MASTER

**CONCEPTUAL DESIGN AND ECONOMIC ANALYSIS
OF A
LIGHT WATER REACTOR FUEL ENRICHER/REGENERATOR**

**FY 1978
YEAREND REPORT**

DEPARTMENT OF NUCLEAR ENERGY

**BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.**

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CONCEPTUAL DESIGN AND ECONOMIC ANALYSIS OF A LIGHT WATER REACTOR FUEL ENRICHER/REGENERATOR

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YEAR-END REPORT

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ABSTRACT

A study has been performed to evaluate the use of high-energy particle accelerators as nuclear fuel enrichers and nuclear fuel regenerators. This builds on ideas that have been current for many years. The new study has, however, explored some novel approaches that have not been examined before. A specific conceptual system chosen for more detailed study would stretch the energy available from natural uranium by a factor of about 3, reduce the separative work requirements by a factor of about 4, and reduce the volume of spent fuel to be stored by a factor of 2, compared to the current once-through light water reactor (LWR) fuel cycle. The concept avoids the need for chemical reprocessing of spent fuel, and would permit continued use of LWR's beyond the time when limitations on fuel resources might otherwise lead to their being phased out. This concept, which is called the Linear Accelerator Fuel Enricher/Regenerator, is therefore viewed as offering a practical means of stretching the use of the nuclear fuel resource in the framework of the existing light water reactor fuel cycle.

This report describes and analyzes the concept referred to. An explanation of the principles underlying the concept is given. Particular attention is devoted to engineering feasibility, proliferation resistance, and economics. It is seen that the concept draws on only proven technology as regards both accelerator design and the fuel irradiation process, and is adapted to existing LWR designs with no change except in fuel-handling practices. A preliminary evaluation of radiation damage, coolant options, and power conversion systems is provided. Neutronic, thermal-hydraulic, and burnup calculations are presented. An analysis is made of fuel economy. Approximate costs of electric power produced using this concept are evaluated and discussed. Estimated development costs of commercialization are provided.

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Linear accelerator nuclear fuel enricher/regenerator.

1. SUMMARY

1.1 Introduction

Nuclear energy is an important element of our total energy system and will become even more so in the future. However, there is a growing awareness that nuclear energy as produced in light water reactors (LWR) using the once-through fuel cycle is wasteful in uranium, and that, at the rate of use presently projected, our more economically attractive uranium resources may be depleted in a few decades. The development and commercialization of the liquid metal fast breeder reactor (LMFBR) have been viewed until recently as the answer to full use of our uranium resources, thus providing the capability to supply nuclear energy for centuries to come. However, LMFBR development has been slower and costlier than first estimated. In addition, because the reprocessing facilities required for the LMFBR system are now seen as augmenting the risk of nuclear weapons proliferation, we have seen a change of policy which has, for all practical purposes, halted progress toward commercialization of fast breeder reactors in this country.

For these reasons, a program has been initiated by DOE to evaluate fuel cycles and alternative nuclear energy systems which could be made more "proliferation resistant" and which at the same time would avoid rapid depletion of uranium resources. This program is conducted under DOE's Nonproliferation Alternative Systems Assessment Program (NASAP), organized for that purpose.

A number of candidates are being explored, a few examples being: use of heavy water reactors (HWR), high-temperature gas-cooled reactors (HTGR), molten salt reactors, thorium ^{233}U fuel cycles (which might allow the use of a more proliferation-resistant fuel and increase the utilization of fuel resources because of the higher conversion ratio attainable in the ^{233}U cycle), and other fuel cycles. Of course ^{235}U would still be needed initially even in the thorium cycle, since ^{233}U does not exist in nature.

It is held by many that controlled thermonuclear fusion with direct conversion to electricity will ultimately supply all the energy needs of the future. It is not clear, however, that fusion if it is shown to be technically feasible might ever become economical or reliable enough to replace fission as a source of electrical energy. Even if a fusion machine does not become a net power producer, the fusion process (producing 14-MeV neutrons) may be used with a

fertile blanket to produce ^{239}Pu or ^{233}U to fuel nuclear reactors. The fusion-fission reactor or fusion-hybrid, as it is called, might be closer to reality than the use of fusion to produce electricity, because of its lowered performance requirement. However, the principle of sustained magnetically confined fusion is still undemonstrated and the magnetic-fusion-hybrid systems with realistic design are likely to be large in size and difficult to maintain.

The linear accelerator fissile-fuel-producer (linac-breeder) is one alternative which like the fusion-hybrid is capable of producing fissile fuel from a fertile material, without requiring an initial inventory of fissile fuel. It constitutes an independent means of producing either plutonium or the ^{233}U fuel envisaged for a proliferation-resistant denatured thorium fuel cycle. Unlike the fusion-hybrid, the linac fuel producer is a direct extension and application of existing technology. In addition, the physical separation of the accelerator from the target leads to a low radiation environment around the accelerator proper. Hence, the accelerator can be directly maintained by "hands-on" methods.

Under NASAP funding, the Department of Nuclear Energy at Brookhaven National Laboratory has studied the use of particle accelerators for producing fissile material to use in proliferation-resistant fuel burner systems. For some background on the history of the concept, see refs. 1.1 to 1.4.

1.2 Summary of the Report

In the present report we describe the conceptual design of a linear accelerator fissile fuel enricher and fuel regenerator consisting of a 1.5-GeV proton linear accelerator using a flowing liquid lead-bismuth primary target surrounded by a blanket (secondary target) containing fertile uranium or thorium. The accelerator would operate with a continuous proton current of 0.3 A, producing about 10^{20} neutrons per second in the primary target. These neutrons will be multiplied by some fast fission in the surrounding fertile material blanket, being finally absorbed primarily in the fertile material. Conservative calculations indicate that this facility could produce more than one ton of fissile material per year.

No basic new technology is needed in the design and construction of the fissile fuel production facility. The parameters of the linear accelerator are

straightforward extrapolations and combinations of those of existing designs. The design problems of the liquid lead-bismuth target and its circulating system were solved a number of years ago in studies of nuclear reactors using heavy liquid metal cooling, and are also similar to some of the problems that have been solved in most respects through developing reactor cooling using flowing alkali metals. The Pb-Bi liquid system is in principle much less difficult to handle than the more chemically reactive liquid sodium system needed in the LMFBR. The blanket with its fertile material utilizes existing LWR and/or HWR technology. With reasonable engineering development, a demonstration facility could be built in the interval 1990-2000, at a cost of \$1500.

The linear accelerator fuel enricher/regenerator would consist of two distinct parts: the 1.5-GeV linear accelerator or linac, and the target-blanket assembly with its power generation system.

The head-end of the linear accelerator would consist of an Alvarez drift-tube linac, operating at about 200 MHz up to an energy of 150 MeV. The Alvarez structure is followed by a $\pi/2$ mode coupled cavity structure, up to the final energy. An acceleration rate of 1.5 MeV/m appears to be close to optimum. The 1.5-GeV linac is about 1000 m long.

The design of the accelerator has not as yet been worked out in detail, but the specification lies well within the state of the art. No fundamental problems are anticipated. Experience with linear accelerators indicates that such machines can be operated at about 80% plant factor, without appreciable beam loss during acceleration, so that residual radiation will not handicap maintenance of the machine.

The target system is separated from the accelerator by a distance of 100 m. This drift length, containing isolation valves and instrumentation, will keep the accelerator free of any contamination by the high-radiation target area. Figure 1.1 shows the design principle of the target. It consists of a primary lead-bismuth target stopping the entire proton beam, and a secondary target (blanket) containing the fertile uranium.

The primary target consists of multiple falling liquid lead-bismuth columns or jets arranged to provide for an evenly distributed neutron source. This is accomplished by providing a variable target density with respect to target depth, and by stepping the target to intercept different parts of the proton beam at different points within the neutron source area. The liquid lead-bismuth jets operate in the vacuum of the containment vessel, which is directly

Figure 1.1. Target-blanket configuration.

connected to the accelerator via the beam transport system. No window material is required between the accelerator and the target.

The fertile material in the blanket consists of PWR or HWR fuel assemblies located in individual pressure-tubes. These surround the primary target neutron source in a manner consistent with minimization of neutron leakage and parasitic absorption, and with mechanical design constraints. The assemblies of fuel, coolant, and moderator are designed to remain highly subcritical under any circumstance. During an irradiation cycle, fuel shuffling is required to achieve good control of reactivity buildup. For this reason, the blanket will need instrumentation to monitor the fuel being irradiated.

Neutron multiplication occurs in the blanket, generating thermal energy. This energy, as well as that generated by the proton beam in the lead-bismuth target, is used to generate electric power by a conventional steam-turbine system.

The safety considerations applicable to the target of the fuel enricher/regenerator are the same as those for an LWR, with the notable exception that the target is subcritical. However, the entire target-blanket and primary coolant systems are housed in a containment building similar to that of a conventional LWR, with all its safety features.

A preliminary economic analysis indicates that the fuel cycle cost of an accelerator fuel enricher/regenerator is higher than that of present-day LWR's.

However, any analysis simply in terms of the cost of fissile fuel produced is misleading. In spite of this high estimate for fuel cycle cost, the cost of electricity produced with the LAFER/LWR combination is calculated to be only 25% higher than that expected from the conventional LWR fuel cycle system. This cost difference might well disappear in the next 20 years, depending in part on the available supply of natural uranium. Moreover, this potential cost penalty has to be weighed against the enormous cost benefit of tripling the uranium fuel resource and reducing the amount of radioactive waste by half within the context of the present LWR nuclear power system.

The technology for commercial exploitation of the LAFER/LWR system could be in place by the year 2000. This objective can be accomplished with predictable performance and a reasonable R&D effort.

The following sections present the concept of the linear accelerator fuel enricher/regenerator in detail. Particular emphasis is given to this concept as being highly promising in support of installed pressurized water reactor (PWR) capacity. Two other options are also discussed briefly. The fuel producer version of the system which requires reprocessing of irradiated uranium is discussed but not treated in any detail. The driven reactor is also discussed for the sake of completeness; although this system has some attractive features, the disadvantages appear at present to outweigh the benefits.

The final sections present a simplified economic analysis and an economic comparison with the current LWR system. They also outline the costs and schedules for the development and introduction of the fuel enricher/regenerator in our nuclear energy program.

2. JUSTIFICATION OF NEED FOR THE FUEL ENRICHER/REGENERATOR

The nuclear power industry in the United States is at present based on light-water-cooled thermal burner reactors, either of the pressurized water type or of the boiling water type (BWR). These reactors have shown good performance in providing reliable and economical power, and they actually supply almost 10% of the electric power generated in the U.S. today. The LWR's use low-enriched ²³⁵U fuel (LEU). Enrichment is accomplished in government-owned gaseous diffusion plants where natural uranium containing 0.7% ²³⁵U is enriched to ~3% ²³⁵U content for commercial use. This concentration is chosen for economic reasons. About 6 tons of natural uranium are needed to produce 1 ton of LEU at

3% concentration. The 1 ton of LEU is fabricated into fuel elements and burned in the LWR at a conversion ratio of approximately 0.6 (fissile material produced to fissile material burned). Burnup proceeds to a total of $\sim 30,000$ MWD/ton of the LEU. The limitation on burnup in the LWR is controlled by the reduction with time of the reactivity needed for supporting the self-sustaining chain reaction and is not necessarily related to radiation damage to the fuel cladding material (Zircaloy). The spent-fuel element, upon removal from the LWR, contains $\sim 2\%$ fissile material by weight, about half of which is ^{239}Pu and half is the remaining unburned ^{235}U . If no recovery of this fissile material for reuse by reprocessing is assumed, the nuclear fuel cycle described above requires that ~ 6300 tons of natural uranium be sent to the isotope separation plant over the full 30-yr production lifetime of each 1000-MW(e) LWR. The loss into tails at the isotope separation plant leads to an effective net burnup of only 5000 MWD/ton of the original 6300 tons of natural uranium. Approximately 30% less fuel would be required if recovery of the ^{239}Pu were obtained from the spent-fuel element by chemical reprocessing. The requirement would then decrease to ~ 4300 tons of natural uranium.

The natural uranium resource in the U.S. has been estimated to be on the order of 3×10^6 tons. This estimate includes uranium which can be reasonably recovered at present at a cost of less than \$100/pound of yellow cake (U_3O_8).^{2.1} This resource can support a maximum of only 480,000 MW(e) of nuclear power over the lifetime of the reactors. There are other estimates^{2.2} which, by the same definition, state that the uranium reserve consists of only 1.5×10^6 tons, which would support only 240,000 MW(e) of nuclear power. A number of estimates indicate that the U.S. will need an electric generating capacity of about 1,000,000 MW(e) by the turn of the century and that as many as 400 reactors (1000 MW(e) each) will be needed to share in fulfilling this requirement along with other power sources such as coal, oil, and solar. It is clear that LWR's operating on the present nuclear fuel cycle are not a long-term solution to the U.S. electrical energy problem. These considerations have contributed to a reluctance on the part of utility executives to invest in another generation of LWR's. The fast breeder reactor (FBR) has been put forward as absolutely necessary to ensure the long-term establishment of a nuclear power economy.

The fast breeder reactor would permit the conversion of essentially all the naturally occurring ^{238}U to fissile material. With the FBR, the above estimated uranium resource could ultimately be extended by a factor of 200 and with

higher priced reserves becoming economical, an almost unlimited energy source would become available. However, the implementation of the FBR is handicapped by the following disadvantages.

1. The fissile material inventory in FBR's is high. LWR's must supply the initial Pu inventory for the FBR's.
2. A new technology must be adopted if FBR's are to replace the present LWR's. The FBR's will be either liquid metal cooled (sodium) or gas cooled (He), which leads to higher unit capital investments and additional safety considerations.
3. Reprocessing of fuel from the proposed FBR's is an absolute necessity for their fuel cycle.

The present U.S. administration policy on nuclear power is to postpone reprocessing indefinitely as a means of averting the possible proliferation of nuclear weapons. This policy, at once, further limits the nuclear fuel resource for nuclear power generation and fast breeder reactor fuel cycles. Studies have therefore been initiated in the U.S. to investigate alternative nuclear fuel cycles, some of which may not depend on nuclear fuel reprocessing. These studies are being conducted under the Nonproliferation Alternative Systems Assessment Program (NASAP). Another series of studies is also being conducted by an international group under the International Fuel Cycle Evaluation Study (INFCE). Some of the fuel cycles proposed to date as alternatives include the following:

1. Heavy-water-moderated reactors (such as the CANDU) which would utilize natural uranium fuel and might attain a net burnup of up to 10,000 MWD/ton of natural uranium. This burnup would be almost twice that which can be obtained with the LEU-LWR fuel cycle. However, the heavy-water power reactor technology is not readily available in the U.S.
2. Variations of heavy-water-moderated reactors. It would be possible to start operation with heavy water and then shift to light water in a spectral shift reactor. A second system is a "hand-me-down cycle," burning the element first in an LWR to 2% fissile fuel content and then further burning it in an HWR to < 1% fissile content.
3. Use of a thorium cycle in an LWR, with a higher conversion ratio (>0.70). The thorium fuel resource in the U.S. is somewhat higher than the uranium resource. However, in order to utilize thorium, pro-

duction and use of fissile ^{233}U would be necessary, which would require reprocessing in addition to converting and breeding.

4. Finally, it has been suggested that reprocessing for converters and breeders be allowed in safeguarded nuclear fuel centers.

We propose, as an alternative to the above, the development and use of the linear accelerator fuel enricher/regenerator (LAFER). This would permit retention of a nuclear economy based on use of the LWR, and would introduce a new dimension to the LWR fuel economy. The LAFER would produce fissile fuel in fabricated fuel rods by means of an external source of neutrons. It could stretch the nuclear fuel resource by a factor of 3.6 or more without resorting to reprocessing, and thus would provide a long-term supply of fuel to the LWR power system. Subsequent use of the LAFER with protected reprocessing, or use of the related linear accelerator fuel producer (also with protected reprocessing), could stretch the fuel supply by a factor of nearly 200 over that estimated from the ^{235}U availability alone. It would achieve the same purpose as the FBR but with the notable exception that it could also efficiently produce ^{233}U from thorium, thus opening up use of the thorium fuel resource. Figure 2.1 shows the potential benefit of the accelerator fuel regenerator and fuel producer in our nuclear energy system. They can effectively supply our energy needs well into the 21st century.

Still another option is open: electrical power can be produced directly with a linear accelerator driven reactor (LADR). As will be pointed out, this option may be less attractive to utilities, in part because of the complexity introduced by the need for a linear accelerator at each power plant.

The option we believe to be most viable and attractive at this time is the linear accelerator fuel enricher/regenerator, because (1) it would not require reprocessing of spent fuel, (2) it would ensure a continuing LWR economy. The LAFER is not a power reactor or a breeder; it is a fuel generator. When applied to fabricated fuel, it would serve as an enricher in the same functional sense as a diffusion plant. It would, however, avoid the disadvantage of diffusion plant operation - of severely depleting the natural uranium resource. One LAFER could support three or more LWR's. The fuel regenerator would not impose new technology on the utility company. The fuel regenerator would be independent of the utility. It would supply the utility with conventional LWR fuel, although this fuel would be very radioactive. More detailed examination of the LAFER fuel cycles and economics is made in later sections of this report.

Figure 2.1. Nuclear power growth patterns. Effect of LAFER systems on uranium resource.

The only device based on use of an external neutron source which might be a competitor to the LAFER is the fusion-fission hybrid. The long-term possibility of replacing the LAFER by fusion-fission hybrids depends on the way the latter system develops. The successful demonstration of the feasibility of an adequate controlled fusion reaction is still some years away. However, there are definite operational advantages to the LAFER, compared with fusion hybrids, related to ease of maintenance. The accelerator fuel producer would be only an extension of present day technology. In this sense the accelerator is unique and really has no near-term competitor. It appears to supply a missing link in the LWR nuclear power cycle chain.

3. PHYSICS CHARACTERISTICS

3.1 The Neutron Spallation - Evaporation Process

For protons or deuterons above ~ 50 MeV, the principal types of interactions with atoms are electron ionization-excitation and inelastic collisions with individual nucleons within the nucleus. Elastic scattering is negligible at these energies. The inelastic collision and ionization-excitation process is

usually called the "spallation-evaporation" or "intranuclear-cascade evaporation" process.

Inelastic scattering reactions between complex nuclei and protons, neutrons, or mesons possessing energies large compared to nucleon binding energies in the target nucleus occur by direct collisions between the projectiles and the individual nucleons in the nucleus.^{3.1-3.3} An incoming high-energy proton will penetrate the Coulomb barrier and collide with an individual nucleon to set off an intranuclear cascade of nucleon-nucleon collisions. If the energy involved in a collision within the nucleus is sufficient, meson production may occur, setting off a subsequent intranuclear cascade of meson-nucleon and meson-meson collisions. Some of the intranuclear cascade collisions will result in protons, neutrons, or mesons escaping, or being "spalled-out," of the nucleus. Ultimately, the energy of the incident particle not carried away by the escaping particles will be shared by the remaining nucleons of the target nucleus, leaving the residual nucleus in a highly excited state. This is followed by a statistical "evaporation" process in which nucleons and gamma rays are emitted until the residual nucleus is de-excited. The energy of the spallation products may be sufficient to induce further intranuclear cascades in other target nuclei. Thus, neutron multiplication occurs as a result of this intranuclear process in any target material. Heavy materials, however, will result in larger neutron yields. Additional neutron multiplication will occur in fissionable material (e.g., Th or ^{238}U because of the fast fission produced by neutron capture.

Even at high energies, the incoming charged particle, which loses energy through the intranuclear cascade process, also loses part of its energy through electron ionization. As the proton slows down, energy transfer by electron ionization becomes more important. Below 500 MeV, it becomes predominant and sets the total range or stopping length of the incoming charged particle in the target material. The energy lost through ionization remains in the target and is converted to thermal energy.

3.2 Neutron Multiplication

The yield of neutrons produced in the spallation-evaporation process by energetic protons has been measured experimentally. Figure 3.1 shows the results of experiments carried out at the 3-GeV Cosmotron at Brookhaven.^{3.4} The

Figure 3.1. Yield measurement of neutrons obtained by bombardment of a heavy metal target with high-energy protons.

neutron yields obtained are for targets of the dimensions shown for various materials. It is interesting to note that the neutron yield for ^{238}U is twice that obtained for Pb. The difference is due to fast fissions occurring in the ^{238}U .

Figure 3.2 shows the calculated energy spectra of neutrons produced by 800-MeV protons incident on the axes of solid targets of Pb and ^{238}U , 15 cm diameter and 30 cm long.^{3.5} Here again the difference introduced by fission neutrons is clearly seen. The yields and energy spectra resulting from the intranuclear cascade alone ($E \gtrsim 5$ MeV) are about the same for both materials.

In the course of this study, the neutron yield produced in the spallation-evaporation process by a 1-GeV proton beam stopping in a target has been calculated for purposes of comparison with the experimental results. The calculation is carried out by the Monte Carlo method with the Nuclear Meson Transport Code (NMTC).^{3.6} NMTC computes the transport of nucleons and mesons up to 3.5 GeV on the basis of the intranuclear-cascade evaporation model,^{3.7} which takes into account both elastic and inelastic scattering. These processes are computed statistically by the Monte Carlo method. Table 3-1 shows the calculated results for various materials of interest. In these calculations, it was assumed that 1-GeV protons are injected into the center of an infinite medium. NMTC calculates all induced reactions down to the energy of 15 MeV; therefore, the neutron yields shown do not include reactions <15 MeV. These results were

Figure 3.2. Neutron energy spectra from 800-MeV protons incident on Pb and ^{238}U targets.

obtained statistically, taking the average yield of 10 batches of 25 protons each.

The calculation for the $\text{H}_2\text{O}/\text{UO}_2$ reactor lattice shown in Table 3-1 shows very small differences over a range of a factor of 6 in moderator/fuel ratio (neutron yield ranges from 28 to 30 neutrons per proton). Since the sampling is small, the statistical error can be large, and the results are somewhat qualitative. However, we can note that the neutron yield in the tightest lattice is not very different from that in the most open lattice. These calculated values

Table 3-1
 Monte Carlo Calculations for Pb and UO₂
 Neutrons Produced by Reactions $\geq$ 15-MeV Excitation Energy
 Neutrons/1-GeV Proton

Target mat.					Av yield*
Pb					35
UO ₂					35
1.4 H ₂ O/UO ₂ effective vol. ratio					28
0.7	"	"	"	"	29
0.3	"	"	"	"	30

Infinite medium, 1-GeV protons injected into center.
 *Does not include fission neutrons.

suggest that the total neutron yield, including neutrons produced by fission and other processes in the energy range <15 MeV, may be somewhat larger than the value of 50 neutrons/1-GeV proton used in earlier evaluations.^{3.8,3.9} However, it must be pointed out that these calculations did not treat a real target geometry, which will include a number of loss mechanisms (neutron leakage, parasitic absorption in structural materials, etc.).

The base calculations described above were carried out for a 1-GeV proton beam. However, the neutron yield is strongly dependent upon the incident particle energy. Figure 3.3 shows the relationship between the neutron and incident particle energy as calculated by Barashenkov et al.^{3.8} Above 1 GeV, the neutron yield increases monotonically with beam energy. It is interesting to note that in the 1-GeV region, deuterons present a gain in yield of $\sim 20\%$ due to the high-energy neutrons produced by the stripping reaction. Although this indicates that a deuteron accelerator might be advantageous, the higher cost and the design constraints of a deuteron accelerator might negate the advantage. The same gain factor may be achieved more readily by increasing the proton beam energy. As a matter of fact, the conservative estimate to be presented below for a regenerator that would support three power reactors requires a 1.5-GeV proton linear accelerator.

Figure 3.3. Relative yield of captured neutrons per unit incident energy vs incident particle energy for protons and deuterons entering a large (infinite size) block of natural uranium.

3.3 Target-Blanket Calculations

Calculations have been carried out to determine the fissile fuel production rate and thermal power density distribution for a number of reactor lattices and a realistic target geometry.

The target blanket as engineered for the fuel regenerator provides for a technically conservative solution to the problems of potential radiation damage to materials during irradiation and peak thermal power densities that will occur near the primary target region.

The primary target consists of falling liquid lead-bismuth columns or jets so distributed along the path of the proton beam as to form a long uniformly distributed "slab" neutron source. No window separates the target region from the accelerator. The target (Pb-Bi jets) operates in a vacuum environment. The elongated "slab" target is flanked by Zircaloy pressure-tubes containing the fuel assemblies to be enriched. The pressure-tubes are required to contain the high-pressure cooling medium used for heat transfer.

Figure 3.4 shows a schematic of the target assembly used in the calculations. The primary target "slab" is 0.3 m wide and 3 m high.

Figure 3.4. Target-blanket geometry used for fissile material production rate calculations.

Figure 3.5. Diagram of target-blanket calculation procedure used at BNL.

Figure 3.5 shows the computational procedure used to calculate the neutron yields, fuel production rates, and peak power density distributions and power generated in the system.

The NMTC code is used to calculate the spallation and evaporation reactions above 15-MeV. energy by the Monte Carlo method. The intra- and intercascade processes are recorded as an events file. The SOURCE program is

Table 3-2
Initial Fissile Fuel Production Rates for
Pb-Bi Target - Fuel Elements Blanket
(0.3-A 1.5-GeV Proton Accelerator)

Moderator/fuel volume ratio = 0.8.

