A Validation Of Neutron Fluxes Through The CANDU® 6 End Shield

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1. ABSTRACT

Worldwide in the radiation shielding community, the two-dimensional (2-D) discrete-ordinates code DORT [1] remains one of the codes used to calculate the transport of neutrons and photons through thick shields. In the Canadian nuclear industry, DORT continues to be used to calculate the neutron-flux, photon-flux, heating, activation, dose-rate, and damage profiles across the different types of CANDU® reactor end shields and radial shields. Including its predecessor DOT-IV that evolved to what is now DORT, this 2-D discrete-ordinates code has been used in the design of many CANDU reactors, i.e., Pickering Nuclear Generating Stations (NGS), Bruce NGS, Darlington NGS, CANDU 6 stations, and the CANDU 3, CANDU 9, and the new ACR-1000™ designs.

Hitherto the validation of DORT applications for CANDU plants relied on limited CANDU station measurements and foreign benchmark experiments. The DORT code is often used to calculate particle fluxes across the reactor shields. Validation exercises that have been performed to date have yet to address the application of the code to calculating deep penetration through the primary shields of a CANDU reactor. This paper provides some validation results of neutron fluxes through a CANDU 6 end shield to bridge the gap in the validation studies.

2. BACKGROUND

The Point Lepreau Nuclear Generating Station was shut down in 1997 January to locate a leak in the primary heat transport system. The leak location was found at the first bend of the outlet feeder pipe of fuel channel S-08. As a result, the section of the feeder pipe between the feeder coupling and the first bend was replaced.

To study the cause of the crack, a section of the feeder that had been removed from the reactor was sent to AECL Chalk River Laboratories (CRL) for analysis. In performing the failure analysis at CRL, the section of the feeder pipe was further dissected into smaller pieces. Figures 1 to 3 identify the different dissections.

