CALCULATION AND COMPARISONS WITH MEASUREMENT OF FAST NEUTRON FLUXES IN THE MATERIAL TESTING FACILITIES OF THE NRU RESEARCH REACTOR

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Abstract

The NRU reactor at Chalk River provides three irradiation facilities to study the effects of fast neutrons (E> 1 MeV) on reactor materials for assessing material damage and deformation. The facilities comprise two types of fast neutron rods (Mark 4 and Mark 7), and a Material Test Bundle (MTB) irradiated in a loop site. This paper describes the neutronic simulation of these testing facilities using the WIMS-AECL and TRIAD codes, and comparisons with the fast neutron flux measurements using iron-wire activation techniques. It also provides comparisons of flux levels, neutron spectra, and size limitations of the experimental cavities between these test facilities.

1. Introduction

The National Research Universal (NRU) reactor at Chalk River began operation in 1957. It is used to carry out research in basic science and in support of the CANDU power reactor programs, such as the fuel bundle and material development programs. It is also a major supplier of medical radioisotopes in Canada and the world. The NRU reactor is heavy water cooled and moderated, with on-line refueling capability. It is licensed to operate at a maximum power of 135 MW, and has a peak thermal flux of approximately 4.0×10^{18} $n.m^{-2}.s^{-1}$. Figure 1 shows an NRU core lattice, with 31 rows and 18 columns (A to S, with no column “I”). The hexagonal lattice pitch is 19.685 cm.

The NRU reactor provides three facilities to study the effects of fast neutron irradiation of reactor materials. Materials of specific interest are zirconium and its alloys, which are used in the pressure tubes and calandria tubes of CANDU reactors. Various in-reactor material test programs have been conducted in NRU for many years, including experiments on irradiation induced creep and growth, on corrosion and on fracture properties. Between 2009 May and 2010 August, the NRU reactor was shut down 15 months for vessel repair, and since restart of the reactor, it has been operating safely and efficiently. At present, the AECL Nuclear Laboratories are seeking new opportunities for material testing and research in NRU from universities and industry.

This paper describes the materials testing facilities of the NRU reactor at the high energy range (E> 1 MeV) of the neutron spectrum. These comprise two types of fast neutron rods (Mark 4 and Mark 7), and a Material Test Bundle (MTB) irradiated in a loop site under CANDU coolant conditions. This paper also presents the methods of neutronic simulation of these testing facilities using the WIMS-AECL [1] and TRIAD codes [2,3]. Comparisons of the simulation results with flux measurements using iron-wire activation techniques are given.

