual parameter ranges that are qualified are summarized on data sheets or data cards. These are used to guide the actual fabricators during production. These data cards alone are insufficient to define acceptable production. However, their use allows a single set of procedures to be used for multiple operating envelopes. With such an arrangement, the data cards can be updated without modifying the individual procedures.

All of the documents described above may be classified as qualification-related. A second set of documents may be classified as fabrication, or quality assurance/control (QA/QC). These documents are denoted by dotted-line boxes in Fig. B.2. The primary QA/QC document is the QA plan. It is generally a top-level document that outlines a QA program meeting the legal requirements as set forth in Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50), Appendix B.

10 CFR 50 Appendix B identifies 18 criteria for acceptable QA programs: (1) organization; (2) QA program; (3) design control; (4) procurement document control; (5) instructions, procedures, and drawings; (6) document control; (7) control of purchased material, equipment, and services; (8) identification and control of materials, parts, and components; (9) control of special processes; (10) inspection; (11) test control; (12) control of measuring and test equipment; (13) handling, storage, and shipping; (14) inspection, test, and operating status; (15) nonconforming materials, parts, or components; (16) corrective action; (17) QA records; and (18) audits. While not strictly applicable to fuel fabrication facilities, Appendix B has historically been applied to nuclear fuel fabricators as suppliers to the licensee. Criterion (9) requires process qualification for special production, inspection, and test processes. Often, fuel fabricators go beyond the strict legal requirements—qualifying additional processes other than those that are legally required because of the QC improvements that can result.

The top-level QA program document usually does not include sufficient information for implementing an acceptable program. Detailed implementation of QA is documented in the QC plan or schedule, through which the QA program interfaces with both the production route and process specifications. The QC plan satisfies the overall QA program requirements. The document(s) identify the points in the fabrication process in which QC is required, including inspection and test plans. For each check point, the specific test to be performed, the characteristics to be measured, the frequency of the test, the number of samples, and the acceptance criteria are identified. Furthermore, the QC plan identifies the responsible tester, either manufacturing or QA personnel.

An important feature of the QC plan is that the level of testing agreed to by both the fuel designer and fabricator depends on the experience of the fabricator.[1] During initial startup activities (either for the plant or for a new product), key process parameters should be monitored, and key characteristics of intermediate and final products should be measured. This intensive effort is required to ascertain the variability in the process and the product. During the qualification period, a smaller number of process parameters is monitored based on the information obtained during the startup period. During qualification, a high sampling rate on intermediate and final products should continue to be implemented until the product variability is understood. During the final period of normal fabrication, a much reduced set of parameters is monitored as part of the QC program. Only those characteristics of intermediate or final products specified by the

customer or found indispensable by the QA personnel are measured. It is incorrect to assume that implementation of the QA/QC program for a large commercial fabrication plant in a development facility will ensure a quality product. In fact, the QA/QC requirements for any development facility should be more stringent to ensure product quality consistent with the well-characterized commercial processes.

The fabrication and inspection procedures comprise another important series of fabrication/QA/QC documents. Procedures are developed for both manufacturing and inspection activities. Each procedure describes the operation of the specified piece of equipment, with instructions to personnel concerning startup of the equipment, qualified levels or settings, safety issues, and calibration/maintenance requirements. A procedure is a step-by-step set of instructions for the operator. It may also provide the basis for operator or inspector training. Procedures may also identify the QC inspection and release points in the process.

The final part of the fabrication documentation is the traveler and data record on which all important information concerning the fabrication process is documented. These records are generally maintained on the largest quantity of uniform material, such as a blending powder batch or sintering batch of pellets. The traveler or routing card contains the sequence of processing, inspection, and test operations for the component. For each of these operations, a signature of the person completing the operation and the date of completion are placed on the traveler. The detailed results of the tests and inspections are maintained on separate data records.

REFERENCE

1. *Guidebook on Quality Control of Water Reactor Fuel, Technical Reports Series No. 221,* STI/DOC/10/221, International Atomic Energy Agency, Vienna (1983).

