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FURTHER DOSIMETRY STUDIES AT RHODE ISLAND NUCLEAR SCIENCE CENTER

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DISCLAIMER

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ABSTRACT: The RINSC is a 2 mega-watt, light water and graphite moderated and cooled reactor that has a graphite thermal column built as a user facility for sample irradiation. Over the past decade, after the reactor conversion from a highly-enriched uranium core to a low-enriched one, flux and dose measurements and calculations had been performed in the thermal column to update the ex-core parameters and to predict the effect from in-core fuel burn-up and rearrangement. The most recent data from measurements and calculations that have been made at the RINSC thermal column since October of 2005 are reported.

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. The main building of RINSC houses a 2 mega-watt (MW) thermal power critical reactor immersed in demineralized water within a shielded tank. In 1986, RINSC was temporarily shutdown to start a US DOE-directed core conversion project for national security reasons. 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, ≤20% uranium-235 in the total uranium) elements [1]. The reactor first went critical after the core conversion was achieved in 1993, and a feasibility study on a core upgrade was completed in 2000 [2].

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 extensively utilized as a neutron-and-photon dual source for nuclear-specific research in the 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 instruments), the reactor has become more compact and thus more effective in generating a high neutron flux in both the in-core and ex-core regions for advanced research.

In this paper, thermoluminescent dosimeter (TLD) measurements using TLD-100 (natural Li of 7.5% ^{6}Li and 92.5% ^{7}Li for both the neutron and gamma dose rate measurements), TLD-600 (enriched ^{6}Li to 95.6% for neutron dose rate measurement only) and TLD-700 (enriched in ^{7}Li to 99.93% for gamma dose rate measurement only) dosimeters were performed in the 1.5 m wide, 1.5 m high and 3 m long graphite formed thermal column. The results were compared to earlier measured dose rates and to the Monte Carlo MCNP code [3] calculated values. These data that were supplemented with the recent measurements made on the flux density of thermal and epithermal neutrons using the bare and cadmium-covered gold foils, are presented.

In additional to these measurements at RINSC, a separate set of test measurements were performed at the Brookhaven National Laboratory (BNL). A calibrated neutron source was used to verify the neutron dose as measured by the TLD dosimeters. Results indicate that, in the range from 5 mSv to 35 mSv, the dose measured by the TLD dosimeters was about 5% to 10% higher than the delivered dose.

Reactor Core and Irradiation Facility

The RINSC reactor, as shown in Figures 1 and 2, is a 2 MW thermal power, university-type research reactor immersed in a 136,800-liter 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 6.75-m long, 2.55-m wide, and 9.6-m deep. The high power section is circular with a diameter of 2.55-m and a depth of 9.6-m. Neutron beam ports, 15-cm and 20-cm in diameter, penetrate the biological shield and terminate at the core face. Two pneumatic systems confined within 5-cm diameter tubes also

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terminate at the core face. A 3-m 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 is configured from individual fuel elements placed in a core grid box located at 7.8-m 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^{TM}); 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 are attached to drive motors above the pool surface. The normal method for shutting down the reactor is to insert the blades with their drive motors. In the event of emergency, electric current can be turned off to the electromagnets and the blades will drop freely into the core.

