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LINEAR ACCELERATOR REACTORS

YEAR END REPORT FY 1977

NON-PROLIFERATION
ALTERNATIVE
SYSTEMS
ASSESSMENT
PROGRAM

DEPARTMENT OF NUCLEAR ENERGY

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

UNDER CONTRACT NO. EY-76-C-02-0016 WITH THE

UNITED STATES DEPARTMENT OF ENERGY

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LINEAR ACCELERATOR REACTORS
(LARs)

Non-Proliferation Alternative Systems Assessment Program
(NASAP)

Year End Report FY 1977
to September 30, 1977

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Printed in the United States of America

Available from

National Technical Information Service

U.S. Department of Commerce

5285 Port Royal Road

Springfield, VA 22161

Price: Printed Copy \$6.00; Microfiche \$3.00

January 1978

275 copies

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LINEAR ACCELERATOR REACTORS
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Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

I. Introduction

Under the Nuclear Alternative Systems Assessment Program (NASAP), Brookhaven National Laboratory has initiated a study of Linear Accelerator Assisted Reactors to assess their potential and feasibility in a nuclear energy scenario which will minimize the risk of weapons proliferation. This report covers the period from the inception of the project to the end of FY 1977 (July-Sept. 1977).

Earlier work ⁽¹⁻³⁾ has shown that a Linear Accelerator Assisted reactor, fed with fissile fuel bred from fertile material with the help of a linear accelerator, has in principle the potential for:

- a) increasing the amount of energy which can be extracted from natural uranium by a substantial factor.
- b) improving the intrinsic safety of the reactor in the case of the driven, subcritical system.
- c) maintaining a relatively low risk of weapons proliferation by foregoing the need for chemical reprocessing of bred fuel and even possibly the need for additional enrichment facilities.

The study is directed at two basic systems which do not require reprocessing:

1) Linear Accelerator Driven Reactors (LADR).

This system involves the use of a high energy proton or deuteron beam from a linear accelerator, impinging on a target consisting of a lattice of metal clad rods of fertile material in water and/or heavy water, with characteristics maximizing the time-integrated production of thermal energy for conversion to power. The fertile elements of interest include metal clad enriched UO_2 , 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. This system is referred to as the Linear Accelerator Driven Reactor (LADR) because the reactor target would always be operated in a strongly subcritical mode, and the generation of power would be the result of amplifying the initial neutron source generated by the beam, by subcritical multiplication. There are two promising configurations here; one uses zircaloy clad uranium oxide elements and the other thorium metal. Light water is used as a coolant. The use of thorium metal is particularly attractive as the fertile material irradiated by the accelerator because of the following:

- (a) Thorium metal has a high thermal conductivity and melting point, and appears to have very good radiation resistance;
- (b) the neutronics of the Th-232/U-233 fuel cycle are favorable, and in some designs would require a less intense beam of particles from the accelerator; and
- (c) thorium has a high natural abundance and reduction of metal from the ore is accomplished using well-known existing technology. The concentration of fissile material in the target fuel elements would not exceed several percent. The lattice parameters and coolant

arrangement would be chosen so as to maintain subcriticality. The elements would not be removed until a substantial fraction of the original fertile material is burned, after which they would be removed from the target reactor for permanent storage in their original solid form without further processing.

2) Linear Accelerator Regenerative Reactor (LARR).

In this system the Linear Accelerator beam interacts with a target to produce a high flux of neutrons. The target would consist of, or be surrounded by reactor fuel elements which in capture of neutrons from the beam-target interaction would in time come to contain the required enrichment of Pu 239 or U 233 depending on the scenario chosen. These fuel elements would then be burned in separate thermal reactors, without intervening reprocessing. This system is referred to as the Linear Accelerator Regenerative Reactor (LARR).

Regenerative systems optimize fuel breeding rate rather than power production rate. It is conceivable, however, that the target thermal gain of a LARR might be sufficient for power production as well. Alternative target designs for the LARR system would include LWR elements with liquid metal and water coolants. Thorium metal will also be evaluated as the fertile material for the LARR system.

The objective of the program is to conduct the basic physics and engineering studies necessary to establish feasibility, to evaluate possible performance and economics of conceptual systems and to identify the problems attached to each that require solutions before the concept could become reality.

II. Spallation Neutron Source and Reactor Physics

Up to now, the major effort has been directed at a parametric survey of reactor lattices and coolant systems to determine which of the many options being investigated show the most promise. The figures of merit chosen are the multiplication factor (k) and the conversion ratio (CR = fissile fuel produced/fissile fuel destroyed). Both of these factors have a strong influence on the economical desirability of the system.

2.1. Power multiplication in the linear accelerator driven

The electric power multiplication in the linear accelerator driven reactor can be expressed by

$$P_{TA} = \epsilon_a \epsilon_p (y_n 0.2 \frac{k}{U} \frac{1}{1-k} + 1.0)$$

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

k is the multiplication factor of the driven reactor,

U is the number of neutrons per fission,

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.

Using typical values for these parameters of $y_n = 50$, $k = 0.9$, $U = 2.5$, $\epsilon_p = 0.35$, and $\epsilon_a = 0.5$, P_{TA} becomes 8.05. Therefore, for obtaining reasonably high power ratios, the values of y_n and k should be sufficiently high (~ 0.9) when dealing with the driven reactor concept. The value of k varies as a function of fuel burnup so that not only is

the initial k value at the beginning of burnup important, but also the history of the k value as a function of burnup should be studied. In this report, a conventional PWR-type reactor and a tight packed heavy water PWR-type reactor were analyzed as target assemblies. The analyses were carried out by varying the parameters of volume ratio $V_{\text{Mod}}/V_{\text{Fuel}}$, density of moderator and initial fuel enrichment. The types of reactor targets analyzed are tabulated in Table II-1. The calculation was performed by the EPRI-CPM code which uses the collision probability method for the thermal neutron transport and the THERMOS-type calculational method for the thermal neutron spectrum calculation. The infinite multiplication factor, k_{∞} , the conversion ratio, CR, and the concentrations of U^{235} , U^{238} , Pu^{239} , Pu^{240} , Pu^{241} , and Pu^{242} are computed as a function of burnup (exposure) from 0 to 150 GWD/ton of fuel. The details of each reactor are discussed in the following section.

2.2 Light water reactor with natural uranium

In order to study the fuel cycle of the light water driven and regenerative reactor with a neutron source produced by a linear accelerator, the light water UO_2 infinite lattice system with various volume ratios of V_{H_2O}/V_{UO_2} and various H_2O densities are calculated by using the burnup code EPRI-CPM. The specification of the light water lattice system is shown in Table II-2; these parameters conform to the present PWR reactor specifications except for fuel enrichment. In this case, we started from natural uranium instead of the current commercial fuel enrichments of 2.25, 2.8, and 3.3%, a point which is discussed later.

