DUKE COGEMA
STONE & WEBSTER

Mr. Andrew Persinko, MOX Project Manager
U.S. Nuclear Regulatory Commission
MS T8A33
Washington, DC 20555

14 July 2000
DCS-NRC-000021

SUBJECT: Transmittal of Information

Dear Mr. Persinko:

Please find enclosed two reports: ORNL/MD/LTR-78, *Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan*; and ORNL/TM-13428, *Survey of Worldwide Light Water Reactor Experience with Mixed Uranium-Plutonium Oxide Fuel*. NRC personnel requested a copy of these reports during a 29 July 2000 conference call with DCS regarding fuel performance benchmarking. Please let me know if you require any additional information.

Sincerely,

Peter S. Hastings
Licensing Manager

xc (without enclosure):

Richard H. Clark, DCS
Robert H. Ihde, DCS
Larry Losh, FCF
Steve P. Nesbit, DPC
Jon Thompson, USDOE-HQ
PRA/EDMS:\Corresp\Outgoing\NRC\DCS-NRC-000021

PO Box 31847
Charlotte, NC 28231-1847

400 South Tryon Street, WC-32G
Charlotte, NC 28202

OAK RIDGE
NATIONAL LABORATORY

MANAGED BY UT-BATTELLE
FOR THE DEPARTMENT OF ENERGY

ORNL/MD/LTR-78
Level 1

Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan

Revision 2

B. S. Cowell
S. A. Hodge

May 2000

Fissile Materials Disposition Program

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

UT-BATTELLE
ORNL-27 (4-00)

Revision History

Revision Number	Date Issued	Reason for Revision
0	July 1997	
1	February 1998	To revise Table 3 to reflect actual and current best-estimate milestone dates.
2	May 2000	Revise Sections 6.0 and 8.0 to incorporate plans for extension of burnup to 50 GWd/MT for some capsules. Also, miscellaneous text changes as necessary to update to current status.

OAK RIDGE NATIONAL LABORATORY

MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

Engineering Technology Division
P.O. Box 2009
Oak Ridge, TN 37831-8057
Tel.: (865) 574-0573
Fax: (865) 574-8216
hodgesa@ornl.gov

May 8, 2000

Distribution

Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan—ORNL/MD/LTR-78, Level 1 (Revision 2)

Revision 0 of this Test Project Plan for the irradiation of MOX fuel prepared from weapons-derived plutonium at representative average linear heat generation rates within the Advanced Test Reactor (ATR) was issued in July 1997. This Project constitutes Task 4.1.1 "Irradiation Tests and Post Irradiation Examination," as described in the FY 2000 Annual Operating Plan for the Fissile Materials Disposition Program under the Department of Energy's Office of Fissile Materials Disposition.

Revision 1 was issued in February 1998 to revise the estimated milestone completion dates previously listed in Table 3 to reflect the actual milestone status at that time.

Revision 2 is now issued to incorporate plans for the extension of burnup for some capsules from the previously planned 30 GWd/MT to 50 GWd/MT. These changes include modifications to cost and milestone estimate tables as appropriate. The text is also modified throughout the report as necessary to reflect the current status of the test irradiation.

It is important to note that this Test Project Plan continues to pertain only to the average power test. A separate document will be prepared for any other test to be conducted under the auspices of this project in the future.

The text modifications incorporated by Revision 2 were approved by Bob Pedersen at INEEL, Don Spellman at ORNL, plus Jon Thompson and Pat Rhoads at DOE. The signature sheet circulated with Revision 0 is considered to remain applicable and is included with this release of Revision 2. This is a Level-1 document as defined in Section 7.0, *Project Documentation Hierarchy and Control* of this Test Project Plan. Any future modifications to this average power test plan will be promulgated only through appropriate revisions to this document.

Sincerely,

Stephen A. Hodge

Stephen A. Hodge, Manager
MOX Irradiation Test Project

SAH:rcp

Enclosure: As Stated

Distribution:

D. Alberstein, LANL
R. G. Ambrosek, INEEL
G. S. Chang, INEEL
K. Chidester, LANL
B. S. Cowell
S. R. Greene
S. B. Grover, INEEL
D. W. Heatherly
W. R. Hendrich
P. Kasik, MPRA
L. Losh, FCF
S. B. Ludwig
C. R. Luttrell
D. C. Mecham, INEEL
W. A. Miller
R. N. Morris
L. J. Ott
J. V. Pace
R. C. Pedersen, INEEL
P. T. Rhoads, DOE
R. A. Roesener, INEEL
J. M. Ryskamp, INEEL
D. J. Spellman
W. K. Terry, INEEL
J. Thompson, DOE
K. R. Thoms
G. T. Yahr
Files–SAH–NoRC

ORNL/MD/LTR-78
Level 1

Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan Revision 2

B. S. Cowell
S. A. Hodge

May 2000

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
managed by
UT-BATTELLE, LLC,
for the
U. S. DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan

Revision 0

Oak Ridge National Laboratory

Signature sheet initially distributed with first issue of this document.

B. S. Cowell

S. A. Hodge

Concurred by	John M. Ryskamp (INEEL)	7/7/97 Date
Concurred by	David Alberstein (LANL)	7/2/97 Date
Submitted by	Sherrell R. Greene (ORNL)	7/15/97 Date
Approved by	Patrick T. Rhoads (DOE/MD)	7/22/97 Date

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
managed by
LOCKHEED MARTIN ENERGY RESEARCH CORP.
for the
U. S. DEPARTMENT OF ENERGY
under contract DE-AC05-96OR22464

Table of Contents

Page

1.0 Introduction 1

2.0 Purpose and Scope of the Irradiation Test Project 1

3.0 Goals, Assumptions, and Requirements 2

4.0 Technical Issues Addressed 3

5.0 Organizational Roles and Responsibilities 4

6.0 Detailed Description of the Demonstration 5

7.0 Project Documentation Hierarchy and Control 6

8.0 Cost and Schedule Estimates 8

9.0 Summary 10

Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan

1.0 Introduction

The United States Department of Energy Fissile Materials Disposition Program (FMDP) has announced that reactor irradiation as mixed uranium-plutonium oxide (MOX) fuel is being pursued for disposal of surplus weapons-usable plutonium (Pu). Although MOX fuel is not currently utilized domestically, it is widely employed in a number of foreign countries. MOX fuel utilization is supported by a large body of MOX fuel irradiation experience that has been generated through research, development, and deployment programs since the mid-1950s. MOX fuel has been utilized domestically in test reactors and on an experimental basis in a number of commercial light water reactors (CLWRs). Over 300,000 MOX fuel rods have been successfully irradiated in the United States and Europe. Most of this experience has been with reactor-grade plutonium, which is derived from spent low enriched uranium (LEU) fuel. To pursue disposition of surplus weapons usable plutonium via reactor irradiation, it must be demonstrated that the unique properties of the surplus weapons-derived or weapons-grade (WG) Pu do not compromise the applicability of this MOX experience base.

2.0 Purpose and Scope of the Irradiation Test Project

One of the challenges facing the FMDP is to demonstrate that substitution of WG Pu for the reactor-grade (RG) Pu in commercial MOX fuel does not affect deleteriously the fuel performance, and that the commercial MOX experience base is therefore applicable. It is the purpose of this test project to contribute new information concerning the response of WG Pu under irradiation. The philosophy behind most MOX fuel development and qualification efforts is that MOX fuel is 95 percent UO_2, and from a materials standpoint, should therefore behave similarly to UO_2 fuel. This philosophy has been adopted and modified for the FMDP mission—plutonium constitutes but a small fraction of the material in MOX fuel, so WG MOX fuel behavior should be similar to that of both RG MOX fuel and UO_2 fuel.

The MOX fuel irradiation demonstration described herein is an initial step toward confirmation of this assertion. The primary focus of the irradiation tests is to address some important outstanding technical issues for the deployment in CLWRs of MOX fuel cycles based upon weapons-derived plutonium. Initial test planning included the provision that the technical objectives of the demonstration project would be limited to those generic issues that could be addressed without biasing programmatic procurement activities.

The LWR MOX fuel irradiation tests will irradiate MOX fuel produced in the TA-55 facility at the Los Alamos National Laboratory (LANL). Two types of MOX fuel pellets are being

irradiated in order to investigate some unresolved generic fuel development/qualification issues. Simple, uninstrumented, drop-in capsules with local flux monitor wires are inserted in the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL). Postirradiation examination (PIE) of this fuel is performed in the Irradiated Fuels Examination Laboratory at the Oak Ridge National Laboratory (ORNL).

The project's goals, assumptions, and requirements are described in Section 3. Section 4 contains a discussion of the technical issues addressed in the demonstration. Section 5 explains the organizational roles and responsibilities. Technical details of the project test matrix are provided in Section 6. Section 7 describes the hierarchy of project documentation and the methods for document control. Cost and schedule estimates are outlined in Section 8. Finally, Section 9 provides a brief summary of the overall project.

3.0 Goals, Assumptions, and Requirements

The four top-level ***goals*** of this Project are:

1) Demonstrate the utilization of Pu derived from weapons components* in a light water reactor (LWR) environment.
2) Contribute experience with irradiation of gallium-containing fuel to the data base required for resolution of generic LWR WG MOX fuel design issues.
3) Initiate irradiation of LWR WG MOX fuel in CY 1997.
4) Exercise the infrastructure necessary to promote WG MOX fuel irradiation by successfully demonstrating abilities to convert Pu metal from weapons components to oxide, fabricate MOX fuel, transport the fresh fuel, irradiate the fuel, transport the irradiated fuel, and perform the postirradiation examination.

The test will emphasize the development of additional information toward the resolution of generic performance issues to assist in mission fuel licensing and utility acceptance. Several additional ***requirements*** imposed on the test activity are summarized below:

1) All test fuel will be produced in the TA-55 facility at LANL.
2) The test will not assess issues related to inclusion of burnable poisons in MOX fuel.
3) The test fuel will be fabricated to meet a generic LWR MOX fuel pellet specification developed by ORNL using process specifications developed by LANL.
4) The test will include a comparison of the behaviors of test fuels with and without thermal treatment for removal of gallium.

* The surplus plutonium inventory contains other WG Pu material besides the weapons components, but all references to WG Pu in this plan pertain only to material derived directly from dismantled weapons components.

5) The plutonium for the test fuel will be derived from one or more weapons components. At least a portion of this material will be derived from components containing the maximum available gallium concentration (~1 weight percent).
6) Test conditions will reproduce LWR operating temperatures (clad and centerline) to the extent possible, as explained in the detailed Design, Functional, and Operational Requirements Document. (The thermal gradient across the fuel is implicitly determined by the specified values for the pellet surface temperature and the linear heat generation rate.)
7) The selection of fuel dimensions, cladding, fuel specifications, and burnup will be accomplished in a manner that does not bias future programmatic procurement activities.
8) The test fuels can be removed from the reactor at selected points within a range of burnups.
9) Domestic facilities will be used for fabrication, irradiation, and PIE.

4.0 Technical Issues Addressed

Several issues must be resolved prior to implementation of WG MOX use in CLWRs. Because this demonstration project was initiated before selection of a mission fuel design and fabrication process, only generic issues are addressed here. Three generic issues that must be considered in the final fuel design are: (1) the effects of gallium, (2) the specific isotopics of WG Pu, and (3) the use of hydride-derived PuO_2 in lieu of aqueous-derived PuO_2. These three issues are addressed in the planned irradiation test activities.

Gallium is an alloying agent present in WG Pu at concentrations up to approximately one weight percent. The technical issue is whether the small quantities of gallium present in the feed plutonium metal and the finished MOX fuel will adversely affect either MOX fuel fabrication or irradiation performance. *Residual gallium concentration is the primary variable of interest in the MOX fuel types to be produced by LANL.* One batch of fuel was fabricated from Pu feed that contains approximately one weight percent gallium and without special treatment for removal of impurities. The second fuel batch was made with a nominal one weight percent gallium feed processed with a PuO_2 powder thermal-conditioning step to remove the gallium. The gallium concentration was measured as necessary to map its evolution through the various processes.

The second generic issue to be addressed in this demonstration project is the specific isotopics of WG Pu. Early MOX fuel was made from plutonium recovered from low burnup UO_2 fuel, or from military stocks. However, only a very limited quantity of MOX was made from this high grade plutonium (low Pu^{240} content). Almost all of the commercial MOX fuel experience is with RG Pu, which is recovered from high burnup UO_2 fuel and contains an appreciable quantity of the higher isotopes (primarily Pu^{240} and Pu^{241}). Differences in nuclear characteristics are apparent between fuels made with the different Pu feeds. The fabrication, handling, performance prediction, and actual irradiation performance of WG MOX fuel will be demonstrated by this Project.

The final generic issue to be addressed is the determination of the effects of variations in the metal-to-oxide conversion process. The RG PuO_2 used as feed in commercial MOX is produced almost exclusively through precipitation of plutonium oxalate from aqueous nitric acid solution. The resulting powder has a uniform and well-characterized morphology, which assists in the achievement of a uniform finished MOX fuel product. Several dry pyroprocesses have also been investigated for possible application to the FMDP mission. The baseline pyroprocess, identified by the ARIES project, was used to convert the weapons components into the feed oxide powder for the two MOX test fuels.

5.0 Organizational Roles and Responsibilities

This test project is sponsored by the Department of Energy Office of Fissile Materials Disposition (DOE-MD), which is responsible for oversight to ensure that the goals of the project are consistent with the FMDP program objectives. DOE-MD arranges funding authority for the accomplishment of the test. Oak Ridge National Laboratory (ORNL), as lead laboratory for reactor alternatives for fissile materials disposition, manages the program for DOE-MD. As lead laboratory, ORNL coordinates and oversees the activities of the other parties to ensure success of the project and report on progress, schedule, and cost for the entire project to DOE.

Acting as the program manager, ORNL has the principal responsibility to ensure that the irradiation test program meets the approved goals, requirements, and technical issues outlined in Sections 3 and 4. ORNL has developed the pellet and fuel pin specifications, designed and fabricated the irradiation basket and capsules, and predicted the fuel behavior and ultimate burnup. ORNL has the continuing responsibility to perform the postirradiation examination, package and dispose of the waste materials, and report the test results.

As lead laboratory for MOX fuel fabrication, Los Alamos National Laboratory (LANL) is responsible for developing the process parameters and producing the test MOX fuels in accordance with the pellet specifications and drawings. LANL obtained a source of weapons-derived PuO_2 feed stock including the available technical data showing its processing history and characteristics. In this effort, Lawrence Livermore National Laboratory (LLNL) has assisted LANL as a support laboratory, reporting to ORNL and DOE through LANL.

LANL loaded the MOX fuel pellets into the fuel cladding supplied by ORNL and performed the seal welding of the fuel pins. LANL provided the required data on the characteristics of the finished fuel pellets as specified in the Fabrication, Inspection, and Test Plan. LANL was also responsible for the packaging, transportation, safeguards and security, emergency response, and appropriate notifications of the finished fuel transfer to Idaho National Engineering and Environmental Laboratory (INEEL).

INEEL is the operator of the Advanced Test Reactor (ATR) where the test is conducted, and as such, has the responsibility for ensuring that the test is designed and operated in compliance with all of the applicable safety and regulatory requirements. INEEL loaded the

fuel pins obtained from LANL into the stainless steel capsules and performed the seal welding of these capsules. Specifications, drawings, reactor data, and other guidance provided by INEEL was used by LANL and ORNL as a basis for ensuring that the test design meets the requirements of the ATR. INEEL performs the necessary tests and calculations and provides the documentation to permit the test insertion, conduct the test irradiation, and remove test rods at prescribed burnups. INEEL is responsible for the packaging, safeguards and security, emergency response, appropriate notifications, and transportation to ORNL of the irradiated test fuels for disassembly and PIE. Funding authorization for the INEEL activities is executed through ORNL.

To facilitate decision-making and communications, the three laboratories involved in the planning and implementation of the LWR demonstration test have designated representatives and alternates to participate in teleconferences normally conducted weekly. These teleconferences are led by ORNL, which, as part of its overall project management role, is responsible for obtaining consensus on issues emerging during implementation of the test. Any issues that cannot be resolved among the project participants will be brought to DOE-MD for resolution.

6.0 Detailed Description of the Demonstration

The LWR demonstration test matrix, Table 1, describes the two test-fuel types that address the generic issues described above. The compositions of these test fuels have been established to focus primarily on the behavior and acceptability of gallium impurities in the fuel. These tests will be used to confirm and extend the results of the out-of-pile Ga-clad corrosion tests conducted by researchers at ORNL and at the Amarillo National Resource Center for Plutonium and will complement the gallium removal and fuel fabrication research activities conducted at LANL and LLNL.

Test fuel 1 is fabricated from weapons components containing about 1 weight percent gallium and converted to oxide through the hydride process identified by the ARIES project. Test fuel 2 is identical to test fuel 1 with the addition of a gallium-removal thermal-processing step for the plutonia powder.

Both fuels are fabricated to the generic LWR MOX fuel pellet specification developed specifically for these tests. Generic zirconium alloy (Zircaloy) is utilized for cladding. The pellet size and shape are generic, representative of LWR pellet geometry but identical to none of the commercial fuels. The pellet diameter is determined by the available sizes of off-the-shelf zirconium alloy cladding tubes. The uranium diluent for the test fuels was derived from a single lot of depleted uranium powder converted via the ammonium diuranate (ADU) process.

Both of the WG MOX test fuels contain a nominal 5 percent total plutonium (measured as mass of plutonium metal in mass of total metal), which is equivalent to 4.7 percent fissile Pu. The nominal linear heat generation rate for the tests is 8 kW/ft; however, the test objectives can be satisfied for heat rates within the range 2–10 kW/ft as described in the Design, Functional, and Operational Requirements Documents.

One test fuel pin of each type was removed from the reactor after reaching about 8 GWd/MT, to provide an opportunity for early indication of the effects of residual gallium at low burnup. One additional fuel pin of each type was removed after reaching about 20 GWd/MT, to provide intermediate indication of any developing trends. The remaining seven fuel pins will be irradiated to approximately 30, 40, or 50 GWd/MT in accordance with Table 1.

Table 1. Test Matrix

Fuel Type	Description[a]	Initial Feed[b]	Pu Purification	Pu to PuO_2 Conversion	BU[c] [GWd/MT]
Test 1A	5% WG Pu MOX	1% Ga WG Pu	none	hydride	8
Test 1B	5% WG Pu MOX	1% Ga WG Pu	none	hydride	20
Test 1C	5% WG Pu MOX	1% Ga WG Pu	none	hydride	30
Test 1D	5% WG Pu MOX	1% Ga WG Pu	none	hydride	40
Test 1E	5% WG Pu MOX	1% Ga WG Pu	none	hydride	50
Test 1F	5% WG Pu MOX	1% Ga WG Pu	none	hydride	50
Test 2A	5% WG Pu MOX	1% Ga WG Pu	thermal	hydride	8
Test 2B	5% WG Pu MOX	1% Ga WG Pu	thermal	hydride	20
Test 2C	5% WG Pu MOX	1% Ga WG Pu	thermal	hydride	30
Test 2D	5% WG Pu MOX	1% Ga WG Pu	thermal	hydride	40
Test 2E	5% WG Pu MOX	1% Ga WG Pu	thermal	hydride	50

[a] The plutonium concentration values are ± 1% relative (0.0500±0.0005).

[b] The 1% Ga feed specification is a nominal value.

[c] These are the target burnups. Irradiation began in February 1998, and cumulative calendar time requirements to reach actual burnups achieved by end FY 2000 are: 8.6 GWd/MT – 7 months, 20.9 GWd/MT – 19 months, and 29.6 GWd/MT – 29 months. Estimates for succeeding burnups are: 40 GWd/MT – 50 months, and 50 GWd/MT – 69 months.

With respect to NEPA requirements, these tests are subject to a categorical exclusion according to 10CFR, Part 1021, Subpart D, Appendix B, Section B.3.10, to wit: "Small-scale research and development projects and small-scale pilot projects conducted (for generally less than two years) to verify a concept before demonstration actions, performed in an existing structure without major modification."

7.0 Project Documentation Hierarchy and Control

The document hierarchy for this project comprises three levels as necessary to provide for the appropriate degrees of oversight while maintaining, at the working level, the flexibility essential to timely completion of the project milestones.

Level 1: Project Plan

This is the top-level, controlled document that completely identifies and defines the Mixed Oxide Fuel Irradiation Test Project. For maximum effectiveness, length is restricted to no more than ten pages. The level of detail includes:

- top level purpose for and description of project
- test matrix in table format
- identification of participating organizations and their respective roles
- major milestones
- cost estimates

Approval: DOE-MD
Concurrence: Lab leads and/or project leads at ORNL, LANL, and INEEL
Other review: MPR Associates

Level 2: Controlled Working Documents

These are the controlled detailed working documents that specifically guide the performance of the various steps of the test project. The Level-2 documents address all features of fuel fabrication, irradiation, and PIE and include

- INEEL Project Management Plan
- Design, Functional, and Operational Requirements
- Thermal/hydraulic Calculations
- Design Calculations (Stress Analyses)
- Technical Specification: Mixed Oxide Pellets for the Light-Water Reactor Irradiation Demonstration Test
- Capsule Loading and Operation Schedule
- Fabrication, Inspection, and Test Plans
- Purchase Orders for Pellets, Pins, and Capsules
- Transportation Plan
- Post Irradiation Examination (PIE) Plan
- Quality Assurance Plan per DOE Order 414.1
- Technical Evaluation Report(s)

Approval: Line management at originating organization
Concurrence: Lab leads and/or project leads at ORNL, LANL, or INEEL, as appropriate
Information Copies: DOE-MD, MPR Associates.

Level 3: Non-controlled Working Documents

These are the working-level documents most susceptible to requirements for rapid generation, implementation, and change. They include communications (e-mail, fax, letter) between participating organizations as well as working documents subject to purely internal controls within a particular organization. For the latter category, which comprises documents such as internal project plans, monthly reports, and meeting minutes, internal organizational approval practices will be followed. Some of these Level-3 documents may be transmitted to DOE-MD for information purposes, but for these, neither approval nor concurrence will be sought.

This three-level approach is consistent with the preferred approach described in the Operations Manual (OM) under which DOE-MD approval items are separated from lower-level information into a document hierarchy. This Level-1 document satisfies the deliverable for a project plan for the LWR Demonstration as described in the FY 1997 and subsequent Annual Operating Plans.

8.0 Cost and Schedule Estimates

All expenditures for this Project during any year are in accordance with the FMDP Annual Operating Plan (AOP) for that year, which is a controlled document and is periodically revised. The actual costs for FYs 1997–1999 and the current cost *estimates* for the remaining period of the LWR Demonstration Project are summarized in Table 2.

Table 2. Cost Estimates ($M)

	FY 1997	FY 1998	FY 1999	FY 2000	FY 2001
ORNL					
Direct LWR Demonstration	1.23	1.39	1.25	0.80	0.70
PIE Preparation and Performance	0.09	0.55	0.98	1.00	1.10
Fuel/Reactor Vendor Subcontracts	0.21	0.23	0.13	0.10	0.10
LANL					
Direct LWR Demonstration	1.50	1.26	0.47	—	—
INEEL	0.71*	1.18*	0.91	0.80	0.75
Total	3.74	4.61	3.74	2.70	2.65

* Includes $0.23M (FY 1997) and $0.02M (FY 1998) direct funding from DOE-MD.

	FY 2002	FY 2003	FY 2004	FY 2005
ORNL				
Direct R&D Demonstration	0.60	0.60	0.50	0.40
PIE Preparation and Performance	1.20	1.30	1.30	0.80
Fuel/Reactor Vendor Subcontracts	0.10	0.10	0.10	0.10
LANL				
Direct LWR Demonstration	—	—	—	—
INEEL	0.75	0.75	0.60	0.10
Total	2.65	2.75	2.50	1.40

The estimated costs for FYs 2000–2001 include allowances for the planning of additional irradiation of existing fuel capsules in the ATR. (There is no allowance, however, for

costs for design, fabrication, and irradiation of additional test fuel.) The bases for the INEEL cost estimates are explained in the INEEL Project Management Plan (PMP), which is a Level-2 document. Allowances for contingencies are not included in Table 2. Details of any future test irradiations not covered by this Test Project Plan will be described by a dedicated Plan to be prepared for that purpose.

As with the cost estimates, schedule estimates are taken from the AOP. Each year's AOP will contain an updated set of milestones and due dates that will serve as the official means of tracking progress. The milestones are summarized in Table 3.

Table 3. LWR Demonstration Project Milestones

Activity	Start Date	Completion Date
Hold organization meeting	Oct. 8, 1996	Oct. 9, 1996
Prepare requirements document	Oct. 8, 1996	Jun. 6, 1997
Issue approved test plan	Mar. 1, 1997	Jul. 23, 1997
Initial fuel fabrication	May 1, 1997	Nov. 13, 1997
Design review meeting	May 28, 1997	May 28, 1997
Fresh fuel shipping plan (ORNL)	Jun. 1, 1997	Sep. 17, 1997
Prepare PIE plan	Jun. 15, 1997	Sep. 26, 1997
Basket and capsule fabrication	Jun. 20, 1997	Dec. 15, 1997
Capsule loading and seal welding	Jan.5, 1998	Jan. 21, 1998
Irradiation	Feb. 5, 1998	Jan. 2004
Irradiated fuel transportation plan (ORNL)	Feb. 16, 1998	Oct. 16, 1998
Initial fuel removal	Sep. 1998	Sep. 15, 1998
PIE activities	Nov. 1998	Mar. 2005
Quick Look report (8 GWd/MT)	Dec. 1998	Jan. 22, 1999
PIE report on early withdrawals	Feb. 1999	Nov. 18, 1999
Intermediate fuel removal	Sep. 1999	Sep. 27, 1999
Quick Look report (21 GWd/MT)	Jan. 2000	Mar. 2000
PIE report on intermediate withdrawals	Mar. 2000	Dec. 2000
Design review for burnup extension	Jun. 2000	Jun. 2000
Fuel removal at 30 GWd/MT	Jul. 2000	Jul. 2000
Quick Look report (30 GWd/MT)	Nov. 2000	Feb. 2001
PIE report on 30 GWd/MT withdrawals	Feb. 2001	Sep. 2001
Fuel removal at 40 GWd/MT	Apr. 2002	Apr. 2002
Quick Look report (40 GWd/MT)	Aug. 2002	Nov. 2002
PIE report on 40 GWd/MT withdrawals	Nov. 2002	Jun. 2003
Achieve burnup of 45 GWd/MT	Mar. 2003	Mar. 2003

Table 3. (continued)

Final fuel removal at 50 GWd/MT	Nov. 2003	Nov. 2003
Quick Look report (50 GWd/MT)	Mar. 2004	Jun. 2004
PIE report on 50 GWd/MT withdrawals	Jun. 2004	Jan. 2005
Prepare final test summary report	Jun. 2004	Apr. 2005

The planned date (November 2003) for final fuel removal is subject to revision depending on the date for the next ATR Core Internal Changeout, which will require a reactor shutdown of about four months. Any significant change to the current schedule will be promulgated by means of a future revision to this document.

9.0 Summary

The plan described in this paper defines the FMDP Light Water Reactor MOX Fuel Irradiation Demonstration tests conducted in the Advanced Test Reactor (ATR). This activity demonstrates the resolve of the DOE to move forward with WG plutonium disposition. Furthermore, these tests, in conjunction with the ongoing gallium evolution and corrosion studies, investigate several of the outstanding technical issues facing the reactor disposition option including the effects of residual gallium impurities, and the performance under irradiation of weapons-derived plutonium.

ORNL provides overall program management for DOE-MD, coordinates the various aspects of the project, and provides comprehensive reporting. ORNL designed the test vehicle to be utilized in the ATR. LANL fabricated all of the fuel. INEEL supplies irradiation services, provides capsule-design assistance as requested, and performs independent safety reviews.

The test fuels reaching 50 GWd/MT will be irradiated for approximately four effective full-power years in the ATR. All of the irradiated fuel will be shipped to ORNL for PIE. Examination will focus on the behavior of gallium and its interaction with the cladding. Measurements will also be taken to verify the predicted performance of the WG MOX fuel. All waste streams will be incorporated into the existing waste collection/disposal systems at ORNL.

ornl

ORNL/TM-13428

OAK RIDGE
NATIONAL
LABORATORY

LOCKHEED MARTIN

SURVEY OF WORLDWIDE LIGHT WATER REACTOR EXPERIENCE WITH MIXED URANIUM-PLUTONIUM OXIDE FUEL

B. S. Cowell
S. E. Fisher

MANAGED AND OPERATED BY
LOCKHEED MARTIN ENERGY RESEARCH CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL-27 (3-96)

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

ORNL/TM-13428

SURVEY OF WORLDWIDE LIGHT WATER REACTOR EXPERIENCE WITH MIXED URANIUM-PLUTONIUM OXIDE FUEL

B. S. Cowell
S. E. Fisher

FEBRUARY 1999

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831–6285
managed by
LOCKHEED MARTIN ENERGY RESEARCH CORP.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05–96OR22464

ABSTRACT

The United States (U.S.) and the Former Soviet Union (FSU) have recently declared quantities of weapons materials, including weapons-grade (WG) plutonium, excess to strategic requirements. One of the leading candidates for the disposition of excess WG plutonium is irradiation in light water reactors (LWRs) as mixed uranium-plutonium oxide (MOX) fuel. A description of the MOX fuel fabrication techniques in worldwide use is presented. A comprehensive examination of the domestic MOX experience in U.S. reactors obtained during the 1960s, 1970s, and early 1980s is also presented. This experience is described by manufacturer and is also categorized by the reactor facility that irradiated the MOX fuel. A limited summary of the international experience with MOX fuels is also presented.

A review of MOX fuel and its performance is conducted in view of the special considerations associated with the disposition of WG plutonium. Based on the available information, it appears that adoption of foreign commercial MOX technology from one of the successful MOX fuel vendors will minimize the technical risks to the overall mission. The conclusion is made that the existing MOX fuel experience base suggests that disposition of excess weapons plutonium through irradiation in LWRs is a technically attractive option.

TABLE OF CONTENTS

ABSTRACT i

TABLE OF CONTENTS iii

LIST OF ACRONYMS vii

1. INTRODUCTION 1

2. MOX FUEL MANUFACTURING PROCESSES 5

2.1 BASIC MANUFACTURING STEPS 5
2.1.1 Oxide Production 5
2.1.1.1 Direct Calcination 6
2.1.1.2 Oxalate Process 6
2.1.1.3 Peroxide Process 7
2.1.1.4 Microwave Denitration 7
2.1.1.5 Coprecipitation 7
2.1.1.6 AUPuC Process 8
2.1.1.7 Hydride Oxidation 8
2.1.1.8 Direct Metal Oxidation 9
2.1.2 Oxide Blending and Milling 9
2.1.2.1 Comilling 10
2.1.2.2 Preparation of a Master Mix or Blend 10
2.1.3 Fuel Constitution 11

2.2 INTEGRATED COMMERCIAL PROCESSES 12
2.2.1 Reference Comilling 12
2.2.2 Micronized Master Blend (MIMAS) and Advanced MIMAS 13
2.2.3 Optimized Comilling (OCOM) 17
2.2.4 AUPuC-Integrated Process 17
2.2.5 Short Binderless Route (SBR) 19
2.2.6 Particle Fuel Technology 21

3. DOMESTIC MOX IRRADIATION EXPERIENCE 23

3.1 U.S. GOVERNMENT DEVELOPMENT 23
3.1.1 Plutonium Utilization Program (PUP)—Early Test Reactor Irradiations 24
3.1.2 Saxton Program 26

3.2 WESTINGHOUSE MOX DEVELOPMENT 27
3.2.1 EEI/Westinghouse Plutonium Recycle Demonstration Program 27
3.2.2 Overseas Westinghouse LTA Programs 30
3.2.3 Ginna LTAs 30

3.3 GENERAL ELECTRIC (GE) MOX EXPERIENCE 32
3.3.1 GE's Early MOX Testing 32
3.3.2 EEI/GE Plutonium Utilization in BWRs Program 32
3.3.3 Other GE MOX Testing 35

3.4 ABB COMBUSTION ENGINEERING (CE) MOX EXPERIENCE 36

3.5 EXXON NUCLEAR MOX EXPERIENCE 36

3.6 GULF UNITED NUCLEAR CORPORATION (GUNC) MOX EXPERIENCE 37

3.7 BABCOCK AND WILCOX (B&W) MOX EXPERIENCE 37

3.8 OTHER DOMESTIC MOX EXPERIENCE 38

4. FOREIGN MOX IRRADIATION EXPERIENCE 39

4.1 BELGIAN MOX EXPERIENCE 40
4.1.1 Early Experience in BR3 Reactor 40
4.1.2 Other Fuel Development Irradiations 41
4.1.2.1 Garigliano Irradiations 43
4.1.2.2 Dodewaard Irradiations 43
4.1.2.3 Chooz A Irradiations 43
4.1.2.4 Oskarshamn LTAs 44
4.1.2.5 Test Reactor and Non-LWR Irradiations 44
4.1.3 Commercial BN MOX Experience 44

4.2 ITALIAN MOX EXPERIENCE 46
4.2.1 Garigliano MOX Irradiations 47
4.2.2 Trino MOX Irradiations 48

4.3 GERMAN (FORMER FEDERAL REPUBLIC OF GERMANY) MOX EXPERIENCE 48

4.4 FRENCH MOX FUEL EXPERIENCE 50

4.5 SWISS MOX EXPERIENCE 52

4.6 UNITED KINGDOM MOX EXPERIENCE 52

4.7 JAPANESE MOX EXPERIENCE 54

4.8 MOX EXPERIENCE IN THE RUSSIAN FEDERATION AND FORMER SOVIET UNION 55

5. SUMMARY AND CONCLUSIONS 57

5.1 SUMMARY OF FINDINGS 57

5.2 CONCLUSIONS 58

REFERENCES 61

APPENDIX A. U.S. MOX EXPERIENCE TABLE A-1

APPENDIX B. MOX EXPERIENCE EXCERPT FROM GESMO B-1

APPENDIX C. BIBLIOGRAPHY C-1

LIST OF ACRONYMS

Acronym	Definition
ABWR	Advanced Boiling Water Reactor
ADU	Ammonium Diuranate [$(NH_4)_2U_2O_7$]
AEC	U.S. Atomic Energy Commission
AECL	Atomic Energy of Canada, Limited
AGR	Advanced Gas Reactor
ALKEM	Alpha Kemistry and Metallurgy GmbH, now part of Siemens AG
ANRCP	Amarillo National Resource Center for Plutonium
ARHCO	Atlantic Richfield Hanford Company
ARIANE	Actinide Research in A Nuclear Element
ARIES	Advanced Recovery and Integrated Extraction System
ATR	1) Advanced Test Reactor (Idaho National Engineering and Environmental Laboratory) 2) Advanced Thermal Reactor (Japan)
AUC	Ammonium Uranyl Carbonate [$(NH_4)_4UO_2(CO_3)_3$]
AUPuC	Ammonium Uranyl-Plutonyl Carbonate [$(NH_4)_4(U,Pu)O_2(CO_3)_3$]
B&W	Babcock and Wilcox
BN	Belgonucléaire
BNFL	British Nuclear Fuels Limited, plc
BNWL	Battelle Pacific NorthWest Laboratories
BR2	Belgian Reactor 2
BR3	Belgium's first PWR, designed by Westinghouse
BRP	Big Rock Point BWR
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium Reactor
CE	ABB Combustion Engineering
CEA	Commissariat à l'Énergie Atomique
CEC	Commission of European Communities
CFCa	le Complexe de Fabrication des combustibles au plutonium de Cadarache, a pilot MOX fuel facility at Cadarache, France

Acronym	Definition
CNA	Centrale Nucléaire des Ardennes (sometimes used to refer to Chooz A)
CNEN	Comitato Nazionale per l'Energia Nucleare
COCA	Cobroyage-Cadarache, a comilling fabrication procedure used in CFCa facility until recently
COGEMA	Compagnie Générale des Matières Nucléaires
COGEPEL	Coprecipitation process
COMMOX	Joint Belgonucléaire-COGEMA MOX marketing firm
COPRECAL	Coprecipitation and calcination process
CPS	Cold-pressed and sintered
DCA	Deuterium Critical Assembly
DEMOX	Dessel MOX plant at Mol, Belgium, more commonly known as the P0 plant.
DOE	Department of Energy
EBWR	Experimental Boiling Water Reactor
ECN	Energieonderzoek Centrum Nederland
EDF	Électricité de France
EEI	Edison Electric Institute
ENC	Exxon Nuclear Company, Inc.
ENEA	Ente per le Nuove tecnologie l'Energia e l'Ambiente
ENEL	Ente Nazionale per l'Energia Elettrica
EPRI	Electric Power Research Institute
ETR	Engineering Test Reactor
FBFC	Société Franco-Belge de fabrication de combustibles
FERTF	Fuel Element Rupture Test Facility
FFTF	Fast Flux Test Facility
FMDP	Fissile Materials Disposition Program
FRAGEMA	FRAMATOME Nuclear Fuel Division, a joint venture of FRAMATOME (50%) and COGEMA (50%)
FRAMATOME	French NSSS and fuel supplier
FSU	Former Soviet Union
GE	General Electric Company
GESMO	Generic Environmental Statement on the use of Recycle Plutonium in

Acronym	Definition
	Mixed Oxide in light water cooled reactors
GETR	General Electric Test Reactor
GKN	Gemeinschaftskernkraftwerk Nekar, also refers to Neckarwestheim-1
GUNC	Gulf United Nuclear Corporation
HFR	High Flux Reactor, Petten
HYDOX	Hydride-oxidation metal-to-oxide conversion
IFA	Italian fuel assembly
IFBA	Integral Fuel Burnable Absorber, a ZrB_2 coating developed by Westinghouse
INEEL	Idaho National Engineering and Environmental Laboratory
IS	Item Summary
JAERI	Japan Atomic Energy Research Institute
JAPCO	Japan Atomic Power Company
JNC	Japan Nuclear Cycle Development Institute
KBR	Kernkraftwerk Brokdorf GmbH
KKG	Kernkraftwerke Göesgen
KKP	Kernkraftwerk Philippsburg
KKU	Kernkraftwerk Unterweser
KRB-A	Kernkraftwerke Gundremmingen Betriebsgesellschaft mbH
KWL	Kernkraftwerk Lingen GmbH
KWO	Kernkraftwerk Obrigheim PWR
KWU	Kraftwerk Union AG
LANL	Los Alamos National Laboratory
LEU	Low enriched uranium
LMFBR	Liquid Metal Fast Breeder Reactor
LMR	Liquid metal reactor
LMR	Liquid Metal Reactor
LTA	Lead Test Assembly
LWR	Light Water Reactor
MAGNOX	Magnesium-alloy-clad, oxide-fueled, CO_2 cooled early UK reactor design
MDF	MOX Demonstration Facility
MELOX	COGEMA MOX plant at Marcoule

Acronym	Definition
MIMAS	Micronized Master blend
MIR	Materials Irradiation Reactor
MOX	Mixed uranium-plutonium oxide
MTR	Materials Test Reactor
MZFR	Mehrzweckforschungsreaktor
NMD	Nuclear Materials Division of The Babcock and Wilcox Company
NOK	Nordostschweizerische Kraftwerke
NPD	Nuclear Power Demonstration (reactor)
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OCOM	Optimized Comilling (sometimes written as OKOM)
OD	Outside diameter
OKG	Oskarshamnsverkets Kraftgrupp Aktielbolag
OKG1	Refers to the first reactor built by Swiss Oskarshamnsverkets Kraftgrupp AB—Oskarshamn 1
ORNL	Oak Ridge National Laboratory
PDS	Plutonium Disposition Study
PFDF	Plutonium Fuel Development Facility
PFFF	Plutonium Fuel Fabrication Facility
PFPF	Plutonium Fuel Production Facility
PHWR	Pressurized heavy water reactor
PIE	Postirradiation Examination
PRDP	Plutonium Recycle Demonstration Program
PRTR	Plutonium Recycle Test Reactor
PUP	Plutonium Utilization Program
PUREX	Plutonium Uranium Recovery by Extraction
PVA	polyvinyl alcohol
PWR	Pressurized Water Reactor
RBG	Reaktor Beteilingungsgesellschaft, now RBU
RBU	Reaktor-Brennelement Union GmbH
RG	Reactor Grade

Acronym	Definition
RG&E	Rochester Gas and Electric
SBR	Short Binderless Route
SCK-CEN	Studiecentrum voor Kernenergie–Centre d'Étude de l'Énergie Nucléaire
SENA	Société d'Énergie Nucléaire Franco-Belge des Ardennes (utility operator for Chooz A, sometimes used as the reactor pseudonym)
SMP	Sellafield MOX Plant
SPERT	Transient Irradiation Test at Idaho National Engineering and Environmental Laboratory
SPHEREPAC	Sphere compaction
SS	stainless steel
T.D.	Theoretical Density
U.S.	United States
UK	United Kingdom
UKAEA	United Kingdom Atomic Energy Authority
UNH	Uranyl nitrate hexahydrate, $UO_2(NO_3)_2 \cdot 6H_2O$
VAK	Versuchsatom Kraftwerk
VENUS	SCK-CEN zero-power critical facility
VIPAC	Vibrationally compacted particle fuel
VVER	Soviet pressurized water reactor
W	Westinghouse
WG	Weapons-grade

Survey of Worldwide Light-Water Reactor Experience With Mixed Uranium-Plutonium Oxide Fuel

Authors: B. S. Cowell
S. E. Fisher

1. INTRODUCTION

The United States (U.S.) and the Former Soviet Union (FSU) have both declared significant quantities of weapons materials (highly enriched uranium and plutonium) surplus to strategic requirements. The U.S. Department of Energy (DOE) has completed the Plutonium Disposition Study (PDS) and initiated the follow-on Fissile Materials Disposition Program (FMDP) to address this material. While the excess uranium can be easily denatured (made unfit for weapons use) through blending with either depleted or natural uranium, the excess plutonium is not so easily dispositioned. One of the leading candidates for disposition of excess weapons-grade (WG) plutonium is irradiation in light water reactors (LWRs) as uranium-plutonium oxide (MOX) fuel. The nuclear fission process, often referred to as "burning" even though it is nuclear rather than chemical, converts the fissionable plutonium into nonfissile fission products. Plutonium has previously been used to fuel domestic reactors and currently fuels a number of foreign reactors.

Research, testing, and commercial utilization of plutonium have resulted in the existence of a vast experience base for MOX in general. This irradiation experience provides the foundation on which to build a disposition program. While the plutonium irradiation data base is extensive, the portion that is directly applicable to the FMDP is limited. Most of the plutonium that has been used as fuel is reactor grade (RG), which has a lower fissile isotope concentration than WG material. Furthermore, much of the irradiation experience is treated as proprietary information by the fuel vendors, reactor vendors, and/or utilities that sponsored the irradiations. Under these conditions, accurate projections of the additional irradiation experience that may be required to support the U.S. Nuclear Regulatory Commission (NRC) licensing of MOX fuel use in commercial reactors become more difficult.

Following standard nomenclature, MOX fuel refers only to LWR plutonium-uranium oxide fuel. Other plutonium-uranium oxide fuels are referenced in this report using alternative nomenclature such as plutonium fuels or fast reactor fuels.

From the early days of the nuclear era, plutonium has been recognized by many as a valuable reactor fuel. Although present in some uranium deposits in minute quantities, plutonium is generally viewed as a man-made element, created through neutron capture in uranium. Because LWRs use low-enriched fuel that contains a large fraction of ^{238}U, they breed a significant amount of plutonium during operation, part of which is fissioned in-situ. Early in the nuclear era, it was assumed that the plutonium bred in the LWRs would be used to fuel liquid metal fast breeder reactors (LMFBRs), which can breed additional plutonium more efficiently. Light water burner and converter reactors were viewed as interim options that would eventually be

supplanted by LMFBRs. Nevertheless, plutonium recycle into LWRs was also thought to be practical.

Several government- and industry-sponsored programs in the United States during the 1960s and 1970s demonstrated the feasibility of MOX fuels. Test irradiations of fuel specimens in research and test reactors led eventually to lead test assembly (LTA) irradiations in commercial reactors. However, President Carter's 1977 nuclear nonproliferation policy announcement, which called for indefinite deferral of domestic commercial reprocessing and recycling of plutonium, effectively ended all domestic recycling efforts (Ref. 1).

During the same period, many foreign countries began investigations into the use of MOX fuel. The Belgians irradiated their first MOX test assembly in 1963. Other European countries, including Germany, France, the United Kingdom (UK), Switzerland, Italy, Sweden, and the Netherlands eventually followed suit. A wealth of information about these programs exists in the open literature, and much more is known to exist in proprietary files. Other countries are known to have MOX irradiation experience (Japan, China, India, the FSU), but only limited open literature documentation on their programs has been located. While the United States' unilateral ban on reprocessing ended domestic MOX research in the late 1970s, several of these foreign countries continued to pursue plutonium recycle in LMFBRs and/or LWRs. MOX fuel is now widely used in several countries, and its use is expected to expand rapidly in both France and Japan in the near future.

This document summarizes the MOX fuel irradiation experience and, by necessity, is primarily limited to a review of information that is documented in the open literature. In this report, irradiation experience refers to the irradiation itself and any examinations performed during and after the irradiation. A comprehensive summary of MOX experience cannot be assembled because of the proprietary nature of much of the information. Most of the open literature information consists of descriptions of the material that was irradiated and the irradiation conditions. Results from the postirradiation examination (PIE) programs are in general unavailable.

Chapter 2 of this report summarizes MOX fuel manufacturing technology. Because the irradiation performance depends heavily on the particular manufacturing techniques employed, a basic understanding of the processes is needed to understand the subsequent sections of this report. Chapter 3 contains a description of the domestic irradiation experience, as reported in the open literature. Overseas MOX irradiation experience is then described, country by country, in Chapter 4. The irradiation experience in most cases consists of a description of the fuel that was tested and the test conditions without elaboration on the purpose of the test or the PIE results due to data unavailability. The lessons learned from all the irradiation experience are summarized in Chapter 5. Appendix A consists of a tabular summary of the domestic MOX experience. Appendix B contains a reprinted summary description of the U.S. experience from the Generic Environmental Statement on MOX (GESMO) report. In addition to the list of cited references, an extensive Bibliography is included in Appendix C.

CANDU (Canadian Deuterium Uranium Reactor) reactor experience with MOX fuel is not described in the body of this report. Differences between CANDU fuel and typical LWR fuel render the LWR MOX data base somewhat inapplicable to the CANDU MOX experience.

Atomic Energy of Canada, Limited (AECL), the CANDU vendor, has investigated a number of advanced fuel cycles, including MOX fuel use, for CANDU reactors. As is the case with the other vendors, most of the data resulting from the tests and examinations for CANDU reactors are proprietary.

2. MOX FUEL MANUFACTURING PROCESSES

As with standard urania (UO_2) fuels, the physical characteristics and irradiation performance of MOX fuels depend on the particular manufacturing process used to produce them. A number of processes have been developed for MOX fuels, and they are discussed in this chapter. Although some of the newer processes may be considered "advanced," no firm evidence was located in the open literature to prove that the irradiation performance of these advanced fuels is greatly superior to that of the fuels produced domestically in the 1970s. The advances have in fact focused more on simplification of the manufacturing process and suitability of the product for subsequent recycling than on in-reactor performance improvements. The suitability of the fuel depends on the program requirements. However, the "advanced" fuels in general seem to have lower fission gas release, better dissolution characteristics (important for reprocessing considerations), and more robust ability to withstand power changes.

This MOX fuel manufacturing discussion is not meant as an exhaustive review. It is included as an introduction to provide sufficient information to understand the issues raised in later sections of this report.

2.1 BASIC MANUFACTURING STEPS

The MOX fuel manufacturing process is similar to the UO_2 fuel manufacturing process, with a few additions. In fact, all of the specific MOX processes are adaptations of either standard UO_2 fuel processes, fast-reactor-fuel processes, or a combination of the two. According to ALKEM (Alpha Kemistry and Metallurgy GmbH), the German plutonium fuel manufacturer, the classic UO_2 production process was modified "only where dictated by the peculiar properties of plutonium. Major deviations occur in powder blending, sintering in diluted hydrogen to prevent explosions, outside contamination control of the fuel rods, and autogammagraphy of finished fuel rods" (Ref. 2).

The generic fabrication procedure has been broken down, for the purposes of this discussion, into three stages: oxide production, blending, and fuel consolidation. The first stage often takes place in a separate facility, but it directly affects the powder preparation required in the second stage. Not all of the integrated MOX manufacturing processes employ all three stages. Some, like the coprecipitation processes, combine oxide production and blending into a single step.

2.1.1 Oxide Production

In the existing commercial MOX fabrication plants, the plutonium that is utilized (and in some cases the uranium also) results from Plutonium Uranium Recovery by Extraction (PUREX) reprocessing operations and is in the form of an aqueous nitrate solution. The uranium streams contain uranyl nitrate hexahydrate, $UO_2(NO_3)_2 \cdot 6H_2O$, (UNH). The corresponding chemistry of the plutonium stream is not as simple.

Multiple plutonium oxidation states are stable. The standard methods of conversion of UNH to oxide must, therefore, be modified for application to plutonium nitrate solutions. The commonly used methods of plutonium oxide production are discussed in this chapter. Four techniques (direct calcination, oxalate, peroxide, and microwave denitration) for converting plutonium nitrate to oxide and two techniques (coprecipitation and Ammonium Uranyl/Plutonyl Carbonate) for coconverting plutonium/uranium mixtures are described.

Most of the surplus plutonium is in the form of metallic weapons parts or pits. Two dry processes are under consideration for conversion of this material to PuO_2. The first is hydride oxidation (HYDOX) in which the metal is contacted with diluted hydrogen to form hydride that is subsequently nitrided and finally oxidized. The second is direct metal oxidation (DMO) in which metal is heated in an oxidizing atmosphere and allowed to burn. The key determinant in the choice of oxide production processes is the suitability of the final oxide powder to the fuel manufacturer. Subsequent powder conditioning may be used to obtain desired powder characteristics, but such conditioning requires additional time, expense, and facilities.

2.1.1.1 Direct Calcination

Calcination is a process in which a material, such as plutonium nitrate solution, is heated to a temperature below its melting point to effect a decomposition, such as oxidation. Direct calcination is the preferred method of conversion from UNH to UO_3 in the United States. It is also the simplest method of converting plutonium nitrate solution to oxide (Ref. 3, p. 443). This method is insensitive to the beginning oxidation state, which is important for plutonium nitrate solutions as several oxidation states, including Pu^{3+}, Pu^{4+}, PuO_2^{+}, PuO_2^{2+}, PuO_2^{3+}, can exist (Ref. 3, p. 436). Direct denitration of plutonium nitrate solution has been investigated at the lab scale using a continuous liquid phase screw-type calciner (Ref. 4). Other methods of direct calcination include flame calcination and fluidized-bed calcination.

One of the disadvantages of direct calcination is that it provides no appreciable decontamination of the plutonium. Any impurities present in the nitrate solution are carried through to the oxide, so clean feed is required. Also, the calcination temperature and time at temperature must be carefully controlled to obtain desirable oxide powder properties. A sufficiently low calcination temperature must be used to prevent sintering or fusing, which results in a product that is difficult to homogenize in the subsequent milling process.

2.1.1.2 Oxalate Process

The oxalate process is another process for conversion of plutonium nitrate to oxide. Depending on the oxidation state of the plutonium in the feed nitrate solution, one of two oxalate processes can be used (Ref. 3, p. 442–443). In both processes, oxalic acid ($H_2C_2O_4$) is added to the nitrate solution. Depending on the feed nitrate solution, either $Pu_2(C_2O_4)_3 \cdot 9H_2O$ (for Pu^{3+}) or $Pu(C_2O_4)_2 \cdot 6H_2O$ (for Pu^{4+}) is precipitated. The oxalate precipitate is filtered, dried, and calcined to form the oxide. This is considered by many to be the standard conversion process. It combines the advantages of high quality product and plutonium decontamination with formation of relatively safe intermediate compounds and solids. Unlike the direct calcination process, the oxalate process provides some separation between impurities in the nitrate solution and the final oxide product.

2.1.1.3 Peroxide Process

The peroxide process is a third process for converting plutonium nitrate to oxide. By adding peroxide to plutonium nitrate at low temperature, large crystals of plutonium peroxide can be formed. These crystals are subsequently filtered from the solution, and the filter cake is dried in air and calcined to the oxide. The peroxide process yields better decontamination than the oxalate process, but it does so at the expense of safety. Impurities present in the nitrate feed can catalyze explosive decomposition of peroxide compounds (Ref. 5, p. 557). In its favor, the peroxide process results in excellent decontamination of cationic impurities. Nevertheless, because of the safety issues associated with the impurity-driven peroxide decomposition, the peroxide process has not been used commercially but remains a useful laboratory conversion process.

2.1.1.4 Microwave Denitration

The Japanese Power Reactor and Nuclear Fuel Development Corporation (PNC) and its successor Japan Nuclear Cycle Development Institute (JNC) have developed the microwave denitration process for application to plutonium nitrate or mixed-nitrate feed streams (Ref. 6). Because it was developed primarily to enhance the diversion resistance of the PUREX process, it is most often discussed for mixed-nitrate feed. Microwave denitration is similar in many respects to direct denitration/calcination. The principal difference is the use of microwave heating.

2.1.1.5 Coprecipitation

Coprecipitation is a method for converting mixtures of plutonium nitrate and uranyl nitrate to MOX. Coprecipitation has been pursued for two reasons: the homogeneity of the resulting MOX powder and the potential for use in reprocessing plants in which no separated plutonium stream exists. The principal coprecipitation process is an adaptation of the ammonium diuranate (ADU) process that is commonly used for conversion of uranyl nitrate to oxide. (A similar aqueous process known by the same name is used to convert UF_6 to UO_2.)

In the ADU process, ammonia is added to uranyl nitrate, forming the ammonium diuranate precipitate, $(NH_4)_2U_2O_7$. The precipitate is then filtered, dried, and calcined. In the coprecipitation process, ammonia is added to a dilute solution of plutonium and uranyl nitrates. The ammonia precipitates out both heavy metals simultaneously—the plutonium as plutonium hydroxide and the uranium as ammonium diuranate. The precipitates are filtered from the solution together, and the resulting filter cake is dried and calcined. An alternative is concentration and direct calcination of the slurry in a fluidized-bed reactor (Ref. 7, p. 132).

The resulting powder is a uniform blend of the two oxides, although some powder milling may be required to deliver the required fuel fabrication feed specifications. With coprecipitation, agglomerates of plutonium hydroxide can form, which must be broken up during the milling stage to prevent their occurrence in the finished MOX product. Coprecipitated material may be used directly if the concentrations of plutonium and uranyl nitrate are controlled such that the final plutonium concentration in the oxide is correct. Alternatively, the coprecipitated material

may be blended with additional uranium oxide powder to obtain the desired plutonium concentration.

A coprecipitation process known as COGEPEL has been investigated by Belgonucléaire (BN), although the details of the process are not described in the available reference (Ref. 8). The General Electric Company (GE) also investigated the coprecipitation and calcination (COPRECAL) process (Ref. 9).

2.1.1.6 AUPuC Process

The Ammonium Uranyl-Plutonyl Carbonate (AUPuC) process is another method of coconverting mixed-plutonium nitrate, uranyl nitrate solutions to MOX. The AUPuC process, like coprecipitation, is an adaptation of a uranyl nitrate conversion process. In the ammonium uranyl carbonate (AUC) process, ammonia and carbon dioxide are added to uranyl nitrate to precipitate ammonium uranyl carbonate. (A similar aqueous conversion process known by the same name is used to convert UF_6 to UO_2.) Filtration, drying, and calcination of the AUC precipitate produce oxide powder.

Because of the good physical properties of UO_2 derived from the AUC process (sometimes referred to as ex-AUC material), a similar process that would coprecipitate both uranium and plutonium was sought. The difficulty with such a process is the different oxidation states of the two metals. In mixed plutonium/uranium nitrate solutions, the metals usually exist in the Pu^{4+} and U^{6+} oxidation states. Researchers at ALKEM developed a successful process by oxidizing the Pu^{4+} to Pu^{6+}, then adding ammonia and CO_2 (Ref. 10). Ammonium uranyl/plutonyl carbonate is formed. The crystalline precipitate, $(NH_4)_4(U,Pu)O_2(CO_3)_3$, is filtered and calcined to form the oxide.

The resulting oxide powder has an average particle size of 40 μm. The Germans have used this process to a limited extent to produce feed for their MOX fabrication. However, use of the AUPuC process is dependent on collocated reprocessing and oxide conversion facilities because of restrictions on shipment of plutonium solutions.

2.1.1.7 Hydride Oxidation

Hydride Oxidation (HYDOX) is a dry process under development by Los Alamos National Laboratory (LANL) and Lawrence Livermore National Laboratory as part of the Advanced Recovery and Integrated Extraction System (ARIES). The ARIES system integrates a number of processes (cutting the pits apart, separating the plutonium from other components with gas, converting the plutonium to an oxide form, packaging it in sealed containers, decontaminating and determining the characteristics of the resulting product) necessary to remove plutonium from the cores of surplus nuclear weapons and converts the plutonium into an unclassified form that is suitable for international inspection, long-term storage, and disposition.

The reference HYDOX process is commonly referred to as the 3-step HYDOX process. Plutonium metal is subjected to hydrogen gas that is diluted in an inert carrier gas. Plutonium hydride spalls from the metal surface, exposing unreacted metal. The plutonium hydride is collected and subsequently exposed to nitrogen gas. The hydride is thus converted to plutonium

nitride. The plutonium nitride is then oxidized through exposure to oxygen gas diluted in an inert carrier gas. The three step HYDOX process avoids the simultaneous use of hydrogen and oxygen gases. PuO_2 produced through HYDOX retains a crystalline structure from the hydride that is slate-like in appearance.

2.1.1.8 Direct Metal Oxidation

Direct metal oxidation (DMO) is the second dry process under consideration for application to the surplus plutonium mission. DMO has been utilized previously for production of feed material for MOX fuel fabrication. It has also been utilized to convert pyrophoric plutonium metal into stable oxide for storage and/or dissolution for subsequent purification.

DMO may be performed in any oxidizing atmosphere. The complexity of a DMO system can vary from a heated reaction vessel to a sealed vessel with temperature and atmospheric control. Unalloyed plutonium metal will oxidize slowly even at room temperature, especially in the presence of moisture. More rapid oxidation occurs above about 300°C.

It has been demonstrated that moisture enhances the oxidation process, so the reactant gas in a DMO system is usually bubbled through water. The water is best described as a catalyst. It reacts with the surface plutonium metal to release hydrogen that is retained in the metal as plutonium hydride. As additional oxygen is made available, this hydrogen is released for reaction with additional metal. A hydride reaction front thus moves through the metal, followed by an oxide front. As a result, PuO_2 produced through DMO is similar in its crystalline makeup to PuO_2 produced through the HYDOX process.

DMO has also been tried in dry air. However, ignition under dry conditions requires higher temperatures. Some experimenters also describe the reaction as more difficult to control under dry conditions.

Most of the experiments performed with DMO suggest that relatively high surface area is required in the feed to obtain reasonable reaction rates at the low temperatures (< 500°C) that are desirable. The chemical reactivity of the product is affected by the reaction temperature. At low reaction temperatures, residual metal, hydride, or hydroxide can be found in the product. At higher reaction temperatures, large sintered particles known as clinkers can be produced.

2.1.2 Oxide Blending and Milling

In most reprocessing plants, plutonium nitrate is converted to oxide powder. This PuO_2 must be subsequently blended with UO_2 for fabrication into MOX fuel. The blending process is very important because the presence of large particles or agglomerates of PuO_2 (especially those $\geq 100\ \mu m$) has been found to be detrimental to both performance of the fuel under irradiation and subsequent solubility of the irradiated fuel in nitric acid. Even those oxide-production processes that combine the plutonium and uranium nitrate solutions (coprecipitation, AUPuC) often require some blending and/or milling to produce MOX that meets the particle size specifications. Several methods of blending/milling have been used for MOX production, and each is discussed in this chapter.

2.1.2.1 Comilling

Comilling is the "reference" process for blending PuO_2 and UO_2 powder and for adjusting the size of the particles and/or agglomerates in the MOX powder. The PuO_2 powder produced using one of the methods discussed in Section 2.1.1 is placed in a mill along with UO_2 powder. The UO_2 powder is typically produced from either the ADU, AUC, or direct dry-conversion process, depending on the fuel manufacturer. The two powders are milled together for up to 72 hours to ensure good homogeneity and to reduce the average particle size.

Although comilling was used until recently (with satisfactory results according to the manufacturers), fuel produced using this technique is often not as homogeneous as that produced by other methods. The milling time required to obtain a homogeneous product in a ball mill may be as long as tens of hours. This milling time and the associated pickup of metallic impurities from the milling media places a practical limitation on the achievable homogeneity. Based on the limited open literature information, the irradiation performance of this comilled fuel has not differed greatly from that produced from other methods.

One potential problem with comilled fuel is low solubility in nitric acid. Low solubility can cause large losses in subsequent reprocessing, although this is not detrimental for the WG plutonium disposition mission. Another difficulty with comilling is the large energy requirement that results from the need to mill all the MOX powder and from the inefficiency of the ball mill. Advances in comilling have focused on different types of mills that achieve better homogeneity in a shorter time. The hammer mill, jet mill, and high energy attritor mill have been investigated as potential improvements over the ball mill.

The ability of a mill to comminute MOX powder is limited by the buildup of electrostatic charges in the powder. These charges in effect create a practical limit of a few microns on the achievable particle size. Addition of a milling aid is necessary to comminute below this size. Zinc stearate [$Zn(CH_3\text{-}(CH_2)_{16}COO)_2$] is commonly utilized as a milling aid. A better milling aid is 1,3-propane diol [$HOCH_2\text{-}CH_2CH_2OH$], which is distributed better than zinc stearate because of its higher vapor pressure. Addition of milling aids allows comminution to proceed to submicron particle sizes.

