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ABSTRACT

Quantitative, experimentally based assessments of the biases of the methods used to develop the neutronics design of the FTR are presented together with brief descriptions of the design methods. Uncertainties in biases have been established that are sufficiently small to allow a high degree of confidence in the nuclear design. Experimental data for these assessments have been developed in full-scale zero-power mockups of the final design of the reactor, except for Doppler data from SEFOR. Temperature, power coefficient, and stability methods evaluations are necessarily deferred to acceptance testing during initial startup of the FTR. Sodium voiding and small sample worths continue to be the technical areas of greatest complexity with least experiment-theory correlation. Critical mass, Doppler effects, control rod worth, and spatial power distribution have generally good experiment-theory correlations.

INTRODUCTION

Design and construction of the Fast Flux Test Facility (FFTF) is predicated on extensive testing and measurement programs to gain maximum assurance that all plant components will function reliably as designed. In keeping with this approach, the FTR Critical Experiments Program was undertaken to evaluate methods used in calculating the neutronic parameters for the Fast Test Reactor (FTR).[1] Experiments were conducted in Argonne National Laboratory critical assemblies ZPR-III, ZPPR, and ZPR-9 by the ANL staff under the direction of Hanford Engineering Development Laboratory. Experiments spanned the period from 1966 through 1974 although planning was initiated a year earlier. The scope of the entire program is shown in Table I. Beginning with Phase C, Engineering Mockup Critical (EMC) experiments converged on a configuration to reproduce the actual design as closely as possible. Out of this FTR-EMC Program came final evaluations of FTR nuclear design methods and important parameters related to safety.

Analysis and evaluation of the data obtained and application to FTR safety analysis is complete except for the final series of measurements involving fuel containing Light Water Recycle (LWR) plutonium (high-^{240}Pu). These latter data are expected to be applied to future FTR driver fuel reloads beyond Core 4. It is the purpose of the present paper to discuss only the significant nuclear safety parameters applied to the initial FTR cores (low-^{240}Pu).

TABLE I

FTR Critical Experiments Program

Program		Facility	Schedule
Phase A	Assess validity of control rod worth calculations	ZPR-III	11/66-03/67
Phase B	Verify preliminary nuclear parameters for PSAR	ZPR-III, ZPPR, ZPR-9	08/67-11/70
Phase C	Verify detailed nuclear parameters for fuel specifications and FSAR	ZPR-9	12/70-11/72
Phase D	Additional nuclear information for FSAR and startup procedures	ZPR-9	12/72-05/74
LWR Pu	Measurement of effects on operation and safety with fuel made from LWR plutonium	ZPR-9	06/74-12/74

Program Results - Summary

Parameters of greatest importance to the safety analyses for FTR include Doppler effect, reactivity worths of sodium, fuel and steel, absorber worths and interaction effects, and neutron source strength. For Doppler effects, measurements from FTR-EMC were used to augment earlier data obtained from SEFOR experiments.

As a part of the FTR-EMC program to measure nuclear design characteristics, safety parameters of the FTR were investigated experimentally. Statistically significant results were obtained that contributed to lower uncertainty and greater confidence in FTR safety analyses. Results further confirm that reactivity effects of core constituents are well within operational margins of safety. Thus, the FTR Critical Experiments Program has provided the necessary confidence that startup and initial operation can be conducted as designed. Acceptance testing during startup operation of the plant will provide final confirmation that nuclear design goals have been achieved.

Neutronics parameters of concern in assessing the safety of FTR are listed below in Table II together with the uncertainties. Confidence levels are approximately one standard deviation (1σ).

TABLE II

FTR Safety-Related Neutronics Parameters

	Value	Uncertainty
Doppler Constant (BOL) ($T\frac{dk}{dT}$)	-0.0050	±20%
Sodium Worth (BOL) Maximum center column	+4.0¢/kg	≤±0.6¢/kg
Fuel Worth (central assembly)	3$	∿±2.3%
Absorber (B_4C) Worth (Row 5)	3.8$	±1.5%
Steel Worth (central)	-2.1¢/kg	±0.2¢/kg
Neutron Source Strength (assembly)	10^8 to 10^{10} n/sec	30% (Estimate)

Program Results - Details

In the following assessment, measurements and calculations pertinent to each neutronic parameter are presented. These results provide the basis for the parameter value, its uncertainty, and the calculation-to-experiment (C/E) bias factors (if any) to be applied in calculational extrapolation to an actual reactor configuration.

