ASSESSMENT OF PWR PLUTONIUM BURNERS FOR NUCLEAR ENERGY CENTERS

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GLOSSARY

AC	Alternating Current
AGNS	Allied - General Nuclear Services
AOO	Anticipated Operational Occurrence
APR	All mixed Oxide Fueled Reactor (Plutonium Burner)
BNWL	Battelle Northwest Laboratories
BOC/BOL	Beginning of Cycle
BWR	Boiling Water Reactor
CE	Combustion Engineering
CEA	Control Element Assembly
CESSAR	Combustion Engineering Standard Safety Analysis Report
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EEL	Edison Electric Institute
EOC/EOL	End-of-Cycle (End of reactivity lifetime)
Equilibrium Cycle	A fuel cycle condition reached after many years of plant operation in which the nature of the feed assemblies, fuel placement, and cycle length remain fixed from cycle to cycle
GE	General Electric
GESMO	Generic Environmental Statement on Mixed Oxide Fuel
HFP	Hot Full Power
HM	Heavy Metal
HZP	Hot Zero Power
KWO	Kraftwerk Oerbrighiem
KWU	Kraftwerk Union

GLOSSARY (Cont.)

LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MOC/MOL	Middle of Cycle
MTC	Moderator Temperature Coefficient
NEC	Nuclear Energy Center
NFS	Nuclear Fuel Services
NSSS	Nuclear Steam Supply System
PWR	Pressurized Water Reactor
SGR	Self-Generated Plutonium Recycle
SIS	Safety Injection System
SLB	Steam Line Break (Secondary System)
SWU	Separative Work Unit
VIPAC	A Process for Manufacturing Fuel Rods Involving Vibrational Compaction of Mixed Oxide Powder
W/O	Weight Percent

The technical feasibility of Nuclear Energy Centers (NEC) and their potential for accelerating the introduction of nuclear power and contributing to domestic energy independence has since 1974 been the subject of continuing studies by a number of federal agencies. Although there are a number of alternative NEC concepts, one generally recognized as attractive is the clustering of nuclear power plants with reprocessing and mixed-oxide fabrication facilities on a single site. Recovered plutonium could then be used to fuel reactors located only at the NEC, eliminating the offsite shipment of plutonium. This concept could prove to be an attractive alternative to providing and paying the additional cost of safeguards required during the transportation and use of mixed-oxide fuel at dispersed sites. Reactors located at a NEC should be designed to accept high loadings of plutonium, so that the plutonium present in the spent fuel from reactors located off-site could be accommodated.

Combustion Engineering, under ERDA sponsorship, has undertaken a program to study the performance and safety characteristics of PWR plutonium burners, and to identify modifications to current PWR designs to enhance plutonium utilization. As a first step, the changes in core physics parameters which result from increased plutonium loadings and the implications of these changed parameters on reactor design and the safety analysis were identified. C-E's System 80TM plant, which has been designed to accommodate plutonium loadings up to and including that of an all-mixed-oxide-fueled core (APR), served as the basis for this phase of the study. Plants of the System 80TM type, although designed largely for dispersed siting and optimized for lower levels of plutonium utilization, might effectively be employed at NECs, since their advanced state of design would allow early introduction. The capability of accommodating various levels of plutonium utilization, from all-uranium fueling through the APR, would also provide the option of returning to uranium burning in later years, when accelerated LMFBR deployment might shift the demand for plutonium away from light-water reactors.

Although the characteristics of the all mixed oxide fueled reactor are qualitatively not very different from those of a plant operating in the self-generated recycle mode (SGR), changes in a number of system requirements and components are large enough that the design bases are substantively affected. In order to establish the bases for APR design, the basic core characteristics, the most important of which are moderator and Doppler coefficients, soluble boron and control rod worth, and delayed neutron fraction, are compared between SGR and APR operation. Lastly, the plant modifications required to provide full APR capability in a plant nominally designed for SGR levels of recycle are identified; these modifications include additional control rods, equipment designed to accommodate higher heat loads, modifications necessary to accommodate a higher fast neutron fluence and modifications to the balance of plant equipment.

Reactors designed primarily for dispersed site application, although possibly provided with the capability for a full mixed oxide core loading; are generally optimized for UO₂ and SGR operation, since optimization for higher plutonium loadings typically results in poor economy and operational difficulties with uranium fueling. At a nuclear energy center, reactors can be dedicated to plutonium fueling and optimization of the lattice to improve plutonium utilization might be desirable. Lattice cell survey calculations have been performed in which the lattice spacing,

fuel rod diameter, and fuel element design are varied to optimize for various NEC objectives. The optimal design depends, of course, on the specific objectives of the NEC; for example, designs with increased water-to-fuel ratios significantly reduce the fuel inventory for a given electrical capacity, while the different objective of reduced plutonium consumption per unit of energy output requires a core design with a decreased water-to-fuel ratio.

The sections which follow provide a detailed discussion of the subjects outlined in the preceding paragraphs. To round out the study, a survey of industrial experience with mixed oxide fueled systems from the standpoint of fuel performance in power reactors, design methodology development, and materials performance was also conducted. This survey has served to identify those areas which require further study or experimental confirmation. These are discussed in the final sections of this report and specific recommendations are made for additional work in the affected areas.

The major conclusions and recommendations resulting from this study are briefly discussed below. The reader is referred to the appropriate sections in the body of the text and to Section 6.0 for a comprehensive discussion of these items.

2.1

Summary and Conclusions

1. The purpose of this study was to explore the performance and safety characteristics of PWR plutonium burners, to identify modifications to current PWR designs to enhance plutonium utilization, to study the problems of deploying plutonium burners at Nuclear Energy Centers, and to assess current industrial capability for the design and licensing of such reactors.
2. A plutonium burner is defined to be a reactor which utilizes plutonium as the sole fissile addition to the natural or depleted uranium which comprises the greater part of the fuel mass. Plutonium burners, with lattice configurations typical of modern PWRs, have a specific power of 1.0 to 1.7 Mwth/kg fissile plutonium, depending on cycle, and for a typical 1300 Mw(e) plant, require a feed of approximately 450 kg fissile plutonium each year to replace the plutonium consumed in operation.
3. The results of this study and the design analyses performed during the development of C-E's System 80TM plant indicate that the use of suitably designed plutonium burners at Nuclear Energy Centers is technically feasible.
4. Although the characteristics of the PWR plutonium burner are not radically different from similar plants operating in the self-generated recycle mode, changes in a number of system requirements and components are large enough that the design bases are substantially affected. The most significant change in basic core characteristics is a reduction in soluble boron reactivity worth and the worth of individual control rods.
5. The consequences of anticipated operational occurrences and postulated accidents are no more severe in the plutonium burner than in the uranium or self-generated recycle modes of operation, provided the reactivity control systems are modified to compensate for the reduction in control poison effectiveness. In fact, an examination of plant safety characteristics indicates that in some respects the system responses under postulated accident conditions are more favorable for APR operation than for SGR.
6. PWRs employed as plutonium burners must be specifically equipped for this application. Additional equipment, which must be supplied or which require consideration of this operational mode during their design include: additional control rods, drives, nozzles and associated equipment to provide the required shutdown margin; equipment modifications to provide and accommodate higher soluble boron concentrations; core and spent fuel cooling equipment sized to accommodate the higher decay heat loads of irradiated mixed-oxide fuel; design of reactor vessel and internals to accommodate a greater flux of high energy neutrons; modifications to radwaste systems to accommodate higher tritium concentrations in the primary coolant; increased strength neutron sources; and the design of fresh fuel handling and storage, and spent fuel

storage facilities to accommodate mixed-oxide fuel. While it may be possible to modify existing PWRs to accommodate some of the items listed above, others would be difficult and/or prohibitively expensive if not incorporated at the design stage.

7. The lattice geometry of the plutonium burner can be optimized to contribute to achieving specific objectives of the Nuclear Energy Center such as uranium ore and separative work conservation, reduced power costs, and improved fuel management flexibility. However, the optimum design generally differs depending on objective, and, in some cases, modifications to achieve one particular objective can compromise other objectives.
8. Some improvement in uranium ore and separative work utilization can be achieved by decreasing the water-to-fuel ratio. The improvements which can be realized with modest changes in lattice spacing, however, are small, approximately a 2% reduction in requirements. Lattice configurations atypical of current PWRs are required to achieve significant reductions in uranium ore and separative work requirements.
9. Reductions in fuel cycle and power costs can be realized by increasing lattice pitch, but such optimization is at the expense of fuel management flexibility, i.e., of retaining the capability for uranium fueling. Annual savings in the cost of power of approximately 6.6 million dollars per plutonium burner, however, can be realized by increasing lattice pitch so as to obtain a water-to-fuel ratio of 3.0.
10. Fuel management flexibility, or the ability to fuel the plutonium burner with slightly enriched uranium fuel, is considered important to the orderly deployment of plutonium burners at Nuclear Energy Centers for the following reasons:
 - a. Plutonium requirements to start up a plant are larger than the annual makeup requirements, and this limits the rate of plutonium burner startup. A faster rate of plant startup can be achieved if the plutonium burners are initially fueled with uranium, with a transition to mixed-oxide fueling as plutonium becomes available.
 - b. In the absence of fuel-management flexibility, the plutonium burners would be dependent on the reprocessing and fabrication facilities for their annual fuel requirements. Large quantities of plutonium fuel would have to be stored to insure a continued fuel supply in the event of extended outages at one of these facilities. Such storage requirements are both costly, and can also limit the rate of plutonium burner deployment.
 - c. After the year 2000, it may be necessary to convert some plutonium burners to uranium fueling in order to provide sufficient plutonium for FBR deployment.
11. The use of low effective density fuel is an attractive alternative to increasing lattice spacing. With such fuel, a higher water-to-fuel ratio can be obtained without modifying lattice geometry or thermal hydraulic design, and hence the capability of uranium fueling is retained. While the power cost savings is not

quite so large as when the lattice spacing is modified, most of the advantage of a higher water-to-fuel ratio is retained. Annual savings in power cost with reduced density fuel is about 5.4 million dollars per plutonium burner, compared to a savings of 6.6 million dollars at an equivalent water-to-fuel ratio when lattice spacing is modified.

Low effective density fuel can be obtained either by the use of annular pellets or by the use of diluent material having a low neutron absorption cross section. Both concepts appear feasible, but both require further development for PWR plutonium burner applications.

12. It is concluded that PWR plutonium burners for Nuclear Energy Center siting should employ a lattice design similar to that of current PWRs, at least during the early phase of Nuclear Energy Center deployment. This conclusion derives from the importance placed upon retention of the uranium fueling option, and the likelihood that a majority of the economic gains identified in the lattice optimization study could be achieved with low effective density fuel. Further economic gains, or small improvements in uranium ore and separative work utilization which could be achieved by modifying the reactor lattice are viewed as insufficient to warrant the costs of developing and licensing modified lattice designs.
13. Based upon an assumed reprocessing facility size of 1500 MT/yr, a mature Nuclear Energy Center in addition will contain a 450 MT/yr mixed-oxide fabrication facility and fifteen 1300 Mw(e) plutonium burners. The center will service thirty-five 1300 Mw(e) uranium-fueled reactors (assumed to be located at dispersed sites) which provide the annual makeup plutonium required to fuel the plutonium burners. The NEC plutonium burners can be deployed at a reasonably rapid rate, two units per year except for the first and last two years when one unit per year is placed into operation, provided that a stockpiling of plutonium fuel is not required as a contingency for extended reprocessing or fabrication outages. If such stockpiles are required, the rate of plutonium burner startup must be reduced as the NEC matures, or alternately, initial plutonium burner deployment must be delayed in order to develop the required plutonium stockpile.
14. Deployment of about seven Nuclear Energy Centers, each eventually containing 15 plutonium burners, could be initiated before the year 2000. By the year 2000, about half the eventual number of plutonium burners would be installed, with the remainder placed into operation after that year. The primary impediment to achieving this level of deployment will be the difficulty of obtaining sufficient construction and operating experience at the initial NEC to support the timely deployment of additional NECs.
15. Sufficient plutonium to provide for the anticipated deployment of FBRs through the year 2000 is available even with the NEC and plutonium burner installation rate discussed above. Sometime after the year 2000, it will be necessary to reduce the rate of construction of plutonium burners, installing FBRs in their place; the evolution of the NEC after year 2000 is consequently toward a mixture of plutonium burners and FBRs. If the rate of FBR expansion is large, or if the FBR proves to have large specific inventories or long doubling times, it may be necessary to convert some plutonium burners to uranium-fueled operation.

16. Although a large quantity of core behavior data has been obtained on mixed-oxide fuels, both in the U.S. and in other countries, current industrial experience is not viewed as totally adequate to support the licensing and operation of plutonium burners. This conclusion results from the fact that commercial experience has been limited to the operation of reactors containing relatively small quantities of mixed oxide fuel. Experience with respect to fuel irradiation performance, fuel performance during simulated transient and accident conditions, and the design methods used to predict local power distributions within mixed-oxide assemblies are viewed as adequate, since these aspects have been demonstrated for mixed oxide fuel and are independent of the fraction of the reactor core which is comprised of such fuel.

Additional experience, however, is required for those aspects of plutonium burner design and operation which depend strongly on the quantity of mixed-oxide fuel in the core, such as reactivity coefficients and the prediction of global power distributions. Adequate experience in these areas should be obtained as commercial recycle begins and the quantities of mixed-oxide fuel are increased to, or possibly beyond, the level of self-generated recycle.

17. The use of thorium fuel enriched with plutonium appears to be an attractive alternative to the uranium-plutonium fuel cycle for plutonium burners located at the NEC. The use of the thorium fuel cycle reduces the number of plutonium burners required to utilize the plutonium produced by a given installed nuclear capacity by about one-third; this could be important in reducing the size and environmental impact of the NEC.

2.2 Recommendations

1. If plutonium burner Nuclear Energy Centers are to contribute significantly to meeting the nation's electrical demand by the 1990s, it is essential that experience be obtained with the construction and operation of the initial NEC at an early date, so as to provide the support necessary for further NEC deployment. It is recommended that the next increment of reprocessing capacity be sited so as to be suitable for expansion into an NEC. This will afford the possibility of developing an NEC at an early date, once the decision to deploy such centers is made.
2. The deployment of Nuclear Energy Centers containing plutonium burners, and the impact of the FBR on these centers have been evaluated through the year 2000, the period over which ERDA predictions of nuclear expansion are available. Although it is recognized that predictions of nuclear growth past the year 2000 are at best uncertain, NEC growth scenarios should be extended well past this date to more clearly delineate their development in that period.
3. It appears that the likely evolution of the NEC concept after the late 1990s is to NECs containing a mixture of plutonium burners and FBRs. If the economies of the NEC are to be fully realized with a mixture of LWRs and fast breeders, the reprocessing and mixed-oxide fabrication facilities located at the NEC will probably have to serve both reactor types. The feasibility and economics of such dual purpose facilities need to be explored.

4. Although a review of currently available experimental information indicates no reason to believe that current design methods are inadequate for the prediction of plutonium burner core parameters, a substantive commitment to validate current design methods for heavily loaded mixed-oxide systems must be made as early as possible in order to support the level of plutonium recycle employed in the plutonium burner concept. The development of a high level of confidence in the prediction of the following parameters with core burnup is an essential requirement: (1) Control rod worth, as well as the worth of other control systems, primarily soluble boron; (2) Doppler, moderator and power coefficients, and the prediction of the fuel temperature variation with core operating parameters; (3) Power peaking and distribution on both a 2D and 3D basis. Pilot demonstration programs and the commercial recycle of plutonium in existing reactors should be pursued to provide the required data base. Irradiation demonstration programs should be expanded to include fuel containing plutonium recovered from previously irradiated mixed-oxide assemblies. Such demonstrations would provide the experimental information necessary to evaluate design methods for plutonium isotopic compositions typical of continued recycle.
5. In addition, better characterization of mixed-oxide-fuel performance at the high plutonium loadings encountered in equilibrium cycle APR fuel is required. This includes confirmation of predicted temperature distributions within the fuel pellet, microstructure effects, and fuel-clad chemical interactions. Additional experimental work is required to quantify APR fuel performance under anticipated transient and accident conditions. Mixed-oxide-fuel demonstration programs in power reactors will supply fuel performance data for mixed-oxide assemblies operating at normal power densities; however, additional experimental information will be required to validate fuel performance models at high power densities and under transient conditions.
6. As noted in Section 2.1, there appears to be a significant economic incentive for the use of low effective density fuel, annular pellets or fuel containing a diluent such as cerium oxide. Further investigation of these alternatives should be pursued in order to firmly establish their feasibility and economics in PWR plutonium burners.
7. The use of thorium fuel enriched with plutonium appears to be an attractive alternative to the uranium-plutonium fuel cycle for plutonium burners located at the NEC. The use of the thorium fuel cycle reduces the number of plutonium burners required to utilize the plutonium produced by a given installed nuclear capacity by about one-third; this could be important in reducing the size and environmental impact of the NEC. The U-233 produced in the thorium cycle could be diluted with natural or depleted uranium and employed at dispersed sites; a 15-20% increase in the energy produced per unit plutonium consumed is realized with this application. Alternately, the U-233 could be used to support LWBR deployment; in this application, the plutonium burners located at NECs serve as the pre-breeder for the LWBR.

The use of the thorium fuel cycle in NEC plutonium burners should be further evaluated. These studies should include the development of material flows and deployment scenarios, evaluations of operational safety and licensability aspects of thorium fueling, economic assessments, and consideration of modifications to existing PWR designs to enhance U-233 production (cf. Section 6.4.2).

8. The feasibility, economics, and potential for uranium ore conservation of alternate types of plutonium burners which employ basic PWR technology should be evaluated. These alternates include the high conversion ratio plutonium burner proposed by Edlund (cf. Section 6.4) and plants employing the spectrum shift concept.

3.0 TECHNICAL CHARACTERISTICS OF PLUTONIUM-BURNING PWRs

3.1 Overview

PWR nuclear steam supply systems have in the past been designed to operate with mixed-oxide fuel loadings at or slightly in excess of that which is found in the self-generated mode of recycle (SGR). In this mode of operation, approximately one-third of the fuel loading is comprised of mixed-oxide fuel assemblies. Since roughly two-thirds of the fuel in this mode of operation is slightly enriched uranium, the reactor system performance characteristics are quite similar to those of a normal uranium-fueled reactor.

As the loading of fissile plutonium is increased beyond the SGR level, the important nuclear characteristics of the reactor core, including those which affect the safety analyses, change further toward those characteristic of mixed-oxide fuel alone. Nuclear characteristics of the all-mixed-oxide-fueled reactor (APR), while not radically different from the uranium-fueled reactor, are sufficiently different to cause changes in a number of system requirements.

The analyses performed by Combustion Engineering to date on the System 80TM PWR design, have demonstrated that mixed-oxide fuel loadings, up to and including the APR, are feasible in a suitably designed PWR system. The actual capability of other PWR systems to accept mixed-oxide-fuel loadings depends strongly on the design of the particular reactor and the inherent margins to design constraints and safety limitations of the overall NSSS design. The diversity of current PWR designs and the detailed and exacting analyses required to identify design margins preclude a generic assessment of the capability of current PWR designs to accommodate large loadings of mixed-oxide fuel to be undertaken within the scope of the current study. Consequently, design analyses performed for the System 80TM plant have largely formed the basis for the material presented in this section. The design analyses performed for System 80TM, however, can be employed to establish the characteristics and design criteria which must be met by any PWR operating as a plutonium burner.

Plants of the System 80TM type, although designed primarily for dispersed siting and optimized for lower levels of plutonium utilization than envisaged for nuclear energy center based plutonium burners, nevertheless appear quite suitable for this application. As discussed in Sections 3.2.2 and 3.2.3, the nuclear characteristics of these plants, when operating with a full core loading of mixed-oxide fuel, are not markedly different from those found in SGR operation. An examination of plant safety characteristics (cf. Section 3.3) indicates that in some respects the system responses under postulated accident conditions are more favorable for APR operation than for SGR. This is particularly true in the case of the control rod ejection accident. Moreover, the presently advanced state of design of PWR systems of this type makes them amenable to early introduction. The capability of plants of this type to accommodate various levels of plutonium utilization would also provide the option of returning to uranium-burning in later years, should accelerated LMFBR deployment shift the demand for plutonium fuel away from light-water reactors.

3.2 Nuclear Characteristics of Existing Designs

3.2.1 PWR Reference Design

The reference NSSS design chosen for this study is the Combustion Engineering System 80TM pressurized water reactor. The reasons for choosing this particular plant design as a basis are two-fold. First, the System 80TM PWR was designed for maximum fuel management flexibility and can accommodate mixed oxide fuel loadings up to and including an all mixed oxide core. Secondly, the broad base of experience gained in developing this plant design provided a valuable starting point with regard to the performance characteristics one might expect from PWR plutonium burners optimized for nuclear energy park-siting.

The pertinent design characteristics of the System 80TM PWR are summarized in Table 3-1 and in Figures 3-1 and 3-2. The System 80TM PWR employs a 241 assembly core, each assembly having a 16 x 16 fuel pin array with five guide tubes. The four outer guide tubes are for control element fingers, the center tube for in-core instrumentation. The in-core instruments are bottom inserted. This, along with a specially designed upper guide structure, makes it possible to rod and instrument each fuel assembly without restriction. The control element assemblies (CEAs) may have either 4, 8 or 12 "fingers". This flexibility in control rod strength makes it possible to provide the required reactivity shutdown margin when operating with significant loadings of mixed-oxide fuel.

The basic PWR is supplied with a complement of 89 CEAs which provide sufficient rod worth for plutonium utilization at the self-generated recycle (SGR) level. The basic NSSS hardware design incorporates all features necessary to accommodate a full mixed oxide core loading. An option package, comprising additional control element assemblies and two additional shutdown tanks to increase boron concentration during refueling, is available to permit operation with all mixed oxide fuel.

A more complete discussion of the basic plant design and engineered safety features will be found in the Combustion Engineering standard safety analysis report for the System 80TM NSSS (CESSAR), Docket Number STN-50-470.

3.2.2 Fuel Management

The recycle of plutonium in pressurized water reactors has been studied most extensively in the self-generated recycle (SGR) mode, in a manner appropriate to an isolated nuclear power plant. In self-generated recycle, the plutonium feed is limited to plutonium produced by the reactor itself in previous operating cycles. PWR plutonium burners sited at nuclear energy centers would employ plutonium feed generated both from their own operation and from previous operating cycles of uranium-burning plants located either on or off site. In order to limit the number of plutonium burners required at a nuclear energy center, the plutonium burners will more than likely be operated with all-mixed-oxide fuel. This section provides an assessment of the fuel management aspects of all-mixed-oxide operation.

Table 3-2 presents a comparison of the basic fuel management characteristics of equilibrium cycle operation for all uranium and SGR systems with those of the first, second, and equilibrium cycles of an all plutonium fueled reactor (APR). Scanning this information, one notes that the core average plutonium inventory for first cycle APR is not very markedly different from equilibrium cycle SGR, but increases substantially in later cycles. This arises primarily because the isotopic mix of the plutonium feed changes with time.

It has been assumed, for the purpose of this study, that a plutonium burner will be initially fueled with plutonium discharged from uranium burning PWR's similar to the reference design and, further, that the plutonium discharged from a given plutonium burner will be recycled in that plant. Under these assumptions, the plutonium isotopics evolve as shown in Figure 3-3. The fraction of fertile isotopes increases with time, and a greater mass of plutonium is required to meet the fissile feed requirements. This evolution of plutonium isotopics has an effect on the fissile requirements due to increased capture in the fertile materials and has a deleterious effect on control rod effectiveness, as will be discussed in Section 3.2.3.

The buildup of fertile isotopes also has a deleterious effect on plutonium replacement value. It is estimated that the replacement value of the plutonium relative to fully enriched uranium decreases from about 0.8 in the first years of plant operation to nearly 0.6 in the equilibrium cycle.

The question raised by this result concerns the possible merits of segregating batches of discharged plutonium according to the number of irradiation cycles to which this material has been subjected and providing for the eventual retirement of the plutonium after a certain number of irradiation cycles. Such a procedure might be contemplated for an energy park when the park size is such that a sufficient number of similar PWR plutonium burners are operating in parallel to yield economic batch sizes for segregated reprocessing and fabrication runs. The advantage of this segregation lies largely in the ability to retire plutonium which no longer retains significant value in thermal reactors. This approach was not considered in the present study, but is worthy of consideration as a means for optimizing plutonium utilization in energy parks containing a mix of both thermal and fast reactors.

Figures 3-4 through 3-19 present fuel management patterns and expected power distributions for the fuel cycles listed in Table 3-2. It will be noted that the achievable global power distributions and local power peaking do not differ substantively as one changes over from all uranium to all plutonium fueled operation.

A special fuel management problem arises in SGR operation due to the considerable difference in fission cross section between the mixed oxide and UO_2 assemblies. Figure 3-10 depicts the mixed oxide assembly enrichment zoning scheme used to control this problem in the SGR core examined for this study. A low enrichment zone at the assembly periphery is required to control local power peaking in the mixed oxide assemblies located adjacent to uranium assemblies. This requirement arises due to the strong flux gradients present in the flux transition region between the UO_2 and mixed oxide assemblies. While additional mixed oxide enrichments could be used to reduce power peaking further, two enrichments have been found adequate for self-generated recycle in the reference design. It is estimated that each additional mixed oxide enrichment increases the fabrication cost of the mixed oxide assemblies by roughly 10%, depending on the lot size. This provides a significant economic incentive to minimize the number of enrichments employed.

Local enrichment zoning of the mixed oxide assemblies used in all mixed oxide cycles appears to be unnecessary. In fact, finding acceptable fuel management patterns for these cores appears to be easier than for a UO_2 core due to the somewhat different reactivity depletion characteristics of the mixed oxide fuels. This effect is illustrated in Figure 3-20. The lower initial reactivity requirement in the fresh mixed oxide fuel relative to the residual reactivity in the burned fuel leads to more easily managed flux gradients between the fresh and burned fuel assemblies.

For the APR second cycle core, there are indications that matching the nominal second cycle length of 10,000 MWD/TeM achieved in UO_2 cores may be difficult. The cycle length of 8,325 MWD/TeM achieved in the second cycle design shown in Figures 3-14 through 3-16, was the longest second cycle length achieved in three design attempts. Each of these began with a first cycle 13,200 MWD/TeM in length.

One factor affecting the achievable second cycle length in UO_2 cores is the burnable poison residual carried over from the first cycle. For a given first cycle length, higher initial enrichments are required in order to compensate for the shim residual worth. These higher enrichments result in increased reactivity being carried over into the second cycle. It is estimated that approximately 400 MWD/TeM of the 10,000 MWD/TeM UO_2 core second cycle arises from this "hidden" reactivity carried over from the first cycle which returns as the shim residual completely burns out during the second cycle. This higher residual reactivity in the burned fuel also permits a higher feed enrichment to be employed in the second cycle for a given power peaking constraint; this is estimated to contribute an additional 300 Mwd/TeM to the second cycle length. The APR first cycle shown does not require (and hence does not use) burnable poisons for moderator temperature coefficient control. It, therefore, does not benefit from these effects.

A number of options, such as the use of burnable poisons in the feed batch, might be explored as a means to introduce a higher enrichment feed into the APR second cycle and thus lengthen it. Such investigations, however, lie outside the scope of the present study.

3.2.3 Safety Related Physics Parameters

Table 3-3 presents the evolution of a number of safety related core parameters with increasing core average plutonium loading. The evolution of Doppler and moderator reactivity coefficients with increasing plutonium loading is shown in Table 3-4.

A major effect of increased plutonium loadings is a decrease in the thermal neutron diffusion length which affects the soluble boron worth, control rod worth, and prompt neutron lifetime. This effect is illustrated graphically in Figure 3-21. The APR equilibrium cycle, which represents the highest expected plutonium loading, exhibits the shortest thermal neutron diffusion length and hence the highest soluble boron concentrations, lowest soluble boron worth, shortest prompt neutron lifetime, and lowest available control rod worth for a given rod pattern. These limiting properties are important in a number of design basis accidents, as will be discussed in section 3.3.