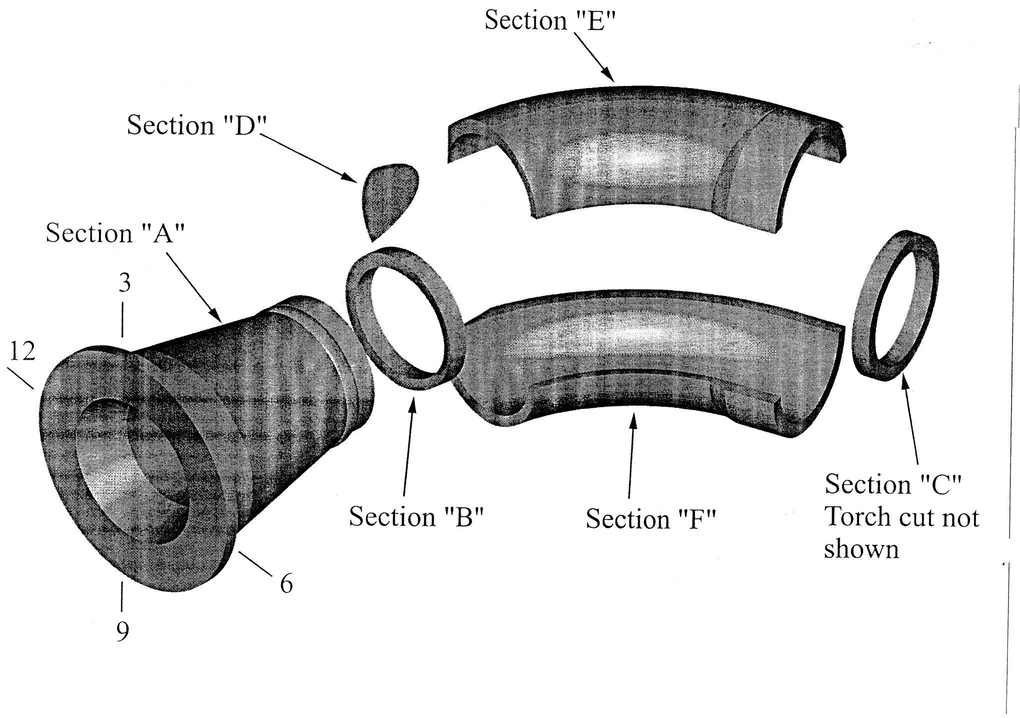

Figure 1 Cut Sections of Feeder Pipe S-08

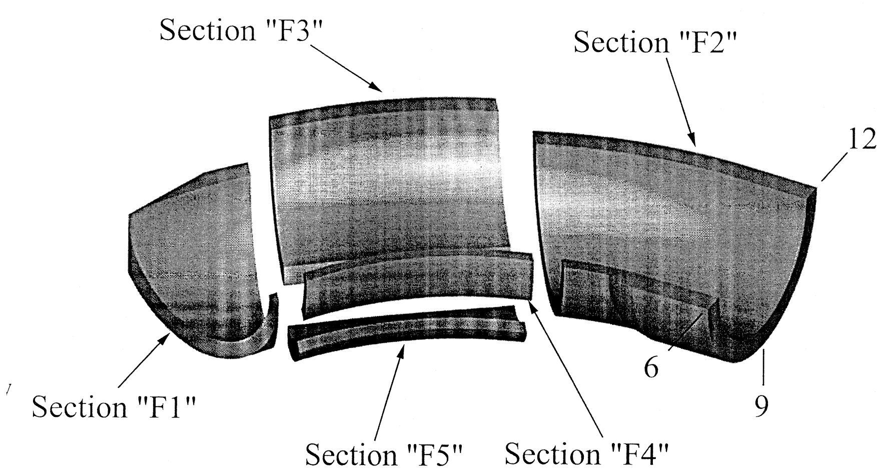

Figure 2 Detailed Cutting of Section F

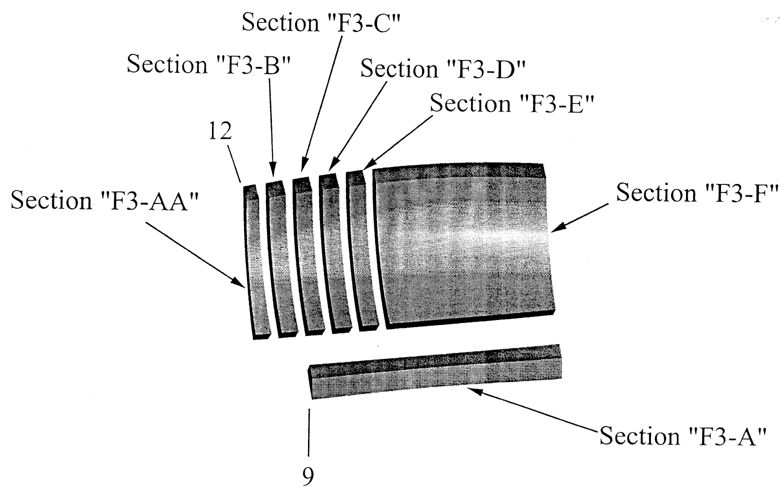

Figure 3 Details of Section F3

In 2000 October, a six-gram sample of the F3-F section of the feeder was ground and chemically cleaned for gamma spectrometry at CRL. That is, all contamination from deposited oxide activity on the surfaces of the sample was removed. As a result, the activities measured from the sample are the activities induced in the feeder pipe material resulting from neutron bombardment when the sample was at the reactor face.

The activity measurements can be viewed as surrogate flux measurements. That is, section F3-F represents a sample used in a foil activation experiment. Since the absorption cross sections are well established, derivation of the activities at the feeder location can provide a good indication of the accuracy of the calculated neutron fluxes after seven orders of attenuation through the end shield.

The gamma spectrometry identified two principal isotopes in the sample, i.e., ^{54}Mn and ^{60}Co. Activities of ^{54}Mn (1.516×10^{-6} μCi/g) and ^{60}Co (1.780×10^{-4} μCi/g) were measured to ±24% and ±1% (one-sigma) uncertainty, respectively, over a seven-hour counting period.

^{54}Mn is created from the ^{54}Fe(n,p) reaction. This is a high-energy (fast) neutron reaction with a higher probability of interaction starting at about 1 MeV. The activity of ^{54}Mn is therefore an indirect measurement of the fast neutron flux. ^{60}Co is created from the ^{59}Co(n,γ) reaction. This is both a low-energy (thermal) and an intermediate-energy neutron interaction. For the same level of thermal and intermediate neutron fluxes, the ^{60}Co productions are of the ratio of 7 to 1. Thus the activity of ^{60}Co is an integrated measurement of the thermal and intermediate neutron fluxes.