Design No.	Fertile material	Coolant	Density of coolant, g/cc	Initial neutron yield Y_n (includes fission reaction)	Initial production rate of fuel material,* kg/yr
1	UO ₂ (Nat.U)	D ₂ O	0.7	53.8	Pu 1010
2	ThO ₂	D ₂ O	0.7	46.6	²³³ U 850
3	UO ₂ (Nat.U)	H ₂ O	0.7	74.1	Pu 1000
4	UO ₂ (Nat.U)	H ₂ O	0.35	65.8	Pu 1050
5	UO ₂ (Nat.U)	H ₂ O	0.175	64.9	Pu 1070
6	ThO ₂	H ₂ O	0.7	46.5	²³³ U 850
7	ThO ₂	H ₂ O	0.35	48.6	²³³ U 890
8	ThO ₂	H ₂ O	0.175	49.0	²³³ U 900

*Based on 80% plant factor.

used to calculate the neutron source distribution as a function of space, energy, and angle from the events file created by the NMTC code. This neutron source is used in the two-dimensional transport code TWOTRAN^{3.10} in calculating the low-energy neutron reactions in the energy range below 15 MeV. The collapsed group cross-section sets used in the TWOTRAN calculation were derived from the DLC-2 100-group cross-section set^{3.13} which is being used in fusion reactor work. The collapsing of the 100 groups into 10 groups is done by using the format change code "Tapemaker" and the one-dimensional transport ANISN code.

The TWOTRAN calculation gives the expected neutron yield for a given geometry, and an initial fuel production rate and peak power distribution. However, this code does not calculate the enrichment/burnup history of fuel being irradiated over a period of time. This calculation is carried out with the 2DB code and in some cases even more accurately with the PDQ code.

These complex codes require long computer runs, and were not used for earlier parts of the scoping study. The early irradiation/burnup calculations were done with the SIZZLE code, which is faster and cheaper but less accurate. SIZZLE is a one-dimensional burnup code developed for fast reactor analysis.^{3.11} It was modified to accept an external neutron source, and additional neutron

Table 3-3

Fissile Material Net Production Rate (kg/yr)*
Integrated over a 1-yr Period
Starting with Different Initial Enrichment Values

Pb-Bi target, fuel elements blanket
(0.3-A 1.5-GeV proton accelerator).

Case No.	Initial enrichment, %			
	0	0.7	2	3
1	-	1010	1030	1040
2	850	-	750	745
3	-	1000	1000	1000
4	-	1050	1060	1070
5	-	1070	1100	1120
6	850	-	740	720
7	890	-	780	770
8	900	-	790	790

*Assumes 80% plant factor.

cross sections obtained from ENDF/B-IV file were added to complement the original HANSEN-ROACH cross sections^{3.12} used in the code. The cross sections used with SIZZLE are not the best. The results are not very accurate and become even less so with increased neutron moderation (lower energy spectrum). However, the use of SIZZLE leads to conservative estimates of fuel production rates, and these preliminary results have been used for the scoping study.

Table 3-2 shows the initial fuel production rate for 8 different reactor lattices, as calculated with TWOTRAN.

Table 3-3 shows the net increase in the amount of fissile material after a 1-yr irradiation period for three initial enrichment values. This calculation was performed with SIZZLE for the same lattice cases and operating conditions given in Table 3-2.

As expected, the fissile fuel production rate is greater with lower moderation (harder neutron energy spectrum). It is also interesting to note that in the UO_2 -Pu cases, the production rate remains essentially unchanged with increased enrichment, whereas in the ThO_2 - ^{233}U cases the production rate decreases with increased enrichment. This is explained by the smaller η value for ^{233}U in the 1- to 10-MeV energy range of interest. On the other hand, the

Table 3-4

Initial and Final Blanket Thermal Power
for a 1-yr Irradiation Period
Starting with Different Enrichment Values

Pb-Bi target, fuel elements blanket
(0.3-A 1.5-GeV proton accelerator).

Case No.	Initial enrichment, %			
	0	0.7	2	3
1	-	300	430	570
		460	620	780
2	55	-	420	850
	220		610	1040
3	-	380	530	620
		550	660	740
4	-	380	530	650
		540	680	810
5	-	450	640	810
		620	830	1030
6	50	-	750	1030
	500		1000	1260
7	60	-	550	770
	310		710	940
8	80	-	560	820
	260		730	1040

thermal power generated in the blanket is strongly dependent upon the enrichment level. Table 3-4 shows the initial and final (after 1-yr irradiation) power levels due to the fission reactions in the blanket fuel assemblies for the case of a 0.3-A 1.5-GeV proton linac. These cases are calculated with the SIZZLE code and HANSEN-ROACH cross sections; thus the values in the table are somewhat lower (10 to 20%) than those calculated with the TWOTRAN code with the DLC-2 cross-section library. As the enrichment in the assembly increases, so does the thermal power generated. Table 3-4 shows only the initial and final power levels, without intervening fuel shuffling. Table 3-5 shows the total burnup during 1 year under the same conditions. Because of the nature of the neutron energy spectrum produced in the primary target, the result of this calculation is quite different from that obtained with a fission (fast or thermal) reactor.

Figure 3.6 shows a typical peak power density distribution (Case 1 of Table 3-2) as calculated by TWOTRAN. As expected, the power peaks closest to

Table 3-5

Average Burnup (MWD/ton)*
for a 1-yr Irradiation Period
Starting with Different Initial Enrichment Values

Pb-Bi target, fuel elements blanket
(0.3-A 1.5-GeV proton accelerator).

Case No.	Initial enrichment, %			
	0	0.7	2	3
1	-	1150	1580	2010
2	350	-	1340	2470
3	-	1450	1790	2040
4	-	1420	1840	2170
5	-	1630	2220	2750
6	750	-	2280	2990
7	500	-	1650	2230
8	450	-	1700	2460

*Assumes 80% plant factor.

the primary target and decreases rapidly with distance. No attempt was made at this time to flatten the power distribution through fuel shuffling strategies, though considerable improvement could be made this way. These preliminary calculations do suggest, however, that in the moderated (driven reactor) case the peak power density will be very high, and this would be a difficult problem with the system, at least with this specific target configuration. However, for the undermoderated cases, the peak power density appears to be reasonable (200 to 300 kW/liter).

Figure 3.7 shows the increase in ^{239}Pu in the fuel with respect to time and to distance from the neutron source. It is assumed that no fuel shuffling takes place. The plot covers a 1-yr irradiation period starting with 2% enriched fuel. Plutonium concentration reaches more than 4% during the period.

3.4 Burnup Calculations

The previous discussion considered the fissile fuel production process involved in the target blanket of the linear accelerator. If the fissile fuel so

Figure 3.6. Power density distribution in blanket.

Figure 3.7. ^{239}Pu buildup in blanket.

produced is to be burned in a conventional reactor without intervening processing, the effect of fission product buildup in the burn cycle is of primary concern. In the symbiosis of the LAFER with burner reactors there are many fuel cycle variations possible. For the purpose of this study we have chosen to utilize the LAFER to provide UO_2 -based fuel for conventional LWR's to achieve 60,000 MWD/ton burnup.

The scenario for the fuel cycle studied which includes the LAFER in the equilibrium mode is as follows. As is normal LWR practice, one-third of a core is replaced at each fuel reloading (roughly each year).

For any one-third of a reactor core:

1. Enrich fresh fuel at conventional enrichment plant to 2% ^{235}U .
2. Fabricate fuel.
3. Increase fissile material content in the LAFER to 3.2%.
4. Burn fuel in LWR for 30,000 MWD/ton ($\sim$ 2% fissile material content in discharged fuel).
5. Reenrich burnt fuel in the LAFER and enhance fissile material content to 3.2%.
6. Burn regenerated fuel in LWR for 30,000 MWD/ton.
7. Store spent fuel after second cycle (total 60,000 MWD/ton burn).

In principle, one could consider irradiating fuel in the LAFER through more than two cycles, with correspondingly better fuel utilization and better economics. However, uncertainties concerning the effects of fission product buildup, and of radiation and corrosion damage to materials, indicate that we should not consider more than about 60,000 MWD/ton burnup at this time.

The effects of fission product buildup during the LAFER irradiation cycle and the subsequent LWR burn cycle have been calculated with the EPRI CELL code.^{3,14} The calculation followed the history of the fuel during the two cycles described above to the total of 60,000 MWD/ton burnup in the LWR. The numerical code does not take into account fuel shuffling. Figure 3.8 shows the variation of the multiplication factor k_∞ for three different lattices whose moderator-to-fuel-volume ratios are 2.5, 1.67, and 1.17, respectively. The tighter lattices (less moderation) yield lower k_∞ values. As expected, k_∞ drops with burnup time and it is also lower during the second burn cycle because of the preceding accumulation of fission products in the fuel. For the LWR case, where $\text{VH}_2\text{O}/\text{VUO}_2 = 1.67$, k_∞ drops to <1 during the second burn cycle indicating the need for the addition of fresh fuel and good fuel shuffling strategy. The

Figure 3.8. Multiplication factor variations during enrichment and burn cycle.

Figure 3.9. Fissile material buildup and neutron absorption due to fission products during enrichment and burn cycle.

k_{∞} value is much lower during the LAFER irradiation cycle (0.5 to 0.6) because of the low moderation of the coolant used.

Figure 3.9 shows the concentrations of ^{235}U , ^{239}Pu , and ^{240}Pu together with the neutron absorption due to the fission products during that same fuel buildup/burnup history. ^{239}Pu builds up in the LAFER irradiation cycle and is burned subsequently in the LWR. ^{240}Pu , on the other hand, builds up continuously during the irradiation and burn cycles. It is interesting to note that thermal and epithermal neutron absorption due to the fission products is decreased by the LAFER irradiation cycle. This indicates that some of the fission products are transformed from strongly neutron-absorbing isotopes to weaker absorbing ones during the process.

4. FUEL CYCLES AND SYSTEMS OPTIONS

4.1 Fuel Cycle Options

In the course of this study, a number of fuel cycle options were evaluated to assess their potential advantages over the conventional LWR and HWR systems. This survey indicated that although many different reactor types can be considered, none in conjunction with the fuel regenerator presents any overwhelming advantage over the conventional LWR or HWR fuel cycles. Because of the enormous incentive to remain within existing technology, we have concentrated on those conventional systems, mainly the light-water-cooled UO_2 -fueled system (PWR or BWR) and the heavy-water-cooled UO_2 -fueled system (CANDU). In addition, we have investigated the production and use of the ^{233}U denatured ThO_2 fuel cycle with these two types of reactors. The use of ThO_2 fuel promises much better fuel utilization in the burner reactor (0.8 to 0.9 conversion factor) than conventional UO_2 fuel. Also, this cycle possesses certain features regarded as attractive in relation to proliferation resistance, which is one of the stated goals of this study.

The symbiosis of the linac with the fuel cycles and reactors mentioned above can lead to three different options: the driven reactor (power producer), the fuel enricher/regenerator (in situ fuel enricher), and the fuel producer (which requires reprocessing). These three systems have been evaluated, and it has been concluded that all are technically feasible. However, the driven reactor option seems less desirable technically and economically than the other

two approaches. On the other hand, the fuel producer option, although the simplest technically, is not further developed here because of the fuel reprocessing requirements. The fuel regenerator appears to be the most promising of the three systems, combining the advantages of stretching our natural uranium resource and providing for the possibility of operating with denatured fuel in a nonreprocessing cycle. In addition, the fuel regenerator appears to offer economic advantages over the fuel producer.

The following is a brief description of each of the three systems, listing their advantages and disadvantages.

4.2 Linear Accelerator Driven Reactor (LADR)

This concept involves the use of a high-energy proton beam from a linear accelerator impinging on a primary Pb-Bi target, thus producing a large neutron flux. The primary target is surrounded by a PWR lattice of metal-clad rods of fertile-fissile material cooled with the appropriate moderator. The fuel assembly is contained in individual pressure-tubes. The system is characterized by the optimization of the time-integrated production of thermal energy for conversion to power. The fertile-fissile elements of interest include metal-clad natural and depleted uranium, and thorium metal. The elements would be burned until they could no longer contribute the necessary rate of power production to the system in a once-through cycle without reprocessing. This system is referred to as the linear accelerator driven reactor because the production of power would be the result of amplifying the initial neutron source generated by the beam, by subcritical multiplication.

The linear accelerator driven reactor systems studied are shown in Figure 4.1. The caption lists their main advantages and disadvantages. It should be pointed out that in addition to the main disadvantage of requiring each power reactor to have its own accelerator tied to the power grid, the system suffers from a basic engineering limitation created by the nature of the neutron source. Since the neutrons are created outside the reactor fuel, the interface between the source and the fuel sees a large neutron flux. Because the system is optimized for power production, fuel near the interface will be subject to very high power density, despite the fact that the neutron source design provides for a diluted flux of $\sim 10^{14}$ neutrons $\text{cm}^{-2} \text{sec}^{-1}$. Peak power densities have been calculated to reach 1.5 MW/liter in conventional PWR fuel-moderator assemblies.

Figure 4.1. Linear accelerator driven reactor (LADR). Advantages: (1) once through, no reprocessing; (2) power from natural uranium fuel in a H₂O- or D₂O-moderated subcritical assembly; (3) subcritical = safety advantages; (4) uranium resources stretched; (5) enriched fuel not required. Disadvantages: (1) an accelerator required by each power reactor; (2) increased complexity; (3) long burnup.

This problem may be specific to the target design we have analyzed; other approaches to the design might circumvent the difficulty.

The economics of the driven reactor concept depend strongly on the power gain achieved in the target-reactor assembly. The electric power multiplication in the linear accelerator driven reactor can be expressed by

$$P_{TA} = \epsilon_a \epsilon_p (Y_n 0.2 \frac{k}{v} \frac{1}{1-k} + 1.0) ,$$

where

- P_{TA} is the ratio of total electric power produced by the target to the electric power used for the linear accelerator,
- Y_n is the neutron yield/proton from the accelerated beam,
- v is the number of neutrons per fission,
- k is the multiplication factor of the driven reactor,
- ϵ_p is the efficiency of producing electricity from thermal energy,
- ϵ_a is the accelerator efficiency for converting electricity to a beam of protons or deuterons.

It has been shown^{4.1} that for the linac driven reactor to be economically viable, P_{TA} should be more than 5. Assuming $Y_n = 50$, the value of k , integrated over the burn cycle, has to be $\sqrt{0.9}$ if this requirement is to be met.

Figure 4.2 shows the value of k_∞ for several CANDU-type reactors. The UO₂ fuel cycle appears to lead to values of k_∞ somewhat below the criterion whereas, in this design, the thorium cycle can be driven to criticality.

Figure 4.2. Multiplication factors of driven CANDU reactor.

4.3 Linear Accelerator Fuel Enricher/Regenerator (LAFER)

This system utilizes the same physical arrangement of target and blanket as the driven reactor. It is different, however, in that instead of optimizing power production by the proper choice of moderator and moderator density in the lattice, this system optimizes the fissile material production rate in the lattice, at the expense of power production. This is achieved by removing as much of the moderator as possible, while maintaining cooling, thus hardening the neutron energy spectrum within the lattice. The lattice consists of fabricated fertile fuel assemblies whose reactivity would be increased by neutron capture and plutonium production to the desired level for subsequent burnup in a separate power reactor. After a burn cycle, the fuel elements would be cycled again through the regenerator for reenrichment for further utilization in the power reactor. The process would be repeated to the extent allowed by limitations of fission product buildup and radiation damage to cladding and structural materials. No reprocessing is involved.

The fuel enricher/regenerator concepts studied are shown in Figure 4.3. The advantages and disadvantages are listed in the caption. The greater portion of this report deals with the fuel regenerator system, for it is felt that this

Figure 4.3. Linear accelerator fuel enricher/regenerator (LAFER). Advantages: (1) no reprocessing; (2) no departure from existing power reactor practice; (3) provision of means of ^{233}U denatured production; (4) several reactors can be supplied by one accelerator; (5) uranium resources stretched; (6) fuel producer and power reactor kept separate, with resulting reliability benefit; (7) enrichment facilities' requirement reduced; (8) relatively little extrapolation required by technology in target and power reactors (LWR-BWR). Disadvantages: long burnup options ($>60,000/\text{MWD/ton}$) may be limited by fission product buildup and radiation damage to cladding and structural materials.

concept offers the most advantages and benefits for our nuclear economy within the context of extending the fuel supply of an existing PWR reactor system and of addressing the issue of nonproliferation.

It is to be noted here that power density in the fuel regenerator does not reach a level causing a severe problem, because of the lack of coolant moderation and resulting low fission rate.

4.4 Linear Accelerator Fuel Producer (LAFP)

This third system has been considered for use, in conjunction with reprocessing facilities in secured areas, for the production of ^{233}U . As for the fuel regenerator, the production rate of fissile material is optimized. How-

Figure 4.4. Linear accelerator fuel producer (LAFP). Advantages: (1) no departure from existing reactor technology; (2) provision of means of ^{233}U production; (3) several reactors can be supplied by one accelerator; (4) full use made of fuel resources as breeders. Disadvantages: reprocessing required.

ever, the fuel producer does not have the constraint of irradiating the specific geometry of prefabricated fuel assemblies, so that in principle higher rates of fissile material production can be achieved.

The fuel producer is outlined in Figure 4.4. Since this option would involve reprocessing, it has not been studied in detail and is not being considered in this report. However, should reprocessing be contemplated in a denatured fuel cycle economy, the linear accelerator fuel producer is a good candidate for the production of the ^{233}U fuel required for this cycle. As the constraints of irradiating fabricated fuel assemblies to be used afterwards in LWR's are removed, we would expect to be able to increase the fuel production rate somewhat above those quoted for the fuel enricher/regenerator (see Tables 3-1 and 3-2).

5. ENGINEERING

5.1 Choice of the Accelerator Type and Its Parameters

Protons and heavier particles can be accelerated to energies >1 GeV most economically in synchrotrons. These accelerators are limited, however, to

Table 5-1
Parameters of Existing Relevant Accelerators

Accelerator		Ion type	$E_{\max}$, MeV	$I_{\max}$, A	Duty factor, %	Status
Laboratory	Type					
LASL-LAMPF	Linac	Proton	800	0.020	12	Op
LLL - MTA	Linac	Deut.	500	0.320	100	Prop
BNL-AGS	Linac	Proton	200	0.2	0.5	Op
FNAL	Linac	Proton	200	0.3	0.2	Op
HEDL-HFNS	Linac	Deut.	35	0.1	100	Prop
CERN-ISR	Storage rings	Proton	3×10^4	40.1	-	Op
BNL-ISA	Storage acceler.	Proton	4×10^5	6.0	-	Prop

pulsed operation and very low average beam intensity. The linear accelerator is the only type of accelerator capable of accelerating a steady, continuous beam current of over 100 mA. With linear accelerators, proton currents of 0.3 A have been accelerated, and the possibility is available of accelerating even higher currents if necessary. Table 5-1 lists a number of relevant accelerators (most of them operating) with their parameters of interest. Storage rings are listed to show the very large currents (large for accelerators) that can be circulated and contained for very long time periods essentially without beam loss.

Besides its unique capability of accelerating high currents, the linear accelerator has the added advantage of: (a) providing the most efficient means of energy conversion (electrical energy to beam energy), and (b) having the lowest beam loss factor (particles lost during the acceleration process).

Minimization of beam loss is an important criterion for this application. In all circular accelerators a certain amount of beam loss is inevitable at injection and extraction, which makes all equipment located in these areas highly radioactive. For the very high beam power application considered here, even the lowest fractional beam loss that has been achieved is still several orders of magnitude higher than that which is acceptable. With a linac, there is no extraction loss, and since the injection energy is typically less than 1 MeV, in-

jection losses are manageable. In addition, because the beam travels in a natural straight line, the loss during the acceleration is much lower than in all other types of accelerators. These simple considerations lead directly to the conclusion that the linac is the optimum choice.

The charged particles are produced in an ion source. They are first accelerated to an energy which is typically about 750 keV in a dc electrostatic accelerator. The 750-keV beam is then injected into the linac and accelerated to the final energy by the rf field in the cavities. New developments in linear accelerators indicate that the injection energy required for high beam currents might be relaxed. In particular, work carried out in the USSR,^{5.1,5.2} shows that 100-kV injection might be possible. Many different linac cavity designs are available, each having its own merits for special applications. The choice of specific structure and rate of acceleration per unit length is based on cost, beam-loading efficiency, and other construction and operating features.

Above the lower limit of beam energy set by ionization loss, the neutron yield in the target is roughly proportional to the beam power, which is in turn proportional to the rf power for a given beam loading. A major part of the cost of the linac is that of the rf supply. This cost is proportional to the beam power, hence the fuel yield. Thus for a given fuel yield the total accelerator cost is primarily affected by the cost of the radio-frequency system. The remaining costs, such as those of the cavity structure, the control and monitoring system, the linac housing, etc., amount to about 30% of the total. The costs of these components are generally proportional to the length, hence the energy, of the linac. Therefore, for a given neutron yield, the total cost of the linac is somewhat lower for lower energy and higher current. Appendix E describes the accelerator costs parametrically.

A beam current of 300 mA has been obtained at Fermilab, though only in a pulsed mode. However, during the pulse a steady state is reached insofar as the transfer of power from the rf supply to the beam is concerned. The pulse length is limited only by the size of the rf supply. Hence, we consider a 300-mA cw beam current a realistic goal attainable with high confidence. At LASL a particle energy of 0.8 GeV has been achieved in a linac, and further acceleration will require only iteration of stages that are now in use. This demonstrates the total practicality of attaining energy >1 GeV. Our conceptual design is based on an accelerator that will provide a beam of 1.5-GeV protons at 300 mA to produce, conservatively, more than 1 ton of fissile fuel per year. This is

enough fissile material to support three reactors. These parameters are not optimized, and it is likely that optimization will result in a somewhat different design.

Higher beam energies are well within the expectations of linac technology. A higher energy, for the same beam current, will yield a correspondingly higher fuel production rate with little or no reduction in certainty of success. However, the unit cost of fuel produced will not be significantly lower.

It is to be noted here that linear accelerators at BNL, CERN, Fermilab, and LASL have impressive records for operational reliability (>90% of scheduled time with an overall plant factor of better than 80%). It is expected that the linac part of the fuel regenerator facility will at least match this performance.

5.2 Comparison of Proton and Deuteron Beams

Figure 3.3 shows that at the same energy an incident deuteron causes a higher neutron yield than a proton. This is especially so for energies <1 GeV. The difference is due to the neutron in the deuteron, which suffers no ionization loss in the target, and which has no Coulomb barrier in penetrating the target nuclei. At 1 GeV the advantage of deuterons over protons appears to be more than 20%, and at 2 GeV this advantage decreases to about 5%. More extensive and exact computations as well as direct experimental verifications are required for a definitive conclusion.

In the linac, the higher mass and correspondingly lower velocity of the deuteron lead to stronger defocusing by the rf and the space charge forces. Therefore, stronger-focusing quadrupole magnets are needed. The most straightforward approach is to lower the rf frequency by a factor of $\sqrt{2}$, which then, in spite of the lower velocity, allows the use of longer and stronger quadrupoles. Lower frequency requires proportionally larger cavity diameters and reduces somewhat the beam-loading efficiency. These factors increase the capital cost of the accelerator somewhat over that of a proton linac (see Appendix E). However, the major concern with a deuteron beam is, perhaps, the much higher radiation and activation levels it can produce at lower energies.

Activation of the accelerator structure by stray beam components must be kept low enough for "hands on" maintenance. Additional costs associated with remote handling facilities and increased downtime due to performing maintenance

and repairs remotely would jeopardize the viability of the concept. In evaluating the levels of radiation and activation, the beam losses can be classified in three groups. First, to minimize the beam loss at higher energies in the linac, a sizable fraction of the low-energy beam which cannot be manipulated into the acceptance of the linac must be stopped at the injection point. Second, even with extreme care in injection matching, some beam loss must be expected in the first cavities of the linac. We estimate this loss to be $\sim 1 \mu\text{A/m}$ of length along the linac up to an energy of ~ 10 MeV. Third, above 10 MeV the beam loss must be minimal. A loss rate of no more than 1 nA/m is required and should be achievable.

For a proton beam, the losses below 10 MeV will yield no discernible radiation and residual radioactivity. However, low-energy deuterons will produce 3.27-MeV neutrons via the reaction, $d + d \rightarrow {}^3\text{He} + n$. The residual radioactivity produced by this reaction will come mainly from the 312-day half-life isotope ${}^{54}\text{Mn}$ produced in iron by the reaction ${}^{54}\text{Fe}(n,p){}^{54}\text{Mn}$. The equilibrium dose rate is estimated to be no more than 1 mrem/hr. This also holds true for the beam loss at <10 MeV, so that this problem does not preclude the deuteron linac.

The problems of radiation and induced activity due to 1 nA/m of beam loss beyond 10 MeV are similar for protons and deuterons. This loss rate will yield a residual radioactivity of 10 to 50 mrem/hr close to the linac structure during operation, which will have decreased substantially after a few hours of cooldown.

It is clear that a deuteron beam, for a given energy, will yield a higher rate of fissile material than a proton beam. However, for this conceptual design, we are assuming a proton linac because of its lower cost and fewer uncertainties. The use of a deuteron linac is not entirely ruled out at this time. The potential 20% gain in production rate of fissile fuel is attractive and must be weighed against the added linac cost and complexity. Also, definitive experiments are needed to assess the relative neutron yields of protons and deuterons.

5.3 Linear Accelerator Design

a. The Injection System. Although no large-scale 100% duty factor accelerator exists today, the technology for producing the dc ion beams required

Figure 5.1. Maximum allowable beam current vs linear accelerator frequency.

for these linacs has been under development for the last 20 years, spurred by the need of neutral beam heating for fusion devices and the construction of DT neutron sources.

A 350-mA molecular H_2^+ source was developed at ORNL in the 1960s as part of the DCX project. The total current was limited by that of the H_2^+ source. At the same laboratory, proton currents up to 1 A were obtained, with more than 80% of the beam composed of protons.^{5.3} These beams were, however, not accelerated, and no measurement was made of the emittance of the beam. An ion source under development for the RTNS project at Livermore^{5.4} has produced a 150-mA dc beam of d^+ at 250 keV. The normalized emittance of 0.02 cm mrad obtained from this source is more than adequate for injection into a linac.

