Nuclear Heating Of In-Core Components For CANFLEX Low-Void- Reactivity Fuel In Bruce B

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ABSTRACT

A nuclear heat balance showing the distribution of nuclear energy for a Bruce B reactor fuelled with CANFLEX® Low Void Reactivity Fuel (LVRF) bundles has been assembled.

The nuclear energy balance in the Bruce-B reactor components is obtained from balance tables that take into consideration the different aspects of nuclear heating: release (production), deposition and transfer. The distribution of nuclear energy—inside the core and outside the core—generated in the fission process in the fuel is calculated with a different methodology for each region.

1 INTRODUCTION

A nuclear heat balance was derived for an LVRF fuelled core with an output of 2702.8 MW(th) of nuclear heat generated in the fuel channels. Heat deposition can affect the design and performance of a number of in-core components and this assessment was undertaken to evaluate whether cooling loads for various core components would change significantly when LVRF is implemented at Bruce B.

The work was performed for CANFLEX LVRF bundles containing 43 UO_2 fuel elements with a slight U-235 enrichment for fuel bundle rings 2,3, and 4, and natural uranium (NU) plus Dysprosium burnable poison as Dy_2O_3 in the central pin.

This work consisted of generating and performing the analysis of a heat balance for the main components of the Bruce B reactor with LVRF fuel. The heat balance is

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presented as tables that show the distribution of energy (generated and deposited) for the main components of the primary systems of the reactor: heat transport system (HTS), moderator system, and shield cooling system (SCS).

Originally, Bruce B was fuelled against the flow, with 13 fuel bundles in the channel, and the first and last bundles approximately halfway into the core. The refuelling direction in the Bruce B reactors is being changed to fuelling with flow, with a 12 bundle fuel string. In this configuration, the last bundle in the channel at the downstream end will be only halfway in the core, and at the upstream end, there will be approximately a half-bundle length of coolant before the first bundle in the channel. This will result in a checkerboard arrangement of axial reflectors at alternate ends of adjacent channels.

The distribution of nuclear energy released in the fission process in the fuel and in the neutron capture in fuel and other components, and the distribution of the neutron, gamma, and beta energy deposited throughout the fuel channels and associated moderator in each lattice cell was calculated in 3 dimensions with the Monte Carlo transport code MCNP4C [1]. The neutron and gamma radiation escaping through the axial and radial core shields was calculated in 1 dimension with the transport code ANISN [2].

2 METHODOLOGY

The nuclear energy released in the fission process appears as the kinetic energy of recoiling fission fragments, of prompt and delayed neutrons, and as the energy of prompt gammas from fission. In addition, nuclear energy is released in the beta-particle decay of radioactive fission fragments carried by beta-particles, photons and antineutrinos and some nuclear energy is released in (n,γ) reactions.

The nuclear energy released per fission of the different components: prompt (fission fragments, prompt gamma rays, and prompt neutrons), and delayed (beta particles, delayed gamma rays, antineutrinos, and delayed neutrons) for the main fissionable isotopes in the fuel (^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu) was obtained from Reference [3]. The weighted-average values were obtained using the fission rates calculated for a mid-burnup lattice cell, calculated with WIMS-AECL, Reference [4].

The transport code MCNP4C (Reference [1]) was used to calculate the distribution of prompt-neutron and gamma energy deposited inside the core regions: the fuel channel region including fuel, clad, coolant, pressure tubes; and the moderator region, including moderator and calandria tubes. The delayed-gamma contribution was also calculated with MCNP4C, using a gamma source calculated with the code ORIGEN-S (Reference [5]) for a fuel composition corresponding to an average mid-burnup value.

The neutron and gamma prompt heating and the delayed-gamma heating deposited in the core are the fraction of energy deposited in the different components of the core, calculated with MCNP4C. The MCNP results assume that the fission fragment (FF) energy and the beta energy are completely deposited in the fuel. As the results obtained from MCNP did not distinguish between FF and neutron energy deposited in the fuel, the fraction of prompt neutron energy deposited in the fuel was obtained from the following equation:

$$fF_N = (E_{FFN} * fF_{FFN} – E_{FF}) / E_N$$

where fF_N is the fraction of prompt neutron energy deposited in the fuel,
E_{FFN} is the neutron plus FF energy released per fission,
fF_{FFN} is the fraction of neutron plus FF energy deposited in the fuel,
E_{FF} is the FF energy released per fission and deposited in fuel, and
E_N is the neutron energy released per fission.