2.1.2.2 Preparation of a Master Mix or Blend

As discussed above, one of the disadvantages of comilling is an intensive milling requirement. One method of oxide blending developed to help alleviate this concern is known as either master mix or optimized comilling. In this process, approximately 20% of the final MOX powder, including all the plutonium, is micronized in a mill. This creates a homogeneous master mix that is approximately 30% plutonium. The milling effort is thus reduced approximately by a factor of five. The homogeneous MOX powder is subsequently mixed with additional UO_2 powder to generate the final MOX blend. Large agglomerates of the master blend are not as detrimental to fuel performance as are large agglomerates (or particles) of PuO_2, because the master blend agglomerates are 70% UO_2.

An additional benefit of master-mix preparation is the ability to utilize the free-flowing properties of the diluent UO_2 to avoid granulation. UO_2 produced via the AUC process is free-

flowing. If the quantity of master mix is limited to 20% or so of the final blend, the free-flowing properties of the UO_2 dominate the behavior of the final blend such that it is also free-flowing. The optimized comilling (OCOM) and AUPuC processes developed by ALKEM (now Siemens) are based on this principle.

A similar process can be used with either coprecipitated or AUPuC-derived MOX powder. The coprecipitated or AUPuC material is homogeneous but often requires some milling to achieve the desired particle sizes. The high-plutonium-concentration powder can be milled and then later blended with free-flowing UO_2 powder to achieve the final desired plutonium assay.

2.1.3 Fuel Constitution

Fuel constitution comprises a series of steps in which MOX powder is conditioned, formed into cylindrical pellets, sintered, ground to size, clad, and compiled into completed fuel assemblies. Most of these MOX operations are based on the corresponding operations utilized for production of LEU (low enriched uranium) fuel.

Once a homogeneous mixture of MOX powder with the desired plutonium assay has been obtained through milling and blending, the MOX powder must be conditioned for pellet pressing. Press feed must be free-flowing to ensure complete, uniform, and repeatable filling of the die cavity. Most MOX feed powders must be conditioned because the milling step for homogenization renders them too fluffy to flow. The flowability is readily achieved through granulation of the fine powder into larger agglomerates. Three techniques for granulation have been utilized: wet binder addition, mechanical granulation, and self-agglomeration.

Wet binder addition, which is no longer commonly practiced with MOX, is described because it was utilized for fabrication of some of the early MOX fuels. In wet binder addition, an organic binder suspended in a solvent is mixed with the MOX powder forming a slurry. The slurry is then spray-dried or evaporated into a paste and finally into coarse granules.

In an effort to streamline the MOX fabrication process, alternatives to wet binder addition were developed. One of the alternatives, mechanical granulation, may be accomplished in one of several ways. In one granulation technique referred to as slugging, the powder blend is pressed at a low pressure (to 30%–50% theoretical density) into large compacts or slugs. These are then crushed and screened to form granules. The slugs can be produced in a high speed pellet press or even in a roll compactor (Ref. 11). In some processes the slugging is eliminated through use of forced sieving. Forced sieving uses a blade to force the powder through the screen, in effect pressing and sizing it simultaneously. An organic binder such as polyvinyl alcohol (PVA) is commonly utilized to increase the strength of the granules regardless of their method of fabrication.

A further refinement to the fabrication process is elimination of mechanical granulation in lieu of self-agglomeration. As described in Section 2.1.2.1, finely ground MOX powder will self-agglomerate due to the buildup of electrostatic charge in the powder. This is the root cause of the lower limit on achievable particle sizes in a mill without the addition of a milling aid. Self-agglomeration is enhanced through tumbling of the powder. Tumbling can be accomplished either in specialized equipment such as the British Nuclear Fuel Limited (BNFL) spheroidizer, or

through judicious design of the mill itself such that milling and agglomeration are possible in a single vessel. If a volatile milling aid is utilized, one can thus comminute the MOX powder below 1 μm for good homogenization, and then eliminate the milling aid through volatilization resulting from the powder heating caused by the input of milling energy.

Following production of a free-flowing MOX powder, die lubricant may be added to the press feed or added directly to the die. The powder is then pressed to > 50% theoretical density (TD) to form green pellets. If a binder has been added, the green pellets are subjected to a binder removal treatment as part of the sintering cycle. Sintering to temperatures as high as 1750°C increases the pellet density to between 93%–96% TD.

Incomplete die filling and wall friction result in nonuniformity in the density distribution in green pellets. These nonuniformities are removed during sintering, but necessarily result in geometric distortion of the cylindrical pellets. Sintered pellets take on the shape of an hourglass. This geometric distortion is removed through centerless grinding. Criticality concerns usually restrict the use of water as a grinding coolant/lubricant, so most MOX pellets are dry ground. Cleaned pellets are then formed into fuel column stacks and loaded into fuel cladding tubes. Fuel pin assembly is completed through attachment of the upper end cap, pressurization with helium, and seal welding.

2.2 Integrated Commercial Processes

The leading commercial MOX-fabrication processes are described in greater detail in this chapter. These processes are (1) Comilling, (2) Micronized Master Blend (MIMAS), (3) Optimized Comilling (OCOM), (4) AUPuC, (5) Short Binderless Route (SBR), and (6) particle compaction. While the basic flowsheets for the processes are available in open literature, details about individual process steps are proprietary and unavailable.

It is believed that MOX fuel manufactured by any of the leading pellet processes could be irradiated satisfactorily in U.S. LWRs, although the effort required to license the fuel would depend on the particular process and on the willingness of the fuel manufacturer to share (or sell) the corresponding irradiation data base for similar fuels.

2.2.1 Reference Comilling

This is the integrated MOX-production process that was used previously by the domestic fuel manufacturers. It was also used previously by several foreign fuel manufacturers, especially for LMFBR fuel. The term comilling derives from the method of oxide blending/milling, described in Section 2.1.2, that is utilized. PuO_2 and UO_2 powders are added to a mill in the correct concentrations to yield the desired fissile concentration. Either natural, depleted, or recycled UO_2 may be used. The powders are milled together for several hours—typically in a ball, jet, or hammer mill—to reduce the average particle size and to ensure uniformity in the MOX powder. It is important to reduce the maximum plutonium particle or agglomeration to less than 100 μm to prevent hot spots in the fuel. The blended powder is usually mixed with an organic binder. Slugging and granulation are then used to form the desired free-flowing MOX granules, suitable for introduction to the pellet press. The green pellets are sintered and centerless ground to the

final dimensions. All of the other pellet fabrication processes may be considered variations of comilling. One set of improvements is based on use of advanced powder mills. Hammer mills, vibromills, jet mills, and attritor mills have been used to reduce the required milling time.

BN produced several thousand fuel rods using variations of the comilling process. The earliest variation is a textbook example of comilling in which fine, comilled MOX powder is granulated into press feed. Subsequent attempts were made to simplify the fabrication process. To reduce the amount of handling of fine plutonium-bearing powders, PuO_2 was blended (not milled) into granulated UO_2. This simplified the fabrication process, but the microstructure produced was unsatisfactory. In the final evolution of comilling at BN known as the "Reference Process," PuO_2 was blended into free flowing UO_2 produced via the AUC process.

ALKEM also utilized variations of the comilling process during early development. Over 9000 MOX fuel rods were fabricated using their "Former Standard" process. ALKEM's experience was similar to BN's in that the fuel performance was adequate while the powder handling aspects of the fabrication and the solubility of the product were not. Two advanced processes, OCOM and AUPuC, were developed to overcome these difficulties.

The French used a variation of the Reference Comilling process in COGEMA's (Compagnie Générale des Matières Nucléaires) CFCa (le Complexe de Fabrication des combustibles au plutonium de Cadarache) MOX fabrication facility located at the Cadarache Nuclear Center until quite recently. One of the fabrication lines at the plant was converted from production of liquid metal reactor (LMR) fuel to MOX fuel in 1989. Due to its LMR heritage, the plant utilized a variation of the reference comilling process that is known as the COCA (Cobroyage-Cadarache) process. The flow sheet for this process is given in Figure 1 (Ref. 12, p. 71). During 1995, the CFCa facility converted to the MIMAS process described in the following section.

2.2.2 Micronized Master Blend (MIMAS) and Advanced MIMAS

The MIMAS process is the latest integrated MOX production process developed and used by BN. BN's MOX-manufacturing development is summarized in Table 1 (Ref. 13, p. 14). Early fuel was produced using one of several variations of the reference comilling process described in Section 2.2.1. As indicated in the table, BN used their "reference" process for MOX manufacture in their Dessel fabrication plant, P0, from 1973 through 1984. While the irradiation performance of this fuel was claimed to be satisfactory by BN, the fuel was unsatisfactory in that it lacked similarity to UO_2 fuel, and in that the plutonium it contained was not sufficiently soluble during reprocessing. Up to 5% of the plutonium remained undissolved, even in boiling nitric acid. It was recognized at the time that large plutonium particles might be responsible for this behavior, because pure PuO_2 is difficult to dissolve in nitric acid, while the solid solution of UO_2 and PuO_2 is not. The MIMAS process, developed in 1984, is an attempt to remedy difficulties with the "reference" fuel.

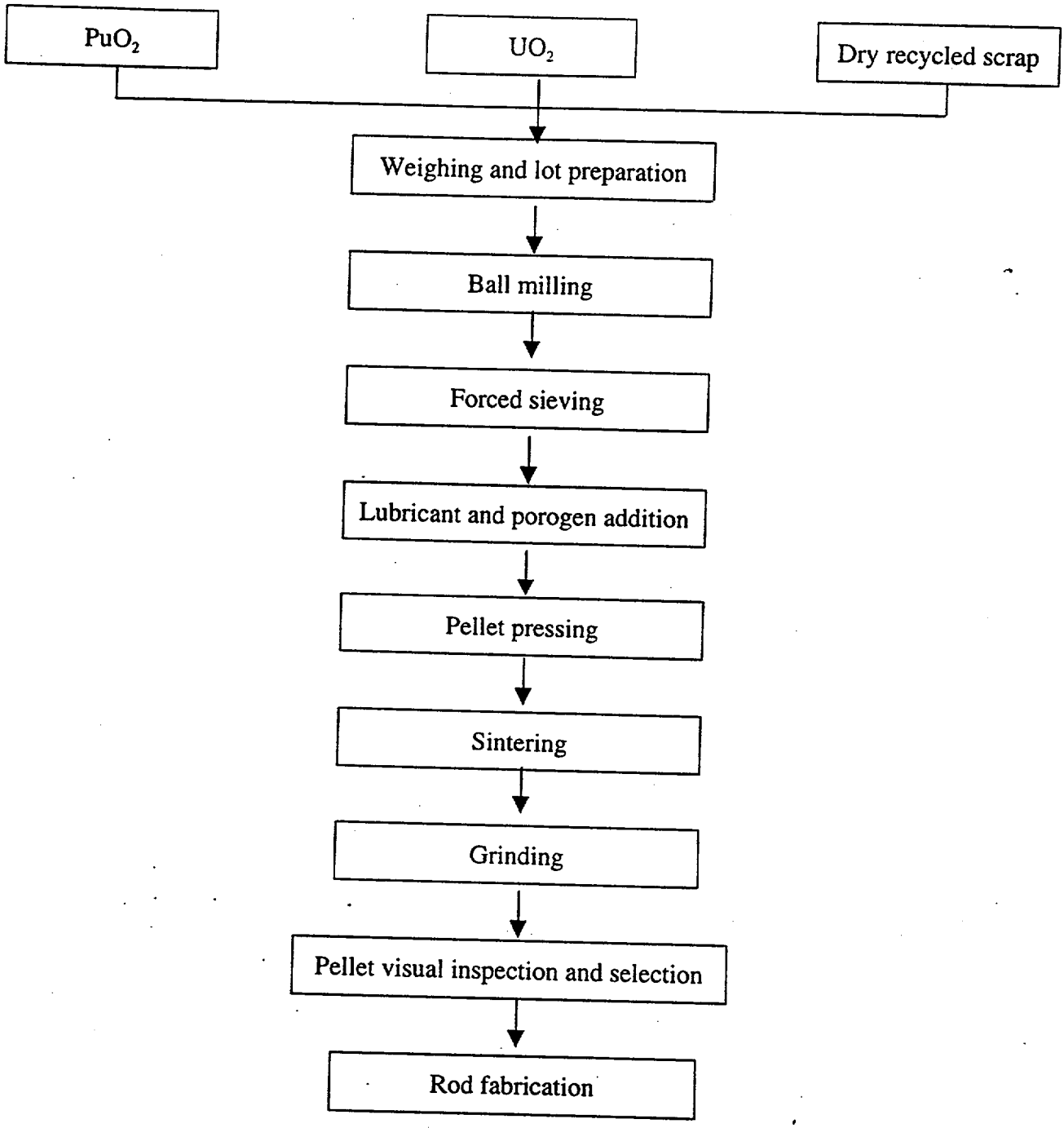


Figure 1. COCA MOX fabrication flow sheet.

Table 1. Evolution of the MOX fuel fabrication techniques of BN

Fuel Type	Period	Advantages	Disadvantages
Granulated (UO_2 + PuO_2) blend	1960–1962 + 1965–1969 (laboratory)	- Assumed best similarity to UO_2 fuel	- Contamination levels, personnel exposure, and waste resulting from complex handling of fine powder
PuO_2 blended into granulated UO_2	1967–1975 pilot facility	- Simplified handling of fine powder	- Significant departures from UO_2 fuel behavior - Unfavorable thermal conductivity - High fission gas release - Large Pu-rich agglomerates
"Reference," i.e., PuO_2 blended into free-flowing UO_2	1973–1984 (fabrication plant)	- Fuel microstructure governed by UO_2 matrix microstructure	- Occurrence of Pu-rich agglomerates - Too large proportion of the Pu in insoluble residues (reprocessing problems)
"MIMAS," i.e., mixing of free-flowing UO_2 and a micronized (UO_2 + PuO_2) primary blend	1983–present (fabrication plant)	- Same advantages as the "reference" MOX and applicability of its data base - Disappearance of the Pu-rich agglomerates issue due to dilution - Resolution of the reprocessing issue	

In the MIMAS process, a PuO_2 - UO_2 mixture of approximately 20%–30% plutonium is prepared. This mixture is milled or micronized, forming a fine uniform powder that is not free-flowing. This master mix is then diluted and mixed with free-flowing ex-AUC UO_2 to the final plutonium concentration. The overall flow sheet is shown in Figure 2 (Ref. 8, p. 66). By micronizing only the master mix, the milling energy requirements may be reduced by a factor of five or more, depending on the final plutonium concentration of the fuel. Also, the addition of a large fraction of free-flowing ex-AUC UO_2 results in a free-flowing MOX powder that is suitable for direct pellet pressing without binder addition or granulation. The resulting pellets consist of small plutonium-rich regions interspersed throughout a UO_2 matrix. This distribution more closely resembles that found in irradiated UO_2 fuel. Thus, the irradiation behavior is quite similar to that of UO_2 fuel. Furthermore, the dissolution characteristics of MIMAS fuel are

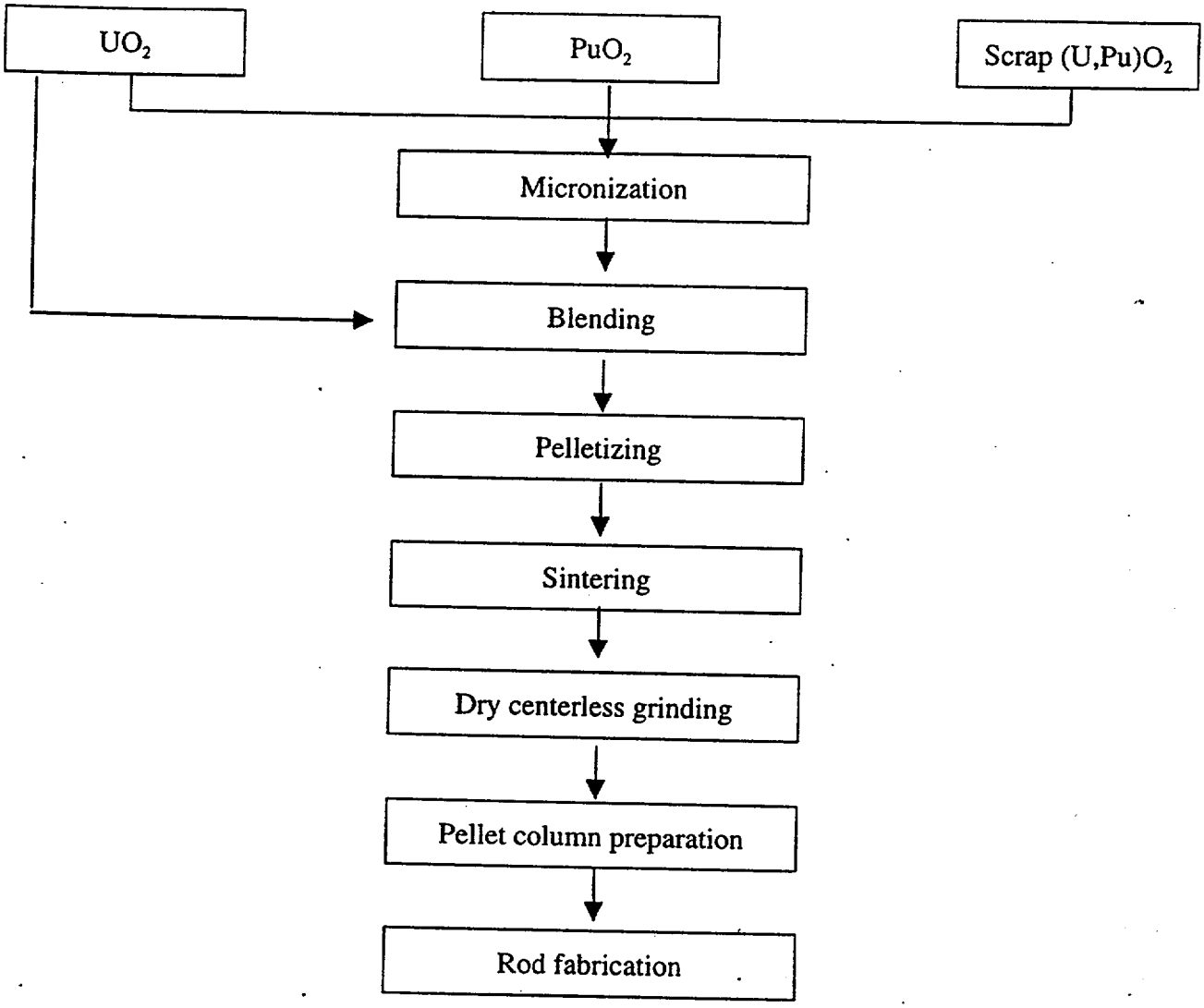


Figure 2. MIMAS flow sheet.

satisfactory to fuel reprocessors. While agglomerates of the master mix are encountered due to the self-agglomeration tendency of the micronized material, these agglomerates are more easily accommodated than pure plutonium agglomerates because of the diluting UO_2 in the master-mix particles.

COGEMA has adopted BN's MIMAS process for production of MOX fuel in France. The process as implemented by COGEMA is known as the advanced MIMAS process but is in most respects identical to BN's original. Advanced MIMAS has been implemented in the new MELOX plant and has been retrofitted into the CFCa plant at Cadarache. One important distinction is the use of UO_2 powder from the TU2 plant rather than the ex-AUC UO_2 utilized originally by BN (Ref. 14, p. 50). TU2 UO_2 is described by the developers as a modified ADU process that produces free-flowing product. Another difference is the use of forced screening of the MOX blend to produce press feed. In addition to allowing use of nonflowable UO_2, inclusion of forced screening allows inclusion of more primary blend in the final blend. Although forced

screening is not considered granulation by COGEMA, the net effect of forced screening is production of loosely held agglomerates. The agglomerates thus produced are harder than those produced by self-agglomeration, but softer than those produced through pressing and crushing.

2.2.3 Optimized Comilling (OCOM)

MOX research in Germany began in 1965. Early fabrication efforts utilized variations of Reference Comilling for pellet production. This "Former Standard" process produced satisfactory fuel that exhibited irradiation performance and reliability roughly equivalent to that of contemporaneous UO_2 fuel. However, as with BN's "Reference Process," fuel produced through the "Former Standard" process was found to be somewhat insoluble in nitric acid. The influence of reprocessors forced evolution of the MOX fabrication process.

The OCOM (or OKOM) process was developed by ALKEM as an improvement to the "Former Standard" process. It is similar in many respects to the MIMAS process. A primary blend is prepared by dry ball milling UO_2, PuO_2, and dry recycle MOX. The plutonium content of this primary blend is maintained at the upper end of the range in which uranium and plutonium form a solid solution ($\geq$ 30%). This primary blend is then diluted with an eight to ten fold larger quantity of free-flowing ex-AUC UO_2. The primary blend is tumbled prior to its dilution to produce free-flowing granulate via self-agglomeration. The agglomerates are allowed to grow to a size that roughly corresponds to that of the diluent ex-AUC UO_2 to enhance the dry blending. By restricting the primary blend to <15% of the secondary blend, the ceramic properties of the ex-AUC UO_2 are retained. This allows direct pressing, with direct die lubrication, without prior granulation. The OCOM flowsheet is shown in Figure 3 (Ref. 15, p. 178).

The key to OCOM is the reliance on the ceramic properties of ex-AUC UO_2. This same material has been utilized to produce many tons of LEU fuel. OCOM MOX fuel is similar in microstructure to this LEU, and has similar behavior under irradiation. The primary blend agglomerates form a solid solution during sintering so that the fuel mass itself has acceptable solubility in nitric acid. These agglomerates are also sufficiently diluted with UO_2 that larger particles of primary blend are acceptable than are acceptable for pure PuO_2 particles.

2.2.4 AUPuC-Integrated Process

The AUPuC process, named after the particular method of nitrate-to-oxide conversion utilized, helps to address reprocessor concerns of MOX solubility.

Because of the favorable experience with UO_2 derived from the AUC process, development of a similar process that would coprecipitate both uranium and plutonium was begun. One of the difficulties with such a coprecipitation is the different oxidation states of uranium and plutonium in nitrate solution. The uranium usually exists in the U^{+6} state, while plutonium is usually in the Pu^{+4} state. The approach was to adjust the oxidation state of either the uranium or plutonium prior to precipitation. Difficulties were encountered with reducing uranium from U^{+6} to U^{+4}, because U^{+4} in turn reduces Pu^{+4} to Pu^{+3}. Therefore, the plutonium is oxidized from Pu^{+4} to Pu^{+6}. This difficult valence adjustment must be complete to produce a filterable product. Residual Pu^{+4} causes precipitation of fines that cannot be filtered easily.

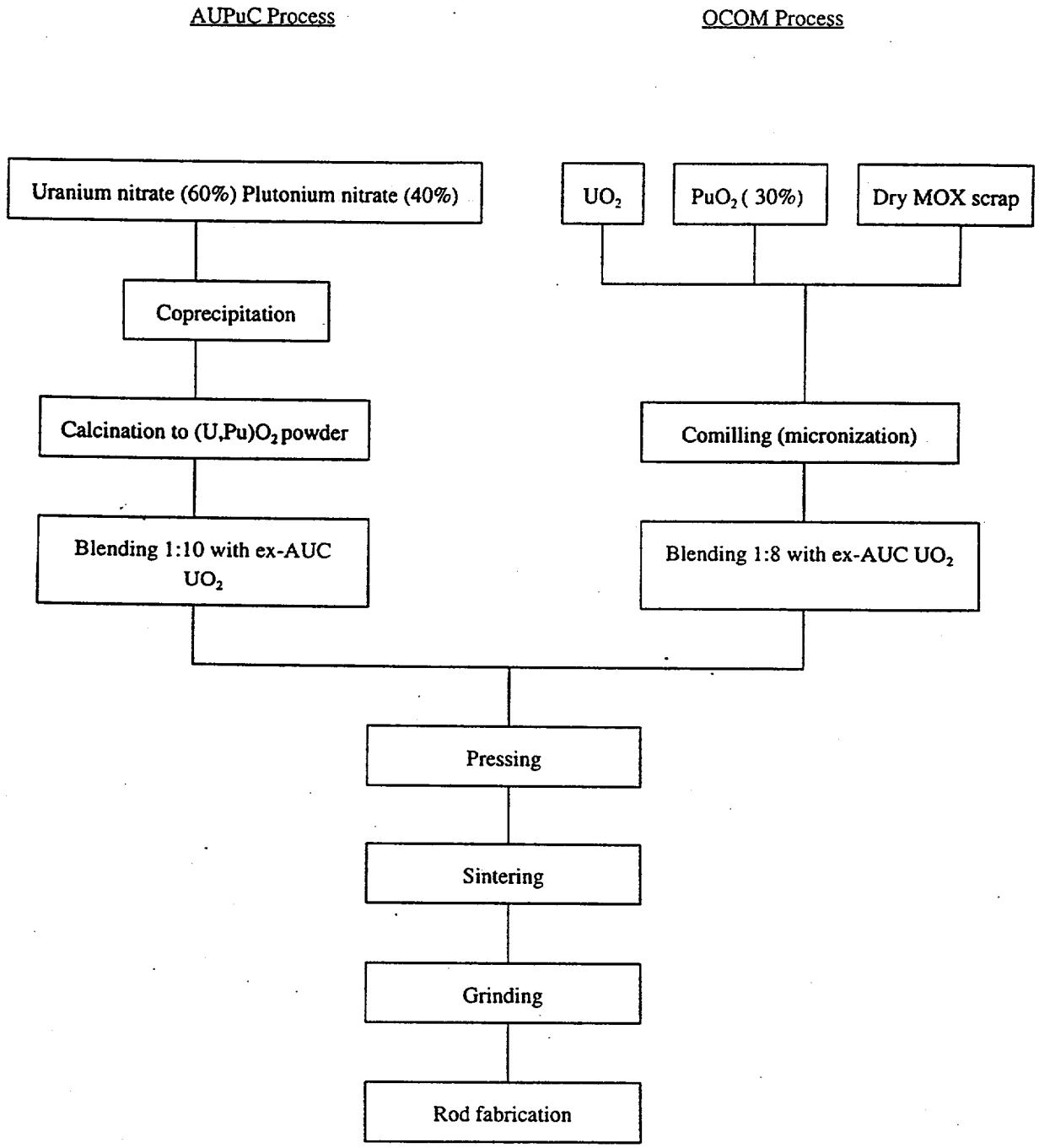


Figure 3. Flow sheets for AUPuC and OCOM MOX fabrication processes.

With both uranium and plutonium in the $(U,Pu)^{+6}$ state, ammonia and CO_2 are added to the solution. A mixed uranium and plutonium complex, ammonium uranyl/plutonyl carbonate or $(NH_4)_4(U,Pu)O_2(CO_3)_3$, is formed. This crystalline precipitate is filtered, dried, calcined, and reduced (Ref. 10). The resulting MOX powder is treated in much the same way as the master

mix in the OCOM flow sheet. Free-flowing ex-AUC UO_2 powder is added to the MOX powder to dilute it to the desired plutonium concentration. The resulting free-flowing powder is sent directly to the pellet press. The AUPuC-process flowsheet is shown in Fig. 3 (Ref. 15, p. 178).

Fuels were first produced using the AUPuC process in 1981. According to ALKEM, the irradiation performance of these fuels was satisfactory, and the solubility in nitric acid was much better than that of the "Former Standard" fuel. However, use of the AUPuC process requires the availability of plutonium nitrate solution, and the lack of collocated reprocessing and MOX plants has prevented widespread use of AUPuC.

2.2.5 Short Binderless Route (SBR)

The SBR is the integrated MOX-manufacturing process used by BNFL in their MOX Demonstration Facility (MDF), which has a nominal capacity of 8 MT/yr. This demonstration plant and the large-scale follow-on Sellafield MOX Plant (SMP), expected to come on-line in mid-1999 with a nominal capacity of 120 MT/yr, use a modified version of the comilling process described in Section 2.2.1 that is known as the SBR. The United Kingdom Atomic Energy Authority (UKAEA) and its successor, BNFL, have manufactured MOX since the early 1960s. However, a large portion of this MOX, especially that produced during the 1970s and 1980s, was fuel for the Dounreay fast breeder reactor. As the prospects for a fast breeder reactor economy dimmed in the mid- to late 1980s, interest in LWR MOX use expanded. BNFL and the UKAEA combined their MOX expertise and developed the SBR process as a collaborative effort.

Short in SBR refers to the time required for comilling. All the MOX powder, including all the UO_2 required to adjust the plutonium concentration to its final value, is milled together. However, instead of a conventional ball mill, a high energy attritor mill is used. The attritor mill is more efficient, allowing the entire powder charge to be milled to a suitably small size (maximum plutonium agglomerate size <100 μm with the average size less than 30 μm) while ensuring homogeneity of the MOX powder. Milling times may be reduced from 3–4 hours to approximately 30 minutes. Although details about the mill and spheroidizer are proprietary, they are shown diagrammatically in Figure 4 (Ref. 16, p. 175). The mill consists of a fixed outer casing, a rotating paddle assembly, and a ball charge.

The *binderless* term refers to the absence of organic binder addition and slugging/granulation. The milled powder coming from the attritor mill is not free-flowing and has a strong tendency towards self-agglomeration. It is, therefore, sent to a spheroidizer, the details of which are proprietary to BNFL. As suggested in Figure 4, the spheroidizer apparently tumbles the fine MOX powder to enhance self-agglomeration. Through control of milling aids and moisture content, similar processes have been used to produce soft agglomerates that are free-flowing and suitable for direct compaction.

The attritor mill and spheroidizer are vertically oriented. They both utilize fixed, static cases. This arrangement greatly reduces the risk of alpha contamination during transfer of the MOX powder from the mill through the powder conditioner to the pellet press.

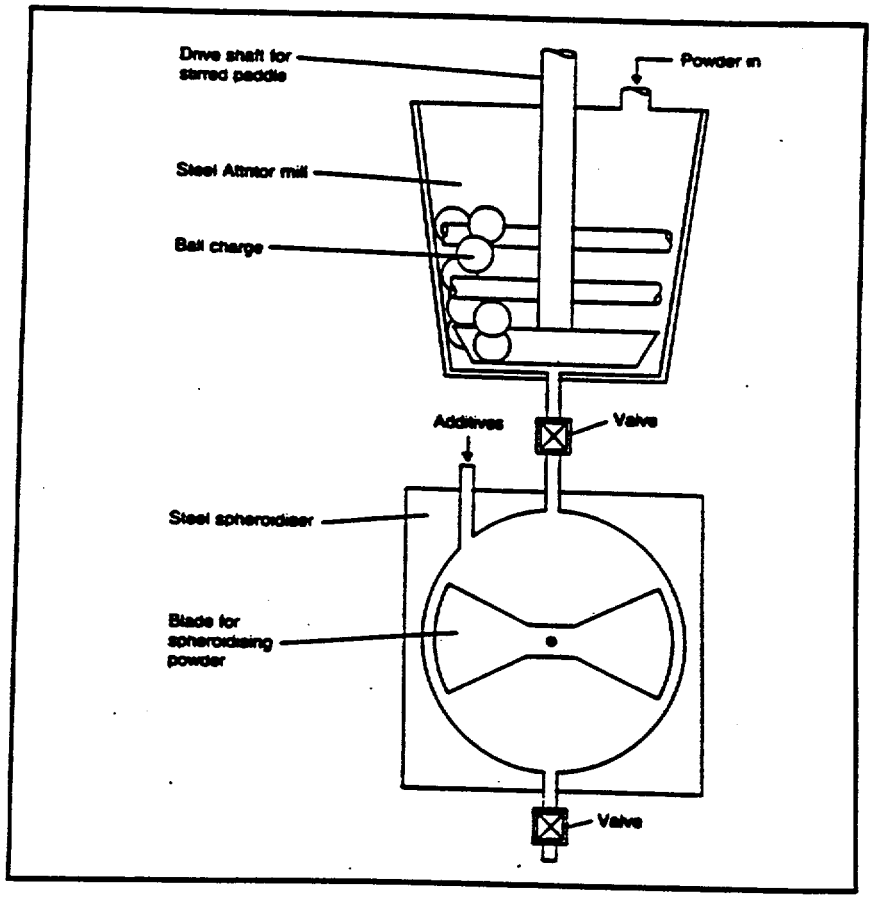


Figure 4. Attritor mill and spheroidizer used in BNFL's SBR.

BNFL claims extremely good homogeneity with their SBR. The autoradiographs used in their public relations and sales brochures support this claim. Although their fuel specifications are proprietary, the maximum plutonium particle (or agglomerate) size allowed is on the order of 100 μm. The average sizes experienced have been less than 20–30 μm (Ref. 17). This is better than the reported experience with Master-Mix-type fuels. However, this comparison may not be appropriate. SBR agglomerates may consist of pure PuO_2, whereas master-mix agglomerates include only 20%–40% plutonium.

2.2.6 Particle Fuel Technology

Although the vast majority of MOX fuel has been produced in pellet form, some alternative fabrication technologies utilizing particle fuel have been attempted. Many particle fuel fabrication processes have been investigated. The more important of these are described in this section. These processes all differ from those described in Sections 2.2.1–2.2.5 in that fuel pellets are not produced. Rather, fuel particles are loaded directly into the cladding tube prior to compaction. These particles are then compacted through one of several techniques directly in the cladding tube. Particle fuel technologies have been investigated because of their promise of greatly simplified fuel fabrication, which translates to lower cost and adaptability to remote operation. As pellet fuels gained dominance during the 1960s, interest in particle fuel technology waned. Nevertheless, limited particle fuel research continues to this day.

One of the most studied of these particle fuel fabrication processes is vibratory compaction (VIPAC). Fully dense feed material (i.e., produced through arc melting and crushing) is separated into 3–5 size ranges by screening. Precise amounts of each size fraction are then mixed to form a feed of known particle size distribution. This feed is then poured into a cladding tube. Vibration, often with the help of a static load placed atop the powder, is then utilized to compact the powder. Smear densities as high as 80% TD are readily achievable. An improved VIPAC process developed by the Research Institute of Atomic Reactors in Dimitrovgrad, Russia, reportedly produces smear densities as high as 88% TD. This is nearly as high as the smear density of a pellet fuel rod, taking into account the dish, chamfer, and fuel/clad gap.

Another particle fuel fabrication type often referenced in early MOX fuel literature is swage compaction. In swage compaction, particles are loaded into an oversize cladding tube, which is then sealed with an upper end cap. The entire fuel rod is then swaged (mechanical reduction in diameter) to the desired fuel pin diameter. In the process, the particle fuel is compacted. Development of this process was abandoned based on both fabrication and irradiation difficulties.

One other important particle fuel technology is sphere compaction (SPHEREPAC). Microspheres produced by sol-gel techniques are used as feed to a VIPAC process. The primary advantage of SPHEREPAC is its elimination of dusting processes. Limited research on SPHEREPAC continues to this day.

3. DOMESTIC MOX IRRADIATION EXPERIENCE

One of the primary difficulties with assessing the state of the irradiation experience base in the United States is that the information is somewhat dated, sometimes incomplete, and spread among a variety of historical documents. A number of different fuel manufacturing processes were tried by U.S. fabricators in test reactors, ultimately leading to the selection of oxide pellets for use in MOX LTAs and even in partial core loads in commercial reactors. Many of the deficiencies that were identified by earlier irradiation tests have since been resolved. The advantage of the U.S. MOX documentation is that much of it is publicly available.

This chapter attempts to provide as complete a summary of the domestic MOX-irradiation experience base as can be developed from the open literature. The experience can be categorized by fuel vendor or by the reactor that irradiated the fuel. In this chapter, a historical description of the irradiation experience is given by fuel vendor. This is somewhat confusing in several cases, because while one vendor designed the fuel, a second vendor fabricated the fuel. For example, some of the MOX fuel irradiated in the Garigliano boiling water reactor (BWR) was designed by GE, but manufactured by BN. This experience is primarily attributed to BN, but is also described under GE's experience.

Appendix A has the U.S. MOX experience categorized by the reactor that irradiated the MOX. This appendix contains a description of the methodology that was used and the table that contains detailed data for the irradiations. In this approach, an emphasis was placed on the rods that were nondestructively and/or destructively examined. The table identifies approximately fifty characteristics of the irradiation. The major areas covered are reactor specifics, MOX assembly design and fuel isotopics, fabrication techniques, a summary of the examinations conducted, and miscellaneous fuel performance observations. The table also contains references that support the data so that the reader can find additional information. The advantage of categorizing irradiations by plant is that a uniform cross-comparison of the characteristics of the irradiation in different U.S. reactors can be made. The table also contains some detail that is not provided in the discussion below. The table does not include the Plutonium Recycle Test Reactor (PRTR), Materials Test Reactor (MTR), Engineering Test Reactor (ETR), or Experimental Boiling Water Reactor (EBWR) irradiations, but these are discussed below.

3.1 U.S. GOVERNMENT DEVELOPMENT

The United States Atomic Energy Commission (AEC) began plutonium recycling studies in 1956. Its efforts were concentrated in two programs, the Plutonium Utilization Program (PUP) managed by the Battelle Pacific Northwest Laboratories (BNWL) and the Saxton Program managed by Westinghouse (W). Each of these programs is described in the following two subsections.

In the early 1970s, the AEC decided that additional government support of plutonium recycling was not justified because private industry was fully capable of commercializing the technology. One of the best summaries of the AEC work, and other early plutonium recycle work, is located

in the *Draft Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in LWRs, WASH 1327* (Ref. 18, Vol. 2, pp. II-34 through II-60) and the follow-on *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors, NUREG-0002* (Ref. 19, Vol. 2, pp. II-11 through II-40). The latter summary is reproduced for the convenience of the reader in Appendix B. In addition to the PUP and Saxton programs, the AEC participated to a limited extent in commercial and international plutonium recycle programs.

3.1.1 Plutonium Utilization Program (PUP)—Early Test Reactor Irradiations

The PUP began in 1956 at the BNWL and was funded by the AEC. The purpose of the program was to develop the necessary technology to implement plutonium recycle in thermal reactors. The time frame of the program must be kept in mind during review of the program results, because the nuclear industry was in its infancy during the first half of the program. Tests were performed in the MTR, the ETR, the EBWR, and the PRTR. Some of the capsule tests performed in the MTR, ETR, and PRTR are described in Table 2 (Ref. 20, p. 673).

The bulk of the testing under the PUP was performed in the PRTR, which was built specifically for this purpose. This reactor is heavy-water moderated and cooled, with a thermal power of 70 MW (Ref. 21, p. 126). The reactor fuel elements, which consisted of groups of rods strapped together, were contained in vertical pressure tubes. The reactor design is similar in many respects to the modern CANDU design. The reactor also included the Fuel Element Rupture Test Facility (FERTF), in which "high-risk" experiments could be performed without risking contamination of the entire system. The FERTF included its own coolant system and was used for tests such as the intentionally defected fuel tests.

At the time of PRTR construction and during the early years of its operation, it was not clear that the now common, cold-pressed and sintered (CPS) pellet fuels would dominate the nuclear industry. Thus, testing under the PUP did not focus on CPS fuel. In fact, most of the testing was for other fuel types including hot-pressed pellets, swage-compacted, and VIPAC. The nonpellet fuels were believed to hold great promise for lowering fuel production costs without reducing fuel performance. As the nuclear industry expanded, however, and pellet fuels began to dominate, research into these alternative processes dwindled and finally ceased in the United States.

In the Saxton program (discussed in the following section), approximately 23% of the fuels tested were VIPAC, with the remainder being CPS fuels. This seems to reflect the general shift in MOX fuel manufacturing philosophy towards the CPS fuels. The fuel types are too different to extrapolate performance data from one to another. In fact, extrapolation of even the CPS data to modern fuel is questionable due to the great advances in MOX fuel production that have been made in the interim.

The irradiation experience resulting from the PRTR MOX fuel tests are summarized in Table 2 (Ref. 20, p. 127). One of the CPS rods did develop a defect during irradiation due to internal hydriding, which was a common cause of failure in the early MOX development programs. Additional tests on metallic plutonium fuels and on UO_2 fuels were also performed in the PRTR.

Table 2. Summary of the U.S. PUP early test reactor experience

Reactor	Fuel Type	Number of Elements	Pu Concentration (wt. %)	Peak Linear Heat Rating (W/cm)	Peak Burnup (GWd/MTM) [1E20 fissions/cm^3]
PRTR	VIPAC	20	0.5	525 16 kw/ft	18.5
PRTR	Swaged	61	0.5	425	12.5
PRTR	VIPAC	16	1.0	445	11.5
PRTR	Swaged	33	1.0	510	13.5
PRTR	?	1	1.5	145	13.0
PRTR	VIPAC	79	2.0	655 20 kw/ft	7.8
PRTR	Swaged	2	2.0	655	8.1
PRTR	Hot and Cold Press Pellet	2	2.0	705	1.8
PRTR	VIPAC	1	2.0	560	1.2
PRTR	?	1	4.0	885	3.5
MTR	MCO*	2	0.03	496 & 560	[0.33 and 0.93]
MTR	MM**	10	0.03 to 5.7	220 to 1250	[0.3 to 2.6]
MTR	MCO	2	0.03	624 and 786	[0.47 and 1.0]
MTR	MM	10	0.2 to 7.5	108 to 885	[0.1 to 4.2]
MTR	High-energy-impacted fuel	2	2.5	1290 39.3 kw/ft	[0.3]
ETR	MM and high-energy-impacted fuel	4	2.5	460	[0.37]
MTR	High-energy-impacted fuel	32		1035	[6.8]
MTR	MCO	1	1.0	570	[0.2]
ETR	MM	1	0.6	460	[0.8]
ETR	MM	1	0.5	404	[0.2]
ETR	MM	1	0.8 to 3.0	440	[0.3]
ETR	?	1	?	550	[0.2]

* Mixture of UO_2 and $(U,Pu)O_2$ coprecipitate

** Mechanically mixed UO_2 and PuO_2

As shown in Table 2, most of the pellet irradiation experience from the early test reactors involved very low burnup as compared to the high burnup obtained with more recent fuels. In addition, the majority of the early experience is with alternative fabrication techniques that are not relevant to the FMDP. PIEs were performed on several irradiated rods in the PRTR.

It is concluded that the experience from the very early test reactors, while interesting in the historical context of MOX fuel development, is for the most part irrelevant to the domestic plutonium disposition program.

3.1.2 Saxton Program

Based on the early success of MOX irradiation experiments described in the previous section, the Saxton Program was initiated. The Saxton program was an AEC program that was subcontracted to Westinghouse. Westinghouse provided nine LTAs for irradiation in the second core of the Saxton pressurized water reactor (PWR). The Saxton reactor was rated at 23.5 MWth with an active fuel length of approximately one meter. The MOX assemblies, first loaded in 1965, contained 638 MOX rods with a plutonium concentration of 6.6%, such that a total of 23 kg of plutonium was loaded. Most of the rods were clad with Zr-4, but 30 rods were clad with 304 SS (stainless steel). The majority of rods contained CPS pellets, but a sizable fraction were VIPAC (~23%). One additional distinction about the Saxton fuel is that the plutonium contained in both the pellet and VIPAC fuels was converted from metal reduction buttons to oxide through DMO.

These rods were irradiated for nearly three years (known as Saxton Core II) at peak linear heat rates of 18.7 kW/ft to a peak pellet burnup of 28 GWd/MT. Of the 638 rods initially inserted into Saxton, 250 were reconstituted into a looser lattice and reintroduced for additional irradiation. These 250 rods (known as Saxton Core III) remained in-core for an additional 2.5 years, and Saxton ceased operation on May 1, 1972. The peak linear heat rate achieved during the second irradiation was 21.1 kW/ft. The peak pellet burnup achieved was 51 GWd/MT.

No fuel failures occurred during the initial irradiation during Core II. PIE of several of the rods indicated satisfactory performance. However, during the second irradiation, there were 37 identified failures out of the 250 MOX rods initially placed in Core III. All the failures occurred in high-power rods between the middle and end of Core III. Accelerated oxidation was seen on the defected rods, and hydriding was noted in the metallography of several rods. In any event, none of the 37 rod failures were attributed to any generic fuel problem intrinsic to the use of MOX. Westinghouse obtained a great deal of MOX experience from the Saxton irradiations (Ref. 22 and Ref. 23). More information on the extensive PIE program at Saxton is presented in the tables contained in Appendix A.

In addition to the large amount of fuel performance information generated in the Saxton program, a significant amount of overall operating experience was obtained with a partial MOX core (9 out of 21 assemblies were MOX). In addition, significant core physics measurements such as control rod worth and temperature coefficients were measured and compared to calculations. The plutonium used in the Saxton program was 91.4% fissile.

With respect to the FMDP, specifically PWR irradiations, the Saxton experience is relevant. The fuel performance information gathered on pellet fuels was rather extensive. This experience was the springboard into the manufacture of LTA's for insertion in commercial reactors. It is true, however, that the techniques used to make the fuel are somewhat outdated as compared to modern fabrication techniques, expertise, and fabrication equipment. However, the fact that 91.4% fissile plutonium was used may help provide answers to questions concerning the differences in physics models for RG plutonium versus WG plutonium.

3.2 WESTINGHOUSE MOX DEVELOPMENT

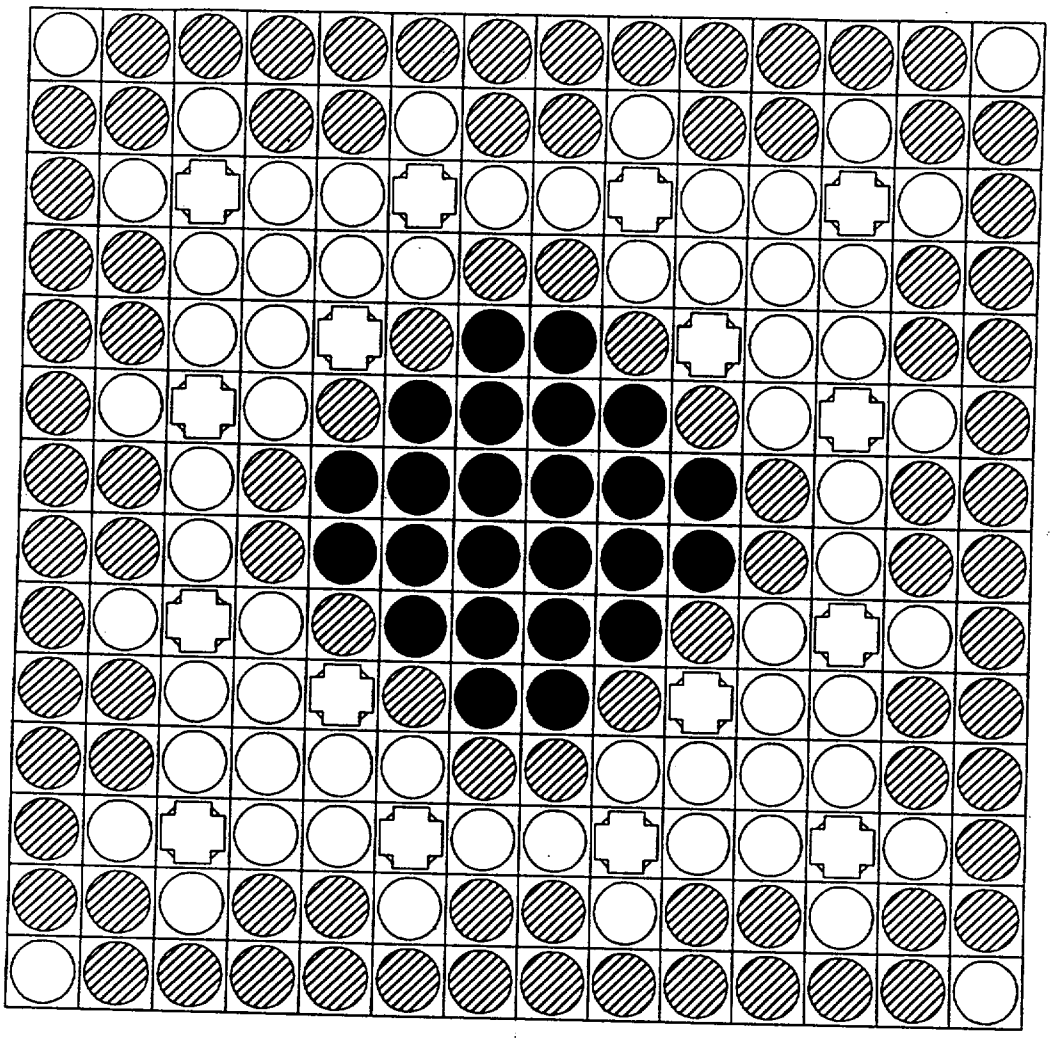
At the conclusion of the Saxton Program and the PUP, the AEC decided that additional government support of plutonium recycle was not justified because the utilities, reactor vendors, and fuel manufacturers were fully capable of commercializing the technology. Westinghouse, having been the AEC's subcontractor for the Saxton Program, was in a position to take the lead in these commercialization efforts. Westinghouse had also been involved in a parallel development program, a joint program between the Edison Electric Institute (EEI), a consortium of utilities, and Westinghouse, known as the EEI/Westinghouse Plutonium Recycle Demonstration Program (PRDP).

3.2.1 EEI/Westinghouse Plutonium Recycle Demonstration Program

The PRDP was conceived to study and demonstrate the economical use of MOX in PWRs. It consisted of two phases—an initial analytical phase and a follow-on MOX irradiation demonstration. The initial phase, which began in 1967, studied the technical issues and economics of MOX use in PWRs. Many of the facets of MOX use were explored using analytical techniques and critical experiments. Among other issues, plutonium's influence on control rod and soluble-boron worth were studied. Computer codes were modified to accommodate the characteristics of MOX fuel, bringing the modeling capabilities for MOX cores to the same level as those for UO_2 cores.

During the second phase of the PRDP, a reference MOX core design was developed. The core, based on the concept of self-generated recycle, was fueled with 1/3 MOX assemblies. (Self-generated recycle refers to the situation in which plutonium recovered from previous core discharges is reinserted as MOX fuel. No additional plutonium is placed in the core.) With all the MOX located in discrete assemblies, the 1/3-core fraction could be accommodated within the existing design limits without changes to the control system because none of the MOX bundles were located in control positions.

Another portion of Phase II of the PRDP consisted of manufacture and irradiation of MOX LTAs in the San Onofre Unit-1 reactor. San Onofre 1 is an early three-loop Westinghouse PWR with an electric capacity of 456 MW. A total of 720 MOX rods were placed in four LTAs (the assemblies consisted of all MOX rods). These four LTA's were the first PWR MOX assemblies irradiated in a U.S. commercial reactor. The plutonium (approximately 86% fissile) was supplied by the AEC at a reduced cost, as partial sponsorship of the study. Figure 5 shows the assembly design.



Center Rods: 3.85 wt% total plutonium

Edge Rods: 3.65 wt% total plutonium

Corner Rods: 3.30 wt% total plutonium

Figure 5. MOX assembly design for San Onofre 1.

At the conclusion of San Onofre's first cycle, fifty-two of the 157 original UO_2 assemblies were off-loaded and replaced with fresh fuel. The four MOX LTAs were included in this new fuel and were placed on the core periphery in nonpeaking positions such that core symmetry was maintained. At the end of Cycle 2, the LTAs had reached an average assembly burnup of 6,450 MWd/MT, and a peak linear heat rate of 6.8 kW/ft. The LTAs were visually inspected. Two of the four LTAs contained 52 removable fuel rods each. At the end of Cycle 2, four rods were removed and subjected to nondestructive examinations. Two of these rods were subjected to destructive examinations.

The assembly was reconstituted using UO_2 rods and then reloaded in the core for an additional cycle. At the conclusion of Cycle 3 (after two irradiation cycles), the four LTAs were again removed. It was originally intended that they be irradiated for three cycles, but contemporaneous failures in unpressurized UO_2 rods led to the conservative decision to remove them. Fuel densification in UO_2 fuels had, in some cases, led to cladding collapse onto the fuel. Rod pressurization eventually solved this problem. While the MOX rods were not pressurized, they showed no indications of cladding collapse at the end of Cycle 3.

At the end of the third cycle, six rods were nondestructively examined, and two rods were destructively examined. These rods were from the same assembly used in Cycle 2, but ended the third cycle with an average assembly burnup of 19,000 MWd/MT and a peak MOX pellet burnup of 23,500 MWd/MT. The results of these examinations are especially useful because of the comparison they provide between two competing fuel manufacturing processes. Of the two rods destructively examined at the end of each cycle, one contained pellets produced through comilling, and the other contained pellets produced via a coprecipitation process. The PIE results identified some advantages of coprecipitated fuel. Appendix A provides further details of this irradiation.

The results of the San Onofre irradiations provide a comparison between the performance of comilled MOX and coprecipitated MOX. Although neither of these fuels is typical of the high quality, high density fuels produced today, the relative performance is important nonetheless. One of the differences in the fuel behavior was densification, which is of much less interest today as a result of improved fuels. Another difference, which is closely related to the densification behavior, is fission gas release.

In MOX fuel, the majority of the power production takes place in the plutonium. In comilled fuels, the plutonium exists as discrete particles surrounded by a UO_2 matrix. Power production is highly localized in the plutonium particles. In coprecipitated fuel, the plutonium is distributed into plutonium-enriched regions surrounded by a UO_2 matrix. The power production is, therefore, more diffuse. The more homogeneous coprecipitated fuel more closely resembles the plutonium distribution seen in irradiated UO_2 fuel, and its behavior is (not surprisingly) more similar to that of UO_2 fuel.

In comilled MOX pellets, the fission products tend to coalesce around the plutonium particles in which most fissions occur. In the tested coprecipitated fuel, the fission products are distributed through a much larger fraction of the surrounding UO_2 matrix due to the uniformity of the coprecipitated master mix. Therefore, fission gas bubbles form more readily in the comilled

fuel. It is the formation of bubbles that eventually leads to fission gas release from the fuel matrix. Fission gas release from the comilled San Onofre MOX fuel was nearly twice as high as that from the coprecipitated fuel. The San Onofre data provided early evidence of the desirability of a high degree of homogeneity in MOX fuel. Reference 24 provides the original documentation for the San Onofre irradiation.

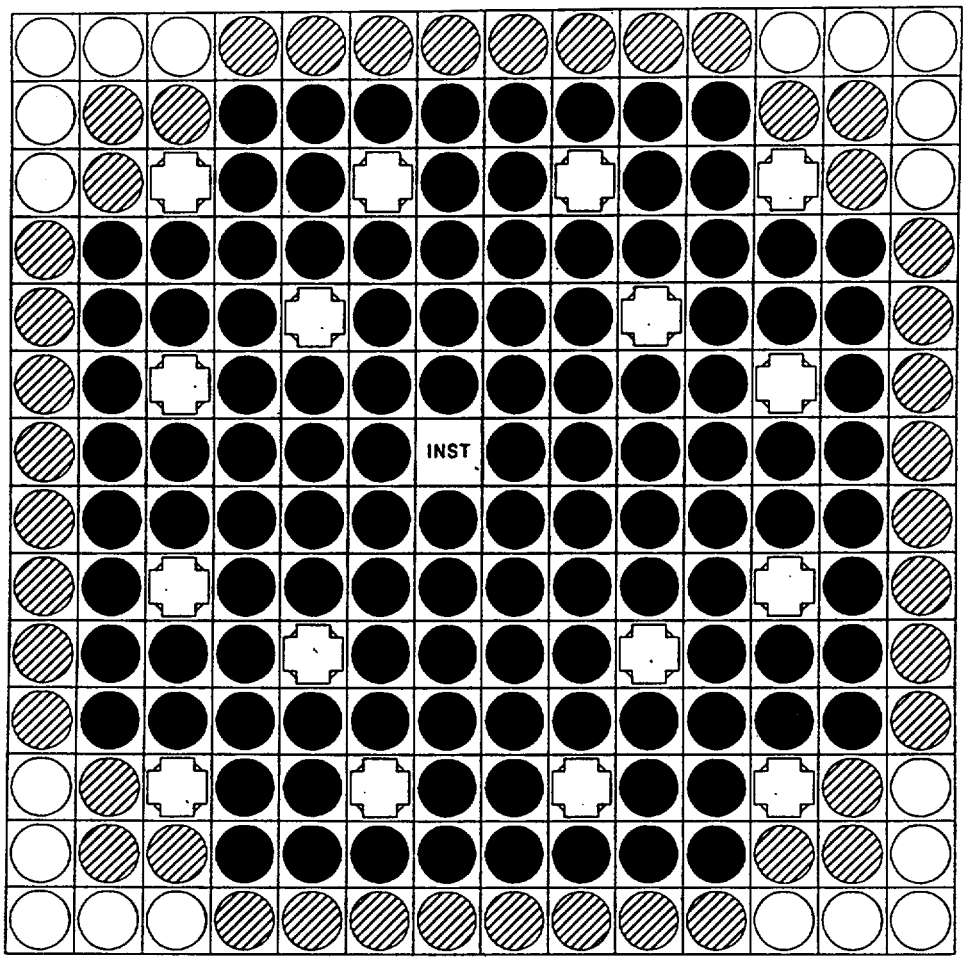
3.2.2 Overseas Westinghouse LTA Programs

Following the successful irradiations in Saxton and San Onofre, Westinghouse participated in a number of collaborative research efforts with foreign participants. Westinghouse produced a total of sixteen LTAs for irradiation in the Italian Trino reactor, the Swiss Beznau-1 reactor, and the Japanese Mihama-1 reactor. Each of these reactors had a Westinghouse-supplied nuclear steam supply system (NSSS). The only additional domestic MOX programs in which Westinghouse participated are the Electric Power Research Institute (EPRI) Plutonia Fuel Study (Ref. 25) and the Ginna MOX irradiations.

3.2.3 Ginna LTAs

The Ginna irradiations are believed to be a continuation of the EEI/W - EPRI program described in Section 3.2.1. MOX rods sufficient to fill four assemblies were manufactured by Westinghouse in 1974 for irradiation in Rochester Gas and Electric (RG&E) Company's Ginna nuclear power station. Ginna is an early-design Westinghouse PWR with an electric output of 498 MW. The assemblies were originally to be loaded in 1975, but other issues facing RG&E forced a delay in the MOX licensing activities. These rods were manufactured when utilities viewed MOX use as an eventual certainty. However, by the time the rods were ready for insertion, domestic MOX use had essentially been abandoned as a result of the Executive Order on nonproliferation issued by President Carter. The licensing revision to permit insertion of the LTAs, in fact, faced the issue of whether or not their irradiation was permitted under the associated NRC policies. The unassembled rods were stored in Westinghouse's Cheswick facility until 1979, when they were transferred to Exxon Nuclear Company, Inc. (ENC) for assembly. It was determined by RG&E management that irradiation of the MOX rods was the best method of dispositioning the rods, and this argument was made to the NRC. They were eventually loaded into the reactor for Cycle 10 in 1980.

The four 14 × 14 LTAs contained a total of 736 MOX rods utilizing approximately 83% fissile plutonium. Figure 6 shows the assembly design for these LTAs. The LTAs were irradiated until 1985 to an average assembly burnup of approximately 40,000 MWd/MT. No operational problems were reported for the LTAs. In fact, these LTAs operated successfully through Ginna's steam generator tube rupture in 1982. No PIE work was done on the assemblies, and the assemblies are currently stored intact in the Ginna fuel pool. These assemblies were the last commercial MOX assemblies irradiated in the United States. Information on the Ginna irradiation is taken from Refs. 26 and 27.



14x14
179 MOX rods

Center Rods: 3.20 wt% total plutonium

Edge Rods: 3.00 wt% total plutonium

Corner Rods: 2.60 wt% total plutonium

INST: Instrumentation tube

Crosses: represent rod cluster control guide tubes

Figure 6. The Ginna LTA MOX assembly.

3.3 GENERAL ELECTRIC (GE) MOX EXPERIENCE

The other domestic reactor and fuel vendors also had active MOX research programs during the 1960s and 1970s. In particular, GE joined the EEI in a joint program to pursue MOX use in BWRs. This program was parallel to the EEI/W PWR program described in Section 3.2.1. GE also has some MOX experience from capsule tests. Eventually, GE assisted in a number of international MOX studies and in EPRI-MOX work. All these activities are described in the following sections. Open literature documentation is available for the EEI/GE Big Rock Point (BRP) tests and the EPRI Quad Cities 1 irradiations.

3.3.1 GE's Early MOX Testing

General Electric, the U.S. AEC, and Euratom jointly conducted a BWR MOX program in the early 1960s. The program consisted of irradiation of a number of MOX rods in the Vallecitos BWR. Vallecitos was an early BWR prototype reactor, which had a thermal output of 30 MW. The reactor was fueled with plate-type fuel, but was capable of operating with rod bundles for testing (Ref. 28). The AEC-GE-Euratom program focused upon obtaining basic operating data for MOX fuel in thermal reactor environments. The data were used to develop a model for predicting the isotopic and reactivity behaviors of MOX fuel.

Sixteen MOX test rods were inserted in the Vallecitos reactor. The rods were clad with Zircaloy and measured 1.07 cm OD (outside diameter). The MOX, which contained 1.5% plutonium of unreported isotopic composition, was irradiated to a maximum burnup of 1.4E20 fissions/cm^3 (Ref. 20, p. 669). Some PIE, including measurement of the plutonium distribution in the irradiated fuel, was performed. Detailed results have not been located in the open literature, however.

GE also participated in irradiations of LMR fuel (20% plutonium) in a number of reactors, including their General Electric Test Reactor (GETR). The extent of GE participation in the PRTR tests is not clear. However, according to Ref. 29, the 80 rods containing hot- and cold-pressed pellets that were irradiated to ~9 GWd/MT, at up to 21.5 kW/ft, were manufactured by GE. This indicates that the PRTR pellet irradiations are more applicable to BWRs than to PWRs.

Ref. 29 and Ref. 30 list GE as the manufacturer of four rods that were tested in the Dresden-1 nuclear power station. Four fuel bundles, with a single MOX rod per bundle, were inserted in 1967. The rods contained hot-pressed pellet fuel and utilized Dresden self-generated plutonium that was ~80% fissile. GE is also known to have been a participant in EPRI's Plutonia Fuel Study (Ref. 25) described in Section 3.8.

3.3.2 EEI/GE Plutonium Utilization in BWRs Program

As with the parallel EEI/W PWR program, the initial stage of the EEI/GE program was analytical in nature. The promising results of the analytical work led rapidly to plans for irradiation testing in the Consumers Power Company BRP BWR. Four types of fuel consisting of annular, solid, and dished pellets with different plutonium loadings were tested. The tests

focused on the behavior of annular fuel, with several solid and solid-dished pellet rods included for comparison. All of the test rods contained cold-pressed and sintered MOX pellets produced from mechanically blended powder in a process similar to that described in Section 2.2.1. The sixteen bundles were loaded for Cycle 7 in May 1969.