The injection system of present day linear accelerators is inefficient. In order to match the dc beam produced by the ion source to the acceptance of the linac, the beam has to be preaccelerated to ~750 keV, then manipulated to give it the proper characteristics for injection. Figure 5.1 shows the maximum theoretical beam current vs cavity resonant frequency for a 750-keV injection of both protons and deuterons. In practice, allowing for some safety margin, we design for currents of only half this magnitude. As the cavity frequency decreases, its physical size and beam aperture increase, allowing higher currents. Correspondingly, the accelerator costs go up with increased size. To

achieve these parameters, the beam must first be scraped clear of all halos and must be precisely tailored in emittance and distribution in the 6-dimensional phase space to match into the linac acceptance. Standard schemes for beam matching and collimating to do this exist. However, the beam loss in the process can be as high as 50%.^{5.5}

New developments in linear accelerators indicate that this problem may be solved with a short section of a new type of accelerator^{5.1,5.2} promising great improvement in its ability to accept more than 90% of the ion source dc beam.^{5.6} At the same time, the injection energy required would be lowered to about 100 keV. This preaccelerator would accelerate the beam to 3 to 4 MeV and would tailor the beam characteristics for injection into the conventional drift-tube linac. The ability to raise the injection energy into the drift-tube linac is of great importance. The maximum current that a given linac structure may accelerate is determined to the first order by the injection energy. Thus, being able to inject at higher energy (> 1 MeV) makes for a safer, more conservative design.

The successful design and development of the injection system is of prime concern, because it is a determinant in the generation of beam losses along the linac.

b. Low- β Linac Section. For particle velocities below about one half the speed of light ($\beta = v/c = 0.5$), ($E \lesssim 150$ MeV), the standard drift-tube cavity structure^{5.7-5.9} with post-coupler compensation similar to that of the linacs at LASL and Fermilab, or the multistem compensation used at BNL, is close to optimum for beam-loading efficiency.

To handle the high radiation level resulting from the high beam current, it is essential that only radiation-resistant inorganic materials be used in the linac cavities. Only metal vacuum seals should be used. All electromagnetic quadrupole magnets must be constructed with ceramic insulators and be water cooled for dc operation.

With the much higher rf power dissipation in the cavities, the cooling and temperature control become important. The temperature of the cavity walls and drift-tubes must be stabilized to within 1°C in order to avoid excessive drifts of the cavity resonant frequency. This is most demanding during turnon and turnoff, when power dissipation varies between zero and the maximum level. It is very important to achieve cooling parameters that eliminate the possibility of transients with long time constants.

Figure 5.2. Brookhaven 200-MeV proton linear accelerator.

Figure 5.3. Linear accelerator facility cross section.

The choice of rf frequency and acceleration rate is based on considerations of cavity excitation power, beam loss during acceleration, reliability of operation, and overall cost. For protons, a first look gives values in the neighborhood of 200 MHz and 1.5 MeV/m, and for deuterons, 100 MHz and 1 MeV/m. Figure 5.2 shows the 200-MeV 200-MHz Brookhaven proton linac (viewed from the low-energy end).

Figure 5.3 shows a cross section of a linac facility, the linac housed in a shielded tunnel with its adjacent power equipment and service gallery. This cross section is typical for the entire linear accelerator, both low- β and high- β sections.

c. High- β Linac Section. Above $\beta = 0.5$ ($E \approx 150$ MeV), the shunt impedance of the drift-tube linac structure degrades rapidly, and one must switch to a different cavity structure, one similar to that of the side-coupled cavities used at LAMPF. However, for this application, because of the large amount of power that is transferred to the beam, a structure with tighter and more symmetric coupling between cavities is required. A $\pi/2$ -mode disc-and-washer structure^{5.10} has been operated at the Radiotechnical Institute, USSR, and shown to have a very high coupling coefficient. The π -mode cross-bar structure^{5.9} developed at the Rutherford Laboratory has similar advantages. These structures have large bandwidths and are better able to cope with high power levels and high beam loading. The same considerations of temperature control and of using radiation-hardened materials as for the drift-tube section apply to this coupled cavity section as well.

Because of the higher initial particle velocity, one can operate this section at an rf frequency several times higher than that of the drift-tube section, thereby obtaining a higher acceleration rate and higher beam-loading efficiency. Initial examination shows that an acceleration rate of 2 MeV/m and a beam loading of more than 70% can be attained.

Although drift-tube linacs have operated at high currents (200 to 300 mA), the experience with high- β structures at high currents is limited. However, electron analogs have been operated successfully at Chalk River.^{5.11} Since this section of the accelerator is by far the largest (90% of the linac), a substantial development effort will be required to optimize its cost and reliability.

d. Radio-Frequency Systems. High-power rf systems using both klystron and gridded tubes have been used for pulsed accelerator applications with good success. At low frequencies gridded tubes are preferred, and at frequencies

>200 MHz the klystron is preferable. High-power gridded tubes and klystrons are available commercially. These tubes give output powers of 0.5 MW average with more than 20,000-hr lifetimes. However, because of the amount of rf power required, and the incentive for high conversion efficiency, tubes expressly developed for this application will be needed. The development effort will be focused on obtaining tubes with high ac-to-rf power conversion efficiency and high output power. The capital cost of rf systems decreases with increasing output power per amplifier unit; therefore, output powers of 1- to 2-MW power tubes should be feasible and desirable.

An entirely new type of rf amplifier called the gyrocon has been developed at the Institute of Nuclear Physics in Novosibirsk, USSR, and is being studied at Los Alamos.^{5,12} In this device, rf power is derived in a circular waveguide from a rotating dc electron beam instead of in a linear cavity from a bunched beam as in the klystron. The overall conversion efficiency from dc to rf power may be well above 90%. In addition, because of the circular geometry the range of practically attainable rf frequency is rather broad. Frequencies from 50 to 400 MHz have been obtained. If the gyrocon proves to be practical as an rf power supply, it may greatly reduce the cost of the rf system.

Compared with the presently operating pulsed systems, continuous wave operation brings tremendous simplification to parts of the rf system. Energy storage requirements are minimal, and feedback control of cavity amplitude and phase is greatly simplified. However, fast crowbar and other protection features may be more difficult. These must be investigated in detail and provided in the design. Proper provision should also be made for cavity sparking and the subsequent system recovery.

Redundancy in the rf system is essential for reliable operation. Therefore, it is important to examine the manner in which rf power is fed from the supply units to the linac cavities. Each cavity may be fed directly from individual rf supplies, or it may be fed from a manifold which is, in turn, fed by the supplies. The latter arrangement has the advantage that the excitation pattern of the cavity is not affected by the failure of a redundant supply unit. Both the rf feed mode and the optimal degree of redundancy should be studied in detail.

e. Control and Diagnostic Systems. Computer control systems for accelerators are well developed on existing machines, and should prove no problem for this application. Distributed intelligence systems, where peripheral

Figure 5.4. PWR-type target-blanket assembly.

minicomputers or microprocessors carry out control functions at the location of the device and control is distributed by a central processor, appear to be the most promising of many possible configurations. Fast control, for protection against beam loss and for the control of cavity phase and amplitude levels, must be accomplished locally with analog systems. Adequate design for these systems has also been demonstrated on existing accelerators.

Beam diagnostic instrumentation which allows monitoring of the beam parameters during accelerator operation must be developed for dc beams. Continuous information on beam position, size, and phase relative to the cavity rf phase is essential for reliable operation. The general technology is available for instrumentation to perform these functions.

5.4 Target-Blanket Assembly

The configuration of the LAFER depends on the type of fuel assembly to be enriched and/or regenerated. Figure 5.4 shows an overall view of the target-blanket configuration used for enriching or regenerating PWR fuel assemblies. Similarly, Figure 5.5 shows the configuration based on CANDU-type fuel

Figure 5.5. HWR-type target-blanket assembly.

assemblies. Horizontal cross sections through the PWR and the CANDU versions are shown in Figures 5.6 and 5.7, respectively.

In both cases, the blanket consists of fuel assemblies contained in pressure-tubes (arranged vertically for PWR fuel and horizontally for CANDU fuel) surrounding the primary target. This, in turn, consists of liquid metal Pb-Bi columns that fall from the jet spray nozzles at the top of either target-blanket assembly. The proton beam, initially circular in shape, is converted into a narrow ellipse by the final set of beam transport magnets and enters a thin vertical slot in the target-blanket assembly where it interacts with the liquid Pb-Bi columns. An alternative to defocusing the beam in this fashion could be to sweep the beam vertically to cover the entire target area.

Apart from the obvious differences in the physical configurations of these two LAFER designs, the main operating difference is that the CANDU version allows on-line refueling whereas the PWR version requires shutdown for refueling and/or reshuffling. In the former case, the fuel assemblies would be loaded and unloaded at the ends of the horizontal pressure-tubes by fuel-handling machines similar to those used in CANDU reactors. If it proves sensible to load and unload fuel on line, as in CANDU reactors, there would be substantial benefits in terms of operating plant factor.

Figure 5.6. PWR-type target-blanket assembly cross section.

Figure 5.7. HWR-type target-blanket assembly cross section.

Because of the exclusive use of light water reactors in the U.S., this study has been limited primarily to the analysis of enriching and regenerating LWR fuel, in general, and PWR fuel (Westinghouse RESAR-3 fuel assembly), in particular.

A practical target-blanket assembly has to satisfy three fundamental constraints:

1. The power density in the blanket fuel assemblies must be compatible with available cooling methods.

2. The neutron source current at the target-blanket interface should be as uniform as possible. This minimizes power peaking and variations in fissile material buildup.
3. The neutron leakage from the target-blanket assembly should be small, so that leakage losses do not significantly reduce the fissile fuel production rate.

Because of the short range of protons in Pb-Bi (90 cm at 1.5 GeV), a Pb-Bi target at normal density would generate excessively high blanket power densities in the first part of the target unless a very large target area were used. This would then cause a very large neutron leakage in the backward direction, as well as a very nonuniform neutron source distribution. The maximum neutron current from a Pb-Bi target consistent with good cooling capability in the blanket is on the order of 10^{14} n/cm² sec. For 450 MW of beam power and a neutron yield of 45 per proton at 1.5 GeV, the surface area of the corresponding target would be ~ 100 m². A single-disk Pb-Bi target of 100 m² area and 90 cm thick would not be feasible, because of the high neutron leakage and nonuniformity of the source.

A practical Pb-Bi target-blanket assembly thus requires that the interaction distance of the primary protons in the target be much greater than the stopping distance in liquid Pb-Bi. This is possible if the effective Pb-Bi density along the target path is much less than that of the liquid. Figure 5.6 shows a cross-sectional view of the target-blanket assembly with the Pb-Bi liquid columns. The effective density of the liquid columns increases with distance along the beam path. This spatial distribution compensates for the decrease in neutron production rate as proton energy is lost by interactions with the Pb-Bi columns. In addition to varying the spatial density of the Pb-Bi columns along the proton beam path, the Pb-Bi columns can be "stepped," forming a wedge or triangle pattern to intercept successive portions of the primary beam. These two techniques, used either in conjunction or separately, should allow one to shape the neutron source function along the beam path to any desired level.

The PWR fuel assemblies are contained in 12-in. i.d. Zircaloy pressure-tubes as shown in Figure 5.8. The pressure-tubes are headered at the bottom of the target-blanket assembly. Fuel assemblies are loaded and unloaded from the top of the pressure-tubes into fuel-handling casks. The pressure-tubes project through the tube sheet at the top of the assembly, and are welded to the tube

Figure 5.8. PWR-type pressure-tubes.

sheet to provide a vacuum-tight seal. Plugs at the top of each pressure-tube can be removed for the insertion or withdrawal of fuel assemblies.

Coolant flows up through the crescent-shaped regions between the square PWR fuel assembly and the round pressure-tubes, and then down through the assembly, cooling the fuel elements. The pressure drop of the coolant acts to keep the fuel assembly centered on the guide structure in the bottom of the pressure-tube. A shroud structure is necessary to prevent short circuiting of the coolant flow from the inlet flow regions to a PWR fuel assembly, since the assembly has no external box around it. Such a shroud is shown in the cross-sectional view of Figure 5.8, with attached ribs that support it in the pressure-tube.

Since there is no window between the accelerator and the target-blanket assembly, the pressure-tubes must be vacuum tight and there must be a vacuum envelope surrounding the pressure-tube assembly. This envelope will in effect be a relatively large vacuum tank, on the order of 10 m in length, $\sqrt{4}$ m high, and $\sqrt{3}$ m wide, with the external pressure nominally at 1 atm. Because of the relatively thick structure required to carry external pressure over large flat areas, the reflector and thermal shield will probably have to be placed inside the vacuum enclosure, rather than outside as shown in Figure 5.6. Design of this portion of the target-blanket assembly will not be examined in detail until neutronic studies define the nature of the reflector and shielding required.

The vapor pressure of Pb-Bi inside the vacuum tank will be $\sim 10^{-5}$ Torr. A precise value is difficult to estimate since different portions of the Pb-Bi jets will be at different temperatures (e.g., the inlet temperature at the spray nozzle might be $\sim 300^\circ\text{C}$, and the outlet temperature $\sim 500^\circ\text{C}$), and the pressure-tubes will be at relatively lower temperatures, i.e., $\sim 300^\circ\text{C}$. The colder surfaces will tend to act as condensing "cold fingers" for Pb-Bi vapor, so that the actual vapor pressure will depend on local position within the target-blanket assembly. There will be some transport of Pb-Bi vapor up the accelerator tube, but this can be easily condensed by cooling panels and returned to the liquid Pb-Bi circuit. At 10^{-5} Torr, the mass flow of Pb-Bi into the beam tube is only ~ 1 kg/day per square meter of area.

The design flow rate for the Pb-Bi target jets is set by one of three criteria:

1. Beam-stopping requirements.
2. ΔT of the Pb-Bi flow.
3. Pumping power for the Pb-Bi flow.

The flow rate based on the first criterion can be expressed as

$$(V_T)_1 = f_s W_s x_s v_o,$$

where

x_s = range of protons in Pb-Bi at 100% density cm (~ 90 cm at 1.5 GeV),

f_s = safety factor for proton stopping (e.g., 2),

W_s = effective width at 100% density for good target performance (e.g., 10 cm),

v_o = initial velocity of Pb-Bi as it leaves the jet nozzle, cm/sec.

The initial velocity should be comparable to the free-fall velocity increment acquired after falling through the height of the target, so that the density variation along the Pb-Bi column will not result in an excessive variation in neutron source current. v_o can be derived for a given target height, h_T :

$$v_o = (v_f^2 - 2gh_T)^{1/2}.$$

For representative conditions $x_s = 90$ cm, $f_s = 2$, $W_s = 10$ cm, $v_f/v_o = 2$, and $h_T = 300$ cm, $(V_T)_1 = 1.58 \times 10^6$ cm³/sec, and $v_f = 880$ cm/sec.

The flow rate based on the second criterion can be expressed as

$$(V_T)_2 = \frac{(P_B)_T}{C_P \rho \Delta T},$$

where

- $(P_B)_T$ = target power deposited in Pb-Bi jets in watts,
- C_P = heat capacity of Pb-Bi, J/g °K,
- ρ = density of Pb-Bi, g/cm³,
- ΔT = temperature rise in Pb-Bi jets, °K.

The beam power deposited in the Pb-Bi target can be approximated by

$$(P_B)_T = (P_B)_O \left(\frac{E_B - Y_n E_n}{E_B} \right),$$

where

- $(P_B)_O$ = beam power before striking target in watts,
- E_B = primary beam particle energy, MeV,
- Y_n = neutron yield from target per primary beam particle,
- E_n = average energy of neutrons as they leave the target, MeV.

The above expression neglects energy losses from gamma rays (resulting from various nuclear excitation events) that escape the target. Typically, $(P_B)_T/(P_B)_O$ will be in the range of 0.8 to 0.85. For acceptable conditions $(P_B)_O = 450$ MW, $(P_B)_T/(P_B)_O = 0.85$, $C_P = 0.16$ J/g °K, $\rho = 10$ g/cm³, and $\Delta T = 300^\circ\text{K}$, $(V_T)_2 = 0.77 \times 10^6$ cm³/sec.

For the third criterion, the pumping power losses due to discharging the Pb-Bi jets from the top of the target-blanket assembly are:

$$(P_B)_T = (V_T)_3 \left(h_T \rho (980) + \frac{1}{2} \rho v_o^2 \right) \times 10^{-7}.$$

The first term represents the gravity head loss due to fall through the height of the target, while the second term represents the velocity head loss due to the inlet velocity v_o . In principle, these losses could be avoided by

isentropic deceleration of the Pb-Bi jets, with recovery of the velocity head energy as pressure. In practice, however, it is probably not practical to shape the collecting trough well enough to achieve a significant pressure recovery. The pumping losses will appear as heat in the Pb-Bi circuit, which can be recovered as work at the power cycle efficiency, e.g., 30 to 35%. This recovery should essentially balance the inefficiency in the mechanical pumps that will provide the Pb-Bi pressure head. For this study, the Pb-Bi surface in the collecting trough at the bottom of the target-blanket assembly is assumed to be a free surface with $P = 0$, and with all the kinetic energy of the falling Pb-Bi converted to heat. For acceptable pumping power conditions $(P_p)_T \leq 10^7$ W, $h_T = 300$ cm, $\rho = 10$ g/cm³, $v_o = 440$ cm/sec, and $(V_T)_3 \leq 25.6 \times 10^6$ cm³/sec.

The first criterion appears to be the most restrictive, since it requires the highest flow rate. For the parameters shown, the $(V_T)_1$ rise is 144°K and the pumping power is 0.62 MW. Substantially higher flow rates would be acceptable from the standpoint of pumping power. This would allow higher values of v_o , which would decrease v_f/v_o and make the neutron source density more uniform. The temperature rise also would be reduced. A clear-cut value for the maximum acceptable final velocity of the Pb-Bi jets is not known; it will principally be determined by splash effects. Excessive splash could disrupt the Pb-Bi jets. For the present, v_f is taken as 8.8 m/sec, which corresponds to a v_o of 4.4 m/sec.

Extensive work was carried out on Bi, Pb, and Pb-Bi liquid metal coolants as part of the LMFBR reactor project at BNL. These materials could be satisfactorily circulated for long periods in Croloy piping systems, provided that the maximum temperature differential in the system did not substantially exceed 100°K. If the temperature differential were too large, material would tend to dissolve from hot pipe regions and to redeposit in cold pipe regions. Over long periods this could result in plugging of the piping system.

The maximum temperature in the LAFER system will be in the Pb-Bi collecting pool at the bottom of the target-blanket assembly, while the minimum temperature will be at the exit of the heat exchanger. This temperature differential can be set to an acceptable level either by using the appropriate flow through Pb-Bi jets, or by mixing the hot Pb-Bi jets with colder Pb-Bi in the collecting pool (the colder Pb-Bi would be supplied by a flow bypass around the jet nozzle system). The second alternative would reduce pumping power losses, as well as permit lower flow velocities for the Pb-Bi jets.

Table 5-2
Blanket Coolant Options

Option	Advantages	Disadvantages
Helium	Nonreactive, nonmoderating	Low thermal cycle efficiency with Zircaloy-clad temperature limitations
D_2O Liquid	Intermediate moderation, no dryout concern	Additional cost
2 Phase	Low moderation	Possible dryout concern
H_2O Liquid	Low cost, no dryout concern	Strongly moderates neutrons
2 Phase	Low moderation	Possible dryout concern
Sodium	Low pressure, high thermal efficiency, nonmoderating	Reactive, not compatible with Zircaloy, can be used only in fuel producer

5.5 Blanket Cooling Considerations

The neutronics and burnup calculations carried out for the cases shown in Table 3-2 and Appendix B indicate that, for undermoderated blanket assemblies, the peak power density to be expected in the highest power region is $\sim 200 \text{ W/cm}^3$. This power density drops very rapidly with increasing distance from the primary target. However, the calculated thermal power gradients do not take into account possible fuel shuffling strategies nor the use of strategically located reflectors. Judicious design choices utilizing these power-flattening methods should permit us to achieve a fairly flat power density distribution in the target which at the same time will enhance the thermal power production.

Highly moderated blanket assemblies (H_2O cooled), at the expense of some reduction in the net fuel production rate, can yield peak power densities, $> 1000 \text{ W/cm}^3$ at the primary target-fuel interface. This power density is much greater than that allowed in LWR fuel elements ($\sim 300 \text{ W/cm}^3$ max). Although other design approaches might alleviate the problem, we are not considering these designs at this time.

For the purpose of this document, we assume an undermoderated (fast neutron spectrum) system with a peak power density of 200 W/cm^3 at the primary target-fuel interface and some flattening of the overall power distribution. The total thermal power in the blanket is conservatively estimated to average 900 MW. Proper orificing of individual pressure-tube cooling loops will be used to optimize the power conversion system efficiency. Preliminary calculations indicate that the average power produced in the blanket can be maintained relatively constant during the irradiation cycle.

Since the LAFER optimization for fuel production requires the existence of a hard neutron spectrum in the blanket, the choice of blanket cooling options is limited by moderation considerations. Coolant options for the blanket fuel assemblies are summarized in Table 5-2 which also shows their relative advantages and disadvantages. Of the six options listed, only four really appear to be practical:

1. Helium
2. Liquid D_2O
3. Two-phase H_2O
4. Two-phase D_2O

Liquid H_2O does not appear to be acceptable because it will strongly moderate neutrons from the target and cause excessive fission power, excessive power peaking, and variation in fissile content in the blanket fuel assemblies. Sodium does not appear to be acceptable, although it is an attractive coolant, since it may not be possible to maintain a protective film on the Zircaloy cladding of the LWR fuel elements if they are subjected to a sequence of different coolant environments. That is, each cycle in the LAFER would involve exposure to sodium followed by exposure to water in a PWR.

Helium is an attractive coolant option. However, because of the potential for oxidative corrosion with oxygen and water impurities in the helium, the maximum operating clad temperature in He will probably be comparable to that in a PWR, that is, on the order of 700°F . The helium temperature can be increased further, however, in a Pb-Bi/helium heat exchanger to remove the heat energy deposited in the target. Because of temperature limitations for the liquid Pb-Bi, the maximum helium temperature can thus be increased to about 750°F . This is enough, however, to enable a small degree of steam superheat in the power conversion cycle and thus the cycle efficiency compares favorably with two-phase or liquid water coolants.

If liquid D_2O is acceptable from the neutronics standpoint, it is relatively easy to use it as the blanket cooling. Pressure drops would be

Figure 5.9. Steam volume fraction in two-phase flow vs inlet quality and velocity.

acceptable and thermal cycle efficiency would be comparable to that in HWR's, i.e., ~30%. The additional cost of D_2O coolant would not significantly affect overall system economics. The D_2O inventory in a LAFER is much less than in an existing CANDU reactor, first because it is not required as a moderator, and second because the thermal power of the blanket is much smaller than that of a CANDU reactor. In addition, since one LAFER regenerates fuel for three LWR reactors the cost of D_2O for a LAFER, expressed as an equivalent additional fuel cycle cost for the LAFER/LWR system, will be negligible (i.e., much less than 1 mill/kWh).

Two-phase (vapor plus liquid) mixtures are an attractive option. Either H_2O or D_2O two-phase mixtures could be used. The effective density of the two-phase mixture will be ~10% that of the liquid, and both H_2O and D_2O two-phase mixtures should result in relatively "hard" neutron spectra. The suitability of two-phase mixtures for cooling the blanket of an LAFER is discussed in detail in Appendix D.

Figure 5.10. Pressure drop in two-phase flow vs inlet quality and heat flow.

In general, two-phase mixtures appear to have good characteristics for cooling the LAFER blanket. The effective steam volume fraction in the two-phase mixture is on the order of 90% (equivalent to an effective density equal to $\sim 10\%$ of liquid density) over the range of inlet quality, inlet velocity, and heat flux expected for the blanket fuel assemblies (Figure 5.9). The pressure drop of the two-phase mixture through the blanket is a relatively small fraction of the base pressure, e.g., 10 to 30 psi compared to ~ 1000 psi for the range of conditions expected for the LAFER (Figure 5.10). Further, heat transfer coefficients for the two-phase mixtures are very high, on the order of several thousands of $\text{Btu/hr ft}^2 \text{ }^\circ\text{F}$; and the corresponding film temperature drops are very small, on the order of several degrees F.

The only reservation with regard to the use of two-phase coolants is the possibility of local "dryout" of fuel element surfaces. The excellent heat transfer properties of two-phase coolants depend on evaporation of thin liquid films on the heat transfer surfaces, which are continually replenished by transfer of liquid from the two-phase mixture. Experience with two-phase coolant mixtures indicates that this does not appear to be of serious concern unless very high exit qualities (>0.8) are employed. The exit quality of the two-phase mixtures in a LAFER would be kept well below 0.8, so that "dryout" should not occur.

5.6 Power Conversion Systems

The structural makeup and the relative arrangement of equipment for heat transport and power conversion depend on the cooling option selected for the LAFER blanket and target. For the purpose of this study, three options of blanket cooling (helium, liquid D_2O , and two-phase H_2O) have been analyzed in some detail. The primary target used in all cases consists of liquid Pb-Bi.

The thermal energy from the LAFER reactor is taken equal to 1350 MW(t) and is obtained as a combined output from the blanket (900 MW(t)) and the Pb-Bi target (450 MW(t)). Indirect cycles are used for the helium and liquid- D_2O blanket cooling options. A direct cycle is used for the two-phase H_2O blanket cooling.

In all cases, the LAFER and its associated heat exchangers, piping, pumps, etc. would be located in a containment shell similar to that for LWR's. Figure 5.11 shows one possible arrangement of a LAFER for the regeneration of PWR fuel assemblies using two-phase H_2O for blanket cooling. Penetration through the containment vessel, the accelerator beam tube, and steam lines could be closed by fast-acting isolation valves, if necessary. Construction of the LAFER and its associated systems inside containment would be similar to that for LWR's.