For the distribution of the nuclear energy outside the core, the nuclear heating distribution through the primary shields—axial and radial—was calculated in 1 dimension with ANISN (Reference [2]). The neutron and the gamma energies escaping from the core were also extracted from ANISN.

From the ANISN calculations of the nuclear heating generated in the axial and radial primary shields, the net current at the outer boundary of the core region—in particles per cm^2 per second—was used to evaluate the neutron and gamma energy escaping from the core. The value obtained in this way was multiplied by a form factor in each case (axial or radial). The axial form factor calculated from the shape of the bundle-power distribution in the peripheral channels from a reactor physics refuelling simulation was used in the calculation of the radial energy escaping from the core. The radial form factor calculated from the bundle-power distribution for the bundles in position 1 was used in the calculation of the axial energy escaping from the core. Both distributions were taken from a time-average power distribution generated for a reference fuel design.

3 HEAT BALANCE: FISSION RATE CALCULATION

To assemble a heat balance that deposits the appropriate nuclear energy in the fuel channels, the absolute fission rate in the core has to be accurately calculated. Thus, the total fission rate (in number of fissions per second) in the reactor is calculated using the nuclear energy deposited in the fuel channels that takes into account the nuclear energy released inside the core less the neutron and gamma energies escaping from the core.

If we consider only the components of the energy released per fission event that are deposited inside the channels, or escape but will be recovered in the primary or secondary cooling systems—only the antineutrino energy is completely lost—, the energy per fission generated inside the fuel channels (E_{CH}) is:

$$E^{CH} = E_{FFN}{}^{CH} + E_{\gamma CP}{}^{CH} + E_{\gamma DEL}{}^{CH} + E_{\beta}{}^{CH}$$

where $E_{FFN}{}^{CH}$ is the prompt neutron and FF energy generated in the channels:

$$E_{FFN}{}^{CH} = E_{FF} + fC_N * E_N$$

with E_{FF} is the FF energy released per fission and deposited in fuel,
fC_N is the fraction of neutron energy deposited in the channel, and
E_N is the neutron energy released per fission.

$E_{\gamma CP}{}^{CH}$ is the prompt gamma plus the capture gamma energy deposited in the channel:

$$E_{\gamma CP}{}^{CH} = [E_\gamma + E_{CP} * (\nu - 1 - L)] * fC_\gamma$$

where E_γ is the prompt gamma energy generated per fission,
E_{CP} is the capture gamma energy released per neutron (Reference [6]),
fC_γ is the fraction of prompt gamma energy deposited in the channel, and
$(\nu - 1 - L)$ is the number of captures per fission;

with ν is the number of neutrons released per fission, and
$L = N_L / N_{TOT}$ is the leakage factor;

and N_L is the neutron leakage,
$N_{TOT} = P_{COOL} / (k * E_{FISS} / \nu)$ is the total number of neutrons per s,
P_{COOL} is the total core power to channels,
E_{FISS} is the energy released per fission inside channels (= E^{CH}), and
k is a unit conversion factor (= 1.602177 E-19 MW•s/MeV).

$E_{\gamma DEL}{}^{CH}$ is the delayed gamma energy deposited in the channel:

$$E_{\gamma DEL}{}^{CH} = E_{\gamma DEL} * fC_{\gamma DEL}$$

with $E_{\gamma DEL}$ is the delayed-gamma energy released per fission, and
$fC_{\gamma DEL}$ is the fraction of delayed-gamma deposited in the channel.

and the beta-particle energy released per fission is

$E_\beta{}^{CH}$ The beta energy is all deposited in the fuel.

To calculate the neutron and gamma heating losses from the core channels (to obtain the fission rate in the reactor), the values of the neutron and gamma energy deposited in the reactivity devices used are estimated.