3. VALIDATION EXERCISE

The validation exercise was performed by extracting flux data from an existing CANDU 6 end-shield calculation done for the vicinity of the feeder-pipe bend, and applying the absorption cross sections to derive the ^{54}Mn and ^{60}Co activities. Based on the comparison, one can infer the level of accuracy of the activities across the end shield, hence the fuel channel. With several CANDU reactors being or planned to be refurbished, an understanding of the accuracy of the fuel-channel activities is essential to tool design, radiation protection, and resource management, to attain an overall low person-Sievert for the refurbishment work.

At AECL, activities of the fuel-channel components are calculated using the two-dimensional discrete-ordinates code DORT from the DOORS suite of codes [1]. For the validation exercise, the neutron fluxes in the vicinity of the feeder are extracted from a DOT 4.2 analysis performed for Wolsong-2 NGS. DOT 4.2 is a predecessor of the current DORT computer code that solves the Boltzmann transport equation using the same discrete-ordinates approach. Thus validating the analysis results generated by DOT 4.2 is analogous to validating the results generated by DORT.

In the DOT model, a fuel-channel lattice was modelled. The model represented the fuel channel lattice as a cylinder of equivalent area. The radial i-mesh in the problem starts from the fuel-channel centreline and goes radially out to the edge of the lattice cell. The axial j-mesh in the problem spans the space between the plane at the end of the end-fitting plane (E-face) and two-bundle lengths into the reactor. Meshes at the lattice edge of the analysis model are about 152 mm from the centreline of the fuel channel.

In the original DOT 4.2 analysis, there was no attempt to model the feeders or the positioning assembly. Thus neutron fluxes taken from a radial mesh at the lattice edge are located in an air region where the feeder would have been located. The fast, intermediate, and thermal neutron fluxes extracted from the lattice edge are listed in Table 1. In this table, neutron fluxes at three mesh locations are reported; they are all 152 mm from the fuel-channel centreline but represent locations that are 305 mm inboard from the E-face, 152 mm inboard from the E-face, and at the E-face. One can see from these fluxes that the activation reactions in the feeder sample will be sensitive to the exact location of the sample. Note that the feeder coupling is about 305 to 457 mm inboard from the E-face.

Table 1 Neutron fluxes at feeder region (values are quoted in air), neutrons·cm^{-2}·s^{-1}

		Neutron Fluxes from the DOT 4.2 Model		
Neutron Flux Group	Energy Boundary	E-face	152 mm away	305 mm away
Fast	0.821 – 14.9 MeV	7.361E5	8.240E5	1.919E6
Intermediate	0.414 eV – 0.821 MeV	1.788E7	1.934E7	3.197E7
Thermal	< 0.414 eV	1.516E6	1.532E6	1.584E6

To derive the ^{60}Co activity, the fluxes in Table 1 were input to the isotope generation and depletion code ORIGEN-S [3]. In the calculation, the sample was irradiated at a constant flux for 12.4 years (corresponding to 89% reactor capacity factor since in-service date). The cobalt content was based on the result of an activation analysis of the same feeder pipe; any differences observed in the comparison are therefore a result of the accuracy of the flux calculation.

The ^{54}Mn activity was calculated using the fine-group ^{54}Fe cross sections from VITAMIN-B6 [4] together with six groups of fast fluxes from the DOT 4.2 analysis, see Table 2.

Table 2 Fast neutron fluxes 305 mm inboard from the E-face

Energy Group	Energy Range, MeV	Neutron Flux, n/cm^2.s
1	14.9 – 12.2	9.625E2
2	12.2 – 11.1	1.256E3
3	11.1 – 6.06	5.574E4
4	6.06 – 3.68	1.242E5
5	3.68 – 2.23	2.900E5
6	2.23 – 1.35	5.131E5

The ^{54}Fe(n,p) reaction cross sections from VITAMIN-B6 are represented in 70 energy groups, from 0.672 to 19.64 MeV. The corresponding 70-group neutron fluxes are derived by generating an equation based on a curve fit through the six groups of fast-flux data in Table 2, shown in Figure 4 as a solid line. The sum of the cross product of the cross sections and fluxes gives ^{54}Mn reaction rates.