The downward trend in the effective delayed neutron fraction with increased plutonium loading which accompanies the reduction in prompt neutron lifetime is also worthy of notice. While this result would seem to portend a deleterious effect upon short period transients such as the rod ejection accident, the overall consequence is moderated by the lowered reactivity worth of the ejected rod and a decreased sensitivity of the power distribution in the blacker APR core to local reactivity perturbations.

The reduction in available control rod worth at higher plutonium loadings necessitates the use of additional control element assemblies to achieve adequate shutdown margin for the all mixed oxide core. For the PWR reference design, eight additional CEAs are employed to bring the net available rod worth in later APR cycles up to that normally required for UO₂ and SGR operation.

3.3 Safety Characteristics of All Mixed Oxide Cores

3.3.1 General Safety Consideration

As stated in Section 50.34 of 10CFR Part 50, each applicant requesting a construction permit or operating license for a nuclear power plant is required to provide an analysis and evaluation of the design and performance of the structures, systems, and components of the plant, with the objective of assessing the risk to public health and safety resulting from operation of the facility. These analyses demonstrate the safety margins during normal operations and anticipated transient conditions, and the adequacy of structures, systems and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The postulated events considered in this section are grouped, as in ANSI 18.2, into three classes, described as follows:

1. Condition II Occurrences:

These are events which are assumed to occur one or more times during the lifetime of the plant.

2. Condition III Occurrences (Infrequent Incidents):

These are events of small probability with potential for small radioactive release from the plant.

3. Condition IV Occurrences (Limiting Faults):

Postulated events of extremely low probability, which are analyzed to establish the performance requirements of engineered safety features.

Condition I occurrences, which are not listed, incorporate normal operational occurrences which are expected to occur during startup, shutdown, power operation and maneuvering. These events do not place unusual constraints on the design which are peculiar to the all-mixed oxide fueled core and hence are not reviewed here.

Table 3-5 lists the events under each class normally considered in the design of PWRs. It is required that Condition II occurrences such as loss of coolant flow result in a reactor shutdown with no radioactive release from the fuel, and with the plant capable of readily returning to power after corrective action is taken. For Condition III occurrences, such as a small LOCA, it must be shown that the engineered safety features and containment barriers function effectively to reduce to an insignificant level the potential for radioactive release to the environment. Consideration of the highly unlikely Condition IV occurrences, such as a major steam line break and large LOCA, provides design bases for the engineered safety features. It must be shown that the potential consequences of these events present an acceptably low risk to the public health and safety, as defined in 10CFR Part 100.

In order to obtain an operating license for any fuel cycle, including plutonium cycles, a detailed safety analysis has to be performed which demonstrates compliance with the requirements outlined in the previous paragraph. Such a detailed analysis lies outside of the scope of this study. This section, therefore, endeavors to identify the changes in plant characteristics due to the increased plutonium loadings which impact the safety and licensing characteristics of the plant. The changes in plant characteristics important to safety such as control rod worth, power coefficient, soluble boron worth, etc., which arise due to increased plutonium loadings are discussed. Implications of these changes on the various design basis accidents and transients are assessed on a qualitative basis first so as to identify those accidents and transients which are more limiting and which consequently bear a more detailed examination. In order to provide perspective, the APR characteristics and their safety implications are compared to those of self-generated recycle (SGR), which represents the level of plutonium recycle toward which most design studies have been directed in the past. Finally, selected results from analyses of the design basis accidents and transients which were identified in the qualitative assessment to have potentially more adverse consequences in the APR are presented.

3.3.2 Safety Implications of APR Nuclear Characteristics

Control Rod Worth and Control Requirements

Control requirements important to safety are those required to limit the effects of anticipated operating occurrences and accidents. These requirements are normally determined by the reactivity defects which are associated with moderator cooldown in the steam line break accident and which increase slightly in the APR. For a given CEA pattern, the reactivity worth of a CEA or a group of CEAs is significantly reduced for APR cycles compared to SGR or UO₂ operation. Under normal operating conditions, this reduction can be as much as 25-30% of the total CEA worth (cf. Section 3.2.3). For the PWR reference design (cf. Section 3.2.1), this reduction of CEA worth and increase in control requirements requires the use of additional CEAs for APR operation relative to the number required for SGR. The reduced CEA worth also has beneficial effects as it results in a decrease in the worst stuck CEA worth, dropped CEA worth, and ejected CEA worth. This serves to increase safety margins in a number of design basis transients and accidents.

Soluble Boron Worth

The soluble boron worth, which is an inverse function of the core absorption cross section, is smaller in APR cores than in SGR and UO_2 cores. The safety implications of this reduction are primarily related to the use of boric acid in the safety injection system, which provides an auxiliary reactivity control system in the event of major accidents, and in the refueling water tank which supplies the reactivity control required to maintain the reactor subcritical during refueling. The reduced boron worth necessitates maintenance of a higher concentration of boric acid which in turn affects the sizing of certain components (cf. Section 3.4) and complicates the boron precipitation problem following a LOCA. A detailed analysis of this latter effect is presented in Section 3.3.5.

Delayed Neutron Fraction and Prompt Neutron Lifetime

The effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*), which are important to the analysis of short term ($\sim$ seconds) power transients, are decreased in the APR core. While this result of itself would appear to have a deleterious effect upon short period transients such as the rod ejection accident, the overall consequence is moderated by the lowered reactivity worth of the ejected rod and a lowered sensitivity of the core power distribution to local reactivity perturbations.

Moderator Temperature Coefficient

For the equilibrium cycle of the PWR reference design (cf. Table 3-4), the MTC of the APR is calculated to be more negative at BOL and less negative at EOL than that of SGR, which means that the consequence of any incident limited by either the most positive MTC or the most negative MTC will be less adverse in the APR equilibrium core than in similar SGR cores. The early APR cycles exhibit calculated moderator temperature coefficients which are slightly more negative than the SGR equilibrium cycle. Consequently, incidents for which the most negative MTC is limiting such as the steam line break accident present more limiting circumstances in the earlier APR cycles. A detailed analysis of the steam line break accident has been performed for the APR and is discussed in Section 3.3.5.

3.3.3 Qualitative Evaluation of Condition II Occurrences

The consequences of Condition II occurrences for APR cores in comparison with SGR cores have been assessed on a qualitative basis taking account of the change in nuclear characteristics with increased plutonium loadings.

The withdrawal of CEAs from a critical condition adds positive reactivity to the reactor core, causing the core power level to increase. The severity of consequences of the incident is primarily determined by the amount of the positive reactivity insertion, the rate of reactivity addition, and local power peaking. Since these factors are reduced significantly in APR cores, the consequences of this incident are less severe. CEA misoperation includes a misaligned CEA, a stuck CEA, or a dropped CEA. The limiting cases for this incident are those which result in the maximum reactivity addition to the core and/or the highest local power peaking. As for the CEA withdrawal incident, the consequences of CEA misoperation will be less severe.

Uncontrolled boron dilution is defined as a decrease in reactor coolant system boron concentration due to the inadvertent addition of unborated water. Since the soluble boron worth is reduced substantially for APR cores, the reactivity addition rate to the core is reduced accordingly, making the incident less severe.

For incidents that result in an increase in the reactor coolant system temperature, thereby reducing the margin to DNB, the primary concern is the rate of temperature increase in the primary system. The type of fuel in the core does not materially influence the system responses. The changes in the reactivity coefficients with increased plutonium loading do not change the consequences of these incidents to any significant degree. Incidents of this kind include loss of flow, loss of AC power, loss of load/turbine trip, and loss of normal feedwater.

For those incidents which result in a decrease in coolant temperature, specifically idle loop startup and excess load, a positive reactivity addition occurs when the moderator temperature coefficient is negative. For the equilibrium cycle of the PWR reference design, the most negative MTC computed for the APR is $-3.1 \times 10^{-4}/^{\circ}\text{F}$ compared to $-3.7 \times 10^{-4}/^{\circ}\text{F}$ for SGR operation. Therefore, the consequences of these incidents will be less severe for the APR in equilibrium cycle operation. As discussed earlier, the moderator temperature coefficient in the early APR cycles is slightly more negative than in the SGR equilibrium cycle (for SGR operation the most negative MTC occurs at the end of equilibrium cycle). Hence, these cooldown incidents will present somewhat more severe consequences in the earlier APR cycles.

In brief, safety implications of APR cores for Condition II occurrences can be summarized as follows:

1. For those incidents caused by an inadvertent reactivity perturbation such as CEA withdrawal and boron dilution, the lowered CEA worth, power distribution sensitivity, and soluble boron worth in APR cores tend to make the consequences of these incidents less severe.
2. For those incidents which are characterized by a decreased margin to DNB caused by malfunctions in the primary or secondary coolant system, the plutonium content of the fuel does not affect the consequences of these incidents to any significant degree.
3. For those incidents that result in a decrease in coolant temperature, the limiting case (one with the most negative MTC) for equilibrium cycle operation is less severe in APR cores than in SGR cores. These incidents will present somewhat more limiting consequences in the earlier APR cycles.

3.3.4 Qualitative Evaluation of Condition III and Condition IV Occurrences

These events are postulated so that design bases for the engineered safety features and containment barriers can be formulated. For PWRs, the design basis accidents include (1) LOCA (loss of the reactor coolant resulting from postulated ruptures in the primary coolant system piping), (2) CEA Ejection (potential fuel damage resulting from withdrawal of control rods at maximum speed), (3) Steam Line Break (rupture in the reactor secondary or steam system piping), and (4) damage to reactor fuel during refueling operation such as dropping an irradiated fuel assembly. The analysis of each of these accidents is performed to determine whether there is assurance that adequate engineered safety features have been incorporated into the design to limit the radiological consequences of the accident.

In case of a loss of coolant accident (small LOCA or large LOCA), the consequences are not expected to be significantly different between APR and SGR cores. System responses are primarily determined by short term decay heat removal requirements in the reactor core which are not markedly different between APR and SGR cores. The potential boron precipitation problem in the long-term cooling phase, during which the borated water in the refueling water tank is recirculated through the core from the containment sump, is greater for the APR, since the boron concentration in the refueling water tank is higher in APR cores. A detailed analysis of this problem is presented in Section 3.3.5.

The steam line break accident is potentially more severe in the APR core largely because of the decrease in CEA worth. An analysis of this accident has been performed and is presented in Section 3.3.5.

A qualitative assessment of the CEA ejection accident for the APR presents some difficulties since lowered ejected CEA worth and local power peaking are advantageous while a smaller delayed neutron fraction, shorter prompt neutron lifetime, and a less negative Doppler coefficient are detrimental. A detailed analysis of this accident has been performed and is presented in Section 3.3.5. The results of this analysis indicate that this accident presents considerably less adverse consequences for APR operation compared to SGR.

The power mismatch and peaking factors resulting from the inadvertent loading of a fuel assembly in an improper position depends on the local fuel burnup and fuel management scheme. No general implications can be identified as a result of the increased plutonium loading in APR cores.

The consequences of a steam generator tube rupture are not substantially different for the APR than for an SGR or UO_2 fueled core. Radiological release as a result of the penetration of the barrier between the reactor coolant system and the main steam system will differ slightly due to the different fission product yields in plutonium and uranium fission.

The consequences of a fuel handling accident during refueling will not be substantively different for the APR than for SGR cores since in the analysis of the latter, it is conservatively assumed that all fuel damage occurs in the mixed oxide assemblies.

3.3.5 Detailed Analysis

Post LOCA Boric Acid Build-up

One of the problem areas identified in Section 3.3.4 is the build-up of boric acid in the core region during long-term emergency core cooling following a LOCA. As discussed in Section 3.3.2, the soluble boron worth is decreased due to the increased plutonium loading in APR cores. As a result, the boron concentration in the refueling water tank for the APR cores is increased to maintain soluble boron control capability.

For the PWR reference design, the boron concentration in the refueling water tank is 6200 ppm and 4400 ppm for APR and SGR operation, respectively. Long-term core cooling in the PWR reference design is accomplished by recirculating the water in the containment sump using high pressure safety injection pumps. The borated water in the containment sump comes primarily from the refueling water tank which is emptied for short term cooling. As the water boils off in the core, and the boron concentration increases, the long-term cooling may be impaired by precipitation of boric acid in the core. In order to avoid this, the ECCS in the PWR reference design is designed to be manually realigned to provide simultaneous hot-leg and cold-leg injection following a LOCA. The hot-leg injection provides a flushing flow which keeps the boron concentration at any point in the core below the precipitation value, which is calculated to be about 31 w/o at 228°F.

Boric acid build-up curves for the limiting case of a double-ended cold leg break for the equilibrium cycle, assuming no intervention are presented in Figure 3-22. As shown in the figure, boric acid precipitation begins at approximately five hours and three hours following the LOCA for the SGR and APR equilibrium cycles, respectively. For the PWR reference design, the boron buildup condition is terminated 90 minutes following the LOCA by initiating hot-leg injection.

Steam Line Break Accident

The major secondary system pipe break is the primary incident adversely affected by the more negative moderator temperature coefficient which characterizes mixed oxide fueled cores. A major pipe rupture in the main steam system reduces the steam generator pressure, increases the flow rate from the steam generators, and causes cooldown of the reactor coolant system. Cooldown of the reactor coolant system, coupled with a negative moderator temperature coefficient, results in a positive reactivity addition and may cause significant return-to-power and possible fuel damage after scram if the shutdown reactivity is not sufficient. With the higher moderator defect in the early APR cycles, the positive reactivity addition due to the cooldown of the reactor coolant system following rupture of a main steam line is therefore increased and a higher shutdown reactivity is required to meet NRC criteria.

An analysis of this accident was performed assuming that the transient was initiated by a doubled ended rupture of one steam line just outside of the containment. A simultaneous loss of AC power was also assumed, resulting in a simultaneous loss of load, loss of feedwater, and loss of primary system coolant flow. The results of this analysis indicate that the minimum required negative reactivity insertion to meet NRC criteria for the early APR cores is slightly larger than that required for the SGR. For the PWR reference design, sufficient shutdown margin is provided to assure that NRC criteria are met for steam line break transients in the SGR core. Since the reactivity worth of each CEA is reduced significantly in the APR core, additional CEAs are required to assure that a similar margin is provided for the early APR cycles. For the PWR reference design, (cf. Section 3.2.1) a CEA pattern including eight additional CEAs is available as part of the APR option package to provide the required shutdown margin for APR operation.

CEA Ejection Accident

The CEA ejection accident is defined as rapid ejection of a CEA from the core under critical conditions, resulting in a nearly instantaneous positive reactivity addition. In the CEA ejection accident, nuclear characteristics of the APR core such as lower ejected CEA worths and local power peaking are advantageous while the smaller values of β_{eff} and ℓ^* and less negative Doppler coefficient of reactivity are detrimental. A quantitative comparison has been performed for the CEA ejection accident in SGR and APR cores for the PWR reference design. This analysis has been performed at both hot full power (HFP) and hot zero power (HZP) conditions at end of equilibrium cycle. The first regulating CEA group was assumed to have been fully inserted at HFP and all regulating banks were assumed inserted for the HZP analysis. It was assumed also that the ejected CEA was the one having the highest reactivity worth for the postulated initial rod insertion and reactor operating state. Table 3-6 gives the ejected CEA worths and local power peaking factors used in the analyses.

The results of the HFP CEA ejection analyses are presented in Figure 3-23. For these cases, a high power trip setpoint of 129% of full power was used and was reached only in the SGR ejection. The control rods start entering the core 0.71 seconds following the trip and are 90% inserted at 3.71 seconds. Due to the lowered worth of the ejected rod in the APR core, the power transient in the APR rod ejection accident is self-limiting and does not reach the high power trip setpoint. The negative reactivity insertion due to increasing fuel temperature offsets the positive reactivity insertion from the ejected rod and the power level stabilizes at 103% of full power. For the APR case the power level increase during the transient is such that the plant would not be expected to experience significant fuel damage.

The results of the HZP CEA ejection analysis are shown in Figure 3-24. This accident does not result in a reactor trip since the fuel temperature reactivity feedback limits the core power level to well below normal operating levels. In the SGR core the power level stabilizes at 15% of nominal full power. The APR power level stabilizes at a slightly lower value of 11% of full power.

In summary, it appears that increasing the plutonium loading of the reactor core from SGR to APR levels results in a more favorable system response for the CEA ejection accident.

3.4 Design Modifications for Plutonium Burning

Design modifications to both the NSSS and balance of plant equipment are required to accommodate large loadings of mixed oxide fuel in pressurized water reactors. These modifications include additional control rods to provide required shutdown margin, equipment modifications to accommodate higher soluble boron concentrations, core and spent fuel cooling equipment sized to accommodate the higher decay heat loads associated with irradiated mixed oxide fuel, design of the reactor vessel and internals to tolerate a greater flux of high energy neutrons than arises in uranium fueled operation, modifications to the radwaste systems to accommodate higher tritium activity in the primary coolant, increased strength neutron sources, and design of fresh and spent fuel storage and fuel handling facilities to safely accommodate mixed oxide fuel.

A summary of the design modifications required to provide APR capability in a plant designed primarily for SGR or uranium fueled operation is provided in Table 3-7. Many of these modifications such as those which affect the design of the reactor internals or the provision of additional control rods, are not readily backfit. For example, redesign of the core shroud to improve heat removal would likely involve a change to the geometry of the girth ribs. In similar fashion, the addition of control rods could involve adding nozzles to the reactor vessel and redesign of the upper guide structure. One is therefore lead to the conclusion that PWRs employed as plutonium burners at nuclear energy centers must be specifically equipped for this application.

The considerations which give rise to the design modifications listed in Table 3-7 are as follows:

1. There is a six percent increase in the rate of high energy (>1 MeV) neutron irradiation of the reactor vessel and core internals with a full loading of mixed oxide fuel. This is due to both an increase in the number of prompt neutrons emitted in plutonium fission and a slightly higher average energy of the fission neutrons.
2. Gamma ray sources are approximately twenty percent higher in mixed oxide fueled operation. This leads to a corresponding increase in heating of the reactor internals.
3. The tritium concentration in the primary coolant during mixed oxide fueled operation will be substantially higher than in uranium fueled operation. The majority of the tritium in the reactor coolant is produced by the $B^{10}(n, 2\alpha)H^3$ reaction which has an energy threshold near 1 MeV. In mixed oxide fueled operation, the soluble boron concentration in the primary coolant is approximately twice that encountered in uranium fueled operation and the neutron flux above the threshold energy somewhat higher. This results in nearly a seventy percent increase in the concentration of tritium in the primary system, which is sufficient to influence both plant operational procedure and the design of both the liquid and gaseous radwaste systems.
4. The higher soluble boron concentration in the primary coolant required for mixed oxide fueled operation requires the chemical and volume control system (CVCS) and safety injection system (SIS) to be sized accordingly.
5. Control rod reactivity insertion requirements are slightly larger for APR operation than for SGR cycles. At the same time, the reactivity worth of each control rod is reduced significantly in the APR core, and consequently additional control rods are required.
6. Irradiated mixed oxide fuel exhibits higher long term decay heat generation rates and longer decay times than uranium fuel. This higher heat load must be accommodated by the auxiliary core cooling and spent fuel cooling systems. In addition, spent mixed oxide assemblies will require longer cooling times prior to reprocessing and specially designed shipping casks. Decay heat generation in the spent fuel is compared for all uranium, SGR, and APR operation in Figure 3-25.

7. Shielding is required for fresh fuel handling and inspection as a result of the neutron and gamma ray sources which arise in the fresh fuel due to plutonium radioactive decay.
8. In mixed oxide cores, the thermal neutron flux arising from a given sized fixed neutron source is expected to be approximately a factor of three less than in the corresponding uranium fueled core. This results in a reduction in effective source lifetime or, alternatively, a requirement for additional or stronger neutron sources.

The C-E System 80TM design was specifically developed to accommodate mixed oxide fuel loadings up to and including the APR. Consequently, design modifications required for APR operation have been incorporated into the basic NSSS hardware. The basic NSSS is supplied with a complement of eighty-nine control element assemblies (CEAs) which provide sufficient rod worth for plutonium utilization at the self-generated recycle (SGR) level. An option package, comprising eight additional CEAs and associated hardware and two additional shutdown tanks to provide increased soluble boron concentrations during refueling, is available to permit operation with all mixed oxide fuel.

TABLE 3-1

PWR REFERENCE DESIGN - GENERAL CHARACTERISTICS

Reactor Core

Number of Fuel Assemblies	241
Number of Control Element Assemblies (CEAs)	89
Number of Fuel Rods	56,876
Core Equivalent Diameter	143 in. (363.2 cm)
Active Fuel Length	150 in. (381.0 cm)
Total Core Heat Output	3800 MW
Average Linear Heat Rate	5.34 kw/ft (175.2 w/cm)
Primary System Pressure	2250 psia (15513.2 kPa)
Core Inlet Temperature	565°F (569.3°K)
Core Outlet Temperature	621°F (600.4°K)
Average Full Power Moderator Temperature	594°F (585.4°K)
Hot Standby Moderator Temperature	564°F (568.7°K)
Fuel Management	3 Batch, Mixed Central Zone
Average Cycle Burnup	11,400. MWD/TeM

Fuel Assembly

Fuel Rod Array, square	16 x 16
Assembly Outside Dimensions, fuel rod to fuel rod	7.977 in. (20.26 cm)
Spacing Between Fuel Assemblies, fuel rod to fuel rod	0.203 in. (0.516 cm)
Fuel Rod Pitch	0.5063 in. (1.286 cm)
Clad Material	Zircaloy-4
Spacer Grid Material	Zircaloy-4
CEA Guide Tube Material	Zircaloy-4
Number of CEA Guide Tubes	5
Pellet Theoretical Density	10.96 g/cm ³
Pellet Density (% Theoretical)	95.0
Stack Height Density	10.054 g/cm ³
H ₂ O/Oxide Volume Ratio	2.04

TABLE 3-2

PWR REFERENCE DESIGN - FUEL MANAGEMENT CHARACTERISTICS

	Equilibrium Cycle UO ₂	Equilibrium Cycle SGR	First Cycle APR	Second Cycle APR	Equilibrium Cycle APR
Cycle Length (MWD/TeM)	11,400	11,400	13,200	8,325	11,400
Average UO ₂ Feed Enrichment (w/o U-235)	3.29	3.62	--	--	--
Average Mixed Oxide Feed Enrichment (w/o Fissile Pu)	--	3.05	1.92	3.03	4.57
Number of UO ₂ Assemblies	241	157	0	0	0
Number of Mixed Oxide Assemblies	0	84	241	241	241
Core Plutonium Inventory (Total Pu)					
Beginning-of-Cycle	421.2 Kg	2228.1 Kg	2877.5 Kg	3511.5 Kg	8439.1 Kg
End-of-Cycle	740.0 Kg	2148.9 Kg	2591.0 Kg	3270.0 Kg	7824.4 Kg
Core Plutonium Inventory (Fissile Pu)					
Beginning-of-Cycle	336.9 Kg	1205.6 Kg	1978.1 Kg	2295.5 Kg	4279.9 Kg
End-of-Cycle	561.8 Kg	1233.7 Kg	1617.3 Kg	2038.5 Kg	3829.0 Kg

TABLE 3-3

PWR REFERENCE DESIGN - SAFETY RELATED PHYSICS CHARACTERISTICS

	Equilibrium Cycle UO ₂	Equilibrium Cycle SGR	First Cycle APR	Second Cycle APR	Equilibrium Cycle APR
Beginning of Cycle Reactivity (CEAs Withdrawn, No Dissolved Boron)					
Hot Standby	0.137	0.122	0.126	0.075	0.083
Full Power, No Xenon	0.121	0.103	0.108	0.053	0.064
Full Power, Equilibrium Xenon	0.101	0.081	0.089	0.038	0.055
Dissolved Boron Requirements					
PPM Dissolved Boron for Criticality - CEAs Withdrawn					
BOC Hot Standby	1589	1820	2749	1743	3189
BOC Full Power, No Xenon	1400	1539	2351	1227	2450
BOC Full Power, Equilibrium Xenon	1170	1208	1934	893	2100
Requirement for Refueling (5% Subcritical)	1955	2383	3153	2977	4203
Inverse Boron Worth (PPM/% $\Delta\rho$)					
Full Power BOC	116	149	218	233	383
Full Power EOC	101	130	150	190	331
Neutron Kinetics Parameters					
Prompt Neutron Lifetime (μsec)					
Beginning-of-Cycle	21.3	17.0	--	--	6.8
End-of-Cycle	24.8	19.5	--	--	7.9
Effective Delayed Neutron Fraction					
Beginning-of-Cycle	0.00625	0.00567	--	--	0.00442
End-of-Cycle	0.00546	0.00518	--	--	0.00447
Available Control Rod Worth					
Total (% $\Delta\rho$)	13.8 ^a	13.5 ^a	13.4 ^a	12.9 ^a	10.3 ^a (12.6) ^b
Net ^c (% $\Delta\rho$)	10.2 ^a	9.9 ^a	10.6 ^a	9.4 ^a	8.4 ^a (9.8) ^b

(a) 89 CEA's

(b) 97 CEAs

(c) Highest worth CEA held out.

TABLE 3-4

PWR REFERENCE DESIGN - FUEL AND MODERATOR TEMPERATURE COEFFICIENTS OF REACTIVITY

Fuel Cycle	Fuel Temperature Coefficient	Moderator Temperature Coefficient
Equilibrium Cycle UO_2 (BOC)	$-1.24 \times 10^{-5}/^\circ\text{F}$ ($-2.23 \times 10^{-5}/^\circ\text{K}$)	$-0.59 \times 10^{-4}/^\circ\text{F}$ ($-1.06 \times 10^{-4}/^\circ\text{K}$)
Equilibrium Cycle UO_2 (EOC)	$-1.25 \times 10^{-5}/^\circ\text{F}$ ($-2.25 \times 10^{-5}/^\circ\text{K}$)	$-3.24 \times 10^{-4}/^\circ\text{F}$ ($-5.83 \times 10^{-4}/^\circ\text{K}$)
Equilibrium Cycle SGR (BOC)	$-1.08 \times 10^{-5}/^\circ\text{F}$ ($-1.94 \times 10^{-5}/^\circ\text{K}$)	$-0.95 \times 10^{-4}/^\circ\text{F}$ ($-1.71 \times 10^{-4}/^\circ\text{K}$)
Equilibrium Cycle SGR (EOC)	$-1.17 \times 10^{-5}/^\circ\text{F}$ ($-2.11 \times 10^{-5}/^\circ\text{K}$)	$-3.73 \times 10^{-4}/^\circ\text{F}$ ($-6.71 \times 10^{-4}/^\circ\text{K}$)
First Cycle APR (BOC)	--	$-0.70 \times 10^{-4}/^\circ\text{F}$ ($-1.26 \times 10^{-4}/^\circ\text{K}$)
First Cycle APR (EOC)	--	$-3.80 \times 10^{-4}/^\circ\text{F}$ ($-6.84 \times 10^{-4}/^\circ\text{K}$)
Second Cycle APR (BOC)	--	$-2.00 \times 10^{-4}/^\circ\text{F}$ ($-3.60 \times 10^{-4}/^\circ\text{K}$)
Second Cycle APR (EOC)	--	$-4.10 \times 10^{-4}/^\circ\text{F}$ ($-7.38 \times 10^{-4}/^\circ\text{K}$)
Equilibrium Cycle APR (BOC)	$-1.01 \times 10^{-5}/^\circ\text{F}$ ($-1.82 \times 10^{-5}/^\circ\text{K}$)	$-1.00 \times 10^{-4}/^\circ\text{F}$ ($-1.80 \times 10^{-4}/^\circ\text{K}$)
Equilibrium Cycle APR (EOC)	$-1.09 \times 10^{-5}/^\circ\text{F}$ ($-1.96 \times 10^{-5}/^\circ\text{K}$)	$-3.10 \times 10^{-4}/^\circ\text{F}$ ($-5.58 \times 10^{-4}/^\circ\text{K}$)

TABLE 3-5

Events Analyzed for PWRs

Condition II Occurrences

1. CEA Withdrawal
2. CEA Misoperation
3. Uncontrolled Boron Dilution
4. Loss of Coolant Flow
5. Idle Loop Startup
6. Loss of Load or Turbine Trip
7. Loss of Normal Feedwater
8. Loss of AC Power
9. Excess Load

Condition III Occurrences

1. Small LOCA
2. Minor Steam Line Break
3. Inadvertent Loading of a Fuel Assembly in a Improper Position

Condition IV Occurrences

1. Large LOCA
2. Steam Generator Tube Rupture
3. Major Steam Line Break
4. CEA Ejection
5. Fuel Handling Accident

TABLE 3-6

Input Parameters
for CEA Ejection Analysis
End of Equilibrium Cycle

	SGR (89 CEAs)	APR (97 CEAs)
Hot Full Power Analysis		
Ejected Rod Worth	0.13% $\Delta\rho$	0.08% $\Delta\rho$
Radial Power Peaking Factor (prior to ejection)	1.46	1.40
Radial Power Peaking Factor (post ejection)	2.41	1.65
Regulating Bank Worth Initially Inserted	0.45% $\Delta\rho$	0.41% $\Delta\rho$
Hot Zero Power Analysis		
Ejected Rod Worth	0.53% $\Delta\rho$	0.43% $\Delta\rho$
Radial Peaking Factor (prior to ejection)	2.49	1.63
Radial Peaking Factor (post ejection)	9.11	4.81
Regulating Bank Worth Initially Inserted	3.5% $\Delta\rho$	3.8% $\Delta\rho$

TABLE 3-7

PWR Design Modifications for Plutonium Burning

System or Component	Altered Design Bases	Design Modification Options
Auxiliary Core Cooling System	Higher Decay Heat Removal Requirement	(1) Size cooling system components accordingly (2) Reduce plant power rating
Chemical and Volume Control System (CVCS)	Increased Maximum Soluble Boron Concentration	(1) Size CVCS components accordingly (2) Employ shorter refueling cycles to reduce reactivity control requirement (4) Restrict plant maneuvering
Control Rod Complement	Reduced Control Rod Worth; Increased Requirements	Add additional control rods and drives
Fresh Fuel Handling	Gamma and Neutron Source in Fresh Fuel Due to Plutonium	Provide required shielding
Fresh and Spent Fuel Storage Facilities	Criticality Safety Characteristics	(1) Redesign facility for high enrichment mixed oxide fuel (2) Store less fuel in existing facility and build additional capacity if required
Neutron Sources	Increased Thermal Neutron Capture	(1) Employ stronger sources (2) Employ additional sources (3) Replace sources more frequently
Radwaste System	Increased Tritium Production In Primary System	(1) Install tritium removal system (2) Employ shorter refueling cycles to reduce soluble boron requirement in primary system (3) Employ burnable poisons to reduce soluble boron requirement in primary system

TABLE 3-7 (Cont'd.)

