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Calculations Supporting the Shipment of Irradiated CANFLEX® Demonstration Fuel Bundles

by

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1.0 INTRODUCTION

At the Point Lepreau Generating Station (PLGS), CANFLEX® bundles were irradiated in the reactor core as part of the demonstration irradiation for the CANFLEX® prototype fuel bundle. Some of these bundles were selected for shipment to the Chalk River Laboratories (CRL) in Ontario for post-irradiation examination. The Atomic Energy Control Board (AECB) and Atomic Energy of Canada Limited (AECL) require estimated values of several fuel parameters to satisfy the shipping regulations [1]. The frequency of such calculations has increased over the last few years as NB Power continues its involvement with and support of CANDU® fuel research and development. An analysis tool has been developed to determine the "best-estimates" for the required fuel parameters.

This paper briefly describes the irradiation of CANFLEX® fuel, the shipping requirements, the analysis tool, the methods used to determine the estimates of the various spent fuel parameters and a summary of the supporting calculations for two CANFLEX® bundles.

2.0 IRRADIATION HISTORY

At PLGS, channels Q20W and S08W were each fuelled with eight CANFLEX® fuel bundles and were irradiated nominally beginning 1998-09-03 to 1999-03-26 and 1999-08-08, respectively. The bundles discharged from positions five and eight of channel Q20W and position eight of channel S08W were eventually selected for shipment to the CRL. The data and analysis tool results for two of the CANFLEX® demonstration bundles (FLX019Z and FLX024Z), which were candidates for shipment to CRL, are provided in Section 5.0 of this paper. The irradiation history (i.e., reactor power and bundle power throughout the irradiation period) for bundles FLX019Z, and FLX024Z can be found in Figure 1.

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3.0 SHIPMENT REQUIREMENTS

The transport requirements are stated in the AECB Licensing Certificate [1]. The specific areas of interest to these calculations are: total activity, decay heat load, minimum storage time and maximum burnup. Other important areas are the activity of ^{131}I and the fuel sheath temperature. The following regulations contain only a subset of the shipping regulations, but concern the specific calculations listed above.

Total Activity

The AECB Licensing Certificate [1] for the current irradiated material transportation package states: "This package is authorized to contain:

a) irradiated CANDU® natural uranium fuel bundles or elements with an activity of **2,000 TBq (54,000 Ci)** maximum, consisting of :
   i) one 28-element fuel bundle assembly or up to 28 such elements,
   ii) **one 37-element fuel bundle assembly or up to 37 such elements**, or
   iii) **one 43-element fuel bundle assembly or up to 43 such elements, or**
   iv) up to 28 elements for any combination of i), ii), or iii)."

Decay Heat Load

"The radioactive decay heat for contents a)… shall not exceed **160 watts**." [1]

Minimum Storage Time and Maximum Burnup

"The minimum storage time for contents a) shall not be less than **180 days**, and the burn-up shall not exceed **1159 GJ/kg of initial uranium**." [1] (**321.9 MW·h/kgU**)

Additional Limitations By AECL – CRL

Chalk River Laboratories (CRL) have stated limitations [2] on ^{131}I and fuel sheath temperature as follows:

i) For non-defective fuel, the maximum fuel sheath temperature must be less than 350°C with an ^{131}I inventory of less than 1000 Ci/element.

ii) For defective fuel, the maximum fuel sheath temperature must be less than 250°C with an ^{131}I inventory of less than 100 Ci/element.

4.0 ANALYSIS TOOL

The analysis tool was designed as an Excel workbook with a simple data input process to ensure a user-friendly tool. The user specifies the uranium type (natural or depleted), the fuel bundle design (37- or 43-element), the end-of-irradiation fuel burnup, the average bundle power and the maximum bundle power generated during the irradiation, and the best-estimate decay power for the bundle (provided by a separate detailed calculation [3]). Each worksheet of the spreadsheet was conditionally formatted to clarify the few cells requiring data input. Cells corresponding to those fuel properties used on several worksheets have been linked, requiring only one input by the user. The analysis tool was validated and verified including conditional formatting, conditional data transfer, and algorithms.