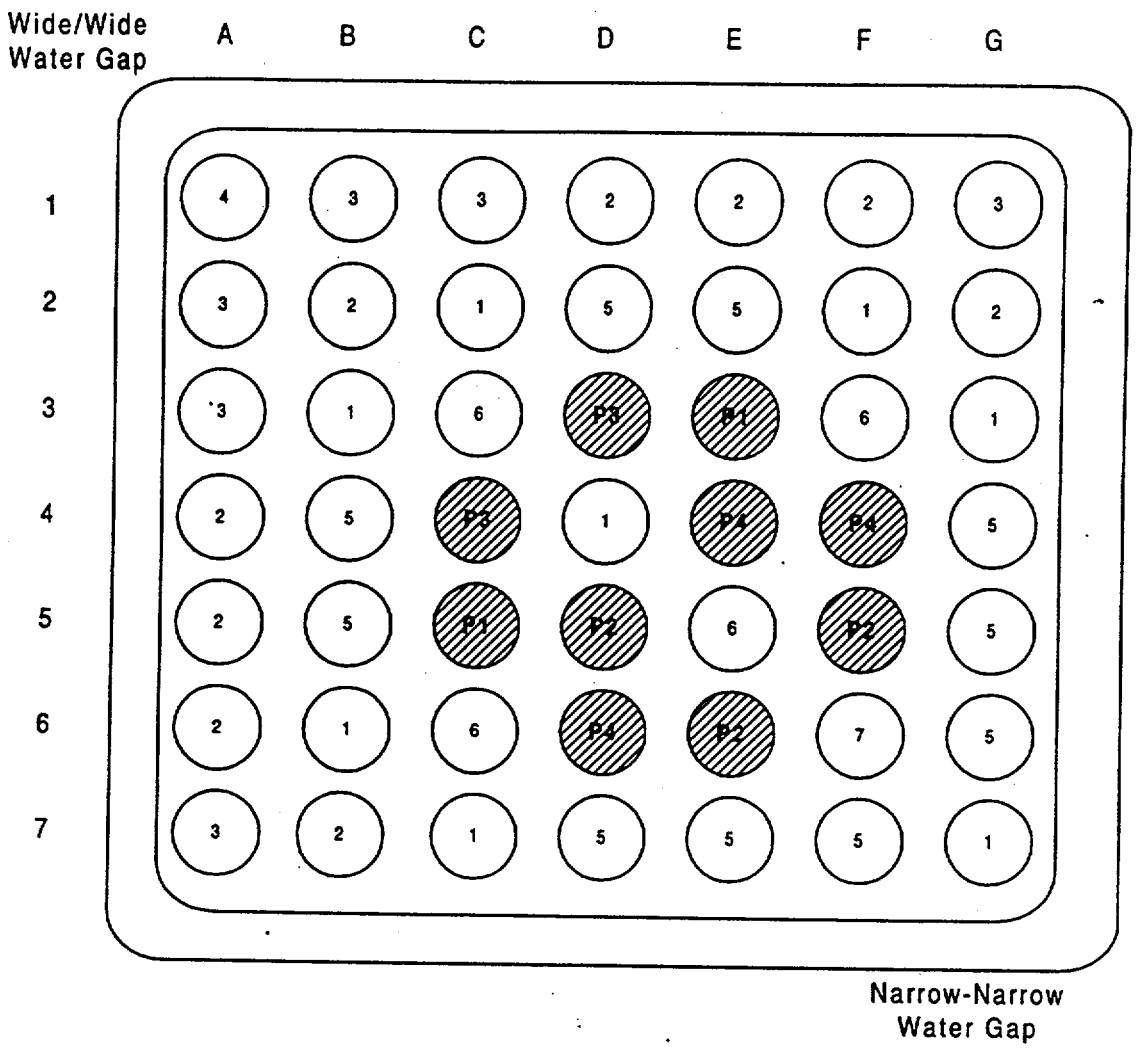
At the conclusion of BRP Cycle 7, the MOX bundles were removed from the core and sipped. One of bundles was suspected of containing a leaking rod based on the results of the sipping tests. Four rods were permanently removed and sent to the Vallecitos Nuclear Center for PIE (Ref. 31). The remaining 28 rods (including six of the ones on which profilometry was performed) were returned to the reactor for additional irradiation and eventually removed in the mid-1970s. Appendix A shows some information regarding the 32-rod irradiation. However, as shown in this appendix, detailed rod-by-rod information on many rods could not be located.

Three reload MOX bundles were introduced into BRP for Cycle 8. Each of the 9 × 9 bundles contained 68 MOX rods with a maximum plutonium loading of 9.1 wt%. Rod descriptions for the three reload bundles are given in Ref. 32, p. 10. All of the fuel was manufactured via cold-pressing and sintering of comilled powder. Natural uranium was used as the MOX diluent, and both 80% and 90% fissile plutonium was utilized. The 90% fissile plutonium used in 180 of the 204 rods was obtained from the AEC. The 80% fissile plutonium was obtained from Dresden-1 recycled fuel.

The MOX reload bundles operated during Cycles 8, 9, and 10. At the conclusion of Cycle 10, sipping tests indicated leaking rods in two of the three bundles. Due to a nonnuclear primary system problem, high rates of crud formation were experienced on all fuel in the reactor during this period. Two rods in the third MOX bundle failed during operations to remove the adherent crud during the shutdown between Cycles 10 and 11. The two failed rods were replaced, and this single MOX bundle was reinstalled in the reactor for Cycle 11.

All of the GE MOX rods were removed from BRP at the end of Cycle 11. As with the two-rod-per-bundle tests, only limited data is available for the PIE that is known to have occurred on the BRP MOX fuel. Some of the data is believed to be documented in EPRI reports, with the remainder in proprietary GE documents. EPRI took over sponsorship of the EEI/GE MOX program in 1974, and it is likely that all the EPRI reports from this program have not been located. It is important to note that most of the BRP data are for annular fuel.

Following the initial success of the BRP reload bundles, GE proceeded to prepare MOX rods for the initial core of the Vermont Yankee reactor. Due to some licensing difficulties, these rods were eventually constituted into five bundles (four center and one periphery) that were loaded into Quad Cities 1 with the first core reload in 1974. Each of the four central bundles contained ten MOX rods in an island arrangement. The MOX island was surrounded by UO_2 rods and was slightly offset towards the water gap corner to isolate the MOX from the control blade. The peripheral bundle contained eight MOX rods scattered near the reflector corner of the bundle. As with the BRP MOX, two plutonium assays were used: Dresden recycle (~ 80% fissile) and AEC plutonium (~ 90% fissile). The isotopic makeup of the plutonium was similar to, but not identical to, that used in the BRP LTAs. All of the pellets were cold-pressed and sintered from comilled powder, with natural uranium used as the diluent. Figure 7 below shows the assembly design for a central MOX bundle at Quad Cities.



Numbers: UO_2 or UO_2-Gd pins of various enrichments

P1: solid 80% fissile MOX pin, 2.14 wt% fissile plutonium

P2: solid 90% fissile MOX pin, 3.52 wt% fissile plutonium

P3: annular 80% fissile MOX pin, 2.34 wt% fissile plutonium

P4: annular 90% fissile MOX pin, 3.62 wt% fissile plutonium

Figure 7. Quad Cities central MOX bundle.

The MOX bundles were irradiated for Cycles 2, 3, 4, and 5 with various reconstitutions taking place. EPRI sponsorship of the program ended at the conclusion of Cycle 5. Nevertheless, GE and Commonwealth Edison continued the program, irradiating a reconstituted central assembly and the peripheral assembly for an additional cycle. The extent of GE proprietary information on the fuel examinations is not known but is likely to be substantial.

A wealth of information was obtained as a result of the Quad Cities' irradiations and the PIEs. The Quad Cities' program demonstrated that extended burnup of BWR MOX fuel was feasible. The bundles were irradiated for up to eight years, to maximum burnups of 39.9 GWd/MT bundle average and 57 GWd/MT peak pellet. The MOX fuel operated without failure at peak linear heat rates up to 15.5 kW/ft. Appendix A shows the examinations that were conducted and the associated EPRI reference documents. Ref. 33 provides a good overall summary of the program and the fuel performance results.

3.3.3 Other GE MOX Testing

GE has been involved in a number of additional MOX programs. In their recent study on plutonium disposition (Ref. 34), GE lists two foreign MOX programs: the Halden Plutonium Program and the Kritz Program. Only brief descriptions are given. Furthermore, GE performed a number of transient tests on MOX fuel in the SPERT [transient irradiation test reactor at Idaho National Engineering and Environmental Laboratory (INEEL)] facility. They also designed some of the MOX fuel (manufactured by BN) that was irradiated in the Italian Garigliano BWR. Finally, GE contributed to EPRI's MOX fuel densification study (Ref. 25).

GE's involvement in the Halden Plutonium Program, as described in Ref. 34 consisted of ramp testing of MOX fuel rods containing annular pellet fuel. The tests were performed at linear heat rates of 17 to 22 kW/ft. The results are described as "the effects of pellet-clad interaction showed acceptable behavior of the MO_2 rods." According to Ref. 29, two rods were tested in Halden. Both solid and annular pellets were tested to peak pellet burnup of 16 GWd/MT.

The Kritz Program is described in the vendor report (Ref. 34) as the preirradiation testing of the Garigliano MOX reload bundles, which showed that the existing analytical models were adequate for prediction of MOX neutronic performance. The available references suggest that these bundles were manufactured by BN to GE's specifications.

Additional testing occurred in the GETR, which was a test reactor that was used for fuel and materials development (Ref. 35). Two rods containing annular MOX fuel were irradiated to ~22 GWd/MT at up to 18 kW/ft (Ref. 29). It is probable that a number of additional tests were performed by GE in support of MOX development. Irradiation of MOX fuel samples was performed in the GETR as part of the EPRI Plutonia Fuel Study as described in Section 3.8.

The available information on the Garigliano irradiations is incomplete. The available open literature information indicates that GE fabricated 96 MOX fuel rods contained in four demonstration island-type assemblies. These assemblies were introduced into the reactor in 1968. This fuel included 12 VIPAC rods, 24 hot-pressed pellet rods, and 60 CPS rods. An assembly average burnup of ~25 GWd/MT at a design peak linear heat rate of 15 kW/ft (Ref. 29,

p. XV-2) was achieved. Two plutonium concentrations were used in the fuel: 2.0% and 3.2%. Selected rods were removed after one, two, and three irradiation cycles corresponding to 15, 21, and 25 GWd/MT. PIE was performed under the auspices of a CNEN/ENEL (Comitato Nazionale per l'Energia Nucleare/Ente Nazionale per l'Energia Elettrica) program (described further in Section 4.2 below) at the Risø Laboratory in Denmark (Ref. 36). It is not known whether GE was granted access to any or all of the PIE results, but some of the preliminary results have been reported openly. BN produced 204 MOX rods, contained in four assemblies designed by GE, that were loaded at the same time (Ref. 37).

In 1975, a full reload of MOX island assemblies was introduced into the Garigliano core. This reload consisted of 46 assemblies, each of which contained 32 MOX rods. The MOX rods were manufactured by BN and designed by GE. The bundles remained in the core through 1981, to a peak burnup greater than 25 GWd/MT (Ref. 34). Details about the irradiation performance and/or PIE of this fuel have not been located; however, it is likely that GE has access to at least a portion of this information.

GE is currently involved with the Japanese in a program to load full-MOX cores into advanced BWRs. No information on this work is available. However, it is thought that this work would have direct applicability to the FMDP.

3.4 ABB Combustion Engineering (CE) MOX Experience

In their Plutonium Disposition Study report (Ref. 38), CE does not reference any original MOX experience. They provide a brief summary of the domestic and foreign MOX programs, but none are CE programs. Based on this information and on the lack of MOX fabrication facilities at the height of the domestic MOX development (Ref. 22, p. 5–2), it seems that CE has no MOX experience of its own. However, CE reportedly participated in joint experiments and testing with ALKEM and KWU (Kraftwerk Union AG) during the 1970s (Ref. 39).

3.5 Exxon Nuclear MOX Experience

Exxon Nuclear Company, Inc. (ENC) had an extensive MOX development program during the 1970s. Available reports of Exxon's activity are specific to BWR fuel. No reports of irradiation of ENC PWR MOX have been found. Two 9 x 9 MOX bundles were introduced into the BRP core in 1972. These bundles were tested at the same time as the GE BRP bundles. BRP's owner, Consumers Power Co., considered the plant a fuel test bed during this period. Two additional bundles were introduced at the next refueling in 1973. Based on the early success of these test bundles, partial-core MOX reloads were initiated.

Eighteen 11 × 11 bundles, each containing 24 MOX rods, were introduced in 1974. Eight additional bundles of the same design were introduced in 1976. This experience is the closest to commercial recycling of MOX that has occurred domestically. The license for MOX use in BRP limited the plutonium content of the MOX to 50 kg total. It was expected at the time that with the successful completion of the GESMO proceedings, this limit would be raised (Ref. 19, p. II-32).

In addition to the BRP experience, ENC manufactured eight MOX LTAs for the German Kahl BWR. No detailed descriptions of the LTAs, their irradiation performance, or PIE have been found for the ENC MOX fuel. It is likely that this information remains proprietary. ENC is not currently involved in the FMDP. However, Siemens Power Corp., a wholly owned subsidiary of Siemens AG, now owns what was ENC. Unfortunately, the purchase of ENC specifically excluded the mixed-oxide fuel data.

ENC participated in EPRI's Plutonia Fuel Study (Ref. 25). The extent of their participation is not clear, however, as described in Section 3.8 below. Further information can be found in Appendix A.

3.6 GULF UNITED NUCLEAR CORPORATION (GUNC) MOX EXPERIENCE

From the limited information available, it appears that GUNC's MOX experience is limited to that obtained from the joint GUNC/Commonwealth Edison program in Dresden 1. Self-generated plutonium was reintroduced into Dresden 1 in 1971. Starting with Cycle 7, eleven MOX bundles (each containing ~77% fissile plutonium in nine MOX rods) were introduced into the core. All the MOX rods contained 2.3% plutonium in natural uranium (Ref. 40). Reports of the remainder of the irradiation (Cycle 10) and the expected PIE on the fuel have not been found. Commonwealth Edison may have access to some of this information as a party to the original research. Further information on this irradiation can be found in Appendix A.

3.7 BABCOCK AND WILCOX (B&W) MOX EXPERIENCE

B&W manufactured fuel for the Fast Flux Test Facility (FFTF). They also provided fuel pellets for EPRI's densification and homogenization experiments as part of the Plutonia Fuel Study (Ref. 25). The MOX pellets were manufactured using a process similar to that used for the FFTF fuel. Ross and Benson (Ref. 41) summarize the process:

> "Pellets for the EPRI program were fabricated using the same process and equipment used to fabricate the FFTF fuel except that the lot size was 5 kg (instead of 50 kg). For the EPRI program only centerless ground fuel pellets were to be supplied by NMD. Encapsulation for irradiation testing and other physical and chemical tests are to be performed by BNWL.
>
> "The PuO_2 as received from ARHCO was used without any calcining treatment. The UO_2 used to manufacture the EPRI pellets was depleted UO_2 as prepared by the ADU process at NMD's Uranium Fuel Fabrication Plant.
>
> "After V-blending the PuO_2 and UO_2 powders in a P-K blender using an intensifier bar, the material was jetmilled using a Troust Fluid Energy Mill and then screened through a 60 mesh screen. No recycle material was added to this powder as would be done during commercial operation.
>
> "As a result of the jet-milling process, the mixed oxide becomes very active [meaning it sinters to a very high density], especially when there is no

recycle present, and it is necessary to add a burnable organic pore former such as carbowax to the powder to lower the density. The sintered density of jet-milled material, with no organic added to the powder, was 97–97.5% T.D.

"Carbowax pore former was added as -120/+200 mesh granules to the powder and the powder is slugged to a nominal density of 46% T.D., granulated and screened to -20 mesh granules. A die lubricant of -200 mesh sterotex (0.3%) is added prior to pelletizing. The pellets were pressed using a Hydramet hydraulic press with a single cavity punch and die set. The green pellets were 0.440" diameter × 0.480" long and were compacted at 5.7 tons to 55% T.D.

"The pellets were...sintered...and dry centerless ground using a Royal Master grinder to 0.3640 ± 0.0005" diameter.... The average density of the pellets was 94.315% T.D. and the one-sigma standard deviation is 0.188% T.D.... The sintered pellets were evaluated for plutonium homogeneity using standard alpha-radiographic techniques. The pellets do meet the B&W specification for MOX zones, which for 25% PuO_2-UO_2 particles, can be as large as 849 microns."

Reported irradiation experience with this B&W MOX fuel is limited to that available in the Plutonia Fuel Study reports. However, the fuels are intentionally referred to as fuel Type 11, 12, etc., such that one cannot readily extract the behavior of a single manufacturer's fuel. No additional reports of production or irradiation of B&W MOX have been found.

3.8 Other Domestic MOX Experience

In addition to the domestic programs described in Sections 3.1–3.7, Nuclear Fuel Services manufactured four MOX demonstration assemblies that were irradiated in the BRP BWR in the early 1970s (Ref. 19, p. II-32). Detailed descriptions of the assemblies, irradiation experience, and PIE have not been located. No additional references to Nuclear Fuel Services MOX research have been found.

EPRI instituted the Plutonia Fuel Study in 1975 under sponsorship by B&W, BNFL, Central Research Institute of Electric Power Industry (Japan), CE, ENC, GE, and Westinghouse. The program was meant to reproduce the UO_2 densification study (Ref. 42). Fuel was manufactured by a number of the participants, and possibly by all. The fuels are not labeled as to their manufacturer in the program write-ups. It may be possible, however, to reconstruct this information from other sources. The fuels underwent extensive preirradiation examination and then were irradiated in the GETR. PIEs were performed. The purpose of these studies was to examine the thermal- and irradiation-induced densification of MOX fuels for comparison with the behavior of UO_2 fuels. If a correlation between the fuels and their manufacturer/method can be deduced from other data or obtained from project reports, the data may provide a useful comparison of the behavior of the various fuels.

4. FOREIGN MOX IRRADIATION EXPERIENCE

MOX research in Europe, as with nuclear power research in general, followed developments in the United States closely during the 1960s and 1970s. For many European countries, MOX use promised energy independence—a long-sought goal. The Belgians irradiated their first MOX assembly in 1963, and they were soon followed by a number of other countries.

In December 1974, the Council of Ministers of the European Communities initiated the Research and Development Pluriannual Programme on Plutonium Recycling in Light Water Reactors. Ten of the forty-eight contracts issued to carry out the program investigated MOX fuel performance through irradiation and PIE (Ref. 43). This program brought together the numerous domestic programs throughout Europe and led to additional MOX irradiations and reloads. The ten programs are summarized in Table 3, taken from Ref. 43.

Table 3. CEC plutonium recycling program PIEs

Contract Number	Contractor	Reactor	Purpose	Assembly Burnup (GWd/MT)	Laboratory
1	BN - SCK-CEN	BR3/PWR	MOX rods in standard assembly (Z0–100)	32	SCK-CEN
2	INTERFUEL/ECN	HFR/PWR	Three MOX rod bundles in rig (vibrasol)	4–9	ECN/Petten
3	BN/GKN	Dodewaard/BWR	MOX rods in standard assembly (B201)	20	SCK-CEN
4	ENEA/ENEL	Garigliano/BWR	MOX rods in standard assembly	14, 21	RISØ
5	KWU	Lingen/BWR	Thorium-plutonium rods in standard assembly	4–20	TUI
6	ENEA/ENEL	Garigliano/BWR	MOX rods in standard assembly	7, 25	RISØ
7	BN - SCK-CEN	BR3/PWR	MOX rods in standard assembly	20–43	SCK-CEN
8	BN/GKN	Dodewaard/BWR	MOX rods in standard assembly	28	ECN/Petten
9	BCR	CNA/PWR	MOX rods in standard assembly	27	SCK-CEN and CEA/Saclay
10	FRAGEMA	CNA/PWR	MOX rods in standard assembly	27	CEA/Saclay

The Commission of the European Communities (CEC) program continued throughout the 1970s. However, when the program came up for renewal in 1980, both France and the United Kingdom believed that rapid deployment of LMRs would, in the very near future, eliminate all plutonium surpluses. The inaction of these two key countries kept the program from being continued, and shortly thereafter many of the national programs ended. Only two of the countries involved in the CEC program, Belgium and the Federal Republic of Germany, continued research into LWR MOX use.

As the 1980s progressed, the LMR revolution did not materialize as expected. In addition, success with reprocessing plants led to large and growing plutonium surpluses. France decided to pursue plutonium recycle in their existing LWRs. In the United Kingdom, BNFL reinstituted MOX research and planned to enter the commercial MOX fabrication arena. By the end of the 1980s, MOX use was expanding rapidly in Europe. Large MOX fabrication facilities were being planned and constructed. At the present time, the MOX industry in Europe has become a fully developed, commercialized enterprise.

The Europeans continued MOX development efforts after the 1977 Executive Order postponing reprocessing indefinitely in the United States. This Order, which effectively ended U.S. MOX research, was meant to influence the other nuclear powers to halt their plutonium recycle programs. However, despite the Executive Order and U.S. policy, other countries proceeded with the technical development and commercial use of RG MOX fuel. This continued development has led to advancements that partially eclipse the earlier domestic experience. Not only does the foreign experience base comprise more irradiated rods, but also most of the foreign experience is for modern high-density, homogeneous fuel.

Due to the widespread use of commercial MOX overseas, the foreign MOX irradiation experience with RG material is extensive. However, much of the data is proprietary, and a central table summarizing the examinations conducted on MOX fuel in foreign countries was not feasible to produce. Chapter 4 does, however, contain a general discussion of the foreign MOX experience.

Not surprisingly, extensive utilization overseas has resulted in improvements in MOX fuel performance. An example is fission gas release, which has been shown to be highly dependent on the fuel microstructure. Early fuels were rather inhomogeneous, containing relatively large plutonium particles and/or agglomerates. Also, the pellet density associated with earlier fuels was not nearly as high as that currently obtained. The net result was higher fission gas release from MOX than from UO_2. MOX produced with the current processes that result in more homogeneous fuel with smaller plutonium particles and higher density has shown much lower fission gas release rates, although these rates remain somewhat higher than those of UO_2 fuels.

4.1 Belgian MOX Experience

The Belgians entered the commercial nuclear power arena early, as evidenced by the presence of a Belgian team at the startup of the Shippingport PWR (Ref. 44, p. 13). They became interested in LWR recycle of MOX shortly after their domestic nuclear research began. In a cooperative effort between the fuel vendor BN and the Belgian Nuclear Research Centre at Mol—Studiecentrum voor Kernenergie-Center d'étude de l'Énergie Nucléaire (SCK-CEN)—the Belgians introduced their first MOX assembly into the BR3 reactor in 1963. In the years since, they have become one of the major players in MOX fuel development and manufacture.

4.1.1 Early Experience in BR3 Reactor

BR3 was the first PWR built outside the United States. It was basically an early Westinghouse design with a thermal output of 41 MW. The MOX bundle introduced into BR3 in 1963 was the

world's first MOX assembly introduced into a commercial reactor (Ref. 45). The island-type assembly utilized 12 stainless steel-clad VIPAC rods. The MOX rods reached burnups of only 3 GWd/MT assembly average and 6 GWd/MT peak pellet (Ref. 37, p. 184).

In 1965, the BR3 reactor core was redesigned with a triangular lattice to accommodate additional experimental and instrumentation capabilities. Included in the updated BR3/Vulcain core was a single MOX bundle containing 18 pins filled with pellet fuel, and 19 pins filled with vibrocompacted MOX powder (Ref. 46). The assembly reached an average burnup of 25 GWd/MT. The pellet-filled rods reached a peak pellet burnup of 44 GWd/MT (Ref. 37). The vibrocompacted rods were placed in lower power positions and reached only 26 GWd/MT peak burnup.

Two additional MOX assemblies, each containing 18 MOX rods, were loaded into the BR3 core in 1969. Half of these rods contained VIPAC fuel, and the other half contained pellet fuel. In subsequent reloads, the MOX core fraction increased steadily, reaching almost 50% with the final reload, which was 70% MOX. The BR3 reactor was shut down in 1987 after 25 years of MOX research. Although it is probable that substantial PIE was performed on the BR3 fuel, only limited documentation of these efforts has been found. The BR3 MOX data provided BN a firm foundation from which to launch their commercial MOX venture.

4.1.2 Other Fuel Development Irradiations

In addition to the BR3 MOX fuel irradiations, BN either led or participated in a number of additional MOX fuel development irradiations that eventually led to commercial use of their MOX fuel. These programs inserted MOX fuel into the Garigliano and Dodewaard BWRs, the Chooz A PWR, and the NPD (nuclear power demonstration) heavy water reactor. The fuel development and demonstration program irradiations are summarized in Table 4 taken from Ref. 47.

The fuels manufactured by BN during this period, from the mid-1960s to the mid-1970s, were produced using one of several experimental processes. Two types of vibrocompacted fuel were tested: homogeneous and heterogeneous. In the homogeneous VIPAC fuel, particles of homogeneous $(U,Pu)O_2$ were loaded into the cladding. In the heterogeneous fuel, the two larger particle fractions were UO_2, and the finest particle fraction consisted of PuO_2. The heterogeneous VIPAC fuel offered the promise of minimal plutonium handling.

The early pellet MOX fuels were manufactured by one or more processes referenced in more recent literature as "previous" techniques. These may also be categorized as homogeneous and heterogeneous. The earliest technique, which is a variation of comilling, pressed a homogeneous blend of granulated $(U,Pu)O_2$. A later technique, which was used in the pilot facility operated from 1967 to 1975, blended PuO_2 into granulated UO_2. The later tests used fuel from the commercial fabrication plant, which used a process now referred to as the "Reference" process, from 1973 to 1984. The "Reference" process blended PuO_2 powder into free-flowing UO_2 powder, avoiding the granulation step used in earlier processes.

Table 4. BN fuel irradiation

Reactor Type	Reactor	Years Operation	Assemblies	MOX Rods					Peak pellet burnup (GWd/MT)	Actual Programs
				BN* previous	BN* reference	BN MIMAS	CFCa	Others		
PWR	BR3	1963–87	129	512	1092	-	-	25	82	
		1984–87	24	-	224	178	-	10	42	PRIMO
	CAP	1985–87	2	-	48	-	40	-	24	
	CNA	1974–78	4	-	144	-	16	-	39	
	Beznau 1	1990-present	2	-	-	22 segments	-	2 segments	55	FIGARO/NOK PIE program
	BR2	1992–94	-	-	-	2	-	-	45–50	CALLISTO
BWR	Dodewaard	1971–87	7	84	-	-	-	-	47	
		1988–93	5	-	-	25 segments	-	15 segments	58	DOMO
	Garigliano	1970–84	51	204	1426	-	-	48	26	
	Oskarshamn 1	1974–79	3	-	51	-	-	-	19	
	Total		227	800	2985	225	56	100		

*See discussion in Section 4.1.2.

4.1.2.1 Garigliano Irradiations

Garigliano is an early-generation GE BWR fueled with standard 8 × 8 fuel bundles. BN produced 204 rods of MOX pellet fuel for the Garigliano core (Ref. 37). These rods, contained in four reload assemblies, were part of a larger group (total of 600 rods contained in 12 bundles) introduced into the core in 1968 (Ref. 37). The successful irradiation of these prototype bundles led eventually to commercial recycle of MOX in the Garigliano reactor in 1975.

4.1.2.2 Dodewaard Irradiations

A second demonstration of BN MOX fuel took place in the Dodewaard BWR. Like Garigliano, Dodewaard is an early GE BWR with a thermal output of 183 MW. A total of seven LTAs containing BN fuel were irradiated in Dodewaard from 1971 through 1987. The first two assemblies contained a total of 30 MOX rods; 27 rods containing pellet fuel and 3 rods containing VIPAC fuel. The VIPAC fuel was used in the three highest power rods in an effort to reduce the peak power in those rods through lower fuel density (Ref. 48). Both types contained 2.7% plutonium (2.5 wt % fissile) in natural uranium. These assemblies were irradiated at linear heat generation rates up to 440 W/cm (13.4 kW/ft) during Cycles 2–5 to assembly average burnups of 20 GWd/MT. PIE was performed on selected rods from these assemblies as part of the CEC plutonium investigations at SCK-CEN (Ref. 43).

Four additional MOX assemblies, containing a total of 56 MOX rods (again both VIPAC and pellet), were loaded in 1973. These assemblies, as with the first two, were of the island type. They contained 2.9% plutonium in natural uranium. After the first cycle of irradiation, sipping indicated a possible failure in one of the MOX assemblies. Two of the four were reloaded. One was irradiated three additional cycles, reaching an assembly average burnup of 28 GWd/MT. It subsequently underwent destructive examination in Petten. The other intact MOX bundle was irradiated four additional cycles to 33 GWd/MT assembly average burnup. After replacement of the leaking rod(s), the other two MOX assemblies were reintroduced to the core. They both operated for four additional cycles to assembly average burnups of 32 and 33 GWd/MT.

The seventh bundle, which contained 12 MOX rods (again VIPAC and pellet), was loaded at the beginning of Cycle 6 in 1974. It operated for a total of six cycles with two interruptions, with final discharge in 1987, reaching an assembly average burnup of 42 GWd/MT (Ref. 13).

4.1.2.3 Chooz A Irradiations

Chooz A is an early-design Westinghouse PWR located in France. It is often referred to in the relevant literature as the SENA (Société d'Énergie Nucléaire Franco-Belge des Ardennes) or CNA (Centrale Nucléaire des Ardennes) reactor. As part of a joint program between BN, the Commissariat à l'Énergie Atomique (CEA), and Reaktor Beteilingungsgesellschaft [RBG, now Reaktor-Brennelement Union GmbH (RBU)], four MOX demonstration assemblies were loaded in the reactor in 1974 (Ref. 49). The four island assemblies contained a total of 160 MOX rods containing 5% plutonium in natural uranium (Ref. 43). However, only 144 of these rods contained fuel produced by BN (Ref. 37). The remainder were produced by the CEA at Cadarache. The rods were all clad with 304 SS, as were all fuel rods in the CNA reactor. The

assemblies were irradiated for three cycles to an average discharge burnup of 27 GWd/MT. PIE was performed on the rods at both SCK-CEN and CEA/Saclay.

A fifth MOX assembly manufactured by FRAMATOME was irradiated in Chooz A simultaneously. These irradiations were meant to lead to commercial self-generated recycle in Chooz A. In fact, an order for 112 MOX assemblies, sufficient for three successive reloads, was placed at about the time the five MOX bundles were originally loaded into the reactor (Ref. 49). However, no mention of this subsequent commercial use has been found. According to Ref. 13, this failure to proceed with commercial reloads was a result of the political decision by the United States to defer reprocessing indefinitely, although other sources have attributed the temporary decline of European MOX research to the rise of breeder reactor programs.

4.1.2.4 Oskarshamn LTAs

Three LTAs containing MOX fuel produced by BN were irradiated in the Swedish Oskarshamn 1 reactor. This 1375-MWt Asea-Atom BWR is also referred to in BN literature as the OKG1 reactor, for the owning utility (OKG Aktiebolag). Three island assemblies containing a total of 51 MOX rods were irradiated from 1975 to 1979.

4.1.2.5 Test Reactor and Non-LWR Irradiations

In addition to the demonstrations performed in commercial reactors, BN conducted a number of irradiations in various test reactors including the High Flux Reactor (HFR) at Petten, the Halden BWR, and the Belgian Reactor 2 (BR2). The VENUS critical facility was used to validate calculational methods. BN provided some fuel to Atomic Energy of Canada, Limited (AECL) for irradiation in the NPD-2 reactor. The available data do not indicate whether this was prototype fuel for CANDU reactors, but the pellet dimensions suggest that this is the case. The preliminary results reported indicate that an assembly average burnup of 16 MWd/MT had been reached, with the irradiation continuing (Ref. 37).

4.1.3 Commercial BN MOX Experience

Based on the successful irradiation of the MOX demonstration assemblies, a full quarter-core reload of MOX island assemblies was introduced into the Garigliano core in 1975. This reload consisted of 46 assemblies, each of which contained 32 MOX rods. The MOX rods were manufactured by BN to GE specifications. The bundles remained in the core through 1981, to a peak burnup greater than 25 GWd/MT (Ref. 34). Details about the irradiation performance and/or PIE of this fuel have not been located; however, it is likely that GE has access to at least a portion of this information.

By the early 1980s, BN had accumulated an appreciable experience base with their "reference" fuel. However, the solubility of the "reference" fuel was unsatisfactory, as up to 5% of the contained plutonium could not be dissolved in nitric acid during the head-end reprocessing operations. The MIMAS process was therefore developed by BN. Tests on MIMAS fuel expanded BN's MOX experience rapidly. Development of the MIMAS process happened to coincide with Électricité de France's (EDF's) decision to pursue commercial MOX use. A joint company, COMMOX, was formed by COGEMA and BN to market MOX production from BN's

P0 plant, COGEMA's CFCa plant, and the planned MELOX plant (Ref. 50). As a result, a large portion of P0's production went to French PWRs including Saint-Laurent B1, Saint-Laurent B2, Gravelines 3 and 4, and Dampierre 1 and 2. These irradiations are discussed further in Section 4.4.

In addition to the Franco-Belgian COMMOX venture, BN has supplied MIMAS fuel to a number of German reactors including Unterweser, Grafenrheinfeld, Philippsburg, Brokdorf, and Gundremmingen. BN has also provided MOX fuel for the Swiss Beznau 1 reactor, described more fully in Section 4.5 below. A summary of BN's commercial MOX deliveries with MIMAS fuel is contained in Table 5 taken from Ref. 8.

Table 5. BN's MIMAS commercial MOX deliveries

Delivery Year	Reactor	MOX Reloads
1987	EDF	1
	CNA	2
1988	EDF	2
	Beznau 1	1
1989	EDF	4
	Beznau 1	1
1990	EDF	4
	Beznau 1	1
1991	EDF	4
1992	EDF	2
	Beznau 1	2
	Unterweser	1
1993	EDF	2
	Grafenrheinfeld	1
	Philippsburg	1
1994	EDF	4
	Brokdorf	1

According to BN, irradiation experience with their MOX fuel has been quite good, with only ten recorded rod failures, only one of which, a hydride failure, can be attributed to the fuel itself. These failures are described in Ref. 51:

> "1. A single end plug weld leak was observed in 1972 on a BR3 MOX fuel assembly after 1243 days at power and a peak pellet burnup of 59000 MWd/tonne HM. It was also a common type of defect in UO_2 fuel in these early days. This deficiency has been corrected by improving the welding technology and revising the welding specifications.
>
> "2. A single hydride failure occurred in 1973 in a Dodewaard assembly. It was also a common mode of failure of UO_2 BWR fuel at that time. A

revision of the pellet specification and of the fabrication technology has taken care of this problem.

"3. Coolant/cladding interaction failure was experienced in 1980 on six MOX rods in BR3 as a result of excessive crud deposition. This failure mode has been recognized to be due to inadequate primary water chemistry in conjunction with the average primary coolant temperature, which is lower than in modern PWRs, and to the surface heat flux level as a consequence of the irradiation conditions of those rods just below the maximum acceptable limits. It has no relation to the type of fuel inside the rod and disappeared with the adoption of a more restrictive permissible heat flux level in the core design criteria.

"4. Two failed MIMAS fuel rods were identified in 1990 in one MOX fuel assembly in Beznau-1. The two failed rods were located in two positions close to each other, and the investigations indicate a failure due to fretting by debris carried around by the coolant." [These failures are independent of the use of MOX, as similar failures were experienced in neighboring UO_2 assemblies.]

An interesting observation is that with the exception of the experimental BR3 reactor, no Belgian reactors were fueled with MOX until 1995. MOX fuel was introduced into two commercial Belgian reactors, Doel 3 and Tihange 2, in 1995.

4.2 ITALIAN MOX EXPERIENCE

The Italians undertook a substantial MOX development program in 1966. Two Italian organizations, Comitato Nazionale per l'Energia Nucleare (CNEN) and Ente Nazionale per l'Energia Elettrica (ENEL), pursued reprocessing and MOX recycle efforts simultaneously. Under this program, numerous test rod irradiations were performed across Europe. These early test irradiations, including the UO_2 tests used to baseline the MOX performance, are summarized in Table 6 (Ref. 52). The data obtained provided sufficient experience to undertake commercial MOX irradiations. ENEL operated two BWRs based on the GE NSSS, a PWR based on the Westinghouse NSSS, and a gas-cooled reactor during this period. Two of these reactors, the Garigliano BWR and the Trino PWR, were used for MOX development and demonstration.

The CNEN/ENEL program was one of the most comprehensive MOX research programs of its time. All facets of plutonium recycle were investigated. A pilot MOX fuel fabrication facility was constructed at the Cassacia Center in which research on pellet, powder VIPAC, and microsphere VIPAC fuel was conducted (Ref. 52, p. 364).

Table 6. CNEN/ENEL test reactor irradiation experiments

Test	Reactor	Number of Rods	Fuel	Linear Power (kW/ft)	Burnup (GWd/MT) (as of 4/72)
IFA-130	Halden	12	UO_2	16.0	1.5
IFA-131	Halden	12	UO_2	16.0	31.5
IFA-124	Halden	2	UO_2-PuO_2	13.0	0.5
IFA-132	Halden	2	UO_2	27.5	11
IFA-133	Halden	2	UO_2	27.5	11
IFA-136	Halden	8	UO_2	16.0	15
IFA-170	Halden	8	UO_2-PuO_2	16.0	18
IFA-178	Halden	8	UO_2	16.0	20
SK-1	R-2	42	UO_2	8–40	not complete
SK-2	R-2	30	UO_2-PuO_2	8–40	not complete
AP-1	Ågesta	12/4	UO_2/UO_2-PuO_2	16.0	11
P-22	Kahl	36	UO_2-PuO_2	13.0	7
P-41	Kahl	36	UO_2-PuO_2	13.0	not complete
HFR-Gd	HFR	12	UO_2/UO_2-PuO_2	16.0	not complete

The test fuel manufactured in the Cassacia pilot plant represents the only known experience with burnable poison in MOX fuel. An all-plutonium 6 x 6 assembly manufactured for irradiation in the Kahl BWR contained two burnable poison rods: one enriched UO_2 rod and one $(U,Pu)O_2$ rod. The only reference to this assembly indicates that a burnup of 3500 MWd/MT had been reached with irradiation continuing (Ref. 36). An additional test in CEA's SILOE reactor discussed in the same reference contained four "rods containing either 1% or 2% Gd_2O_3 in either enriched uranium-oxide or MOX matrices, totaling four types of rods. For each rod type, irradiations are performed up to three different levels to check the gadolinium evolution at different stages." (Ref. 36, p. 270)

4.2.1 Garigliano MOX Irradiations

Four fuel fabricators were contracted by ENEL to manufacture a total of 16 fuel bundles containing 860 MOX fuel rods for the Garigliano BWR (Ref. 53). The plutonium was recovered from spent fuel from the Latina gas reactor. BN and ALKEM jointly supplied four bundles, GE supplied four bundles, and the United Kingdom Atomic Energy Authority supplied eight bundles (Ref. 52, p.363). The bundles were designed to have the same performance parameters as the UO_2 assemblies that they replaced. Twelve of the sixteen bundles were inserted into the reactor in 1968. In 1970, two of these twelve were removed for inspection, and the remaining four (of the original 16 bundles) were loaded. The four GE fuel assemblies included 12 VIPAC rods, 24 hot-pressed pellet rods, and 60 cold-pressed and sintered rods. The GE fuel was irradiated to ~25 GWd/MT assembly average burnup at a design peak linear heat rate of 15 kW/ft (Ref. 29, p. XV-2). All of the 204 rods produced by BN and ALKEM contained pellet fuel.

In 1975, a full reload of MOX-island assemblies was introduced into the Garigliano core. This reload consisted of 46 assemblies, each of which contained 32 MOX rods. GE prepared the fuel design, and contracted with BN for the actual MOX fuel fabrication. The bundles remained in the core through 1981 and were irradiated to a peak burnup greater than 25 GWd/MT (Ref. 34). Details about the irradiation performance and/or PIE of this fuel have not been located.

4.2.2 Trino MOX Irradiations

Westinghouse manufactured eight LTAs for irradiation in the Trino reactor. Trino is a small (270 MWe) PWR with a Westinghouse-supplied NSSS. The MOX assemblies were irradiated from 1975 to 1978 to a peak pellet burnup of approximately 35 GWd/MT (Ref. 22).

4.3 German (Former Federal Republic of Germany) MOX Experience

MOX fuel development in Germany was conducted originally by Alpha Kemistry and Metallurgy GmbH (ALKEM). ALKEM was eventually purchased by Kraftwerk Union Aktiengesellschaft (KWU). Siemens Aktiengesellschaft eventually purchased KWU in 1988. Thus, references to MOX fuel development by the same group over a period of years can be attributed to either ALKEM, KWU, or Siemens. For simplicity, most of the work described in this section is attributed to ALKEM because of the historical nature of this summary.

ALKEM's original investigations into MOX fuel fabrication and utilization began in the 1960s. A single MOX fuel assembly was inserted into the Versuchsatomkraftwerk (VAK) Kahl BWR in 1966. Additional MOX insertions into VAK were made through the 1960s and early 1970s. ALKEM's first PWR MOX fuel, designed by KWU, was inserted into the Kernkraftwerk Obrigheim (KWO) in 1972. MOX fuel was also supplied by ALKEM for the Mehrzweckforschungsreaktor (MZFR), a prototype pressurized heavy water reactor. Commercial MOX use was expanded to German BWRs in 1974 with the insertion of 16 MOX assemblies into the Kernkraftwerk Gundremmingen. MOX utilization in these reactors slowly expanded during the 1970s and eventually expanded to additional German reactors.

ALKEM utilized their "Former Standard" pellet fabrication process for all the fuel fabricated through 1981. Table 7, reproduced from Ref. 54, summarizes ALKEM's experience with this early MOX fuel. Fuel performance with "Former Standard" fuel was adequate according to the manufacturer. Some early fuel failures were attributed to local hydriding that resulted from excessive moisture in the fuel. Improvements to the pellet-drying process subsequently eliminated this type of failure (Ref. 39). The generally good MOX fuel behavior is consistent with the experience of the Belgians from irradiation of fuel produced using their similar "Reference" process. However, as with the early Belgian fuel, the irradiated MOX was not sufficiently soluble in nitric acid to satisfy reprocessing requirements. The AUPuC and OCOM processes were, therefore, developed to address this insolubility.

Table 7. Siemens/KWU-ALKEM experience with "Former Standard" MOX fuel

Reactor (Type)	Year of First Insertion	Number of All-MOX Fuel Assemblies	Number of MOX Island Assemblies	Total Number of Rods	Maximum Assembly Burnup Achieved (GWd/MT)
VAK, Kahl (BWR)	1966	7	88	972	21
KWL, Lingen (BWR)	1970		1	15	26
KRB-A, Gundremmingen (BWR)	1974	64		2240	20
MZFR, Karlsruhe (PHWR)	1972	8		296	14
KWO, Obrigheim (PWR)	1972	33		5940	35
Total		112	89	9463	

ALKEM produced MOX fuel using the OCOM and AUPuC processes from 1981 to 1991. The 35-MT/yr pilot fabrication line was then shut down in anticipation of the opening of the 120-MT/yr second generation MOX plant that has since been abandoned. ALKEM's experience base with OCOM and AUPuC fuel is summarized in Table 8, taken from Ref. 54. The performance of the two fuels is quite similar, as one would expect from their very similar microstructures, and is basically equivalent to that of modern UO_2 fuel.

Table 8. Siemens/KWU-ALKEM experience with OCOM and AUPuC MOX fuel

Reactor (Type)	Year of First Insertion	Number of All-MOX Assemblies	Total Number of Rods	Fuel Type	Maximum Assembly Burnup Achieved (GWd/MT)
KWO, Obrigheim (PWR)	1981	29	5220	OCOM/AUPuC	35
GKN-1, Neckarwestheim (PWR)	1982	32	6560	OCOM	42
KKU, Unterweser (PWR)	1984	20	4720	OCOM/AUPuC	37
	1987	28	6496	OCOM/AUPuC	25
BZN-2, Beznau-2 (PWR)	1984	52	9308	OCOM/AUPuC	36
KKG/BAG, Grafenrheinfeld (PWR)	1985	16	3776	OCOM/AUPuC	34
		16	3712	OCOM/AUPuC	34
KKP-2, Philippsburg (PWR)	1988	12	2784	OCOM/AUPuC	18
KWG, Grohnde (PWR)	1988	20	4640	OCOM/AUPuC	27
KBR, Brokdorf (PWR)	1989	20	4640	OCOM/AUPuC	13
Total		245	51856		

Although Siemens no longer has MOX fabrication capabilities, MOX utilization continues in a number of German reactors. The MOX fuel is still fabricated to specifications provided by Siemens, but the fuel itself is fabricated by BN, COGEMA, and BNFL. None of these

fabricators produces OCOM or AUPuC fuel. In fact, MIMAS fuel supplied by COMMOX is currently loaded in several German reactors.

In addition to the standard irradiation tests described above, ALKEM performed PIEs on numerous irradiated MOX rods (Ref. 15). These examinations on thirteen "Former Standard" rods, six AUPuC rods, and four OCOM rods (as of 1987) indicate the performance similarities between MOX and UO_2 fuels. The dimensional changes as a result of irradiation (both densification and swelling) have been less with MOX. The fission gas release is higher, although the modern OCOM and AUPuC fuels have lower fission gas release rates than the earlier "Former Standard."

Siemens also performed ramp testing on MOX fuel in the KWO and HFR Petten reactors. The tests in general indicate that MOX fuel is actually superior to UO_2 fuel with respect to response to rapid power increases.

> "1. In KWO, 14 test rods with former standard MOX fuel, preirradiated at KWO to a burnup range of 9.0 to 21.8 MWd/kg (m), were ramp tested starting at 160 to 360 W/cm and leading to 270 to 420 W/cm. Thereafter, two of these ramp-tested rods were ramp tested a second time after a further preirradiation period at 165 to 230 W/cm and a burnup of 17 to 27 MWd/kg (M) to 260 W/cm.
>
> "2. In the High Flux Reactor (HFR), Petten, The Netherlands, 10 test rods with former standard MOX fuel, preirradiated at KWO to a burnup range of 9.3 to 32.1 MWd/kg (M), were ramp tested between 290 and 480 to 560 W/cm.
>
> "3. In the HFR Petten, three test rods with AUPuC-MOX fuel, preirradiated to ~35 MWd/kg (M), were ramped from 250 W/cm to 420 to 490 W/cm" (Ref. 15).

No defects resulted from these ramp tests. At the time of the publication (1987), additional ramp tests with both AUPuC and OCOM fuels were planned. Although Siemens' experience, like that of BN, is well documented in open literature, the company undoubtedly maintains a much more complete, albeit proprietary, data base.

4.4 French MOX Fuel Experience

The French nuclear industry is quite complex to the outside observer. A brief explanation of the relationships between the different companies is therefore useful for understanding the follow-on discussion. Development through the early 1980s was concentrated within the Commissariat à l'Énergie Atomique (CEA). As part of a 1983 reorganization, CEA-Industrie was formed as a holding company, which owns CEA's industrial interests. As part of this reorganization, COGEMA was created for fuel cycle activities. FRAMATOME is the French NSSS designer, originally as a Westinghouse licensee but independent since 1981. FRAMATOME and COGEMA jointly fabricate LEU fuel and fuel assembly hardware through their subsidiary

Société Franco-Belge de fabrication de combustibles (FBFC). Marketing of LEU fuel assemblies is also controlled through a joint subsidiary, FRAGEMA. COGEMA is solely responsible for the fabrication of MOX fuel, but design and fabrication of the fuel assembly hardware is performed by FRAMATOME and FBFC. COGEMA and BN jointly market the MOX fuel production from BN's P0 and from COGEMA's CFCa and MELOX MOX plants through COMMOX. EDF is the French national utility.

CEA and its successor COGEMA have been involved in MOX fuel research for many years. However, the early focus in France on plutonium utilization focused on in-situ utilization through extended burnup in their gas-cooled reactors. It was also planned that plutonium produced in these gas-cooled reactors would be utilized as fuel for fast reactors. Plutonium utilization in thermal reactors was investigated as a contingency. CEA collaborated with BN and RBG for insertion of four MOX LTAs into the CNA reactor in 1974. These four island assemblies contained a total of 160 MOX fuel rods, 16 of which were fabricated by CEA at Cadarache. This project was conducted under one of the 48 contracts among the Commission of the European Communities (CEC) and various research and industrial organizations. FRAMATOME supplied two full-MOX LTAs containing rods produced by CEA at Cadarache to CNA for irradiation starting in 1975.

Following this early investigation, French interest in thermal recycle dropped. The lack of French interest in thermal recycle was instrumental in stopping the renewal of the CEC program on plutonium recycle in LWRs. Plutonium fuel work in France continued, but was focused exclusively on fast-reactor applications.

By 1985, the expected fast-reactor fuel cycle had not materialized as expected. Simultaneous success with reprocessing plants had created a substantial and growing surplus of separated plutonium. EDF responded to this situation by announcing its decision to pursue MOX utilization in the 900 MWe series of PWRs. Because of the historical cooperation between the French fuel suppliers and BN through FBFC, the adoption of BN's MIMAS process for production of the necessary MOX fuel was straightforward. The first reload quantity of MOX fuel was delivered to St. Laurent B1 in 1987. Additional reloads have followed in many of EDF's other 900-MWe class reactors.

MOX fuel assemblies for the French reactors have been produced in the three facilities that supply fuel to COMMOX: P0, CFCa, and MELOX. Some of the early fuel produced in CFCa was not MIMAS, because the plant was only recently converted to this process. CFCa formerly utilized a comilling process known as COCA that was developed for fast-reactor fuel fabrication. No breakdown in the type of fuel utilized in EDF's reactors has been located, but the vast majority has clearly been MIMAS. The early fuel was fabricated with ex-AUC UO_2, but more recent fuel has been produced with TU2 UO_2.

French operational experience with MOX fuel has been reported more frequently in the last few years as postirradiation results from the early MOX fuel assemblies have become available. However, it is clear that EDF and COGEMA have relied extensively on BN's experience base. In addition to participation in many of BN's international programs, FRAMATOME, EDF, and CEA have conducted a joint analytical and experimental program to investigate MOX fuel performance (Ref. 55). The results of this study were consistent with BN's findings that

MIMAS MOX fuel performance is roughly equivalent to that of similar LEU fuel and adequate for the 1/3 core refueling strategy being utilized. Fission gas release late in fuel life was found to be the only topic warranting additional intensive study. Continued investigation under the FRAMATOME-EDF-CEA program focused on fuel performance under both steady-state and transient conditions (Ref. 56). The experimental results from the various experimental projects conducted under the auspices of the FRAMATOME-EDF-CEA program have been utilized to improve, extend, and validate the computational fuel performance codes. The studies have investigated fission gas release, thermal stability, irradiation-induced densification, pellet-clad mechanical interaction, and reactivity insertion accidents (Ref. 57).

The future plans for MOX utilization in France are quite ambitious. EDF plans to obtain licensing authority to load MOX fuel in any or all of the 28 existing 900 MWe class PWRs in France. The recent approval for MOX introduction and utilization in the four Chinon units brings the total number of French reactors that are licensed to burn MOX fuel to 20. This number is able to absorb the licensed production from MELOX, currently 100 MTHM/yr. An extension to MELOX, the MELOX West Fitting Building, is expected to be commissioned during 1999. The claimed full capacity of MELOX including this extension is 250 MTHM/yr.

4.5 Swiss MOX Experience

Nordostschweizerischen Kraftwerke (NOK) operates a total of five reactors including the two Westinghouse-designed Beznau units. Four MOX LTAs were loaded into Beznau-1 in 1978 (Ref. 22). Routine MOX utilization began in 1984 with the insertion of a MOX reload into Beznau-2. This fuel and subsequent reloads were fabricated by ALKEM using the OCOM and AUPuC processes (Ref. 54). MOX fuel was also utilized in Beznau-1, starting in 1988 with the insertion of a reload of MIMAS MOX fuel supplied by COMMOX (Ref. 51). One noteworthy aspect of this MIMAS fuel is that some of it is in the form of segmented rods; these rod segments have been utilized in several international programs conducted by BN. Beznau-1 has also irradiated SBR MOX LTAs fuel fabricated by BNFL in the MOX Demonstration Facility at Sellafield. NOK plans to continue MOX utilization in Beznau through the end of their operating lives. In addition, MOX fuel use was expanded to the Göesgen reactor in 1997 (Ref. 58).

4.6 United Kingdom MOX Experience

In the early 1960s, limited plutonium fuel fabrication capacity was installed in a laboratory at the UKAEA's Windscale complex (now Sellafield). This laboratory produced limited quantities of fuel for a variety of reactors including BR3 and Garigliano. The emphasis was on LMR fuel production, however, because of faith in the rapid implementation of an LMR fleet. The UKAEA did participate in the CEC program investigating plutonium recycling in thermal reactors. However, the UK's emphasis on LMR development helped prevent the extension of the CEC program when it came up for renewal in 1980.

Throughout the 1970s and early 1980s, various LMR fuel fabrication processes were researched at Windscale. High-density annular pellet fuel was chosen over low-density solid fuel, VIPAC, and SPHEREPAC. Additional fabrication capacity was required to supply the fuel required for

the planned irradiation program. Design and construction of a new facility for enhanced throughput (FACET) proceeded from the mid-1980s.

FACET was ready for commissioning in 1988. Changes in energy policy in the United Kingdom were reflected through reduced LMR budgets. With this change, the need for FACET evaporated. The facility was never fully commissioned. Concurrent developments in the rest of Europe resulted in the decision by the UKAEA and BNFL to convert FACET to the MDF. This conversion was completed in 1993.

One of the key aspects of MDF's development was the adaptation of the lessons learned in LMR fuel development to LWR MOX applications. The wet binder granulation process was abandoned in favor of the SBR because of the simplifications it entailed (Ref. 59).

Although a detailed accounting for MDF's production has not been found, it is known that both LTAs and reload assemblies have been supplied to Beznau from MDF. Fuel has also been produced for irradiation in test reactors as part of the qualification of SBR fuel. Furthermore, the extensive LMR fuel development that led to development of the SBR provides additional irradiation performance information.

A large second-generation SBR fabrication plant is under construction at Sellafield. This 120 MT/yr SMP is based on the technology proven in MDF. SMP is nearly complete and is awaiting authorization to start uranium commissioning.

Because BNFL is a relative newcomer to the LWR MOX market, their experience base with SBR fuel is limited compared to that of COMMOX and Siemens. Nevertheless, SBR test fuel has been successfully irradiated, and the associated irradiation results reported (Ref. 60). Additional information will become available when the Beznau-1 LTAs are subjected to PIE over the next 1–2 years (Ref. 61). One of the four MDF LTAs supplied to Beznau-1 included a group of 8 specially characterized rods, which will be the focus of the PIE.

BNFL has claimed that the SBR process produces a more homogeneous fuel than the master mix processes. The fabrication experience obtained in MDF has demonstrated that good homogeneity is obtained (Ref. 61). Electron probe microanalysis has been utilized to confirm the results of alpha autoradiography. The highest reported plutonium concentration in the plutonium-rich zones is nearly that of master mix, about 30% using a 1 μm spot size for analysis. The plutonium-rich zones in SBR fuel are typically less than 30 μm. This homogeneity is roughly equivalent to that found in master-mix fuel. MIMAS fuel typically contains plutonium-rich zones that are 50 μm or less in equivalent diameter, and these plutonium-rich zones consist of master-mix containing 30%–40% plutonium. The issue of whether SBR or master-mix processes produce better homogeneity cannot be resolved until statistically significant quantities of SBR fuel have been irradiated and subjected to PIE. The fact remains that both the SBR and master-mix processes produce fuel containing no plutonium particles large enough to cause problems during power transients.

Although BNFL and the UKAEA have extensive LMR fuel experience and experience with thermal plutonium recycle, plutonium is not utilized in Sizewell B, the only PWR in the United Kingdom.

4.7 JAPANESE MOX EXPERIENCE

Plutonium fuel development work in Japan has historically been performed under the auspices of the Power Reactor and Nuclear Fuel Development Corporation (PNC), now the Japan Nuclear Cycle Development Institute (JNC). A small laboratory, the Plutonium Fuel Development Facility (PFDF), was constructed in 1965 by PNC at the Tokai Works to study the basic properties of MOX fuel, to develop fabrication processes, and to produce fuel for irradiation testing. Both LMR and thermal MOX fuels were fabricated in this laboratory. Several fabrication processes were investigated in the PFDF including comilling, master-mix, coprecipitation, VIPAC, and SPHEREPAC. Fuel pin tests with these fuels were conducted in both the GETR and the Japan Research Reactor. Fuel assembly tests were conducted in Saxton, the Halden BWR, Mihama-1, and Tsuruga-1.

In cooperation with the Japan Atomic Energy Research Institute (JAERI), PNC fabricated two 3 x 3 assemblies for irradiation in the Halden BWR. One of the assemblies contained SPHEREPAC fuel and the other contained comilled fuel (Ref. 62). This test demonstrated no safety-significant differences between the fuel behaviors. PNC also fabricated one complete fuel assembly plus 4 additional fuel rods for irradiation in Saxton Core III. All of the Saxton fuel rods contained comilled pellet fuel. PNC also fabricated 2 rods incorporated into 2 of the 4 MOX fuel LTAs fabricated by Westinghouse and inserted into Mihama-1 (Ref. 63). PNC produced 48 MOX fuel rods for incorporation into 2 island-type MOX fuel LTAs inserted into Tsuruga-1 under a joint program between PNC and Japan Atomic Power Company (JAPCO). Additional MOX fuel rods were produced in PFDF for fast reactor irradiation tests and for critical experiments.

A larger facility, the Plutonium Fuel Fabrication Facility (PFFF), was completed in 1972. This facility was constructed near the PFDF at Tokai Works to produce larger quantities of fuel for both fast and thermal reactors. Two separate lines were installed: one for production of Advanced Thermal Reactor (ATR) fuel and a second for production of LMR fuel. The ATR line was utilized to produce over 9 MT of MOX fuel for use in the Deuterium Critical Assembly (DCA). It has since been utilized to produce MOX fuel for the Fugen ATR. The LMR line has produced test fuel for irradiation in test and prototype LMRs including GETR, Japan Materials Test Reactor, Rapsodie, Joyo, and Dounreay (Ref. 64).

In 1988, the Plutonium Fuel Production Facility (PFPF) was completed for supplying Monju fuel. PFPF is a large pilot facility complete with a pusher furnace. The plant is highly automated and experienced some operational difficulties during early operation as a result. These problems have been resolved, but Monju's sodium leak in December 1995 and the political fallout resulting from that leak have drastically reduced the need for fuel from PFPF.

Plutonium utilization policy in Japan is currently under governmental review. In February 1997, the Japanese Cabinet approved a nuclear policy on "Utilization of Plutonium in LWRs" that called for the introduction of MOX fuel into a single BWR and PWR during 1999. The subsequent accident in the Bituminization Facility in PNC's Tokai reprocessing plant resulted in a postponement in this schedule.

Tokyo Electric Power Co. and Kansai Electric Power Co. originally planned to load a BWR and PWR, respectively, during 1999 and to add one additional unit each during 2000. Longer term policy is for all utilities to implement MOX utilization by 2010. The political uncertainty resulting from the Monju leak and the Tokai fire will likely delay this implementation schedule. Japanese utilities currently store more than 10 MT of separated plutonium at Sellafield and La Hague. The MELOX extension and SMP are both being constructed in part to produce MOX fuel from this plutonium for utilization in Japanese reactors. Most plans for MOX utilization in Japanese reactors call for 30% MOX in the cores, but plans also exist for construction of one or more GE Advanced BWRs at Ohma that will burn full MOX cores.

4.8 MOX Experience in the Russian Federation and Former Soviet Union

Plutonium utilization in the Russian Federation has been focused on LMR applications. Plutonium fuel has been tested in four LMRs: the BR-10 at Obninsk, the BOR-60 at Dimitrovgrad, the BN-350 at Aktau, and the BN-600 at Beloyarsk (Ref. 65). Four fabrication technologies have been investigated: comilling, VIPAC, sol-gel, and coprecipitation. The total quantity of fuel produced is over 1 MTHM. Pellet technology has been developed by researchers at the A. A. Bochvar All-Russian Research Institute of Inorganic Materials (VNIINM) and implemented at the large laboratory scale in the Paket laboratory at the Production Association Mayak. VIPAC technology and the associated pyroelectrochemical reprocessing technology used to produce high density VIPAC feed have been developed by researchers at the All-Russian Research Institute of Atomic Reactors (NIIAR). The VIPAC technology has been demonstrated extensively in the BOR-60 LMR at NIIAR.

MOX fuel has never been utilized in Russian PWRs. Less than 10 rods containing MOX test fuel have been irradiated in the Materials Irradiation Reactor (MIR) at NIIAR. The fabrication processes utilized have been taken from LMR development work.

Ambitious plans for implementation of a plutonium economy have been delayed and modified as a result of economic difficulties. The second generation reprocessing plant at Krasnoyarsk (known as RT-2), capable of reprocessing PWR fuel, has not progressed beyond the conceptual design stage. Limited construction work has been performed, but the design has never even been finalized. Some recent reports by officials of the Russian Ministry of Atomic Energy (MINATOM) indicate that RT-2 has been canceled.

The existing reprocessing plant at Chelyabinsk has separated more than 30 MT of plutonium. The large MOX fuel fabrication plant planned to convert this plutonium into MOX fuel has only been partially completed (<20%). No credible schedule for completion of this Complex-300 plant has been located. Therefore, MOX fabrication capacity in Russia is limited to the existing laboratories at PO Mayak and at NIIAR. Neither lab is equipped for production of MOX fuel meeting PWR specifications.

5. SUMMARY AND CONCLUSIONS

The worldwide MOX experience constitutes a vast body of evidence suggesting that irradiation of surplus WG plutonium as MOX fuel in commercial nuclear reactors is a technically viable disposition option.

