REACTIVITY FEEDBACK COEFFICIENTS FOR USE IN CRBR TRANSIENT ANALYSES

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ABSTRACT

Reactivity Feedback coefficients for the Clinch River Breeder Reactor (CRBR) are developed for use in system transient analyses, reactor stability analyses, and accident analyses. The calculational methods and associated uncertainties are described and design problem areas are delineated.

I. INTRODUCTION

This paper describes the predicted reactivity feedback coefficients for the Clinch River Breeder Reactor and the corresponding calculational techniques used. The reactivity feedback coefficients presented include: Doppler (with and without sodium), radial and axial expansion, sodium density and sodium voiding. Where appropriate comparisons with existing experiments and with other reactor designs are presented. The major problems encountered in arriving at viable results (with uncertainties) for the CRBR are in geometric modeling and in interpreting experiments, such as the plate-to-pin heterogeneity corrections necessary in the ZPPR criticals (1). The analyses and experiments are continuing to refine the analysis presented herein.

II. CALCULATIONAL METHODS

The basic ENDF/B-III cross section data is reduced to 30 neutron energy groups in the Bondarenko (2) format using the ETOX program (3). The XSRES program, an unpublished code using methods similar to SPHINX-II (4), applies energy self-shielding factors and provides the corrected cross sections to the IDX program (5). The IDX program computes and applies fine flux weighting factors to account for the heterogeneous distribution of materials. For homogeneous resonance self-shielding, the cross sections are collapsed to 21 groups in IDX. For heterogeneously treated cross sections, the 30 group cross sections are collapsed to 21 groups in ANISN (6) which has been modified to perform cell flux weighting (4).

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These 21 group cross sections are subsequently used in the 2DB (7) (two-dimensional diffusion) program and in PERT-V (8), a perturbation theory program. The 2DB program in RZ geometry is used to obtain flux and adjoint solutions for use in PERT-V and to obtain integral reactivities for gross region perturbations in computing reactivity coefficients by the direct Δk method. PERT-V, also in RZ geometry, is used to obtain spatial distributions and integral values of reactivity for computing reactivity coefficients. The RZ geometry was chosen to provide detailed axial distributions of Doppler constants and sodium voiding reactivity worth required for the HCDA analysis which used 20 axial nodes per radial "row" of fuel and radial blanket assemblies. The RZ modeling introduces distortions in the radial direction since the hexagonal cylinder geometry is represented as a circular cylinder. In particular, the control rods and control rod channels are represented as annuli in RZ geometry.

III. RESULTS AND DISCUSSION

A. Doppler Constant

The Doppler constant ($T \frac{dk}{dT}$) is the most important negative reactivity feedback coefficient in the CRBR. It is a prompt feedback mechanism and its large magnitude contributes significantly to the safety and stability of the CRBR. Table I shows the Doppler constants for each major fueled reactor region at the beginning-of-cycle (BOC) and the end-of-cycle (EOC) for both the first core and equilibrium core. Doppler constants for these core configurations are also shown with sodium removed from the fuel and radial blanket assemblies. These Doppler constants are developed, assuming a T^{-1} dependence, with an increase of 250K above the region average fuel temperatures at full-power operating conditions. The generally lower (in magnitude) core values of the Doppler constant in the equilibrium core relative to the first core reflect the changes in material (fissile fuel, fertile fuel and fission products) isotopic composition and its spatial distribution. Between BOC and EOC for each core, the insertion of control rods at BOC generally result in a Doppler constant of smaller magnitude. The effect of removal of sodium is to reduce the magnitude of the Doppler constant substantially because of the harder neutron spectrum in the voided reactor.

