Neutronic Calculations for Central Test Loop (CTL) in Hypothetical Heavy Water Research Reactor

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ABSTRACT*: One of the experimental facilities of the heavy water research reactor is the central test loop (C.T.L). It is located along the central axial line of the vessel and therefore will highly affect the neutronic parameters of the reactor. So, from the neutronics point of view, C.T.L is the most important facility. It is mainly designed for fuel testing, radioisotope production and neutron activation. All the simulations were performed by MCNPX2.6.*
As a first step towards C.T.L analysis, the effect of D2O-filled, H2O-filled, and He-filled C.T.L on the effective multiplication factor (Keff.), has been evaluated. According to results, H2O-filled C.T.L has a higher number of thermal neutrons, while He-filled C.T.L includes more resonance neutrons. In the next step, thermal and total axial neutron fluxes were calculated and used as the comparison parameters. The core without C.T.L (C.T.L replaced by heavy water) is selected as the reference case, and the effect of all other cases is calculated according to that.

KEYWORDS*: Heavy Water Reactor, Neutronic Calculations, Central Test Loop, Neutron Activation*.

I. INTRODUCTION

Nuclear reactors, designed specifically to serve as source of neutrons for experimental purposes, and with powers ranging from about 10 kW to a few MW, are known as research reactors. Much research is done with thermal neutrons, and also thermal reactor control is facilitated because of slower response to perturbations and inherent shutdown mechanisms connected with the moderator. Therefore, most of the research reactors available today are thermal, that fission occurs predominantly at thermal energies Refs. 2, 3.

II. MONTE CARLO METHOD

II.A. Introduction

The target of all transport calculations are, in general, the determination of the particle flux and the related terms such as reaction rates, doses, currents, etc (Ref. 1). The flux is often obtained as the solution of the Boltzmann transport equation and is grasped as a deterministic quantity because it is a solution of a deterministic equation relating exact quantities. By contrast, Monte Carlo does not solve an explicit equation, but rather obtain answer by simulating individual particles and recording some aspect (tallies) of their average behavior. The average behavior of particles in the physical system is then inferred from the average behavior of the simulated particles (Ref. 3). Therefore, one can say that Monte Carlo solves a transport problem by simulating particle histories rather than by solving an equation. Since the Monte Carlo method involves the generation of a large number of particle histories, using a considerable amount of nuclear data, its use is very strongly dependent on computer time and memory.

II.B. MCNP Transport Code

The Monte Carlo N-Particle transport code system (MCNP) developed by Los Alamos National Laboratory (LANL), is a general-purpose code for calculating the time-dependent continuous-energy transport for neutrons, photons, electrons, or coupled Neutron/photon/electron in three-dimensional geometry (Ref. 3). The code includes the capability of calculating eigenvalues for critical systems. Point wise cross-section data are used. For neutrons, energy region is from 10-11 to 20 MeV, and all reactions given in a particular cross-section evaluation (such as ENDF/B-VII) are accounted for thermal neutrons are described by both free gas and S (α,β) models. The energy regime for electron is the same as that for photons . MCNP is used for many applications such as, nuclear criticality safety, radiation shielding, nuclear safeguards, detector design and analysis, accelerator target design, medical physics and radiotherapy including BNTC, PET and neutron and photon oncology.

III. RESULTS AND DISCUSSION

II.A. Structure of research heavy water reactor

The research heavy water reactor is a tank-type reactor with pressure tubes, using heavy-water coolant and moderator, and19-pin fuel bundles in a hexagonal lattice, reactor fueled with natural uranium. Its experimental facilities include a vertical thimble along the central axial line of the tank (Central Test Loop), one vertical 15-centimeter-diameter medical beam tube, and three verticals 10-centimeter-diameter research beam tubes. By means of these facilities some activities such as fuel testing, radioisotope production, neutron radiography, neutron depth profiling (NDP), neutron activation analysis (NAA), and many other experiments requiring beam facility are possible. In addition to these provided data, this analysis assumes that the pressure tubes and major structural elements of the reactor fuel bundles are made of Zircaloy-2. It is also assumed that the center tube, which forms a key structural element of the fuel bundle, has the same inside and outside dimensions as the fuel-cladding tubes and is filled with coolant. The schematic model of reactor, as simulated by MCNPX2.6, is depicted in Fig. 1 and Fig. 2.

