13th International Conference on CANDU Fuel
Holiday-Inn Waterfront Hotel
Kingston, Ontario, Canada, 2016 August 15-18

APPLICATION OF THE WANO FUEL RELIABILITY INDICATOR FOR PHWRS

B.J. Lewis,[1] C. Banica[2] and P. Chan[3]
[1]Lewis Consultants, 1090 Dunham St., Kingston, Ontario, Canada K7P 2K1
(613 893-2166, lewibre@gmail.com)
[2]Ontario Power Generation - Darlington Nuclear, Reactor Physics, 1 Holt Road South, Bowmanville, ON L1C 3Z8
[3]Royal Military College of Canada, Department of Chemistry and Chemical Engineering, PO Box 17000, Kingston, ON K7K 7B4

ABSTRACT – The methodology used by WANO to develop fuel reliability indicators for defective fuel has been examined in terms of fundamental fission product release theory with a scaling to reference values of the linear heat rating and coolant purification flow for CANDU operation. The new fuel reliably indicator has been applied to several known defect occurrences in CANDU reactors where the number of failures are known.

1. Introduction

A fuel reliability indicator (*FRI*) has been proposed by the World Association of Nuclear Operators (WANO) [1,2] to monitor industry progress in achieving and maintaining high fuel integrity. With defective fuel, there is a release of fission products into the primary heat transport system (PHTS) as well as possible fuel debris. The element performance is also affected with a reduced fuel thermal conductivity and enhanced fission-product diffusivity. Restrictions exist on the amount of I-131 in the PHTS that could potentially force the unit to shut down. Hence, the presence of failed fuel can have a detrimental effect on the operating cost and performance of the reactor, with an increased radiological hazard to plant workers. Consequently, WANO has defined an indicator to assess fuel integrity based on a steady-state measurement of the coolant activity of I-131. This indicator is corrected for the tramp uranium contribution, and normalized to a common purification rate and power level.

This paper examines the underlying theory for WANO's performance indicator for fuel reliability and compares this methodology to fission product release theory for CANDU defective fuel. A discussion of the applicability of the WANO methodology to CANDU fuel operation and integrity assessment is presented including an example of its use with some historical cases from CANDU reactors with known failures.

2. Fission Product Release Theory for Defective CANDU Fuel

Above about 1000°C, fission products migrate primarily by thermally-induced diffusion in the bulk UO_2: [3-10].

$$\frac{R_{dif}}{B} = 3\sqrt{\frac{D'}{\lambda}} \qquad \textbf{(1)}$$

Here R is the release rate and B the birth rate where $B = FY$ [F is the fission rate (fissions s^{-1}) and Y is the cumulative fission product yield (atom $fission^{-1}$)], D' is the empirical diffusion

coefficient in the UO_2 fuel [= D/a^2 where D is the diffusion coefficient and a is the grain radius] (s^{-1}) and λ is the radioactive decay constant (s^{-1}). Below 1000°C, the fission product release is generally independent of temperature but is activated as a result of the fission event. In particular, fission gas can be released by athermal diffusion in accordance with Eq. (1). In addition, gas release can also result at the external surface of the pellet by direct fission recoil as well as by knockout when either a primary fragment or energetic particle created in a collision cascade interacts elastically with a fission product atom. As previously shown, releases predicted for the recoil and knockout mechanisms were considerably less than measured releases of noble gas for a defective fuel rod with secondary hydriding irradiated at a low power rating of 23 kW/m in an X-2 loop experiment, particularly after taking into consideration any holdup in the gap [3]. This observation arises since fuel oxidation can affect the vacancy-enhanced diffusion rate of fission products in the fuel matrix of defective elements [3,11].

Release of fission products from the gap of the defective rod into the coolant can occur by a first-order rate process, where the coolant release rate (R_c) is proportional to the available inventory in the gap, N_g:

$$R_c = \nu N_g \tag{2}$$

and ν is the gap escape-rate constant (s^{-1}). The subsequent mass balance in the gap at steady state is:

$$\frac{dN_g}{dt} = R_{dif} - \lambda N_g - \nu N_g = 0 \tag{3}$$

Thus, with diffusion as the dominant release mechanism from the fuel, the observed release fraction into the coolant is:

$$\left(\frac{R_c}{B}\right) = \left\{\frac{\nu}{\lambda+\nu}\right\} 3\sqrt{\frac{D'}{\lambda}} \tag{4}$$

The term in curly brackets describes the holdup of fission products in the gap during their transport to the defect opening. Equation (4) yields a release dependence on the decay constant of $\lambda^{-1/2}$ to $\lambda^{-3/2}$ in agreement with observations in defective fuel experiments [3].