Overall containment requirements will probably be comparable to those for LWR's though the lower thermal rating of the LAFER's, as well as the use of separate pressure-tubes for the fuel assemblies, may result in somewhat less demanding requirements.

a. Helium Cooling. Figure 5.12 shows the flow diagram of the indirect cycle assumed for the helium cooling of the blanket. The pressure-tubes containing the fuel assemblies are divided into two separate banks that are connected in series on the helium side. Because of fuel cladding temperature constraints, the helium temperature at the outlet of each bank of pressure-tubes is limited to 700°F. This is accomplished by intermediate cooling of helium in the subcooled portion of the steam generator, before entering the second bank of pressure-tubes. Upon exiting the second bank, helium is again cooled in the second portion of the steam generator, providing the bulk of the latent heat of vaporization for the water. The helium is then sent to a Pb-Bi helium heat exchanger where it removes the target heat (450 MW(t)), increasing its temperature to about 747°F. Finally, the helium loop is closed through the superheating portion of the steam generator. The two-stage cooling of the

Figure 5.11. LAFER target system and containment building.

Figure 5.12. Flow diagram for helium cooling.

blanket permits balancing the relative thermal inputs from the fuel assemblies which have different power densities. There are two primary loops, each having a helium circulator. To obtain good heat transfer characteristics, the helium pressure is maintained at about 100 atmospheres. The helium circulators operate at 480°F and are located upstream from the second bank of pressure-tubes of each loop.

The superheated steam is supplied to the throttle valves (975 psia) of a tandem-compound four-flow turbine which develops the mechanical energy to drive the electrical generator.

Because of the small degree of superheat, steam reheat is necessary so that some main steam is supplied to the reheater. A turbine bypass system is provided to pass up to 50% of full steam flow directly to the condenser to reject heat during start-up, shutdown, and other transient conditions where part or all of the steam must be bypassed. A four-stage regeneration system is provided to heat the feedwater to 350°F.

The gross and net electrical outputs are 486 MW(e) and 455.4 MW(e). The overall plant efficiency is 33.7%.

b. Liquid-D₂O Cooling. The flow diagram of the indirect cycle assumed for the liquid-D₂O cooling of the blanket is shown in Figure 5.13. The primary system was assumed to consist of two identical loops, although for simplicity only one loop is represented. Heavy water at 2000 psia is circulated through a Pb-Bi/D₂O heat exchanger, then in series through the pressure-tubes of the LAFER blanket, and, finally, through the steam generator. The primary pump is located on the cold leg of each loop between the steam generator and the Pb-Bi/D₂O heat exchanger. One pressurizer is connected to the hot leg of one primary loop.

Figure 5.13. Flow diagram for liquid-D₂O cooling.

The heavy water temperature increases from 540.9°F at the outlet of the steam generator, to 560°F after the Pb-Bi/D₂O heat exchanger, and, finally, to 600°F at the outlet of the pressure-tubes.

The heavy water inventory was estimated to be 178,570 lb which is about 18% of the inventory required for the 2180-MW(t) CANDU reactor (620 MW(e)). The difference is because of the LAFER blanket thermal output is only 900 MW(t) and no D₂O is required for moderating purposes. From the steam generator, the dry saturated steam is supplied to the throttle valves (875 psia) of a tandem-compound four-flow turbine which develops the mechanical energy to drive the electrical generator. Some main steam is also supplied to the moisture separator reheaters, which remove the water droplets from the exhaust of the high-pressure turbine and superheat the steam slightly.

A turbine bypass system is provided to pass up to 50% of full steam flow directly to the condenser to reject heat during start-up, shutdown, or other transient conditions when part or all of the steam flow must be bypassed. The regenerative system consists of four feedwater heaters which preheat the water to 368°F.

The gross and net electrical outputs are 430.3 MW(e) and 402.3 MW(e), respectively. The overall plant efficiency is 29.8%.

c. Two-Phase H₂O Cooling. The flow diagram of the direct cycle for two-phase H₂O cooling of the blanket is shown in Figure 5.14. The hot Pb-Bi exiting the reactor is used to heat the feedwater (@ 2600 psia) which has been previ-

Figure 5.14. Flow diagram for two-phase H_2O cooling.

ously heated by the regenerative system up to near the saturation temperature ($\sim 670^\circ F$).

Upon expanding the feedwater to about 1060 psia, a two-phase mixture ($x \approx 0.3$) is obtained. This in turn is fed to the first stage of pressure-tubes, where a portion of the power generated in the blanket raises the steam quality to $x \approx 0.7$. A pressure drop of ~ 20 psia is assumed to take place in the pressure-tubes. After leaving the first stage of pressure-tubes, the two-phase mixture is sent to a moisture separator where dry saturated steam is separated and sent to the turbine circuit. A portion of the steam is retained to form a two-phase mixture of 0.3 quality, which then enters the second stage of pressure-tubes. The exiting two-phase mixture, again at a quality of ~ 0.7 , enters the final moisture separator. The two-stage operation permits balancing the relative thermal inputs from the blanket (900 MW(t)) and the target (450 MW(t)), in the operation of the coolant system.

The remaining moisture (30% of the mass flow rate) is removed from the moisture separator and is then diverted to the HP feedwater heater on the steam side. The dry saturated steam is supplied to the turbine throttle valves (@ 975 psia) of a tandem-compound four-flow turbine which develops the mechanical energy to drive the electrical generator. Normally, two thirds of this steam passes through the entire turbine and exhausts into the condenser at a vacuum

Table 5.3

Comparison of Alternate Power Conversion Systems

	Blanket coolant		
	Helium	Liquid D ₂ O	Two-phase H ₂ O
Thermal rating, MW(e) (total/blanket/target)	1350/900/450	1350/900/450	1350/900/450
Electric output, MW(e), gross	486	430.3	471.9
net	455.5	402.3	450.8
Overall plant efficiency, %	33.7	29.8	33.4
Throttle valve steam conditions			
pressure, psia	975	875	975
temperature, °F	720	532	541
Feedwater temperature, °F	350	368	440
Balance of plant capital cost,* \$10 ⁶ (1/1/78)	375.5	428.7** 411.4†	387.9
Unit cost (BOP),* \$/kW	825.3	1065.5** 1022.6†	860.5

*Includes all plant costs with the exception of the linear accelerator costs.

**With D₂O, including inventory costs of \$17.3 x 10⁶.

†Without D₂O.

condition where the waste heat is rejected. The remaining one third of the steam flow is exhausted at various stages from the turbine for heating the feedwater being pumped back to the reactor via the expansion valve.

Some main steam is also supplied to the moisture separator reheaters, which remove water droplets from the exhaust of the high-pressure turbine and superheat the steam slightly. In this manner, excessive erosion of the turbine blades is avoided and the thermal efficiency is improved.

A turbine bypass system is provided to pass up to 50% of full steam flow directly to the condenser to reject heat during start-up, shutdown, or other transient conditions where part or all of the steam flows must be bypassed.

Condensate is pumped back from the condenser hot-wells by three 50% capacity condensate pumps through 100% deep bed polishing demineralizers, the steam packing exhaustor, and the low-pressure heaters.

Two 50% capacity turbine-driven feedwater pumps supply water to the high-pressure heater and further to the Pb-Bi/water heat exchanger. Two emergency feedwater pumps (one motor driven and one turbine driven) and one start-up feedwater pump are also provided. The latter pump takes suction from the condensate storage tanks and, subsequently, from the condenser hot-wells, and pumps the water through the fossil fuel start-up boiler.

The gross and net electrical outputs are 471.9 MW(e) and 450.8 MW(e), respectively. The overall plant efficiency is 33.4%.

d. Comparison of Alternate Power Conversion Systems. The major characteristics of the three power conversion systems analyzed are summarized in Table 5-3. Also included are the total capital cost and the unit cost for the balance of plant, that is, all plant costs except the cost of the linear accelerator. Detailed breakdowns of the capital cost estimates both at 2-digit and 3-digit accounting levels are given in Appendix E.

According to the results presented it appears that the helium cooling alternative is both most efficient and least expensive, followed, in order, by the two-phase H_2O -cooling and the liquid D_2O -cooling alternatives.

5.7 Materials

Within the context of the linear accelerator fuel enricher/regenerator, the ability of materials to sustain long burnups in the power reactor ($\sim 60,000$ MWD/ton) and the irradiation cycle of the regenerator are perhaps the key to the successful feasibility of the concept. As such, the effects of protons and hard neutrons, and of the long burn cycle, are of great concern.

As the regenerator concept addresses itself specifically to existing LWR and HWR technology, most of the attention was given to Zircaloy as the material of interest for both cladding and structural purposes. It is quite clear that the problem would be greatly reduced if stainless steels were considered. Stainless steel will be investigated further for the pressure-tubes.

The approach taken in this evaluation has led to three different parallel paths as follows.

a. Evaluation of Existing Experience. Zircaloy is today the standard cladding material used in LWR's and HWR's. A great amount of experience and operating data have been gained and accumulated over the last 20 years, and the Zircaloy used in earlier days has been improved to the point that cladding and

structural damage from radiation is practically nonexistent in operating reactors.

An evaluation of this experience and knowledge of the state of the art has been carried out for BNL by R.J. Teitel Associates and Westinghouse Electric Corp., Research and Development Center. Particular attention was given to those parameters that might influence the longer burnups required for the fuel regenerator. Appendix C describes the present knowledge and experience with Zircaloy. The report is encouraging. Indications are that at operating reactor temperatures, radiation-induced losses of mechanical properties occur during the first 12,000 MWD/ton burnup and then remain practically unchanged to the end of the burn cycle. Also, fuel elements have been exposed to 60,000 MWD/ton without noticeable deleterious effects.

This assessment leads us to believe that the linear accelerator fuel enricher/regenerator in conjunction with burner reactors can be projected on a two-cycle burn of 60,000 MWD/ton total as a realistic design parameter. This is the parameter used in this report. Higher burnups, to 100,000 MWD/ton, might be possible in the future. However, experience at this time is scanty so that material performance at such long exposure cannot be predicted.

b. Target-Blanket Design. Because of concern about the potential effects of high-energy protons and neutrons on materials, the target-blanket design has been directed toward minimizing these effects. The proton beam generated by the accelerator will impinge on a liquid lead-bismuth target without first having to traverse a window. The proton beam will be stopped completely in the liquid lead-bismuth. By the same token, primary neutron multiplication will also take place in the Pb-Bi so that the intensity of the higher-energy part of the neutron spectrum will be reduced in the region of structural materials, fuel, and fuel cladding. The calculated neutron energy spectrum reaching the "first-wall" pressure-tubes peaks at 5 to 6 MeV. Some high-energy protons and neutrons do reach the first wall from multiple scattering; however, this flux is very small.

Also, because of blanket peak power distribution considerations as well as potential radiation damage, the total primary neutron flux at the first wall will be kept to a level of the order of 10^{14} n cm⁻² sec⁻¹. This is accomplished by the variable density target described in the previous section.

c. Radiation Damage Analysis. The LAFER-LWR system assumed in the analysis consists of an initial fuel enrichment in LAFER, LWR fuel burn followed by a second LAFER fuel enrichment with a final LWR fuel burn. Thus, the fuel

will experience four separate irradiation cycles. The radiation environment in the two stages of the fuel cycle are quite different. A great deal of experience with thermal reactor irradiations has shown that Zircaloy performs well as a fuel cladding. Even under this irradiation it is the fission or fast component of the neutron spectrum that dominates in producing the radiation damage. The radiation damage mechanism will therefore be very similar during the LAFER irradiation cycle.

The neutron spectrum from the target will look like a degraded fission spectrum quite similar to that produced in a fast reactor but with a high-energy tail extending up to the energy of the proton beam. Most of the neutrons, and as a result most of the displacement-type damage, will come from neutrons with energy < 20 MeV. The importance of the neutron flux component above ~ 10 MeV is its potential in producing helium and hydrogen impurities in the structural alloys and new impurity products in the fuel itself. Further, unlike the evaporation or fissionlike part of the spectrum which has a more or less isotropic source distribution, the high-energy component will be more forward peaked with respect to the momentum of the proton beam. Potentially this homogeneity in the spectrum field may have an impact on the radiation damage problem.

The primary target is also a source of protons originating from similar nuclear processes. The leakage proton flux and energy distribution are strongly influenced by the target geometry. The leakage proton flux spectrum will have an impact on the radiation damage environment. In this case, as with the neutron spectrum, the low-energy component ($E < 20$ MeV) and the high-energy component ($E > 20$ MeV) present different radiation damage problems. The low-energy protons will stop in a small amount of material (~ 0.1 cm), whereas the high-energy protons will penetrate long distances (~ 10 to 20 cm). The high-energy proton flux will be forward peaked, and thus will be a nonuniform source of damage.

A relationship between damage exposure in one radiation environment and that in another is required so that relative damage levels can be estimated. Two parameters of interest are displacement damage and gas production. Although the effects of displacements and gas production on damage are coupled, they will be described separately. The best way to describe displacement damage is to calculate a spectrum-averaged displacement cross section for the material and irradiating particle of interest. This cross section incorporates the energy spectrum of the irradiating particle field so that displacement levels from dif-

ferent irradiations can be compared by a single exposure index. Specifically, proton and neutron irradiations can be directly compared. For the present analysis we have assumed (see Appendix C) that for LWR conditions, 1.5×10^{21} n/cm² ($E > 1.0$ MeV) $\bar{\sigma}$ 12,000 MWD/MT $\bar{\sigma}$ 1.9 dpa, where the damage is expressed in units of displacements per atom (dpa). Since in the reactor the fuel burnup can be related to the fluence via a fission cross section, we can also relate fuel burnup to dpa.

This displacement cross section for the primary neutron spectrum has been estimated by calculating this cross section, using two spectra obtained from the neutron irradiation effects experiments at Los Alamos Meson Physics Facility (LAMPF). The spectra were calculated for positions at 90° and 135° with respect to the center-of-mass of neutron production. Using ENDF/B nuclear data for Zircaloy-2, the results are

$$\sigma_{\text{dpa}} (135^\circ) = 0.94 \times 10^{-21} \text{ cm}^2, \quad (1)$$

$$\sigma_{\text{dpa}} (90^\circ) = 1.14 \times 10^{-21} \text{ cm}^2, \quad (2)$$

where we use $E_\phi = 25$ eV after Carpenter.^{5.13} For the present calculations we assumed:

$$\sigma_{\text{dpa}} (\text{LAFER}) = 1 \times 10^{-21} \text{ cm}^2. \quad (3)$$

It is interesting to note that, using the results of (1) and (2), and the assumption (3),

$$\sigma_{n,p} (\text{LAFER}) = 2 \times 10^{-27} \text{ cm}^2, \quad (4)$$

$$\sigma_{n,\alpha} (\text{LAFER}) = 2 \times 10^{-28} \text{ cm}^2. \quad (5)$$

For the LAFER radiation field, the neutron and proton flux and fluence parameters result from target calculations, and the proton gas production cross sections were estimated from calculations by Coulter et al.^{5.14}

- 1) LAFER neutron flux = 10^{14} n/cm² s.
- 2) Enrichment dose = 2.5×10^{21} n/cm².
- 3) Low-energy ($E < 20$ MeV) proton flux = 10^{13} p/cm² s.
- 4) High-energy ($E > 20$ MeV) proton flux = 3×10^{12} p/cm² s.
- 5) High-energy proton hydrogen production = 8×10^{-25} cm².
- 6) High-energy proton helium production = 1×10^{-25} cm².

In analyzing the response of Zircaloy to the irradiation conditions expected for the LAFER-LWR cycle, it is important to review three facts derived from the performance of Zircaloy in water reactors. One reason for projecting a long life for Zircaloy cladding is that the radiation-induced changes in strength and ductility tend to saturate at fluences of $1-2 \times 10^{21} \text{ n/cm}^2$ ($\sim 12,000 \text{ MWD/MT}$), which is early in the service life of the fuel. Secondly, Zircaloy picks up hydrogen by corrosion processes and should have a hydrogen concentration less than 200 to 500 ppm. Finally, and perhaps most importantly, Zircaloy cladding in reactor service does not form voids. This critical observation is most likely related to the fact that large dislocation networks are not observed and that the amount of helium generation is low ($\sim 0.3 \text{ ppm}$ at 10^{22} n/cm^2). The following analysis concentrates on how the LAFER radiation environment may affect these three critical properties of Zircaloy.

The displacement damage level that will be reached by the cladding will be a result of two LAFER enrichment exposures and two fuel burns in a reactor. Each LAFER irradiation will produce an exposure of 2.5 dpa. At a burnup of 30,000 MWD/MT the reactor exposure will be 4.7 dpa. Therefore, the total exposure will be 14.4 dpa, which is equivalent to 96,000 MWD/MT fuel burnup. This is higher by roughly a factor of 2 than that experienced by high burnup fuels.

Although direct comparisons between dpa levels in one material to those in another are not valid because the damage potential of a dpa is strongly material dependent, it is instructive to note that an exposure of 14.4 dpa is considerably less than that required for stainless steel in a fast reactor. In fast reactors, exposure limits are in the range of 50 to 100 dpa. From the calculated exposure level and its relationship to present exposures in water reactors and exposure of stainless steel in fast reactors, there is nothing in displacement damage alone to suggest that there will be a major change in the response of Zircaloy to the radiation environment in the LAFER-LWR cycle as compared to that in the LWR alone.

The presence of the high-energy neutron component produces a major change in the production cross sections for helium and hydrogen gas. It has been shown that helium can play an important role in void formation in Zircaloy.^{5.13} The potential effects of hydrogen in Zircaloy are clear. The amount of hydrogen generated in two LAFER irradiations will be about 10 ppm, considerably less than what is picked up through corrosion. The helium generation will be about 1 ppm for the two LAFER cycles. During a typical fuel burn in a water reactor, ~ 0.1

ppm is produced. The helium produced in the LAFER does not appear to present a major problem. However, since the specific role that helium may play in void formation in Zircaloy has not been identified, the presence of higher helium concentrations, cycled irradiation, and higher exposures may result in effects that cannot be predicted at this time.

In this analysis, the displacement effects of protons have been ignored. In a more complete analysis where more details of the proton flux and spectrum are known, proton displacement effects should be included. It is expected that on the average they will contribute at most a 10% addition to neutron displacement damage. However, the low-energy proton flux would produce a concentration of ~ 400 ppm hydrogen/day of LAFER irradiation in this layer. Even if the low-energy proton flux is 1% of the estimate, the implanted hydrogen is not tolerable. It appears unavoidable to conclude that a low-energy proton filter must be provided to prevent this unacceptable problem; this can be accomplished by installing a thin (~ 0.2 cm) steel curtain around the primary Pb-Bi target.

6. SAFETY AND LICENSING CONSIDERATIONS

Inasmuch as the linear accelerator fuel enricher/regenerator concept represents a new technology, although derived in part from existing technologies, licensing issues would naturally be focused in areas for which uncertainties are directly related to potential health and environmental hazards. For this reason, a licensing review of LAFER design would be most favorable if appropriate safety-related conservatisms are factored into the design at an early stage in those areas for which there are large uncertainties in the technology.

Perhaps an instructive precedent for this approach is the licensing review and related guidance associated with the Clinch River Breeder Reactor. The basic position taken by the Nuclear Regulatory Commission^{6.1} in the licensing review of the Clinch River Breeder Reactor is that it should achieve a level of safety comparable to that of the current generation (commercial U.S.) of light water reactor plants, and that the design approach to achieve this level of safety should be similar or analogous to the practice observed with light water reactors. Although there are differences in reactor concept and experience between the light water reactor, the Clinch River Breeder Reactor, and the rather less well-defined LAFER, the above objective and the implied design safety approach appear to define a useful procedure in each case.

The design safety approach is a three-level concept in which nuclear power plants are: 1) designed and constructed so that there is a large tolerance for operator errors, off-normal operation, and component malfunctions, and a high probability that they will operate without failures or malfunctions that could lead to accidents; 2) designed to provide measures and features to cope with events which are anticipated to occur during the plant lifetime; 3) designed in a conservative manner so that the public is protected from the consequences of certain highly unlikely events. The postulated events in the third level of safety are used to establish a set of design basis accidents, and systems and features are designed to control these accidents so that the consequences would be within the radiological dose guidelines of the Code of Federal Regulations, Chapter 10 (10CFR100).

Probabilistic risk analysis would be a helpful design tool for the accelerator fuel regenerator in assessing the relative risks associated with alternative design features. It should be cautioned, however, that probabilistic methods should be used only to assess relative risk and not to predict absolute risk for the new technology. It is interesting to note that the guideline set forth by the Nuclear Regulatory Commission^{6.1} for the Clinch River Breeder Reactor is that there is no greater than one chance in a million per year for consequences greater than the 10CFR100 dose guidelines.

For the accelerator fuel enricher/regenerator, the major emphasis must be placed on the prevention of accidents that may lead to the meltdown or disruption of the blanket and loss of containment system integrity. Two general approaches can be taken in this area. The probability of accident initiators may be reduced to an acceptably low level, and/or mitigating safety features may be incorporated into the system design in order to prevent the accident from progressing to the core melt stage.

An example of the first approach would be the assurance that the probability of an increase in proton beam current is sufficiently low that the contribution to the transient overpower accident need not be considered. An example of the second approach would be to design a sufficiently redundant and diverse shutdown heat removal system such that a blanket meltdown due to a loss of heat sink is unlikely.

Electrical power requirements for effective shutdown heat removal for the LAFER may pose a special problem. Because of the possibility of short unscheduled accelerator outages, and the frequent downtime needed for fuel re-

placement and shuffling, the shutdown heat removal system will be correspondingly challenged. Apart from turbine driven auxiliary feedwater pumps, the shutdown heat removal system relies on off-site and/or on-site electric power to achieve its mission. From light water reactor operating experience, it is known^{6.2} that the reactor trip event itself poses a challenge to the integrity of off-site electric power. Furthermore, on-site ac power, supplied by diesels, is limited by the reliability of the latter. Currently, diesel availability is less than 0.99 per demand. Therefore, the total electric power system for the facility must be designed with these considerations in mind.

Another area of licensing concern may be the design of the reactor shutdown system. For the Clinch River Breeder Reactor, the Nuclear Regulatory Commission recommended at least two independent, diverse, and redundant reactor shutdown systems which would satisfy the requirement that the reactor power level be quickly and reliably reduced whenever plant conditions require such actions.

If a pressure-tube concept is employed in the LAFER blanket, consideration should be given to methods for detection and mitigation of the consequences of pressure-tube failure. In particular, protection against failure propagation should be provided.

Coolant boundary systems should be of high integrity and should be assured throughout the lifetime of the plant. In particular, pipe ruptures on the main headers to the pressure-tubes may be of concern in regard to a possible undercooling event.

For the fuel regenerator, the buildup of actinides may require that for source term guidelines the specifications for site suitability should include specific dose guidelines for the actinides above and beyond those given in 10CFR100.

Of central concern in much of the foregoing is the possibility of a blanket meltdown accident and its potential role in the reactor licensing process for the fuel regenerator. In the case of the Clinch River Breeder Reactor, it is interesting to note that, although the Nuclear Regulatory Commission did not consider the core disruptive accident as a design basis accident,^{6.2} it did require that containment integrity be provided for at least 24 hours following a postulated core disruptive accident. For an accelerator fuel regenerator based on light or heavy water, containment issues^{6.2} may be

associated with steam and hydrogen explosions, fuel-coolant interactions, and fuel-concrete interactions. Examples of design safety features that may be considered for the LAFER at least in its early stages are: a core catcher (or, appropriately, blanket catcher) and filtered venting systems.

The above considerations may present an unnecessarily pessimistic view of the design features required to ensure safety of the LAFER, though adoption of this conservative view at this stage is the prudent course. On the positive side, the low power density in the LAFER and the pressure-tube design which reduces likelihood of a loss of coolant that might cause extensive meltdown of fuel could serve to relax limits on LAFER designs for safety purposes.

However, the linear accelerator fuel enricher/regenerator does contain unique features in comparison to currently licensed U.S. light water reactors. Consequently, prudence dictates that an initial, demo plant accelerator fuel regenerator design contain sufficient safety margins to account for uncertainties in those areas in which public health is at risk. Further, it is planned that appropriate safety features be included as an integral part of the design at an early stage, certainly before the formal licensing application is filed with the Nuclear Regulatory Commission.

7. PROLIFERATION CONSIDERATIONS

7.1 Introduction

Three types of accelerator fuel and power producers are considered:

1. Linear accelerator fuel enricher/regenerator - LAFER.
2. Linear accelerator driven reactor - LADR.
3. Linear accelerator fuel producer - LAFP.

All three of these systems employ either natural uranium, low enriched uranium, or thorium as the fresh fuel charge. Studies to date indicate that the most efficient operation results from employment of fuel containing uranium at ~2% ²³⁵U content.

LADR's and LAFER's operate without reprocessing, while the LAFP cycle requires reprocessing. The LADR employs an accelerator "on-line" with the driven reactor. This is to be contrasted with the LAFER and LAFP, where the accelerator is "off-line" and services a number of reactors. The LAFER cycle is compatible with LWR's currently in service and makes maximum use of existing LWR

technology. It is as proliferation resistant as the LWR once-through cycle. As this report concerns mainly the linear accelerator fuel enricher/regenerator, our proliferation analysis concentrates on the LAFER system.

7.2 Fuel Enricher/Regenerator

The LAFER increases the fissile content of 2% LEU ($^{235}\text{U}/\text{U}$) to 3.2% ($^{235}\text{U} + \text{Pu}$) by conversion of ^{238}U to Pu. The exposure of the fuel in the LAFER corresponds to $\sim 3,000 = \text{MWD/ton}$. The plutonium produced in the fuel rods is both diluted by uranium-238 and contaminated with fission products. This aspect of the LAFER cycle provides protection similar to that of the proposed irradiation of fresh MOX fuel in a reactor before shipment. However, in the LAFER case this irradiation is part of the enriching and regenerative process and therefore an additional facility is not required.

