A METHODOLOGY USED IN CALCULATING NUCLEAR HEAT BALANCE

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ABSTRACT

In this paper, a methodology to calculate the heat balance of a reactor core has been formulated. The methodology is generic and can be used for all CANDU® reactor designs including the Advanced CANDU Reactor™ (ACR).

Hitherto the calculation of the heat balance of a CANDU reactor relies on results generated by different computer codes. The methodology developed in this paper aims at using one code to derive all parameters used in the heat balance. The code of choice is MCNP5 [1], which is also used in deriving many of the reactor-physics parameters. The methodology allows the heat balance to be calculated using the same computer code and cross-section data that are used in reactor-physics calculations.

METHODOLOGY

The heat balance was assembled by treating the reactor core as a repeating array of lattice cells and calculating the nuclear energy deposited within lattice cells that simulate beginning of life or fresh fuel, mid-burn-up and end-of-life or exit-burn-up. Nuclear heat generation in each fuel lattice was calculated using the Monte Carlo transport code MCNP5 [1]. Simple analytical treatments were used to calculate energy transfer between reactor components due to radiation, convective, and conductive heat transfer.

In this paper, the methodology was applied to a preliminary design concept for the ACR, a pressure-tube reactor with pressurized light-water coolant in 520 fuel channels with heavy-water moderator.

The methodology was developed to provide more accurate heat-balance calculations and minimize the use of results generated by other codes. Since MCNP is being used for reactor-physics-analysis work, the use of this methodology to derive the heat balance would provide consistency within a project. Moreover, parameters like particle leakage, and nuclear heating in reactivity devices, calandria shell, and shield-cooling system can now be generated using the same code.

Monte Carlo Analysis

In the Monte Carlo simulation, a 3x3 array of fuel lattices for 2 fuel-bundle lengths was modeled. The fuel channels in this preliminary version of the ACR reactor are arranged in a 245 mm-square lattice within a 7.5m-diameter calandria shell. The ACR uses a variation of the CANFLEX fuel bundle [2]. Each fuel bundle comprises 43 elements with dysprosium in the central zirconia element and uniform enrichment of ^{235}U in the outer 3 rings of 42 elements. The boundary conditions were periodic at the boundaries of the fuel lattices, and reflective at the ends of the fuel bundles.

The fuel compositions for the mid- and exit-burnup fuel were generated using WIMS-IST [3] at 10057 MWd/MgU and 21822 MWd/MgU, respectively. When the depletion module of the MCNP code has been validated, one can generate the fuel compositions of a fuel bundle at different burnup using MCNP. The fuel-type arrangement is shown in Figure 1: one fresh bundle, one exit-burn-up bundle, and seven mid-burn-up bundles. There are a total of 18 lattice cells in the model, 9 lattice cells with the exact fuel-type arrangement end on to the 9 bundles shown. This 18-cell arrangement was chosen to simulate closely to the operating core condition and to accommodate any effects that may have caused by reactivity devices like the zone-control unit.

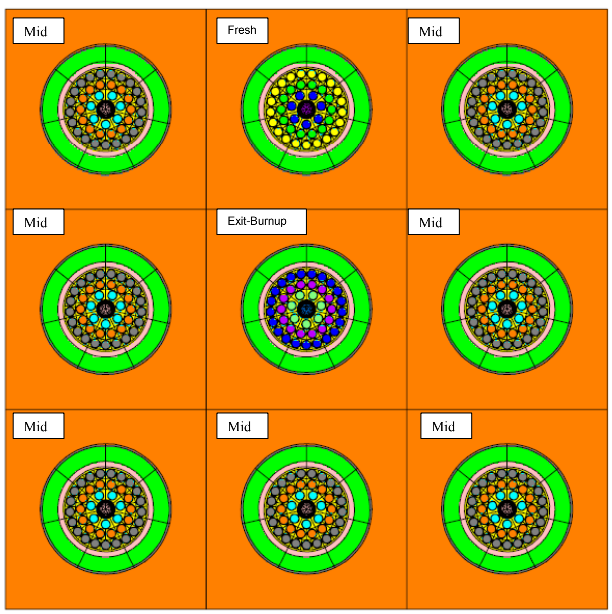

Figure 1
Fuel-Bundle Types Arrangement in a 3x3 Grid

Since the current MCNP5 cross-section data do not contain decay components. Decay energies due to nuclear disintegration of fission products in the fuel and activation products from structural components were calculated separately. The approach to account all energies of the fuel in the lattice was to run MCNP separately in two modes: prompt (*kcode*) and fixed-source (*sdef*) modes.