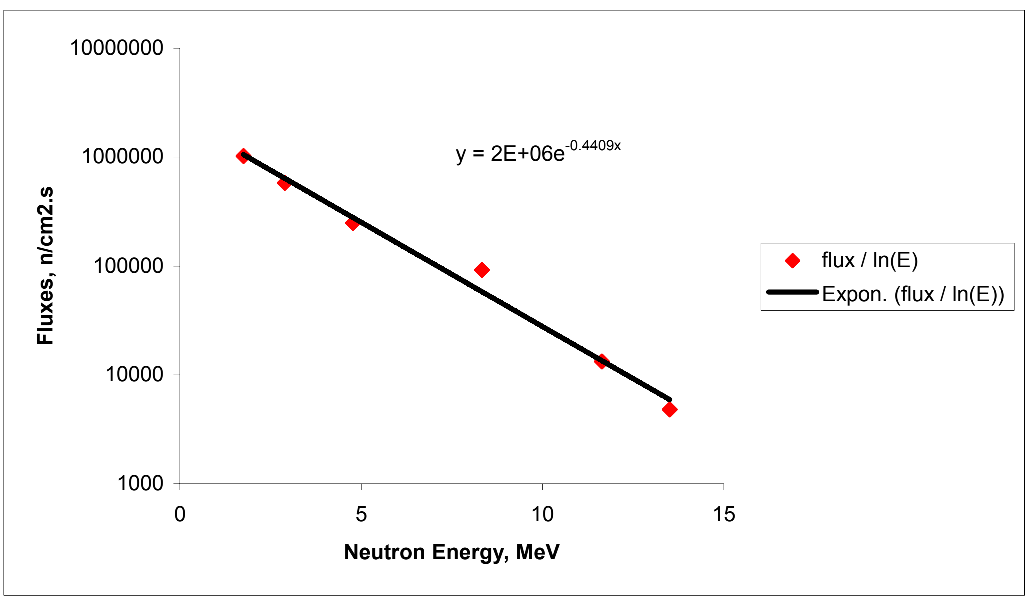

Figure 4 Fast Neutron Fluxes 305 mm Inboard from the E-face

4. ASSUMPTIONS

In generating the final activities, the following assumptions were applied. The assumptions are implemented directly to correct the inadequacies in the model, which includes the full geometry of the end shield but not the feeder piping.

- Fuel Bundle Power

  The end-bundle power for channel S-08 is 90% of the time-averaged value at the centre of the reactor used in the DOT 4.2 calculations for the central fuel channel. This assumption accounts for the radial power profile.

- Shielding

  The analysis model did not include the feeder piping. The structural material of the feeder piping will lower the neutron fluxes by 50%.

- Geometry

  The flux distribution within the feeder varies from location to location. Sample F3 is located to one side of the pipe when joined to the feeder coupling; the location can receive only about 70% of the flux that reaches the edge of the lattice cell.

- Self Attenuation

  The fluxes extracted from the model are in an air region. A flux depression of about 5% is expected had the calculated flux in a steel region been available.

5. RESULTS

Incorporating the above assumptions, the calculated activities for ^{54}Mn and ^{60}Co, compared to the gamma spectroscopy measurements, are listed in Table 3.

Table 3 Comparison of the calculated ^{54}Mn and ^{60}Co activities to gamma spectroscopy Measurement

Isotope	Measurement, μCi/g	Calculated, μCi/g
^{54}Mn	$1.516 \times 10^{-6} \pm 24\%$	24.5×10^{-6}
^{60}Co	$1.780 \times 10^{-4} \pm 1\%$	2.19×10^{-4}

The comparison of the ^{60}Co activities shows good agreement; the difference between the two activities is less than 50%. This indicates that the thermal and intermediate fluxes used in calculating the ^{60}Co activities are reliable.

The comparison of the ^{54}Mn activities shows a large discrepancy; the calculated activity is a factor of 16 too high. Even if the 24% uncertainty in the measurement is considered in the comparison, the discrepancy is still a factor of 13, implying that the fast flux in the DOT 4.2 calculation was grossly overestimated.

However, as mentioned above, the fast flux was extracted in an air region where no feeder piping was modelled. Had the feeder been modelled with heavy water included in the three-inch pipe, the fast flux would have been reduced substantially due to neutron moderation; a 20-cm slab of heavy water would give roughly a factor-of-10 attenuation in the fast flux. Because the F3-F sample of the feeder pipe is close to the bend where it is partially shielded from fast neutrons leaving the reactor face, neutron fluxes that can reach section F3-F of the feeder would have gone through a couple of feeder pipes that contain heavy water (see illustration in Figure 5).