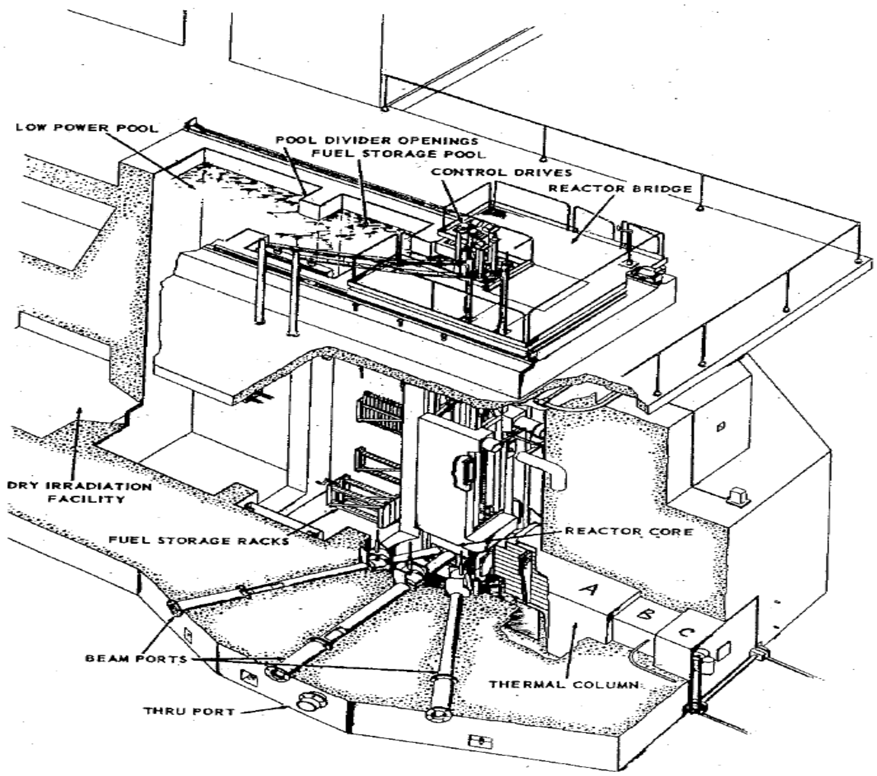

Figure 1. The RINSC reactor facility includes the reactor core, horizontal beam ports, control rod system, and graphite thermal column (A-B-C). The thermal column is developed for sample irradiation.

The reactor's thermal column is a graphite pile, which is 1.5-m wide by 1.5-m high by 3-m long within a concrete biological shield. There is a 7.5-cm thick lead shield and a 5-cm thick aluminum cooling plate (78-cm 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 2.5-cm steel framing). The thermal column is made up of consecutive graphite blocks, each being 10-cm by 10-cm in cross section. Near the centerline of the thermal column, a 5-cm by 5-cm air beam hole has been designed to accommodate samples and apparatus for experimental usage. In 1996, measurements at 100 kW power with LEU fuel elements in the core, the ex-core peak flux of thermal neutrons (<5.3 keV) and epithermal neutrons (5.3 to 821 keV) were, respectively, $3.9x10^9$ n/cm^2-sec and $1.3x10^7$ 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 the model). Since neutrons in the thermal energy region (<5.3 keV) 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 a 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 7.5-cm thick lead shield in an aluminum tank (0.5-cm thick wall) next to the reactor core, only the photons at >1 MeV can be detected within an acceptable statistical uncertainty (<10% at the 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 (which is 15-cm from the core edge, 6.25-cm behind the 7.5-cm lead shield) will be ~7.8×10^3 Gy/hr at 1 MW operation. At the aluminum cooling plate, (nearly 1-m downstream from the core edge), the gamma-ray dose rate will be reduced to 82 Gy/hr. At 1.5-m and at 2.5-m from the face of the thermal column, the predicted dose rate will be reduced to 9.2 Gy/hr and 0.43 Gy/hr, respectively.

