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RE-EVALUATION OF THE 1962 MARK VI-B DYNAMIC TESTS REGARDING REACTIVITY COEFFICIENTS (U)

by N. T. Hightower, III, et al.

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by

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ABSTRACT

RE-EVALUATION OF THE 1962 MARK VI-B DYNAMIC TESTS
REGARDING REACTIVITY COEFFICIENTS

by
Toby Burnett
N.T. Hightower III

The 1962 Mark VI-B rod tests have been re-evaluated. A significantly lower (more negative) coolant coefficient is now ascribed to that test, -1 pcm/Deg-C vs the previously reported +2 pcm/Deg-C. The change from the previous value is because of revisions to delayed neutron constants and accounting for spatial effects. The new value is in reasonable agreement with the currently calculated value of -2 pcm/Deg-C, considering measurement and calculational uncertainties. Therefore, we conclude that the current analytic models for physics and transient analysis methodology (GLASS, GRIMHX & TRIMHX, AND MARY) are fully consistent with 1962 test observations, and that there is no basis for assigning a calculational bias or increasing uncertainty allowances.

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1. INTRODUCTION

In 1962, transient tests were conducted in the L-6 charge (composed almost entirely of Mark VI-B assemblies), in which special "delta-k rods" were used to introduce a small, rapid reactivity perturbation (ref 1). The change in measured flux was used to infer a set of reactivity coefficients. A +2 pcm/Deg-C coolant coefficient, when combined with the point kinetics and thermal kinetics models in the BURP code (a predecessor of AA3), gave the closest fit to the observed flux response in the first few seconds after the reactivity perturbation.

As part of establishing the Reactor Physics and Safety Analysis methodology in 1989-90, SRL committed to revisiting this 1962 test with current calculational methods used for physics and for transient analysis (ref 2), for the purpose of justifying the uncertainty assigned to current calculations of reactivity coefficients.

Current GLASS calculations (ref 3) predict an effective coolant coefficient of -2.3 pcm/Deg-C (fuel, coolant, and target are computed as 0.0, -2.2, and -0.1 pcm/Deg-C, respectively).

The purpose of this report is to document the re-evaluation of the 1962 test results, and compare with the predictions based on current analytic models.

2. SUMMARY

Current calculations are fully consistent with the 1962 VI-B test measurements. In fact, the difference between current test inference and calculations is smaller than might be expected based on various uncertainties in test and analytic results.

We must emphasize that the 1962 tests did NOT measure reactivity coefficients. The test measured the transient response to a reactivity perturbation of unknown magnitude. Reactivity coefficients can be inferred from the test results, based on analytic models for the transient response and spatial effects. Errors in these models -- or in the test measurements -- would cause error in the inferred coefficient. The 1962 tests were not accurate enough to establish an uncertainty in the combination of analytic models. They were, however, useful to assist in confirming the uncertainty assigned to the analysis methodology based on other considerations.

The best fit to the 1962 test data is for a coolant coefficient of -1.3 pcm/Deg-C, or about 1 pcm/Deg-C more positive than the current calculated coefficient. This 1 pcm/Deg-C difference is the sum of numerous error contributors, several of which could cause the difference by themselves. In any event, 1 pcm/Deg-C is less than the model uncertainty assigned to calculation of prompt reactivity coefficients.

Sources of error include:

Test Measurement and Documentation: Test Conditions (power, temperature, rod pattern & power shapes); detector location; timing of reactivity perturbation; noise in flux signals and accuracy of reading them; plotting inaccuracies.

Analytic Models:

Reactivity coefficient calculations (GLASS & GRIMHX);

Spatial redistribution (TRIMHX) to compare detector and core average response;

Dynamic models:

Delayed neutron constants, flow and power sharing, moderator flow patterns, region-wise power depositions, spatial feedback weighting.

The current coolant coefficient inferred from the 1962 tests is 3 pcm/Deg-C more negative than historic inferred values. The more negative value is caused by:

- o Revised delayed neutron constants (this effect alone decreases the apparent coolant coefficient by roughly 2 pcm/Deg-C).
- o Spatial effects (decreases the apparent coolant coefficient by about 1 pcm/Deg-C).
- o Improved transient models (the calculated fuel and target coefficient are essentially zero for the Mark VI-B, so the improved transient model makes little difference).

Both the size of the reactivity insertion, Δk , and the feedback effects are unknowns that must be inferred from the test observations. The current inferred results, compared to the 1962 inferences, are tabulated below for the size of the reactivity insertion and the coolant coefficient:

Exposure (MWD)	Δk , pcm		Coolant Coef, pcm/Deg-C	
	ref 1	Present	ref 1	Present
14,000	28.5	36.8	+1.7	-1.3
91,000	28.5	35.7	+0.5	-2.4
199,000	32.5	39.5	-2.0	-4.7

These test results give a good fit to the data -- better than had been obtained in the 1962 data reduction. However, there are still several areas of significant uncertainty, primarily regarding the delayed neutron data, test measurement accuracy and the spatial effects. Based on engineering judgement and the sensitivity studies reported herein, the uncertainty is estimated to be roughly 1.5 pcm/Deg-C, in addition to the roughly 1 to 1.5 pcm/Deg-C uncertainty in the calculated coefficient. Thus, the total expected measurement and calculational error are significantly larger than the difference. Further quantification is not justified by the quality of the 1962 test results.

We therefore conclude that the current analytic models -- the combination of physics and transient methodology (GRASS, GRIMHX & TRIMHX, and MARY) -- are fully consistent with the 1962 test observations, and that there is no basis for assigning a calculational bias or increasing uncertainty allowances.

Additional evaluation of the 1962 tests is recommended as part of designing any new delta-k rod tests. Further such studies will help establish test measurements and techniques to minimize test uncertainty.

3. DISCUSSION

3.1 CALCULATIONAL PROCEDURE

GENERAL

In the 1962 tests, a small, local, rapid reactivity insertion was made near the middle of the core, and the resulting flux monitored on the ex-core power-range neutron monitors. The shape of the curve is determined by:

1. The size of the reactivity insertion (unknown)
2. The reactivity feedback coefficients (unknown)
3. The neutronics; e.g., delayed neutron constants (assumed to be known, but less well known than expected)
4. The relation between the core average power and the observed flux (assumed to be constant in the original evaluation, and analytically determined with TRIMHX in this evaluation)
5. The thermal kinetics (simulated with the BURP code in 1962, and with the MARY code (ref 12 and 13) in this evaluation)

Unfortunately, the 1962 tests did not measure the magnitude of the reactivity insertion (from zero-power tests), nor were before-and-after core power distribution measurements recorded, nor were accurate thermal power measurements from before and after the perturbation made. Therefore, data reduction is merely selection of a reactivity insertion and feedback coefficients to fit the shape of the curve.

DELAYED NEUTRON EFFECTS

The historic delayed neutron data used at SRS is the Keepin data for the six U-235 groups (ref 7), plus several of the photoneutron groups (with two of the shorter photoneutron groups combined with the U-235 groups as a mid-frequency approximation.) Recent estimates of delayed neutron constants have lead to marked revisions in the shortest U-235 group (ref 4). For example, the yield of the shortest delayed neutron group has been increased by a factor of four. These changes significantly affect the calculated shape of the neutron flux response to a step reactivity change. Since the Mark VI-B coolant coefficient was inferred by analyzing the shape of the observed flux response, the delayed neutron constants have a significant effect.

The base case in this report uses the delayed neutron constants for thermal fission in U-235 from the Brady-England paper (1989). The effect of other sets of delayed neutron constants, such as ENDF/B-VI, are discussed in Appendix B, and Appendix A reports results of sensitivity studies with various assumptions concerning delayed neutrons.

SPATIAL EFFECTS

The delta-k rods were located near the center of the reactor core. Therefore, the relative core power increased more in the center, and less at the periphery. The ex-core power range flux monitor will also increase less than the average core power. To the extent that reactivity is associated with the core average power, the change in the relation between the flux detector and the core power must be considered.