Prompt Calculation

Prompt analysis is analogous to an eigenvalue-search calculation or a criticality-search calculation. In a *kcode* calculation mode, prompt neutron and gamma particles are tracked. The energy deposited along their *random walk* is also recorded. The cross-section library provides coupled neutron and photon data so tracking of both neutrons and photons is done within one run. For example, in a fission event both gamma particles (γ) and fission neutrons are released from the fission site; MCNP5 will track particles from that release site to their next interaction.

To obtain nuclear-energy distribution in all regions of the lattice model, energy deposition in all regions of the model are scored using the *F6* tally. MCNP5 tallies the energy deposition based on the product of the flux ϕ, the cross section σ_i of reaction i of the material, the associated energy release H_i of reaction i, the number density N of the material, and the reciprocal of the material mass m.

$$F6 = \frac{N}{m} \sigma_i \phi \mathrm{H}_i$$

In the *F6* tally, energy is deposited only when there is an interaction and this energy is a direct physical simulation of nuclear energy deposition. More precisely, the *F6* tally represents the kinetic energy carried by a neutral particle into the interaction plus any energy released by the interaction minus the kinetic energy carried away by the recoiling particle and any particles arising from the interaction. This tally option was used to derive the energy deposited in all regions of the lattice cell. The sum of the *F6* tallies for neutrons, E_{np}, and gamma particles, $E_{\gamma p}$, is the total nuclear heat in the lattice cell due to prompt radiation.

Fixed-Source Calculation

Within the fuel lattice, nuclear heating due to activation decay γ particles in zirconium and in Dy is insignificant (both are less than 0.1% of the total recoverable fission heat) when compared to γ particles from fission products and actinides (referred hereto as *FP*). Therefore the fixed-source calculation (*sdef* mode) focuses only on the decay of γ particles from *FP*.

There are three fuel types in the model. Decay spectra for mid- and exit-burnup fuel were generated at 10057 MWd/MgU and 21822 MWd/MgU, using WIMS-IST and ORIGEN-S [4] in a coupled calculation. For fresh fuel, the spectra were generated at 1-h irradiation; this allows *FP* build-up in the fuel but not long enough to have an effect on the reactivity worth of the fuel due to Xe and other poisons. The three decay spectra were input into the MCNP5 model with the fission-power profile generated in the prompt calculation, and the geometry distribution based on the pin areas. Table 1 lists the decay-photon spectra for the mid-burnup bundle. MCNP5 starts γ particles randomly in the fuel region according to the combined distribution assigned to the pins. The γ particles are then tracked from the points of origin to the sites that they are absorbed. Over the entire particle track, all interactions are tallied and finally output as heating based on the *F6* tally; this is the decay-γ heating $E_{\gamma d}$ of the lattice cell.