Since no reprocessing is involved, the LAFER system is inherently more proliferation resistant than CIVEX.^{7.1,7.2} It is interesting to note that reprocessing in the CIVEX scheme will probably require streams containing pure plutonium, because the input fuel from LWR's, breeder blankets, and burned breeder cores will not contain the concentration of plutonium required for fresh fuel to be recycled in LWR's or to be loaded into breeder cores. In the LAFER scheme, the plutonium concentration in fuel removed from the reactor is exactly that desired for insertion in the LWR. Thus any reprocessing that might be allowed at a future date would not require any streams of pure plutonium.

After being burnt to 30,000 MWD/ton in the LWR, the fuel (originally generated by the LAFER) is returned to the LAFER for regeneration. A second 30,000-MWD/ton cycle would then take place in the LWR. At this stage, the spent fuel (which has gone through 60,000 MWD/ton burnup for power production and about 6,000 MWD/ton for regeneration) is highly contaminated with fission products.

The plutonium from a LAFER/LWR cycle contains a higher fraction of ^{240}Pu and ^{242}Pu than standard LWR spent fuel. This degrades the weapons quality of the plutonium further below that of the typical LWR discharge, and tends to compensate for the somewhat increased fraction of total plutonium in the spent LAFER/LWR fuel over that of typical LWR discharge.

The LAFER/LWR fuel cycle is on balance somewhat more proliferation resistant than the once-through LWR fueled with uranium.

7.3 Driven Reactor

The LADR achieves a high degree of safeguardability by virtue of the fact that plutonium production and utilization takes place in situ. The fuel never leaves the reactor until the end of the cycle, when it is ready for storage and disposal. Thus, the proliferation resistance of the LADR is also dependent on the lack of required chemical processing of the fuel.

Fuel can be burnt in the LADR to a lower fissile content than in the LWR once-through cycle. The discharged fuel has a very high burnup with a consequent increase in the fission product inventory and hence very high radioactivity. An advanced reprocessing technology would be required to extract the plutonium from such high burnup fuel. In addition, the increased fraction of higher plutonium isotopes degrades the weapons quality of the high burnup fuel.

On balance, the LADR fuel cycle is somewhat more proliferation resistant than a once-through LWR fueled with uranium.

7.4 Fuel Producer

Operation of the LAFP in conjunction with LWR's or HWR's will require reprocessing and fuel fabrication involving plutonium or ^{233}U generated by the LAFP. If ^{233}U is produced, it would be consumed along with thorium, and presumably plutonium would be mixed with uranium for subsequent burning. In either case, proliferation resistance of the cycle would be equivalent to that achieved in a plutonium recycle situation. This equivalence would be maintained in the presence of institutional constraints, such as safeguarded fuel cycle centers that might include reprocessing facilities and LAFP's.

Production of ^{233}U might be done through use of a denatured fuel cycle. If the fuel used in the LAFP consists of a mixture of uranium and thorium, the irradiated fuel will include ^{233}U in thorium and uranium and some plutonium. For use in a subsequent reactor, the benefits of denaturing can only be retained if the isotopic ratio of the ^{233}U in uranium approaches 20%. This seems to call for long exposure in the LAFP, but we have not explored the full implication of this point.

A denatured fuel cycle could be supported if ^{233}U is made by irradiating thorium alone in a LAFP. Denatured fuel could then be made through mixing this ^{233}U with natural uranium and thorium within the safeguarded fuel cycle center.

This option would provide the same proliferation resistance as a fast reactor producing ^{233}U through use of a uranium-plutonium core and a thorium blanket. The proliferation resistance would be comparable to that of the once-through LWR cycle if the security of the safeguarded fuel cycle center were ensured.

7.5 Conclusions

Detailed safeguards analyses have been carried out by several groups for many of the major subsystems. For example, the safeguardability of the regional fuel cycle center is being addressed in a Department of Energy study and has also been considered in an International Atomic Energy Agency study.^{7.3}

The safeguardability of reprocessing for the thorium-uranium fuel cycle is being considered in the Barnwell Applicability Study,^{7.4} and the safeguardability of the denatured thorium-uranium fuel cycle is being considered in the Department of Energy study entitled Interim Assessment of the Denatured ^{233}U Fuel Cycle.^{7.5} Analysis of the proliferation profile for accelerator fuel and power production has been carried out by Brookhaven National Laboratory for the Department of Energy.^{7.6,7.7}

In summary, none of the three systems (LAFER, LADR, LAFP) requires the employment of plutonium reprocessing, MOX fabrication, plutonium recycle, or breeder reactors. All three of the systems are more resource efficient than the LWR once-through cycle. LAFER's and LADR's are at least as safeguardable as the LWR once-through cycle, and LAFP's (DTFC) are at least as safeguardable as certain versions of the denatured thorium fuel cycle currently being considered.^{7.5}

8. FUEL CYCLE AND ECONOMIC ANALYSIS

8.1 Fuel Cycle

By its nature, the linear accelerator fuel enricher/regenerator is suited to a variety of fuel cycles. Our economic analysis is restricted to its use with the uranium-plutonium cycle, because the technical analysis of other choices has not yet been done.

The conventional LWR fuel cycle is shown in Figure 8.1. 6300 metric tons of natural uranium (0.7% ^{235}U) are required as feed to isotopic enrichment, to

Figure 8.1. Conventional LWR fuel cycle. Natural uranium requirement for 1000-MW(e) (75% plant factor) over 30-yr production = 6300 tons. Fuel burnup = 30,000 MWD/ton enriched U. Net burnup = 5000 MWD/ton natural U.

Figure 8.2. LAFER-LWR fuel cycle; no reprocessing. Natural uranium requirement for 1000-MW(e) (75% plant factor) LAFER-LWR over 30-yr production = 1750 tons. Burnup in power reactor; two burn cycles at 30,000 MWD/ton. Exposure in LAFER = 3000 MWD/ton max. per cycle. Net resource gain = 3X.

fuel one 1000-MW(e) power reactor operating at 75% plant factor over a 30-year lifetime. Of these 6300 tons of natural uranium, 5250 tons end up as depleted uranium (still with $\sim 0.2\%$ ^{235}U) and the other 1050 tons are used in the power reactor. Without reprocessing, the 1050 tons of fuel, still containing about 2% fissile material, end up after reactor use as spent fuel which must be disposed of.

Table 8-1

Linear Accelerator Fuel Regenerator for LWR Economy,
One LAFER Feeding Three LWR's

Natural uranium resource gain over that of present LWR economy
(30-yr lifetime; 1000 MW(e); 6300 tons natural UO_2).

No. of burn cycles at 30,000 MWD/ton ea.	Total LWR burnup, MWD/ton	Tons/1000 MW(e) of nat. UO_2 needed, 1 LAFER/3 LWR	Natural U resource gain
1	30,000	3,500	1.8
2	60,000	1,750	3.6
3	90,000	1,170	5.4
4	120,000	875	7.2
5	150,000	700	9.0

The proposed fuel cycle including the LAFER is shown in Figure 8.2. The cycle shown is one option among many possibilities. The system is based on partial preenrichment (2%) of new fuel. The amount of preenrichment is predicated upon the more or less arbitrary decision to have one LAFER provide fuel for three power reactors on an equilibrium basis. This preenrichment level, given a LAFER design capacity, determines a priori two quantities: (1) how much extension of the natural uranium resource can be achieved, and (2) how much reactor capacity one LAFER can support. Each can be somewhat improved at the expense of the other.

The scenario for the fuel cycle which includes the LAFER in the equilibrium mode is as follows. As is normal LWR practice, one third of a core is replaced at each fuel reloading (roughly each year).

For any one third of a reactor core, it is necessary to:

1. Enrich fresh fuel at a conventional enrichment plant to 2% ^{235}U .
2. Fabricate fuel.
3. Increase fissile material content in the LAFER to 3.2%.
4. Burn fuel in LWR for 30,000 MWD/ton (2% fissile material content in discharged fuel).
5. Reenrich burnt fuel in the LAFER and enhance fissile material content to 3.2%.
6. Burn regenerated fuel in LWR for 30,000 MWD/ton.
7. Store spent fuel after second cycle (total 60,000 MWD/ton burn).

Figure 8.3. Linear accelerator fuel regenerator with light water reactor (1 LAFER/3 LWR). Total natural uranium requirement for 30-yr supply = 5250 tons for 3 LWR. Two burn cycles in six years (60,000 MWD/ton).

In principle, one could consider irradiating fuel in the LAFER through more than two cycles, with correspondingly better fuel utilization and better economics. However, as previously described, uncertainties concerning the effects of fission product buildup, and of radiation and corrosion damage to materials, indicate that we should not consider more than about 60,000 MWD/ton burnup at this time.

The natural uranium requirement to fuel an LWR operating in conjunction with a LAFER is calculated to be 1750 tons, over its 30-year life. This is 3.6 times less than the 6300 tons presently required. The factor provides a substantial improvement in the utilization of our natural uranium resource, because it is also the factor of increase in electric power achievable from the resource. It should also be pointed out that, because of the longer integrated fuel burnup (60,000 MWD/ton), the rate of generation of spent fuel is only half that of the conventional fuel cycle (500 tons vs 1000 tons over the 30-year life of the reactor). This is an important additional factor.

The effect of additional burn cycles for the UO_2 -LWR system is shown in Table 8-1. It would be highly desirable to be able to extend the fuel burnup cycle to $> 100,000$ MWD/ton.

Figure 8.3 shows the schematic of one total system consisting of one linear accelerator fuel enricher/regenerator supporting three power reactors (UO_2 -fueled LWR's). The thermal power generated in the target assembly has been estimated to be 1350 MW(t): 450 MW(t) direct beam deposition in the lead target and 900 MW(t) in the surrounding blanket from fast fission multiplication. This power would be sufficient to generate 450 MW(e) at 33% power cycle efficiency, or enough to supply about half the energy needed for the accelerator power supply at 50% efficiency. Definitive calculations are yet to be performed concerning the power cycle. For the purpose of this analysis, we have assumed that 450 MW will have to be purchased from the power grid. Any changes in this ratio will not materially affect the economic picture.

8.2 Economic Analysis

The following is a first-order economic analysis of the fuel cycle with the LAFER and LWR's, fueled in the uranium cycle. The system analyzed is that just described, in which one fuel regenerator enhances the fissile content of fuel for three LWR's. Analysis is made as of a 1986 introduction of the system, although this early achievement could not be realized.

The capital cost of three 1000-MW(e) LWR's if purchased today would be about $\$1,800 \times 10^6$, neglecting interest on investment during construction and escalation. A recent industry analysis by Bechtel concludes that the interest and escalation would increase the cost in 1986 to $\$3,300 \times 10^6$, a factor of 1.8 above current cost. The capital cost of the LAFER could be considered as increasing this estimate, but we prefer to treat the LAFER and its operation as contributing to fuel cycle costs.

The 1.5-GeV 0.3-A linear accelerator is estimated to cost M\$470. This estimate is based on a cost study carried out at BNL. The costs were derived from costs of existing linear accelerators (BNL 200-MeV linac, LASL 800-MeV linac) properly adjusted to account for different parameters and escalation to 1978. The linac costs are given in Section 9 and in more detail in Appendix E.

The target-blanket complex cost has been estimated by United Engineers of Philadelphia. The estimate based on the design presented in this report was

Table 8-2

Comparison of LWR and LAFER
Fuel Cycle Costs - Mills/kWh(e)
(Escalated to 1986)

	Unit cost in 1977 dollars	LWR	Reduction factor	2-Cycle LAFER-LWR
Yellowcake	\$ 50/lb U_3O_8	3.94	3.62	1.09
Conversion	\$ 11/kg	0.29	3.62	0.08
Enrichment	\$100/SWU	2.90	4.33	0.67
Fabrication	\$200/kg	2.05	2.0	1.03
Storage and carrying charge	\$400/kg HM	3.87	2.0	1.94
Transportation	\$ 30/kg HM	0.31	0.5	0.62
Amortization of LAFER (15%, 80% P.F.)		-		11.63
Electrical power to LAFER (450 MW)		-		8.60
Operation & maintenance		-		2.00
		13.36		27.66
		Mills/kWh(e)*		Mills/kWh(e)

*Industry Report (1976-7) - Bechtel estimate.

arrived at by utilizing LWR and HWR costing procedures as developed for commercial reactor application. To be sure, there are appreciable differences between the target-blanket of the LAFER and an LWR reactor core. The LAFER has no thick heavy pressure vessel. It operates at a lower peak power density than an LWR. It has no control rod system since it always operates in a highly subcritical mode. The pressure-tube design should be favorable as regards costs connected with safety, since fuel core meltdown or even meltdown of a substantial part of the core is less plausible. On the other hand, the pressure-tube design and associated piping are more complicated than those of an LWR. There is an extra circuit: the circulating lead-bismuth. Also, additional shielding is required to stop the high-energy particles present. However, appropriate allowances have been made for these differences. The total cost of the target-blanket system has been estimated to be M\$350. It is interesting to note that of that amount, M\$275 represents the balance of plant (BOP) costs which are well understood and readily available. Therefore, the greater uncertainty attached to the cost of

Table 8-3

Comparative Economics of Linear Accelerator Fuel Enricher/Regenerator
With Light Water Reactors

<u>Capital cost</u>	<u>Conv. LWR</u>	<u>LAFER-LWR</u>
Base capital cost \$/kW (gross)	\$ 600	600
Completion cost \$/kW (gross) escalated for 1986 operation	1100	1100
<u>Resource</u>	<u>Tons</u>	
Fuel requirement over 30 yr, nat. U	6300	1750
<u>Power generation cost (av for first 10 yr)</u>	<u>Mills/kWh(e)</u>	
Capital charges 15%, and 70% P. F.	26.9	26.9
Fuel, 5% escalation/yr	13.4	27.7
Operation and maintenance	3.3	3.3
Total	43.6*	57.9
*ANS industry report (1976-7) - Bechtel estimate. Conclusion: LAFER/3 LWR reasonably competitive with LWR.		

the proposed target-blanket assembly affects only a small portion of the total cost.

It is, therefore, estimated that a linear accelerator fuel enricher/regenerator as outlined in this report will cost $\$850 \times 10^6$. The Bechtel factor for escalation and interest to 1986 raises this to $\$1500 \times 10^6$. This is the capital cost to be reflected in fuel cycle costs.

Table 8-2 presents an estimate of conventional fuel cycle costs for LWR's and for LWR's associated with operation of a LAFER. The second column shows current unit costs of the various components of fuel cycle cost, as taken from various industry sources. The third column shows the Bechtel estimates of the corresponding anticipated costs per kWh(e) in 1986. The fourth column is the reduction factor associated with use of the LAFER. The last column is the anticipated cost of the fuel cycle with use of the LAFER. The added items in fuel cycle cost of the LAFER-LWR are amortization of the $\$1500 \times 10^6$ capital cost at 15% interest and at 80% plant factor (shared among the three reactors served), the cost of power purchased for the accelerator, and an additional amount for operation and maintenance of the facility. The final estimate is that the fuel

Figure 8.4. Reactor capacity supported by 1 LAFER vs reactor conversion ratio.

cycle cost of the LAFER-LWR will be 27.7 mills/kWh(e) in 1986, compared to 13.4 mills for the LWR alone.

Table 8-3 shows the total cost of electrical power estimated for the two cycles. The estimated cost for the LAFER-LWR cycle exceeds that projected for the LWR alone by 33%. However, the analysis really should be carried out as of a reasonable date for introduction of the LAFER technology. This may be about the year 2000. Little reliance can be placed on fuel cycle estimates extrapolated to that date. The cost of yellowcake should be very much higher than the estimated value for 1986. The cost of enrichment will depend on the state of enrichment technology at that date. Other components of cost are also unpredictable.

In any case, the LAFER must be seen not as a competitor to the LWR, with power costs that must be compared to those of the LWR. It is a companion to the LWR fuel cycle, which would permit continued use of the LWR for a long period of time. In this sense it has no real competitor. In a breeder sense, it is competitive in its ability to produce central station power over the longer term with the breeders, and with the fusion, and the fission-fusion reactors.

The preceding analysis has been based on the continued use of the LWR with the plutonium fuel cycle. The LAFER might be used in conjunction with the Th- ^{233}U cycle. The use of thorium fuel would lead to a higher conversion ratio, and as Figure 8.4 shows, this would make it possible for a LAFER to support more reactors at one time. We have not explored this possibility, which is more complex because of the lack of a naturally fissile component of thorium. There is

no simple counterpart to pre-enriching fuel to be further irradiated in the LAFER. One possibility would be to create fuel of a mixture of thorium and 10% enriched uranium, for introduction into the LAFER. Exposure in the LAFER would build in ^{233}U to the level needed for a full fuel cycle operation. A preliminary analysis of one point in the design spectrum indicates that one LAFER could support five LWR's, fueled with a denatured fuel consisting of 2.5% ^{235}U , 1.5% ^{233}U , 12.5% ^{238}U , and 83.5% Th. This fuel would be fabricated initially of 15% uranium (20% enriched) and 85% thorium, and would be enhanced in fissile content by exposure in the LAFER (a small amount of Pu buildup during the process has been neglected). Fuel could withstand a single core exposure of nearly 60,000 MWD/ton before discharge. There would be no recycle because full burnup would be achieved in the first cycle. The enrichment of the uranium to an initial 20% would consume about 20% more Separative Work Units (SWU) than the normal input of 3.2% uranium to the reactor. But the feed rate of natural uranium to the diffusion plant to support this fuel cycle would be reduced by a factor of $\sqrt{2}$, and this implies a fuel stretching by the same factor. It is possible that other choices of fuel could be more attractive in implications for the fuel cycle. In addition, the discharged fuel would contain predominantly ^{235}U and ^{233}U as fissile components, and even less plutonium than the customary fraction usually calculated to be present in fuel discharged from a reactor on the denatured cycle (the initial ^{238}U content of the fresh fuel to be inserted in the LAFER is only about 60% that usually assumed in denaturing). If reprocessing is allowed in a denatured fuel scenario, the fuel discharged from the reactor is ideal for simple removal of fission products and reentry into the LAFER with essentially no other change in chemical and isotopic composition.

8.3 Conclusions

The economic analysis presented in the previous pages addresses itself to one specific system. This system, one linear accelerator fuel enricher/regenerator supporting three UO_2 -fueled LWR's, was picked as the most obvious choice answering a specific need, that of supplying fuel to an existing installed reactor capacity.

Conclusions reached concerning this system are as follows:

- a. The cost of electrical power produced by this system is about 33% higher than that of the power produced by the corresponding LWR alone. This

Table 9-1
LAFER
Engineering Parameters

Accelerator

Ion source: 100-kV dc accelerator
H-mode preaccelerator: 0.1 to 4 MeV
Low- β Alvarez accelerator: 4 to 150 MeV, 200 MHz, 1.5 MeV/m
High- β $\pi/2$ -mode accelerator: 150 to 1500 MeV, 600 MHz, 1.5 MeV/m
200-MHz rf, 50 MW cw
600-MHz rf, 500 MW cw

Target blanket

Target: Lead-bismuth eutectic
Flow 5 m/sec, 6,000 m³/hr
Head $\sim$ 10 m total
Blanket: 240 PWR fuel assemblies
240 pressure-tubes set in 3 x 3 x 12-m vacuum vessel
Power generator sized for 450 MW(t) in target
900 MW(t) in blanket

increase in power cost is based on conservative assumptions, especially as regards the target-blanket neutron yield and corresponding fissile fuel production rate. Also, a linear accelerator efficiency of 50% has been used (beam power to ac input power ratio). It is hoped that this efficiency might be improved to $\sim$ 60%.

b. The difference in cost is entirely in the cost of the fuel cycle. So the economic desirability of the system is resource dependent.

c. The LAFER/LWR scenario considered here (with 60,000 MWD/ton burnup, no reprocessing, throwaway fuel cycle) will:

1. extend the natural uranium fuel resource by a net factor of 3;
2. reduce the enrichment plant capacity required by a factor of 4.3;
3. reduce the rate at which spent fuel must be stored by a factor of 2.

These factors can be greatly improved if longer burnups are considered, or if a reactor with higher conversion ratio is used.

d. The LAFER is not a futuristic concept. It is a device that utilizes available technology, which can be assessed, costed, and evaluated with a high degree of accuracy and certainty. Although some questions remain concerning the validity of neutron yield calculations which have an influence on the economic promise of the LAFER, relatively simple experiments can provide the information necessary to close that gap.

Table 9-2

LAFER
Cost Estimate Summary
1978 dollars

	M\$
Accelerator	
Injection system	10
Low- β accelerator	20
Low- β rf system	40
High- β accelerator	70
High- β rf system	320
Beam transport	10
Subtotal	470
Target-blanket system	
Target-blanket vessel	75
Containment bldg.	75
Power-generating equip.	
Turbine plant	65
Electric equip.	40
Heat rejection system	10
Indirect cost	115
Subtotal	380
Total	850

e. Because of the reliance on existing technologies, the LAFER can be developed at a reasonable cost. We estimate the R&D costs to be about \$3000 million dollars including the cost of a prototype plant. By the same token, development can take place in a relatively short time scale. We estimate no more than 20 years to commercialization (see Section 9).

9. COSTS AND SCHEDULES

9.1 Capital Cost

A preliminary cost estimate for a linear accelerator fuel enricher/regenerator has been calculated on the basis of the proposed design presented in this report. Appendix E gives the details of the estimate. The costs are given in 1978 dollars without escalation.

Table 9-3

RD&D Cost and Time Schedule for the LAFER Development

Activity	1980s										1990s										Cost 1978 M\$
	0	1	2	3	4	5	6	7	8	9	0	1	2	3	4	5	6	7	8	9	
LAFER physics	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	200
Reactor physics	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	150
Yield exper.	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	10
Eng. design study	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	40
Lattice exper.	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	150
Accel. devel.	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	150
Target devel.	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	150
Materials devel.	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	150
Demo. plant design	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	200
construction	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	600
operation	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	100
Safety, licensing	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	100
Proto. plant design	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	300
construction	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	1000
operation	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	-----	
COST 1978 M\$ x 10	1	5	12	13	14	22	30	25	10	10	10	12	12	12	20	35	35	20	12	10	3200
																					TOTAL

The estimated unit costs were developed from the cost history of existing accelerator facilities properly adjusted for escalation to 1978. The facilities surveyed were the Brookhaven 200-MeV linac and the Los Alamos LAMPF 800-MeV linac. The estimate of unit costs for the target-blanket assembly, containment system, and power generation system was developed by United Engineers of Philadelphia under contract to BNL. The details of their estimate are found in Appendix E.

The engineering cost estimate is based on information available today. Its purpose is to provide a basis for an economic analysis of the system and for the evaluation of its desirability. As the program develops, its costs will be regularly reviewed and refined to input into the ongoing economic and cost benefit analysis.

Table 9-1 outlines the linear accelerator fuel enricher/regenerator parameters used for the cost estimate.

Table 9-2 outlines the cost estimate for one typical LAFER commercial facility in 1978 dollars.

9.2 Research and Development Program

As part of this study, estimates have been made of the research and development required to achieve commercialization of the system. Although the concept is untried, it is based on existing technologies. No new undemonstrated principle is involved. Also, the design approach taken to date indicates that all parameters chosen are within the state of the art. Thus, the engineering development activity is heavily dependent on borrowing from the existing knowledge and experience of linear accelerator facilities, LWR's, reactor materials laboratories, etc.

Within this context, it is estimated that the first commercial facility can be in place within the next 20 years at a cost of about \$3000 x 10 (1978 dollars). Table 9-3 scopes the RD&D cost and time schedule required to achieve this goal.

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APPENDIX A

NEUTRON SOURCE AND POWER DISTRIBUTION CALCULATION

The classified MTA project was initiated in the early 1950s. Its purpose was the production of fissile material for nuclear weapons. The project was discontinued when a competitive method of fissile material production was found to be cheaper and more readily available.

The yield of neutrons from bombardment of heavy targets by high-energy protons, which is a critical factor for the production of fissile material, was experimentally determined, using the 184-in. cyclotron at Berkeley. A semi-empirical analysis was performed to estimate the yield above the energy range for which the experiment was performed.

Recent knowledge of high-energy nucleon reactions enables us to calculate reasonable estimates of the neutron yield from fundamental reaction cross sections of the nucleon-nucleon and nucleon-meson type. Several Monte Carlo calculations of neutron yield have been performed in the BNL studies, based on cross sections for spallation and subsequent evaporation reactions. The results are in reasonable agreement with the experimental values. The Monte Carlo method presently used to calculate the neutron yield produced by high-energy particles (>15 MeV) ignores high-energy fission. Fissions are not negligible, however, even in very high-energy reactions. The calculated neutron yield is in reasonable agreement with the experimental one. The uncertainty of the calculated neutron yield is presently estimated to be within $\pm 20\%$.

The Nucleon Meson Transport Calculation (NMTC) code was adopted from the several available Monte Carlo codes, because of its ability to calculate not only intranuclear cascades but also internuclear cascades of nucleons with energies >15 MeV. For the transport of neutrons below the 15-MeV threshold limit of the NMTC code, the two-dimensional analytical code TWOTRAN was used. The code system adopted in the calculation of neutron yield and flux distribution for the linear accelerator driven reactor (LADR) and the linear accelerator fuel enricher/regenerator (LAFER) conceptual designs are shown in Figure 3.5.

Figure A.1 shows the x-z geometries for which the NMTC and TWOTRAN calculations were performed. In the NMTC calculation, the proton beam is injected in the z direction (into the page). A controlled penetration of the incident proton beam in the target can be made by orienting the plane of the primary lead

Figure A.1. Geometry of target blanket used for calculation purpose.

target at an acute angle with the beam direction or by varying the target density through a varied spatial density of liquid lead jets. The neutrons produced by the high-energy protons are then distributed through the target, and the neutron leakage from the beam entrance surface can be minimized. Furthermore, since the neutron source is spread over a larger area, the design contributes to a lower peak fission power density near the primary target. Spreading the proton beam in the y direction by means of magnets also assists in lowering the peak fission power.

Table A-1 shows the types of fuel and coolant used in the calculation. The fuel assembly geometry chosen resembles that of the CANDU reactor, with the pressure-tubes in a tighter array.

