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Environmental Impact of Extended Burnup Fuel Cycles in Calvert Cliffs Units 1 and 2

1.0 Introduction

Baltimore Gas and Electric (BG&E) is planning to extend both the cycle length and the discharge fuel burnup in Calvert Cliffs Units 1 and 2. Since the original Calvert Cliffs Environmental Report was based on lower fuel burnups, this report has been prepared to address generically those environmental impacts which are affected by the eighteen-month extended discharge burnup fuel cycle. Detailed safety analysis results are not presented but will be addressed as required on a cycle-by-cycle basis in the reload submittals. The conclusion reached herein is that there are no significant or substantive increases in adverse environmental impacts due to the planned actions; in fact, there are several areas where the environmental impact of the Calvert Cliffs fuel cycle is reduced.

The planned action is shown schematically in Figure 1 which is the projected operation schedule for Calvert Cliffs Units 1 and 2 for the next several years. Generally, through 1980, the units have been operating on annual reload cycles with approximately a three-batch fuel management scheme. Beginning in late 1980 for Unit 1 and early 1981 for Unit 2, a transition is planned to an 18-month fuel cycle also with approximately three-batch fuel management. It is anticipated that the equilibrium batch average fuel discharge burnup will reach about 43,000 MWD/T and that the maximum rod average burnup will be about 50,000 MWD/T.

This report is organized as follows. The technical bases for assuring successful fuel performance at extended burnups are discussed in Section 2. The environmental impact due to the effects of longer cycles and extended burnup on the uranium fuel cycle are addressed in Section 3. In Section 4, the radiological effects of the eighteen-month extended burnup fuel cycle are presented. Section 5 summarizes the above discussions and draws a conclusion regarding the environmental impact of the planned action.

2.0 Fuel Performance

Successful fuel performance is required for the environmental impact to remain acceptable as a result of increases in the fuel burnup. As discussed below, technical evaluations performed to date using available operational and experimental data indicate a high likelihood that the current C-E fuel design, or slight modifications thereof, can successfully be operated to higher burnups than are now standard practice without increasing the propensity for fuel rod failure.

C-E currently has over 220,000 standard fuel rods operating at a defect level of approximately 0.03 percent (Reference 1). Thus, far, over 18400 rods have been successfully operated to between 32,000 and 44,000 MWD/T. Since the upper limit of primary coolant activity (Technical Specification limit) for operation of Calvert Cliffs 1 and 2 is equivalent to approximately a 0.25 percent defect level, a significant margin exists to cover the possibility of increased failures should they occur from extended burnup. However, this is not expected to be the case. As discussed in the following sections, C-E has in place numerous irradiation programs to demonstrate that fuel rods can be successfully operated to higher discharge exposures with the same or better performance than is typical of standard PWR fuel.

Since the great majority of the technical bases for fuel design applicable to the operating conditions at Calvert Cliffs is limited to burnups up to about 37,000 MWD/T, some technical uncertainties at extended burnups do exist in such areas as pellet clad interaction, external clad corrosion, fuel assembly dimensional changes, and fission gas release. The first two areas are those which determine the effects of extended burnup on the propensity for fuel failure; the latter two areas are not directly related to fuel rod integrity but are included for completeness of the discussion.

2.1 Pellet Clad Interation (PCI)

PCI has been identified as the cause of a limited number of fuel rod failures due to rapid power changes (ramps) in light water reactors. The propensity for fuel failures due to PCI is dependent on whether the fuel is in a condition susceptible to PCI, and the probability that a change in power of sufficient magnitude and rate will occur.

Considerable work is being done to define the major causes of PCI, that is, the effects of power change rate, final power, power increase and burnup. PCI failure has not been observed in fuel with very low burnups. Initially, fuel becomes more susceptible to PCI failure as burnup increases because of the closure of the as-fabricated gap between fuel and cladding as a result of clad creepdown, pellet swelling and relocation and because of the increased availability of iodine and cesium (fission products) which are considered to be the primary corrosive species contributing to stress corrosion cracking. [

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Currently, the Studsvik International Cooperative Over-Ramp Program and the CE/KWU Ramp Program at Petten have perhaps the most applicable data on ramp testing as a function of burnup (see Figure 2). [

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Figure 2 shows C-E's acquisition schedule for high burnup PCI data. [

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Confirmation of acceptable behavior at high burnups will thus be available well in advance of the time when full scale extended burnups are attained in Calvert Cliffs.

2.2 External Clad Corrosion

Clad corrosion involves a reaction of the clad with water, leading to the formation of an oxide on the external cladding surface. Although corrosion on the outside surface of PWR fuel rods has not been a cause of fuel rod failures, this phenomenon is of potential concern for extended burnup fuel since technical uncertainties exist when extrapolating present data to extended exposures. Figure 3 shows that corrosion data from Calvert Cliffs 1 for exposures up to 46,000 MWD/T are currently available. These data show no reason for not extending burnup because of external clad corrosion. Additionally, data from the joint EPRI/CE/KWU Corrosion Program will provide data for exposures of up to 50,000 MWD/T during 1981, well in advance of the time when full scale extended burnups are attained in Calvert Cliffs.

2.3 Fission Gas Release

Extending the burnup increases the amount of released fission gas from the fuel pellets which increases the internal fuel rod pressure. Should the pressure increase sufficiently, a possible concern arises with respect to the performance of the fuel cladding. [

] However, even if future analyses show that this condition is predicted, the criterion of no clad lift-off will continue to be maintained. Specifically, the fuel design criterion will continue to include a requirement that the cladding will not creep away from contact with the fuel. Thus, no adverse consequences would be expected.

The availability of data on the effect of high burnup on fission gas release is shown in Figure 4. As can be seen, data are already available up to 37,000 MWD/T and five sources of data in the 40-45,000 MWD/T range will be available well before full scale extended burnups are encountered in Calvert Cliffs.

2.4 Fuel Assembly Dimensional Changes

Extending fuel burnup gives rise to two concerns in the area of fuel assembly dimensional changes, viz., bowing and reduction of axial clearance.

Due to a combination of stress relaxation, growth and creep, fuel has been found to bow. Excessive bowing could affect power peaking, DNB and assembly handling. The extent of bowing in C-E fuel has been minor, even after burnups of up to 46,000 MWD/T as shown by recent data from the Calvert Cliffs-1 reactor (see Figure 3). C-E data obtained after 1, 2, 3 and 4 cycles of operation show that the effect of burnup on the extent of bowing beyond approximately two cycles of exposure is very small. In particular, the extent of fuel rod bowing appears to saturate after 20,000 MWD/T. Therefore, it is expected that the incremental bowing due to extended burnup will not be significant.

Zircaloy clad fuel rods increase in length during reactor service due to irradiation induced growth of the Zircaloy cladding. This is a well characterized effect and requires only the proper design allowance between the assembly upper end fitting and the top of the fuel rods.

Figure 3 shows the projected schedule for acquiring additional data on dimensional stability. This indicates that the C-E/EPRI/Calvert Cliffs-1 program has already provided data up to 46,000 MWD/T. These data indicate no dimensional stability limitation in achieving high fuel assembly burnups. In addition, data for exposures of up to 52,000 MWD/T will be available well before such burnups are attained in Calvert Cliffs.