APPENDIX C

LWR RELOAD CORE LICENSING

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APPENDIX C. LWR RELOAD CORE LICENSING

For domestic LWRs, the NRC must approve a new fuel design prior to its insertion and use in a commercial reactor unless the new fuel design falls within a preapproved or licensed envelope. The NRC performs its review and grants approval according to the legal requirements set forth in the *Code of Federal Regulations* (CFR) and the individual operating licenses. The overall process by which a utility pursues regulatory approval to reload the core is shown via a logic diagram in Fig. C.1. This logic diagram is an expanded version of part of the operation given in Fig. 1 in Sect. 3.4 as "Implement mission fuel use." Figure C.1 makes clear the advantage of an uncontested reload application. Utilities have, therefore, become experienced in developing reload applications that avoid major regulatory hurdles.

Because the requirements promulgated in the CFR are legally binding and, in some instances, complex in nature, the NRC has issued a series of regulatory guides that provide applicants and licensees with an acceptable interpretation of the legal requirements. The regulatory requirements for evaluating the safety impact of licensee-contemplated changes in the reactor design, including design changes associated with core reloads at refueling, are given in Title 10, Sect. 50.59, Part 50 of the CFR (10 CFR 50.59). A final regulatory guide for refueling of commercial LWRs has never been published, but NRC has issued a number of documents that can be used by the licensees for guidance in performing safety analyses for determining the need for a license amendment application or that are used in NRC's review for preparing safety evaluations of license amendment applications:

- NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, July 1981 as revised through October 1990 (the acceptance criteria for changes in fuel design are found principally in Chaps. 4, 9, and 15; NUREG-0800 is the subject of an on-going NRC effort to update).
- "Guidance for Proposed License Amendments Relating to Refueling," Enclosure 1 to NRC untitled letters sent individually to each LWR licensee and signed by K. R. Goller on either June 18 or 23, 1975.
- Draft *Regulatory Guide SC 521-4*, "LWR Core Reloads; Guidance on Applications for Amendments to Operating Licenses and on Refueling and Startup Tests," September 1979 [see *Federal Register* 44 (195), p. 57541, October 5, 1979].
- NRC Generic Letter 83-11, "Licensee qualifications for performing safety analyses in support of licensing actions," February 8, 1983.
- NRC Generic Letter 84-20, "Scheduling guidance for licensee submittals of reloads that involve unreviewed safety questions," August 20, 1984.

NRC is currently seeking to provide unified guidance on the use of 10 CFR 50.59. The current proposal was submitted by the NRC to staff of the Commission in SECY-97-035, "Proposed guidance with regard to implementation of 10 CFR 50.59," February 12, 1997. Pending revised guidance, the Draft *Regulatory Guide SC 521-4* forms the basis for much of the following description. The NRC issued SC 521-4 as a means of easing the regulatory burden associated with routine refueling applications. Under 10 CFR 50.59 (a)(1),

> the holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test, or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question.