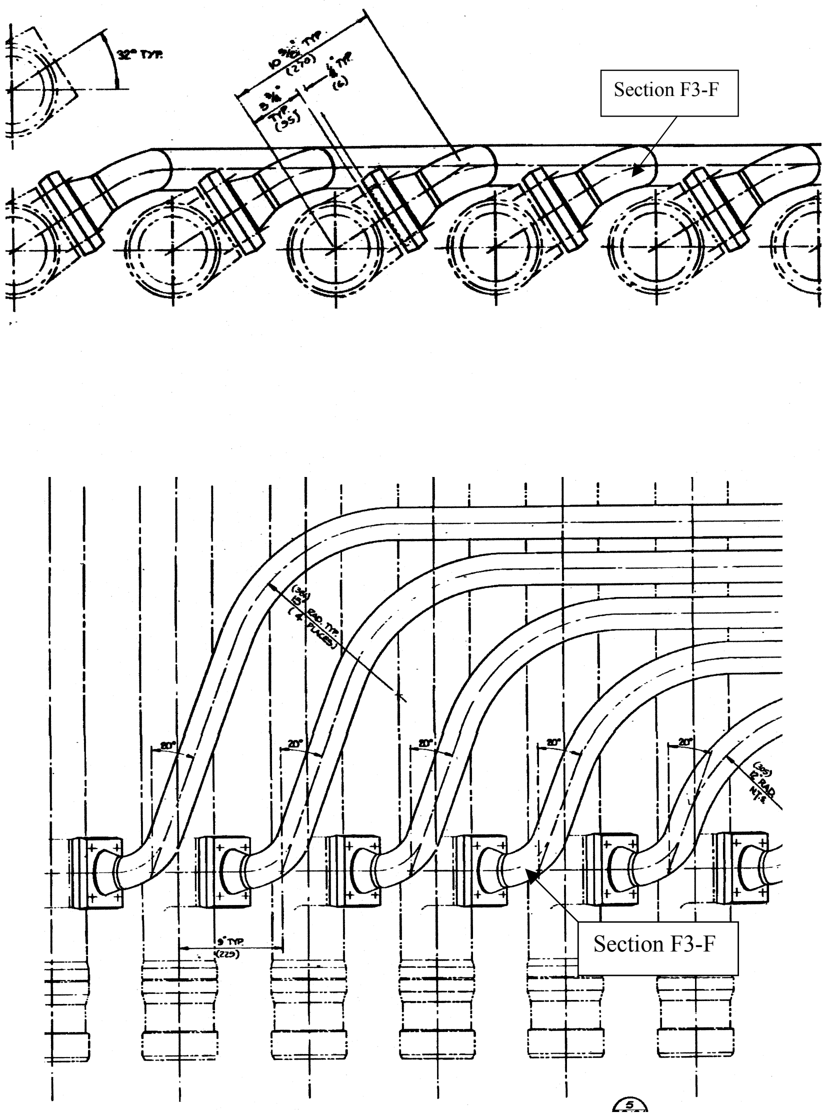

Figure 5 Diagram of the Connection of Feeders to End-Fitting

As a result, the fast flux that was used to calculate ^{54}Mn activities at the sample should be at least a factor of six or more lower. This will drop the overestimation of the calculated ^{54}Mn activities closer to a factor of two.

6. CONCLUSIONS

This paper presented an integrated comparison of the activities using gamma spectrometry measurements and available discrete-ordinates neutron fluxes calculated by DOT 4.2. While the calculation results only represent an approximation of the true geometry, the comparison nevertheless can be viewed as a partial validation of the neutron-flux calculation using the discrete-ordinates code DOT 4.2 in a bulk-shield environment.

Based on the comparison, the fast, intermediate, and thermal fluxes that were used to calculate the fuel-channel-component activities are shown to be conservative. The conservatism is shown to be a factor of two too high for the activities calculated. Thus one can conclude that activity estimates in the fuel-channel components at the reactor face will be too high. Nevertheless, the discrepancy for ^{60}Co activity is relatively small (less than 50% overestimated), indicating that thermal capture activities in the fuel channel are generally within 50% accurate. Considering the neutron fluxes are extracted in a region that do not correspond to the true physical arrangement of the feeders, the resulting difference of a factor of two is within acceptable limits.

Future validation work can be extended to using the latest version of the two-dimensional discrete-ordinates codes DORT with a revised 66-group cross-section library. While the DORT calculation methodology is the same as DOT 4.2, the updated 66-group library may have an impact on the calculation results. Photo-neutron activation needs to be included, although the authors believe it is insignificant. In addition, when artefacts from refurbishment are obtained and analysed, the neutron-flux profile across the end shield will be able to be validated with greater accuracy and confidence.

7. ACKNOWLEDGEMENT

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8. REFERENCES

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