PWR Design Modifications for Plutonium Burning

System or Component	Altered Design Bases	Design Modification Options
Reactor Internals	Increased Heating Due to Larger Gamma Ray Dose	<ol style="list-style-type: none"> (1) Redesign internals to reduce thermal stresses for given heat load (2) Redesign internals to improve cooling geometry
Reactor Vessel	Increased Fast Neutron Irradiation	<ol style="list-style-type: none"> (1) Increase thickness of core support barrel (2) Reduce plant power rating (3) Anneal vessel in situ and/or reduce plant service life (4) Increase size of vessel
Safety Injection System (SIS)	Increased Maximum Soluble Boron Concentration	<ol style="list-style-type: none"> (1) Size SIS components accordingly (2) Employ shorter refueling cycles to reduce reactivity control requirement
Spent Fuel Pool Cooling System	Higher Decay Heat Removal Requirement	<ol style="list-style-type: none"> (1) Size cooling system components accordingly (2) Reduce plant power rating

FIGURE 3-1
 REACTOR CORE CROSS SECTION
 AND
 STANDARD CONTROL ELEMENT ASSEMBLY PATTERN (89 CEAs)

FIGURE 3-2
CONTROL ELEMENT ASSEMBLY CONSTRUCTION
AND
UPPER GUIDE STRUCTURE DESIGN

ISOTOPIC COMPOSITION OF PLUTONIUM FEED VS. TIME
ALL MIXED OXIDE FUELED REACTOR

FIGURE 3-3

16

Fuel Batch

0.00	Box Peaking Factor
0.00	Pin Peaking Factor

Maxima
 Box Peaking Factor: 1.18
 Pin Peaking Factor: 1.26

K - Fresh Uranium Fuel
 J - Uranium Fuel Irradiated One Cycle
 I - Uranium Fuel Irradiated Two Cycles
 H - Uranium Fuel Irradiated Three Cycles

						K	K	K	K
						0.67	0.83	0.88	0.87
						1.03	1.10	1.12	1.11
					K	K	I	K	I
					0.68	0.93	1.10	0.90	1.18
					1.06	1.17	1.22	0.97	1.26
				K	K	I	J	I	J
				0.75	1.07	0.96	1.07	0.96	1.05
				1.10	1.19	1.02	1.12	0.99	1.08
			K	K	I	J	I	K	I
					0.87	1.07	1.01	1.18	0.97
					0.92	1.11	1.07	1.25	1.00
			K	I	J	J	J	I	J
						1.14	1.12	0.96	1.16
						1.21	1.20	0.98	1.25
		K	K	J	I	J	I	J	J
						1.00	1.16	0.99	1.13
						1.06	1.22	1.06	1.21
		K	I	I	K	I	J	J	I
							1.06	1.14	0.99
							1.08	1.21	1.04
		K	K	J	I	J	I	J	J
								1.00	1.13
								1.07	1.18
		K	I	I	J	J	J	I	J
									H
									0.91
									0.93

FIGURE 3-6

UNRODDED PLANAR POWER DISTRIBUTION
 UO₂ EQUILIBRIUM CYCLE - EOC 11,400 MWD/T_{em}

0.00 —
0.00 —

0.00 — Box Peaking Factor

0.00 — Pin Peaking Factor

Maxima

Bcx Peaking Factor: 1.24

Pin Peaking Factor: 1.41

K - Fresh Uranium Fuel

K*-Fresh Mixed Oxide Fuel

J/J*-Fuel Irradiated One Cycle

I/I*-Fuel Irradiated Two Cycles

H - Uranium Fuel Irradiated
Three Cycles

FIGURE 3-7

UNRODDED PLANAR POWER DISTRIBUTION SGR EQUILIBRIUM CYCLE - BOC NO XENON

3-28

ing Factor	ing Factor				K	K*	K	K
					0.71	0.80	0.96	0.88
1.24	1.41				1.81	1.20	1.31	1.13
el	Cycle	Cycles	ated		K*	K*	K	J*
					0.61	0.90	1.22	1.11
1.41					1.15	1.30	1.41	1.36
el	Cycle	Cycles	ated		K*	K	J	I*
					0.73	1.18	1.12	0.99
1.41					1.32	1.39	1.24	1.14
el	Cycle	Cycles	ated		K*	K	I*	I
					1.02	0.95	0.99	1.23
1.41					1.18	1.08	1.17	1.38
el	Cycle	Cycles	ated		K*	J	I	J*
					1.09	1.05	1.21	1.08
1.41					1.28	1.23	1.41	1.29
el	Cycle	Cycles	ated		K	K	I*	J
					0.99	1.15	1.21	1.07
1.41					1.08	1.35	1.34	1.15
el	Cycle	Cycles	ated		K*	J*	I	K
					0.79	0.99	0.98	
1.41					0.83	1.19	1.12	1.12
el	Cycle	Cycles	ated		K	J	I*	I
					1.17	1.05	1.19	1.19
1.41					1.32	1.19	1.19	1.19
el	Cycle	Cycles	ated		K	I	K	I
					0.89	1.05	1.05	1.05
1.41					1.05	1.05	1.05	1.05

FIGURE 3-10

MIXED OXIDE ASSEMBLY ENRICHMENT ZONING
SGR EQUILIBRIUM CYCLE

Fuel Batch

0.00 — Box Peaking Factor
0.00 — Pin Peaking Factor

Maxima

Box Peaking Factor: 1.17

Pin Peaking Factor: 1.34

A - Fresh Mixed Oxide Fuel

B - Fresh Mixed Oxide Fuel

C - Fresh Mixed Oxide Fuel

						C	C	C	C				
						0.58 0.91	0.73 1.01	0.77 1.08	0.77 1.03				
						C	C	A	C				
						0.57 0.93	0.83 1.14	1.03 1.25	1.14 1.33	0.93 1.03	1.17 1.34		
						C	C	B	B	A	B	A	
						0.64 1.00	0.98 1.26	1.02 1.14	1.10 1.18	1.02 1.08	1.13 1.19	1.03 1.08	
						C	C	A	B	A	B	A	
								0.95 1.04	1.12 1.18	1.03 1.08	1.15 1.19	1.04 1.09	1.15 1.19
						C	B	B	A	B	A	B	A
									1.03 1.08	1.15 1.19	1.04 1.09	1.15 1.19	1.05 1.09
						C	C	B	A	B	A	B	A
										1.05 1.09	1.15 1.19	1.05 1.09	1.15 1.19
						C	C	A	B	A	B	A	B
											1.04 1.09	1.15 1.19	1.04 1.09
						C	A	B	A	B	A	B	A
												1.04 1.09	1.15 1.19
						C	C	A	B	A	B	A	B
													1.04 1.09

FIGURE 3-13

UNRODDED PLANAR POWER DISTRIBUTION
APR FIRST CYCLE - EOC 13,200 MWD/TeM

0.00—
0.00—

0.00 — Box Peaking Factor

0.00 — Pin Peaking Factor

Maxima

Box Peaking Factor: 1.27

Pin Peaking Factor: 1.41

D - Fresh Mixed Oxide Fuel

C - Mixed Oxide Fuel Irradiated
One Cycle

B - Mixed Oxide Fuel Irradiated
One Cycle

A - Mixed Oxide Fuel Irradiated
One Cycle

ing Factor ing Factor					D 0.62 1.05	D 0.77 1.10	D 0.80 1.15	D 0.83 1.19		
					D 0.64 1.07	D 0.91 1.27	D 1.14 1.40	C 0.97 1.12	B 0.79 0.87	D 1.21 1.37
1.27 1.41			D 0.75 1.20	D 1.13 1.41	B 0.85 0.93	B 0.88 0.96	C 1.08 1.23	B 0.87 0.94	B 0.85 0.92	
rel adiated			D	D	B 0.89 0.99	C 1.13 1.24	C 1.18 1.32	B 0.93 1.03	C 1.15 1.30	C 1.13 1.25
adiated			D	B	C	C 1.21 1.32	B 1.02 1.11	C 1.24 1.37	B 0.99 1.08	C 1.17 1.24
adiated	D	D	B	C	B	C 1.26 1.39	B 1.03 1.11	D 1.23 1.38	B 1.00 1.07	
	D	C	C	B	C	B	D 1.26 1.40	B 1.02 1.11	C 1.19 1.31	
	D	B	B	C	B	D	B	C 1.22 1.34	C 1.27 1.38	
	D	D	B	C	C	B	C	C	A 0.96 1.03	

FIGURE 3-14

UNRODDED PLANAR POWER DISTRIBUTION
APR SECOND CYCLE - BOC NO XENON

0.00—
0.00—

0.00—Box Peaking Factor

0.00 — Pin Peaking Factor

Maxima

Box Peaking Factor: 1.27

Pin Peaking Factor: 1.38.

D - Fresh Mixed Oxide Fuel

C - Mixed Oxide Fuel Irradiated
One Cycle

B - Mixed Oxide Fuel Irradiated
One Cycle

A - Mixed Oxide Fuel Irradiated
One Cycle

ing Factor ing Factor					D 0.61 0.99	D 0.75 1.06	D 0.80 1.13	D 0.82 1.13	
			D 0.61 1.00	D 0.88 1.22	D 1.09 1.33	C 0.98 1.14	B 0.83 0.91	D 1.19 1.38	
1.27 1.38			D 0.70 1.10	D 1.07 1.35	B 0.86 0.95	B 0.91 0.99	C 1.12 1.25	B 0.92 0.99	B 0.91 0.97
el adiated	D		D	B 0.89 0.98	C 1.13 1.22	C 1.21 1.32	B 0.98 1.06	C 1.19 1.31	C 1.18 1.27
adiated	D		B	C	C 1.22 1.31	B 1.05 1.12	C 1.26 1.35	B 1.01 1.08	C 1.17 1.23
adiated	D	D	B	C	B	C 1.27 1.37	B 1.03 1.08	D 1.17 1.27	B 1.00 1.05
	D	C	C	B	C	B	D 1.18 1.28	B 1.01 1.07	C 1.16 1.25
	D	B	B	C	B	D	B	C 1.18 1.27	C 1.23 1.33
	D	D	B	C	C	B	C	C	A 0.96 1.02

FIGURE 3-15

UNRODDED PLANAR POWER DISTRIBUTION
APR SECOND CYCLE - MOC 4000 MWD/TeM

APR SECOND CYCLE - MOC 4000 MWD/TEM

0.00 —
0.00 —

0.00 — Pin Peaking Factor

Pin Peaking Factor: 1.44

H - Mixed Oxide Fuel Irradiated
Three Cycles

					K	K	K	K
					0.56 0.97	0.68 1.02	0.71 1.04	0.72 1.05
			K	K	K	K	I	K
1.23			0.52	0.79	1.04	1.15	0.95	1.11
1.44			0.91	1.15	1.35	1.41	1.11	1.36
		K	K	J	I	J	J	I
el		0.59	0.95	1.02	1.01	1.16	1.20	1.07
adiated		0.98	1.24	1.19	1.13	1.35	1.44	1.20
	K	K	I	I	J	I	J	I
adiated			0.90	1.01	1.14	1.11	1.23	1.13
			0.99	1.14	1.35	1.25	1.42	1.25
adiated	K	J	I	K	I	J	I	J
				1.15	1.02	1.16	1.11	1.23
				1.39	1.13	1.35	1.21	1.41
K	K	I	J	I	I	J	J	I
					0.94	1.13	1.15	1.08
					1.02	1.30	1.31	1.19
K	K	J	I	J	J	I	I	J
						0.99	1.08	1.16
						1.09	1.22	1.33
K	I	J	J	I	J	I	J	J
							1.06	1.12
							1.19	1.28
K	K	I	I	J	I	J	J	H
								0.93
								1.01

UNRODDED PLANAR POWER DISTRIBUTION
APR EQUILIBRIUM CYCLE - BOC NO XENON

Fuel Batch

0.00	Box Peaking Factor
0.00	Pin Peaking Factor

Maxima

Box Peaking Factor: 1.22

Pin Peaking Factor: 1.34

K - Fresh Mixed Oxide Fuel

J - Mixed Oxide Fuel Irradiated
One Cycle

I - Mixed Oxide Fuel Irradiated
Two Cycles

H - Mixed Oxide Fuel Irradiated
Three Cycles

						K	K	K	K
						0.56	0.67	0.68	0.67
						0.90	0.93	0.94	0.93
				K	K	K	K	I	K
				0.59	0.82	0.97	1.03	0.88	1.00
				0.93	1.11	1.19	1.22	1.01	1.18
		K	K	J	I	J	J	I	
		0.63	0.97	1.05	0.99	1.09	1.11	1.02	
		0.97	1.21	1.17	1.08	1.25	1.30	1.12	
	K	K	I	I	J	I	J	I	
			0.96	1.05	1.14	1.11	1.20	1.11	
			1.03	1.14	1.30	1.20	1.32	1.20	
	K	J	I	K	I	J	I	J	
				1.17	1.06	1.17	1.14	1.22	
				1.33	1.15	1.33	1.21	1.34	
	K	K	I	J	I	I	J	J	I
						1.02	1.18	1.19	1.11
						1.08	1.30	1.30	1.19
	K	K	J	I	J	J	I	I	J
							1.07	1.13	1.18
							1.14	1.23	1.31
	K	I	J	J	I	J	I	J	J
								1.12	1.15
								1.22	1.27
	K	K	I	I	J	I	J	J	H
									0.99
									1.05

FIGURE 3-19

UNRODDED PLANAR POWER DISTRIBUTION
APR EQUILIBRIUM CYCLE - EOC 11,400 MWD/T_{em}

RELATIVE REACTIVITY DEPLETION RATES IN
URANIUM OXIDE AND MIXED OXIDE FUELS

FIGURE 3-20

FIGURE 3-21

THERMAL DIFFUSION LENGTH VS. PLUTONIUM LOADING
PWR REFERENCE DESIGN

FIGURE 3-22
POST-LOCA BORIC ACID BUILD-UP
FOR APR AND SGR

FIGURE 3-23

HOT AND AVERAGE CHANNEL POWER
EXCURSION AFTER CEA EJECTION, HFP

FIGURE 3-24

HOT AND AVERAGE CHANNEL POWER
EXCURSION AFTER CEA EJECTION HZP

FIGURE 3-25
DECAY HEAT GENERATION IN SPENT FUEL
EQUILIBRIUM CYCLES

4.0 PLUTONIUM BURNER DESIGN OPTIMIZATION

4.1 Overview

Current PWR designs, although possibly provided with the capability for a full mixed oxide core loading, are generally optimized for UO_2 and SGR operation, since optimization for higher plutonium loadings typically results in poor economy and operational difficulties with uranium fueling. At a nuclear energy center (NEC), specific plants could be dedicated to plutonium fueling and optimization of the reactor lattice to improve plutonium utilization might be desirable. The optimal design depends, of course, on the specific objectives of the NEC; for example, designs with increased water-to-fuel ratios significantly reduce the fuel inventory demand for a given electrical capacity, while the different objective of reduced plutonium consumption per unit of energy output requires a core design with a decreased water-to-fuel ratio.

Sections 4.2 through 4.4 discuss the results of a conceptual design study in which survey analyses were conducted varying the reactor lattice water-to-fuel ratio in an effort to determine optimum lattice designs to achieve various NEC objectives. In any optimization study, a relatively large number of alternative designs must be evaluated to clearly identify trends and consideration must be given to the impact of changes in lattice parameters on NSSS and containment size, thermal margins, pumping requirements, secondary system design, and balance of plant design. In order to be able to explore a suitable range of conceptual designs, a simplified method based upon the use of a point reactor model to predict core nuclear performance was employed in these analyses.

The results of the APR lattice optimization study indicate that as the water-to-fuel ratio is increased, the required core fissile inventory and fuel cycle costs decrease. An increase in the lattice water-to-fuel ratio may be achieved by the following means: 1) increasing the lattice pitch; 2) reducing fuel pellet density or diameter; 3) replacement of solid fuel pellets with annular pellets; 4) replacement of a portion of the uranium oxide by a low cross section diluent material. Of the four alternatives, replacing a portion of the uranium oxide by a diluent material and the use of annular pellets appear attractive. The desired increase in lattice water-to-fuel ratio is realized without requiring modifications to the reactor lattice or to the overall plant design and without a degradation of fuel thermal performance.

The number of plutonium burners at a given nuclear energy center must match the available plutonium supply. Nuclear energy centers, as presently envisaged, will comprise a mix of uranium burners (located either at the NEC or at dispersed sites) and plutonium burning reactors, fuel reprocessing facilities, and mixed-oxide fuel fabrication facilities. Since the quantity of plutonium required to fuel an initial mixed-oxide core is substantially larger than the annual makeup requirement, the number of plutonium burners introduced at a given site will gradually increase with time until an equilibrium level is reached at approximately one plutonium burner for every two or three uranium-fueled reactors. In later years, this ratio might be altered by requirements to set aside plutonium for LMFBR deployment and, depending on LMFBR requirements, some of the plutonium burners might be converted to the uranium cycle or phased out entirely. Other factors influencing the use of plutonium burners at nuclear energy centers include the possibility of delays in government

approval of mixed-oxide fuel utilization in light water reactors sited at dispersed locations and the possible limitation of burning plutonium produced off-site only in plutonium burners located within the confines of nuclear energy centers.

These considerations are treated in Section 4.5 and their effect on the deployment of LWR plutonium burners at nuclear energy centers examined. Special operational considerations arising from the close proximity of reprocessing, fuel fabrication, and nuclear power plant facilities at nuclear energy centers are discussed in Section 4.6.

4.2 Fuel Cycle Cost Considerations

In Section 4.4, fuel cycle and power cost analyses are discussed in order to establish the optimum lattice configuration from the standpoint of economic considerations. In performing these analyses, it is necessary to estimate the costs of materials and fuel cycle services in the 1990s and beyond, the period when nuclear energy centers will be in operation. The methodology and basis for estimating these costs are described in this section and in Section 4.3. It is recognized, of course, that such cost projections are highly speculative, but the bases described in Section 4.2 and 4.3 at least provide a reasonable starting point for the economic evaluations. However, in view of the uncertainty in these projections, the cost evaluations of Section 4.4 have been performed parametric in the important cost parameters. While fuel cycle and power costs depend directly on the economic assumptions, the parameter of interest, the optimum lattice configuration, proved to be relatively insensitive to reasonable changes in the economic assumptions.

4.2.1 U₃O₈ Prices

U₃O₈ prices, after having remained relatively stable within the \$7 to \$12 per lb. price range between 1948 and 1973, have recently increased sharply. Since mid-1973, prices have been increasing 40 to 45% every six months such that the spot price for uranium is currently about \$30/lb U₃O₈, and uranium contracted today for delivery in the 1980s costs more than \$40/lb U₃O₈.

The recent price increases are caused in part by the depressed price of uranium in previous years, the large projected national uranium demand which cannot be met with present production facilities, and price increases in other forms of energy. In view of the recent instability in uranium price, straightforward projections into the future are difficult and uncertain.

Although ore prices are likely to fluctuate over the next several years, reflecting perceptions of the supply and demand situation, in the longer term uranium ore prices should be determined by the full costs of recovering uranium, plus a reasonable profit for this service.

Cost Plus Reasonable Profit Price Forecast

U₃O₈ reserves are normally quoted in categories of "forward costs". The forward cost concept allows for direct comparison of various ore bodies which have different concentrations of U₃O₈, lie at various depths in differently shaped veins and in different host materials. However, forward costs are only those costs necessary to recover U₃O₈ and do not include "sunken costs", such as property acquisition, exploration, mine development, cost of money, return on investment and taxes.

Although there are many divergent estimates of what the latter add to the price, a factor of 1.8 is often used to account for "sunken costs". In addition, "forward costs" need to be escalated to the appropriate supply date in order to get contemporary dollars in any given year.

To determine how "forward costs" will vary with time, it is necessary to compare U_3O_8 requirements as a function of time with resources available in the various "forward" cost categories.

Reserve requirements are based on the fact that a mine/mill complex is built on a reserve that is large enough to support, for many years, the facility's production. A common rule-of-thumb is that the initial reserve must at least equal cumulative production for the next eight years. Necessary additions to reserves are made by exploration and discovery of low cost potential resources or by considering higher cost reserves. The price of uranium, in turn, is directly related to the "forward cost" of the lowest grade (highest cost) ore body which has to be included to satisfy the reserve requirement of the production facility.

The eight-year production base requirement is shown in Figure 4-1, along with projections of variously priced reserves for the U.S. In this figure, resources are assumed to be converted into reserves at 8%/year.

The following is an example of how prices are determined. In 1993, the eight-year forward reserve requirement is 1.55 million tons of yellowcake. To achieve that much reserve in 1993, all \$15 per pound and cheaper reserves have to be in some stage of production. Thus, the "forward cost" of produced uranium would be at least \$15 per pound and the selling price, including reasonable profit, is 1.8 times or \$27 per pound. The 1975 price of \$27 per pound is then escalated for eighteen years at 6% per year to give a final value of \$77 per pound (in 1993 dollars) for U_3O_8 produced and delivered in 1993. Other prices forecast by this method are shown on the insert of Figure 4-1.

Since a majority of the plutonium burners will not be deployed until the mid to late 1990s, a U_3O_8 price of \$40/lb (1976 dollars) was used in the economic analyses. In recognition of the fact that supply and demand considerations could exert an upward pressure on ore prices, the economic analyses were repeated using a \$60/lb ore price, to determine the sensitivity of the optimal design to ore price assumptions.

4.2.2 Reprocessing Costs

Capital Costs

Commercial LWR reprocessing plant construction experience is limited to the NFS, AGNS, and GE installations. Currently, none of these facilities is in operation, although fuel has been reprocessed at NFS prior to the shutdown of the facility for expansion. The AGNS facility is considered typical of the reprocessing plants which will be installed at NECs. Detailed design and cost data have not been published for this project; however, recognition must be taken of the AGNS plant investment requirements (recently announced as \$600 million in 1975 \$) in projecting future reprocessing costs for 1500 MT/yr facilities. This includes the cost of:

1. storage of fuel at the reprocessing site
2. shearing of spent fuel

3. dissolution
4. separation of U and Pu from fission products
5. separation of U and Pu and purification of each product
6. conversion of recovered U to UF_6
7. conversion of recovered Pu to PuO_2
8. storage of recovered U and Pu

Capital costs are assumed to be based on a capital structure of 90% equity with an after tax annual earning rate of 15% and a 10% debt with an annual interest rate of 8%. Plant lifetime is 20 years, Federal tax rate is 50% while state and local taxes, insurance, licensing, interim replacements and other miscellaneous costs are 3%. The fixed charge rate corresponding to this financing structure is 26% on depreciable (straight line) capital. Assuming that operating costs are about 10% of the total capital costs, it is expected the AGNS reprocessing costs, based on the above estimates, will be approximately \$133/kg heavy metal in 1976 \$, not including charges for ultimate waste disposal, safeguards or transportation.

Waste Management

One major uncertainty in the cost of reprocessing is the ultimate requirement for radioactive waste management. These requirements will affect reprocessing plant facilities as well as other fuel-cycle considerations such as storage and conversion. It is currently anticipated that high-activity wastes (those wastes generated in the first cycle of solvent extraction or equivalent) will be accumulated as liquids at the reprocessing plant, converted to a solid, and directly shipped to a salt mine repository within 10 years after generation. However, detailed evaluations by ERDA of waste treatment alternatives and the specific solid waste forms and/or process requirements are still in progress. Also being considered is the separation of the very long-lived actinide (transuranium, Am, Cm, etc.) constituents from the bulk of the high-level fission product wastes to reduce the volume required for permanent storage. Implementation of various waste treatment alternatives can substantially affect reprocessing plant waste management costs.

Published regulations have not clearly defined management requirements for other wastes generated in the reprocessing step (low-level wastes), but attention must be given to the ultimate disposition of wastes containing very long-lived alpha emitters, such as certain Pu isotopes. A proposed amendment to 10CFR20, "Standards for Protection Against Radiation", was published in 1974 which would require that waste material contaminated with transuranium elements be solidified and packaged, as soon as practicable after generation and within 5 years, for treatment and disposal. An appropriate fee would be charged the generator of the waste to cover the costs of such activities. Because of the very low-level of alpha activity (10 nC/g) proposed for classifying wastes as transuranium contaminated, practically all process wastes from fuel reprocessing plants may fall in this category. The final form in which this regulation is issued will have substantial impact on future reprocessing costs.

Cost estimates for the treatment and ultimate disposal of wastes generated at the reprocessing facility are difficult to estimate because the criteria for their handling has not yet been established. Reference (1) indicates that, until criteria are firmly established, a reasonable figure to be used for the cost of waste disposal is \$50/kg.

4.2.3 Safeguards Costs

The term "safeguards", as employed here, denotes the steps taken to prevent diversion of fissile material to unauthorized use. These safeguards include considerations of physical security for the facility and accountability of the plutonium.

Currently, safeguards regulations are under NRC review, and some modification and strengthening of these regulations are anticipated in the future. A complete assessment of the cost of safeguards, of course, cannot be made until these new regulations are established. Some guidance, however, can be obtained from the recent E. R. Johnson Associates study prepared for the Atomic Industrial Forum(2), which estimated the costs involved in establishing and maintaining at a reprocessing facility an extensive physical security system and a comprehensive accountability system. The cost calculations show that the total annual cost for the physical security system would amount to \$4.2 million, and the total annual cost for the accountability system would amount to \$2.7 million. On a combined basis, the total cost of the improved safeguards at the reprocessing facility would only amount to roughly \$5/kg heavy metal in the spent fuel. Within the context of the Nuclear Energy Center, safeguards costs should be somewhat less, since the physical security system would be shared with the fabrication plant and operating reactors.