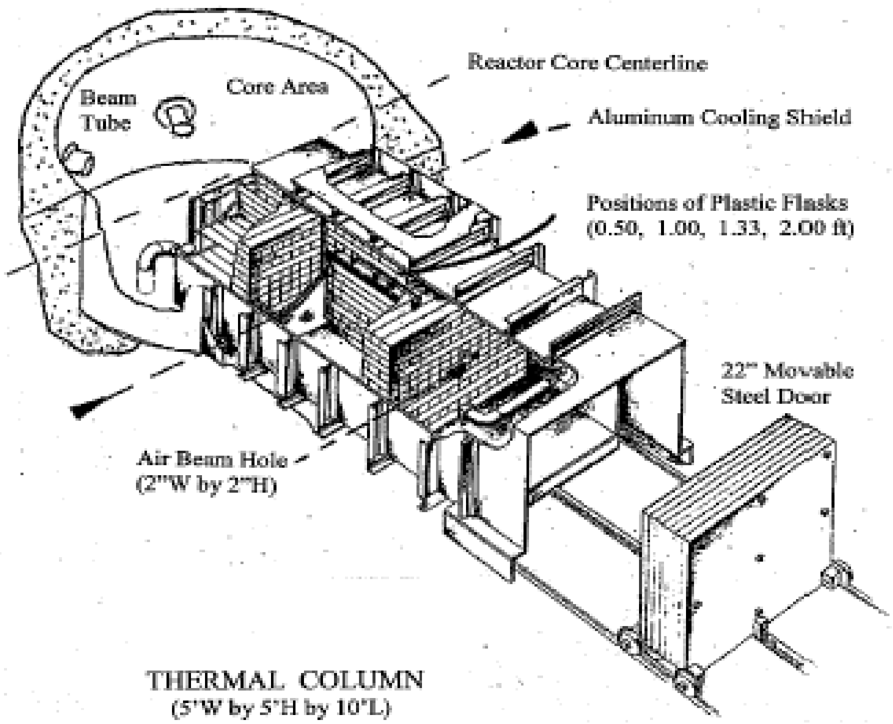

Figure 2. The graphite thermal column at RINSC is modeled by the MCNP for flux and dose prediction along the center air beam hole.

MCNP 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 a 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, there are multiple material regions through which most of the core particles must have been slowed down via inelastic scattering before reaching the target samples or the TLDs (unless they have been absorbed or they have leaked from the system). In the MCNP model, a homogenized reactor core can be set up, followed by segmented material zones, to expedite the code run, while maintaining the source strength to the areas further down the thermal column. Checks on the core criticality and the flux distribution (based on the present reactor status of the uranium consumed and the sample used) are essential in order to judge the adequacy of the model simplification.

The MCNP model developed for the in-core and ex-core analysis can be observed in Figure 3, where the left figure represents a full-core geometry comprised of fuel plates (F) and control blades (B) in the central elemental cells (C), surrounded by the reflector (beryllium, R) and the moderator (graphite, G) cells located symmetrically around the core mid-plane. The right two figures, respectively, represent a simplified RINSC core and a full-reactor geometry containing homogenized regions and the segmented thermal column.

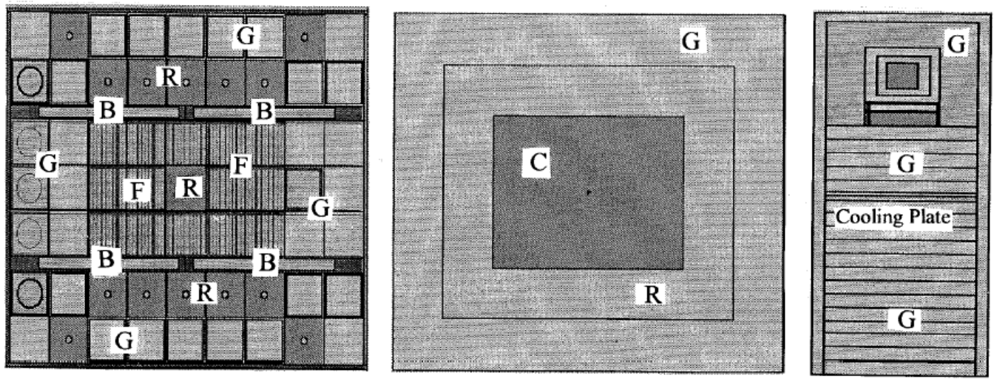

Figure 3. The left figure shows the MCNP geometry for the in-core parameter analysis and the right two figures show the homogenized core and the segmented thermal column for the ex-core tally estimation.

Results

Measurements were made in 2005, using both bare and cadmium-covered gold foils, to determine both the thermal and epithermal neutron flux. They were made at the duplicate positions from the 1996 measurements. The measured data at the various distances from the cooling plate are presented in Tables 1 and 2, where the calculated flux by MCNP is also included. The flux in tables is given in unit of neutrons/cm^2-sec.