At PLGS there exist Departmental Procedures that pertain to specific department activities. In support of the spent fuel calculations, a departmental procedure was prepared [4]. The analysis tool was prepared in parallel to accompany the procedure. This procedure can also be applied as a user's manual.

The analysis tool provides estimates for the following spent fuel parameters: total fission product inventory (fissile and non-fissile masses, and discharge activity), ^{131}I inventory, and maximum sheath temperature. The spreadsheet is divided into four worksheets corresponding to the above listed parameters. A brief description of each sheet follows.

4.1 Fissile and Non-Fissile Masses

The most recent data regarding fission product inventory were calculated using the program WIMS-AECL version 2-5b-7 [5] for CANDU®-6 37-element natural or depleted fuel, and CANFLEX® 43-element natural or depleted fuel. These data were embedded in the spreadsheet, and are conditionally transferred for the various isotopic mass calculations. The isotopic concentrations are given as a function of discharge burnup. Given a known discharge burnup of a particular fuel bundle, and using linear interpolation, the various isotope concentrations can be determined. The following categories are currently reported:

i) Fissile Uranium - ^{235}U and ^{236}U,
ii) Non-Fissile Uranium - ^{234}U and ^{238}U,
iii) Fissile Plutonium – ^{239}Pu and ^{241}Pu
iv) Total Plutonium – the sum of ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu.

4.2 Discharge Activity of Fuel Bundle

Fission product inventory is dependent on the power history of the bundle. Therefore, no simple analytic expressions can be derived for the total fission product inventory. The storage period prior to shipping reduces the importance of a detailed power history. Hence, discharge burnup and average power level become the dominant independent variables. Several runs of the code ORIGEN-2 produced a table of activities for decay times from 1 second to 20 years. This data table is also embedded in the workbook. The total activity can be interpolated either logarithmically as a function of burnup or linearly as a function of time, for a specified burnup.

4.3 Iodine-131 Inventory

The inventory of ^{131}I can be estimated for the whole fuel bundle and the outer, intermediate, inner, and centre elements using the following equation:

$I = I_o\, e^{-\lambda t}$, where:
I = ^{131}I inventory (Ci)
I_o = initial iodine at discharge (Ci)
λ = decay constant for ^{131}I (days^{-1})

t = cooling time (days).

The initial iodine inventory at discharge can be calculated as follows:

$I_o = K \cdot P_{max} \cdot CF_i$, where: K = ^{131}I equilibrium coefficient (Ci/kW)
CF_i = fractional power per element.
P_{max} = maximum bundle power (kW).

The equilibrium coefficient (K = 29.17 Ci/kW) for ^{131}I was based on an upper bound safety analysis correlation [6] number used in the CURIES code.

The decay constant is:

$\lambda = \frac{\ln 2}{t_{1/2}}$, where: $t_{1/2}$ = half-life of ^{131}I (days) ($t_{1/2}$ = 8.040 days [7]).

4.4 Maximum Sheath Temperature Calculation

The maximum sheath temperature is determined by the element with the highest heat flux (usually outer element), which is calculated with a value of decay power after cooling [3]. A simplified model uses natural convection as the only heat transfer mechanism governing the temperature of the fuel within the flask. The model used in determining the maximum sheath temperature was based on Newton's Law of Cooling for a cylindrical object:

$$q = h \cdot A \cdot \Delta T$$

where q is the element power (in Watts), h is the fuel-sheath-to-coolant (air) convective heat transfer coefficient (5 $W \cdot m^{-2} \cdot K^{-1}$) [8,9], A is the sheath's surface area (m^2), and ΔT is the temperature difference between the heated surface of the sheath and the ambient air temperature (38 °C is recommended [10]). Post-irradiated, cold dimensions were assumed when calculating the sheath surface area.