We define the neutron and gamma energy losses from the core channels as

$$E_{N,LOSSES}^{CH} = (E_{N,REACT\text{-}LOSSES} + E_{N,LOSSES}) * fC_N$$

with $E_{N,REACT\text{-}LOSSES}$ is the neutron energy generated in the reactivity devices,
$E_{N,LOSSES}$ is the neutron energy escaped from the core (from ANISN), and
fC_N is the fraction of neutron energy deposited in the channel.

$$E_{\gamma,LOSSES}^{CH} = (E_{\gamma,REACT\text{-}LOSSES} + E_{\gamma,LOSSES}) * fC_\gamma$$

with $E_{\gamma,REACT\text{-}LOSSES}$ is the gamma energy generated in the reactivity devices,
$E_{\gamma,LOSSES}$ is the gamma energy escaped from the core (from ANISN), and
fC_γ is the fraction of prompt gamma energy deposited in the channel.

Finally, the overall power in the reactor is

$$P_{TOTAL} = P_{COOL} + E_{N,LOSSES}^{CH} + E_{\gamma,LOSSES}^{CH}$$

with P_{COOL} is the total core power delivered to fuel channels

and, the fission rate is

$$FR = P_{TOTAL} / (k * E_{FISS})$$

with E_{FISS} is the energy released per fission, and
k is a unit conversion factor.

The fission rate obtained was $8.5 \cdot 10^{19}$ fissions/s.

4 HEAT TRANSFER OF THE NUCLEAR ENERGY GENERATED IN THE CALANDRIA SHELL AND TUBESHEET

The Bruce-B reactor was designed with 13 NU fuel bundles in each of the 480 fuel channels (Reference [7]). The Bruce B reactor with LVRF will have 12 bundles per channel. With LVRF, all the channels have half of one end bundle penetrated into the end shield, and the other end bundle separated by half bundle length of axial reflector (D_2O coolant) from the end shield. Each end shield side of the reactor core has half of the channels penetrated through the end shield (case ‘Axial-2’), and the other half separated from it by the axial reflector (case ‘Axial-1’).

First, the neutron and gamma heating generated in the heavy water reflector, in the calandria shell, and in the light water inside the shield tank are calculated with a radial ANISN model. Then, the neutron and gamma heating generated in the axial D_2O reflector, in the calandria-side tubesheet (CSTS), the end-shield H_2O, the baffle plate, and in the steel-balls/light-water region ("60/40" region) are calculated with an axial ANISN model (case 'Axial-1'). Also, the neutron and gamma heating generated in the calandria-side tubesheet (CSTS), the end-shield H_2O, the baffle plate, and in the steel-balls/light-water region ("60/40" region) are calculated with an axial ANISN model (case 'Axial-2'). The nuclear heating deposited in the fuel machine-side tubesheet (FMTS) for the axial models, and in the shield tank for the radial model are not included in the present analysis because the values are negligible.

To obtain the fraction of the nuclear heat deposited in the calandria shell and in the calandria-side tubesheet transferred to the light water inside the shield tank (part of the shield cooling system), and the fraction transferred to the moderator system, a simple heat transfer treatment was applied (Reference [8]).

The values for the volumetric heat source prior to attenuation (in MW/cm^3) and the linear attenuation coefficient (in cm^{-1}) in the CSTS and in the calandria shell are obtained from the graphs of heat density as a function of thickness in the ANISN output.

After calculating the location of the maximum temperature in the CSTS and in the calandria shell, the total heat deposited in each one is split accordingly to that location, where the energy deposited on either side of that maximum temperature is assumed to be transferred down the temperature gradient.

5 HEAT LOSSES

The nuclear heating transferred from the fuel channel region in the core (comprising the fuel, clad, coolant and pressure tubes) to the moderator region (calandria tubes plus moderator) through the gas annulus (made of low density CO_2) has two main components: the natural convection heat transfer and the radiant heat transfer.

The natural convection heat transfer across the annulus gap was determined (Reference [9]) by

$$Q_{COND} = N_c * A_c * h * \Delta T$$

where N_c is 480 channels,
A_c is the inner surface area of the calandria tube,
ΔT is the difference between HTS and moderator temperatures, and
$h = Nu * k / x$;

with k is the CO_2 thermal conductivity,
x is the thickness of the gas annulus, and
Nu is the Nusselt number.

The radiant heat transferred was calculated with the formula from Reference [10]:

$$Q_{RAD} = \sigma * A_1 * \mathcal{F}_{12} * (T_1^4 - T_2^4)$$

where σ is the Stefan-Boltzmann constant,
A_1 is the PT outer area for all channels,
T_1 and T_2 are the PT and CT temperatures, and
$\mathcal{F}_{12}$ is the area factor for grey surface in enclosure

The total nuclear heating from the fuel channels transferred to the moderator system through the gas annulus is

$$Q(\text{fuel chan} \rightarrow \text{moder}) = Q_{COND} + Q_{RAD}$$

The heat loss to end shields, 3.67 MW(th), was estimated from a CANDU 6 value but normalized to Bruce-B using the number of channels and the average HTS temperature for each reactor.