4.2.4 Transportation Costs

Current estimates of spent fuel shipping costs are somewhat higher than those quoted in previous years. This increase has resulted from the increase in shipping cask cost, due to the necessity of accommodating more severe hypothetical accident conditions, and to additional regulations and precautions during transportation.

Reference (3) shows that present estimates of total spent fuel transportation costs, based on the use of special trains, range from \$35/MTU-mile for short hauls to \$7/MTU-mile for long-distance hauls. For a typical large reactor 1000 miles from a reprocessing plant, the total cost would likely be about \$25/kgU.

4.2.5 SWU Prices

The ERDA plans to change from "cost of recovery" to "commercial" pricing beginning in 1978 will lead to much higher SWU prices in the future, compared to prices in effect today. In the past, the price has been calculated to recover the government's costs over a reasonable period of time, and has not included allowance for such things as taxes, insurance and risk. In order to make the anticipated shift to private-industry enrichment easier, ERDA is proposing new pricing criteria which would make ERDA's prices closer to those which would be charged by private industry.

Cost estimates for new enriching capacity can be converted into unit costs, and ultimately prices, if assumptions are made with regard to factors such as capital recovery rates, desired product inventory levels, and structure and financing of the enterprise.

At present, the gaseous diffusion and gas centrifuge processes appear to be the only realistic means for expansion of U.S. enrichment capacity on a time scale consistent with the required timing. Other methods reported, e.g., the Becker nozzle; and the South African Atomic Energy Board (AEB) process, although reported to have reached the pilot plant stage, have not yet advanced enough to be commercial realities until the mid 1980s, if then. Other methods under current development, such as laser enrichment, could potentially be developed in the 1980s, but have not yet reached the pilot stage. In light of the above, price estimates developed here are based on the only two proven reliable technologies, i.e., gaseous diffusion and gas centrifuge.

To estimate what the SWU price will be in the future, it is assumed that the next increment of SWU capacity will be a government-owned gaseous diffusion add-on facility, the subsequent capacity addition, a privately-owned gaseous diffusion plant utilizing advanced technology, and following this, privately-owned gas centrifuge plants. With this scenario, and the financial assumptions shown in Tables 4-1 and 4-2, SWU prices are estimated to be \$85/SWU (1976 \$) in the 1980-1995 time period.

4.2.6 Fabrication Costs

An essential element of the fuel cycle is the fabrication of mixed-oxide fuel assemblies for use in the Pu burners at the energy park. It is assumed that collocated fuel reprocessing facilities will provide the necessary input Pu for the mixed-oxide fuel fabrication plant.

Although the experience to date in the fabrication of mixed-oxide fuel for LWR power stations in pilot facilities is relatively extensive, fabrication processes and facility requirements for fabricating mixed-oxide fuel have not been fully defined for large-scale production. The fact that more restrictive regulatory requirements have evolved since the early prototype facilities were designed and licensed reduces the importance of mixed-oxide fabrication experience to date as a basis for projecting future facility requirements and costs.

Regulatory requirements in the areas of process material confinement, operating and accident forces (earthquakes, etc.), and nuclear criticality safety are generally well-defined. However, continued tighter regulatory requirements are indicated for occupational radiation exposure and for protection against fissile material diversion and loss of material control by sabotage. Although the final form of these regulations has not yet been established, indications are that they could have a substantial bearing on future facility designs and costs.

The fuel fabrication facility design used in developing fabrication cost estimates assumes that process systems will be mechanized so as to limit personnel access and to reduce operating costs. A mechanized facility was selected since it was felt that it would be difficult and costly to achieve anticipated future personnel protection standards when processing high-exposure plutonium with glove-box operated and contact maintained facilities.

Capital costs for mixed-oxide fuel fabrication plants will be higher than many previously published estimates, not only because of additional provisions required for plant safety and personnel protection and nuclear safeguards, but also because of higher construction costs. Reference 4 estimates that, at a design capacity of 300 metric tons of mixed-oxide fuel per year, capital costs on the order of \$100 million (Jan. 1975 dollars) are indicated where support facilities are shared with a collocated reprocessing plant.

Fabrication costs for mixed-oxide fuel will vary with capacity and will entail cost penalties, relative to UO_2 fuel, of 250% in small plants having annual throughputs of 30 to 80 MT/yr. However, for larger plants (300 to 500 MT/yr) fabrication penalties are expected to be much less. The results of Reference 4 show that unit costs for mixed-oxide fuel rod fabrication and bundle assembly functions, including capital charges, are expected to be in the range of \$150 to \$570/kg UO_2 for plant throughputs ranging from 500 MT/yr to 80 MT/yr, respectively.

Here, a fabrication price of \$225/kg heavy metal was chosen as the reference cost for fuel cost economic evaluations.

4.2.7 Summary

Table 4-3 provides a summary of the cost parameters used to evaluate the fuel-cycle economics of plutonium burners at Nuclear Energy Centers. It is recognized that the reprocessing and mixed-oxide fabrication costs may be higher or lower than those shown, depending on the number of plutonium burners at the site. This number determines the throughput of fuel and, therefore, the ultimate size and cost of the collocated reprocessing and fabrication facilities. However, based on presently available cost information, the values shown from a reasonable basis to determine the design characteristic of economically optimized plutonium burners.

4.2.8 References

1. Walton A. Rodger, "Throwaway Fuel Cycle," paper presented at Fuel Cycle Conference '75, Atlanta, Georgia.
2. E. R. Johnson Associates, Inc., "An Upper Estimate of Safeguards for handling Pu" (July 21, 1975), prepared for Atomic Industrial Forum.
3. Reuben W. Peterson, "Shipment of Spent Fuel and Recovered Products," Paper presented at ANS Topical Meeting, Toronto, April 30, 1975.
4. NSF-75-500, "Assessment of Energy Parks versus Dispersed Electric Power Generating Facilities, Final Report (May 30, 1975).

4.3 Power Cost Considerations

The objective of this section is to assess the change in cost of electric power generation for three different ways of changing the water-to-fuel ratio of the Pu burner lattice. The water-to-fuel ratio can be modified by: 1) changing the lattice pitch (spacing between fuel rods); 2) by changing fuel rod diameter; and 3) by decreasing the effective fuel density. The first two options represent the most straightforward procedure, but necessitate major modifications to the physical size

of the reactor and/or thermal hydraulic design. The third option, that of low-density fuel, is attractive since the increase in water-to-fuel ratio is realized without requiring a modification to the core or plant design and without a degradation of thermal margin.

For Options 1) and 2) above, consideration must be given to the impact of changes on NSSS and containment size, pumping power, secondary design, and balance of plant design in order to determine the effect that these changes have on the capital cost of the plant. Table 4-4 shows how the effective core diameter changes for these two design options, while Table 4-5 shows which components of the capital cost are affected. The PWR reference design described in Section 3.2 and outfitted for APR operation provides the base case so that no additional capital costs are required for CEAs, CEA drive mechanisms, or safety injection tanks.

Power costs consist of two components: fixed charges on capital investment and operating expenses. Once a piece of capital equipment is operational, the fixed charges on the investment begin and continue for the life of the equipment. An annual carrying charge may be determined by applying an appropriate levelized fixed charge rate. This annual cost, the product of the total capital cost and the levelized fixed charge rate, is assessed each year during the life of the equipment. The levelized fixed charge rate includes the following components:

1. Depreciation
2. Return on Investment
3. Federal Income Taxes
4. State and Local Taxes
5. Insurance

The fixed charge rate therefore depends upon the lifetime of the plant as well as the utility financial position, federal and state tax rate, etc.

For a private investor owned utility, the fixed-charge rate of 16% is assumed for the power generation equipment with a 30-year lifetime. Other assumptions used to calculate power costs are shown in Table 4-6.

The second component of the power cost is the annual operating expense. The operating expenses are comprised of the fuel costs and operating and maintenance expense. Assumptions regarding fuel costs were described in the previous section. Operating and maintenance expenses are assumed to be 2 mills/kwhr(e) (1976\$).

For a fixed-charge rate of 16%/year, and with the assumptions shown in Table 4-6, a change in capital cost of \$276,000 (1976\$) is equivalent to a .01 mill/kwhr(e) change in power cost. For a 20%/year fixed-charge rate, a change in capital cost of \$221,000 (1976\$) is equivalent to a .01 mill/kwhr(e) change in power cost.

The parameter of most interest in the economic optimization of the plutonium burner is the change in the cost of power from the plutonium burner as the lattice design is modified. This parameter, however, does not completely characterize the impact of plutonium burner design modifications since the number of uranium-fueled reactors required to provide the makeup fissile plutonium for the plutonium burners changes as the lattice design of the plutonium burner is modified. A second parameter of interest is the average, or "system," power cost associated with the plutonium burners and the uranium-fueled reactors which serve to supply the required feed of plutonium.

An average "system" power cost may be defined to be the average of the power cost of the Pu Burner at the energy park and the cost of operating the reactors which feed Pu to the Pu burners. Expressed mathematically,

$$E = \frac{(E_c^U + E_f^U) N_u + (E_c^P + E_f^P) N_p}{N_u + N_p} \quad (4.3-1)$$

where:

- E = Average system power cost
- N_u = No. of all U Feeders
- N_p = No. of Pu Burners
- E_c^U = Capital and Operating cost of U Feeders
- E_f^U = Fuel Cost of U Feeder
- E_c^P = Capital and operating cost of Pu Burner
- E_f^P = Fuel Cost of Pu Burner

In the discussion that follows, "system" costs are expressed in terms of the total power cost per Pu Burner at the energy park so that $N_p = 1$ in equation 4.3-1 or:

$$E = \frac{(E_c^U + E_f^U) N_u + (E_c^P + E_f^P)}{N_u + 1} \quad (4.3-2)$$

The number of U feeders, N_u , required to supply Pu to one burner is a function of the water-to-fuel ratio of the lattice. Figure 4-2 shows the annual initial fissile Pu loading and fraction of fissile Pu consumed in a Pu burner as a function of water-to-fuel ratio for several different rod diameters. It is seen that the fissile loading required decreases while the fraction of fissile Pu consumed increases as the water-to-fuel ratio is increased. For a rod diameter of .382", loading requirements decrease by 38%, while the fraction of fissile Pu consumed increases by 100% as the water-to-fuel ratio is varied from 1.6 to 2.8. The net effect is an increase in the fissile Pu makeup requirements as the water-to-fuel ratio is increased.

Figure 4-3 shows the number of U feeders required per Pu burner for different water-to-fuel ratios of the Pu burner lattice when the wetness of the lattice is changed by varying the lattice pitch; the number of U feeders required increases with water-to-fuel ratio to provide the increased fissile Pu makeup requirements.

When the water-to-fuel ratio is increased by reducing the pellet density, fissile Pu loading requirements decrease and the fraction of Pu consumed increases, as Figure 4-4 shows. With this option, the number of U feeders required per Pu burner for a given water-to-fuel ratio is greater (Figure 4-5) than the number required when the water-to-fuel ratio is changed by varying lattice pitch.

In order to evaluate "system" costs and to determine the effect of modifications of lattice design on the economics of plutonium utilization, it is necessary to define a Pu value so that E_f^U and E_f^P in Equation 4.3-2 may be computed. The typical procedure for establishing a Pu value is to relate the price of plutonium to the value of the uranium fuel which is displaced. This "indifference value" is that plutonium price which yields equal fuel cycle costs in uranium and plutonium fueled cores. In this case where two different reactor types are involved—the standard uranium fueled plutonium producer (feeders) and the optimized plutonium burner—the definition of indifference value is ambiguous, since plutonium will have a different value in the optimized burner than if it were employed in the feeder reactors.

Establishing the plutonium indifference value based upon plutonium worth in the optimized burner is useful in order to establish how the economic value of plutonium is enhanced by optimization, and to establish the maximum price one could afford to pay for plutonium and still remain competitive with plutonium utilization at dispersed sites. This indifference value is established by determining the plutonium price such that the fuel cycle cost of the plutonium burner for each lattice configuration is equal to that of uranium fueling in the reference design, taking appropriate credit for the plutonium produced, i.e.,

$$E_f^U (W/F = 2) \equiv E_f^P (W/F) \quad (4.3-3)$$

Plutonium values determined by this method are given in Figure 4-6. Plutonium values are quoted in terms of a "parity value" in this figure. This parity value is the fraction of the cost of 90 w/o enriched uranium, and is employed because it is less sensitive to economic assumptions than the indifference value itself. The indifference value can be obtained by multiplying the parity value by the cost of enriched uranium; for the base case economic assumption, a parity value of unity corresponds to an indifference value of 35\$/gm fissile Pu. As may be seen from Figure 4-6, plutonium value increases with increasing water-to-fuel ratio, increasing, for example, by 43% between a water-to-fuel ratio of 1.6 and 3.0. Note that, for a given water-to-fuel ratio, higher ore prices enhance the value of plutonium while higher carrying charges decrease its value. This curve also represents the price the plutonium burner could afford to pay for plutonium, at each water-to-fuel ratio, and still remain competitive with the fuel-cycle cost of the uranium reactors. Since, by definition, uranium and plutonium burner fuel cycle costs are equal, this method of defining plutonium value does not fully reflect the gains in economy realized by the plutonium burner, other than as reflected in a reduction in overall system (uranium plus plutonium burners) power costs.

An alternate approach, and one which more clearly delineates the economic incentive for optimization, is to define the plutonium value based on its worth to the uranium burner reactor, or, equivalently, in plutonium burners of the PWR reference design. This is accomplished by determining a plutonium value such that

$$E_f^U (W/F = 2) = E_f^P (W/F = 2) \quad (4.3-4)$$

This plutonium value which is obtained from Figure 4-6 at a W/F ratio of 2.0, represents the price that the uranium-fueled reactors could sell plutonium to the plutonium burners and remain competitive with self-utilization. Since this

indifference value is not dependent on plutonium burner design, uranium burner fuel cycle costs are constant, and any improvements in plutonium utilization in the plutonium burners is observed directly as a reduction in plutonium burner fuel cycle cost.

In the economic evaluations presented in Section 4.4, the latter definition of indifference value has been employed in order to obtain a more direct understanding of the impact of plutonium burner optimization on the economics of the NEC.

4.4 Selection of Optimum Plutonium Burner Designs

4.4.1 Minimum Power Cost

For each of the options available to modify the water-to-fuel ratio of the plutonium burner lattice, plutonium burner and system power costs were determined as a function of lattice water-to-fuel ratio and compared to the system power cost obtained with the PWR reference design (c.f. Section 3.1.2). Here the "system" is defined as a nuclear energy center comprising an arbitrary number of plutonium burners and the associated number of uranium burning plants of the PWR reference design required to provide plutonium feed.

Figure 4-7 shows the change in plutonium burner and system costs as the water-to-fuel ratio is modified by changing the lattice pitch. The lower curve gives the change in capital costs versus water-to-fuel ratio. This component increases as the water-to-fuel ratio becomes larger because of the increasing size of the reactor. The second curve from the bottom shows that fuel cycle costs decrease with water-to-fuel ratio. As the water-to-fuel ratio increases, the required initial fissile plutonium inventory decreases while the plutonium makeup requirements increase slightly (cf. Section 4.3). Fuel cycle costs decrease with increasing water-to-fuel ratios because of reduced carrying charges on the lower inventory of plutonium, which proves to outweigh the added costs associated with the slight increase in plutonium consumption. The second from the top curve shows the change in plutonium burner power cost as the lattice water-to-fuel ratio is varied; this total power costs is simply the sum of the change in capital and fuel cycle costs given in the lower two curves. The top curve gives the change in average system power cost. This curve was constructed from Equation 4.3-2 using the change in plutonium burner power cost from the second curve in Figure 4-7 and the power cost of the uranium fueled reactors.

As indicated in Figure 4-7, the lowest plutonium burner or system power cost is obtained at the highest water-to-fuel ratio evaluated. At this water-to-fuel ratio, 3.2, plutonium burner and average system power costs are reduced by 0.71 and 0.27 mills/kw-hr respectively. Most of this reduction, however is realized at a water-to-fuel ratio of 3.0, and higher ratios are unattractive because of moderator temperature coefficient considerations (cf. Section 4.4.3).

For the reference NEC (cf. Section 4.5), which services a total of fifty 1300 Mw(e) reactors (fifteen plutonium burners and thirty-five associated uranium-fueled reactors), the use of such optimized plutonium burners would result in an annual power cost savings of about \$106 million. This saving is also equivalent to a \$345 million reduction in the capital costs of the NEC's complement of plutonium burners. Although these savings are not insignificant, they are still a relatively small fraction of the total cost of power or of NEC construction.

The influence of changing carrying-charge rates and uranium ore price assumptions is illustrated in Figures 4-8 and 4-9, respectively. Increasing the carrying charge rate tends to increase the benefits of higher water-to-fuel ratios slightly, due to reduced plutonium inventory at higher water-to-fuel ratios, and results in an optimum at a slightly higher water-to-fuel ratio. Increases in the assumed price of uranium ore tends to move the optimum to higher water-to-fuel ratios and to increase the benefits realized by optimization. In any case, however, the optimum lattice water-to-fuel ratio is 3.0 or larger, and hence, from economic considerations, the plutonium burner should have as high a water-to-fuel ratio as possible, consistent with technical (moderator temperature coefficient) constraints.

Figure 4-10 shows similar data for plutonium burner and systems costs when the water-to-fuel ratio is held fixed but the rod diameter is changed. Capital costs increase with an increase in rod diameter because the lattice pitch must increase to maintain the water-to-fuel ratio. Table 4-7 shows the evolution of fresh assembly batch size and fuel assembly residence time as the rod diameter is increased from .3427 in. to .4512 in. The fissile loading requirements and fabrication costs decrease as the rod diameter increases, while fuel cycle costs increase because of the higher carrying charges which accrue on the fissile plutonium inventory which is carried in core.

Therefore, when the rod diameter is increased, the fuel cycle cost savings which result from a smaller plutonium loading requirement and a lower fabrication cost are overshadowed by higher carrying charges on the fuel. Note that the lowest system power cost is obtained for the smallest rod diameter, and that the savings are smaller than that realized by changing the water-to-fuel ratio.

Figure 4-11 shows the change in system fuel cycle cost when the water-to-fuel ratio is changed by reducing the fuel density in the plutonium burner. The corresponding change in plutonium burner fuel cycle cost is given in Figure 4-12. The fissile loading requirements decrease and the fraction of plutonium consumed increases as the water-to-fuel ratio increases. Although there is a substantial increase in the plutonium consumption rate, fuel cycle costs decrease as the pellet density is reduced because of the lower initial fissile loading requirements and the correspondingly lower carrying charges on the plutonium inventory. By using this method to increase the water-to-fuel ratio, no capital cost penalty is incurred because the reactor lattice geometry remains unchanged. As a result, the change in power cost is simply equal to the change in fuel cycle cost. Note that the lowest fuel cycle cost is obtained at the lowest effective pellet density.

From the standpoint of minimizing power costs, increasing the water-to-fuel ratio by reducing the pellet density appears to be the most promising method. Although the cost reductions realized by reducing pellet density are slightly smaller than that realized by increasing the lattice pitch, 0.19 versus 0.25 mills/kw-hr system costs, respectively at a water-to-fuel ratio of 3.0, reduced density fuel has the advantage of not requiring modifications to the lattice geometry and of preserving fuel management flexibility options. The cost savings realized by reduced density fuel (70% dense fuel, or a water-to-fuel ratio of 3.0) is equivalent to a 262 million dollar reduction in capital costs or an 80 million dollar annual power cost saving for a fifty-reactor (uranium and plutonium burners) NEC.

4.4.2 Minimum Ore and Separative Work Requirements

Combined ore and separative work requirements for a 1300 Mw(e) plutonium burner and its associated uranium burning plants as a function of plutonium burner water-to-fuel ratio, fuel rod diameter, and fuel pellet density are shown in Figures 4-13 through 4-15. These curves are based upon a 3.29 w/o U-235 feed requirement for the uranium burning plants, 0.3 w/o tails, and recycling (with a 3% reprocessing loss) of the uranium discharged from the UO_2 plants.

Although increasing the plutonium burner lattice water-to-fuel ratio by reducing the mixed oxide pellet density results in the lowest system power costs, this method is shown to place the greatest demand on uranium ore and separative work resources. Increasing the water-to-fuel ratio by increasing the plutonium burner lattice pitch results in higher system power costs but the lowest ore and separative work requirements. Thus, selection of an optimum plutonium burner design will depend upon nuclear energy center objectives. Uranium ore and separative work utilization can be maximized by the use of an extremely dry lattice while minimum power costs are obtained by the use of an extremely wet lattice. There quite possibly exists an optimum lattice design from a cost/benefit standpoint, however, a detailed exploration of this possibility lies beyond the scope of the present study.

4.4.3 Design for Maximum Fuel Management Flexibility

Fuel management flexibility, i.e., the ability to fuel the plutonium burner with alternate types of fuel, in particular, slightly enriched uranium, is an important consideration in plutonium burner design. In particular, the capability of accommodating uranium fueling is desirable for the following reasons:

1. In the absence of such flexibility, the plutonium burner would be totally dependent on the continued operation of the reprocessing and mixed-oxide fabrication facilities for their fuel supply. In the event of extended outages at one of these facilities, the electrical capacity of the NEC plutonium burners could be lost for lack of a fuel supply. In order to guard against this situation, plutonium and fabricated fuel would have to be stockpiled in order to insure a supply of mixed-oxide fuel during a reprocessing or fabrication plant outage.

Such stockpiling of plutonium fuel is not particularly attractive because of the cost of storage and inventory carrying charges. Stockpiling can also delay the rate of plutonium burner installation (cf. Section 4.5). In addition, there would be no way of replenishing the plutonium stockpile consumed during an outage, unless excess reprocessing and fabrication capacity were provided, and this would not appear to be economically attractive. If plutonium burners are designed with the flexibility of accommodating uranium fueling, uranium fuel could be substituted during periods of plutonium supply shortage.

2. As noted in Section 4.5, initial core loadings require more plutonium and fabricated fuel than is required for the annual reloads (about one-third of the core is reloaded each year). These higher initial requirements limit the rate of introduction of plutonium burners. A faster plant startup rate, to meet, for example, a high electrical demand, might be accommodated by initially fueling the plutonium burners with uranium, and making the transition to mixed-oxide fueling as plutonium becomes available.

3. In the later years of NEC operation, after the year 2000, it may be necessary to convert plutonium burners to UO_2 -fueled operation in order to provide additional plutonium for FBR deployment (cf. Section 4.5).

The primary consideration affecting the design of PWRs having the desired level fuel management flexibility is the moderator temperature coefficient of reactivity (MTC). The MTC should remain negative throughout the cycle, in order to avoid operation problems and an adverse impact on certain accidents, such as ATWS. The MTC becomes less negative as the water-to-fuel ratio is increased. A larger water-to-fuel ratio can be accommodated in a mixed-oxide-fueled core while maintaining an acceptable MTC because the plutonium fuel compensates for the increased moderation. However, if the option of uranium fueling is to be preserved, the water-to-fuel ratio is limited by MTC constraints imposed by the uranium fueling requirement.

The water-to-fuel ratio of the System 80TM reactor, which has been employed as the reference design in this study, was determined based on MTC considerations for the initial uranium core loading; lattice shims are employed in this cycle to achieve a sufficiently negative MTC. If the option of initially fueling the plutonium burner with uranium is to be retained, little increase in the lattice water-to-fuel ratio would be acceptable.

In the long term, where it may be necessary to convert the plutonium burner to UO_2 -fueled operation to provide plutonium for the FBR, some increase in water-to-fuel ratio could be accommodated since the plutonium and fission products resident in the burned fuel will tend to make the MTC more negative than in the case for an initial uranium core loading. In this case, it is estimated that an increase in water-to-fuel ratio to somewhere between 2.2 to 2.3 would be acceptable.

Greater latitude is possible if fuel-management flexibility is to be provided solely as a contingency against the interruption of plutonium supply. In this case, only a fraction of the core, perhaps one reload, would be expected to be comprised of uranium fuel, and since the uranium fueling is expected to be only temporary, the addition of burnable poison shims to make the MTC more negative could be considered. In this instance, increasing the lattice water-to-fuel ratio to between 2.5 and 2.6 might be acceptable.

4.4.4 Selection of An Overall Optimum Design

Optimum designs to achieve specific objectives of the NEC were discussed in Sections 4.4.1 through 4.4.3. In general, it was found that design modifications can contribute to achieving specific objectives, but that the optimum design generally differs depending on the objectives. In some cases modifications to achieve one particular objective can compromise other objectives. The question remains, then, as to which design represents the best overall compromise. There is no simple answer to this question since the conclusion depends on the relative importance placed on each objective, but some general observations can be made.

Decreasing the water-to-fuel ratio contributes to the objective of conserving uranium resources. The saving in uranium ore and separative work requirements, however, is relatively small. A reduction of less than 2% is achieved with respect to the PWR

reference design at a plutonium burner water-to-fuel ratio of 1.6. Further improvements could, of course, be made by employing still drier lattices, but an extremely dry lattice would be required to make substantial gains; such a lattice would be a marked departure from current PWR technology. Since the initial fissile plutonium loading increases as the water-to-fuel ratio is reduced, the cost savings attending uranium ore and separative work conservation cannot be fully realized. Consequently, it is concluded that the objective of improved uranium ore and separative work utilization realized by reduced water-to-fuel ratios should not be a major consideration in selecting the overall optimum plutonium burner design.

NEC power and fuel cycle costs can be reduced by increasing the water-to-fuel ratio by expanding the lattice pitch (cf. Section 4.4.1) at the expense of only a small increase in uranium ore consumption. While this savings represents several millions of dollars per plutonium burner, it is small compared to the overall cost of power, or to the cost of the plutonium burners themselves. Of concern also is the maintenance of fuel management flexibility which can be compromised at high water-to-fuel ratios. Lastly, it was found that most of the economic benefit observed by increasing the lattice pitch could also be realized using fuel of a low effective density; since full density UO_2 fuel could be employed, if required, this option preserves fuel management flexibility.

Consequently, it is concluded that the plutonium burner design representing the best overall compromise should have a lattice design similar to that of current PWRs. Such designs preserve fuel management flexibility and reduced fuel cycle and power costs could still be realized by employing reduced effective density fuel. Improvements in uranium ore and SWU utilization, or further fuel cycle, or power cost reductions realized by modifying lattice pitch, are not viewed as sufficiently large to warrant the cost associated with developing and licensing these designs.

4.5 Consideration of Plutonium Utilization and Siting Scenarios

4.5.1 General Description of Nuclear Energy Centers for Plutonium Utilization

The basic Nuclear Energy Center design considered in this study is shown schematically in Figure 4-16. In this concept, a spent fuel reprocessing facility, a mixed oxide fuel fabrication facility, a number of reactors designed for plutonium utilization, and related support facilities such as waste treatment and spent fuel and plutonium storage are located at a single site. Spent fuel from the plutonium burners and from associated uranium fueled reactors are reprocessed at the energy center to provide the plutonium which fuels the nuclear energy center's plutonium burning reactors. The associated uranium fueled reactors may either be located on the NEC or at dispersed sites with spent fuel shipped to the NEC for reprocessing.

The primary advantage of this concept, in addition to the economics generally attributed to the NEC concept, is that plutonium utilization is confined to the NEC. This eliminates the need for the off-site shipment of plutonium, either in the pure oxide form or as fabricated fuel, and hence eliminates the need for the safeguarding of plutonium shipments. Adequate security measures for safeguarding plutonium located at the NEC should be more simple and less costly than that required when reprocessing plants, fabrication facilities, and plutonium burners are located at separate and dispersed sites.

In the sections which follow, the concept of a unit NEC is first developed, and then deployment of these energy centers to meet the projected nuclear growth during the remainder of the century are explored.

4.5.2 Unit Nuclear Energy Centers

The unit NEC is defined to consist of a 1500 MT/yr reprocessing facility and mixed oxide fabrication, plutonium storage, and an installed plutonium burner capacity of sufficient size to utilize the recovered plutonium. The uranium burning reactors which supply the spent uranium fuel input to the reprocessing facility may be located at the NEC or at dispersed sites.

