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Reactor Dosimetry Study of the Rhode Island Nuclear Science Center

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INTRODUCTION

The Rhode Island Nuclear Science Center (RINSC), located on the Narragansett Bay Campus of the University of Rhode Island, is a state-owned and US NRC-licensed nuclear facility constructed for educational and industrial applications [1]. The main building of RINSC houses a two-megawatt (2 MW) thermal power critical reactor immersed in demineralized water within a shielded tank. As its original design in 1958 by the Rhode Island Atomic Energy Commission focused on the teaching and research use of the facility, only a minimum of 3.85 kg fissile uranium-235 was maintained in the fuel elements to allow the reactor to reach a critical state. In 1986 when RINSC was temporarily shutdown to start US DOE-directed core conversion project for national security reasons, all the U-Al based Highly-Enriched Uranium (HEU, 93% uranium-235 in the total uranium) fuel elements were replaced by the newly developed U_3Si_2-Al based Low Enriched Uranium (LEU, $\leq$20% uranium-235 in the total uranium) elements [2]. The reactor first went critical after the core conversion was achieved in 1993, and feasibility study on the core upgrade to accommodate Boron Neutron-Captured Therapy (BNCT) was completed in 2000 [3].

The 2-MW critical reactor at RINSC which includes six beam tubes, a thermal column, a gamma-ray experimental station and two pneumatic tubes has been extensive utilized as neutron-and-photon dual source for nuclear-specific research in areas of material science, fundamental physics, biochemistry, and radiation therapy. After the core conversion along with several major system upgrade (e.g. a new 3-MW cooling tower, a large secondary piping system, a set of digitized power-level instrument), the reactor has become more compact and thus more effective to generate high beam flux in both the in-core and ex-core regions for advance research. If not limited by the manpower and operating budget in recent years, the RINSC built "in concrete" structure and control systems should have been systematically upgraded to a 5 MW power facility to further enhance its experimental capability while still maintaining its safe margin as designed [2].

In this paper theoretical prediction of the flux and dose of neutrons and photons transported from the in-core region to the ex-core irradiation facility (thermal column) is presented, based on Monte Carlo results obtained from the processing of MCNP code [4]. The 5-ft wide 5-ft high and 10-ft long thermal column which was constructed by piled graphite blocks has been fully simulated by code to complement earlier MCNP model that only includes the in-core geometry for the fuel conversion analysis [3]. Dosimetry data collected from thermo-luminescent dosimeters (TLD badges containing ^{6}LiF or ^{7}LiF chips) and bare

and cadmium-covered gold foils are also presented through comparison with computed results. Details of the code model and dosimeter utilization are described in the paper.