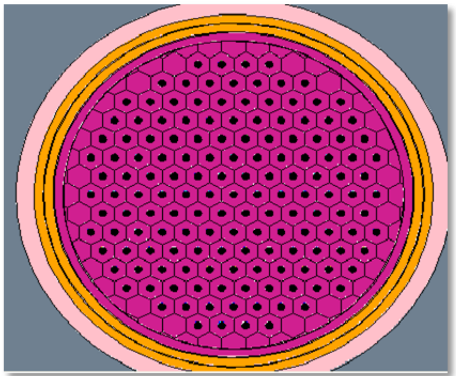

Fig.1 Reactor Core Layout

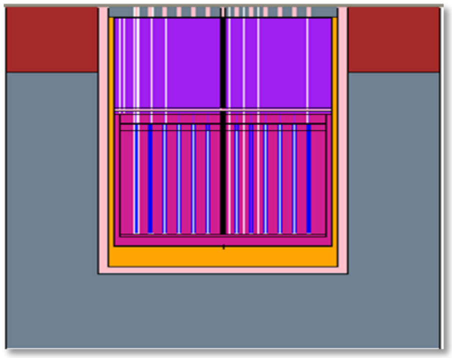

Fig. 2. Schematic Model of Reactor as Simulated By MCNPX

II.B. CENTRAL TEST LOOP RELATED CALCULATIONS

II.B.1. Effect on Multiplication Factor

One of the experimental facilities of the Reactor is the Central Test Loop (C.T.L). The geometrical description of C.T.L is given below in the tableI.

TABLE I. Geometry of C.T.L under Normal Condition

Zone Number	External Radius (Cm)	Material
1	1.80	Coolant
2	1.95	Steel
3	2.10	Coolant
4	2.25	Steell
5	2.75	Coolant
6	3.25	Steel
7	3.40	Vacuum
8	3.80	Steel
9	4.00	Coolant
10	4.10	EH-125

Considering fuel testing, zones 1, 3, and 5 of C.T.L may fill by heavy and light water, and any inert gas such as Helium. Each option has been treated accurately in this study.

All the Control and Protection System elements were withdrawn from the reactor core. The calculations were made at the beginning of life for cold and clean core. In the input file all 150 single cell, as described before, have been simulated. The extension of the fuel assembly to the vessel top, and down to the bottom of the calandria has been modeled exactly. Calandria, vessel, steel plate inside the vessel, heavy concrete on the vessel and all other region have been simulated according to the real geometry.

The results of calculation of multiplication factor are presented in table II. Evaluation of the effect of each case and related error has been done using the following and Error Propagation formula:

$$\Delta\rho = \frac{K - K_{ref.}}{KK_{ref.}} \times 100 \qquad (1)$$

Considering table II, it is clear that all cases have negative reactivity related to the reference case. Especially it is noteworthy that, inserting seven fuels pin (4.4% enriched) in the C.T.L will also impose a negative reactivity, however slightly less than other cases. The negative reactivity is mainly due to the presence of three steel cylinder in the C.T.L. Steel is capable of removing thermal neutrons and at the same time thermalizing fast neutrons. Having in mind that the main portion of neutrons in the core are thermal, the three steel cylinder will reduce thermal neutrons effectively, and therefore there are less thermal fission in the enriched fuel pins to compensate the negative reactivity.

Table II. Effect of C.T.L on Multiplication Factor in Various Configurations

Core Configuration	Keff. ± S.D.	Δρ% ± S.D.
No C.T.L	1.07008 ± 0.00013	_
C.T.L Filled With D2O	1.05601 ± 0.00014	-1.245 ± 0.017
C.T.L Filled With D2O+Fuel	1.05870 ± 0.00013	-1.005 ± 0.016
C.T.L Filled With H2O	1.05553 ± 0.00014	-1.288± 0.017
C.T.L Filled With He	1.05582 ± 0.00014	-1.262 ± 0.017