The 1500 MT/yr size was chosen for the reprocessing facility as it was felt that this size generally represents the optimum throughput for such facilities; the size of the NEC can be adjusted, of course, by scaling each component to match reprocessing throughput.

Other considerations which should be borne in mind while discussing the deployment of plutonium burners at nuclear energy centers are identified below.

- a. Demand for reprocessing and mixed oxide fabrication should either increase or remain constant as the NEC matures; scenarios which require a temporary high reprocessing or fabrication load with subsequent decommissioning of facilities should be avoided as uneconomical.
- b. Similar considerations apply to plutonium storage. However, it may be desirable to require that some minimum quantity of mixed oxide fuel be stored for the plutonium burners as a contingency against temporary outages of the reprocessing or fuel fabrication facilities.
- c. Plutonium burners should be introduced in a continuous manner in order to take advantage of the construction economies associated with maintaining a stable construction work force.

4.5.3 Deployment of Plutonium Burners at a Nuclear Energy Center

The characteristics of the plutonium burners used in developing deployment scenarios are given in Table 4-8. It has been assumed that these reactors are operated as plutonium burners from startup, i.e., that the initial core loading would be comprised of mixed oxide fuel. Since roughly three times as much fuel must be fabricated to supply the initial core loading as is required for a reload batch, the startup of a new plutonium burner places an initially high demand on plutonium supply and fabrication capacity. This in turn limits the possible rate of introduction of the plutonium burners.

Figure 4-17 gives the annual fissile plutonium requirements for the reference plutonium burner; the diluent is assumed to be natural uranium. As this figure illustrates, a larger quantity of plutonium is required to fuel the initial core than is required for a reload batch (approximately one-third of the fuel is replaced each year). The annual quantities of reload plutonium required also increases until an equilibrium is reached after about twenty years of operation. The annual quantity

of plutonium recovered from the plutonium burners is shown in Figure 4-18. The amount of plutonium recovered gradually increases from about 300 kg, for the low enrichment batch in the first core which is discharged after one cycle of irradiation, to about 840 kg in the equilibrium cycle.

The reference plutonium burner deployment scenario is shown in Figures 4-19 a-f. As shown in Figure 4-19 a, the first plutonium burner is placed in operation in year zero. Two reactors per year are added for the next six years, and then one reactor in years seven and eight, reaching a total of fifteen plutonium burners eight years after the installation of the initial reactor. The reprocessing plant is assumed to begin operation three years prior to the startup of the first plutonium burner (Figure 4-19 b); it is assumed to operate at its full capacity of 1500 MT/yr during subsequent years. Mixed oxide fabrication capacity is shown in Figure 4-19 c. The fabrication facility begins operation one year prior to startup of the initial plutonium burner with a throughput of 200 MT/yr, and increases capacity as additional burners are placed in operation until the ultimate capacity of 450 MT/yr is reached.

The balance between plutonium production and demand for this scenario is shown in Figure 4-19 d. Plutonium demand begins two years prior to startup of the first plutonium burner in order to provide a stockpile of plutonium for fabrication which begins in the subsequent year. Plutonium demand increases rapidly for the next five years, largely because of the requirements for initial cores, slows somewhat in years six and seven (plutonium demand in these years results from reload requirements plus that required for new cores which are now being introduced at the reduced rate of one per year), and then gradually increase in subsequent years reflecting the increase in reload feed requirements as the Pu burners mature (see Figure 4-17). Plutonium production reaches a value of 10,350 kg/year after the reprocessing plant reaches full capacity. At full capacity, the reprocessing plant can accommodate the spent fuel from 50 reactors (in this time period, only fuel from uranium fueled reactors is available for reprocessing). In subsequent years, as the plutonium burners discharge fuel, the quantity of recovered plutonium increases. After year three, plutonium demand exceeds production. This is due to the demand from new core loadings and the increases in reload requirements which leads the increase in recovered plutonium. During these years, excess plutonium stored from prior years of operation, when supply exceeded demand, must be employed. Finally, in year 15, an equilibrium condition is reached in which annual plutonium production essentially matches demand.

The quantities of recovered plutonium which are in storage, either as plutonium oxide or in fabricated fuel, is given in Figure 4-19 e.

The quantity of plutonium in storage increases during the years when production exceeds demand, and then this stored plutonium is used to supplement production in subsequent years before an equilibrium is reached. As this figure illustrates, there is essentially no plutonium left in storage when the equilibrium situation is achieved. The last figure, 4-19 f, gives the number and type of reactors associated with the NEC. Once the reprocessing facility reaches full capacity, a total of fifty reactors are always associated with the energy center, since this number of reactors provides the required amount of fuel to keep the reprocessing plant operating at full throughput. Initially, before the plutonium burners begin operation and discharge fuel, the

reprocessing plant services only uranium fueled reactors. As the NEC's plutonium burners are introduced, they displace uranium fueled reactors, so that the total number of plants accommodated remains constant.

There are a number of other considerations and constraints on plutonium burner deployment which are briefly discussed below:

1. Other rates of plutonium burner deployment than that shown in Figure 4-19 are, of course, possible. The particular rate chosen appears logical in that it provides for the continuous use of a trained labor force over a relatively long period of time, thus contributing to possible construction economies and stabilizing the labor situation at the NEC.

A longer deployment period could be easily accommodated; for example, one reactor per year over a period of fifteen years. In this case, the buildup of reprocessing and fabrication capacity would also be delayed, so as to balance plutonium supply with demand, and thus avoid developing large plutonium stockpiles. A more rapid rate of introduction presents a more difficult situation, since the reprocessing and fuel fabrication facilities could not supply sufficient plutonium for the initial loadings.

If a more rapid introduction is required, for example, to meet electrical capacity requirements, one of the following options could be elected:

- a. Reprocessing plant operation could be started earlier in order to develop a stockpile of plutonium to supplement annual production in years where demand is high because of rapid expansion. As noted above, some storage of plutonium is required in the reference scenario (cf. Figure 4-19 e); in the case of a more rapid expansion, this stockpile would temporarily be larger, resulting in additional storage costs.
 - b. Plutonium, recovered and possibly fabricated at a dispersed site, could be shipped into the NEC to supply the additional plutonium required during the years of rapid expansion. This option, of course, assumes that safeguards considerations permit the shipment of plutonium and that dispersed site reprocessing facilities exist.
 - c. Reprocessing plant capacity could be increased to provide the plutonium as required. The disadvantage of this approach is that the expanded reprocessing capacity would not be needed once equilibrium cycle operation is achieved, and hence this option is not likely to be economically attractive.
 - d. Some reactors at the NEC could be fueled initially with enriched uranium, with a transition to mixed oxide fueling when permitted by the plutonium supply. This option assumes that the plutonium burners have sufficient fuel management flexibility to operate on the uranium cycle.
2. In the reference deployment scenario discussed above, no additional plutonium is held in storage, except that held in temporary storage to accommodate expansion in those years where plutonium demand exceeds production, i.e.,

there is essentially no plutonium in storage after year twelve (cf. Figure 4-19 d). There consequently is no stockpile of plutonium available to provide fuel for the burners in the event that an operational problem at the reprocessing or fabrication facilities temporarily terminates the plutonium fuel supply. Such a situation would be acceptable only in the case where the plutonium burners have sufficient fuel management flexibility to accommodate uranium fueling in the event that the plutonium supply is interrupted.

In the event that the plutonium burners are not provided with such fuel management flexibility, stockpiling of plutonium to prevent an interruption of fuel supply is probably a necessity. The stockpile should probably be sufficiently large to provide the plutonium required for the subsequent year of operations, since it appears reasonable to assume that the entire plutonium supply could be interrupted for this period due to an outage at the reprocessing plant. A similar situation exists with respect to the fabrication facility, except that it is anticipated that these facilities will employ a number of 50-100 MT/yr parallel lines, and so the total fabrication capacity should not be interrupted. Consequently, a stockpile of fabricated fuel equal to about 25% of the subsequent years requirement should be sufficient.

One method of maintaining such a stockpile is to add plutonium burners only when sufficient plutonium is in storage to cover the subsequent years requirements. This modification to the basic deployment scenario is illustrated in Figures 4-20 a to c. The rate of reprocessing capacity deployment (Figure 4-20 b) is identical to the reference case (Figure 4-19 b). The introduction of plutonium burners, Figure 4-20 a, initially proceeds as in the reference scenario, but in year six the rate of introduction must be slowed to one unit every 2 to 4 years; by year sixteen only 14 of the 15 units have been added. The total plutonium in storage is given in Figure 4-20 c. Storage requirements peak at 25 MT, about equal to those of the reference scenario, and then level off at about 20 MT, the approximate annual requirement of the NEC at equilibrium.

A second alternative is illustrated in Figures 4-21 a to c. Here, the installation rate of the reference scenario is maintained, but reprocessing capacity is brought on line two years earlier in order to develop the required stockpile before plutonium burner deployment begins. Figure 4-21 c shows the total plutonium storage requirements and, as this figure indicates, storage requirements peak at about 46 MT before decreasing to the equilibrium requirements of 20 MT. There is, of course, some economic penalty associated with providing this additional storage.

Of the alternatives discussed the deployment of plutonium burners with the flexibility of accommodating uranium fueling appears to be the most attractive. The other alternatives, both of which rely on plutonium storage, are somewhat unsatisfactory since there is no way of replenishing the stockpile (short of installing excess reprocessing capacity) following an outage, and hence no latitude to accommodate a second outage. As noted above, these alternatives also increase plutonium storage requirements or slow the introduction of plutonium burners. A more detailed evaluation of the economic considerations and tradeoffs involved in guaranteeing the plutonium supply should be performed to more clearly identify the most advantageous deployment scenario.

4.5.4 Deployment of Nuclear Energy Centers

In the previous section, the deployment of reprocessing, mixed oxide fuel fabrication, and installed plutonium burner capacity was examined to illustrate the evolution and requirements of a single NEC complex. In this section, considerations of the deployment of the unit NEC to meet the forecasted growth in electrical demand are explored. Throughout this section, the reference plutonium burner deployment scenario, illustrated in Figure 4-19, is employed.

The forecast growth of nuclear electric generating capacity is shown in Figure 4-22. This growth scenario corresponds to ERDA moderate - low predictions (1). The data for thermal reactors includes the HTGR. For the purpose of this study, HTGR capacity was assumed replaced by LWR deployment; this was done because of the uncertainty in HTGR availability, and in any case has only a small impact on the deployment scenarios since even in the year 2000 HTGRs represent a relatively small fraction of the installed capacity, less than 13%. As Figure 4-22 illustrates, the anticipated installed FBR capacity is much smaller than that expected for thermal reactors, even in the year 2000; in the year 2000, the annual installation of thermal reactors is still 50% greater than FBR additions. The impact of FBR deployment on the evaluation of the NEC is discussed in Section 4.5.5.

Figure 4-23 gives the number of 1500 MT/yr reprocessing plants (solid line) which will be required, based on the nuclear growth given in Figure 4-22. In theory, each reprocessing facility offers the potential of adding a NEC containing plutonium burners, since each reprocessing facility will recover sufficient plutonium to support plutonium burner deployment. However, for logistical reasons, actual NEC deployment is expected to be more gradual.

Two reprocessing facilities are currently under construction or awaiting modification; the AGNS facility at Barnwell, N.C., and the NFS facility at West Valley, N.Y. Since these facilities were not sited for NEC application, it is doubtful whether these sites would prove suitable. The projected NEC deployment, shown in Figure 4-23, has consequently been assumed to begin with the addition of the third unit of reprocessing capacity in 1985. The first plutonium burner would then begin operation in about 1987. This schedule would seem to provide a reasonable amount of time for site selection and licensing. The next unit of reprocessing capacity is needed in 1988. If a nuclear energy center were to be deployed at this date, planning would have to begin by 1978. Since there will be only a very limited amount of licensing experience with the initial NEC at this date, it is assumed that this reprocessing plant will be located at a dispersed site and is not associated with a nuclear energy center.

The second energy center is assumed to be introduced in 1990 in conjunction with the fifth increment of reprocessing capacity. Additional NECs are introduced in subsequent years each time an additional unit of reprocessing capacity is required. It should be noted however that experience with a large number of operating plutonium burners at the first NEC will not be available until the early 1990's (7 units are deployed by 1990, and it is not until 1995 that all 15 units in the first NEC become operation). If a decision on additional NECs must await operational experience at the initial NEC, additional NEC deployment could be deferred until 2000 or beyond. It has been assumed here that a tentative commitment to deploy

NECs, i.e., a decision to build reprocessing and fabrication facilities on sites suitable for NECs, need not await such experience since the number of plants constructed at a site could be limited if required. Nevertheless, the NEC deployment scenario given in Figure 4-23 is probably optimistic, representing a maximum rather than a typical rate of introduction.

The total nuclear capacity and installed plutonium burner capacity is given in Figure 4-24 for the NEC deployment of Figure 4-23. Also shown on this Figure is the captive UO_2 fueled capacity, i.e., those reactors which depend on NECs for reprocessing and which, in turn, supply the plutonium feed required by the plutonium burners. The substantial installed capacity, approximately 250 GW(e) after 1990, which remains independent of NECs is comprised of those plants which utilize the three dispersed site reprocessing facilities and those which are in their first cycle of operation. As Figure 4-24 illustrates, a relatively small fraction of the total nuclear generating capacity is comprised of plutonium burners. In the year 2000, seven NEC will be various stages of construction. Although the ultimate, or equilibrium, plutonium burner capacity at these centers will be 136 GW(e), only 61 GW(e) will actually be operating by 2000.

4.5.5 Impact of the FBR on Plutonium Burner Deployment

Projected FBR installed capacity is given in Figure 4-22. Except for the startup of a small demonstration unit in 1983, FBR deployment does not begin until the early 1990s. The installation rate is expected to remain small until 1997, after which the rate increases more rapidly, reach 23 GW(e) installed in year 2000.

In evaluating the LWR generating capacity necessary to provide a fuel supply for the introduction of the breeder, it is necessary to make a number of assumptions regarding the characteristics of these plants. Two model FBRs were selected for this purpose, an early oxide breeder and a fairly optimistic carbide breeder,⁽²⁾ to represent the range of reactor characteristics attainable in the early and later years of plant installations. The important parameters of these reactors are summarized in Table 4-9. The oxide breeder, with a 27.5 year doubling time and a specific inventory of 5.17 kg fissile Pu/Mw(e), was employed through 1996. In 1997 and beyond, the advanced breeder with a 11.4 year doubling time and a 3.0 kg fissile Pu/Mw(e) specific inventory was assumed to be installed to achieve the predicted FBR growth.

The cumulative fissile plutonium required to fuel the FBRs is given in Figure 4-25. Also shown for comparison is the plutonium production from that segment of the installed capacity which is not associated with NECs; plutonium from these reactors could be diverted to fuel the initial cores of FBRs without affecting the deployment of plutonium burners at NECs. As Figure 4-25 illustrates, plutonium production from these non-NEC associated reactors is more than sufficient to support FBR growth through the year 2000; the required plutonium inventory would be available if plutonium from these reactors was diverted to FBR use beginning in about 1990, and about twice the predicted year 2000 FBR capacity could be installed if all plutonium from these reactors was stockpiled for this purpose. However, Figure 4-25 also shows that the annual plutonium demand, represented by the slope of the cumulative requirement curve, is increasing rapidly. Consequently, some time after the year 2000, plutonium production from these reactors will be insufficient to meet FBR demands. Once this occurs, plutonium employed in the NEC plutonium burners would have to be diverted for FBR use.

The plutonium supply and demand picture is summarized in Table 4-10 for the year 2000. The annual plutonium production from non-NEC associated reactors supports the addition of 14 GW(e) of advanced FBR capacity. An additional 11 GW(e) capacity could be realized by halting the installation of new plutonium burners and diverting the plutonium which had been reserved for their initial fueling to FBRs; in this case plutonium would still be available to fuel the NEC plutonium burners which are operating in the year 2000. If still greater quantities of plutonium are required to support FBR deployment, the plutonium burners could be converted to UO₂ operation, assuming their design allows such conversion. This would supply sufficient plutonium to fuel another 16 GW(e)/year of advanced FBR capacity. (The FBR capacity quoted above is in addition to the expansion possible from plutonium breeding in the FBR).

The likely evolution of the NEC after the year 2000 consequently is the gradual reduction in the number of plutonium burners constructed, and the installation of FBRs in their place. Since the specific inventory of the advanced FBR is approximately the same as that of a PWR plutonium burner, a one-for-one replacement with this type of FBR would be possible. If the installation rate of FBRs becomes high compared to its doubling time, or if specific inventory requirements are high, it may prove necessary to convert existing plutonium burners to uranium fueled operation. The mix of FBR and plutonium burners at the NEC will depend, of course, upon the rate of FBR deployment and on the specific characteristic of the FBRs deployed in the Post-2000 period.

There are a number of remaining questions on the impact of FBR deployment which should be addressed:

1. As noted above, plutonium supply is sufficient through the year 2000, but that diversion of plutonium from NECs or conversion of plutonium burners to uranium fueled operation may be necessary in subsequent years, depending on the rate of increase in FBR capacity and on FBR characteristics. Although it is recognized that predictions of nuclear energy growth past the year 2000 are, at best, uncertain, NEC growth scenarios should be extended well past this date to more clearly delineate their development in that period.
2. It appears that the likely evolution of the NEC concept after the late 1990's is to NECs containing a mixture of plutonium burners and FBRs. If the economies of the NEC are to be fully realized with a mixture of LWRs and fast breeders, the reprocessing the mixed oxide fabrication facilities located at the NEC will probably have to serve both reactor types. The feasibility and economics of such dual purpose facilities need to be explored.

References

1. ERDA United States Moderate Growth (70%), February 1975
2. R. L. Hellens, "Future Role of Converter Reactors for Electric Power Generation, Magnitude and Deployment Schedule of Energy Resources Conference, July 21-23, 1975

A number of potential advantages have been identified for Nuclear Energy Centers themselves, and for the concept of clustering plutonium burners at these centers. The advantages include construction economies, more effective land utilization, shared facilities, the elimination of the cost of plutonium and plutonium fuel shipment, and associated cost of safeguards during shipment, and the potential for achieving a higher level of physical security and safeguards at the NEC than would be possible at dispersed sites for the equivalent cost. In addition, there are a number of less salient advantages to employing plutonium burners at NECs, which are briefly mentioned below:

1. Throughout this report, a two-year out-of-reactor time has been assumed between spent fuel discharge and the re-incorporation of the plutonium into the burners as mixed-oxide fuel. During this period, the spent fuel is allowed to cool for approximately 60 days, shipped to the reprocessing plant for reprocessing, the recovered plutonium oxide shipped to the fabrication facility, the fuel fabricated, and the fabricated fuel shipped to the reactor site. The two-year time period is the assumption typically employed in the industry, and reflects the time required for these operations, as well as lag times to accommodate reprocessing and fabrication plant schedules.

Within the context of the NEC, it should be possible to significantly reduce this out-of-reactor time, because of the reduced shipping requirements and because it should be possible to more closely schedule reactor refuelings with reprocessing and fabrication plant operations. Plutonium can also be effectively traded among plants, so that it can be utilized as available, rather than stored until the next refueling of the source plant. It appears feasible to reduce the average out-of-pile time by perhaps as much as one year. A one-year reduction would result in about a 10% reduction in fuel cycle costs, because of lower carrying charges on inventory, and would also allow an increased rate of plutonium burner deployment.

2. We have assumed that the spent fuel discharged from the plutonium burners is completely mixed during reprocessing. This assumption was made since it would be uneconomical to segregate fuel from different generations (plutonium which has been cycled through the reactor various times) during reprocessing of fuel from a single reactor. Segregated reprocessing could be economical at the NEC since the large number of plutonium burners in operation would allow adequate batch sizes for separate reprocessing and fabrication runs for each of the recycle passes.

The advantage of pass segregation lies largely in the ability to retire higher pass material which no longer retains significant value in thermal reactors. This degradation of value results from the buildup of the higher isotopes of plutonium, which is the reason the required plutonium loadings increase with time. The retired plutonium would be stored for FBRs since it retains a significant value for this application.

3. The deployment of FBRs, of course, requires the development of the plutonium fuel cycle. FBRs are likely to benefit naturally from the NEC concept, since their specific inventories and doubling times are sensitive to the out-of-reactor fuel time, and hence it is advantageous to locate FBRs at or near reprocessing the fabrication facilities so as to minimize the period in which plutonium is out of pile. The deployment of plutonium burners at NECs can, consequently, help to pave the way for the introduction of FBRs by an early solution to the problems of NEC development and plutonium supply.

TABLE 4-1

SWU PRICE FINANCIAL ASSUMPTIONS

1. Capacity of new plants - 8.75×10^6 SWUs/yr.
2. Capital Investment and Operating Expenses - Table 4-2.
3. Cost of Power - 16.0 mills/kw hr in 1980 and escalated 6%/yr.
4. Method of Amortization - Sum of the year digits.
5. Amortization period - 20 years private, 15 years gov't. owned.
6. Operating Capacity - 95%.
7. R&D Expenses - 20% of Operating Expenses.
8. Working Capital - -10% of the combined cost of the capital charge, tax, power and operating expenses carried for 2 months
9. Insurance, licensing and other miscellaneous costs - 3% of the total capital investment.
10. Feed and Product Flywheel - 7.0×10^6 SWUs and the required feed carried for 3 months.
11. Preproduction Stockpile - 25×10^6 SWUs in 1980, 21×10^6 SWUs in 1982-1985, declining to zero SWUs in 1990.

Gov't. Owned Gaseous Diffusion	Privately Owned Gaseous Diffusion	Privately Owned Centrifuge
12. Power - 2400 MW	Power - 2400 MW	Power - 240 MW
13. No Cost of Technology	Royalty fee - 3%	Same as private Diffusion plant
14. No Federal Taxes	Federal Tax - 48%	" "
15. No Profits	Return on equity - 15%	" "
16. 100% debt at 5.5% interest	Debt/equity ratio - 50/50 Cost of debt - 8%	" "

TABLE 4-2

ENRICHMENT FACILITY CAPITAL COSTS VS. TIME

8.75 x 10⁶ SWU/Yr Plants

	Gaseous Diffusion CIP/CUP Technology						
	1980	1985	1990	1995	2000	2005	2010
Capital Cost*, \$/SWU	269	376	530	743	1043	1459	2049
Operating Expense**, \$ x 10 ⁶	24	35	51	75	110	161	237
	Gaseous Diffusion Advanced Technology						
Capital Cost*, \$/SWU	230	322	454	637	893	1250	1755
Operating Expense**, \$ x 10 ⁶	24	35	51	75	110	161	237
	Centrifuge First Plant						
Capital Cost*, \$/SWU	328	460	647	908	1273	1782	2502
Operating Expenses**, \$ x 10 ⁶	109	248	365	536	788	1157	1701
	Centrifuge Mature						
Capital Cost*, \$/SWU	217	304	428	600	842	1178	1654
Operating Expense**, \$ x 10 ⁶	123	181	266	391	575	845	1242

* Escalated at 7%/yr from 1975

** Escalated at 8%/yr from 1975

Power Costs vs Time (Escalated 6%/yr from 1980)

Cost, mills/kwhr	16.0	21.44	28.64	38.40	51.36	68.64	91.84
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TABLE 4-3
FUEL CYCLE COST ASSUMPTIONS
1976 DOLLARS

Ore, \$/lb U_3O_8	40,60
U_3O_8 UF_6 Conversion, \$/kg U	5.91
Pu Fabrication, \$/kg HM	225
U Fabrication, \$/kg HM	80
Separative Work, \$/SWU	85
Reprocessing, Shipping, Waste Management, Safeguards	213
Pu Value, \$/gm	Indifference Value
Carrying Charge Rate, %	14, 18
Tails, w/o	.3

TABLE 4-4
EQUIVALENT CORE DIAMETER (IN.) FOR
GIVEN ROD O.D. AND WATER-TO-FUEL RATIO

ROD O.D. (IN.)	Water-to-Fuel Ratio						
	1.6	1.8	2.0	2.2	2.4	2.6	2.8
.3427			127.72	131.43		138.56	
.3820	133.86	138.27	142.53	146.70	150.74	154.68	158.51
.3992			149.06				
.4209		152.49	157.22	161.81	166.27		
.4340			162.15				
.4512	158.35	163.58	168.66	173.58	178.38		

TABLE 4-5
CAPITAL COST COMPONENTS AFFECTED BY
CHANGES IN CORE GEOMETRY

Item	Design Sensitivity
Reactor Internals	Core Diameter
Reactor Vessel	Core Diameter
Cavity	Core Diameter
Containment	Core Diameter
Insulation	Core Diameter
Pressurizer	Core Volume
Safety Injection Tanks	Core Volume
Reactor Coolant Pumps	Pump Power
Primary System Piping	Vessel Flow

TABLE 4-6
CAPITAL COST ASSUMPTIONS

Plant Thermal Capacity, MW(t)	3800
Initial Plant Cost	\$600 x 10 ⁶
Reference Date for Escalation	8 yrs prior to COD
Heat Rate, BTU/kwh	10342
Fraction Paid for Annual Insurance Premiums	.025
Corporate, State, Federal, Income Tax Rate	.50
Plant Lifetime, yrs.	30
Annual Escalation Rate	6%
Annual Discount Rate	8%
Fixed Charge Rate	16%
Capacity Factor	.745
Operating and Maintenance Cost, mills/kwhr	2.0

TABLE 4-7

FUEL BATCH SIZE AND LENGTH OF FUEL RESIDENCE IN CORE
AS A FUNCTION OF MIXED OXIDE ROD DIAMETER

ROD OD (in)	Feed Batch Size No. of Assemblies	Cycles of Core Residence
.3427	88	2.73
.3820	71	3.40
.3992	65	3.72
.4209	58	4.15
.4340	55	4.41
.4512	50	4.78

TABLE 4-8

PLUTONIUM BURNER CHARACTERISTICS

Initial Loading, TeM	102
Diluent	Natural UO ₂
Capacity Factor	.75
Annual Loading, TeM	30
Discharge Exposure, Mwd/TeM	34,200

TABLE 4-9
CHARACTERISTICS OF POSTULATED OXIDE AND
CARBIDE BREEDERS

1000 MW(e) Capacity

	Oxide FBR	Carbide FBR
Specific Power (Mwth/kg)	.7	1.4
Thermal Efficiency	.38	.417
Capacity Factor	.75	.75
Years in Reactor	4.0	2.0
Years in Reprocessing	1.5	1.5
Fifa	.75	.75
Breeding Ratio	1.20	1.30
Gain/Year	.0252	.068
Specific Inventory, by fissile Pu/MW(e)	5.17	3.00
Doubling Time, years	27.5	11.4

TABLE 4-10
PLUTONIUM SUPPLY AND DEMAND
Year 2000

Plutonium Production, MT Pu Fissile	
Plutonium Burners	25.2
Associated UO ₂ Reactors	<u>55.7</u>
Total NEC Production	80.9
Plutonium Burner Reload Requirements, MT Pu Fissile	<u>47.2</u>
Net Pu Available from NEC	33.7
Plutonium Production from Other Reactors, MT Pu Fissile	41.2
Achievable FBR Deployment, GW(e)*	
Plutonium from Other Reactors	14
Net Pu Available from NEC	11
Converting Plutonium Burners to UO ₂ Operation	16

*FBR specific inventory of 3.0 kg fissile Pu/MW(e)

FIGURE 4-1

ESTIMATED UNITED STATES URANIUM RESERVES AND REQUIREMENTS

FIGURE 4-2

FEED AND FRACTION FISSILE PLUTONIUM CONSUMED VS.
W/F RATIO FOR DIFFERENT ROD DIAMETERS

FIGURE 4-3

NUMBER OF FEEDERS VS. PU BURNER W/F RATIO
CONSTANT ROD DIAMETER - LATTICE PITCH VARIED

FIGURE 4-4

FEED AND FRACTION FISSILE PLUTONIUM CONSUMED VS.
W/F RATIO FOR DIFFERENT PELLETT DENSITIES

FIGURE 4-5

NO. U FEEDERS VS. PU BURNER PELLET DENSITY

FIGURE 4-6
 PLUTONIUM VALUE VS. W/F RATIO

FIGURE 4-7

POWER COST VS. WATER-TO-FUEL RATIO
 PLUTONIUM VALUE FIXED AT INDIFFERENCE VALUE FOR PWR REFERENCE DESIGN
 LATTICE PITCH VARIED

FIGURE 4-8

POWER COST VS. WATER-TO-FUEL RATIO
EFFECT OF CARRYING CHARGE RATE

FIGURE 4-9

POWER COST VS. WATER-TO-FUEL RATIO
EFFECT OF URANIUM ORE PRICE

FIGURE 4-10

POWER COST VS. FUEL ROD DIAMETER
 CONSTANT WATER-TO-FUEL RATIO

FIGURE 4-11

CHANGE IN SYSTEM FUEL CYCLE COST VS. PELLET DENSITY

FAB = \$225/KGHM
 AT 95% THEORETICAL DENSITY
 (CORE FAB COST = CONSTANT)
 REPROCESSING = \$213/KGHM
 CCR = 14%

FIGURE 4-12

CHANGE IN PLUTONIUM BURNER FUEL CYCLE COST VS. DENSITY

FIGURE 4-13

SYSTEM ORE AND SWU REQUIREMENTS
VS. PLUTONIUM BURNER W/F RATIO

FIGURE 4-14

SYSTEM ORE AND SWU REQUIREMENTS
VS. MIXED OXIDE
ROD DIAMETER

FIGURE 4-15

SYSTEM ORE AND SWU REQUIREMENTS
VS. MIXED OXIDE
PELLET DENSITY

FIGURE 4-17
FISSILE PLUTONIUM REQUIREMENTS
1300 MW_e APR
75% CAPACITY FACTOR

FIGURE 4-18
FISSILE PLUTONIUM DISCHARGED
1300 MW_e APR
75% CAPACITY FACTOR

FIGURE 4-19
FUEL CYCLE CHARACTERISTICS OF REFERENCE NUCLEAR ENERGY CENTER FOR
PLUTONIUM UTILIZATION

FIGURE 4-19 (Cont'd.)