The most important goal of the effort documented in this report is to determine what is known worldwide about MOX fuel, and what additional information is needed to irradiate the excess weapons plutonium safely in domestic commercial reactors.

5.1 SUMMARY OF FINDINGS

MOX fuel has been investigated for almost forty years in the United States and abroad. Domestic MOX fuel development, by 1975, had progressed to the commercial demonstration stage. The plutonium used in much of the domestic research and development had a high fissile content, similar to that of surplus WG plutonium. Commercial use of MOX fuel in the United States was delayed by completion of the *Final Generic Environmental Statement on the Use of Recycle Plutonium in MOX Fuel in Light Water Cooled Reactors*, commonly known as GESMO, and the availability of plutonium from commercial reprocessing plants. Domestic MOX research ended by 1980 as a result of President Carter's Executive Order on nonproliferation that indefinitely postponed commercial reprocessing. At the time, overseas MOX research for the most part lagged behind domestic efforts. In the intervening years, however, continued development and commercial utilization overseas has eclipsed the domestic experience.

Overseas development of MOX did not end in the 1970s as hoped by domestic policy makers. However, it received less attention than did the utilization of plutonium fuels in LMRs. Several European countries researched MOX during this period, but Belgium and Germany continued appreciable MOX programs despite the expected emergence of a plutonium-fueled LMR economy. The Belgian and German programs led to the development and commercial utilization of state-of-the-art MOX fuels by the mid-1980s.

Later in the 1980s, as it became apparent that the expected LMR deployment schedule would either be delayed or canceled, France and then the United Kingdom renewed their respective MOX programs. The French licensed BN's MIMAS process and planned for rapid expansion of their utilization of MOX fabricated by this process. In the United Kingdom, the UKAEA and BNFL combined their efforts to develop the SBR process for MOX manufacture.

Because of various developments and programs in Europe, commercial MOX use is a reality in a number of European countries. The experience base resulting from such commercial use dwarfs the previous U.S. MOX experience in terms of both breadth and depth. The gap between domestic and overseas knowledge continues to grow.

In addition to the European efforts, the Japanese have actively pursued MOX continuously since the 1960s. One of the latest developments in the Japanese MOX program is the plan to build

new Advanced Boiling Water Reactors fueled entirely with MOX. A Japanese MOX fuel manufacturing plant is also planned.

It is important to remember, when reviewing the worldwide experience, that the goals and requirements of the FMDP are different from those driving the commercial MOX programs throughout the world. Commercial MOX programs are meant to close the nuclear fuel cycle by providing economical fuel from reprocessed nuclear materials. In the commercial context, plutonium is a valuable resource, not a troublesome legacy. The goal of commercial MOX programs is therefore to maximize the energy obtained from it.

In the FMDP, plutonium disposition—not economic energy production—is the goal. Irradiation is meant to make the plutonium self-protecting through the generation of an intense radiation field. Because the goals of the FMDP differ from those of commercial plutonium recycle, optimization results in distinct approaches. For example, in existing commercial MOX programs, MOX fuel is limited to a fraction of the core load—typically 30% to 50%. In the FMDP, because high-plutonium throughput rate is one of the most important requirements, full-core loads of MOX have been proposed. If the FMDP requirements fall outside that covered by existing data, the merits and costs of required development programs will have to be weighed against alternatives that closely mimic existing commercial MOX programs. These issues are more fully explored in the FMDP report on fuel qualification issues (Ref. 66).

5.2 CONCLUSIONS

One of the most important lessons learned during this review is that MOX fuel performance information is specific to a particular fuel fabrication process. Thus, to apply one of the existing experience bases to licensing efforts, the corresponding production process would have to be utilized for manufacturing the mission fuel. For a number of production processes, the existing performance data are sufficient to justify overseas commercial utilization of the MOX fuel. This is the case with the Belgian (and now French) MIMAS fuel, the German OCOM and AUPuC fuel, and BNFL's SBR fuel. Although licensing regimes overseas are not exactly compatible with NRC regulations, it is possible that if the data are sufficient for licensing in these European countries, they are sufficient to support the majority of NRC licensing concerns.

Another important conclusion drawn from this work is that development (and qualification) of a new domestic MOX manufacturing process would likely entail a lengthy and expensive program. Without assistance from one of the overseas MOX manufacturers, many years and hundreds of millions of dollars would likely be required to recover the previous experience base, adapt the production procedures to meet modern requirements (in terms of worker safety, environmental compliance, and product quality), and satisfy both licensing and end-user requirements.

Full MOX usage in a reactor has never been demonstrated. In cores limited to less than 50% MOX, the excess reactivity can be compensated through alternative means, including incorporation of burnable poison into the UO_2 rods. No technical impediments to incorporation of burnable poison in MOX have been found, and indeed some preliminary studies of such incorporation have been performed overseas. However, the fuel development program necessary to include burnable poison in MOX would likely require many years and tens of millions of

dollars. Foreign research (either ongoing or planned for the near term) could help accelerate the U.S. program, if cooperation were to be arranged.

Gallium is present in weapons plutonium and will have to be dealt with. Gallium removal from the MOX is desirable in terms of overall fuel irradiation performance because gallium behavior under irradiation is unknown. ORNL, LANL, and INEEL are currently investigating the array of issues associated with gallium.

Americium impurities, caused by decay of ^{241}Pu, have also been identified as a potential source of irradiation behavior difficulties. However, upon closer inspection, it was determined that americium is of more concern during fuel fabrication than during irradiation. Commercial MOX often contains much more americium than does weapons plutonium. Finally, research performed under this program suggests that the americium ceases to represent an exposure problem once it is blended with UO_2.

Two other potential concerns, the proposed use of higher plutonium concentrations than those used in commercial MOX and the use of depleted rather than natural uranium as the diluent in MOX, have been identified. Plutonium concentrations in the range proposed for the FMDP mission will almost certainly be within the span of the foreign experience base. Also, depleted uranium has been used previously as the diluent in MOX and is not expected to present any difficulties for the FMDP. In the commercial arena, the choice of diluent material has been driven primarily by availability (and cost) of the material. Thus, these two issues are not of major significance to the FMDP. The concerns identified do not present insurmountable obstacles to the use of WG plutonium in LWRs.

The current direction of the FMDP program is (1) rapid initiation of plutonium disposition, (2) low technical risk associated with initial plutonium disposition, (3) decoupling of fuel development success from mission success, (4) maximum use of commercial technology, and (5) gradual movement from known parameter space (partial core loads/no burnable poison) into a more desirable range (full core loads/burnable poison) or the use of more reactors. A detailed fuel development program plan may be developed when the necessary decisions regarding program goals and fuel performance requirements have been made by the DOE.

In conclusion, the existing MOX fuel experience base suggests that disposition of excess weapons plutonium through irradiation as MOX in LWRs is a technically viable option. This conclusion was reached through a review of the open foreign literature and a categorization and reorganization of the domestic experience base. Most of the detailed irradiation performance data remains proprietary and is unavailable. Based on the available information, it appears that adoption of foreign fuel technology and MOX fuel use patterns from one of the successful MOX fuel vendors will minimize the risks to the overall mission.

REFERENCES

1. Nuclear Regulatory Commission's Order of December 23, 1977, concerning its proceedings on the Generic Environmental Statement on Mixed Oxide Fuel (GESMO) and matters related to reprocessing and the recycling of uranium and plutonium in MOX, Federal Register Vol. 42, No. 251 (December 30, 1977), 65334.

2. H. Stehle, N. Eickelpasch, H. Schenk, et al., "Experience With Plutonium Recycle Fuel For Large Light-Water Reactors in the Federal Republic of Germany." *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2–13 May 1977. IAEA, Vienna, 1977.

3. M. Benedict, T. H. Pigford, and H. W. Levi, *Nuclear Chemical Engineering*, McGraw-Hill, 1981.

4. F. D. Fisher, *Reactivity of Plutonium Dioxide Prepared by Direct Calcination of Nitrate Solutions*, U.S. AEC Report HW-77846, Hanford Atomic Products Operation, General Electric Co., Oct. 15, 1963.

5. O. J. Wick (ed.), *Plutonium Handbook, A Guide to the Technology*, American Nuclear Society, 1980.

6. M. Koizumi et al., "Development of a Process for the Co-conversion of Pu-U Nitrated Mixed Solution to Mixed-Oxide Powder Using a Microwave Heating Method," *Nucl. Tech.*, **61**, 55 (April 1983).

7. M. K. Jones and W. J. Jenkins. "Slurry Calcination Process for Conversion of Aqueous Uranium and Plutonium to a Mixed-Oxide Powder." *Proc. ANS Top. Mtg. Fuel Cycles for the 80's*, Gatlinburg, Tennessee, 29 Sep.–2 Oct. 1980. American Nuclear Society, La Grange Park, 1980.

8. V. Vanderborck, D. Haas, P. Deramaix, and A. Vandergheynst. "Commercial MOX Fuel Fabrication and Irradiation Experience in Belgium." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16–19, 1994. Kluwer, 1995.

9. C. B. Kincaid, E. A. Aitken, I. N. Taylor, D. C. Wadekamper, and J. J. Zimmer, COPRECAL: Co-Conversion of Pu, U Mixed Nitrate to Mixed Oxide. GEFR-SP 168, General Electric Co., Sunnyvale, California, October 19, 1979.

10. V. Schneider, F. Herrman, and W. G. Druckenbrodt, "The AUPuC-Process: A Co-precipitation Process with Good Product Homogeneity to the Full Scale of Plutonium Concentration," *Trans. Am. Nucl. Soc.*, **31**, 176 (1979).

11. R. M. Horgos and M. L. Masch. *Remote Mixed Oxide Fabrication Facility Development Volume 1: Reference Facility, WCAP-9471*. Westinghouse Electric Corp., Pittsburgh, May 1979.

12. D. Haas, A. Vandergheynst, J. Van Vliet, R. Lorenzelli, and J. Nigon, "Mixed-Oxide Fuel Fabrication Technology and Experience at the BN and CFCa Plants and Further Developments for the MELOX Plant," *Nuclear Technology*, **106**, April 1994.

13. H. Bairiot, and P. Deramaix, "MOX Fuel Development: Yesterday, Today, and Tomorrow," *Journal of Nuclear Materials*, 188 (1992).

14. J. van Vliet, E. Pelckmans, N. Mostin, R. Lorenzelli, M. Moulard, and C. Mares, "Industrial Qualification of TU2 Powder Use for MOX Fuel Fabrication," *Nuclear Europe Worldscan*, **11–12** (1994).

15. H. Roepenack, F. U. Schlemmer, and G. J. Schlosser, "Development of Thermal Plutonium Recycling," *Nuclear Technology*, **77**, May 1987.

16. G. Macdonald, "The MOX Demonstration Facility - the Stepping Stone to Commercial MOX Production," *Nuclear Energy*, Vol. 33, Num. 3, June 1994.

17. BNFL, *MOX Technology in BNFL*, undated BNFL marketing brochure.

18. U.S. Atomic Energy Commission, *Draft Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors*, WASH-1327, August 1974.

19. U.S. Nuclear Regulatory Commission, *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors*, NUREG-0002, August 1976.

20. M. D. Freshley, "Irradiation Behavior of Plutonium Oxide Fuels," *Plutonium Handbook*, American Nuclear Society, 1980.

21. M. D. Freshley, "Mixed-Oxide Fuel Irradiations in the Plutonium Recycle Test Reactor," *Nuclear Technology*, **15**, August 1972.

22. Boltax, "Mixed Oxide Fuel for a LWR Plutonium Disposition Reactor (PDR600): A Review of Fuel Technology," *Plutonium Disposition Study*, DOE/SF19683–3.

23. Westinghouse Electric Corporation. "Saxton Plutonium Program," WCAP-3385 SERIES (Reports spanning series 1–57), (1965–1974.)

24. Westinghouse Electric Corporation. "EEI-Westinghouse Plutonium Recycle Demonstration Program," WCAP-4167 SERIES (Reports spanning series 1–7), (1970–1974)

25. M. D. Freshley et al., *EPRI Plutonia Fuel Study (RP396) - Final Summary Report*, EPRI NP-637, 1977.

26. Letter from J. J. Shea, Nuclear Regulatory Commission, to Rochester Gas & Electric Company, "Docket No. 50–244, Summary of Meeting Held on November 1, 1979."

27. Letter from D. Ziemann, Nuclear Regulatory Commission, to L. D. White, Jr., RG&E, "Docket No. 50–244, Safety Evaluation and Notice of Issuance for Four MOX Assemblies."

28. L. Kornblith Jr. et al., "Vallecitos Boiling Water Reactor," Presentation to the 1958 Nuclear Congress, Chicago, Ill., March 18, 1958.

29. Lee Jr., "Quad Cities-1 Nuclear Power Station First Reload Supplement for Assemblies Containing 48 EEI-GE Mixed Oxide Fuel Rods," Letter to J. F. O'Leary, U.S. AEC, dated June 15, 1973.

30. J. H. Hughes, "Request for Amendment of Operating License DPR-2, as amended, Dresden Nuclear Power Station, including Exhibit I," Letter to P. A. Morris, U.S. AEC, dated January 9, 1967.

31. General Electric, *Plutonium Utilization in Boiling Water Reactors Phase II: Post-Irradiation Examination of Four Fuel Rods After One Cycle of Operation in Big Rock Point Reactor*, NEDC-10387, July 1971.

32. G. J. Walke, "Request for Change to the Technical Specifications License No. DPR-6," Letter to P. A. Morris, U.S. AEC, dated Dec. 22, 1969.

33. General Electric Company, "Quad Cities Nuclear and Fuel Performance Measurement," EPRI NP-3568, Electric Power Research Institute, July 1984.

34. General Electric, *Study of Pu Consumption in Advanced Light Water Reactors: Evaluation of GE Advanced Boiling Water Reactor Plants*, NEDO-32292, May 13, 1993.

35. General Electric, "General Electric Test Reactor," GEZ-4128A. Marketing brochure for GETR irradiation capabilities, 1966.

36. Mangiagalli, F. Cicognani, F. Pistella, et al. "Plutonium Utilization in Fast-Breeder and Light-Water Reactors in Italy." *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2–13 May 1977. IAEA, Vienna, 1977.

37. H. Bairiot, M. Gaube, and B. Van Outryve D'Ydewalle, "Irradiation Behavior of LWR Plutonium Fuel Rods," *Nuclear Energy Maturity: Proceedings of the European Nuclear Conference*, Paris, April 21–25, 1975.

38. Combustion Engineering, *DOE Plutonium Disposition Study Screening Study for Evaluation of the Potential for System 80+ to Consume Excess Plutonium*, April 30, 1994.

39. H. Markl and H. Stehle, "KWU Plutonium Recycle Fuel and Core Design, Performance, and Experience," *Proceedings Topical Meeting on The Plutonium Fuel Cycle*, ANS, Bal Harbour, Florida, May 2–4, 1977.

40. H. E. Bliss, "Proposed Change No. 22 to Appendix A, DPR-2, to permit operation of Cycle 8," Letter to P. A. Morris, U.S. AEC, dated January 18, 1971.

41. W. J. Ross and E. M. Benson, "Babcock and Wilcox Plutonium Fuel Fabrication Experience," *Proceedings Topical Meeting on The Plutonium Fuel Cycle*, ANS, Bal Harbour, Florida, May 2–4, 1977.

42. M. D. Freshley, D. W. Brite, J. L. Daniel, and P. E. Hart, "Irradiation-Induced Densification of UO_2 Pellet Fuel," *Journal of Nuclear Materials*, 62:138–166, 1976.

43. M. Lippens and N. Cadelli. "A European Investigation of MOX Fuel Performance in LWRs." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17–20 April 1988. ANS, La Grange Park, 1988.

44. P. D. Dresser (ed.), *Nuclear Power Plants Worldwide*, Gale Research, 1993.

45. Haas and J. Van Vliet, "BN MOX Fuel Fabrication," *Nuclear Europe Worldscan*, Vol. XII, No. 5–6, May/June 1992.

46. J. Storrer A and D. H. Locke, "High Burnup Irradiation Experience in Vulcain," *Nuclear Engineering International*, Vol. 15, No. 165, Feb. 1970.

47. Y. Vanderborck, D. Haas, P. Deramaix, A. Vandergheynst, "MOX Fuel Fabrication and Irradiation Experience in Belgium," Proc. Second Annual International Policy Forum: Management and Disposition of Nuclear Weapons Materials, Lans downe, VA (1995).

48. H. Bairiot, P. Deramaix, C. Vandenburg, et al. "Plutonium Assemblies in Reload 1 of the Dodewaard Reactor." *Nucl. Tech.*, **33** (April 1977) 184.

49. J. Basselier, J. Debrue, L. Leenders, and H. Van den Broeck. "Experimental Data as Reference For Plutonium Recycling in the SENA Reactor." *Nucl. Energy Maturity, Proc. European Nuclear Conf.*, Paris, France, April 21–25, 1975.

50. G. LeBastard, "Recycling and MOX Fabrication: French and Belgian Achievements," Proc. of 9th Pacific Basin Nuclear Conference, Sydney (1994).

51. P. Deramaix, D. Haas, and J. Van de Velde, "In-Pile Performance of Mixed-Oxide Fuel with Particular Emphasis on MIMAS Fuel," *Nuclear Technology*, Vol. 102, April 1993.

52. G. Schileo, "A Review of the Italian Activities for the Thermal Recycling of Plutonium," Nucl. Tech., **16**, November 1972.

53. A. Chamberlain and C. Melches, "Review of the Status of and Assessment of the Prospects for the Establishment of Plutonium Recycle in the Thermal Reactors in the FORATOM Countries," Proc. of International Conference on Nuclear Power and Its Fuel Cycle, Salzburg, Austria, IAEA, May 1977.

54. W. Goll, H. Fuchs, R. Manzel, and F. U. Schlemmer, "Irradiation Behavior of UO_2/PuO_2 Fuel in Light Water Reactors," *Nuclear Technology*, Vol. 102, April 1993.

55. P. Blanpain, X. Thibault, and M. Trotabas, "MOX Fuel Experience in French Power Plants," *Proc. 1994 International Topical Meeting on Light Water Reactor Fuel Performance*, West Palm Beach, Florida, ANS (1994).

56. P. Blanpain, X. Thibault, and J.-P. Pagès, "Recent Results from the In Reactor MOX Fuel Performance in France and Improvement Program," *Proc. 1997 International Topical Meeting on Light Water Reactor Fuel Performance*, Portland, OR, ANS (1997).

57. L. Caillot, et. al, "Analytical Studies of the Behavior of MOX Fuel," *Proc. 1997 International Topical Meeting on Light Water Reactor Fuel Performance*, Portland, OR, ANS (1997).

58. H. Bairiot, "MOX Fuel: the Accomplishment and the Future," TOPFUEL'97, BNES (1997).

59 H. MacLeod and G. Yates, "Development of Mixed-Oxide Fuel Manufacture in the United Kingdom and the Influence of Fuel Characteristics on Irradiation Performance," *Nuclear Technology*, **102** (1993).

60. J. Mullen, C. Brown, I. D. Palmer, and P. Morris, "Performance of SBR MOX Fuel in the Callisto Experiment," TOPFUEL'97, BNES (1997).

61. J. Edwards, R. D. Grimoldby, S. J. Marshall, and R. W. Stratton, "BNFL Supply of MOX Fuel Assemblies to the Beznau-1 PWR of NOK," *Proc. of IAEA Technical Committee Meeting on Recycling of Plutonium and Uranium in Water Reactor Fuel*, Newby Bridge, Windermere, U.K. (1995).

62. H. Tsuruga, S. Matsuura, T. Suzaki, and R. Yumoto, "Non-destructive Gamma-Ray Spectrometry on Irradiated Mixed Oxide Fuel Rods," *Proc. of a Panel on Plutonium Utilization in Thermal Power Reactors*, Karlsruhe, IAEA-175, IAEA (1975).

63. T. Amasuma, M. Akutsu, and M. Koisumi, "Review on Plutonium Recycling Experiments and Program in Japan," *Proc. of a Panel on Plutonium Utilization in Thermal Power Reactors*, Karlsruhe, IAEA-175, IAEA (1975).

64. T. Muto, M. Koizumi, K. Naruki, T. Itaki, and T. Amanuma, "Status of the Development of Mixed-Oxide Fuels in the Power Reactor and Nuclear Fuel Development Corporation (PNC)," Proc. *of an International Conference on Nuclear Power and Its Fuel Cycle*, Salzburg, IAEA (1977).

65. N. N. Egorov et al., "Management of Plutonium in Russia," *Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Kluwer (1995).

66. B. S. Cowell, G. L. Copeland, D. L. Moses, Fuel Qualification Issues and Strategies for Reactor-Based Surplus Plutonium Disposition, ORNL/TM-13405, August 1997.

APPENDIX A

U.S. MOX EXPERIENCE TABLE

Page intentionally blank

APPENDIX A. U.S. MOX EXPERIENCE TABLE

This appendix provides a map of the U.S. domestic PWR and BWR MOX experience. The experience has been organized by first categorizing the reactor in which the MOX was irradiated. For each plant, the experience was diagrammed for each major campaign conducted by the fuel fabricator or sponsor of the research. This is followed by irradiation characteristics and a description of the examinations that were conducted. The column titles are generally self-explanatory; however, the considerations and hints below may be helpful to the user. The reader will notice several columns that have little or no data in them (such as "Beyond Design Basis Testing"). This column was set up from the beginning to reserve a slot for information. However, the MOX rods in the reactor campaigns were not subjected to transient testing in the reactor used for steady-state irradiation. It is also noteworthy that the current Advanced Test Reactor irradiations on MOX fuel are not mentioned. As of this writing, the first of these MOX rods (which utilized WG plutonium) had just come out of the reactor and were, therefore, not mapped. Any future updates of the data base should include these rods as information becomes available. Two tables and a corresponding reference list are presented. One table is used for PWRs and another for BWRs. The following sheets comprising these tables can be taped together to provide a collective "map" of the U.S. MOX experience. ORNL maintains the electronic version of this data base.

ITEM: In the category marked "ITEM," a number was assigned to each plant arbitrarily. This number is followed by a decimal and then a "batch" number. The definition of a batch number is somewhat arbitrary. However, it was used to refer to a group of rods or assemblies that went into a reactor at the same time, or it could also be thought of as a testing campaign number. The item number within that campaign is then sequentially assigned.

The "ITEM DESCRIPTION" column is important. This field tells the user whether the input for the entire row is for a Batch of MOX Assemblies (B), a single assembly or an arbitrary grouping of rods (A), an individual rod (R), or an irradiation item summary (IS). This field governs the interpretation of the data columns all along the row.

The term "IS" in the ITEM DESCRIPTION column refers to a summary of a campaign that was previously mapped. This merely offers a means to provide the reader with an overview of previously presented material. As an example, if one wants to total the number of MOX rods, then the IS rods should be taken out, since the information would otherwise be counted twice. The use of the IS option is demonstrated in the Quad Cities irradiation merely to inform the reader that core-wide gamma scans were conducted on both MOX assemblies and UO_2 assemblies.

REACTOR SPECIFICS: The column titles are self-explanatory. Probably the most important column to note is the "No. of MOX Assys in this ITEM."

ITEM ASSY MOX DESIGN AND FUEL ISOTOPICS The column titles are self-explanatory. The idea is to give the reader a picture of the type of assembly design and fuel rod characteristics that were used in this irradiation.

FABRICATION ASPECTS: An attempt is made to describe briefly the plutonium and uranium oxide powder process and the pellet fabrication techniques used for the fuel.

MAX LHGR: This is the maximum linear heat generation rate for the rod or assembly that was found in the literature. The literature was searched for the maximum value that the rod or assembly was thought to have seen. In the creation of the table, some interpretation and selection of the literature values had to be conducted, and, sometimes, a value was judged simply to be "representative" or close to what the specific maximum heat generation rate was. Oftentimes, the authors in the literature will quote vague or somewhat nondescript values.

FIRST SET/SECOND SET MEASUREMENTS: In mapping an irradiation, up to two total examinations (at a specific burnup) can be used to describe the examinations. There are several options for the description. One example might be the case where assemblies have MOX rods that are pulled at the end of a cycle (the 1st set is used to describe this). The assembly could be reconstituted and put back into the core for more cycles. The 2nd set is used to describe a subsequent PIE when the assembly comes out. The 1st set could also be used just to point out a simple visual exam. The second set could be used to describe the final exams at discharge burnup. This also works for single rod descriptions. However, if the rod is destructively examined at the first set of measurements, obviously the second set of measurements is "NA". In the case of a single rod description, the "Peak MOX rod burnup" and the "Avg Assy" burnup were both simply set to the average MOX rod burnup. The peak pellet of this rod was then given. For assembly descriptions the column titles are more self-explanatory.

MISCELLANEOUS: The most important of these columns are the last and next to the last column. In the "Overall Performance Notes," a brief overview of what was found for the rod is given, paying particular attention to whether the rod failed or not. The last column provides any miscellaneous information thought to be useful to the reader.

REFERENCES: Just below each line, a reference key is given which provides the full reference where the information in the cell was found. This reference is very valuable to the reader that wants to know more about a specific MOX rod assembly or irradiation. This reference key is used in conjunction with the reference list.

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
1							DOMESTIC PWR IRRADIATION													
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<				ITEM ASSY MOX DESIGN AND				FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-individual Rod, IS-irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239
5																				
6			GINNA IRRADIATION																	
7																				
8	1.1	1	B	Ginna	EPRI	PWR	1980	1985	10	4	NP1-4	14X14	716	?	?	3.1	3	92.9	0.29	79.56
9							ORTIZ-1	ORTIZ-1	ORTIZ-1	ORTIZ-1	ORTIZ-1	SHEA-1	SHEA-1			VOIGT-1	VOIGT-1	VOIGT-1	VOIGT-1	VOIGT-1
10																				
11			San Onofre-1																	
12																				
13	2.1	1	A	San Onofre-1	EEI	PWR	1970	1973	2	1	D51X	14X14	180	?	?	3.85	3	95.8	?	90.6
14							WCAP-4167-7	WCAP-4167-7				WCAP-4167-2	WCAP-4167-2			WCAP-4167-2	WCAP-4167-2	WCAP-4167-2		WCAP-4167-2
15	2.1 *	2	B	San Onofre-1	EEI	PWR	1970	1973	2	3	D49X,D50X,D52X	14X14	540	?	?	3.85	3	95.8	?	90.6
16							WCAP-4167-7	WCAP-4167-7				WCAP-4167-2	WCAP-4167-2			WCAP-4167-2	WCAP-4167-2	WCAP-4167-2		WCAP-4167-2
17			END OF COMMERCIAL PWRS																	
18																				
19		START OF PWR TEST RX ROD IRRADS																		
20																				
21			SAXTON CORE II RODS																	
22			"Four Low Burnup Rods"																	
23	3.1	1	R	Saxton	AEC	PWR-TEST	1965	1966	2 (core II)	NA	X-1	3X3 subassy in 9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
24							WCAP-3385-1	WCAP-3385-12			WCAP-3385-	WCAP-3385-51				WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
1																	
2	I S O T O P I C S																
3								>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)
5																	
6																	
7																	
8	16.06	3.36	0.73	N	Exxon/West.	100	High enr-center,med&low on edge/corner	NA	Boron	?	?	?	D	Zr-4	two assy - incore det. two assy-incore thermocouples	?	39,800
9	VOIGT-1	VOIGT-1	VOIGT-1	VOIGT-1	VOIGT-1	VOIGT-1	VOIGT-1		VOIGT-1				VOIGT-1	VOIGT-1	VOIGT-1		ORTIZ-1
10																	
11																	
12																	
13	13.4	5.2	0.8	N	West.	100	High fissile in center diamond, med and low interspersed and surrounding	NA	West. NFD Manuf	TOTAL Irrad-59.8 Kg Pu nitrate->50 Kg-oxalate precip:8kg coprecip u&Pu with ammonia yielding 35% Mast Mix (West. PFDL at Cheswick)	compiled with Eldorado Specification ENL-1	12 Kg Vee blend, 68Kg lotspressed into 1" slugs,crushed and roll mixed, pressed and sintered, wet centerless ground, vacuum dried	?	Zr-4	Some in-core power comparisons were made-not clear exactly which Pu assys	6.81	6,450
14	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2		WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2		WCAP-4167-2	WCAP-4167-3	WCAP-4167-7	WCAP-4167-7
15	13.4	5.2	0.8	N	West.	100	High fissile in center diamond, med and low interspersed and surrounding	NA	West. NFD Manuf	TOTAL Irrad-59.8 Kg Pu nitrate->50 Kg-oxalate precip:8kg coprecip u&Pu with ammonia yielding 35% Mast Mix (West. PFDL at Cheswick)	compiled with Eldorado Specification ENL-1	12 Kg Vee blend, 68Kg lotspressed into 1" slugs,crushed and roll mixed, pressed and sintered, wet centerless ground, vacuum dried	?	Zr-4	Some in-core power comparisons were made-not clear exactly which Pu assys	?	?
16	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2		WCAP-4167-2	WCAP-4167-2	WCAP-4167-2	WCAP-4167-2		WCAP-4167-2	WCAP-4167-3		
17																	
18																	
19																	
20																	
21																	
22																	
23	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel	NA	Zr-4	NA	10.6	4500
24	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8		WCAP-3385-12		WCAP-3385-12	WCAP-3385-12

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
1																
2	ET BU>>>>>		<<< FIRST SET MEASUREMENTS			>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND SET	M EASURE	MENTS	>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or-N batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
5																
6																
7																
8	?	?	?	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	?	NA	?
9																
10																
11																
12																
13	?	6,700	assembly visual, 4 rods-visual,rod length,gamma scan, diameter	2	Visual, length, profilometry(at H.C.), Fission Gas, clad hydrogen, clad & fuel metallography, gamma scan, autoradiographs,fast fluence, tensile/burst test, fuel dens & metallography, profilometry, burnup & transuranics-on 1 mech mixed rod and 1 master mix	Battelle	19,000	?	23,500	assembly visual, 6 rods-visual, rod length,rod diameter	2	Visual, length, profilometry(at H.C.), Fission Gas, clad hydrogen, clad & fuel metallography, gamma scan, autoradiographs,fast fluence, tensile/burst test, fuel dens & metallography, profilometry, burnup & transuranics-on 1 mech mixed rod and 1 master mix	Battelle	Y	NA	shipped to San Onofre in West Container Serial 26
14		WCAP-4167-	WCAP-4167-7	WCAP-4167-7	WCAP-4167-7	WCAP-4167-7			WCAP-4167-7			WCAP-4167-7	WCAP-4167-7			WCAP-4167-2
15	?	?	assembly visual	NA	NA	NA	?	?	?	Assembly Visual	NA	NA	NA	Y	NA	shipped to San Onofre in West Container Serial 26 and Serial 21
16			WCAP-4167-7													WCAP-4167-2
17																
18																
19																
20																
21																
22																
23	4500	6100	Visual, microscope, dimensional, gamma, determine peak power	1	Fission gas release and analysis, metallography(photomicrographs), H2 content in clad, Cs-137, Sr-90, Nd148, U&Pu isotopes	?	NA	NA	NA	NA	NA	NA	NA	Y	N	Y
24	CAP-3385-1	CAP-3385-1	WCAP-3385-12		WCAP-3385-12											

	BB	BC
1		
2		MISCELLANEOUS >>>
3		
4	Overall Performance notes (failure, crudding etc.)	Miscellaneous Info
5		
6		
7		
8	No fuel failures noted	NOTE: The actual fuel pellets were manufactured by Westinghouse, Exxon was responsible for rod loading and overall core analysis; average particle size was 10 to 20 microns
9		*VOIGT-1*
10		
11		
12		
13	No apparent anomalies, solid and gaseous fission products visible in mech mixed above 6,000 Mwd/t and in master mix above 21,500. Fission gas release of <2%. Comparisons between master mix and mech mix were made, master mix showed advantages	Manufacturing, irradiation, and PIE were fairly well documented in WCAP 4167 series of reports. Measured instrumentation showed that MOX assy powers were higher than calculated.
14	*WCAP-4167-7*	*WCAP-4167-2, MOMSEN-1*
15	No fuel failures noted	Manufacturing, irradiation, and PIE were fairly well documented in WCAP 4167 series of reports. Measured instrumentation showed that MOX assy powers were higher than calculated.
16	*WCAP-4167-7*	*WCAP-4167-2, MOMSEN-1*
17		
18		
19		
20		
21		
22		
23	No failure noted, cladding shows higher hydrogen content than pellet rod, inner surface of the cladding had 0.6 mil thick reaction layer. More moisture identified in the Vipac fuel. Slight increase in length.	difference in power discrepancy, calculated burnup higher than measured
24	*WCAP-3385-56, WCAP-3385-10*	*WCAP-3385-56*

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant.batch)	#RRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239
25	3.1	2	R	Saxton	AEC	PWR-TEST	1965	1966	2 (core II)	NA	X-5	3X3 subassy in 9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
26							WCAP-3385-1	WCAP-3385-12			WCAP-3385-	WCAP-3385-51				WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
27	3.1	3	R	Saxton	AEC	PWR-TEST	1965	1966	2 (core II)	NA	D	3X3 subassy in 9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
28							WCAP-3385-1	WCAP-3385-12				WCAP-3385-51				WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
29	3.1	4	R	Saxton	AEC	PWR-TEST	1965	1966	2 (core II)	NA	E	3X3 subassy in 9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
30							WCAP-3385-1	WCAP-3385-12				WCAP-3385-51				WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
31			"Rod OverPower Test"																	
32	3.1	5	R	Saxton	AEC	PWR-TEST	1965	1967	2 (core II)	NA	A	3X3 mod w/increased lattice pitch in 9X9	1	?	?	6.6	1	91.4	?	90.49
33							WCAP-3385-56	WCAP-3385-56				WCAP-3385-56				WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
34	3.1	6	R	Saxton	AEC	PWR-TEST	1965	1967	2 (core II)	NA	B	3X3 mod increased lattice pitch in 9X9	1	?	?	6.6	1	91.4	?	90.49
35							WCAP-3385-56	WCAP-3385-56				WCAP-3385-56				WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
36	3.1	7	R	Saxton	AEC	PWR-TEST	1965	1967	2 (core II)	NA	D' (not same rod as above)	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
37							WCAP-3385-56	WCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
38	3.1	8	R	Saxton	AEC	PWR-TEST	1965	1967	2 (core II)	NA	CH	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
39							WCAP-3385-56	WCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
40			"14 Rod Core II PIE"																	

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S							>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
3																	
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular,S-solid, D-dished,U-undished)	Cladding utilized	Any Instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)
25	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel					
26	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8	NA	Zr-4	NA	10.6	4500
27	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground					
28	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	D	Zr-4	NA	10.6	4500
29	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground					
30	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	D	Zr-4	NA	10.6	4500
31																	
32	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground					
33	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	D	Zr-4	flux wire, in/out T/C, flow meas for Overpower test	18.7	15,400
34	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground					
35	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	D	Zr-4	flux wire, in/out T/C, flow meas for Overpower test	18.7	15,400
36	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground					
37	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	D	Zr-4	?	?	9,300
38	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground					
39	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	D	Zr-4	?	12.0	20,500
40																	

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET ME		A SUREMENTS	>>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND S ET	M EASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
25	4500	6100	Visual, microscope, dimensional, gamma, determine peak power	1	Fission gas release and analysis	?	NA	NA	NA	NA	NA	NA	NA	Y	N	Y
26	/CAP-3385-1	/CAP-3385-1	WCAP-3385-12		WCAP-3385-10											
27	4500	6100	Visual, microscope, dimensional, gamma, determine peak power	1	Fission gas release and analysis, metallography(photomicrographs), H2 content in clad, Cs-137, Sr-90, Nd148, U&Pu isotopes	?	NA	NA	NA	NA	NA	NA	NA	Y	N	Y
28	/CAP-3385-1	/CAP-3385-1	WCAP-3385-12		WCAP-3385-12											
29	4500	6100	Visual, microscope, dimensional, gamma, determine peak power	1	Fission gas release and analysis	?	NA	NA	NA	NA	NA	NA	NA	Y	N	Y
30	/CAP-3385-1	/CAP-3385-1	WCAP-3385-12		WCAP-3385-12											
31																
32	15,400	?	Extensive Visual, gamma scan, profilometry and length	1	Fission gas release and analysis, Metallography w/ alpha and beta/gamma autoradiographs,oxide film	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No, but this was a deliberate overpower test	?
33			WCAP-3385-56		WCAP-3385-56, WCAP-3385-20	WCAP-3385-16										
34	15,400	?	Extensive Visual, gamma scan, profilometry and length	1	Fission gas release and analysis, Metallography w/ alpha and beta/gamma autoradiographs,oxide film	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No, but this was a deliberate overpower test	?
35			WCAP-3385-56		WCAP-3385-56, WCAP-3385-20	WCAP-3385-16										
36	9,300	?	Extensive Visual, gamma scan, profilometry and length	1	Fission gas release and analysis, Metallography w/ alpha and beta/gamma autoradiographs,oxide film	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	This was an Overpower test reference rod	?
37			WCAP-3385-56		WCAP-3385-56, WCAP-3385-20	WCAP-3385-16										
38	20,500	?	Extensive Visual, gamma scan, profilometry and length	1	Fission gas release and analysis, Metallography w/ alpha and beta/gamma autoradiographs,oxide film	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	This was an Overpower test reference rod	?
39			WCAP-3385-56		WCAP-3385-56, WCAP-3385-20	WCAP-3385-16										
40																

	BB	BC
2		MISC ELLANEOUS >>>
3		
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info
25	No failure noted	No comment
26		
27	No failure noted or significant performance characteristics	difference in power discrepancey, measured burnup higher than calculated
28	*WCAP-3385-56*	*WCAP-3385-56*
29	No failure noted	No comment
30	*WCAP-3385-56*	
31		
32	Light crud deposits, no failures noted. Fission product predpitates visible in high temperature central regions. Both overpower test rods showed higher fission gas release than reference rods	This rod was removed from the core during early 1967 shutdown (13,400 Mwd/t calc) and later placed in the test fixture for the overpower test. Overpower test was conducted in steps. Power steps followed by visual exams and flux wire measurements. The absence of columnar grain growth or well defined central void was not consistent with ~21.5 kw/ft, review lowered peak to 18.7 kw/ft
33	*WCAP-3385-56*	*WCAP-3385-56, WCAP-3385-16*
34	Light crud deposits, no failures noted. Fission product predpitates visible in high temperature central regions. Both overpower test rods showed higher fission gas release than reference rods	This rod was removed from the core during early 1967 shutdown (13,800 MWd/t calc) and later placed in the test fixture for the overpower test. Overpower test was conducted in steps. Power steps followed by visual exams and flux wire measurements. The absence of columnar grain growth or well defined central void was not consistent with ~21.5 kw/ft, review lowered peak to 18.7 kw/ft
35	*WCAP-3385-56*	*WCAP-3385-56, WCAP-3385-16*
36	Surface was black and lustrous, little evidence of crud deposits	NOTE: This is an overpower reference rod, e.g. similar burnup and LHGR to rods A&B, prior to the overpower test
37	*WCAP-3385-56*	*WCAP-3385-56*
38	Free of anomalous surface conditions, except for stain-like areas probably from corrosion products	NOTE: This is an overpower reference rod, operated at highest linear power levels outside the overpower test assembly
39		*WCAP-3385-56*
40		

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if multi blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-239	%Pu-239
41	3.1	9	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	TI	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
42							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
43	3.1	10	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	TP	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
44							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
45	3.1	11	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	TT	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
46							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
47	3.1	12	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	QE	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
48							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
49	3.1	13	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	TE	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
50							wCAP-3385-56	wCAP-3385-56			wCAP-3385-56					WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
51	3.1	14	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	LA	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
52							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
53	3.1	15	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	MY	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
54							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
55	3.1	16	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	RI	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
56							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
57	3.1	17	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	JF	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
58							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S							>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or fi batch-Max assy)
41	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet, blended, pressed, no bindersintered and ground	D	Zr-4	?	?	17,400
42	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
43	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steamd oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	19,700
44	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
45	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	20,000
46	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
47	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	20,700
48	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
49	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	20,700
50	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
51	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	17,800
52	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
53	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	14,900
54	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
55	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	14,900
56	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56
57	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	14,800
58	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			WCAP-3385-56

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET MEASUREMENTS			>>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND SET	MEASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or N batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
41	17,400	?	Visual examination, Length, gamma scan	1	Fission gas release and analysis, some autoradiography, oxide film, clad hydrogen, clad tensile test, clad burst test	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
42			WCAP-3385-56		WCAP-3385-56											
43	19,700	?	Visual examination, Length, gamma scan	1	Fission gas release, clad tensile test,clad burst	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
44			WCAP-3385-56		WCAP-3385-56											
45	20,000	?	Visual examination, Length, gamma scan	1	Fission gas release, oxide film, fast flux dosimetry, clad tensile, clad burst,	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
46			WCAP-3385-56		WCAP-3385-56											
47	20,700	?	Visual examination, Length, gamma scan	1	Fission gas release, oxide film, clad hydrogen, fast flux dosimetry,	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
48			WCAP-3385-56		WCAP-3385-56											
49	20,700	?	Visual examination, Length, gamma scan	1	Fission gas,oxide film, tensile - test,clad burst, U/Pu isotope, Nd-148	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
50			WCAP-3385-56		WCAP-3385-56											
51	17,600	?	Visual examination, Length, gamma scan	1	Fission gas, oxide, fast flux dosimetry,tensile test, clad burst test	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
52			WCAP-3385-56													
53	14,900	?	Visual examination, Length, gamma scan	1	Fission gas, U/Pu, Nd-148,Pu-236,Pu238, Np-237, Am-241,Am-243	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
54			WCAP-3385-56		WCAP-3385-56											
55	14,900	?	Visual examination, Length, gamma scan	1	Fission gas, tensile test, U/Pu, Nd-148,Pu-236,Pu-238,Np-237, Am-241, Am-243, Cs137,multi&single channel gamma scan, macro-photo&auto-radiograph	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
56			WCAP-3385-56		WCAP-3385-56											
57	14,600	?	Visual examination, Length, gamma scan	1	Fission gas,tensile test, U/Pu isotopes, Nd-148, Pu-236/Pu-238	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
58			WCAP-3385-56		WCAP-3385-56											

	BB	BC
2		
3		MISCELLANEOUS >>>
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous info
41	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
42	WCAP-3385-56	WCAP-3385-56
43	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
44	WCAP-3385-56	WCAP-3385-56
45	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
46	WCAP-3385-56	WCAP-3385-56
47	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
48	WCAP-3385-56	WCAP-3385-56
49	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
50	WCAP-3385-56	WCAP-3385-56
51	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
52	WCAP-3385-56	WCAP-3385-56
53	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature. This rod was checked also because of rod ML anomaly (see additional core II rods). It is from batch#3
54	WCAP-3385-56	WCAP-3385-56
55	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
56	WCAP-3385-56	WCAP-3385-56
57	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
58	WCAP-3385-56	WCAP-3385-56

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (e commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239
59	3.1	18	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	K4	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
60							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
61	3.1	19	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	N3	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
62							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
63	3.1	20	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	N1	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
64							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
65	3.1	21	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	S1	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
66							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
67	3.1	22	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	P0	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
68							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
69		"End of Core II Additional Rods"																		
70	3.1	23	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	ML	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
71							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
72	3.1	24	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	LI	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
73							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
74	3.1	25	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	MG	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
75							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
76	3.1	26	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	RA	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
77							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
78	3.1	27	R	Saxton	AEC	PWR-TEST	1965	1968	2 (core II)	NA	TM	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S							>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, B-solid, D-dished, U-undished)	Cladding utilized	Any instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)
59	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel	NA	Zr-4	?	?	20,400
60	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8		WCAP-3385-12			WCAP-3385-56
61	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel	NA	Zr-4	?	?	19,300
62	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8		WCAP-3385-12			WCAP-3385-56
63	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel	NA	Zr-4	?	?	20,700
64	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8		WCAP-3385-12			WCAP-3385-56
65	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel	NA	Zr-4	?	?	15,700
66	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8		WCAP-3385-12			WCAP-3385-56
67	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal buttons to thin strips oxidized at 550C, calcined at 950C-powder through a 44 micron sieve	Arc-fused UO2 mixed w/U3O8	VIPAC-BNL's Nupac process-four size fractions sieved and mixed, vibrated and tamped at 50g accel	NA	Zr-4	?	?	15,300
68	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53, WCAP-3385-8		WCAP-3385-12			WCAP-3385-56
69																	
70	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no binder-sintered and ground	D	Zr-4	?	?	?
71	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			
72	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	treated in a Fitz mill, wet blended, pressed, no binder-sintered and ground	D	?	?	?	?
73	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				
74	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no binder-sintered and ground	D	Zr-4	?	?	?
75	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			
76	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	treated in a Fitz mill, wet blended, pressed, no binder-sintered and ground	D	?	?	?	?
77	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				
78	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	treated in a Fitz mill, wet blended, pressed, no binder-sintered and ground	D	?	?	?	?

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET MEASUREMENTS			>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND SET	MEASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
59	20,400	?	Visual examination, Length, gamma scan	1	Fission gas,oxide film, clad hydrogen, tensile test, burst test,	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
60			*WCAP-3385-56*		*WCAP-3385-56*											
61	19,300	?	Visual examination, Length, gamma scan	1	Fission gas, oxide film, tensile test, burst test	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
62			*WCAP-3385-56*		*WCAP-3385-56*											
63	20,700	?	Visual examination, Length, gamma scan	1	Fission gas, oxide film, tensile test, burst test	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
64			*WCAP-3385-56*		*WCAP-3385-56*											
65	15,700	?	Visual examination, Length, gamma scan	1	Fission gas,tensile test, U/Pu isotopes, Nd-148, Pu-236/Pu-239	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
66			*WCAP-3385-56*		*WCAP-3385-56*											
67	15,300	?	Visual examination, Length, gamma scan	1	Fission gas, tensile test, U/Pu, Nd-148,Pu-236,Pu-239	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
68			*WCAP-3385-56*		*WCAP-3385-56*											
69																
70	?	?	Fission gas collection	1	Fission gas, metallography	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
71			*WCAP-3385-23*		*WCAP-3385-23*											
72	?	?	Visual?	1	Metallography, others?	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
73																
74	?	?	Visual?	1	Metallography, others?	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
75																
76	?	?	Visual?	1	Metallography, others?	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
77																
78	?	?	Visual?	1	Metallography, others?	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?

	BB	BC
2		MISCELLANEOUS >>>
3		
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info
59	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
60	*WCAP-3385-56*	*WCAP-3385-56*
61	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
62	*WCAP-3385-56*	*WCAP-3385-56*
63	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
64	*WCAP-3385-56*	*WCAP-3385-56*
65	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
66	*WCAP-3385-56*	*WCAP-3385-56*
67	For this 14 rod irradiation: No failures or other significant anomalies noted. Most rods covered with thin, dark-gray, crud film.	Fairly extensive performance comparison (based on the 14 rod irradiation) between pellet fuels and VIPAC fuels is in the literature.
68	*WCAP-3385-56*	*WCAP-3385-56*
69		
70	Rod did not fail. However, local hydriding, thick oxide layer, clad cracked extending towards outer surface for 70% of wall thickness.	This rod selected because of anomalous dimensions in core III reconstitution. It led to closer looks at rods LI, MG, RA, &TM. Cause of ML hydriding believed to be stray hydrogenous contamination
71	*WCAP-3385-23*	*WCAP-3385-23*
72	?	Done because of rod ML anomaly
73		*WCAP-3385-23*
74	Like Rod ML, this pellet fuel showed significant hydriding. It was from same manufacturing batch #3 as ML	Done because of rod ML anomaly
75		*WCAP-3385-23*
76	?	Done because of rod ML anomaly
77		*WCAP-3385-23*
78	?	Done because of rod ML anomaly

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (& commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239
79							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
80	3.1	28	R	Saxton	AEC	PWR-TEST	1965	1969	2 (core II)	NA	MO	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
81							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
82	3.1	29	R	Saxton	AEC	PWR-TEST	1965	1969	2 (core II)	NA	MZ	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
83							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
84			REMAINING Estimated Core II rods (arrived at by deduction: (639 total -250=389 rods total that went to end of core II-29 rods accounted for above=359 rods																	
85	3.1	30	A	Saxton	AEC	PWR-TEST	1965	1969	2 (core II)	NA	NA	9X9 (mod)	359	?	?	6.6	1	91.4	?	90.49
86							wCAP-3385-56	wCAP-3385-56								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
87		SAXTON CORE III "Loose Lattice" RODS																		
88			Midlife Core III rods (includes RR -early core III and 7 MOL rods from variety of assembly positions)																	
89	3.2	1	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	RR	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
90							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
91	3.2	2	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	LZ	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
92							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
93	3.2	3	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	BO	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
94							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
95	3.2	4	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	NI	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
96							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
97	3.2	5	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	RD	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
98							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S							>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
3																	
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular,S-solid, D-dished,U-undished)	Cladding utilized	Any instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)
79	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				
80	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	?
81	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			
82	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zr-4	?	?	?
83	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-12			
84																	
85	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	This row represents both VIPAC and pellet fuels, do not know the mix	This row represents both VIPAC and pellet fuels	This row represents both VIPAC and pellet fuels	?	This row represents both 304 stainless steel and zirc clad rods	?	?	?
86	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51										
87																	
88																	
89	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	Zirc-4	?	16.0	25,100
90	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57
91	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	17.3	33,700
92	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57
93	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	21.0	25,800
94	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57
95	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	16.3	27,000
96	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57
97	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	17.1	26,100
98	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET MEASUREMENTS			>>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND SET	MEASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
79																
80	?	?	Visual?	1	Metallography, others?	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
81																
82	?	?	Visual?	1	Metallography, others?	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
83																
84																
85	?	?	?	?	?	?	NA	NA	NA	NA	NA	NA	NA	Y	None noted	?
86																
87																
88																
89	25,100	33,200	Visual, profilometry, gamma scann	1	Fission gas release, metallography, Beta/gamma autoradiograph, clad hydrogen, oxide thickness, Nd-148, U/Pu	West PIE Hot Cell	NA	NA	NA	NA	NA	NA	NA	Y	No	?
90		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
91	33,700	41,100	Visual, profilometry, length, gamma scan	1	Fission Gas, Metallography, clad hydrogen, alpha, beta gamma autoradiography, Nd-148, U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
92		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
93	25,800	34,000	Visual, profilometry, length, gamma scan	1	Fission gas release, metallography, tensile, burst, Mn-54 (fast flux), clad hydrogen, alphaBetagamma auto radiograph, Nd-149, U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
94		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
95	27,000	33,500	Visual, profilometry, length, gamma scan	1	Fission gas release, metallography,Clad burst, Mn-54(fast flux), clad H2, alpha&beta gamma autoradiography, Nd-149, U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
96		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
97	26,100	32,000	Visual, profilometry, length, gamma scan	1	Fission gas release, metallography, tensile, clad burst, Mn-54 (fast flux), clad hydrogen, Nd148,U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
98		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										

	BB	BC
2	____	MISC ELLANEOUS ____ >>>
3		
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info .
79		*WCAP-3385-23*
80	Unlike Rod ML, this pellet fuel showed no significant hydriding; it was from same batch #3 as ML	Done because of rod ML anomaly
81	*WCAP-3385-23*	*WCAP-3385-23*
82	Unlike Rod ML, this pellet fuel showed no significant hydriding; do not know if it was from same batch #3 as ML	Done because of rod ML anomaly. This was known as an archive rod (for ML)
83	*WCAP-3385-23*	*WCAP-3385-23*
84		
85	No failures noted	NOTE: THIS ROW REPRESENTS AN ESTIMATED 359 MOX RODS USED IN CORE II and removed following core II.
86		
87		
88		
89	No major defects noted. Good performance. Larger fraction fuel w/equiaxial grain growth & fission product precipitate	Removed early in Core III, as a lead indicator
90	*WCAP-3385-57*	*WCAP-3385-57*
91	No failure noted	No comment
92		
93	No failure noted	No comment
94		
95	No failure noted	No comment
96		
97	No failure noted	No comment
98		

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant,batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (e commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-239	%Pu-239
99	3.2	6	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	FS	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
100							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
101	3.2	7	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	GL	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
102							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
103	3.2	8	R	Saxton	AEC	PWR-TEST	1965	1970?	2 (core III)	NA	MQ	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
104							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
105																				
106			End of Core III rods																	
107	3.2	9	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	IM	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
108							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
109	3.2	10	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	PF	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
110							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
111	3.2	11	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	BE	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
112							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
113	3.2	12	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	BK	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
114							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
115	3.2	13	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	GM	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
116							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
117	3.2	14	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	LS	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
118							wCAP-3385-56	wCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S																
3								>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or M batch-Max assy)
99	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	15.4	25,500
100	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57
101	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	15.3	26,500
102	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57
103	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	10.7	28,800
104	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53			WCAP-3385-29	WCAP-3385-57
105																	
106																	
107	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet	D	?	?	?	39,000
108	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
109	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	37,500
110	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
111	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	30,200
112	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
113	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	33,400
114	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
115	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	31,900
116	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
117	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	33,700
118	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET MEASUREMENTS			>>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND SET	MEASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to PX and to PIE?
99	25,500	32,000	Visual, profilometry, length, gamma scan	1	Fission gas release, metallography, tensile, clad burst, Mn-54 (fast flux), clad hydrogen, Nd148,U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
100		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
101	26,500	33,000	Visual, profilometry, length, gamma scan	1	Fission gas release, metallography, tensile, clad burst, Mn-54 (fast flux), clad hydrogen, Nd148,U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
102		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
103	28,800	36,000	Visual, profilometry, length, gamma scan	1	Fission gas release, metallography, tensile, clad burst, Mn-54 (fast flux), clad hydrogen, alpha,beta gamma autoradiography Nd148,U/Pu (radially sampled)	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	?
104		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
105																
106																
107	39,000	50,500	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst,	Battelle Mem Inst	NA	NA	NA	NA	NA	NA	NA	Y	No	
108		WCAP-3385-5	WCAP-3385-57			WCAP-3385-57										
109	37,500	51,000	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst, Mn-54 (fast flux), clad hydrogen, alpha & beta gamma autoradiography, Nd148, U/Pu, Pu-236, Pu-238, Np-237, Am-241, Am-	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
110		WCAP-3385-5	WCAP-3385-57			WCAP-3385-57										
111	30,200	43,000	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst, Mn-54 (fast flux), clad hydrogen, alpha & beta gamma autoradiography, Nd148, U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
112		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
113	33,400	41,900	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst, Mn-54 (fast flux), clad hydrogen, alpha & beta gamma autoradiography, Nd148, U/Pu, Pu-236, Pu-238, Np-237, Am-241, Am-243, Cm-242,Cm-244	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
114		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
115	31,900	41,500	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst, Mn-54 (fast flux), clad hydrogen, Nd148, U/Pu	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
116		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
117	33,700	42,000	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst, Mn-54 (fast flux), clad hydrogen, Nd148, U/Pu,Pu-236, Pu-238, Np-237, Am-241, Am-243, Cm-242, Cm-	Battelle Mem Inst (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
118		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										

	BB	BC
2		MISC ELLANEOUS >>>
3		
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info
99	No failure noted	No comment
100		
101	No failure noted	No comment
102		
103	Suspected failed rod based on low fission gas release	No comment
104	*WCAP-3385-57*	
105		
106		
107	No failure noted	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
108		*WCAP-3385-57*
109	Several radial cracks originating from the inner surface, thick exterior oxide rim, solid hydride, extensive inner clad oxide	Statement is made that this rod was examined as part of another program. Performance comparisons for end of core III rods discussed in WCAP-3385-57
110	*WCAP-3385-57*	*WCAP-3385-57*
111	Experienced abnormally high rapid oxidation outside of clad surface	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
112	*WCAP-3385-57*	*WCAP-3385-57*
113	No failure noted, experienced columnar grain growth	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
114	*WCAP-3385-57*	*WCAP-3385-57*
115	No failure noted, external corrosion films were as expected	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
116	*WCAP-3385-57*	*WCAP-3385-57*
117	No failure noted, external corrosion films were as expected	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
118	*WCAP-3385-57*	*WCAP-3385-57*

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<	___	___	REACTOR SPECIFICS				>>>>>>>>	<<<<<<<	___	___	ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239
119	3.2	15	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	FI	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
120							WCAP-3385-56	WCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
121	3.2	16	R	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	JX	9X9 (mod)	1	?	?	6.6	1	91.4	?	90.49
122							WCAP-3385-56	WCAP-3385-57								WCAP-3385-51		WCAP-3385-51	WCAP-3385-51	WCAP-3385-51
123		Rest of Core III rods (250-16=234)																		
124	3.2	17	A	Saxton	AEC	PWR-TEST	1965	1972	2 (core III)	NA	NA	9X9 (mod)	234	?	?	6.6	1	91.4	?	90.49
125																				
126		Japan's PNC MOX in Saxton Core III																		
127	3.3	1	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	C-1	9X9(mod)	1	?	?	4.0	1	?	?	?
128									AKUTSU-1							AKUTSU-1				
129																				
130	3.3	2	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	C-3	9X9(mod)	1	?	?	4.0	1	?	?	?
131									AKUTSU-1							AKUTSU-1				
132																				
133	3.3	3	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	25	9X9(mod)	1	?	?	4.9	1	?	?	?
134									AKUTSU-1							AKUTSU-1				
135																				
136	3.3	4	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	41	9X9(mod)	1	?	?	4.9	1	?	?	?
137									AKUTSU-1							AKUTSU-1				

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S							>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
3																	
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)
119	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	31,000
120	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
121	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	Pu metal cut into small cubes, steam oxidized. Oxide was ball milled to pass through 325 mesh	ceramic grade oxide	Pellet-NUMEC, Vee blended treated in a Fitz mill, wet blended, pressed, no bindersintered and ground	D	?	?	?	?
122	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-51	WCAP-3385-8		WCAP-3385-51			WCAP-3385-53	WCAP-3385-53	WCAP-3385-53	WCAP-3385-53				WCAP-3385-57
123																	
124	0.57	0.09	0.04	N	West.	100	uniform MOX	NA	West.	This row represents both VIPAC and pellet fuels, do not know the mix	?	?	?	This row represents both 304 stainless steel and zirc clad rods, do not know mix	?	?	?
125																	
126																	
127	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	14.9	?
128																AKUTSU-1	
129																	
130	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	9.8	?
131																AKUTSU-1	
132																	
133	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	11.3	?
134																AKUTSU-1	
135																	
136	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	6.1	?
137																AKUTSU-1	

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET MEASUREMENTS			>>>>>>>>>	<<<SECOND SET BU>>>>>			SECOND SET	MEASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
119	31,000	40,900	Visual, profilometry, length, gamma scan	1	Fission gas release,metallography,tensile, burst, Mn-54 (fast flux), clad hydrogen, Alpha and beta gamma autoradiography, Nd148, U/Pu	Battelle Mem Inst. (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
120		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
121	?	44,000	Visual (profilometry, length, gamma scan?)	1	(No fission gas since it's a failed rod) Metallography, Mn-54 (fast flux), clad hydrogen, Nd148, U/Pu	Battelle Mem Inst. (Columbus Ohio)	NA	NA	NA	NA	NA	NA	NA	Y	No	
122		WCAP-3385-5	WCAP-3385-57		WCAP-3385-57	WCAP-3385-57										
123																
124	?	?	Visual checks done at reconstitution at the end of core II. Approximately 250 rods in total selected for insertion in Core III for "extended burnup testing". Thus this line item represents app 250-16 or approx 235 rods	NA	NA	NA	?	?	?	Leak testing and other non-destructive examinations, performed as part of a West. funded program, identified 37 defected rods from high power locations	NA	NA	NA	Y	No	
125																
126																
127	?	8,050	Length	1	Fission gas, alpha &beta gamma autoradiograph, metallography, U/Pu burnup sample	Battelle Mem. Institute	NA	NA	NA	NA	NA	NA	NA	Y	No	?
128		AKUTSU-1	AKUTSU-1		AKUTSU-1	AKUTSU-1										
129																
130	?	5,890	Length	1	Fission gas, metallography, U/Pu burnup sample (not isotopics)	Battelle Mem. Institute	NA	NA	NA	NA	NA	NA	NA	Y	No	?
131		AKUTSU-1	AKUTSU-1		AKUTSU-1	AKUTSU-1										
132																
133	?	?	Length, profilometry	1	Fission gas, Mn-54 clad, clad hydrogen, clad burst	Battelle Mem. Institute	NA	NA	NA	NA	NA	NA	NA	Y	No	?
134		AKUTSU-1	AKUTSU-1		AKUTSU-1	AKUTSU-1										
135																
136	?	3,430	Length, profilometry	1	Fission gas, alpha, beta gamma autoradiograph, U/Pu burnup, metallograph, radial burnup	Battelle Mem. Institute	NA	NA	NA	NA	NA	NA	NA	Y	No	?
137		AKUTSU-1	AKUTSU-1		AKUTSU-1	AKUTSU-1										

	BB	BC
2		MISCELLANEOUS >>>
3		
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info
119	No failure noted, external corrosion films were as expected	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
120	*WCAP-3385-57*	*WCAP-3385-57*
121	Rod failed. Thick oxide spalling, extensive hydriding with cracks. Concluded that failure mode was accelerated oxidation from outside of clad between mid-life and end of life, columnar grain growth. Failure not attributed to MOX	Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
122	*WCAP-3385-57*	*WCAP-3385-57*
123		
124	There were 37 identified failed rods. One of these (JX) was selected for destructive PIE and is shown above. Thus, there were 36 failed rods out of 235. Many of these failed rods had operated at a high power level throughout Core III. Some failed rodshad linear power increases in excess of 10kw/ft over their Core II levels	NOTE: THIS ROW REPRESENTS AN ESTIMATED APPROXIMATE 235 MOX RODS USED IN CORE III consisting of both VIPAC and Mech mixed Saxton rods. Performance comparisons and results of End of Core III rods discussed in WCAP-3385-57
125	*WCAP-3385-57*	*WCAP-3385-57*
126		
127	No failure noted	No comment
128	*AKUTSU-1*	
129		
130	No failure noted	No comment
131	*AKUTSU-1*	
132		
133	No failure noted	No comment
134	*AKUTSU-1*	
135		
136	No failure noted	No comment
137	*AKUTSU-1*	