Figure 2.2.1 shows the infinite multiplication factor k_{∞} as a function of burnup. The case of B1 (2.0) 0.7 (where the value in

parenthesis (2.0) is the volume ratio of V_{H_2O}/V_{UO_2} and the value after the parenthesis (0.7) is the density of H_2O , and Bl is the specification of fuel and clad size) shows that k_{∞} starts from 0.86 and reaches almost 0.9 at 2.5 GWD/ton burnup. After this burnup k_{∞} gradually decreases and becomes 0.68 at 150 GWD/ton.

By decreasing V_{H_2O}/V_{UO_2} , the initial k_{∞} values becomes smaller, but k_{∞} at 150 GWD/ton burnup increases. For the case of Bl(0.5) 0.175 tight lattice, k_{∞} starts at 0.47 and reaches 0.94 after 150 GWD/ton. The case of Bl(0.5) 0.7 shows a rather flat k_{∞} varying from 0.75 to 0.80.

Figure 2.2.2 shows the conversion ratios (CR). The highest conversion ratio is 3.8 for Bl(0.5) 0.175 tight lattice, the CR decreases rapidly to 1.5 at 30 GWD/ton and approaches 1 after long burnup. The CR in the loose lattice of Bl(2.0) 0.7 starts at 1.0, then after a small dip to 0.9 at 3 GWD/ton burnup, increases to 1.

Figure 2.2.3 shows the change of U^{235} concentration as a function of burnup. It starts from 0.71% for the natural uranium concentration and rapidly decreases as burnup increases. The U^{235} concentration in a reactor with a large V_{H_2O}/V_{UO_2} decreases more rapidly than the one with a small volume ratio, because of the softer spectrum for the large V_{H_2O}/V_{UO_2} designs.

Figure 2.2.4 shows the U^{238} concentration which decreases almost linearly with burnup. The U^{238} concentration for the tight lattice decreases even more rapidly because of the larger conversion ratio in the tight lattice.

Figure 2.2.5 shows the Pu^{239} concentration buildup. In the case of B1 (2.0) 0.7, the Pu^{239} concentration increases to 0.4% at 30 GWD/ton and then remains constant at a value of 0.4% until 150 GWD/ton burnup. The Pu^{239} concentration of B1 (1.0) 0.7 case is always larger than the case of B1 (2.0) 0.7 and reaches 1.3% after 150 GWD/ton. In the cases of B1 (0.5) 0.7, 0.35, and 0.175, larger concentration values of 3.7, 5.3, and 6.3% respectively are reached after 150 GWD/ton burnup.

Figures 2.2.6, 2.2.7, and 2.2.8 show the Pu^{240} , Pu^{241} , and Pu^{242} concentrations. The concentration of Pu^{240} indicates a similar tendency to that of Pu^{239} . However, in the case of B1 (0.5) 0.7, the concentration of Pu^{241} at 150 GWD/ton shows larger values than the tighter lattices of B1 (0.5) 0.35 and 0.175, whereas the concentration of Pu^{242} in the tight lattice is smaller than that of the open lattice.

2.3. Light water reactor with U^{235} enrichment

As shown earlier, the initial k_{∞} for natural uranium with the possible exceptions of the open lattice B1 (2.0) 0.7 is small so that a high intensity external neutron source (linear accelerator) is required. By increasing the initial enrichment of U^{235} , the k_{∞} value in the early burnup period can be increased. To study the behavior of the initial enrichment as a function of the several parameters such as, U and Pu concentration, the calculation for the B1 series water lattices with U^{235} enrichments of 2.25, 2.8, and 3.3 percents have been carried out. These enrichments of U^{235} are the same as those

used in present day PWR reactors. Figure 2.3.1 shows k_{∞} as a function of burnup; the loose lattices of $V_{H_2O}/V_{H_2O} = 2.0$ and 1.0 give initial k_{∞} values > 1.0 .

By increasing the initial enrichment, the k_{∞} value increases until a 50 GWD/ton burnup is reached. However, after this burnup time, the effect of the initial enrichment on the k_{∞} value decreases. The effect of initial enrichment disappears after 90-100 GWD/ton, and the asymptotic value of k_{∞} depends only on the volume ratio (V_{H_2O}/V_{UO_2}).

For the open lattice case, k_{∞} decreases sharply during the first 40 GWD/ton burnup. This decrease is due mostly to the high burnup of the U^{235} isotope. For existing PWR reactors with V_{H_2O}/V_{UO_2} of about 1.5, k_{∞} becomes less than 1.0 after about 20 GWD/ton. By shuffling the fuel elements, burnups of 30 GWD/ton can be achieved.

For the tight lattice of $V_{H_2O}/V_{UO_2} = 0.5$, $\rho = 0.175$, k_{∞} increases from 0.47 for natural uranium to 0.78 for 3.3% enriched fuel. If we use an initial enrichment of 8-9% in this tight lattice, k_{∞} can be maintained constant at about 0.95 over the long burnup period of 150 GWD/ton.

Figure 2.3.2 shows the conversion ratio (CR) for these lattices. The CR of the tightest lattice (e) with 2.25% enrichment starts from 1.6 and gradually decreases to the final value of 1.0. On the other hand, contrary to the tight lattice, the CR of the open lattice (a) with enrichment of 3.3% starts from 0.4 and gradually increases to 1.

Figures 2.3.3 and 2.3.4 show the U^{235} and U^{238} concentration. The concentration of U^{238} in the open lattice decreases more rapidly

than that of the tight lattice, because the CR in the tight lattice is higher. Figure 2.3.5 shows the effect of the enrichment on the Pu^{239} concentration. It indicates that the Pu^{239} concentration is almost entirely determined by the volume ratio of $V_{\text{H}_2\text{O}}/V_{\text{UO}_2}$. Figures 2.3.6 and 2.3.7 show the concentrations of Pu^{240} and Pu^{241} . These are also rather independent of the initial enrichment of U^{235} .

2.4 Heavy water tight lattices.

As shown in the light water UO_2 lattice cases, the tight light water UO_2 lattice shows higher conversion ratios and k_∞ at 150 GWD/ton burnup than that of the open lattices. This is due to the hardening of the neutron spectrum. To achieve long fuel burnups, the tight lattice is preferable. However, heat removal from the tight lattice is more difficult. From this point of view, a D_2O moderator, which has less moderating influence than H_2O , is more advantageous and thus may be preferable.

In this section, the D_2O moderated UO_2 lattices which have the same geometry and isotope compositions of fuel and clad as the H_2O moderated UO_2 lattice, are analyzed. Figure 2.4.1 shows k_∞ for the case when the initial uranium enrichment is 0.71% (natural uranium). As to be expected, the initial k_∞ is very small. Therefore, the intensity of the linac neutron source has to be greater than for the corresponding light water case. In order to increase the initial k_∞ , the U^{235} concentration in the fuel should be increased. Figure 2.4.2 shows the change of k_∞ for the B1 series reactors with initial enrichments of U^{235} of 3.3, 5.0, and 7.0%. By increasing the initial enrichment, k_∞ increases during the early burnup time. After 100 GWD/ton, k_∞

approaches an asymptotic value which depends only on V_{D_2O}/V_{UO_2} and not on the initial enrichment. The asymptotic k_{∞} value at 150 GWD/ton burnup for case a (2.0) 0.7, b (1.0) 0.7, c (0.5) 0.7, d (0.5) 0.35, and e (0.5) 0.175, are respectively 0.76, 0.84, 0.92, 1.0, and 1.06. These values are about 10 to 12% higher than the corresponding H_2O lattice cases.