In summary, significant factors affecting fuel rod integrity and design at extended burnup are continuing to be closely investigated. In addition, where technical uncertainties exist, C-E has been and is currently participating in extensive irradiation programs that provide data to assure successful fuel performance at extended fuel burnups. In all cases, an adequate high burnup data base will be available prior to achieving full scale hign burnup in the Calvert Cliffs Units.

3.0 Uranium Fuel Cycle

There is a trend in the industry toward eighteen-month fuel cycles. This trend is motivated in part by an anticipated improvement in plant capacity factor (due to the reduced number of refueling outages) and the associated decrease in required replacement power. The planned action for Calvert Cliffs is a change from annual to eighteen-month fuel cycles and an increase in fuel discharge exposure which is consistent with successful fuel performance. This section examines the differential environmental impact of the planned action and concludes that the impact is favorable.

Reload fuel management schemes to date for Calvert Cliffs have employed batch average discharge exposures of about 29,000 MWD/T; the planned extended burnup design employs a batch average discharge exposure of up to about 43,000 MWD/T. For a given refueling interval, the higher discharge fuel exposure plan shows a smaller fraction of the core being replaced at each refueling. Table 1 provides a comparison of estimated fuel management information for the planned 18-month cycles with extended fuel burnup versus operation of 18-month cycles with lower fuel burnups. The low burnup design assumes that one-half of the core will be replaced at each 18-month refueling outage; fuel assemblies would be burned for three years (similar to the current practice for annual cycles). The extended burnup design shown assumes that one-third of the core is replaced at each 18-month refueling outage and that fuel assemblies are burned for four and one-half years (a 50% increase from current practice). This three-batch design yields the highest burnup for a range of fuel management plans being considered; designs with somewhat lower average burnups may actually be implemented in Calvert Cliffs.

The environm tal impact associated with the out-of-core uranium fuel cycle includes effects due to uranium mining and milling, UF_6 production, enrichment, fuel fabrication, transportation, and spent fuel storage and disposal (Reference 8). Implementation of the planned extended burnups will reduce the environmental impact of the Calvert Cliffs uranium fuel cycle a consequence of lower head-end and tail-end fuel cycle requirements. In comparing cycles of the same length, it is observed from Table 1 that increased fuel exposure is expected to reduce U_3O_8 requirements (and hence uranium mining and milling and UF_6 production requirements) by up to 13%, to reduce separative work requirements by up to 4%, and to reduce the number of assemblies fabricated (and hence conversion, fabrication, fuel transportation and spent fuel storage requirements) by up to 33%. Implementation of extended burnups will reduce land and water use and liquid, solid and radiological effluents associated with the above requirements and hence result in a reduced environmental impact from these effects.

Comparison of the eighteen-month extended burnup cycle with continued operation of an annual low burnup cycle shows somewhat smaller reductions but still results in an overall decreased environmental impact. As shown in Table 2, while the reduction in the number of assemblies fabricated and discharged continues at up to 33%, the reduction in U_3O_8 requirements for this comparison drops to 4%. The increase in separative work requirements entails an increased consumption of electrical energy at the enrichment facility. However, this increase can be favorably compared with the increase in energy production which may be obtained by the reduced number of refueling outages with an 18-month fuel cycle.*

Two aspects of higher burnup fuel which should be discussed with regard to storage design requirements are decay heat considerations and criticality conditions. Decay heat considerations should not adversely

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*The additional SWUs consume 48 GWhe/year but the reactor produces an additional 231 GWhe/year.

affect fuel storage requirements. This is because the short term (first few weeks after reactor shutdown) decay heat production from high burnup fuel is virtually identical to the decay heat production from lower burnup fuel (since almost all of the decay heat produced in this time interval is from short-lived fission products which reach a saturated concentration during the first year of irradiation). Higher burnup fuel does not exhibit significantly higher decay heat production than lower burnup fuel until longer periods after shutdown at which time the decay heat production is significantly lower (by an order of magnitude or more) than the decay heat production at shorter time intervals after shutdown (see Figure 5). Heat removal requirements for the spent fuel storage pool are thus unaffected since cooling requirements are based principally on the earlier (larger) heating rates which are unaffected by increased fuel exposure. Since the initial enrichment of fuel designed for extended burnups is higher than that of fuel designed for lower burnups (see Table 1), criticality evaluations of shipping, handling, and storage activities are being addressed in licensing proceedings separate from this report. As discussed above for fabrication, the successful implementation of higher burnup fuels will result in a reduction in the number of spent fuel assemblies to be stored of up to 33%.

4.0 Radiological Impact of Eighteen-Month Extended Burnup Fuel Cycles

This section provides an assessment of the radiological impact of operating Calvert Cliffs with a planned fuel cycle design consisting of an 18-month interval between refuelings and increased discharge exposures. Consideration is given primarily to fission product related nuclides since they are the only radioisotopes significantly affected by the planned fuel cycle. A comparison of radiological data for a low burnup annual fuel cycle is made with values for the planned extended burnup fuel cycle. From this comparison, it has been determined that the source terms are dominated by the short-lived radionuclides, the concentrations of which saturate during the first cycle of irradiation. Therefore, with the high specific activity short-lived nuclides constituting the predominant radiological concentrations and energy levels, operation of the planned extended fuel cycle with the expected equivalent fuel performance (i.e., no increase in the propensity for fuel failures) has a negligible additional radiological impact. Furthermore, a basis for such operation will be the continued use of the same radiological activity technical specification limits as are presently being employed.

4.1 Plant Releases

An assessment has been made of the environmental impact of anticipated radiological releases from the Calvert Cliffs units operating with the planned eighteen-month extended burnup fuel cycle. The source of radiological activity for both gaseous and liquid effluents is the Reactor Coolant System (RCS).

The concentrations of nuclides in the RCS were determined through the use of a computer coded mathematical model developed by C E which solves the mass balance for nuclide production and removal in the fuel pellet region and the reactor coolant region. In the fuel pellet region, production mechanisms include direct fission yield, parent fission product decay, and neutron activation, while removal includes decay, neutron activation, and escape to the coolant. In the coolant region, production mechanisms include escape from the fuel (through defective cladding), parent decay, and neutron activation; removal is by decay, coolant purification, leakage, and by feed and bleed operations for startups, shutdowns, load follow, and reactivity depletion.

For radionuclides deemed to be of concern with respect to environmental impact, Table 3 shows the incremental changes in RCS fission product nuclide concentration between the planned fuel cycle and the annual low burnup cycle calculated using the above mathematical modelling. Increases in the longer lived isotopes are due to the higher core average burnup. Generally, the short-lived isotopes reach the same equilibrium concentration in either cycle; however, there are increases in some of the shorter lived isotopes due to the lower thermal flux in the more highly enriched core and to the buildup of longer lived precursors.

4.1.1 Gaseous Releases

A summation of the total annual gaseous releases from the Calvert Cliffs units for the years 1978 and 1979 is provided in Table 4A as presented in the semiannual report to the NRC. As shown, the gaseous effluents are a small fraction of the plant technical specifications. The source of the radioactive gases is the RCS and thus the activity released from the plant is a function of the reactor coolant specific activity.

From Table 3, it is concluded that since the significant iodine isotope concentrations in the reactor coolant do not show an increase, the planned fuel cycle does not result in an increase in the site boundary thyroid dose. The increase in fission and activation gas effluent activity, primarily due to the increase in the Kr-85 reactor coolant concentration, would be approximately 6 percent and would remain a small fraction of the plant technical specification.

