Neutron Transport Equation Solution of Discrete Ordinates Method

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Abstract: This article is a verification of the one-dimensional slab geometry neutron transport equation solver – STAR (**S**$_N$ neutron **T**ransport **A**ccelerate solve**R**). This code is based on the Discrete Ordinates Method to discrete the angular variable which is recorded as $\mathbf{\Omega}$. STAR computes the three problems, and the results are in good agreement with the reference values.

Problem 1: Fixed source problem with uniform vacuum boundary [1]

Description: The first problem is a 1-D, one energy group with fixed source problem. The fixed source is isotropic and uniformly distributed throughout the slab. Boundary condition of left side is reflective boundary and the right side is vacuum boundary condition. In this case, only P0 scattering source term is considered.

Table 1. Macroscopic cross section

Parameter	Σ_t	Σ_s	Q_e
Value	1.0cm^{-1}	0.5cm^{-1}	1.0cm^{-3}·s^{-1}

Geometry: The length of the problem is 5cm, and it is shown in Fig 1.

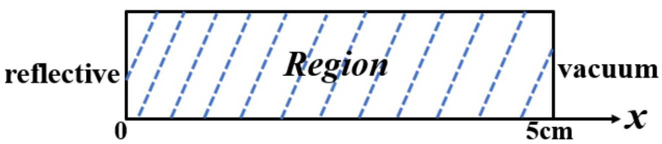

Figure 1. Geometry of problem 1

Result: The scalar neutron flux is shown in Fig 2. Use the S$_N$ method to discrete the angular parameters, which N is 8. Result fits well with the reference.

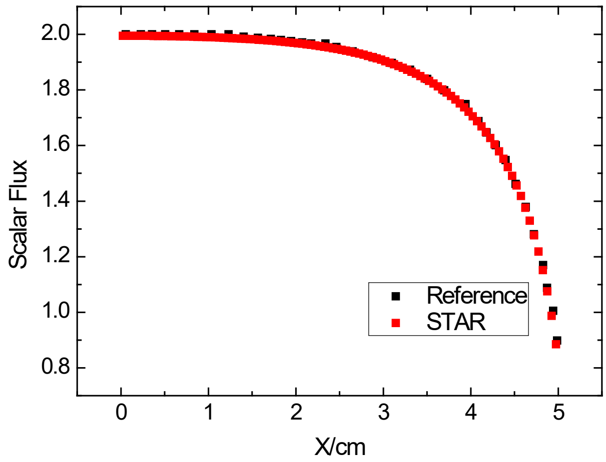

Figure 2. Scalar neutron flux distribution

Problem 2: Fixed source problem of MOX fuel in two zones [1]

Description: The second problem is a 1-D, one energy group with fixed source problem. In this problem, there are two different regions with different macroscopic cross section parameters. Left and right side are both reflective boundary condition. In this case, only P0 scattering source term is considered.

Table 2. Macroscopic cross section

Region	Material	Σ_t	Σ_s	Q_e
1	MOX	0.83333 cm^{-1}	0.60272 cm^{-1}	1.0
2	UO_2	0.83333 cm^{-1}	0.74091cm^{-1}	0.65

Geometry: The length of each region is 10.71cm in this problem, which is shown in Fig 3.

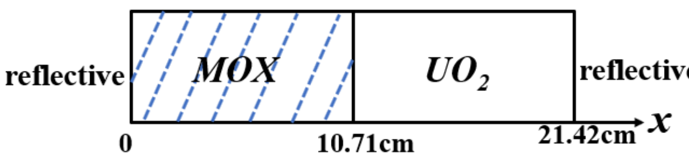

Figure 3. Geometry of problem 2

Result: The scalar neutron flux is shown in Fig 4. Also meets the referenced result well.