Table 1
***FP* Decay γ Spectra for Mid-Burnup Fuel (photons/s)**

	Mid-Burnup Fuel at 10057 MWd/MgU			
Mean E (MeV)	Centre	Ring 2	Ring 3	Ring 4
0.02	7.61E+14	5.45E+15	1.90E+16	2.96E+16
0.045	3.41E+14	2.59E+15	9.70E+15	1.42E+16
0.105	6.65E+14	5.16E+15	1.64E+16	2.75E+16
0.225	3.07E+14	3.40E+15	1.20E+16	1.83E+16
0.375	1.16E+14	1.84E+15	6.92E+15	1.03E+16
0.575	1.42E+14	2.82E+15	1.08E+16	1.59E+16
0.85	1.59E+14	3.36E+15	1.31E+16	1.88E+16
1.25	8.71E+13	1.82E+15	7.16E+15	9.99E+15
1.75	3.25E+13	6.98E+14	2.73E+15	3.88E+15
2.25	1.61E+13	3.56E+14	1.41E+15	1.94E+15
2.75	7.86E+12	1.72E+14	6.79E+14	9.41E+14
3.5	3.78E+12	8.60E+13	3.47E+14	4.57E+14
4.5	1.62E+12	3.78E+13	1.56E+14	1.95E+14
5.5	3.05E+11	6.91E+12	2.83E+13	3.57E+13
6.5	2.59E+10	4.29E+11	1.69E+12	2.18E+12
7.5	1.21E+09	1.41E+10	5.22E+10	6.82E+10
9	8.56E+07	1.36E+09	5.30E+09	6.94E+09
12	4.33E+03	1.59E+04	3.62E+04	4.81E+04

Fission-Produced Decay β Calculation

Electrons released from *FP* β-decay are assumed to deposit all their energies locally in the fissile material. This heating contribution is derived directly from the output of the WIMS-ORIGEN-S coupled-calculation for the fixed-source calculation.

Table 2 lists the *FP* decay β energy per fission of the three types of burnup fuel used in the model. This energy, $E_{\beta d}$, is proportional to the number of fission events in the fuel because they are induced by fission. The values shown implicitly include the appropriate weighting for fissions in the different fissile species.

Table 2
***FP* Decay β Energy in the ACR Fuel Bundle, MeV/fission**

		Centre	Ring 2	Ring 3	Ring 4
Fresh	Light Element	1.28E-06	4.09E-06	9.38E-06	1.28E-05
	Actinides	2.10E-03	1.19E-02	2.75E-02	5.79E-02
	Fission Products	3.29E-02	4.90E-01	2.13E+00	3.32E+00
	Total	0.035	0.502	2.162	3.374
Mid-Burnup	Light Element	1.58E-06	4.32E-06	9.73E-06	1.32E-05
	Actinides	6.03E-03	2.90E-02	6.88E-02	1.48E-01
	Fission Products	6.74E-02	6.22E-01	2.44E+00	3.58E+00
	Total	0.073	0.651	2.513	3.725

		Centre	Ring 2	Ring 3	Ring 4
Exit-Burnup	Light Element	1.93E-06	5.03E-06	1.09E-05	1.42E-05
	Actinides	7.46E-03	3.64E-02	9.05E-02	1.95E-01
	Fission Products	1.06E-01	7.60E-01	2.42E+00	3.15E+00
	Total	0.113	0.797	2.508	3.349

Output from Monte Carlo Analysis

The results of the *F6* tallies from *kcode* and *sdef* calculation modes can be combined using the number of fission events to normalize the two calculations to derive the nuclear heat distribution in the fuel lattices. In Table 3, each region of the lattice cell has been assigned a variable to facilitate the derivation of the heat balance in the following sections.

Using the assigned variables, nuclear heat per fission from 18-lattice cell *i* calculated by MCNP5 is represented as follows:

$$(E_{np} + E_{\gamma p} + E_{\gamma d} + E_{\beta d}) \quad = \quad \nu \sum_{j,i}^{i=18} \left(t_{ji} + e_{ji} \right)$$

Burnup j will take values from 1 to 3, representing fresh, mid-, and exit-burnup fuel. Variables t and e are defined in Table 3. Variable ν is the number of neutrons per fission from the MCNP calculation.

