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POSTANALYSIS OF THE CNPS CRITICAL EXPERIMENTS

by

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ABSTRACT

The Compact Nuclear Power Source (CNPS) was designed to produce electric power for remote sites where fuel logistics and costs would justify a remotely sited nuclear power plant. Since the reactor was of novel design with no appropriate benchmarks a series of critical experiments was carried out at LANL.

This paper describes the methodology and reports the results of the postanalysis that was performed on the critical experiments, which included several distinct critical configurations, the measurement of the isothermal temperature coefficient of reactivity and various material worths. Comparisons with measurements indicate that current methods and cross sections are adequate for calculating at least the beginning of life conditions in low enriched ^{235}U-graphite cores.

INTRODUCTION

The Compact Nuclear Power Source was the result of a joint design effort by Los Alamos National Laboratory and Atomic Energy of Canada Limited, Whiteshell for the production of 20KW_e at remote sites where fuel logistics and costs would justify such deployment. Originally the CNPS was developed as a possible power supply for the North Warning System, a replacement for the Distant Early Warning radar system that extends across Alaska, Canada, and Greenland near the Arctic Circle. The CNPS demonstration plan was to assemble the core and reflector materials at the LANL Critical Facilities for zero power physics experiments and then ship it to Whiteshell where full power tests would begin after connecting to an organic rankine cycle. For various reasons, not associated with the technical aspects of the design, the program was terminated after assembly at LANL. However, it was recognized that the system was a unique one and that the critical experiments program would provide a very useful reactor physics benchmark so it was decided to complete the experimental program.

This paper describes the neutronic modeling and methodology that went into the postanalysis of the CNPS critical experiments and compares the results with the experimental values. Details of the experimental program are given in a companion paper in these Proceedings[1].

THE REACTOR DESIGN

A previous paper on CNPS[2] described two conceptual designs, one with a thermoelectric converter and control drums in the reflector, and the other with an organic rankine converter and in-core control rods. The latter was chosen for the demonstration reactor, but further modifications were made, such as the replacement of zirconium - 2.5% niobium by stainless steel for the heat pipe material, when testing of the former indicated that there might be long-term problems with the wall integrity. The use of SS (a factor of 10 loss in reactivity relative to Zr) placed some further uncertainty on the ability of the reactor to operate at power for 20 years without refueling.

The CNPS core consists of a heterogeneous arrangement of unclad cylindrical fuel compacts in a graphite blocks. Figure 1. shows a 45° sector of the 492 fuel compact holes, five control rod holes and the twelve larger holes for heat pipes that make up the reactor core. The fuel compact is made up of 20% enriched triso particles in a graphite matrix; the composition of the fuel compact is given in Table I. Twenty extra holes, having the same diameter as the fuel holes, were drilled out in the vicinity of the core center; normally plugged with graphite, these would be filled with BeO in the event of a reactivity shortfall. Because of the criticality limited nature of CNPS, the core was designed to be at the peak of the K_{eff} vs. $C/^{235}U$ atomic ratio curve and so additional fuel compacts in these 20 extra holes would give comparatively little reactivity increase. A number of moderating BeO rods were ordered to fill these extra holes to provide more contingency reactivity if needed.

The core dimensions are 113 cm high by 120 cm diameter. A section view of the core and reflectors is shown in Fig. 2. The aluminum platen is part of the critical assembly machine MARS and serves to raise the core and bottom reflector into position. A 4 mm gap between the core and radial reflector provides clearance for such movement but would not be present in the final power reactor . The graphite density of the central core block is 1.82 g/cc; that for the outer core block and the radial reflector is 1.80 g/cc. The densities for the bottom and upper reflectors are 1.75 g/cc; the top reflector is at 1.75g/cc.

CALCULATIONAL METHODS

The neutronic postanalysis was carried out using a variety of codes to process cross sections and calculate the reactor characteristics. A computational flow diagram for the study is shown in Fig. 3. Multigroup cross sections sets were produced using the ENDF/B-V library and the NJOY[3] and TRANSX[4] processing codes. The NJOY code produces multigroup Doppler-broadened cross sections with Bondarenko-type self-shielding corrections at different temperatures in the MATXS format. TRANSX is a utility code producing results from the MATXS data transport tables that are compatible with the discrete-ordinate (S_n) codes ONEDANT[5] and TWODANT[6].

Sixty nine group cross sections based on the Evaluated Nuclear Data File ENDF/B-V formed the basis of the discrete-ordinate calculations. Forty thermal groups were used with twenty group upscattering.