Doppler Effect

The Doppler effect in FTR is quantified by the Doppler constant (-0.005 at beginning of life) which is equal to $T^n(\frac{dk}{dT})$ with n = 1.0. This choice for n is based primarily on calculations for FTR but is supported by the results of the analysis of the SEFOR experiments.[2,3] Small sample experiments in a variety of critical assemblies yield an n of ~0.8. The resonance energy flux incident on these samples, however, is relatively unaffected by the local temperature change. When the entire core is heated, this flux should be altered and a larger value of n would be anticipated.

The uncertainties in the nominal calculated Doppler constant for a particular core configuration have been evaluated with two techniques:

1. Method I--The variation of the calculated Doppler coefficient was established for changes in ^{238}U resonance parameters, calculational methods, and models of FTR, about best nominal values and conditions.

2. Method II--The uncertainty in the calculated Doppler coefficient was established in analyses of both small sample UO_2 Doppler measurements made in the FTR-EMC program and the superprompt critical Doppler experiments performed in SEFOR Cores I and II.[3,4]

Method I results led to the following sources and values of independent 1σ confidence level uncertainties in the calculated Doppler coefficient to FTR due to uncertainties in the

- ^{238}U resonance parameters, ±11%.
- Radial reflector/core interface neutron spectra, ±3%.
- Axial reflector/core interface neutron spectra, ±2%.
- 1/T dependence of the Doppler coefficient, ±5%.
- Neglect of the possible value of the ^{239}Pu Doppler effects, ±8%.

Statistically combined value of the uncertainties given above is ±15%.

Method II uncertainty evaluation utilized the C/E ratios of the UO_2 small sample Doppler measurements in FTR criticals and the Doppler measurements of SEFOR Cores I and II. Extensive small sample Doppler measurements were performed in the FTR-EMC program.

These measurements were performed at several locations: the core center and off-center, near an inserted and withdrawn control rod, and at the core/reflector interface. Altogether twenty measurements were performed in the engineering mockup of FTR. In addition, the results of analyses of small sample Doppler experiments in early FTR criticals (ZPR-3/Assy. 48, 51 and ZPR-9/FTR-3) were also included in the statistical analysis. The C/E values for the early FTR criticals ranged from 0.80 to 0.92 indicating that the calculations underpredicted the experimental results by 10 to 20%. The analysis of the FTR/EMC core center Doppler measurements gave a range of C/E values from 0.85 to 0.98, whereas the off-center Doppler measurements yielded C/E values in the range of

$\sim$1.0 to 1.30 with an average of $\sim$1.10. The primary reasons for this higher C/E value in the case of off-center Doppler measurements is due to the complexity in the analysis of these experiments. A statistical analysis of all the C/E values yielded a mean value of 1.00 with a standard deviation of ±0.12.

SEFOR superprompt critical experiments were analyzed using the FTR methods to yield a nearly direct assessment of the bias and uncertainty of the FTR methods. Core 2 results (most applicable because of the absence of BeO) yielded a small and negligible bias of 2% and an uncertainty of ±11%. Hence the SEFOR analyses indicate that the FTR model is correct with a 1σ confidence interval of ±11%. Extrapolation to FTR, to account for differences between SEFOR and FTR (pin size, Pu/U ratio, reflector configuration, and poison) yields an additional independent uncertainty of ±8%. Combining these two values, 11% and 8%, in quadrature, yields a total uncertainty of ±14% in the calculated FTR Doppler effect. This value is in good agreement with that obtained through Method I.

From the results obtained through these two methods, the expected uncertainty in the calculation of the FTR Doppler constant for any particular core configuration is taken as ±15% at the 1σ level.

Because the calculated Doppler constant is sensitive to variations in core configuration such as the amount of absorber present, an additional allowance is made for possible future departures from the nominal core configuration for which the reference Doppler calculations were made. To accommodate such variations, the total expected Doppler variation is increased to ±20% for purposes of design and safety analysis.

