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ATLANTA ANS TOPICAL PROCEEDINGS

NUCLEAR DESIGN OF THE CLINCH RIVER BREEDER REACTOR SYSTEM

D.R. Riley, A.R. Buhl, United States
Atomic Energy Commission, Germantown, Maryland
P.W. Dickson, R.A. Doncals, Westinghouse Advanced
Reactors Division, Madison, Pennsylvania
P.W. Greebler, General Electric
Sunnyvale, California

1. REACTOR SYSTEM DESIGN DESCRIPTION

An elevation view of the Clinch River Breeder Reactor (CRBR) is shown in Figure 1. This figure shows the location of the core and axial and radial blankets in relation to the reactor and reactor enclosure components. The reactor is sodium cooled with an inlet temperature of 730°F and an outlet mixed mean temperature of 995°F. This ΔT, coupled with a flow of 41.6 X 10^6 gallons per hour, results in a power output of 975 MWt.

Figure 2 shows the reactor plan. There are 198 fuel assemblies which are subdivided into two core zones. The outer zone has a higher enrichment than that used in the inner zone to minimize the radial power peaking factors. The next 2 1/2 rows are upper and lower axial blankets which contain depleted uranium. Outside the radial blanket assemblies are 4 rows of removable shield assemblies. The outer row of shield assemblies react against the former rings of the core restraint system at two elevations to provide gross radial position control of the core, blanket, and control assemblies. Reactivity control is achieved through the movement of 19 control assemblies which are subdivided into two control systems, called the primary and secondary systems.

A fuel assembly is shown in detail in Figure 3. It's a hexagonal structure of 316 stainless steel (SS) about 4 1/2 inches flat to flat. The duct contains enlarged regions called load pads. These pads enable the structure to be held rigidly in place while providing a space for the assemblies to swell. The fuel assembly contains 217 fuel rods. Each rod is 0.23 inches in diameter and has a cladding thickness of 15 mils. The rods are separated by wrapping the rods with wires to serve as spacers. The fuel portion of the rod is 36 inches long containing uranium-plutonium dioxide pellets. Above and below the uranium-plutonium dioxide region are 14 inches of depleted uranium dioxide pellets. There is a plenum volume 48 inches long to accommodate fission product gases. The lower part of the fuel assembly contains a 20 inch stainless steel shield to protect the lower internals from neutron irradiation damage for a 30-year life.

The blanket assemblies are similar to the fuel assemblies except that they have only 61 rods as opposed to the 217 of the fuel assemblies. The stainless-steel rods are 0.52 inches in diameter with a cladding thickness of 15 mils. Each rod contains depleted uranium dioxide pellets spanning the same axial region as the combined length of the fuel and axial blankets.

The control assemblies use B_4C as the absorber, while the shield assemblies are made of stainless steel.

All of these assemblies fit into inlet modules. Each module holds seven assemblies and fits into a support plate. Inlet modules can either hold seven fuel assemblies, or a combination of blanket, fuel or control assemblies. These inlet modules fit into liners which are installed in the lower support plate as shown in Figure 4. The "leakage" flow from the inlet module supplies a low pressure plenum in the lower support structure. The coolant then flows to a region above the lower support structure where it either cools the shield assemblies or the in-vessel storage thimbles. The latter coolant then flows into an annulus between the vessel and the vessel liner to hold the vessel temperature below 900°F while the sodium inside the liner is at 995°F to 1015°F.

The upper internals, Figure 5, are also modularized. However, they are parallelograms that channel the flow of nine to sixteen assemblies into the individual chimneys. The central modules are centered so that a control assembly location is in the center of any chimney in which a control assembly is located. The coolant flows from the chimneys into the outlet plenum and finally exits via the vessel outlet nozzles.

2. NUCLEAR DATA, COMPUTATION METHODS AND CRITICAL EXPERIMENTS

The overall analytical approach used in the analyses of CRBRP is summarized in Figure 6. The cross section data used in these nuclear design calculations is the ENDF/B-III data file (1). The ENDF/B-III pointwise data and resonance parameters were first processed into a multigroup library in the Bondarenko format (2) by the ETOX code (3). This neutron processing code calculates group constants for nuclear reactor calculations from currently available microscopic neutron data files such as the Evaluated Nuclear Data File. Output from this code includes "infinitely dilute" group cross sections, inelastic transfer matrices and temperature, as well as resonance dependent self-shielding factors for arbitrary values of σ_0 (total cross section per absorber atom). The punched card output from ETOX in the Bondarenko format is then input to the XSRES-1DX code (4). Resonance self-shielded cross sections in 30 groups were calculated using these infinite dilute cross sections and resonance self-shielding factors. The specific resonance self-shielding factors which were applied to given compositions were obtained by an interpolation scheme. Elastic removal corrections were also applied using an interactive process similar to the method used in the 1DX code (5).