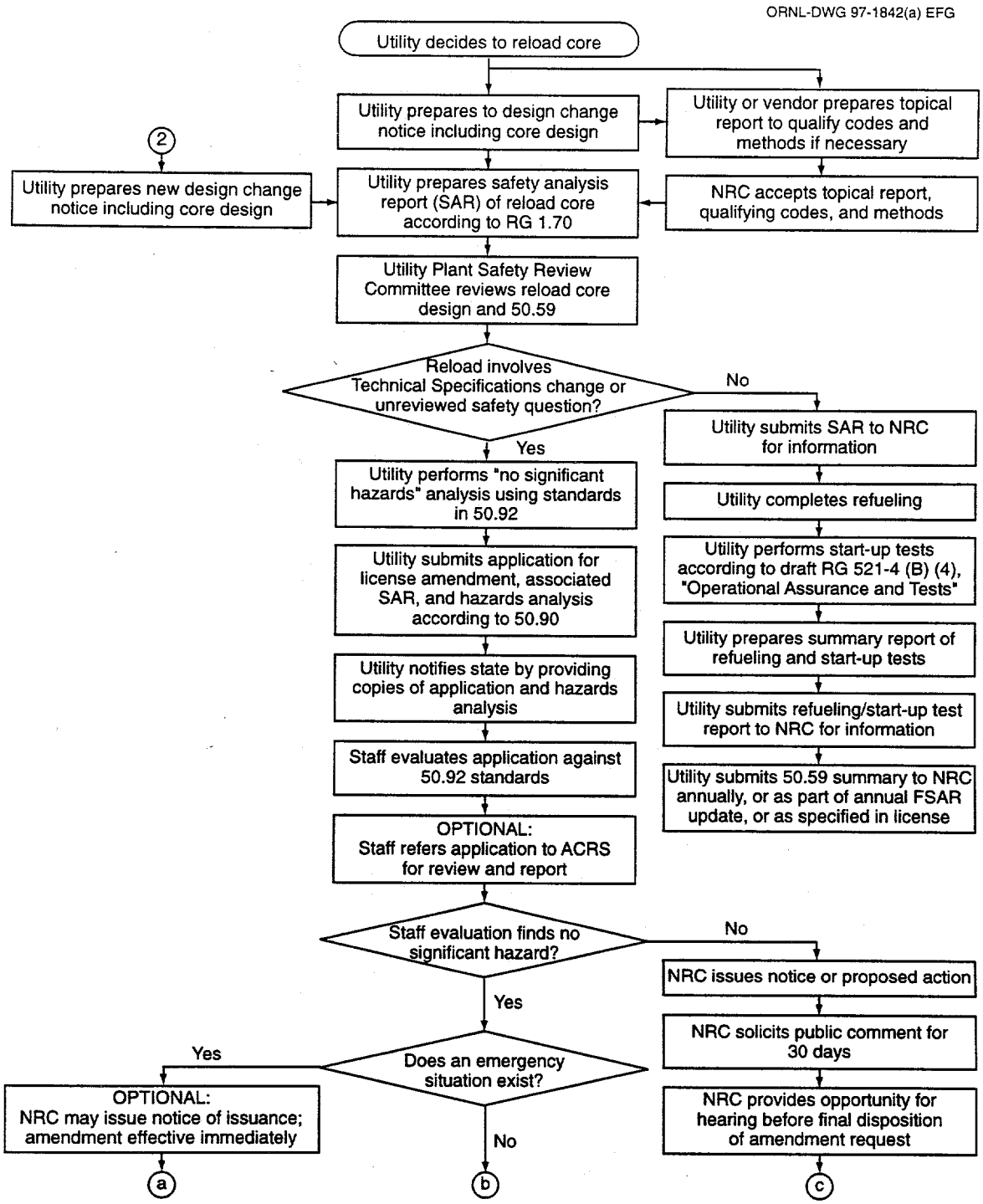

Fig. C.1. NRC reload licensing process.