The element decay power was determined by the DCYPWR code [3]. This element decay power was based on the radial power distribution within the bundle, relative to the bundle average power level. A lower bound convective heat transfer coefficient was assumed for each ring of elements, and was considered to be constant over all fuel elements. The radial temperature distribution across the fuel bundle was assumed to be due to the radial decay power distribution and not to thermal "shielding" by neighbouring fuel elements within the bundle. Diametral and axial thermal expansion of the fuel sheath was not considered, as this effect would be small at the temperatures of interest. The shipping flask is made of a composite of lead and stainless steel, both of which have relatively high heat conduction properties. A relatively large uncertainty of 65 % (refer to section 4.5) has been applied to the sheath temperature calculation to account for the above contributing factors. This accuracy level is justified by the large margin to existing limits imposed by the transport regulations.

Further work on the sheath temperature calculations would involve a more exact model. The maximum sheath temperature calculation should be determined using a finite difference model. Boundary conditions should be established for the system, including ambient air temperature. Heat transfer phenomena through the system should be estimated using characteristics and dimensions of the fuel bundle components and the shipping flask, as well as the air surrounding the system. To simplify this model the bundle may be considered as one cylindrical unit rather than including individual elements. However, this assumption would need validation. Both radial and azimuthal directions (r, θ) should be considered to complete the 2-D model. This temperature distribution model could be used not only to determine the maximum sheath temperature, but also the temperature at various radial positions throughout the shipping flask.

At elevated temperatures, air oxidation may affect UO_2 morphology, in the case of a failed fuel sheath, which in turn would increase the fission products released from the fuel. These factors should be incorporated into the model described above.

4.5 Uncertainties

The equilibrium coefficient (K) for ^{131}I, selected for the Iodine-131 inventory calculations, was an upper bound value extracted from the PLGS Safety Report [6].

A 65 % uncertainty of the sheath temperature was previously estimated. It is applied to the calculation described in the simplified model of section 4.4 before reporting the maximum temperature. The uncertainties are briefly described below:

i) The decay power calculations for each bundle are based on the ANSI/ANS-5.1 (1994) Standard. The uncertainties, dependent on cooling time, are stated as below [11]:

Cooling Time	**Uncertainty**
$t_{cooling} < 10^3$ seconds	+ 20%, - 40%
10^3 seconds < $t_{cooling} < 10^7$ seconds	+ 10%, - 20%
$t_{cooling} > 10^7$ seconds	+ 25%, - 50%

ii) A lower-bound convective sheath-to-air heat transfer coefficient was selected from the range 5 to 30 $W \cdot m^{-2} \cdot K^{-1}$ [8,9]. Although no calculations were performed, a ±20 % uncertainty on maximum sheath temperature has been judged sufficient.

iii) A ±20 % uncertainty is attributed to the radial temperature distribution; experimental tests by J. Schenk of CRL have shown higher sheath temperatures, on those elements located towards the centre of the bundle, by between 10 °C and 40 °C [12]. The higher sheath temperatures may be due to “thermal shielding” of neighbouring fuel elements.

5.0 CALCULATIONS

The spent fuel calculations analysis tool was used to determine the spent fuel properties of various bundles, in particular CANFLEX® bundles FLX019Z and FLX024Z. Table 1 provides bundle properties and required shipping parameters that were produced using the analysis tool. For comparison, calculations for a 37-element CANDU®-6 fuel bundle (Y10000Z) have also been presented. The irradiation history for this bundle has also been shown in Figure 1. An analysis tool demonstration of the calculations of fissile and non-fissile masses, and bundle discharge activity for bundle FLX024Z are provided in Tables 2 and 3. Similarly, Tables 4 and 5 display the Iodine-131 inventory and the maximum sheath temperature calculations for bundle FLX019Z.

6.0 SUMMARY

1. The spent fuel analysis tool facilitates the calculations of estimates for several parameters required for compliance with the AECB and AECL transportation regulations.
2. The tool can be utilized for both 37-element and CANFLEX® fuel bundles.
3. Calculations supporting the shipment of two irradiated CANFLEX® fuel bundles were presented. For comparison, calculations for a CANDU®-6 fuel bundle were also given.