6 RESULTS

To balance the nuclear heating in the core components, it is assumed that all the kinetic energy of the fission fragments and all the beta-particle energy released per fission event in the fuel are fully deposited in the fuel.

The following definitions are needed to build the heat balance table for the lattice components.

$PF_{FF} = FR * k * E_{FF}$ is the FF (fission fragments) energy deposited per second in the fuel, where
FR is the fission rate,
k is a unit conversion factor, and
E_{FF} is the FF energy released per fission and deposited in fuel.

$PX_N = FR * k * E_N * fX_N$ is the neutron energy deposited per second in the component X, where
E_N is the neutron energy released per fission, and
fX_N is the fraction of neutron energy deposited in component X (MCNP).

$PX_{N\text{-}LOST} = E_{N\text{-}LOST} * fX_N$ is the neutron energy lost per second from the component X, where

$E_{N\text{-}LOST}$ is the neutron energy lost from the core.

$PX_\gamma = FR * k * [E_\gamma + E_{CP} (\nu - 1 – L)] * fX_\gamma$ is the prompt-gamma energy deposited per second in X, where

E_γ is the prompt-gamma energy released per fission,
$(\nu – 1 – L)$ is the amount of neutrons captured per fission,
E_{CP} is the capture-gamma energy per $(\nu – 1 – L)$ neutron, and
fX_γ is the fraction of prompt-gamma energy deposited in X.

$PX_{\gamma\text{-}LOST} = E_{\gamma\text{-}LOST} * fX_\gamma$ is the gamma energy lost per second from the component X, where

$E_{\gamma\text{-}LOST}$ is the gamma energy lost from the core

$PX_{\gamma DEL} = FR * k * E_{\gamma DEL} * fX_{\gamma DEL}$ is the delayed-gamma energy deposited per second in X, where

$E_{\gamma DEL}$ is the delayed-gamma energy released per fission, and
$fX_{\gamma DEL}$ is the fraction of delayed-gamma energy deposited in X.

$PF_\beta = FR * k * E_\beta$ is the β-particle energy released per fission and deposited per second in the fuel.

A table derived assembling these terms (Table 6.1) shows the distribution of energy in the cell components for an average lattice.

The next table (Table 6.2) shows the different heat contributions for the different systems of the reactor. The table shows the energy generated through different components, and subsequently, the transfer of energy, showing the heat deposited in the heat transport system, moderator system, and shield cooling system.

For the heat transferred to the shield cooling system, the following definitions are used: I) The heat originated in the calandria-side tube sheets and deposited in the end shields, is the total of the two different axial configurations considered, 'axial-1' and 'axial-2', and II) The heat from fuel channels is the same as the heat loss to end shields from the moderator system.

The results will not be significantly affected (see reference [11]) if the calculations are performed using an updated version of the MCNP cross-section library, which includes gamma production for the dysprosium (only in the central pin, with 12.91 wt% Dy/U). It should be underlined that the reactions (n,γ) for dysprosium were already included in the library used for this work.

TABLE 6.1
NUCLEAR HEATING DEPOSITION IN THE LATTICE CELL COMPONENTS [MW(th)]

Component:	**P_{FF}**	**P_N**	**P_N (losses)**	**P_N (net)**	**P_γ**	**P_γ (losses)**	**P_γ (net)**	**$P_{\gamma DEL}$**	**P_β**	**Totals**
Fuel	2346.05	22.59	0.151	22.44	152.85	4.930	147.92	62.44	84.80	2663.65
Clad		0.095	0.001	0.094	6.392	0.206	6.186	1.722		8.00
Coolant		9.700	0.065	9.635	5.170	0.167	5.003	1.432		16.07
PT		0.147	0.001	0.146	12.276	0.396	11.88	3.052		15.08
Sub-total	**2346.05**	**32.53**	**0.22**	**32.31**	**176.69**	**5.70**	**170.99**	**68.65**	**84.80**	**2702.80**
CT		0.088	0.001	0.088	4.921	0.159	4.763	1.301		6.15
Moderator		38.65	0.258	38.39	50.15	1.618	48.54	12.609		99.53
Sub-total		**38.74**	**0.26**	**38.48**	**55.07**	**1.78**	**53.30**	**13.91**		**105.68**
Total	**2346.05**	**71.27**	**0.48**	**70.79**	**231.77**	**7.48**	**224.29**	**82.56**	**84.80**	**2808.49**