Table 3
Variables of Different Regions of the Fuel Lattice

	Energy per Fission Neutron at Burnup j	Energy Fraction
Central pin	t_{j1}	f_1
Ring 2	t_{j2}	f_2
Ring 3	t_{j3}	f_3
Ring 4	t_{j4}	f_4
Fuel clad	t_{j5}	f_5
Endplates	t_{j6}	f_6
Coolant	t_{j7}	f_7
Pressure Tube	t_{j8}	f_8
Gas Annulus	e_{j1}	f_9
Calandria Tube	e_{j2}	f_{10}
Moderator	e_{j3}	f_{11}

Correction for an Over-Critical System

Nuclear heat distributions from MCNP5 simulate a closed infinite system. The system does not allow any leakage of particles nor does it allow absorption of particles in reactor structures. Thus, the system multiplication factor, k, over-estimates the true system reactivity of a critical reactor core. Before the nuclear heat distributions from MCNP5 can be used to derive the heat balance, an adjustment is needed to correct for an over-critical system.

To correct the system multiplication factor k, a neutron leakage term l_n needs to be included for the fuelled region of the core. That is, the 520 fuel lattices of the ACR reactor have a system multiplication factor of $(1+l_n)$ before allowance for leakage. In practice the system multiplication factor must be greater than $(1+l_n)$ to account for neutron absorption events that must be allowed in reactivity control devices such as the mechanical zone control units.

The total prompt neutron and the *FP* decay energy deposited in the lattice cell will be proportional to the system multiplication factor k. Consequently, to normalize the prompt neutron and *FP* decay energy deposited to a system multiplication factor $(1+l_n)$, the reduction to the prompt neutron energy and energies due to *FP* decay is

$$\Delta(E_{np} + E_{\gamma d} + E_{\beta d}) = (E_{np} + E_{\gamma d} + E_{\beta d})\frac{\delta}{k}$$

δ is defined as $k - (1 + l_n)$, the excess reactivity that accounts for leakage. E_{np}, $E_{\gamma d}$, and $E_{\beta d}$ depend directly on fission events therefore they can be treated as a group by applying the same correction. Adjustment of the prompt γ terms however is different because not all prompt γ particles originate from fission. To adjust $E_{\gamma p}$, excess reactivity beyond $(1+l_n)$ initially treated as fission events must be treated as capture events, generating prompt-capture γ.

To make this adjustment, $E_{\gamma p}$ is split into two contributions: fission events and capture events. That is,

$$\begin{aligned} E_{\gamma p} &= E_{\gamma p}(fission) + E_{\gamma p}(capture) \\ &= E_{\gamma p}\frac{Q_{fiss}}{Q_{fiss} + (\nu - k)\,Q_{cap}} + E_{\gamma p}\frac{(\nu - k)\,Q_{cap}}{Q_{fiss} + (\nu - k)\,Q_{cap}} \end{aligned}$$

Q_{fiss} is the total prompt energy in the lattice including fission fragments, neutrons, and γ except prompt-capture-γ energy Q_{cap}. Note that Q_{cap} is the energy due to prompt absorption reactions, not solely to capture reactions.

Then, the contribution due to fission is adjusted by the same correction factor as E_{np}, $E_{\gamma d}$, and $E_{\beta d}$. The contribution due to capture is adjusted by assuming excess neutrons are now captured, generating γ particles. The change in $E_{\gamma p}$ is therefore represented as follows:

$$\Delta E_{\gamma p} = E_{\gamma p}\frac{Q_{fiss}}{Q_{fiss} + (\nu - k)\,Q_{cap}}\left(1 - \frac{1 + l_n}{k}\right) + E_{\gamma p}\frac{(\nu - k)\,Q_{cap}}{Q_{fiss} + (\nu - k)\,Q_{cap}}\left(1 - \frac{\nu - (1 + l_n)}{(\nu - 1)\,k}\right)$$

Effective Nuclear Heat of the System

After the correction, the effective nuclear heat per fission E_{fiss} from all regions in the MCNP5 model is as follows:

$$E_{fiss} = (E_{np} + E_{\gamma d} + E_{\beta d})\left(1 - \frac{\delta}{k}\right) + \frac{E_{\gamma p}}{Q_{fiss} + (\nu - k)\,Q_{cap}}\left(\frac{Q_{fiss}\,(k-\delta)}{k} + \frac{(\nu-k)(\nu-k+\delta)\,Q_{cap}}{(\nu-1)\,k}\right)$$

The effective nuclear heat is the average total nuclear heat generated in a lattice of a critical core; the heating includes those fission events required to sustain a critical core with some neutron leakage.