Figure 1 shows axial distributions of the Doppler constant for each radial "row" of fueled assemblies for the BOC of the first core. A "row" is annular representation of the hexagonal assemblies at approximately the same radius. These distributions are illustrative of those obtained for other core configurations in Table I. The discontinuity in the Doppler constant distribution at the core-axial blanket interfaces is due to the discontinuity in the fuel loadings at those interfaces. The axial asymmetry in the distributions (particularly "rows" 2 and 7) is due to the partially inserted control rods (withdrawn approximately 12 inches from the bottom of the core) which have a relatively stronger influence at resonance energies than at higher energies.

Experiments performed in the Southwest Experimental Fast Oxide Reactor (SEFOR) provide a major source of experimental data on the Doppler effect in

an LMFBFR. These data have been analysed in detail by several investigators (e.g. references (9) and (10)) using methods similar to the methods employed in the design of the CRBR. The measured value of -0.006 for core II is close to the core value for CRBR (-0.0058) and to the Fast Test Reactor value of -0.0050 . An uncertainty of $\pm 15\%$ covers the results obtained in the analyses of the SEFOR Doppler constant. The uncertainty for the CRBR Doppler constant is increased to $\pm 20\%$ to account for small differences in calculational methods, for differences in materials inventory and fission products and for the range of control rod configurations.

B. Uniform Axial Core Expansion Coefficient

The axial expansion coefficient defines the relationship between reactivity and changes in the length of the active core (fuel pellet stack height). It is assumed that the axial expansion is dependent on the temperature at the radial surface (shoulder) of the dished fuel pellet. Implicit in this assumption for the uniform axial expansion coefficient is the assumption of free movement of the fuel pellets within the clad tubes. Later in life with possible fuel pellet degradation and/or swelling under irradiation, the mode of axial expansion could be altered significantly with a tendency toward reducing the magnitude of the expansion coefficient. Calculationally, the core was expanded axially while maintaining the same mass as in the initial calculation. A value of -0.12¢/mil was obtained for the first cycle. Relating this linear expansion coefficient to the change in fuel surface temperature between refueling and full-power conditions gives -0.023¢/°F . Because the mechanism of core axial expansion is dependent on the pellet and clad behavior under irradiation, a relatively large uncertainty of $\pm 50\%$ is assumed.

C. Radial Core Expansion Coefficient

The radial core expansion coefficient is composed of two parts. The uniform portion defines the relationship between reactivity and changes in the effective (equivalent circular) radius of the core. This uniform radial expansion coefficient is dependent upon the change in dimensions of the lower core support structure which in turn depends on the inlet coolant temperature. The non-uniform portion (fuel assembly bowing) is that portion over and above the contribution from the uniform expansion. This separation is convenient since the uniform radial expansion coefficient can be simply obtained without detailed knowledge of the mechanical motion of the fuel assemblies which is a complex function of both the local temperature and neutron flux irradiation. The local temperature dependence is a function of the absolute temperature within and the temperature gradients across the fuel assembly ducts and fuel pins. The irradiation induced swelling is a complex function of the flux magnitude, spectrum and residence time.

Table II is an RZ reactivity/motion map showing the reactivity per inch of inward radial motion (hence, for expansion ΔR is negative) for each "row" of fuel and radial blanket assemblies. The uniform radial expansion coefficient by "row" is given at the bottom of the table. This table is specifically for the BOC of the first core with rods withdrawn.

This is an unrealistic condition but is given since the reactivity effects are largest for this case. Additional maps have been developed with different rod patterns for BOC and EOC of the first core and equilibrium core. Development of these maps involved both direct Δk and perturbation calculations to obtain the axial distribution of radial motion coefficients.

Between refueling temperature and full-power conditions the resulting uniform linear radial expansion coefficient for the expected change in core radius (ΔR) is -0.55¢/mil . Relating this change in radius to the corresponding change in inlet temperature gives -0.20¢/°F . The corresponding reactivity change between refueling and full-power conditions is -66¢ .

