**SHORT-TERM DYNAMIC EFFECTS
ON THE UTILIZATION OF PLUTONIUM
AS A REACTOR FUEL**

JAMES R. SHEFF

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OF PLUTONIUM AS A REACTOR FUEL

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INTRODUCTION

With the availability in the near future of reactor fuels containing plutonium enrichment, it is desirable to estimate quantitatively the effects produced on several of the characteristic parameters of the reactor associated with its control and inherent safety. Calculations have been made of the average delayed neutron fractions, the average delayed neutron decay constants, the mean neutron lifetime, the prompt neutron decay constant, and a parameter representative of the control strength required for fuel burnup. These calculations were done for two specific reactor types-- a Boiling Water Reactor (BWR) and a Pressurized Water Reactor (PWR); for two basic levels of plutonium enrichment; and for two reactor fueling modes. The results were obtained as a function of goal fuel exposure.

DISCUSSION

Efficient analysis of a wide range of reactor systems requires a simplified model of the systems. Suitable simplifications, which are not tolerable for the reactor designer, are frequently quite adequate for the intercomparisons of an optimization study.

In this study of the short-term dynamics of plutonium-containing reactors, the simplifications are 1) that a zero space-dimensional model is adequate, and 2) that the probable actual fueling situations can be bracketed by two limiting fueling schemes.

As a consequence of the first simplification, only homogeneous representations of the system are possible. The limiting fueling schemes considered (Assumption 2) are batch and fully-graded. Batch means that fuel of only one age is in the reactor at a time, while fully-graded means that fuels of all ages, between initial loading and goal exposure, are in the

reactor in equal amounts at all times. These two schemes are limits of the actual case where at any time in the reactor there is fuel of a number of discrete ages which also is usually located heterogeneously.

The effect of the heterogeneous placement can be quite important, as shown by Figure 1, which gives the k_{excess} for an infinite array of groups containing three fuel assemblies each. The center of each group is new fuel and the adjacent assemblies are one-third and two-thirds burned fuel. The larger the number of rods in an assembly, the more heterogeneous is the array. The magnitude of the effect depends also upon the enrichment.

Heterogeneous placement of the fuel will probably require heterogeneous placement of the controls, and will affect all of the parameters associated with the kinetic behavior of the systems. These effects have not been included in the calculations.

Two reactors are considered at this time--a 400 MW_e BWR (Oyster Creek), and a 1000 MW_e PWR (Indian Point). The properties of these reactors are representative of the current state-of-the-art for such reactors.

The average delayed neutron fraction will be discussed first, followed by the mean neutron lifetime, the prompt decay constant, the average delayed neutron decay constants, and the amount of reactivity excess required for burnup.

As is frequently the case in zero-dimensional survey calculation, the descriptions for the figures which follow vary in a cyclic and quite repetitious fashion over the pattern of the systems studied. In such cases, obvious advantages accrue if the notation in titles of the figures can be kept short by use of a selected list of adjectives to describe the systems. Following is such a list for this study.

1. Type of Fuel

- Pure-U. Reactor is initially fueled with U-235 enriched U-238.
- Recycle-Pu. Reactor is at first partially fueled with plutonium at an amount about equal to that found in discharged fuel of the type above. The remainder is U-235 enriched U-238.

FIGURE 1. Effect of Fuel Element Assembly Size on Initial k_{∞} for a Graded Fuel Loading

- Pu-Nat U. The reactor is initially fueled with plutonium-enriched natural uranium.

2. Fueling Scheme

- Batch. The loaded reactor is operated to some point, shutdown and all of the fuel replaced in a batch fashion.
- Graded. This is a limiting approximation to the case where fuel of several discrete ages is kept in the system at any given time. It is assumed that fuel of all ages is present in a manner which allows for complete interaction of the neutrons. This is analogous to the analytical concept of continuous refueling of the reactor.

The abbreviations used for the plutonium fuel compositions will be introduced and discussed at a later point (p. 10).

