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# APPLICATION OF THREE-DIMENSIONAL TRANSPORT CODE TO THE ANALYSIS OF THE NEUTRON STREAMING EXPERIMENT

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# APPLICATION OF THREE-DIMENSIONAL TRANSPORT CODE TO THE ANALYSIS OF THE NEUTRON STREAMING EXPERIMENT

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## ABSTRACT

This paper summarizes the calculational results of neutron streaming through a Clinch River Breeder Reactor (CRBR) Prototypic coolant pipe chaseway. Particular emphasis is placed on results at bends in the chaseway.

Calculations were performed with three three-dimensional codes: the discrete ordinates radiation transport code TORT and the Monte Carlo radiation transport code MORSE, which were developed by Oak Ridge National Laboratory (ORNL), and the discrete ordinates code ENSEMBLE, which was developed in Japan. The purpose of the calculations is not only to compare the calculational results with the experimental results, but also to compare the results of TORT and MORSE with those of ENSEMBLE.

In the TORT calculations, two types of difference methods, weighted-difference and nodal, were applied. Only the weighted-difference method was applied in ENSEMBLE calculation.

Both TORT and ENSEMBLE produced nearly the same calculational results, but differed in the number of iterations required for converging each neutron group. Also, the two types of difference methods in the TORT calculations showed no appreciable variance in the number of iterations required. However, a noticeable disparity in the computer times and some variation in the calculational results did occur.

The comparisons of the calculational results with the experimental results, showed for the epithermal neutron flux generally good agreement in the first and second legs and at the first bend where the two-dimensional modeling might be difficult. Results were fair to poor along the centerline of the first leg near the opening to the second leg because of discrete ordinates ray effects. Additionally, the agreement was good throughout the first and second legs for the thermal neutron region.

Calculations with MORSE were made. These calculational results and comparisons are described also.

## INTRODUCTION

A calculation of the neutron streaming through an experimental mockup of a Clinch River Reactor (CRBR) prototypic coolant pipe chaseway has been revisited with three-dimensional discrete ordinates codes. The experiment was conducted at the Tower Shielding Facility (TSF) at Oak Ridge National Laboratory (ORNL) in 1976 and 1977.<sup>1</sup>

The measurement of the neutron flux, using Bonner Ball detectors, indicated nine orders of magnitude attenuation in the empty pipeway, which contained two right-angle bends and was surrounded by concrete walls. The measurement data were originally analyzed<sup>2</sup> using the DOT3.5 two-dimensional discrete ordinates radiation transport code.<sup>3</sup> However, the results did not agree with measurement data at the bend, probably because of the difficulties in modeling the three-dimensional configurations using two-dimensional methods. The two-dimensional calculations used a three-step procedure involving a separate calculation in each of the three legs of the configuration.

The experiment has been recently analyzed at ORNL with the TORT three-dimensional discrete ordinates radiation transport code<sup>4</sup> to compare the calculational results with the experimental results. Additionally, a prior analysis of the chaseway was made in Japan using DOT3.5, MORSE,<sup>5</sup> and ENSEMBLE,<sup>6</sup> which is a three-dimensional discrete ordinates radiation transport code developed in Japan. The TORT results were also compared with results from these calculations.

## CALCULATIONAL CONDITIONS

The experimental arrangement, the detector traverse lines, and the portion of the configuration calculated are shown in Fig. 1. The geometry model for the three-dimensional calculation and the number of the spatial mesh intervals are shown in Fig. 2. The number of spatial mesh intervals in this calculation is 60, 65, and 30 in x-, y-, and z-dimensions, respectively, and represents a total of 117000 mesh cells. The 21-group cross sections used in the calculations were obtained by using shielding factors in the ANISN<sup>7</sup> one-dimensional discrete ordinates radiation transport code to collapse 100-group data. The P<sub>5</sub> 100-group nuclear data were obtained from the ENDF/B-IV DLC2D library.<sup>8</sup>