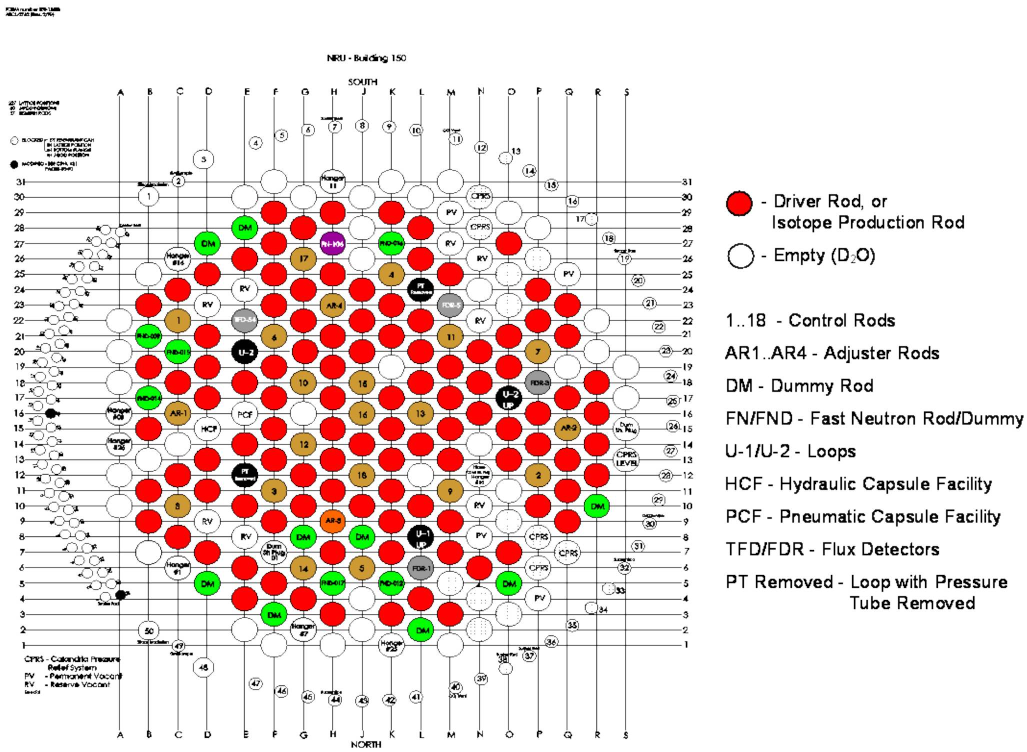

Figure 1 The NRU Core Lattice

2. Materials Testing in the Fast Neutron Flux Region

2.1 Mark 4 Fast Neutron Rods

Figure 2 shows a Mark 4 Fast Neutron (Mk4 FN) rod, which has an overall length of 7.825 metres. It consists of an outer flow tube and an inner flow tube, with its bottom end closed to form a cavity. Experimental inserts enclosed in pressure tubes are placed inside the central cavity. At the top of each experimental insert is an integral shielding plug, with tubes for the passage of service leads and water piping for cooling.

A Mk4 rod contains approximately 32 kg of natural uranium, in the form of sintered UO_2 fuel arranged in three 15-element bundles, each approximately 49.5 cm long. The fuel pins are located in the annulus formed by outer and inner flow tubes, and the fuel is cooled by heavy water. The maximum operating power for a Mk4 FN rod is 1.65 MW.

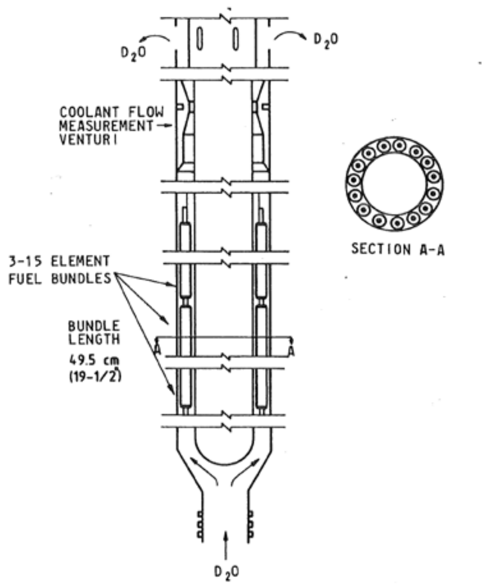

Figure 2 The Mk4 FN Rod.

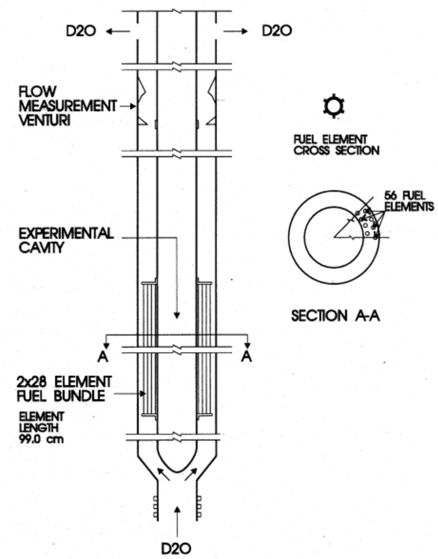

Figure 3 The Mk7 FN Rod

2.2 Mark 7 Fast Neutron Rods

Figure 3 shows a Mark 7 Fast Neutron (Mk7 FN) rod, which has an overall length of 7.822 m. It consists of an outer flow tube and an inner flow tube with bottom-end closed to form a cavity. The experimental insert is enclosed in a pressure tube and placed inside this cavity. To cool specimens in the experimental insert, the coolant flows down through a central flow tube and upward to cool the specimens.

The fuel for the Mk 7 FN rod is made up of 56 elements, arranged in two rings of 28, containing low enriched uranium (LEU) fuel, the same as the fuel in an NRU driver rod. The LEU fuel alloy is dispersion-type Al-61.4 wt% U_3Si, consisting of particles of U_3Si dispersed in a continuous aluminum matrix, with 19.75 % U-235 in uranium. Each fuel element is 99.06 cm long, with a fuel length of 81.28 cm, located at the mid-height of the reactor. The fuel provides a uniform axial fast-neutron flux region approximately 60 cm in length inside the experimental cavity. The Mk 7 FN rod is designed to operate with an experimental insert at a maximum power of 2.64 MW to the coolant.