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AND					FUEL
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																			
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if multi blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-239	%Pu-239
138	3.3	5	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	43	9X9(mod)	1	?	?	4.9	1	?	?	?
139									AKUTSU-1							AKUTSU-1				
140	3.3	6	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	C-2	9X9(mod)	1	?	?	4.0	1	?	?	?
141									AKUTSU-1							AKUTSU-1				
142	3.3	7	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	C-4	9X9(mod)	1	?	?	4.0	1	?	?	?
143									AKUTSU-1							AKUTSU-1				
144	3.3	8	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	37	9X9(mod)	1	?	?	4.9	1	?	?	?
145									AKUTSU-1							AKUTSU-1				
146	3.3	9	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	40	9X9(mod)	1	?	?	4.9	1	?	?	?
147									AKUTSU-1							AKUTSU-1				
148	Rem. Japan's PNC MOX -Saxton Core III (p324, Akuto-68+4=72 MOX tot)																			
149	3.3	10	R	Saxton	PNC (Japan)	PWR-TEST	1971	1972	Core III	NA	NA	9X9(mod)	63	?	?	4.9	1	?	?	?
150									AKUTSU-1				AKUTSU-1			AKUTSU-1				
151																				
152		ADDITIONAL IRRADIATIONS TO BE MAPPED:																		
153																				
154		Plutonium Recycle Test Reactor																		
155																				
156																				
157																				
158		Capsules in GE Test Reactor, MTR, ETR																		
159																				

	U	V	W	X	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK
2	I S O T O P I C S							>>>>>>	<<<	F A B R I C A T I O N A S P E C T S					>>>>	Max LHGR	<<<FIRST S
4	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)
138	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	11.3	?
139																AKUTSU-1	
140	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	?	?
141																	
142	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	?	?
143																	
144	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	?	?
145																	
146	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	?	?
147																	
148																	
149	?	?	?	?	?	100	uniform MOX	NA	?	?	?	?	?	?	?	?	?
150																	
151																	
152																	
153																	
154																	
155																	
156																	
157																	
158																	
159																	

	AL	AM	AN	AO	AP	AQ	AR	AS	AT	AU	AV	AW	AX	AY	AZ	BA
2	ET BU>>>>>		<<< FIRST SET ME		A SURE	MENTS	>>>>>>>>>	<<<SECOND SET BU>>>>>		SECOND SET	M EASUREMENTS		>>>>>>> > >	<<<<<<<<<		
3																
4	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?
138	?	5,570	Length	1	Fission gas, clad burst, clad tensile, metallography, clad hydrogen	Battelle Mem. Institute	NA	NA	NA	NA	NA	NA	NA	Y	No	?
139		AKUTSU-1	AKUTSU-1		AKUTSU-1	AKUTSU-1										
140	?	?	Length-at Battelle and GE Vallecitos, Length also via neutron radiograph	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	Y	No	?
141			AKUTSU-1		AKUTSU-1	AKUTSU-1										
142	?	?	Length-at Battelle and GE Vallecitos, Length also via neutron radiograph	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	Y	No	?
143			AKUTSU-1		AKUTSU-1	AKUTSU-1										
144	?	?	Length-at Battelle and GE Vallecitos, Length also via neutron radiograph, profilometry	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	Y	No	?
145			AKUTSU-1		AKUTSU-1	AKUTSU-1										
146	?	?	Length-at GE Vallecitos based on neutron radiograph	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	Y	No	?
147			AKUTSU-1		AKUTSU-1	AKUTSU-1										
148																
149	?	?	Length-at GE Vallecitos based on neutron radiograph	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	Y	No	?
150			AKUTSU-1		AKUTSU-1	AKUTSU-1										
151																
152																
153																
154																
155																
156																
157																
158																
159																

	BB	BC
2		MISC ELLANEOUS >>>
3		
4	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info
138	No failure noted	No comment
139	*AKUTSU-1*	
140	No failure noted	No comment
141	*AKUTSU-1*	
142	No failure noted	No comment
143	*AKUTSU-1*	
144	No failure noted	No comment
145	*AKUTSU-1*	
146	No failure noted	No comment
147	*AKUTSU-1*	
148		
149	No failure noted	Reference cites 68 rods of 4.9% blend. Subtracting the 5 rods, gives 63 rods for the remainder which were not examined at all for this line item
150	*AKUTSU-1*	*AKUTSU-1*
151		
152		
153		
154		
155		
156		
157		
158		
159		

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	X
1							DOMESTIC BWR IRRADIATION																	
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<	___	___	REACTOR SPECIFICS				>>>>>>>>	<<<<<<<	___	___	ITEM ASSY MOX DESIGN AN					FUEL	ISOTOPICS			___
3	NOTE: blank-irrad was not mapped, ?-no date found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																							
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/ (U+Pu) in ITEM (Max if multi blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)
5																								
6			QUAD CITIES EPRI																					
7	1.1	1	A	Quad Cities	EEI/EPRI	BWR	1974	198?	2	1	GEB-161	7X7	10	?	?	4.0	4	90.0/80.1	0.12/0.25	67.16/75.66	10.06/18.49	2.38/4.47	0.28/1.13	N
8	REF						EPRI-240		EPRI-240			EPRI-240	EPRI-240			EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2
9	1.1	2	A	Quad Cities	EEI/EPRI	BWR	1974	198?	2	1	GEB-162	7X7	8	?	?	4.0	4	90.0/80.1	0.12/0.25	67.16/75.66	10.06/18.49	2.38/4.47	0.28/1.13	N
10	REF															EPRI-72-2		EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	
11	1.1	3	A	Quad Cities	EEI/EPRI	BWR	1974	198?	2	1	GEB-159	7X7	10	?	?	4.0	4	90.0/80.1	0.12/0.25	67.16/75.66	10.06/18.49	2.38/4.47	0.28/1.13	N
12	REF															EPRI-72-2		EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	
13	1.1	4	A	Quad Cities	EEI/EPRI	BWR	1974	198?	2	1	GEB-158	7X7	10	?	?	4.0	4	90.0/80.1	0.12/0.25	67.16/75.66	10.06/18.49	2.38/4.47	0.28/1.13	N
14	REF															EPRI-72-2		EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	
15	1.1	5	A	Quad Cities	EEI/EPRI	BWR	1974	198?	2	1	GEB-160	7X7	10	?	?	4.0	4	90.0/80.1	0.12/0.25	67.16/75.66	10.06/18.49	2.38/4.47	0.28/1.13	N
16	REF															EPRI-72-2		EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	
17	1.1	NA	IS	Quad Cities	EEI/EPRI	BWR	1974	198?	2	5	NA	7X7	48	?	?	4	4	90.0/80.1	0.12/0.25	67.16/75.66	10.06/18.49	2.38/4.47	0.28/1.13	N
18	REF															EPRI-72-2		EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	EPRI-72-2	
19																								
20			BRP-32 ROD EEI																					
21																								
22	2.1	1	R	Big Rock Pt	EEI/EPRI	BWR	1969	1979	7	NA	DY0001	9X9	1	?	?	1.40?	1	90	?	?	?	?	?	N
23	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			EPRI 72-2	EPRI 72-2	EPRI 72-2						EPRI 72-2
24	2.1	2	R	Big Rock Pt	EEI/EPRI	BWR	1969	1970	7	NA	DY0002	9X9	1	?	?	1.40	1	90	?	67.33	9.92	2.49	0.27	N
25	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			NEDC-10387	EPRI 72-2	NEDC-10387		NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387
26	2.1	3	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0003	9X9	1	?	?	?	1	90	?	?	?	?	?	N
27	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
28	2.1	4	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0004	9X9	1	?	?	?	1	90	?	?	?	?	?	N
29	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2

	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK	AL	AM	AN	AO	AP	AQ	AR	AS
1																					
2	—	—	—	>>>>>>	<<<	FABRICATION ASPECTS				—	>>>>	Max LHGR	<<<FIRST SET BU>>>>>			<<< FIRST SET MEASUREMENTS			>>>>>>>>	<<<SECOND SET B	
3																					
4	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument.	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or ll batch-Max assy)	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or ll batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT
5																					
6																					
7	GE	20.4	Island	Gd in U	GE	?(Vallecitos)	?	cold pressed, mechanically mixed powder	A,S,D,U	Zr-2	No-but TIPS nearby	?	9160	?	13400	gamma scan-power dist,	4	burnup and transuranics	Battelle Columbus	29230	?
8	EPRI-240	EPRI-240	EPRI-240	EPRI-240									EPRI-3568		EPRI-3568	NP-2307LD,EPRI NP-214	3568, EPRI NP-	EPRI NP-3568, EPRI NP-2307LD,EPRI NP-214	3568, EPRI NP-	EPRI NP-3568	
9	GE	16.9	corner-MOX	Gd in U	GE	?(Vallecitos)	?	cold pressed, mechanically mixed powder	A,S,D,U	Zr-2	No-but TIPS nearby	?	2980	?	4060	gamma scan-power dist,	4	burnup and transuranics	Battelle Columbus	12600	?
10													EPRI-3568		EPRI-3568	EPRI NP-3568, EPRI NP-2307LD,EPRI NP-	EPRI NP-3568, EPRI	EPRI NP-3568, EPRI NP-2307LD,EPRI NP-214	EPRI NP-3568, EPRI	EPRI NP-3568	
11	GE	20.4	Island	Gd in U	GE	?(Vallecitos)	?	cold pressed, mechanically mixed powder	A,S,D,U	Zr-2	No-but TIPS nearby	15.5	9160	?	?	gamma scan-power dist,	?	NA	NA	31340	?
12												EPRI-3568	EPRI-3568			EPRI NP-3568, EPRI NP-2307LD,EPRI NP-				EPRI NP-3568	
13	GE	20.4	Island	Gd in U	GE	?(Vallecitos)	?	cold pressed, mechanically mixed powder	A,S,D,U	Zr-2	No-but TIPS nearby	14	9160	?	?	gamma scan-power dist,	?	NA	NA	39900	?
14												ILRNC	EPRI-3568			EPRI NP-3568, EPRI NP-2307LD,EPRI NP-				EPRI NP-3568	
15	GE	20.4	Island	Gd in U	GE	?(Vallecitos)	?	mechanically mixed powder	A,S,D,U	Zr-2	No-but TIPS nearby	?	9160	?	?	gamma scan-power dist,	?	NA	NA	31350	?
16													EPRI-3568			EPRI NP-3568, EPRI NP-2307LD,EPRI NP-				EPRI NP-3568	
17	GE	20.4	Island	Gd in U	GE	?(Vallecitos)	?	cold pressed, mechanically mixed powder	A,S,D,U	Zr-2	No-but TIPS nearby	15.5	NA	NA	NA	scan of MOX and uranis bundles across core were taken to obtain core wide measurements	NA	NA	NA	NA	NA
18												ILRNC		ILRNC							
19																					
20																					
21																					
22	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	9.03	18,100	18,100	25,100	neutron radiography, gamma scans,rod length, rod diameter	?	?	?	NA	
23										GESMO-F		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2					
24	GE	NA	?	?	GE	?	?	?	6	Zr-2		11.7	?		7000	visual, gamma scan, neut radiographs, rod diam & length,	?	fission gas release, burnup & Uranium/Plutonium isotopes, cladding samples	Vallecitos	NA	
25										GESMO-F		NEDC-10387	NEDC-10387			NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387		
26	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
27										GESMO-F		GESMO-F									
28	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
29										GESMO-F		GESMO-F									

	AT	AU	AV	AW	AX	AY	AZ	BA	BB	BC
1										
2	U>>>>>	SECOND SET	MEASUREMENTS		>>>>>>>>>	<<<<<<<<<				
3										MISCELLANEOUS >>>
4	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?	Overall Performance notes (failure, crudding etc..)	Miscellaneous info
5										
6										
7	?	gamma scan (EOC 2,3,4), rod length, visual	?	?	?	Y	NA	?	?	Annular & solid fuels tested
8		EPRI NP-3568								
9	?	gamma scan (EOC 2,3,4), rod length, visual, profilometry, oxide thickness	?	?	?	Y	NA	?	?	Annular & solid fuels tested
10		EPRI NP-3568								
11	39330	gamma scan (EOC 2,3,4), rod length, visual, neutrography-burnup gamma (EOC5)	2	fission gas release at EOC5	Battelle Columbus	Y	NA	Nat. Lead Ind. 1/2 cask to hot cell	?	Annular & solid fuels tested
12	EPRI NP-3568	EPRI NP-3568		EPRI NP-3568	EPRI NP-3568			EPRI NP-3568		
13	?	length, visual, neutrography-burnup gamma (EOC5), profilometry, oxide thickness	1	fission gas release at EOC5	Battelle Columbus	Y	NA	Nat. Lead Ind. 1/2 cask to hot cell	?	Annular & solid fuels tested
14		EPRI NP-3568		EPRI NP-3568	EPRI NP-3568			EPRI NP-3568		
15	36780	gamma scan (EOC 2,3,4), rod length, visual	?	?	?	Y	NA	?	?	Annular & solid fuels tested
16	EPRI NP-3568	EPRI NP-3568								
17	~57000	5 fission gas measurements taken, 3 MOX rods, the other 2 were on uranium rods	3	NA	NA	Y-however, some GE prop. data may exist	Prompt transient failure tests at SPERT&Vallecito s & Pu particle size effects-	?	Fission gas measurements showed that solid pellets performed better than annular. No fuel failures reported	Annular & solid fuels tested. Gd depletion (in urania) which was near MOX rods was tested at four axial levels. Some urania rods subjected to fission gas release
18	ILRNC						EPRI NP-3568		EPRI NP-3568	EPRI NP-2307
19										
20										
21										
22	NA	NA	NA	NA	NA	Very little public information	NA	NA	No unique problems identified for solid, dished, or annular pellets	Non-dest exam documented in NEDO-12552, do not know if dest exam conducted
23									EPRI 72-2	
24	NA	NA	NA	NA	NA	Y	?	spent fuel in AEC-owned TREAT-2 cask	?	
25								NEDC-10387		
26	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 5 known PIE rods); multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
27										
28	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 5 known PIE rods); multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
29										

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	X
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<	___	___	REACTOR SPECIFICS				>>>>>>>>	<<<<<<<	___	___	ITEM ASSY MOX DESIGN AN					FUEL	ISOTOPICS			___
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																							
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/(U+Pu) in ITEM (Max if multi blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)
30	2.1	6	R	Big Rock Pt	EEI/EPRI	BWR	1969	1970	7	NA	DY0005	9X9	1	?	?	1.40	1	90	?	87.33	9.92	2.49	0.27	N
31	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			NEDC-10387	EPRI 72-2	NEDC-10387		NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387
32	2.1	6	R	Big Rock Pt	EEI/EPRI	BWR	1969	1973	7	NA	DY0006	9X9	1	?	?	1.40?	1	90	?	?	?	?	?	N
33	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			EPRI 72-2	EPRI 72-2	EPRI 72-2						EPRI 72-2
34	2.1	7	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0007	9X9	1	?	?	?	1	90	?	?	?	?	?	N
35	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
36	2.1	8	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0008	9X9	1	?	?	?	1	90	?	?	?	?	?	N
37	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
38	2.1	9	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0009	9X9	1	?	?	?	1	90	?	?	?	?	?	N
39	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
40	2.1	10	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0010	9X9	1	?	?	?	1	90	?	?	?	?	?	N
41	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
42	2.1	11	R	Big Rock Pt	EEI/EPRI	BWR	1969	1970	7	NA	DY0011	9X9	1	?	?	1.51	1	90	?	87.42	9.84	2.47	0.27	N
43	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			NEDC-10387	EPRI 72-2	NEDC-10387		NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387
44	2.1	12	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0012	9X9	1	?	?	?	1	90	?	?	?	?	?	N
45	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
46	2.1	13	R	Big Rock Pt	EEI/EPRI	BWR	1969	1973	7	NA	DY0013	9X9	1	?	?	1.51?	1	90	?	?	?	?	?	N
47	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			EPRI 72-2	EPRI 72-2	EPRI 72-2						EPRI 72-2
48	2.1	14	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0014	9X9	1	?	?	?	1	90	?	?	?	?	?	N
49	REF*				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
50	2.1	15	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0015	9X9	1	?	?	?	1	90	?	?	?	?	?	N
51	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
52	2.1	16	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0016	9X9	1	?	?	?	1	90	?	?	?	?	?	N
53	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
54	2.1	17	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0017	9X9	1	?	?	?	1	90	?	?	?	?	?	N
55	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
56	2.1	18	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0018	9X9	1	?	?	?	1	90	?	?	?	?	?	N
57	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
58	2.1	19	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0019	9X9	1	?	?	?	1	90	?	?	?	?	?	N
59	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2

	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK	AL	AM	AN	AO	AP	AQ	AR	AS
2				>>>>>>	<<<	FABRICATION ASPECTS					>>>>	Max LHGR	<<<FIRST SET BU>>>>>			<<<	FIRST SET MEASUREMENTS		>>>>>>>>	<<<SECOND SET B	
3																					
4	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any Instrument.	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT
30	GE	NA	?	?	GE	?	?	?	D	Zr-2		11.7	?		7000	visual, gamma scan, neut radiographs, rod diam & length,	1	fission gas release, burnup & Uranium/Plutonium isotopes	Vallecitos	NA	
31										GESMO-F		NEDC-10387		NEDC-10387			NEDC-10387	NEDC-10387	NEDC-10387		
32	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	8.91	17,900	17,900	26,200	neutron radiography, gamma scans,rod length, rod diameter	?	?	?	NA	
33										GESMO-F		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2					
34	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
35										GESMO-F		GESMO-F									
36	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
37										GESMO-F		GESMO-F									
38	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
39										GESMO-F		GESMO-F									
40	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
41										GESMO-F		GESMO-F									
42	GE	NA	?	?	GE	?	?	?	A (.1"ID)	Zr-2		11.70	?	?	7000	visual, gamma scan, neut radiographs, rod diam & length,	1	fission gas release, burnup & Uranium/Plutonium isotopes	Vallecitos	NA	
43										GESMO-F		NEDC-10387		NEDC-10387		NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387		
44	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
45										GESMO-F		GESMO-F									
46	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	10.85	20,700	20,700	28,000	neutron radiography, gamma scans,rod length, rod diameter	?	?	?	NA	
47										GESMO-F		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2					
48	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
49										GESMO-F		GESMO-F									
50	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
51										GESMO-F		GESMO-F									
52	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
53										GESMO-F		GESMO-F									
54	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
55										GESMO-F		GESMO-F									
56	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
57										GESMO-F		GESMO-F									
58	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
59										GESMO-F		GESMO-F									

	AT	AU	AV	AW	AX	AY	AZ	BA	BB	BC
2	U>>>>>	SECOND SET	M E A S U R E M E N T S		>>>>>>>> > >	<<<<<<<<<				MISC ELLANEOUS >>>
4	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?	Overall Performance notes (failure, crudding etc..)	Miscellaneous info
30	NA	NA	NA	NA	NA	Y	?	spent fuel in AEC-owned TREAT-2 cask	?	
31								NEDC-10387		
32	NA	NA	NA	NA	NA	Very little public information	NA	NA	No unique problems identified for solid, dished, or annular pellets	Non-dest exam documented in NEDO-12552, do not know if dest exam conducted
33									EPRI 72-2	
34	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
35										
36	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
37										
38	?	?	?	?	?	?	?	?	Severe Crud spalling and external clad corrosion prevent further irradiation	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
39									EPRI 72-2	
40	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
41										
42	NA	NA	NA	NA	NA	Y	?	Spent fuel in AEC-owned TREAT-2 cask	?	
43								NEDC-10387		
44	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
45										
46	NA	NA	NA	NA	NA	Very little public information	NA	NA	No unique problems identified for solid, dished, or annular pellets	Non-dest exam documented in NEDO-12552, do not know if dest exam conducted
47									EPRI 72-2	
48	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
49										
50	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
51										
52	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
53										
54	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
55										
56	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
57										
58	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 8 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000MwdMT for many rods-projected by EPRI 72-2
59										

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	X
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<			REACTOR SPECIFICS				>>>>>>>>	<<<<<<<			ITEM ASSY MOX DESIGN AN					FUEL	ISOTOPICS			
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ####-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																							
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/ (U+Pu) in ITEM (Max if mult blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-236	%Pu-238	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)
60	2.1	20	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0020	9X9	1	?	?	?	1	90	?	?	?	?	?	N
61	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
62	2.1	21	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0021	9X9	1	?	?	?	1	90	?	?	?	?	?	N
63	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
64	2.1	22	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0022	9X9	1	?	?	?	1	90	?	?	?	?	?	N
65	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
66	2.1	23	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0023	9X9	1	?	?	?	1	90	?	?	?	?	?	N
67	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
68	2.1	24	R	Big Rock Pt	EEI/EPRI	BWR	1969	1970	7	NA	DY0024	9X9	1	?	?	1.72	1	90	?	87.48	9.8	2.45	0.27	N
69	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			NEDC-10387	EPRI 72-2	NEDC-10387		NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387
70	2.1	25	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0025	9X9	1	?	?	?	1	90	?	?	?	?	?	N
71	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
72	2.1	26	R	Big Rock Pt	EEI/EPRI	BWR	1969	1973	7	NA	DY0026	9X9	1	?	?	1.72?	1	90	?	?	?	?	?	N
73	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2			EPRI 72-2	EPRI 72-2	EPRI 72-2						EPRI 72-2
74	2.1	27	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0027	9X9	1	?	?	?	1	90	?	?	?	?	?	N
75	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
76	2.1	28	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0028	9X9	1	?	?	?	1	90	?	?	?	?	?	N
77	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
78	2.1	29	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0029	9X9	1	?	?	?	1	90	?	?	?	?	?	N
79	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
80	2.1	30	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0030	9X9	1	?	?	?	1	90	?	?	?	?	?	N
81	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
82	2.1	31	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0031	9X9	1	?	?	?	1	90	?	?	?	?	?	N
83	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
84	2.1	32	R	Big Rock Pt	EEI/EPRI	BWR	1969	1977	7	NA	DY0032	9X9	1	?	?	?	1	90	?	?	?	?	?	N
85	REF				EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2				EPRI 72-2	EPRI 72-2						EPRI 72-2
86																								
87			BRP-3 BUNDLE EEI																					
88																								
89	2.2	1	A	Big Rock Pt	EEI/EPRI	BWR	1970	1973	6	1	CE-P1	9X9	68	?	?	9.1	5	90/80	0.10/0.27	80.92/75.36	10.16/18.24	2.53/4.96	0.26/1.18	N

	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK	AL	AM	AN	AO	AP	AQ	AR	AS
2	—	—	—	>>>>>>	<<<	FABRICATION ASPECTS				—	>>>>	Max LHGR	<<<FIRST SET BU>>>>>			<<<	FIRST SET MEASUREMENTS		>>>>>>>>	<<<SECOND SET B	
4	Fuel Designer	% of MOX in ITEM (red %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular,S-solid, D-dished,U-undished)	Cladding utilized	Any Instrument.	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT
60	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
61										GESMO-F		GESMO-F									
62	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
63										GESMO-F		GESMO-F									
64	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
65										GESMO-F		GESMO-F									
66	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
67										GESMO-F		GESMO-F									
68	GE	NA	?	?	GE	?	?	?	A (.2"ID)	Zr-2		11.70	?	?	7000	visual, gamma scan, neut radiographs, rod diam & length.	?	fission gas release, burnup & Uranium/Plutonium isotopes	Vallecitos	NA	
69										GESMO-F		NEDC-10387	NEDC-10387			NEDC-10387	NEDC-10387	NEDC-10387	NEDC-10387		
70	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
71										GESMO-F		GESMO-F									
72	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	10.77	21,700	21,700	30,200	fission radiography, gamma scans,rod length, rod diameter	?	?	?	NA	
73										GESMO-F		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2					
74	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
75										GESMO-F		GESMO-F									
76	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
77										GESMO-F		GESMO-F									
78	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
79										GESMO-F		GESMO-F									
80	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
81										GESMO-F		GESMO-F									
82	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
83										GESMO-F		GESMO-F									
84	GE	NA	?	?	GE	?	?	?	?	Zr-2	?	5->15?	?	?	?	?	?	?	?	?	
85										GESMO-F		GESMO-F									
86																					
87																					
88																					
89	GE	84	Octant sym. Pattern	Gd	?	oxalate -derived	"Eldorado"?	cold pressed sintered from mechanically blended powders	A	Zirc-2	?	?	15,416	?	?	visual insp.slipped	?	?	?	NA	NA

	AT	AU	AV	AW	AX	AY	AZ	BA	BB	BC
2	U>>>>>	SECOND SET	MEASUREMENTS		>>>>>>> > >	<<<<<<<<<				MISCELLANEOUS >>>
3										
4	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?	Overall Performance notes (failure, crudding etc..)	Miscellaneous Info
60	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
61										
62	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
63										
64	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
65										
66	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
67										
68	NA	NA	NA	NA	NA	Y	?	spent fuel in AEC-owned TREAT-2 cask	?	
69								NEDC-10387		
70	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
71										
72	NA	NA	NA	NA	NA	Very little public information	NA	NA	No unique problems identified for solid, dished, or annular pellets	Non-dest exam documented in NEDO-12552, do not know if dest exam conducted
73									EPRI 72-2	
74	?	?	?	?	?	?	?	?	Severe Clad Spalling and external clad corrosion prevent further irradiation	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
75									EPRI 72-2	
76	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
77										
78	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
79										
80	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
81										
82	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
83										
84	?	?	?	?	?	?	?	?	?	GENERIC data used for 24 rods (there are 6 known PIE rods), multiple burnup reconstitutions, 2 different annular fuels tested:~20,000Mwd/MT for many rods-projected by EPRI 72-2
85										
86										
87										
88										
89	NA	NA	NA	NA	NA	No	?	spent fuel shipped in NFS (M100) cask	Failed Rods in the bundle, spalling reported	Cobalt rods were placed for dual purpose: make Co60 and depress power in corners

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	X
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<	___	___	REACTOR SPECIFICS				>>>>>>>>	<<<<<<<<	___	___	ITEM ASSY MOX DESIGN AN					FUEL	ISOTOPICS			___
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																							
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/ (U+Pu) in ITEM (Max if multi blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)
90								EPRI 72-2	EPRI 72-2			WALKE-1	WALKE-1			EPRI 72-2	WALKE-1	EPRI 72-2	WALKE-1	WALKE-1	WALKE-1	WALKE-1	WALKE-1	EPRI 72-2
91	2.2	2	A	Big Rock Pt	EEI/EPRI	BWR	1970	1973	6	1	CE-P2	9X9	68	?	?	9.1	5	90/80	0.10/0.27	86.92/75.36	10.16/18.24	2.53/4.96	0.28/1.18	N
92								EPRI 72-2	EPRI 72-2			WALKE-1	WALKE-1			EPRI 72-2	WALKE-1	EPRI 72-2	WALKE-1	WALKE-1	WALKE-1	WALKE-1	WALKE-1	EPRI 72-2
93	2.2	3	A	Big Rock Pt	EEI/EPRI	BWR	1970	1974	8	1	CE-P3	9X9	68	?	?	9.1	5	90/80	0.10/0.27	86.92/75.36	10.16/18.24	2.53/4.96	0.28/1.18	N
94								EPRI 72-2	EPRI 72-2			WALKE-1	WALKE-1			EPRI 72-2	WALKE-1	EPRI 72-2	WALKE-1	WALKE-1	WALKE-1	WALKE-1	WALKE-1	EPRI 72-2
95																								
96		BRP-EXXON-2 9X9 BUNDLE																						
97																								
98	2.3	1	A	Big Rock Pt	Private?	BWR	1972	?	?	1	?	9X9	?	?	?	?	?	?	?	?	?	?	?	?
99							GESMO-F					GESMO-F												
100																								
101	2.3	2	A	Big Rock Pt	Private?	BWR	1972	?	?	1	?	9X9	?	?	?	?	?	?	?	?	?	?	?	?
102							GESMO-F					GESMO-F												
103																								
104		BRP-EXXON-2 11X11 BUNDLE																						
105																								
106	2.4	1	A	Big Rock Pt	Private?	BWR	1973	?	?	1	?	11X11	?	?	?	?	?	?	?	?	?	?	?	?
107							GESMO-F					GESMO-F												
108																								
109	2.4	2	A	Big Rock Pt	Private?	BWR	1973	?	?	1	?	11X11	?	?	?	?	?	?	?	?	?	?	?	?
110							GESMO-F					GESMO-F												
111																								
112		BRP-EXXON-COMMERCIAL Reload No. 1																						
113																								
114	2.5	1	B	Big Rock Pt	Private?	BWR	1974	?	?	16	?	11X11	432	?	?	?	?	?	?	?	?	?	?	?
115							GESMO-F			GESMO-F		GESMO-F	GESMO-F											
116																								
117		BRP-EXXON-COMMERCIAL Reload No. 2																						
118																								
119	2.6	1	B	Big Rock Pt	Private?	BWR	1976	?	?	8	?	11X11	192	?	?	?	?	?	?	?	?	?	?	?
120							GESMO-F			GESMO-F		GESMO-F	GESMO-F											
121																								
122		BRP-NFS-4 BUNDLE																						
123																								
124	2.7	1	A	Big Rock P	Private?	BWR	1973	?	?	1	?	?	73	?	?	?	?	?	?	?	?	?	?	?
125							GESMO-F			GESMO-F														
126																								
127	2.7	2	A	Big Rock P	Private?	BWR	1973	?	?	1	?	?.	73	?	?	?	?	?	?	?	?	?	?	?

	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK	AL	AM	AN	AO	AP	AQ	AR	AS
2				>>>>>>	<<<	FABRICATION ASPECTS					>>>>	Max LHGR	<<<FIRST SET BU>>>>>			<<<		FIRST SET MEASUREMENTS	>>>>>>>>	<<<SECOND SET B	
3																					
4	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular,S-solid, D-dished,U-undished)	Cladding utilized	Any instrument.	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT
90		WALKE-1	WALKE-1	WALKE-1		WALKE-2	WALKE-2		EPRI 72-2	EPRI 72-2			EPRI 72-2			EPRI 72-2					
91	GE	84	Octant sym. Pattern	Gd	?	oxalate -derived	"Eldorado"?	cold pressed sintered from mechanically blended powders	A	Zirc-2	?	14.7	16,712	19,900	24,700	visual insp,slipped, neutron radiography, gamma scans,rod length, rod diameter-7 rods	?	?	?	NA	NA
92		WALKE-1	WALKE-1	WALKE-1		WALKE-2	WALKE-2		EPRI 72-2	EPRI 72-2		EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2	EPRI 72-2					
93	GE	84	Octant sym. Pattern	Gd	?	oxalate -derived	"Eldorado"?	cold pressed sintered from mechanically blended powders	A	Zirc-2	?	?	15,026	?	?	visual insp,slipped	?	?	?	17,491	?
94		WALKE-1	WALKE-1	WALKE-1		WALKE-2	WALKE-2		EPRI 72-2	EPRI 72-2			EPRI 72-2			EPRI 72-2				EPRI 72-2	
95																					
96																					
97																					
98	Exxon?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
99	GESMO-F																				
100																					
101	Exxon?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
102	GESMO-F																				
103																					
104																					
105																					
106	Exxon?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
107	GESMO-F																				
108																					
109	Exxon?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
110	GESMO-F																				
111																					
112																					
113																					
114	Exxon?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
115	GESMO-F																				
116																					
117																					
118																					
119	Exxon?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
120	GESMO-F																				
121																					
122																					
123																					
124	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
125																					
126																					
127	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?

	AT	AU	AV	AW	AX	AY	AZ	BA	BB	BC
2	U>>>>>	SECOND SET	MEASUREMENTS		>>>>>>>>>	<<<<<<<<<				MISCELLANEOUS >>>
3										
4	(2nd Set) Peak MOX Pellet MWd/MT	(2nd Set) Non-dest exams performed - description	(2nd Set) No. MOX rods PIE	(2nd Set) summary of PIE measurement	(2nd Set) Where PIE performed	Was any data made public?	Beyond design basis/transient tests performed? If so, which ones.	How transported to RX and to PIE?	Overall Performance notes (failure, crudding etc..)	Miscellaneous info
90								VOLL-1	EPRI 72-2	
91	NA	NA	NA	NA	NA	No	?	spent fuel shipped in NFS (M100) cask	Failed MOX rods in the bundle, spalling, Axial fission product migration toward cooler upper and lower ends of rods for annular pellets in excess of 2wt% PuO2	Cobalt rods were placed for dual purpose: make Co60 and depress power in corners. PIE information provided in NEDO-12552, GE Proprietary
92								VOLL-1	EPRI 72-2	EPRI 72-2
93	?	slpped	NA	NA	NA	No	?	spent fuel shipped in NFS (M100) cask	Spalling reported, assy was sound upon final disharge from the core	Cobalt rods were placed for dual purpose: make Co60 and depress power in corners. Scrape of oxide nodule from two rods caused a breach. NOTE: This bundle was reconstituted at the end of cycle 10 and put in for an additional cycle using 2 rods from CEP1. This lowers the discharge burnup (MwdA/T)
94								VOLL-1	EPRI 72-2	EPRI 72-2, VOLL-1
95										
96										
97										
98	?	?	?	?	?	No	?	?	?	Expected to reach 20GWd/t, variety of non-destructive and destructive exams were planned, do not know if they were actually done
99										VOLL-1, GESMO-F
100										
101	?	?	?	?	?	No	?	?	?	Expected to reach 20GWd/t, variety of non-destructive and destructive exams were planned, do not know if they were actually done
102										VOLL-1, GESMO-F
103										
104										
105										
106	?	?	?	?	?	No	?	?	?	Expected to reach 20GWd/t, variety of non-destructive and destructive exams were planned, do not know if they were actually done, prelude to all MOX reload-Some short pellets tested, experiments aimed at fuel densification and PCI problems
107										VOLL-1, GESMO-F
108										
109	?	?	?	?	?	No	?	?	?	Expected to reach 20GWd/t, variety of non-destructive and destructive exams were planned, do not know if they were actually done, prelude to all MOX reload-Some short pellets tested, experiments aimed at fuel densification and PCI problems
110										VOLL-1, GESMO-F
111										
112										
113										
114	?	?	?	?	?	No	?	?	No operational or planned increase in primary system coolant activity from urania cores was noted due to use of MOX	First U.S. commercial scale MOX reload-which brought BRP to the licensed limit of 50kg of plutonium
115									VOLL-1, GESMO-F	VOLL-1, GESMO-F
116										
117										
118										
119	?	?	?	?	?	No	?	?	?	Little information was found concerning this batch
120										VOLL-1, GESMO-F
121										
122										
123										
124	?	?	?	?	?	No	?	?	?	Expected to reach 20GWd/t, no plans for destructive examination
125										VOLL-1, GESMO-F
126										
127	?	?	?	?	?	No	?	?	?	Expected to reach 20GWd/t, no plans for destructive examination

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	X
2	<<<<<<	ITEM	>>>>>>>>>>>>>>	<<<<<	___	___	REACTOR SPECIFICS				>>>>>>>>	<<<<<<<	___	___	ITEM ASSY MOX DESIGN AN					FUEL	ISOTOPICS			___
3	NOTE: blank-irrad was not mapped, ?-no data found in refs for this column, ##?##-judgment made or incomplete data, NA-does not apply (or thought not to be performed) or nothing to report for (IS) summary																							
4	LINE ITEM (plant.batch)	IRRADIATION ITEM (in the batch)	ITEM DESCRIPTION B-Batch of Assys, A-Single Assembly or group of rods, R-Individual Rod, IS-Irradiation Summary (a commentary)	Reactor & Unit	Project Sponsors	Reactor Type	Year First Insertion	Year of Final Discharge	Reactor Cycle Inserted	No. of MOX Assys in this ITEM (NA if rod)	Batch, Assy or Rod Name	Assy Design	TOTAL No. of MOX rods in ITEM	Mass of MOX in ITEM	Mass of Pu in ITEM	Pu total % Pu/ (U+Pu) in ITEM (Max if multi blends)	No. of MOX pin types/blends in ITEM	Pu Wt % Fissile	%Pu-238	%Pu-239	%Pu-240	%Pu-241	%Pu-242	Diluent Uranium (D or N)
128							GESMO-F			GESMO-F														
129																								
130	2.7	3	A	Big Rock P	Private?	BWR	1973	?	?	1	?	?	73	?	?	?	?	?	?	?	?	?	?	?
131							GESMO-F			GESMO-F														
132																								
133	2.7	4	A	Big Rock P	Private?	BWR	1973	?	?	1	?	?	73	?	?	?	?	?	?	?	?	?	?	?
134							GESMO-F			GESMO-F														
135																								
136		Dresden Unit No. 1-11 bundles																						
137																								
138	3.1	1	B	Dresden-1	GUNC/ComEd	BWR	1969	?	7	11	TypeVII-Pu	6X6	99	?	?	2.90%	1	77.4	0.4	71.3	20.6	6.1	1.6	N
139					GESMO-F		GU-5304		GESMO-F		BLISS-1	GU-5309	GESMO-F			GESMO-F	GESMO-	GESMO-F	GESMO-F	GESMO-F	GESMO-F	GESMO-F	GESMO-F	BLISS-1
140																								
141	.	Dresden Unit No. 1- 4 rod irradiation																						
142	3.2	1	B	Dresden-1	EEVGE	BWR	1967?	?	5?	4	Type III-F	?	4	?	?	1.70%	3	80	?	75	19	5	1	N
143							HUGHES-1		HUGHES-1		HUGHES-1					HUGHES-1	HUGHES	HUGHES-1		HUGHES-1	HUGHES-1	HUGHES-1	HUGHES-1	HUGHES-1
144																								
145			END OF COMMERCIAL BWRS																					
146																								
147		ADDITIONAL IRRADIATIONS TO BE MAPPED:																						
148																								
149		Experimental Boiling Water Reactor (Test Rx)																						
150																								

	Y	Z	AA	AB	AC	AD	AE	AF	AG	AH	AI	AJ	AK	AL	AM	AN	AO	AP	AQ	AR	AS
2	—	—	—	>>>>>>	<<<	FABRICATION ASPECTS				—	>>>>	Max LHGR	<<<FIRST SET BU>>>>>			<<<		FIRST SET MEASUREMENTS	>>>>>>>>	<<<SECOND SET B	
3																					
4	Fuel Designer	% of MOX in ITEM (rod %)	Rod Pattern Desc.	B. Poison (in U of assy)	Fuel Bundler (where)	PuO2 conversion process (where)	UO2 conversion process (where)	Pellet Fabrication process	Pellet Types (A-annular, S-solid, D-dished, U-undished)	Cladding utilized	Any instrument.	Max LHGR (kW/ft)	(1st Set) Avg Assy MWd/MT (single or if batch-Max assy)	(1st Set) Peak MOX rod MWd/MT	(1st Set) Peak MOX Pellet MWd/MT	(1st Set) Non-dest. exams performed-description	(1st Set) No. MOX rods PIE	(1st Set) summary of PIE measurement	(1st Set) Where PIE performed	(2nd Set) Avg Assy MWd/MT (single or if batch-Max assy)	(2nd Set) Peak MOX rod MWd/MT
128																					
129																					
130	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
131																					
132																					
133	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?	?
134																					
135																					
136																					
137																					
138	Gulf United?	25	central MOX island	Gd	?	Powder from AEC	NUKEM Natural powder	Master blend, cold form pellets, sintered, wet centerless ground	?	Zr-2	2 in MOX Assy, 4 adjoining UO2	14	17,470	?	22,830	at end of cyc 8, visual, all 11 assys slp tested for fission gas leakage, 6 were leakers, both UO2 and MOX rods	?	?	?	>19440?	?
139	GESMO-F	BLISS-1	GU-5309	BLISS-1		GU-5309	GU-5308	GU-5308				GU-5309	GESMO-F		GESMO-F	GESMO-F				GA-A13813	
140																					
141																					
142	GE?	?	?	?	?	?	?	hot pressed	?	Zr-2	?	?	?	?	?	?	?	?	?	?	?
143		HUGHES-1						HUGHES-1		HUGHES-1											
144																					
145																					
146																					
147																					
148																					
149																					
150																					

REFERENCE LIST

REF KEY	WHERE FOUND	FULL REFERENCE
	T-Texas Database P-Paper (also fiche) at ORNL	
		Addition to Original Bibliographic List Addition and/or Consolidation of bibliographic series report into a sequential listing
AKUTSU-1	P-only	Akutsu, H and M. Koizumi. "Irradiation Performance Test of PuO2-UO2 Fuels by Saxton Reactor" , IAEA-175, Plutonium Utilization in Thermal Power Reactors, IAEA conference i
BLISS-1	P	Bliss, H. E. "Proposed Change No. 22 to Appendix A, DPR-2, to Permit Operation of Cycle 8." NRC Docket 50010-77, (Jan. 18, 1971).
EPRI-72-2	P	Electric Power Research Institute. Use of Plutonium Fuel in Boiling Water Reactors, EPRI 72-2. Electric Power Research Institute, Palo Alto, (1975).
EPRI-240	P	Electric Power Research Institute. Core Design and Operating Data for Cycle 1 and 2 of Quad Cities 1, NP-240. Electric Power Research Institute, Palo Alto, (1976).
EPRI-NP2307	P	Electric Power Research Institute. Burnup and Transuranium Element Composition in Irradiated UO2, UO2-Gd2O3, and UO2-PuO2 Rods from the Quad Cities-1 Reactor, NP-23
EPRI-3568	P	General Electric Co. Quad Cities Nuclear and Fuel Performance Measurements, EPRI NP-3568. Project 497-1, Final Report. Electric Power Research Institute and General Ele
WCAP-4167-2	P, T	Haley, J. EEI-Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period Ending December, 1970, WCAP 4167-2. (1971).
GU-5308	T	Hower, L. D., Jr. Mixed Oxide Fuel Pellet Fabrication and Fuel Rod Assembly for Dresden I, Batch 7 Reload, GU-5308. Gulf United Nuclear Corp., (Jan. 1974).
HUGHES-1	P	Hughes, J. H. "Request for Amendment of Operating License DPR-2, as Amended, Dresden Nuclear Power Station, Including Exhibit I." NRC Docket 50010-83. (Jan. 9, 1967).
GU-5304	T	Krysinski, T. L. Dresden I Plutonium Recycle Program, Semi-Annual Report for Period Ending 6/30/73, GU-5304. Gulf United Nuclear Corp., (Jun. 1973).
GU-5309	T	Krysinski, T. L. Dresden I Plutonium Recycle Program, Semi-Annual Report for Period Ending 12/31/73, GU-5309. Gulf United Nuclear Corp., (Jan. 1974).
WCAP-4167-4	P	Melehan, J. B. EEI - Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period 10/71 - 3/72, WCAP-4167-4, (Dec. 1972).
MOMSEN-1	P	Momsen, Bruce F., "An Analysis of Plutonium Recycle Fuel Elements in San Onofre - I", Submitted in Partial Fulfillment of the Requirements for the Degrees of Master of Science
NEDC-10387	P	Ogawa, S. Y. and D. L. Zimmerman. Plutonium Utilization in Boiling Water Reactors Phase II: Post-Irradiation Examination of Four Fuel Rods After One Cycle of Operation in Big
ORTIZ-1	P	Ortiz, Jaime (RG&E). "Fax to S. E. Fisher, Lead Test Assemblies Design Parameters", 2/18/97.
SHEA-1	P	Shea, J. J. "Summary of Meeting Held on November 1, 1979." Letter to Rochester Gas & Electric Company (RG&E) dated Nov. 16, 1979. Nuclear Regulatory Commission (197
GESMO-F		U. S. Nuclear Regulatory Commission. Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors, NUREG-0
VOIGT-1	P	Voigt, H. H.,"Re: Rochester Gas and Electric Corporation R. E. Ginna Nuclear Power Plant, Unit No. 1 Docket No. 50-244", Letter w/enclosures to Harold Denton (NRC), Decemb
VOLL-1	P	Voll, R. W. "Plutonium Recycle R&D and Operating Experience at Big Rock Point." Proc. Topical Mtg. on the Plutonium Fuel Cycle, Bal Harbour, Florida, May 2-4, 1977. ANS-F
WALKE-1	P	Walke, G. J. "Request for Change to the Technical Specifications License No. DPR-6." NRC Docket 50155-16. (Dec. 22, 1969).
WALKE-2	P	Walke, G. J. "Additional Information: Change No. 19, Request for Change to the Technical Specifications License No. DPR-6." NRC Docket 50155-22. (Jan. 28, 1970).
WCAP-4167-6	P	Wallace, T. W. EEI - Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period 4/73 - 9/73, WCAP-4167-6. (Dec. 1973).
WCAP-4167-7	P	Wallace, T. W. EEI - Westinghouse Plutonium Recycle Demonstration Program, WCAP-4167-V7. (1975).
GA-A13813	T	Wan, M. Y. Dresden I Plutonium Recycle Program Report for the period Juanuary 1, 1974 through June 30, 1975, GA-A13813, December 1975.

GU-5314	T	Washington, C.E. "Shipping of Plutonium Dioxide Powder and Mixed Oxide Fuel Rods", GU-5314, Gulf Nuclear Fuels Company, May 1974
WCAP-3385-8	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program: SemiAnnual Progress Report for the Period Ending June 30,1966, WCAP-3385-8, (1966)
WCAP-3385-10	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Semiannual Progress Report for the Period Ending December 31,1966, WCAP-3385-10. (1967).
WCAP-3385-12	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Semi-Annual Progress Report for the Period Ending June 30, 1967, WCAP-3385-12. (1967).
WCAP-3385-16	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Semi-Annual Progress Report for the Period Ending June 30, 1968, WCAP-3385-16. (1968).
WCAP-3385-22	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Semi-Annual Progress Report for the Period Ending December 31,1969, WCAP-3385-22. (1970).
WCAP-3385-23	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Quarterly Progress Report for the Period Ending March 31,1970, WCAP-3385-23. (1970).
WCAP-3385-29	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Quarterly Progress Report for the Period Ending September 30,1971, WCAP-3385-29. (1971).
WCAP-3385-51	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Nuclear Design of the Saxton Partial Plutonium Core, WCAP-3385-51. (1965).
WCAP-3385-53	T,P	Westinghouse Electric Corporation. Saxton Plutonium Program : Materials Design and Fabrication of the Saxton Partial Plutonium Core, WCAP-3385-53. (1965).
WCAP-3385-56	T,P	Westinghouse Electric Corporation. Saxton Core II Fuel Performance Evaluation, WCAP-3385-56 Parts 1 and 2. (1971).
WCAP-4167-1	P	Westinghouse Electric Corporation. EEI-Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period Ending April, 1970, WCAP 4167-1. Westingh

n Karlsruhe, Nov. 1974, (1975)

307-LD. Electric Power Research Institute, Palo Alto, (March 1982).

ctric Co., (July 1984).

in Nuclear Engineering and Nuclear Engineer at the Massachusetts Institute of Technology", May 1974.

Rock Point Reactor, NEDC-10387. General Electric Co., San Jose, (July 1971).

9).

002. Vol. 1-5. U. S. Nuclear Regulatory Commission, (Aug. 1976).

er 20,1979.

uel Cycle Division (1977).

ıouse Electric Corporation (1970).

APPENDIX B

MOX EXPERIENCE EXCERPT FROM GESMO

Page intentionally blank

3.0 PLUTONIUM RECYCLE IN LWR's

3.1 Development and Testing of Mixed Oxide Fuels

The initial development of technology for plutonium recycle in LWR fuel was sponsored by the USAEC, with follow-on programs financed by utility companies and nuclear reactor manufacturers; in some cases, programs had joint sponsorship. Development of the technology of plutonium recycle in reactor fuels began with the AEC sponsored Plutonium Utilization Program (PUP) at Hanford in 1956, and is continuing, mainly with mixed oxide fuel performance demonstrations in LWR's. After supporting the PUP program at Hanford and the Saxton MOX fuel development and testing program, the U.S. Government concluded that further development of plutonium recycle technology could be carried out by industry.

*d/m.g. = disintegrations/minute/gram

The Federal government-supported research and development program on plutonium recycle was essentially completed by the year 1970, with only a small program wrapup phase extending to 1972. Major industry programs were initiated in the year 1967 with the Edison Electric Institute supporting mixed oxide fuel development and testing performed by Westinghouse and General Electric, followed by the mixed oxide fuel performance demonstration programs in commercial reactors. As early as the year 1959, demonstrations of plutonium recycle were also initiated in foreign reactors. Each of the major programs carried out to establish the viability of plutonium recycle in LWR's is discussed below.

As a result of the experience acquired and the technology developed in various plutonium recycle programs, both in the United States and abroad, it has been demonstrated that plutonium recycle is technically feasible. This conclusion is based on successful irradiations of fuel in the Plutonium Recycle Test Reactor at Hanford, and in the Saxton, San Onofre, Big Rock Point, and Dresden Unit No. 1 U.S. reactors. Foreign experiments have involved tests of mixed oxide fuel in a number of reactors, but especially at Garigliano in Italy. The mixed oxide fuels were irradiated to specific power levels and to burnups typical of those expected in LWR's. The irradiations showed no abnormalities with respect to fuel behavior or predicted reactor control and core performance characteristics.

3.1.1 Plutonium Utilization Program

The Plutonium Utilization Program (PUP) sponsored by the AEC at its Battelle Pacific Northwest Laboratories (PNL) in Richland, Washington, to develop the technology for plutonium recycle in thermal reactors, began in the year 1956, about one year before the first demonstration nuclear plant began operation at Shippingport, Pennsylvania, in 1957.

It was not known in the 1950's what type of nuclear power reactors would dominate the commercial market or what type of fuel would be used; therefore, a great deal of the effort of the Plutonium Utilization Program was devoted to development and testing of fuels other than the mixed oxide pellet type which, if plutonium recycle proceeds, would be used in LWR's.

As a part of PUP, the 70 MWth Plutonium Recycle Test Reactor (PRTR) was built at PNL (formerly the Hanford Laboratory) for fuel performance tests; operating characteristics are shown in Table II-3. The PRTR was a heavy water moderated and cooled reactor with 85 vertical pressure tubes which contained individual fuel assemblies. It also included a Fuel Element Rupture Test Facility (FERTF) which was a test loop with a separate light water cooling system to be used for conducting high risk experiments with elements having intentional defects.

The major efforts in PUP were concentrated on development of mixed oxide fuels, their irradiation in the PRTR, and experimental and calculational neutronics studies. Other efforts included studies of chemical reprocessing, economic optimization, and reactor decontamination.

Table II-3

OPERATING CONDITIONS FOR THE PLUTONIUM RECYCLE TEST REACTOR

Reactor Operating Pressure	1050 psi
Coolant Surface Velocity*	15 ft/sec
Inlet Coolant Temperature	235°C (455°F)
Outlet Coolant Temperature	275°C (527°F)
Coolant pH	6.0 to 7.5
Peak Linear Heat Rating*	20.1 kW/ft (464,000 $\frac{Btu}{hr\ ft}$)
Axial Peak-to-Average Power Ratio	1.27
Average Linear Heat Generation Rate of Maximum Rod*	16.1 kW/ft
Maximum Allowable Fuel Temperature	Incipient melting
Design Peak Burnup	35,000 MWd/MTHM
Film Coefficient (Calculated Value for 20 kW/ft)*	6520 Btu/hr/ft^2°F
Peak Cladding Surface Heat Flux*	475,000 Btu/hr/ft^2
Maximum Allowable Cladding Surface Heat Flux	650,000 Btu/hr/ft^2
Boiling Burnout Ratio*	1.85
Pressure Tube (Inside Diameter)	3.25±.01 in.
Equivalent Diameter*	0.3 in.
Flow Area*	12.11 in^2
Maximum Allowable Tube Power	1800 kW

*HPD PRTR fuel element. Nineteen-rod cluster of .565 in Zircaloy clad rods containing vibrationally compacted UO_2 with 2 wt% PuO_2 fuel.

In the area of fuel development and irradiation, the PUP program was directed almost exclusively toward vibratory packed particles rather than the pellet type UO_2 fuels which later came to be exclusively utilized in commercial LWR's. Also, in the earlier part of the program, metallic Pu-Al alloy elements were fabricated and irradiated, but these were of very little economic interest. The large effort on the vibratory packed (Vipac) particle fuel was spurred by what appeared to be potential economic advantages of this method and its adaptability to remote operation. But, as the program progressed, the economic advantage of Vipac fuel appeared marginal. For this reason and because of the good performance and general acceptance of pellet type fuel in commercial power reactors, the fuel fabricators designed their LWR fuel facilities to produce pellet-type fuel; and the Vipac fuel became a possible alternative. Table II-4 summarizes the fuel irradiation experiments performed in the PRTR during the Plutonium Utilization Program. These irradiations included 2 assemblies (38 rods) of mixed oxide hot pressed pellets and 13 individual rods of cold pressed and sintered pellets. The mixed oxide fuel designs tested in PRTR (Zircaloy clad fuel which was heterogeneously and homogeneously enriched) included three chronological phases as noted below. Some types were vibratory compacted and some were swage compacted types of fuel.

The irradiations carried out in the PRTR were classified into three chronological phases:

Phase I - Startup - experimental elements

Phase II - Continuation of tests - modified phase I elements

Phase III - Batch core experiments

In Phase I, which extended from the startup of the Plutonium Recycle Test Reactor in July 1961 to January 1965, a large variety of experimental elements was irradiated. The element types included Al Pu alloy elements; UO_2 elements fabricated by vibratory compaction and by swaging; and heterogeneously enriched (incrementally loaded) and homogeneously enriched mixed oxide fuels fabricated by swaging and by vibratory compaction. Peak burnups of 13,000 MWd/MTHM at peak linear heat rating of 12 kW/ft were attained during Phase I operation. The first plutonium produced in PRTR was recycled back into the reactor as a swage compacted UO_2 -0.5 wt% PuO_2 element in May 1963.

During the Phase I irradiation, 38 mixed oxide rods developed in service defects. With one exception, these defects were attributed to internal gas phase hydriding of the Zircaloy-2 cladding, caused by impurities in the fuel material. Three types of impurities were identified:

- Residual fluoride contamination in the plutonium oxide

- Absorbed moisture in the fuel

Table II-4

FUEL ELEMENTS IRRADIATED IN PLUTONIUM RECYCLE TEST REACTOR

Fuel Element Type	Number of Fuel Elements	Peak Linear Heat Rating (kW/ft)	Reactor Peak Burnup (MWd/MTHM)
Al - Pu	75	15.1	(80% of Pu)
UO_2	68		
Vipac	1	10.1	2,500
Swaged (1 not swaged)	65	14.1	15,300
Vipac Tubular	1	-	1,700
Vipac Inverted Cluster	1	-	170
UO_2 - PuO_2	216		
UO_2 - 0.5 wt% PuO_2	(81)		
Vipac	20	16.0	18,500
Swaged	61	13.0	12,500
UO_2 - 1.0 wt% PuO_2	(49)		
Vipac	16	13.6	11,500
Swaged	33	15.6	13,500
UO_2 - 1.5 wt% PuO_2	(1)	4.4	3,500
UO_2 - 2.0 wt% PuO_2	(84)		
Vipac	79	20.0	13,000
Swaged	2	20.0	7,800
Pellet (hot press)	2	21.6	3,150
Vipac Salt Cycle	1	17.1	1,800
UO_2 - 4.0 wt% PuO_2	(1)	27.0	1,250
Pellets (cold press)			
0.5 wt% PuO_2	9 rods	12.0	11,700
2.0 wt% PuO_2	4 rods	15.7	2,300

- Traces of hydrocarbons (oil) introduced in the fuel by leakage from mechanical processing equipment

Although hydriding of the cladding led to severe localized embrittlement and loss of cladding fragments in some instances, little or no fuel loss into the coolant resulted, and no severe reactor operating difficulties were experienced. By concerted efforts to identify and correct this problem, the impurity induced failures were confined to a short period of time, after which fuel materials of improved quality eliminated the problem.

Irradiation of Phase I first generation elements in PRTR was continued in Phase II. In Phase II (January 1965 through September 1965) irradiation tests were performed on modified design mixed oxide fuel elements which were developed to provide for operation at high power density, high burnups (~20,000 MWd/MTHM), and high linear heat ratings (~20 kW/ft). It was expected that these fuel elements would be used for a full fuel loading in the Batch Core Experiment under Phase III.

During Phase II of PRTR operation, peak burnups of 15,000 MWd/MTHM were attained on Phase I first generation mixed oxide fuel. Also during Phase II operation, peak burnups of about ~6,500 MWd were achieved on prototype high power density (HPD) fuel at peak linear heat ratings of about 21 kW/ft and maximum fuel temperatures above melting. Phase II operation was terminated as a result of the rupture of an intentionally defected mixed oxide element under irradiation in the Fuel Element Rupture Test Facility loop in the PRTR at a peak linear heat rating of about 27 kW/ft with significant fuel melting at the plane of the defect.

The Batch Core Experiment (BCE) was conducted in Phase III, which extended from January 1967 through July 1968. At the start, the PRTR was loaded with 66 fresh HPD UO_2 -2 wt% PuO_2 elements designed to operate at high specific powers to high burnups. The irradiation of selected first generation Phase I and Phase II elements was continued in the fringe positions of the BCE during this experiment.

At the end of Phase III operation, peak burnups of 13,000 MWd/MTHM were achieved on high-power density mixed oxide fuels that operated at nominal maximum peak heat ratings of 19 kW/ft with maximum fuel temperatures near melting. Peak heat ratings as high as 21.4 kW/ft with fuel temperatures above melting were achieved for short periods of time. Peak burnups of about 18,500 MWd/MT were attained on first generation mixed oxide fuels at maximum peak heat ratings of 17.1 kW/ft.

A large number and variety of experimental fuel elements were successfully irradiated in the PRTR to evaluate the irradiation performance of mixed oxide fuels suitable for plutonium utilization in water cooled reactors. Mixed oxide fuels were irradiated to peak burnup levels above 18,000 MWd/MTHM and to linear heat rates about 20 kW/ft.

The behavior of the various experimental mixed oxide fuel types operating under high performance conditions was generally excellent. Fabrication problems associated

with PRTR mixed oxide fuel elements of the first-generation design resulted in fuel rod defects which provided some of the first experience with gas phase hydriding defects in Zircaloy clad oxide fuel rods, and resulted in an improved understanding of the phenomenon. Consequently, improved fuel fabrication techniques were developed and no fuel rod defects occurred in the more advanced vibratory compacted HPD design mixed oxide elements irradiated in PRTR during the BCE.

The PUP placed major emphasis on packed particle fuels, and most of the fuel irradiations in PRTR were not demonstrations representative of the pellet type mixed oxide fuels planned to be utilized in current LWR's. However, it should be noted that the test results all indicated that the MOX pellet fuel currently in use would perform adequately under commercial LWR operating conditions.

Further details, summaries and references on PUP are available in Nuclear Technology (August 1972 and May 1973).[10,11]

3.1.2 Saxton Program

The Saxton Program was carried out by Westinghouse under an AEC contract to supplement the work at PNL and develop information on utilization of mixed oxide fuel in pressurized water reactors. Primary objectives were to

- Perform pilot-scale tests of plutonium enriched fuel in a pressurized water reactor environment
- Compare the performance of mixed oxide fuel fabricated by two economically promising techniques: pelletized versus vibratory compacted (Vipac)
- Obtain nuclear data of interest to plutonium recycling, especially in depletion and generation of transuranic isotopes
- Provide a preliminary basis for selection and design of plutonium fuel for a commercial PWR

The project included design and fabrication of mixed oxide fuel elements, reactor irradiation of the fuel, and post irradiation evaluation. The guidelines for mechanical, thermal, and hydraulic design of the mixed oxide fuel elements were

- 20,000 MWd/MT peak rod average burnup
- 16 kW/ft maximum design heat rating in the rods
- Internal gas pressure at end of design life to be less than external reactor operating pressure
- Fuel rod outside diameter, length, and lattice spacing to be the same as for the standard UO_2 fuel rods

The project was initiated early in the year 1964, and full power operation of the Saxton PWR with standard UO_2 fuel elements began in January 1966. The characteristics of the Saxton Reactor during the period of Core II operation are summarized in Table II-5.

Table II-5

SAXTON CORE II DESIGN OPERATING CONDITIONS

Reactor Type	PWR
Maximum Power Level	23.5 MWt
Maximum Linear Power Density	16 kW/ft
Maximum Heat Flux	531,400 Btu/hr-ft^2
Average Coolant Temperature	277°C (530°F)
System Pressure	2,000 psia
Maximum Clad Surface Temperature	339°C (642°F)
Average Clad Temperature at Hot Spot (stainless steel)	356°C (674°F)
Average Clad Temperature at Hot Spot (Zircaloy 4)	367°C (692°F)
Maximum Fuel Central Temperature	2,200°C (3992°F)
Peak Rod Average Burnup	25,000 MWd/MTHM
Chemical Shim, Beginning of Life	2,000 ppm boron
Initial Loading - MOX	345 kg in 9 assemblies
Initial Loading - UO_2	525 kg in 12 assemblies

Saxton Core I used standard UO_2 fuel elements to establish a core performance base line. Saxton Core II fuel loading consisted of nine central mixed oxide fuel assemblies (638 rods) and twelve outer fuel assemblies of standard UO_2. The mixed oxide contained 6.6 wt% PuO_2 in natural UO_2. The UO_2 assemblies were enriched to 5.7 wt% ^{235}U. Of the nine plutonium assemblies, two contained vibratory compacted (Vipac) fuel; the remaining seven assemblies, pelletized fuel. With the exception of some thirty fuel rods which were clad with 304 stainless steel, the mixed oxide fuel rods were clad with Zircaloy 4. Important elements requisite to nuclear operations analysis and fuel performance evaluation were:

- Analyses of at power boron (soluble neutron absorber) and control rod worths (ability to absorb neutrons to control reactor power level and shut down reactor during emergencies), temperature and power coefficients, core depletion rate (rate that fissile atoms fission; i.e., the rate of fuel burnup), and core flux wire* and detector maps**

*Flux wire - a special wire that can be inserted into the core for a short irradiation period. The wire, when withdrawn and passed by a radiation scanner, provides provides data that is indicative of the core neutron flux at the wire location. Neutron flux is a measure of the number of neutrons per square centimeter/second.

**Detector maps - in core radiation mapping, sensors positioned within the core produce three dimensional measurements of neutron density (radial axial flux maps).