Direct Δk calculations were also performed to verify the magnitude of the uniform radial expansion coefficient obtained from Table II. A uniform increase in the pitch of the fuel assembly hexagonal lattice gave -0.47¢/mil . A uniform expansion of the core at constant mass calculation gave -0.41¢/mil . This calculation precluded the additional sodium (as in the previous cases) entering the core upon expansion. These results compare favorably with the -0.55¢/mil (which was expected to be larger in magnitude) obtained from Table II.

A preliminary mechanical motion model was used to estimate the net "bowing" reactivity using Table II. A value of -3.3¢ was obtained. Table II along with similar tables for other core conditions can be used with other mechanical models of fuel assembly mechanical motion to evaluate the reactivity effects.

D. Sodium Density Reactivity Coefficient

The sodium density reactivity coefficient defines the relationship between reactivity and sodium density changes brought about by temperature changes. Table III shows the sodium density reactivity coefficient for each major region of the reactor for the first core. The sodium density change between refueling temperature and full-power operating conditions is relatively small (≈ 10 percent), and the corresponding reactivity effect is also small relative to the Doppler effect. The change in the distribution and the change in magnitude of the sodium density coefficient for other core configurations generally follow that of the sodium voiding worth discussed in the next section. The uncertainty is also assumed to be the same as for the sodium void worth uncertainty.

E. Sodium Void Worth

Small, distributed voids such as gases entrained in the coolant are adequately treated by use of the sodium density reactivity coefficient in the preceding section. Large voids in a region of the reactor or complete voiding of the reactor is an abnormal situation and is treated separately. In the following discussion, the reactivity associated with this latter type of voiding is developed.

Although detailed spatial distributions of the sodium void worth have been developed for the HCDA analysis for the same core configurations as for the Doppler constants, only the integral values of reactivity for the major fueled reactor regions are given in Table IV. In Table IV, major changes in sodium void worth are noted only in the core regions between the first and equilibrium cycles. The substantial buildup of fission products ($\sim 10\%$) combined with the higher fissile loading in the equilibrium core harden the spectrum and account for the majority of the differences noted between the first and equilibrium cores. In addition, the radial flux distribution in the equilibrium core is more uniform due to the buildup of fissile material in the radial blankets and fission products in the core.

If, in addition to voiding only the fuel and radial blanket assemblies, the control rods and control rod channels are also voided full length (64 inches, including the rod and channel portions in the axial blankets), then the net reactivity effect is negative as shown in Table V. The positive contribution of voiding control rods is evident in comparing BOC (with rods partially inserted) to EOC conditions (rods withdrawn to top of core).

The ZPPR critical experiments provide an initial source of sodium voiding reactivity worth data for the CRBR. Figure 2 shows experimental results of sodium voiding in the ZPPR-3 and ZPPR-3 Modified criticals. Figure 2 was provided by the General Electric Company (11) based on data from Argonne National Laboratory in reference (1). Voiding was performed axially for ± 12 inches about the core midplane and progressively in the radial direction out to about 80 centimeters. The ZPPR-3 Modified critical is a close simulation to CRBR in its simulation of control rod pattern and control rod positions.

Selected voiding patterns were calculated for these experiments using the same methods as used for the CRBR calculations except for the heterogeneous corrections for the plate-geometry.

Thus, calculations of the sodium void worth in these ZPPR criticals can provide a direct estimate of the uncertainty in the CRBP data generated for the HCDA analysis. Table VI shows the results of 21 energy group RZ Δk calculations using heterogeneously treated cross sections. The general trend in both criticals is for an increasing C/E with increasing voiding. As noted for the maximum sodium voiding condition, the error is 33 percent in ZPPR-3 Modified and 56 percent in ZPPR-3. In this region of maximum discrepancy, the individual control rods have been represented as annular rings. It is felt that the error is strongly influenced by the modeling of these control rods. The control rod pattern in ZPPR-3 Modified is similar to that of CRBR. In modeling only one ring of six rods is annularized instead of two rings of rods (3 in one and 6 in the other). Additional analysis (XYZ geometry) and supporting experiments are planned to reduce the C/E ratio for large void configurations.