Appropriate cell and/or reactor spectra are generated by means of the diffusion theory code 1DX (5) or the transport theory code ANISN (6) in order to space/energy collapse to 21 or 9 neutron groups. As indicated, 21 group cross sections were primarily used to generate the CRBRP reactivity coefficients while 9 groups were used in criticality, power distribution, control rod worth and breeding ratio predictions. The two dimensional diffusion theory code 2DB (7) was used in the calculations of power distributions, fuel depletion, criticality and control rod worths. Reactivity coefficients such as Doppler and sodium void coefficients were obtained using 2DB and PERT V (8).

In order to assess the uncertainties in important nuclear parameters and to generate necessary bias factors, analysis of various ZPPR critical assembly experiments have been performed.

3. FUEL CYCLE ASPECTS

The CRBRP operating and fuel performance requirements for annual refueling are shown in Table I. As noted, the capacity factor is scheduled to increase from 0.35 in the first cycle to 0.75 in the third and subsequent cycles. It also has been assumed that the fuel assembly peak burnup capability increases for the charged assemblies from 80,000 MWd/T in the first cycle to 150,000 MWd/T in the fifth cycle. A constant feed enrichment scheme has also been incorporated in the fuel management. The fissile loadings in the initial core have been set to provide sufficient excess reactivity for 128 full power days of operation. The feed assemblies in all subsequent cycles have an enrichment which is compatible with the equilibrium operating cycle length of 275 full power days. For the first core loading, the enrichments are 18.7 percent and 27.1 percent Pu/U+Pu for the inner and outer core zones, respectively. For the equilibrium cycle, the feed enrichments are 22 and 32 percent for the inner and outer core zone, respectively.

In the transition period in going from the first to the equilibrium cycles, the fuel and blanket management has been prescribed to provide sufficient excess reactivity for the given cycle length without exceeding the fuel performance limits shown in Table I. In these early cycles, approximately one-half of the core is discharged either due to these reactivity requirements or burnup limits.

Once equilibrium conditions have been achieved, approximately one-third of the core fuel assemblies and one-sixth of the radial blanket assemblies are discharged at the end of each cycle. This reference refueling scheme is illustrated in Figure 7 for the equilibrium cycles (275 full power days). At the end of a particular cycle, those fuel assemblies which have resided in the core for three cycles are removed and replaced by fresh assemblies, e.g., those labelled A in Figure 7. At the end of the following cycle those labelled B are replaced and the remaining one-third, labelled C, are replaced after the succeeding cycle. During the equilibrium cycles each radial blanket assembly remains in the reactor for six operating-cycles. Many of these assemblies are shuffled outward every other cycle while others are shuffled once after three years and a few are not shuffled at all. This blanket management scheme is also illustrated in Figure 7.

4. CONTROL CONSIDERATIONS

Two independent reactivity control systems are utilized in the CRBRP. The primary system serves both a safety and an operational function assuming the failure of any single active component (i.e. a stuck rod), to shut down the reactor from any operating condition and to maintain subcriticality over the full range of coolant temperatures expected during the shutdown. Allowance must also be made for the maximum reactivity fault associated with any anticipated occurrence. In addition, the primary control system is designed to meet the fuel burnup and load follow requirements for each cycle as well as to compensate for criticality and refueling uncertainties. The other reactivity control system, which is identified as the secondary system, must have sufficient worth at any time in the reactor cycle, assuming the failure of any single active component (i.e. a stuck rod), to shut down the reactor from any operating condition to the hot shutdown temperature of the coolant. Allowance must also be made for the maximum reactivity fault associated with any anticipated occurrence.

The primary and secondary control systems operate independently such that the capability of either system to fulfill its safety function in no way depends on the operation (or failure) of the other system. Design diversity and separation are provided to protect against common mode failures. Reactivity compensation is accomplished in the primary and secondary control rods by poisoning the core with neutron absorbing boron carbide (B_4C) containing either naturally occurring or enriched concentrations of the B-10 isotope.

The locations of the primary and secondary control rods are illustrated in Figure 2. The primary control system consists of 15 control rods which are grouped into banks according to their designated mode of operation. The two primary control rods in row 4 are designated as startup rods which are parked above the core during power operation. The parked rod position is defined such that the bottom of the absorber pellet stack is aligned with the top of the core (at the core-upper axial blanket interface). The central rod, plus the twelve rods in row 7, constitute the burnup and load-follow operating group which can be operated either as two separate banks or as one large bank containing all 13 rods. In the two-bank mode, the first bank consists either of the center rod plus the six rods on the hex flats in row 7 or the center rod plus the six rods in the corners of the hex in row 7, whereas the remaining six rods in the corners or the flats of the hex in row 7, respectively, constitute the second bank. The rod pattern which minimizes power peaking factors consists of the center rod grouped with the six row 7 flat rods as the primary operating bank with the six row 7 corner rods operating as a second bank. Individual rod banks are fractionally inserted in the core as dictated by excess reactivity requirements at any time-in-life. The safety shutdown margin is provided by the two startup rods plus the remaining worth in the partly withdrawn burnup and load-follow rods.