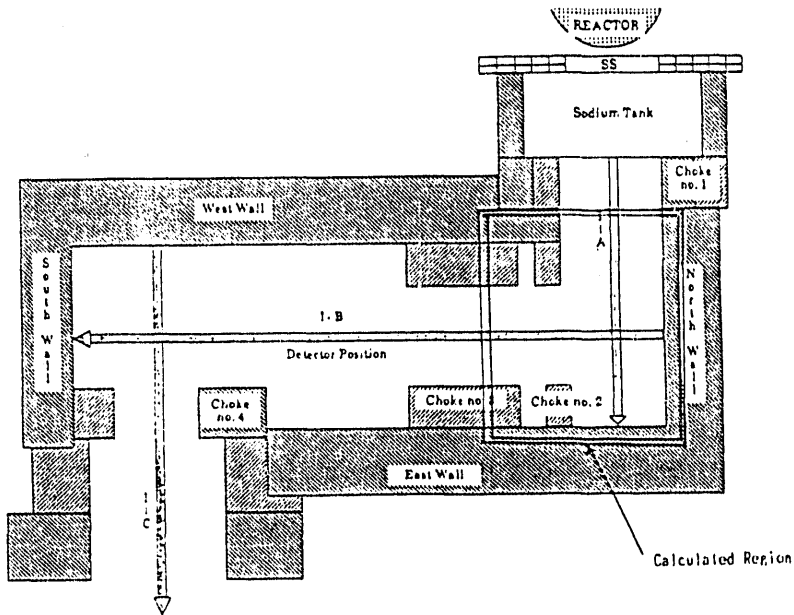


Fig. 1. Experimental Arrangement and Bonner Ball Detector Position

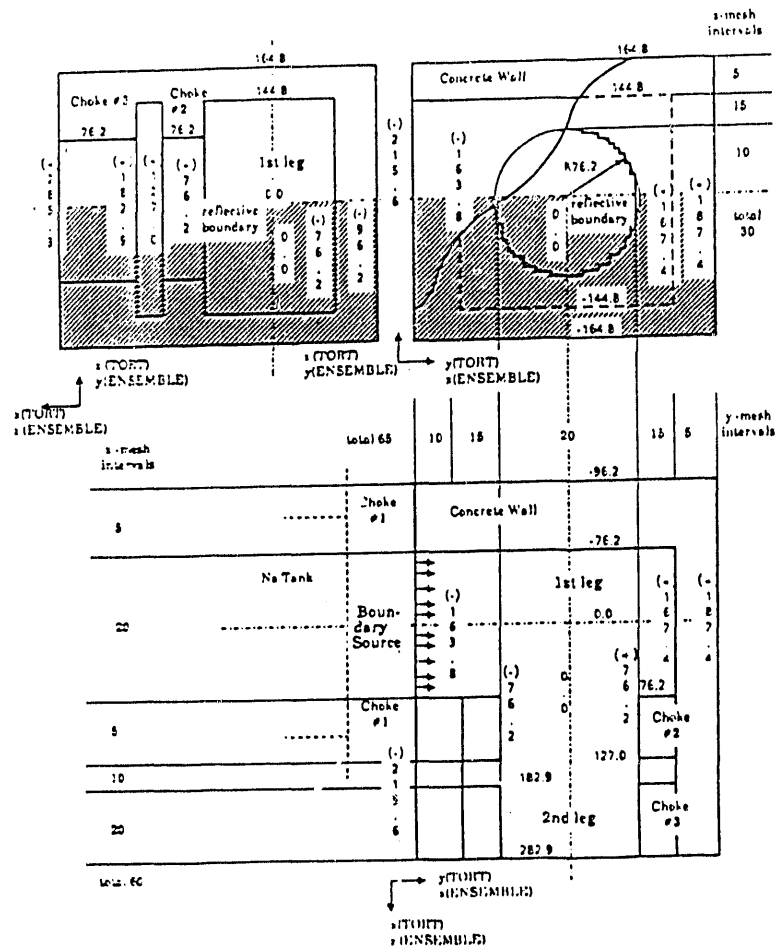


Fig. 2. Geometry Modelings for 3-Dimensional Discrete Ordinates Calculations

In the TORT calculation, the weighted-difference and nodal methods were used. On the other hand, in the ENSEMBLE calculation only the weighted-difference method, which is equal to the zero-weighted model in TORT, was used. The acceleration method in the TORT calculation was "stabilized partial current rebalance" and that used in the ENSEMBLE calculation was "coarse mesh rebalance." But in the ENSEMBLE calculation, the acceleration method was turned off after a number of iterations based on some convergence condition. The weighted-difference method and the acceleration method were selected based on previous calculational experience. It was not verified whether those models and methods were the best selections. Convergence criteria of 1% for the TORT calculation and 0.1% for the ENSEMBLE calculation were selected. All groups converged to within 1% except the thermal group. In order to save computer time, the thermal group was not converged further.

The neutron boundary source for the three-dimensional calculations was input at the exit surface of choke no. 1. A small amount of measured neutron leakage from the concrete of choke no. 1 was neglected in the calculations.

The boundary source was produced by transforming - using the nearest neighbor method\* - the first-leg angular flux from the 124-direction biased angular quadrature of the two-dimensional calculation to the 60-direction angular quadrature used in the three-dimensional calculation. In this calculation the neutron current at the source boundary was preserved.

## CALCULATIONAL RESULTS

Along the centerline of the first leg, the TORT calculation-to-experiment (C/E) values of the bare Bonner Ball (thermal neutron flux detector) indicated an improvement over the previous two-dimensional results as shown in Fig. 3. The better agreement resulted from two effects. First, in the TORT calculation, thermal neutrons produced by the scattering of fast neutrons in the concrete minimized ray effects. Second, the opening of the first leg into the second leg in the DOT3.5 calculation was approximated by surrounding the opening with the wall of the first leg. Such an approximation caused an underestimation of the measured result. The TORT calculation required no such approximation.