Energy Loss to Leakage

Both neutron and γ particles can escape the reactor core and deposit their energies in regions like the reflector, end shields, calandria shell, etc. In the previous section the over-critical correction to the MCNP5 results was adjusted to $(1+l_n)$ to allow energy loss due to neutron leakage. No adjustment was required for γ leakage since it has no effect on system reactivity; gamma leakage only affects the available gamma energy in the system.

Neutrons leaving the core will lower the fission rates of the fuel, but E_{fiss} was calculated for a value corresponding to a critical system plus leakage. The energy loss to leakage has two components: energy loss to axial (L_{An}) and radial (L_{Rn}) leakage. If the leakage does not affect the overall neutron heat distribution in the lattice, derivation of L_{An} and L_{Rn} can be based on the prompt kinetic neutron energy that leaves the fuelled region of the core, i.e., the lattice cells.

$$(L_{An} + L_{Rn}) \quad = \quad E_{fiss}\,\frac{Q_n}{Q_{fiss}}\,l_n$$

Q_n is the fission-neutron energy. This is a straight reduction from the total energy E_{fiss}. The prompt-neutron nuclear heat in the system is now reduced to the following:

$$\left(1 - \frac{\delta}{k}\right)E_{np} \; - \; E_{fiss}\,\frac{Q_n}{Q_{fiss}}\,l_n$$

Similar to neutron leakage, γ energy reduction is based on the prompt γ energy that leaves the fuelled region of the core.

$$(L_{A\gamma} + L_{R\gamma}) \quad = \quad E_{fiss}\,\frac{Q_\gamma}{Q_{fiss}}\,l_\gamma$$

Q_γ is the fission-gamma energy. If the energy is drawn evenly from all regions, the energy loss is a reduction from the total γ energy due to prompt and delayed γ particles. That is, the γ heating in the system is now the following:

$$\left(1-\frac{\delta}{k}\right)E_{\gamma d} - E_{fiss}\,\frac{Q_\gamma}{Q_{fiss}}\,l_\gamma + \frac{E_{\gamma p}}{Q_{fiss} + (\nu-k)\,Q_{cap}}\left(\frac{Q_{fiss}\,(k-\delta)}{k} + \frac{(\nu-k)(\nu-k+\delta)\,Q_{cap}}{(\nu-1)\,k}\right)$$

Nuclear Heat in Lattice Cell after Leakage

After the correction due to an over-critical system and the subtraction of the energy loss due to leakage, the net nuclear heat available in the lattice E_c, per fission, is represented as follows:

$$E_c = \left(1-\frac{\delta}{k}\right)\left(E_{np} + E_{\gamma d} + E_{\beta d}\right) - \frac{E_{fiss}}{Q_{fiss}}(Q_n\,l_n + Q_\gamma\,l_\gamma) + \frac{E_{\gamma p}}{Q_{fiss} + (\nu-k)\,Q_{cap}}\left(\frac{Q_{fiss}\,(k-\delta)}{k} + \frac{(\nu-k)(\nu-k+\delta)\,Q_{cap}}{(\nu-1)\,k}\right)$$

This is the total nuclear energy per fission in 18 bundles. There are two components to the energy E_c, i.e., F/C and non-F/C components.

To normalize the total nuclear energy per fission to a particular reactor fission power P in MW, the number of fissions per second, N, for P needs to be calculated.

$$N = \frac{P}{E_c\;1.602x10^{-19}}$$

Spatial distribution of the nuclear energy for a particular burnup lattice is obtained by applying the fraction of the total heating of that particular burnup lattice.

The total nuclear energy generated in a core for a fission power of P is therefore equal to $H = N\left\lfloor E_c - (L_{An} + L_{Rn}) - (L_{A\gamma} + L_{R\gamma})\right\rfloor$.

Heat Transfer between Reactor Components

H is the available energy in the reactor core. The distribution of H in the fuel lattice is based on the MCNP5 calculations. The amount of H transferred from one component to another as heat is treated separately.