The secondary control system consists of four control rods whose only designated function is to provide a shutdown margin to standby temperature under postulated emergency conditions at any time-in-life. The secondary control rods, located in row 4, are parked above the core during all power operations.

During the approach to power from the hot-refueling temperature, the four secondary safety control rods are withdrawn first, one at a time, and latched above the core in the parked position. Then the two row 4 startup rods in the primary system are fully withdrawn and parked above the core. The reactor is then brought up to critical, at a low power level, by partially withdrawing some combination of the six row 7 corner rods and the center plus the six row 7 flat rods operated as banks. Control rods operated in the banked mode are ganged such that the bank, or group of rods, serves a single function. The rods in a designated bank are stepped out successively in 0.025 inch increments in a staggered pattern such as to maintain a relatively uniform power distribution across the core. All the rods in a control bank are inserted within ±1 inch of the nominal average bank height in the core over the full range of operating powers in order to avoid local power tilting variations in different sections of the core.

Further power increases from the zero-power-critical state are accomplished by successively withdrawing the row 7 corner rod bank and the center plus row 7 flat rod bank, as required. At the beginning of the first cycle, it is anticipated that full power conditions will be achieved with the center plus row 7 flat control bank withdrawn roughly 1/3 and with the row 7 corner control bank fully withdrawn. At the beginning of the equilibrium cycles, where a larger amount of start-of-cycles excess reactivity is present, the row 7 corner control bank withdrawal will be stopped at roughly the 2/3-withdrawn position, and the reactor will be brought up to full power conditions by withdrawing the center plus row 7 flat control bank roughly 1/3. In a normal or planned shutdown, the reactor will be brought down to a low power level by inserting one or both of the center and row 7 control banks while simultaneously decreasing the coolant pump speed to 20% of full flow. Below 20% power, the coolant temperature rise across the reactor decreases until, at essentially zero-power-critical, the remaining rods are tripped and the reactor is driven subcritical.

The various contributions to the control worth requirements are listed in Tables II and III for the primary and secondary systems, respectively. These total requirements are based on the preceeding design criteria. Table IV shows a comparison of these primary and secondary control reactivity requirements with the available CRBRP rod worths. It is important to note that these requirements are satisfied even under the extremely pessimistic, hypothetical accident assumptions that: the highest worth burnup and load-follow rod is uncontrollably withdrawn, one of the two independent shutdown control systems fails to operate, and the highest worth control rod in the operating system remains stuck in the fully withdrawn position.

5. BREEDING CHARACTERISTICS

The reactor breeding ratio is defined as the ratio of the production of fissile plutonium to the destruction of fissile material. In this definition, the mass of U-235 up to the weight fraction found in depleted uranium is not considered as a fissile material. A principal goal of the CRBRP is to obtain a breeding ratio of 1.2. The inference of this breeding ratio in terms of compound system doubling time was calculated for the CRBRP maximum fuel rod linear power rating of 14.5KW/FT. These calculations resulted in a compound system doubling time of 25 to 30 years if the CRBRP breeding ratio goal of 1.2 is achieved in all cycles.

Detailed breeding ratio calculations have been performed for the first and equilibrium cycles. The breeding ratio for CRBRP was calculated as 1.2 and 1.23 for the beginning and end of the first cycle. Thus, the breeding ratio goal of 1.2 is achieved in the first core. However, assuming design changes are not made to the fuel assembly, the breeding ratio is reduced in the equilibrium cycle due to the buildup of fission products, higher fissile loading and power buildup in the radial and axial blankets. The introduction of advanced fuel and radial blanket assembly designs will be required to maintain the breeding ratio at a value of over 1.2. Preliminary evaluations have shown that the CRBRP system (pump head) is compatible with designs which incorporate slightly larger rods in the core (0.23 to 0.245 inches) and radial blanket (0.485 to 0.511 inches) assemblies. In addition, it is possible to increase the fertile material (U-238) contained in the reactor by changing the pellet theoretical density from 91.3 to 95 percent in the core assemblies. Incorporation of these changes in later fuel and radial blanket assembly designs would permit achievement of the 1.2 breeding ratio in the equilibrium cycles.

6. NUCLEAR DESIGN IMPACT ON SAFETY PARAMETERS

The CRBRP is designed to provide a safe, stable, reliable and controllable source of energy by incorporation of the safety requirements identified in the "General Safety Design Criteria for the Clinch River Breeder Reactor Plant". The nuclear design features of the plant contribute substantially to meet these requirements.