F. Temperature Defect

Table VII illustrates the reactivity contribution from the reactivity coefficients developed in the preceding sections. Shown in Table VII are

the components of the temperature defect (the cumulative change in reactivity in going from one steady-state temperature condition to a higher temperature condition) from hot-standby conditions to full-power conditions for the first cycle. By far the longest contributor to the temperature defect is the core Doppler effect. The average power coefficient of -0.223c/MWt is also shown in Table VII. The prompt power coefficient about hot-full-power conditions is composed primarily of the Doppler effect (instantaneous) and the relatively fast-acting axial fuel expansion effect. Of the -0.10c/MWt prompt power coefficient about SS is due to Doppler.

IV. CONCLUDING REMARKS

A relatively complete set of reactivity feedback coefficients for the Clinch River Breeder Reactor have been developed for use in system transient analyses, reactor stability analyses and in accident analyses. Most of the data is reported herein along with the methods used and the associated uncertainties. In order to reduce the uncertainties (particularly for sodium voiding) and refine the reactivity feedback coefficients, a continuing analysis effort is planned with emphasis on the problem areas found such as geometric modeling.

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TABLE I DOPPLER CONSTANTS BY REACTOR REGION

First Core

Reactor Region	DOPPLER CONSTANT, $-T \frac{dk}{dT} \times 10^4$			
	Beginning-of-Cycle		End-of-Cycle	
	With Na	Without Na	With Na	Without Na
	With Na	Without Na	With Na	Without Na
Lower Axial Blanket	4.4	3.8	3.9	3.9
Radial Blanket	9.7	7.0	8.1	8.1
Inner Core	44.1	23.1	51.1	27.5
Outer Core	13.7	8.1	14.6	9.0
Upper Axial Blanket	1.1	1.0	1.5	1.2
Total	72.7	43.0	79.2	49.7

Equilibrium Core

Reactor Region	DOPPLER CONSTANT, $-T \frac{dk}{dT} \times 10^4$			
	Beginning-of-Cycle		End-of-Cycle	
	With Na	Without Na	With Na	Without Na
	With Na	Without Na	With Na	Without Na
Lower Axial Blanket	5.2	3.4	5.7	3.9
Radial Blanket	11.0	10.2	12.1	11.9
Inner Core	34.4	16.3	37.0	19.2
Outer Core	10.7	6.1	13.1	7.7
Upper Axial Blanket	1.7	1.0	1.7	1.5
Total	62.4	37.0	69.6	44.2

TABLE II Reactivity Worth for Inward Radial Motion of Fuel Assemblies (Beginning of First Cycle - All Control Rods Parked Above Core)

Axial Location (inches above lower core support plate)	Cents per Inch of Inward Radial Motion										
	Inner Enrichment Zone							Outer Zone		Radial Blanket	
	Row 2	Row 3	Row 4	Row 5	Row 6	Row 7	Row 8	Row 9	Row 10	Row 11	Row 12
	Row 2	Row 3	Row 4	Row 5	Row 6	Row 7	Row 8	Row 9	Row 10	Row 11	Row 12
Upper Axial Blanket	93.0	0	0	0.1	0.1	0.2	0.3	-0.1	-0.2	0.1	0
	56.0	0	0.1	0.3	0.4	0.5	0.8	-1.6	-2.1	-1.5	-0.1
	79.0	0	0.3	0.7	1.1	1.4	2.3	6.6	6.9	2.8	0.2
	74.0	0.1	0.5	1.1	1.8	2.3	3.7	9.4	13.5	4.9	0.5
	69.0	0.2	1.6	3.6	5.7	7.4	12.0	31.3	43.0	14.9	1.5
	64.0	0.3	2.1	4.9	7.6	10.1	16.2	43.0	59.4	19.3	1.9
	58.0	0.2	1.6	3.6	5.6	7.4	11.9	37.7	60.6	10.4	1.1
	53.0	0.2	1.1	2.6	4.0	5.3	8.5	27.7	49.9	8.7	0.8
	48.0	0.1	0.7	1.7	2.6	3.4	5.5	16.2	30.2	5.7	0.5
	43.0	0	0.3	0.7	1.1	1.3	2.2	0.1	6.0	1.4	0.2
Lower Axial Blanket	36.0	0	0.1	0.2	0.4	0.6	0.7	-2.7	-2.7	-0.8	0.1
	29.0	0	0	0	0	0.1	0.1	-0.3	-0.3	0.1	0
Row Total	1.1	8.4	19.5	30.8	40.0	44.2	67.3	248.2	65.6	6.7	