Sodium Worth

The reactivity worth of sodium in FTR is calculated to be -0.05¢/°F for small isothermal temperature changes and -0.02¢/MW for power level changes at constant flow and inlet temperature. This worth includes the changes in sodium content associated both with the temperature of the sodium itself as well as the volume available for coolant in the reactor core. Quite obviously, these effects are small and generally can be neglected in the calculation of normal reactor transients. In large transients such as hypothetical unprotected accidents, voiding the sodium from the central channels of the reactor may yield noticeable positive reactivity feedback. The reactivity worths of such voids are calculated using first-order perturbation theory with a 3-D(Hex-Z) model of the core.[5] Comparison of this technique with sodium void experiments conducted in FTR-EMC[6] indicates that the calculation is apparently biased by $\sim$0.15¢/kg (overpredicts the positive effect). The uncertainty in calculated worth was found to depend on the nature of the voided region but could be as high as 0.60¢/kg. For comparison, the central sodium void worth in FTR is $\sim$4.0¢/kg.

Fuel Worth

Fuel subassembly removal, accompanied by sodium inflow, will be a frequent gross reactivity event in FTR refueling. A bias of 0.067 ± 0.024% has been established for 2-D 12-energy group diffusion theory k-difference method of calculating this exchange.[7]

Cross sections are from the ENDF/B-III nuclear data file. Heavy metal isotopes are heterogeneously resonance self-shielded and adjusted to account for spatial structural heterogeneities. Corrections for axial buckling changes, not normally accounted for in 2-D planar calculations, are included via direct changes in core average bucklings for the 'before' and 'after' eigenvalue calculations. All reactivity values for FTR beginning-of-life conditions are based upon the conversion factor 0.322% $\Delta k/k/\$$, e.g., $\beta = 0.00322$.

Computations of the central subassembly worth in the FTR were performed in like manner using the same neutron energy group structure and axial buckling adjustment. Resonance self-shielded cross sections were adjusted for fuel pin lattice interaction effects using the Sauer-Dancoff correction. Also, triangular geometry, using six triangles per FTR hexagonal subassembly, was selected for the reactor calculations. The resulting computed FTR central fuel subassembly worth was 3.223$.

When adjusted using the bias factor established by the EMC results, the FTR-computed worth becomes 3.021 ± 0.068$ as is summarized in Table III. The C/E value of 1.067 for this parameter is based on ENDF/B IV delayed neutron data and represents considerable improvement in the ability to calculate central fuel worth. The reported uncertainty is due to the uncertainty in the bias factor. Other contributions to the uncertainty in the computed worth originate from the uncertainties in:

- Conversion factors from measured to computed units of reactivity worth.
- Heterogeneity difference between the platelet structure of the EMC and the array of pins of the FTR.
- Mesh size difference between the EMC square array and the FTR triangles.

The uncertainty in the conversion factors due to uncertainties in delayed neutron data tends to cancel out by using the same delayed neutron data to compute these factors in both reactors. None of these contributions, however, is expected to significantly increase the reported uncertainty of the FTR computed worth.

TABLE III

Reactivity Worth Measurements and Calculations
Summary of Results

Material Removed	Material Inserted	EMC Measured Worth ($)	EMC Calculated Worth ($)	C/E	FTR Calculated Worth ($)	FTR Calculated Worth Adjusted by C/E ($)
Fuel Portion of Central Subassembly	Sodium Channel Composition	-3.11 ± 0.07	-3.3181 ± 0.0013	1.067 ± 0.024	-3.223	-3.021 ± 0.068
Row 3[a] Control Rod	Sodium Channel Composition	6.795 ± 0.049	6.667 ± 0.013	0.981 ± 0.007	17.5[b]	17.8 ± 0.1
Row 5[a] Control Rod	Sodium Channel Composition	3.818 ± 0.013	3.748 ± 0.013	0.982 ± 0.012	21.8[c]	22.2 ± 0.3
-	Small[d] Sample of Stainless Steel at Core Center	-0.01879 ± 0.00012	-0.02604	1.39 ± 0.01[e]	-0.029	-0.021

a Average of many rod worth measurements.

b Worth computed for bank of row 3 control rods.

c Worth computed for bank of row 5 control rods.

d Stainless steel worths listed in units of $/kg.

e Quoted uncertainty in C/E is due to uncertainty in experiment only.

Absorber Worth

Worths of simulated FTR primary and secondary control rods were measured in the EMC for many different control rod configurations to obtain individual worths and to examine the effects of control rod interactions.[8] Table III lists worths of 6.795 ± 0.049$ and 3.818 ± 0.042$ for the Row 3 and Row 5 rods, respectively.