One should note that with slightly higher enrichment, the cases described could have their k_{∞} values of 0.76, 0.84, 0.92, 1.0 remain almost constant during the entire burnup time, up to 150 GWD/ton.

Figure 2.4.3 shows the conversion ratio (CR) as a function of burnup. The CR of a tight lattice (3.3% enrichment) starts at 2.55 and approaches 1.0 after 100 GWD/ton. The CR for an open lattice (7% enrichment) starts at 0.83 and increases to 1 after 10 GWD/ton.

Figures 2.4.2 and 2.4.3 indicate that in order to keep k_{∞} constant during long burnup times (from 0 to 150 GWD/ton), the conversion ratio should be kept very close to 1 during this entire period. Figures 2.4.4 and 2.4.5 show the U^{235} and U^{238} concentrations. The concentration change of U^{238} is rather independent on the volume ratio and initial enrichment.

Figure 2.4.6 shows the Pu^{239} concentration, which is more sensitive to the initial enrichment of U^{235} than in the light water reactor. However, after long burnup times, the concentration tends to converge to a value independent of the initial enrichment. The Pu^{239} concentration for a, b, c, d, and e lattices at 150 GWD/ton are 5.9, 6.6, 7.4, 7.9, and 7.9% respectively. These values are higher

than for the light water reactor cases.

Figures 2.4.7, 2.4.8, and 2.4.9 show the concentrations of Pu^{240} , Pu^{241} , and Pu^{242} which like that of Pu^{239} , depend on the initial enrichment of U^{235} .

From the large number of burnup calculations for the several reactor types considered, we can make the following comments. The tight lattice, especially the D_2O tight lattice, gives high k_∞ increases to ≥ 1 . Initial U-235 enrichment contributes to increasing the k_∞ mainly at early burnup times. The asymptotic value of k_∞ at high burnup times is almost independent of the initial enrichment and depends only on the volume ratio of moderator to fuel. These trends are very important for evaluating the design criteria of the Linac driven and regenerative reactors. From the neutronics point of view, the tight D_2O lattice with slight initial enrichment fuel is preferable, but the heat removal may be difficult. Other mechanical characteristics of the target and the clad materials need to be considered in parallel with the neutronic calculations.

Table II-1

Reactor Types Analyzed in Burnup Calculation

1. H_2O - UO_2 Zr-Clad Reactor

Enrichment: Natural uranium, 2.25%, 2.8%, and 3.3%

Lattice specification is the same as in a typical

PWR reactor

2. D_2O - UO_2 Zr-Clad Reactor

Enrichment: Natural uranium, 3.3%, 5%, and 7%

Lattice specification is the same as Case 1.

Table II-2

PWR Type Reactor Specifications

Pellet radius	0.18345"	UO_2 fuel
Zr-clad inner radius	0.1867"	
Zr-clad outer radius	0.2110"	
Power	400 W/cm fuel element	
Initial enrichment of fuel	Natural uranium 0.71% U^{235}	

LINAC REACTOR LWR BI

Figure 2.2.1

LINAC REACTOR LWR BI

Figure 2.2.2

LINAC REACTOR LWR BI

Figure 2.2.3

LINAC REACTOR LWR BI

Figure 2.2.4

LINAC REACTOR LWR BI

Figure 2.2.5

LINAC REACTOR LWR BI

Figure 2.2.6

LINAC REACTOR LWR BI

Figure 2.2.7

LINAC REACTOR LWR BI

Figure 2.2.8

LINAC REACTOR LWR BI

Figure 2.3.1

LINAC REACTOR LWR BI

Figure 2.3.2

LINAC REACTOR LWR BI

Figure 2.3.3

LINAC REACTOR LWR BI

Figure 2.3.4

LINAC REACTOR LWR BI

Figure 2.3.5

LINAC REACTOR LWR BI

Figure 2.3.6

LINAC REACTOR LWR BI

Figure 2.3.7

LINAC REACTOR LWR BI

Figure 2.3.8

LINAC REACTOR HWR BI NATURAL URANIUM

Figure 2.4.1

LINAC REACTOR HWR BI

Figure 2.4.2

LINAC REACTOR HWR BI

Figure 2.4.3

LINAC REACTOR HWR BI

Figure 2.4.4

LINAC REACTOR HWR BI

Figure 2.4.5

LINAC REACTOR HWR BI

Figure 2.4.6

LINAC REACTOR HWR BI

Figure 2.4.7

LINAC REACTOR HWR BI

Figure 2.4.8

LINAC REACTOR HWR BI

Figure 2.4.9

2.5 Neutron Yield

Yield of high energy neutrons produced by the proton beam stopping on a target are obtained with the NMTC code ⁽⁵⁾. The nucleon meson transport code (NMTC) computes the transport in matter of nucleons and mesons up to 3.5 GeV. The calculation is based on the intranuclear cascade evaporation model ⁽⁶⁾ which takes into account both elastic and inelastic scattering. These processes are computed statistically by the Monte Carlo method.

The NMTC code (with standard subroutine called NEL=2) is only valid for "heavy" nuclei where $A \geq 5$. It can also treat the hydrogen case ($A = 1$) with a special subroutine called NEL=1. Calculations for nuclei with masses $5 > A > 1$ give erroneous results. Therefore, the numbers quoted below for the D_2O case cannot be trusted.

In order to test the NMTC code for various cases and to obtain a rough estimate of the neutron yield from various target compositions, infinite media targets of O_2 , H_2O , D_2O , Pb, $U^{238}O_2$, UO_2-D_2O , and UO_2-H_2O were studied. In these calculations, it was assumed that 1 GeV protons are injected into the center of an infinite medium. The yield of high energy neutrons (which is the number of neutrons slowed down to < 15 MeV energy) was calculated by using 10 batches each with 25 incident protons.

The number of neutrons produced by each proton in a batch are shown in the tables together with their value averaged over the 10 batches. These yields for O_2 , H_2O , D_2O , and UO_2-D_2O medium are shown in Table II-3; the values for Pb and UO_2 are given in Table II-4. The neutron yield from the O_2 system is 5.58, which is about 1/7 of the neutron

yield from Pb. In the case of H_2O , the calculation using the NMTC code with the standard subroutine NEL=2 gives a neutron yield of 0.39. When the NEL=1 option is used in which hydrogen nuclei are treated as protons, the neutron yield becomes 4.10.