- Evaluation of nuclear parameters in zero power tests, based on measurement of boron and control rod worths, temperature and pressure coefficients, minimum shutdown reactivity and xenon decay

- Nondestructive and destructive post-irradiation examinations of the fuel

Core II achieved 9,360 effective full power hours, corresponding to a core average burnup of 10,940 MWd/MTHM with an average burnup of 17,400 MWd/MTHM in the central region which contained the mixed oxide fuel. Subsequent measurements and data reduction showed that the burnups of the peak plutonium rod were 21,000 MWd/MTHM (rod average) and 28,000 MWd/MTHM (peak pellet).

Extensive examination of PuO_2 and UO_2 mixed oxide fuel rods after the Core II irradiation led to the following conclusions:

- Mixed oxide fuel performed satisfactorily, with no evidence of fuel rod failures, thus confirming the adequacy of design and fabrication procedures.

- The fuel rods exhibited good dimensional stability, with a maximum of 0.23 percent length increase and, with the exception of one rod, changes in mean diameter no greater than 0.003 inch.

- Pellet and vibratory compacted fuel performed equally well, although length increases with Vipac fuel were slightly less, and center fuel temperatures in peak power Vipac rods were somewhat higher than in highest power pellet fuel rods.

- The cold-reduced and stress-relieved Zircaloy 4 cladding employed in the plutonium region of Saxton Core II performed well. Oxidation of the outer surface was highly variable and, in some areas, greater than had been predicted from out of pile testing. Hydrogen uptake by the Zircaloy during operation was less than 50 ppm, which indicated few chemically reactive impurities in the fuel. The resulting hydrides were randomly or circumferentially oriented. Mechanical tests of clad samples indicated moderate irradiation strengthening but retention of significant ductility (at least 2.5 percent uniform elongation as measured in tensile tests).

- One fuel rod, which had anomalous dimensions, exhibited local massive hydriding but did not fail. The source of the excess hydrogen was probably stray contamination introduced during fabrication.

- None of the changes in dimensions, microstructure, or properties was of sufficient consequence to impose basic operation limits for MOX fuel in irradiation environments similar to those of Saxton Core II. As a result, it was determined that the mixed oxide irradiations could be carried to peak pellet burnups approaching 50,000 MWd/MTHM by reconstituting the

mixed oxide fuel rods into a looser lattice configuration employing 250 rods. This lattice change was made before starting Core III operations.

Core III remained in operation until May 1, 1972, at which time the peak pellet burnup ranged from 40,000 to 51,000 MWd/MTHM and a peak linear power of 21.2 kW/ft had been achieved.

Analysis and evaluation of the mixed oxide rods from Core III indicated good overall performance, even at the peak pellet burnups experienced. Progressive changes were observed in rod length, fuel microstructure, fuel clad interaction, corrosion of the cladding material, and mechanical properties of cladding. However, these changes were consistent with increased irradiation time and had no apparent effect on fuel performance. Profilometer scans, fission gas collection, and hydrogen analysis showed results similar to those observed at the end of Core II irradiation.

In the Core III irradiation, 33 rods developed defects when the burnup reached 40,000 to 42,000 MWd/MTHM. The defects were limited to rods near the upper end of the power spectrum and were associated with an anomalous crud condition not seen previously on any Saxton fuel rods. The defective rods were not considered indicative of an inherent power, burnup, or other performance limitation in Zircaloy clad mixed fuel, but appeared to be related to the presence of significant quantities of adherent crud, which suggested a change in core environment, such as water chemistry, after the midlife shutdown.

With respect to core reactivity, effectiveness of the control rods, and the prediction of neutron flux patterns and power densities, the methods of calculation which had been used for reactor cores with uranium fuels proved readily adaptable and accurate when used in computing the characteristics and performance of reactor cores with mixed oxide fuels. The predicted values were always within 5% of measured values and, for most parameters, within 2%.

It was shown that a reactor core designed for uranium fuels can accept mixed oxide fuel without change in the mechanical design, and that it can achieve longer life (with mixed oxide fuels) if the lattice spacing is increased.

AEC support of the Saxton Plutonium Program was terminated in 1972. Reports of the work are available in the documents listed in References 12-20 at the end of this chapter.

3.1.3 Edison Electric Institute/Westinghouse

Industry participation in mixed oxide fuel development was provided by the Edison Electric Institute (EEI), an investor owned electric utility company trade association organization through which support is provided for research and development projects of interest to the industry. A contract was entered between EEI and Westinghouse Electric Corporation (W) for a plutonium utilization development program directed toward the use of mixed oxide fuel in pressurized water reactors. The AEC contributed

to the work under this program by providing plutonium at a reduced charge. The EEI/W program was initiated in the year 1967, with the first phase of the work devoted to study factors that influence the economics of Pu recycle and the distinctive characteristics of plutonium fueled pressurized water reactors.

Using analytic and semiempirical adjustments to reactor core calculational techniques, Westinghouse improved the computer codes to make it possible to calculate the initial criticality of plutonium fueled systems with an accuracy consistent with that for uranium-fueled systems. In cores containing both plutonium fuel and enriched uranium fuel, calculations indicated that the use of separate and distinct core regions for each fuel type would be the most promising method for fuel loading. This could be accomplished either by arranging two distinct fuel regions in each assembly, or by use of individual assemblies of each fuel type.

Calculations and critical experiments showed no inherent limits which would restrict the use of a full plutonium core in a pressurized water reactor. However, in the core lattice configuration of existing reactors, a core consisting entirely of mixed oxide rods would experience a reduction in control rod worth. This results from the already noted fact that the fission cross sections for the fissile plutonium isotopes are about twice that of ^{235}U; consequently the same power density may be maintained with about half the neutron flux. Thus, with mixed oxide fuel, control rods have only about half as many neutrons to act on. This phenomenon is somewhat complicated by the fact that nonfissile plutonium isotopes have very high neutron absorption resonances in the thermalization neutron energy range, further reducing the number of neutrons available for the control rods to act on. The EEI/W experiments showed that, with a core design incorporating a larger number of control rods and a more open lattice spacing, a core with all mixed oxide rods could be operated safely. In a standard PWR core employing both mixed oxide fuel rods and rods containing UO_2 only, adequate control rod worth can be assured by positioning the UO_2 rods adjacent to all control rods and positioning the mixed oxide rods so as to obtain the desired power distribution. See CHAPTER IV, Section C-3.0, for a more complete treatment of this subject.

During the early studies and experiments under the EEI/W program, certain areas were identified as requiring a continuing effort:

- In the fabrication studies, the nature and extent of the shielding (primarily for neutrons) required for the high burnup plutonium fabrication, and the effects of this shielding on the cost of performing fabrication operations needed further resolution.

- Although initial criticality could be calculated satisfactorily, the depletion characteristics of large mixed oxide fueled cores contained uncertainties which could be resolved only through actual irradiation and subsequent destructive examination of the mixed oxide fuel. In addition, nuclear design uncertainties remained in the calculation of the power

distribution characteristics and control rod requirements for cores containing both UO_2 fuel rods and mixed oxide fuel rods. A demonstration fuel loading for a large PWR would assist in resolving this area of uncertainty.

- Based on the work at PNWL, the inpile materials performance of mixed oxide fuel was expected to be similar to and as satisfactory as that of uranium fuel; however, before this performance could be established with a high level of confidence, large quantities of plutonium fuel would have to be irradiated in the typical PWR environment.

In the preliminary core region design study conducted under Phase 2 of the EEI/W program, a 1,000 MWe four loop plant with a core containing both mixed oxide and UO_2 fuels was selected as the reference case. Calculations were made for the reference core and an identical core fueled with UO_2 only. The self-generated recycle mode of refueling with plutonium was assumed; this required each region reload to include both mixed oxide and uranium oxide fuel. To simulate equilibrium recycle conditions, operation with a 1/3 loading of mixed oxide fuel introduced in sequential loadings* was analyzed and compared with a UO_2 core that had operated for four fuel cycles. These analyses yielded several conclusions:

- An average enrichment of 4.2 wt% Pu is required to achieve the 33,000 MWd/MTHM burnup reached in the reference UO_2 core.

- Using the discrete assembly concept (all rods in single assembly contain either mixed oxide or UO_2), self-generated Pu recycle can be accomplished with all mixed oxide rods located in assemblies that do not contain control rods. This can be done without increasing the peak power density and without reducing the core power capability or lifetime.

- At equilibrium, with one-third of the core containing mixed oxide fuel elements, it is not necessary to install additional control rods or to position mixed oxide rods in assemblies containing the control rods.

- The moderator temperature coefficient for the core containing mixed oxide is 6.5% more negative, with the result that, as the reactor core temperature increases, the control rod worth decreases slightly.

- Natural uranium shows an economic advantage over depleted uranium as the mixed oxide diluent.

*The calculations were based on introducing all of the mixed oxide fuel (1/3 loading) over a 3 year period.

As a part of phase 2 of the EEI/W Plutonium Recycle Demonstration Program, a total of 720 PuO_2 and UO_2, Zircaloy 4 clad fuel rods in four assemblies were irradiated in San Onofre Nuclear Generating Station Unit No. 1. All rods in each fuel assembly contained mixed oxide pellets. The fuel rods were of three different plutonium enrichments and positioned so as to control local power. The number of fuel assemblies selected for the demonstration program was representative of the initial loading for self-generated plutonium recycle. These four assemblies, containing 45 kg of plutonium (fissile) in the mixed oxide rods, were inserted into the San Onofre reactor during the first refueling in November 1970. The demonstration mixed oxide assemblies were operated through two normal reactor cycles (San Onofre Cycles 2 and 3). The original plan was to irradiate these assemblies for three cycles. However, because of the possibility of problems identified subsequent to their insertion, caused by fuel densification in UO_2 fuels and the consequent limitation on power operation, irradiation was restricted to two cycles. One of the ways in which the UO_2 fuel densification problems was manifested was by the collapse of the clad material onto the pellets. This was compensated for by pressurizing the fuel rods. The mixed oxide rods were not pressurized and rather than risk fuel densification problems, it was decided to remove the mixed oxide assemblies even though they had shown no signs of trouble.

The San Onofre core consists of 157 fuel assemblies. During the first refueling at the end of cycle 1, 105 fuel assemblies from the initial core were reloaded according to the standard plan, and 52 fresh fuel assemblies were added, consisting of 48 UO_2 assemblies and the four PuO_2 and UO_2 demonstration assemblies. Two of the mixed oxide assemblies each had 52 removable and 128 nonremovable fuel rods. At the end of cycle 2, removable mixed oxide rods were examined and returned to the core. Four of these were replaced with natural UO_2 rods. Two of the four discharged rods were subjected to postirradiation examinations.

After one cycle, the peak pellet burnup on these assemblies was 12,600 MWd/MTHM, and the highest rod average burnup was 10,500 MWd/MTHM. Visual examination of the four assemblies and eight of the removable rods showed them to be in excellent condition.

After two cycles of irradiation were completed in June 1973, the peak pellet burnup on these assemblies was 25,050 MWd/MTHM and the highest rod average burnup was 21,050 MWd/MTHM. The assembly average burnup was 18,950 MWd/MTHM. Visual examination of the four assemblies and of six removable rods showed them to be in excellent condition. Although there was one indication of possible local clad hydriding on a peripheral rod in one of the assemblies, the rod was still intact, with no evidence of mechanical degradation. Rod length, diameter, and ovality measurements were made on six rods, four of which had previously been measured after one cycle of irradiation. The measurements showed no unusual conditions.

Two rods irradiated for one cycle and two rods irradiated for two cycles were selected for a program of nondestructive and destructive postirradiation examination. The examinations showed no anomalous conditions.

Details on the EEI/W Plutonium Program are available in the documents listed in References 21-25 at the end of this chapter.

3.1.4 Edison Electric Institute/General Electric

The Edison Electric Institute also sponsored work by the General Electric Company (GE) on plutonium utilization in boiling water reactors. AEC contributed by providing plutonium at a reduced charge. The EEI/GE mixed oxide fuel investigation was initiated in the year 1957 and pursued in parallel with the PWR mixed oxide program. The first phase was a study of the technical and economic aspects of plutonium recycle in BWR fuel. The following conclusions were reached on the work to be performed in subsequent parts of the program:

- It is technically and economically feasible to utilize recycle plutonium in BWR's.
- The fabrication method (hot pressed vs cold pressed pellets) needs to be evaluated with respect to effects on cost and performance.
- In reactor operating experience on fuel is needed.
- Further work is required concerning the trend toward reduced control margins when plutonium is utilized.
- Nuclear calculational methods require improvement.
- Fast transient tests are required to evaluate safety, because plutonium segregation within the fuel rod is possible as a result of diffusion or some other mechanism.

Under the development and testing phase of the EEI/GE plutonium recycle demonstration program, mixed oxide fuels of several types were tested in operating reactors to evaluate their performance. Major tests were planned for the Big Rock Point nuclear power reactor. In addition, four assemblies which were fabricated originally for use in the first core of Vermont Yankee were instead now being irradiated in the Quad Cities Unit No. 1 reactor. Optimization of mixed oxide fuel element design was continued, taking into account improvements in methods, technology, and economic studies.

The irradiations of mixed oxide fuel carried out in Big Rock Point under this program began in March 1969, and included 32 rods; irradiation of three bundles, each containing 68 mixed oxide rods, was initiated in March 1970. These tests are described subsequently.

3.1.4.1 Rod Irradiations

The fuel rod tests were designed to compare the performance of

- Rods containing mixed oxide pellets with flat ends so that the pellets would stack within the cladding to make a solid rod

- Rods containing pellets with dished ends which would create about 3% voids within the fuel rods

- Rods containing annular pellets

These fuel forms would be compared with each other and with standard UO_2 fuel. The major differences in the fuel rod designs are shown in Table II-6. In this test, emphasis was placed on annular fuels in which the annular pellets are stacked within the rod cladding so that there is a vertical hole sealed inside the rod. This has the effect of reducing the plutonium in the core without changing the fuel rod size or spacing. The four rods containing cylindrical solid pellets and the four containing dished pellets supplied the performance link between UO_2 fuel of current design and mixed oxide fuel. These eight rods were also designed to show incremental performance differences between the three pellet geometries. See Table II-8. Tables II-7 and II-9 present fuel rod design information.

The plutonium content in each rod was originally designed to be constant, and the plutonium concentration was varied to make up for changes in fuel density and geometry. Thus, the linear power characteristics of each rod were similar. The rods were positioned initially in the Big Rock Point reactor core to maintain these similar power characteristics.

The 32-rod irradiation began with Cycle 7 of the Big Rock Point reactor in March 1969 and continued through Cycle 11, which ended in March 1974. It should be noted that during the early operation of Big Rock Point, the copper-nickel tubes in the feedwater heater led to high crudding rates on the surface of all fuel rods in Big Rock Point. This in turn led to a restriction of 70% of rated power on the reactor power level and consequent derating of the specific power of the mixed oxide fuel rods during part of the operating period. Nevertheless, the mixed oxide rods operated between 5 and 15 kW/ft. The 32 rods were examined visually after Cycles 7, 8, 9, 10, and 11.

Four rods were given destructive examinations after Cycle 7. Additional rods were removed after each cycle for possible destructive examination. Both the visual examinations and the destructive tests in the hot cells showed no flaws or inadequacies in any of the fuel rods. Peak rod exposures of 23,100 MWd/MTHM were achieved.

Table II-6

EEI/GE - BIG ROCK POINT REACTOR
32 ROD PROGRAM EXPERIMENT DESIGN

	Solid	Dished	Annular Hole 0.1 in.	Annular Hole 0.2 in.
Density, % of Theoretical	92	95	92	92
Enrichment, %	1.22	1.22	1.36	1.59
Hole Size, diam., inches	-	-	0.100*	0.200*
Dishing, %	-	3.0	-	-
Rods, No.	4	4	12	12

*Hole sizes of 0.10 and 0.20 inches are calculated to assure no melting at peak calculated linear heat generation rates of 21.6 and 26.9 kW/ft, respectively.

Table II-7

EEI/GE - BIG ROCK POINT
MOX FUEL DATA

	MOX Rods
Fuel	
Material	UO_2 and PuO_2
Pellet Diameter, in.	0.471
Active Length, in.	68.62
Density, % of Theoretical	92-95
Cladding	
Material	Zircaloy-2
Thickness, in.	0.040
Outside Diameter, in.	0.5625
Rod Pitch, in.	0.707
PuO_2 and UO_2 Rods per Bundle	2
Plutonium Fissile content (Weight % in PuO_2 and UO_2)	
1.22 Nondished	
1.22 Dished	
1.36 0.1-in. Annular Hole	
1.59 0.2-in. Annular Hole	

Table II-8

EEI/GE - BIG ROCK POINT MIXED OXIDE FUEL

THERMAL PERFORMANCE CHARACTERISTICS

	MOX Fuel			
			Annular Hole	
	Solid	Dished	0.1 in.	0.2 in.
Fuel Pellets				
Outside Diameter, in.	0.471	0.471	0.471	0.471
Inside Diameter, in.	0.0	0.0	0.1	0.2
Cladding				
Thickness, in.	0.040	0.040	0.040	0.040
Outside Diameter, in.	0.5625	0.5625	0.5625	0.5625
Incipient Melting Temperature of UO_2, °F	5080	5080	5080	5080
Fuel Density, % of Theoretical	94	94	94	94
Centerline Temperature				
at 500,000 Btu/h-ft^2-°F	5080	5080	4850	3950
at 410,000 Btu/h-ft^2-°F	4600	4350	4100	3250
Heat Flux for Incipient Melting, Btu/h-ft^2	465,000	490,000	530,000	670,000
Area Fraction Molten at Peak Heat Flux	0.09	0.03	0	0

Table II-9

EEI/GE - BIG ROCK POINT

FUEL PELLET SPECIFICATIONS, THREE BUNDLES

No. of MOX Rods	204
Diameter (in.)	0.471 ± 0.002
Annulus (in.)	0.150 ± 0.005
Density (% TD)	92.0 ± 1.5
Enrichment (%)	
Pu Fissile/Pu + U	1.46
	2.30
	4.95
	8.16
	2.04
^{235}U	0.7
Oxygen to Heavy Metal Ratio	1.98 - 2.02
Gas Content	
Moisture (μℓ/g)	<12
Gas (μℓ/g)	28
Homogeneity	100% <500 μm, at 95% confidence level
	95% <100 μm, at 95% confidence level

The fuel rod examination phase responsibility was assumed by the Electric Power Research Institute (EPRI). It was expected that fuel rod characterization, metallographic examination of fuel rod sections and microprobe analysis for fission product transport would be included in the laboratory examination.

3.1.4.2 Bundle Irradiation Demonstrations

Three bundles containing 204 mixed oxide rods were designed to demonstrate the performance of complete mixed oxide fuel bundles in the Big Rock Point reactor. The normal UO_2 bundle mechanical design was used. MOX fuel loading was designed to be interchangeable with the UO_2 fuel, with respect to performance and exposure capability. Bundles contained MOX rods of four different plutonium concentrations designed to provide the desired power distribution for operation in the reactor through four cycles. The peak fuel bundle exposure achieved was 17,500 MWd/MTHM. Special rods were included for irradiation of some 80% fissile plutonium from the Dresden reactor.

The MOX rods all contained cold pressed and sintered fuel pellets of annular design prepared from mechanically blended ceramic grade PuO_2 and UO_2 powders. The annular hole was 0.150 inch diameter and the fuel matrix was nominally 92% of theoretical density. The only rod to rod variation was the plutonium enrichment and the removability of four of the rods.

Each bundle contained four of the removable fuel rods which could be examined to monitor the performance of the fuel. The four cobalt corner rods were also removable. Twice the usual number of burnable poison (Gd_2O_3) rods were used because of reduced worth in a mixed oxide fuel bundle. Table II-9 shows the fuel pellet specifications for the MOX rods.

The bundle irradiations were initiated with Cycle 8 in the year 1970 and continued through Cycle 10 with all three bundles. Only one bundle was reinstalled for Cycle 11, as decribed later. Irradiation of this bundle continued through Cycle 11.

It has been reported informally that the fission product leakage tests showed evidence of rod failures in two of the bundles. Two rods in the third bundle failed in a decrudding operation during the Cycle 10 shutdown. With replacement of these two rods,* the bundle was returned to the reactor for continued irradiation during Cycle 11.

On the basis of performance evaluation to date, the investigators felt that the mixed oxide fuel in these three bundles, as well as in the 32 individual rods previously irradiated, behaved similarly to UO_2 fuel--with no abnormal behavior resulting from the use of mixed oxide fuel.

*These rods also were to be examined by EPRI. See paragraph 3.1.4.1.

3.1.4.3 MOX Fuel Irradiation - Quad Cities Unit No. 1

The reload plutonium recycle fuel bundle was designed with the same envelope dimensions as the initial core fuel. See CHAPTER IV, Section C-2.0. It could, therefore, be inserted, without restriction, into all locations within the reactor core at Quad Cities or other similar BWR cores. The basic lattice arrangement of 49 rods in a seven by seven array is the same as the initial core fuel, with a centrally located spacer capture rod, and eight tie rods located symmetrically around the periphery of the fuel bundle.

Prototype MOX fuel bundles were of the same general mechanical configuration that GE had been designing and manufacturing for the past 12 years, with gadolinium for reactivity control augmentation. Gadolinium containing reload fuel had been the subject of past AEC safety analyses for Dresden Unit No. 1, Big Rock Point, Humboldt Bay Unit No. 3, Dresden Units No. 2 and No. 3, Quad Cities Units No. 1 and No. 2, Nine Mile Point and others, and had been approved for use in each case. The mixed oxide fuel bundles also incorporated design improvements which had also demonstrated their value in initial core fuel for Browns Ferry Unit No. 1, Peach Bottom Unit No. 2, and Cooper Station.

Two types of mixed oxide fuel assemblies were designed. Four assemblies of Type A31 contained 40 of the 48 rods and were designed to be loaded in the central reactor positions around the center control blade. The uranium enrichments in the UO_2 fuel rods were the same as the standard UO_2 reload fuel, with the exception that 10 Type 5 high enrichment UO_2 rods were introduced to improve power distribution. The four identical Type A31 assemblies were designed to be irradiated under well controlled conditions in the center of the reactor. This would maximize the benefits of possible following program gamma scans and isotopic measurements.

Two types of plutonium were utilized in the mixed oxide fuel assemblies: Dresden Unit No. 1 recycle Pu (80% fissile) and AEC Pu (90% fissile). The Dresden Unit No. 1 recycle plutonium was used in reduced concentration in mixed oxide rods at the outside of the mixed oxide rod island and provided some flattening of local power peaking as well as improving the steam void dependence of the local power peaking. The remaining eight MOX rods were incorporated in a special peripheral fuel assembly design, Type A32--two rods of each of the four mixed oxide rod fuel types. Irradiation of the Type A32 assembly provides a directly comparable low power environment for fuel rods identical to those located in the central fuel assemblies, for future evaluations of the observed fuel performance. The environment at the periphery also results in the coolest possible BWR neutron spectrum and will provide reactor physics data of significance. These BWR prototype fuel assemblies were inserted in Quad Cities Unit No. 1 core in July 1974. The average burnup for the four center fuel assemblies was nearly 8,000 MWd/MTHM as of January, 1976 when the assemblies were visually examined during a reactor refueling outage. The peripheral fuel assembly reached a burnup of about 3,000 MWd/MTHM.

Reports covering the EEI/GE program are listed as References 26-38 at the end of this chapter.

3.1.5 Gulf United Nuclear Corporation/Commonwealth Edison

In the year 1957, Gulf United Nuclear Corporation and Commonwealth Edison joined in sponsoring a plutonium recycle demonstration program in the Dresden Unit No. 1 nuclear power reactor for the overall purpose of gaining experience in all aspects of the recycle operation. Objectives of the Dresden Plutonium Recycle Demonstration Program were to

- Establish the adequacy of a full-size plutonium recycle assembly under actual operating conditions
- Fabricate mixed oxide fuel assemblies on a semiproduction scale
- Establish fuel cycle costs for MOX assemblies under commerical conditions
- Evaluate reactor performance for a core containing a significant quantity of mixed oxide fuel
- Verify the adequacy of analytical models for calculating reactivity and power distributions in mixed oxide assemblies
- Obtain measured reactivity and local power distributions for mixed oxide assemblies by critical experiments prior to irradiation
- Obtain postirradiation isotopic and burnup data from hot cell examinations of removed rods

The available plutonium for fabricating the demonstration assemblies had the isotopic composition shown in Table II-10. This isotopic distribution is characteristic of Dresden fuel at 12,000 MWd/MTHM--rather than at discharge (23,000 MWd/MTHM), which would have been preferable for demonstration assemblies. The total assembly plutonium fissile content, 0.45 wt%, was maintained even though the isotopic composition differed from equilibrium discharge plutonium.

Table II-10

ISOTOPIC COMPOSITION OF PLUTONIUM FOR DRESDEN PLUTONIUM RECYCLE ASSEMBLIES

	Composition of Plutonium wt%
^{238}Pu	0.4
^{239}Pu	71.3
^{240}Pu	20.6
^{241}Pu	6.1
^{242}Pu	1.6

Percent fissile = 77.4 wt%

At 77.4 wt% fissile, the total plutonium contained in the 11 demonstration assemblies was 6.6 kg.

It was desirable from a fabrication and economic standpoint to use the standard Dresden UO_2 fuel rods in non-plutonium bearing rod locations. The number of mixed oxide fuel rods and their location were established on the basis of utilizing self generation plutonium (0.45 wt% fissile plutonium per assembly) in the minimum number of rods while still meeting the local power peaking limitations. Nine mixed oxide rods were chosen as a compromise between power peaking and fabrication penalty. With nine mixed oxide rods at a fissile plutonium content of 1.78 wt%, a beginning of life-peak-to-local power ratio of 1.28 was calculated for the assembly--the same as the reference UO_2 beginning of life peak.

The specific locations of the mixed oxide assemblies in the Dresden reactor core at the beginning of Cycle 7 were selected primarily to distribute these elements throughout the core. This permitted core uniformity and eliminated distortion of the core by any unexpected performance of the mixed oxide elements. The two instrumented assemblies were placed incore at locations along the north south axis. Four other mixed oxide elements were loaded adjacent to instrumented UO_2 assemblies. Thus, any effects of the mixed oxide assemblies on their uranium neighbors could be observed in the instrument responses. The thermal hydraulic characteristics of the plutonium bearing assemblies were identical to those of the UO_2 fuel assemblies which constituted the major portion of the reload batch.

After two cycles in the reactor core, the mixed oxide assemblies had attained an average exposure of 15,900 MWd/MTHM, a highest assembly exposure of 17,470 MWd/MTHM and a peak pellet exposure of 22,830 MWd/MTHM. At that time all eleven mixed oxide assemblies were tested for fission product gas leakage: six appeared to contain leaking rods and were given detailed visual inspection by closed circuit television. The inspections revealed end plug weld fractures in both the standard UO_2 and the MOX fuel rods. Clad blisters and a major rod fracture were also observed in UO_2 rods. Similar failures have been observed in the same rod locations in fuel assemblies containing only UO_2 rods.

End plug weld failures were the most common visual evidence of failure in all types of fuel rods and the only observed fault in the mixed oxide rods.

The five assemblies that did not show indication of leaks were reinserted for Cycle 9, which started in March 1974. They were examined in September 1975 at the end of the cycle. Two fuel assemblies with fuel clad leaks were removed. The average burnup exposure of the 23 mixed oxide rods in the three fuel assemblies was 15,000 - 17,000 MWd/MTHM.

Present plans are to perform post irradiation examination, including isotopic composition measurements of two rods from Cycle 8 and two rods from Cycle 10. Another fuel inspection is planned at the end of Cycle 10, expected in November 1976.

Details of the GUNC/Commonwealth Edison Program are available in the documents listed in References 39 through 41 at the end of this chapter.

3.1.6 Big Rock Point/Exxon/NFS

Exxon Multiple Cycle Plutonium Utilization

One of the ongoing programs at Big Rock Point includes two uranium assemblies and four mixed oxide assemblies. The four mixed oxide assemblies contain a total of 96 plutonium-bearing rods. Two of these MOX assemblies with the 9 x 9 rod matrix design which characterized commercial uranium fuel designs prevalent at that time, were inserted into the Big Rock Point core in May 1972. The maximum assembly exposure achieved so far is 17,800 MWd/MTHM. The remaining two assemblies, incorporating the 11 x 11 fuel rod matrix design with smaller fuel pins and more heat transfer area, were inserted in April 1973. The 11 x 11 design served as a forerunner to the commercial mixed oxide fuel design discussed later, and was first inserted in the Big Rock Point core in July 1974. The maximum assembly exposure reached on this design is 15,400 MWd/MTHM. The two uranium assemblies provide a standard for reference in evaluating the four Exxon mixed oxide fuel assemblies. All six fuel assemblies have remained in the core since first inserted. It is expected that assembly exposures exceeding 20,000 MWd/MTHM will be achieved by the end of the year 1976 (Cycle 14).

Examination of the rods in the program is primarily nondestructive in nature. Typical poolside examinations include visual inspection and measurements:

- Visual inspection by periscope (individual fuel rods and overall assembly)
- Rod diameter measurements by profilometer
- Cladding integrity testing by eddy current
- Pellet column length by gamma scan, eddy current, and plenum gauge methods
- Pellet column continuity verification by gamma scan
- Relative rod power measurements by gamma scan
- Cladding growth measurements by mechanical fixture

Destructive examinations are planned, however, for isotopic analysis and for features revealed by the nondestructive examination. Four rods with only 672 hours of irradiation will be destructively examined for densification data.

As the rods from this program are discharged, the plutonium fuel will be recovered, refabricated, and reinserted into the reactor. This will allow gathering isotopic data on multiple recycle plutonium.

NFS Demonstration Assemblies

Four demonstration assemblies manufactured by Nuclear Fuel Services were inserted in the Big Rock Point core in February 1973. Each assembly contains 73 mixed oxide rods; the first assembly has accumulated an exposure of 13,700 MWd/MTHM. All of these

mixed oxide assemblies have remained in the core since first inserted. Burnup to 20,000 MWd/MTHM is planned unless fuel integrity is compromised. No plans have been made for destructive examination of these assemblies.

Exxon Commercial Irradiation

Irradiation of mixed oxide assemblies on a commercial scale began in July 1974 with the insertion of eighteen assemblies, each of the 11 x 11 design with 24 mixed oxide rods. Eight additional mixed oxide assemblies were inserted during the spring refueling of 1976. Commercial irradiation of plutonium at Big Rock Point is currently restricted to 50 kg.*

Experience

Experience with both developmental and commercial mixed oxide fuel at Big Rock Point has been extremely good. Off-gas activity--an indicator of fuel integrity--has shown a downward trend over the last several years. The recently completed cycle 13 had the lowest off-gas activity of any full length cycle. Examinations of the fuel at the end of this cycle revealed no leaking mixed oxide assemblies.

3.1.7 The Belgian Plutonium Recycle Program[43]

The Belgian plutonium recycle program was initiated in the year 1959 under EURATOM sponsorship. It was described in 1971 as a "1000 man-year effort." The program has emphasized plutonium recycle for LWR's and has included extensive testing as well as research and development. The Belgian 11.5 MWe BR-2 reactor was the first LWR to be loaded with plutonium fuel; it has since been supplemented with additional core loadings which carried burnup tests as high as 50,000 MWd/T. Most recent tests involve fuel elements in the BWR's at Dodewaard and Garigliano, Italy. PWR fuel tests are being conducted in the SENA reactor.

Some of the conclusions reached by the Belgians are summarized as follows:

- In equilibrium recycle cores, the water to fuel ratio should be increased to achieve better plutonium utilization and to compensate for control rod worth decreases; this water to fuel ratio increase is limited by the associated decrease in temperature coefficients.

- There is an economic incentive to increase the burnup of mixed oxide fuel beyond that which would be optimum for enriched uranium to compensate for higher mixed oxide fuel fabrication costs.

- The first generation PWR plants can advantageously make use of plutonium recycle.

*Memo and Order from the USNRC, August 1, 1975.

- The so called "plutonium island" fuel assembly type (plutonium zone surrounded by enriched uranium only fuel) is recommended for some core configurations where the shutdown margins remain practically unaffected; in the SENA case the relative control rod worth is decreased by less than 2%.

- Plutonium utilization in BWR's appears economically less attractive than in PWR's, but several BWR characteristics favor progressive conversion into plutonium burners.

- Relatively independent behavior of the fuel assemblies inside individual shrouds is observed when assemblies are separated by large water gaps (flexibility to adapt the water moderator/metal ratio).

- The practice of power distribution flattening by control rod movements is recommended.

- Routine utilization of multiple enrichments within the fuel assemblies gives a lower relative penalty for plutonium fuels.

3.1.8 CNEN/ENEL Plutonium Utilization Programs in Italy[44]

In the year 1966, the Italians launched a major program of study and development related to plutonium utilization. The ENEL (Ente Nazionale per l'Energia Elettrica) program investigated the feasibility of plutonium recycling by loading mixed oxide fuel rods into operating reactors. CNEN (Comitato Nazionale per l'Energia Nucleare) worked on mixed oxide fuel technology, including physics, fuel element design, and fabrication methods.

Under the CNEN program, a pilot plant at Saluggia undertook reprocessing of irradiated mixed oxide fuels to separate both uranium and plutonium. The Reactor Physics Laboratory at Casaccia Nuclear Research Center investigated the neutronic behavior of MOX fuels in cooperation with Battelle Pacific Northwest Laboratory. The mixed oxide fuel rods for the initial work in Italy were provided by the USAEC. A new plutonium laboratory was completed at Casaccia in the year 1968 and used thereafter for fuel element fabrication research and development.

Mixed oxide fuel pins were irradiated in reactors in Sweden, Norway, Germany, England and France as well as in Italy. Many irradiations involved single rods for research investigations; the Swedish and German reactors accepted complete fuel assemblies for tests to burnups of 15,000 and 25,000 MWd/MTHM, respectively. The activities of the CNEN program provided a basis for planning experiments with mixed oxide fuels in the ENEL nuclear power reactors.[42]

The Garigliano BWR power station was used by ENEL for a plutonium recycle demonstration program which began in the year 1968. A total of 600 mixed oxide fuel rods was incorporated into fuel assemblies for the Garigliano reactor. Critical experiments were performed with mixed oxide fuel assemblies, and irradiated fuel assemblies

were examined in detail to determine how closely the calculated values agreed with measured values. Agreement was considered to be good, proving the validity of computer codes for use with mixed oxide cores. Examination of 12 assemblies after the first shutdown showed no abnormal conditions. This examination included both a fission product gas leakage analysis and a visual examination. Postirradiation metallurgical examination of a mixed oxide fuel rod after 10,000 MWd/MTHM peak pellet burnup showed the plutonium distribution to be similar to the pre-irradiation distribution.

Four reload assemblies containing 96 mixed oxide fuel rods were provided by General Electric in the year 1968 as part of a group of 24 plutonium bearing fuel elements for irradiation in the Garigliano reactor. Four of these were discharged from the reactor in the year 1975, and 46 new mixed oxide fuel assemblies added. The new elements were fabricated by Fabbricazioni Nucleari at Bascomarengo, Italy, using fuel rods fabricated by Belgonucleaire. All of the new assemblies are of the plutonium island type. To date, irradiated fuel from the Italian reactors has been processed at Windscale, England, and Mol, Belgium, for separation and purification of plutonium. Fabrication of fuel rods containing plutonium initially was performed for the Italians by contractors in the United States, Germany, and other European countries, but the CNEN mixed oxide fabrication plant at Casaccia, Italy, is now in operation. Italy's current plans are to recycle no more plutonium in the LWR's, but to recover the plutonium and save it for use in fast breeder reactors. The Italian experimental fast breeder is scheduled for startup in the year 1978.

3.1.9 Obrigheim Reactor Demonstration of Mixed Oxide Fuel

In a cooperative program with the West Germany Kraftwerk Union (KWU), Combustion Engineering (CE), through ALKEM, fabricated mixed oxide fuel assemblies in Europe for the Obrigheim reactor. The demonstration began in the year 1972 with insertion of a single demonstration assembly. Eight additional mixed oxide assemblies were added during the September 1973 refueling. Since that time, more mixed oxide fuel assemblies have been added and some have been removed. The announced intention is to continue adding mixed oxide rods until the equivalent of self generation levels is achieved. As of early 1976, one mixed oxide fuel assembly is in its third cycle, 11 are in the second cycle, and 8 are in the first cycle. Because of a cooperative agreement with KWU, CE has complete access to data from this program. In addition to the Obrigheim demonstration, there have been other CE/KWU programs to determine the irradiation performance and densification properties of mixed oxides and a program to dynamically measure fuel properties, including densification, in the Halden, Norway, reactor.

The 111 assemblies, representing a collective exposure of 230 operating cycles, have only developed one leaking assembly--the prototype assembly in KWO. This assembly was shown by postirradiation examination to have failed by internal hydriding and from a failure that was characteristic of similar failures in UO_2 fuel assemblies. The visual inspection of the mixed oxide assemblies together with destructive postirradiation examination of 12 mixed oxide fuel rods did not show any significant differences from rods from UO_2 assemblies. The accumulated burnup of mixed oxide assemblies to date is shown in Table II-11.

Table II-11

SUMMARY OF THE IRRADIATION OF MIXED OXIDE FUEL ASSEMBLIES OPERATED IN KRAFTWERK UNION (KWU) SUPPLIED PLANTS

Nuclear Power Plant*	Year of Insertion	Number of Inserted Fuel Assemblies	Number of Inserted Fuel Rods	Amount of Fissile Pu, kg	Burnup, MWd/MTU	Number of Cycles	Matrix Material
VAK	1966	41	557	18.4	15,000	4	Natural uranium
KRB	1974	40	1400	94.0	22,000	2	Natural uranium
KWL	1970	1	15	1.0	18,000	5	Natural uranium plus 232Th
MZFR	1972	8	296	11.8	12,000	4	Natural uranium
KWO	1972	21	3780	158.9	28,500	3	Natural uranium

*VAK: Versuchsatomkraftwerk Kahl
KRB: Kernkraftwerk RWE Bayernwerk (Gundremmingen)
KWL: Kernkraftwerk Lingen
MZFR: Mehrzweckforschungsreaktor (Karlsruhe)
KWO: Kernkraftwerk Obrigheim

The KWU mixed oxide fuel rods were fabricated by ALKEM, while the assembly was carried out by Reaktor-Brennelement Union (RBU); both organizations being affiliates of KWU. The current capacity of ALKEM is 20 metric tons of heavy metal per year and will increase to 40 metric tons in about 1980.

The KWU experience with mixed oxide fuel assemblies was summarized by CE as follows:

- 6,048 fuel rods in 111 fuel assemblies
- 1 defected fuel assembly
- No significant restrictions in fuel cycle management
- No licensing restrictions

Thus, from a technical point of view, KWU's experience is that the performance of mixed oxide fuel assemblies is essentially equivalent to that of uranium oxide fuels.

3.1.10 Worldwide Plutonium Utilization Plans and Programs

Many countries have been developing and testing the technology required for recycle of plutonium in thermal reactors. A large amount of plutonium is expected to have been produced in commercial reactors around the world by the year 1980. Because most countries do not yet have an established reprocessing industry, it is uncertain how much of this plutonium will be separated from spent fuel and purified in a form suitable for recycling in nuclear fuel. To date, most national programs have concentrated on mixed oxide fuel irradiations, demonstration and large reload programs, design studies, critical experiments and economic and environmental assessments. The fuel reprocessing aspects of the plutonium recycle studies are generally not so far

advanced. Although fuel reprocessing plants have operated in the past, there are no commercial plants now in operation anywhere in the world. A reprocessing plant in France may start up late in the year 1976. Others in England and the United States may be started up a few years later but operations today are limited to pilot plants or special noncommercial fuel reprocessing facilities.

In the United States, assuming favorable regulatory decisions, Allied-General Nuclear Services' Plant at Barnwell, South Carolina, is expected to start operations in the early 1980's; similarly, Nuclear Fuel Services' Plant at West Valley, New York, is expected to start up again in the early 1980's after completing planned modifications.

In November 1974, the International Atomic Energy Agency's Panel on Plutonium Utilization in Thermal Reactors met in Karlsruhe, Germany, to review the current status of plans and programs for plutonium utilization in the participating countries. The 1974 status reports for the various countries are summarized in the following paragraphs, adapted from a report prepared for the Electric Power Research Institute[45] in Palo Alto, California, with updates from other sources.

Belgium: Belgium has a well established plutonium recycle development program. An industrial facility capable of producing 900 to 1,000 kg/week of mixed oxide fuel has been in operation since the year 1973. The Eurochemic fuel reprocessing plant processed 120 tons of fuel in the years 1973 and 1974, but has been shut down since that time. Demonstrations of the behavior of plutonium fuels have been in progress for several years in PWR and BWR plants. In parallel, a few samples were and are being irradiated in material testing reactors to assess particular details of the specifications or to investigate the fuel behavior at extreme conditions. Belgium has purposely followed a policy of scaling up its mixed oxide fuel manufacturing capacity in order to fulfill not only its needs but to allow it to act as a subcontractor for foreign reload suppliers.

Canada: The plutonium utilization program in Canada is directed towards solving the technical problems of plutonium recycle in CANDU (natural uranium, heavy water) reactors and establishing conditions for economic viability. To provide a focus for these investigations, the Canadians have performed a design study which used a conceptual design for a 1,200 MWe CANDU BLW reactor as the basis for an examination of all aspects of the reactor system and fuel cycle. Similar studies are in progress, to examine plutonium recycle in the CANDU PHW and the use of plutonium as the initial fissile feed for a thorium ^{233}U fuel cycle in CANDU reactors.

A 3-ton per year pilot facility for the fabrication of mixed oxide fuel was completed in the year 1974. The plant is being operated to fabricate 200 to 300 CANDU fuel bundles or 3.2 to 4.8 tons of fuel (Th and Pu). The intent is to obtain sufficient experience to permit reliable fuel fabrication cost estimates and to demonstrate the successful operation of mixed oxide fuel bundles in Canadian Power Reactors. Canada has no fuel reprocessing plant at present.

Federal Republic of Germany: Up to the year 1975, work in the Federal Republic of Germany concentrated on successful demonstration of recycle fuel behavior in thermal power reactors. This included fuel fabrication at prototype scale, elements testing under irradiation and the necessary applied software development. Phase I ended in the year 1974 with design and initiation of testing of full Pu-reload cores following the self generation concept in both a PWR and BWR.

Phase II of plutonium recycle in the FRG for the years 1975 to 1980 will be conducted by a joint venture of utilities, the nuclear fuel recycling industry, and the government. The primary goal of this program is to advance the technology of commercial plutonium recycling. Additional aims are to demonstrate technology by which the environmental impact of plutonium can be held as low as possible and to develop technology needed for fast breeder fuel element production. Present plans call for operation of a fuel reprocessing plant about the year 1985.

Plutonium utilization in thermal power reactors is considered to be a necessity at least in the next decade. The first core loads for fast breeder reactors are not anticipated prior to the year 1990. An immediate recycling of plutonium in thermal power reactors will improve the economy of the nuclear fuel cycle because stored Pu has a high financial value. The Federal Republic of Germany does not plan to consider the alternative of plutonium storage, either in purified form after chemical separation or in the form of spent fuel elements after discharge from the reactor. The key objectives in Pu recycling, the demonstrations of Pu technology, and the technical and economical aspects of Pu handling are also directed toward the development and introduction of fuel fabrication technology for fast breeder reactors.

France: France has decided to concentrate on the development of fast breeder reactors; thus interest in the recycle of plutonium as fuel in thermal reactors is secondary and at a low level. A few years ago it appeared that for about 10 years (1980 to 1990), France would have a great quantity of available plutonium and only a few fast breeder reactors. Today, it appears that spent fuel reprocessing has fallen behind schedule and plutonium accumulation during this period will not be large. A fuel reprocessing plant at Le Havre with a capacity of about 800 tons of fuel per year has been constructed and may start up near the end of the year 1976.

India: India plans to utilize the plutonium produced in CANDU type reactors as fuel for fast breeders when they become available. A 40 MWth fast breeder test reactor is presently under construction at the Reactor Research Center near Madras to gain experience with sodium cooled fast reactors.

The annual spent fuel discharge from the Tarapur Atomic Power Station (two BRW's at 200 MWe each) contains about 120 kg of Plutonium. The CANDU type power stations at Rajasthan, Madras and Navora will produce spent fuel containing about 150 kg plutonium per station per year. The fuel discharged up to this time from the Tarapur station contains about 200 kg of plutonium.

To operate the Tarapur Atomic Power Station, enriched fuel is imported from the United States, but India is taking serious note of the developments being made in the technology of plutonium recycle. The capability to reprocess spent fuel is being developed at Tarapur, though no firm decision has been made to utilize MOX technology. A plant is being set up to fabricate the fuel elements for the fast breeder test reactor.

Japan: The Power Reactor and Nuclear Fuel Development Corporation (PNC) is now planning to initiate plutonium recycling at an early stage. It plans to irradiate plutonium fuel assemblies in JPDR (PWR, 90 MWth). Another program is under way to load four plutonium fuel assemblies in MIHAMA-1 (PWR, 340 MWe) by the year 1977 or later. In the Advanced Thermal Reactor (ATR, 165 MWe), reactor physics experiments have been carried out since the year 1972; ATR is scheduled to be critical in 1976. A reprocessing facility (PNC, 200 tons/year) has been operated on a test basis since the year 1975. The reprocessing facility is not currently scheduled for production operation. In addition, a conversion facility at PNC is scheduled to be operative in the year 1977 for conversion of plutonium nitrate produced from PNC's reprocessing facility to PuO_2 which is used for fabrication of FBR, ATR, and Pu-thermal reactor fuel. The present fabrication capacity is insufficient for Pu fuel assembly loading programs, and therefore expansion is under consideration. The total amount of Pu produced from thermal reactors in Japan will increase to about 14 tons by the year 1980.

The Netherlands: At present there are two operational thermal power reactors in the Netherlands. One is at Dodewaard (BWR, 50 MWe), the second at Borssele (PWR, 450 MWe).

At the start of the second cycle of the BWR plant at Dodewaard, two prototype Pu-island elements were loaded. They remained in the core during Cycles 2, 3, 4, and 5. Average burnup on removal was about 20,000 MWd/MTHM. At the start of Cycle 5, four Pu-island elements were loaded (two with gadolinium as burnable poison). At the start of Cycle 6, one fresh Pu-island element was added with gadolinium burnable poison.

In the near future, Dodewaard will most probably sell its plutonium. The production rate at equilibrium is about 12 kg fissile Pu per year. The Borssele plant will probably recycle its own plutonium--with the exception of the plutonium of the first discharge. The production rate is about 78 kg fissile Pu per year at equilibrium (assuming no Pu recycling).

The sol-gel processes are being evaluated for application in producing spherical fuel particles as feed material for vibratory compaction--the Vibrasol process. It has been successfully applied to production of about 100 UO_2 fuel rods for irradiation purposes and has now been further developed for mixed oxide rods. Mixed oxide Vibrasol rods are at present under irradiation in the High Flux Reactor (HFR) at Petten. It is felt that the Vibrasol process has distinct advantages as a fabrication method, especially for mixed oxide. Furthermore, as indicated by the irradiation of instrumented fuel assemblies in the Halden Reactor, Vibrasol fuel rods may have better operating behavior, due to less interaction between the fuel and the cladding.

United Kingdom: The major research and development effort of the United Kingdom Atomic Energy Authority (UKAEA) is directed towards the exploitation of the sodium cooled fast reactor (SCFR). However, adequate expertise and manufacturing capacity for producing plutonium bearing fuels for experimental purposes for either gas or water cooled thermal reactors are being maintained by both the UKAEA and British Nuclear Fuels Limited (BNFL). This could form the basis of development programs for plutonium recycling should the UK Electricity Generating Board require that option. If the UK decides to develop the plutonium recycle option, the earliest date at which large scale recycling could commence is the year 1986. This timescale is set primarily by the steam generating heavy water reactor (SGHWR) commissioning program and the desirability of a few years of successful operation experience with uranium fuel before introducing plutonium recycle as fuel on a large scale. A demonstration plutonium recycle program would involve the irradiation of a series of trial assemblies beginning about the year 1975, initially to check validity of possible manufacturing routes arising within the fabrication plant development program, and later to include studies of the operational and fuel management aspects of recycle. Fuel for the initial stages of a demonstration program would be manufactured in laboratory and development facilities which have already provided mixed oxide fuel that has been irradiated in a number of different types of reactors. The fuel reprocessing plant in England has been shut down since a chemical explosion that occurred in 1973.[46] When that explosion occurred, the plant was starting up for a new processing campaign using the tritex (dibutyl carbitol) solvent extraction process. Fission product residues (mainly ruthenium-106) from previous processing operations were released inside the building and 35 employees received fission product contamination of skin and lungs. No health effects have been observed, and no offsite contamination occurred. Current plans call for a 1,000 metric ton per year plant to be started up in the early 1980's.

Sweden: The accumulated plutonium from Swedish nuclear power plants is estimated to be 1.4 tons by 1980 and 15 to 18 tons by the year 1990. Because it appears improbable that breeder reactors will be introduced commercially before the 1990's, it is likely that the plutonium will be recycled as fuel. This is not expected to start before the year 1979.

Development work is in progress along several different lines. The critical facility KRITZ at Studsvik is large enough to accommodate full length assemblies, and measurements can be performed at different temperatures up to 250°C. At the plutonium laboratory at Studsvik, mixed oxide pellets have been produced for 10 years for internal experiments and, more recently, for AECL. Experimental fuel pins have been irradiated with the aim of studying fabrication parameters.

Demonstration irradiations of plutonium fuel started in the Agesta PHWR in the year 1966, in cooperation with the UKAEA. The first plutonium fuel to be used in an LWR is represented by three assemblies which have been loaded into Oskarshamn I. ASEA-ATOM is responsible for the design and manufacture of the island-type assemblies, but since there is currently no fabrication of such elements on a commercial scale in Sweden, the mixed oxide rods were obtained from Belgonucleaire.

APPENDIX C

BIBLIOGRAPHY

Page intentionally blank

APPENDIX C. BIBLIOGRAPHY

Abe, T., et al. "Results From the Load Follow Experiment of MOX Fuel." *Proc. Enlarged Halden Programme Group Mtg.*, Bolkeso, Norway, Feb. 11-15, 1990.

Abe, T., et al. "Failure Behavior of Plutonium-Uranium Mixed Oxide Fuel Under Reactivity-Initiated Accident Conditions." *J. of Nucl. Mat.*, **188**, (1992) 154-161.

Abel, J. S. "Quad Cities Unit 1 First Reload, Supplement 2 - Mixed Oxide Fuel - Docket Nos. 50-254." NRC Docket 50254-304, (Dec. 14, 1973).

Abel, J. S. "Quad Cities Station, Unit 1, Report of Startup Testing with 8x8 Array Fuel, AEC Docket 50-254." NRC Docket 50254-473, (Sept. 10, 1974).

Adelfang, P., et al. "Fabricación Y Control De Barras Combustibles De Oxidos Mixtos $(U,Pu)O_2$, Para Ensayo De Irradiación En El Reactor HFR-Petten." XIV Reunión Científica de la AATN, Córdoba, España, 1986 (60).

Aitken, E. A., S. K. Evans, and D. C. Wadekamper. *Behavior of Discrete PuO2 Particles in a Thermal Recycle Fuel During SPERT Power Transients, NEDC-12146.* (Nov. 1970).

Aitken, E. A., et al. *A Thermodynamic Data Program Involving Plutonia and Urania at High Temperatures, GEAP-12229.* General Electric Co. (1971).

Akie, H., et al. "A New Fuel for Once-Through Weapons Material Burning." *Nucl. Tech.*, **107**, 182 (1994).

Albright, D., F. Berkhout, and W. Walker. *World Inventory of Plutonium and Highly Enriched Uranium 1992.* Oxford Univ. Press, London (1993).

Alldred, K. "Development and Fabrication Experience of MOX Fuel in the UK." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Aloia, A., A. Ariemma, A. Buono, and F. Velona. "Safety Aspects of Plutonium-Fueled Fast and Thermal Reactors." *Proc. Int. Conf. on the Safety of Fast Reactors*, Aix-en-Provence, 19-22 Sep. 1967. CONF-670916.

Ankney, R. D., and D. Biswas. "Weapons-Grade Plutonium Disposition in Existing Westinghouse PWRs." *Trans. Am. Nucl. Soc.*, **71**, (1994) 488.

Anderson, H. J. and H. R. Wisely. *Specification for Nuclear Grade Mixed Oxides (UO_2-PuO_2), Compactable Powder, HW-78876.* Hanford Atomic Products Operation, General Electric Co. (1963).

Arcelli, G., A. Ariemma, M. Cometto, et al. "Italian Progress on LWR Fuel Design, Manufacturing, and Testing and Management." *Proc. Int. Conf. on Nuclear Power and its Fuel Cycles*, Salzburg, Austria, 2 May 1977. IAEA-CN-36/310, CONF-770505-161.

Arcipiani, B., U. Caldognetto, A. Gibella, and F. Pistella. "Calculations and Experiments for a Pu Fueled Microlattice With Gd as a Burnable Poison." *Energ. Nucl.* (Milan), **17**: 312-16 (May 1970).

Ariemma, A., U. Belelli, and L. Sani. "Assessment and Experimental Investigation of Plutonium Potential in ENEL Water Reactors." *Energ. Nucl.* (Milan) **14**: 401-12 (July 1967).

Ariemma, A., U. Belelli, and L. Sani. "Assessment and Experimental Investigation of Plutonium Potential in ENEL Water Reactors." *Proc. Use of Plutonium as a Reactor Fuel*, Brussels. IAEA, Vienna (1967).

Ariemma, A., G. Lesnoni, M. P. Gualandi, et al. "Accuracy of Power-Distribution Calculation Methods for Uranium and Plutonium Lattices Based on Recent Experiments and ENEL Reactor Operating Data." *Trans. ANS*, **11**: 590-1 (Nov. 1968).

Ariemma, A., L. Bramati, U. Cammorota, et al. "Measurements of Prototype Plutonium Elements for the Garigliano Reactor." *Ing. Nucl.* **10**: No. 4, 23-37 (Jul-Aug 1969).

Ariemma, A., et al. "Accuracy of Power Distribution Calculation Methods for Uranium and Plutonium Lattices Based on Recent Experiments and ENEL Reactor Operation Data." Proc. Amer. Nucl Soc. Inter. Conf., Washington, D.C., Nov. 1968. Amer. Nucl. Soc. (1969).

Ariemma, A., U. Belelli, G. Lesnoni, et al. *Results into the Inquiry into the Prospects for the Use of Plutonium in ENEL Water Reactors*. AEC-tr-6946.

Ariemma, A., L. Bramati, U. Cammarota, et al. *Open-Vessel Experiments on Plutonium Prototype Elements in the Garigliano Reactor*. EUR-4475. Commission of the European Communities, Brussels (Belgium), 1970.

Ariemma, A., G. Lesnoni La Parola, M. P. Gualandi, et al. "Accuracy of Power-Distribution Calculations Methods for Uranium and Plutonium Lattices." *Nucl. Appl. Tech.*, **8**: 328-54 (Apr. 1970).

Ariemma, A., M. P. Gualandi, I. Rosa, and B. Zaffiro. "Evaluation of Alternative Solutions for Plutonium Utilization in Light-Water Reactors." *Trans. ANS*, **15**, No. 1, 112-13 (Jun. 1972).

Ariemma, A. "Main Results of ENEL Garigliano BWR Plutonium Demonstration Program up to 1971." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Ariemma, A., et al. "Italian Experience on Fuel Performance." *Proc. Fourth Int. Conf. Peaceful Uses of Atomic Energy*, A/CONF 49/P-188. United Nations and International Atomic Energy Agency, Vienna (1972).

Ariemma, A., U. Belelli, M. P. Gualandi, et al. *Plutonium Recycle in ENEL's Light Water Reactors.* EUR-4794. Ente Nazionale per l'Energia Elettrica, Rome, Italy, 1972.

Ariemma, A. *ENEL's Viewpoint on Pu Recycle in Water Reactors.* INIS-mf-830. Ente Nazionale per l'Energia Elettrica, Rome, Italy, 1973.

Ariemma, A., M. Paoletti, B. Zaffiro, et al. "Contribution to Evaluation of Mixed Oxide Core Performance From Data Obtained in Italian Experimental Programs." *Proc. IAEA Conf. on Plutonium Utilization in Thermal Power Reactors*, Karlsruhe, 25-29 Nov. 1974. IAEA-175, IAEA-PL-632/10. IAEA, Vienna, 1975.

Ariemma, A. "Main Achievements in Italian Reactors on Plutonium Recycle in Thermal Reactors." *Proc. IAEA Conf. on Plutonium Utilization in Thermal Power Reactors*, Karlsruhe, 25-29 Nov. 1974. IAEA-175, IAEA-PL-632/9. IAEA, Vienna, 1975.

Ariemma, A., G. M. Paoletti, C. Lepscky, et al. *Experience of Plutonium Recycle in Italy.* INFCE/DEP/WG-4/51. IAEA.

Ariemma, A., C. Lepscky, A. Linari, et al. "Experience on Plutonium Recycle in Italy." *Proc. ANS Top. Mtg. on the Plutonium Fuel Cycle*, Miami Beach, FL, USA, 2 May 1977. ANS, La Grange Park, IL, 1977. CONF-770554.

Ariemma, A., M. Cuzzaniti, B. Zaffiro, et al. "ENEL's Experience in the Management of Irradiated Fuel. Old and New Problems Encountered by Nuclear Station Operators." *Proc. Int. Conf. on Nuclear Power Experience*, Vienna, 13 Sep. 1982. CONF-820914-Vol. 3, IAEA-CN-42/343. IAEA, Vienna, 1983.

Ariemma, A., S. Cao, E. Crispino, and P. Venditti. "Italian Reprocessing Activities and Policy." *Proc. ANS Int. Top. Mtg. on Fuel Reprocessing and Waste Management*, Jackson Hole, WY, USA, 26 Aug. 1984. CONF-840802. ANS, La Grange Park, IL, 1984.

Asashi, K., et al. "Irradiation and Post Irradiation Testing Program of BWR MOX Fuel Rods." *Proc. ANS Int. Top. Mtg. On Light Water Reactor Fuel Performance*, West Palm Beach, Florida, April 17-21, 1994.

Atkins, R. J., et al. "Preparation and Properties of Spherical Particulate Nuclear Fuels for Plutonium-Fueled Reactors." *Proc. Plutonium as a Power Reactor Fuel*, Richland, Washington, Sept. 13-14, 1962, HW-75007. Hanford Atomic Products Operation (1962).

Bahl, J. K. and M. D. Freshley. "Plutonium and Fission Product Migration in Mixed-Oxide Fuels During Irradiation." *Trans. Am. Nucl. Soc.*, **13**, 599 (1970).

Bahl, J. K. and M. D. Freshley. "Plutonium and Fission Product Redistribution in Mixed-Oxide Fuels During Irradiation." *Nucl. Tech.*, **15**, (Aug. 1972) 114.

Bailey, W. J. *Investigation of Plutonium-Bearing Fuel Elements for the Plutonium Recycle Test Reactor, Part I, USAEC Report HW-70158.* Hanford Atomic Products Operation, General Electric Co., (June 1961).

Bailey, W. J., M. D. Freshley, and R. E. Sharp. "The Irradiation Testing of a Plutonium Fuel Element Concept at Hanford." *Trans. Am. Nucl. Soc.*, **5** (1), (1962) 252. [Also USAEC Report HW-SA-2425].

Bailey, W. J. and T. D. Chikalla. "Irradiation of UO_2/PuO_2 Capsules." *Trans. Am. Nucl. Soc.*, **5** (2), (1962) 467. [Also *USAEC Report HW-SA-2675*].

Bailey, W. J. and T. D. Chikalla. "Irradiation of Uranium-Plutonium Oxide." *Trans. Am. Nucl. Soc.*, **6** (2), (1963) 350. [Also USAEC Report HW-SA-3129].

Bailey, W. J. "Irradiation Testing in Support of the Plutonium-Recycle Demonstration Experiment in EBWR." *Trans. Am. Nucl. Soc.*, **7** (2), (1964) 389.

Bailey, W. J. and M. D. Freshley. *Irradiation Properties of High Energy Rate Pneumatically Impacted UO_2-PuO_2 Fuels, BNWL-356.* Battelle-Northwest Laboratory (1967).

Baily, W. E. *Use of Plutonium in Boiling Water Reactors Phase II Extension Interim Report Covering Period July 1972 to December 1974, EPRI-72-2.* Electric Power Research Institute, Palo Alto, (June 1975).

Bairiot, H. and A. Lhost. *Plutonium Recycle in Thermal Reactors – Present Status of the Belgian Program, BN-6905-06.* Belgonucléaire (April 1969). Also in *Trans. Am. Nucl. Soc.*, **12**, (1969) 35.

Bairiot, H., et al. Belgian Demonstration Program on Pu Recycle – Part I: Pu Recycle in the BR3 Reactor, BN-7106-06. SCK/CEN-Belgonucléaire Association, Brussels (June 1971). Also in *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Bairiot, H. and H. Van den Broeck. "Plutonium Recycling in Light Water Reactors." *Proc. Fourth Int. Conf. Peaceful Uses of Atomic Energy*, A/CONF 49/P-299. United Nations and International Atomic Energy Agency, Vienna (1972).

Bairiot, H., P. Deramaix, N. Hoppe, and B. van Outryve d'Ydewalle. "Irradiation of Plutonium Bearing Fuel Clad With Zircaloy." *B.N.E.S. Conf. Proc. 61.1/61.5*, Oct. 1973.

Bairiot, H., et al. "Pu-Coated Particle Development." *Nucl. Tech.*, **23** (1974).

Bairiot, H., M. Gaube, and B. van Outryve d'Ydewalle. "Irradiation Behavior of LWR Plutonium Fuel Rods." *Nucl. Energy Maturity, Proc. European Nuclear Conf.*, Paris, France, April 21-25, 1975.

Bairiot, H., P. Deramaix, C. Vandenburg, et al. "Plutonium Assemblies in Reload 1 of the Dodewaard Reactor." *Nucl. Tech.*, **33** (April 1977) 184.

Bairiot, H. "Laying the Foundations for Plutonium Recycle in Light Water Reactors." *Nucl. Eng. Int.*, **29**, 27 (1984).