4.1.2 Liquid Releases

The plant operating data summarized in Table 4B provides the quarterly liquid activity releases from the plant for the years 1978 and 1979. As shown, the liquid effluents are a small fraction of the applicable limit. As in the case of gaseous effluents, the liquid releases are a function of the activity in the reactor coolant.

It is anticipated that there will be an increase in tritium production due to the higher soluble boron concentration required as a result of extending the fuel cycle length to 18 months. The radiological impact of tritium released in liquid effluents is negligible, however, since tritium contributes less than 1.0 percent to the total whole body dose. The increases in long-lived isotopes due to the planned fuel cycle will likewise contribute negligibly to the whole body dose which is, in any case, well below the 40 CFR190 limits. Table 4C summarizes the whole body dose contributions resulting from gaseous and liquid releases from the plant for the years 1978 and 1979. The thyroid dose projected to result from the liquid activity releases under the planned fuel cycle will not change from that projected for the annual fuel cycle.

4.2 Plant Accidents

This section presents a general review of plant accidents reported in the FSAR and the Environmental Report. For the purpose of discussion here, plant accidents are divided into categories involving release of reactor coolant system activity, fuel rod gas gap activity and reactor core activity. The conclusion is that there is no significant incremental adverse impact caused by the radiological effects of the planned eighteen-month extended discharge burnup fuel cycle.

4.2.1 Accidents Involving Release of RCS Activity

Accidents in this category are those for which it has been determined that fuel cladding integrity is maintained during the accident. Therefore, the activity available for release is dependent only upon the activity within the RCS.

Examination of Table 3 indicate an increase in RCS noble gas specific activity for Kr-85, Xe-131m, and X -135. However, the whole body dose is dominated by the high specific activity of Xe-133 which is not changed for the extended fuel cycle. The thyroid dose is determined by the iodine activity; however, the only iodine activity which increases is that of the long-lived I-129 which makes a neglibible contribution since it has a low specific activity (see Table 5). Therefore, it is concluded that there is no significant increase in radiological consequences due to extended burnup fuel for accidents involving release of RCS activity.

4.2.2 Accidents Involving Fuel Rod Gas Gap Activity Release

Accidents in this category involve loss of integrity of the fuel rod cladding and the subsequent release of the activity contained in the fuel rod gas gap. For accidents or transients which involve potential fuel failure, the quantities (curies) of radioactive gases and volatiles in the fuel gas gap are pertinent. Although larger quantities of long-lived fission products will be present in the fuel, review of the applicable source terms indicates that the doses are due almost exclusively to short-lived isotopes of radioactive gases and volatiles. This is because the short-lived isotopes have high specific activities, while the low specific activities of long-lived isotopes result in a small contribution to the dose. Since these short-lived gases and volatiles reach equilibrium concentration after about one year of irradiation, there is no significant increase in site boundary dose due to the release of radioactive gases and volatiles from extended burnup fuel as compared to lower burnup fuel.

As an example, the radiological consequences of the fuel handling accident are addressed here because it is representative of this type of accident. Table 6 presents a comparison of the significant radioisotopes for this accident for low burnup (28,800 MWD/T) and extended burnup (43,200 MWD/T) spent fuel assemblies as calculated by the ORIGEN computer code (Reference 10). Since the dose is dominated by the high specific activity of Xe-133 which has not changed, there is no significant change to the whole body dose from this accident. Furthermore, since there is no significant change in the iodine isotope I-131, there is no significant change in the thyroid dose.

4.2.3 Accidents Involving Release of Reactor Core Inventory

The FSAR provides an analysis of the radiological consequences of a postulated maximum hypothetical loss of coolant accident (LOCA). Although increased fuel failures during the LOCA event are not anticipated, it should be noted that regulations require that site boundary doses be evaluated under the conservative assumption of 100% fuel failure; thus, any tendency for increased fuel failures during LOCA with high burnup fuel will not alter the perceived consequences of the loss-of-coolant accident. This accident is postulated to result in the release of 100% of the noble gas, 50% of the iodine and 1% of all other fission product isotopes present in the reactor core at the end of the fuel cycle. Site boundary doses are due almost exclusively to the short-lived fission products which reach saturated concentrations during the first year of irradiation. Thus, the inventory (curies) of isotopes which are significant in the calculation of site boundary dose during the loss-of-coolant accident is virtually independent of burnup (see Table 7). In particular, when the source terms in Table 7 as calculated by the ORIGEN computer code (Reference 10) are weighted by the Dose Factors of Reference 9, it is clear that there is no significant increase in the dose due to either the noble gas or the iodine source terms. It is thus concluded that there will be no additional significant radiological impact resulting from extended fuel cycle operations for these events as postulated in the FSAR.

4.3 Fuel Handling During Refueling

Dose rates in the spent fuel pool and refueling cavities are a function of the sources within the irradiated fuel and the fission and corrosion products in the pool water. The dose rate is dominated by the high specific activity short-lived isotopes which reach approximately the same equilibrium level. Furthermore, the dose rate is required to be within technical specifications.

The actual dose is determined by the dose rate and the duration of the exposure. Since the planned action involves a refueling outage every eighteen months rather than every twelve months, the dose component due to refueling is reduced by about one-third.

4.4 Transportation of Fuel

The primary impact of extended fuel exposures on the transportation of fuel to and from the Calvert Cliffs nuclear power plant site is the substantial reduction in the number of fuel assemblies to be transported. As discussed in Section 3.0, a reduction in the number of fuel assemblies of up to 33% would be expected as a result of the planned actions. Somewhat higher enrichments ($\leq$ 4.1 w/o U-235) will be utilized, and criticality evaluations of shipping, handling and storage activites are being addressed separately from this report. The dose and heat rates of newly discharged fuel assemblies are dominated by the high specific activity short-lived isotopes whose concentrations are unaltered by burnup extensions. As the fuel cools down, the contribution of the long-lived isotopes becomes relatively more important, and slight increases in the cooldown time are required to match the dose or heat rate of lesser exposed fuel. Figure 6 shows, for example, that a fuel assembly exposed to 42,000 MWD/T would require an additional 30 days of cooldown to match the gamma energy release of fuel exposed to 28,000 MWD/T at 90 days cooldown. When the dose and heat rates have been matched, the environmental impact of transporting extended burnup fuel depends only upon the number of fuel assemblies being transported; as noted earlier, up to 33% fewer fuel assemblies are required as burnup is increased. There is therefore a reduced environmental impact due to the increase in fuel discharge exposure.

5.0 Summary and Conclusion

Baltimore Gas and Electric is planning to operate the Calvert Cliffs units on an 18-month fuel cycle with average fuel discharge burnups extended from about 29,000 MWD/T up to about 43,000 MWD/T. Fuel demonstration programs are in place to verify acceptable fuel performance at these extended burnups. The environmental impact of the Calvert Cliffs uranium fuel cycle is generally reduced due to reductions in uranium mining and milling, UF_6 production, fuel fabrication, transportation, and spent fuel storage and disposal. Radiological effects are not significantly changed since doses are generally dominated by short-lived fission products which reach equilibrium levels at lower burnups.

Based on the above statements, it is concluded that the planned operation of Calvert Cliffs Units 1 and 2 on an 18-month cycle with extended fuel burnup results in a net reduction in environmental impact.