Bill Graves has calculated the ratio, (core average power)/(leakage flux), for the two detector locations which may have been used for the test measurement (ref 5). These came from TRIMHX runs for 0 to 2.1 seconds. This ratio is also referred to as "tilt". As expected, the tilt showed a prompt jump, followed by an slower increase as the initial delayed neutron precursor distribution decayed. The behavior of the tilt, and the core average power as calculated from the observed detector response, is in Appendix C, and several sensitivity cases are reported in Appendix A.

TRANSIENT MODEL

The L-6 charge was approximated based on available information from the references (ref 6). Although the Mark VI-B assembly is similar to a Mark 22 assembly, the L-6 core had some significant differences relevant to thermal kinetics from a Mark 22 core.

The L-6 core had about 20% more assemblies than typical Mark 22 cores, and the flow per assembly was also lower. However, the flow area in the Mark VI-B is less than in the Mark 22, giving a higher coolant velocity and better heat transfer from fuel to coolant. Compared to a Mark 22 fuel assembly, the Mark VI-B has lower fuel temperatures, a faster fuel thermal time constant, and a faster coolant thermal time constant. At equal core powers, both charges have similar coolant temperatures.

REGRESSION ANALYSIS

The MARY code permits inputting the core nuclear power, and having the reactivity calculated -- as well as the transient temperatures of the various regions. This feature was used, and a regression analysis run (Lotus 1-2-3) to correlate change in temperature in the coolant and moderator upflow region to the reactivity.

Data was available from the 1962 tests only for the first 10 seconds. The regression analysis used only the period from 0.2 to 5 seconds, to minimize the effects of feedback from the moderator downflow region. For the purposes of data reduction, the axial weighting was set equal to the value used for safety analysis of the K-14.1 core. The effort to obtain a calculated value specific for the beginning of the L-6 charge is not justified by the quality of the 1962 data.

3.2 POTENTIAL ERROR CONTRIBUTORS

We approach the discussion of error contributions with a sense of humility. In 1962 -- and up until 1991 -- no one voiced any suspicion that errors in effective delayed neutron constants could cause a shift of 2 pcm/Deg-C in the apparent coolant temperature coefficient. We are reluctant to claim that no significant further error could be caused by some unexpected source.

However, those potential sources of error that we can identify are discussed below.

The values of uncertainty and error given in this report are those which are not believed likely to be exceeded. They can therefore be considered as corresponding to a confidence level of 95%, although such mathematical precision may imply more rigor than is warranted. In the few places where possible (e.g., scatter in regression analysis), the uncertainty was taken as two standard deviations.

DELAYED NEUTRON CONSTANTS

Details of delayed neutron constants and their effects are discussed in Appendices A and B to this report. Despite the advances in delayed neutron data, we are not confident that adequate accuracy has been obtained. Error contribution comes from several potential sources.

First, it is not clear which are the "best" set of delayed neutron constants. The status on this topic is discussed in detail in Appendix B. Second, there are literally hundreds of delayed neutron precursors, which are -- by convention -- collapsed into six sets. The shape of neutron flux within the first few seconds may not be adequately described by the six groups (only two of which have time constants of one second or less). Note that the evaluation is pegged to the prompt jump between 0 and 0.2 seconds, and there may be delayed neutron precursors not properly accounted for at that short time. Third, each of the delayed neutron constants has its own uncertainty, which tends to increase for the shorter-lived precursors. Again, the shorter-lived precursors have a significant effect within the first few seconds.

As an estimate, further uncertainties in delayed neutron constants, as they would affect the shape of the flux response curve in the interval 0.2 to 5 seconds, are considered to be in the magnitude of 1 pcm/Deg-C.

SPATIAL EFFECTS

The TRIMHX calculations and model are subject to some uncertainty, but the reasonably-expected amount appears pretty small from the results in Appendices A and C, probably no more than a few tenths of a pcm/Deg-C. TRIMHX has been compared to SRS reactor tests and the results published in refereed journals (ref 14). However, there is no experience base in using it for relating detector response to core average power for spatial transients. Therefore, much larger errors than expected (non-Gaussian) are possible.

MEASUREMENT UNCERTAINTIES

In doing a 1991 analysis, one wishes that more test measurements were available from the 1962 tests. These areas are noted here for the dual objectives of estimating measurement error, and suggesting improvements if similar tests are made in the future.

Delta-k rod tests should also be made at zero power to determine the response without reactivity feedback (looking only at the reactivity worth of the delta-k rods and the delayed neutron parameters). Comparison of those results with the at-power results would isolate the feedback effects and eliminate most of the uncertainty due to delayed neutron constants and dk. Accurate initial and final power and temperature measurements would have permitted a check of the overall power coefficient. Multiple detectors, with different geometries with respect to the delta-k rods should be used for diversity and self-checking of the spatial effects.

The actual power at the time of the 1962 beginning of charge test is uncertain. Reference 1 reports the core power as 2075 MW in the summary, and 2125 MW in reference 6.

Measurement error includes: (a) scatter in flux data points, including plotting and digitizing error; (b) uncertainty in timing and shape of reactivity insertion; and (c) uncertainty in initial test conditions; and (d) detector location. Considering the sensitivity cases reported in Appendix A and the standard error in the coolant coefficient from the regression analysis, measurement error (narrowly defined as these four components) is estimated to be in the range of 0.5 to 1 pcm/Deg-C.

TRANSIENT T&H MODEL

This area considers all deviations between the MARY T&H model and the actual L-6 reactor charge. A significant uncertainty is the moderator flow patterns (the extent of the moderator upflow region, the fraction of assembly effluent rising into the moderator upflow region, and the lateral flow out of the upflow region into the downflow region). For safety analyses, conservative parameters can be selected. For this study, best-estimate (really, best-guess) parameters were used. Other uncertainty contributors are the weighting functions for axial and radial variations in coolant and moderator temperatures, and (to a much smaller extent) the thermal kinetics in the channel. (It's hard to make an error in the channel thermal kinetics that makes a significant difference in reactivity feedback when the fuel and target coefficients are negligible.) We estimate the transient model also has an uncertainty in the range of 0.5 to 1 pcm/Deg-C.

Since the measurement error and the transient T&H model used to evaluate the data are tightly coupled, we estimate the combined uncertainty (measurement plus transient T&H model) as roughly 1 pcm/Deg-C.

CALCULATED COEFFICIENTS

Two types of error exist in the calculations for reactivity coefficients: model uncertainties and parameter uncertainties. Model uncertainties refer to approximations in the diffusion and transport models used in the GLASS and GRIMHX computer codes, and errors in the cross-section libraries on which they're based. Parameter uncertainties refer to errors caused by differences between model input and physical parameters, specifically including manufacturing tolerances (dimensions and isotopic concentrations), heavy water purity, flux shape, burnup uncertainty, and so on.

Parameter uncertainty in calculated coefficients for safety analysis can be quite large, since these calculations must bound variations encountered in actual operation. For the K-14.1 charge, 2.5 pcm/Deg-C has been established as the parameter uncertainty for the effective coolant coefficient for the K-14.1 charge at beginning of cycle; i.e., from -2.8 to -7.8 pcm/Deg-C (see Table 2 of ref 8). For post hoc examination of test conditions, parameter uncertainty should be significantly less, even considering the poorer manufacturing control of lithium contents in use for the 1962 L-6 charge. Greg Cefus reports (ref 3) that, for his calculation of the VI-B temperature coefficients, he had to adjust the lithium contents by nearly 10% in order to match the reported critical rod position. He apportioned the discrepancy equally between the inner and outer target. However, other apportionments, or other parameter variations, may have caused the reactivity discrepancy, and other variations may not have had the same influence on temperature coefficients. On the basis of engineering judgement, an uncertainty of 0.4 pcm/Deg-C is considered by Reactor Physics to be a reasonable estimate for the uncertainty due to parameter variation in the L-6 charge.