The first core negative Doppler constant (approximately $T \frac{dk}{dT} = -0.006$ with sodium present and -0.0035 with sodium voided) provides a dominant, prompt, negative power coefficient of reactivity for all regions of the core. The Doppler constant is also negative in all the blankets. This feedback mechanism contributes to the stable operation of the core as well as mitigating the consequences of postulated reactivity addition transients. The magnitude of the Doppler constant for fast, plutonium fueled cores has been demonstrated in the SEFOR experimental program. The effect of the reactivity control system of the CRBR on the spatial distribution of the Doppler constant has been studied in the ZPPR-3 critical experiment program. The conclusion resulting from comparison of these measured values against predicted values is that the Doppler constant for CRBR can be predicted with a 2σ confidence to ±20% for the first core.

The sodium void reactivity for the first core has a predicted maximum positive worth of only +3$ for the inner core zone and +2$ for the total core. Voiding of either the radial or axial blankets results in a 0.5$ reactivity decrease of a 1.0$ decrease for voiding all blankets. Voiding of a single fuel assembly results in less than 0.03$ reactivity addition. A detailed series of experiments has been conducted in the ZPPR-3 critical experimental program to measure the sodium void worth for CRBRP. Comparison of the predicted and measured sodium void worth values shows that the predicted positive worth values are approximately 50 percent greater than the measured values at the radius of maximum voiding. While this agreement is not as good as desired and additional analytical effort is worthwhile, the comparison does show that there will be no greater reactivity addition from sodium voiding than currently predicted.

The reactor core and radial blanket is held in position by a passive core clamping mechanism. The interaction of this clamping system with the thermal and irradiation induced bowing of the fuel assemblies produces a negative fuel bowing reactivity feedback. For the fresh core a small positive bowing coefficient is predicted which is overridden by the strong negative Doppler constant. The interaction of all the reactivity feedback mechanisms provide a predicted power coefficient of reactivity of -0.10¢/MWt at full power and -0.22¢/MWt average from startup to full power for the first core.

The CRBRP is provided with both a primary and alternate reactivity control system. Both of these systems are designed to reliably shut the reactor down in response to a scram signal. The designs of these systems are diverse to provide additional shutdown reliability.

The reactivity monitoring system, described in more detail in Section 7, provides assurance that the reactor is sufficiently subcritical during refueling to prevent any accidental reactivity incident. This system also provides a means of detecting anomalous reactivity during startup and power operation. Early detection of anomalous reactivity behavior will allow corrective action to be taken to prevent serious damage of core components from several postulated malfunctions.

7. CORE MONITORING REQUIREMENTS

The Low Level Flux Monitor (LLFM) system provides continuous neutron flux monitoring for very lower power operation (source range) and during all subcritical operations. This system is used to provide assurance that the reactivity addition from the worst postulated refueling error, ~5$, caused by replacement of a control rod with a fuel assembly, would not make the reactor critical. Three BF_3 neutron detectors are located at 120° intervals outside the reactor vessel and are surrounded by graphite to enhance the detector neutron sensitivity. The reactor subcriticality is determined from the count rates of these detectors using the Modified Source Multiplication (MSM) technique. An inverse kinetics rod drop measurement at a few dollars subcritical is used to calibrate the source multiplication equation. The effects caused by changes in spontaneous neutron source strength and flux distribution are accounted for by appropriate correction factors in the modified source multiplication equation. The LLFM count rate from three contingency storage locations in the vessel for failed fuel is limited to <10% of the count rate from the core. The LLFM minimum count rate occurs at the shutdown state (~23$) prior to power operation of the first core. The count rate is greater than 10 per second at this reactor condition. At shutdown after a year's operation, the LLFM rate is greater than at the beginning of the cycle even though the reactor is at a greater degree of subcriticality. This effect is caused by the large inherent neutron source from Cm-242 and Cm-244 built up in the irradiated fuel during operation.

As the reactor is brought to power, another set of neutron monitors, the wide range system, becomes active and overlaps the low level monitor by over three decades. This system provides logarithmic readout of count rate or logarithmic or linear readout of mean square voltage depending upon power level and operator options. From 1% to 125% of full power, a compensated ion chamber also provides power measurement. The ranges of each system are shown in Figure 8. The signals from these instruments are fed to the control room data handling and display system where the power level and its rate of change are displayed. All systems except the Low Level Flux Monitor are also fed to the plant protection system which will cause a reactor trip on either overpower or power to flux mismatch.

The system is controlled by both neutron instrumentation and the in-core thermocouples. The control thermocouples consist of 21 scattered around the reactor core. Only 7 of these thermocouples are required for control; the number of 21 was selected for redundancy.

