
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.

^ad/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 $\frac{\text{Btu}}{\text{hr ft}}$)
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 UO₂ with 2 wt% PuO₂ 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

<u>Fuel Element Type</u>	<u>Number of Fuel Elements</u>	<u>Peak Linear Heat Rating (kW/ft)</u>	<u>Reactor Peak Burnup (Mwd/MTHM) (80% of Pu)</u>
A1 - Pu	75	15.1	
<u>UO₂</u>	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
<u>UO₂ - PuO₂</u>	216		
<u>UO₂ - 0.5 wt% PuO₂</u>	(81)		
Vipac	20	16.0	18,500
Swaged	61	13.0	12,500
<u>UO₂ - 1.0 wt% PuO₂</u>	(49)		
Vipac	16	13.6	11,500
Swaged	33	15.6	13,500
<u>UO₂ - 1.5 wt% PuO₂</u>	(1)	4.4	3,500
<u>UO₂ - 2.0 wt% PuO₂</u>	(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
<u>UO₂ - 4.0 wt% PuO₂</u>	(1)	27.0	1,250
<u>Pellets (cold press)</u>			
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 UO₂ 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/MTM
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 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/MTM 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/MTM. 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/MTM 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 cruding 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.	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

	<u>MOX Rods</u>
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 PuO₂ and UO₂)	
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			
	Solid	Dished	Annular Hole	
			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 described 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 commercial 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.

Big Rock Point/Exxon/NFSExxon 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 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/MTM 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 remained at a low level 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 Program⁴³

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 Italy⁴⁴

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 KWU. 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		Amount of Fissile Pu, kg	Burnup, Mwd/MTU	Number of Cycles	Matrix Material
		Fuel Assemblies	Fuel Rods				
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 ²³² Th
MZFR	1972	8	296	11.8	12,000	4	Natural uranium
KWO	1972	21	3780	158.9	28,500	3	Natural uranium

*VAK: Versuchatomkraftwerk 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 Institute⁴⁵ 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 ²³³U 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 MWe). 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/MTM. 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 tri-tex (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.