The GLASS computer code, and the cross-section data which it uses, have been in use at Savannah River for many years. Based on subcritical isothermal tests, recent comparisons with other codes, and engineering judgement, errors in the effective coolant temperature coefficient are believed to be about 1 pcm/Deg-C (ref 10 and 11).

The total uncertainty in the calculated coefficient, due to L-6 parameter uncertainty and calculational uncertainty, is therefore expected to be in the range of 1 to 1.5 pcm/Deg-C.

TOTAL UNCERTAINTY

The total of the above contributors to error appears to be in the neighborhood of 2 pcm/Deg-C. (The RMS value of the error contributors is 1.8 to 2.2 pcm/Deg-C, and the possibility exists that two of the major contributors could be simultaneously near their limit of maximum expected error.) Therefore, if the discrepancy between test observation and calculation were significantly more than 2 pcm/Deg-C, additional effort would be justified to identify potential sources. Since the deviation is only about 1 pcm/Deg-C (and comparable to the uncertainty allowance in the calculated coefficient alone), further effort is not considered justified.

4. RESULTS

The results of the regression analysis for the coolant and moderator upflow region are shown in the table below. The fit is surprisingly good, based on the R-squared value, considering the scatter in the data. Little credence should be placed in the value of the moderator upflow coefficient (except perhaps in the trend). The short time period, and uncertainties in the moderator flow patterns and in the axial flux shape and weighting, prevent any conclusions regarding the absolute value of the moderator upflow coefficient.

Regression Results for 0.2-5 Sec Fit to Coolant and Mod Upflow

	14,000 MWD	91,000 MWD	199,000 MWD
	-----	-----	-----
Estimated Delta-k, pcm	36.8	35.7	39.5
Std Err of Est dk, pcm	0.56	0.32	0.49
R Squared	0.989	0.997	0.994
No. of Observations	17	16	19
Degrees of Freedom	14	13	16
Coolant Coef, pcm/Deg-C	-1.27	-2.36	-4.65
Std Err of Coef.	0.36	0.22	0.33
Mod Upflow Coef, pcm/Deg-C	-8.05	-9.74	-12.86
Std Err of Coef	0.34	0.22	0.30

The results are shown graphically on Figures 1 through 6. The top graph of Figures 1 to 3 show the core power (calculated from observed detector response), and the reactivity needed to produce that power. Figures 4 to 6 show the calculated temperature response (for the core power shown in the top graph of Figures 1 to 3) of the coolant, fuel, target, and moderator upflow. (These are weighted average temperatures over the axial and radial direction, as is done for SAR safety analysis.) These temperatures were used in the regression analysis. The bottom graph of Figures 1 to 3 is a check solution, in which the calculated reactivity and coefficients were put into MARY to compute a power transient.

Two curves are shown on the bottom graph of Figure 1, corresponding to the best fit coefficients and the calculated coefficients. (The apparent reactivity addition was 36.8 and 38.1 pcm for the best fit and for the calculated coefficients, respectively.) The best fit coefficients are clearly a better fit, at least between 0.2 and 0.5 seconds. The difference, however, could be attributed to either error in the combination of measurement plus T&H transient model used to evaluate the data, or to error in the calculated coefficients, or even to error in the effective delayed neutron constants, or to some combination of these.

5. CONCLUSIONS

The current coolant coefficient inferred from the 1962 tests is 3 pcm/Deg-C more negative than historic inferred values. The more negative value is caused by:

- o Revised delayed neutron constants. Estimated delayed neutron parameters have shifted considerably in the last two decades (ref 4). In particular, the shortest delayed neutron group fractions have increased considerably (increasing the total beta for U-235 from .0065 to .0073). This has little effect on safety analysis, but has a marked effect on estimating the feedback effects from a reactivity step of unknown magnitude. The change in delayed neutron constants alone decreases (makes more negative) the inferred coolant coefficient by 2 pcm/Deg-C.
- o Spatial effects. Prior evaluations of the 1962 tests assumed that the detector response was proportional to core power. That's not strictly true. The delta-k perturbation was near the center of the core, changing the relation between detector and core average power by (probably) about 0.7% (current TRIMHX calculation). That's a far bigger effect than it appears. For example, where the detector reported a 4.3% increase, the core average power probably increased 4.3% plus 0.7%, or 5.0% -- at least a 15% larger change than previously considered. This effect decreases the inferred coolant coefficient by about 1 pcm/Deg-C.
- o Improved transient models. Current transient models in the MARY code permit explicit modeling of each tube in the fuel assembly, and also cross flow in the moderator tank. Because the fuel temperature changes both more and faster than does the coolant temperature, explicit modeling of fuel temperature feedback has been found necessary. In this particular instance, these improvements had no effect. The calculated fuel coefficient is zero for the Mark VI-B at beginning of cycle, and the calculated target coefficient is very small (-0.1 pcm/Deg-C), so the fuel and target tubes need not be explicitly modeled. Also, the quality of the data, and the lack of knowledge of the moderator flow patterns and axial weighting does not support accurate predictions of the moderator upflow and downflow coefficients.

We therefore conclude that the current analytic models -- the combination of physics and transient methodology (GRASS, GRIMHX & TRIMHX, and MARY) -- are fully consistent with the 1962 test observations, and that there is no basis for assigning a calculational bias or increasing uncertainty allowances.

Additional evaluation of the 1962 tests is recommended as part of designing any new delta-k rod tests. Further such studies will help establish test measurements and techniques to minimize test uncertainty. In particular, we recommend attempts to determine fuel, target, and moderator upflow and downflow coefficients from the longer-term test results. These attempts will be frustrating, but should provide valuable insight on the test data that should be recorded in future tests.

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FIGURE 1

FIGURE 2

FIGURE 3

FIGURE 4

FIGURE 5

APPENDIX A

SENSITIVITY STUDIES

A series of variations were considered, to observe the sensitivity in the inferred coolant coefficient to various assumptions. All of these cases considered the beginning of life test, and assigned a moderator downflow coefficient of -5.35 pcm/deg-C (This had a very small effect in the time period of interest.) The regression analysis was done to fit the data from 0.2 to 5 seconds. Except for case F, the reactivity was assumed to be added as a ramp between 0 and 0.1 seconds.)

SPATIAL CORRECTION VARIATIONS

The delta-k rods are located near the center of the reactor core. Therefore, the core power increased more in the center, and less at the periphery. The ex-core power range flux monitor will also increase less than the average core power. To the extent that reactivity is associated with the core average power, the change in the relation between the flux detector and the core power must be considered.

The correction factors, or "tilts", are discussed and calculated in Appendix C. The correction for the base case is

$$\text{Core total power as fraction of initial} = \{\text{Tilt (detector B)}\} \times \{\text{Detector B indication as fraction of initial}\}$$

$$\text{where Tilt (detector B)} = 1 + .0072 [1 - .098 * F(t)]$$

$$F(t) = \frac{\text{Delayed neutron relaxation function}}{\text{Sum}\{ \text{Beta}(i) * \text{Exp}[-\text{Lambda}(i)*t] \} / \text{Sum}\{ \text{Beta}(i) \}}$$

$$\text{Tilt (detector A)} = 1 + .00812 [1 - F(t)]$$

Correction factors for several cases are computed in Appendix C. These are:

Case

- A. Detector B and the Graves' (ref 5) TRIMHX factors (base case, see formula above). This case was taken as the reference for the sensitivity studies. It differs slightly from the early-in-life case cited on page 9 because of the inclusion of a moderator downflow coefficient (of -5.35 pcm/Deg-C).
- B. Detector A and the Graves' TRIMHX factors (see formula above).

- C. This case assumed a large inaccuracy in the TRIMHX calculation of the static tilt caused by the delta-k rod perturbation, and assumes the detector tilt fits the equation

$$\text{Tilt (detector)} = 1 + .015 [1 - .098 * F(t)]$$

(instead of $1 + .0072 [1 - .098 * F(t)]$).