7.0 REFERENCES

1. "Radioactive Material Type B(U) Package Design Approval Certificate No. CDN/2061/B(U)-85, Rev.4 AECB 1999 July 23.

2. Nuclear Safety Advisory Committee, "Principles and General Rules for the Operation of the Building 234 Universal Cells", AECL-Misc 191, 1986 August.

3. D.F. Basque, J.A. Walsworth, R.A. Prime, R.W. Sancton and E.G. Young, "The DCYPWR Code: Fuel Decay Power Calculations for CANDU® Fuel and Reactor Cores", to be presented at the 21st CNS Annual Conference, Toronto, 2000 June.

4. M.-J. Basque and J.A. Walsworth, "Calculations to Support Spent Fuel Shipments", Irradiated Fuel Properties, DP-01368-03103-7 Appendix 1 Revision 1, 2000 February 16.

5. F. Ardeshiri, unpublished data to C. Newman of NB Power, AECL-Sheridan Park Memorandum, 1999 February 9.

6. "Inventory to Element Power Ratios for the Isotopes Considered in the CURIES Code", Point Lepreau Generating Station Safety Report (1997 Edition, Rev 0, HTML Version) Part 3, Appendix III, Table III.I-1.

7. F.W. Walker, J.R. Parrington, F. Feiner, "Nuclides and Isotopes, Fourteenth Edition", GE Nuclear Energy, San José, California, 1989.

8. M.A. Wright, M.S. Quraishi, "PLGS Analysis of the Consequences of an End Fitting Failure", TTR-153, Section 3.2.2, 1985 May.

9. Babcock and Wilcox, "STEAM its generation and use", Chapter 4 (Table 2), 40th Edition, 1992, p. 4-3.

10. M. Mersereau, Personal Communication – Fax, 1999 November 18.

11. "American National Standard for Decay Heat Power in Light Water Reactors", ANSI/ANS-5.1-1994, American Nuclear Society, La Grange Park, Illinois, 1994 August 23.

12. J.R. Schenk, unpublished experimental data, AECL-Chalk River Laboratories, 1993 August.

Table 1: Spent Fuel Properties Calculated via Spreadsheet Tool

	Bundle Serial No.		
	FLX024Z	**FLX019Z**	**Y10000Z**
Bundle Type (37-Element CANDU / 43-Element CANFLEX)	43-Element CANFLEX	43-Element CANFLEX	37-Element CANDU
Fuel Type: (natural/depleted)	natural	natural	natural
Decay Power (kW)	0.134	0.113	0.102
Pre-irradiation uranium mass content (kg)	18.600	18.591	19.322
Fissile Mass at discharge (g) (^{235}U, ^{236}U, ^{239}Pu, ^{241}Pu)	112.268	112.15	112.97
Remaining Mass at discharge (kg) (Pre-irradiation U – Fissile)	18.488	18.479	19.209
Fissile U Mass at discharge (g) (^{235}U, ^{236}U)	64.413	64.268	54.115
Non-Fissile U Mass at discharge (kg) (^{234}U, ^{238}U)	18.359	18.349	19.020
Fissile Pu Mass at discharge (g) (^{239}Pu, ^{241}Pu)	47.855	47.884	58.852
Total Pu Mass at discharge (g) (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu)	62.147	62.224	82.582
Non-Fissile Mass at discharge (kg) (^{234}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{240}Pu, ^{242}Pu)	18.373	18.364	19.044
Total Activity (Bq)	9.43×10^{14}	7.73×10^{14}	7.08×10^{14}
^{131}I Activity (Bq)	3.84×10^{6}	1.6×10^{5}	3.10
Estimated maximum sheath temperature of outer element (°C)	125	116	115

Table 2: Example Fissile and Non-Fissile Mass Calculations for Bundle FLX024Z From Reference 4

Irradiated Fuel Properties
DP-01368-03103-07
APPENDIX 1:
CALCULATIONS TO SUPPORT SPENT FUEL SHIPMENTS

STEP 7 : **TOTAL FISSION PRODUCT INVENTORY**

A) and B): Fissile and Non-Fissile Masses

Notes: Cells requiring user input are indicated by *capital letters in italics* and are highlighted yellow.
"-" or #VALUE! indicates a cell formula that will be automatically updated.