Table 1. Thermal neutron flux measured in 1996 and 2005 versus the flux calculated by the MCNP code at 2MW operating power. (MCNP calculations are normalized from the cooling plate)

Distance to the Cooling Plate	Thermal Flux (1996)	Thermal Flux (2005)	Thermal Flux (MCNP)
0 cm	35.26 E+8		35.26 E+8
15 cm	32.07 E+8		20.80 E+8
30 cm	11.49 E+8	11.49 E+8	10.22 E+8
45 cm	6.79 E+8		6.79 E+8
60 cm	3.99 E+8	5.42 E+8	5.11 E+8
75 cm	2.40 E+8		3.07 E+8
90 cm	1.40 E+8	2.11 E+8	1.80 E+8

Table 2. Epithermal neutron flux measured in 1996 and 2005 versus the flux calculated by the MCNP at 2 MW operating power. (MCNP calculations are normalized from the cooling plate)

Distance to the Cooling Plate	Epithermal Flux (1996)	Epithermal Flux (2005)	Epi thermal Flux (MCNP)
0 cm	13.2 E+6		50.7 E+7
15 cm	2.14 E+6		14.2 E+7
30 cm	1.16 E+6	44.7 E+6	36.6 E+6
45 cm	0.60 E+6		16.3 E+6
60 cm	0.39 E+6	12.0 E+6	7.2 E+6
75 cm	0.28 E+6		
90 cm	0.22 E+6	0.47 E+6	

(Thermal neutron flux measurements from 1996 and 2005 are in better agreement than the epithermal flux measurements. This may be due to differences in the respective cadmium covers from the two experiments, which would affect the cutoff energy.)

In tables 3 and 4, the gamma-ray dose rate and neutron dose rate calculated by the MCNP code is compared with the dose rates measured during the 2004 and 2005 experiments, at various locations in the thermal column. Chips of TLD-100 (natural Li) and TLD-700 (enriched in ^{7}Li to 99.93%) were placed along the center air beam hole (5 cm wide by 5 cm high), which starts from the cooling plate and horizontally extends to the outer face of the thermal column wall.

Table 3. The gamma-ray dose rate measured in 2004 and 2005 versus the values calculated by MCNP at 2 MW (MCNP calculations are normalized from the cooling plate)

Distance to the Cooling Plate	2004 TLD-100	2005 TLD-100	2005 TLD-700	γ - Dose Rate (MCNP)
30 cm	13.0 Gy/hr	11.4 Gy/hr		9.8 Gy/hr
60 cm	3.70 Gy/hr	3.9 Gy/hr		2.8 Gy/hr
75 cm	2.30 Gy/hr			1.8 Gy/hr
90 cm	1.25 Gy/hr	3.3 Gy/hr	3.60 Gy/hr	1.0 Gy/hr
120 cm			0.14 Gy/hr	
150 cm			0.04 Gy/hr	

Table 4. The neutron dose rate measured in 2004 and 2005 versus the values calculated by MCNP at 2 MW (MCNP calculations are normalized from the cooling plate)

Distance to the Cooling Plate	2004 TLD-100	2005 TLD-100	2005 TLD-600	Neutron dose rate (MCNP)
30 cm	140 Sv/h	51.5 Sv/h		45.0 Sv/hr
60 cm	48 Sv/h	14.1 Sv/h		11.0 Sv/hr
75 cm	47 Sv/h			7.3 Sv/hr
90 cm	35 Sv/h	11.6 Sv/h	7.0 Sv/h	4.9 Sv/hr
120 cm			5.8 Sv/h	3.6 Sv/hr
150 cm			2.1 Sv/h	2.2 Sv/hr

BNL Neutron Test Measurements

An experiment was conducted at BNL to check the linearity of the neutron response (^{6}Li and ^{7}Li isotopes) from TLD badges at exposures from 0.05 to 0.35 mSv. Seven sets of 3 TLDs each were exposed to an unmoderated ^{252}Cf neutron source. The nominal delivered exposures for each 3-TLD set varied from 0.05 to 0.35 mSv in increments of 0.05 mSv.

An average of the reported dose at each delivered exposure was calculated, along with a one standard deviation for each average. These data are presented in Table 5. An average bias (±) of the reported dose versus the nominal delivered dose was calculated and is presented.