The following variables will be used in the paper referring to the energies deposited in various components:

E_{Rf} energy deposited in reflector,

E_{Rk} energy deposited in vault water, and

E_{As} energy deposited in steel balls and water of the end shields.

Nuclear heating in the calandria shell E_{Rc} and in the calandria-side tube sheet E_{At} will transfer to the reflector and to the shield cooling system (SCS), respectively. The split between reflector and SCS is based on the presence of the exponential heating source in the calandria shell and the calandria-side tube sheet (CSTS), and the locations of the maximum temperature.

There is a finite amount of energy transferred from the fuel channel (F/C) through the gas annulus (G/A) to the moderator. The transfer process comprises of heat convection Q_{nc}, radiation Q_{rad}, and conduction Q_{con}. The sum of the energy ($Q_{nc} + Q_{rad} + Q_{con}$) will be subtracted from H that is available for the F/C and added to the moderator heat load.

Convective heat transfer occurs in the CO_2 gas gap between the calandria tube (C/T) and the pressure tube (P/T). Thermal radiation heat (radiant heat) transfer also occurs across the gas annulus as a function of the temperature difference between the P/T and the C/T. The G/A of the ACR reactor design has a bigger inner diameter. The garter springs in each channel positioning the P/T in the C/T are therefore larger. As such, heat conduction is an added heat-transfer path from the F/C to the moderator; this contribution is insignificant in the CANDU 6 reactor heat-balance calculation.

There is also heat transfer from the F/C to the end shields across the bearings supporting the end fitting (E/F). The heat transferred is derived from the product of a number of F/C and temperature difference between the coolant and end shield (E/S). This heat is subtracted from the total energy deposited in the fuel channel as lost.

REACTOR HEAT BALANCE

Based on the derivation above, fission-energy distribution in a reactor core can be summarized in Table 4. The heat-transfer balance is listed in Table 5. The parameters L_{HTS}, L_{PIC}, G_{HTS}, G_{Mpump}, L_{Mpipe}, $G_{SCSPump}$, and $L_{SCSPipe}$ in Table 5 are assigned by the process-system engineers. These are external sources not calculated by MCNP5.

Table 4
Equations of Nuclear Heat Deposition

F/C Components	Nuclear Heat	Non-F/C Components	Nuclear Heat
Central pin	$H'f_1$	G/A	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\ f_9$
Ring 2	$H'f_2$	C/T	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\ f_{10}$
Ring 3	$H'f_3$	Moderator	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\ f_{11} + (Q_{nc}+Q_{rad}+Q_{con})$
Ring 4	$H'f_4$	MZCU	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\ f_{12}$
Fuel clad	$H'f_5$		
Endplates	$H'f_6$	Reflector	E_{Rf}
Coolant	$H'f_7$	Shield Tank	E_{Rk}
P/T	$H'f_8$	SCS	E_{As}
		Calandria	E_{Rc}
		CSTS	E_{At}

Table 5
Heat-Transfer Balance Equations

		Loss		Gain
Fuel Channel	End shield	$HLES$		
	Moderator	$Q_{nc}+Q_{rad}+Q_{con}$		
	HTS Piping	L_{HTS}[*]		
	Pressure & Inventory Control System	L_{PIC}[*]		
			HTS pumps	G_{HTS}[*]
Moderator			Nuclear heat	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\, f_{11} + (Q_{nc}+Q_{rad}+Q_{con})$
			Reflector	E_{Rf}
			C/T	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\, f_{10}$
			MZCU	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\, f_{12}$
			G/A	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\, f_{9}$
			Calandria transfer	$E_{Rc}\, F_c^+$
			CSTS transfer	$E_{At}\, F_s^+$
			F/C transfer	$Q_{nc}+Q_{rad}+Q_{con}$
			Pump	G_{Mpump}[*]
	Moderator-Piping	L_{MPipe}[*]		
SCS			CSTS	$E_{At}\,(1-F_s^+)$
			End Shield	E_{As}
			F/C	$HLES$
			Shield tank	E_{Rk}
			Calandria	$E_{Rc}\,(1-F_c^+)$
			Pump	$G_{SCSPump}$[*]
	SCS Piping	$L_{SCSPipe}$[*]		

The heat loads at different reactor components are summarized in Table 6. In the table, $G_{capture}$ can be obtained from a full-core MCNP5 calculation that includes the primary shields. The thermal-to-fission power ratio of the reactor is defined as the ratio of the net heat load to the coolant to the total fission heat ($\frac{H_{cool}}{P'}$) as indicated in Table 6.