Bairiot, H., et al. "LWR MOX Fuel Experience in Belgium and France With Special Emphasis Placed on the Results Obtained in the BR3 PWR." *Proc. Int. Symp. on Improvements in Water Reactor Fuel Technology and Utilization*, Stockholm, Sweden, Sept. 15-19, 1986. IAEA (1987).

Bairiot, H., et al. "Collaboration Between France and Belgium on Plutonium Recycle," IAEA-SM 294/10. Presented at Int. Symp. on the Back-End of the Nuclear Fuel Cycle – Strategies and Options, Vienna, Austria, May 11-15, 1987.

Bairiot, H., and G. Lebastard. "Recent Progress of MOX Fuels in France and Belgium." Presented at Int. Conf. On Nuclear Power Performance and Safety, Vienna, Sept. 28 - Oct. 2, 1987. IAEA-CN-48-101.

Bairiot, H. and C. Vandenburg. *Use of MOX Fuels, Nuclear Fuel Cycle in the 1990s and Beyond the Century: Some Trends and Foreseeable Problems.* Technical Report Series No. 305, IAEA (1989).

Bairiot, H. and E. van der Bemden. "MOX Fuel Delivers the Goods." *Nucl. Eng. Int.*, (March 1990).

Bairiot, H., et al. "Foundations for the Definition of MOX Fuel Quality Requirements." *Proc. Int. Conf. On Characterization and Quality Control of Nuclear Fuels*, Karlsruhe, Germany, June 19-21, 1990. Also in *J. Nucl. Mater.*, **178**, 187 (1991).

Bairiot, H., et al. "30 Years Experience on MOX Fuel, From R&D to Industrial Utilization." *Proc. 3rd Int. Conf. Nuclear Fuel Reprocessing and Waste Management*, Sendai, Japan, April 15-18, 1991.

Bairiot, H. and P. Deramaix. "MOX Fuel Development: Yesterday, Today, and Tomorrow." *J. Nucl. Mater.*, **188** (1992) 10-18.

Bairiot, H., et al. *Proc. Nuclex, Technical Mtg. 5*, 7.

Bairiot, H. and C. Vandenberg. "Use of MOX Fuels: The Reasons to Start." Publication information unknown, (~ 1987 - 1989).

Barbrault, P. "A 100% Plutonium-Fueled PWR For 21st Century." *Nuclear Europe Worldscan*, **3-4**, 1994.

Barbrault, P. "A Plutonium-Fueled PWR 2000." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Bardsley, R. E. and R. E. Sharp. "Fabrication of UO_2-PuO_2 Swage-Compacted Fuel Elements." *Trans. Am. Nucl. Soc.*, **5** (2), (1962) 452.

Bariteau, J. P., et al. "MOX Fuel Fabrication at the CEA Cadarache Complex." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Baron, D., C. Forat, and E. Maffeis. "FRAGEMA Experience on Fission Gas Release Under Base and Transient Operating Conditions." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17-20 April 1988. ANS, La Grange Park, 1988.

Basselier, J., J. Debrue, L. Leenders, and H. Van den Broeck. "Experimental Data as Reference For Plutonium Recycling in the SENA Reactor." *Nucl. Energy Maturity, Proc. European Nuclear Conf.*, Paris, France, April 21-25, 1975.

Basselier, J., et al. "Analysis and Synthesis of the Theoretical Studies Performed on the Control and Safety of LWR's Burning Plutonium Fuel." *EUR 8118 EN.* Commission of the European Communities (1982).

Basselier, J., Th. Maldague, and M. Lippens. "Validation of MOX Fuel Through Recent Belgonucléaire International Programmes." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, UK, July 3-7, 1995.

Bassler, E. A., et al. *Mechanical, Thermal, and Hydraulic Design of Saxton Partial Plutonium Core, WCAP-3385-52*, (Dec. 1965).

Bates, J. L. and W. E. Roake. *Irradiation Testing of Plutonium-Uranium Oxide Nuclear Fuel, USAEC Report HW-61142.* Hanford Atomic Products Operation, General Electric Co., (July 1959).

Battelle-Northwest Laboratory. *Ceramics Research and Development Operation Quarterly Report, January-March 1965, BNWL-91.* Battelle-Northwest Laboratory (1965).

Battelle-Northwest Laboratory. *Ceramics Research and Development Operation Quarterly Report, July-September 1965," BNWL-198.* Battelle-Northwest Laboratory (1966).

Battelle-Northwest Laboratory. *Plutonium Utilization Annual Report – Fiscal Year 1967, BNWL-624.* Battelle-Northwest Laboratory (1968).

Battelle-Northwest Laboratory. *Plutonium Utilization Program Technical Activities Quarterly Report – September, October, November 1969, BNWL-1289.* Battelle-Northwest Laboratory (1970).

Battelle-Northwest Laboratory. *Plutonium Utilization Program Technical Activities Quarterly Report – December 1969, January, February 1970, BNWL-1358.* Battelle-Northwest Laboratory (1970).

Battelle-Northwest Laboratory. *Plutonium Utilization Program Technical Activities Quarterly Report – March, April, May 1970, BNWL-1442.* Battelle-Northwest Laboratory (1970).

Baumier, J. "Economic Aspects of Plutonium Recycling in PWRs." *Proc. Int. Symp. Back End of the Nuclear Fuel Cycle: Strategies and Options*, Vienna, Austria, 1987. IAEA-SM-294/33, p. 529, IAEA (1987).

Bay, H., et al. "MOX Fuel Reloads in Beznau-I." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Bay, H., et al. "Debris Induced Fuel Failures on U and MOX Fuel in Beznau-1." *Proc. Tech. Com. Mtg. on Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms, and Management*, Dimitrovgrad, Russian Federation, May 26-29, 1992.

Bean, C. H., R. E. Sharp, and W. J. Bailey. "The EBWR Plutonium Recycle Demonstration Experiment." *AIME Symposium on Plutonium Fuels Technology*, Scottsdale, Arizona, (Oct. 1967).

Beauvy, M. "Nonideality of the Solid Solution in $(U,Pu)O_2$ Nuclear Fuels." *J. of Nucl. Mat.*, **188** (1992) 232-238.

Beer, O., and G. J. Schlosser. "Advanced Uranium Utilization in LWR by Recycling of Uranium and Plutonium." *Proc. Technical Committee and Workshop on Advanced Light and Heavy Water Reactor Technology*, Vienna, Austria, November 1984. IAEA-TECDOC-344, p. 91, IAEA, Vienna (1985).

Beer, O., G. Schlosser, and F. Spielvogel. "Experience With Thermal Recycling of Plutonium and Uranium." *Trans. Am. Nucl. Soc.*, **49**, Suppl. 1, (1985) 224.

Bekiarian, A. and G. Le Bastard. "MOX Manufacturing Perspectives in a Fast Growing Future and the MELOX Plant." *Proc. RECOD '91*, Sendai, Japan, Sept. 14-18, 1991.

Bekiarian, A. and J. L. Nigon. "MOX and MELOX Nuclear Fuels." *Revue de l'Énergie*, **433**, (1991), 606.

Belgonucléaire. *General Description of the Belgian Program on Pu Recycling in Thermal Reactors, BN-7106-02*. Belgonucléaire (1971). Also in *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Bell, B. T., et al. "The Development and Manufacture of MOX Fuels in the United Kingdom." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Benesovsky, F. *Powder Metallurgy in the Nuclear Age*. Metallwerk Plansee AG., Reuttle/Tyrol (1962).

Bennett, C. L. *Physics Calculations for Mixed PuO_2-UO_2 NPR Loadings, HW-83193*. Hanford Atomic Products Operation, Richland, (June 1964).

Bennett, R. A. and L. C. Schmid. *Approach to Critical and Calibration Experiments in the Plutonium Recycle Critical Facility, HW-80206*. General Electric Company (1964).

Bergeron, J., L. Martin Deidier, and A. Darraud. *Les combustibles a oxydes mixtes. L'adaptation du coeur et des chaudieres rep aux recharges MOX. Les aspects physiques dans la*

conception des assemblages plutonium, CEA-CONF-9259. Presented at SFEN/BNES meeting – technical session on nuclear fuel cycle, Paris, Oct. 7-10, 1987.

Berthet, A., et al. "Actions et experimentations menees en France pour l'introduction du plutonium dans les reacteurs a eau pressurisee (REP)," IAEA-SM 294/75. Presented at Int. Sump. on the Back End of the Nuclear Fuel Cycle – Strategies and Options, Vienna, Austria, May 11-15, 1987.

Berthet, A., M. Morel, and J. F. Marin. *Recyclage du plutonium dans les reacteurs a eau sous pression: programme de qualification du combustible a oxyde mixte (MOX), CEA-CONF-9356.* Presented at Int. Conf. On Nuclear Power Performance and Safety, Vienna, Sep. 28 - Oct. 2, 1987.

Berthet, A., et al. "Qualification Program for Mixed Oxide Fuel." Proc. SFEN/BNS Mtg. Tech. Session on Nuclear Fuel Cycle, Paris, Oct. 1987. CEA-CONF-9352.

Biancheria, A. "Characterization Methods for Ceramic-Grade Uranium Dioxide, Plutonium Dioxide, and Mixed Uranium-Plutonium Dioxide Powder." *Proc. Characterization of Uranium Dioxide*, Oak Ridge, Tennessee, Dec. 12-13, 1961. USAEC Report TID-7637, Oak Ridge National Laboratory (Oct. 1962).

Biancheria, A., et al. *Materials Design and Fabrication of the Saxton Partial Plutonium Core, WCAP-3385-53*, (Dec. 1965).

Bibilashvili, Yu. K., and F. G. Reshetnikov. *Russia's Nuclear Fuel Cycle: An Industrial Perspective. IAEA Bulletin 3*, 1993.

Billaux, M. and J. van Vliet. "Impact of Fuel Heterogeneities on Fission Gas Release for LWR U-Pu Mixed Oxide Fuels." *Res Mechanica*, **17**, (1986) 41-57.

Bindler, L., et al. "Calcul de reseaux UO_2-PuO_2-H_2O." *Proc. Use of Plutonium as a Reactor Fuel*, Brussels, March 1967. IAEA, Vienna (1967).

Bindler, L., et al. "A Program of Experiments with UO_2-PuO_2 Fuels in a Thermal Critical Facility and Their Interpretation." Presented at Deutsches Atomforum Mtg., Frankfurt (April 1969).

Bindler, L., et al. "Some Highlights of Three-Year Belgian Critical Experiment and Code Calibration Work on UO_2-PuO_2-H_2O Lattices." Presented at Deutsches Atomforum Mtg., Frankfurt (April 1969).

Bindler, L., L. Leenders, and H. Van den Broeck. "Experimental and Theoretical Physics Work on Plutonium Enriched LWR's Lattices." Euratom Rep. EUR 4034 (1970). Also reported as Blg-440 (1970) and as BN-7006-03, Belgonucléaire (1970).

Bindler, L., et al. "Belgian Plutonium Recycle Program – Final Experimental Assessment of the Methods of Physics Calculations." *Trans. Am. Nucl. Soc.*, **15**, (1972) 954.

Bishop, D. M. and W. E. Baily. *Non-destructive Examination of Plutonium Recycle Fuel Rods Irradiated to 30,000 MWd/t, NEDO-12562 [or NEDO-12552].* General Electric Co., San Jose, (Dec. 1974).

Biswas, D., R. W. Rathburn, and H. D. Garkisch. "Characteristics of a 100% MOX-Fueled PWR Design for Weapons Plutonium Disposition." *Trans. Am. Nucl. Soc.*, **69**, (1993) 95.

Biswas, D., et al. "Weapons-Grade Plutonium Disposition in Pressurized Water Reactors." *Nucl. Sci. and Engr.*, **121**, (1995) 1-16.

Black, T. J., S. Y. Ogawa, and J. S. Armijo. *Use of Plutonium Fuel in Boiling Water Reactors, EPRI-NP-223.* Electric Power Research Institute, Palo Alto, (Aug. 1976).

Black, T. J., et al. *Use of Plutonium in Boiling Water Reactors Destructive Examination of PuO2-UO2 Fuel Rods Irradiated in Big Rock Point to 30,000 MWd/T, EPRI-NP-233.* Electric Power Research Institute, Palo Alto, (August 1976).

Bladini, F., C. Lepcsky, F. Pistella, et al. *Post-Irradiation Examinations of 5 Fuel Rods From MO Bundles Irradiated in the Garigliano BWR up to 15 MWd/kg. Topical Report on Nondestructive and Destructive Results.* EUR-7144-EN. Commission of the European Communities, Luxembourg.

Blanchon, T. *Plutonium Recycling: Electricite de France's Point of View.* Electricite de France, 1994.

Blanpain, P., X. Thibault, and M. Trotabas. "MOX Fuel Experience in French Power Plants." *Proc. ANS Int. Top. Mtg. On Light Water Reactor Fuel Performance*, West Palm Beach, Florida, April 17-21, 1994.

Blanpain, P., et al. "Plutonium Recycling in French Power Plants: MOX Fuel Irradiation Experience and Behavior." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, UK, July 3-7, 1995.

Bliss, H. E. "Proposed Change No. 22 to Appendix A, DPR-2, to Permit Operation of Cycle 8." NRC Docket 50010-77, (Jan. 18, 1971).

Bloomster, C. H., R. E. Bardsley, and W. T. Ross. "An Incremental-Loading Process to Control Plutonium Enrichment in the Mechanically Mixed Oxide-Fuel Systems." *Trans. Am. Nucl. Soc.*, **5** (2), (1962) 452.

Bober, M., C. Sari, and G. Schumacher. "Plutonium Migration in a Thermal Gradient in Mixed (U,Pu) Oxide Fuels." *Trans. Am. Nucl. Soc.*, **12**, (1969) 603.

Bober, M., C. Sari, and G. Schumacher. "Redistribution of Plutonium and Uranium in Mixed (U,Pu) Oxide Fuel Materials in a Thermal Gradient." *J. Nucl. Mater.*, **39**, (1971) 265.

Boczar, P. G., I. J. Hastings, and A. Celli. *Recycling in CANDU of Uranium and/or Plutonium From Spent LWR Fuel, AECL-10018.* Also in *Proc. Tech. Committee Mtg. On Recycling of*

Plutonium and Uranium in Water Reactor Fuels, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Boczar, P. G., et al. "Plutonium Burning in CANDU." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Bohm, W., G. Schlosser, and H. Spierling. "Nuclear Core Design and Performance of KWU LWRs With Plutonium Recycle." *Proc. IAEA Panel on Plutonium Recycling*, Karlsruhe, Nov. 1974. IAEA, Vienna (1974).

Bonet, H., et al. "Plutonium Recycle Concepts For RCC Type PWRs." *Nucl. Energy Maturity, Proc. European Nuclear Conf.*, Paris, France, April 21-25, 1975.

Brandberg, S. G. "The Conversion of Uranium Hexafluoride to Uranium Dioxide." *Nucl. Tech.*, **18**, (May 1973) 177.

Brite, D. W. and C. A. Burgess. "High Energy Rate Pneumatically Impacted UO_2-PuO_2 Fuels." *Trans. Am. Nucl. Soc.*, **7**, (1964) 408.

Brite, D. W. "Plutonium Fuel Technology Part I: Plutonium Fuel Fabrication." *Nucl. Tech.*, **18**, (May 1973) 87.

Brown, C. L. and D. R. Oden. "Critical Parameters for 0 to 7% PuO_2-U(Nat)O_2 Rods in Water." *Trans. Am. Nucl. Soc.*, **12**, (1969) 873.

Brown, C. L., L. C. Davenport, and D. P. Oden. "Plutonium Fuel Technology Part III: Nuclear Criticality Safety Considerations in LWR (Pu,U)O_2 Fuel Fabrication." *Nucl. Tech.*, **18**, (May 1973) 109.

Brown, J. H., and Nilson. "A Survey of Plutonium Recycling in the US." *J. Brit. Nucl. Energy Soc.*, **13**, 2, (1974) 141.

Brzoska, B., G. H. Chalder, and P. Dewes. *Proc. Enlarged Halden Program Group Mtg.*, Sanderstolen, 1977.

Buck, P, W. Brooks, and E. Fass. *PLATR Critical Experiments on Mixed-Oxide Fuel Assemblies for BWR's and Comparison with Calculations, GU-5289.* Gulf-United Nuclear Fuels Corp. (1973).

Buckner, M. R., et al. *Strategies for Denaturing the Weapons-Grade Plutonium Stockpile, WSRC-RP-92-1004.* Westinghouse Savannah River Company (1992).

Burley, T. B. and M. D. Freshley. "Internal Gas Pressure Behavior in Mixed-Oxide Fuel Rods During Irradiation." *Nucl. Appl. Tech.*, **9**, 233 (1970).

Burnham, J. B., L. G. Merker, and D. E. Deonigi. *Comparative Costs of Oxide Fuel Elements – Vol. I – The Computed Costs of Fuel Elements (Vibrational Compacted or Pellet) Enriched with*

^{235}U *or Low Exposure Plutonium or High Exposure Plutonium, BNWL-273, Vol. I.* Battelle-Northwest Laboratory (1966).

Bussey, P. "The Plutonium Based Fuel Production Programmes in France." *Plutonium as a Power Reactor Fuel – Proc. Am. Nucl. Soc. Top. Mtg*, Richland, Washington, Sept. 13-14, 1962. USAEC Report HW-75007, Hanford Atomic Products Operation, General Electric Co., and USAEC, (Dec. 1962).

Butler, E. M. and R. O. Meyer. "Diffusion of Plutonium and Uranium in Irradiated Mixed-Oxide Fuel." *J. Nucl. Mat.*, **47**: (1973) 229-240.

Butterfield, L. D., Jr. "Proposed Change to Appendix A, DPR-2, to Permit Operation of Cycle 9." NRC Docket 50010-124. (Jan. 3, 1973).

Cadelli, N., and M. Lippens. "An European Investigation of MOX Fuel Performance in LWR's." Presented at Int. Conf. on Nuclear Power Performance and Safety, Vienna, 1987. IAEA-CN-48-157.

Caillot, L., et al. "Thermal and In-Pile Densification of MOX Fuels: Some Recent Results." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, UK, July 3-7, 1995.

Calabro, R. *Test of Plutonium Recycle Nuclear Prediction Methods for Heterogeneous Lattices and Reactor Experiments, UNC-5211.* (April 15, 1968).

Caldwell, C. S. "Relationship Between Process Variables and Final Properties of Ceramic Grade UO_2, UO_2•PuO_2." *Proc. Characterization of Uranium Dioxide*, Oak Ridge, Tennessee, Dec. 12-13, 1961. USAEC Report TID-7637, Oak Ridge National Laboratory, (Oct. 1962).

Caldwell, C. S., et al. "A Continuous Coprecipitation Process for Production of Reactor-Grade (U-Pu)O_2 Mixed Oxide Powder." *Trans. Am. Nucl. Soc.*, **6** (2), (1963) 394.

Caldwell, C. S. and K. H. Puechl. "Plutonium-Uranium Dioxide Powder and Pellet Manufacture." *AIME Symposium on Plutonium Fuels Technology*, Scottsdale, Arizona, (Oct. 1967).

Carlson, M. J. C. *Water and Sorbed-Gas Content of Hypostoichiometric Mixed Oxide Fuel Pellets, WHAN-SA-77*, (1970).

Carroll, R. M. and O. Sisman. "Behavior of Mixed Oxide (U,Pu)O_2 Fuels as a Function of Temperature and Burnup." *Nucl. Tech.*, **11**, (Aug. 1971) 578.

Carver, J. G., et al. *Program for the Development of Plutonium Recycle for Use in Light Water Moderated Reactors. Program Summary. Sixth Quarterly Report, July 1 - September 30, 1962, USAEC Report GEAP-4081.* United States - Euratom Joint Research and Development Program, (Oct. 15, 1962).

Carver, J. G. *Plutonium Subcritical Experiment Program – Final Report, GEAP-5357.* General Electric Co. (1967).

Castelli, R., et al. "Le combustible au plutonium FRAGEMA dans les reacteurs a eau sous pression," IAEA-SM 294/18. Presented at Int. Symp. on the Back-End of the Nuclear Fuel Cycle – Strategies and Options, Vienna, Austria, May 11-15, 1987.

Castelli, R. and A. Chotard. "FRAGEMA MOX Fuel in Pressurized Water Reactors." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17-20 April 1988. ANS, La Grange Park, 1988.

Castelli, R., M. Fremont, and J. Pelet. "First Feedbacks of Plutonium Recycling and Experimental Validation." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Caye, T. E. *Saxton Plutonium Project – Semi-Annual Progress Report for the Period Ending March 31, 1971, WCAP-3385-27.* Westinghouse Electric Corp. (1971).

Caye, T. E. *Saxton Plutonium Project, Quarterly Progress Report for the Period Ending 3/31/72, WCAP-3385-31.* Westinghouse Electric Corp. (Nov. 1972).

Caye, T. E. *Saxton Plutonium Project – Quarterly Progress Report for the Period Ending June 30, 1972, WCAP-3385-32 (or 12?).* Westinghouse Electric Corp. (1972).

Celnik, J., et al. *Evaluation of Plutonium Recycle Nuclear Calculation Methods by Comparison with Experimental Data, UNC-5168*, (Feb. 28, 1967).

Ceramics Research and Development Operation. *Specification for Swage-Compacted, Mixed-Oxide (UO_2-PuO_2), Fuel Elements for the PRTR, (Mark I-M), HW-79290.* Hanford Atomic Products Operation (1963).

Ceramics Research and Development Operation. *Specifications for Vibrationally Compacted, Mixed-Oxide (UO_2-PuO_2), Fuel Elements for the PRTR, (Mark I-L), HW-79291.* Hanford Atomic Products Operation (1963).

Chalder, G. H., P. Dewes, and D. B. Scott. *Proc. Enlarged Halden Program Group Mtg.*, Sanderstolen, 1976.

Chamberlain, A. and C. Melches. "Prospects for the Establishment of Plutonium Recycle in Thermal Reactors in the Foratom Countries: Status and Assessment." *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2-13 May 1977. IAEA, Vienna, 1977.

Charlier, A., et al. "Use of BR3 for the Belgian Plutonium Recycle Irradiation Program." *Proc. Specialists' Mtg. Improved Utilization of Water Reactor Fuel With Special Emphasis on Extended Burnups and Plutonium Recycling*, Mol, Belgium, May 1984. IWGFPT/20, p. 208, IAEA, Vienna (1984).

Charlier, A., J. Basselier, and L. Leenders. "VENUS International Program (VIP), A Nuclear Data Package for LWR Pu Recycle." *Proc. Int. Conf. Physics of Reactor Operation, Design, and Computation (PHYSOR '90)*, Marseille, France, April 23-26, 1990.

Chikalla, T. D. *Melting Behavior in the System UO_2-PuO_2, USAEC Report HW-SA-2460.* Hanford Atomic Products Operation, General Electric Co., July 1962. [Also in *J. Am. Ceram. Soc.*, **46**, (1963) 323.]

Childs, G. D. and E. P. Gerstenhaber. *IFRC-MOX Product Mix, Fuel Supply Requirements and Interfacing Fuel Cycle Service Supply Activities, IFRC-77-259.* Westinghouse Electric Corp., Pittsburgh, PA (1977).

Chotard, A., et al. "PROMOX – The French R&D Program on MOX Fuel." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Christensen, D. E., et al. *Plutonium Recycle Burnup Data and Results from the Batch Core Experiment, BNWL-1674.* Pacific Northwest Laboratories (1971).

Cina, B. "Some Segregation Effects in Sintered UO_2-PuO_2 Pellets." *J. Nucl. Mat.*, **9**, (1963) 111.

Clark, H. J. and J. M. Boswell. *LWR Fuel Recycle Facilities – Capacity of Conceptual Design, DPSTD-LWR-76-100.* Savannah River Laboratory, E. I. du Pont de Numours and Co., Aiken, SC (1976).

Cleveland, J. M. *The Chemistry of Plutonium.* New York: Gordon and Breach, Science Publishers, (1970).

Clemente, M. and S. Langenbuch. "Untersuchungen zum Einsatz von MOX-Brennelementen. (Investigations of the use of MOX fuel elements)." BMU-1991-277, GRS-A-1758. BMU, Bonn, 1991.

COGEMA. *The Nuclear Fuel Cycle.* Undated marketing publication.

COGEMA. *The Reprocessing Division.* Undated marketing publication.

Coluccio, G., M. Galliani, and A. Linari. "Experience in the Fabrication of Fuel Bundles for the Garigliano Power Plant and Their Performance in Operation." *Proc. IAEA Int. Conf. on Fabrication of Water Reactor Fuel Elements*, Prague, Czechoslovakia, 6 Nov. 1978. CONF-781196, IAEA-SM-233/11.

Cornet, G., et al. "European Experience and Potential in Use of Plutonium in LWRs." *European Nuclear Soc. TOPNUX93 Mtg.*, The Hague, Apr. 25-28, 1993.

Corona, R., et al. "Messa a punto di un metdod per il calcolo di noccioli arricchiti in plutonio ai fini del riciclo del plutonio nei reattori termici." EUR-3891. European Atomic Energy Community (1968).

Coudron, P. E., R. P. Matsen, and L. C. Schmid. "The Determination of Ratios of Effective Cross Sections from Burnup Data from the Saxton Plutonium Program." *Nucl. Tech.*, **15**, (Aug. 1972) 229.

Couty, Y. and R. Lorenzelli. "MELOX: Impact of Material and Operational Constraints on the Design of a Large Capacity Plant." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Crain, H. H. *Preliminary Criteria for the Design of a Fuel Fabrication Facility, FPER-77-137.* Westinghouse Electric Corp., Pittsburgh, PA (1977).

Crowther, R. L. *Plutonium Utilization in Boiling Water Reactors Phase II Semi Annual Report Jan.-June 1971, NEDC-10586.* General Electric Co., San Jose, (Dec. 1971).

Crowther, R. L. *Plutonium Utilization in Boiling Water Reactors Phase II Semi Annual Report July-Dec. 1971, NEDC-10747.* General Electric Co., San Jose, (Nov. 1972).

Crowther, R. L. *Plutonium Utilization in Boiling Water Reactors Phase II Semi Annual Report Jan.-June 1972, NEDC-10796.* General Electric Co., San Jose, (Jan. 1973).

Crowther, R. L., et al. "BWR Pu Recycle." *Trans. Am. Nucl. Soc.*, **16** (2), (Nov. 1973).

Cutrone, M. B. and G. F. Valby. *Gamma Scan Measurements at Quad Cities Nuclear Power Station Unit 1 Following Cycle 2, NP-214.* Electric Power Research Institute, Palo Alto, (July 1976).

Dawson, F. G., et al. *Analysis of Plutonium Fueled Light Water Reactors, BNWL-SA-926.* Also in *Proc. Use of Plutonium as a Reactor Fuel*, Brussels. IAEA, Vienna (1967).

Dawson, F. G., et al. "The EBWR Plutonium Demonstration Experiment." *Trans. Am. Nucl. Soc.*, **10**, (1967) 500.

Dawson, F. G. *Program Analysis and Plans – Plutonium Utilization Program – FY-67 through FY-70, BNWL-298.* Battelle-Northwest Laboratory (1969).

Dawson, F. G., et al. *Plutonium Utilization in Light Water Reactors, BNWL-SA-3849.* Battelle-Northwest (1971).

Dawson, F. G. "Thermal Reactor Plutonium Fuel Irradiation Programs in the United States." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972). Also *BNWL-SA-3855*, Battelle-Northwest Laboratory (1971).

Dawson, F. G., et al. "Results from USAEC Plutonium Utilization Program Conducted by Battelle-Northwest." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972). Also *BNWL-SA-3865*, Battelle-Northwest Laboratory (1971).

De Coster, M., and F. Pistella. *Radial Power Distribution Within the Pins of Light Water Moderated UO_2/PuO_2 Fuelled Lattices.* RT/FI-69-50. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1969.

De Waegh, F. *Pu Recycle in Thermal Reactors – Demonstration Program in the BR3 Nuclear Power Plant, BN-240.00/151/n/018.* Belgonucléaire (Oct. 1969). Revised 1970.

De Waegh, F., P. Deramaix, and J. Debrue. *Belgian Demonstration Program on Pu Recycle – Part II: Recycle Concept for the SENA Reactor, BN-7106-08.* SCK/CEN-Belgonucléaire Association, Brussels (June 1971). Also in *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Debrue, J., et al. "Determination experimentale des caracteristiques neutroniques de reseaux UO_2-PuO_2-H_2O." *Proc. Use of Plutonium as a Reactor Fuel*, Brussels, March 1967. IAEA, Vienna (1967).

Debrue, J., P. Deramaix, and F. de Waegh. "Plutonium Recycle Studies for the SENA PWR Reactor." *Nucl. Appl. Tech.*, **9**, (1970) 516.

Debrue, J., et al. "A Basic Experimental Reactor Physics Program on UO_2-PuO_2-H_2O Lattices." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972). Also in *Trans. Am. Nucl. Soc.*, **14**, (1971) 377.

Decoster, M. and F. Pistella. *Radial Power Distribution Within the Pins of Light Water Moderated UO_2-PuO_2 Fueled Lattices. RT/FI(69)50.* Comitato Nazionale per L'Energia Nucleare, Rome (1969).

Denton, H. R. "Storage of Mixed Oxide Fuel." Letter to NRC Commissioners dated Feb. 1, 1980. Nuclear Regulatory Commission (1980).

Deonigi, D. E. "The Value of Plutonium Recycle in Thermal Reactors." *Nucl. Tech.*, **18**, (May 1973) 80.

Depestel, M., et al. "MOX Fuel Experience in the CNA Plant." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Deramaix, P., et al. "Measurements in Cold Critical Assemblies Performed at Mol." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972) 99.

Deramaix, P., et al. "Belgian Demonstration Programme on Plutonium Recycle Part II – Loading of Two Plutonium Assemblies in the Dodewaard Reactor." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Deramaix, P. and A. Charlier. "MOX Fuel Concepts for PWRs and BWRs." *Proc. Improvements in Water Reactor Fuel Technology and Utilization*, Stockholm, Sweden, IAEA (1987).

Deramaix, P., et al. "MIMAS Fuel Performance in Commercial Reactors." *Proc. ANS-ENS Topical Mtg.*, Avignon, France, April 21-24, 1991.

Deramaix, P., D. Haas, and J. Van de Velde. "In-Pile Performance of Mixed-Oxide Fuel With Particular Emphasis on MIMAS Fuel." *Nucl. Tech.*, **102**, (1993) 47.

Devezeaux de LaVergne, J. G. and L. F. Durret. "Plutonium: Economy of Recycle." *Revue generale nucleaire*, **1**, (1995), 40.

Devillers, C., et al. "Safety Problems Relating to Plutonium Recycling in Light Water Reactors." Proc. ENS/ANS/FORATOM Joint European Nuclear Congress and Technology/Industry Exposition: Looking into Nuclear's Future in the 21st Century, Lyon, Sep. 23-28, 1990.

Dibbert, H. J., J. Huber, and S. L. Winnik. "Technische und wirtschaftliche Aspekte der Plutonium-ruckfuhrung." *Atomwirtsch.*, **33**, (1988) 452.

Doshi, P. K. and W. Brooks. *Dresden Plutonium Demonstration Program, Proof Test Facility – Experiments and Analysis, GU-5285.* (Oct. 1972).

Durret, M. F. *Plutonium Recycle in CANDU-Type PHW Heavy Water Reactors, AECL-3910.* Atomic Energy of Canada, Limited (1971).

Durret, L. F. "The Nuclear Fuel Cycle." *RGN*, No. 4, (July-August 1994).

Edlund, M. C. "Plutonium Fuel Development Facilities at Babcock and Wilcox." *Proc. Amer. Nucl. Soc. Plutonium Information Mtg.*, March 1-2, 1966, CONF-660308. U.S. Atomic Energy Commission

Edwards, J., et al. "The Conversion of Recycled Pu into Thermal Reactor MOX Fuel in the UK." *Proc. RECOD '91*, Sendai, Japan, April, 1991.

Egorov, N. N., et al. "Management of Plutonium in Russia." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Eickelpasch, N., H. Schenk, H. Stehle, et al. "Experience With Plutonium Recycle Fuel For Large Light-Water Reactors in the Federal Republic of Germany." *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2-13 May 1977. IAEA, Vienna, 1977.

Electric Power Research Institute. *Use of Plutonium Fuel in Boiling Water Reactors, EPRI 72-2.* Electric Power Research Institute, Palo Alto, (1975).

Electric Power Research Institute. *Core Design and Operating Data for Cycle 1 and 2 of Quad Cities 1, NP-240.* Electric Power Research Institute, Palo Alto, (1976).

Electric Power Research Institute. *Core Design and Operating Data for Cycle 3, NP-552.* Electric Power Research Institute, Palo Alto, (1978).

Electric Power Research Institute. *Mixed Oxide and Uranium Fuel Surveillance at Quad Cities Nuclear Power Station Unit 1 Following Cycle 3, NP-2304-LD.* Electric Power Research Institute, Palo Alto, (1982).

Electric Power Research Institute. *Fuel Inspection Data Summary Report for Initial Mixed Oxide Examination at Quad Cities 1, January, 1976, NP-2305-LD.* Electric Power Research Institute, Palo Alto, (1982).

Electric Power Research Institute. *Mixed Oxide and Uranium Fuel Surveillance at Quad Cities Nuclear Power Station Unit 1 Following Cycle 4, NP-2306-LD.* Electric Power Research Institute, Palo Alto, (1982).

Electric Power Research Institute. *Burnup and Transuranium Element Composition in Irradiated UO_2, UO_2-Gd_2O_3, and UO_2-PuO_2 Rods from the Quad Cities-1 Reactor, NP-2307-LD.* Electric Power Research Institute, Palo Alto, (March 1982).

Ennerst, K., J. Krellmann, and H. Vietzke. "State-of-the-Art Fuel Assembly Fabrication With Fuel Containing Plutonium." *Atomenergie Kerntechnik*, **47** (2), (1985) 68.

Eschbach, E. A. and S. Goldsmith. "Using Plutonium in Thermal Reactors." *Nucleonics*, **21** (1), (1963) 48.

Esteban, A., et al. "Precipitacion de oxalato de plutonio (III) y calcinacion a oxido de plutonio. (Precipitation of plutonium (III) oxalate and calcination to plutonium oxide)." INIS-MF-12948. Presented at Annual Mtg. of the Argentine Association of Nuclear Technology, Buenos Aires, Dec. 4-7, 1989.

Esteve, B., et al. "Recycled Fuels: French Experience and Program." *Proc. Int. Uranium Fuel Seminar*, Beaver Creek, USA.

Esteve, B. and A. Gloaguen. "Recycling of Reprocessed Materials: EDF's Views and Current Policy." *Proc. The Uranium Institute Annual Symposium 1994*, London, Sept. 1994.

Esteve, B. and A. Gloaguen. "RepU and MOX: EDF's Views and Current Policy." *Nuclear Europe Worldscan*, **3-4**, (1995) 49.

Esteve, B. "Plutonium: Reprocessing and Recycle of Irradiated Fuels." *Revue generale nucleaire*, **1**, (1995), 50.

Evans, S. K., D. L. Fischer, and W. V. MacNabb. "Plutonium Utilization in Boiling Water Reactors." *Proc. Use of Plutonium as a Reactor Fuel*, Brussels. IAEA, Vienna (1967).

Evans, S. K., et al. "Behavior of Pu Recycle Fuel in a SPERT Power Transient." *Trans. Am. Nucl. Soc.*, **13** (2), (Nov. 1970) 551.

Evans, S. K. and E. A. Aitken. "Behavior of Large PuO_2 Particles in a Rapid Power Transient." *Trans. Am. Nucl. Soc.*, **13** (2),(Nov. 1970) 553.

Fairburn, G. T., W. G. Pettis, and N. L. Snidow. *Plutonium Lattice Experiment in Uniform Test of UO_2-1.5% PuO_2 Fuel, BAW-1357.* Babcock and Wilcox Co. (1970).

Faust, L. G., et al. "Radiation Dose Rates from UO_2-PuO_2 Thermal Reactor Fuels." *Nucl. Tech.*, **15**, (Aug. 1972) 249.

Federici, E., P. Blanpain, and P. Permezel. "A Model for the Description of the Evolution of Pu Agglomerates in MOX Fuels." *Proc. IAEA Tech. Committee Mtg. On Water Reactor Fuel Element Modeling at High Burnup and Its Experimental Support*, Windermere, UK, September 19-23, 1994.

Ferrari, A., R. DeLayre, H. Guillet, et al. "L'Utilisation du Plutonium." *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2-13 May 1977. IAEA, Vienna, 1977.

Fischer, D. L. *Plutonium Utilization in Boiling Water Reactors Phase I Final Report, GECR 5566.* General Electric Co., (Nov. 1967).

Fischer, D. L., et al. "Nuclear and Fast Transient Aspects of Plutonium Particle Size in Thermal Recycle Fuel." *Trans. Amer. Nucl. Soc.*, **12** (1), (1969).

Fischer, U. "Mehrfache Ruckfuhrung von Plutonium in Thermischen Reaktoren." *Atomwirt.*, **31**, (Nov. 1986) 548.

Flipot, A. J., J. J. Huet, and E. Vanden Bemden. *Progress on Fabrication Techniques of Plutonium Fuel, BN-7106-05.* Belgonucléaire, Brussels (1971). Also in *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Flipot, A. J., I. Lafontaine, and J. M. Leblanc. "Development and Production of Plutonium Fuel." *Nucl. Energy Maturity, Proc. European Nuclear Conf.*, Paris, France, April 21-25, 1975.

Fougeras, P., et al. "Optimization of a Calculational Scheme for the Treatment of Plutonium Recycling in Pressurized Water Reactors." *Nuclear Science and Engineering*, **121**, (1995) 32-40.

Fournier, W. and G. Dalverny. "Plutonium: MOX Fuel and the MELOX Plant." *Revue generale nucleaire*, **1**, (1995), 34.

Francis, K. E., and R. G. Sowden. *Microstructure of PuO_2 Prepared by Different Methods.* United Kingdom Atomic Energy Authority, Harwell (internal document), (1959).

Freshley, M. D. *Plutonium Spike Fuel Elements for the PRTR, Part I – The Mark I-G, HW-69200.* Hanford Atomic Products Operation (1961).

Freshley, M. D. "The Irradiation of Prototype UO_2/PuO_2 Fuel Elements." *Trans. Am. Nucl. Soc.*, **5** (2), (1962) 467. [Also *USAEC Report HW-SA-2668*].

Freshley, M. D., et al. "PRTR Fuel Element Experience." *Trans. Am. Nucl. Soc.*, **7** (2), (1964) 388.

Freshley, M. D., F. E. Panisko, and R. E. Skavdahl. "The Irradiation Behavior of UO_2-PuO_2 Fuels in the PRTR." *AIME Symposium on High Temperature Nuclear Fuels*, Delavan, Wisconsin, (Oct. 1966).

Freshley, M. D. and S. Goldsmith. "Operating Experience with Plutonium Fuels in PRTR." *Nucl. Met.*, **13**, 96 (1967).

Freshley, M. D. and F. E. Panisko. *The Irradiation Behavior of UO_2-PuO_2 Fuels in PRTR, BNWL-366.* Battelle-Northwest Laboratory (1967).

Freshley, M. D. and T. B. Burley. *Vibrationally Compacted Ceramic Fuels, BNWL-SA-2412.* Battelle-Northwest Laboratory (1969).

Freshley, M. D., et al. "Behavior of Discrete PuO_2 Particles in Thermal Recycle Fuel During Rapid Power Transients." *Trans. Am. Nucl. Soc.*, **13** (2), (Nov. 1970) 552.

Freshley, M. D. *Mixed-Oxide Fuel Irradiations in PRTR, BNWL-SA-3981.* Battelle-Northwest Laboratory (1971).

Freshley, M. D. "Mixed-Oxide Fuel Irradiations in the Plutonium Recycle Test Reactor." *Nucl. Tech.*, **15**, (Aug. 1972) 125.

Freshley, M. D. "The Defect Performance of UO_2-PuO_2 Thermal Reactor Fuel." *Nucl. Tech.*, **15**, (1972) 209-228.

Freshley, M. D., et al. "Behavior of Discrete Plutonium-Dioxide Particles in Mixed-Oxide Fuel During Rapid Power Transients." *Nucl. Tech.*, **15**, (1972) 239-248.

Freshley, M. D. "UO_2-PuO_2 – A Demonstrated Fuel for Plutonium Utilization in Thermal Reactors." *Nucl. Tech.*, **18**, (May 1973) 141-170.

Freshley, M. D., et al. "The EPRI Plutonia Fuel Study." *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977, ANS-Fuel Cycle Division (1977).

Freshley, M. D. and L. J. Harrison. "The Transient Behavior of Vipac and Pellet Thermal Reactor Oxide Fuels." *Nucl. Tech.*, **15**, 177 (Aug. 1977).

Freshley, M. D., et al. *EPRI Plutonia Fuel Study (RP 396): Fuel Pellet Fabrication and Characterization.* Electric Power Research Institute, Palo Alto, (1977).

Freshley, M. D., et al. *EPRI Plutonia Fuel Study (RP 396): Postirradiation Examination and Characterization Results*. Electric Power Research Institute, Palo Alto, (1977).

Freshley, M. D., et al. *Plutonia Fuel Study, EPRI NP-637*. Research Project 396. Electric Power Research Institute, Palo Alto, (Jan. 1978).

Freshley, M. D., et al. "Irradiation-Induced Densification and PuO_2 Particle Behavior in Mixed-Oxide Pellet Fuel." *J. of Nuclear Materials*, **81**, (1979).

Frost, B. R. T., P. G. Mardon, and L. E. Russell. "Research on the Fabrication, Properties, and Irradiation Behaviour of Plutonium Fuels for the UK Reactor Programme." *Plutonium as a Power Reactor Fuel – Proc. Am. Nucl. Soc. Top. Mtg*, Richland, Washington, Sept. 13-14, 1962. *USAEC Report HW-75007*, Hanford Atomic Products Operation, General Electric Co., and USAEC, (Dec. 1962).

Gaafer, M., F. Orestano, and F. Pistella. *Assessment of the Reliability of the CNEN's Methods for the Neutronic Design of PWR's. Part III. Analysis of Several Westinghouse Experiments.* RT/FI-73-23. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1973.

Gaafer, M. A., et al. "Assessment of the Reliability of the CNEN's Calculational Methods for the Neutronic Design of PWRs." *Nucl. Tech.*, **27**. (Oct. 1975) 187.

Gabrielli, P. G., and F. Pistella. *Modifications in Cell Cross Sections in the Epithermal and Fast Energy Range due to the Presence of Gd as a Burnable Poison (BWR, PWR).* RT/FI-74-39. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1974.

Galloway, G. D., G. C. Martin, and R. A. Wolters. *Cold Critical Measurement Performed Prior to Cycle-4 Startup of the Quad Cities-1 Reactor, NP-2303-LD*. Research Project 497-1, Electric Power Research Institute, Palo Alto, (March 1982).

Gaube, M. and G. Smissaert. "The Belgian Nuclear Industry." *Nuclear Europe Worldscan*, Vol. XII, No. 5-6 (1992).

General Electric Co. *Plutonium Utilization in Boiling Water Reactors, Phase I Final Report, GECR-5566*. General Electric Co., (Nov. 1967).

General Electric Co. *General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring, 1974, NEDO-20103*. General Electric Co., (Sept. 1973).

General Electric Co. *STR Bundle Submittal, Quad Cities-1 Segmented Test Rod Bundle, NEDE-20236*. General Electric Co., (Jan. 1974).

General Electric Co. *Quad Cities 1 Nuclear Power Station First Reload License Submittal, NEDO-20292*. General Electric Co., San Jose, (1974).

General Electric Co. "Quad Cities MO_2 Bundle Reload Supplement." AEC Docket No. 50-254, 74. NEDE, General Electric Co., San Jose, (1974).

General Electric Co. *Quad Cities Nuclear and Fuel Performance Measurements, EPRI NP-3568.* Project 497-1, Final Report. Electric Power Research Institute and General Electric Co., (July 1984).

General Electric Co. *Neutron Radiography of Fuel Rods Irradiated for Four Cycles at the Quad Cities-1 BWR, EPRI NP-3782.* Electric Power Research Institute, Palo Alto (Nov. 1984).

Giacchetti, G. and C. Sari. "Behavior of Molybdenum in Mixed-Oxide Fuel." *Nucl. Tech.*, **31**, (Oct. 1976) 62.

Gibello, A., F. Orestano, F. Pistella, and E. Santandrea. "Physics Experiments in a PuO_2-UO_2 Microlattice for the Design of Fuel Elements in LWRs." *J. Nucl. Energy*, **25**, No. 12, 623-5 (Dec. 1971).

Gibby, R. L. "The Effect of Plutonium Content on the Thermal Conductivity of $(U,Pu)O_2$ Solid Solution." *J. Nucl. Mater.*, **38**, (1971) 163.

Gloaguen, A. and G. A. Lebastard. "Nuclear Fuel: Recycling?" *Proc. USCEA Fuel Cycle Conference*, Boston, MA, Mar. 21-23, 1994.

Golinelli, C. "Synthesis Report of a Reactor-Physics Programme Devoted to the Qualification of the Project Calculation Methods Applied to the Mixed (U,Pu) Fueled Reloadings." Commission of the European Communities, EUR-7603-EN, (1982).

Goll, W., et al. *Qualification of Readily Soluble Mixed Oxide Fuels. Final Report, SIEMENS-KWU-U9-414/88/006.* SIEMENS/KWU, Erlanger, 1988.

Goll, W., et al. "Untersuchungen an bestrahlten MOX-Testbrennstaeben mit Plutonium der 2. Generation. (Examination of irradiation MOX test fuel rods with second-generation plutonium)." SIEMENS-KWU-U9-414/80/010. SIEMENS/KWU, Erlanger, 1989.

Goll, W., et al. "Irradiation Behavior of UO_2/PuO_2 Fuel in Light Water Reactors." *Nucl. Tech.*, **102** (1993) 29.

Gouffon, A. and J. P. Merle. "Safety Problems Related to the Use of MOX Assemblies in PWRs." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35. IAEA, (1990).

Gould, T. H., Jr. and R. T. Huntoon. *Technical Data for FFF – Separate Stream Case, DPST-AFCT-77-150.* Savannah River Laboratory, E. I. du Pont de Numours and Co., Aiken, SC (1977).

Grifoni, S., R. Lloret, and F. Pistella. "Some Problems Encountered in Dosimetry of Irradiations of Burnable Poisons." *Proc. ASTM-EURATOM Symposium on Reactor Dosimetry: Dosimetry Methods for Fuels, Cladding, and Structural Materials*, Palo Alto, CA, USA, 2 Oct. 1977. NUREG/CP-0004 (Vol. 1).

Grison, E., W. B. H. Lord, and R. D. Fowler (eds.). *Plutonium 1960.* Cleaver-Hume Press Ltd., London, 1961.

Grochowski, G., G. Hamilton, and S. Israel. *Nuclear Design, Economics, and Safety Analysis of the Dresden I Mixed Oxide Demonstration Assemblies, GU-5291.* Gulf United Nuclear Corp., (Dec. 1972).

Gruber, K., G. Latzel, and H. Roepenack. "Fabrication and Quality Control of Mixed Oxide LWR Fuel With Regard to Homogeneity of Fissile Content." *Proc. Seminar Practical Experience in the Application of Quality Control in Water Reactor Fuel Fabrication, KfK-Bericht 3777, IAEA-SR-102/23.* IAEA, Vienna, (July 1984) 353.

Guais, J. C., G. Lebastard, and L. F. Durret. "Uranium and Plutonium Recycling in Light Water Reactors." *Revue generale nucleaire*, **5**, (1989), 377.

Gue, J. P., et al. "French Experience in MOX Fuel Dissolution." *Proc. Global '93: Future Nuclear Systems – Emerging Fuel Cycles and Waste Disposal Options*, Seattle, Sep. 12-17, 1993.

Guldner, R. and H. Schmidt. "Optimization of Process Parameters For the Sintering of MOX Fuel." *J. Nucl. Mater.*, **178**, (1991) 152.

Guyot, F., P. Mahe, and S. de Perthuis. "Framatome's Experience in Nuclear Design of MOX Fuel Assemblies." *Jahretagung Kerntechnik '95*, Nurnberg, May 16-18, 1995.

Haas, D. "Status of the PRIMO MOX Fuel Research and Development Programme." *Proc. Int. Topl. Mtg. LWR Fuel Performance*, Avignon, France, April 21-24, 1991. American Nuclear Society.

Haas, D. "Belgonucléaire MOX Fabrication Experience for Light Water Reactors." Proc. Belgian Technical Seminar, Tokyo, Nov. 1991.

Haas, D. and J. Van Vliet. "Belgonucléaire MOX Fuel Fabrication." *Nuclear Europe Worldscan*, Vol. XII, No. 5-6 (1992).

Haas, D., et al. "Mixed-Oxide Fuel Fabrication Technology and Experience at the Belgonucléaire and CFCa Plants and Further Developments For the MELOX Plant." *Nucl. Tech.*, Vol. 106, (April 1994).

Haas, D., J. van Vliet, and A. Vandergheynst. "Experience With MOX Fuel." *Proc. Fuel Cycle 95*, San Diego, CA, April 2-5, 1995. Nuclear Energy Institute (1995).

Haley, J. *EEI-Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period Ending December, 1970, WCAP 4167-2.* (1971).

Hamilton, G. *Dresden I Plutonium Recycle Program, Semi-Annual Report for Period Ending 6/30/72, GU-5287.* Gulf United Nuclear Corp., (Jan. 1973).

Hamilton, G. *Dresden I Plutonium Recycle Program, Semi-Annual Report for Period Ending 12/31/72, GU-5290.* Gulf United Nuclear Corp., (Jan. 1973).

Hanford Atomic Products Operation. *The Plutonium Recycle Program – A Resume of the Concept, Program, and Facilities, USAEC Report HW-50700.* Hanford Atomic Products Operation, General Electric Co., (June 1957).

Hanford Atomic Products Operation. *Ceramics Research and Development Operation Quarterly Report, July-September 1963, HW-76303.* Hanford Atomic Products Operation (1964).

Hanford Atomic Products Operation. *Ceramics Research and Development Operation Quarterly Report, October-December 1963, HW-76304.* Hanford Atomic Products Operation (1964).

Hanford Atomic Products Operation. *Ceramics Research and Development Operation Quarterly Report, January-March 1964, HW-81600.* Hanford Atomic Products Operation (1964).

Hanford Atomic Products Operation. *Ceramics Research and Development Operation Quarterly Report, April-June 1964, HW-81601.* Hanford Atomic Products Operation (1964).

Hanford Atomic Products Operation. *Ceramics Research and Development Operation Quarterly Report, July-September 1964, HW-81602.* Hanford Atomic Products Operation (1964).

Hanford Atomic Products Operation. *Ceramics Research and Development Operation Quarterly Report, October-December 1964, HW-81603.* Hanford Atomic Products Operation (1965).

Hanthorn, H. E. *Calculated Cost of Fabrication of Plutonium Enriched Fuel Elements, HW-74304.* Hanford Atomic Products Operation (1962).

Hardman, J. R., and A. R. Dastur. "CANDU Nuclear Power Plant Optimization With MOX Fuel." *ANP'92: International Conference on Design and Safety of Advanced Nuclear Power Plants*, Oct. 25-29, 1992, Tokyo, Japan. (CONF-921003), pp. 19.1/1 - 19.1/17.

Hauth, J. J., and R. J. Anicetti. *Vibrational Compaction of Ceramic Powders for Nuclear Fuel Elements," HW-SA-1735.* Hanford Atomic Products Operation (1959).

Hauth, J. J. *Vibrationally Compacted Ceramic Fuels, USAEC Report HW-67777.* Hanford Atomic Products Operation, General Electric Co., (Jan. 1961).

Hauth, J. J. "Vibrational Compaction of Nuclear Fuels," in *Vibratory Compacting Principles and Methods.* H. H. Hausner, K. H. Roll, and P. K. Johnson (eds.), Plenum Press, New York (1967).

Hellens, R. L. "Problems in Recycle of Plutonium in Pressurized Water Reactors." *Trans. Am. Nucl. Soc.*, **14**, (1971) 820.

Hibbs, M. "Proponents Mount Last-Ditch Drive to Save Hanau Plant For Weapons Pu." *NuclearFuel*, (June 5, 1995).

Hnilica, K., H. Holley, and K. Lahner. "Determination of Nuclear Characteristics of U/Pu Fuel Elements in LWR's." *Atomwirtschaft*, 2, (1974) 69.

Holzer, R. and H. Knaab. "Recent Fuel Performance Experience and Implementation of Improved Products." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17-20 April 1988. ANS, La Grange Park, 1988.

Horgos, R. M. *Safeguards Criteria for a Large-Scale Mixed Oxide Fuel Fabrication Facility, RPDE-78-068.* Westinghouse Electric Corp., Nuclear Fuels Division, Pittsburgh, PA (1977).

Houston, D., et al. *Review of Plutonium Recycle Demonstration to be Conducted in the San Onofre Reactor.* Prepared for Southern California Edison Company by Battelle Memorial Institute, Columbus, Ohio.

Howell, J. P., D. T. Rankin, and J. D. Scarbrough. *Technical Data Summary – Mixed Oxide Fuel Fabrication Facility Coprocessing Case, DPSTD-AFCT-77-12; Rev. 1.* Savannah River Laboratory, E. I. du Pont de Numours and Co., Aiken, SC (Aug. 1979).

Hower, L. D., Jr. *Mixed Oxide Fuel Pellet Fabrication and Fuel Rod Assembly for Dresden I, Batch 7 Reload, GU-5308.* Gulf United Nuclear Corp., (Jan. 1974).

Hughes, J. H. "Request for Amendment of Operating License DPR-2, as Amended, Dresden Nuclear Power Station, including Exhibit I." NRC Docket 50010-83. (Jan. 9, 1967).

Hughes, J. H. "Request for Change of Operating License DPR-2, as Amended, Dresden Nuclear Power Station, Including Exhibit I." NRC Docket 50010-85. (Jan. 31, 1967).

Huntoon, R. T., D. T. Rankin, and J. D. Scarbrough. *Mixed Oxide Fuel Fabrication Facility. DPSTD-LWR-77-4.* Savannah River Laboratory, E. I. du Pont de Numours and Co., Aiken, SC (1977).

Huntoon, R. T., et al. *Technical Data Summary – Mixed Oxide Fuel Fabrication Facility Coprocessing Case, DPSTD-AFCT-77-12.* Savannah River Laboratory, E. I. du Pont de Numours and Co., Aiken, SC (1977).

Huttmann, A., M. Billaux, J. E. Christiansson, et al. "New Way of Recycling Plutonium in Boiling Water Reactors." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17-20 April 1988. ANS, La Grange Park, 1988.

Inoue, S., et al. "Light Water Reactor Core Design Code System CASMO/SIMULATE(13), Verification for Beznau Unit 2 Loaded With MOX Fuel on Commercial Basis." *Proc. Spring Mtg.*, H35, Hiratsuka, Japan, March 28-30, 1992. Atomic Energy Society of Japan, (1992).

International Atomic Energy Agency. *Guidebook on Quality Control of Mixed Oxides and Gadolinium Bearing Fuels for Light Water Reactors, IAEA-TECDOC-584.* IAEA, Vienna, (Feb. 1991).

Japan Atomic Energy Research Institute. Achievements of Japanese Fuel Irradiation Experiments in HBWR, JAERI-M-92-155. JAERI, Tokyo, (Oct. 1992).

Janin, R. "Plutonium Recycling in France." *Atomwirtschaft-Atomtechnik*, **32**, *10*, (Oct. 1987), 502.

Jeffs, A. T. *Oxygen:Metal Ratio Effects in $(U,Pu)O_2$ Fuels, AECL-4063.* Atomic Energy of Canada, Limited (1970).

John, C. D., Jr. *Identification of Specialized Fabricating Equipment for the Manufacture of Mixed Oxide Fuels.* Westinghouse Electric Corp., Nuclear Fuels Division, Pittsburgh, PA (1977).

Johnson, A. A., et al. *Commercial Fabrication of Plutonium Fuel, DP-838.* Savannah River Laboratory (1963).

Jones, L. V. and L. J. Wittenberg. "Plutonium Reactor Fuels Research Program at Mound Laboratory." *Proc. Plutonium as a Power Reactor Fuel*, Richland, Washington, Sept. 13-14, 1962. *HW-75007.* Hanford Atomic Products Operation (1962).

Jones, M. K. and W. J. Jenkins. "Slurry Calcination Process for Conversion of Aqueous Uranium and Plutonium to a Mixed-Oxide Powder." *Proc. ANS Top. Mtg. Fuel Cycles for the 80's*, Gatlinburg, Tennessee, 29 Sep. - 2 Oct. 1980. ANS, La Grange Park, 1980.

Joo, H. K. and S. Y. Oh. *Recycling of MOX Fuel for LWRs, KAERI/AR-336/91.* Korea Atomic Energy Research Institute, Daeduk, (Jan. 1992).

Kajiyama, T., et al. "Irradiation Performance of MOX Fuel in Fugen." *PNC Technical Review No. 67*, (1988) pp. 20-37.

Kanda, K., et al. "Effective Utilization of Uranium and Plutonium in LWR Cores – General Remarks." *Trans. Am. Nucl. Soc.*, **66**, (1992) 459.

Kaneda, K., et al. "Fission Gas Release of Mixed Oxide Fuel." *HPG Mtg. On Fuel Performance Experience and Analysis*, Sanderstolen, Norway, March 2-7, 1986.

Kaneda, K. "Plutonium Utilization Experience in Japan." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Kier, P. H. *Analysis of the Initial Critical Experiments of the EBWR Plutonium Recycle Program, ANL-7368.* Argonne National Laboratory (1967).

Kinchin, G. H. "Physics of Plutonium Recycling in Thermal Reactors." *Proc. Use of Plutonium as a Reactor Fuel*, Brussels, March 1967. IAEA, Vienna (1967).

Kinchin, G. H., I. Johnstone, and J. G. Tyror. "Thermal Reactor Physics Measurements Using Pu-Bearing Fuel." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Knoglinger, E. and H. Schmale. "The VAK-Kahl Plutonium Recycle Demonstration Program." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Koler, R. K. and R. E. Sharp. "Fabrication of Plutonium-Bearing Ceramic Fuel Elements by Vibrational Compaction." *Trans. Am. Nucl. Soc.*, **5**, (1962) 453.

Kosanke, H. D., et al. *Gamma Scan Measurements at Quad Cities 1 Following Cycle 4, NEDC-25492*. General Electric Co., San Jose, (Nov. 1981).

Kramer, E. and S. Langenbuch. "Sicherheitstechnische Untersuchungen zur Leistungsverteilung in MOX-Brennelementen beim Einsatz in LWR-Kernen. (Safety-related investigations on power distribution in MOX fuel elements in LWR cores)." BMU-1991-302, GRS-A-1407. BMU, Bonn, 1991.

Krebs, W. D., and G. J. Schlosser. "Status of Fuel Assembly Design and Core Management Experience With MOX Fuel in the FRG for Siemens/KWU Type LWRs." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35. IAEA, (1990).

Krellmann, J. and W. Stoll. "Remote MOX Fuel Fabrication: Facts and Inferences." *Proc. Seminar on Remote Handling Equipment for Nuclear Fuel Cycle Facilities*, Oxford, UK, Oct. 2-5, 1984. OECD, (1985).

Krellmann, J. "Plutonium Processing at the Siemens Hanau Fuel Fabrication Plant." *Nucl. Tech.*, **102**, (1993) 18.

Krellmann, J. "MOX Fuel Technology and MOX Performance in Germany – Plutonium Handling Experience." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Krysinski, T. L. *Dresden I Plutonium Recycle Program, Semi-Annual Report for Period Ending 6/30/73, GU-5304*. Gulf United Nuclear Corp., (Jun. 1973).

Krysinski, T. L. *Dresden I Plutonium Recycle Program, Semi-Annual Report for Period Ending 12/31/73, GU-5309*. Gulf United Nuclear Corp., (Jan. 1974).

LaBar, M. P. *Mechanical Design of the Dresden I Mixed Oxide Demonstration Assemblies, GU-5292*. Gulf United Nuclear Corp., (March 1973).

Lackey, W. J., F. J. Homan, and A. R. Olsen. "Porosity and Actinide Redistribution During Irradiation of $(U,Pu)O_2$." *Nucl. Tech.*, **16**, (1972) 120.

Lassman, K., et al. "The Radial Distribution of Plutonium in High Burnup UO_2 Fuel." *J. Nucl. Mater.* **208** (1994), 223-231.

Layman, W. H., C. R. Montgomery, and H. J. Williams. "Utilization of a Zoned Loading of Plutonium Fuel in the Saxton Reactor." *Trans. Am. Nucl. Soc.*, **10**, (1967) 498.

Leamer, R. D., et al. *PuO_2-UO_2 Fueled Critical Experiments, WCAP-3726-1.* Westinghouse Electric Corp. (1967).

Lebastard, G. and H. Bairiot. "MOX Fuel Fabrication Industry." *Proc. Int. Conf. On Nuclear Fuel Reprocessing and Waste Management*, Paris, Aug. 1987, p. 841.

Lebastard, G. "MELOX Plant to Double Pu Recycle Capacity." *Modern Power Systems*, **9**, 2, (Feb. 1989), 25.

Lebastard, G. "Building MELOX to Meet Future Demand." *Nucl. Engr. Int.*, **35**, (Dec. 1990), 32.

Lebastard, G. "Recycling and MOX Fabrication: French and Belgian Achievements." *Trans. Am. Nucl. Soc.*, **70**, (1994) 797.

Lee, B., Jr. "Quad Cities-1 Nuclear Power Station First Reload, Supplement for Assemblies Containing 48 EEI-GE Mixed Oxide Fuel Rods." NRC Docket 50254-215. (June 15, 1973).

Lee, B., Jr. "Quad Cities 1 Nuclear Power Station First Reload License Submittal." NRC Docket 50254-318. (Feb. 28, 1974).

Leenders, L., and H. Van den Broeck. *DAtF, Reaktortagung Bericht*, **171** (1974).