FUEL CYCLE CHARACTERISTICS OF REFERENCE NUCLEAR ENERGY CENTER
FOR PLUTONIUM UTILIZATION

FIGURE 4-20
EFFECT OF PLUTONIUM STORAGE REQUIREMENT
ON PLUTONIUM BURNER DEPLOYMENT

FIGURE 4-21
EFFECT OF PLUTONIUM STORAGE REQUIREMENT
ON REPROCESSING PLANT DEPLOYMENT

FIGURE 4-22

PROJECTED NUCLEAR GENERATING CAPACITY
LOW - MODERATE GROWTH

FIGURE 4-23

REPROCESSING PLANT AND NUCLEAR ENERGY CENTER DEPLOYMENT

FIGURE 4-24
IMPACT OF A SINGLE REFERENCE
NUCLEAR ENERGY CENTER

FIGURE 4-25
COMPARISON OF AVAILABLE PLUTONIUM SUPPLY
WITH REQUIREMENT FOR FAST BREEDER DEPLOYMENT

The industrial capability for successfully and safely recycling Pu in LWRs has been summarized in the draft "Generic Environmental Statement on Mixed Oxide Fuel (GESMO) in LWR's", issued in August 1974.¹ It was concluded that a great quantity of core behavior data has been developed and that computations on the more complex plutonium-uranium systems can now be performed with an accuracy approximating those for the usual uranium-fueled system. In the limited recycle of plutonium (GESMO addresses plutonium recycle only up to 115% of SGR), it is possible to segregate the plutonium and uranium fuel rods such that there is no substantial change in reactor performance relative to the all-uranium reactor. This means that fuel management schemes can be found for the SGR cores that preserve the basic properties of the all-uranium reactor. This is not entirely true for the all-plutonium reactor since in this limiting case calculations of control rod worth and peak pin power, for example, will be more dependent on the adequacy of the calculational methods and cross section data.

Very little work of relevance has been published since the draft version of GESMO was released. In this section, the topics covered by GESMO are expanded as to their relevance to the design of the APR. This information is supplemented where required using information developed by Combustion Engineering as part of its continuing plutonium utilization research and development programs.

Industrial experience with mixed-oxide-fueled lattices falls into two major categories:

1. Zero power lattice experiments, including both exponential and critical assemblies, which provide a data source for the validation of calculational methods.
2. Prototypical irradiations in power reactors. So far, the mixed-oxide fuel has usually represented only a small fraction of the total fuel loading and thus has only a small effect on the nuclear properties of the reactor core. An early demonstration of mixed-oxide fuel performance was undertaken in the Saxton Project, where approximately one-third of the core was comprised of mixed-oxide fuel. Unfortunately, the size of the reactor made anything more than a rough nuclear design evaluation difficult and expensive.

The review of reactor physics experience presented in the following sections indicates that current design methods, mixed-oxide fuel irradiation performance, and safety and licensing development are adequate to the extent that experimental information is available for verification, and licensing regulations appropriate to mixed-oxide fuel have been formulated. However, additional experience in operating reactors with higher loadings of mixed-oxide fuel is clearly required to confirm this tentative conclusion for the plutonium burner concept. Such experience will be gained through demonstration programs and, once federal regulations permit commercial recycle, by the gradual introduction of mixed-oxide fuel up to, or perhaps beyond, the SGR level in current LWRs.

In reviewing the adequacy of industrial experience for plutonium burner applications, it is useful to note that there are two types of considerations: those dependent on the presence of mixed-oxide fuel itself, independent of the quantity of plutonium present in the reactor, and those which depend on the quantities of plutonium present. Mixed-oxide fuel irradiation performance, the performance of the fuel during transient and accident conditions, and the adequacy of design methods to predict power distributions within the mixed-oxide assemblies are examples of the former. Since these aspects are essentially independent of the fraction of the reactor core which is comprised of mixed-oxide fuel, past experience with demonstration and other special experiments, and the additional confirmation which will be available when SGR begins, are all directly applicable to the plutonium burner concept. Consequently, industrial experience with irradiation and transient fuel performance and local power predictive capability, which has been evaluated with respect to SGR operation and is generally considered adequate for this level of plutonium utilization¹, should also be adequate for the design of LWR plutonium burners.

Industrial experience with respect to plutonium burner characteristics which depend strongly on the quantity of mixed-oxide fuel present in the core must be viewed as less extensive and not completely satisfactory because of the limited mixed-oxide loadings employed to date. The capability of current methodology to predict reactivity coefficients, reactor and system responses under transient and accident conditions, and the ability to predict global power distributions in an all-mixed-oxide-fueled PWR, which could be influenced by the plutonium loading, must be verified for the plutonium loadings of interest. Of these, the ability to predict global power distributions is probably of least concern, although this conclusion should be confirmed with experiment, since the absence of the strong assembly-to-assembly flux gradients present in SGR simplifies theoretical considerations for the analysis of the plutonium burner. Many of the recommendations for further study presented in Section 6.0 are directed toward the development of an experimental base which can be employed to assess the adequacy of current industrial practice for those aspects of the PWR design and safety analyses which are influenced by the core plutonium loading.

5.1 Reactor Physics

5.1.1 Introduction

The area of reactor physics represents one of the more basic areas of concern when considering the design of PWR plutonium burners for nuclear energy center deployment. The technical feasibility of PWR plutonium burners was discussed in Section 3.0. What remains to be answered in the area of reactor physics is the question of the precision of neutronics calculations in plutonium cores relative to uranium cores.

To a first approximation, the required accuracy for neutronics calculations for a plutonium recycle core is the same as that required for a uranium core. The adequacy of any calculational method that is used to design reactor cores must be tested by applying the method to a variety of reactor experiments. To do a satisfactory evaluation of methods, one therefore needs an extensive body of experimental data. When considering plutonium recycle in reactors presently fueled with uranium, one cannot assume that calculational methods that have been tested

and normalized for the uranium cores will be adequate for plutonium recycle cores. There are enough differences neutronicly between uranium and plutonium that one needs to demonstrate the reliability of calculational methods by testing them against the available experimental data for plutonium lattices. In this section, we review the available experimental data, concentrating on data applicable to PWRs.

Significant experimental programs concerned with plutonium recycle physics have been carried out at many laboratories throughout the world. The USAEC has sponsored a program, the Plutonium Utilization Program (PUP)¹⁹, which began in 1957 and terminated in June 1972. The primary objective of this program was to provide a firm technical base for the safe, economical use of plutonium in commercial thermal power reactors. An important segment of this program dealt with reactor physics problems encountered in using plutonium fuel in thermal reactors. Extensive experimental and analytic physics studies were carried out.

Several industrial firms within the U.S. have also carried out experimental programs investigating the physics problems of plutonium recycle. Programs have been carried out by the Westinghouse Electric Corporation,^{2,3} General Electric Company,⁴ the Babcock and Wilcox Company,^{5,6} Gulf United Nuclear Fuels Corporation,^{7,8} and Gulf General Atomic.^{9,10} Reviews of the U.S. experimental program are given in Refs. 1, 11, 12, 13 and 14.

Significant experimental programs have also been carried out in several foreign countries, notably Belgium¹⁵ and Italy.¹⁶ The British¹⁷ have done extensive work with plutonium lattices, although much of their work has not been aimed specifically at plutonium utilization, but rather at investigating the problems of uranium fuel after significant amounts of plutonium have built up in these fuels. For information concerning experimental work that is not specifically mentioned here, we refer the reader to reports of recent international conferences and panels,^{12, 18} GESMO,¹ and the many published reviews of experimental programs in this area.^{11, 13, 14}

In this section we summarize the major programs mentioned above. Our intent is not to give a detailed description of all the work that has been done along these lines, but rather to review the major programs applicable to the physics design of mixed-oxide-fueled PWRs.

Lattice studies in either subcritical or low power critical facilities have comprised the major part of the available experimental physics data. Some irradiation studies, in which plutonium-enriched fuels have been irradiated under actual power reactor conditions, have also been conducted. The data in this area, however, are not nearly as extensive as the data in the area of lattice studies.

5.1.2 Physics Experience With Mixed-Oxide-Fuel in Pressurized Water Reactors

A number of irradiation experiments have been initiated by both U.S. and European reactor vendors to gain experience with mixed-oxide fuel in power reactors. Although such experiments have provided experience in mixed-oxide-fuel irradiation behavior, they are of limited value for judging the accuracy of reactor physics calculational methods as applied to the design of all plutonium-fueled reactors. The number of mixed-oxide assemblies is small and hence only a very small effect upon reactor core behavior is seen. Moreover, unless the mixed-oxide assemblies are specifically instrumented, it is difficult to differentiate the effect of the mixed-oxide assemblies on core power distribution from the uncertainties which normally arise in the calculation of global reactor properties.

Industrial experience with mixed-oxide fuel in pressurized water reactors is comprised mainly of results obtained from the following programs:

Saxton Plutonium Project (Refs. 20-25)

An irradiation program was conducted in the Saxton reactor under the sponsorship of USAEC, Euratom, the Westinghouse Electric Corporation and the Saxton Nuclear Experimental Corporation. The basic objective of this experiment was to develop information concerning the utilization of plutonium-enriched fuel in a typical pressured water reactor system through design, fabrication, and operation of a partial core of mixed-oxide fuel in the Saxton reactor. The total Saxton Core-II fuel loading contained nine mixed-oxide fuel assemblies; two contained vibrationally compacted fuel and seven contained pelletized fuel. There were a total of 638 mixed-oxide rods and the peak burnup attained was about 29,000 MWD/TeM. The mixed-oxide fuel contained UO_2 - 6.6 wt% PuO_2 . A series of physical, metallurgical, nuclear, and fuel systems analysis examinations were carried out. At Core-II startup, zero and low power physics measurements were made to verify reactivity, control rod and boron worth, flux and power distributions, and temperature, pressure, and power coefficients. Similar data were obtained during periodic shutdowns and at the end of core operation to determine the change in characteristics. Using flux wires and fission detectors, power distribution measurements were made periodically during operation. The critical boron concentration was monitored throughout core life. A number of irradiated fuel rods were gamma-scanned after shutdown to provide a power burnup map in the plutonium region. Mass spectrometric and radiochemical analyses were performed to measure the isotopic content of these fuel samples. The radial distribution of isotopes was measured for one of the irradiated pellets. Isotopic data obtained from the destructive analysis of the mixed-oxide rods irradiated in Core-II of Saxton have been used at BNWL to derive ratios of effective cross sections of the heavy isotopes.

As part of a continuing program to evaluate mixed-oxide fuel at higher burnups, selected rods from Core-II were reinserted into the Saxton reactor and were irradiated as part of Core-III. Approximately 250 fuel rods from Core-II were assembled into seven Core-III assemblies. The objectives of this extended burnup test (Core-III) were to (a) demonstrate the performance of mixed-oxide fuel at linear power and burnup levels consistent with modern PWR technology and (b) to obtain design information on the depletion and transuranic isotope generation characteristics of plutonium fuel at higher burnup.

EEL-Westinghouse Plutonium Recycle Demonstration Program (Refs. 26-29)

Under Phase I of this Edison Electric Institute program, Westinghouse performed a study of the economics of plutonium recycle and the characteristics of mixed-oxide-fueled PWRs. Phase II of the program included an analytical study of SGR plutonium recycle in a 1000 MWe reactor together with the irradiation of four mixed-oxide assemblies in Cycles 2 and 3 of the San Onofre reactor, a 450 MWe PWR located in Southern California. Zircaloy cladding was employed for the mixed-oxide fuel, while the remainder of the core employed stainless steel clad fuel. The assemblies were irradiated to a maximum rod average burnup of 21,050 Mwd/TeM without anomalous conditions developing in any of the rods. The location of the mixed-oxide demonstration assemblies in San Onofre Cycles 2 and 3 and the basic design of the mixed-oxide assemblies is shown in Figures 5-1 through 5-3.

Kraftwerk Union Mixed-Oxide-Fuel Demonstration

This program has, to date, involved the irradiation of twenty-one mixed-oxide-fuel assemblies in the 300 MWe Obrigheim PWR (KWO) located on the Neckar River near Heidelberg, Federal Republic of Germany. The program commenced with the insertion of a single mixed-oxide assembly in the fourth reactor cycle and was followed by eight additional demonstration assemblies in Cycle 5. The mixed-oxide assembly loaded in Cycle 4, and 7 of the eight fresh mixed-oxide assemblies inserted in Cycle 5 were discharged at the end of this cycle and four fresh mixed-oxide assemblies loaded. At the end of Cycle 6, seven of the mixed-oxide assemblies discharged at the end of Cycle 5 were reinserted along with eight fresh assemblies, bringing the mixed-oxide loading in Cycle 7 to twenty assemblies--approximately 50% of the SGR loading for this plant. The location of the mixed-oxide demonstration assemblies in KWO Cycles 4-7 and the basic design of the mixed-oxide assemblies is shown in Figures 5-4 through 5-8.

Analysis of Power Reactor Measurements

The influence of increasing mixed-oxide fuel loadings on the ability to predict reactivity and power distribution in pressurized water reactors has been examined in some detail using measurements obtained during San Onofre Cycle 2 and KWO Cycles 4 and 5. The comparisons were made using calculational techniques identical to those normally applied in the design of all uranium-fueled cycles and thus provide a measure of current capabilities with regard to the analysis of mixed-oxide-fueled systems.

A comparison of calculated versus measured core reactivity over several refueling cycles for the San Onofre and KWO reactors is provided in Figures 5-9 and 5-10. The calculations were carried out at measured critical conditions so that any deviation from criticality provides a direct measure of the calculational uncertainty. As there appears to be no discernable trend with increased mixed-oxide loading, one is led to conclude that current calculational techniques adequately predict core reactivity in systems containing a relatively small loading of mixed-oxide fuel.

Information concerning the global power distribution in the KWO and San Onofre cores can be inferred from in-core instrument signals. A comparison of calculated versus measured instrument signals for San Onofre Cycle 2 and KWO Cycles 4 and 5 has been made at selected burnup points with the results shown in Figures 5-11 through 5-19. The agreement in global power distribution shows no noticeable deterioration with increased mixed-oxide loading, and, as shown in Figure 5-12, the capability to predict axial power shape within a mixed-oxide assembly also appears quite satisfactory. One is again led to the conclusion that current calculational techniques are adequate for the limited range of mixed-oxide loadings examined.

While the San Onofre and KWO results provide a valuable early test of the ability of current methods to predict global power distribution and reactivity in PWR cores containing mixed-oxide fuel, it should be recognized that the number of mixed-oxide assemblies present is substantially smaller than would be present during SGR operation. It is a relatively large extrapolation to assume that these conclusions will apply to cores having mixed-oxide fuel loadings approaching the APR level.

5.1.3 Zero Power Lattice Experiments

As indicated in the preceding section, the available physics data on mixed-oxide-fueled power reactors is not very extensive. Consequently, one must rely heavily on the results of zero power critical experiments to supply the measure of calculational uncertainties associated with mixed-oxide systems. A large number of the early mixed-oxide lattice experiments were performed to aid in the evaluation of reactor physics design methods. These have encompassed both uniform lattices and lattices containing simulated heterogeneities more typical of actual LWR fuel assemblies. Unfortunately, the majority of these experiments are difficult to analyze with confidence due to their small size and high neutron leakage. An extensive survey of the experimental data available from these experiments is presented in Reference 14. The types of experimental data available and the organizations which have performed these experiments are summarized in Table 5-1.

Results from the analyses of a large number of mixed-oxide lattice experiments are reported in the open literature. These experiments are for the most part small in size, and hence subject to the introduction of considerable uncertainties in their analysis, due to the careful attention that must be paid to neutron leakage effects. In reviewing the literature, one must, in addition, take care to differentiate those organizations or individuals using unadjusted nuclear data in their analyses and those which have used semi-empirical data adjusted to improve agreement between calculations and measurements made on a particular system (cf. References 26 and 30 for examples of these procedures.)

A representative analysis of plutonium-fueled lattice experiments performed at room temperature in a light-water moderator using unadjusted data is presented by Liikala et al. in Reference 31. Uncertainties in calculated k_{eff} of ± 1.3 and $\pm 1.7\%$ $\Delta K/K$ were found for UO_2 and mixed-oxide-fueled lattices, respectively. The authors concluded that the principal deficiencies in the analysis resided in the calculation of U-238 resonance absorption and in the treatment of neutron thermalization in the plutonium-bearing lattices. The former problem is widely recognized within the nuclear industry and has been resolved for uranium lattices through use of the empirical correlations based upon the resonance integrals measured by Hellstrand (Reference 32). The source of the basic cross section data used in this study was the Battelle Northwest Master Library which contains cross section data similar to ENDF/B-I.

An analysis of the ESADA lattice experiments using ENDF/B-IV data was recently completed by Brookhaven National Laboratory and is reported in Reference 33. Uncertainties in calculated k_{eff} of $\pm 0.36\%$ $\Delta K/K$ are reported for water-to-fuel ratios ranging from 1.1 to 8.3, with a maximum discrepancy of only 0.49% $\Delta K/K$. The results of this analysis are quite similar to those obtained by Goldstein and Windsor (Reference 34) at Combustion Engineering using ENDF/B-II data for the plutonium isotopes.

Neutron kinetics parameters for mixed-oxide lattices have been inferred using noise analysis techniques (cf. References 21 and 22). The measurements were obtained in uniform lattices over a range of mixed-oxide loadings and in a 4 w/o Pu mixed-oxide lattice with a varying number of water holes. Calculated values of β_{eff}/λ^* were found to agree with the measurement to within $\pm 5\%$.

To supplement the experimental information available in the open literature, Combustion Engineering has undertaken proprietary experimental programs. The information obtained from these programs has been used extensively as a basis for the validation of physics calculational techniques. Other reactor manufacturers and fuel vendors have also followed this path. As a result, a substantial body of information now exists which does not appear in the open literature. The program undertaken by Combustion Engineering has emphasized measurements on prototypical mixed-oxide assembly designs to provide a data base for validating physics design methods. Major items of interest include the ability to predict reactivity, power distribution, control rod worth, boron worth, and reactivity coefficients in mixed-oxide-fueled lattices.

Room temperature critical experiments were performed for C-E and several other U.S. nuclear vendors under contract with Battelle Northwest Laboratories, to provide a data base for use in the development and evaluation of physics calculational techniques. The lattices examined included both all UO_2 and all mixed-oxide assembly configurations. The mixed-oxide configuration examined in the C-E proprietary experiments included a mockup of an all-mixed oxide fuel assembly, an assembly consisting of a central zone of mixed-oxide fuel surrounded by UO_2 fuel rods. The fuel assembly types simulated in these experiments are illustrated in Figure 5-20. Measurements for many lattices were performed both with and without inserted control rods to quantify the effect of mixed-oxide fuel on control rod worth. Pinwise fission rate distribution and axial buckling measurements were also performed.

In addition to the experimental program carried out under contract with Battelle Northwest Laboratories, a series of lattice experiments at elevated temperatures was conducted in the Kritz facility of A. B. Atomenergi in Studsvik, Sweden. These experiments simulated all-mixed-oxide fuel assembly designs similar to those examined in the C-E/BNWL experiments and were designed to extend the base of experimental information to moderator temperatures and mixed-oxide-fuel enrichments more characteristic of those that would be encountered in a mixed-oxide-fueled PWR. The experimental program was divided into two phases. The initial phase employed relatively low enrichment fuel, 1.38 w/o fissile Pu (the C-E/BNWL experiment employed 1.52 w/o fissile Pu) and uranium fuel having 1.86 w/o U-235. The second phase of the program endeavored to expand the experimental data base into the range of plutonium enrichments which would be encountered in pressurized water reactors. For these experiments, the simulated mixed-oxide assembly employed two zones of enrichment, 2.0 and 3.2 w/o fissile plutonium. Power distribution, boron worth, and control rod worth measurements were obtained over a temperature range from 68 to 473°F (293-518°K).

Detailed analyses of the C-E/BNWL and Kritz lattice experiments have been made using calculational techniques identical to those normally applied in the design of uranium-fueled reactors. The cross section data employed is based largely upon ENDF/B-II, with U-238 resonance capture based on the measurements of Hellstrand (cf. Reference 32). Corrections were made to the Pu-239 and Pu-240 resonance integrals and thermal cross sections to account for self-shielding effects in the PuO_2 grains (Ref. 34).

A tabulation of calculated reactivities for each of the mixed-oxide assembly configurations examined and a UO_2 reference lattice is presented in Tables 5-2 and 5-3 for the C-E/BNWL and Kritz lattice experiments, respectively. Over the series of experiments, which were performed at rather low plutonium loadings, no clear trends emerge. Uncertainties in calculated k_{eff} of ± 0.64 and $\pm 0.31\%$ $\Delta K/K$ are observed for the mixed-oxide and all-uranium assembly configurations, respectively, with some slight hint of an increasing reactivity bias with plutonium enrichment in the Kritz lattices. The latter trend is consistent with known biases in the ENDF/B-II plutonium cross sections used for these analyses.

The ability to calculate the pin-by-pin power distribution within a mixed-oxide assembly and at interfaces between mixed-oxide and uranium fuel requires independent confirmation since this fine structure cannot be measured in an operating reactor. Plutonium fuel, relative to U-235, exhibits a quite different spectrum dependence of the fission cross section, so that good agreement with calculations at room temperature does not ensure similar agreement at the elevated moderator temperatures encountered in an operating PWR. With this in mind, C-E has had performed two series of experiments in the Kritz facility at Studsvik in which power distributions have been measured at temperatures between 68°F and 473°F ($273\text{--}518^\circ\text{K}$). As described earlier, the initial phase of this program employed relatively low enrichment fuel, with a central region of 1.38 w/o fissile Pu in depleted uranium surrounded by 1.86 w/o uranium fuel. The deviations between measurements and calculations showed no trend with moderator temperature and is within the experimental uncertainty of $\pm 2\%$. For all of the fuel rods measured, the standard deviation is 1.86%, roughly comparable to the agreement observed in similar UO_2 critical experiments. The distribution of deviations between measured and calculated pin powers in the Phase I lattice experiments is shown in Figure 5-21.

The Kritz experiments were extended into the range of fuel enrichments pertinent to pressurized water reactors in Phase II of the program. Power distribution measurements were obtained on a simulated assembly containing enrichment zones of 2.0 and 3.2 w/o fissile plutonium surrounded by 3.1 w/o UO_2 fuel. Results from the analysis of these measurements at two moderator temperatures are shown in Figures 5-22 and 5-23. A similar comparison for a reference all-uranium lattice is shown in Figure 5-24. Again the agreement between measurement and calculation is similar for the mixed-oxide and UO_2 assemblies; no discernable trend with moderator temperature is observed.

5.1.4 Conclusions

As a result of the numerous experimental and demonstration programs that have been, or are being conducted, the technical feasibility of plutonium recycle at the SGR level has been adequately demonstrated. To the extent that the physics design methods have been verified by these programs, there is reason to believe that the physics calculational uncertainties associated with plutonium utilization at the APR level will not be significantly greater than those associated with SGR operation.

Although the review of currently available experimental information indicates no reason to believe that current design methods are inadequate for the prediction of plutonium burner core parameters, confirmation of this conclusion must await experimental information from systems containing more typical mixed-oxide loadings. Specific recommendations for additional experimental work are presented in Section 6.2.

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5.2 Fuel Irradiation Performance

5.2.1 Introduction (Refs. 1, 2)

Numerous experimental demonstrations of mixed oxide irradiation performance were and are currently being conducted in this country and in Europe. The purpose of these programs is to demonstrate that: 1) Mixed oxide fuel has a performance capability comparable to UO_2 fuels in LWR applications; 2) There exists a degree of design flexibility in current LWRs sufficient to accommodate a mix of UO_2 and mixed oxide fuel assemblies; 3) Adequate materials and fabrication technology exists to fabricate mixed oxide fuel on a commercial scale; and 4) Fuel cycle economics for mixed oxide fuels are comparable to UO_2 fuel.

Irradiation performance data to date indicates that Zircaloy clad LWR mixed oxide fuel does satisfy the above criteria. There do not appear to be any intrinsic performance limitations that result from the additions of PuO_2 to UO_2 . There are, however, important plutonium characteristics which must be factored into the design and fabrication techniques for mixed oxide fuel. These considerations include the migration behavior of the plutonium fuel and its fission products in the fuel matrix, the potential for locally higher fission densities due to inhomogeneous blending of UO_2 and PuO_2 powders, and the slightly higher creep rate and lower melting point for the mixed oxide fuel. Tables 5-4 and 5-5 list the various experimental demonstration programs completed or in progress. The results obtained from each of these programs is discussed in the following sections.

5.2.2 Battelle Northwest Laboratories PRTR Program (Refs. 3, 4)

The plutonium recycle demonstration program in the Plutonium Recycle Test Reactor (PRTR) was initiated at BNWL in 1956 and continued until the late 1960's. A total of some 2500 Vipac and Swage rods and over 1700 pelletized rods were irradiated to burnups in the 13,000 to 18,000 MWD/TeM range at linear heat ratings in the range of 9.1 to 19.8 kw/ft (300-650 watts/cm). Although Vipac and Swaged fuel is no longer of interest to the industry, the irradiation data obtained along with the pelletized rod data have provided a basis for the design of many follow-on demonstration experiments. These early irradiations established the basic irradiation characteristics of mixed oxide fuels and showed that mixed oxide fuel performance under irradiation is very similar to UO_2 fuel performance. They also established that fuel formed by mechanically blending UO_2 and PuO_2 powder performs as well as fuel formed by coprecipitating the uranium and plutonium.

5.2.3 Saxton Plutonium Project (Refs. 5-7)

The Saxton plutonium project, conducted for the AEC by Westinghouse had the following objectives: 1) To obtain a pilot scale evaluation of mixed oxide fuel performance in a PWR environment; 2) to compare the performance of pelletized and Vipac mixed oxide fuel; and 3) To obtain nuclear data of interest to plutonium recycling such as depletion and generation of transuranic isotopes.