[*] This is an assigned value from other process-system engineers.

Table 6
Summary of Heat-Load Equations

Nuclear energy deposited in F/C	H'
Nuclear energy deposited outside F/C	$(L_{An} + L_{Rn}) + (L_{A\gamma} + L_{R\gamma}) + [H' - (Q_{nc}+Q_{rad}+Q_{con})]\ (f_9 + f_{10} + f_{11} + f_{12})$
Capture energy due to leakage	$G_{capture}$ *
Available reactor nuclear power	Sum of the above, P'
Energy from F/C to coolant	H'
Loss to Moderator & shield	$-[\ HLES + (Q_{nc}+Q_{rad}+Q_{con})]$
Net fission energy to coolant	Sum of the above, $H_{coolant}$
Loss to HTS Piping & P&IC	$-[L_{HTS} + L_{PIC}]$
Net fission heat to S/G & pre-heaters	$H_{coolant} - (L_{HTS} + L_{PIC})$
Energy gain from pumps	G_{HTS}
Total heat load to S/G & pre-heaters	$H_{coolant} - (L_{HTS} + L_{PIC}) + G_{HTS}$
Net fission heat to moderator	$[H' - (Q_{nc}+Q_{rad}+Q_{con})]\ (f_9 + f_{10} + f_{11} + f_{12}) + E_{Rc}\, F_c^{+} + E_{At}\, F_s^{+} + (Q_{nc}+Q_{rad}+Q_{con})$
Net fission heat to SCS	$HLES + E_{At}\ (\ 1\text{-}F_s^{+}\ ) + E_{As} + E_{Rk} + E_{Rc}\ (\ 1\text{-}F_c^{+}\ )$
Fission heat to HTS & P&IC	$L_{HTS} + L_{PIC}$
Total fission heat	Sum of the above, P'

CASE STUDY

The methodology was applied to the ACR reactor normalized to a thermal reactor power of 3177 MW. Based on the reactor specifics, the corresponding nuclear heat rates in the different reactor components are tabulated in Table 7.

Table 8 shows the nuclear heat balance based on heat loads derived in Table 7 and Table 9 shows the heat transfer balance of the reactor. The thermal efficiency of the ACR reactor, that is the ratio of the *Net Coolant Heat Load* to the *Available Nuclear Power*, was calculated to be 95.6%.

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* Derived from external calculation.

Table 7
Nuclear Heat Deposition in the ACR Reactor Core

F/C Components	MW	Non-F/C Components	MW
Central Element	48.8	G/A	0.02
Ring-2 Elements	302.5	C/T	14.7
Ring-3 Elements	1111.6	Moderator	101.2
Ring-4 Elements	1642.8	MZCU*	6.1
Fuel Cladding	13.4	Total	122.02
Fuel Gap	0		
Pressure Tube	30.5	Reactor Components	MW
Coolant	25.8	Reflector	10.78
Endplate Regions	1.60	Calandria Shell	1.86
Total	3177	Vault Water	0.95
		CSTS	1.25
		SCS	0.54
		Total	15.38

Table 8
Heat Load Summary of the ACR Reactor

	MW
Nuclear Energy in F/C	3177.0
Nuclear Energy in non-F/C (including leakage)	126.6
Capture γ Energy due to Neutron Leakage	10.8
Available Nuclear Power	**3314.4**
Energy Transfer from F/C to Coolant	3177.0
Loss to Moderator & Shield	-9.1
Net Coolant Heat Load	**3167.9**
Loss to Piping & P&IC	-17.5
Net Nuclear Power to S/G & Pre-Heaters	3150.4
Energy gain from pumps	33.2
Total Heat Load to S/G & Pre-Heaters	**3183.6**
Net Nuclear Power to S/G & Pre-Heaters	3150.4
Net Nuclear Heat in Moderator	139.4
Net Nuclear Heat in SCS	7.1
Nuclear Energy to P&IC System	17.5
Total Nuclear Heat	**3314.4**

* Not calculated, assumed 5% of non-F/C heat rate.