The TORT calculations, with both the nodal and the weighted-difference methods, were in good agreement with the ENSEMBLE calculations (within 10%), especially in the thermal neutron energy region. The nodal method required more computer time than the weighted-difference method for the same size mesh, thus no advantage was gained in using the nodal method in this case.

For the 3-inch Bonner Ball (epithermal neutron flux detector), both TORT and ENSEMBLE gave results (Fig. 4) that underestimated the flux behind the opening of the second-leg. The number of spatial mesh and quadrature directions was limited by a computer-time constraint. Because of the low-order angular quadrature, ray effects were observed in the calculations. It is believed that these ray effects, which are a characteristic problem in the discrete ordinates codes, caused the results to be underestimated. The agreement, however, is good in the first leg. DOT3.5 showed much better agreement than the 3-D codes near the opening to the second leg.

The MORSE calculation gave results for which the C/E values are low near the sodium tank because of the forward biasing of the neutron source. In general, TORT, MORSE, and ENSEMBLE agreed fairly well with each other both in shape and magnitude.

In the second leg DOT3.5 underestimated the thermal neutron flux everywhere. The epithermal neutron flux is overestimated near the entrance of the second leg through the second choke of Fig. 1. Then the flux is underestimated. TORT and ENSEMBLE gave better agreement with measured results in the second leg than did DOT3.5. Thus, the 3-D discrete ordinates methods give improved calculational results at bends in ducts.

## CONCLUSION

General conclusions deduced from the comparison with the three-dimensional calculations, which were performed using TORT, MORSE, and ENSEMBLE, and the comparison of those calculations with measurement data follow.