- D. This case assumed that 30% of the tilt followed delayed neutron relaxation (instead of 9.8%); i.e.,

$$\text{Tilt (detector)} = 1 + .0072 [1 - .30 * F(t)]$$

- E. Combination of cases C and D, assuming the detector tilt fits the equation

$$\text{Tilt (detector)} = 1 + .015 [1 - .30 * F(t)]$$

These cases (A through E above) were used to evaluate the data from the 1962 test near beginning of charge life. The resulting reactivity coefficients are tabulated on Table A-1, and the results shown on Figures A-1 to A-5.

Use of detector "B" to evaluate the test data causes an inferred coolant coefficient 0.12 pcm/Deg-C more negative than detector "A". Doubling the detector tilt, to 15% (from 7.2%) causes the inferred coolant coefficient to be 1.1 pcm/Deg-C more negative, while increasing the effect of the delayed neutrons on the spatial transient leads to a smaller inferred coolant coefficient.

TIME OF REACTIVITY INSERTION

Case F assumed that the reactivity was added as a ramp between 0.1 and 0.2 seconds (instead of between 0 and 0.1 seconds). As shown on Figure A-6, and the statistics parameters on Table A-1, the fit is not good.

This case illustrates the sensitivity to the assumed time during which the reactivity was added.

EFFECT OF CHANGE IN DELAYED NEUTRON CONSTANTS

As discussed in Appendix B, the six-group delayed neutron constants have been revised significantly since 1962. The revision in these constants, more than any other factor, explains the previous large discrepancy between calculated and test-inferred values of the reactivity coefficient.

The reference case for delayed neutron constants assumed zero precursor concentration for the five longest-lived photo-neutron delayed neutron groups -- those with half-lives of 27 minutes or longer -- appropriate for a reactor that had started up rapidly and had just reached full power. (This is a common assumption in safety analysis when small delayed neutron fractions are conservative.) Case G assumed all delayed neutron precursors were in

equilibrium, and determined a slightly less negative (0.09 pcm/Deg-C) coolant temperature coefficient. Although Case G is more appropriate to the test conditions than is Case A, the difference is trivial.

Case H increased the shortest two delayed neutron group fractions by 30% and 10%, respectively. (The values were chosen arbitrarily.) The effect is to make the apparent coolant coefficient more negative by 0.54 pcm/Deg-C. This is significant, since the shortest delayed neutron group parameters are subject to the most uncertainty when collapsed from their current 271 groups down to 6.

Case I used the old delayed neutron constants; i.e., the Keepin data in vogue in the 1962 era. Note that these constants caused an apparent coolant coefficient 1.8 pcm/Deg-C more positive than the reference case with the newer delayed neutron constants.

Cases K and L considered the ENDF/B-VI delayed neutron constants for fast and thermal fissions, respectively (refer to Appendix B for discussion.) The thermal fission constants produce results virtually the same as those with the Brady-England constants. The fast fission constants give a noticeably less negative prompt coefficient.

The dependence on total beta is examined in Cases M, N, and O, with the conclusion that the inferred prompt coefficient is not highly sensitive to total beta (for small changes). Reducing all betas by 20% causes an inferred prompt coefficient 0.44 pcm/Deg-C less negative.

SCATTER IN THE DATA

All test data (detector response vs time) discussed in this report came from Bailey's 1962 report (ref 1), except that used for Case J. Bailey's report has hand sketched curves, and some additional noise is introduced in reading the values of the points on the curves, in addition to plotting errors in the curves themselves. For example, to the best of our ability to read the plots in reference 1, the data seems to be plotted at .42, .52, .60, .71, .79, etc seconds. The possibility exists that readings at 0.1 seconds were intended. The February 1962 Monthly Progress Report includes two better-quality plots, with apparently different values. These were digitized, and analyzed as case J. The results are virtually identical to case A, although the scatter is significantly less. Case J lends confidence that plotting/digitizing errors were not significant.

TABLE A-1 SENSITIVITY STUDIES

CASE --->	A	B	C	D	E
Estimated Delta-k, pcm	36.7	37.4	42.7	35.5	40.2
Std Err of Est dk, pcm	.55	.55	.60	.52	.56
R Squared	.988	.989	.990	.988	.989
Coolant Coefficient	-1.19	-1.33	-2.35	-0.67	-1.52
Std Err of Coef.	.35	.35	.35	.34	.33
Mod (up) Coefficient	-7.78	-7.79	-7.61	-7.79	-7.52
Std Err of Coef.	.33	.32	.32	.32	.31

CASE --->	F	G	H	I	J
Estimated Delta-k, pcm	39.6	36.8	38.2	32.7	36.5
Std Err of Est dk, pcm	1.24	.55	.55	.47	.38
R Squared	.955	.988	.990	.986	.992
Coolant Coefficient	-2.83	-1.10	-1.73	+0.63	-1.21
Std Err of Coef.	.65	.35	.35	.31	.22
Mod upflow Coefficient	-7.60	-7.78	-7.95	-7.10	-7.76
Std Err of Coef.	.74	.33	.33	.29	.20

- A - Reference 1962 VI-B test at 14,000 MWD
- B - Detector B assumed (the record was unclear as to whether detector "A" or "B" had been used)
- C - Assuming the detector-to-core-average tilt was 1.5%, rather than the 0.72% estimated from TRIMHX.
- D - Assuming delayed neutrons contributed 30% of the detector-to-core-average tilt, rather than the 9.8% predicted by TRIMHX.
- E - Combination of case C & D
- F - Assuming the reactivity from the delta-k rod was added between 0.1-0.2 seconds, instead of between 0-0.1 seconds.
- G - Equilibrium delayed neutrons (rather than only those with half-lives less than 30 minutes).
- H - As G, except shortest two delayed neutron groups increased by 30% and 10%.
- I - Old delayed neutron constants (the Keepin data in vogue in 1962)
- J - Data from Feb 1962 Monthly Progress Report rather than Bailey Report

TABLE A-1 SENSITIVITY STUDIES
(Continued)

CASE --->	K	L	M	N	O
Estimated Delta-k, pcm	34.8	34.8	36.8	32.9	29.1
Std Err of Est dk, pcm	.52	.52	.55	.52	.50
R Squared	.988	.988	.988	.987	.984
Coolant Coefficient	-0.22	-0.97	-1.10	-0.91	-0.66
Std Err of Coef.	.34	.34	.35	.33	.33
Mod (up) Coefficient	-7.98	-7.37	-7.79	-7.03	-6.29
Std Err of Coef.	.31	.32	.33	.31	.30

K - ENDF/B-VI (fast) delayed neutron constants, per WSRC-TR-91-115-Addendum (all 9 photoneutron groups included).

L - As case K, except thermal U-235 delayed neutrons constants used, per Brady-England 1989

M - Repeat of Case G (Brady-England constants, 9 photoneutron groups)

N - Same as Case M, except all betas reduced to 90%

O - Same as Case M, except all betas reduced to 80%

FIGURE A-1

FIGURE A-2

FIGURE A-3

FIGURE A-4

FIGURE A-5

FIGURE A-6

FIGURE A-7

FIGURE A-8

FIGURE A-9

FIGURE A-10

APPENDIX B

EFFECTS OF DELAYED NEUTRON CONSTANTS

EFFECT OF CHANGE IN DELAYED NEUTRON CONSTANTS

The quality and quantity of delayed neutron precursor data have greatly increased since 1962. However, we are not at all sure what constants are "correct".

Results cited in the main section of this report are based on the delayed neutron constants cited in the 1989 Brady-England paper (ref A1), and noted in WSRC-TR-115 (ref A2). Those estimates are based on 271 discrete precursors, plus additional parents, for a total of nearly 400 fission products. Constants for 6 groups were found by fitting curves of neutron emission after a fission pulse as calculated by the CINDER code. Comparisons were also made with dynamic tests done by Keepin (A3). The Brady-England work concluded that the fastest delayed neutron groups were significantly larger than estimated by Keepin (nearly four times higher for the fastest group). Differences in the four slowest groups were relatively minor. The total delayed neutron yield therefore was larger than Keepin's, but also with a larger uncertainty. (These data are based on the Brady-England calculated value of 1.78 as the number of delayed neutrons per 100 thermal fissions in U-235. The uncertainty in the calculated value of 1.78 stems from uncertainty in knowledge of the decay modes of the shorter-lived isotopes.)