II.B.2. Effect on Multiplication Factor

The spectrum characteristic at the center of Reactor core, when there is no C.T.L in the core, are illustrated in Fig3.In this figure the spectrum has been normalized by the flux density in the 0 to 0.009 eV energy bins Refs. 2, 4. The shape in the fast-neutron region is basically a fission spectrum at the higher energies, but degraded by scattering in the moderator as evident for energies below 2 MeV. The fast region joins the intermediate (epithermal, resonance, or 1/E) region at roughly 0.5 MeV. As indicated, the intermediate spectrum is usually very close to:

$$\phi(E)d(E) = \phi_0 \frac{d(E)}{E} \qquad (2)$$

This means there is an equal flux density in each logarithmic energy interval. The quantity is proportional to the fast neutron source density and inversely proportional to the slowing-down power of the moderator. The shape of the spectrum in the thermal region is approximately Maxwellian, degrading into the 1/E tail. Because of the absorption and other effects, the spectrum is slightly harder, i.e., contains a greater proportion of higher energy neutrons, than Maxwellian at the temperature of the moderator. The hardening effect has caused the most probable energy of the thermal neutrons to be near 0.03 eV. Calculated spectrums in the C.T.L filled by D2O, H2O, and He are illustrated in Fig4. All of them show the same behavior as described above. It should be noted that the graphs have been normalized by the no-C.T.L flux magnitude in the 0 – 1E-9 energy bin, so actually they are relative flux per unit energy (Ref. 1).

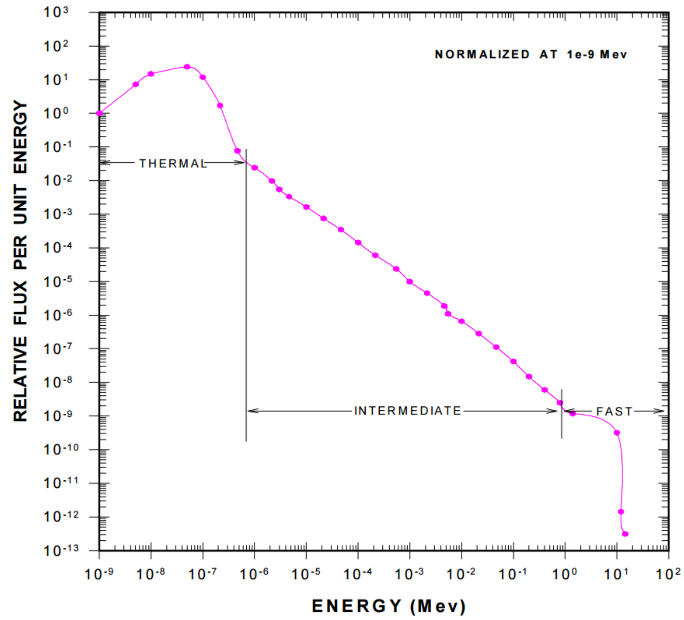

Fig. 3 Neutron spectrum in the center of core without CTL

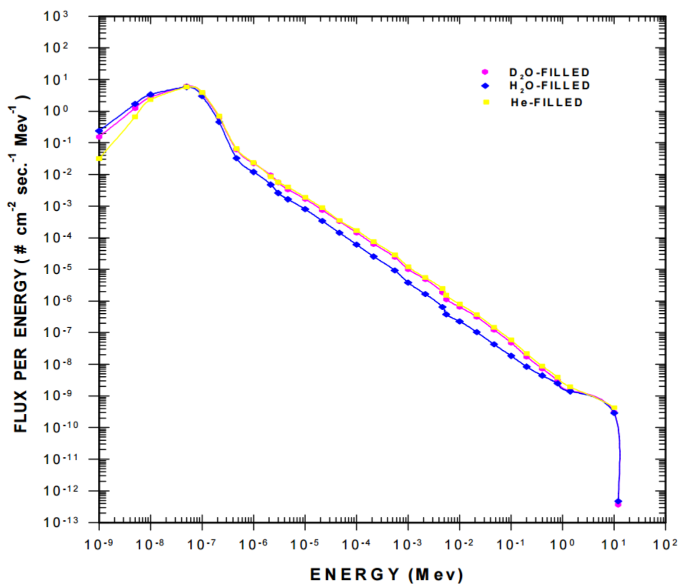

Fig. 4 Comparison between neutron spectrums in various CTL configurations

In table III, the neutron fractions in each energy interval and for various C.T.L configurations have been compared.