Lenain, R., et al. *Analysis of a Control Rod Ejection Transient in a MOX (Mixed Oxide) Fueled PWR (Pressurized Water Reactor), CEA-CONF-9655.* Also in *Proc. ANS Int. Reactor Physics Conf.*, Jackson-Hole, Sept. 18, 1988.

Lepscky, C., and F. Pistella. *Experiments on the Use of Gd_2O_3 as Burnable Poison for PWR. Final Project of Sample Irradiation in Siloe.* RT/FI-73-41. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1973.

Levina, I. K., V. V. Saprykin, and A. G. Morozov. "The Safety Criteria and VVER Core Modification for Weapon Plutonium Utilization." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Lhost, A. and H. Bairiot. *Evolution of Fuel Under Irradiation – Belgian Experience, BN-7106-04.* Belgonucléaire, Brussels (June 1971). Also in *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Liikala, R. C. "A Comparative Study of Unit Cell Homogenization for PuO_2-UO_2-H_2O Lattices." *Physics Research Quarterly Report, July, August, September 1964, HW-84369.* General Electric Co. (1965).

Liikala, R. C., U. P. Jenquin, and W. A. Reardon. *The PuO_2-UO_2 Experiment in the EBWR, BNWL-126.* Pacific Northwest Laboratories (1965).

Liikala, R. C., V. O. Uotinen, and U. P. Jenquin. "Lattices of Plutonium-Enriched Rods in Light Water Part II – Theoretical Analysis of Plutonium-Fueled Systems." *Nucl. Tech.*, **15**, (1972) 272.

Liikala, R. C., V. O. Uotinen, and U. P. Jenquin. *Uncertainties in the Analysis of Plutonium Fueled Light Water Moderated Assemblies, BNWL-1656.* Pacific Northwest Laboratories (1973).

Lippens, M. "Densification of UO_2-PuO_2 Fuel." *Journal of Nuclear Materials.* **81,** (1979).

Lippens, M. and N. Cadelli. "A European Investigation of MOX Fuel Performance in LWRs." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17-20 April 1988. ANS, La Grange Park, 1988.

Lippens, M., J. Wilson, A. Garlick, et al. *Summary of Post-Irradiation Examinations of Uranium/Plutonium Oxide Fuel Rods Irradiated in LWRs, BN 8610-03.* Belgonucléaire, 1988.

Loizzo, P., et al. *Experimental and Calculated Results for UO_2 and UO_2-PuO_2 Fueled H_2O-Moderated Loadings, BNWL-1379.* Pacific Northwest Laboratories (1970).

Lorenzelli, R. "MELOX: Mixed Oxide Fuel Fabrication." *Revue generale nucleaire*, **4**, (1991), 298.

Lussie, W. G. *The Response of Mixed-Oxide Fuel Rods to Power Bursts, IN-ITR-114.* Idaho Nuclear Corporation (1970).

Macdonald, A. G. "The MOX Demonstration Facility – The Stepping Stone to Commercial MOX Production." *Nucl. Energy*, **33**, No. 3 (1994).

Macherey, R. E. "History of EBWR Materials and Fuel Element Development." *Trans. Am. Nucl. Soc.*, **18**, (1974) 112.

MacLeod, H. M., G. R. Chilton, and J. A. Dodd. "A Versatile Facility at Windscale, United Kingdom, for the Fabrication of Experimental Uranium-Plutonium-Oxide Nuclear Fuel." *Nucl. Tech.*, **88**, (1989) 190.

MacLeod, H. M., and G. Yates. "Development of Mixed-Oxide Fuel Manufacture in the United Kingdom and the Influence of Fuel Characteristics on Irradiation Performance." *Nucl. Tech.*, **102**, (April 1993).

Madic, C. "Plutonium: An Introduction to the Nuclear and Physicochemical Properties." *Revue generale nucleaire*, **1**, (1995), 7.

Magnin, D. and M. Moulie. "Adaptation of MOX Design for Equivalent MOX/UO2 Performance in PWR Reactors." *Proc. Am. Nucl. Soc. Int. Top. Mtg. Safety of Operating Reactors*, Sept. 17-20, 1995, Seattle, WA., (1995) 800.

Mangiagalli, D., F. Cicognani, F. Pistella, et al. "Plutonium Utilization in Fast-Breeder and Light-Water Reactors in Italy." *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2-13 May 1977. IAEA, Vienna, 1977.

Marin, J. F., et al. Programme de qualification du combustible a oxyde mixte, CEA-CONF-9352. Presented at SFEN/BNS Meeting – technical session on nuclear fuel cycle, Paris, Oct. 7, 1987.

Marino, A. C., P. Adelfang, and J. Spino. "Experiencias con oxidos mixtos $(U,Pu)O_2$. (Irradiaciones en los reactores MZFR y HFR-Petten)." XIV Reunion Cientifica de la AATN, Cordoba, 1986 (59).

Marino, A. C. "Barras Combustibles MOX irradiadas en Petten. Analisis del comportamiento de la barra A.1.4 ("pathfinder")." Informe interno GACC-DPGT No. 002.09/93.

Marino, A. C. "Barras Combustibles MOX irradiadas en Petten. Analisis del comportamiento de las barras A. 3 (con Iodo) y A.4 (con CsI)." Informe interno GACC-DPGT No. 002.10/93.

Marino, A. C., E. E. Perez, and P. Adelfang. "Argentine Nuclear Fuels MOX Irradiated in the Petten Reactor, Experiment Analysis with the BACO Code." *Proc. Tech. Committee Mtg. on Water Reactor Fuel Element Modelling at High Burnup and its Experimental Support*, Windermere, UK, Sept. 19-23, 1994.

Marino, A. C., E. Perez, and P. Adelfang. "Irradiation of Argentine MOX Fuels. Post-Irradiation Results and Analysis." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, UK, July 3-7, 1995.

Markl, H., and H. Stehle. "KWU Plutonium Recycle Fuel and Core Design, Performance, and Experience." *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977. ANS-Fuel Cycle Division (1977).

Marshall, A., and A. S. Shoebottom. "The Homogeneity of PuO_2-UO_2 Fuel Made by Dry Blending PuO_2 and UO_2 Powders in Ball Mills." British Nuclear Fuels Limited (internal document), 1967.

Marshall, A., G. A. Holburt, and A. S. Shoebottom. "The Effect of Calcination Conditions on the Properties of Plutonium Dioxide." British Nuclear Fuels Limited (internal document), 1976.

Marshall, R. K. and D. W. Brite. *EPRI Plutonia Fuel Study (RP 396): Fuel Pin and Capsule Design, Fabrication, and Irradiation.* Electric Power Research Institute, Palo Alto, (1977).

Martin, D. G. "A Re-Appraisal of the Thermal Conductivity of UO_2 and Mixed (U,Pu) Oxide Fuels." *J. of Nucl. Mat.*, **110**, (1982) 73.

Martin, D. G. "The Thermal Expansion of Solid UO_2 and (U,Pu) Mixed Oxides – A Review and Recommendations." *J. of Nucl. Mat.*, **152**, (1988) 94.

Martin Deidier, L., A. Darraud, and J. Bergeron. "Mixed Oxide Fuels. Adaptation of PWR Core and Vessel to the Mixed Oxide Fuels. The Physical Aspect in the Conception of the Plutonium Assemblies." Proc. SFEN/BNS Mtg. Tech. Session on Nuclear Fuel Cycle, Paris, Oct. 1987. CEA-CONF-9259, CONF-8710325.

Matsen, R. P. *EBWR Burnup Data, BNWL-978.* Pacific Northwest Laboratories (1968).

Matsen, R. P. *Burnup Data from UO_2-0.48 wt.% PuO_2 Fuel, BNWL-1488.* Pacific Northwest Laboratories (1970).

Mattuci, A., F. Pistella. *Burnup Calculations for LWR Fuel Elements Containing Gd Poisoned Pins.* RT/FI-71-33. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1971.

McLachlan, A. "EdF Takes the MOX Recycle Plunge to Utilize Reprocessed Pu and U Stocks." *Nucleonics Week*, (June 1985).

Melehan, J. B. *EEI - Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period 10/71 - 3/72, WCAP-4167-4*, (Dec. 1972).

Melin, P., B. Gautier, and P. Combette. "Behavior of FRAGEMA Fuel in Power Reactors." *Proc. Int. Topl. Mtg. LWR Fuel Performance*, Avignon, France, April 21-24, 1991. American Nuclear Society (1991).

Merker, L. G., L. G. Faust, and W. J. Bailey, et al. *Battelle-Northwest Fuel Fabrication Experience with High Exposure Plutonium, BNWL-27.* Battelle-Northwest (1965).

Merth, D. W. *Gamma Scan Measurements at Quad Cities Nuclear Power Station Unit-1 Following Cycle 3, NP-2302-LD.* Research Project 497-1. Electric Power Research Institute, Palo Alto, (March 1982).

Meyer, R. O., D. R. O'Boyle, and E. M. Butler. "The Effect of Oxygen-to-Metal Ratio on Plutonium Redistribution in Irradiated Mixed-Oxide Fuels." *J. Nucl. Mater.*, **47**, (1973) 265.

Meyer, R. O. "Analysis of Plutonium Segregation and Central-Void Formation in Mixed-Oxide Fuels." *J. Nucl. Mater.*, **50**, (1974) 11.

Meyer, R. O., C. R. Hann, and D. D. Lanning. "Effects of Fissile Atom Segregation in Light Water Reactor Plutonium Recycle Fuels." *Nucl. Tech.*, **27**, (Nov. 1975) 389.

Michel, A., A. Vandergheynst, and J. van Vliet. "Applying the MOX Industrial Experience to the Recycling of Weapon-Grade Pu." *Proc. IAEA Radioactive Wastes & Environmental Restoration*, Berlin, Sept. 3-9, 1995.

Miller, R. S., et al. "Operating Experience with the Saxton Reactor Partial Plutonium Core II," *AIME Symposium on Plutonium Fuels Technology*, Scottsdale, Arizona, (Oct. 1967).

Moncassoli, A. "Analisi preliminare di un nocciolo a miscela di ossidi di U e Pu." Incontro sulia Fisica del Plutonio nei reattori termici, Torino 30-31 maggio, 1966.

Mondot, J. et al. "EPICURE: An Experimental Programme Devoted to the Validation of the Calculational Schemes for Plutonium Recycling in PWRs." *Proc. Int. Conf. Physics of Reactor Operation, Design and Computation (PHYSOR '90)*, Marseille, France, April 23-26, 1990, Vol. I, P. VI-53. American Nuclear Society French Section (1990).

Moore, J. B. "Description of Proposed Change and Safety Analyses Proposed Change No. 21 to the Technical Specifications Provisional Operating License DPR-13." Letter to J. F. O'Leary dated June 15, 1973. Southern California Edison Company (1973).

Morel, M., P. Melin, and A. Dumont. "Irradiation Experience Leads Fragema to New Fuel Designs." *Nucl. Engr. Int.*, Vol. 37, No. 459, Oct. 1992.

Morgan, J. N. and W. D. Krebs. "Siemens Experience With Large Scale Mixed Oxide Fuel Assembly Insertion in Light Water Reactors." *Proc. ANS Topical Mtg. on LWR Fuel Performance*, West Palm Beach, Florida, (1994).

Mori, M., M. Kawamura, and H. Nishioka. "CASMO Benchmarking Against VIP-PWR MOX Fuel Critical Experiment." *Proc. Autumn Mtg.*, E48, Kobe, Japan, Oct. 9-11, 1993. Atomic Energy Society of Japan.

Mori, M., M. Kawamura, and K. Yamate. "CASMO-4/SIMULATE-3 Benchmarking Against High Plutonium Content Pressurized Water Reactor Mixed-Oxide Fuel Critical Experiment." *Nucl. Sci. and Engr.*, **121**, (1995) 41.

Morris, P. A. "Docket No. 50-155, Amendment to Facility License, License No. DPR-6, Amendment No. 3." NRC Docket 50155-11, April 18, 1969.

Mulford, R. N. R. and F. H. Ellinger. "UO_2-PuO_2 Solid Solutions." *J. Am. Chem. Soc.*, **80**, (1958) 2023.

Murphy, E. S., G. Manca, and D. E. Christensen. *Nondestructive Analysis of Fuel Irradiated in the EBWR by Gamma Scanning, BNWL-1005*. Pacific Northwest Laboratories (1969).

Muto, T., M. Koizumi, K. Naruki, et al. Status of the Development of Mixed-Oxide Fuels in the Power Reactor and Nuclear Fuel Development Corporation (PNC). *Proc. Int. Conf. on Nuclear Power and Its Fuel Cycle*, Salzburg, 2-13 May 1977. IAEA, Vienna, 1977.

Nakamura, Y. "The Plutonium Fuel Development Laboratory of the Atomic Fuel Corporation." *Nippon Genshiryoku Gakkaishi*, **8**, (1966) 41.

Nakamura, Y., et al. "Plutonium Fuel Development in Japan for Fast and Thermal Reactors." *Proc. Fourth Int. Conf. Peaceful Uses of Atomic Energy*, A/CONF 49/P-821. United Nations and International Atomic Energy Agency, Vienna (1972).

Nakano, H., M. Yamamoto, and K. Kono. "Japanese Approach to Attaining Experience in MOX Fuel Fabrication." *Proc. OECD/NEA-JAEC-IAEA Symp.*, Tokyo, Japan, April, 1989.

Naruki, K:, et al. "Fabrication of PuO_2-UO_2 Mixed-Oxide Fuel for the Initial Loading of Heavy Water Reactor 'FUGEN':" *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977. ANS-Fuel Cycle Division (1977).

Neill, J. M., P. d'Oultremont, and C. J. Rindfleisch. *Reanalysis of Plutonium Lattice Spectral Data, GA-10116 (Rev.).* Gulf General Atomic, Inc. (1970).

Newman, D. F. "Measurement of Microscopic Neutronic Parameters in UO_2-0.9 wt% PuO_2 Rods Moderated by H_2O." *Reactor Physics Quarterly Report, April-June, 1968, BNWL-887.* Pacific Northwest Laboratories (1968).

Newman, D. F. "Measurement of Microscopic Neutronic Parameters in UO_2-PuO_2 Rods Moderated by Light Water." *Reactor Physics Quarterly Report, July-September, 1968, BNWL-921.* Pacific Northwest Laboratories (1968).

Newman, D. F. and C. R. Gordon. "Interpretation of k_∞ for Water Moderated UO_2-PuO_2 Lattices." *Reactor Physics Quarterly Report, BNWL-1381-2.* Battelle Memorial Institute, Pacific Northwest Laboratories (1970).

Newman, D. F. and C. R. Gordon. *Measurement and Analytical Correlation of Neutronic Parameters for a Water-Moderated Lattice Containing UO_2-PuO_2 Particulate Fuel, BNWL-1603.* Pacific Northwest Laboratories (1971).

Newman, D. F. "Measurement of k_∞ and Relative Reaction Rates in an H_2O Moderated UO_2-PuO_2 Particulate Fueled Lattice." *Nucl. Tech.*, **15**, (1972) 192.

Nigon, J.-L., and C. Golinelli. "MOX in France: Domestic Programme and MELOX Plant." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Nodvik, R. J. *Saxton Core II Fuel Performance Evaluation, Part II: Evaluation of Mass Spectrometric and Radiochemical Analyses of Irradiated Saxton Plutonium Fuel, WCAP-3385-56, Pt. II.* (July 1970).

Novikov, A. N., et al. "Use of MOX (R-Pu and W-Pu) Fuel in VVER-1000 (Neutral-Physical Aspects of Possibilities)." *Proc. Managing the Plutonium Surplus: Applications and Options*, The Royal Institute of International Affairs, London, Jan. 24-25, 1994.

Nuclear Engineering Development Corporation. *Plutonium Utilization in Boiling Water Reactors - Phase II Semi-Annual Report, January - June, 1971, NEDC-10586.* General Electric Co., San Jose, (1971).

Nuclear Engineering Development Corporation. *Plutonium Utilization in Boiling Water Reactors Phase II Semi Annual Report January - June 1972, NEDC-10796.* General Electric Co., San Jose, (Jan. 1973).

Nuclear Engineering Development Corporation. *Plutonium Utilization in Boiling Water Reactors Phase II, 1969 Annual Report, NEDC-12111.* Nuclear Engineering Development Corporation (July 1970).

Nuclear Engineering Development Corporation. *Plutonium Utilization in Boiling Water Reactors - Phase II January - June, 1970, NEDC-12152.* General Electric Co., San Jose, (1970).

Nuclear Engineering Development Corporation. *Plutonium Utilization in Boiling Water Reactors Phase II – Semi-annual Report – January-June 1970, NEDC-12153.* Nuclear Engineering Development Corporation (1970).

Nuclear Engineering Development Corporation. *Core Design and Operating Data for Cycle 4, NEDC-25490.* General Electric Co., San Jose, (1981).

Nuclear Engineering Development Corporation. *Core Design and Operating Data for Cycle 5, NEDC-25491.* General Electric Co., San Jose, (1981).

Nuclear Materials and Equipment Corp. *Development of Plutonium-Bearing Fuel Materials – Progress Report, Period July 1 through September 30, 1961, USAEC Report NUMEC-P-80.* Nuclear Materials and Equipment Corp., (1961).

Nuclear Materials and Equipment Corp. *Development of Plutonium-Bearing Fuel Materials – Progress Report, Period April 1 through June 30, 1962, USAEC Report NUMEC-P-101.* Nuclear Materials and Equipment Corp., (1962).

Nuclear Materials and Equipment Corp. *Development of Plutonium-Bearing Fuel Materials – Progress Report, Period April 1 through June 30, 1963, USAEC Report NUMEC-P-105.* Nuclear Materials and Equipment Corp., (1963).

Nuclear Materials and Equipment Corp. *Development of Plutonium-Bearing Fuel Materials – Progress Report, Period July 1 through September 30, 1963, USAEC Report NUMEC-P-106.* Nuclear Materials and Equipment Corp., (1963).

O'Neill, G. L. *Plutonium Utilization in Boiling Water Reactors Phase II, Jan. - Jun. 1970, NEDC-12152.* General Electric Co., San Jose (Sept. 1970).

O'Neill, G. L. and R. L. Crowther. *Plutonium Utilization in Boiling Water Reactors Phase II Semi Annual Report July-Dec. 1970, NEDC-12198.* General Electric Co., San Jose, (May 1971).

Ogawa, S. Y. and D. L. Zimmerman. *Plutonium Utilization in Boiling Water Reactors Phase II: Post-Irradiation Examination of Four Fuel Rods After One Cycle of Operation in Big Rock Point Reactor, NEDC-10387.* General Electric Co., San Jose, (July 1971).

Oi, N., H. Maki, T. Kawada, et al. "BWR Fuel Performance and Recent Trend of R&D in Japan." *Proc. ANS Int. Topical Mtg. on LWR Fuel Performance*, Williamsburg, 17-20 April 1988. ANS, La Grange Park, 1988.

Olson, R. E. and A. J. Nardi. "Westinghouse Plutonium Fuels Development Laboratory Design Philosophy and Startup Experience." *Proc. Conf. Remote Systems Technology, 18th*. Amer. Nucl. Soc. (1970).

Orestano, F., and F. Pistella. *Heterogeneous Mockup of Plutonium With Europium in a Lattice of Uranium*. RT/FI-69-23. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1969.

Orestano, F., and F. Pistella. *Power Distributions and Reactivity Worths for Different Patterns in a UO_2-PuO_2 Microlattice*. RT-FI-69-49. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1969.

Orestano, F., and F. Pistella. *Power Distributions and Reactivity Worths for Different Patterns in a UO_2-PuO_2 Microlattice. Part II*. RT-FI-70-51. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1970.

Orestano, F., and F. Pistella. *Neutron Calculations for the Setup of a Plutonium Lattice in the Ritmo Reactor*. RT/FI-70-45. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1970.

Orestano, F., and F. Pistella. *Recycle of Plutonium. Physics Experiments Carried out in the Casaccio Reactor*. RT/FI-70-48. Comitato Nazionale per l'Energia Nucleare, Rome, Italy, 1970.

Orestano, F., F. Pistella, and G. Testa. "Significant Items on the Features of Plutonium Recycle form Physical and Technological Standpoints." *Trans. ANS*, **14**, No. 2, 824-5 (Oct. 1971).

Orestano, F., and F. Pistella. "Evaluation of Neutron Spectral Effects in a Plutonium-Fueled Lattice by Foil Activation." *Nucl. Sci. Eng.* **46**, No. 3, 376-84 (Dec. 1971).

Orestano, F., and F. Pistella. "Application of Gd as a Burnable Poison for Light Water Reactors." *Energ. Nucl.*, (Milan), **19**: No. 5, 337-41 (May 1972).

Orestano, F. and F. Pistella. *Assessment of the Reliability of the CNEN's Methods for the Neutronic Design of PWR's, Part I: Analysis of the Belgian Plutonium Recycling Program Experiments. RT/FI(72)45*. Comitato Nazionale per L'Energia Nucleare, Rome (1972).

Organization for Economic Cooperation and Development. *Pu Fuel - An Assessment*. OECD, Nuclear Energy Agency, Paris, France (1989).

Orr, W., et al. *Nuclear Design of the Saxton Partial Plutonium Core, WCAP-3385-51*. Westinghouse Electric Corp., Pittsburgh, (Dec. 1965).

Orr, W. L. "Westinghouse Pu Recycle Experience and Design." *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977. ANS-Fuel Cycle Division (1977).

Pacific Northwest Laboratories. *Plutonium Utilization Program Technical Activities, Quarterly Report, September-November, 1967, BNWL-654*. Pacific Northwest Laboratories (1968).

Pacific Northwest Laboratories. *Plutonium Utilization Program Technical Activities, Quarterly Report June-August 1968, BNWL-907*. Pacific Northwest Laboratories (1968).

Pacific Northwest Laboratories. *Plutonium Utilization Program Technical Activities, Quarterly Report September-November 1968, BNWL-963*. Pacific Northwest Laboratories (1969).

Pacific Northwest Laboratories. *Plutonium Utilization Program Technical Activities, Quarterly Report June-August 1969, BNWL-1224*. Pacific Northwest Laboratories (1969).

Pacific Northwest Laboratories. *Plutonium Utilization Program Technical Activities, Quarterly Report March-May 1970, BNWL-1442*. Pacific Northwest Laboratories (1970).

Pajot, J. "Fuel Fabrication Plants for Mixed Oxide Fuels." Proc. SFEN/BNS Mtg. Tech. Session on Nuclear Fuel Cycle, Paris, Oct. 1987. CEA-CONF-9351.

Parkes, P., and W. Baxter. "The Short Binderless Route for Mixed Oxide Fuel Production." *Proc. European Nucl. Soc. Conf.*, Lyon, France, Sept. 1990.

Parrat, D., Y. Musante, and A. Brissaud. "Mixed Oxide Fuel in Defective Experimental Rod EDITHMOX 1: Irradiation Results and Metallographic PIE." *Proc. Tech. Committee Mtg. On Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms, and Management*, Dimitrovgrad, Russia, May 26-29, 1992.

Parrat, D., et al. "Behavior of a Defective MOX Fuel Rod in a PWR." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, UK, July 3-7, 1995.

Patterson, J. "Re-Using Plutonium in Thermal Reactors." *Nucl. Engr. Int. Special Publication: British Reprocessing*, (Oct. 1990), 8.

Perolat, J. P. "MOX Pellet Reference Material." Proc. Annual Symp. Of the European Safeguards Research and Development Association (ESARDA) Symposium on Safeguards and Nuclear Material Management (13th), Avignon, May 14-16, 1991.

Peterson, R. E. Neutron Kinetic Problems Associated with Mixed-Oxide Fuels, HW-81259. General Electric Co. (1964).

Philipponeau, Y. "Thermal Conductivity of $(U,Pu)O_{2-x}$ Mixed Oxide Fuel." *J. Nucl. Mater.*, **188** (1992) 194.

Pistella, F. "A Review of CNEN's Reactor-Physics Work on Pu Recycling in LWRs." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Plucinski, C. E. *Water and Gas Determinations in Mixed Oxides, WHAN-SA-62*. (1970).

"Plutonium Utilization in Commercial Power Reactors." *Nucl. Tech.*, Vol. 15, No. 2, (Aug. 1972) 107-317.

Porrot, E., et al. "Analytical In-Pile Experiments on MOX." *Proc. European Nuclear Conf.*, Lyon, France, Sept. 23-28, 1990.

Porsch, D., et al. "Spaltstoffrueckfuehrung in Druckwasserreaktoren. Weiterentwicklung der Auslegung und Verifikation an Anwendungsfaellen. Abschlussbericht. (Fuel recycling in pressurized water reactors. Development of design and verification in different nuclear power stations. Final report.)" KWU-B-314/90/410. Siemens A. G., 1991. NTIS No. TIB/A94-04686/HDM.

Powell, D. J. "UK Prepares for the MOX Market." *Nucl. Engr. Int.*, **37**, No. 451 (1992).

Puechl, K. H. "Plutonium Fuel Development Programs at Nuclear Materials and Equipment Corporation." *Plutonium as a Power Reactor Fuel – Proc. Am. Nucl. Soc. Top. Mtg*, Richland, Washington, Sept. 13-14, 1962. *USAEC Report HW-75007*. Hanford Atomic Products Operation, General Electric Co., and USAEC, (Dec. 1962).

Raleigh, H. D. "Economics of Plutonium Recycle in Thermal Reactors." *Reactor Fuel Reprocessing Technol.*, **12**, *4* (1969).

Rand, M. H. and T. L. Markin. "Some Thermodynamic Aspects of $(U,Pu)O_2$ Solid Solutions and Their Use as Nuclear Fuels." *Proc. Symp. Thermodynamics of Nuclear Materials with Emphasis on Solution Systems*, Sept. 4-8, 1967. IAEA, Vienna (1967).

"A Review of Plutonium Utilization in Thermal Reactors." *Nucl. Tech.*, Vol. 18, No. 2, (May 1973) 78-176.

Rim, C. S., et al. *Performance Characteristics of Mixed PuO_2-UO_2 Fuels in Pressurized Water Reactors, WCAP-8388.* (1975).

Rippon, S. E. "MOX Technology For New Nuclear Fuel Fabrication." *Modern Power Systems*, Vol. 14, No. 1, (Jan. 1994).

Robkin, M. A. *Program for Development of Plutonium Recycle for Use in Light Water Moderated Reactors, Final Report, GEAP-4097 or 4907.* (Aug. 1965).

Roepenack, H. "MOX Fuel Rods for LMRs and LWRs." *Atomwirtschaft*, **143** (Mar. 1983).

Roepenack, H., F. Schlemmer, and G. Schlosser. "KWU/ALKEM Experience in Thermal Pu-Recycling." *Proc. Specialists' Mtg. Improved Utilization of Water Reactor Fuel With Special Emphasis on Extended Burnups and Plutonium Recycling*, Mol, Belgium, May 1984. IWGFPT/20, p. 208, IAEA, Vienna (1984).

Roepenack, H., V. Schneider, and W. G. Druckenbrodt. "Experience With the AUPuC Co-Conversion Process for Mixed Oxide Fuel Fabrication." *Am. Ceram. Soc. Bull.*, **63**, (1984) 1051.

Roepenack, H., F. U. Schlemmer, and G. J. Schlosser. "Development of Thermal Plutonium Recycling." *Nucl. Tech.*, **77** (1987) 175-186.

Roll, J. B. "Commercial Plutonium Recycle." *Power Eng.*, (Nov. 1970) 34.

Rome, M., M. Salvatores, and M. Lebars. "Plutonium Reload Experience and Perspective in French PWRs." *Trans. Am. Nucl. Soc.*, **60**, (1989) 114.

Rome, M.; et al. "Plutonium Reload Experience in French Pressurized Water Reactors." *Nucl. Tech.*, **94**, (April 1994) 87.

Rome, M., M. Lebars, and M. Francillon. "Plutonium Recycling Experience in French PWR Plants, Neutronic Results." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Rose, P. C. *Neutron Radiographic Measurements on Fuel Rods Removed from the Quad Cities-1 Reactor Following Cycle 2, NEDC-25493*. General Electric Co., San Jose, (July 1982).

Roser, T. "Hanau MOX: Why Are We Waiting?" *Nucl. Engr. Int.*, (1992) 36-37.

Ross, W. J. and E. M. Benson. "Babcock & Wilcox Plutonium Fuel Fabrication Experience." *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977. ANS-Fuel Cycle Division (1977).

Rossett, F.-D., M. Soldevila, C. and C. Bangil. "Increase in the Plutonium Recycling Ratio in French PWRs." *Trans. Am. Nucl. Soc.*, **66**, (1992) 458.

Runnalls, O. J. C. "Plutonium Fuel Program in Canada." *Proc. Plutonium as a Power Reactor Fuel*, Richland, Washington, Sept. 13-14, 1962. *HW-75007*. Hanford Atomic Products Operation (1962).

Runnalls, O. J. C. *Plutonium Fuels for Thermal Reactors, AECL-2102*. Atomic Energy of Canada, Limited (1964).

Russell, L. E., et al. "Observations on Phase Equilibria and Sintering Behavior in the PuO_2-UO_2 System." *J. Nucl. Mat.*, **5**, (1962) 216.

Russell, L. E. and N. H. Brett. "The Sintering Behaviour and Stability of $(Pu,U)O_2$ Solid Solutions." *Trans. Brit. Ceram. Soc.*, **62** (1963) 97.

Saji, E. and H. Shirayanagi. "Analyses of Boiling Water Reactor Mixed-Oxide Critical Experiments with CASMO-4/SIMULATE-3." *Nucl. Sci. and Engr.*, **121**, (1995) 52.

Sakata, T. "Nuclear Fuel Recycling in Japan." *Proc. Economics of U and Pu Recycling and Status of MOX Fuel Utilization*. OECD, NF-92/4, 1992.

Sakurai, S., et al. "Effective Utilization of Uranium and Plutonium in LWR Cores – Plutonium Utilization Improvements in BWRs." *Trans. Am. Nucl. Soc.*, **66**, (1992) 463.

Sani, L. "Considerations on Plutonium Utilization in Power Reactors." *Proc. IAEA Panel on Plutonium Utilization*, Vienna, Sept. 1968. International Atomic Energy Agency.

Sani, L. and G. Schileo. "Italian Programs for Plutonium Utilization." Foratom Symposium on Industrial Aspects of the Nuclear Fuel Cycle in Europe, Stockholm, Sept. 1970.

Sani, L. and G. Schileo. "Plutonium Utilization in Thermal and Fast Reactors." *Proc. Fourth Int. Conf. Peaceful Uses of Atomic Energy*, A/CONF 49/P-185. United Nations and International Atomic Energy Agency, Vienna (1972).

Sanoit, J. "Study of New U(VI) and Pu(VI) Coprecipitation Methods for the Preparation of $(U,Pu)O_2$." CEA, Paris, 1990.

Sasaki, M., T. Iwase, and K. Kanda. "Effective Utilization of Uranium and Plutonium in LWR Cores – Plutonium Utilization Improvements in PWR Cores." *Trans. Am. Nucl. Soc.*, **66**, (1992) 464.

Sauvage, T. *Redistribution During Irradiation of the Constituents and Fission Products in a Uranium-Plutonium Mixed Oxide Fuel, CEA-R-4649*. Commissariat a l'Energie Atomique, Paris (1975).

Sayers, J. B. and J. H. Worth. "Comparison of the Irradiation Behavior of 1% PuO2 in UO2 and Stoichiometric UO_2." *Proc. Power Reactor Experiments*, Vienna, Oct. 23-27, 1961. IAEA, Vienna (1962).

Schenk, H. J., K. L. Huppert, and W. Stoll. "Experience With the Use of Recycle Plutonium in Mixed-Oxide Fuel in Light Water Reactors in the Federal Republic of Germany." *Nucl. Tech.*, **43**, (1979) 174.

Schileo, G. "CNEN's Activities and Facilities for the Plutonium Utilization." *Comit. Naz. Energia Nucl., Notiziario*, **15**, *2*, (1969) 61.

Schileo, G. "A Review of the Italian Programe for the Recycling of Plutonium in Thermal Reactors." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Schileo, G. "A Review of the Italian Activities for the Thermal Recycling of Plutonium." *Nucl. Tech.*, **16**, (Nov. 1972) 360.

Schlemmer, F.U., H.-P. Fuchs, and R. Manzel. "Status of the Irradiation Experience with Recycled Fuel Materials in the FRG for Siemens/KWU Type Fuel Assemblies." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Schlosser, G., U. Fries, and M. Knorer. *Kuhlmitteltemperaturkoeffizienten in Plutonium- und Urangittern und ihre Anwendung auf Plutonium- und Urangittern und ihre Anwendung auf Plutonium-Ruckfuhrungszustande im Kernkraftwerk Obrigheim, KWU 397/73*. Kraftwerk Union (1973).

Schlosser, G., W. Bohm, and P. Uyttendaele. "Plutoniumruckfuhrung im Umfang der Eigenerzeugung am Beispiel eines 1300 Mwe Druckwasserreaktors." Presented at Reactor Conference, Karlsruhe (1973).

Schlosser, G., and R. Manzel. "Recycling of Plutonium in Light Water Reactors, Part 1, Demonstration Programs." *Siemens Forsch. Entwicklungsber.*, **8**, (1979) 108.

Schlosser, G., and P. Uyttendaele. "Recycling of Plutonium in Light Water Reactors, Part 2, Prospective Commercial Recycling." *Siemens Forsch. Entwicklungsber.*, **8**, (1979) 153.

Schlosser, G., et al. "Plutonium Recycling in LWRs: KWU Experience and Commercial Designs." *Trans. Am. Nucl. Soc.*, **31**, (1979) 281.

Schlosser, G. J. and S. Winnik. "Thermal Recycling of Plutonium and Uranium in the Federal Republic of Germany: Stategy and Current Status." *Proc. Int. Symp. Back End of the Nuclear Fuel Cycle: Strategies and Options*, Vienna, Austria, 1987. IAEA-SM-294/33, p. 541, IAEA (1987).

Schlosser, G. J., W-D. Krebs, and P. Urban. "Experience in PWR and BWR Mixed-Oxide Fuel Management." *Nucl. Tech.*, **102**, (1993) 54.

Schlosser, G. J. "MOX Fuel Utilization in Existing German LWRs: Design and Safety Aspects, Core Performance, and Irradiation Experience." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Schlosser, G. J. and D. Porsch. "Safe Utilization of Weapons Grade Plutonium in Existing PWRs – Supported by German MOX Recycling Experience." *Proc. Am. Nucl. Soc. Int. Top. Mtg. Safety of Operating Reactors*, Sept. 17-20, 1995, Seattle, WA. (1995) 785.

Schmid, L. C., et al. *Reactor Physics Data for the Utilization of Plutonium in Thermal Power Reactors, BNWL-801*. Pacific Northwest Laboratories (1967).

Schmid, M. "Rueckfuehrung von Plutonium sus hochabgebrannten Kernbrennstoffen in den Brennstoffkreislauf (Mixchoxid-Brennelementfertigung). (Recycling of plutonium of high burn-up nuclear fuel into the fuel cycle (fuel fabrication))." KFK-4862. KfK, Karlsruhe, (Sept. 1991).

Schmiedel, P. "Recycling of U and Pu." *Nucl. Eur. Worldscan*, **3-4** (1990).

Schmitz, F. and A. Marajofsky. "Autodiffusion du Plutonium Dans $(U,Pu)O_{2-x}$." *Thermodynamics of Nuclear Materials*, 1974, I. IAEA, Vienna, (1975) 457.

Schneider, V. W. "Problem of the Fabrication and Use of Pu-Bearing Fuel Elements." *Proc. 2nd International Fair and Technical Meetings for Nuclear Industries*, Basel, Switzerland, Oct. 6-11, 1969.

Schneider, V., F. Herrmann, and W. G. Druckenbrodt. "The AUPuC-Process: A Coprecipitation Process With Good Product Homogeneityto the Full Scale of Plutonium Concentration." *Trans. Am. Nucl. Soc.*, **31**, (1979) 176.

Schneider, V., R. Gueldner, and G. Braehler. "MOX-Fuel Fabrication in the Federal Republic of Germany." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35, IAEA (1990).

Sha, W. T., P. R. Huebotter, and R. K. Lo. "The Effect of Plutonium Migration on Allowable Power Rating and Doppler Broadening." *Trans. Am. Nucl. Soc.*, **14**, (1971) 183.

Sharp, R. E. *Plutonium Spike Fuel Elements for the PRTR, Part II – The Mark I-H, HW-69200.* Hanford Atomic Products Operation (1961).

Shea, J. J. "Summary of Meeting Held on November 1, 1979." Letter to Rochester Gas & Electric Company (RG&E) dated Nov. 16, 1979. Nuclear Regulatory Commission (1979).

"Siemens Abandons Hanau MOX Plant." *Nuclear News*, (August 1995).

Smalley, W. R. *Saxton Plutonium Program – Semi-Annual Progress Report for the Period Ending June 30, 1969, WCAP-3385-20.* Westinghouse Electric Corp. (1969).

Smalley, W. R. *Saxton Core II Fuel Performance Evaluation Part I Materials, WCAP-3385-56, Pt. 1.* (1971).

Smalley, W. R. *Evaluation of Saxton Core III Fuel Materials Performance, WCAP-3385-57.* Westinghouse Electric Corp. (1974).

Smith, R. C., L. G. Faust, and H. H. van Tuyl. *The Effect of Radiation Levels from Plutonium on Fuel Fabrication Process Design, BNWL-SA-3139.* Battelle-Northwest (1970).

Smith, R. C., et al. *Critical Experiments with the UO_2-2 wt.% PuO_2 Batch Core in the PRTR, BNWL-1553.* Pacific Northwest Laboratories (1971).

Smith, R. C., L. G. Faust, and L. W. Brackenbush. "Plutonium Fuel Technology Part II: Radiation Exposure from Plutonium in LWR Fuel Manufacture." *Nucl. Tech.*, **18** (May 1973) 97.

Sofer, G. R., J. R. Tomonto, and J. S. Tulenko. "Plutonium as an Alternate Fuel in Large Water Cooled Reactors." Presented at Winter Mtg., Nov. 1969, Los Angeles, California. Amer. Soc. Mech. Engr. (1970).

Sommer, D. *Einfluss der Kuhlmitteleintrittstemperatur auf die lokalen Leistungen in Uran- und Plutonium-Brennelementen, KWO-75-N-4.* Kernkraftwerk Obrigheim (1975).

Sommer, D. "Betriebserfahrungen mit Plutonium-Brennelementen und Stretch-out im 5. Zyklus." Presented at Reactor Conference, Nurnberg (1975).

Sommer, D. "Comparison of the Power Distribution in Uranium and Plutonium Fuel Elements in the KWO Core." *Nucl. Tech.*, **32**, (Mar. 1977) 257.

Southern California Edison Company and San Diego Gas & Electric Company. "San Onofre Nuclear Generating Station Unit 1 – Part I Operating History and Verification of Design Objectives, Appendix A." Southern California Edison Company and San Diego Gas & Electric Company.

Southern California Edison Company and San Diego Gas & Electric Company. "Application of Southern California Edison Company and San Diego Gas & Electric Company for a Class 104 (b) License to Acquire, Possess, and Use a Utilization Facility as a Part of Unit No. 1 of the San Onofre Nuclear Generating Station of Southern California Edison Company and San Diego Gas & Electric Company – Amendment No. 19." Southern California Edison Company (July 28, 1970).

Stehle, H., H. Abmann, and F. Wunderlich. *Nucl. Engr. and Design*, **33** (1975) 230.

Stehle, H., et al. "Experience With Pu-Recycle Feul for Large Light Water Reactors in the Federal Republic of Germany." *IAEA International Conference on Nuclear Power and its Fuel Cycle*. Salzburg, Austria, (May 1977).

Steinert, C. and M. Dillig. "Parametric Study of Plutonium Recycling in Obrigheim Power Station." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Stinson, W. P. "Subcritical Measurements with UO_2-2 wt.% PuO_2 Fuel." *Plutonium Utilization Program Annual Report Fiscal Year 1967, BNWL-624*. Pacific Northwest Laboratories (1968).

Stinson, W. P. and J. H. Lauby. "Approach to Critical Experiments with UO_2-4 wt.% PuO_2-H_2O Lattices." *Plutonium Utilization Program Technical Activities Quarterly Report September - November 1968, BNWL-963*. Pacific Northwest Laboratories (1969).

Stinson, W. P. and V. O. Uotinen. *Destructive and Nondestructive Analysis of Plutonium Fuels Used in Lattice Experiments, BNWL-1627*. Pacific Northwest Laboratories (1971).

Stoll, W. "Plutonium Technology." *Atomwirt., Atomtech.*, **15**, (1970) 580.

Stoll, W. "Experience in Fabricating Plutonium Recycle Fuel." *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977. ANS-Fuel Cycle Division (1977).

Storrer, J. and D. H. Locke. "High Burnup Experience in Vulcaine." *Nucl. Engr. Int.*, Vol. 15, No. 165 (1970).

Takahashi, Y. "Tsuruga MOX Fuel Utilization Program." *J. of Atomic Energy Society of Japan*, **28**, (1986) 203.

Tanaka, K., S. Uematsu, and S. Sato. "PCMI Behavior of MOX Fuel at Middle of Fuel Life in IFA-554-555." *Proc. Enlarged Halden Programme Group Mtg.*, Bolkeso, Norway, Feb. 11-15, 1990.

Taylor, E. G. *Saxton Plutonium Program Critical Experiments for the Saxton Partial Plutonium Core, EURAEC-1493/WCAP-3385-4.* Westinghouse Electric Corp. (1965).

Taylor, E. G. *Critical Experiments for the Saxton Partial Plutonium Core, WCAP-3385-54.* Westinghouse Electric Corp., Pittsburgh, (Dec. 1965).

Teske, K., et al. *J. Nucl. Mater.*, **188** (1992) 226.

Thiesen, R. and D. Vollath. "Plutonium Distribution and Diffusion in UO_2-PuO_2 Ceramics." *Proc. Use of Plutonium as a Reactor Fuel*, Brussels, March 1967. IAEA, Vienna (1967).

Thomas, I. D. "Plutonium Fuel Experience at Hanford." *Proc. Plutonium as a Power Reactor Fuel*, Richland, Washington, Sept. 13-14, 1962. *HW-75007.* Hanford Atomic Products Operation (1962).

Thomas, W. and U. Hesse. "Analyse spezieller Sicherheitsfragen bei der thermischen Rezyklierung im Brennstoffkreislauf fuer Druckwasserreaktoren. (Analysis of special safety questions for thermal recycling in the fuel cycle of pressurized water reactors)." BMU-1991-285, GRS-A-1329. BMU, Bonn, 1991.

Thomas, W. and K. Hummelsheim. "Kernbrennstoffmengen bei der thermischen Rezyklierung – typische. (Fissile material masses during thermal recycling – typical data)." BMU-1991-321, GRS-A-1611. BMU, Bonn, 1991.

Thomas, W. "Use of Mixed Oxide Fuel in Existing Light Water Reactors in Germany." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Thompson, R. L., A. J. Impink, and R. W. Colombos. *Saxton Core II Nuclear Operations Summary, WCAP-3385-55.* (Dec. 1970).

Tinturier, B. "Nuclear Power Research and Development Trends in France at the Beginning of the 21st Century." *Revue de l'energie*, **41**, *423*, (1990), 420.

Tomonto, J. R., J. S. Tulenko, and A. Veras. "A Demonstration Program of Plutonium Recycle in a Boiling Water Reactor." *Trans. Amer. Nucl. Soc.*, **12**, (1969) 232.

Toppel, B. J., P. J. Vogelberger, Jr., and E. A. Wimunc. *Safety Analysis Associated with the Plutonium Recycle Experiment in EBWR, ANL-6841.* Argonne National Laboratory (1964).

Toppel, B. J. and R. Avery. *Some Physics Aspects of the Plutonium Experiment in the EBWR, ANL-7019.* Argonne National Laboratory (1965).

Trauwaert, E., et al. "Recent Progress in Manufacturing of Plutonium Fuel for Recycling in LWR," IAEA-SM 294/9. Presented at Int. Symp. on the Back-End of the Nuclear Fuel Cycle – Strategies and Options, Vienna, Austria, May 11-15, 1987.

Trauwaert, E., N. Mostin, and R. Lefevre. "MOX Fabrication Experience at Dessel." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Cadarache, France, Nov. 13-16, 1989. IWGFPT/35. IAEA (1990).

Tulenko, J. S. and G. T. Hamilton. *A Demonstration Program for Plutonium Utilization in a Boiling Water Reactor, NED-681*. United Nuclear Corp. (1970).

U. S. Atomic Energy Commission. *Draft – The Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in LWR's – GESMO, WASH-1327*. Vol. 1-4. U. S. Atomic Energy Commission, (Aug. 1974).

U. S. Nuclear Regulatory Commission. *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors, NUREG-0002*. Vol. 1-5. U. S. Nuclear Regulatory Commission, (Aug. 1976).

U. S. Nuclear Regulatory Commission. "Mixed Oxide Fuel – Order." *Federal Register*, **42**, No. 251, (Friday, Dec. 30, 1977), 65334.

U. S. Nuclear Regulatory Commission. "Mixed Oxide Fuel – Memorandum of Decision." *Federal Register*, **43**, No. 93, (Friday, May 12, 1978), 20575.

U. S. Nuclear Regulatory Commission. "Rochester Gas and Electric Corporation Docket No. 50-244 R. E. Ginna Nuclear Power Plant Amendment to Provisional Operating License – Amendment No. 32." Nuclear Regulatory Commission (1980).

U. S. Nuclear Regulatory Commission. "Safety Evaluation by the Division of Operating Reactors Concerning Rochester Gas and Electric Corporation's Application for Amendment to its Operating License to Store Four Plutonium-Uranium Oxide Fuel Assemblies in the Spent Fuel Pool at the R. E. Ginna Nuclear Power Plant Docket No. 50-244." Nuclear Regulatory Commission (1980).

U. S. Nuclear Regulatory Commission. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. to Provisional Operating License No. DPR-18 Rochester Gas and Electric Corporation R. E. Ginna Nuclear Power Plant Docket No. 50-244." Nuclear Regulatory Commission (1980).

Uckert, J., et al. "Spaltstoffrueckfuehrung in Siedewasserreaktoren. Abschlussbericht. (Fuel recycling in boiling water reactors. Final report)." SIEMENS-KWU-B-33/90/0808. Siemens/KWU, Offenbach, 1990.

Uematsu, S., K. Tanaka, and M. Matsumoto. "ATR MOX Fuel Performance Under Load Follow Operation." *Proc. ANS-ENS Topical Mtg.*, Avignon, France, April 21-24, 1991.

Uematsu, S., et al. "Recent Results of PIE on MOX Fuel For Demonstration ATR." *ANP'92: International Conference on Design and Safety of Advanced Nuclear Power Plants*, Oct. 25-29, 1992, Tokyo, Japan. (CONF-921003) pp. 19.1/1 - 19.1/17.

Uotinen, V. O. and L. D. Williams. *Experiments and Calculations for H_2O-Moderated Assemblies Containing UO_2-2 wt% PuO_2 Fuel Rods, BNWL-SA-1107.* Pacific Northwest Laboratories (1967). Also in *Trans. Am. Nucl. Soc.*, **10**, (1967) 186.

Uotinen, V. O., et al. "Reactor Kinetic Parameters of Lattices of Plutonium and Uranium in Water." *Nucl. Sci. Eng.*, **44** (1971) 66.

Uotinen, V. O., W. P. Stinson, and S. P. Singh. "Kinetic Parameters in a Plutonium Lattice Containing Water Regions." *Nucl. Tech.*, **15**, (Aug. 1972) 109.

Uotinen, V. O., et al. "Lattices of Plutonium-Enriched Rods in Light Water – Part I: Experimental Results." *Nucl. Tech.*, **15**, (Aug. 1972) 257.

Uotinen, V. O., B. R. Leonard, Jr., and R. C. Liikala. "The Neutronics of Plutonium Recycling." *Nucl. Tech.*, **18**, (May 1973) 115.

"Use of Plutonium as a Fuel in Nuclear Reactors, Quarterly Report No. 11, April 1 - June 30, 1963." United States - Euratom Joint Research and Development Program, Report EURAEC-790. Centre d'Etude d l'Energie Nucleaire and Societe Belge pour l'Industrie Nucleaire, Brussels, (July 1963). [BN-6307-03, translation].

van der Meer, K., et al. "Venus: Determining the Neutronic Behavior of Mixed Oxide Fuel." *Proc. Am. Nucl. Soc. Int. Top. Mtg. Safety of Operating Reactors*, Sept. 17-20, 1995, Seattle, WA. ANS, (1995) 792.

Van Dievoet, J., et al. "Plutonium Recycle in LWRs - Problems and Prospects." *Nucl. Energy Maturity, Proc. European Nuclear Conf.*, Paris, France, April 21-25, 1975.

Van Dievoet, J. P. "Using Plutonium in Water Reactors." *Trans. Am. Nucl. Soc.*, **24**, (1976) 9.

Van Dievoet, J., et al. "MOX Fuel and Its Fabrication in Europe." *Trans. Am. Nucl. Soc.*, **61**, Suppl. 1, (1990) 293.

Van Vliet, J., et al. "Industrial Qualification of TU2 Powder Use For MOX Fuel Fabrication." *Nuclear Europe Worldscan*, 11-12, 1994.

Vanden Bemden, E. "Description of the Belgonucléaire-C.E.N. Plutonium Project." *Proc. Plutonium as a Power Reactor Fuel*, Richland, Washington, Sept. 13-14, 1962. *HW-75007.* Hanford Atomic Products Operation (1962).

Vanden Bemden, E. "General Description of the Belgian Programme on Plutonium Recycling in Thermal Reactor." *Proc. IAEA Panel on Plutonium Recycling in Thermal Power Reactors*, Vienna, Austria, June 21-25, 1971. IAEA, **143**, (1972).

Vanderborck, Y., D. Haas, P. Deramaix, and A. Vandergheynst. "Commercial MOX Fuel Fabrication and Irradiation Experience in Belgium." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Vandergheynst, A. and J. van Vliet. "MOX Fuel Fabrication, a Mature Industry." *Proc. The Uranium Institute Annual Symposium 1994*, London, Sept. 1994.

Vandergheynst, A. and M. Debauche. "Key Issues of MOX Fuel Plant Engineering." *Proc. ICONE 3*, Kyoto, Japan, Apr. 23-27, 1995.

Veeder, J. and R. Didsbury. *A Catalogue of Advanced Fuel Cycles in CANDU PHW Reactors, AECL-8641*. Atomic Energy of Canada, Limited, (1985).

Vielvoye, A. and H. Bairiot. "Economic Optimization of MOX Fuel." *Nucl. Eur. Worldscan*, **1-2** (1991).

Vincent, F. "Gaining Good Experience With MOX at French PWRs." *Nucl. Engr. Int.*, **35**, (Dec. 1990), 35.

Voll, R. W. "Plutonium Recycle R&D and Operating Experience at Big Rock Point." *Proc. Topical Mtg. on the Plutonium Fuel Cycle*, Bal Harbour, Florida, May 2-4, 1977. ANS-Fuel Cycle Division (1977).

Votinen, V. O., B. R. Leonard, Jr., and R. C. Liikala. "The Neutronics of Plutonium Recycling." *Nucl. Tech.*, Vol. 18, No. 2, (May 1973) 115-140.

Wakerley, M. W., G. D. Burholt, and A. Palit. *Waste Management and Radiological Safety Implications of Using Plutonium as a Fuel in UK Thermal Reactors, DOE-RW-89.089*. Department of the Environment, London, (Aug. 1989).

Walke, G. J. "Request for Change to the Technical Specifications License No. DPR-6." NRC Docket 50155-16. (Dec. 22, 1969).

Walke, G. J. "Additional Information: Change No. 19, Request for Change to the Technical Specifications License No. DPR-6." NRC Docket 50155-22. (Jan. 28, 1970).

Walker, C. T., et al. "Irradiation Behaviour of MOX Fuel: Results of EPMA Investigation." *Nucl. Engr. Des.*, **131** (1991), 1-16.

Walker, C. T., W. Goll, and T. Matsumura. "MOX Fuel Irradiation Behaviour: Results From X-Ray Microbeam Analysis." *Proc. Tech. Committee Mtg. On Recycling of Plutonium and Uranium in Water Reactor Fuels*, Newby Bridge, UK, July 3-7, 1995.

Walkup, P. C. *PRTR Fuel Element Rupture Test Facility Capabilities and Project History, BNWL-40*. Battelle-Northwest Laboratory (1965).

Wallace, T. W. *EEI - Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period 4/73 - 9/73, WCAP-4167-6.* (Dec. 1973).

Wallace, T. W. *EEI - Westinghouse Plutonium Recycle Demonstration Program, WCAP-4167-V7.* (1975).

Wan, M. Y. *Dresden I Plutonium Recycle Program Report, GA-A14106.* General Atomics, (July 1976).

Watteau, M. "Advances in Fuel Design and Fuel Management Using Reprocessed Materials." *Proc. TOPFUEL 95*, Wurzburg, Mar. 12-15, 1995.

Westinghouse Electric Corporation. *Economic Utilization of Plutonium in Pressurized Water Reactors, Final Report, WCAP-7160.* Westinghouse Electric Corporation, (1968).

Westinghouse Electric Corporation. *Saxton Plutonium Program Semiannual Progress Report for the Period Ending December 31, 1969, WCAP-3385-22.* Westinghouse Electric Corporation (1970).

Westinghouse Electric Corporation. *EEI-Westinghouse Plutonium Recycle Demonstration Program Progress Report for the Period Ending April, 1970, WCAP 4167-1.* Westinghouse Electric Corporation (1970).

Westinghouse Electric Corporation. *Saxton Plutonium Project Quarterly Progress Report for the Period Ending December 31, 1970, WCAP-3385-26.* Westinghouse Electric Corporation (1971).

Westinghouse Electric Corporation. *Saxton Plutonium Project Quarterly Progress Report for the Period Ending June 30, 1971, WCAP-3385-28.* Westinghouse Electric Corporation (1971).

Westinghouse Electric Corp. *System Description for DIS Mixed-Oxide Fuel Fabrication Plant – Separate Feed Streams (PuO2-UO2) Case.* Contract No. EY-77-C-09-0945. Westinghouse Electric Corp., Nuclear Fuels Division, Pittsburgh, PA (1978).

Westinghouse Electric Corp. *System Description for DIS Mixed-Oxide Fuel Fabrication Plant – Co-Processed Feed (COMOX) Case.* Contract No. EY-77-C-09-0945. Westinghouse Electric Corp., Nuclear Fuels Division, Pittsburgh, PA (1978).

Westinghouse Electric Corp. *PDR Plutonium Disposition Study Phase II Final Report, DOE/SF/19683-5.* Westinghouse Electric Corp., Pittsburgh, PA (1994).

White, L. D., Jr. "Responses to Questions on Mixed Oxide Fuel (Non-Proprietary)." Letter to D. L. Ziemann dated Feb. 20, 1980. Rochester Gas & Electric Corporation (1980).

Wick, O. J., T. C. Nelson, and M. D. Freshley. "Plutonium Fuels Development." *Proc. Second United Nations Int. Conf. On the Peaceful Uses of Atomic Energy*, Geneva, 1958. United Nations, Geneva, (1958).

Wick, O. J. (ed.). *Plutonium Handbook.* USAEC. New York: Gordon and Breach, Science Publishers, 1967.

Wick, O. J. (ed.). *Plutonium Handbook – A Guide to the Technology – Volumes I & II.* American Nuclear Society (1980).

Wiese, H-W. "Investigation of High-Burnup LWR Uranium, MOX and Recycled-Uranium Fuel on the Basis of JEF-1 Data Validated at KfK." *Proc. Int. Conf. Physics of Reactors*, Marseille, France, April 23-26, 1990. Vol. 1, p. VI-44.

Wiese, H-W. "Investigation of the Nuclear Inventories of High-Exposure PWR Mixed-Oxide Fuels With Multiple Recycling of Self-Generated Plutonium." *Nucl. Tech.*, **102**, (April 1993) 68.

Wilcox, P. "The Use of Plutonium in the UK." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Williamson, H. E. and D. C. Ditmore. *Experience with BWR Fuel Through September 1971, NEDO-10505.* General Electric Co., (May 1972).

Windsor, H. and R. Goldstein. "Analysis of Lattices Containing Mixed-Oxide Fuel in Particular Form." *Trans. Am. Nucl. Soc.*, **15**, (1972) 107.

Wisely, H. R. *Specifications for Plutonium Dioxide, BNWL-42.* Battelle-Northwest Laboratory (1965).

Worden, J. R., R. C. Liikala, and W. A. Reardon. "An Analysis of Plutonium-Light Water Critical Experiments." *Physics Research Quarterly Report, January, February, March 1964, HW-81659.* General Electric Co. (1964).

Yamamoto, K., et al. "Melting Temperature and Thermal Conductivity of Irradiated Mixed Oxide Fuel." *J. of Nucl. Mat.*, **204**, (1993) 85.

Yamanaka, R. et al. "Post Irradiation of MOX Fuel." *Proc. IAEA Specialists' Mtg.*, Tokyo, Japan, 1994.

Yanagisawa, K. *Behavior of Water Reactor Fuel Rod, JAERI-M-90-120.* Japan Atomic Energy Research Inst., Tokyo, (Aug. 1990).

Yokote, M., Y. Kondo, and S. Abeta. "PWR Fuel in Japan – Progress and Future Trends." *Nucl. Eng. Int.*, **39**, (June 1994) 479.

Yomano, T. "Japanese Nuclear Fuel Recycle Plutonium Utilization Policy." *Proc. Mixed Oxide Fuel (MOX) Exploitation and Destruction in Power Reactors*, Obninsk, Russia, Oct. 16-19, 1994. Kluwer, 1995.

Zaloudek, F. R., and G. M. Hesson. *Calculations of the Surface Heat Flux and Fuel Temperatures for PRTR MK-I PuO2-UO2 Fuel Element with Segregated PuO2, HW-76115.* Hanford Atomic Products Operation, Richland, (Jan., 1963).

Ziemann, D. L. "Request for Additional Information (Ginna MOX) Docket No. 50-244." Letter to L. D. White, Jr. of RG&E dated Feb. 13, 1980. Nuclear Regulatory Commission (1980).

Zolotar, B. A. and J. T. A. Roberts. "EPRI Plutonium Recycle Demonstration Program." *Trans. Am. Nucl. Soc.*, **23**, (1976) 238.

Zwicky, H. U., et al. "Radial Plutonium and Fission Product Isotope Profiles in Mixed Oxide Fuel Pins Evaluated by Secondary Ion Mass Spectrometry." *Journal of Nucl. Mat.*, **202** (1993).

ACKNOWLEDGEMENTS

The authors would like to thank the many contacts in the fuels industry as well as the ORNL librarians who have helped the authors locate the numerous articles that are publicly available. The authors would also like to thank Roxanne Puglisi for her superb assistance in the editing and preparation of this manuscript.

INTERNAL DISTRIBUTION

1.	B. B. Bevard	20.	D. L. Moses
2.-6.	B. S. Cowell	21.	D. G. O'Connor
7.	W. G. Craddick	22	R. T. Primm, III
8.	R. J. Ellis	23.	D. J. Spellman
9.-13.	S. E. Fisher	24.-25.	Laboratory Records (for submission to OSTI)
14.	E. C. Fox	26.	Laboratory Records-RC
15.	J. C. Gehin	27.	Central Research Library
16.	S. R. Greene	28.	ORNL Patent Section
17.	S. A. Hodge	29.	D. L. Williams
19.	G. E. Michaels	30.	K. A. Williams

EXTERNAL DISTRIBUTION

31. M. L. Adams, Dept. of Nuclear Engineering, Texas A&M University, Zachry 129, College Station, TX 77843
32. D. Alberstein, Los Alamos National Laboratory, P. O. Box 1663, MS-K575, Los Alamos, NM 87545
33. J. F. Baker, Office of Fissile Materials Disposition, Dept of Energy, MD-1, 1000 Independence Avenue SW, Forrestal Bldg., 6G-050, Washington, DC 20585
34. C. A. Beard, 600 S. Tyler, Suite 800, Amarillo, TX 79101
35. J. Buksa, Los Alamos National Laboratory, P. O. Box 1663, Los Alamos, NM 87545
36. H. R. Canter, Office of Fissile Materials Disposition, Dept of Energy, MD-1/2, 1000 Independence Ave SW, Building 3F043, Washington, DC 20585
37. S. Chang, INEEL, P. O. Box 1625, MS-3885, Idaho Falls, ID 83415-3885
38. T. Cremers, Los Alamos National Laboratory, P. O. Box 1663, Los Alamos, NM 87545
39. Cygelman, Department of Energy, 1000 Independence Ave. SW, Forrestal Building 3F043, Washington, DC 20585
40. George A. Davis, ABB-Combustion Engineering, 2000 Day Hill, MC-9310-0422, Windsor, Connecticut 06095
41. Ed Ehrlich, General Electric, 175 Curtner Avenue, San Jose, CA 95125
42. L. Groves, Sandia National Laboratories, P. O. Box 969, Livermore, CA 94551
43. D. Harrison, Department of Energy, 101 Convention Center Drive, Suite P200, Las Vegas, NV 89109
44. C. Jaeger, Sandia National Laboratories, P. O. Box 5800, Albuquerque, NM 87185-0759
45. J. D. Nulton, Office of Fissile Materials Disposition, Dept. of Energy, MD-4, 1000 Independence Avenue SW, Washington, DC 20585
46. Odelli Ozer, Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94304
47. S. L. Passman, Sandia National Laboratories, 1401 Wilson Blvd., Suite 1050, Arlington, VA 22209

48. L. Peddicord, Dept. of Nuclear Engineering, Texas A&M University, Zachry 129, College Station, TX 77843
49. D. Peko, Department of Energy, 1000 Independence Ave. SW, Forrestal Building 3F042, Washington, DC 20585
50. P. T. Rhoads, Office of Fissile Materials Disposition, Dept of Energy, MD-4, 1000 Independence Ave. SW, Forrestal Building 3F043, Washington, DC 20585
51. S. S. Sareen, Sandia National Laboratories, 2650 Park Tower Drive, Suite 800, Vienna, VA 22180
52. J. H. Thompson, Office of Fissile Materials Disposition, Dept. of Energy, MD-4, 1000 Independence Ave. SW, Washington, DC 20585
53. Michael L. Travis, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, PA 15230
54. Rosa Yang, Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, CA 94304