References

1. M. G. Andrews, et al., "The Performance of C-E Fuel in Operating PWRs," TIS-6171, April 30, 1979.

2. "Final Report on the Studsvik Over-Ramp Project," STOR-37, (Proprietary) to be published.

3. "Examination of Calvert Cliffs I Test Fuel Assembly After Cycle 3," CE/EPRI Report NPSD-87, September 1979.

4. "Fort Calhoun Poolside Inspection Programs at End of Cycles 4 and 5," CE/DOE Report CEND-383, to be published.

5. "CE/KWU/EPRI Waterside Corrosion of Zircaloy Clad Fuel Rods, RP-1250-01, Task A, Review of Fuel Rod Waterside Corrosion Behavior," NPSD-79, June 1979.

6. "CE/EPRI/KWU Waterside Corrosion of Zircaloy Clad Fuel Rods, Review of PWR Fuel Waterside Corrosion Behavior," EPRI-NP-1472, August 1980.

7. "Gas Release and Microstructural Evaluation of Three-Cycle Fuel Rods from Calvert cliffs-1," CE/EPRI Report NPSD-119, December 1980.

8. 10 CFR Part ~~52~~ 51, Table S-3.

9. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Practicable' for Radioactive Material in Light-Water Cooled Nuclear Power Reactor Effluents," WASH-1258, Table A-4 of Annex A, July 1973.

10. M. J. Bell, "ORIGEN, - The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

TABLE 1

Comparison of Fuel Management Information for 18-Month Low Burnup and Extended Burnup Fuel Cycles

	Low Burnup 18-Month Cycles	Extended Burnup 18-Month Cycles	Relative Change
Fraction of Core Replaced per Refueling	0.50	0.33	
Batch Average Discharge Burnup (MWD/MTU)	28,800	43,200	
Equilibrium Cycle Enrichment (w/o)	3.01	3.87	
Number of Assemblies Fabricated/Discharged			
Equilibrium Cycle	108	72	
30-Year Cumulative	2160	1440	-33%
U_3O_8 Requirements (ST)*			
Equilibrium Reload	305	266	
30-Year Cumulative	6096	5309	-13%
Separative Work Requirements (10^3 SWU)*			
Equilibrium Cycle	184	176	
30-Year Cumulative	3673	3535	-4%
Heavy Metal Discharged in Spent Fuel (MTM)			
Equilibrium Cycle	42	28	
30-Year Cumulative	839	560	-33%

*Tails Composition 0.2 w/o

TABLE 2

Comparison of Fuel Management Information
for Annual and Extended Fuel Cycles

	Low Burnup Annual Cycles	Extended Burnup 18-Month Cycles	Relative Change
Fraction of Core Replaced per Refueling	0.33	0.33	
Batch Average Discharge Burnup (MWD/MTU)	28,800	43,200	
Equilibrium Cycle Enrichment (w/o)	2.76	3.87	
Equilibrium Cycle	72	72	
30-Year Cumulative	2160	1440	-33%
U_3O_8 Requirements (ST)*			
Equilibrium Reload	185	266	
30-Year Cumulative	5549	5309	-4%
Separative Work Requirements (10^3SWU)*			
Equilibrium Reload	107	176	
30-Year Cumulative	3212	3535	+10%
Heavy Metal Discharged in Spent Fuel (MTM)			
Equilibriuim Cycle	28	28	
30-Year Cumulative	839	560	-33%

*Tails composition 0.2 w/o

TABLE 3

Incremental Changes in Reactor Coolant System Fission Product Specific Activities From the Low Burnup Annual Fuel Cycle to the Eighteen-Month Extended Burnup Fuel Cycle

Nuclide	Fractional Change	Nuclide	Fractional Change
Kr-85m	0.0	Te-129	0.01
Kr-85	0.62	I-129	0.27
Kr-87	0.0	I-131	0.0
Kr-88	0.0	Te-132	0.0
Xe-131m	0.05	I-132	0.0
Xe-133	0.0	I-333	0.0
Xe-135	0.12	Te-134	0.0
Xe-138	0.0	I-134	0.0
		Cs-134	0.19
Br-84	0.0	I-135	0.0
Rb-88	0.0	Cs-136	0.0
Rb-89	0.0		
Sr-89	0.19	Cs-137	0.24
Sr-90	0.53	Cs-138	0.0
Y-90	0.48	Ba-140	0.0
Sr-91	0.0	La-140	0.0
Y-91	0.13	Pr-143	0.0
Zr-95	0.23	Ce-144	0.15
Mo-99	0.0		
Ru-103	0.0		
Ru-106	0.19		

TABLE 4A

Total Annual Airborne Radiological Effluents

	1978			
A. Fission and Activation Gases	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
1. Total Release, Ci	2.2(+4)	8.4(+2)	2.9(+3)	1.7(+3)
2. Average Release Rate, Ci/sec	2.8(+3)	1.1(+2)	3.6(+2)	2.2(+2)
3. Percent of Quarterly Technical Specification Limit, %	1.6(+1)	6.2(-1)	2.1	1.3
B. Iodines				
1. Total Release, Ci	5.7(-3)	1.4(-2	5.3(-2)	5.0(-2)
2. Average Release Rate, Ci/sec	7.2(-4)	1.7(-3)	6.7(-3)	6.3(-3)
3. Percent of Quarterly Technical Specification Limit, %	4.1(-1)	1.1	4.2	4.0
C. Particulates				
1. Particulates With Half Lives Greater Than 8 Days, Ci	1.6(-3)	8.4(-4)	7.9(-3)	3.3(-3)
2. Average Release Rate, Ci/cc	2.1(-4)	1.1(-4)	1.0(-3)	4.2(-4)
3. Percent of Quarterly Technical Specification Limit, %	*		*	*

*Percent of technical specification limits for iodines includes particulates with half lives greater than 8 days.

TABLE 4A (continued)

Total Annual Airborne Radiological Effluents

	1979 1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
A. Fission and Activation Gases				
1. Total Release, Ci	2.5(+3)	8.8(+2)	5.7(+3)	1.0(+3)
2. Average Release Rate, Ci/sec	3.2(+2)	1.1(+2)	7.9(+2)	1.3(+2)
3. Percent of Quarterly Technical Specification Limit, %	2.2	3.0	4.0	7.3(-1)
B. Iodines				
1. Total Release, Ci	1.6(-1)	6.0(-2)	5.1(-2)	2.9(-2)
2. Average Release Rate, Ci/sec	2.0(-2)	7.6(-3)	6.4(-3)	3.7(-3)
3. Percent of Quarterly Technical Specification Limit, %	1.3(+1)	4.7	4.0	2.3
C. Particulates				
1. Particulates With Half Lives Greater Than 8 Days, Ci	1.7	2.9(-2)	4.8(-2)	5.9(-3)
2. Average Release Rate, Ci/cc	2.1(-1)	3.7(-3)	6.1(-3)	7.4(-4)
3. Percent of Quarterly Technical Specification Limit, %	*	*	*	*

*Percent of technical specification limits for iodines includes particulates with half lives greater than 8 days.