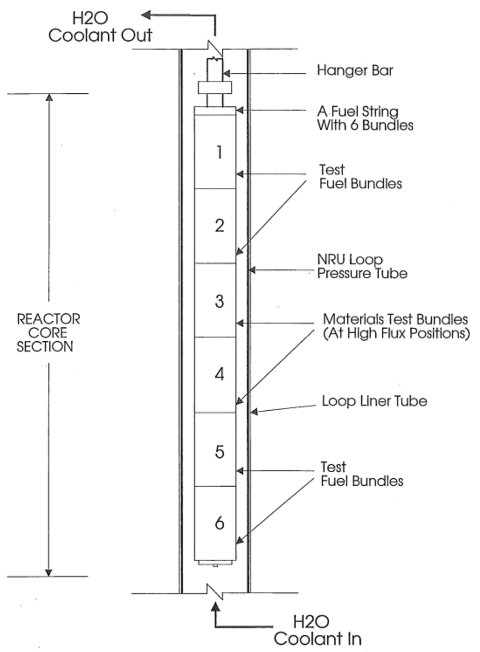

Figure 4 Material Test Bundles in a Loop Site.

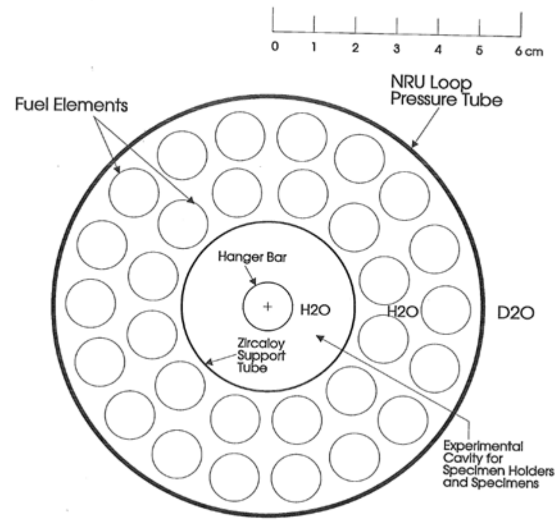

Figure 5 A Material Test Bundle.

2.3 Materials Test Bundles

The Materials Test Bundle (MTB) is designed to enable metallurgical specimens to be irradiated in a high-flux position in an NRU loop, such as the two middle positions on a loop fuel string, as shown in Figure 4. The NRU loops are high temperature and high pressure test facilities, in which test elements may be subjected to conditions simulating those existing in power reactors. The loop test section is confined to a 10.3 cm ID pressure tube, made of Zr 2.5% Nb or other zirconium alloys. The design pressure of the U-1 and U-2 loops is 13.8 MPa. The loops are cooled by light water. The MTB is a 30-element bundle, similar to a 37-element CANDU fuel bundle with the 7 centre elements removed. The overall length of the MTB is 482 mm. Its outer fuel ring has 18 elements with uranium enriched to 1.25 wt% U-235, and its inner ring has 12 elements with uranium enriched to 1.7 wt% U-235. The centre seven elements of the bundle are replaced by a 41.7 mm ID, 43.2 mm OD Zircaloy tube welded to the webs of the end plates. The tube provides internal support for the bundle and serves as a guide for the specimen holder assembly.

3. Method of Neutronic Simulation

3.1 Fast Neutron Spectrum inside the Experimental Cavities

Calculations of the fast-neutron flux spectra in the experimental cavities of the Mk7 and Mk4 FN rods and the MTB were performed using the WIMS-AECL code [1], with its associated ENDF/B-V derived data base. WIMS-AECL is a multi-group transport code with two-dimensional capabilities using the “Pij” collision probability method. The main transport calculations were performed using 34 energy groups, with a nine group subdivision above 1 MeV.

The experimental section of the material testing facility was modeled as a super-cell, which included the cell-of-interest and a representation of its neighbouring fuel rods. The neighbouring fuel rods provide the correct driving spectrum for the FN rod, and are modeled as two fuel rings located at radii of 20.2 cm and 39.9 cm from the centre of the cell. The fuel loading in these two rings was about 2.39 g/cm and 4.34 g/cm of U-235. A boron-10 ring was added outside the fuel rings, in order to keep the super-cell k-effective close to 1.000.