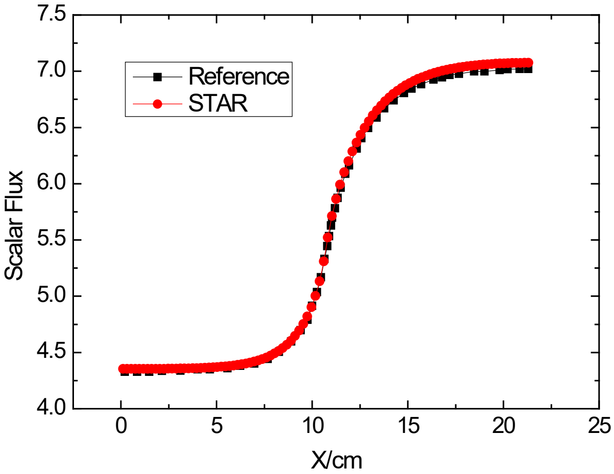

Figure 4. Scalar neutron flux distribution

Problem 3: Inhomogeneous vacuum fission source problem (ISSA benchmark) [2]

Description: The final problem is a k-eigenvalue problem. This problem contains two different zones with different parameters. Left side is reflective boundary and the right side is vacuum boundary. In this problem, we should to find the relative neutron flux and k-eigenvalue.

Table 3. Cross section of ISSA benchmark

Region	Σ_t	Σ_s	$\nu\Sigma_f$
1	1.0 cm^{-1}	0.5 cm^{-1}	1.0
2	0.8 cm^{-1}	0.4 cm^{-1}	0.0

Geometry: The length of region 1 is 2.0cm, and length of region 2 is 3.0 cm. As shown in Fig 5.

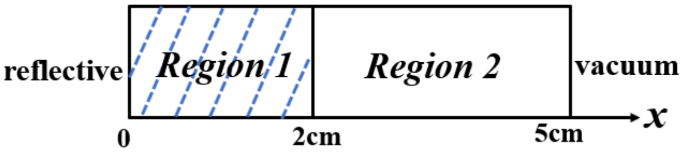

Figure 5. Geometry of ISSA benchmark

Result: The normalized scalar neutron flux is shown in Fig 6. We can know about there are infinite eigenvectors to a fixed eigenvalue problem from the linear algebra. So, we only can know about the relative neutron flux distribution in the eigenvalue problem. And then we obtain the normalized scalar neutron flux distribution and it fits well with the reference result.

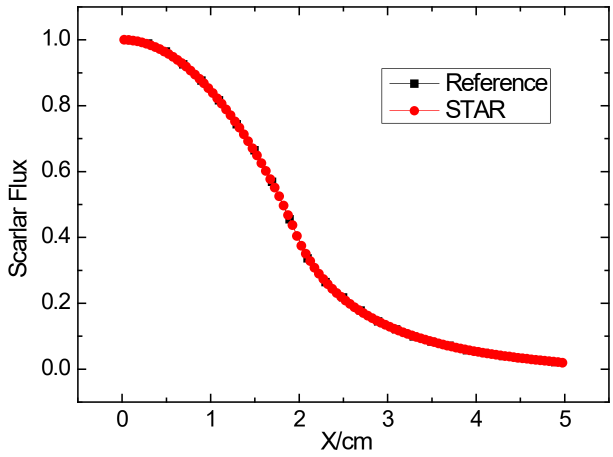

Figure 6. Normalized neutron flux

Table 4 shows the k-eigenvalue which calculated by STAR and MCNP. We believe that the result which is calculated by MCNP is the benchmark solution. And the result of STAR is increased 300pcm to MCNP. The reason is the one energy group of discrete energy variable may cause the big error. So, the result of STAR is acceptable.

Table 4. K-eigenvalue of ISSA benchmark

Code	keff	Absolute error
MCNP	1.67869	0
STAR	1.68169	+300pcm

References

[1]. ZHENG Zheng, WU Hong-chun et al. *Coupled P_N-DP_N Method for Solving the Neutron Transport Equation of Planar Geometry* [J]. Nuclear Power Engineering **vol 31** (2010).

[2]. Issa J G., Riyait N S, Goddard A J H, et al, *Multigroup Application of the Anisotropic FEM code FELTRAN to One, Two, Three-Dimensional and R-Z problem* [J]. Progress in Nuclear Energy. **18(1):251-246**, (1986).