TABLE III: Thermal, Epithermal and Fast Neutron Fractions in C.T.L Configurations

Energy Interval	D2O-Filled C.T.L	H2O-Filled C.T.L	He-Filled C.T.L
0 – 0.465 eV	0.997024	0.998462	0.996643
0.465ev – 0.4 MeV	0.002976	0.001528	0.003358
0.4 – 20 MeV	3.3E-10	2.8E-10	4.6E-10

II.B.3. Neutron Flux Distribution

The accurate calculation of flux density is basic to almost every calculation in reactor physics, i.e., knowing the absolute neutron flux density is one of the main objects of reactor physics calculations. In a research reactor, it is often required to know the absolute neutron flux density along the experimental channel or loop.

Axial neutron flux distribution in three different C.T.L configuration of cold and clean core has been calculated. For this, we have used the F4 tally (Track estimate of cell flux) in many cylindrical cells in C.T.L. The dimensions of tally cell have been chosen so large to include as many neutrons as possible. On the other hand, cell dimensions should be small enough to avoid large flux gradient in the cell volume. Thermal and total axial neutron fluxes, in different C.T.L configurations, has been calculated and compared to each other in Fig.5 and Fig.6.

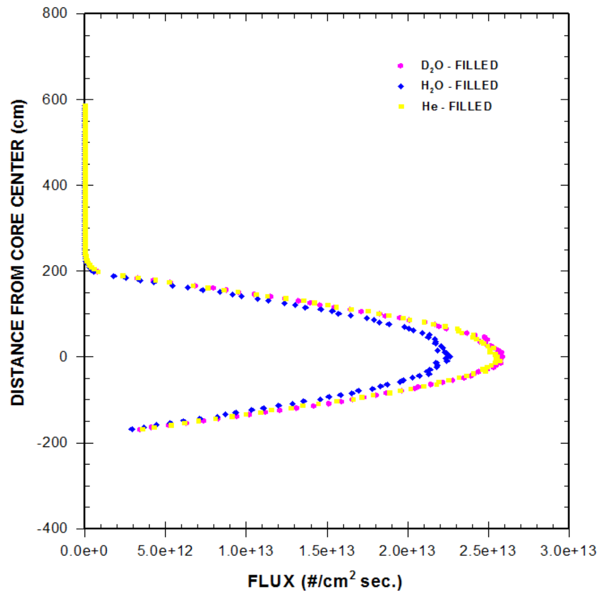

Fig. 5 Thermal Neutron Axial Flux Distribution in Various C.T.L Configurations

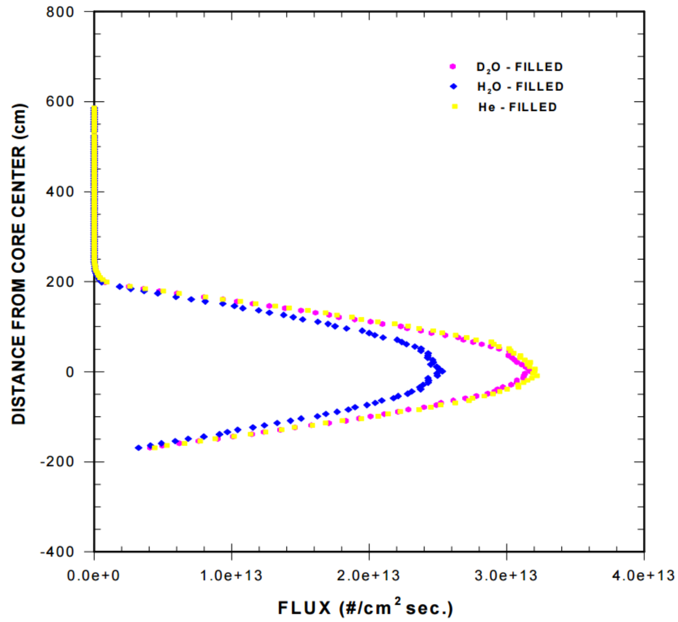

Fig 6. Total Neutron Axial Flux Distribution in Various C.T.L Configurations

Radial neutron flux distribution for H2O-filled, D2O-filled, and He-filled C.T.L has also been determined. The radial flux distributions in thermal and fast energy region are illustrated and compared in Fig.7 andFig.8.