AVERAGE DELAYED NEUTRON FRACTION

The average delayed neutron fraction for a fuel mixture is of interest because it represents in reactivity units the difference between the delayed and prompt critical states. The latter, of course, must be avoided in operation, and thus the delayed neutron fraction is important information for evaluation of both reactor designs and reactor fueling concepts. The scope of the problem can be illustrated by inspection of the isotopic delayed neutron fractions, shown in Table I. The quantities shown in parenthesis in Table I are estimates based on either the known value in the opposite energy group or on an experimental correlation.⁽¹⁾ The comparison usually made is that between U-235 ($\beta = 0.0065$) and Pu-239 ($\beta = 0.0021$). Clearly, this comparison is an oversimplification of the problem and exaggerates the effect. In most U-235 systems, there is plutonium buildup, and in plutonium systems, there is usually at least the U-235 in depleted uranium as well as Pu-241 and U-238. Because of its large β and the large quantities usually present, the U-238 provides a substantial boost to the effective β for both uranium and plutonium systems if the neutron spectrum is reasonably hard.

1. G. Robert Keepin. "Physics of Nuclear Kinetics," Addison-Wesley Publishing Company, 1965.

TABLE I
 DELAYED NEUTRON FRACTIONS⁽¹⁾

	<u>Thermal</u>	<u>Fast</u>		<u>Thermal</u>	<u>Fast</u>
U-233	0.0027	0.0027	Pu-238	(0.0017)	(0.0016)
U-235	0.0065	0.0064	Pu-239	0.0021	0.0020
U-238	-----	0.0148	Pu-240	(0.0027)	0.0027
Th-232	-----	0.0203	Pu-241	0.0049	(0.0049)
			Pu-242	(0.0095)	(0.0088)

The average beta, Equation (1), is obtained by weighting the individual isotopic betas by the relative isotopic neutron generation rates.

$$\bar{\beta} = \frac{\sum_{ij} \beta_{ij} \nu_{ij} \sigma_f^{ij} \phi_j}{\sum_{ij} \nu_{ij} \sigma_f^{ij} \phi_j} \quad (1)$$

where the quantities have their usual meanings. The summation over i is over the range of isotopes present and the summation over j is over the energy groups present. This representation is correct in the sense that it correctly preserves the definition, "the fraction of neutrons emitted as delayed," and in the sense that it correctly preserves the form of the reactor equations when the usual definitions of a macroscopic fission cross section and an average $\bar{\nu}$ value are used for multicomponent situations.

This definition is also fully consistent with an interpretation in terms of the dynamic behavior of the reactor. This is true even with respect to the concept of "importance" weighting unless the energy mesh is made sufficiently fine that the neutrons are not all born in a single high-energy group. Even then, it is still a very good approximation.

For a typical case, Figure 2 shows the delayed neutron fraction, β , averaged over the reactor composition via Equation (1), as a function of reactor operation time expressed as fuel exposure in MWd/MT. Three curves are shown: one is initially pure U-235 enrichment; the second is enriched U-235 plus approximately that amount of recycle plutonium available from a previous core; and the third is plutonium enrichment of natural uranium. The percent isotopic concentrations for the plutonium, shown in parentheses, are representative of first recycle plutonium from a power reactor. The fuel concentrations, and hence the initial multiplication constants, were adjusted in this case to yield identical final average multiplication constants at 35,000 MWd/MT. The data shown in Figure 2 are for graded operation.

The two most prominent features in Figure 2 are the wide variation of the delayed neutron fraction with exposure for Pure-U as opposed to the small variation in the Pu-Nat U fuel, and the closeness of all three curves near the end of core life. The Pure-U and the Recycle-Pu curves differ insignificantly throughout the latter half of fuel life.

The choice of exactly what information to pick to adequately describe the system in terms of a single global parameter is somewhat tricky. For batch operation the initial and final values of β will be used since they provide boundaries, but in the limit of fully graded operation the appropriate value would be an average over the core life. In terms of the realistic case of a small number of fuelings per full core cycle, one should proceed as follows: If the number of fuelings per core cycle is, for example, four, and one takes a linear average of the β 's, that is, an average for constant flux, the resulting β behaves as shown by the dashed line in Figure 2 for the Pure-U case, and as shown by the adjacent portion of the plutonium curves for the plutonium cases. The latter situation results because the dashed curves would be superimposed over the solid curves since the plutonium curves are almost linear. One notes that the β at midexposure is a reasonable approximation to consider as a global parameter and that the variation in β for the Pure-U case is considerably greater than for the plutonium cases. Completely ignored both in this calculation involving

FIGURE 2. Delayed Neutron Fraction for a Simulated BWR

linear averaging and also in the basic underlying calculations are the almost certain effects of inhomogeneity and the effects produced by fuel shuffling. The consideration of such effects would require one-dimensional calculations, probably a scheme of flux flattening through control, and then certainly flux and probably flux-adjoint flux weighting of the results from each region. Some efforts along this path are planned for the future.