Table A-2 shows the yield of neutrons created in the reactions by energy excitation >15 MeV and the multiplication of the neutron yield by fissions

Table A-1

Composition of Subcritical Assembly
Used in Calculations

Case No.	Fissile material	Coolant	Coolant density
1	UO ₂	D ₂ O	0.77
2	Th	D ₂ O	0.77
3	UO ₂	H ₂ O	0.7
4	UO ₂	H ₂ O	0.35
5	UO ₂	H ₂ O	0.175
6	Th	H ₂ O	0.7
7	Th	H ₂ O	0.35
8	Th	H ₂ O	0.175

Table A-2

Neutron Yield and Initial Fissile Material Production Rate

Case No.	Neutron yield (without fission)	Multiplication factor by fission	Total neutron yield	Initial production rate of fissile material ton/yr
1	28.24 ± 2.15	0.2708	35.88	0.8405
2	29.79 ± 4.31	0.04386	31.09	0.709
3	28.24 ± 2.15	0.7510	49.44	1.158
4	27.82 ± 2.57	0.5798	43.94	1.029
5	29.54 ± 2.67	0.4672	43.33	1.015
6	29.79 ± 4.31	0.04147	31.02	0.708
7	30.85 ± 3.91	0.05093	32.42	0.740
8	30.81 ± 3.19	0.06139	32.70	0.747

Calculated for 1-GeV proton and 100% plant factor.

induced by neutrons with energies <15 MeV. The fourth column is the total neutron yield. The neutron yield of UO₂ fuel is a little larger than that of thorium fuel because the fission cross section of ²³⁸U is about 5 times larger than that of ²³²Th in the energy range from 1 to 10 MeV. The larger neutron yield in a highly moderated UO₂ fuel assembly with H₂O coolant, compared with a less moderated one, is caused by the low-energy fission of ²³⁵U in the natural uranium. The last column shows the initial fissile material production rate per

Figure A.2. Fission neutron distribution (case 1).

year. In the case of UO_2 fuel, plutonium is produced at the expense of some of the ^{235}U present in natural uranium. This burnup is not accounted for in the calculation. The net fissile material production rate in the less moderated assembly is larger than that in the highly moderated assembly, because the harder neutron spectrum leads to more fissioning of ^{238}U .

Figures A.2 and A.3 show the fission neutron distributions in the x-z geometry produced by the neutron source calculated with NMTC. The results are normalized to one source neutron. The TWOTRAN calculation gives the fission distribution in each region as shown in Figure A.2-a in a histogram form. In order to simplify the display, Figure A.2-b shows the surface figure which is made by connecting the values at the center of each cell. Both A.2-a and A.2-b are for case 1 of Table A-1.

Figure A.3 shows cases 2 to 8. In case 2 the depression between the two peak values is due to the statistical error produced by the small samples used in the NMTC Monte Carlo calculation. In the NMTC calculation, 10 batches were computed, each batch containing 25 protons.

Figure A.3. Fission neutron distribution (cases 2 to 8). Neutron energy spectra with respect to radial position in target-blanket assembly.

(Case 2)

(Case 3)

(Case 4)

(Case 5)

(Case 6)

(Case 7)

(Case 8)

Figure A.4. Group flux vs energy (cases 1 to 8; region (I-6)).

Cases 3 to 8 show a rather steep gradient, because of the short neutron migration length in these subcritical assemblies. The steepness of the gradient is less pronounced for the less moderated assemblies.

Figure A.4 shows the neutron energy spectra with respect to radial position in the target-blanket assembly. The target assembly is divided radially into 10 zones (I-1 to I-10). I-1 is the Pb zone, I-2 is a vacuum, I-3 to I-10 are eight 25-cm-wide blanket zones.

Comparison between the spectra in UO_2 and Th fuels shows that they are very similar. The spectra for the D_2O ($\rho = 0.7$) coolant cases (1 and 2) show the same order of spectrum hardness as those in the H_2O ($\rho = 0.175$) coolant cases (5 and 8), but the spectrum in the lower energy range for D_2O decreases more rapidly than for H_2O . As expected, the spectra in the case of H_2O ($\rho = 0.7$) cases (3 and 6) are quite soft, contributing substantially to low-energy ^{235}U fission in the natural uranium oxide fuel, hence, the rather large multiplication factor shown in Table A-2.

APPENDIX B

PRELIMINARY BURNUP CALCULATIONS

In Appendix A, the initial production rate of fissile materials such as ^{239}Pu and ^{233}U was computed using the NMTC and TWOTRAN codes. These codes are limited in that they cannot take into account the effects of burnup and fission product buildup. As the fuel material burns, the composition of the materials changes and fission products accumulate. These changes will affect the production of fissile material and also the fission power distribution in the assembly. In order to study these effects, burnup calculations were performed. Although two- or three-dimensional burnup calculations are desirable, they require long computer runs. Therefore, the preliminary scoping calculation was done by using the one-dimensional burnup code SIZZLE. The original SIZZLE code was modified to accommodate the neutron source produced by the accelerator. The cross-section set for this calculation is based on the HANSEN-ROACH cross sections. Additional cross sections were necessary; these were obtained from the ENDF/B-IV file using the ETOG code. The HANSEN-ROACH cross-section set is reasonably accurate for the analysis of a hard spectrum fission reactor. However, for the analysis of the LAFER, the set of cross sections used for the calculation in the energy range of interest tends to underestimate the result. Thus, the net fissile material production in this calculation is on the low side. Nevertheless, the calculation shows the general trend, which should not be too different from that of the actual case.

The target-blanket geometry used in this calculation is shown in Figure B.1. UO_2 and Th fuel assemblies were studied with D_2O and H_2O coolants. These assemblies are the same as those used in the TWOTRAN calculations of Appendix A, with the composition given in Figure B.1.

The neutron source spectrum used in the SIZZLE calculation is provided by the NMTC calculation. This code yields also the spatial distribution of the source, not only in the liquid lead target region, but also in the surrounding blanket. However, the neutron source is mostly concentrated in the liquid lead target region. To simplify the SIZZLE calculation, it is assumed that the neutron source is uniformly distributed in the liquid lead region only. The total number of neutrons produced by the high-energy reactions of spallation and evaporation in the primary Pb-Bi target is assumed to be 45 for all cases.

Figure B.1. Geometry used in the SIZZLE calculation (one dimensional with $24 \times 10^4 \text{ cm}^2$ area).

Table 3-3 shows the net production rate of fissile material integrated over a one-year period, starting with different initial enrichment values. These values were extrapolated from the data calculated in the case of a Pb target and a 1-GeV proton beam. Table 3-3 indicates that a reactor produces less fissile material as the amount of moderator increases. This is because fission of ^{238}U decreases with the amount of moderation introduced in the lattice. It is also interesting to note that in the case of ^{239}Pu production in UO_2 fuel, the production rate appears to be independent of initial enrichment. However, for the case of Th fuel, the net production of ^{233}U decreases with ^{233}U content. The production rate of ^{233}U is also smaller because the Th fissile cross section is only 20% of the ^{238}U cross section.

Table 3-4 shows the initial and final (after one year) power levels due to the fission reaction in the blanket region. As the fuel is irradiated, the power level increases with increased fissile content. Table 3-5 shows the corresponding total burnup during one year of operation.

Outputs of the SIZZLE code calculation were processed to determine the fission distribution, mass distribution of heavy elements, and fission products at each time step. Figures B.2 to B.9 show (for case 5 with 2% ^{239}Pu initial enrichment) the fission distribution, which is proportional to the power distribution (events/cm) and the mass distributions of fission products produced per ^{235}U fission. The distribution of the material density (in kg) is approximated by the line connecting the densities at the middle of each region.

Figure B.2. Fission neutron source distribution.

Figure B.3. Mass of fission product.

Figure B.4. Mass of ^{235}U .

Figure B.5. Mass of ^{240}Pu .

Figure B.8. Mass of ^{236}U .

Figure B.9. Mass of ^{239}Np .

Figure B.10. Total power level (MW).

Figure B.11. Neutron multiplication factor = $\nu \Sigma_f \phi / s$.

Figure B.12. Burnup (MWD/ton).

Figures B.10 to B.12 show the total power level, multiplication factor of the neutron source, and total burnup. These data were calculated for half the blanket, so that the total power should be twice the value shown in the figure.

APPENDIX C

MATERIALS LIMITATIONS TO HIGH FUEL BURNUP IN THE LAFER-LWR SYMBIOTIC CONCEPT

1. INTRODUCTION

Brookhaven National Laboratory is investigating the feasibility of employing a linear accelerator to convert fertile material to fissile material. In this scheme, a high-energy linear accelerator is used to produce a high-intensity beam of protons which are impinged on a target of heavy elements to produce neutrons. These neutrons in turn are absorbed in ^{238}U or ^{232}Th to produce fissile ^{239}Pu or ^{233}U . Although uncertainties, including those of economics, exist, the claimed advantages of this concept include extension of the uranium resource base and greater proliferation resistance than afforded by current fuel reprocessing and recycle schemes.

One scenario of the LAFER concept, which exploits the safeguards advantage, involves the breeding of either ^{239}Pu or ^{233}U from appropriate fertile material in the LAFER target-blanket assembly reactor, and subsequent use of the bred elements in light water reactors without intervening reprocessing. After burnup of the fissile inventory in the LWR, the fuel elements would be rejuvenated in the LAFER and returned to the LWR for additional burnup. Such sequential burnup and rejuvenation would result in appreciable extension of the total burnup of the fuel, and levels as high as 100,000 MWD/MTU are anticipated. As such high burnups represent an appreciable increase over the $\sim 30,000$ -MWD/MTU life typical of commercial LWR fuel, the impact of this increased burnup on fuel performance warrants consideration, and it is the purpose of this report to address the potential consequences of this extended burnup on fuel cladding behavior.

2. DISCUSSION

The details of the LAFER target and blanket parameters have not yet been established. Therefore, for the purpose of this report it will be assumed that the environment in the LAFER will be similar to those in LWR's. That is, the

atmosphere will be composed of pressurized water or steam, and cladding temperatures will be in the range of 300° to 400°C. Furthermore, it will be assumed that the neutron flux spectrum to which the cladding is exposed will be similar to that of LWR's, and that the cumulative burnup of the fuel in the combined LAFER-LWR exposure will be 100,000 MWD/MTU. The geometry of the fuel is assumed to be that of the usual Zircaloy-clad stacked pellets.

With the above assumptions, the most serious impacts on fuel behavior are expected to be those caused by the effects of irradiation on mechanical properties and also by the limitations imposed by corrosion of the cladding in the water and/or steam environments. Therefore, the effects of irradiation on mechanical properties and corrosion behavior will be considered in the following sections. In addition, because the target burnup in the LAFER-LWR concept is 100,000 MWD/MTU, existing experience with high burnup rods in LWR's will be briefly reviewed so as to set the baseline from which extrapolations to the target values must be made.

2.1 High Burnup Experience in LWR's

Current commercial LWR's are operated such that the region average burnup is ~33,000 MWD/MTU with peak burnups to ~36,000 MWD/MTU. However, both commercial and test reactors have experienced burnups appreciably above these values, and design values¹ for Westinghouse Pressurized Water Reactors (PWR) are 50,000 and 55,000 MWD/MTU for peak rod average and peak pellet burnups, respectively. Region-3 assemblies, exposed for three cycles, were discharged from the Point Beach Unit 2 Nuclear Generating Station after achieving an average burnup of 36,100 MWD/MTU. The peak assembly average burnup was 39,300 MWD/MTU, and the peak rod average burnup was 42,800 MWD/MTU. These burnups are among the highest achieved in a Westinghouse commercial plant.

An "extended burnup test" was performed under USAEC sponsorship on mixed-oxide fuel in the Saxton Reactor.³ The peak pellet burnup achieved in a peak burnup rod was 51,000 MWD/MTU. Although failures were detected in certain high-power rods during the second half of Core-III operation, the defects did not appear related to any burnup limitation of the mixed oxide fuel.

Additional PWR experience at high exposures was reported¹ for UO₂ fuel exposed in the Zorita (Jose' de Cabrera) reactor. Rod average burnups of up to 57,000 MWD/MTU were achieved for three-cycle fuel at discharge. The correspond-

ing peak pellet burnups reached 65,000 MWD/MTU. PIE examinations of the high burnup rods revealed them to be in generally excellent condition.

Compared with the burnup values cited for commercial and experimental fuel, the target burnup of 100,000 MWD/MTU for the LAFER-bred fuel represents a significant, though not unreasonable, increase in fuel duty.

2.2 Corrosion Considerations

Zircaloy-2 and -4 were developed as fuel cladding material for use in PWR's,⁴ and out-of-pile data indicate that both alloys have similar corrosion kinetics and that the $H_2O(l) \rightarrow H_2O(g)$ phase change has no effect on corrosion behavior. In-pile, however, the situation is less clear. Appreciable data⁴⁻¹¹ have been generated on the effect of irradiation on the corrosion behavior of zirconium alloys, and, although various degrees of irradiation enhancement have been reported, the data sets have been too sparse to form the basis of a corrosion model. Recently, however, Dalgaard¹² has shown that the long-term corrosion behavior of Zircaloy-4 in commercial PWR's with forced convective heat transfer can be adequately described by use of the following out-of-pile rate equation:

$$\ln\left(\frac{dw}{dt}\right) = 16.73 - \frac{22200}{RT},$$

where

w = weight of oxygen reacted per unit area (mg/dm^2),

t = time in days.

Thus, it is believed that irradiation enhancement does not occur in commercial PWR's under normal operating conditions.

The situation regarding steam environments has not been as well modeled, but it is generally agreed that some form of corrosion enhancement does occur in the combined presence of steam and irradiation.^{7,9,11,13} In general, the accelerated corrosion kinetics are manifested in two forms^{11,13}; one form is that of accelerated but uniform growth of the oxides, and the second is of a very rapid film growth in localized regions, termed nodules, with eventual growth and overlap of areas of the locally thick oxides. Experimental evidence^{7,11} suggests that high oxygen concentrations in the steam phase and the

Figure C.1. Illustration of the degree of corrosion enhancement of Zircaloy-2 fuel cladding in the boiling water reactor, VBWR. Taken from ref. 13.

simultaneous presence of a neutron flux are required for this type of enhancement.

The magnitude of the radiation enhancement of corrosion in the coolant of a BWR is shown in Figure C.1, taken from ref. 13. The solid curves represent standard autoclave corrosion kinetics for Zircaloy-2, and the points and dashed curve represent data obtained from the Vallecitos boiling water reactor (VBWR) cladding operating such that the exit steam quality was 1 to 5%. These data indicate that the post transition corrosion curve for Zircaloy-2 cladding in VBWR, with a clad surface temperature of 300°C (575°F), is equivalent to that obtained from out-of-pile autoclave tests performed at $\sim$ 380°C (715°F). In addition, the in-pile data were obtained from measurements of the thickness of the uniform oxide film and thus do not reflect the even greater acceleration that occurs at nodules.

The above data are by no means complete. However, they do indicate that corrosion in the liquid coolant is more reproducible than in the mixed phase, and that the former is sufficiently well understood and documented to allow model representation. Relative to the LAFER concept, the data indicate that time and temperature considerations should be more important limitations than burnup per se. Therefore, if a flux spectrum similar to those of LWR's is assumed, design parameters such as coolant temperature, flow rate; fuel temperature, rod power, heat flux, coolant chemistry, and total exposure time will dictate the corrosion performance.

Because a portion of the hydrogen liberated by the corrosion reaction is absorbed by the cladding, and the presence of $\text{ZrH}_{1.6-2.0}$ precipitates could seriously affect mechanical behavior,¹⁴ it is necessary to minimize hydrogen pickup during corrosion. In out-of-pile tests, Zircaloy-2 is observed⁴ to pick up ~70% of the corrosion-liberated hydrogen, and Zircaloy-4 ~30%. From the standpoint of mechanical properties, the hydrogen concentrations should be less than ~200 to 500 ppm. Data from PWR's¹² with Zircaloy cladding indicate hydrogen concentrations of 30 to 100 ppm (average of 60 ppm) for 0.024-in. wall cladding after 1300 days of exposure. Assuming an average clad temperature of 600°F, this concentration would correspond to a hydrogen pickup of ~15% of the theoretical value. Experience¹³ with Zircaloy-2 in BWR environments indicates that fractional hydrogen pickups are ~15%.

Apparently, fractional hydrogen pickup tends to be somewhat suppressed in-pile; particularly under the conditions of enhanced oxidation observed in BWR's. Therefore, mechanical properties of LWR cladding are not limited by hydride precipitation.

2.3 Mechanical Behavior

Neutron irradiation significantly influences the mechanical behavior of zirconium-based alloys, and the general effects have been reviewed.¹⁵ The most notable effects of irradiation are an increase in yield strength and a corresponding decrease in ductility; and these changes are undoubtedly caused by interaction of the irradiation-induced damage with glide dislocations and the normal deformation processes. Because of the dependence of mechanical properties on the nature of the damage state, a brief summary of the nature of irradiation damage in zirconium-based alloys is warranted.

From experimental investigations of the nature and crystallography of irradiation-induced defects formed in zirconium and dilute alloys, it is concluded that most, if not all, defect clusters are small, geometrically simple, perfect dislocation loops. Although loop morphologies have been relatively well characterized, their vacancy or interstitial character is subject to debate, as is their exact mechanism of formation.

The majority of the investigations which have been performed report loops that possess Burgers vectors of the type $(a/3) \langle 1120 \rangle$.¹⁶⁻¹⁹ However, Burgers vectors with a nonbasal component have also been observed.²⁰ A wide range of loop orientations has been identified,^{16-18,20} indicating that there is no

Figure C.2. Effect of neutron fluence on damage accumulation in Zircaloy-2 irradiated at 300°C. ρ = volume density of visible defects. N_d = calculated number of atom sites involved in the damage, assuming that the defects are monolayer planar loops $\{1T00\}$. Taken from ref. 21.

unique habit plane. Instead, it appears that a range of orientations is possible, the exact position of any given loop being a function of its size in the manner described by Kelly and Blake.¹⁸ One observation in which there is unanimous agreement is that dislocation loops do not form on (0001) basal planes.

An area in which there is no agreement concerns the nature of the loops. Of the investigations which attempted to analyze dislocation loop character, one reported only interstitial loops,¹⁶ a second reported only vacancy loops,¹⁷ and the remaining, which are currently believed to be the most reliable, have reported the presence of both.¹⁸

An investigation aimed at determining the effect of neutron fluence on damage accumulation was performed by Williams and Gilbert.²¹ They examined Zircaloy-2 pressure-tube material after irradiation at 300°C to doses of 4×10^{19} to 8×10^{21} n/cm² ($E > 1.0$ MeV) and after postirradiation annealing experiments. Defect structures in Zircaloy-2 irradiated at 500°C to 2 to 4×10^{21}

Figure C.3. Change in yield stress as a function of neutron fluence for Zircaloy-2. The data best fit a saturation model. Taken from ref. 15.

n/cm^2 ($E > 1.0$ MeV) were also characterized. The most pertinent observations made in the structure irradiated at $300^\circ C$ were the following:

1. Dislocation loops remain geometrically simple and do not interact with each other.
2. Defect number densities saturate at $\sim 3 \times 10^{16}/cm^3$ at fluences greater than $5 \times 10^{20} n/cm^2$ (Figure C.2).
3. Fluences greater than $5 \times 10^{20} n/cm^2$ increase the average diameter of existing defects only slightly, the size range at $\sim 8 \times 10^{21} n/cm^2$ being 25 to 150 Å.

The defect structure in samples irradiated at $500^\circ C$ was very similar to that observed after the $300^\circ C$ irradiations, although there was a greater fraction of large loops at the higher irradiation temperature.

Annealing experiments performed at 300° to $550^\circ C$ revealed that the annealing behaviors of materials irradiated at 300° and $500^\circ C$ were similar. The 1-hr anneal at $450^\circ C$ caused a slight change in the size distribution of the

Figure C.4. Effect of neutron fluence and test temperature on the longitudinal necking strain of Zircaloy-2 irradiated at $280^\circ C$. Taken from ref. 15.

Figure C.5. Tensile and yield stress as functions of fluence for Zircaloy-2 plate tested at 250°C after irradiation at 327°C. Taken from ref. 23.

defects and some growth, such that the median diameter was increased to 150 Å. The 1-hr anneal at 500°C dissolved almost all of the discrete dislocation loops, with the retained loops being ~ 300 Å in diameter, while 1 hr at 550°C left only a dislocation network. These indications of observable recovery at temperatures of 450°C and above suggest that dynamic recovery should occur at 400°C and probably lower.

Just as the data on defect damage, Figure C.2, indicate a saturation effect, so does the dependence of strength and ductility on fluence. Evidence to support a saturation model includes the data of Bement,¹⁵ Harbinson and Baroch,²² and Rieger and Lee.²³ Bement's data¹⁵ on the effect of neutron fluence on the change in yield strength and necking strain of Zircaloy-2 are

Figure C.6. Total and uniform elongations as functions of fluence for Zircaloy-2 plate material tested at 250°C after irradiation at 327°C. Taken from ref. 23.

shown in Figures C.3 and C.4, respectively. The yield stress data in Figure C.3 are for Zircaloy-2 irradiated at 280°C and tested at room temperature. These data fit the following relationship:

$$\Delta\sigma_y = A(1 - \exp(-B\phi t))^{1/2},$$

where

$\Delta\sigma_y$ = change in yield stress,

ϕ = neutron flux,

t = time,

B = a constant related to the strength of the obstacles,

A = constant.

A saturation effect is also displayed by the ductility data shown in Figure C.4, which is a representation of the necking strain as a function of

Figure C.7. Reduction in area as a function of fluence for Zircaloy-2 plate tested at 250°C after irradiation at 327°C. Taken from ref. 23.

fluence for Zircaloy-2 irradiated at 280°C and tested at both room temperature and 300°C. The strength and ductility data of Rieger and Lee, Figures C.5 to C.7, for Zircaloy-2 irradiated at 327°C and tested at 250°C further corroborate

the saturation effects. It is seen from Figures C.5 and C.6 that the strength and ductility are essentially at their saturation values at fluences of $\sim 2 \times 10^{21}$ nvt (n/cm^2).

With regard to the effect of irradiation on mechanical properties, it should be noted that the radiation hardening can be recovered or annealed out by thermal treatment, and the temperature for recovery in 1-hr annealing treatments varies from 300° to 500°C , depending upon irradiation temperature fluence.^{15,24} These temperatures are consistent with those required for recognition of microstructural changes in the size and distribution of the loop-type defects. Because the limit of this range, namely, 300° to 400°C , is the range of service temperatures for LWR fuel cladding, it may be speculated that operation of fuel toward or beyond the upper limit of 400°C (in order to take advantage of dynamic recovery processes) would be beneficial in minimizing the ductility loss caused by irradiation. Unfortunately, however, corrosion becomes more severe at these high temperatures, and may become life limiting if extended exposures and high heat fluxes are anticipated. Nevertheless, the saturation effect displayed in the mechanical property data suggests that fuel burnup can be increased without infringing upon the limitations imposed by the intrinsic mechanical behavior of the cladding. Of course, fuel design and duty, as they impact on mechanical stress and strain of the cladding, may impose additional restrictions.

2.4 Pellet-Clad Interaction and Stress-Corrosion Cracking

A subject of contemporary interest is cladding failure caused by pellet-clad interaction (PCI). Such failures are associated with power changes during which stresses are imposed upon the cladding. Although the exact details of the mechanism of failure of Zircaloy cladding during PCI are not clearly understood, the operational parameters associated with and morphological characteristics of the failures²⁵⁻²⁸ have led to the conclusion that the failures are either purely mechanical in nature or the result of chemically assisted mechanical failure, i.e., stress-corrosion cracking. The latter possibility has been the subject of many recent publications, and it has been demonstrated²⁹⁻³⁶ that iodine, a prevalent fission product in UO_2 fuel, can indeed cause stress-assisted failure in Zircaloy under certain laboratory conditions.

The strongest argument linking iodine stress-corrosion cracking to fuel failure under power ramp conditions was provided by Garlick,³³ who showed that

the topography of the fracture surfaces of nonpenetrating cracks in failed fuel contained fracture surface morphologies similar to the surface features of samples cracked in iodine environments in the laboratory. However, the main difficulty in the universal acceptance of iodine stress-corrosion cracking as a dominant fuel failure mechanism is that the iodine concentration in most fuels is below that shown to be the threshold value for iodine-induced cracking. Using values for fission product inventories reported by Lustman,³⁷ an upper bound calculation for the iodine concentration yields about 0.7×10^{-5} g I_2/cm_2 Zr at the burnup rate of 30,000 MWD/MTU. This is about a factor of 3 less than the critical concentration required for cracking in laboratory tests reported by Wood.³² Furthermore, the assumption used in this upper bound calculation is that all the iodine formed migrates to the clad surface, and, surely, some of the iodine would be expected to be combined with other species such as cesium which is known to be concentrated within the fuel and away from the cold surface adjacent to the cladding.³⁸

In addition to laboratory work on iodine, tests have revealed that Zircaloy is susceptible to stress-corrosion cracking in the presence of cesium³⁹ and cadmium,⁴⁰ which are also fission products. Also, postirradiation examination of PWR fuel from the Maine Yankee reactor, in which the failures were purportedly⁴¹ caused by stress-corrosion cracking, revealed the presence of CsI compounds at the inner surface of the cladding, suggesting that possibly cesium and iodine, even in the combined state, may act as an aggressive species. Additional data on fuel failures under power ramp conditions should be forthcoming in the near future, particularly from the EPRI-sponsored programs⁴² which emphasize fission-product-induced stress-corrosion cracking.

2.5 Void Formation in Zirconium and Its Alloys

Because swelling due to void formation has been observed in many metals subjected to high-fluence fast-neutron damage, some concern for such swelling may exist in the LAFER concept should the flux spectrum shift appreciably from that of the spectrum in LWR's. A review of the available data, however, indicates that voids have not been observed in zirconium alloys even when irradiated²¹ in a fast reactor spectrum to 4×10^{21} n/cm².