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TABLE I OPERATING & FUEL PERFORMANCE REQUIREMENTS

Cycle Number	Capacity Factor (Full-Power-Days)	Maximum Allowable Burnup Limit For Reload Core Assemblies Charged At The Start-of-Cycle (MWD/TONNE)
1	0.35(128FPD)	80,000
2	0.55(200FPD)	100,000
3	0.75(274FPD)	125,000
4	0.75(274FPD)	125,000
5	0.75(274FPD)	150,000

TABLE II REACTIVITY WORTH REQUIREMENTS FOR THE PRIMARY CONTROL SYSTEM

		First Cycle Worth Requirement ($)*	Equilibrium Cycle Worth Requirement ($)*
1.	Hot-Full-Power to Hot-Zero-Power.	3.25±1.14	3.25±1.14
2.	Power Stretch Capability.	0.20±0.07	0.20±0.07
3.	Maximum Reactivity Insertion Requiring Control System Response.	2.70±0.35	2.80±0.36
4.	Planned Start-Of-Cycle Excess Reactivity (Consists of excess fuel loading for fuel depletion plus uncertainty, criticality and fissile content uncertainties, and control bite.)	10.52	18.17
5.	Allowance for Underprediction of Criticality.	±1.40	-
6.	Allowance For Fuel Tolerances.	±0.70	±0.35
7.	Allowance for Refueling and Fuel Management Uncertainties.	-	0.80±0.40
	Subtotal	16.67	25.22
	Plus Uncertainties**	1.97	1.31
	Total Requirement	18.64	26.53

**Combined uncertainty is obtained from the square root of the sum of the squares of the individual uncertainties.

TABLE III REACTIVITY WORTH REQUIREMENTS FOR THE SECONDARY CONTROL SYSTEM

		First Cycle Worth Requirement ($)*	Equilibrium Cycle Worth Requirement ($)*
1.	Hot-Full-Power to Hot-Zero-Power	2.40±0.84	2.40±0.84
2.	Power Stretch Capability.	0.20±0.07	0.20±0.07
3.	Maximum Reactivity Insertion Requiring Control System Response	2.70±0.35	2.80±0.36
4.	Allowance for B-10 Burnup (EOC only)	0.10	0.10
	Subtotal (BOC/EOC)	5.30/5.40	5.40/5.50
	Plus Uncertainties**	0.90	0.90
	Total Worth Requirement (BOC/EOC)	6.20/6.30	6.30/6.40

**Combined uncertainty is obtained from the square root of the sum of the squares of the individual uncertainties.

TABLE IV COMPARISON OF CONTROL REQUIREMENTS WITH AVAILABLE PRIMARY AND SECONDARY ROD BANK WORTHS

	First Cycle		Equilibrium Cycle	
	Primary	Secondary	Primary	Secondary
Requirement	$18.64	$6.20	$26.53	$6.30
Total Available Worth	23.04	8.96	31.[illegible]	8.41
Available Worth with Worst Stuck Rod	21.06	6.84	29.[illegible]	6.42

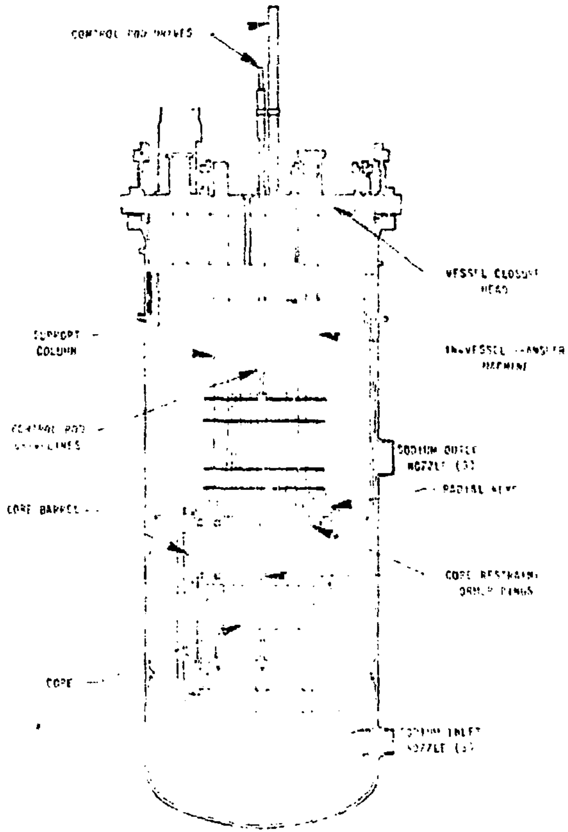
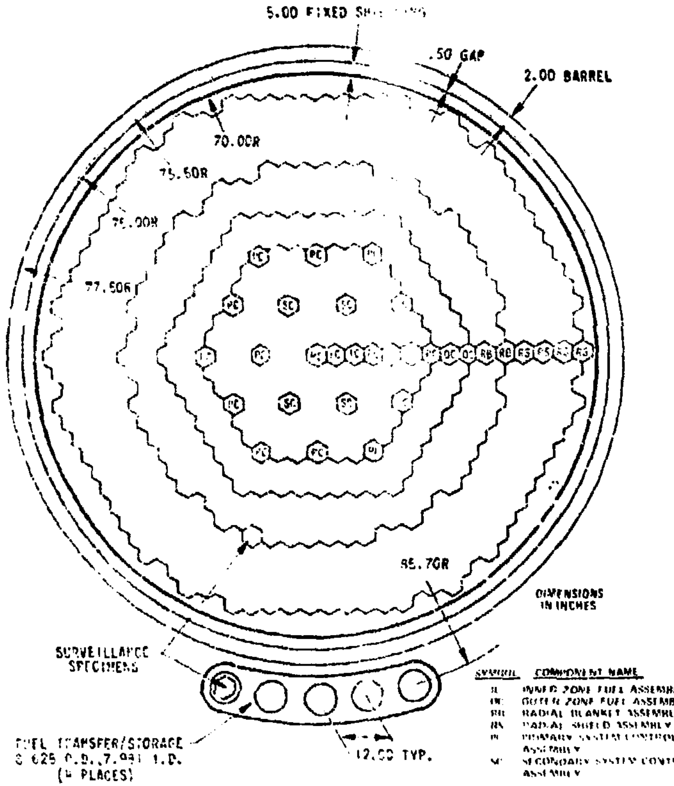