Shown in Figure 3 are the major isotopic contributions to the three curves of Figure 2. Of interest are the significant contributions produced in all three cases by the fast effect in U-238 and the contribution of Pu-241 in the case of Pu-Nat U.

Before proceeding to look further at the results, it is appropriate to discuss the scope of and reasons for the plutonium isotopic compositions used in this study. At the same time, a system of notation is introduced to simplify reference to these compositions.

The plutonium compositions will be written as Pu(a, b, c, d, e) where a, b, c, d, and e will be the percent of the isotopes of atomic mass 239, 240, 241, 242, and 238 respectively. Zeros at the end of the list will be dropped. In this notation, this study will include the following isotopic compositions:

1. Pu (92, 8)
2. Pu (62, 23, 11, 3, 1)
3. Pu (48, 25, 15, 10, 2)
4. Pu (100)
5. Pu (80, 20)
6. Pu (75, 20, 5)
7. Pu (70, 20, 5, 5)
8. Pu (55, 15, 15, 15)
9. Pu (40, 30, 15, 15)

The first three first compositions were chosen because they represent closely the general compositions found at the end of life in low exposure U-235 enriched fuel, high exposure U-235 enriched fuel, and high exposure of recycle level plutonium--U-235 enriched fuel. The other compositions are, in

FIGURE 3. Isotopic Contributions to the Delayed Neutron Fraction for a Simulated BWR

general, reasonably close to one of the first three in overall compositions, but were picked such that through comparisons of pairs one can try to develop generalizations upon the trends of individual isotopes. For example, comparison of 4 and 5 provides information about the effect of Pu-240, comparison of 5 and 6 differs only in replacement of part of the fissile Pu-239 by Pu-240, and 8 and 9 show the effect of Pu-239 being replaced by Pu-240 in a high Pu-241 and Pu-242 environment.

In an attempt to maintain clarity, the graphs will, in several cases, not contain the complete list of isotopes. Some will be relegated to a few comments in the description as to their behavior with respect to those shown.

The fully graded fuel cycle will be considered first for both the 400 MW_e boiling water reactor and the 1000 MW_e pressurized water reactor. All three fueling schemes, pure U-235 enrichment, recycle level plutonium mixed with U-235, and plutonium enrichment of natural uranium, are included.

The average delayed neutron fraction is shown as a function of goal fuel exposure in Figure 4 for the BWR. The points plotted are like the average β 's on Figure 2. The top curve is for the Pure-U feed, and the two lower bands encompass the various plutonium compositions studied. The upper band is for the Recycle-Pu feed, while the lower band is Pu-Nat U data. The Pure-U delayed neutron fraction is only about 6% greater than that for Recycle-Pu and only 30% greater than that for Pu-Nat U.

A corresponding graph for the PWR would be qualitatively the same as Figure 4 with only minor quantitative differences. That this is the case can be seen by detailed inspection of Figures A1 and A2 in the Appendix. The uranium curves overlap, and the plutonium bands are slightly broader and encompass those of the BWR.

The details of the BWR plutonium bands in Figure 4 shown in Figures 5 and 6 for Recycle-Pu and Pu-Nat U respectively. Wherever possible, a standard notation was adopted for the graph lines. The notations, however, were also included if possible to avoid the need for continuous and repeated checking of the list which follows.

FIGURE 4. Average Delayed Neutron Fraction as a Function of Exposure (Graded fueling of the BWR)

FIGURE 5. Average Delayed Neutron Fraction for the Case of Recycle-Pu (Graded fueling of the BWR)

FIGURE 6. Average Delayed Neutron Fraction for the Case of Pu-Nat U. (Graded fueling of the BWR)