However, the number of delayed neutrons per fission is considered to be a directly measured parameter, and the current consensus is that its value is 1.66 ± 0.05 per 100 thermal fissions in U-235 (ref A4), considering measurements as well as calculations. The ENDF/B-VI data set has 1.67 delayed neutrons per 100 fissions (the same as ENDF/B-V), which translates to a value of total beta of 0.006867. Delayed neutron yields for the six groups are obtained by multiplying this value of total beta by the fractions for each of the six groups (alpha values) in the Brady-England paper. The convention is to use fast fission values for both fast and thermal fissions because of the greater statistical precision in measurements with fission neutron spectrum (ref A5). (This practice comes from the format restrictions in the ENDF files. Only one data set can be included. By tradition, the fast fission set is included. However, it is technically appropriate to use the thermal data set for thermal fissions where differences in the results would be obtained.) In this manner, the fast groups of delayed neutrons appear to have increased at the expense of the slower four groups.

The ENDF/B-VI is therefore a 'hybrid', consisting of a total beta based on one set of considerations, and apportioned based on entirely different considerations. Neutron kinetics calculations with the ENDF/B-VI constants will be significantly different than with Keepin's original, and an even larger change from ENDF/B-V.

In power reactor work, reactivity coefficients are frequently measured by dynamic tests. A small reactivity perturbation is introduced, such as by

changing temperature or moving a control rod, and the reactor put on a long period. After spatial effects have died out, the stable period is measured and translated to a reactivity with inverse kinetics. For U-235 thermal fission, the measured reactivity worth (of whatever introduced the perturbation) would differ by about 15% depending on whether ENDF/B-V or ENDF/B-VI delayed neutron constants are used.

At this writing, ENDF/B-VI is in test use. Some concerns have been expressed over the large change from ENDF/B-V (ref A6). Based on telephone discussion with M. C. Brady (ref A7), Mod 1 is expected to be out next year, with improved fission product yields. Little or no change is expected in the total value of beta, and Dr Brady indicated no reason to expect significant change in the fractions (alphas) from her 1989 work with England.

WSRC-TR-91-115-Addendum lists four sets of delayed neutron constants for U-235. A fifth can be generated from the ENDF/B-VI total beta of 0.006867 and the Brady-England fractions for thermal fission (which would seem more appropriate to thermal fissions). These five sets, tabulated on Table B-1, are as follows:

1. Keepin set, used at the Savannah River Site for many years. Per reference A5, these are also based on fast fission constants and applied to thermal fissions.
2. Brady-England set (ref A1), using their alphas and lambdas for thermal fission in U-235, and their value of 1.78 for delayed neutrons per 100 thermal fissions in U-235.
3. ENDF/B-V data, using 1.67 delayed neutrons per 100 fissions.
4. ENDF/B-VI (fast), 1.67 delayed neutrons per 100 fissions, and apportioned between the six groups using the alphas for fast fission from the Brady-England work (the fast lambdas are also used). These are the only constants obtainable directly from ENDF/B-VI.
5. ENDF/B-VI (thermal), 1.67 delayed neutrons per 100 fission, and apportioned between the six groups using the alphas for thermal fission from the Brady-England work (the thermal lambdas are also used).

The point kinetics response to a small step change in reactivity for these five groups is shown on Figure B-1 and Table B-2. For short times, (less than 5 seconds), all five sets differ markedly. At long times (more than 5 seconds), the Brady-England set is virtually indistinguishable from the Keepin set. Also at long times, the fast and thermal ENDF/B-VI sets are similar. Other than those similarities, marked differences between the sets exist.

The response to a 200 pcm step change is also tabulated on Table B-2. Here, differences are larger. The fast and thermal ENDF/B-VI do not cross, as they did for a 40 pcm step.

Because of the way reactivity worths and coefficients are measured on commercial power plants, a comparison of reactivity vs period provides the most insight as to the difference between the various sets. The reactivity causing a given stable period is tabulated on Table B-3 for the five sets. For small

reactivity (such as used in power reactor experiments), the Keepin and Brady-England sets are virtually identical, the ENDF/B-V gives about 5% higher reactivity (or worth), and the ENDF/B-VI(fast) gives about 10% lower worth.

Note that switching from ENDF/B-V to ENDF/B-VI(fast) would give roughly a 15% reduction in the measured value of reactivity worths and coefficients -- a surprisingly large difference.

At this point, the author (me) must confess to not knowing what the "true" delayed neutron fractions are. I must confess to preferring the Brady-England set, on the following bases:

- a. They form a self-consistent set, with both the absolute and relative magnitudes of the delayed neutron yields based on the same considerations.
- b. For small reactivity changes, they indicate little change from Keepin (or even ENDF/B-V) response. That is, the older data has withstood the test of time, and it seems unreasonable to expect large errors in areas that are relatively easy to measure (small perturbations at long times).
- c. Intuitively, one would expect that more detailed information would most likely affect the short-lived delayed neutron precursors, increasing their expected yield. But also intuitively, one would expect that increase to come at the expense of prompt neutrons, rather than at the expense of the longer-lived groups.

For these reasons, I'll stick with the results based on the Brady-England constants. Realizing that the issue is in question and that other opinions differ, the results with the other sets are also reported, and the differences considered in estimating the total uncertainty.

(Also an uncertainty consideration are the following two points: (1) there are not six, but literally hundreds of precursors, and (2) uncertainties exist in the individual yields of the various precursors as well as in their total. These points are important to the 1962 Mark VI-B data evaluation since the entire process depends on the shape of the curve in the first few seconds.)

Table B-4 shows the inferred prompt coefficient from the 1962 Mark VI test with various delayed neutron constants. Other than the Keepin set, a significant difference is caused by use of the fast vs thermal ENDF/B-VI data. Since the Savannah River reactors are predominately thermal fission, the results for fast fission delayed neutron parameters would appear inappropriate.

The substantial difference is caused by the fact that the magnitude of the reactivity perturbation is unknown. Both the size of the reactivity addition and the feedback coefficient must be inferred from the shape of the observed flux in the first few seconds. (Very unfortunately, the 1962 tests did not include a test at zero power, without feedback, so that the change due to feedback could be readily observed.)

PHOTONEUTRONS

In a heavy water reactor, photoneutrons (from the gamma-n reaction with deuterium) are significant. These tend to be longer-lived than those from fissionable elements, and add about 15% to the delayed neutrons from U-235 fission alone. Traditionally, nine delayed photoneutron groups are used. In past Savannah River practice, these nine photoneutron groups were combined with the traditional six for U-235 fission to form 10 groups. (Computer time was a factor.) In current analyses, all 15 groups are used.

No adjustments in the photoneutron constants have been made, and 10 vs 15 groups is not believed to make any significant difference in calculated results.

For reference, the "old" and "new" (with Brady-England) delayed neutron constants used in this study are listed on Table B-5.

REFERENCES

- A1. M. C. Brady and T. R. England, "Delayed Neutron Data and Group Parameters for 43 Fissioning Systems", Nuclear Science and Engineering, : 103, 12-149, Oct 1989
- A2. T. G. Williamson, "Delayed Neutron Yields and Decay Constants (U)", WSRC-TR-91-115, March 1991
- A3. G. Robert Keepin, "Physics of Nuclear Kinetics", Addison-Wesley Publishing Co., Palo Alto, 1965
- A4. J. Blachot, M.C. Brady, A. Fillp, R. W. Mills, D. R. Weaver, "Status of Delayed Neutron Data - 1990", OECD Nuclear Energy Agency, NEACRP-L-323 and NEANDC-299"U", Dec 1990
- A5. T. G. Williamson, "Delayed Neutron Yields and Decay Constants (U)", WSRC-TR-91-115-Addendum, May 1991
- A6. Y. A. Chao, Westinghouse Commercial Nuclear Fuel Division, personal communication, Sept-Oct, 1991
- A7. M. C. Brady/T. G. Williamson/Toby Burnett, personal communication, Oct 1, 1991
- A8. R. W. Waldo, et al, "Delayed Neutron Yields: Time Dependent Measurements and a Predictive Model", Physical Review C, Vol 23, No 3, March 1981.