Cell weighted cross sections were generated for the TWODANT models using TRANSX and ONEDANT in the more detailed flow diagram shown in Fig. 4. For design engineering reasons the fuel- graphite cell arrangement is not constant over the core. In the inner core it is mostly in a square pattern, whereas in the outer core the fuel channels are set in a circular patterns. Again, the $C/^{235}U$ optimization meant that the K_{eff} would not be sensitive to such non-uniformities Nevertheless, separate cell heterogeneity calculations were made for the inner and outer core regions for the discrete-ordinate analysis.

A cylindical unit cell of the fuel rod and the associated surrounding graphite was prepared for the first TRANSX input. The radius of the graphite was based on proportional areas of fuel and graphite in the inner core region. Since TRANSX requires a three region cell for its self-shielding computation - absorber, clad, and moderator - a thin annulus of graphite around the fuel pin was cast as a pseudo-clad. The Bell-Hansen-Sandmeier transport approximation and bound carbon atom treatment at low energies were used. The methods did not allow the treatment of grain heterogeneity which had been previously found to be small[2].

From this first TRANSX run came 69 gp resonance self shielded cross sections for each of the cell components in the P_3 scattering approximation. These cross sections were next used in a ONEDANT infinite cylinder model using the same cell construction but with white boundary conditions at the outer radius of the graphite and with an S_8 angular mesh approximation. The spatial multigroup fluxes from the ONEDANT calculation were then used as cell weighting fluxes in a final TRANSX calculation which produced the 69 group cell averaged macroscopic cross sections. When the cell weighted cross sections were used in a homogenized arrangement of the ONEDANT cell the K_{eff}s of both were found to be identical, thus providing a check of the stages of the cell weighting processes.

This procedure was repeated for each of the cell types, fuel, heat pipe and shim, and mock up control rods, for both the inner and outer core regions. Similar cell calculations were made for the experiments in which 20 beryllium oxide or polyethylene rods were substituted for graphite rods in the extra 20 holes. The resulting sets of macroscopic cross sections were combined to form a master cross section set for subsequent RZ calculations of the reactor using TWODANT.

For situations where two-dimensional discrete-ordinates codes have difficulty modeling the effect being calculated, such as off-center control rods, MCNP[7] was used. MCNP is a general purpose Monte Carlo code that can be used for neutron, photon, or coupled neutron-photon transport, including the capability to calculate eigenvalues for multiplying systems. The code treats an arbitrary three dimensional configuration of materials in geometric cells bounded by first- and second- degree surfaces and some special fourth - degree surfaces. Pointwise cross-section data are used for ENDF/B-V data. The model used for CNPS consisted of over 1300 cells and almost 800 surfaces.

For the K_{eff} calculations the standard deviation is quoted for the error in the tables. For differences in K_{eff}s the square root of the sum of the squares of the standard deviation is used. The latter value may be somewhat conservative since MCNP uses a correlated path technique. The appropriate postprocessor for estimating the error ror K_{eff} differences is not yet available for MCNP.

RESULTS

K_{eff} Calculation

Although there were many intermediate critical configurations, four principal ones were chosen for the detailed CNPS postanalysis. Details of the these are given in Table II which also shows the calculated eigenvalues. The control rod holes are identified by compass directions, N, E, etc. The zirconium tubes which were ordered prior to the heat pipe design change contained less than a 1000 ppm Hf

It can be seen from the table that MCNP only slightly underpredicts K_{eff} in all four cases, whereas the discrete-ordinates path overpredicts by 1 to 2.5%. Surprisingly, the worst TWODANT result is the small clean core which does not have any off-axis control materials. However, calculations with compensating errors can produce fortuitous agreement.

Control Rod Worths

The safety rod in the CNPS critical assembly consisted of stacks of natural B_4C pellets, destined to become the control and safety elements for the demonstration reactor, enclosed in a brass tube. The worth of this rod was measured in the clean and the fully loaded core. In the latter configuration the worth of enriched B_4C was also measured. The measured and calculated results are shown in Table V. Both MCNP and TWODANT overpredict the rod worths, but the standard error on the former encompasses the measured value.

Worth of BeO, Polyethylene, and Graphite Rods

For most of the experiments the 20 extra holes near the core center were filled with graphite rods. A number of replacement experiments in these holes was carried out on the small clean core configuration. The calculated and experimental results, in terms of cents/kg are presentod in Table IV. The interest in BeO stemmed from the possible need for it to enhance the reactivity of the Demonstration unit. Polyethylene was of interest for water immersion accident during transportation scenarios.