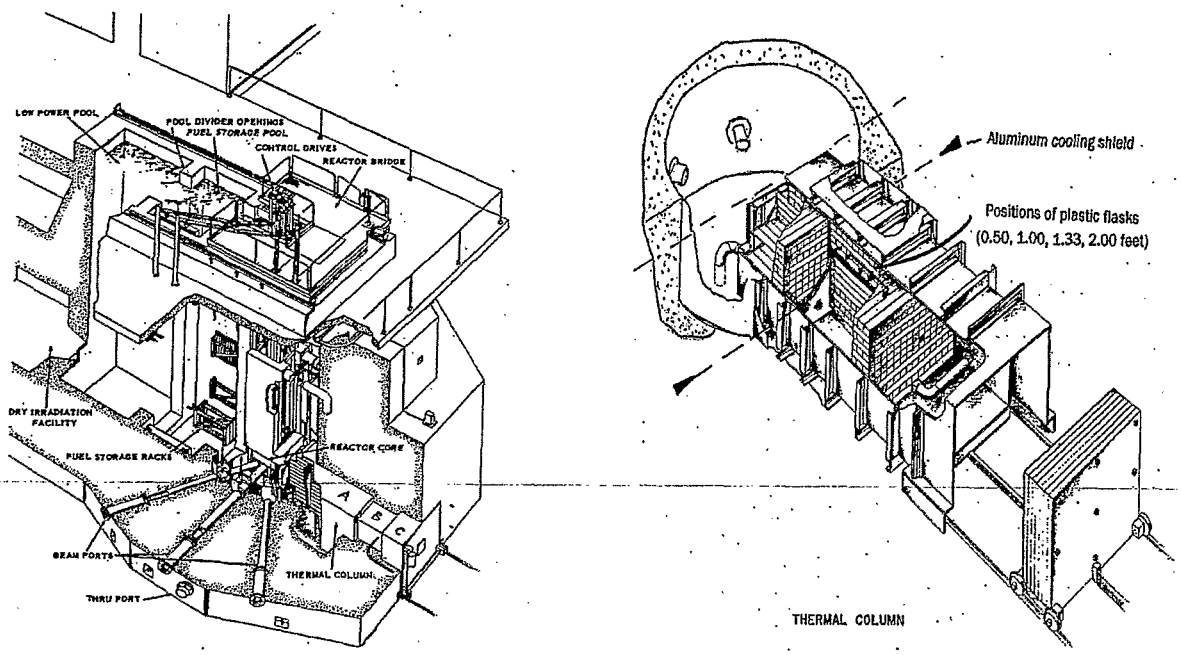

Figure 1. A scaled RINSC facility shown in the left includes the reactor core, horizontal beam ports, and control rod system in the fuel pool. The graphite thermal column (A-B-C in left figure) developed for sample irradiation is shown in the right.

REACTOR CORE AND IRRADIATION FACILITY

The RINSC reactor as shown in Figure 1 is a 2 MW thermal power, university-type research reactor immersed in a 36,000-gallon pool of demineralized water. The pool is contained in an open shielded tank, divided into three sections (low power section, fuel storage section, high power section), that is approximately 22.5-ft long, 8.5-ft wide, and 32-ft deep. The high power section is circular with a diameter of 8.5-ft, and a depth of 32-ft. Neutron beam ports, 6 and 8 inches in diameter, penetrate the biological shield and terminate at the core face. Two pneumatic systems confined within 2-in diameter tubes also terminate at the core face. A 10-ft long thermal column indicated by the regions A, B, and C in Figure 1 extends through the shield to the high power end of the pool.

The reactor core as shown in Figure 2 is configured from individual fuel elements placed in a core grid box at 26-ft beneath the pool surface. The grid box is attached to the base of a suspension frame which in turn is attached above the pool surface to a movable bridge that spans the width of the pool. Control of the reactor is achieved through the use of four shim-safety blades and one regulating blade. The safety blades are an alloy of aluminum and boron (B_4C based boral™); the regulating blade is a hollow, stainless steel, rectangular tube. The shim-safety blades are held up by electromagnets at the ends of aluminum extensions that attached to drive motors above the pool surface. The normal method for

shutting down the reactor is insertion of the blades with their drive motors. In the event of emergency, electric current can be turned off to the electromagnets and blades will drop freely in the core.