TABLE 4B

Total Annual Liquid Radiological Effluents

	1978			
A. Fission and Activation Products	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
1. Total Release, Ci	1.6	6.4(-1)	1.5	2.3
2. Average Diluted Concecentrations Ci/sec	2.8(-9)	1.1(-9)	2.6(-9)	4.0(-9)
3. Percent of Applicable Limit, %	2.8	1.1	2.6	4.0
B. Tritium				
1. Total Release, Ci	8.4(+1)	1.1(+2)	1.4(+2)	1.3(+2)
2. Average Diluted Concentrations, Ci/ml	1.5(-7)	1.8(-7)	2.3(-7)	2.2(-7)
3. Percent of Applicable Limit, %	4.8(-3)	6.1(-3)	7.8(-3)	7.4(-3)
C. Dissolved and Entrained Gases				
1. Total Release, Ci	3.67	2.3	2.1(+1)	1.3
2. Average Diluted Concentrations, Ci/ml	6.3(-9)	4.0(-9)	3.6(-8)	2.2(-9)
3. Percent of Applicable Limit, %	2.1(-1)	1.4(-1)	1.2	7.4(-2)

TABLE 4B (continued)

Total Annual Liquid Radiological Effluents

	1979			
A. Fission and Activation Products	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr
1. Total Release, Ci	1.4	9.7(-1)	9.1(-1)	4.5
2. Average Diluted Concecentrations Ci/sec	2.6(-9)	1.7(-9)	1.7(-9)	7.7(-9)
3. Percent of Applicable Limit, %	2.6	1.7	1.7	7.7
B. Tritium				
1. Total Release, Ci	1.8(+2)	1.2(+2)	1.3(+2)	1.3(+2)
2. Average Diluted Concentrations, Ci/ml	2.4(-7)	2.1(-7)	2.4(-7)	3.1(-7)
3. Percent of Applicable Limit, %	8.0(-3)	7.0(-3)	7.9(-3)	1.0(-2)
C. Dissolved and Entrained Gases				
1. Total Release, Ci	4.4	3.0	6.0	2.3
2. Average Diluted Concentrations, Ci/ml	8.3(-9)	5.1(-9)	1.1(-8)	3.8(-9)
3. Percent of Applicable Limit, %	2.8(-1)	1.7(-1)	3.6(-1)	1.3(-1)

Table 4C

1978 Gamma Immersion Dose at
the Site Boundary in mrem

Location	1st Qtr.	2nd Qtr.	3rd Qtr.	4th Qtr.	Total
SE	9.37E-2	2.92E-3	2.53E-3	3.91E-3	1.03E-1
SSE	6.03E-2	2.93E-3	3.01E-3	4.99E-3	7.12E-2
S	4.83E-2	3.08E-3	1.88E-3	5.31E-3	5.86E-2
SSW	3.34E-2	3.06E-3	1.48E-3	5.69E-3	4.36E-2
SW	1.41E-2	2.29E-3	2.70E-3	1.88E-3	2.10E-2
WSW	1.64E-2	1.70E-3	1.62E-3	1.77E-3	2.15E-2
W	1.26E-2	1.41E-3	3.90E-3	9.27E-4	1.88E-2
WNW	1.41E-2	1.19E-3	4.41E-3	1.70E-3	2.14E-2
NW	1.40E-2	1.20E-3	2.32E-3	7.26E-4	1.82E-2

1978 Total Body Dose in mrem or Liquid Release

1st Qtr.	2nd Qtr.	3rd Qtr.	4th Qtr.	TOTAL
3.48E-3	1.87E-3	2.81E-3	5.12E-3	1.33E-2

Table 4C (continued)
1979 Gamma Immersion Dose at
the Site Boundary in mrem

Location	1st Qtr.	2nd Qtr	3rd Qtr	4th Qtr.	Total
SE	2.58E-2	8.86E-4	1.83E-3	6.88E-3	3.54E-2
SSE	5.53E-3	1.17E-3	3.26E-3	1.82E-3	1.18E-2
S	2.84E-3	1.21E-3	6.69E-3	1.20E-3	1.19E-2
SSW	4.53E-3	8.21E-4	2.00E-2	1.06E-3	2.64E-2
SW	1.69E-3	4.75E-4	2.21E-2	5.46E-4	2.48E-2
WSW	2.02E-3	5.27E-4	1.34E-2	5.00E-5	1.60E-2
W	2.54E-3	5.56E-4	9.79E-3	3.49E-5	1.29E-2
WNW	3.35E-3	1.07E-3	4.55E-2	1.70E-4	5.01E-2
NW	3.88E-3	1.75E-3	4.62E-3	2.35E-4	1.05E-2

1979 Total Body Dose in mrem for Liquid Releases

1st Qtr.	2nd Qtr.	3rd Qtr.	4th Qtr.	TOTAL
3.55E-3	1.20E-3	2.58E-3	7.55E-3	1.49E-2

TABLE 5

Calculated Reactor Coolant System Fission Product Specific Activities for the Annual Cycle

Isotope	Concentration in Reactor Coolant (μCi/cc)
Br-84	0.0466
Kr-85m	1.49
Kr-85	0.885
Kr-87	0.81
Kr-88	2.6
Rb-88	2.55
Rb-89	0.064
Sr-89	5.07×10^{-3}
Sr-90	2.61×10^{-4}
Y-90	1.02×10^{-3}
Sr-91	3.56×10^{-3}
Y-91	0.111
Zr-95	9.35×10^{-7}
Mo-99	2.03
Ru-103	4.13×10^{-3}
Ru-106	2.48×10^{-4}
Te-129	0.0251
I-129	7.21×10^{-8}
I-131	3.97
Xe-131m	1.48
Te-132	0.33
I-132	1.09
I-133	5.66
Xe-133	181
Te-134	0.0262
I-134	0.62
Cs-134	0.10
I-135	2.7
Xe-135	7.53
Cs-136	2.55×10^{-2}
Cs-137	0.32
Xe-138	0.36
Cs-138	0.69
Ba-140	6.11×10^{-3}
La-140	5.85×10^{-3}
Ce-144	0.0040
Pr-143	0.0058

TABLE 6

Activity Release to Spent Fuel Pool Water

Isotope	FSAR Activity Release (Curies) 14 Outer Rods	176 Rods (Complete Assembly)	Percentage Increase for Extended Fuel Cycle*
Kr-85	4.90×10^{2}	2.86×10^{3}	40.5
Xe-131m	6.03×10^{1}	3.20×10^{2}	1.7
Xe-133	1.42×10^{4}	7.34×10^{4}	0.0
I-131	1.20×10^{4}	6.55×10^{4}	1.9

* Increase in specific activities for fuel assemblies at 43,200 MWD/T relative to fuel assemblies at 28,800 MWD/T.