In Figure 5, one sees first that the spread among the various curves is only 6%, a fact which results primarily from the long exposure, the graded operation, and the relatively small amounts of plutonium present initially in recycle level plutonium operation. Although this small range of only 6% precludes any serious justification for adjusting isotopic concentrations to achieve an improved average beta, a brief attempt at understanding the effects observed seems worth while. The β average is increased by the increased Pu-240 concentration, which leads more quickly to near equilibrium quantities of Pu-241 with its larger beta. This is the dominating effect observed in the Pu(100), Pu(92, 8), Pu(80, 20), and Pu(75, 20, 5) series of curves. The difference between the Pu(75, 25, 5) and Pu(70, 20, 5, 5) curves however is a result of the poisoning effect of Pu-242 which increases the U-235 enrichment required and hence increases the U-235 contribution with respect to the plutonium contributions. The beta for Pu(62, 23, 11, 3, 1) was greater than that for either Pu(70, 20, 5, 5) or Pu(75, 20, 5) because of the larger Pu-241 contributions, and was in part also greater than the latter because of the larger enrichment required to overcome the Pu-242. The same ideas apply to the other remaining curves shown as well as to the Pu(55, 15, 15, 15) curve which falls in between Pu(48, 25, 15, 10, 2) and Pu(62, 23, 11, 3, 1).

In Figure 6, the U-235 concentration is fixed while the plutonium varies to achieve the desired exposure. Then, in contrast to Figure 5, the effect of greater enrichment is a reduction in beta. Thus, the enrichment effect and the high Pu-241 effect compete in Figure 6. At an exposure of

35,000 MWd/MT the lower three curves are seen to be as in Figure 5, but at 15,000 MWd/MT exposure, these effects essentially cancel. The Pu(75, 20, 5) and Pu(70, 20, 5, 5) curves are inverted in Figure 6 as expected, and likewise the Pu(40, 30, 15, 15) curve is considerably lower relative to the others.

The similarity between the BWR and PWR obtains also for the results of Figure 5 and 6. In fact the results for the PWR are qualitatively identical to these figures, and the actual values on any pair of corresponding curves agree to within about 4% at the outside. The PWR results are shown in Appendix, Figures A1 and A2.

Then, analogous to Figure 4 above, one has for a batch fueling mode in the 400 MW_e BWR the initial-loading-and final-exposure-delayed neutron fractions as shown in Figure 7. The initial delayed neutron fractions are lower than that for enriched uranium fueling by 17 and 50% for the cases of Recycle-Pu and Pu-Nat U respectively. As would be expected, the final betas for Recycle-Pu are essentially the same as that for Pure U. The Pu-Nat U delayed neutron fractions at the final exposure are down by 25% from that of pure uranium. The general increase in the delayed neutron fraction throughout the burnup for the Pu-Nat U case obtains because of the initial nonequilibrium plutonium compositions with respect to Pu-241.

The detailed effects produced by changes in isotopic composition are in general very similar in form to those described above. Figures 8 and 9 correspond to Figures 5 and 6 and in the Appendix, Figures A3 and A4 correspond to A1 and A2. In Figures 9 and A4, it is interesting to note that although the mid-exposure beta for Pu(100), Pu(92, 8), and Pu(80, 20) shows the same relative behavior as in Figures 6 and A2 the initial and final betas are not in the Figures 6 and A2 order. Instead, one finds the ranges over which the betas change from initial to final exposure are in the same order as Figures 6 and A2, the smallest range being that of the Pu(100) case. Similar behavior is also encountered for the pair Pu(70, 20, 5, 5) and Pu(75, 20, 5).

FIGURE 7. Initial and Final Delayed Neutron Fractions for a Simulated BWR Using Batch Fueling as a Function of Fuel Exposure and Plutonium Concentration

FIGURE 8. Initial and Final Delayed Neutron Fractions Simulated
BWR - Batch Burnup, Recycle Level Plutonium

FIGURE 9. Initial and Final Delayed Neutron Fractions - Batch Burnup for Pu-Nat U Fueling in a Simulated BWR

AVERAGE PROMPT NEUTRON LIFETIME AND PROMPT DECAY CONSTANT

Although the delayed neutron fraction itself gives the required information about the separation in reactivity units between prompt and delayed critical, to assess the behavior of a reactor in a transient state requires additional information in the form of either the prompt neutron lifetime or generation time. With this information, one then obtains the prompt decay constant which, combined with a given input excess reactivity in dollars, is a measure of the speed with which a super prompt critical transient would take place. That is, a measure of how rapidly the flux or power level is changing. In conjunction with temperature coefficients, one can then evaluate to some degree the relative merits of various systems. One may include also, in such an evaluation information about external preventative measures or engineered safeguards.