Wolfenden and Farrell⁴³ have reviewed several possible explanations for the remarkable behavior of zirconium with respect to void formation. Although

their arguments are inconclusive, the most likely explanations were thought to be the following:

1. Dislocation networks, a prerequisite for void formation, do not generally form in irradiated zirconium.
2. Helium, a strong promoter of void formation, is generally formed in only very small quantities in zirconium because of its low (n, α) cross section of 0.055 mb.

The first explanation seems inadequate in view of the fact that cold-worked Zircaloy-2 does not form voids.²¹ However, there is evidence which does support the latter explanation. Carpenter⁴⁴ irradiated crystal bar zirconium in a 1-MeV high-voltage electron microscope at 450°C and a displacement rate of 2.4×10^{-3} dpa/sec. During the irradiation, unfaulted dislocation loops formed directly and grew into a dense tangled structure. If the foils were not preinjected with helium, no void formation was observed up to a dose of 4.3 dpa (3.4×10^{21} n/cm²) ($E > 1.0$ MeV). However, if the foils were preinjected with 10 appm He, voids were observed at a dose of 3.4 dpa (2.7×10^{21} n/cm²) ($E > 1.0$ MeV). At a dose of 8.6 dpa (6.9×10^{21} n/cm²) ($E > 1.0$ MeV), the voids had grown to 200 Å diameter. Carpenter concluded that voids can be produced in zirconium only if an insoluble gas is present to stabilize void nuclei. This is currently the best existing explanation for the absence of void formation in fast-neutron-irradiated zirconium.

3. CONCLUSIONS

Relative to projected burnups of LWR fuel to 100,000 MWD/MTU, as is proposed in the LAFER concept, the existing data base includes appreciable information from commercial plants to 30,000 MWD/MTU and experimental fuel to peak pellet burnups of 65,000 MWD/MTU. Thus, the required extrapolation ranges from 3 to 1.5 times the existing experience. cursory analyses of the available data on the usual life-limiting properties of corrosion and mechanical behavior indicate the following:

- ° In water, under single-phase convective heat transfer, the corrosion kinetics of Zircaloy are essentially unaffected by irradiation in LWR's.
- ° Corrosion kinetics are accelerated under BWR conditions, and the degree of acceleration is sensitive to exposure time and oxygen concentration.

To the author's knowledge, there is no existing model to describe the kinetics under boiling conditions.

- ° Corrosion oxide film buildup, because of the thermal insulation effect, will impose limitations on fuel temperature, heat flux, flow rate, and other thermal hydraulic parameters. The fuel design and environment variables must take these limitations into account.
- ° The strength and ductility of Zircaloy are affected by irradiation, with the former increasing and the latter decreasing. At the temperatures of interest, the effects saturate at LWR fluences of $1 \text{ to } 2 \times 10^{21} \text{ n/cm}^2$, corresponding to burnups $\sim 12,000 \text{ MWD/MTU}$. Therefore, existing LWR fuel operates successfully long after the mechanical properties have reached their saturation values. Thus, there appears to be no intrinsic materials limitation to appreciably increasing fuel burnup. Design considerations, however, must ensure that failure criteria are not exceeded.
- ° Failure due to pellet-clad interaction may be of primary concern if the fuel operates for extended periods after the cladding and pellets are in intimate contact. Design stress limitations may require revision if fission-product-induced stress-corrosion cracking is operative. Limitations on rod power, power changes, and rate of power change may be required.
- ° No problems due to clad swelling caused by void formation are anticipated as long as the neutron flux spectrum is that typical of LWR's.

4. RECOMMENDATIONS

It is difficult to make specific recommendations to minimize material-design conflicts at this stage of the LAFER design. However, using the assumptions made earlier, and after this brief review of existing data, additional thought or studies in the following areas appear warranted:

- ° An assessment must be made of beam heating effects and radiation damage caused by the 1-GeV proton beam if this beam impinges directly on the fuel. If other target material is utilized for neutron generation, then the energy spectrum of the neutrons generated should be defined and compared with that prevalent in LWR's upon which the data base has been established.

- ° A detailed analysis of high burnup fuel should be performed to expand the data base. In particular, mechanical property data should be reviewed (or generated) to ensure that extrapolation of the saturation effect is valid.
- ° An assessment of fission product inventory and release should be performed to ensure an inventory of potentially aggressive species no greater than that in LWR fuel.
- ° If steam is the candidate atmosphere, or boiling is to occur in the LAFER blanket, then more detailed assessment of the corrosion behavior is warranted.
- ° An assessment of the impact of high burnup on the behavior and properties of the ceramic fuel pellets should be performed.
- ° Design of the fuel for the LAFER-LWR concept should be consistent with proven fuel design criteria. This may be accomplished by continued input from and in collaboration with fuel design personnel.

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APPENDIX D

THERMAL-HYDRAULIC CALCULATIONS OF LAFER SYSTEMS

Introduction

Thermal-hydraulic calculations have been carried out for the blankets of accelerator-reactor systems based on the geometry of PWR (Figure 5.4) and CANDU (Figure 5.5) fuel assemblies. The objectives of this study were to investigate the capability of helium and two-phase mixture of steam and liquid water cooling for such assemblies used in LAFER (linear accelerator fuel enricher/regenerator), and to map out the range of desirable operating conditions. Figures D.1 to D.20 illustrate the range of capabilities for cooling PWR and CANDU fuel assemblies with a two-phase mixture of steam and water. Figures D.21 through D.50 illustrate helium cooling capabilities for the same PWR and CANDU fuel assemblies.

Figure D.1. Two-phase mass flow rate versus inlet quality of three inlet velocities for PWR fuel.

Figure D.2. Two-phase exit quality versus inlet quality by various fuel element heat fluxes and inlet velocities for PWR fuel.

Figure D.3. Two-phase exit quality versus inlet quality at two different fuel element heat fluxes and various inlet velocities for PWR fuel.

Figure D.4. Two-phase exit velocity versus inlet velocity at various fuel element heat fluxes for PWR fuel.

Figure D.5. Two-phase exit quality versus average fuel element heat flux at three inlet qualities and inlet velocities for PWR fuel.

Figure D.6. Two-phase pressure drop versus inlet quality at three inlet velocities and various fuel element heat fluxes for PWR fuel.

Figure D.7. Steam volume fraction versus inlet quality at three inlet velocities and various fuel element heat fluxes for PWR fuel.

Figure D.9. Two-phase heat transfer coefficient versus inlet quality at three inlet velocities and various fuel element heat fluxes for PWR fuel.

Figure D.8. Steam volume fraction versus inlet quality at three inlet velocities and two fuel element heat fluxes for PWR fuel.

Figure D.10. Two-phase film temperature drop versus inlet quality at three inlet velocities and various fuel element heat fluxes for PWR fuel.

Figure D.11. Two-phase mass flow rate versus inlet quality of three inlet velocities for CANDU fuel.

Figure D.12. Two-phase exit quality versus inlet quality at two fuel element heat fluxes and three inlet velocities for CANDU fuel.

Figure D.13. Two-phase exit quality versus inlet quality at two different fuel element heat fluxes and three inlet velocities for CANDU fuel.

Figure D.14. Two-phase exit velocity versus inlet velocity at various fuel element heat fluxes for CANDU fuel.

Figure D.15. Two-phase exit quality versus average fuel element heat flux at three inlet qualities and inlet velocities for CANDU fuel.

Figure D.16. Two-phase pressure drop versus inlet quality at three inlet velocities and various fuel element heat fluxes for CANDU fuel.

Figure D.17. Steam volume fraction versus inlet velocity at three inlet velocities and two fuel element heat fluxes for CANDU fuel.

Figure D.18. Steam volume fraction versus inlet velocity at three inlet velocities and two fuel element heat fluxes for CANDU fuel.

Figure D.19. Two-phase heat transfer coefficient versus inlet quality at three inlet velocities and various fuel element heat fluxes for CANDU fuel.

Figure D.20. Two-phase film temperature drop versus inlet quality at three inlet velocities and various fuel element heat fluxes for CANDU fuel.

Figure D.21. Temperature difference ($T_{out} - T_{in}$) versus helium pressure and flow velocities for PWR fuel.

Figure D.22. Helium pressure drop versus total pressure and flow velocities for PWR fuel.

Figure D.23. Film temperature drop versus helium pressure and flow velocities for PWR fuel.

Figure D.24. Helium heat transfer coefficient versus total pressure and flow velocities for PWR fuel.

Figure D.25. Fuel clad temperature versus helium pressure and flow velocities for PWR fuel.

Figure D.26. Temperature difference ($T_{out} - T_{in}$) versus helium pressure and inlet temperatures for PWR fuel.

Figure D.27. Helium pressure drop versus total pressure and inlet temperature for PWR fuel.

Figure D.28. Film temperature drop versus helium pressure and inlet temperatures for PWR fuel.

Figure D.29. Helium heat transfer coefficient versus total pressure and inlet temperatures for PWR fuel.

Figure D.30. Fuel clad temperature versus helium pressure and inlet temperatures for PWR fuel.

Figure D.31. Temperature difference ($T_{out} - T_{in}$) versus helium pressure and fuel element heat fluxes for PWR fuel.

Figure D.32. Helium pressure drop versus total pressure and several fuel element heat fluxes for PWR fuel.

Figure D.33. Film temperature drop versus helium pressure and fuel element heat fluxes for PWR fuel.

Figure D.34. Helium heat transfer coefficient versus total pressure and fuel element heat fluxes for PWR fuel.

Figure D.35. Fuel clad temperature versus helium pressure and fuel element heat fluxes for PWR fuel.

Figure D.36. Temperature difference ($T_{out} - T_{in}$) versus helium pressure and flow velocities for CANDU fuel.

Figure D.37. Helium pressure drop versus total pressure and flow velocities for CANDU fuel.

Figure D.38. Film temperature drop versus helium pressure and flow velocities for CANDU fuel.

Figure D.39. Helium heat-transfer coefficient versus total pressure and flow velocities for CANDU fuel.

Figure D.40. Fuel clad temperature versus helium pressure and flow velocities for CANDU fuel.

Figure D.41. Temperature difference ($T_{out} - T_{in}$) versus helium pressure and inlet temperatures for CANDU fuel.

Figure D.42. Helium pressure drop versus total pressure and inlet temperatures for CANDU fuel.

Figure D.43. Film temperature drop versus helium pressure and inlet temperatures for CANDU fuel.

Figure D.44. Helium heat transfer coefficient versus total pressure and inlet temperatures for CANDU fuel.

Figure D.45. Fuel clad temperature versus helium pressure and inlet temperatures for CANDU fuel.

Figure D.46. Temperature difference ($T_{out} - T_{in}$) versus helium pressure and fuel element heat fluxes for CANDU fuel.

Figure D.47. Helium pressure drop versus helium pressure and fuel element heat fluxes for CANDU fuel.

Figure D.48. Film temperature drop versus helium pressure and fuel element heat fluxes for CANDU fuel.

Figure D.49. Helium heat-transfer coefficient versus total pressure and fuel element heat fluxes for CANDU fuel.

Figure D.50. Fuel clad temperature versus helium pressure and fuel element heat fluxes for CANDU fuel.

Table D-1

Fuel Assembly Data

	PWR*	CANDU**
Number of assemblies (or channel)	205	380
Fuel elements per assembly (or per bundle)	208	37
Bundles per channel	-	12
Length of assembly (or channel), ft	12	19.5
Equivalent diam. of flow passages between elements	0.042	0.0290
Free flow area per assembly (or per channel), ft ²	0.30	0.0360
Heat-transfer area per assembly (or per channel), ft ²	281	96.17
Average heat flux at surface of fuel element, Btu/ft ² hr	197,000	193,000
Volume ratio, in fuel assembly coolant/fuel (or bundle)	1.5	0.8

*Steam/nuclear installations for electric utilities.

**UE&G, ERDA 770630, C00-2477-4.

Calculational Procedure for Two-Phase Flow Coolant

The fuel assembly design data pertinent to the thermal hydraulic calculation are summarized in Table D-1. The ranges of independent input variables used in the analysis are as follows:

- P_o = average heat flux from fuel pins in the blanket fuel assembly
(expressed as a multiple of the average heat flux in either PWR or CANDU elements).
= 0.25, 0.5, 0.75, 1.0 x (average PWR or CANDU heat flux);
- x_i = inlet steam quality to channel
= $\frac{\text{mass of steam}}{\text{mass of steam and water}}$
= 0.3, 0.5, 0.7;
- v_i = inlet velocity of two-phase mixture to channel
= 25, 50, 75, 100 ft/sec;
- P_{2p} = inlet pressure of two-phase mixture to channel
= 1000 psia.

As discussed in Section 5.3 of this report, two-phase mixtures are desirable because they do not moderate neutrons very well by comparison with liquid coolants, and retain good heat transfer/transport capability. The key questions examined in this study are:

- Is adequate cooling of the blanket fuel assemblies achievable at low overall densities of two-phase mixtures (e.g., 10% of liquid density)?
- Are two-phase pressure losses acceptable for the range of heat fluxes expected in the blanket?
- Can hot spots occur as a result of local dryout?

The following dependent variables are calculated for the above range of input variables:

- x_e = exit quality of two-phase mixture from channel (calculated from overall enthalpy balance, assuming pressure drop is negligible);
- v_e = exit velocity of two-phase mixture from channel (calculated from overall mass balance);
- ΔP_{2p} = two-phase pressure drop across channel (based on average quality in the channel);
- α_e = exit steam volume fraction from channel (based on average quality in the channel);

$\bar{h}$ = average heat transfer coefficient in channel (based on average quality);

ΔT_f = film temperature drop (based on average heat-transfer coefficient in channel).

The following correlations are used:

1. Pressure Drop The Martinelli-Nelson two-phase friction factor^{1,2} is used to determine the pressure drop. The two-phase pressure drop is expressed in terms of the single-phase liquid pressure drop as

$$\left(\frac{\Delta P}{\Delta L} \right)_{TPF} = \left(\frac{\Delta P}{\Delta L} \right)_{LPF} \phi_{l\,tt}^2,$$

where the correlation $\phi_{l\,tt}$ parameter is a function of

$$X_{tt} = \left(\frac{\rho_g}{\rho_l} \right)_{SAT}^{1/(2-n)} \left(\frac{\mu_l}{\mu_g} \right)^{n/(2-n)} \frac{(1-x)}{x},$$

n is determined empirically for $f = C/Re^n$, and $n = 0.20-0.25$ for turbulent flow.

2. Void Fraction The steam volume fraction, α , is calculated from the Bankoff Correlation³ of slip ratio,

$S = (1-\alpha)/(K-\alpha)$ = slip ratio,

$K = K_j + (1-K_j)\alpha^\gamma$,

$\gamma = 3.33 + 0.18 \times 10^{-3} P + 0.46 \times 10^{-6} P^2$,

$K_j = 0.71 + 10^{-4} P$,

P = pressure (psia).

This steam fraction is determined from the relationship:

$$\alpha = \frac{1}{[(1 + Sv)_f/v_g][(1-x)/x]},$$

where

v_f = specific volume of water,

v_g = specific volume of steam,

x = flow quality.

3. Heat-Transfer Coefficient The heat-transfer correlation of Schrock and Grossman with Wright constants⁴ is used to determine the two-phase heat-transfer coefficient:

$$h = 6700 \left[\frac{\phi_s}{GH_{fg}} + 0.00035 \left\{ \left(\frac{x}{1-x} \right)^{0.9} \left(\frac{\rho_f}{\rho_g} \right)^{0.5} \left(\frac{\mu_g}{\mu_f} \right)^{0.1} \right\}^{0.66} \right] \\ \times \left[\left(\frac{0.023k_f}{D} \right) \left(\frac{DG(1-x)}{\mu_f} \right)^{0.8} (Pr)^{0.4} \right]$$

The physical properties are evaluated at saturation conditions, where

H_{fg} = heat of vaporization, Btu/lb,

ρ = density, lb/ft³,

k = thermal conductivity, Btu/ft hr °F.

Calculational Procedure for Helium Coolant

The fuel assembly design data for helium coolant is the same (Table D-1) as that for two-phase coolant. The ranges of independent variables used in the analysis are as follows:

P_o = average heat flux from fuel pins

= 0.25, 0.5, 0.75, 1.0 x (average PWR or CANDU heat flux);

v_i = inlet velocity of helium to channel

= 100 to 600 ft/sec;

P_{2p} = inlet pressure of helium to channel

= 50, 75, 100 atm;

T_i = inlet temperature of helium to channel

= 0 to 600°F.

Helium is a very attractive coolant option, since it has the lowest degree of neutron moderation, is relatively nonreactive (impurities in helium may be reactive at higher temperature), and eliminates the possibility of dryout of heat transfer surfaces, which is of potential concern with two-phase coolant. There are several key questions about the suitability of helium coolant, however, which are examined in this study:

- a. Is adequate cooling of the blanket fuel assemblies feasible without excessive pressure drop and/or excessively high cladding temperatures?
- b. Will helium coolant yield acceptable power cycle efficiencies?
- c. Are required pressure levels for good performance too high for practical blanket systems?

The following limiting criteria are taken for helium coolant:

- a. Oxidation limits - 700°F maximum clad temperature to ensure that Zircaloy does not oxidize because of impurities in helium.
- b. A coolant operating temperature range that yields a power cycle efficiency of ~30%.
- c. A pressure drop of no more than 10% of base pressure.

The following dependent variables are calculated for the above range of input variables:

- v_o = exit velocity of helium from channel;
 ΔP_o = helium pressure drop along channel;
 $\bar{h}$ = average heat transfer coefficient in channel;
 T_o = exit temperature of helium from channel;
 T_{max} = maximum clad temperature;
 ΔT_f = film temperature drop.

The standard Dittus-Bolter relationship is used to determine the heat-transfer coefficient for helium:

$$h = 0.23 \frac{k_g}{D} \left(\frac{DG}{\mu_g} \right)^{0.8} \left(\frac{Pr}{g} \right)^{0.4}$$

Discussion

To return to the questions about the two-phase cooling of the blanket listed earlier, the following answers can be formulated:

- a. Adequate cooling of the blanket fuel assemblies appears achievable at low overall two-phase coolant densities, i.e., ~10% of liquid density, at least up to fuel pin heat fluxes that are comparable to average PWR and CANDU heat fluxes (~200,000 Btu/ft² hr). In any case, these heat fluxes would appear to be upper limits, regardless of coolant choice, to ensure good cooling and safe operation. Since the blanket area can be readily designed to keep maximum heat fluxes at below this limit, there appears to be no problem in achieving adequately low coolant densities with two-phase mixtures.

b. Two-phase flow pressure drops with PWR assemblies are quite low and have significant impact on power generation efficiency or coolant properties. With CANDU assemblies, two-phase flow pressure drops are considerably higher, because of lower coolant/fuel volume ratio and longer pressure-tubes. The length considered in the CANDU cases examined, 19.5 ft, appears to be an upper limit to pressure-tube length if the maximum heat flux in the assemblies is on the order of 200,000 Btu/ft² hr. If heat fluxes are lower, pressure-tubes can be correspondingly longer. If longer pressure-tubes are not possible at high heat fluxes, it probably will be necessary to use split flow headering, with the pressure-tube fed at its middle with inlet coolant.

The coolant would then exit at each end of the pressure-tube. There appears to be no fundamental reason why this is not feasible, but it will involve a more complex piping arrangement. With this flow, the length of the blanket pressure-tubes could be doubled, so that pressure-tubes could be 40 ft long at heat fluxes of 200,000 Btu/ft² hr.

c. Gross dryout of the fuel assembly is of no concern, since outlet qualities of the two-phase mixture can readily be kept in the range of 0.4 to 0.7, depending on coolant flow conditions. The question of localized dryout is much more difficult to answer, since it is affected by fuel element geometry, evenness of flow, etc. Ultimately, experiments on fuel assemblies will be required to assure that dryout cannot occur. The margin of safety is measured by going to low inlet qualities and high inlet velocities. For PWR fuel assemblies, an inlet quality as low as 0.2 and a flow velocity of 75 ft/sec yield a pressure drop of 36 psia, which is acceptable. The effective moderator density will be higher, however, on the order of 20% of liquid density. This probably would be acceptable from the standpoint of neutronics. Outlet quality under these conditions would be 0.3, so that the margin of safety would be considerable. Conditions for CANDU fuel are similar, with an inlet quality of 0.2 corresponding to outlet quality of 0.4. It thus appears that a large margin of safety against local dryout can be provided. Experiments on electrically heated simulated fuel assemblies can be carried out to determine local heat transfer conditions, and these should demonstrate that local dryout is not a problem.

With regard to the helium coolant option, it appears feasible for PWR fuel elements at heat fluxes up to 1.0 x the average PWR heat flux (197,000 Btu/ft² hr) and 0.75 x average CANDU heat flux (193,000 Btu/ft² hr). Relatively high

inlet flow velocities of ~ 400 ft/sec and high helium pressure of 100 atm are desirable to meet the criterion of 700°F maximum clad temperature. Helium pressure drops are somewhat higher than with two-phase flow, but still acceptable (~ 20 to 50 psia).

Overall, both helium and two-phase coolant appear feasible for application to LAFER systems, in terms of their thermal hydraulic performance. Helium has a somewhat higher pressure loss, but less neutron moderation and there is no concern about local dryout of fuel element surface. The choice as to which to use will be best made by comparing the overall performance of complete system designs for a LAFER optimized for helium coolant to a LAFER optimized for two-phase coolant.

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APPENDIX E

COST ESTIMATES

1. LINEAR ACCELERATOR COSTS

After consideration of the technical feasibility of the linear accelerator, the next important question to be answered is whether such an accelerator is economically feasible.

A preliminary study was undertaken to determine the cost and power requirements for linear accelerators in the range of currents and energies of interest.

Figure E.1. Proton linac capital costs as a function of beam currents and accelerator gradients.

Figure E.2. Proton linac input power as a function of beam currents and accelerator gradients.

It is clear that, for any of the systems under consideration, the linac will need to be optimized with respect to capital cost (minimum) and power utilization efficiency (maximum). For a given beam current and energy, the cost and efficiency parameters can be optimized by varying the acceleration gradient of the machine. To study this effect, a computer program has been developed, and the results are given on Figures E.1 and E.2 for proton linacs, and in Figures E.3 and E.4 for deuteron linacs.

It is interesting to note from these figures that the capital cost of the machine varies very little between gradients of 0.75 MeV/m to 2.5 MeV/m. Figure E.5 shows this variation for the case of a 300-mA proton linac. However, the ac input power to the linac is more strongly influenced by the accelerating gradi-

Figure E.3. Deuteron linac capital costs as a function of beam currents and accelerator gradients.

ent, as shown in Figures E.2 and E.4. No attempt was made here to weigh the capital cost factor against the input power required for any given machine.

The following gives the parametric basis used for the exercise:

- a. The capital costs given here are in 1978 dollars and do not include: depreciation, escalation, E.D.I. and A, or the cost of real estate.
- b. The accelerator structure shunt impedances used in calculating the excitation power have been taken from accurate calculation done with the MESSYMESH program (MURA) and the LALA program (LASL), and properly adjusted for β (where $\beta = v/c$) and frequency changes.
- c. The accelerating structure frequency choices have been made on the basis of earlier calculations on matched input beams from parameters for high-

Figure E.4. Deuteron linac input power as a function of beam currents and accelerator gradients.

current linacs. The initial frequency is used up to $\beta = 0.5$ (~ 150 MeV) protons, and then it is increased by a factor of 3.

d. For each case, the total cost of a machine includes a M\$20 fixed investment. This amount covers labs and offices, injection and beam transport systems, control room, etc.

Unit Costs

For the purpose of calculation, the linac was divided into five areas, each costed separately on the following basis.

a. Accelerator: This calculation includes the cost of the linac structure, focusing system, vacuum, beam diagnostics, radiation monitors, services distribution, etc. The unit cost used is \$55,000/m for a 300-MHz structure.

Figure E.5. Variation of capital cost for a 300-mA proton linac.

This cost is adjusted for frequency change from \$26,000 at 1350 to \$134,000 at 50 MHz.

b. Radio Frequency: This cost includes the cost of the entire rf system, including input power transformers and rectifiers, amplifier chains, controls, transmission lines, etc. The unit cost used here is \$650,000/MW rf. This unit cost is based on systems available presently. Because of the large amount of rf power required, there will be a real incentive to develop new rf power sources specifically for this application which should yield lower costs.

c. Cooling System: This system includes the cooling towers, pumps, heat exchangers, piping, and water treatment system required for the entire facility. The system is sized on having to cool a 60% efficient rf system plus the excitation power required for the accelerator. The unit cost taken is \$70,000/MW cooling load calculated.

d. Linac Tunnel: A typical tunnel cross section was designed consisting of a 2-ft-thick reinforced concrete boxlike structure supporting a 5-m earth shield. In addition, a retaining wall 8 m high, 5 m distant from the tunnel, and connected to the tunnel by frequent ribs was provided. This wall separates the rf area from the tunnel. The cost also includes that of lighting, power distribution, cranes, etc. The unit cost for the tunnel is \$12,000/m for a 300-MHz, 10 x 10-ft clear tunnel. The cost is adjusted for frequency changes from

\$7,000/m for a 1350-MHz, 8 x 8-ft tunnel to \$30,000/m for a 50-MHz, 20 x 20-ft tunnel.

e. Rf Building: This building houses all the high-power rf equipment, electronics, power supplies, and other hardware necessary to power the accelerator. By and large, the high-power rf system is the major space user of the building (>80%). The structure considered here is of standard construction, a light structure, without crane coverage, and with no air conditioning. We have estimated the cost of this building to be \$80/ft² and with a requirement of 500 ft²/MW rf, we have used \$40,000 MW rf as the unit cost of this structure.