FIGURE 1 REACTOR ELEVATION



SYMBOL	COMPONENT NAME	QUANTITY
IC	INNER ZONE FUEL ASSEMBLY	108
OC	OUTER ZONE FUEL ASSEMBLY	90
RB	RADIAL BLANKET ASSEMBLY	[illegible]
RS	RADIAL SHIELD ASSEMBLY	124
PC	PRIMARY SYSTEM CONTROL ASSEMBLY	[illegible]
SC	SECONDARY SYSTEM CONTROL ASSEMBLY	[illegible]

FIGURE 2 REACTOR PLAN

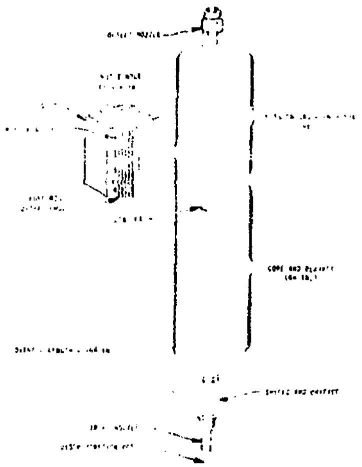


FIGURE 3 FUEL ASSEMBLY

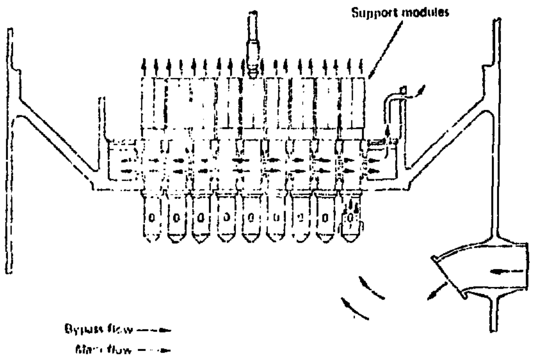


FIGURE 4 MODULAR CORE SUPPORT STRUCTURE

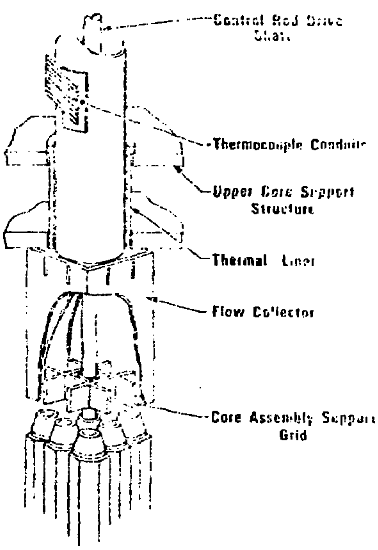


FIGURE 5 UPPER INTERNALS