Since the actual lifetime calculation was obtained in a zero-dimensional analysis, it excludes core leakage effects. It is an infinite prompt neutron lifetime. This is not a serious limitation because (1) only comparisons between similar systems are to be made, and (2) the leakage effects are reasonably small in large systems. A more serious limitation may be the exclusion of adjoint weighting with respect to the energy variable, but for the calculational test cases considered, this also proved to be of secondary importance. This is so because in survey calculations errors of perhaps 10 to 15% are acceptable when comparisons are being made. A reactor designer clearly cannot in general accept such errors if their elimination is reasonably feasible.

In a simple mono-energetic representation of nuclear reactor systems, the infinite medium prompt neutron mean lifetime is

$$\ell^* = \frac{1}{v \Sigma_a} \quad (2)$$

In the more-realistic multi-energy group representation, a single global-average lifetime in some sense must be obtained. Intuitively, one is

immediately lead to a neutron-density reciprocally weighted average value as given in Equation (3)

$$\ell = \frac{\int n dv}{\int n v \Sigma_a dv} \quad (3)$$

Intuitively, as well as rigorously, it would also be desirable to include adjoint or importance weighting into the definition, but this has not been done in the current study. Available in the code used is the average absorption cross section in the Westcott notation, Σ_a^* , which is directly related to the desired quantity by

$$\ell = \frac{1}{v_{2200} \Sigma_a^*} \quad (4)$$

As with the delayed neutron fraction this quantity is a function of fuel exposure and the fuel isotopic composition.

The lifetime as a function of fuel exposure is shown in Figure 10 for Pure-U fueling and Recycle-Pu fueling. These particular data are for batch loadings having two final exposures, but they represent the general effects encountered. The behavior is certainly nonlinear, but since the variation is reasonably small, particularly for the plutonium cases, the value of ℓ at the mid exposure was picked for comparisons of the graded cases. A weighted reciprocal average would be more accurate, but for comparison purposes the mid-exposure point seemed adequate.

The lifetime calculations for the case of the simulated BWR, graded burnup, and for both Recycle-Pu and Pu-Nat U are shown in Figures 11 and 12. As with β , the plutonium curves all fall below that of the uranium fueled case, but their order is predominantly reversed from that for the β 's. This is because an increase in the higher isotopes not only increases the β , but also increases the neutron absorption considerably since the higher isotopes have large absorption cross sections and include the fertile isotopes Pu-240 and Pu-242. Hence the prompt neutron lifetimes for those plutonium curves having higher isotopes are less than those containing mostly Pu-239.

FIGURE 10. The Infinite Prompt Neutron Lifetime as a Function of Fuel Exposure for Pure-U and Recycle-Pu Fueling

FIGURE 11. Average Prompt Neutron Lifetime for the Case of Recycle-Pu (Graded fueling of the BWR)

FIGURE 12. Average Prompt Neutron Lifetime Simulated BWR - Graded Burnup (Plutonium mixtures in Nat-U)

Analogous to and qualitatively similar to Figures 11 and 12, the neutron lifetime calculations for the case of the simulated PWR operated on a graded fuel cycle are presented in Appendix figures A5 and A6.

For the case of batch burnups, the choice of a single parameter to describe the lifetime becomes quite difficult as can be seen by inspection of Figure 10. However to remain somewhat consistent with the practice introduced above for the delayed neutron fractions shown in Figure 13 are the initial, mid, and final exposure values for the same two fuels used in Figure 10. Namely Pure-U and Recycle-Pu (48, 25, 15, 10, 2). The curves for the missing compositions fill the gaps between each pair (i. e., initial, mid, or final) of representative curves in the same order as already observed in Figures 11 and 12.

PROMPT DECAY CONSTANT

The prompt decay constant, $\alpha = \frac{\beta}{\lambda}$, provides a measure of the transient behavior of the system from a safety point of view. In conjunction with the temperature coefficients which provide the transient shutdown mechanism one obtains a general feel for the overall system behavior in an excursion.

The prompt decay constant is shown in Figure 14 for several representative cases, namely Pure-U fueling and Recycle-Pu (48, 25, 15, 10, 2) fueling for the two final batch exposures shown. As above, the value of a mid-exposure will be used in the comparisons for the particular case of graded studies.