The NMTC code cannot treat deuterium nuclei in the same way as hydrogen nuclei, and only the NEL=2 option can be used. The neutron yield calculated from this option gives 0.364 which is unreasonably small as in the case of NEL=2 calculation for H_2O . The neutron yield from a $UO_2 - D_2O$ system (where the isotope composition is the same as the one for the B1-series reactor of $V_{D_2O}/V_{UO_2} = 0.5$ in the burnup calculation) gives 33.82 using the NEL=2 option for deuterium nuclei. This number must be somewhat smaller than the real value, because the neutron yield calculated in NEL=2 option is smaller than the one calculated with the NEL=1 option in H_2O system. Also, the additional contribution to neutron yield due to $D(n,2n)p$ reaction is not negligible.

As mentioned earlier the present version of NMTC codes cannot treat the light nucleus where $A < 5$ except for the case of hydrogen. Therefore, modifications using a reasonable nuclear model theory for the light nuclei reactions are necessary. However, nuclear model theory for the light nuclei reactions is not very well understood. Therefore, we have to substitute hydrogen for the light nuclei and consequently the correction resulting from this substitution must be evaluated from the experimental data available for the light nucleus reaction. This especially applies to deuterium where the nucleus binding energy is small. The resulting cross sections for $D(n,2n)p$ and $D(p,n,p)p$

reactions are therefore large and these reactions are likely to be a major contribution to the neutron yield. These processes should therefore be carefully studied.

Table II-4 shows the results for the infinite Pb and $U^{238}O_2$ target cases. The averaged neutron yields are 35.19 and 35.08, respectively. For UO_2 , the total neutron yield from UO_2 should be increased about twice this value due to the fission process in the energy range from 15 MeV to 1 MeV. The neutron yield in UO_2 is penalized by the presence of O_2 and thus shows very little difference from Pb above 15 MeV.

Table II-5 shows the neutron yield in the UO_2-H_2O system, with the geometry of the B1 series lattice where $V_{H_2O}/V_{UO_2} = 2.0, 1.0, 0.5$, and $\rho_{H_2O} = 0.7, 0.35, 0.175$, in the burnup calculation. The averaged neutron yields are between 28 and 31. Since the number of sampling is small, the statistical error is rather large, so that a detail comparison among these cannot be made, however it can be observed that the neutron yield in the tight lattice is not much larger than that of the open lattice.

These values suggest that the total neutron yield including neutrons produced by fission and other processes in the energy range of less than 15 MeV may be somewhat larger than the value of 50 which has been used up to now in the evaluation of linear accelerator driven and regenerative reactors.

It is to be noted here that all the neutron yield calculations carried out so far have not considered real geometries where leakages and neutron absorbent materials will be important elements in the final multiplication factor. However, in so far as possible these calculations will be correlated with known experimental data and corrections will be introduced to account for these detrimental effects.

Table II-3

Neutron Yield

Batch	O ₂	H ₂ O NEL=1	H ₂ O NEL=2	NEL=2 D ₂ O	UO ₂ D ₂ O NEL=2 .5
1	4.72	2.76	0.44	0.28	41.52
2	5.40	4.80	1.04	0.56	34.80
3	6.00	4.88	0.00	0.08	31.32
4	6.52	3.84	0.08	0.28	36.12
5	6.24	4.56	0.28	0.32	34.52
6	5.28	4.04	0.44	0.92	34.28
7	5.32	3.44	0.24	0.20	33.56
8	6.44	4.36	0.48	0.44	37.68
9	4.28	3.56	0.56	0.16	26.00
10	5.56	4.80	0.36	0.40	28.40
Average Y =	5.576	4.104	0.392	0.364	33.82

NEL=1 special treatment for H.

NEL=2 H is treated in the same way as heavy nuclei.

D " " " " " "

Table II-4

Neutron Yield

<u>Batch</u>	<u>Pb</u>	<u>UO₂</u>
1	37.84	35.24
2	34.32	32.04
3	36.16	34.08
4	33.60	38.80
5	37.96	36.12
6	30.36	37.52
7	30.44	34.84
8	38.76	38.68
9	34.00	29.80
10	38.48	33.68
Average Y =		35.080

Infinite medium, 1 GeV proton injected into center
lowest energy 15 MeV.

Table II-5

Neutron Yield

<u>Batch</u>	<u>(2).7</u>	<u>(1).7</u>	<u>(.5).7</u>	<u>(.5).35</u>	<u>(.5).175</u>
1	28.00	31.80	31.84	29.92	36.24
2	21.84	27.68	23.60	33.52	35.96
3	34.44	28.92	28.48	27.40	29.84
4	29.04	30.40	21.52	27.64	28.76
5	29.16	25.84	33.48	31.16	30.56
6	21.24	29.64	26.84	38.72	26.64
7	31.04	24.76	30.80	28.28	30.72
8	24.64	27.76	28.16	36.24	30.00
9	30.52	28.80	27.16	27.48	34.28
10	26.40	36.16	32.32	27.24	26.32
Average Y =	27.632	29.176	28.420	29.760	30.932

Natural Uranium-Water

Infinite Medium

1 GeV proton injected into center

III. Linear Accelerator Study

After technical feasibility, the next important question to be answered is whether the Linac assisted reactor or regenerative system makes any economic sense.

To this effect, a preliminary study was undertaken to determine the cost and power requirements for linear accelerators in the range of currents and energies of interest. 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 3.1 and 3.2 for proton Linacs, and Figures 3.3 and 3.4 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 3.5 shows this variation for the typical case of a 300 mA proton Linac. However, the AC input power to the Linac is more strongly influenced by the accelerating gradient as shown in Figures 3.2 and 3.4. No attempt was made here to weigh the capital cost factor against the input power required for any given machine.

The following give the parametric basis used for the exercise:

1. The capital cost given here are 1977 dollars and do not include: depreciation, escalation, E.D.I. and A, nor the cost of real estate.

2. The accelerator structure shunt impedances used to calculate the excitation power has 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 frequently changes.

3. The accelerating structure frequency choices have been made on the basis of earlier calculations on matched input beams from parameters for high current Linacs. The initial frequency is used up to $\beta = 0.5$ (≈ 150 MeV) protons and then it is increased by a factor of three.

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

Unit Cost

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

a. Accelerator: This cost includes the Linac structure, focusing system, vacuum, beam diagnostic, radiation monitors, services distribution, etc. The unit cost used is \$55,000/m for a 300 MHz structure. This cost is adjusted for frequency change from \$26,000 at 1350 MHz to \$134,000 at 50 MHz.

b. Radio Frequency: This cost includes the entire system including input power transformers and rectifiers, amplifier chains, controls, transmission lines, etc. The unit cost used here is \$650,000/MWrf for the low energy part of the accelerator (to 150 MeV protons) and \$350,000/MWrf thereafter. This difference is predicated upon the use of triode or tetrode amplifiers at low frequency

and the use of klystrons at the higher frequencies.