Traces of uranium compounds may be found on the surfaces of the heat-transport system primarily from a previous loss from defective fuel elements. The debris is in the form of very fine particles, where fuel is lost after individual grains of fuel are loosened by oxidation along their boundaries. For small particles of fuel at low temperature, fission product release can occur by direct recoil and knockout [12]. However, since the range of the fission fragment ($\mu_f \sim 10$ μm) is comparable to the size of the UO_2 fuel particle itself, each fragment will leave the individual particle of the fissioning U-atom. For instance, experimental ratios of release rates of ^{88}Kr to ^{138}Xe were between 0.4 and 0.7 in a pressurized-water loop containing fuel debris on the piping surface [3]. The release expression for a small particle of fuel of diameter d, deposited onto a solid surface is [12]:

$$\left(\frac{R}{B}\right)_{tramp} = \frac{1}{2}\left\{\frac{1}{\alpha^3} + \frac{3\mu_f}{2d}\left(1 - \frac{1}{\alpha^2}\right)\right\} \tag{5}$$

where $\alpha = 1$ or d/μ_f, whichever is greater. Since B is proportional to the fission yield Y, and the right hand side of Eq. (5) is a constant, a release ratio of $R_{88}/R_{138} = Y_{88}/Y_{138} = 0.55$ is calculated in excellent agreement with the measured range of 0.4-0.7. In this calculation,

the cumulative yield has been used to account for the decay of the precursor products also emitted into the recirculating coolant. Since $\mu_f > d$, $\alpha = 1$ and Eq. (5) reduces to:

$$\left(\frac{R}{B}\right)_{tramp} = \frac{1}{2} \quad \textbf{(6)}$$

This result is to be expected if the range of the fission fragment is greater than the particle size so that any fission fragment born in the fuel grain particle will be released into the coolant (i.e., *R/B* = 1). On the other hand, if the particle is deposited onto a pipe surface, approximately one half of the particles will embed themselves into the underlying surface so that *R/B* = ½ in accordance with Eq. (6).

3. Application of WANO Methodology to Candu Fuel

The WANO methodology converts a measured coolant activity *A* (Bq) into a release rate *R* (atom s^{-1}) for defective fuel. As shown in Figure 1, one can distinguish releases from defective fuel [Eq. (4)] and tramp fuel contamination [Eq. (6)] by examining the release rate *R* (atom s^{-1}) when normalized by the isotopic yield *Y* (atom $fission^{-1}$), where a composite curve results:

$$\left(\frac{R}{Y}\right) = a\lambda^b + c \quad \textbf{(7)}$$

Here *b* ranges from -1/2 to – 3/2 from the fission product release theory in Section 2. The amount of fuel debris in the core can be estimated directly from the fitted parameter $c = ½ F_t$ (~ 1.1 x 10^{12} fissions/s) [3]. The normalized release rate from defective fuel depends on the radioactive decay constant λ, with a typical exponential dependence ranging from $\lambda^{-1/2}$ for a large defect (indicating only diffusion in the fuel) to $\lambda^{-3/2}$ for a small defect (due to diffusion in the fuel with significant holdup in the fuel-to-sheath gap). On a log-log plot, this dependence is simply the slope of the straight line shown in Figure 1. For tramp uranium fuel, as discussed in Section 2, the fuel temperature is too low for diffusion to be important so that only fission product recoil occurs (i.e., the flat line shown in Figure 1). The isotope I-131 has a long half-life and is therefore the predominant isotope with a large diffusional release. On the other hand, the short-lived I-134 isotope will decay before it is released from a defective fuel element so that its contribution is mainly due to a recoil release from tramp debris. Consequently, these two isotopes can be used to distinguish the defective fuel vs. tramp fuel contributions. WANO estimates the defective fuel contribution by initially correcting for tramp uranium effects with an activity corrected for coolant cleanup.

The mass balance for the fission product inventory N_i (atom) released into the coolant for isotope *i* is [8-10]:

$$\frac{dN_i}{dt} = R_i - (\lambda_i + B_a)N_i \quad \textbf{(8)}$$

Here B_a is the actual purification rate constant (s^{-1}) and R_i is the release rate contribution from both defective and tramp fuel:

$$R_i = \underbrace{\underbrace{\left(\frac{\nu}{\lambda+\nu}\right)}_{\text{gap transport}} \underbrace{3\sqrt{\frac{D'}{\lambda}}F_f Y}_{\text{transport in fuel matrix}}}_{\text{defective fuel contribution}} + \underbrace{\tfrac{1}{2}F_t Y}_{\text{tramp fuel contribution}} \quad \textbf{(9)}$$