Figures 15 and 16 show the behavior of α for the BWR (A7 and A8 for the PWR) operating upon a graded fuel cycle with Recycle-Pu and Pu-Nat U. The plutonium concentrations representative of first and second plutonium recycle, Pu (62, 23, 11, 3, 1) and Pu (48, 25, 15, 10, 2), are seen to have larger average prompt decay constants than the Pure-U case. Thus for an equal reactivity input in dollars above prompt critical, the Recycle-Pu system would be on a smaller period. As mentioned above, however, one must also have information about the shut down mechanisms such as for example the temperature coefficients in order to speculate about which system would

FIGURE 13. Initial, Middle, and Final Exposure Prompt Neutron Lifetimes as a Function of Final Fuel Exposure (Batch fueling of the BWR)

FIGURE 14. Prompt Decay Constant as a Function of Exposure for the Case of Batch Operation

FIGURE 15. Average Prompt Decay Constant as a Function of Final Fuel Exposure (Graded Recycle-Pu operation of a BWR)

FIGURE 16. Average Prompt Decay Constant Simulated BWR - Graded Burnup (Plutonium mixtures in Nat. U)

have the less serious transient. Figure 19 shows that the plutonium Doppler coefficients are generally larger in magnitude than that of the uranium case which will tend to offset the effect of a larger decay constant.

Generally, the ordering of the curves, except for pure uranium, is the same as that observed for the delayed neutron fractions. The almost inverse ordering of the lifetime curves producing a spreading effect.

In Appendix A, Figures A9 to A12 show the initial and final prompt decay constants for batch fuel exposure for both the PWR and the BWR. The effects are qualitatively identical to those observed for graded operation above.

REACTIVITY REQUIREMENTS

One characteristic that is of prime importance with respect to the ease of control and to a degree economy of a reactor operating on a graded fuel cycle is the average reactivity change per unit fuel exposure. Multiplication of this quantity by the expected exposure in a single period of a given fueling of the fuel cycle gives an estimate of the reactivity which must be controlled. Again, accuracy of the result is restricted by the assumptions listed above.

Figures 17 and 18 show this characteristic for the two BWR fueling situations considered for graded fuel loadings. The curves falling below that of uranium are generally those containing larger percentages of the higher isotopes, and are the Phoenix fuels. The parallel PWR curves are shown in Figures A13 and A14.

AVERAGE DELAYED NEUTRON DECAY CONSTANT

The average delayed neutron decay constant is another potentially interesting global quantity because of its importance to the time behavior of the system in transient changes between delayed and prompt critical. In particular, it represents the asymptotic inverse period of a reactor having positive reactivity of $\beta/2$. In the range of reactivities between delayed and prompt critical, the transient behavior depends in most cases to a very good approximation only upon the mean delayed neutron lifetime, $1/\lambda$, and not upon the mean prompt neutron lifetime, ℓ .

FIGURE 17. Reactivity Change per Unit Fuel Exposure as a Function of Final Fuel Exposure (Graded recycle-Pu fueling of a BWR)

FIGURE 18. Reactivity Change per Unit Fuel Exposure as a Function of Final Fuel Exposure (Graded Pu-Nat U Fueling of a BWR)

In a straightforward manner, it can be shown that the desirable average delayed neutron lifetime is

$$\frac{1}{\lambda} = \frac{\sum_{ij} \frac{\beta_{ij}}{\lambda_{ij}} v_{ij} \sigma_f^{ij} \phi_j}{\bar{\beta}} \quad (5)$$

Where the notation of Equation (1) has been used.

The results of calculations do verify the above comment concerning the relative size of λ and $1/\lambda$ for the systems considered. Other than this, the quantity turns out to be extremely insensitive to the parameters under investigation. It is a constant to within about 5 to 10% over the fuel isotopes considered and varies only slightly ($\leq 2\%$) with respect to the exposure changes considered.

DOPPLER COEFFICIENT

The Doppler coefficient is one important quantity which aids in evaluating the meaning which should be attached to changes in the prompt decay constant. This quantity for a 100 °C fuel temperature change is part of the usual ALTHAEA output at the initial and goal exposure points.

An example of the results is shown in Figure 19 for the graded exposure case of the PWR operating in the Recycle-Pu mode.

While the curves for the initial exposure case are seen to vary in a uniform systematic manner, the final exposure curves clearly show rather strange behavior. The effects are not yet explained. It is possible that the computations are perhaps not fully converged, but this seems unlikely because of the uniformity of the initial exposure curves.