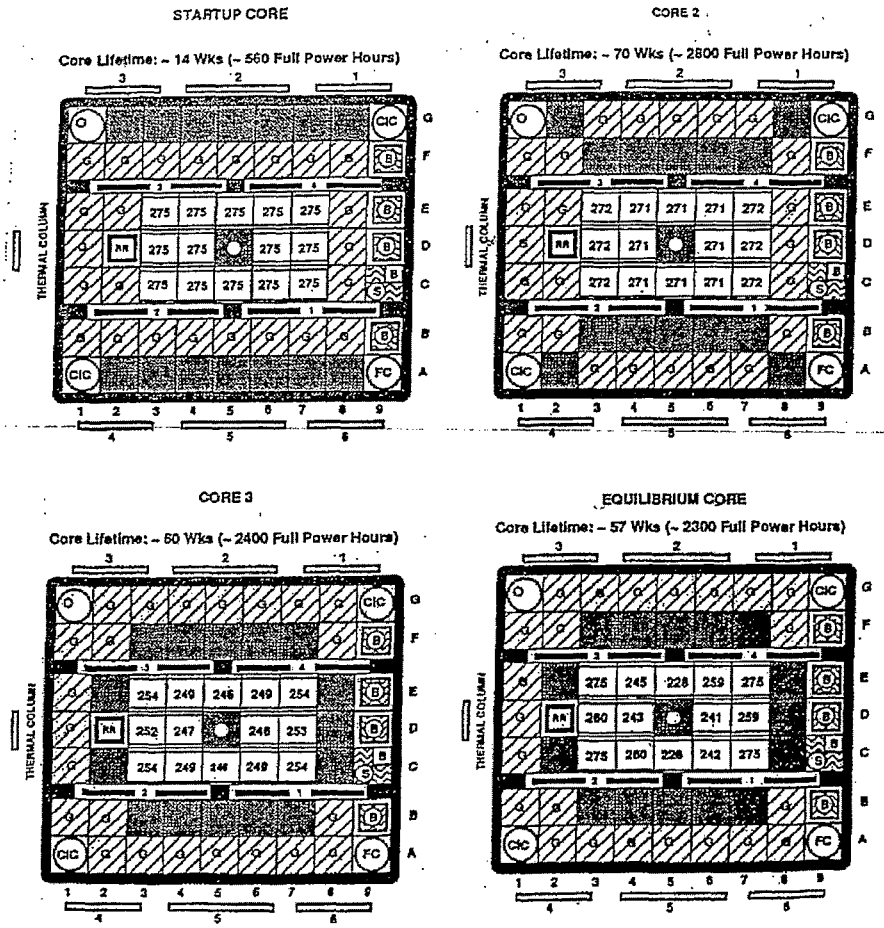

Figure 2. The gram amounts of uranium-235 in 14 HEU elements at 1964 startup, transition (core 2 and 3), and equilibrium core in the RINSC reactor where lifetimes were based on recorded operation of 8 hours per day and 5 days per week.

The reactor's thermal column is a graphite pile, which is 5-ft by 5-ft by 10-ft long within a concrete biological shield. There is a 3-in thick lead shield and a 2-in thick aluminum cooling plate (31-in downstream of the lead shield) between the reactor core and the thermal column, which extends to the outer face of the concrete shield (within a 1-in steel framing). The thermal column is made up of consecutive graphite blocks, each being 4-in by 4-in in cross section. Near the centerline of the thermal column, a 2-in by 2-in air beam hole has been designed to accommodate samples and apparatus for experimental usage. During the criticality operation in 1964 at 1-MW power, the ex-core peak flux of thermal neutrons (<5.3 keV) and fast neutrons (>821 keV) from the HEU fuel elements were, respectively, 1.5×10^8 n/cm^2/sec and 1.5×10^7 n/cm^2/sec at the cooling plate, the location where the air beam hole starts. The ex-core peak flux of thermal neutrons and epithermal

neutrons (5.3-821 keV) measured in 1996 at 100 kW power from the LEU fuel elements were, respectively, 3.9×10^9 n/cm^2/sec and 1.3×10^7 n/cm^2/sec at the cooling plate.

The name of the thermal column reflects the goal of its design and the property of its constituent; in that most of the core neutrons transported through the long beam path in graphite should be efficiently moderated before being absorbed by the material in the target samples (tallies in model). Since neutrons in the thermal energy (<5.3 k eV) dominate the spectrum of the neutron flux up to 10 MeV, the count of fast neutrons (>821 keV) in the thermal column requires the use of flux-reduction technique (e.g. using cadmium to shield thermal neutrons) to improve the accuracy of the fast neutrons. In the gamma-ray dose rate measurements, due to the presence of a 3-in lead shield in an aluminum tank (0.25-in thick wall) next to the reactor core, only the photons at >1 MeV can be detected within an acceptable statistical uncertainty (<10% at one-sigma level). Since the decay of gamma-rays in the thermal column has been confirmed to follow an exponential curve based on the data obtained from both calculations and measurements, the dose rate at the front-face of the thermal column (6-in from the core edge, 2.5-in behind the 3-in lead shield) will be ~7.8×10^8 mR/hr under 1 MW operation. At the aluminum cooling plate, (nearly 3-ft from the core edge), the gamma-ray dose rate will be reduced to 8.2×10^6 mR/hr. At 5-ft and 8-ft from the face of the thermal column, the predicted dose rate will be down to 9.2×10^5 mR/hr and 4.3×10^4 mR/hr, respectively.