Conclusions

It is interesting to note that the major portion of the capital cost is for rf power. For a typical 1.5-GeV 300-mA proton linac, these costs in % of total are as follows:

	<u>Total</u>	<u>Rf</u>	<u>Accel.</u>	<u>Cooling</u>	<u>Tunnel</u>	<u>Rf bldg.</u>
2-MeV/m machine	100%	72%	10%	8%	3%	6%
1-MeV/m machine	100%	68%	17%	6%	4%	5%

From these numbers it is clear that the major attention toward reducing accelerator costs must be directed to the rf system, where any improvement will be translated into major savings in both capital expenditure and input power requirements.

2. BALANCE OF PLANT COSTS

For the purpose of this cost estimate, the Balance of Plant (BOP) is defined as that part of the LAFER facility which is complementary to the linear accelerator and its associated systems. Thus, included in the BOP are the target-blanket assembly, the containment system, and the power generation system.

The material presented in this appendix is organized to correspond to the uniform system of accounts (USAEC Report NUS-531) used for the detailed cost estimate.

A preliminary cost estimate for the BOP has been performed at the two-digit account level. The accounts used are as follows:

<u>Code-of-Account</u>	<u>Title</u>
20	Land and land rights
21	Structures and improvements
22	Reactor plant equipment
23	Turbine plant equipment
24	Electric plant equipment
25	Miscellaneous plant equipment
26	Main condenser heat rejection system
91	Construction services
92	Home office engineering and services
93	Field office engineering and services

The descriptions associated with the accounts 20 through 26 address the land required and the plant equipment design. This portion corresponds to the "direct cost" portion of the cost estimate. The descriptions associated with the accounts 91 through 93 define the construction support activities. This corresponds to the "indirect cost" portion of the cost estimate. The sum of the "direct cost" and the "indirect cost" is the "total case construction cost."

Account 20: Land and Land Rights

The land required for the linear accelerator fuel enricher/regenerator is greater than the amount considered in Account 20 for an equivalent power output because of the land requirement for the linear accelerator. As presented, the Account 20 includes the land required for siting the BOP. This amount of land also includes the exclusion area as required by a conventional reactor. Land rights for right-of-way and approaches to the sites are not included.

Account 21: Structures and Improvements

The primary structure in the BOP is the reactor containment building. This structure is a Seismic Category I reinforced concrete enclosure, consisting

of a vertical cylindrical shell with a hemispherical dome and flat base. A continuously welded plate liner is attached to the inside face of the concrete to ensure a high degree of leak-tightness. The containment and other reinforced concrete Seismic Category I structures house all safety-related equipment essential for safe plant operation, shutdown, and control.

The major Seismic Category I structures of the BOP include:

- Reactor containment building
- Primary auxiliary building
- Waste process building
- Fuel storage building
- Control and diesel generator building
- Emergency feedwater pump building (including electrical tunnels)
- Main steam and feedwater pipe enclosures
- Ultimate heat sink structure

The major Nonseismic Category I structures house Nonseismic Category I equipment and components not required for plant safety or safe shutdown. These structures are designed to assure that their failures will not cause loss of function of a Seismic Category I structure. The major Nonseismic Category I structures include:

- Start-up boiler building
- Administration and service building
- Makeup water intake and discharge structure
- Fire pump house
- Nonessential switchgear building
- Circulating water pump house
- Holding pond
- Chlorination building
- Makeup water pretreatment building

Account 22: Reactor Plant Equipment

The reactor design proposed in this study consists of a flowing Pb-Bi primary target surrounded by a blanket of PWR fuel assemblies located inside vertical pressure-tubes cooled with a two-phase (water-steam) mixture. The reactor plant equipment includes the following:

- Reactor equipment
- Main heat transfer and transport systems (blanket and target)
- Safeguard systems
- Radwaste processing system
- Fuel handling and storage system
- Other reactor plant equipment
- Instrumentation and control system

Account 23: Turbine Plant Equipment

The turbine plant equipment consists of the steam and power conversion system components of the steam cycle. Included are:

- Turbine generator
- Condensing systems
- Feed heating system
- Other turbine plant equipment
- Instrumentation and control
- Turbine plant miscellaneous items

Account 24: Electric Plant Equipment

The electric plant equipment conveys the electric power generated in the plant to the low voltage bushings of the generator step-up (GSU) transformers, controls and meters the electric energy, and protects the components through which the power flows. It is the source of power for the plant auxiliaries and the plant control, for the protection and surveillance systems during operation, and for the plant protection system and engineered safety features during normal operation, abnormal conditions, and accident conditions. The electric plant equipment includes:

- Switchgear
- Station service equipment
- Switchboards
- Protective equipment
- Electrical structures and wiring containers
- Power and control wiring

Account 25: Miscellaneous Plant Equipment

Miscellaneous plant equipment includes systems for maintenance or general supply of plant equipment requirements as follows:

- Transportation and lifting equipment
- Air, water, and steam service system
- Communications system
- Furnishing and fixtures

Account 26: Main Condenser Heat Rejection System

The main heat rejection system is a circulating water system consisting of structures and mechanical equipment which serve the main condensers and service water system to reject the plant heat through mechanical draft wet cooling towers. Makeup water is assumed to be available from a nearby river. The main condenser heat rejection system includes:

- Makeup water intake and discharge structures
- Mechanical equipment

Construction Support Activities Description

The descriptions associated with Accounts 91 through 93 address the construction support activities.

Account 91: Construction Services

This account includes costs for services, functions, expenses, taxes, and other indirect costs related to:

- Temporary construction facilities
- Construction tools and equipment
- Payroll insurance and taxes
- Permits insurance and local taxes

Account 92: Home Office Engineering and Services

Included in this account are:

Home office services
Home office quality assurance
Home office-construction management

Account 93: Field Office Engineering and Service

Field-related activities included in this account are:

Field office expenses
Field job supervision
Field-quality assurance
Test and start-up engineering

BOP Cost Estimates

a. Helium Cooling. The capital cost estimates for the BOP of the proposed LAFER facility when helium is used for blanket cooling is given in 1978 dollars at a two-digit account level in Table E-1 and at a three-digit account level in Table E-4. The numbers shown were estimated on the basis of the capital cost information given in DOE's EEDB Initial Update Study of the 900-MW(e) GCFR, UE&C-ERDA-770930 (COO-2477-16). Proper multipliers were applied to account for the difference in power output and engineering design. The total capital cost for the 455-MW(e) LAFER facility is 375.5 million dollars. This translates to \$825.3/kW(e) compared to \$757.5/kW(e) for the 900-MW(e) GCFR.

b. Liquid-D₂O Cooling. The capital cost estimated for the BOP of the proposed LAFER facility when liquid D₂O is used for blanket cooling is given in 1978 dollars at a two-digit account level in Table E-2 and at a three-digit account level in Table E-5. The numbers shown were estimated on the basis of the capital cost information given in DOE's EEDB Initial Update Study of the 1100-MW(e) PHWR, UE&C-ERDA-770631 (COO-2477-13). Upon applying proper multipliers to account for the difference in power output and engineering design, the total capital cost for the 402-MW(e) LAFER facility was estimated to be 411 million dollars. This translates to \$1022.6/kW(e)* compared to \$682.8/kW(e) for the 1100-MW(e) PHWR.

*When the cost of D₂O inventory is included, the total plant cost is \$428.7 millions and the unit cost is \$1065.5/kW(e).

c. Two-Phase H₂O Cooling. The capital cost estimated for the BOP of the proposed LAFER facility when two-phase H₂O is used for blanket cooling is given in 1978 dollars at a two-digit level in Table E-3 and at a three-digit account level in Table E-6. The numbers shown were estimated on the basis of the capital cost information given in DOE's EEDB Initial Update Study of the 1139-MW(e) PWR UE&C-ERDA-770501 (COO-2477-5). Proper multipliers were applied to account for the difference in power output and engineering design. The total capital cost for the 451-MW(e) LAFER facility is 387.9 million dollars. This translates to \$860.9/kW(e) compared to \$580.9/kW(e) for the 1139-MW(e) PWR.

It should be noted that these cost estimates are purely the "best guesses" at present and they may have a range of +10 to -30%. Further refinement can be accomplished upon developing a complete equipment list and using the CONCISE code at a seven-digit accounting level.

Table E-1

Capital Cost Estimate Summary (Two-Digit Accounts)
 455-MW(e) Helium-Cooled LAFER Balance of Plant
 (In thousands of 1978 dollars)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
20	Land and land rights	\$ -	\$ --	\$ 1,800	\$ 1,800
21	Structures and improvements	3,975	39,961.5	30,546.5	74,483
22	Reactor plant equipment	49,808	20,119	3,001	72,928
23	Turbine plant equipment	39,237	10,138	2,120	51,495
24	Electric plant equipment	8,762	10,334	4,895	23,991
25	Miscellaneous plant equipment	4,912	2,630	438	7,980
26	Main condenser heat reject system	6,095	2,369	1,270	<u>9,734</u>
2	Total direct cost				<u>242,411</u>
91	Construction services	13,786	15,903	21,546	51,235
92	Home office engineering and services	63,268	-	-	63,268
93	Field office engineering and services	16,623	-	1,962	<u>18,585</u>
9	Total indirect cost				133,088
	Coolant cost (not included)				-
	Total base cost				<u>\$375,499</u>
	Unit cost \$/kW	<u>\$375,499</u>		=	\$825.27/kW
		455			

Table E-2

Capital Cost Estimate Summary (Two-Digit Accounts)
 402-MW(e) Liquid-D₂O-Cooled LAFER Balance of Plant
 (In thousands of 1978 dollars)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
20	Land and land rights	\$ -	\$ -	\$ 1,800	\$ 1,800
21	Structures and improvements	4,875	46,879.5	34,440.5	86,195
22	Reactor plant equipment	55,866	22,416	6,311	85,593
23	Turbine plant equipment	42,447	12,962	3,004	58,413
24	Electric plant equipment	8,762	10,334	4,895	23,891
25	Miscellaneous plant equipment	5,860	3,415	509	9,284
26	Main condenser heat reject system	10,285	3,196	978	<u>14,458</u>
2	Total direct cost				<u>279,634</u>
91	Construction services	15,900	16,903	22,546	55,349
92	Home office engineering and service	58,680	-	-	58,680
93	Field office engineering and service	15,727	-	1,962	<u>17,689</u>
9	Total indirect cost				<u>131,718</u>
	Coolant cost (liquid-D ₂ O inventory) @ \$97/lb				<u>17,325</u>
	Total base cost				<u>\$428,677</u>
	Unit cost \$/kW	<u>\$428,677</u> 402.333		=	\$1,065.48/kW

Table E-3

Capital Cost Estimate Summary (Two-Digit Accounts)
 451-MW(e) Two-Phase H₂O-Cooled LAFER Balance of Plant
 (in thousands of 1978 dollars)

Account No.	Account description	Factory equipment cost	Site labor cost	Cost material cost	Total costs
20	Land and land rights	\$ -	\$ -	\$ 1,800	\$ 1,800
21	Structures and improvements	4,100	40,028.5	28,751.5	72,880
22	Reactor plant equipment	55,527	17,695	5,398	79,620
23	Turbine plant equipment	45,320	13,459	3,153	61,932
24	Electric plant equipment	8,762	10,334	4,895	23,891
25	Miscellaneous plant equipment	5,359	2,623	430	8,412
26	Main condenser heat reject system	10,285	3,196	978	<u>14,458</u>
2	Total direct cost				262,993
91	Construction services	14,460	15,826	22,141	52,427
92	Home office engineering and service	54,795	-	-	54,795
93	Field office engineering and service	15,727	-	1,962	<u>17,689</u>
9	Total indirect cost				<u>124,911</u>
	Coolant cost				-
	Total base cost				<u>\$387,904</u>
	Unit cost \$/kW	<u>\$387,904</u> <u>450.813</u>		=	\$860.45/kW

Table E-4

Capital cost Estimate Summary (Three-Digit Accounts)
 455-MW(c) Helium-Cooled LAFER Balance of Plant
 (in thousands of 1978 dollars)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
20	Land and land rights	\$	\$	\$ 1,800	\$ 1,800
211	Yardwork	138	3,684	3,758	7,580
212	Reactor containment building	1,985	13,676	9,613	25,274
213	Turbine room and heater bay	353	3,413	4,626	8,392
214	Security building	36	420	273	729
215	Reactor service and fuel storage	457	3,485	2,239	6,181
216	Radwaste building	-	1,914	1,086	3,000
218A	Control room/D-G building	535	3,943	2,039	6,517
218B	Administration and service building	366	1,644	1,299	3,309
218C	Helium storage area	-	346	102	448
218D	Fire pump house including foundations	18	135	85	238
218E	Emergency feed pump building	15	1,048	387	1,450
218F	Manway Tunnels (RCA tunnels)	2	388	163	553
218G	Electrical tunnels	3	4.5	2.5	10
218H	Nonessential switchgear building	9	155	114	278
218I	Access building	-	327	424	751
218J	Piping penetration building	6	1,391	948	2,345
218K	Reactor auxiliary building	-	803	1,712	2,515
218S	Holding pond	-	116	51	167
218T	Ultimate heat sink structure	31	2,479	1,305	3,815
218U	Ultimate heat sink tunnel	-	501	283	784
218V	Containment room emergency air intake structure	21	89	37	147
21	Structures and improvements	3,975	39,961.5	30,546.5	74,483

Table E-4 (Cont'd)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
220A	Nuclear steam supply system	\$ 35,425	\$ -	\$ -	\$ 35,425
220B	NSSS options	-	-	-	-
221	Reactor equipment	9	6,805	577	7,391
222A	Main heat tr. and transport system (Pb-Bi)	3,229	4,017	430	7,676
222B	Main heat tr. and transport system (helium)				
223	Safeguards system	2,243	3,263	493	5,999
224	Radwaste system	630	718	134	1,482
225	Fuel handling and storage	1,596	477	65	2,138
226	Other reactor plant equipment	5,324	2,703	1,030	9,057
227	Rx instrument and control	1,352	851	15	2,218
228	Reactor plant miscellaneous items	-	1,285	257	1,542
22	Reactor plant equipment	49,808	20,119	3,001	72,928
231	Turbine-generator	23,925	2,265	561	26,751
233	Condensing systems	4,447	1,709	453	6,609
234	Feed heating system	5,900	1,635	174	7,709
235	Other turbine plant equipment	4,037	3,811	386	8,234
236	Instrumentation and control	928	140	12	1,080
237	Turbine plant miscellaneous items	-	578	534	1,112
23	Turbine plant equipment	39,237	10,138	2,120	51,495
241	Switchgear	2,918	521	52	3,491
242	Station service equipment	4,730	855	147	5,732
243	Switchboards	501	139	66	706
244	Protective equipment	-	905	465	1,370
245	Electrical structure and wiring container	-	3,711	1,189	4,900
246	Power and control wiring	613	4,203	2,976	7,792
24	Electric plant equipment	8,762	10,334	4,895	23,991

Table E-4 (Cont'd)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
251	Transportation and lift equipment	\$ 1,373	\$ 260	\$ 83	\$ 1,716
252	Air, water, and steam service system	2,246	2,103	235	4,584
253	Communications equipment	685	203	108	996
254	Furnishings and fixtures	608	64	12	684
25	Miscellaneous plant equipment	4,912	2,630	438	7,980
261	Structures	79	989	640	1,708
262	Mechanical equipment	6,016	1,380	630	8,026
26	Main condensate heat reject system	6,095	2,369	1,270	9,734
2	Total direct costs				242,411
911	Temporary construction facilities	-	13,389	6,458	19,847
912	Construction tools and equipment	-	2,514	14,388	16,902
913	Payroll insurance and taxes	13,786	-	-	13,786
914	Permits, insurance, and local taxes	-	-	700	700
915	Transportation	-	-	-	-
91	Construction services	13,786	15,903	21,546	51,235
921	Home office services	59,284	-	-	59,284
922	Home office Q/A	3,183	-	-	3,183
923	Home office construction management	801	-	-	801
92	Home office engineering and services	63,268	-	-	63,268
931	Field office expenses	-	-	1,962	1,962
932	Field office supervision	12,031	-	-	12,031
933	Field QA/QC	2,336	-	-	2,336
934	Plant start-up and test	2,256	-	-	2,256
93	Field office engineering and services	16,623	-	1,962	18,585
9	Total indirect costs				133,088
	Total base cost				\$375,499
	Unit cost \$/kW	\$375,499 455		=	\$825.27/kW

Table E-5

Capital Cost Estimate Summary (Three-Digit Accounts)
 402-MW(e) Liquid-D₂O-Cooled LAFER Balance of Plant
 (in thousands of 1978 dollars)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
20	Land and land rights	\$ -	\$ -	\$ 1,800	\$ 1,800
211	Yardwork	138	3,684	3,758	7,580
212	Reactor containment building	2,480	21,704	14,228	38,412
213	Turbine room and heater bay	353	3,413	4,626	8,392
214	Security building	36	420	273	729
215	Reactor auxiliary building and tunnels	317	2,768	1,526	4,611
216	D ₂ O upgrading tower structure	90	859	931	1,880
217	Fuel storage building	179	1,419	850	2,448
218A	Control room/D-G building	843	5,477	2,203	8,523
218B	Administration and service building	366	1,644	1,299	3,309
218D	Fire pump house including foundations	18	135	85	238
218E	Emergency feed pump building	15	1,048	387	1,450
218F	Manway tunnels (RCA tunnels)	2	388	163	553
218G	Electrical tunnels	3	4.5	2.5	10
218H	Nonessential switchgear building	9	155	114	278
218J	Main steam and feedwater pipe enclosure	6	1,391	948	2,345
218K	Pipe tunnels	-	182	80	262
218P	Containment equipment hatch missile shield	-	94	29	123
218S	Holding pond	-	72	31	103
218T	Ultimate heat sink structure	20	1,911	2,860	4,791
218V	Control room emergency air intake structure	-	111	47	158
21	Structures and improvements	4,875	46,879.5	34,440.5	86,195

Table E-5 (Cont'd)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
220A	Nuclear steam supply system (NSSS)	\$ 34,425	\$ -	\$ -	\$ 35,425
220B	NSSS options	-	-	-	-
221	Reactor equipment	462	1,446	1,514	3,422
222A	Main heat tr and transport system (Pb-Bi)	5,127	10,111	1,306	16,544
222B	Main heat tr. and transport system (D ₂ O)				
223	Safeguards system	1,765	2,241	300	4,306
224	Radwaste system	3,843	1,289	455	5,587
225	Fuel handling and storage	1,390	799	64	2,253
226	Other reactor plant equipment	3,776	4,937	1,574	10,287
227	Rx instrument and control	4,261	593	50	4,904
228	Reactor plant miscellaneous items	817	1,000	1,048	2,865
22	Reactor plant equipment	55,886	22,416	6,311	85,593
231	Turbine-generator	26,052	2,268	512	28,832
233	Condensing systems	5,670	2,173	329	8,172
234	Feed heating system	5,760	3,628	414	9,802
235	Other turbine plant equipment	4,037	3,811	386	8,234
236	Instrumentation and control	923	140	12	1,080
237	Turbine plant miscellaneous items	-	942	1,351	2,293
23	Turbine plant equipment	42,447	12,962	3,004	58,413
241	Switchgear	2,918	521	52	3,491
242	Station service equipment	4,730	855	147	5,732
243	Switchboards	501	139	66	706
244	Protective equipment	-	905	465	1,370
245	Electrical structure and wiring container	-	3,711	1,189	4,900
246	Power and control wiring	613	4,203	2,976	7,692
24	Electric plant equipment	8,762	10,334	4,895	23,891

Table E-5 (Cont'd)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
251	Transportation and lift equipment	\$ 1,737	\$ 260	\$ 83	\$ 1,716
252	Air, water, and steam service system	2,246	2,103	235	4,584
253	Communications equipment	1,133	988	179	2,300
254	Furnishings and fixtures	608	64	12	684
25	Miscellaneous plant equipment	5,360	3,415	509	9,284
261	Structures	79	989	641	1,708
262	Mechanical equipment	10,206	2,207	337	12,750
26	Main condensate heat reject system	10,285	3,196	978	14,458
2	Total direct costs				279,634
911	Temporary construction facilities	-	13,389	6,458	19,847
912	Construction tools and equipment	-	3,514	15,388	18,902
913	Payroll insurance and taxes	15,900	-	-	15,900
914	Permits, insurance, and local taxes	-	-	700	700
91	Construction services	15,900	16,903	22,546	55,349
921	Home office services	55,470	-	-	55,470
922	Home office Q/A	2,409	-	-	2,409
923	Home office construction management	801	-	-	801
92	Home office engineering and services	58,680	-	-	58,680
931	Field office expenses	-	-	1,962	1,962
932	Field job supervision	12,031	-	-	12,031
933	Field QA/QC	1,881	-	-	1,881
934	Plant start-up and test	1,815	-	-	1,815
93	Field office engineering services	15,727	-	1,962	17,689
9	Total indirect costs				131,718
	Liquid D ₂ O inventory (initial operation only) (@ \$97/lb)	17,325	-	-	17,325
	Total base cost				\$428,677
	Unit cost \$/kW =	\$428,677 402.333			= \$1,065.48/kW

Table E-6

Capital Cost Summary (Three-Digit Accounts)
 451-MW(e) Two-Phase H₂O-Cooled LAFER Balance of Plant
 (in thousands of 1978 dollars)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
20	Land and land rights	\$ -	\$ -	\$ 1,800	\$ 1,800
211	Yardwork	138	3,684	3,758	7,580
212	Reactor containment building	1,985	13,676	9,613	25,274
213	Turbine room and heater bay	353	3,413	4,626	8,392
214	Security building	36	420	273	729
215	Primary auxiliary building and tunnels	317	2,768	1,526	4,611
216	Waste process building	118	3,616	1,951	5,685
217	Fuel storage building	179	1,419	850	2,448
218A	Control room/D-G building	535	3,943	2,039	6,517
218B	Administration and service building	366	1,644	1,299	3,309
218D	Fire pump house including foundations	18	135	85	238
218E	Start-up boiler building	15	1,048	387	1,450
218F	Manway tunnels (RCA tunnels)	2	388	163	553
218G	Electrical tunnels	3	4.5	2.5	10
218H	Nonessential switchgear building	9	155	114	278
218J	Main steam and feed-water pipe enclosure	6	1,391	948	2,345
218K	Pipe tunnels	-	182	80	262
218M	Hydrogen recombiner structure	-	49	38	87
218P	Containment equipment hatch missile shield	-	94	29	123
218S	Holding pond	-	72	31	103
218T	Ultimate heat sink structure	20	1,816	892	2,728
218V	Control room emergency air intake structure	-	111	47	158
21	Structures and improvements	<u>4,100</u>	<u>40,028.5</u>	<u>28,751.5</u>	<u>72,880</u>

Table E-6 (Cont'd)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
220A	Nuclear steam supply system (NSSS)	\$ 34,425	\$ -	\$ -	\$ 35,425
220B	NSSS options	-	-	-	-
221	Reactor equipment	259	624	1,061	1,944
222A	Main heat tr. and transport system (Pb-Bi)	3,229	4,017	430	7,676
222B	Main heat tr. and transport system (steam)	-	-	-	-
223	Safeguards system	2,243	3,263	493	5,999
224	Radwaste system	4,921	2,784	677	8,382
225	Fuel handling and storage	1,596	477	65	2,138
226	Other reactor plant equipment	3,776	4,937	1,574	10,287
227	Rx instrument and control	4,261	593	50	4,904
228	Reactor plant miscellaneous items	817	1,000	1,048	2,865
22	Reactor plant equipment	55,527	17,695	5,398	79,620
231	Turbine-generator	23,925	2,265	561	26,751
233	Condensing systems	5,670	2,173	329	8,172
234	Feed heating system	5,760	3,628	414	9,802
235	Other turbine plant equipment	4,037	3,811	386	8,234
236	Instrumentation and control	928	140	12	1,080
237	Turbine plant miscellaneous items	-	942	1,351	2,293
238	Start-up boiler	5,000	500	100	5,600
23	Turbine plant equipment	45,320	13,549	3,153	61,932
241	Switchgear	2,918	521	52	3,491
242	Station service equipment	4,730	855	147	5,732
243	Switchboards	501	139	66	706
244	Protective equipment	-	905	465	1,370
245	Electrical structure and wiring container	-	3,711	1,189	4,900
246	Power and control wiring	613	4,203	2,976	7,692
24	Electric plant equipment	8,762	10,334	4,895	23,891

Table E-6 (Cont'd)

Account No.	Account description	Factory equipment cost	Site labor cost	Site material cost	Total costs
251	Transportation and lift equipment	\$ 1,373	\$ 260	\$ 83	\$ 1,716
252	Air, water, and steam service system	2,246	2,103	235	4,584
253	Communications equipment	1,132	196	100	1,428
254	Furnishings and fixtures	608	64	12	684
25	Miscellaneous plant equipment	5,359	2,623	430	8,412
261	Structures	79	989	641	1,708
262	Mechanical equipment	10,206	2,207	337	12,750
26	Main condensate heat reject system	10,285	3,196	978	14,458
2	Total direct costs				262,993
911	Temporary construction facilities	-	13,389	6,458	19,847
912	Construction tools and equipment	-	2,437	15,023	17,460
913	Payroll insurance and taxes	14,460	-	-	14,460
914	Permits, insurance, and local taxes	-	-	660	660
915	Transportation	-	-	-	-
91	Construction services	14,460	15,826	22,141	52,427
921	Home office services	51,747	-	-	51,747
922	Home office Q/A	2,247	-	-	2,247
923	Home office construction management	801	-	-	801
92	Home office engineering and services	54,795	-	-	54,795
931	Field office expenses	-	-	1,962	1,962
932	Field job supervision	12,031	-	-	12,031
933	Field QA/QC	1,881	-	-	1,881
934	Plant start-up and test	1,815	-	-	1,815
93	Field office engineering and services	15,727	-	1,962	17,689
9	Total indirect costs				124,911
	Total base cost				\$387,904
	Unit cost (\$/kW)	\$387,904 450.813		=	\$860.45 kW