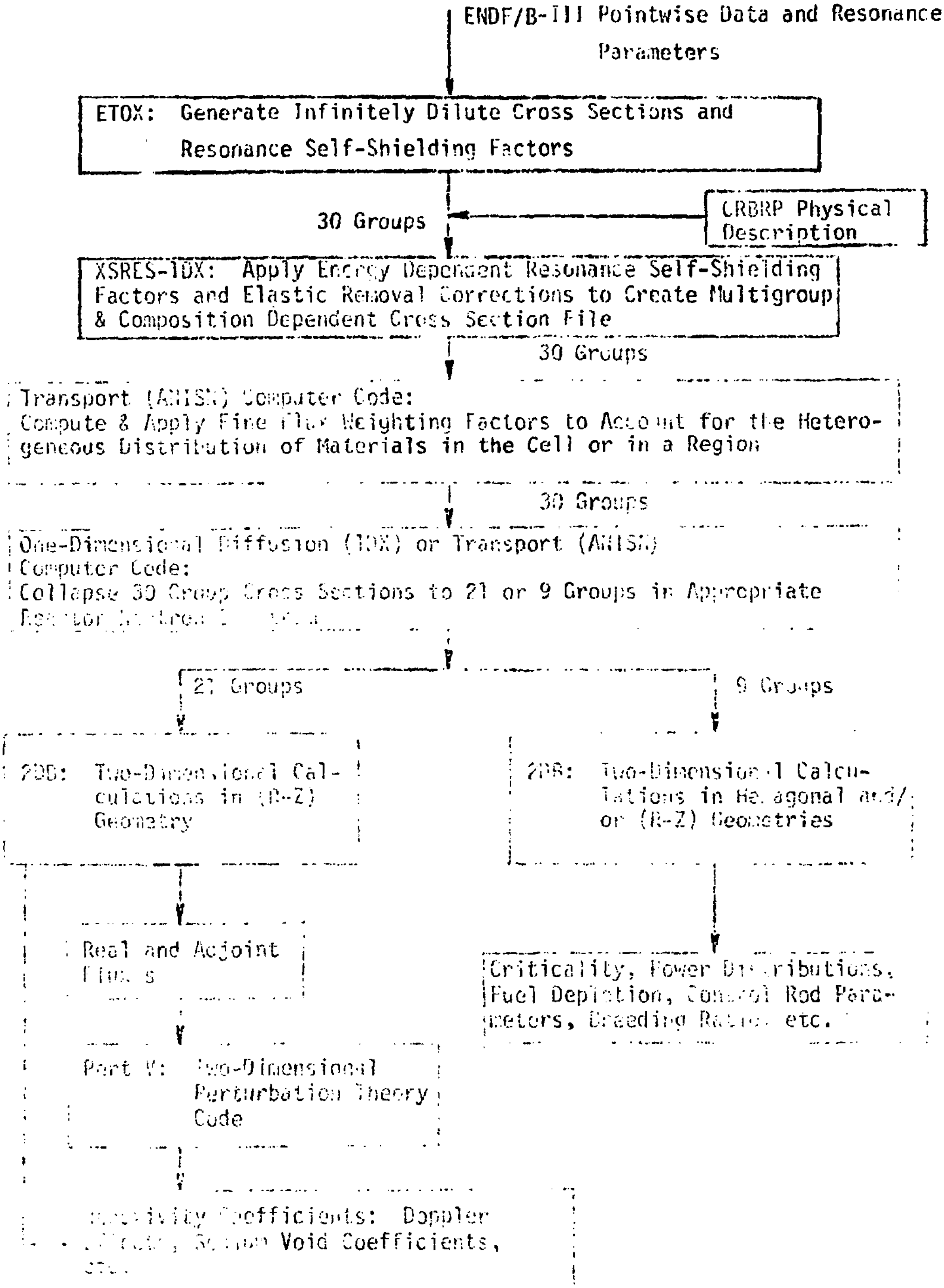


FIGURE 6 COMPUTATIONAL SCHEME FOR ANALYSIS OF

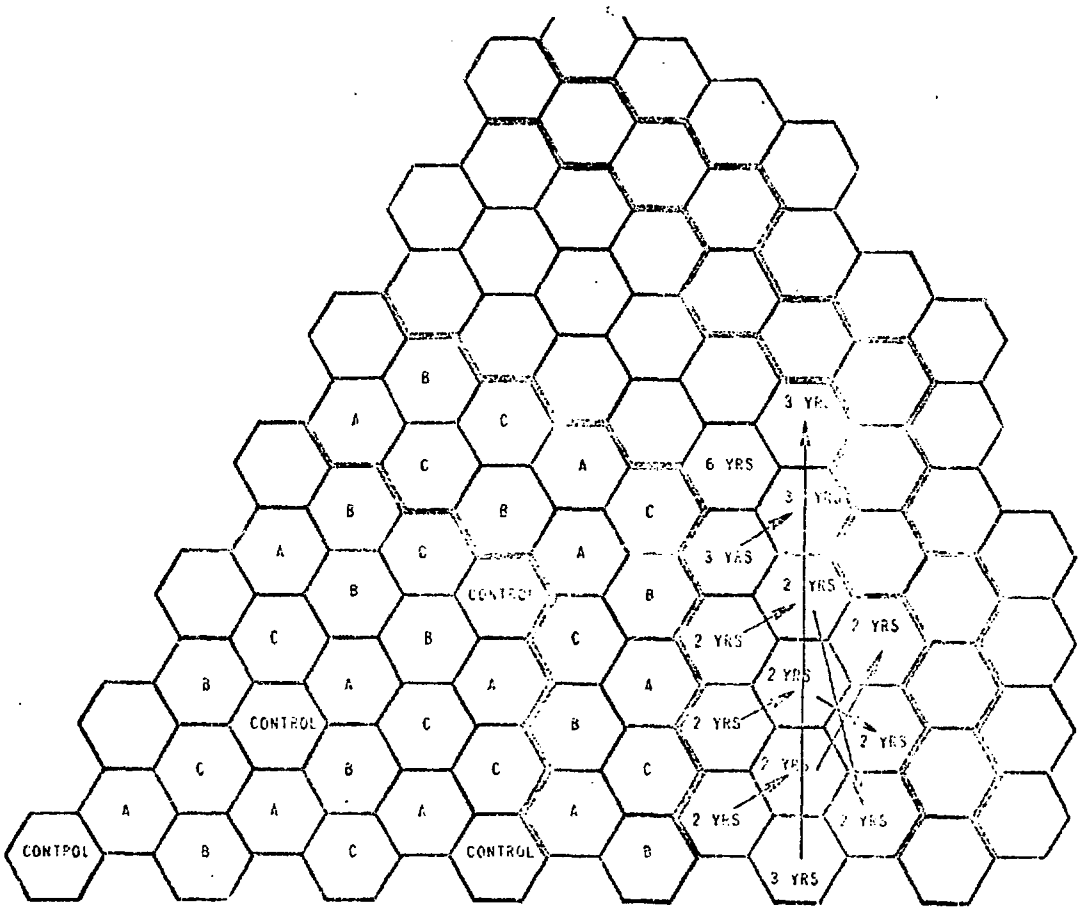


FIGURE 7 CORE AND RADIAL BLANKET REFUELING SCHEME

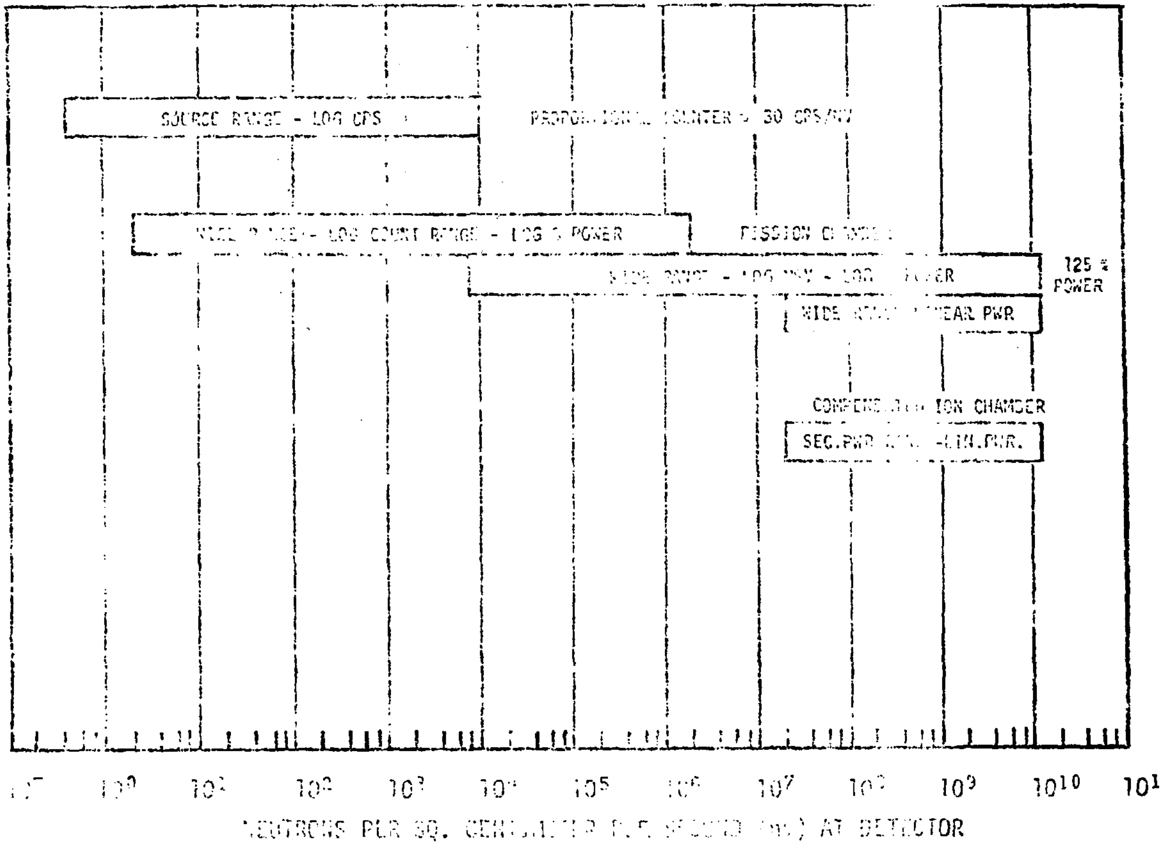


FIGURE 8 CRBRP FLUX MO[illegible] SYSTEM INSTRUMENT RANGE COVERAGE