The experimental insert was modeled as four annuli: the first for the centre flow tube, the second for the inner support tube, the third for the specimen holder and specimens, and the fourth for the enclosing pressure tube. Details of the NRU Mk7 FN rod with a typical experimental insert, as modeled for the WIMS-AECL calculations, are shown in Table 1. Similarly, the experimental inserts for the Mk4 FN rod and the MTB were modeled as annuli. The radii and material composition of the third annulus may vary slightly, depending on which type of specimen holder assembly was used.

3.2 Comparison of Calculated Fast Neutron Spectra for the three Facilities

A reference case was set up to model a typical Mk7 rod, with a zirconium pressure tube thickness of 3.5 mm, and a 3.0 mm gap between the pressure tube and the inner flow tube of the rod. The rod was assumed to be fresh with no fuel burnup. Column 3 of Table 2 lists the WIMS calculated neutron fluxes at the specimen irradiation location inside the experimental cavity of the NRU Mk7 FN rod for the reference case. The neutron fluxes are normalized to the rod design power of 2.64 MW to the coolant. For the LEU fuel used in the Mk7 rod, the fuel power-to-coolant ratio was taken to be 0.94, and the fission power from the Mk7 rod is (2.64/0.94)=2.81 MW. For a fuel length of 81.28 cm, the linear fission power of the rod is 3.45 MW/m. In column 3 of Table 2, the fast neutron flux above 1 MeV at the specimen irradiation location is 7.3×10^{17} $n.m^{-2}.s^{-1}$, the thermal neutron flux below 0.625 eV at the same location is 1.69×10^{18} $n.m^{-2}.s^{-1}$, and the total neutron flux is 5.08×10^{18} $n.m^{-2}.s^{-1}$. In percentages, the fast neutron flux above 1 MeV is 14.4% of the total, and the thermal flux is 33.4%.

Table 1 NRU MK7 FN Rod and Typical Experimental Insert Data
(IR=Inner Radius, OR=Outer Radius)

Fuel	
Number of fuel elements	56
Outer radius of the U-Al fuel core	0.2743 cm
Effective outer radius of the Al clad (Al fins included in the clad)	0.3760 cm
U_3Si-Al density	5.45 g/cm^3
Wt. % of U_3Si in Al matrix, and U-235 enrichment	61.4% 19.75%
Linear mass of U-235	8.51 g/cm
First and second fuel ring radii	4.38 cm, 5.17 cm
FN Rod Flow Tubes	
Inner Al Flow Tube, IR	3.66 cm
“ “ “ “, OR	3.86 cm
Outer Zr Flow Tube, IR	5.72 cm
“ “ “ “, OR	5.85 cm
Typical Experimental Insert	
Centre Flow Tube, IR	0.55 cm
“ “ “, OR	0.63 cm
Inner Support Tube, IR	0.67 cm
“ “ “, OR	0.79 cm
*Specimen Holder & Specimens, IR	1.99 cm
* “ “ “ “ , OR	2.44 cm
Pressure Tube, IR	3.00 cm
“ “ , OR	3.35 cm

* Annulus for reporting fast neutron fluxes

Similar flux spectrum calculations were performed for the NRU Mk4 FN rod, and the results are listed in columns 5 and 6 of Table 2. The neutron flux is normalized to a linear rod power of 1.05 MW/m, or 1.65 MW total rod power. It is noted that the Mk4 FN rod produces less than one-third of the fast neutron fluxes of the Mk7 rod (2.07 vs 7.30 x 10^{17}n.m^{-2}.s^{-1}). Also, for the NRU Mk4 FN rod, the percentage of neutron flux above 1 MeV is only 6.5% of the total.

For the MTB bundle, the calculated neutron flux spectra at the specimen irradiation locations are listed in the last 2 columns of Table 2. The calculated fluxes are normalized to a nominal thermal flux of 2.85 x 10^{18} n.m^{-2}.s^{-1} in the first driver fuel ring, which is a typical bundle flux level in the U-1 or U-2 loop. This is equivalent to normalizing to a linear bundle fission power of 1.73 MW/m, or a total bundle fission power of approximately 834 kW. In Table 2, the fast-neutron flux above 1 MeV in the experimental cavity of MTB is calculated to be 5.31 x 10^{17}n.m^{-2}.s^{-1}. The MTB provides a material irradiation facility at a fast neutron flux level between those of the Mk4 and Mk7 FN rods. For each spectrum in Table 2, the sum of the neutron fluxes of all energy groups was normalized to be 100%. The MTB has the highest percentage of fast neutrons (E> 1 MeV), which is 15.6%, and for the Mk7 and Mk4 FN rods, it is 13.1 and 6.5%, respectively.