Table 5. The TLD delivered dose versus reported dose are shown with standard deviation and averaged bias

Delivered Dose	Reported Dose	Standard Deviation	Averaged Bias
5 mSv	5.7 mSv	0.4 mSv	± 13.3 %
10 mSv	10.9 mSv	0.6 mSv	± 8.6 %
15 mSv	16.4 mSv	1.4 mSv	± 9.3 %
20 mSv	21.7 mSv	0.4 mSv	± 8.4 %
25 mSv	26.4 mSv	2.8 mSv	± 5.6 %
30 mSv	32.1 mSv	1.0 mSv	± 7.0 %
35 mSv	37.3 mSv	1.1 mSv	± 6.6 %

For the TLD badges all but one individual badge (a 23.3 mSv value reported for a 25 mSv exposure) showed a positive bias. The greatest difference between the reported dose and the delivered exposure is 3.8 mSv (a 28.8 mSv value reported for a 25 mSv exposure). The highest percent difference is at the nominal 5 mSv (5.7 mSv reported for a ±13.3% bias). The average bias is 8.4%.

Conclusions

At the RINSC reactor, fission energy neutrons generated from the LEU core are slowed down to epithermal energy neutrons through the core edge regions and further slowed down (moderated) to thermal neutrons energies in the thermal column.

Gold foils and ^{6,7}LiF-TLDs were used to measure the neutron and photon doses and neutron fluxes during 2004-2005. The measured neutron fluxes are shown in Tables 1 and 2. The normalized curves of thermal and epithermal neutron fluxes along the central air beam hole are plotted in Figures 4 and 5, respectively.

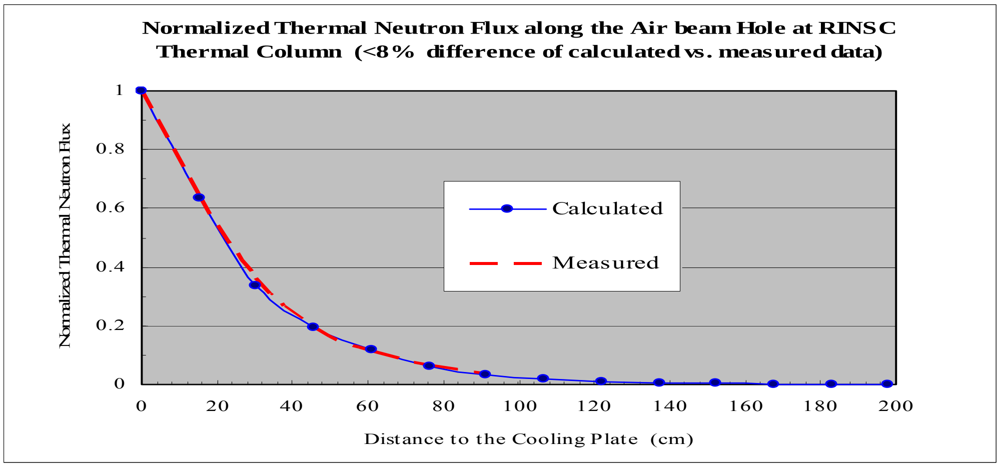

Figure 4. The MCNP calculated thermal neutron flux vs. the gold foil measured flux along the central air beam hole at thermal column.

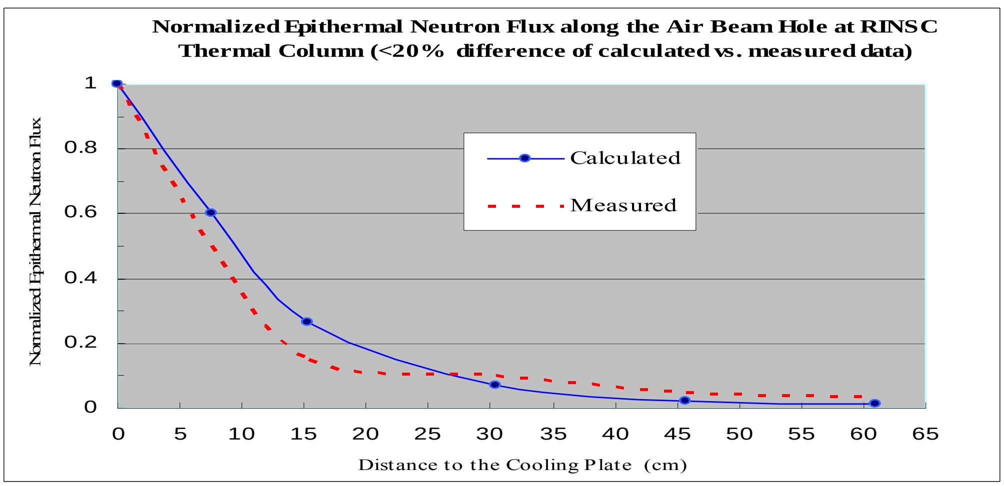

Figure 5. The MCNP calculated epithermal neutron flux vs. the gold foil measured flux along the central beam hole at thermal column.