COMPUTATIONAL MODEL

Simulation of the RINSC reactor for core conversion analysis has been thoroughly performed during 1986-1993 by the use of Monte Carlo particle transport code MCNP [4]. This statistical-based code that has been developed and periodically updated by the Los Alamos National Laboratory is a general purpose Fortran compiled software package, which can be used to model any single particle motion or coupled neutron photon transport in a 3-D geometry consisting of different material regions. For the in-core parameter analysis [3], detailed geometrical configuration of the key elements plus full set of material cross sections (including the cross section sets for thermal neutron treatment) must be incorporated into the model input for code processing in order to obtain results with high accuracy. For the ex-core irradiation studies, due to the presence of multiple material regions through which most of the core particles, if not absorbed by or leaked out from the system, must have been slowed down via inelastic scattering before reaching samples or TLDs, a homogenized reactor core (diminished angular and distant-effect of point sources in the core to remote tallies) followed by segmented material zones (to orient particle transport in geometry by optional weighting factors) can be set up in model to expedite code run while maintaining the source strength to its downbeam areas. To judge the adequacy of model simplification, checking on the core criticality and flux distribution based on the present reactor status (e.g. uranium consumed, sample used) are essential.

The MCNP model developed for the in-core and ex-core analysis can be observed from Figure 3, in which the left one represents a full-core geometry comprising fuel plates and control blades in the central elemental cells surrounded by the reflector (beryllium) and

moderator (graphite) cells in symmetrical to the core midplane. The right two figures, respectively, represent a simplified RINSC core and a full-reactor geometry containing homogenized material regions and segmented thermal column. To expand model geometry from a full core (in-core analysis) to a full reactor (ex-core studies), sets of criticality test-runs according to the actual amount of ^{235}U (3.5-kg in equilibrium core and 3.85-kg in fresh core), ^{238}U (4 times the amount of ^{235}U in LEU fuel), ^{14}Si and ^{27}Al (in atomic ratio of U:Si:Al=3:2:1 for the U_3Si_2-Al fuel), ^{10}B and ^{11}B (in ^{10}B:^{11}B=1:4 for natural boron in 4 control blades), ^{56}Fe (1 regulating blade), ^{1}H and ^{16}O (space-filled light water in ratio of ^{1}H:^{16}O=2:1), ^{12}C and ^{9}Be (reflector elements), concrete (biological shield), and natural Pb (gamma shield) must be performed through adjustment on boron contents in the core to approach K_{eff} = 1.00 + an excess reactivity. The term of excess reactivity is reserved to balance the negative worth measured from target samples and irradiated apparatus. In the code input of material section, the boron alone ($^{10,11}B$) has been set as adjustable because of the fact that all 4 safety blades are actually located outside of the fuel elements and thus are less effective in neutron absorption than that by the uniformly-distributed boron in a homogenized core as set in model. Furthermore, under reactor criticality the 4 safety blades must have been lifted to a fixed bank position below which the fuel elements are free from blade shielding (i.e. no neutron absorption), thus yielding a non-uniform boron distribution in the core which is different from the material homogeneity setup in code. In additional to the criticality test, the reliability of calculated flux by code runs also needs to be checked to ensure the model adequacy. This can be approached by flux normalization in terms of measured data points along the beam path upstream from where flux tallies are normally setup. At RINSC reactor, the reference points locate at the core edge and cooling plate.