Table 2 Comparison of Fast Neutron Spectra inside the Experimental Cavities of a Mk 7 Rod, Mk 4 Rod and MTB.

Group	**Energy Width (MeV)**	**Mk 7 Rod Flux** **($x 10^{16} n.m^{-2}.s^{-1}$)**	**Mk 7 Rod Flux (% of total)**	**Mk 4 Rod Flux** **($x 10^{16} n.m^{-2}.s^{-1}$)**	**Mk 4 Rod Flux (% of total)**	**MTB Flux** **($x 10^{16} n.m^{-2}.s^{-1}$)**	**MTB Flux (% of total)**
1	7.79-10.0	0.551	0.108	0.177	0.055	0.420	0.124
2	6.07-7.79	1.534	0.302	0.500	0.157	1.184	0.348
3	4.72-6.07	3.459	0.680	1.115	0.350	2.675	0.787
4	3.68-4.72	6.361	1.251	1.823	0.573	4.483	1.318
5	2.87-3.68	9.564	1.881	2.729	0.858	6.777	1.993
6	2.23-2.87	12.274	2.415	3.590	1.129	8.956	2.634
7	1.74-2.23	12.786	2.515	3.482	1.095	9.024	2.653
8	1.35-1.74	13.550	2.666	3.695	1.162	9.846	2.896
9	1.05-1.35	12.939	2.545	3.565	1.121	9.744	2.865
Fast E>1.05MeV	Sub-total	73.019	14.365	20.676	6.500	53.109	15.618
Epi-thermal	$0.625x10^{-6}$ -1.05	265.699	52.270	87.474	27.50	164.438	48.35
Thermal	Below $0.625x10^{-6}$	169.606	33.366	209.940	66.0	122.519	36.028
Total		508.324	100.00	318.090	100.00	340.066	100.00

Comparisons of the fast neutron flux levels in the three fast irradiation facility in NRU are summarized in Table 3, together with the limitations of the size of the experimental cavity. Among the three facilities, the Mk7 FN rod provides the highest fast neutron flux level, followed by the MTB and the Mk4 FN rod. The Mk7 FN rod has a moderate size of experimental cavity, slightly less than that of Mk4 FN rod, but larger than the MTB.

Table 3 Comparison of the three Fast Neutron Irradiation Facilities in NRU

Facilities	**Fuel**	**No. of Fuel Elements**	**Experimental Cavity**	**Fast Neutron Fluxes, $n. m^{-2}.s^{-1}$**
Mk4 FN Rod	Natural Uranium	3 x 15	7.2 cm ID by 150 cm height	2.1×10^{17}
Mk7 FN Rod	LEU (20% U235)	56	7.3 cm ID by 81 cm height	7.3×10^{17}
MTB	Enriched Uranium (1.25&1.71%)	30	4.2 cm ID by 48 cm height per bundle	5.3×10^{17}

3.3 Determination of Changes in Flux Level with Fuel Burnup

There is a reduction in the flux level in the test facilities as fuel burns up. For example, the initial high flux level for a fresh Mk7 rod can be maintained with surrounding fuel management for about 2 months, and the flux level will then drop gradually from 7.3 x 10^{17} $n.m^{-2}.s^{-1}$ as the rod burns up. When the rod power drops to 2 MW, the fast flux level above 1 MeV is 5.5 x 10^{17} $n.m^{-2}.s^{-1}$. Over the rod life time, the average operating power is 1.58 MW, and the average fast flux level above 1 MeV provided by the rod is 4.4 x 10^{17} $n.m^{-2}.s^{-1}$. The tracking of the change in the flux level in the test facilities as fuel burnup can be performed by the TRIAD code [2,3]. A brief summary of the TRIAD code is given at the end of this paper in Appendix A.

During the irradiation period in the test facilities, calculations of the axial flux and power levels can also be performed using the TRIAD code. Figure 6 shows a typical axial flux profile in a Mark 7 rod. The high fast neutron flux region extends about 40 cm above and below the centre of the reactor.