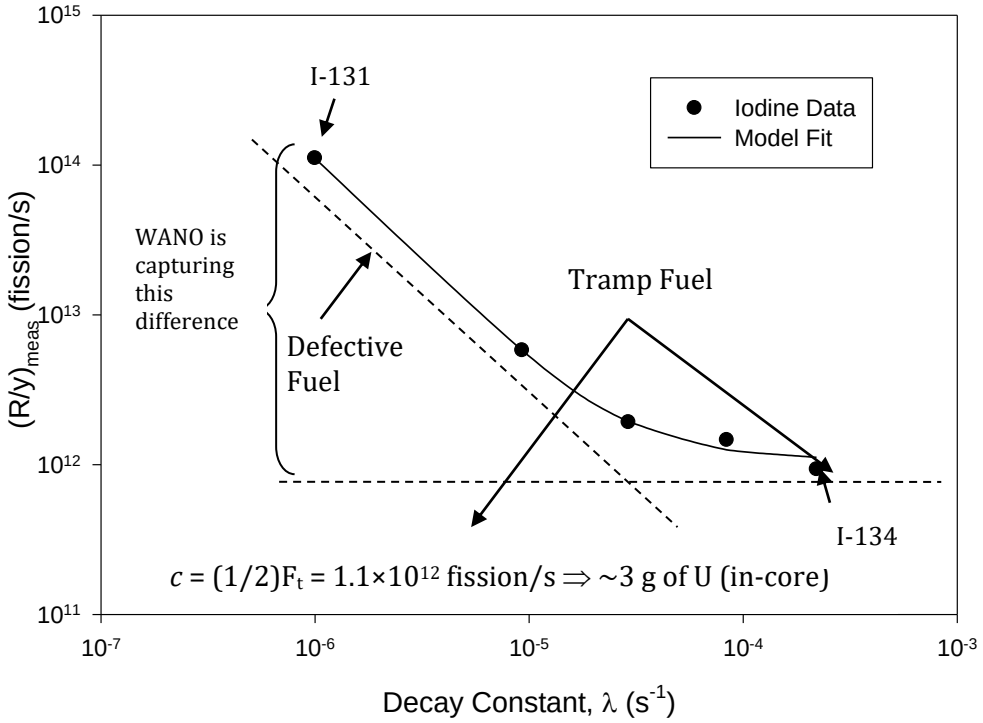

Figure 1. Schematic showing fission product release from defective fuel (sloped line) and recoil release from tramp fuel (flat line independent of the decay constant) with the model fit to data derived from measured coolant activities.

F_f is the fission rate for the defective element (fission s^{-1}) and F_t is the fission rate in the tramp fuel (fission s^{-1}). Equation (9) gives rise to the observed release rate behaviour in Figure 1 when normalized by the fission product yield *Y*. For high mass chain products, the yield *Y* will not change considerably with fuel burnup because of similar yields for the fissile species of U-235 and Pu-239. Hence, no correction is needed for either the iodine or xenon isotopes with fuel burnup, where the U-235 yield can be directly used with minimal error. This insensitivity is a useful property because one does not know *a priori* the respective burnup of the defective fuel rods versus how long fuel debris has resided on in-core surfaces from past defect occurrences.

3.1 FRI Derivation

The *R/Y* value for I-131 is corrected by subtracting the tramp uranium contribution (as estimated by the *R/Y* value for I-134). At steady-state, Equation (8) becomes:

$$\frac{dN_i}{dt} = R_i - (\lambda_i + B_a)N_i = 0 \Rightarrow A_i \equiv (\lambda N)_i = \left(\frac{\lambda_i}{\lambda_i + B_a}\right)R_i \qquad \textbf{(10)}$$

The release rate R_i (in atom s^{-1}) is simply proportional to the coolant activity A_i (in Bq). One can normalize the measured coolant activity by the coolant mass which will shift the I-131 and I-134 data points by the same amount but respect their difference in Fig. 1. Thus, the *FRI* as used by WANO follows from the basic concept:

$$\left(R/Y\right)_{def} = \left(R/Y\right)_{I-131} - \left(R/Y\right)_{I-134} \quad \textbf{(11)}$$