A total of nine assemblies (9 x 9 rod arrays) containing 6.6 w/o plutonium fuel were inserted into the central region of Saxton Core II. The fuel was prepared by a mechanical mixing of the oxides; 148 rods contained 84% theoretical density vibratory compacted fuel and 490 rods containing 94% theoretical density pelletized fuel. The majority of the fuel was zircaloy clad, with a number of stainless steel clad

rods included for comparison; all fuel rods were helium filled and unpressurized. These rods were irradiated to burnups greater than 25,000 MWD/TeM at maximum linear heat ratings of 16 kw/ft (524.9 w/cm) and a maximum fuel centerline temperature of 2200°C (2473°K).

The Saxton Core II program also included a two rod overpower test. Two rods which had operated at peak linear power ratings of 12 kw/ft (393.7 w/cm) and achieved respective burnups of approximately 13,000 and 27,000 MWD/TeM, were escalated to a power level of 15.5-16 kw/ft (508.5-524.9 w/cm) to precondition the fuel and establish a base point for evaluation. The rods were then removed, subjected to a visual and profilometer examination and reinserted into the reactor. They were stabilized at a power level of 15.9 kw/ft (521.7 w/cm) for six days and then ramped to 18.7 kw/ft (613.5 w/cm) over a two minute escalation and then maintained at this power level for 5 days. The reactor was then shut down and then brought back up to power for a repeat ramp test, with 15 days of operation at the 18.7 kw/ft (613.5 w/cm) level. The only effects produced by the overpower operation were an insignificant increase in rod length (<0.1%), an increase in fission gas release caused by the higher fuel temperature and some limited columnar grain growth and incipient central void formation which was also the result of the higher operating temperature.

The mixed oxide fuel performed well in the Saxton II core. There were no rod failures. One rod exhibited local massive hydriding and anomalous dimensions but did not fail. The source of the excess hydrogen was attributed to strong contamination introduced during manufacturing. On the remaining fuel, hydrogen uptake in the zircaloy cladding was less than 50 ppm. Clad ductility at the end of irradiation was fairly high (72.5%). Fission gas release averaged around thirty percent.

The successful demonstration of mixed oxide fuel performance in Saxton Core II led to the Core III "Extended Burnup Test". Approximately 250 rods from Saxton Core II were reconstituted into seven assemblies and irradiated in Core III to peak burnups up to 51,000 MWD/TeM. The objectives of this extended test were to demonstrate mixed oxide fuel capability to high burnups over a wide spectrum of power levels and obtain depletion characteristics and transuranic isotope generation data for high burnup.

Post irradiation evaluation showed that 37 of the 250 rods had failed. The failures were caused primarily by clad corrosion. Crud deposition caused large localized increases in clad temperature and the consequent corrosion. In other respects the fuel pins remained dimensionally stable, fission gas release ranged from 32 to 37%, and some fuel-clad bonding was observed.

In summary the Saxton experiments demonstrated the performance of mixed oxide fuel under typical PWR operating conditions to be equal to that of UO₂ fuel.

5.2.4 The EEI-Westinghouse Plutonium Recycle Demonstration Program (Refs. 8-10)

The purpose of the EEI-Westinghouse plutonium recycle demonstration program was to license, operate, and evaluate a representative number of plutonium fuel rods in a large PWR under actual utility operating conditions. Four assemblies of 180 pins each were fabricated and inserted into San Onofre Unit 1 at the start of the second reactor cycle. The mixed oxide fuel rods were unpressurized and zircaloy clad, and contained pelletized fuel of 91% theoretical density. Selected rods at the end of cycle 2 and cycle 3 were removed and shipped to BMI for hot cell examination. The individual rods operated at linear heat ratings of 5.0 to 6.5 kw/ft (164-213 w/cm) and achieved peak burnups ranging from 21,000 to 24,000 MWD/TeM. At the completion of the third cycle visual and profilometry inspection revealed no evidence of clad or fuel deteriorations. A diametral creep down in the hottest portion of the rods of about 5-6 mils was observed and fuel rod length changes of .22 to .36% occurred.

In summary, the results of this program confirmed the results obtained in the Saxton irradiation, i.e., that mixed oxide fuel performance under PWR operating conditions is similar to that seen for uranium fuel.

5.2.5 EEI-GE BWR Plutonium Recycle Demonstration (Refs. 11, 12)

A total of 236 mixed oxide fuel rods were irradiated in the Consumers Power Company Big Rock Point reactor. Thirty-two of the fuel rods were irradiated to peak pellet burnups ranging from 25,000 - 30,000 MWD/TeM at linear heat ratings ranging from 9.0 to 14.7 kw/ft (295.3 - 482.3 w/cm). Three geometric pellet configurations were used, solid, dished, and annular. Post irradiation examination of eleven select rods indicates that all of the rods operated in a manner similar to UO_2 rods. Fuel rod growth, fuel densification, and fuel swelling rates were consistent with those predicted for UO_2 fuel. No pellet-clad mechanical interaction was observed.

Examination of the annular pellet rods showed no evidence of axial fuel redistributions caused by vapor transport and no evidence of gross fuel movement into and down the center annulus. This information is of significance with regard to the safety and performance of annular fuel pellets. The relatively low linear heat ratings at which the irradiation was conducted preclude any firm conclusions with respect to fuel migration. Axial migration of both ruthenium and cesium to the top and bottom of the fuel column was observed in the fuel rods containing the annular pellets.

5.2.6 Dresden I Plutonium Recycle Program (Refs. 13 - 18)

Eleven assemblies containing nine mixed oxide rods each, with a plutonium content of 2.3% were irradiated during cycles seven and eight in the Dresden I BWR. At the end of cycle eight, six of the eleven demonstration assemblies contained failed rods. The fuel failures with one exception were attributed to end plug weld fracture. This failure mode was found to be common to both the mixed oxide and UO_2 rods in the affected assemblies and has been attributed to manufacturing defects. Fuel exposures achieved by the end of cycle 8 were approximately 16,500 MWD/TeM assembly average and 22,000 MWD/TeM peak pellet. The assemblies operated at relatively low power densities of 3-4 kw/ft (98.4-131.2 w/cm) average and 9.2 kw/ft (301.8 w/cm) peak pellet.

The five unfailed assemblies were reinserted in cycle 9 and assuming no additional fuel failures develop, will be carried over into cycle 10. The final post irradiation work is scheduled for 1977.

5.2.7 Kraftwerk Union Mixed Oxide Fuel Demonstration (Refs. 19-22)

The first mixed oxide assembly was inserted into the Obrigheim reactor in 1972 at the beginning of the fourth reactor cycle. One fuel pin failed during the first cycle of operation and several more during the second cycle. Preliminary analysis indicates that hydriding caused by high fuel moisture content caused the failures. Irradiation of this assembly was terminated at the end of cycle five. Eight additional mixed oxide assemblies were inserted in 1973, four in 1974 and eight in 1975. The present core makeup (cycle 7) consists of eight fresh mixed oxide assemblies, eleven assemblies which have received one cycle of irradiation and one assembly which has received two cycles of irradiation. No additional failures have been reported to date.

Sixteen demonstration assemblies were inserted into the Gundremmingen Reactor in 1974, followed by twenty-four in 1975. These assemblies contain 36 mixed oxide rods each, having a fissile plutonium enrichment of 2.03 w/o and are designed to operate at 15.8 kw/ft (518.4 w/cm) to a burnup of 22,000 MWD/TeM. Performance to date is reported as excellent with no observed fuel failures.

The Kahl reactor was loaded with a single assembly containing nine mixed oxide rods in 1966. A burnup of 16,000 MWD/TeM was achieved. Since then twenty-three additional assemblies containing 36 mixed oxide pins each have been inserted. These assemblies were designed to operate at a low power density, 4 kw/ft (131.2 w/cm), to a burnup of 15,000 MWD/TeM.

Four mixed oxide assemblies were inserted into the MZFR in 1972. No fuel failures have been reported. In addition a two rod, mixed oxide fuel densification test was also performed. One rod contained 90% theoretical density fuel pellets and the second 93% theoretical density pellets. These rods were removed after 4000 MWD/TeM burnup. Fuel densification was found to have occurred and was of a slightly smaller magnitude than has been observed for UO_2 rods of the same initial density.

Three instrumented mixed oxide rods were inserted into the Halden reactor (Experiment IFA-427) to measure fuel centerline temperature, internal gas pressure and changes in fuel stack length. These rods, still in operation, have achieved 7000 MWD/TeM burnup at a 10 kw/ft (328 w/cm) power level. An additional four rods (Experiment IFA-428) were inserted to determine the effect of fuel clad gap and power level on operating parameters and fuel centerline temperature. This fuel was discharged after a burnup of 2000 MWD/TeM; no further data is available.

5.2.8 The Belgo Nucleaire Plutonium Recycle Program (Refs. 23 - 26)

Irradiation of Belgo Nucleaire fabricated mixed oxide fuel has been conducted in the BR-3, Sena, Dodeward and Garigliano reactors. A description of each program follows.

BR-3 Irradiations

Two assemblies comprising twelve and nineteen stainless steel clad vipac and pelletized mixed oxide fuel rods were irradiated in the BR-3 reactor. These assemblies achieved burnups of 6 and 37,000 MWD/TeM respectively. Post irradiation examination revealed slight diameter increase and ridging of the fuel rods containing pelletized fuel.

Two mixed oxide fuel assemblies each containing thirty-six zircaloy clad rods were irradiated in the cores 2 and 3A to peak burnups of 36,000 MWD/TeM. Sipping tests at end of cycle indicated that both assemblies contained failed fuel. Results of a detailed post irradiation examination are not yet available.

Thirteen zircaloy clad assemblies, containing 240 mixed oxide rods and nine assemblies containing 212 mixed oxide rods were inserted in Core 3A and operated at a peak linear heat rate of 18.3 kw/ft (600 w/cm). Mean burnups ranged from 7,000 to 18,000 MWD/TeM with peak rod and peak pellet burnups of 23,700 and 30,100 MWD/TeM, respectively. Sipping tests at end of cycle revealed no fuel failures. All of these assemblies have been reinserted into core 3B for further irradiation.

Sena Irradiation

Four assemblies of 208 fuel rods each, clad with stainless steel, have been irradiated in the Sena reactor. Fuel performance data is not yet available.

Dodeward Irradiation

Two assemblies of 30 and 57 plutonium rods respectively were irradiated at a peak linear heat rate of 15.2 kw/ft (500 w/cm) to an average burnup of 20,000 MWD/TeM. Sipping tests at end of cycle showed that both assemblies contained failed rods. A detailed post irradiation examination is planned.

Garigliano Irradiation

Four assemblies of 204 rods each have been irradiated to 11,000 MWD/TeM and removed from the reactor for examination. No defects or failed fuel rods were found and the assemblies were reinserted for further irradiation.

5.2.9 Conclusions

From the collected results available from the many irradiation programs carried out worldwide the following general conclusions can be drawn.

First, the irradiation behavior of mixed oxide fuels appears to be very similar to that observed for UO_2 fuels. Second, when fuel failures have been observed, the failure rates and failure modes were the same as that observed for UO_2 fuel.

It would appear that the irradiation performance of mixed oxide fuels presents no special problem relative to its use in pressurized water reactors.

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5.3 Fuel and Materials Development

5.3.1 Introduction (Refs. 1, 5, and 6)

The APR lattice optimization studies discussed in Sections 4.3 and 4.4 show that as the water-to-fuel ratio is increased, the required core fissile inventory and fuel cycle costs decrease. An increase in the lattice water to fuel ratio may be achieved by the following means: 1) Increasing the lattice pitch; 2) Reducing fuel pellet density or diameter; 3) Replacement of solid fuel pellets with annular pellets; 4) Replacement of a portion of the uranium oxide by diluent material.

Of the four alternatives, replacing a portion of the uranium oxide by a diluent material appears the most attractive. Increasing the lattice spacing requires a larger pressure vessel or substantial modification to the reactor internals, with a concomitant increase in plant capital cost. Reductions in pellet density or pin diameter are accompanied by a degradation of fuel thermal performance. Replacement of solid pellets by annular pellets raises questions regarding fuel relocation which remain to be resolved, although some encouraging data in this regard has been reported (cf. Section 5.2).

Replacement of a portion of the uranium oxide by a diluent material such as cerium oxide appears to be a promising alternative; other possible diluent materials are zirconium and yttrium oxides. The use of diluent materials as a means of increasing water to fuel ratio in a plutonium burning PWR was briefly discussed in Section 4.1. The results presented in Section 4.4 indicate the possibility of a significant potential savings in fuel cycle costs if this option were chosen. This section presents a review of current industrial experience with cerium oxide as a diluent material in LWR fuel. Recommendations for further study in this area are presented in Section 6.3.

5.3.2 Characteristics of Cerium Oxide

Cerium oxide has a number of attractive properties as a diluent in LWR fuel. These are summarized as follows: 1) Cerium has a very low neutron capture cross section; 2) Mixed oxide powders containing cerium oxide possess a low sintering temperature; 3) Cerium oxide powder is readily available in high purity form at reasonable cost; 4) Cerium is present in spent fuel as a fission product, hence, introducing cerium oxide into the fuel as a diluent should not require substantial modification to the chemical processes used for fuel reprocessing; 5) Cerium oxide exhibits good irradiation stability.

5.3.3 Characteristics of $\text{CeO}_2\text{-UO}_2$ Systems (Refs. 2, 4, and 7)

The microstructure of sintered CeO_2 and $\text{CeO}_2\text{-UO}_2$ powders depends on the sintering atmosphere, sintering time, sintering temperature, and post-sintering heat treatment. Cerium oxide sintered in a hydrogen atmosphere produces a Widmanstätten microstructure as well as a discontinuous grain boundary phase. Addition of approximately twenty atom percent UO_2 to the cerium oxide eliminates both of these problems and produces a single phase solid solution. Also with increasing UO_2 fractions, the sintered material tends to become less brittle and less prone to cracking.

Sintering of cerium oxide powders containing up to thirty atom percent UO_2 produces a single phase, face centered cubic crystalline structure. At compositions approaching fifty atom percent UO_2 , a microstructure containing two distinct face centered cubic phases is produced. This is due to oxygen pick-up and the formation of the ternary system $\text{UO}_2\text{-CeO}_2\text{-O}$. Heat treatment in argon removes the excess oxygen and the sintered material returns to a single phase, face centered cubic structure. Typical sintering temperatures range from 2375° to 2825°F ($1575\text{-}1825^\circ\text{K}$).

Limited data is also available for $\text{CeO}_2\text{-PuO}_2$ systems for compositions containing twenty atom percent PuO_2 . Sintering in a carbon dioxide atmosphere at temperature between 2875°F and 2970°F ($1850\text{-}1900^\circ\text{K}$) produces a stable, single phase solid solution.

No data is currently available on the ternary $\text{UO}_2\text{-PuO}_2\text{-CeO}_2$ system. However, on the basis of the information developed regarding the individual $(\text{CeU})\text{O}_2$ and $(\text{CePu})\text{O}_2$ systems, small additions (~ 5 w/o) of PuO_2 to $\text{CeO}_2\text{-UO}_2$ are expected to result in a single phase face centered cubic microstructure with no appreciable degradation of properties. Apart from the stable single phase microstructures which are obtained, the $\text{CeO}_2\text{-UO}_2\text{-PuO}_2$ system has two major fabrication advantages; hydrogen is not required as the sintering furnace atmosphere and the required sintering temperatures are quite low, 2730 to 2875°F ($1775\text{-}1825^\circ\text{K}$). This low sintering temperature precludes the possibility of plutonium vaporization during the sintering cycle.

The thermal conductivity, melting point, thermal expansion coefficient, and corrosion resistance reported in literature for $\text{CeO}_2\text{-UO}_2$ fuel compositions do not vary significantly from those reported for UO_2 fuel.

5.3.4 Irradiation Experience (Ref. 2)

Although irradiation experience with $\text{CeO}_2\text{-UO}_2$ fuel is very limited, the high burnups achieved in these tests and the careful and detailed post irradiation examinations conducted provide a substantial base of information.

Fuel specimens containing fourteen and thirty atom percent of 93 w/o enriched UO_2 were irradiated to $25,000$ MWD/TeM and $110,000$ MWD/TeM. Post irradiation evaluation data for these specimens and comparative data for $\text{ZrO}_2\text{-UO}_2$ fuel specimens is presented in Table 5-6. The results obtained from these irradiation tests are summarized as follows: 1) the irradiated fuel showed no significant dimensional changes. There was no evidence of clad distortion or fuel-clad chemical interaction; 2) Micro-structural changes in the fuel were, in general, less pronounced than in UO_2 fuel; 3) The fuel retained the single phase, face centered cubic crystal structure during irradiation; 4) Irradiation to very high burnups did not adversely affect the corrosion resistance of the fuel; 5) The fuel swelling values were far below those reported for UO_2 fuel; and 6) The dissolution characteristics in nitric acid of $\text{CeO}_2\text{-YO}_2$ fuels and $\text{CeO}_2\text{-PuO}_2$ fuels are very similar to those of $\text{UO}_2\text{-PuO}_2$ fuel.

5.3.5 Conclusions

The physical properties of $\text{CeO}_2\text{-UO}_2$ fuels are very similar to those of UO_2 . The high sinterability at relatively low temperatures in air or argon atmospheres result in a relatively simple sintering operation. This is highly desirable in a plutonium fuel fabrication facility. The dissolution characteristics in nitric acid are similar to those of $\text{UO}_2\text{-PuO}_2$ fuel. High purity cerium oxide powders are available at reasonable cost. Irradiation properties of $\text{CeO}_2\text{-UO}_2$ systems appear quite favorable. While no data currently exists on the properties of $\text{CeO}_2\text{-UO}_2\text{-PuO}_2$ fuels, the properties of these materials are expected to compare favorably with those of $\text{CeO}_2\text{-UO}_2$ materials.

5.3.6 References

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2. G. Zuromsky, et al, "Irradiation Testing of Ceramic Fuels", CEND 2936-335, March 1968.
3. W. E. Unger, et al, "LMFBR Fuel Cycle Studies Progress Report for September 1971 No. 31", ORNL-TM-3614, October 1971.
4. M. S. Farkas, "The Preparation and Heat Content of Uranium-Plutonium- and Neptunium containing Microspheres for Use as Intrinsic Thermocouple Flux Probes", BMIC 10175, July 21, 1966.
5. "Third Annual Report - High Temperature Materials and Reactor Component Development Programs", GEMP-270 B Volume II - Materials", February 28, 1974.
6. "Sixth Annual Report - High Temperature Materials Program, Part B" GEMP-475-B February 28, 1967.
7. C. A. Elyard, et al, "Sintered Uranium Dioxide", Patent Specification No. 1,116,663 (British) Copyright 1968.

5.4 Safety and Licensing

A typical UO_2 core near the end of an equilibrium cycle derives a substantial fraction of its power (>50%) from the fissioning of plutonium isotopes produced in the core. Thus, the use of plutonium as fuel in LWRs does not represent a new situation. Experience has also been gained through experimental and demonstration irradiation of limited loadings of mixed oxide fuel rods in commercial reactor cores in the United States. However, little experience, if any, with respect to safety analysis and licensing issues of large-scale use of plutonium in commercial PWRs has been gained. Methods, computer codes and evaluations of PWR safety analyses have been developed already for UO_2 cores. Much of this methodology can be taken over directly to the APR since the performance of a mixed oxide core will, by design, be similar to a UO_2 core under the normal steady state and load following conditions. As discussed in Chapter 3, changes in the nuclear and physical properties of mixed oxide cores will somewhat alter fuel behavior during abnormal transients and accidents, but the consequences are not expected to be affected dramatically. No major improvements to existing techniques in analyzing consequences of transients and accidents for mixed oxide cores have been identified as necessary. Admittedly, experimental data on mixed oxide fuels is not as extensive as for UO_2 , but more data will become available as the industry gains experience from the irradiation of mixed oxide fuel assemblies in the SGR mode. The necessary experience in formulating and responding to licensing questions dealing with plutonium fuel should be acquired in the same way. Thus we anticipate that licensing of all-plutonium fueled reactors should not introduce any new substantive issues or problems.

TABLE 5-1

SUMMARY OF LIGHT-WATER MIXED-OXIDE LATTICE EXPERIMENTS
DESCRIBED IN THE OPEN LITERATURE*

Uniform clean criticals (critical size and buckling)	Belgonucleaire-SCK/CEN BNWL Westinghouse
Reactivity measurements in cores containing various perturbations	CNEN ENEL BNWL-CNEN Westinghouse Gulf-United Nuclear Fuels
Power distribution measurements	Belgonucleaire-SCK/CEN ENEL ENEL-UKAEA BNWL BNWL-CNEN Westinghouse Gulf-United Nuclear Fuels
Control rod worth measurements	Belgonucleaire-SCK/CEN CNEN-BNWL BNWL Westinghouse
Temperature coefficient measurements	BNWL CNEN-BNWL Westinghouse UKAEA Gulf-United Nuclear Fuels
Spectrum and reaction rate measurements	Belgonucleaire-SCK/CEN CNEN ENEL-UKAEA General Electric Gulf General Atomic BNWL UKAEA
Measurement of κ_{∞}	Babcock and Wilcox Belgonucleaire-SCK/CEN BNWL Gulf-United Nuclear Fuels
Reactivity effects of PuO_2 particles	BNWL
β/ℓ and Rossi-alpha	CNEN-BNWL BNWL
Gadolinia burnable poison effects	CNEN Gulf-United Nuclear Fuels

*For detailed references see Ref. 14

TABLE 5-2

ANALYSIS OF BNWL ROOM TEMPERATURE MIXED-OXIDE LATTICE EXPERIMENTS

Simulated Fuel Assembly Description	Control Rods	Calculated K_{eff}
All Mixed Oxide Assembly	Out Inserted	1.0086 1.0043
Mixed Oxide Central Island	Out Inserted	1.0064 1.0026
Mixed Oxide Small Island	Out Inserted	1.0137 1.0101
UO ₂ Reference	Out Inserted	1.0055 0.9994

Mixed Oxide Rods: 1.52 w/o Fissile PuO₂

UO₂ Rods: 2.35 w/o U235

TABLE 5-3

ANALYSIS OF KRITZ MIXED OXIDE LATTICE EXPERIMENTS

Simulated Assembly Fuel Enrichment	Control Rods	Moderator Temperature	Calculated K_{eff}
1.38 w/o Fissile Pu	Out	68°F (293°K)	1.0026
1.38 w/o Fissile Pu	Out	338°F (443°K)	1.0005
1.38 w/o Fissile Pu	Out	473°F (518°K)	0.9963
2.71 w/o Fissile Pu	Out	77°F (298°K)	1.0048
2.71 w/o Fissile Pu	Out	446°F (503°K)	1.0024
2.71 w/o Fissile Pu	Inserted	77°F (298°K)	1.0029
2.71 w/o Fissile Pu	Inserted	473°F (518°K)	1.0017
UO ₂ Reference 3.1 w/o	Out	81°F (300°K)	1.0035
UO ₂ Reference 3.1 w/o	Out	442°F (501°K)	1.0015
UO ₂ Reference 3.1 w/o	Inserted	72°F (295°K)	1.0020
UO ₂ Reference 3.1 w/o	Inserted	428°F (493°K)	1.0000

TABLE 5-4

U.S. LWR PLUTONIUM FUEL IRRADIATION EXPERIENCE

Reactor (Type)	Pu Assembly & Rod Characteristics					Type	Clad			Linear Heat Rate KW/FT	Burnup GWD/MTM	Load Date	Discharged Date
	ASBL	Design/Fab	Rods	Split	Composition		Mat.	OD	Thick				
Saxton (PWR)	7	West/West-NUMEC	72	70	6.6% Pu + NU	Pellet	Zr-4	0.391	0.023	16/21.6 pD	29 pA	12/65	Recon
	2	West/West-BNW	72	70	6.6% Pu + NU	Vipac	Zr-4	0.391	0.023	16 pD	5 aA	12/65	66
	7	West/West-NUMEC	72	36	Pu + U	Pellet	Zr-4	0.391	0.023	24.7 pD	50 pD	12/69	72
				36	Zr & SS Water Tube					22.9 pA			
San Onofre (PWR)	4	West/West	180	64	3.30% Pu + NU	Pellet	Zr-4	0.422	0.0243			11/70	
				92	3.65% Pu + NU								
				24	3.95% Pu + NU								
Dresden-1 (BWR)	4	GE/GE	36	1	1.2-1.7% Pu	Pellet	Zr-2	0.5625	0.035			6/67	71
				35	Irrad. U								
	11	UNC/UNC	36	9	2.3% Pu + NU	Pellet	Zr-2	0.5625	0.035	9 pA	8.3 a (9/71)	1/70	
				27	1.77-2.34% EU								
Big Rock Point (BWR)	2	GE/GE	81	2	1.22% Pu	Solid	Zr-2	0.5625	0.04	13.6 pA	19.5 aD	5/69	
				79	EU	Pellet							
	2	GE/GE	81	2	1.22% Pu	Dished	Zr-2	0.5625	0.04	13.6 pA	19.5 aD	5/69	
				79	EU	Pellet							
	6	GE/GE	81	2	1.36% Pu	Annular	Zr-2	0.5625	0.04	13.6 pA	19.5 aD	5/69	
				79	EU	0.1" h							
	6	GE/GE	81	2	1.59% Pu	Annular	Zr-2	0.5625	0.04	13.6 pA	19.5 aD	5/69	
				79	EU	0.2"							
	3	EEI/GE	81	68	1.6-9% Pu + NU	Annular	Zr-2	0.5625	0.04	15.9 pA	20 aD	3/70	
				8	3.4% EU+Gd	0.15"							
				5	Co or Spacer								
	2	Exxon/Exxon	81	24	3.65% Pu + NU		Zr-2	0.5625	0.04	21.6 pD	25.9 aD	3/72	
				53	2.55-4.2% EU				0.05				
				4	Co								
	4	NFS/NFS	121	32	1.03% Pu + 1.56% EU		Zr-2	0.449	0.034			3/73	
			41	2.45% Pu + 2.4% EU									
			4	Co									
			4	Hollow									
			40	Standard									
2	Exxon/Exxon		121	24	4.3% Pu		Zr-2					3/73	

TABLE 5-5

EUROPEAN LWR PLUTONIUM FUEL IRRADIATION EXPERIENCE

Reactor (Type)	ASBL	Pu Assembly & Rod Characteristics				Type	Clad			Linear Heat Rate KW/FT	Burnup GWD/MTM	Load Date	Discharged Date
		Design/Fab	Rods	Split	Composition		Mat.	OD	Thick				
Dodewaard (BWR)	1	Belgo/Belgo	36	15	2.26% Pu + NU	Pellet	Zr-2	0.53	0.031	7.01 aD	25 aD	4/71	
				21	2.5% EU					12.2 pD	40 pD		
	1	Belgo/Belgo	36	12	2.26% Pu + NU	Pellet	Zr-2	0.53	0.031	7.01 aD	14 P	4/71	
				3	2.26% Pu + NU	Vipac				12.2 pD	(7/72)		
Garigliano (BWR)	6	ENEL/BNFL	64	24	3.16% Pu + NU	Dished	Zr-2	0.4684	0.0299	10.4 pD	13.7 a	10/68	12/76
				28	1.55% Pu + NU	Pellet					(10/72)		
				12	0.82% Pu + NU	Pellet							
	3	GE/GE	64	12	3.21% Pu + NU	Vipac	Zr-2	0.4685	0.0299		13.7 a	10/68	12/76
				12	2.00% Pu + NU	Wafer					(10/72)		
				28	2.41% EU								
	4	ENEL/ALKEM Belgo, KRT	64	12	1.83% EU	Pellet	Zr-2	0.594	0.035	12.2 pD	22 aD	9/70	5/78
				23	3.34% Pu + NU	Pellet					30 pD		
				28	1.64% Pu + NU	Pellet					7.2 a		
				12	0.87% Pu + NU	Pellet					(10/72)		
	1	ENEL/BNFL	64	23	3.34% Pu + NU	Pellet	Zr-2	0.594	0.035	12.2 pD	25.4 aD	12/72	12/76
				28	1.64% Pu + NU	Pellet							
				12	0.87% Pu + NU	Pellet							
				1	EU	Pellet							
	1	ENEL/BNFL	64	23	3.34% Pu + NU	Pellet	Zr-2	0.594	0.035	12.2 pD	21.6 aD	12/72	5/78
				28	1.64% Pu + NU	Pellet							
				12	0.87% Pu + NU	Pellet							
				1	EU	Pellet							
	1	GE/GE	64	23	3.34% Pu + NU	Pellet	Zr-2	0.594	0.035	12.2 pD	25.5 aD	6/75	5/78
				28	1.64% Pu + NU	Pellet							
				12	0.87% Pu + NU	Pellet							
				1	EU	Pellet							
	25	BN/RBU	64	31	4.30% Pu + NU	Pellet	Zr-2					6/75	5/81
				33	2% EU	Pellet							
	18	BN/RBU	64	31	4.30% Pu + NU	Pellet	Zr-2					6/75	11/82
				33	2% EU	Pellet							
	3	BN/RBU	64	31	4.30% Pu + NU	Pellet	Zr-2					6/75	5/84
				33	2% EU	Pellet							

TABLE 5-5 (Cont'd.)