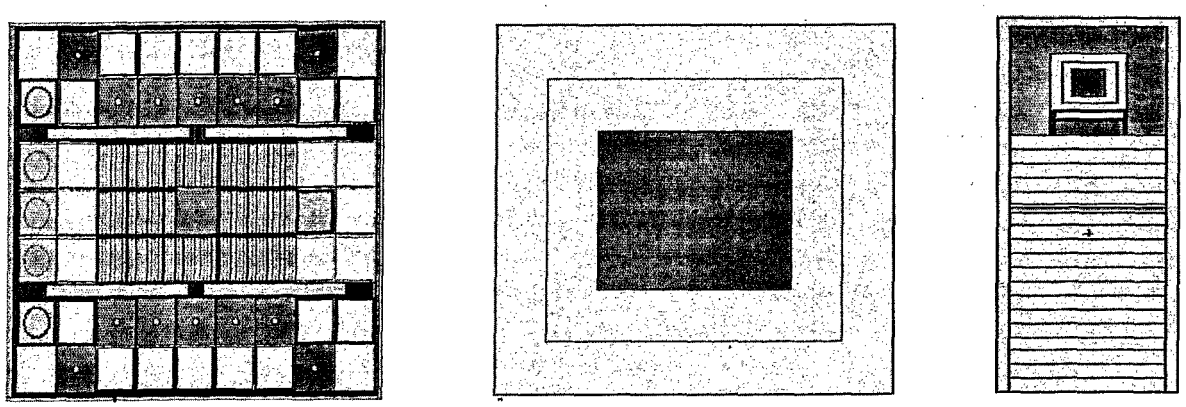

Figure 3. The left figure shows the MCNP geometry for in-core parameter analysis and the right two figures show the homogenized core and segmented thermal column for the ex-core tally (flux and dose) estimate.

SAMPLE IRRADIATION

Using gold foils in the experiment conducted at the thermal column in 1996, the reactor was first brought up to a sub-critical point by removing the four safety shim blades. Then

the wood pole with the gold foil on it was inserted into the beam hole of the thermal column, and subsequently the reactor was brought up to 100 kW for a 1-minute sample irradiation. The reactor was then scrammed for sample removal and counting. Both with and without the cadmium shields, the readings on the gold foils are listed in Table 1 below.

Table 1. Thermal neutron irradiation data from bare and cadmium-covered gold foils at 100 kW

Distance (ft)	Bare Gold Foil (μCi/g)	Cadmium-covered Gold Foil (μCi/g)
0.5	5.159	0.856
1.0	4.922	0.044
2.0	1.267	-------
3.0	0.443	0.238*

Note: ------ indicates that no reading was made at this location.
* indicates that the cadmium-covered gold foil at 3 feet was partially uncovered.

When counted, the data set collected from the four bare foils looks acceptable, but as noted, the data from the cadmium-covered foils at 3-ft seems problematic.

In the latest sample irradiation at RINSC thermal column during November 2004 to January 2005 using the ^{6,7}LiF-based TLD badges, the average dose rates of neutrons and photons obtained from a critical operation of the reactor at 100 to 300 kW power in 2 to 7 minutes exposure periods are listed in Table 2. Note that due to the initial startup of the reactor by lifting the 4 safety blades step-wise from criticality at ~ 10 W until a steady power level of 100 kW is reached, the actual exposure time of the TLD in the thermal column is estimated to be about 1-minute longer (~ 3 to 8 minutes total irradiation period).

Table 2. Neutron and photon dose rates from TLD badges under 100 kW and 3-8 minutes exposure

Distance to Cooling Plate (ft)	Neutron Dose (mR/hr)	Photon Dose (mR/hr)
1.0	1.4E07	1.3E06
2.0	4.8E06	3.7E05
2.5	4.7E06	2.3E05
3.0	3.5E06	1.2E05

CONCLUSIONS

At the RINSC reactor, fission energy neutrons generated from the LEU core are slowed to epithermal energy neutrons through the core edge regions and further slowed down (moderated) into thermal neutrons in the thermal column. When bare and cadmium-covered gold foils were used in the earlier measurements (1993-1996), the measured data

compared to MCNP calculated fluxes of thermal and epithermal neutrons fall within the uncertainty of 10% and 20%, respectively as shown in Figures 4 and 5 below. No photon flux was specifically measured at the time of the neutron flux measurement. The efforts dedicated to the post core conversion analysis (1993-1996), followed by the major facility upgrade needed for the BNCT project preparation, dominated the limited working time along with the on demand operation of the RINSC reactor. From the neutron-and-photon dual mode MCNP calculations, the flux of photons after normalization with respect to the flux tally obtained at aluminum cooling plate (~ 3-ft from the core edge) is shown in Figure 6. Roughly according to an exponential decay, the flux of photons down the thermal column to 200 cm (~ 6.6 ft) is included in the same plot for future reference.