Bundle-Specific Data: Discharge burnup: *143.81* MWh/kgU
Pre-Irradiation mass of U: *18.600* kg

Choose a bundle and fuel type by placing an "x" in the appropriate box:
Note: only one selection is required.

Bundle Type	Type of Fuel: ***Natural Uranium (NU)***	Type of Fuel: Depleted Fuel (DEP)
37-Element CANDU-6		
43-Element CANFLEX	x	

Fissile and Non-Fissile Mass:

Interpolation from DP-01368-03103-7 App. 1, Isotopic Concentrations vs Burnup
In Table ***5***

Discharge Burnup (MWh/kgU)	Specific Isotopic Concentration (g/initial kgU): ^{234}U	^{235}U	^{236}U	^{238}U	^{237}Np	^{238}Pu	^{239}Pu
137.82	0.046	2.915	0.639	987.230	0.019	0.002	2.395
143.81	0.046	2.808	0.655	986.979	0.020	0.002	2.427
157.42	0.045	2.565	0.691	986.410	0.022	0.003	2.500
Concentration in This Bundle (g):	**0.850**	**52.230**	**12.183**	**18357.817**	**0.371**	**0.043**	**45.142**

Discharge Burnup (MWh/kgU)	Specific Isotopic Concentration (g/initial kgU): ^{240}Pu	^{241}Pu	^{242}Pu	Fissile Pu	Total Pu	^{242}Cm	^{241}Am
137.82	0.694	0.136	0.028	2.531	3.255	0.000	0.001
143.81	0.734	0.146	0.032	2.573	3.341	0.000	0.001
157.42	0.825	0.168	0.041	2.668	3.537	0.000	0.001
Concentration in This Bundle (g):	**13.655**	**2.713**	**0.594**	**47.855**	**62.147**	**0.003**	**0.018**

Fissile Mass (g): 112.268 ($^{235}U + ^{236}U + ^{239}Pu + ^{241}Pu$); per PIR 99-011 Rev 0
Remaining Mass (kg): 18.488 (pre-irradiation U - Fissile)

Other Calculations:

Fissile U Mass (g):	64.413	($^{235}U + ^{236}U$); per PIR 99-011 Rev 0
Non-Fissile U Mass (kg):	18.359	($^{234}U + ^{238}U$); per PIR 99-011 Rev 0
Fissile Pu Mass (g):	47.855	($^{239}Pu + ^{241}Pu$); per PIR 99-011 Rev 0
Total Pu Mass (g):	62.147	($^{238}Pu + ^{239}Pu + ^{240}Pu + ^{241}Pu + ^{242}Pu$); per PIR 99-011 Rev 0
Non-Fissile Mass (kg):	18.373	($^{234}U + ^{238}U + ^{237}Np + ^{238}Pu + ^{240}Pu + ^{242}Pu$); per PIR 99-011 Rev 0

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Worksheet: Step7 A & B

Table 3: Example Bundle Discharge Activity Calculations for Bundle FLX024Z From Reference 4

Irradiated Fuel Properties
DP-01368-03103-07
APPENDIX 1:
CALCULATIONS TO SUPPORT SPENT FUEL SHIPMENTS

STEP 7 : TOTAL FISSION PRODUCT INVENTORY

C): Discharge Activity of Fuel Bundle

Notes: Cells requiring user input are indicated by *capital letters in italics* and are highlighted yellow.
"-" or #VALUE! indicates a cell formula that will be automatically updated.