TABLE B-1
FIVE SETS OF U-235 DELAYED NEUTRON CONSTANTS

1 Keepin		2 Brady-England		3 ENDF/B-V		
Beta (%)	Lambda (Sec-1)	Beta (%)	Lambda (Sec-1)	Beta (%)	Lambda (Sec-1)	
0.0169	3.87	0.0651	2.9544	0.01782	3.87	
0.0832	1.4	0.1276	0.9886	0.08773	1.4	
0.2645	0.311	0.251	0.3169	0.27894	0.311	
0.1222	0.115	0.1198	0.1219	0.12885	0.116	
0.1384	0.0317	0.1403	0.0325	0.14598	0.03174	
0.0247	0.0127	0.0278	0.0133	0.02604	0.01272	
0.6499		0.7316		0.68536		<-- Total Beta
12.75226		11.39020		12.72524		<-- avg delay (sec)

4 ENDF/B-VI (Fast)		5 ENDF/B-VI (therm)		
Brady-England Alpha (fast)	Alpha (therm)	Beta (%)	Lambda (Sec-1)	
0.0664	0.0890	0.04561	2.853	
0.1586	0.1744	0.10889	0.84949	
0.3868	0.3431	0.26558	0.30278	
0.1725	0.1637	0.11846	0.12078	
0.1807	0.1918	0.12408	0.032739	
0.0350	0.0380	0.02403	0.013336	
		0.68665		<-- Total Beta
		11.05943		<-- Avg delay (sec)

TABLE B-2
POINT KINETICS RESPONSE FOR FIVE SETS OF
U-235 DELAYED NEUTRON CONSTANTS
40 AND 200 PCM STEP REACTIVITY INSERTION
(No source, prompt neutron cycle time of 20 microseconds)

TIME (Sec)	n(t)/n(0) for 40 pcm step dk				
	1 Keepin	2 Brady-Eng	3 ENDF/B-V	4 ENDF/B-Vif	5 ENDF/B-Vit
.10	1.0683	1.0609	1.0645	1.0646	1.0652
.20	1.0709	1.0639	1.0670	1.0673	1.0684
.50	1.0778	1.0716	1.0735	1.0744	1.0767
1.0	1.0872	1.0820	1.0824	1.0843	1.0878
2.0	1.1028	1.0986	1.0971	1.1010	1.1057
5.0	1.1410	1.1380	1.1330	1.1425	1.1482
20.0	1.2878	1.2844	1.2705	1.3038	1.3076
60.0	1.6520	1.6441	1.6072	1.7130	1.7056
n(60)/n(20)	1.283	1.280	1.265	1.314	1.304

TIME (Sec)	n(t)/n(0) for 200 pcm step dk				
	1 Keepin	2 Brady-Eng	3 ENDF/B-V	4 ENDF/B-Vif	5 ENDF/B-Vit
.0	1.0000	1.0000	1.0000	1.0000	1.0000
.10	1.4687	1.4021	1.4341	1.4347	1.4397
.20	1.4936	1.4283	1.4566	1.4592	1.4691
.50	1.5601	1.4975	1.5167	1.5253	1.5471
1.0	1.6570	1.5961	1.6037	1.6228	1.6590
2.0	1.8322	1.7712	1.7599	1.8030	1.8599
5.0	2.3519	2.2787	2.2159	2.3501	2.4556
20.0	6.4233	6.1152	5.5557	6.8814	7.3924
60.0	73.1727	65.7304	50.4001	94.8814	108.0468
n(60)/n(20)	11.39	10.75	9.07	13.79	14.62

TABLE B-3

STABLE PERIOD FOR FIVE SETS OF
U-235 DELAYED NEUTRON CONSTANTS
(No source, prompt neutron cycle time of 20 microseconds)

STABLE PERIOD (Sec)	Reactivity, pcm				
	1 Keepin -----	2 Brady-Eng -----	3 ENDF/B-V -----	4 ENDF/B-Vif -----	5 ENDF/B-Vit -----
0.01	847.1	927.5	882.2	883.1	882.6
0.02	744.4	823.5	779.3	780.1	778.9
0.04	689.2	765.9	723.8	724.1	721.8
0.1	645.1	715.6	679.0	678.0	672.7
0.2	615.2	677.6	647.9	645.1	636.4
0.4	578.3	629.4	609.3	603.5	590.9
1	510.0	543.3	537.4	525.9	509.9
2	441.5	462.0	465.2	448.6	433.6
4	362.0	373.1	381.3	361.2	350.1
10	254.4	258.7	267.8	247.3	242.7
15	210.3	213.1	221.3	202.3	199.9
20	181.2	183.4	190.7	173.1	172.0
30	144.0	145.5	151.6	136.3	136.5
40	120.5	121.6	126.8	113.5	114.1
70	82.1	82.8	86.4	76.6	77.7
100	62.7	63.2	66.0	58.3	59.3
150	45.2	45.6	47.6	41.9	42.7
200	35.4	35.7	37.3	32.7	33.5
300	24.8	24.9	26.1	22.8	23.4
400	19.1	19.2	20.0	17.5	18.0
700	11.3	11.3	11.9	10.3	10.6
1000	8.0	8.0	8.4	7.3	7.5

TABLE B-4
COMPARISON OF INFERRED PROMPT COEFFICIENT FROM 1962 MARK VI-B TESTS
BASED ON VARIOUS DELAYED NEUTRON CONSTANTS

	Prompt Coefficient ----- -2.3 pcm/Deg-C
Calculated (1991, Cefus)	
Case A Brady-England (without slowest 5 photoneutron groups)	-1.2
Case G Brady-England (with all 9 photoneutron groups)	-1.1
Case K ENDF/B-VI (fast) (with all 9 photoneutron groups)	-0.2
Case L ENDF/B-VI (thermal) (Brady-England alphas and lambdas for thermal U-235, and with all 9 photoneutron groups)	-1.0
Case I Historic AA-3 input (based on Keepin constants)	+0.6

TABLE B-5

DELAYED NEUTRON CONSTANTS USED FOR SRS REACTOR KINETICS ANALYSIS
OF 1962 MARK VI-B TESTS
(INCLUDES PHOTONEUTRONS)

NEW (Brady-England)			OLD*	
	BETA	LAMBDA(sec-1)	BETA	LAMBDA(sec-1)
1	.00065	2.9544	.000169	3.87
2	.00128	0.9886	.000832	1.400
3	.00251	0.3169	.003098	.3055
4	.00120	0.1219	.001222	.115
5	.00140	0.0325	.0013845	.0317
6	.00028	0.0133	.0003888	.01397

	.00732	<- Total		
Photoneutrons				
7	.000430	0.277	.00004864	.00481
8	.0001346	0.0169	.00002335	.001500
9	.0000462	0.00481	.00003065	.0001776
10	.00002218	0.00150	.00000331	.00000409
11	.00001367	0.00043 (39 MIN)		
12	.00001544	0.000117 (2.3 HR)		
13	.00000213	0.000044 (6.3 HR)		
14	.00000068	0.0000036 (77 HR)		
15	.00000033	0.00000063 (18 DA)		
TOTAL BETA = .00799			TOTAL BETA = .00720	

* In the "OLD" set, the nine delayed photonuetron groups were combined with the six groups to form a total of ten groups.

FIGURE B-1

APPENDIX C

CORRECTION FOR SPATIAL EFFECTS

The delta-k rods were located near the center of the reactor core. Therefore, the core power increased more in the center, and less at the periphery. The ex-core power range flux monitor will also increase less than the average core power. To the extent that reactivity is associated with the core average power, the change in the relation between the flux detector and the core power must be considered.