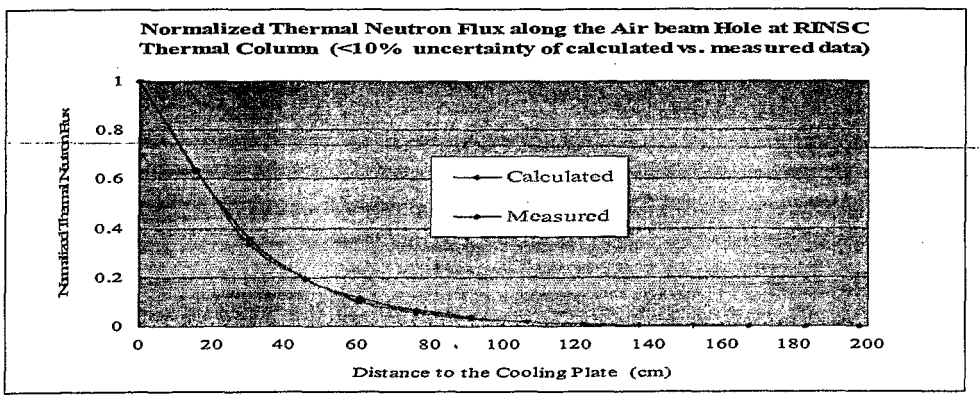

Figure 4. Comparison of MCNP calculated versus foil measured thermal neutron flux along the central air beam hole at RINSC thermal column.

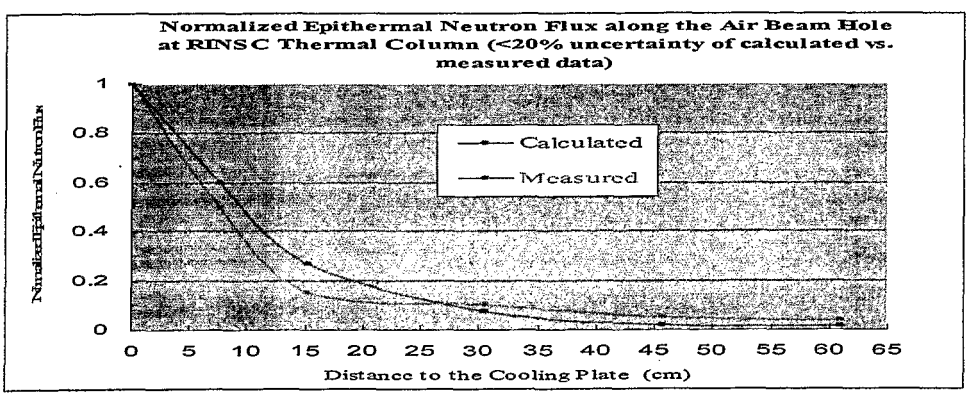

Figure 5. Comparison of MCNP calculated versus foil measured epithermal neutron flux along the central air beam hole at RINSC thermal column.

The use of ^{6,7}LiF-TLDs and gold foils for dose estimation of the neutrons and photons in the thermal column was recently performed during November 2004 until January 2005. Measured data at 100 to 300 kW power for ~ 3 to 8-minute exposure time (including a 1-minute extension for the reactor startup) are presented in Table 2. The normalized curve

plotted along the data (in red, with +/- uncertainty) from equal-distant TLDs setup in line within the air beam hole is depicted in Figures 7 and 8, in which the predicted dose by the code MCNP (in blue) is also included for comparison. In most of the measurements conducted previously and presently, the result's uncertainty has been found to be < 9% of the population mean, while the maximum difference has been 23% which occurred at downstream sample locations (> 3 feet) in the thermal column or occurred due to a short irradiation period (of ~ 2 to 7 minutes). Compared to the calculated dose from MCNP runs, the maximum discrepancy for the neutron dose is ~ 23% and for the gamma dose is ~ 14%. Both of these maxima occur at the furthest location from the cooling plate (> 3 feet).