Along the first- and second-leg, the TORT (both nodal and weighted difference) and ENSEMBLE calculations gave similar neutron fluxes, agreeing within 10 to 15%. But a dip in the neutron flux due to ray effects was observed in the first leg because of the small number of angles in the quadrature set and the small number of spatial mesh intervals. These could not be increased due to computer-time constraints. There is no clear winner among the three 3-D discrete ordinates calculations. In the thermal neutron energy region where convergence was very difficult, the three calculations gave results that were different mainly due to convergence.

In the TORT calculation, the nodal and weighted-difference methods gave the same results. But the nodal method required more computer time than the weighted-difference method. Therefore, no advantage was gained by using the nodal method.

\* For a given output direction, the nearest neighbor is that direction in the input quadrature set which forms the largest dot product with the output direction.

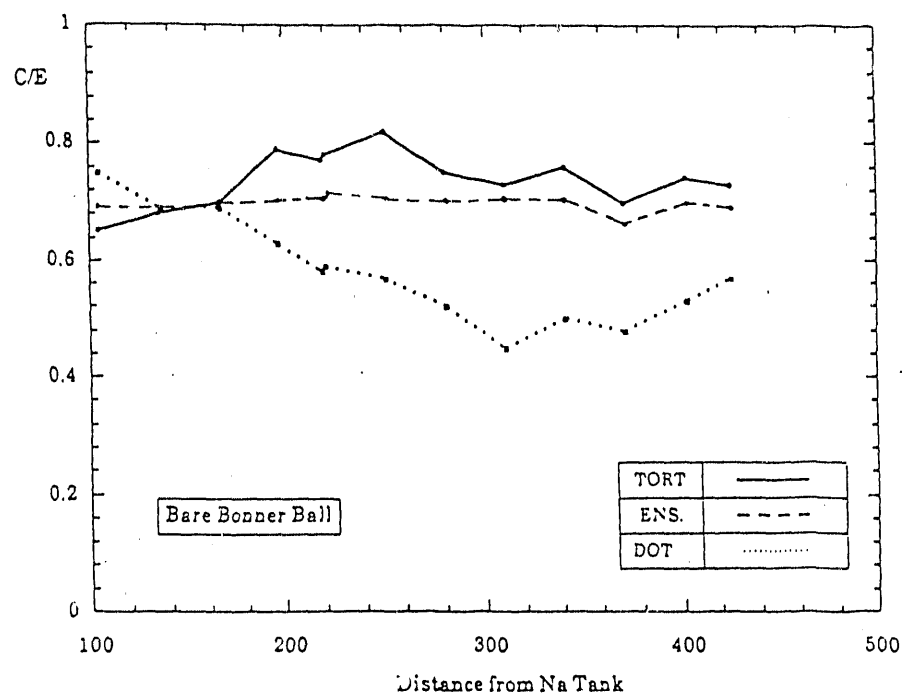


Fig. 3. C/E Values for First Leg Measurement (Bare Bonner Ball)