The calculated worth of BeO agreed reasonably well with experiment, although in the MCNP case the statistical variance is large (but conservative). The graphite is underpredicted by both MCNP and TWODANT, but again the variance is large in the MCNP result. BeO insertion in the Demo. unit would not have given much of a reactivity boost, but fortunately it would not have been needed. For polyethylene, which has not yet been calculated with TWODANT,the MCNP results might suggest that the worth increases with rod diameter, but the variances obscure the trend.

Temperature Coefficient

The temperature range of the experiment, which was carried out in the fully loaded core was 17° to 78°C. Although this is well below the region of interest for accident scenarios, this simple experiment provides a useful check point for temperature coefficient of reactivity calculations. Two cross section sets were generated for TWODANT, one at 300K and the other at 361K. These were applied to core materials only, since the time constant for transmission of heat from core to radial reflector is long and prvious calculations showed the effect of heating up the reflectors to be small.

The calculated value of the reactivity change was -1.22 ¢/°C compared with a measured value -0.953 ¢/°C. Currently, MCNP lacks the appropriate cross sections for calculating the CNPS experiment.

CONCLUSIONS

Overall, the calculations agree reasonably well with measurements. For the four principal configurations the MCNP eigenvalues are quite close to unity. The TWODANT values were somewhat higher, but still satisfactory, considering the heterogeneities in the core. The MCNP model is exceptionally detailed and little improvement in the reactivity worth calculations can be expected without investing in much longer running times. The TRANSX/TWODANT approach needs further work to evaluate the effects of different core zoning and cell approximations on the results. However, the current calculational methods and cross sections seem to be adequate for calculating the beginning of life conditions for low enriched heterogeneous ^{235}U-graphite cores.

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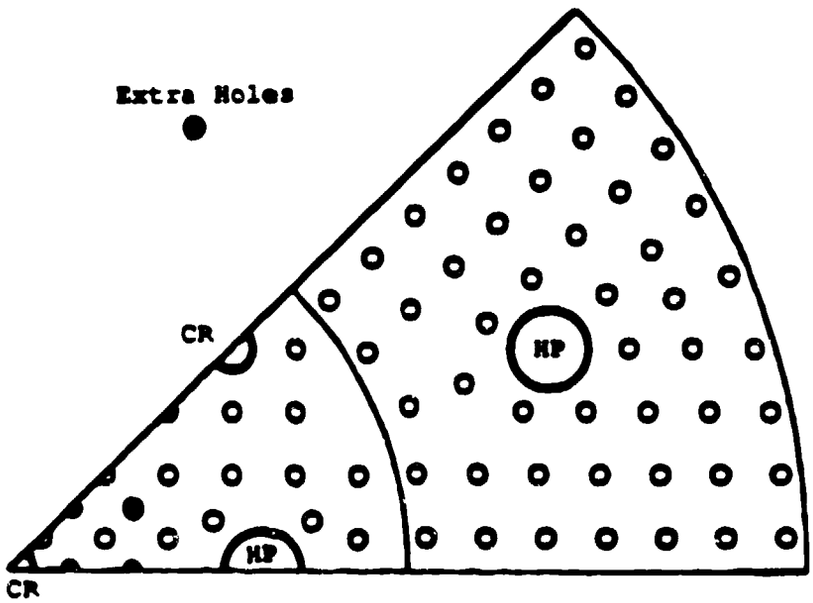