TABLE 6.2
BRUCE B HEAT BALANCE

HEAT BALANCE DATA	MW(th)	MW(th)	MW(th)
1. - HEAT GENERATION BALANCE			
HEAT GENERATED IN:			
Fuel Bundles	2663.65		
Fuel Sheaths	8.00		
Coolant	16.07		
Pressure Tubes	15.08		
Total (Fuel Channels)		**2702.80**	
Moderator	99.53		
Reflector	3.68		
Axial D_2O Reflector	0.43		
Calandria Tubes	6.15		
Reactivity Mechanisms, Guide Tubes & Adjusters	4.14		
Calandria Shell	0.90		
Calandria Tubesheets	0.56		
End Shields	0.46		
Shield Tank Water	0.44		
Total (Other Reactor Components)		**116.29**	
Total Fission Heat			**2819.09**
2. - HEAT TRANSFER BALANCE (PRIMARY)			
2.1. - HEAT TRANSPORT SYSTEM			
Heat from Fuel Channels		**2702.80**	
Heat Loss to Moderator (Convective + Radiation)	-3.53		
Heat Loss to End Shields	-3.67		
Total Heat Loss (Moderator and End Shields)		**-7.20**	
Net Fission Heat to Coolant (before piping heat loss)			2695.60
Total Heat Loss (Piping)		-1.60	
Net Fission Heat to Coolant			2694.00
Total Heat Loss (Purification)		-13.00	
Net Fission Heat to Boiler & Preheaters			2681.00
Pumping Energy Appearing in Coolant		17.30	
Total Heat Transferred to Boilers & Preheaters			**2698.30**
2.2. - MODERATOR SYSTEM			
Heat Generated in Moderator	99.53		
Heat Generated in Reflector	3.68		
Heat Generated in Axial D_2O Reflector (at alternate ends)	0.43		
Heat Generated in Calandria Tubes	6.15		
Heat Generated in Reactivity Mechanisms, Guide Tubes	4.14		
Heat from Calandria Shell	0.42		
Heat from Calandria Tubesheets	0.14		
Heat from Fuel Channels	3.53		

Total Fission Heat to Moderator		**118.03**	

TABLE 6.2
BRUCE B HEAT BALANCE (Cont.)

	MW(th)	MW(th)	MW(th)
2.3. - SHIELD COOLING SYSTEM			
Heat from Calandria Tubesheets to End Shields	0.42		
Heat Generated in End Shields (Axial H2O region)	0.04		
Heat Generated in End Shields ('Baffle Plates' region)	0.28		
Heat Generated in End Shields ('60/40' region)	0.14		
Heat from Fuel Channels	3.67		
Total Fission Heat Appearing in the End Shields		**4.54**	
Heat from Calandria Shell to Shield Cooling System	0.48		
Heat Generated in Shield Tank	0.44		
Total Fission Heat Appearing in the Shield Tank H_2O		**0.92**	
Total Heat to Shield Cooling Heat Exchangers		**5.47**	
2.4. - SUMMATION			
Net Fission Heat to Boilers & Preheaters		**2698.30**	
Net Fission Heat to Moderator		**118.03**	
Net Fission Heat to Shield Cooling System		**5.47**	
Net Fission Heat to Piping & Purification System		1.90	
Total Fission Heat			**2823.07**
Efficiency of Fission Heat Conservation to Boilers & Preheaters (2698.30 / 2823.07 =)			95.6%

7 CONCLUSIONS

The heat balance calculation for Bruce B with LVRF fuel shows no increase in the heat load to the moderator over the moderator heat load for Bruce B with NU fuel [12]. The heat load is the total of the nuclear energy deposited in the moderator, reflector, axial reflector, calandria tubes, and other components, 118 MW(th).

The results for the distribution of neutron and gamma energy within the lattice cell are comparable with previous analysis (Reference [12]) that used a one-dimensional discrete ordinates analysis of order S_8 for the ANISN calculations.

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