Using calculations similar to those described for fuel worths, control rod C/E biases of 0.981 ± 0.007 and 0.982 ± 0.012 for the Row 3 and Row 5 rods, respectively, were found. Under tilted flux conditions, EMC results showed rod worth to vary by a factor of two. Rod worths in tilted fluxes have been calculated with the same accuracy quoted above.[9]

Stainless Steel Worth

A reactivity worth of -0.01878 ± 0.00012$/kg was measured at the core center of the EMC for a small stainless steel sample. The measurement was made using the sample-oscillation method and a conversion factor of 307.28 Ih/$ was used to convert to units of dollars. First-order perturbation theory (FOP) was used to compute a worth of -0.02604$/kg establishing a C/E bias of 1.39 ± 0.01 where the reported uncertainty is due to the uncertainty in the experiment, only.

Stainless steel worth in the FTR was computed in an earlier study using 3-D FOP theory and twelve neutron energy groups.[5,10] Applying a conversion factor of 0.322%(Δk/k)/$, the computed worth was -0.029$/kg. After applying the EMC C/E bias, the adjusted stainless steel worth was -0.021$/kg.

Neutron Source Strength

Currently, it is planned that during refueling of FTR there will be no scram capability. All control rods will be fully inserted. Core components will be handled remotely and mechanical design features and operational procedures have been instituted to prevent refueling errors.

Subcritical reactivity will be monitored during refueling using the Modified Source Multiplication (MSM) technique. With this technique, the relationship of the reactivity to the inverse count rate of a neutron flux monitor is employed, with calculated corrections being made for detection efficiency and/or inherent neutron source strength changes.

For small changes in reactor configuration from that of the reference configuration, the accuracy of the MSM method is good. Experiments in the FTR-EMC showed that the subcritical reactivity could be measured to within about ±5%(1σ) down to the fully shutdown state ($\rho \sim -35\$$). The configurations in the EMC experiments involved primarily small changes in detection efficiency due to spatial flux changes. Effects due to changes in neutron source strength were not studied.

In FTR, the effects of neutron source changes will be large. First, there will be a rapid buildup of the neutron source strength versus irradiation, as shown in Fig. 1. This is due largely to the neutrons from ^{242}Cm, which will be produced from neutron capture in ^{241}Am, which, has in turn, built in from ^{241}Pu decay during fuel pin fabrication and storage.[11] As a result, the replacement of a burned fuel element during refueling can result in significant decreases in the neutron source strength and in the observed count rate of the Low Level Flux Monitors, as shown in Figs. 2 and 3.

Currently, the neutron source strength of each burned fuel element cannot be calculated with good accuracy largely because of uncertainties in the required nuclear data. However, it is clear from Fig. 2 that one must have some knowledge of the source strength in order to be able to infer the reactivity during refueling from measurements of the detector count rates. Hence one would expect the uncertainty in reactivity assessments to increase substantially as refueling progresses following the initial calibration.

The current plan to provide accurate reactivity (∿±10%[1σ]) throughout the whole refueling operation is to recalibrate periodically during refueling. In this fashion, changes in source strength that occur between calibrations can be kept small and the uncertainty in the assessed reactivity due to uncertainty in the source strengths can be limited. At some time in the future, when the nuclear data and methods used to calculate the buildup of the neutron source strength are improved, fewer recalibrations will be required to maintain sufficient accuracy.

CONCLUSIONS

Neutronics parameters important to safety in the FFTF are known with a greater degree of confidence as a result of the FTR Critical Experiments Program. Experience gained will no doubt provide a data base from which future LMFBRs will benefit. For example, data obtained can serve to focus future measurements on more uncertain parameters such as sodium worth and eliminate measurements for which a high degree of calculational confidence has been established.

The uncertainty associated with the Doppler coefficient in FTR is close to ±15% compared with the expected uncertainty at the onset of the program of ±25% (see Ref. 1). K_{eff} biases are less than a percent with an uncertainty of about 0.3%Δk/k. Control rod worth calculations are nearly unbiased with uncertainties of near 5%. Fuel assembly worth calculations are significantly less biased than small-sample worths but are still of concern. Sodium void calculational methods are weak but voiding effects in FTR are experimentally confirmed to be small. Neutron source effects will be difficult to compute initially but are expected to improve based upon correlation with operational experience.

