In-core Fuel Management , Safety , and Thermal Hydraulics Studies for Upgrading TRIGA MARK II Research Reactor

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Bangladesh Atomic Energy Commission has approved a project to upgrade the research reactor to higher flux to meet the growing demand of medical radio-isotopes production and other irradiation facilities. Preliminary studies with the various core parameters showed that it might be possible to create new irradiation flux traps , increase the neutron flux at desired location, and at the same time the fuel burn-up can be made optimal. This will need major reshuffling and reconfiguration of the core with fuel rods initially loaded. The principal objective of this study is focused to make the above improvements in the core without disturbing the safety parameters.

This presentation deals with the neutronic and thermal hydraulic analysis of the 3 MW TRIGA MARK II research reactor to upgrade it to a higher flux. To realize this objective, the overall strategy followed is: (I) generation of problem dependent cross section library from basic Evaluated Nuclear Data Files such as ENDF/B-VI, JENDL 3.2 with NJOY94.10+[1] , (ii) use WIMSD-5 [2] package to generate cell constants for all of the materials in the core and its immediate neighborhood, (iii) use CITATION[3] to perform 3-D global analysis of the core to study multiplication factor, neutron flux and power distribution, power peaking factors, temperature reactivity coefficients, etc., (iv) check the validity of the deterministic codes with the Monte Carlo code MCNP4B2[5], (v) couple output of CITATION with PARET[4] to study thermal hydraulic behavior to predict safety margins, and (vi) reshuffle the current core configuration to achieve the desired objectives. The computational methods, tools and techniques, customization of cross section libraries, various models for cells and super cells, and a lot of associated utilities have been standardized and established/validated for the overall core analysis. Analyses using the 4-group ,and 7-group libraries of macroscopic cross sections generated from the 69-group WIMSD-5 library were performed to study the effect of group structure on neutronics parameters. Various studies showed that a 7-group structure is more suitable for TRIGA calculations considering its LEU fuel composition. The 7-group calculation predicts the experimental value of k_{eff} (1.077459) with an accuracy of 0.2% whereas the 4-group calculation yields k_{eff} with 0.4% accuracy, in the case of wet CT. So the neutronic analyses were performed using 7-group structure. Table I shows a comparison of the values of the flux distributions obtained from the calculations and experiment at irradiation location in the CT. It may be observed that our calculations overpredict in case of the dry CT and underpredict in the case of wet CT. The maximum error with the experiment was found for wet CT at core midplane with 32% underprediction.

Table- I
Comparison of the flux values at Dry & Wet CT with the experimental results

Group	Wet				Dry			
	Maximum Th. Flux (x10^{13})		Epi.Th. Flux (x10^{13})		Maximum Th. Flux \ (x10^{13})		Epi.Th. Flux (x10^{13})	
	Calc.(err.%)	Expt.	Calc. (err.%)	Expt.	Calc. (err.%)	Expt.	Calc. (err.%)	Expt.
4-G	6.596 (36%)	10.4	2.17 (14%)	2.53	5.939 (-7%)	5.56	2.05 (-19%)	1.71
7-G	7.031 (32%)		2.21 (12%)		6.318 (-14%)		2.08 (-21%)	

The power distributions in the X, Y, and Z-planes of the core were studied to locate the Hot-spot i.e., the point at which maximum power density occurs and it was found at the fuel position C4. The axial power distribution (distribution in the Z-plane) was used for the thermal hydraulic analysis of the core. The power peaking factors were calculated with WIMSD-5 and CITATION together. The total peaking factor defined as a product of hot rod factor, axial peaking factor, and radial peaking factor is found to be $f_T = f_{HR} \times f_z \times f_R = 1.875 \times 1.239 \times 2.535 = 5.887$ for wet CT compared to SAR value of $f_T = 1.70x1.25x2.65 = 5.633$. The same for dry CT is $f_T = 1.886x1.236x2.535 = 5.909$.

The fuel temperature coefficient α_f, another important safety parameter, has been calculated through global calculation by using CITATION. The average curve of α_f with temperature at zero burn-up shows that the curve deviates a little with that reported in the SAR for LEU fuel. The temperature curve that has been obtained by averaging the CITATION results was used through least square fitting method in the thermal hydraulic analysis of the core.

The testing of the PARET model calculations has been accomplished through benchmarking the available TRIGA operational data and the SAR values, both for steady state and pulse mode operations. For the steady state analysis the hot channel and an average channel were considered to obtain the most important safety margins. The axial distribution of temperatures of fuel center line, fuel surface, clad outer surface and coolant were calculated. The peak value for the fuel center line temperature in the hot channel is found to be 804.35^0C which is higher than the SAR value of 725^0C. It is observed from our calculation that the bulk coolant temperature from inlet to outlet increases by 6.94^0C and 4.08^0C respectively in the hot and average channel compared to SAR value 4.3^0C in the average channel. Pulsed parameters were analyzed at an insertion of \$1.996 reactivity to benchmark the experimental values available in the SAR. It was observed that the power peaks after 0.272 sec and its value is 844.75 MW which compares reasonably well with the experimental value of 852 MW. The prompt energy released is 19.9 MW-sec compared to an experimental value of 18.3 MW-sec. This agreement establishes the fact that the quality of modeling of TRIGA for in PARET for transient analysis is good.

The maximum temperatures in the fuel centre line, fuel surface, and clad surface are 481.9^0C, 399.8^0C 152.2^0C and 323.6^0C, 267.3^0C, 110.8^0C for the hot channel and average channel respectively.

Monte Carlo calculations were performed to verify the qualitative predictive capability of the deterministic codes used so far in the current analysis. The cross sections used in the MCNP model were from the ENDF/B-VI and JENDL3.2 cross section data. The multiplication factor K_{eff}..predicted were 1.07620 (± 0.00162) and 1.07892 (±0.00160) obtained with ENDF/B-VI and JENDL3.2 cross section libraries respectively and compared very well with the measured value of 1.077459.

The current core was reconfigured keeping in mind the upgrading of the flux in some desired locations. In the reconfigured core, graphite dummy element at C-6 was taken out of the core to create a new irradiation channel. The fuel elements D-8 & D-9 around C-6 were exchanged with graphite dummies at G-12 &G-18. Another new irradiation channel was created at location C-10. The fuel elements at C-10 was moved to G-8 position. The fuel elements D-14 & D-15 around C-10 were exchanged with the dummy elements at G-24 and G-2 locations. Fluxes identical to the current CT were found in the new irradiation channels.

REFERENCES

[1] R.E. MacFarlane, The NJOY Nuclear Data Processing System 94.10+, LANL-12740-M, August1995.

[2] WIMSD-5, A Neutronic Code for Standard Lattice Physics Analysis, distributed by NEA Databank, 1996.

[3] T.B. Fowler and D.R. Vondy, CITATION, A computer program for solving Diffusion Theory treating up to three dimension, ORNL-TM-2496 Rev.2, July1971.

[4] C.F. Obenchain, PARET, A program for the analysis of reactor transients, IDO-17282, January 1969.

[5] J.F.Briesmeister, MCNP-The Monte Carlo N- Particle Transport Code System, version 4B2, LA-12625-M, January1998.