Both the neutron dose and the photon dose measured by TLDs are compared with the MCNP calculations in Figure 6 (neutron) and Figure 7 (photon). Note that based on the data listed in Table 5 (comparing the delivered neutron dose to the dose reported by the same type of TLD used in the RINSC experiments), there is an apparent positive bias averaging +8.4%. The bias seems more pronounced at lower total dose (i.e., those doses measured at greater distances from the cooling plate). This would bring the measured dose into closer agreement with the MCNP calculated neutron dose. There would still be a greater dose discrepancy for neutrons than for photons, since the code embedded flux-to-dose conversion for the epithermal neutrons (5.3 to 821 keV) is up to 30 times higher than that for the thermal neutrons (<5.3 keV), while the variation is much slighter (a factor of 6.73) for conversion of the prompt photons (1 to 15 MeV). To ensure the reliability of calculated dose, two sets of conversion factors are recommended for use [5]. In prior experiments at RINSC, the MCNP flux-to-dose conversion in a single set was up to 30 times higher than for thermal neutrons, with less variation for prompt photons (greater than 1 MeV). The agreement in the experiment reported herein between the MCNP calculated and measured dose is, therefore, closer than in measurements previously reported [5].

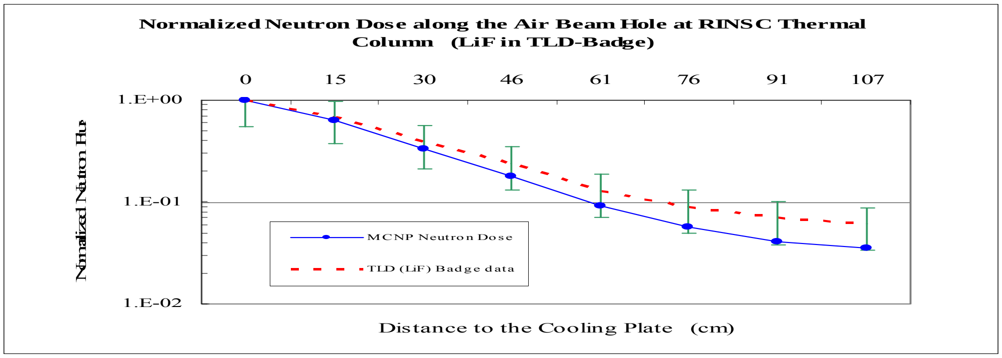

Figure 6. MCNP calculated neutron dose versus TLD measured dose along the central air beam hole at RINSC thermal column.

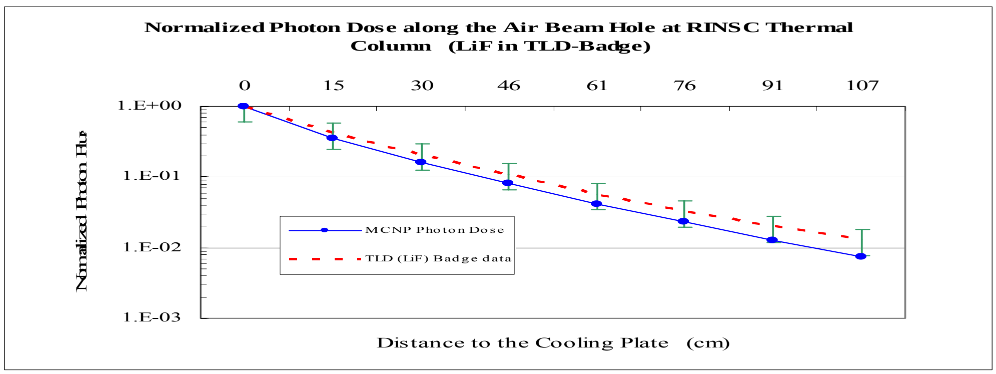

Figure 7. MCNP calculated photon dose versus TLD measured dose along the central air beam hole at RINSC thermal column.

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