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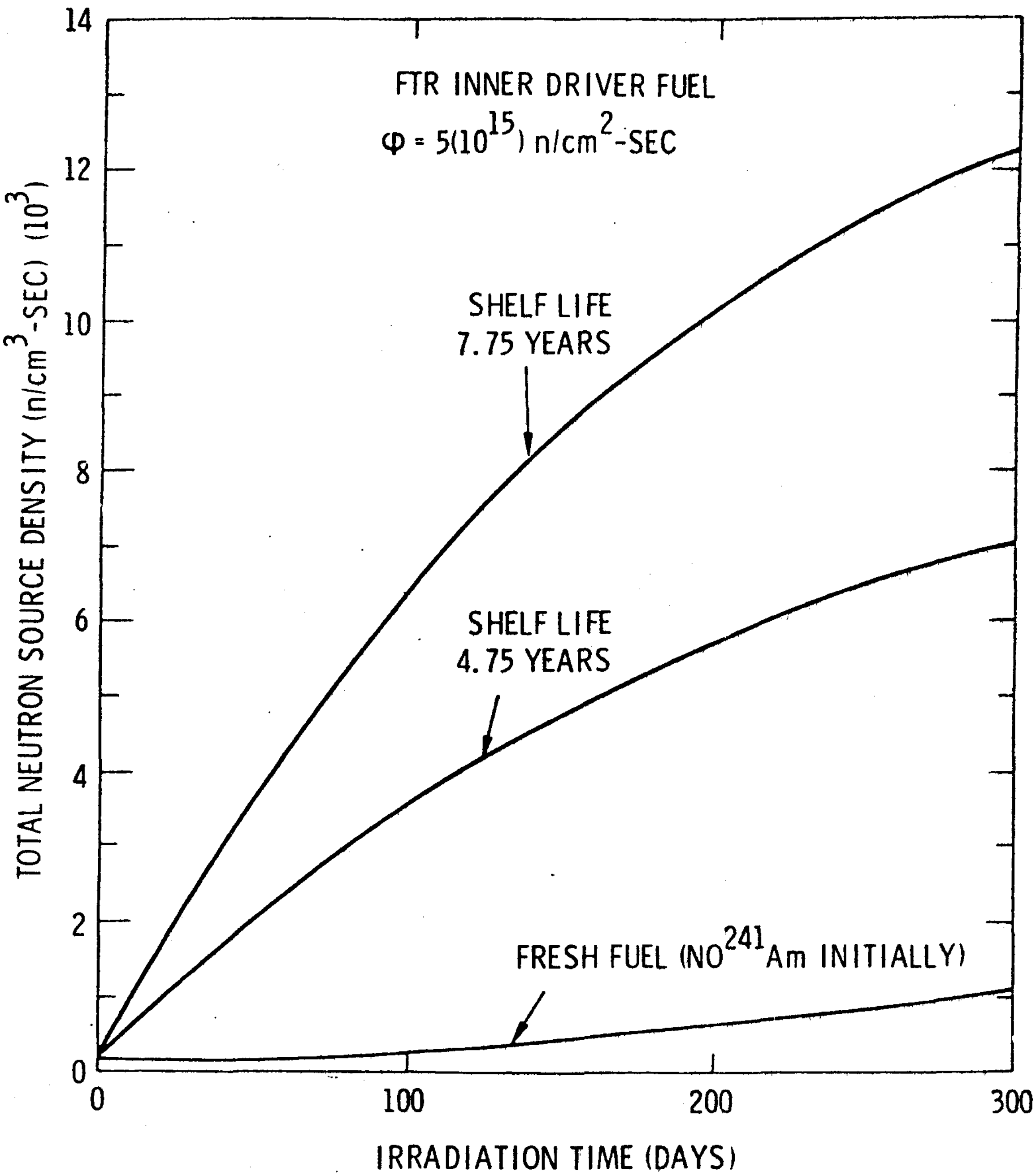

Fig. 1. Increase in neutron source strength with irradiation as a function of fuel storage life.