Fig. 1. 45° Sector of CNPS Core

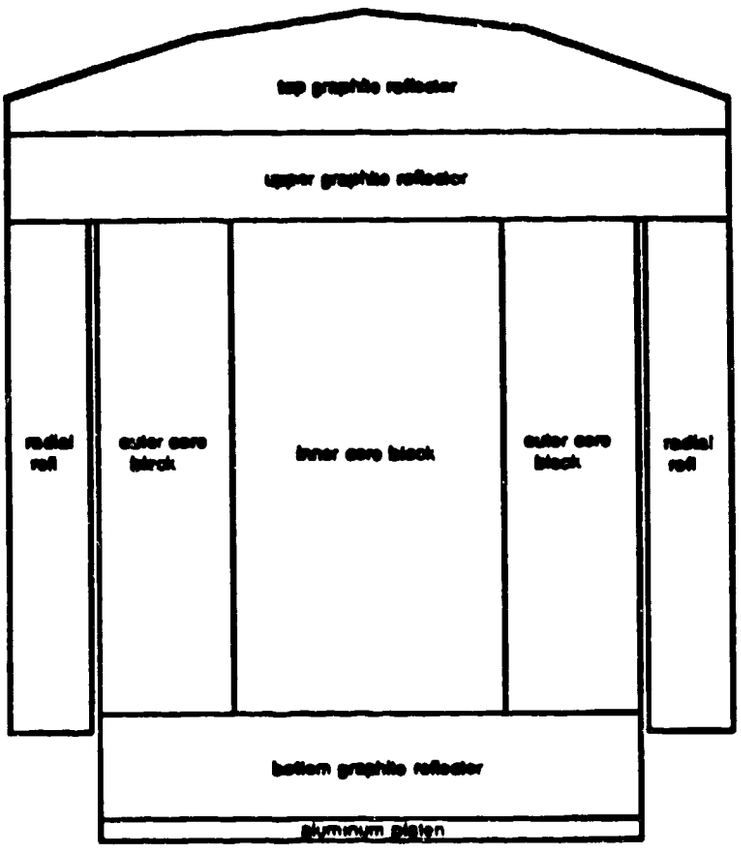

Fig. 2. Front View of Graphite Assembly

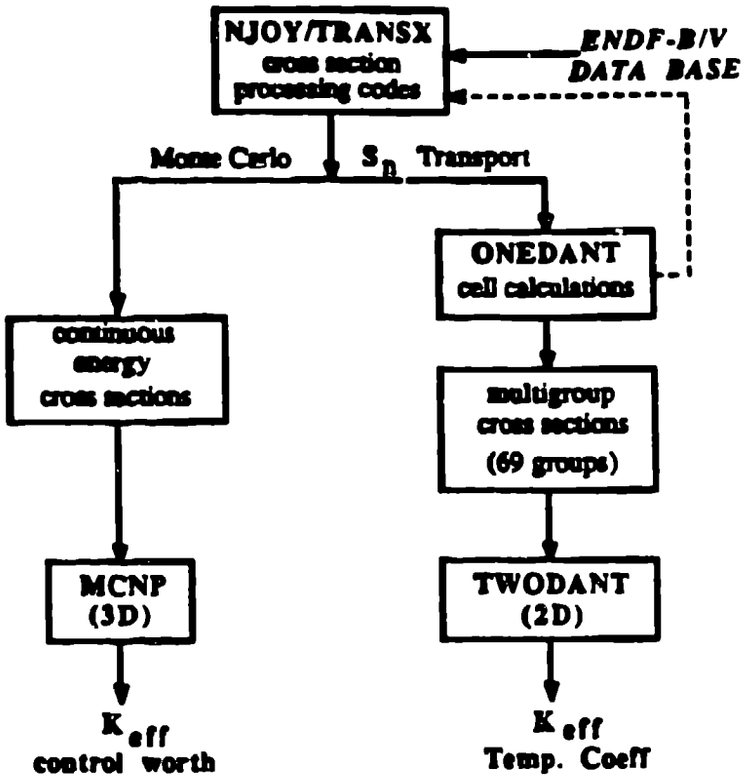

Fig. 3. Calculational Flow Chart

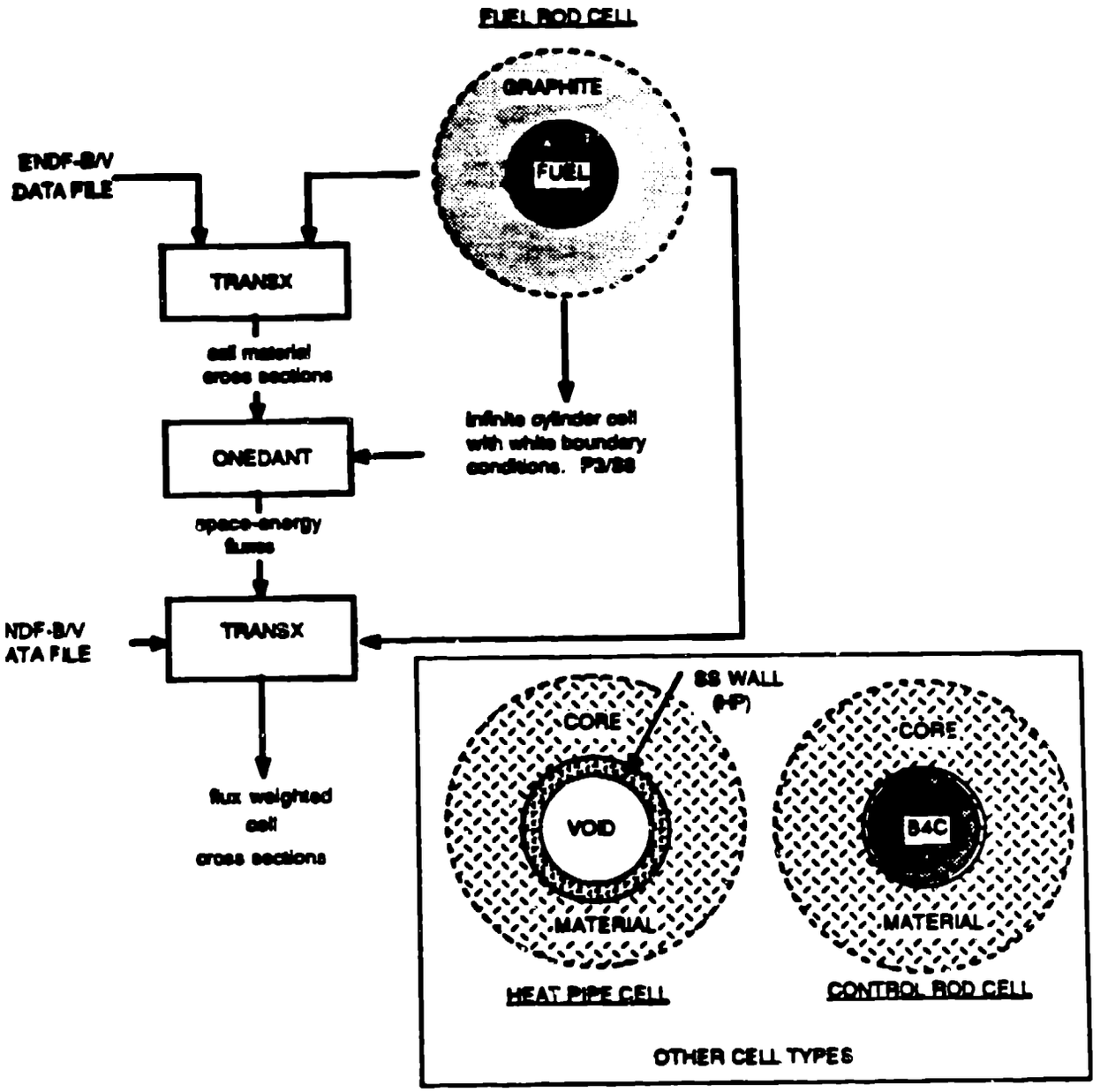

Fig. 4. Generation of Cell Weighted Cross Sections

Table I. Description of Fuel Compact

KERNEL	diameter		506.5 μ
	density		10.62 g/cc
	C/U ratio		0.274
	O/U ratio		1.631
	enrichment		0.1989 (by weight)
COATINGS	C buffer	thick.	79 μ
		density	0.96 g/cc
	PyC	thick.	33 μ
		density	1.87 g/cc
	SiC	thick.	35 μ
		density	3.21 g/cc
	PyC	thick.	35 μ
MATRIX	graphite density		0.70 g/cc
	packing fraction		0.399

Table II. Calculated Eigenvalues for Four Critical Configurations.

CRITICAL CONFIGURATION					CALCULATED K_{eff}	