Bundle-Specific Data:

Cooling Time:	*1.89E+07*	sec
Avg. Bundle Power:	*535.1*	kW
Discharge burnup:	*143.81*	MWh/kgU
Power Output:	28.769	kW/kgU
Pre-Irradiation mass of U:	*18.600*	kg

Activity at Power Output: ***25.419*** **kW/kgU**

From Table 7: Bundle Activity vs. Decay Time, DP-01368-03103-7 App. 1

Discharge Burnup (MWh/kgU)	Time 1 (sec) 1.44E+07	t (sec) 1.89E+07	Time 2 (sec) 2.16E+07
100	1497.2	**1079.8**	889.7
143.81	**1763.6**	**1315.6**	**1105.9**
200	2105.2	**1617.2**	1383.3

Interpolating Logarithmically @ burnup discharge:	**1315.6**	Ci/kgU
Interpolating Linearly @ time t:	**1315.2**	Ci/kgU

Activity at Power Output: ***45.755*** **kW/kgU**

From Table 7: Bundle Activity vs. Decay Time, DP-01368-03103-7 App. 1

Discharge Burnup (MWh/kgU)	Time 1 (sec) 1.44E+07	t (sec) 1.89E+07	Time 2 (sec) 2.16E+07
100	1869.9	**1302.9**	1051.9
143.81	**2295.5**	**1646.7**	**1352.6**
200	2841.4	**2086.9**	1738.2

Interpolating Logarithmically @ burnup discharge:	**1646.7**	Ci/kgU
Interpolating Linearly @ time t:	**1646.4**	Ci/kgU

Interpolating Between Power Outputs:

Power output (kW/kgU)	Total Activity (Ci/kgU)
25.419	1315.6
28.769	**1370.1**
45.755	1646.7

Total Fission Product Inventory/Activity (Ci): 25484

Total Fission Product Inventory/Activity (Bq): **9.43E+14**

$1\ \text{Ci} = 3.7 \times 10^{10}\ \text{Bq}$

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Worksheet: Step7 C

Table 4: Example Iodine-131 Inventory Calculations for Bundle FLX019Z From Reference 4

Irradiated Fuel Properties
DP-01368-03103-07
APPENDIX 1:
CALCULATIONS TO SUPPORT SPENT FUEL SHIPMENTS

STEP 8 : **IODINE-131 INVENTORY**

Notes: Cells requiring user input are indicated by *capital letters in italics* and are highlighted yellow.
"-" or #VALUE! indicates a cell formula that will be automatically updated.

^{131}I Inventory:

Bundle Selection:

Note: only one bundle type should be selected.

Selection	Type of Bundle
	37-Element CANDU
x	43-Element CANFLEX

Data:

Variable	Description	Value	Units
K	^{131}I Equilibium coefficient [20]	29.17	Ci/kW
$t_{1/2}$	Half-Life of ^{131}I [21]	8.040	days
λ	Decay constant for ^{131}I	0.08621	days^{-1}
t	Cooling Time	*256.0*	days
P_{max}	Maximum Bundle Power	*570*	kW

Calculations:

43-Element CANFLEX-NU Bundle

Variable	Description	Entire Bundle	Outer Element	Intermediate Element	Inner Element	Centre Element	Units
$N_{Elements}$	# Elements per Ring	43	21	14	7	1	dimensionless
CF2	Fractional Power per Element	1.000	0.02461	0.02026	0.02507	0.02398	dimensionless
P_{max}	Maximum Power	570	14.0	11.6	14.3	13.7	kW
I_o	Initial Iodine at discharge	16627	409	337	417	399	Ci
I	^{131}I Inventory after cooling	4.32E-06	1.06E-07	8.76E-08	1.08E-07	1.04E-07	Ci
		1.60E+05	**3.94E+03**	**3.24E+03**	**4.01E+03**	**3.84E+03**	Bq

$$\lambda = \frac{\ln 2}{t_{1/2}}$$

$$CF_i = \frac{\text{Ring Linear Power}}{\text{Total Bundle Linear Power}} \times \frac{1}{N_{elements}}$$