Bill Graves has calculated the ratio, (core average power)/(leakage flux), for the two detector locations which may have been used for the test measurement (ref 5). These came from TRIMHX runs for 0 to 2.1 seconds, comparing core average power with the local power in the blanket. As expected, these results showed a prompt jump, followed by a slower increase as the initial delayed neutron precursor distribution decayed. That is, Graves' values can be closely fit by the equation:

$$\begin{aligned}
 \text{Tilt (detector B)} &= \frac{\text{Core total power as fraction of initial}}{\text{Detector B indication at time } = t \text{ relative to } t=0} \\
 &= 1.0072 - .000705 * F(t) \\
 &= 1.0072 * [1 - .000700 * F(t)] \\
 &= 1 + TB(t) \\
 &\quad TB(t) = \text{Change in detector B tilt} \\
 &\quad \quad = .0072 [1 - .098 * F(t)] \\
 F(t) &= \text{Delayed neutron relaxation function} \\
 &= \text{Sum}\{ \text{Beta}(i) * \text{Exp}[-\text{Lambda}(i)*t]\} / \text{Sum}\{ \text{Beta}(i) \} \\
 \text{Tilt (detector A)} &= 1 + TA(t) \\
 &\quad TA(t) = \text{Change in detector A tilt} \\
 &\quad \quad = 1.128 * TB(t)
 \end{aligned}$$

[I know of no one else who has used or written of the delayed neutron relaxation function, $F(t)$; or fit the tilt function to the equation $\text{Tilt}(t) = A * [1 - B * F(t)]$, where A is the steady-state tilt, and B probably depends upon the perturbation, the reactor, and maybe the geometry of the two points. Values of B have been found to vary from 0.07 to 0.4. The equation appears to accurately fit the various cases for which it's been tested. Some further discussion of it is in my memo, NS-RM&OI-TWB-91-028, "Delayed Neutron Effects on Neutron Flux Tilt", 2/2/91. A word of caution: For a reactor driven subcritical, neither the flux nor the delayed neutron precursor distribution will ever approach their critical shapes. Therefore, the equation still holds, except that the value of A is NOT the static tilt, and the equation cannot be used to determine transient shapes during shutdown (e.g., scram response) based on static calculations. - Toby Burnett]

Test data was recorded for only one detector. However, records are not clear whether it was detector A or detector B. Detector B was the most likely choice.

The neutron relaxation function is plotted on Figure C-1.

Comparison between the Graves' TRIMHX values and the formulas above are shown on Figure C-2. The small variations are attributable to numerical noise in the TRIMHX calculations.

The first two columns of Table C-1, C-2, and C-3 show the measured flux detector response to the delta-k rod perturbation at 14,000, 91,000, and 199,000 MWD burnup, respectively. Reactivity from the delta-k rods was assumed to be equivalent to a step reactivity insertion at 0.4 seconds. The third column tabulates the delayed neutron relaxation function at those times. Column 4 & 5 list the "Tilt", or correction factor, to be multiplied by the measured flux response to get the core average power per the equations above, for detectors A and B respectively. The last five columns list the computer core power based on various assumptions:

- A. Using detector A and the Graves' TRIMHX factors (ref 5).
- B. Using detector B and the Graves' TRIMHX factors.
- C. This case allows for inaccuracy in the TRIMHX calculation of the static tilt caused by the delta-k rod perturbation, and assumes the detector tilt fits the equation

$$\text{Tilt (detector)} = 1 + .015 [1 - .098 * F(t)]$$

(instead of $1 + .0072 [1 - .098 * F(t)]$).

- D. This case assumed the 30% was due to the delayed neutrons, i.e.

$$\text{Tilt (detector)} = 1 + .0072 [1 - .30 * F(t)]$$

- E. Combination of cases D and E, assuming the detector tilt fits the equation

$$\text{Tilt (detector)} = 1 + .015 [1 - .30 * F(t)]$$

These cases (A through E above) used to evaluate the data from the 1962 test near beginning of charge life. The resulting reactivity coefficients are tabulated on Table A-1, and the results shown on Figures A-1 to A-5.

Use of detector "A" to evaluate the test data causes an inferred coolant coefficient 0.12 pcm/Deg-C more negative than detector "B". Doubling the detector tilt, to 15% (from 7.2%) causes the inferred coolant coefficient to be 1.1 pcm/Deg-C more negative, while increasing the effect of the delayed neutrons on the spatial transient leads to a smaller inferred coolant coefficient.

In reference 5, Graves also determined the variation in the tilt with increasing radius near the edge of the reactor. The change in tilt in the blanket is only 2% greater than in the outer fuel; i.e., $.00735[1-.098F(t)]$ instead of $.0072[1-.098F(t)]$. He also input the current delayed neutron constants (ref 4) into TRIMHX, and found a further 9% increase. The combination of these two effects is less than the 13% difference between detector "B" and detector "A", and would make the inferred coolant coefficient about -0.1 pcm/Deg-C more negative.

TABLE C-1
CORE AVERAGE POWER CALCULATED FROM DETECTOR RESPONSE
1962 VI-B TEST -- BURNUP = 14,000 MWD

Time (sec)	Meas. Flux	Delayed Neutron Relaxation Function F(t)	Correction		Calc Core Total Power				
			Factor		Avg		Case C	Case D	Case E
			Dect-B	Dect-A	Dect-B	Dect-A	A=.015	B=.3	C & D
0.04	1	1	-	-	1	1	1	1	1
0.21	1.0436	0.919581	1.006551	1.007389	1.05044	1.05131	1.05784	1.04904	1.05494
0.31	1.0456	0.880951	1.006578	1.007420	1.05248	1.05336	1.05993	1.05114	1.05714
0.42	1.0472	0.843962	1.006604	1.007449	1.05412	1.05500	1.06161	1.05283	1.05893
0.52	1.0484	0.814318	1.006625	1.007473	1.05535	1.05624	1.06287	1.05410	1.06028
0.6	1.0511	0.792830	1.006640	1.007490	1.05808	1.05897	1.06564	1.05687	1.06312
0.71	1.0517	0.765950	1.006659	1.007511	1.05870	1.05960	1.06629	1.05753	1.06385
0.79	1.0536	0.748040	1.006672	1.007526	1.06063	1.06153	1.06825	1.05948	1.06586
0.9	1.0542	0.725303	1.006688	1.007544	1.06125	1.06215	1.06889	1.06014	1.06657
1	1.0557	0.706254	1.006701	1.007559	1.06277	1.06368	1.07044	1.06169	1.06818
1.5	1.0606	0.627753	1.006757	1.007621	1.06777	1.06868	1.07553	1.06680	1.07351
2	1.0636	0.567772	1.006799	1.007669	1.07083	1.07176	1.07867	1.06995	1.07684
2.5	1.0675	0.519569	1.006833	1.007708	1.07479	1.07573	1.08270	1.07399	1.08102
3	1.0701	0.479660	1.006861	1.007739	1.07744	1.07838	1.08540	1.07670	1.08384
3.5	1.0701	0.445932	1.006885	1.007766	1.07747	1.07841	1.08545	1.07677	1.08400
4	1.0681	0.416983	1.006905	1.007789	1.07548	1.07642	1.08347	1.07483	1.08212
4.5	1.0678	0.391836	1.006923	1.007809	1.07519	1.07614	1.08320	1.07458	1.08193
5	1.0671	0.369779	1.006939	1.007827	1.07450	1.07545	1.08253	1.07393	1.08133
6	1.0628	0.332929	1.006965	1.007856	1.07020	1.07115	1.07822	1.06969	1.07715
7	1.0574	0.303435	1.006985	1.007880	1.06479	1.06573	1.07279	1.06432	1.07182
8	1.0536	0.279365	1.007002	1.007899	1.06098	1.06192	1.06897	1.06055	1.06808
9	1.0519	0.259391	1.007016	1.007915	1.05928	1.06023	1.06728	1.05888	1.06645
10	1.0478	0.242561	1.007028	1.007928	1.05516	1.05611	1.06314	1.05480	1.06237