TABLE 7

Comparison of the Calculated LOCA Release Source Terms (Curies)

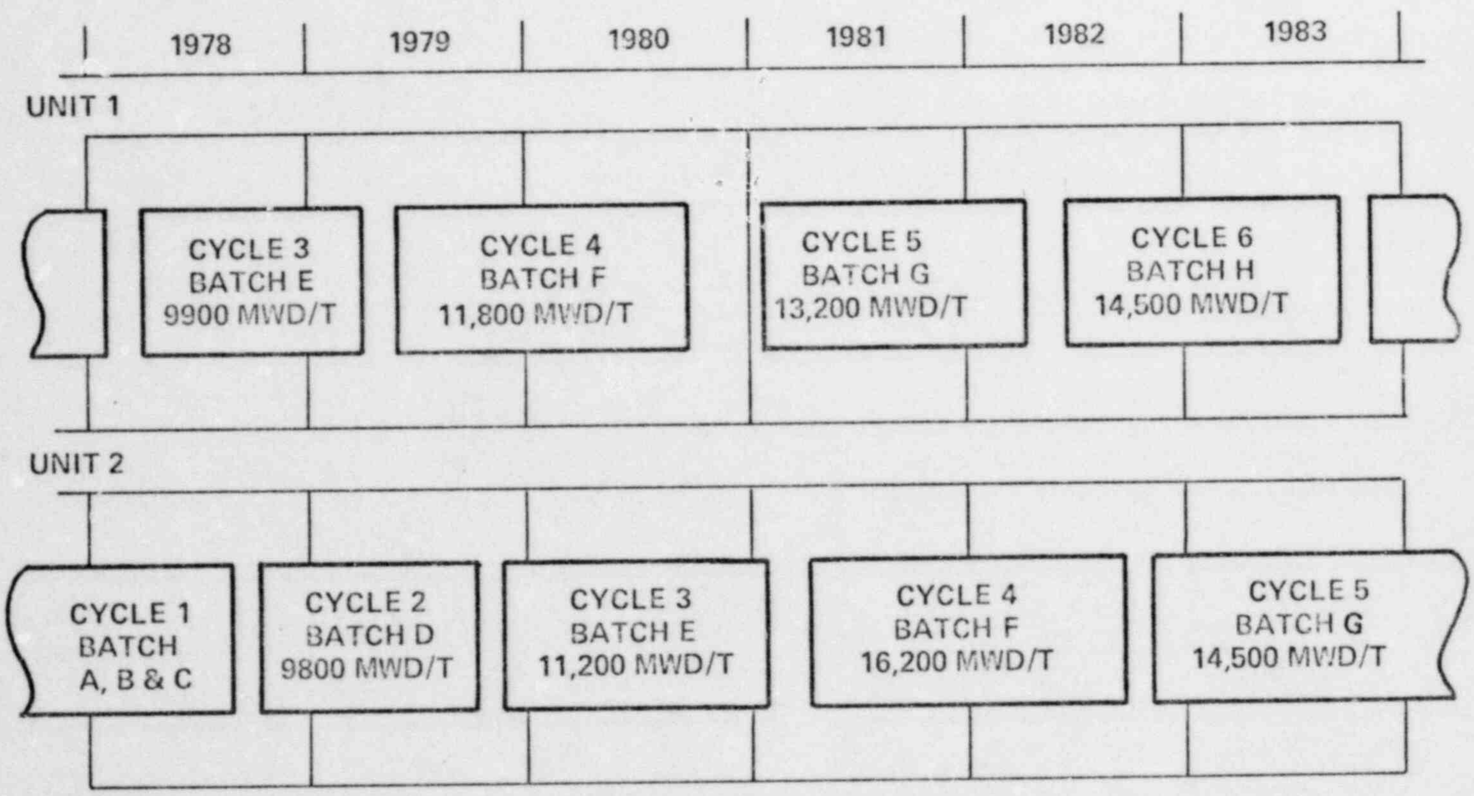
Isotope	Core Inventory for the Annual Cycle*	Core Inventory for the Eighteen Month Fuel Cycle **	% Change
I-131	7.38+7	7.48+7	+1.3
I-132	1.08+8	1.09+8	+0.9
I-133	1.54+8	1.53+8	-0.6
I-134	1.66+8	1.64+8	-1.2
I-135	1.43+8	1.42+8	-0.7
Kr-85m	2.05+7	1.97+7	-3.9
Kr-85	5.17+5	7.51+5	+45.3
Kr-87	3.81+7	3.62+7	-5.0
Kr-88	5.43+7	5.19+7	-4.4
Xe-131m	5.20+5	5.24+5	+0.8
Xe-133	1.53+8	1.53+8	0.0
Xe-135m	3.00+7	3.05+7	+1.7
Xe-135	3.00+7	2.97+7	-1.0
Xe-138	1.25+8	1.22+8	-2.4

* Core average burnup at end of cycle is 19,200 MWD/T

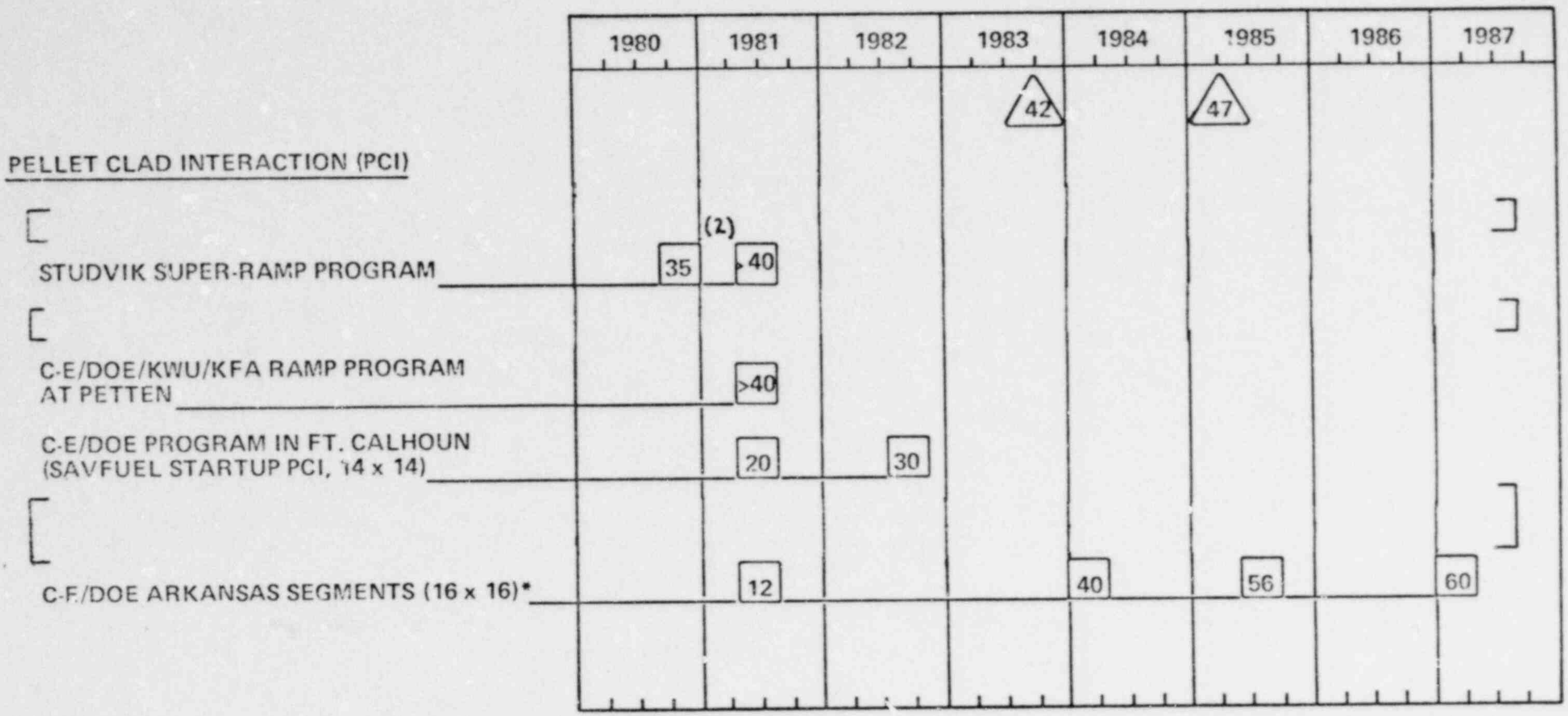
** Core average burnup at end of cycle is 28,800 MWD/T