ORNL-DWG 97-1842(b) EFG

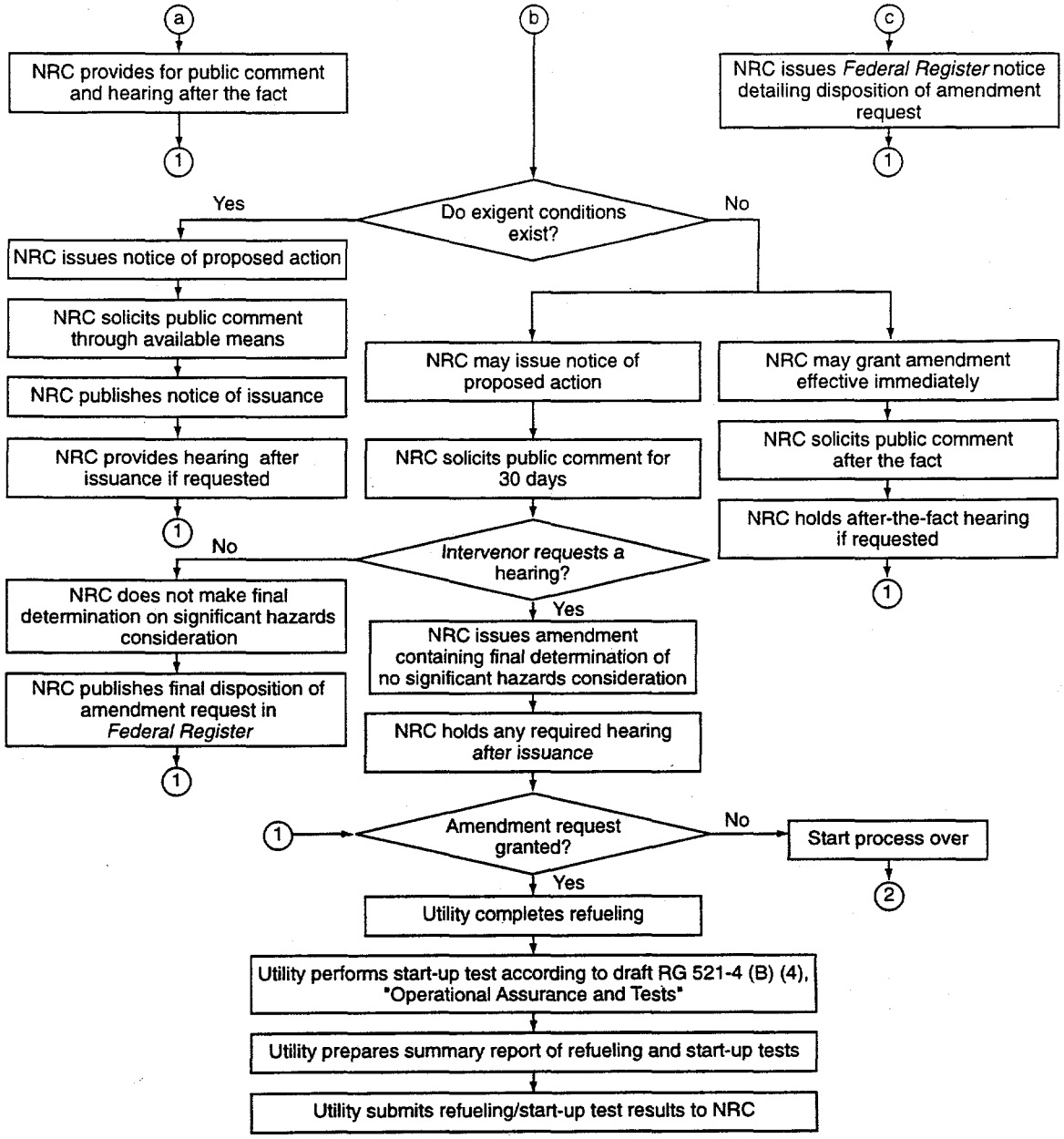

Fig. C.1 (continued).

Routine refuelings involve neither a change in the Technical Specifications nor an unreviewed safety question. The Technical Specifications are additional reactor-specific requirements that are included as an appendix to the operating license and are therefore legal requirements that must be met. An unreviewed safety question is defined in 10 CFR 50.59 (a) (2):

> A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

For a reload involving neither changes to the Technical Specifications nor an unreviewed safety question, the licensee prepares a document, sometimes referred to as a Design Change Notice or a Design Change, that identifies the changes that will accompany the refueling. These may include setpoint changes, control rod changes, documentation updates, the new core map identifying locations for all assemblies, etc. A second document prepared in parallel is the core reload safety analysis report (SAR). The reload SAR in fact provides much of the basis for the determination under 10 CFR 50.59 that the refueling involves neither a Technical Specifications change nor an unreviewed safety question. In many instances, the utility may simply reference a generic SAR prepared by the fuel vendor and already accepted by the NRC. Such a generic SAR would have been provided to NRC in a vendor topical report that NRC would have reviewed and approved using as a minimum the acceptance criteria from the appropriate sections and chapter of the NUREG-0800 *Standard Review Plan*. Only plant-specific issues would be addressed in the utility's SAR. In either case, the licensee then proceeds with the reload without prior NRC approval. This scenario is known as a "50.59 reload" and is the simplest path for accomplishing the task.