Table 9
Heat Transfer Balance in MW for the ACR Reactor

		Loss, MW	Gain, MW
Fuel Channel	Loss to End shield	4.37	
	Loss to Moderator	4.75	
	Sub-Total Loss	9.12	
	Loss to HTS Piping	3.50	
	Loss to P&IC	14.00	
	Sub-Total Loss	17.50	
	Gain from HTS pumps		33.20
Moderator	Nuclear Energy in Moderator		101.17
	Nuclear Energy in Reflector		10.78
	Nuclear Energy in C/T		14.73
	Nuclear Energy in MZCU		6.10
	Nuclear Energy in G/A		0.02
	Energy Transfer from Calandria Shell		1.03
	Energy Transfer from CSTS		0.80
	Energy Transfer from F/C		4.75
	Net Nuclear Heat in Moderator		139.38
	Gain from Pump		1.27
	Loss to Piping	0.54	
	Moderator Heat Load		**140.11**
Shield Cooling System	Energy Transfer from CSTS		0.45
	Energy Transfer from End Shield		0.54
	Energy Transfer from F/C		4.37
	Energy Transfer from Shield Tank		0.95
	Energy Transfer from Calandria Shell		0.83
	Net Nuclear Heat in SCS		7.14
	Gain from Pump		0.15
	Loss to Piping	0.05	
	SCS Heat Load		**7.24**

Note that the nuclear heating, heat load, and heat balance of the ACR reactor as tabulated in Table 7, Table 8, and Table 9 were based on parameters that do not represent the current ACR-1000 design. The main differences are the fuel-lattice pitch, the fuel enrichment, the fuel exit-burnup, reactor thermal power, and the central-pin compositions.

CONCLUSION

The methodology used in deriving the nuclear heat balance has been established based on the Monte Carlo transport code MCNP5. The methodology was used successfully to calculate the heat balance of the ACR reactor. Some key improvements of using this methodology to calculate heat balance are

- the ability to use one code to derive parameters like leakage and structural heating of the reactor,
- the ability to simulate closer to an operating reactor core by including fuel bundles of different burnups, and
- the correction applied to the results generated by an over-critical lattice.

Based on this methodology, the nuclear energy deposited initially in the outer ring of fuel in an ACR lattice is shown to drop gradually with irradiation, see Table 10. The greatest change in nuclear energy deposited in a lattice is in the central pin where the proportion of bundle energy doubles. The thermal to fission power ratio for the ACR reactor is calculated to be 0.956 in this paper.

Table 10
Relative Heat Distributions of Different Burnup Bundles in the Fuel Lattice

	Fresh	10057 MWd/MgU**	21822 MWd/MgU
Central Elements	1.00%	1.37%	2.08%
Ring-2 Elements	7.91%	8.85%	10.81%
Ring-3 Elements	33.47%	33.88%	33.96%
Ring-4 Elements	52.84%	50.33%	46.53%
Pressure Tube	0.77%	0.91%	1.10%
Calandria Tube	0.38%	0.45%	0.56%
Coolant	0.72%	0.78%	0.85%
Moderator	2.54%	2.98%	3.57%
Endplate Regions	0.04%	0.05%	0.05%
Fuel Cladding	0.34%	0.40%	0.48%
Fuel Gap	0.00%	0.00%	0.00%
Gas Annulus	0.00%	0.00%	0.00%
Total	100%	100%	100%

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** Averaged over 7 mid-burnup bundles.

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