(Except for Whole Bundle, where $CF_i = 1.0$)

$$I_o = K \cdot P_{max} \cdot CF_i$$

$$I = I_o\, e^{-\lambda t}$$

$$1\,Ci = 3.7 \times 10^{10}\ Bq$$

37-Element CANDU Bundle				43-Element CANFLEX-NU Bundle			
Table 2.4-6 [13] LINEAR POWER AND POWER DISTRIBUTIONS AT NOMINAL DESIGN BUNDLE POWER (800 kW)				Table 2.4-6 [14] LINEAR POWER AND POWER DISTRIBUTIONS AT CANFLEX-NU NOMINAL DESIGN BUNDLE POWER (800 kW)			
Element Ring	Number of Elements, $N_{Elements}$	Ring Linear Power (kW/m)	Fractional Power per Element, CF1	Element Ring	Number of Elements, $N_{Elements}$	Ring Linear Power (kW/m)	Fractional Power per Element, CF2
Whole Bundle	37	1666	1.000	Whole Bundle	43	1663.54	1.000
Outer	18	915.8	0.03054	Outer	21	859.74	0.02461
Intermediate	12	498.0	0.02491	Intermediate	14	471.94	0.02026
Inner	6	217.7	0.02178	Inner	7	291.97	0.02507
Centre	1	34.33	0.02061	Centre	1	39.89	0.02398

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Worksheet: Step 8

Table 5: Example Maximum Sheath Temperature Calculations for Bundle FLX019Z From Reference 4

Irradiated Fuel Properties
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APPENDIX 1:
CALCULATIONS TO SUPPORT SPENT FUEL SHIPMENTS

STEP 9 : MAXIMUM SHEATH TEMPERATURE

Notes: Cells requiring user input are indicated by *capital letters in italics* and are highlighted yellow.
"-", "~" or #VALUE! indicates a cell formula that will be automatically updated.

$$T_{sheath} = \frac{q}{h \cdot A} + T_{air} \qquad q'' = \frac{q}{A} \qquad T_{sheath} = \frac{q''}{h} + T_{air}$$

Assumptions: $q = P_{element}$

$$P_{element} = P_{bundle} \times CF_i$$

Where: T_{sheath}, surface temperature of hottest element
q, heat generated (W)
h, convective heat transfer coefficient [16,17] (W m^{-2} K^{-1}) = 5
A, element surface area (m^2)
T_{air}, ambient air inside the flask (°C) - suggested 38 °C [18] = *38*
q'' , surface heat flux (W/m^2)
$P_{element}$, element decay power (W)
P_{bundle}, decay power after cooling (W) = *113*
CF_i , fractional power per element (see STEP 8)
Uncertainty value (%) - suggested ± 65 % [5] = *65*

Variable	Description	Entire Bundle	Outer Element	Intermediate Element	Inner Element	Centre Element	Units
$N_{Elements}$	# Elements per Ring	43	21	14	7	1	dimensionless
CF2	Fractional Power per Element	1.00000	0.02461	0.02026	0.02507	0.02398	dimensionless
$P_{element}$	Decay Power per Element	*113*	2.78	2.29	2.83	2.71	W
A	Surface Area	0.771236	0.017356	0.017356	0.020472	0.020472	m^2
q ''	Heat Flux		160.23	131.93	138.40	132.36	W/m^2
T_{sheath}	Sheath Temperature		70.0	64.4	65.7	64.5	° C

Bundle Type: **43-Element CANFLEX-NU Bundle**

Results:
Element with q''_{max} = **outer element**
q''_{max}, maximum heat flux (W/m^2) = 160 W/m^2
T_{max}, best-estimate maximum sheath temperature (°C) = 70.0 ° C

Uncertainty value (%) = 65 %
T_{adj}, adjusted maximum sheath temperature (°C) = **116** **° C**
Note: Enter T_{adj} value on TIM.

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Worksheet: Step 9 A

Figure 1: Reactor Power and Bundle Power as a Function of Date or FPD