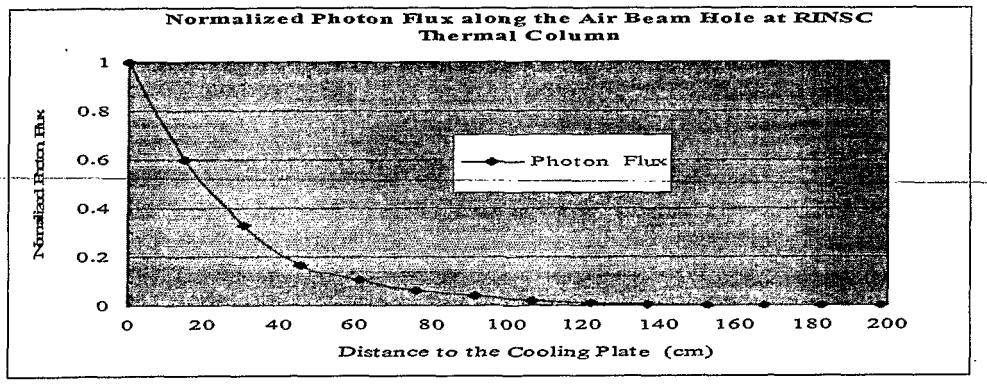

Figure 6. MCNP calculated photon flux along the central air beam hole in thermal column.

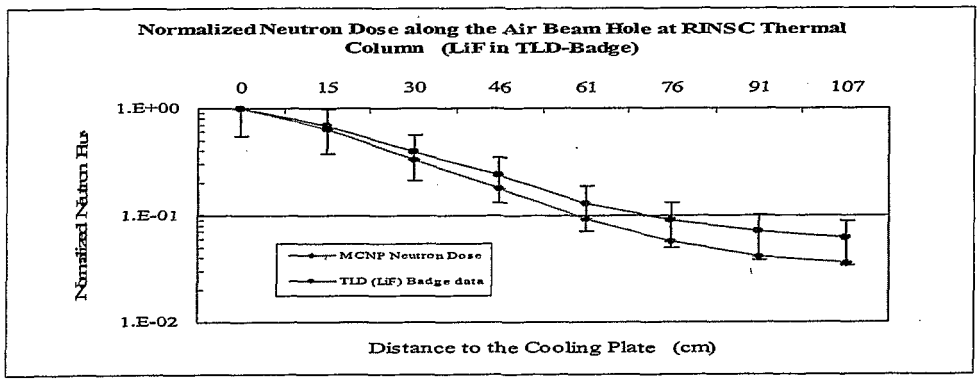

Figure 7. Comparison of MCNP calculated versus TLD measured neutron dose along the central air beam hole at RINSC thermal column.

It is unsurprising that the dose discrepancy larger for neutrons than for photons since the code embedded flux-to-dose conversion for the epithermal neutrons (0.5 eV to 100 keV) is up to 30 times higher than that for the thermal neutrons (< 0.5 eV), while the conversion for prompt photons (1 MeV to 15 MeV) only varies slightly up to 6.73. To ensure the reliability of radiation dose from calculations, 2 sets of conversion factors [5] are recommended to be used in the future.

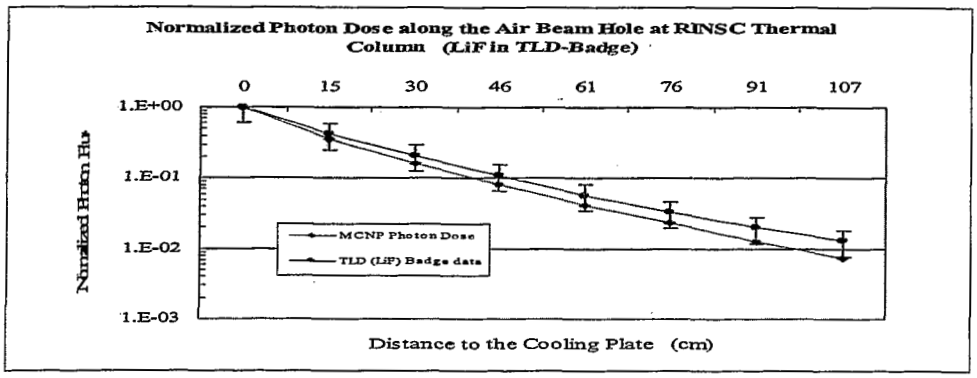

Figure 8. Comparison of MCNP calculated versus TLD measured photon dose along the central air beam hole at RINSC thermal column.

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