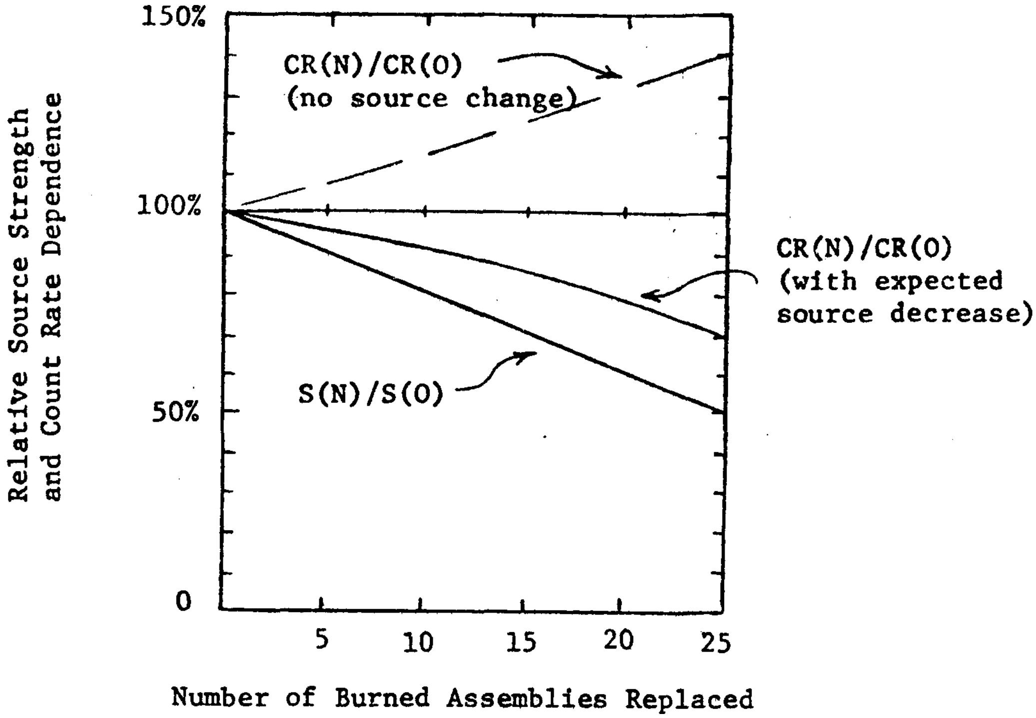

Fig. 2. Relative change in neutron source and count rate during refueling.

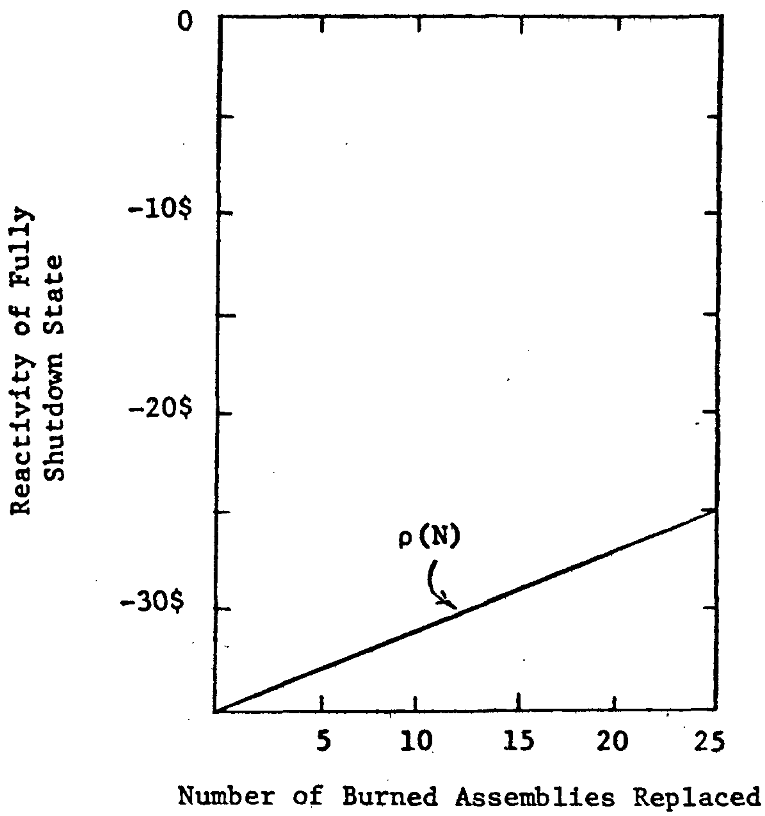

Fig. 3. Change in reactivity during refueling.