TABLE III SODIUM VOIDING REACTIVITY COEFFICIENT BY
REACTOR REGION FOR FIRST CORE, C/°F

Inner Core Zone	28×10^{-3}
Outer Core Zone	-17×10^{-3}
Radial Blanket	-7×10^{-3}
Upper Axial Blanket	-5×10^{-3}
Lower Axial Blanket	-5×10^{-3}

TABLE IV SODIUM VOIDING REACTIVITY IN EACH REACTOR REGION
(Control Rods and Channels Not Voided)

FIRST CORE

Reactor Region	Sodium Voiding Reactivity, cents	
	Beginning-of-Cycle	End-of-Cycle
Inner Core Zone	271	264
Outer Core Zone	-83	-83
Total Core	188	181
Lower Axial Blanket	-25	-22
Upper Axial Blanket	-23	-25
Radial Blanket	-42	-41
Total Core and Blankets	98	93

EQUILIBRIUM CORE

Reactor Region	Sodium Voiding Reactivity, cents	
	Beginning-of-Cycle	End-of-Cycle
Inner Core Zone	344	355
Outer Core Zone	-16	-21
Total Core	328	334
Lower Axial Blanket	-26	-22
Upper Axial Blanket	-18	-24
Radial Blanket	-39	-36
Total Core and Blankets	245	252

TABLE V TOTAL SODIUM VOIDING REACTIVITY
(All Regions Voided)

Core Configuration	Sodium Voiding Reactivity, \$
Beginning of First Cycle	-4.00
End of First Cycle	-5.50
Beginning of Equilibrium Cycle	-2.50
End of Equilibrium Cycle	-4.00

TABLE VI COMPARISON OF EXPERIMENT AND CALCULATIONS FOR
ZPPR-3 AND ZPPR-3 MODIFIED SODIUM VOIDING EXPERIMENTS

Critical Experiments	Number of Drawers Voided	Equivalent Radius, cm.	Measured ⁽¹⁾ Reactivity, \$	Ratio of Calculation/Experiment
ZPPR-3	228	48.15	1.606	1.18
	500	71.29	2.472	1.54
	632*	80.17	2.567	1.56
ZPPR-3M	228	48.15	1.399	0.81
	510*	72.00	2.026	1.33
	632	80.17	1.711	1.60
* Approximately maximum voiding condition				

TABLE VII TEMPERATURE DEFECT-FIRST CYCLE
(Hot-Standby to Full Power Conditions)

COMPONENTS	REACTIVITY, Cents
Doppler	
Inner Core	-130.9
Outer Core	- 29.9
Radial Blankets	- 6.1
Axial Blankets	- 3.1
Axial Fuel Expansion	- 15.4
Radial Core Expansion	- 30.8
Coolant Density	- 1.4
Total Temperature Defect, Cents	-217.6
Power Coefficient (0-100% Power), Cents/MWt	- 0.223

FIGURE 1 DOPPLER CONSTANT DISTRIBUTION FOR BEGINNING OF CYCLE
OF FIRST CORE WITH SODIUM