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 consisting of a 2 ft. thick reinforced concrete box-like structure supporting 5 m earth shield was designed. In addition a retaining wall 8 m high, 5 m distance 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 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 (more than 80%) of the building. The structure considered here consists of standard construction, light structure, without crane coverage and no A/C. We have estimated the cost of this building to be \$80/ft² and with a requirement of 500 ft²/MWrf we have used \$40,000 MWrf as the unit cost of this structure.

Conclusions

It is interesting to note that the major portion of the capital cost is rf power. Typically, for a typical 1 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 effort to make the accelerator more viable has to be directed to the rf system where any improvement will be translated into major savings in both capital expenditure and input power requirements.

Figure 3.1

Figure 3.2

Figure 3.3.

Figure 3.4

Figure 3.5

IV. Critical Parameters of Linear Accelerator Reactors

4.1 Introduction

Linear accelerators could be used to produce fissile fuel in two modes, either with fuel reprocessing or without fuel reprocessing. With fuel reprocessing, the fissile material would be separated from the target and refabricated into a fuel element for use in a burner power reactor. Without reprocessing, the fissile material would be produced in-situ, either in a fresh fuel element or in a depleted or burned element after use in a power reactor. In the latter mode the fissile material would be increased in concentration for reuse in a power reactor. We call this system a Linear Accelerator Regenerative Reactor (LARR).

The linear accelerator breeder (LAB) can also be conceived of operating in a power production mode in which the spallation neutrons would be used to drive a subcritical assembly to produce power. We call this a Linear Accelerator Driven Reactor (LADR).

The Carter Administration's recent nuclear non-proliferation policy has recommended that plans for nuclear fuel reprocessing be stopped for the time being, and that development of the Clinch River Breeder Reactor be halted. A review is under way to determine the technical options available for reducing the opportunities for proliferation of nuclear weapons capability throughout the world. Among the options being explored are the following:

1. eliminate reprocessing
2. limit enrichment

3. limit transport of fuel
4. maintain low concentration of fissile material in power reactor fuel
5. approach a throwaway fuel cycle

These measures could also act in the direction of reducing the total energy available from uranium resources. A number of alternatives are being explored to solve this problem, including the following:

- a. find additional natural uranium resources
- b. reduce enrichment plant tail concentrations
- c. use thorium as a nuclear fuel
- d. use new fuel cycles

Both the Linear Accelerator Driven Reactor (LADR) and the Linear Accelerator Regenerative Reactor (LARR) appear to offer a high potential for extending the nuclear fuel supply while at the same time limiting the risk of misuse of the nuclear power fuel cycle for weapons production.

4.2 Principles of the Linear Accelerator Driven Reactor (LADR)

The LADR consists of a subcritical reactor target containing depleted uranium, natural uranium, slightly enriched uranium, or thorium, producing a net amount of energy when supplied with neutrons generated by stopping a high energy proton or deuteron beam from a linear accelerator. A critical economic parameter is the amount of electrical power needed to drive the accelerator, relative to the power output from the reactor target. The expression for the ratio of power output from the target to power input to the

accelerator (P_{TA}) is given in Table IV-1. Reasonable values have been chosen for the spallation neutron yield (50 neutrons per proton on a uranium target), power cycle efficiency and accelerator efficiency (35% and 50%, respectively), and the value of neutrons per fission from uranium (3 neutrons/fast fission). Based on these assumptions, the power ratio (P_{TA}) was calculated as a function of the multiplication factor (k). The results are given in Table IV-1. As an economic rule of thumb, the power consumption of the accelerator should not be more than about 20% of the net electrical output of the target. At values of k below about 0.9 the power required to be circulated back to the accelerator becomes a significant fraction of the total power output of the target reactor. For example, at $k = 0.8$, the power consumption of the accelerator would be 40% ($100/2.5$) of the electrical output of the target, and at $k = 0.6$, the accelerator would consume about 95% ($1/1.05$) of the target reactor power. It appears that the design of the target assembly must be such as to yield a value of k of 0.9 or greater.

Neutronic burnup calculations as indicated in Section II have been made for target assemblies consisting of fuel elements similar to those used in LWR's. A unit cell of zircaloy-clad UO_2 elements in light water coolant was used with a volume ratio of 2 parts water to 1 part fuel. The calculated infinite multiplication factor (k_∞) is shown in Figure 4.1 as a function of the burnup in GWD/ton (1000 MWD/ton). If the fuel is initially depleted UO_2 (i.e., tails from enrichment plants), the calculated value of k_∞ rises from a very low

value to a peak value of a little above 0.8, and then slowly falls to a value of about 0.7 after a burnup of over 100 GWD/ton. At the beginning of a cycle, starting from all fresh fuel, there would have to be a relatively larger energy input and circulating power investment before higher values of k_{∞} are reached. However, at the peak, the circulating power would still amount to over 40% of the target power. Closer lattice spacing (lower water/fuel ratio) will increase k_{∞} and improve performance somewhat.

The situation is improved if the fraction of fissile U^{235} initially in the U^{238} increases. With natural uranium (0.7% U^{235}) the k_{∞} is initially about 0.6, but it still does not rise much above 0.8 on longer burnups, so that the circulating power would still be relatively large.

Heavy water cooling or liquid metal sodium cooling would help to increase the value of k_{∞} for both depleted, natural uranium and enriched uranium fueled assemblies. With heavy water the k_{∞} would reach values of 0.9 or more on burnup to over 100 GWD/ton. However, the use of heavy water would be a departure from present day US commercial nuclear technology.

On the other hand, the capability of extending the burnup of fissile fuel to between 100 and 150,000 MWD/ton of fuel by means of linear accelerator reactors, means that the nuclear fuel resource is increased by 20 to 30 fold since at present the burnup power reactor fuel cycle utilizes only 0.5% of the uranium supply. Thus there is a large incentive to thoroughly investigate the LAR's.

Another interesting scenario whose implications are being explored is the following:

a. Fuel is initially burned in a conventional LWR with a conversion ratio of 0.6, to the usual burnup of about 30,000 MWD/ton.

b. The elements, which now contain about 1.8% fissile material (Pu^{239} and U^{235}) are placed in the reactor target of an LWR cooled LADR, and burned to an average value of k_{∞} of about 0.9. With fuel shuffling in the LADR, additional burnup by an average of 30,000 MWD/ton would lead to a total burnup of an additional 60,000 MWD/ton of fuel removed from the LWR. This would treble the energy available from uranium, compared to a present throwaway LWR cycle.

Additional neutronic and design calculations are being conducted on the LADR approach. The concept seems to satisfy the objectives of restraining proliferation and stretching the fuel reserves.

4.3 Principles of the Linear Accelerator Regenerative Reactor (LARR) Concept

The LARR is fundamentally a fuel producer, as opposed to the LADR which is primarily a power producer. The LARR is used to irradiate a reactor fuel to the level of reactivity required for burnup in a normal power reactor. The LARR could be used to irradiate depleted, natural, or enriched uranium fuel, as well as thorium fuel.