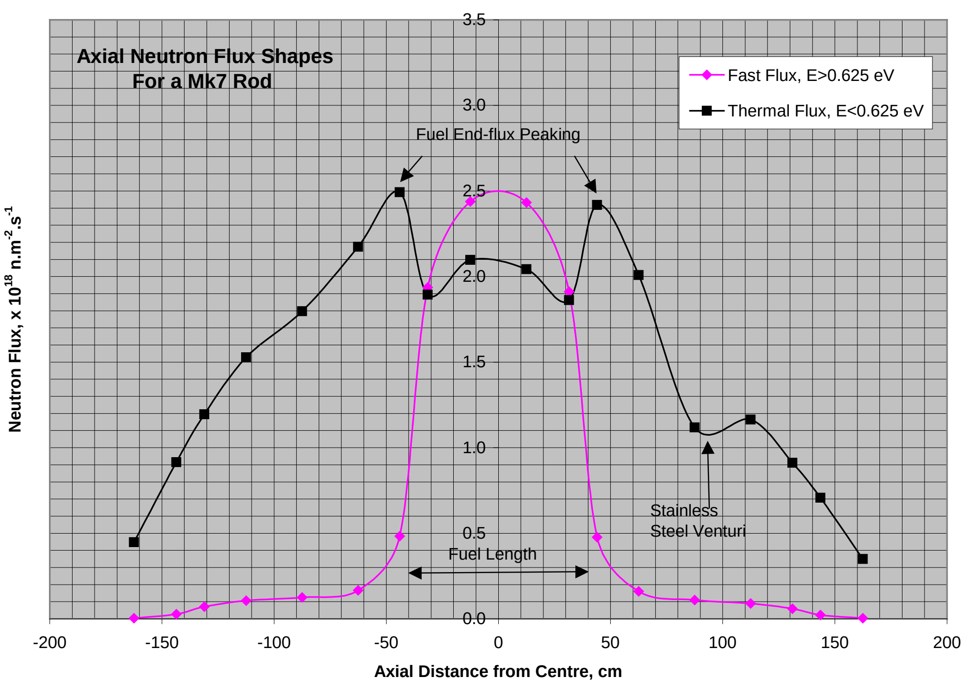

Figure 6 Axial Neutron Flux Profiles in a Mk 7 Rod

4. Fast Neutron Flux Measurement

4.1 Iron Wire Flux Monitors

Iron-wire flux monitors are used for fast neutron flux measurements (E> 1 MeV) in the material testing facilities. The reaction Fe54(n,p)Mn54 has an effective threshold at approximately 1.05 MeV. The measured activity of Mn54, which has 312.5 day half-life, can be used to determine the fast neutron flux above 1 MeV.

The measured activity of Mn54 is first corrected for counter efficiency to obtain the absolute activity. The absolute activity, A, is related to the fast neutron flux $\emptyset$ (E> 1 MeV) by:

$$A = N\sigma_{FE}\emptyset(1-\exp(-\lambda t_1))\exp(-\lambda t_2), \qquad \text{Eq.(1)}$$

Where N is the number of iron atoms per unit mass of wire,

t_1 is the irradiation time in the reactor at an assumed constant flux,

t_2 is the counting delay time after irradiation, and

λ is the decay constant of Mn54.

σ_{FE} is the effective or spectrum averaged iron cross section, which can be written as:

$$\sigma_{FE} = \sum_i \sigma_i \emptyset_i / \sum_i \emptyset_i, \qquad \text{Eq.(2)}$$

where σ_i and $\emptyset_i$ are the group cross sections for iron and group fluxes of the neutron spectrum at the irradiation location. The effective iron cross section inside each of the test facilities may be slightly different because of the difference in neutron spectra. In MTB it is calculated to be 101.5 mb.

4.2 Comparison of Measured and Calculated Fast Fluxes in the Mark 7 Rod

In the past, the measured fast neutron flux per unit rod power in the NRU Mk7 FN rod, EFN701 was determined, based on the measured activities of Mn54 from the iron wire flux monitors. The changes in the rod fuel burnup of the Mk7 were obtained from the TRIAD core following calculation for the period of irradiation of the iron wire monitors. Table 4 shows the fast flux measurements in a Mark 7 rod in the past. In this Table, the average measured fast neutron flux per MW fission is 2.61 x 10^{17} $n.m^{-2}.s^{-1}$/MW for the EFN 701 rod. The measurement error is about $\pm$ 15%. From Table 2 of Section 3.2, the calculated fast neutron flux (E> 1MeV) in a specimen holder at a typical distance from the centre of a Mk7 FN rod is 7.3 x 10^{17} $n.m^{-2}.s^{-1}$ at 2.81 MW fission power, or 2.60 x 10^{17} $n.m^{-2}.s^{-1}$ /MW. This agrees with the EFN701 measurement data of the three phases to within 4.5%. The discrepancies between the measured and calculated results are attributed to uncertainty in material composition of the experimental inserts, and uncertainty in the iron wire flux monitor locations.