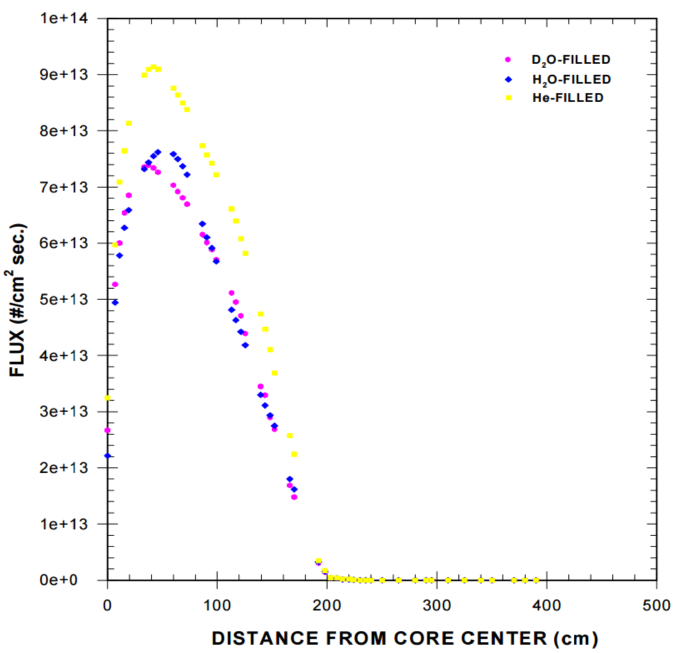

Fig7. Thermal Neutron Radial Flux Distribution in Various C.T.L Configurations

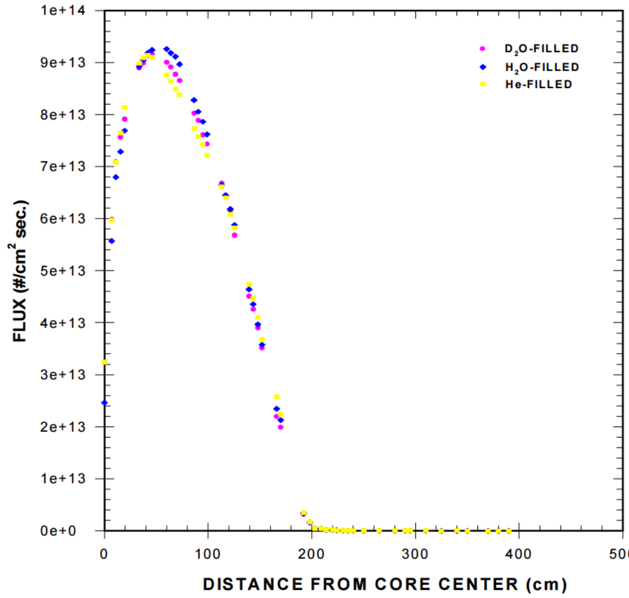

Fig.8.Total neutron radial flux distribution in various C.T.L configurations

IV. CONCLUSION

Neutron spectrums in different C.T.L configuration are sketched in Fig8. It is clear that there is no significant difference in the fast region for three C.T.L configurations. In the intermediate region, the higher absorption cross-section of H2O is responsible of the lower spectrum of H2O-filled C.T.L .This may be a consideration if epithermal neutrons are wanted. Because of the same reason, the spectrum in D2O is less than that for He. In all three cases, the intermediate region, join the thermal region, which is Maxwellian in shape, at about 0.5 eV. The slowing-down power of H2O is more than that of D2O, so more neutrons will scatter from intermediate energy to thermal region by H2O, and hence the thermal spectrum in H2O will be higher than spectrum in D2O and He. Having in mind the very low slowing-down power of He, there is nearly no neutron thermilized by He, and therefore the spectrum of He-filled C.T.L, in thermal region, has been dropped significantly. He-filled C.T.L is the worst case if thermal neutrons are required. Using He-filled C.T.L, there is also another problem with removal the heat generated in enriched fuel. In table3, the neutron fractions in each energy interval and for various C.T.L configurations have been compared. It is apparent that higher thermal neutron will occur in H2O-filled C.T.L, while there are more resonance neutrons in He-filled C.T.L.

It is apparent from Fig.5 and Fig.6 that, thermal and total fluxes in H2O-filled C.T.L are considerably less than D2O and He-filled C.T.L. Radial neutron flux distribution for H2O-filled, D2O-filled, and He-filled C.T.L has also been determined. The radial flux distributions in thermal and fast energy region are illustrated and compared in Fig.7 and Fig.8. Considering these figures, it is clear that inserting H2O-filled C.T.L in the cold core will results in less thermal neutron flux density near the core center. Compared with the two other cases, this effect will cause the maximum flux to be occurring far from the core.

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