Table IV-2 gives the basic equations for the rate of fuel production in the target and the power which can be supported with this fuel. The fuel production is a direct function of the accelerator current and neutron yield per GeV proton or deuteron, and is an

indirect function of the conversion ratios and multiplication factors. For the limiting case of starting with depleted U^{238} , the relative capacity of power reactors which can be supported by the accelerator target reactor becomes a strong function of the power reactor conversion ratio. For a conventional LWR, the conversion ratio is 0.6 and therefore an accelerator capacity of about 22% of the LWR capacity would be required to regenerate the fuel. It is desirable to produce a hard neutron spectrum in the regenerator to maximize the fuel production and therefore either light or heavy water steam coolant in a tight lattice is preferred. If the regenerator supplies fuel to a heavy water cooled power reactor, such as the Canadian pressure tube CANDU reactor, the conversion ratio can be increased and the regenerator can be made smaller.

The LARR could be sited in an energy park, supplying regenerated fuel to a number of reactors at the same site. This would reduce the opportunity for theft of the nuclear material, which would never be transported off-site and would be extremely radioactive in all cases where it contained appreciable amounts of fissile material.

Table IV-1

DRIVEN REACTOR SYSTEM

P_{TA} = Power output from target/power input to accelerator

$$P_{TA} = E_P E_A \left[\left(Y \times \frac{k}{v} \times \frac{1}{1-k} \right) 0.2 + 1 \right]$$

E_P = efficiency of power cycle ~ 0.35

E_A = efficiency of accelerator ~ 0.50

Y = spallation neutron yield from target per 1 GeV proton ~ 50

v = number of neutrons/fission ~ 3.0

k multiplication factor

<u>k</u>	<u>P_{TA}</u>
0.6	1.05
0.7	1.54
0.8	2.51
0.9	5.43
0.95	11.26

Table IV-2

LINEAR ACCELERATOR REGENERATIVE REACTOR

F = production rate of fissile fuel = Kgm/yr

$$F = 0.0786 \text{ IY} \left(\frac{C_T - 1}{C_T + 1} \right) \left(\frac{1}{1 - k_T} \right)$$

I = linear accelerator proton current

Y = spallation neutron yield per GeV proton

$$P_{TA} = E_p E_a \left[Y \times \left(\frac{C_T - 1}{C_T + 1} \right) \left(\frac{1}{1 - k_T} \right) \left(\frac{1}{1 - C_p} \right) \times 0.2 + 1 \right]$$

P_{TA} = $\frac{\text{Total power output from power reactors}}{\text{to power input to accelerator}}$

E_p = efficiency of power cycle

E_a = efficiency of accelerator

C_T = conversion ratio for target reactor

C_p = conversion ratio for power reactor

k_T = multiplication factor for target reactor

For the special case of starting with unenriched U^{238} in the regenerator

$$C_T \rightarrow \infty \text{ and } k_T \rightarrow 0$$

so that $F = 0.0786 \text{ IY}$

Thus for $I = 300 \text{ ma}$ and $Y = 50$

$$F = 1180 \text{ Kgm/yr}$$

$$\text{and } P_{TA} = E_p E_a \left[Y \times \frac{1}{1 - E_p} \times 0.2 + 1 \right]$$

when $E_p = 0.35$, $E_a = 0.50$ and $Y = 50$

C_p	P_{TA}
0.6	4.6
0.7	6.0
0.8	8.9
0.9	17.6
0.95	35.2

LINEAR ACCELERATOR DRIVEN REACTOR (LADR)
MULTIPLICATION FACTOR (k_{∞}) AS A FUNCTION OF BURN UP

Figure 4.1

V. Mechanical/Thermal and Systems Design

Figure 5.1 shows a possible configuration for introducing the accelerator beam into a driven reactor lattice that is inside a pressure vessel. For safety reasons, as well as increased uniformity of beam deposition, it is more desirable to direct the beam into the lattice from the top rather than from the side. The fuel assemblies are locked into a bottom grid plate so that the beam will not have to penetrate a thick steel top grid plate. The assemblies are double ended and can be turned end for end as well as radially shuffled, so as to increase the uniformity of irradiation.

In this configuration, the water depth above the fuel assemblies will have to be relatively shallow compared to PWRs. To adequately penetrate the fuel assembly lattice, which will be approximately 10 feet long, a beam particle energy of ~ 3 GeV will probably be required. To avoid excessive alteration of the beam in the water layer covering the lattice, it will be necessary to restrict the depth to a few feet, compared to ~ 12 feet for PWR's. This should be an adequate depth but this question will have to be examined in more detail. The water density is a factor of ~ 5 lower than the average lattice density. At maximum, only about 10% beam attenuation in the water layer can be tolerated.

If the beam divergence angle is 20 degrees, the beam diffuser will have to be about 20 feet above the top of the lattice. The head of the pressure vessel will have to accommodate a long cone-shaped region for the spreading beam. The mechanical design problems will have to be examined.

If a beam divergence angle of only 10 degrees can be achieved, the diffuser will have to be located about 50 feet above the lattice, and the mechanical problems will become more difficult. The most reasonable solution is probably to flange the beam tube at some point below the diffuser so the pressure vessel head can be removed for fuel loading/unloading and shuffling operations.

The beam losses through tens of feet of high pressure steam will have to be examined. It may be necessary to use a low density inert gas, i.e., He, in part of the beam spreading region.

Figure 5.2 shows a configuration where the window is placed outside of the containment vessel and the beam transported through high pressure steam or inert gas to the diffuser at the top of the driven reactor. The beam losses in the gas filled portions of the beam transport tube are probably too high for this to be practical. It appears likely that the window between the high pressure and vacuum portions of the beam transport tube will have to be placed relatively close to the beam diffuser, e.g., directly above it.

Figure 5.3 shows a window concept that should permit rapid replacement of damaged windows. Window lifetime is not well defined as yet, but it probably will be limited to a few weeks by interstitial gas (He and H) formed by high energy nuclear reactions. The frequent replacement required probably rules out a welded or bolted window. The window in Figure 5.3 shows a window consisting of an annular cylinder enclosed in a tight fitting cylindrical sleeve. The beam transport tube perpendicularly intersects the sleeve, with

the beam passing through the opposite sides of the annular window. Gap leakage from the high pressure to the low pressure side (between the annular window and its enclosing sleeve) is minimized by using either a labyrinth or freeze seal. When the damage to the annular window exceeds allowable levels, the annular window can be repositioned azimuthally or moved along its axis, so that fresh undamaged surfaces are exposed to the beam.

Assuming a typical beam diameter of 6 inches, a diameter of 12 inches for the annular cylinder and an effective length of 10 feet, a given window could be repositioned on the order of 100 times before it would have to be replaced.