4.3 Comparison of Measured and Calculated Fast Fluxes in the MTB

Measurements of the fast neutron fluxes above 1 MeV were obtained from a Material Test Bundle, AHA, in the U-1 loop of NRU at site L08 from 1994 February 12 to 1994

July 9. The experimental insert of the bundle contained 3 iron wire flux monitors in each of the specimen holders attached to their upper, middle and lower sections. The iron wire flux monitor was 0.25 mm in diameter by 2.54 cm long.

Using Eq. (1), the measured fast neutron fluxes above 1 MeV averaged over the irradiation period in the upper, middle and lower sections of the specimen holders were determined to be 4.34, 4.53 and 4.68 x 10^{17} $n.m^{-2}.s^{-1}$, respectively. The accuracy of the measured fast neutron fluxes was estimated to be about 8%, with 5% for the Mn54 activity measurements and 3% for the effective iron cross section.

Table 4 Comparison of Measured and Calculated Fast Fluxes in a Mark 7 Rod

Phase	**1**	**2**	**3**	**Average**
Irradiation Period	87 Dec 08 to 88 Jan 25	88 Apr 16 to 88 May 12	88 May 20 to 88 Jun 06	
Irradiation Time (days)	27.7	23.5	16.8	
Average Rod Fission Power (MW)	2.11±10%	1.60±10%	1.78±10%	
Measured Flux >1 MeV, (x10^{17} $n.m^{-2}.s^{-1}$)	5.56±5%	3.99±5%	4.81±5%	
Measured Fast Flux per MW Fission, (10^{17} $n.m^{-2}.s^{-1}$/ MW)	2.64±15%	2.49±15%	2.70±15%	2.61±15%
Calculated Fast Flux per MW Fission*, (10^{17} $n.m^{-2}.s^{-1}$/ MW)	2.60	2.60	2.60	
Deviation of Measured to Calculated Fluxes,%	+1.5%	-4.2%	+3.8%	

*Note: From Table 2 in Sec. 3.2, Calculated Fast Flux per MW Fission = 7.3 x10^{17} $n.m^{-2}.s^{-1}$/2.81 MW=2.60 x10^{17} $n.m^{-2}.s^{-1}$/ MW

Table 5 Comparison of Measured and Calculated Fast Fluxes in MTB

ELEVATION	MEASURED FAST NEUTRON FLUXES IN 3 IRON-WIRES (x 10^{17} $n.m^{-2}.s^{-1}$)	AVERAGE OF MEASURED FAST NEUTRON FLUXES (x 10^{17} $n.m^{-2}.s^{-1}$)	CALCULATED FAST NEUTRON FLUXES (x 10^{17} $n.m^{-2}.s^{-1}$)	DEVIATION OF MEASURED FROM CALCULATED FLUXES
UPPER	4.27 4.38 4.38	4.34	4.89	-11.2%
MIDDLE	4.49 4.61 4.49	4.53	5.08	-10.8%
LOWER	4.61 4.83 4.61	4.68	5.17	-9.5%

Table 5 shows the comparison of the measured and calculated fast neutron fluxes for the AHA bundle at the 3 elevations. The calculated fast neutron fluxes as determined from the TRIAD code at the three elevations were 4.89, 5.08 and 5.17 x 10^{17} $n.m^{-2}.s^{-1}$, respectively. In general, the calculated and measured values agree to within 12%. The 12% difference appears significant, but its actual significance would need to be evaluated in the context of impact on parameters of interest related to material deformation and damage, such as for pressure tube creep and sag. It might be possible that current needs for aging management would require higher accuracy. Inclusion of simulations using more refined methods, such as MCNP, would be beneficial in the future.