Despite the possible inaccuracies it seems reasonable to conclude that the magnitude of Doppler coefficients for the plutonium cases are larger by up to about 15% than that of the uranium case.

FIGURE 19. Doppler Coefficient for the PWR Operating in a Recycle-Pu Mode for Fully Graded Exposures

CONCLUSIONS

The primary conclusion of this report is that from a kinetics point of view the effects encountered in plutonium fueling, particularly at a recycle level, will present no serious barrier to the operation of systems such as those studied herein. The effects noted were in general quantitatively small and relevant pairs of effects frequently would tend to cancel. Although the quantitative differences would be of interest to a designer, it is not clear that they are sufficiently large to drastically affect adversely conclusions concerning the desirability of a system on the basis economics or conservation. For example, it was found that although the average delayed neutron fraction is smaller for plutonium systems, the problem is not nearly as serious as is often implied when one examines more realistically the reactors state during burnup rather than comparing unburned fuel values or pure isotope values.

Of vital interest is the effect of relaxing the restrictions which are present in this study. It seems probable that this will not greatly affect the results. The effects of introducing heterogeneity will be minimized in part by the usual flux flattening requirements in such regions. It also can be anticipated that importance weighting will probably tend to reduce the changes seen over fuel lifetime while leaving the average values about the same.

A second restriction upon the conclusions is the narrow range of fuels studied and the small number of reactors studied. Not a great deal can be said about either of these effects which would either build or reduce confidence in the primary conclusions given above. For example, for the thorium system, one will have only about one-third as many fast fissions, producing 20% more delayed neutrons than U-238. Thus the net effect will be that the thorium system will get a base contribution to the delayed neutron fraction of only about one-half as much as the U-238 system.

Finally, the principal effects observed in this study were explainable in terms known system properties which then permits speculation about systems other than those investigated.

APPENDIX

The features differentiating the BWR and PWR behavior were very largely quantitative. To shorten the main body of the report for those not interested in the fine quantitative details the results for the PWR are grouped herein.

Also included to a lesser extent are some of the qualitatively similar BWR curves.

FIGURE A1. Average Delayed Neutron Fraction for the Case of Recycle-Pu (Graded fueling of the PWR)

FIGURE A2. Average Delayed Neutron Fraction for the Case of Pu-Nat (Graded fueling of the PWR)

FIGURE A3. Initial and Final Delayed Neutron Fractions
(Batch burnup for Recycle-Pu fueling in a
simulated PWR)

FIGURE A4. Initial and Final Delayed Neutron Fractions Batch Burnup for Pu-Nat (U Fueling in a Simulated PWR)

FIGURE A5. Average Prompt Neutron Lifetime for Recycle-Pu in the PWR (Graded fueling was used.)

FIGURE A6. Average Delayed Neutron Lifetime for Pu-Nat U in the PWR (Graded fueling assumed)

FIGURE A7. Average Prompt Decay Constant as a Function of Final Fuel Exposure (Graded Recycle-Pu operation of a simulated PWR)

FIGURE A8. Average Prompt Decay Constant as a Function of Final Fuel Exposure (Graded Pu-Nat U operation of a simulated PWR)

FIGURE A9. Decay Constant at Initial and Final Burnup Times as a Function of Fuel Exposure and Plutonium Isotopic Composition (Batch fuel loading and recycle level plutonium in uranium)

FIGURE A10. Prompt Decay Constants at Initial and Final Burn-up Times as a Function of Fuel Exposure and Plutonium Isotopic Composition (Batch fuel loading of plutonium enriched Nat-U.)

FIGURE A11. Prompt Decay Constant at Initial and Final Burn-up Times as a Function of Fuel Exposure (Batch fuel loading of a PWR using Recycle-Pu)

FIGURE A12. Prompt Decay Constant at Initial and Final Burn-up Times as a Function of Fuel Exposure (Batch fuel loading of a PWR using Pu-Nat U)

FIGURE A13. Reactivity Change per Unit Fuel Exposure as a Function of Final Reactivity for Recycle-Pu Fueling (Graded loading of a PWR)

FIGURE A14. Reactivity Change per Unit Fuel Exposure as a Function of Final Fuel Exposure (Graded Pu-Nat fueling of a PWR)

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