For a 50.59 reload, the licensee is required to maintain records of the SAR and usually provides a copy of the SAR to the NRC for information only. A summary of the changes, tests, and experiments performed under 50.59 must be prepared by the licensee annually. This report must also include a summary of the safety analysis for each change, test, and experiment. This report may be included as part of the annual final safety analysis report (FSAR) update required by 10 CFR 50.71, or as otherwise specified in the specific operating license.

The use of MOX (mixed uranium-plutonium oxide) fuel in domestic commercial reactors is unlikely to be allowed under 50.59. Most of the plant Technical Specifications include a design description of the core that specifies uranium oxide fuel. For example, the Improved Standard Technical Specifications for Westinghouse plants describe the reactor core in the following manner:

> The reactor shall contain [157] fuel assemblies. Each assembly shall consist of a matrix of [Zircaloy or ZIRLO] fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Although this core description does provide for limited insertion of LTAs, the initial loading of all of the MOX test assemblies that were irradiated domestically required a license amendment to modify the section of the Technical Specifications quoted above. These modifications were granted for Big Rock Point, San Onofre 1, Quad Cities 1, Dresden 1, and R. E. Ginna. Assuming a license amendment is required to address this core description in the Technical Specifications, according to 50.59(c):

> The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in Technical Specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in Technical Specifications, shall submit an application for amendment of his license pursuant to 50.90.

Section 50.90 simply indicates that the licensee must file the appropriate application for amendment:

> Whenever a holder of a license or construction permit desires to amend the license or permit, application for an amendment must be filed with the Commission, as specified in 50.4, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

The form prescribed for original applications, according to Draft *Regulatory Guide SC 521-1*, is outlined in *Regulatory Guide 1.70*, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Revision 3, November 1978. Unfortunately, *Regulatory Guide 1.70* does not address refueling specifically. This is acknowledged in SC 521-1, which provides two criteria the refueling safety analysis must meet:

1. The safety analysis performed on the reload fuel and the refueled core should use currently approved analytical and calculational procedures and should satisfy all aspects of the safety review described in *Regulatory Guide 1.70*.
2. The safety analysis should address differences between reload fuel and irradiated fuel remaining in the core (or being returned to the core) that involve unreviewed safety questions or changes in technical specifications.

In addition to the SAR, 10 CFR 50.91 requires the utility to provide the NRC with the results of a "no significant hazards" analysis as outlined in 10 CFR 50.92. No significant hazard is involved

> ...if operation of the facility in accordance with the proposed amendment would not:
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
> (3) Involve a significant reduction in a margin of safety.

Upon receipt of the utility's application for license amendment, associated SAR, and "no significant hazards" analysis, the NRC's actions are outlined in 10 CFR 50.91, "Notice for Public Comment; State Consultation." The regulations concerning the Commission's required actions are somewhat ambiguous, providing the Commission with several optional paths. The staff makes its own significant hazards determination using the criteria in 10 CFR 50.92. If the proposed staff determination finds a significant hazards issue, the Commission issues a notice of proposed action in the *Federal Register* soliciting public comment. After the 30-day comment period, the

NRC provides an opportunity for a public hearing prior to the final disposition of the amendment request, and then issues a *Federal Register* notice detailing the disposition of the request.