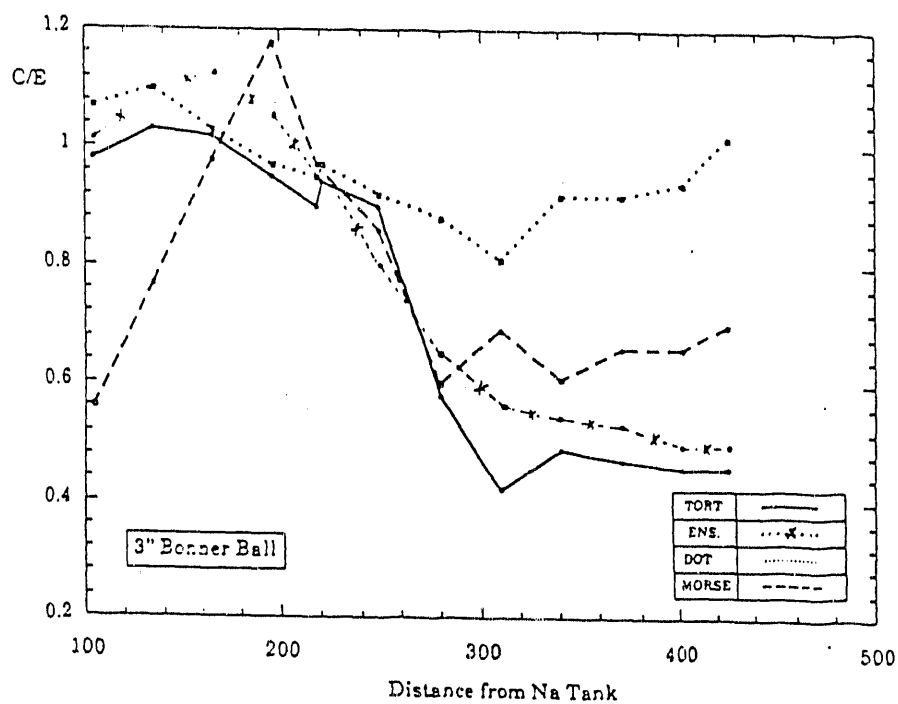


Fig. 4. C/E Values for First Leg Measurement (3-in. Bonner Ball)

The three-dimensional calculations were also compared with the two-dimensional calculations which were performed previously. With two-dimensional calculations it may be difficult to get good results at the bend, which is three-dimensional geometry, and therefore the two-dimensional calculations cannot be expected to give good C/E values. The results from the three-dimensional discrete ordinates codes gave good results in the thermal neutron energy region, where C/E values were improved considerably from the 2-D results. The three-dimensional discrete ordinates codes also compared well with experimental results at the first bend in the duct and along the axis of the second leg of the duct.

The calculational results from MORSE were compared with those from TORT and ENSEMBLE under the same calculational conditions. The MORSE calculation uses random sampling and the calculational results might not be the best estimation. In the first-leg, the MORSE calculation gave results much lower than the discrete ordinates results, but in the second leg, gave generally better results. The computer time with MORSE was about one tenth of that with TORT and ENSEMBLE. But, in the MORSE calculation, the total time required from the preparation (input and user routines) through the completion was as much as in the TORT or ENSEMBLE calculation because the MORSE calculation needed more setup time for the calculation.

Especially in shielding calculations it is desirable to save computer time and problem setup time. While no clear choice of 3-D code can be made on the basis of the calculated results alone, TORT is favored because of its competitive computer time and its relative ease of use.

## REFERENCES

1. F. J. MUCKENTHALER and C. E. CLIFFORD, "A Measurement of the Neutron Streaming in a CRBR Prototypic Coolant Chaseway," *ORNL/TM-5901*.
2. R. E. MAERKER, "Discrete Ordinates Analysis of the TSF Experiment on Neutron Streaming in a CRBR Prototypic Coolant Pipe Chaseway," *ORNL/TM-6497*.
3. W. A. RHOADES and F. R. MYNATT, "The DOT III Two-Dimensional Discrete Ordinates Transport Code," *ORNL/TM-4280*, September 1973.
4. W. A. RHOADES and R. L. CHILDS, "The TORT Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code," *ORNL-6268*.
5. M. B. EMMETT, "The MORSE Monte Carlo Radiation Transport Code System," *ORNL-4972*, February 1975.
6. T. NISHIMURA, K. TADA, H. YOKOBORI, and A. SUGAWARA, "Development of Discrete Ordinates SN Code in Three-Dimensional (X, Y, Z) Geometry for Shielding Design," *Journal of Nuclear Science and Technology* 17, pp. 539-558 (1980).
7. W. W. ENGLE, JR., "A Users Manual for ANISN: A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," *Union Carbide Corporation Nuclear Division Report, K-1699* (March 1967). Code and documentation obtained from "RSIC Computer Code Collection ANISN-ORNL" CCC-254 (September 1988).
8. ORNL Radiation Shielding Information Center, "100 Group Neutron Cross-Section Data Based on ENDF/B," *RSIC Data Library Collection, DLC-2* (January 22, 1975).

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