5. Conclusions

Several conclusions can be drawn from the specific features of the three NRU fast neutron material testing facilities:

1) AECL Nuclear Laboratories provide new opportunities for material testing and research in three fast neutron irradiation facilities different from those available at universities or from industries. The highest fast neutron flux level (E> 1 MeV) provided is $7.3 \times 10^{17} n.m^{-2}.s^{-1}$ in the Mk 7 Rod, followed by $5.3 \times 10^{17} n.m^{-2}.s^{-1}$ in the MTB, and $2.1 \times 10^{17} n.m^{-2}.s^{-1}$ in the Mk 4 Rod.
2) There are size limitations for the experimental cavities of the three facilities. The Mk 4 rod offers the largest cavity size for irradiation, followed by the Mk 7 rod and the MTB has the smallest cavity size.
3) The flux level in the experimental cavities of the facilities during the material testing period can be tracked using the WIMS-AECL and TRIAD codes, and measurements of the fast neutron flux can be achieved using iron wire flux monitors.
4) There was reasonable agreement between the calculated and measured fast neutron fluxes in the Mk 7 Rod and MTB in the past, within ~12%. However, current needs for aging management would require higher accuracy, and improving the simulations using more refined methods, such as MCNP, would be beneficial in the future.

6. References

[1] J.D. Irish and S.R. Douglas, "Validation of WIMS-IST", Proceedings of the 23rd Annual Conference of the Canadian Nuclear Society, Toronto, Canada, 2002 June.
[2] S.R. Douglas, "A calculational model for the NRU reactor", Paper presented at the Canadian Nuclear Society 1985 Annual Conference; also AECL Report, AECL-8841, 1985 June.
[3] T.C. Leung and M.D. Atfield, "Validation of the TRIAD3 Code Used for the Neutronic Simulation of the NRU Reactor", Proceedings of the 30th Annual Conference of the Canadian Nuclear Society, Calgary, Alberta, Canada, 2009 May 31 - June 3.

APPENDIX A: Summary Description of the TRIAD Code

Neutronic simulation of the NRU reactor is performed by the TRIAD code, which is a three-dimensional neutron diffusion code in two energy groups. The code calculates steady-state neutron flux and power distributions for the reactor. The present TRIAD code version incorporates 301 rod sites, each with six triangular cells, with 18 planes, each of variable cell height, in the axial direction. The 18 axial cells representing a rod in NRU can be different cell types, but each type has the same uniform neutronic properties. The detailed flux shapes and neutron spectrum through each type of cell are determined using the WIMS-AECL neutron transport code. The homogenized cell parameters, in two energy groups, are then calculated by flux- and volume-weighting the region material properties. Examples of cell parameters are the diffusion coefficients and the various cross sections, such as absorption, removal and fission.

After the cell parameters are calculated, the flux and power distributions for the cells in the NRU core can be determined using a modified neutron diffusion theory. The modification is the use of cell discontinuity factors (*cdf*) to improve the radial neutron leakage calculation between adjacent cells. The usual inter-cell leakage calculation in the finite-difference diffusion theory uses a simple linear model, which distorts the flux distribution except in relatively uniform reactors. Since the NRU reactor is made up of many different types of rods, some with very different neutronic properties, it is necessary to use *cdf*'s in TRIAD to adjust the neutron current calculation at the homogeneous cell boundaries to minimize these distortions. The *cdf* is calculated from the ratio of the heterogeneous to homogeneous cell boundary flux. The heterogeneous cell boundary flux is determined from the WIMS-AECL flux shape by extrapolating the last 3 mesh point fluxes inside the boundary of the actual cell. The homogeneous cell boundary flux is determined from a cell having the homogenized cell parameters and the same leakage currents as the heterogeneous rod representation.

In the TRIAD code, the two-group diffusion equations in three dimensions are solved numerically using a finite difference method. The difference equations for the group fluxes in each triangular prism of an NRU hexagonal cell are solved using flux iteration techniques, with successive point over-relaxation to accelerate convergence. After the fluxes in all prisms are determined, the flux in each hexagonal cell is calculated as the average of the fluxes of the six triangular prisms for that cell. The power generated from a hexagonal cell is then calculated as the product of the cell flux, cell volume and Q value, which is the linear heat rating per unit flux per unit cell volume.

The TRIAD code has two main modules. The first is the simulation module which calculates steady-state neutron flux and power distributions in the reactor. The second module of TRIAD is the core-following module which tracks reactor assemblies by name, reactor position and fuel rod burnup, or isotopic content for isotope production rods, as they move into and out of the reactor.