If the staff evaluation finds no significant hazards and neither emergency nor exigent conditions exist, the NRC may either (1) grant the amendment request effective immediately, solicit public comment after the fact, and provide for an after-the-fact hearing if requested; or (2) issue a notice of proposed action, solicit public comment, issue the amendment request, and provide for an after-the-fact hearing if requested. The NRC is only required to hold a public hearing prior to disposition of the amendment request if a significant hazard is involved.

If the license amendment request is granted, the utility completes the refueling. A series of startup tests is performed, including those specified in draft *Regulatory Guide 521-4 (B) (4)*, "Operational Assurance and Tests" as follows:

Recommended tests for PWRs include

- Control rod drive tests and drop time tests (hot).
- Comparisons of predicted and measured values of moderator temperature coefficient, critical boron concentration, rod bank worths, and local power at the actual detector locations.
- A check of core symmetry by comparing symmetric detector readings and by comparing worths of symmetric rods.

Recommended tests for BWRs include

- A check of core power symmetry by checking for mismatches between symmetric detectors.
- Withdrawal and insertion of each control rod to check for criticality and mobility.
- Comparison of predicted and measured critical in-sequence rod pattern for nonvoided conditions.

The utility prepares a summary report of refueling and the startup tests and submits it to the NRC.

Another important part of fuel qualification from the regulatory perspective concerns the safety analyses that must be performed for any reload core, regardless of whether a license amendment is required. These safety analyses must be performed using approved codes and methods. No legal requirements concerning the approval process have been found in the CFR or the regulatory guides. However, this should not be taken as an indication of the lack of importance of this activity. Codes and methods approval is a licensing activity. It may also be an integral part of qualification testing. Approval of codes and methods is often not required for new UO_2 fuel designs, but will be required for MOX fuel. The data needed for code validation may be developed through qualification testing, purchased, or developed independently of the qualification testing. Because the codes and supporting data are for the most part proprietary, it is difficult to assess their current applicability to MOX fuel.

The utility's fuel designer or fuel fabricator would prepare a licensing topical report covering the codes and methods to be utilized. This topical report would demonstrate the applicability of the codes and methods through incorporation of validation data. The key is convincing the Commission through this licensing topical report that the licensee has sufficient data and competency to ensure the accuracy of the analyses in the SAR. Because of the similarity between UO_2 and MOX fuels, a licensee might be able to convince the Commission of the applicability of the existing urania codes with minor increases to the safety margin applied to the results. Such an approach is much more likely to be accepted for loading of LTAs than for an entire reload.

An additional consideration is the similarity of the proposed MOX fuel design to currently operating UO_2 designs. Use of identical hardware and operation within the licensed UO_2 envelope is likely to reduce the licensing burden.

In summary, because changes will likely be required for several of the descriptive passages in the plant Technical Specifications (in addition to any substantive changes) for MOX fuel utilization, a license amendment will be required for each utility that utilizes the MD MOX fuel. Thus, the site-specific approach will most likely be required at least for some portion of the overall licensing process leading to the use of MOX fuel in U.S. LWRs. Under this scenario, the key to a timely and successful implementation of the MOX fuel irradiation strategy will be the objective demonstration by the responsible utility/fuel vendor group that the use of MOX fuel in their reactor(s) is, from an overall licensing standpoint, a transparent change; that is, the effects of switching from UO_2 to MOX fuel on the operational safety margins documented in the facility FSAR are technically negligible. Consequently, MD, in its tasking to the utility/fuel vendor groups being considered for executing the MOX fuel irradiation strategy, should require documentation describing (1) the specific changes in plant licensing documentation required to use the MD MOX fuel in their reactor(s), and (2) their approach and schedule for obtaining, where needed, NRC concurrence with the changes. Subsequently, the utility/fuel vendor group selected by DOE should be tasked to submit the necessary licensing documentation changes on a time table that is consistent with U.S. strategic interests for accomplishing the MD mission and, at the same time, will lead to receipt of NRC approval of the changes well in advance of MOX fuel being shipped to the facility for use.

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