EUROPEAN LWR PLUTONIUM FUEL IRRADIATION EXPERIENCE

Reactor (Type)	ASBL	Pu Assembly & Rod Characteristics					Clad			Linear Heat Rate KW/FT	Shutup GWD/MTM	Load Date	Discharged Date
		Design/Fab	Rods	Split	Composition	Type	Mat.	OD	Thick				
Gundrem- mingen-A (BWR)	16	RBU-ALKEM	36	36	2.030 Pu + 0.69% U	Pellet	Zr-2	0.563	0.035	15.8 pD	22 aD	6/74	5/78
	24	RBU-ALKEM	36	36	2.030% Pu + 0.69% U	Pellet	Zr-2	0.563	0.035	15.8 pD	22 aD	7/75	6/79
Kahl (BWR)	1	/BNFL	36	9	1.7% Pu	Pellet	Zr-2	0.497		10 aD	16.2 aA	1/66	
				27	U					10 aD	7 p		
	2	/CNEN	36	36	Pu + U	Pellet	Zr-2			4 aD	(7/72)		
	4	NIS/ALKEM	36	36	1.9% Pu + NU	Pellet	Zr-2			4 aD	15 aD	10/68	
	17	NIS/ALKEM	36	36	1.9% Pu + NU	Pellet	Zr-2			4 aD	15 aD	12/70	
Lingen (BWR)	1	AEG/KRT	36	15	Pu + Th		Zr-2		0.0354			7/70	
				21									
Sena-Chooz (PWR)	4				4.35 Pu + NU		SS					12/74	
Obrigheim (PWR)	1	RBU-ALKEM	180	180	Pu + NU		Zr-4				15 aA	10/72	9/74
	1	RBU-ALKEM	180	180	3.2% Pu + NU		Zr-4					10/73	9/77
	7	RBU-ALKEM	180	180	Pu + NU		Zr-4					10/73	9/74
	4	RBU-ALKEM	180	180	Pu + NU		Zr-4					10/74	9/78
BR3/Vulcan (PWR)	1	Belgo/Belgo	37	12	Pu + U	Vipac	SS	0.343	0.0197	3.96 pA	3.4 aA	63	64
				25							6.0 pA		
	1	Belgo/Belgo	37	18	Pu + U	Pellet	SS	0.335	0.0197	7.62 pA	25 aA	12/66	68
		BNFL		19	Pu + U	Vipac					40 pA		
	2	Belgo/Belgo	36	15	Pu + U	Pellet	Zr-4	0.343	0.024	10.4 pA	41 aD	7/69	70
				3	Pu + U	Vipac					60 pD		
				18	U								
	9	Belgo/Belgo				Pellet	Zr-4	0.343	0.024	10.1 D	22 aD	5/72	74
											39 pD		
	13	Belgo/Belgo				Pellet	Zr-4	0.421	0.028	18.3 D	23 aD	5/72	74
											38 pD		

a = Assembly Average
NU = Natural Uranium

p = Local Peak
EU = Enriched Uranium

A = Achieved Value

* = Selected Rods Removed March 1972 For Hot Cell Examination

TABLE 5-6

SUMMARY OF IRRADIATION CONDITIONS AND POST IRRADIATION

EXAMINATION RESULTS FOR $\text{CeO}_2\text{-UO}_2$ and $\text{ZrO}_2\text{-UO}_2$ FUELS

Specimen No.	Position in Capsule	Burnup Fission/cm ³ a/o x 10 ²⁰	Centerline Irradiation Temperature, °C						Fission Gas Release, %	Fission Gas Remaining in Fuel, Atoms/cm ³ x 10 ²⁰	Portion of Fuel Undergoing Grain Growth, %	Fuel Column * Length Change, %		Clad Diameter Change, %	Fuel Lattice Parameter		
			Avg.	Max.	Avg Last 20% of Irrad.	Max Last 20% of Irrad.	Avg Last Cycle	Max Last Cycle				ΔL_1	ΔL_2		Unirrad.	Fuel Center	Fuel Edge
Z-10-A	1	40	4.4	1675	2150	1670	1790	1620	1790	-	-	1.0(a)	-	0.2	5.149	5.128	(Bulk)
Z-10-B	3	40	4.4	1875	2345	1780	1955	1760	1955	43	-	44	-	0.2	-	-	-
Z-10-C	4	42	4.6	1870	2455	1710	1900	1650	1790	32	-	48	-	0.2	-	-	-
Z-10-D	1	52	5.7	1655	2345	1250	1425	1260	1345	-	-	0.7	-	0.2	-	-	-
Z-10-E	2	55	6.2	1730	2510	1335	1510	1345	1510	37	1.0	44	1.4	5.3(b)	-	5.109	5.134
Z-10-F	3	57	6.3	1805	2555	1545	1790	1705	1790	46	0.8	52	1.4	15.3(b)	-	-	-
Z-10-G	1	70	7.8	1935	2955	1815	2955	2480	2955	53	0.8	-	3.3	38.7	-	-	-
Z-10-H	2	69	7.7	1890	2925	1730	2925	2400	2925	-	-	2.7	38.7	0.4	-	-	-
Z-10-J	3	71	7.9	2025	3040	1815	3040	2345	3040	56	0.9	71	0	46.0	-	5.136	5.150
C-10-A	2	43	6.5	1415	1760	1245	1510	1010	1510	22	-	46	-	-	5.415	5.399	(Bulk)
C-10-B	5	44	6.7	1295	1595	1195	1315	1150	1260	28	-	33	-	-	-	-	-
C-10-C	4	58	8.1	1400	1955	1205	1400	1315	1400	48	1.0	48	0.7	2.0(b)	-	5.391	5.418
C-10-D	5	61	8.6	1200	1650	1050	1175	1120	1175	62	0.7	44	1.4	3.3(b)	-	-	-
C-10-E	4	74	10.6	1400	2120	1245	2120	1730	2120	58	1.1	-	1.3	29.0	-	-	-
C-10-F	5	77	11.0	1295	1955	1165	1955	1730	1955	59	1.0	74	6.7	28.0	-	5.410	5.416
Z-22-M	1	40	10.5	1975	2455	1775	2095	1815	1870	46	-	58	-	-	5.173	5.138	(Bulk)
Z-22-N	3	37	9.2	1955	2650	1710	2290	1930	2065	-	-	-	-	-	-	-	-
Z-22-O	4	40	10.5	2000	2730	1635	2150	1870	1925	52	-	65	-	-	-	-	-
Z-22-P	1	60	15.2	2095	2480	1580	2175	1510	1705	62	1.7	61	4.7	22.0	-	-	-
Z-22-R	3	58	14.7	1905	2260	1200	2095	1120	1620	66	1.3	-	2.0	8.6	-	5.166	5.165
Z-22-Z	4	58	14.6	1760	2095	1145	1980	1345	1455	-	-	-	0	0	-	-	-
Z-22-T	1	73	18.4	1700	2425	1355	1650	1425	1480	-	-	-	2.7	16.0	-	-	-
Z-22-X	2	74	18.5	1630	2425	1200	1540	1290	1375	69	1.4	72	0.7	12.0	-	5.159	5.163
Z-22-Y	3	62	15.6	1660	2510	1145	1425	1205	1260	84	0.6	77	0	11.3	-	-	-
C-25-M	2	44	17.0	1480	1870	1425	1730	1565	1595	74	-	56	-	-	5.421	5.38	(Bulk)
C-25-O	5	41	15.7	1525	1955	1295	1595	1425	1480	79	-	63	2.0	-	-	-	-
C-25-P	2	61	23.0	1660	1900	1500	1815	1260	1400	88	0.7	79	0.7	8.0	-	-	-
C-25-R	5	57	21.5	1315	1595	1175	1455	925	1290	87	0.7	-	2.7	8.0	-	5.396	5.419
C-25-N	4	76	28.4	1405	2040	925	1205	1040	1095	84	1.1	97	3.3	10.7	-	5.413	5.417
C-25-S	5	64	25.8	1195	1815	690	870	650	705	96	0.3	-	0.7	4.7	-	-	-

(a) Measured optically.

(b) Measured from photomicrographs, all other lengths measured from gamma scans.

* ΔL_1 is the fuel length change adjacent to the fuel-clad interface; ΔL_2 is the fuel length change at the center of the fuel column.

FIGURE 5-1

EEI - WESTINGHOUSE MIXED OXIDE FUEL DEMONSTRATION
 LOCATION OF DEMONSTRATION ASSEMBLIES AND IN-CORE INSTRUMENTATION
 SAN ONOFRE UNIT 1 - CYCLE 2

FIGURE 5-2

EEI - WESTINGHOUSE MIXED OXIDE FUEL DEMONSTRATION
 LOCATION OF DEMONSTRATION ASSEMBLIES AND IN-CORE INSTRUMENTATION
 SAN ONOFRE UNIT 1 - CYCLE 3

3.31 W/O FISSILE Pu

WATER

3.10 W/O FISSILE Pu

2.84 W/O FISSILE Pu

FIGURE 5-3

EEI - WESTINGHOUSE MIXED OXIDE FUEL DEMONSTRATION
MIXED OXIDE DEMONSTRATION ASSEMBLY ENRICHMENT ZONING
SAN ONOFRE UNIT 1 - CYCLES 2 AND 3

FIGURE 5-4

KRAFTWERK UNION MIXED OXIDE FUEL DEMONSTRATION
 LOCATION OF DEMONSTRATION ASSEMBLIES AND IN-CORE INSTRUMENTATION
 KWO CYCLE 4

FIGURE 5-5

KRAFTWERK UNION MIXED OXIDE FUEL DEMONSTRATION
 LOCATION OF DEMONSTRATION ASSEMBLIES AND IN-CORE INSTRUMENTATION
 KWO CYCLE 5

FIGURE 5-6

KRAFTWERK UNION MIXED OXIDE FUEL DEMONSTRATION
 LOCATION OF DEMONSTRATION ASSEMBLIES AND IN-CORE INSTRUMENTATION
 KWO CYCLE 6

FIGURE 5-7

KRAFTWERK UNION MIXED OXIDE FUEL DEMONSTRATION
LOCATION OF DEMONSTRATION ASSEMBLIES AND IN-CORE INSTRUMENTATION
KWO CYCLE 7

FIGURE 5-8

KRAFTWERK UNION MIXED OXIDE FUEL DEMONSTRATION
 MIXED OXIDE DEMONSTRATION ASSEMBLY ENRICHMENT ZONING
 KWO CYCLES 4, 5, 6 AND 7

FIGURE 5-9

SAN ONO FRE UNIT 1 - CYCLES 1 AND 2
CALCULATED CORE REACTIVITY VS. EXPOSURE

FIGURE 5-10

KWO CYCLES 1-5
CALCULATED CORE REACTIVITY VS. EXPOSURE

FIGURE 5-11

SAN ONOFRE UNIT I - CYCLE 2
COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS
1750 MWD/TeM CYCLE EXPOSURE

SAN ONOFRE CYCLE 2 - AXIAL SHAPE FROM Mn AEROBALL DETECTOR
MIXED OXIDE TEST ASSEMBLY A-8 - 1750 MW/D/T CYCLE EXPOSURE

FIGURE 5-12

FIGURE 5-13

SAN ONOFRE UNIT I - CYCLE 2
 COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS
 4950 MWD/TeM CYCLE EXPOSURE

FIGURE 5-14

SAN ONOFRE UNIT 1 - CYCLE 2
 COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS
 7400 MWD/TeM CYCLE EXPOSURE

FIGURE 5-15

KWO CYCLE 4

COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS

170 MWD/TeM CYCLE EXPOSURE

FIGURE 5-16

KWO CYCLE 4

COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS

8000 MWD/TeM CYCLE EXPOSURE

FIGURE 5-17
 KWO CYCLE 5
 COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS
 2500 MWD/TeM CYCLE EXPOSURE

FIGURE 5-18

KWO CYCLE 5
 COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS
 6230 MWD/TeM CYCLE EXPOSURE

FIGURE 5-19
 KWO CYCLE 5
 COMPARISON OF CALCULATED VS. MEASURED INSTRUMENT SIGNALS
 6950 MWD/TeM CYCLE EXPOSURE

SMALL MIXED OXIDE ISLANDS

MIXED OXIDE CENTRAL ZONE

ALL MIXED OXIDE LOADING

FIGURE 5-20

MIXED OXIDE FUEL ASSEMBLY CONFIGURATIONS EXAMINED
CE-BNWL PROPRIETARY CRITICAL EXPERIMENTS

KRITZ PHASE I MIXED OXIDE ASSEMBLIES
 DEVIATION BETWEEN MEASURED AND CALCULATED PIN POWERS

FIGURE 5-21

FIGURE 5-22

KRITZ PHASE II MIXED OXIDE ASSEMBLY
 COMPARISON OF CALCULATED VS. MEASURED PIN POWERS
 77°F (298°K) MODERATOR TEMPERATURE

FIGURE 5-23

KRITZ PHASE II MIXED OXIDE ASSEMBLY
 COMPARISON OF CALCULATED VS. MEASURED PIN POWERS
 446° (503°K) MODERATOR TEMPERATURE

FIGURE 5-24

KRITZ PHASE II REFERENCE UO_2 ASSEMBLY
 COMPARISON OF CALCULATED VS. MEASURED PIN POWERS
 442° (501°K) MODERATOR TEMPERATURE

6.0 Recommended Areas for Further Study

6.1 Design Methodology Improvements

To date, the major plutonium recycle demonstration programs have given emphasis to the gradual introduction of mixed-oxide assemblies into uranium-fueled reactors. These assemblies generally contained plutonium recovered from relatively low exposure spent fuel. The relatively small quantities of mixed-oxide fuel which have been incorporated to date into PWRs have not had an observable effect on overall reactor core properties.

The analytical methods employed for the design and subsequent evaluation of these demonstration cycles were generally identical to those employed for uranium-fueled operation. These methods have proven adequate at least for these early demonstrations where the mixed-oxide assembly fractions were low and the core plutonium inventory did not depart markedly from that found in all uranium cycles. The validity of these methods has not been (and currently cannot be) adequately tested in the range of plutonium loadings expected in all-mixed-oxide-fueled operation. The more limited assumption of their adequacy for SGR fuel cycle design should also be subjected to additional confirmation; this confirmation will become possible as the mixed-oxide loadings in commercial LWRs gradually increase following initiation of commercial plutonium recycle. A substantive commitment to validate current design methods for heavily-loaded mixed-oxide systems must be made as early as possible in order to support the level of plutonium recycle employed in the plutonium burner concept. In any such program, it is felt that the following areas, which directly affect predictions of fuel performance and plant safety, should receive a high priority:

A. Basic Technology

1. Evaluation and improvement of the basic cross section data, including fission product yields.
2. Evaluation and improvement of resonance capture calculations in systems containing mixtures of fertile materials, i.e., U-238, Pu-240, and Pu-242.
3. Evaluation and improvement of lattice physics calculation techniques used to produce the averaged cross section data used in core calculations.
4. Evaluation and improvement of methods used to predict fuel temperature behavior in mixed-oxide fuels.

B. Design Methods

The development of a high level of confidence in the prediction of the following parameters with core burnup is an essential requirement: (1) Control rod worth, as well as the worth of other control systems, primarily soluble boron; (2) Doppler, moderator, and power coefficients, and the prediction of the fuel temperature variation with core operating parameters; (3) Power peaking and distribution on both a 2D and 3D basis.

Although the review of currently available experimental information indicates no reason to believe that current design methods are inadequate for the prediction of plutonium burner core parameters, confirmation of this conclusion must await experimental information from more typical mixed-oxide loadings. Confirmation of design methods must be performed both for more substantial plutonium loadings and for plutonium isotopic compositions more typical of continued recycle.

6.2 Experimental Verification of Design Methods

The successful development of PWR plutonium burners for siting both at nuclear energy centers and at dispersed sites depends in large measure on the demonstrated validity of the analytical methods used for their design. In addition, a satisfactory demonstration of the safety and economic feasibility of a PWR completely fueled with plutonium ultimately requires a full-scale operational test. This section attempts to quantify the additional experimental effort which will be required to support the design and licensing of PWR plutonium burners.

The following types of experiments are needed to support the development of safety-related design methodology:

1. Benchmark experiments which substantiate the calculation of basic physical phenomena, such as resonance absorption. While a substantial number of experiments of this type have been conducted, the plutonium concentrations employed have been low and the variation in plutonium isotopics limited. It is important to point out that the analysis of such experiments as do exist has been quite limited and there has been little effort to date to incorporate the findings into the design and safety analysis methodology. Supplemental experiments are required, and a comprehensive analysis effort must be undertaken to translate the experimental results into design methodology improvements.
2. Lattice experiments, such as the proprietary Kritz and BNWL experiments (cf. Section 5.1.3) have played an important role in the validation of the design and safety analysis methods. Although there have been a fairly substantial number of mixed-oxide-fueled-lattice experiments performed, the plutonium loadings have been generally quite low, the lattice configurations not representative of those employed in light-water reactors, and in some instances, the quality of the experimental data has been less than optimum. Additional lattice experiments, which more realistically represent conditions in an all-mixed-oxide-fueled PWR need to be performed. The Kritz experiments provide an example of the type of experimental work that is required.
3. Additional experimental work is required to quantify mixed-oxide-fuel performance under anticipated transient and accident conditions. Mixed-oxide-fuel demonstration programs in power reactors will supply fuel performance data for mixed-oxide assemblies operating at normal power densities; however, additional experimental information will be required to validate fuel performance models at high-power densities and under transient conditions. In addition, better characterization of mixed-oxide-fuel

performance at the high plutonium loadings encountered in equilibrium cycle APR fuel is required. This includes confirmation of predicted temperature distributions within the fuel pellet, microstructure effects, and fuel-clad chemical interactions.

4. Irradiation demonstration programs should include fuel containing plutonium recovered from previously irradiated mixed-oxide assemblies. Such demonstrations would provide the experimental information necessary to evaluate design methods for plutonium isotopic compositions typical of continued recycle.

A comprehensive program of design methodology validation should conclude with a full scale demonstration of mixed oxide fueled reactor operation. Demonstration programs undertaken to date have been limited to the insertion of small numbers of mixed oxide assemblies in reactor cores fueled primarily with uranium assemblies. Because of the small fraction of mixed oxide fuel present in the reactor core, information obtained from these programs has been of very limited value in predicting the behavior of all mixed oxide fueled systems. Clearly, a more ambitious demonstration program is required, leading eventually to the demonstration of all mixed oxide fueled operation in a large PWR.

The most practical approach to a full scale plutonium burner demonstration would involve a gradual transition from all UO_2 to all mixed oxide fueled operation in a LWR designed to be operable with a full core loading of mixed oxide fuel. The C-E System 80TM plant with APR option package fitted (cf. Section 3.2.1) possesses this capability. An alternative approach would involve fueling an entire core initially with mixed oxide fuel to avoid a relatively long transition period. The transition approach, however, would have the following advantages: 1) The program scope and direction could be changed more easily; 2) The transition period would allow time to conduct the methods development and experimental validation programs required to support design and safety predictions; 3) Feedback from reactor operation could be more readily incorporated into the design of subsequent fuel cycles; 4) Safety related design uncertainties would be minimized, since the mixed oxide fuel would be introduced on a gradual basis.

Since the demonstration of a full scale plutonium burner would best be carried out in a utility owned commercial power reactor, it is definitely advantageous to minimize potential licensing problems and reactor shutdowns. A transition approach, perhaps over a five to six year period, would seem to be the most acceptable from this viewpoint.

6.3 Fuels and Materials Development

The benefits to be gained by the use cerium oxide as a diluent in mixed oxide fuels have been discussed in Sections 4.2 and 5.3. As indicated in Section 5.3, there is essentially no data available regarding the properties of $\text{CeO}_2\text{-PuO}_2\text{-UO}_2$ fuels, although inferences can be drawn from the data which has been accumulated with respect to $\text{CeO}_2\text{-UO}_2$ systems. This section outlines basic requirements for additional investigations into the properties of $\text{CeO}_2\text{-PuO}_2\text{-UO}_2$ materials. Information from these investigations is required in order to firmly establish the feasibility of using cerium oxide fuel materials in PWR plutonium burners.

Investigations will be required to confirm that the properties of $\text{CeO}_2\text{-PuO}_2\text{-UO}_2$ fuels are similar to $\text{CeO}_2\text{-UO}_2$ materials in the following respects:

1. Single phase microstructural stability is maintained in the range of plutonium loadings of interest.
2. The fuel material is compatible with the cladding materials currently in use in large PWRs, most notably zircaloy.
3. Corrosion resistance of the fuel material with water and steam is acceptable for the range of conditions expected in large PWRs.
4. The dissolution properties of the fuel material in nitric acid are acceptable in the range of plutonium loadings of interest.
5. Irradiation performance is acceptable.

An investigation of the fuel fabrication aspects of $\text{CeO}_2\text{-PuO}_2\text{-UO}_2$ material would also be of value and would serve to identify and fabrication cost penalties associated with the use of these materials. Compatibility of these fuel materials with the chemical processes used at reprocessing facilities should also be more thoroughly investigated.

6.4 Plutonium Fueled PWR Advanced Converters

There are two plutonium fueled advanced converter concepts which offer the possibility of significantly conserving the nation's uranium ore resources, and which we feel deserve further consideration. These two concepts, the high conversion ratio plutonium burner and the Thorium/Pu converter, are briefly discussed below.

6.4.1 High Conversion Ratio Plutonium Burner

The high conversion ratio plutonium burner, proposed by M. C. Edlund⁽¹⁾ utilizes a close packed lattice having a water-to-fuel ratio of about 0.4, as compared to 2.0 for the standard PWR. As proposed, this type of converter would be realized by modifying existing PWRs; within the context of NECs, new reactors could be constructed, allowing additional flexibility in design.

Annual uranium ore consumption and plutonium fuel requirements and initial plutonium loadings for this concept are compared to those of more conventional PWRs with various water-to-fuel ratios in Figure 6-1. Annual uranium ore requirements are seen to be significantly lower than those of the typical PWR, with a reduction of about 30% realized. Although the initial loadings are higher than those of the PWR, about 10% higher, the loadings are considerably lower than that which would be obtained by extrapolating the PWR curve to a water-to-fuel ratio of 0.4 because of the higher power density of the high conversion ratio reactor.

Although this concept appears to have the potential of extending uranium resources, a number of questions require additional evaluation in order to assess the feasibility of the concept. The main areas of concern are whether such a reactor could be licensed to operate at the proposed power density and the extent to which current technology is applicable, since these considerations strongly influence the likelihood of near-term development.

In particular, the following areas require study:

1. Thermal Margin Considerations
 - a. Adequacy of thermal margin to accommodate the increase in inlet temperature and decrease in flow.
 - b. Thermal margin requirements for various accidents, e.g. LOFA, control rod drop.
 - c. Design modifications and development work required to establish margins.
2. Consideration of high coolant flow velocities (~ 11.5 m/s) and the effect on fuel hold-down and geometric integrity.
3. Adequacy of control rod worth
4. LOCA considerations
 - a. Maximum power peaking
 - b. Emergency cooling and reflood for close packed lattices.
5. An assessment of whether such a reactor could be constructed and licensed using basic PWR technology.

In addition, there are a number of modifications of this basic concept which should be evaluated. These include the use of heavy water, or mixtures of light and heavy water, and the spectrum shift concept. These alternatives might realize similar or greater economies while allowing the lattice spacing to be increased, mitigating many of the technical concerns identified above.

6.4.2 Thorium/Pu Fuel Cycle

The use of the thorium fuel cycle could prove to be an attractive means of plutonium utilization at energy centers, and should be subject to further evaluation. In this fuel cycle, plutonium recovered from the plutonium burners and dispersed site uranium fueled reactor is used to provide thorium oxide fuel enriched in plutonium for use in the NEC plutonium burners. With the thorium fuel cycle, U-233, rather than additional quantities of plutonium, are produced in the plutonium burners. This fuel, because of its superior nuclear characteristics, has a number of potential advantages compared to either U-235 or plutonium.

The U-233 produced by the burners could be mixed with recovered UO_2 at the NEC and shipped to dispersed sites to reduce U-235 requirements. Mixing of U-233 and recovered uranium at the NEC would eliminate the safeguards problem associated with the shipment of either pure U-233 or plutonium and mixed oxide fuel since the U-233 cannot be chemically separated from the mixture. Alternately, the U-233 could be used to enrich thorium; such a fuel cycle would probably be limited to utilization in reactors operating at the NEC. The use of the thorium fuel cycle with U-233 recycle in LWRs would increase the amount of energy produced per unit quantity of plutonium consumed by about 15-20%.

A second alternative would be to employ the U-233 produced in the LWBR. In this instance, the NEC's plutonium burners would provide the initial inventory of U-233 required for LWBR installation.

Simplified schematics of NECs operating on the uranium/plutonium and thorium/plutonium fuel cycles are compared in Figure 6-2. The number of reactors and material flows shown in this figure are characteristic of a mature NEC where all plutonium burners have been deployed and an equilibrium situation has been reached. The plutonium requirements and U-233 production rates were taken from the results reported in Reference 2. The material flows are characteristic of the System 80TM PWR operating on the thorium/plutonium fuel cycle; no lattice optimization has been performed to increase U-233 production.

A total of fifty plutonium burners and associated dispersed site uranium reactors are employed in both systems illustrated in Figure 6-2. In the case of the thorium fuel cycle, 10 plutonium burners are required to utilize the recovered plutonium, as compared to 15 in the case of the uranium/plutonium cycle. This reduction in the number of plutonium burners required at the NEC to serve a given amount of installed nuclear capacity could be important in minimizing the environmental impact of the NEC.

A number of specific items which require further development are summarized below:

1. Detailed material flows and deployment scenarios should be developed for the thorium fueled plutonium burner concept.
2. Considerations which might affect operation, safety, and licensability of the thorium fueled plutonium burners should be evaluated.
3. Modifications to existing PWR designs to increase U-233 production should be explored. Such modifications might include changes in lattice design or the use of a thorium blanket.
4. The LWBR program should be carefully reviewed in order to establish the quantities of U-233 required and the rate of possible LWBR deployment.
5. The potential for uranium ore and SWU conservation possible with the recycle of U-233 in LWRs should be further quantified.
6. The economics of the thorium fueled plutonium burner should be established. Cost development should be performed for reprocessing and fabrication services, and the possibility and economics of dual purpose thorium-uranium facilities should be explored as part of this study.
7. The use of heavy water systems or of the spectrum shift concept to improve plutonium utilization and U-233 production should be evaluated.

URANIUM ORE AND FISSILE PLUTONIUM REQUIREMENTS
VS. LATTICE WATER TO FUEL RATIO
1300 MW_e - 75% CAPACITY FACTOR

FIGURE 6-1

References

1. M. C. Edlund, "High Conversion Ratio Plutonium Recycle in Pressurized Water Reactors," Wingspread Conference, May 14-16, 1975.
2. R. L. Hellens, et al, "A Survey of Thorium Fuel Cycles in PWR's," to be presented, ANS, June 1976.

FIGURE 6-2