- A. Using detector B and the Graves' TRIMHX factors.
- B. Using detector A and the Graves' TRIMHX factors.
- C. Assuming the detector-to-core-average tilt was 1.5%, rather than the 0.72% estimated from TRIMHX.
- D. Assuming delayed neutrons contributed 30% of the detector-to-core-average tilt, rather than the 9.8% predicted by TRIMHX.
- E. Combination of case C & D

TABLE C-2
CORE AVERAGE POWER CALCULATED FROM DETECTOR RESPONSE
1962 VI-B TEST -- BURNUP = 91,000 MWD

Time (sec)	Meas. Flux	Delayed Neutron Relaxation Function F(t)	Correction		Calc Core Total Power				
			Factor		Avg Dect-B	Avg Dect-A	Case C A=.015	Case D B=.3	Case E C & D
			Dect-B	Dect-A					
0.04	1	1	-	-	1	1	1	1	1
0.21	1.0416	0.919581	1.006551	1.007389	1.04842	1.04930	1.05582	1.04703	1.05291
0.31	1.0432	0.880951	1.006578	1.007420	1.05006	1.05094	1.05750	1.04873	1.05471
0.42	1.0444	0.843962	1.006604	1.007449	1.05130	1.05218	1.05877	1.05002	1.05610
0.52	1.0454	0.814318	1.006625	1.007473	1.05233	1.05321	1.05983	1.05109	1.05725
0.6	1.0469	0.792830	1.006640	1.007490	1.05385	1.05474	1.06138	1.05264	1.05887
0.71	1.0477	0.765950	1.006659	1.007511	1.05468	1.05557	1.06224	1.05351	1.05980
0.79	1.0489	0.748040	1.006672	1.007526	1.05590	1.05679	1.06348	1.05476	1.06110
0.9	1.0498	0.725303	1.006688	1.007544	1.05682	1.05772	1.06443	1.05571	1.06212
1	1.0506	0.706254	1.006701	1.007559	1.05764	1.05854	1.06527	1.05656	1.06302
1.5	1.0547	0.627753	1.006757	1.007621	1.06183	1.06274	1.06955	1.06086	1.06754
2	1.0579	0.567772	1.006799	1.007669	1.06509	1.06601	1.07289	1.06422	1.07107
2.5	1.0606	0.519569	1.006833	1.007708	1.06785	1.06878	1.07570	1.06705	1.07403
3	1.0621	0.479660	1.006861	1.007739	1.06939	1.07032	1.07728	1.06865	1.07574
3.5	1.0609	0.445932	1.006885	1.007766	1.06820	1.06914	1.07612	1.06752	1.07468
4	1.0598	0.416983	1.006905	1.007789	1.06712	1.06806	1.07505	1.06648	1.07371
5	1.0551	0.369779	1.006939	1.007827	1.06242	1.06336	1.07035	1.06185	1.06917
6	1.0505	0.332929	1.006965	1.007856	1.05782	1.05875	1.06574	1.05731	1.06468
7	1.0456	0.303435	1.006985	1.007880	1.05290	1.05384	1.06082	1.05244	1.05986
8	1.0439	0.279365	1.007002	1.007899	1.05121	1.05215	1.05913	1.05079	1.05825
9	1.0409	0.259391	1.007016	1.007915	1.04820	1.04914	1.05612	1.04781	1.05530
10	1.0374	0.242561	1.007028	1.007928	1.04466	1.04562	1.05257	1.04429	1.06197

- A. Using detector B and the Graves' TRIMHX factors.
- B. Using detector A and the Graves' TRIMHX factors.
- C. Assuming the detector-to-core-average tilt was 1.5%, rather than the 0.72% estimated from TRIMHX.
- D. Assuming delayed neutrons contributed 30% of the detector-to-core-average tilt, rather than the 9.8% predicted by TRIMHX.
- E. Combination of case C & D

TABLE C-3
CORE AVERAGE POWER CALCULATED FROM DETECTOR RESPONSE
1962 VI-B TEST -- BURNUP = 199,000 MWD

Time (sec)	Meas. Flux	Delayed Neutron Relaxation Function F(t)	Correction Factor		Calc Core Total Power				
					Avg		Case C	Case D	Case E
			Dect-B	Dect-A	Dect-B	Dect-A	A=.015	B=.3	C & D
0.04	1	1	-	-	1	1	1	1	1
0.11	1.0301	0.964261	1.006519	1.007354	1.03682	1.03768	1.04409	1.03537	1.04108
0.19	1.0454	0.927987	1.006545	1.007382	1.05224	1.05312	1.05965	1.05083	1.05672
0.3	1.047	0.884580	1.006575	1.007417	1.05388	1.05477	1.06134	1.05254	1.05854
0.42	1.0474	0.843962	1.006604	1.007449	1.05432	1.05520	1.06181	1.05303	1.05913
0.5	1.0476	0.819983	1.006621	1.007468	1.05454	1.05542	1.06205	1.05329	1.05945
0.6	1.0498	0.792830	1.006640	1.007490	1.05677	1.05766	1.06432	1.05556	1.06180
0.75	1.0494	0.756838	1.006665	1.007519	1.05640	1.05729	1.06397	1.05524	1.06157
0.81	1.0498	0.743753	1.006675	1.007529	1.05681	1.05770	1.06440	1.05567	1.06203
0.92	1.0519	0.721378	1.006690	1.007547	1.05894	1.05984	1.06656	1.05783	1.06426
1	1.0516	0.706254	1.006701	1.007559	1.05865	1.05955	1.06628	1.05757	1.06403
1.09	1.0527	0.690241	1.006712	1.007572	1.05977	1.06067	1.06742	1.05871	1.06522
1.21	1.0542	0.670332	1.006727	1.007588	1.06129	1.06220	1.06897	1.06026	1.06683
1.31	1.0544	0.654841	1.006737	1.007600	1.06150	1.06241	1.06920	1.06050	1.06711
1.42	1.0548	0.638813	1.006749	1.007613	1.06192	1.06283	1.06963	1.06094	1.06759
1.5	1.0566	0.627753	1.006757	1.007621	1.06374	1.06465	1.07147	1.06277	1.06946
2	1.058	0.567772	1.006799	1.007669	1.06519	1.06611	1.07299	1.06432	1.07117
2.5	1.05850	0.519569	1.006833	1.007708	1.06573	1.06666	1.07357	1.06493	1.07190
3	1.05930	0.479660	1.006861	1.007739	1.06657	1.06750	1.07444	1.06583	1.07290
4	1.05390	0.416983	1.006905	1.007789	1.06118	1.06211	1.06906	1.06054	1.06773
5	1.04760	0.369779	1.006939	1.007827	1.05487	1.05580	1.06274	1.05431	1.06157
6	1.04180	0.332929	1.006965	1.007856	1.04906	1.04999	1.05692	1.04855	1.05587
7	1.03800	0.303435	1.006985	1.007880	1.04525	1.04618	1.05311	1.04479	1.05215
8	1.03410	0.279365	1.007002	1.007899	1.04134	1.04227	1.04919	1.04092	1.04831
9	1.03180	0.259391	1.007016	1.007915	1.03904	1.03997	1.04688	1.03865	1.04607
10	1.03080	0.242561	1.007028	1.007928	1.03805	1.03897	1.04589	1.03768	1.04514

- A. Using detector B and the Graves' TRIMHX factors.
- B. Using detector A and the Graves' TRIMHX factors.
- C. Assuming the detector-to-core-average tilt was 1.5%, rather than the 0.72% estimated from TRIMHX.
- D. Assuming delayed neutrons contributed 30% of the detector-to-core-average tilt, rather than the 9.8% predicted by TRIMHX.
- E. Combination of case C & D

FIGURE C-1

FIGURE C-2

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