Figure 1

CALVERT CLIFFS UNITS 1 & 2 PLANNED OPERATING SCHEDULE



BURNUP MILESTONES FOR FUEL IRRADIATION TESTS



*THESE PROGRAMS INCLUDE SEGMENTS AS SEEDS FOR POSSIBLE FUTURE RAMP TESTING

(1) STUDSVIK OVER-RAMP PROJECT FINAL REPORT, STOR-37, PROPRIETARY, TO BE PUBLISHED (REF. 2)

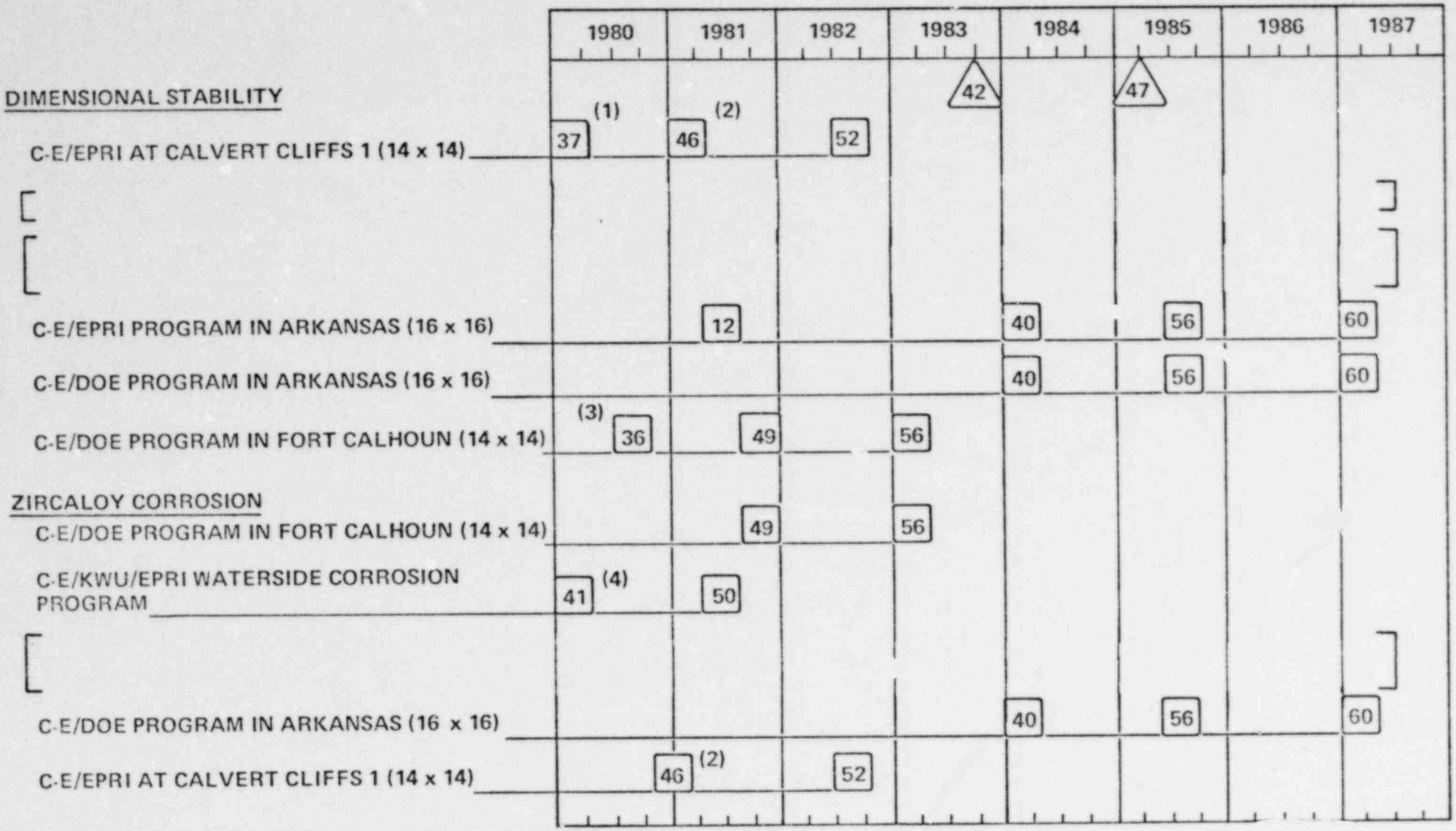
(2) PROPRIETARY DATA NOW AVAILABLE

[xx] CONSERVATIVE ESTIMATE OF DATA AVAILABILITY AT ESTIMATED BURNUP (GWD/T)

△xx CALVERT CLIFFS LEAD ROD BURNUP IN DISCHARGED BATCH (GWD/T)

Figure 2

BURNUP MILESTONES FOR FUEL IRRADIATION TESTS



(1) CE/EPRI REPORT NPSD-87, EXAMINATION OF CALVERT CLIFFS 1 TEST FUEL ASSEMBLY AFTER CYCLE 3, SEPT. 1979 (REF. 3)

(2) DATA UNDER EVALUATION

(3) CE/DOE REPORT CEND-383, FORT CALHOUN POOLSIDE INSPECTION PROGRAMS END-OF-CYCLES 4 AND 5, TO BE PUBLISHED (REF. 4)