It is desirable to have the capability of repositioning windows either on-line when the reactor is operating, or with a very short shut down. This appears feasible with a dual beam transport system, as shown in Figure 5.4. The beam could probably be switched rapidly, i.e., in a fraction of a second, from a given transport line to the other. The window in the non-active line could then be isolated by movable shutter valves and the annular cylinder then repositioned with a labyrinth seal, this probably would take a few minutes; with a freeze seal probably on the order of an hour. In either case, the window in the non-active transport line could be repositioned in a much shorter time than the damage time for the window in the active line.

With two beam lines and ~ 100 positions for each window, the reactor should function for at least a year before the annular cylinder have to be replaced. The most logical time for such replacement would be during reactor fueling or shuffling operations.

A low pressure window is shown in Figure 5.4 between the Linac and the high pressure window. The necessity for this window will depend on the rate and nature of the gas leaking past the high pressure window. If the leakage rate is low and the gas easily condensed, e.g. steam, the low pressure window will probably not be needed. A freeze seal using a low melting alloy (e.g., Wood's metal) should have essentially zero leakage. The fusible alloy could be injected through small holes in the sleeve. Coolant connections to the movable annular window would be through flexible lines with the coolant probably being room temperature water. At an energy of 3 GeV the loss through the window would only be a few tenths of a percent of the beam.

Figure 5.5 and 5.6 show a possible configuration for a fuel element regenerator. The proton or deuteron beam is directed into a target of liquid lead, which is surrounded by several tons of pressure tubes. The tubes contain the fuel elements which absorb the neutrons produced in the lead target. Damage to the pressure tubes by the direct beam and high energy neutrons should be relatively small, since the lead target will be the primary target. The rejuvenated fuel elements will then be returned to a thermal reactor of either the LWR or CANDU type. The pressure tubes will probably be double walled with a gas insulating space, as in the CANDU reactors, since the lead bath is likely to operate at a higher temperature than the coolant for the fuel elements.

No windows would be necessary, since the vapor pressure of lead is very low, on the order of 10^{-4} Torr. The small amount of lead vapor

entering the beam transport can be readily condensed and returned to the lead target.

To achieve effective fuel element rejuvenation, the neutron spectrum should be relatively hard. However, the coolant has to be compatible with the zircaloy cladding of the fuel elements. The best choice from the standpoint of a hard neutron spectrum is steam. The maximum allowable temperature with zircaloy will be relatively low on the order of 700°F. Adequate cooling of the fuel elements with steam appears feasible, given this restriction on maximum temperature. The corresponding allowable temperature rise along the pressure tube will only be ~ 100°F; however, straight flow along the fuel pins of LWR elements does not appear to result in excessive pressure losses. Typically, each pressure tube of the rejuvenator would have an LWR fuel element inside. The steam would flow up the pressure tube on the outer faces of the LWR box assembly, and then flow down the assembly, cooling the fuel pins.

**SUGGESTED APPLICATION OF A PROTON BEAM
TO A SUB-CRITICAL PWR CORE
LINEAR ACCELERATOR DRIVEN REACTOR**

Figure 5.1

LINEAR ACCELERATOR DRIVEN REACTOR
WITH WINDOW AND BEAM TRAIN

Figure 5.2

LADR

WINDOW ARRANGEMENT

LINEAR ACCELERATOR REACTOR

Figure 5.3

LADR
WYE ARRANGEMENT OF WINDOWS

Figure 5.4

PART PLAN VIEW OF CALANDRIA TUBE BANK

Figure 5.5

**CALANDRIA WITH LEAD TARGET
LINEAR ACCELERATOR REGENERATIVE REACTOR**

Figure 5.6

VI. Systems Aspects

Figure 6.1 shows the projected cost and efficiency of the accelerator as a function of beam power, for a beam energy of 3 GeV. These curves and similar ones for a 1 GeV beam were derived from the accelerator cost studies presented earlier in this report. These results can be used to estimate the effects of reactor conversion ratio, target heat recovery efficiency, and neutron yield on a reactor (or reactors) whose fuel is also being rejuvenated by an accelerator-target system. Figures 6.2 through 6.4 illustrate these effects for a rejuvenator with a 3 GeV beam, and 6.5 and 6.6 for a rejuvenator with a 1 GeV beam. The direct cost of the accelerator is shown, together with the fractional power recirculated from the reactors whose fuel is being rejuvenated (all of the power produced by the target is recirculated to the accelerator).

The acceptable limits on recirculating power and accelerator cost are taken as 20% and \$200/KW(e) [unit cost of the accelerator per gross KW(e) output from the reactors that it is rejuvenating fuel for]. Two cases are shown for neutron yield. The first ($y = 20$, $P_T = P_B$) refers conservatively to a yield of 20 neutrons per GeV of beam particle energy with the target thermal power equal to the beam power. This is assumed to be characteristic of a lead target. The second case ($y = 40$, $P_T = 4 P_B$) refers to a yield of 40 neutrons per GeV of beam particle energy and a target thermal power four times the beam power. This is assumed to be characteristic of the fuel elements being the target for the beam. Two variations are shown for each case; one accelerator-rejuvenator per reactor, and four accelerator-rejuvenators per reactor.

It is clear that heat recovery from the target to a power cycle (assumed to be 33% efficient) is essential for a practical rejuvenator system unless the reactor conversion ratio is relatively high, i.e., above 0.8. With heat recovery to a power cycle and a high neutron and high energy yield in the target, reactors can be operated with a conversion ratio down to 0.6 without exceeding the recirculating power limit. With a low neutron/low energy yield, the reactor conversion ratio must be above ~ 0.9 to stay within acceptable recirculating power limits, and recovery of target heat in a power cycle is not necessary.

The accelerator cost tends not to be as restrictive as recirculating power. In all cases, reactors are able to operate at lower conversion ratios within the recirculating power and cost limits if several reactors share one accelerator-rejuvenator. The principal difference between the 3 GeV and 1 GeV beam cases is that several reactors cannot share the same accelerator-rejuvenator because of beam current limitations in the accelerator, unless the reactor conversion ratio is high.

The most important conclusion is that for accelerators to rejuvenate fuel for reactors with conversion ratios of ~ 0.6 , the neutron and heat yields in the target must both be high. Low yields in the target will require reactors to operate at conversion ratios in the 0.8 to 0.9 range. Improvements in accelerator efficiency and cost would help to relax these restrictions. Also, if the target heat could be recovered at greater efficiency, the efficiency restrictions would be relaxed. One interesting possibility is to use the accelerator-rejuvenator as a superheater for steam produced in an LWR or CANDU.

The incremental efficiency for recovery of the heat would then be much higher than 33%.

Figures 6.7 and 6.8 show corresponding unit cost and recirculating power curves for the driven reactor version. The primary variable here is k_{∞} instead of conversion ratio.

With one accelerator per 1000 MW(e) driven reactor, k_{∞} has to be above 0.955 for the high yield case and above 0.975 for the low yield case. This constraint is relaxed somewhat if one accelerator feeds four reactors, but the k_{∞} still has to be well above 0.9. In all cases, the recirculating power condition is more restrictive than the unit accelerator cost condition. An increase in neutron yield as indicated in the earlier neutronics section can relax this k_{∞} requirement.