ID	FUEL CHANNELS	HEAT PIPES	B4C PELLETS	SHIM ROD POSITION	MCNP	TWODANT
A	184	0	0	116 cm	0.993	1.025
B	202	11Zr	0	34.6 cm	0.993	1.018
C	380	12SS	21 in N	70.2 cm	0.999	1.020
D	492	12SS	13 in W,N,E	58.3 cm	0.993	1.010

* distance between bottom of core and lower tip of rod

Table III. Worth of Central Control Rod ($)

CONTROL MATERIAL	CLEAN CORE			FULLY LOADED CORE		
	CALCULATION		EXP	CALCULATION		EXP
	MCNP	TWODANT		MCNP	TWODANT	
Nat B_4C	8.2±0.7	8.6	7.6	4.2±0.5	4.5	4.3
Enr B_4C *				4.6±0.6	4.9	4.8

* enriched to 52.5% in ^{10}B isotope

Table IV. Twenty -Hole Replacement Experiments

MATERIAL	REACTIVITY WORTH (¢/kg) *		
	MCNP	TWODANT	EXP.
BeO	15.3±9.9	11.1	13.5
GRAPHITE	3.3±5.4	5.2	10.7
POLYETHYLENE			
1/4 in. diam	182±83	nc	294
3/8 in. diam	286±44	nc	266

* relative to void