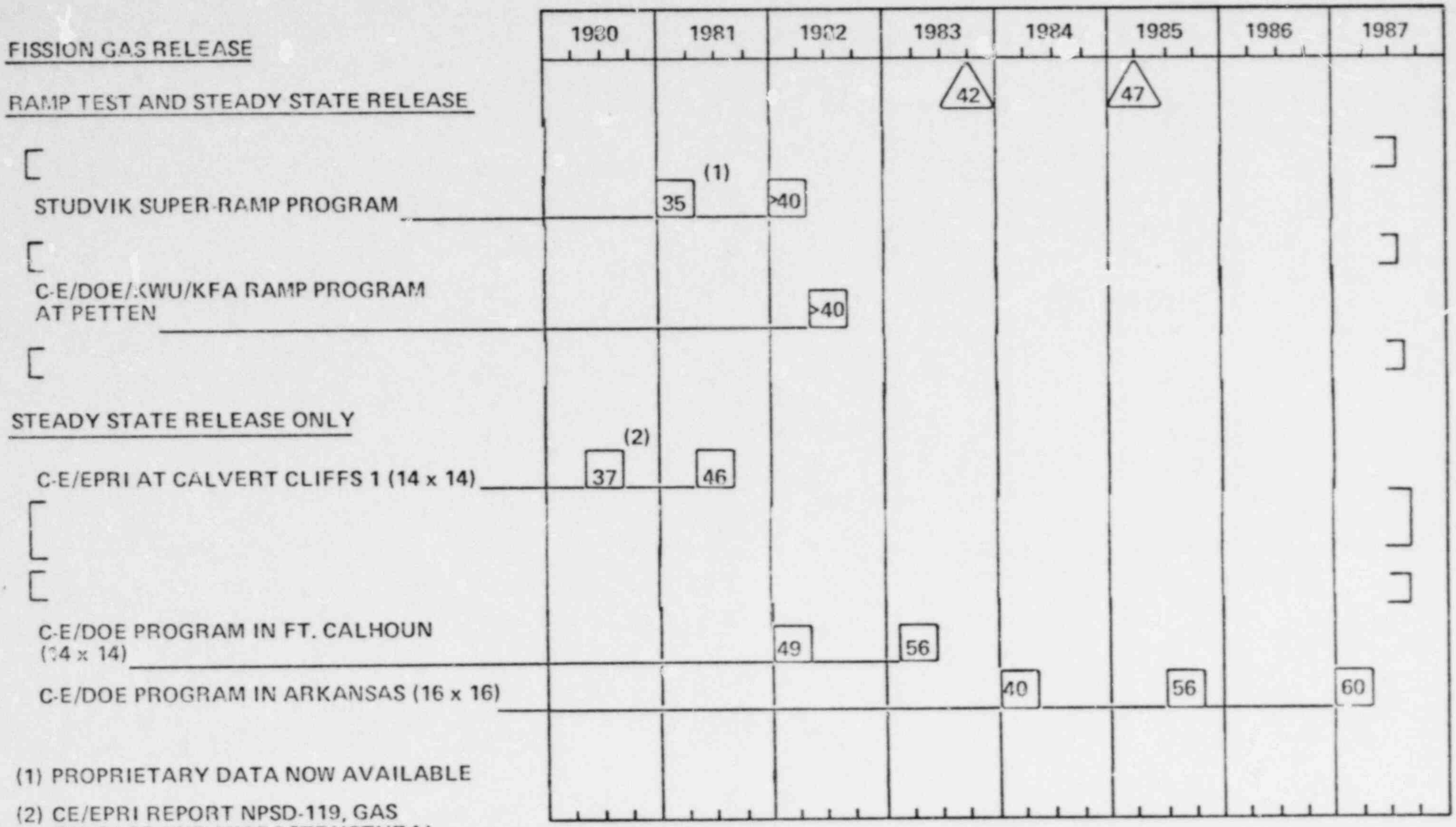
(4) (REFS. 5 and 6)

xx CONSERVATIVE ESTIMATE OF DATA AVAILABILITY AT ESTIMATED BURNUP (GWD/T)

xx CALVERT CLIFFS LEAD ROD BURNUP IN DISCHARGED BATCH (GWD/T)

Figure 3

BURNUP MILESTONES FOR FUEL IRRADIATION TESTS



(1) PROPRIETARY DATA NOW AVAILABLE

(2) CE/EPRI REPORT NPSD-119, GAS RELEASE AND MICROSTRUCTURAL EVALUATION OF THREE-CYCLE FUEL RODS FROM CALVERT CLIFFS - 1, DEC., 1980 (REF. 7)

xx CONSERVATIVE ESTIMATE OF DATA AVAILABILITY AT ESTIMATED BURNUP (GWD/T)

xx CALVERT CLIFFS LEAD ROD BURNUP IN DISCHARGED BATCH (GWD/T)

Figure 4

Figure 5

DECAY HEAT vs COOLDOWN TIME

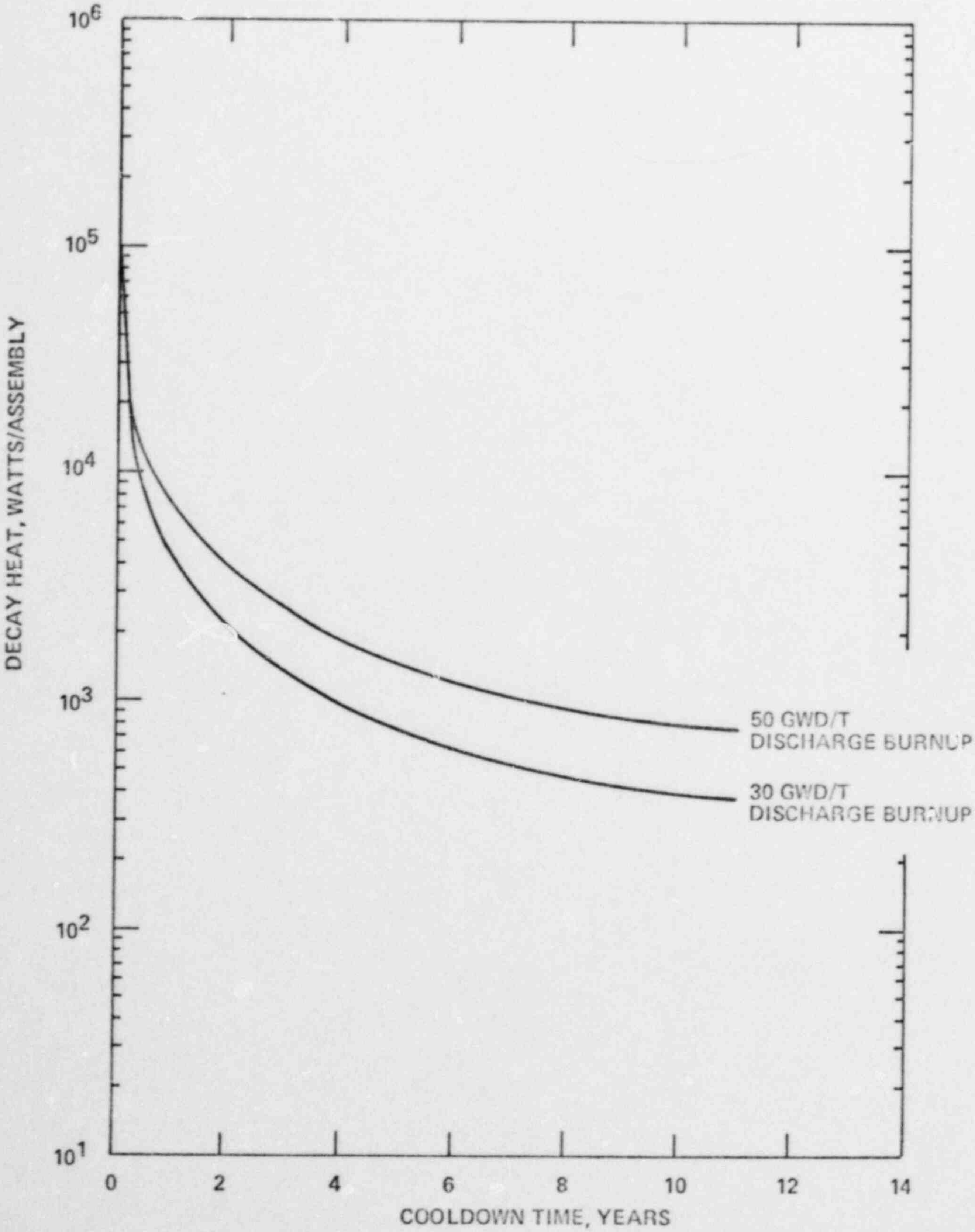


Figure 6

GAMMA ENERGY FOR LOW BURNUP FUEL AND EXTENDED BURNUP FUEL vs COOLDOWN TIME