UNIT CAPITAL COST & EFFICIENCY VS. BEAM POWER
[1.0 MeV/m]

Figure 6.1

Figure 6.2

Figure 6.3

ACCELERATOR UNIT COST VS. REACTOR CONVERSION RATIO [3 GEV BEAM ENERGY]

Figure 6.4

RECIRCULATING POWER VS REACTOR CONVERSION RATIO

Figure 6.5

Figure 6.6

CASE I: 1 ACCELERATOR/1000 MW (e) REACTOR

Figure 6.7

CASE II: 1 ACCELERATOR/4-1000 MW(e) REACTORS

Figure 6.8

VII. Program Plan

A program plan and schedule is given in Table VII-1 and is based on the following tasks:

Task 1. Linear Accelerator Characteristics

1.1 Establish assumptions as to performance and design characteristics desired of the linear accelerator for supplying protons or deuterons to the target assembly. This should be done parametrically, as a function of beam current and energy.

1.2 Identify problems that remain to be solved in design, construction and operation of the accelerator. Recommend steps to be taken to resolve these problems.

1.3 Provide estimates of capital cost and operating cost of the accelerator, as a function of beam current and energy.

Estimates should be made under two general assumptions:

- . The accelerator is a first-of-a-kind device to be built only once, and
- . a number of standardized accelerators are constructed to be used with a number of accelerator breeders.

Task 2 Proton Beam and Target Reactor Physics

2.1 Neutron Yield and Energy Release

. Calculate the yield, energy spectra distribution and power density distribution of neutrons produced by spallation and high energy fission, induced by the proton or deuteron beam in an infinite medium. The cases to be treated are listed in Table VII-2 roughly in the order of most promising cases.

. Calculate the spatial distribution of energy density associated with the nuclear processes leading to the neutron production in a lattice structure for the cases listed in Table VII-2.

2.2 Target Reactor Physics

. Using the EPRI cell code, calculate k_{∞} as a function of burnup for a typical cell for the cases listed in Table VII-2.

2.3 Energy-Dependent Nuclear Data to 50 MeV

. Evaluate nuclear data for fissile and fertile material (^{235}U , ^{233}U , ^{239}Pu , ^{241}Pu , ^{238}U , ^{240}Pu , ^{232}Th , and ^{234}U) and the target material Pb as needed for the analysis under Tasks 2.1, 2.2, and 3.

. Evaluate data on structural materials needed for the analysis under Tasks 2.1, 2.2, and 3. This should include fuel, cladding, pressure vessel steel, beam entry windows, and other structural materials as required.

2.4 Gas-Production Cross-Section for Structural Materials

. Provide cross-sections that may be used to calculate the rate of gas formed in cladding, beam windows, and other important structural materials, as a result of interaction with the primary proton or deuteron beam, and secondary neutrons.

Task 3. Fuel Cycle History

3.1 For a selected driven reactor case from Task 2.1, calculate the spatial and temporal power distribution and the k_{∞} as a function of burnup. Assume fuel shuffling, with heavily burned fuel removed and new fuel added as necessary, according to a predetermined strategy.

3.2 For a selected regenerative reactor case from Task 2.1, calculate the spatial and temporal power distribution and the k_{∞} as a function of burnup for both the target and power reactor. Assume fuel shuffling, heavily burned fuel removal, and new fuel added as necessary, according to a predetermined strategy.

3.3 Calculate the spatial and temporal power distribution and the k_{∞} of a CANDU-type reactor whose input of fuel is regenerated through irradiation according to one of the cases from Task 2.1 Cases should be considered where the core is initially composed of fuel, all of the same composition and is burned to full replacement, and where fuel is replaced on-line according to CANDU shuffling and replacement methods. Extend the analysis to formulate a complete view of the rate of fuel irradiation and of the general design of irradiator to satisfy the reactor input requirements.

Task 4 Thermal-Hydraulic and Mechanical Design

4.1 Driven Reactor

For the driven reactor case of Task 3 conduct the necessary thermal-hydraulic design to establish total system power, orifice requirements, other fuel shuffling options and their impacts, etc.

4.2 Regenerative Reactor

For the most promising regenerative case selected from Task 3, conduct the thermal-hydraulic analysis as described in Task 4.1 above.

Task 5 Materials Selection

5.1 Survey, assess, and recommend materials for the fertile fuel and cladding.

Estimate high energy proton and neutron damage and the fuel element behavior at high burnup conditions. Define a program for obtaining the necessary materials information for use in the design of accelerator reactors chosen in Task 3.

5.2 Survey, assess, and recommend materials for windows and calandria tubes of accelerator reactors and define information needed to incorporate in the design of the accelerator reactors selected in Task 3. Estimate losses, damage, and cooling of windows.

Task 6. Economic Evaluation

6.1 Perform a preliminary parametric study to determine basic criteria controlling economic evaluation of the driven reactor and the regenerative reactor.

6.2 Estimate the capital and operating cost for the driven and regenerative reactor systems selected in Task 3.

Task 7. Non-Proliferation Assessment

Assess the safeguards aspects of the driven and regenerative systems relative to LMFBR-PWR fuel cycle system. Assess the non-proliferation aspects of the driven and regenerative reactor systems from the international development point of view.

LDR AND LRR SYSTEMS

<u>CASE</u>	<u>REACTOR TYPE</u>	<u>TARGET</u>	<u>CLAD AND FUEL</u>	<u>COOLANT</u>	<u>LATTICE SPACING</u>
1	DRIVEN	PV-W-ZR-UO ₂	ZR-UO ₂	LW	TIGHT
2	DRIVEN	PV-W-TH	ZR-TH	LW	TIGHT
3	REGENERATIVE	CAL-ZR-UO ₂	ZR-UO ₂	HW	TIGHT
4	REGENERATIVE	CAL-PB	ZR-UO ₂	STEAM (HW)	TIGHT
5	REGENERATIVE	CAL-PB	ZR-TH	STEAM (HW)	TIGHT
6	DRIVEN	PV-W-ZR-UO ₂	ZR-UO ₂	STEAM (HW)	TIGHT
7	DRIVEN	PV-W-ZR-TH	ZR-TH	STEAM (HW)	TIGHT
8	DRIVEN	CAL-ZR-TH	ZR-TH	HW	OPEN
9	DRIVEN	CAL-ZR-UO ₂	ZR-UO ₂	HW	OPEN

LW - LIGHT WATER

HW - HEAVY WATER

PV - PRESSURE VESSEL

W - WINDOW

CAL - CALANDRIA

PB - LEAD

Table VII-2

Brookhaven National Laboratory

LINEAR ACCELERATOR ASSISTED REACTOR (LAR)

Program Plan - FY 1978

Program Plan - FY 1978 (Continued)

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