Solving for R in Eq. (10), substituting this result into Eq. (11), and normalizing by the mass of the reactor coolant system (m_c in grams) yields:

$$\left(\frac{R/Y_{def}}{m_c}\right) = \mathrm{A}_{M,I-131}\left(\frac{\lambda + B_a}{Y\lambda}\right)_{I-131} - \mathrm{A}_{M,I-134}\left(\frac{\lambda + B_a}{Y\lambda}\right)_{I-134} \quad \textbf{(12)}$$

where A_M is the measured coolant activity concentration in Bq/g (or μCi/g). To normalize to a reference rate constant (B_n), Eq. (12) can be rewritten as:

$$\left(\frac{R/Y_{def}}{m_c}\right) = \left\{\mathrm{A}_{M,I-131}\left(\frac{\lambda_{131} + B_a}{\lambda_{131} + B_n}\right)\right\}\left(\frac{\lambda_{131} + B_n}{Y_{131}\cdot\lambda_{131}}\right) - \left\{\mathrm{A}_{M,I-134}\left(\frac{\lambda_{134} + B_a}{\lambda_{134} + B_n}\right)\right\}\left(\frac{\lambda_{134} + B_n}{Y_{134}\cdot\lambda_{131}}\right) \quad \textbf{(13)}$$

The term is curly brackets is the respective coolant activity normalized to B_n (s^{-1}) where:

$$\left(\frac{R/Y_{def}}{m_c}\right) = \mathrm{A}_{N,I-131}\left(\frac{\lambda_{I-131} + B_n}{Y_{I-131}\cdot\lambda_{I-131}}\right) - \mathrm{A}_{N,I-134}\left(\frac{\lambda_{I-134} + B_n}{Y_{I-134}\cdot\lambda_{I-131}}\right) \quad \textbf{(14)}$$

and (with i = I-131 and I-134):

$$A_{N,i} = \left\{\mathrm{A}_{M,i}\left(\frac{\lambda_i + B_a}{\lambda_i + B_n}\right)\right\}. \quad \textbf{(15)}$$

WANO specifically normalizes to a "reference" purification rate constant of $B_n = 2 \times 10^{-5}$ s^{-1}. *However, for CANDU operation a typical value of $B_n \sim 5 \times 10^{-5}$ s^{-1} is chosen.* Finally, multiplying Eq. (14) through by the factor $(Y_{\text{I-131}}\lambda_{\text{I-131}})/(\lambda_{\text{I-131}} + B_n)$ yields:

$$A_{D,C}(Bq/g) \equiv \left(\frac{Y_{I-131}\cdot\lambda_{I-131}}{\lambda_{I-131} + B_n}\right)\left(\frac{R/Y_{def}}{m_c}\right) = \mathrm{A}_{N,I-131} - \mathrm{A}_{N,I-134}\left(\frac{Y_{I-131}\cdot\lambda_{I-131}}{\lambda_{I-131} + B_n}\right)\left(\frac{\lambda_{134} + B_n}{Y_{134}\cdot\lambda_{131}}\right) \quad \textbf{(16)}$$

or

$$A_{D,C}(Bq/g) = \mathrm{A}_{N,I-131} - \mathrm{k}_C\mathrm{A}_{N,I-134} \quad \textbf{(17)}$$

$$k_C = \frac{\left[Y\lambda/\left(\lambda + B_n\right)\right]_{I-131}}{\left[Y\lambda/\left(\lambda + B_n\right)\right]_{I-134}} \quad \textbf{(18)}$$

Equations (17) and (18) are identical to the WANO methodology except for a slight discrepancy for their proposed k factor:

$$k_C = k\left(\frac{\lambda_{I-131}}{\lambda_{I-134}}\right)^{0.045} \quad \textbf{(19)}$$

This slight difference occurs where WANO assumed a small dependence of the decay constant for the tramp uranium contribution based on empirical observation. As detailed in Section 2, this assumption is not consistent with recoil theory, where use of a flat line for the tramp contribution allows for a good fit of the model to the data in Fig. 1. Typical *R/Y* plots do not reveal this type of dependence; for example, off-gas data from the HATCH-1 plant also show a constant recoil release for the short-lived noble gases [13]. However, the gamma-ray measured for I-134 (i.e., at 847.0 keV) cannot be distinguished from that of the corrosion product Mn-56 (i.e., at 846.8 keV) [14], which would augment the measured activity. In addition, as seen in Fig. 1, there is also a very small diffusional release for the short-lived isotopes in addition to that of recoil. Nevertheless, the ratio of $(\lambda_{\text{I-131}}/\lambda_{\text{I-134}})^{0.045}$ is very close to unity where $k_C/k = 0.784$. Hence, this

difference makes little impact on the calculated *FRI* value especially given the measurement uncertainty in the coolant activity data.

3.1.1 Power Correction and Normalization

Reactor power can impact the I-131 activity concentration, where WANO normalizes A_D to a reference heat generation rate of 18 kW/m assuming a dependence on the linear heat generation rate of $LGHR^{1.5}$. This correction is appropriate for athermal diffusion for the lower-powered PWR fuel. For instance, as shown in Ref. [15], the diffusion coefficient D for fission gases for oxidized fuel consists of three components over different temperature regimes: (i) intrinsic diffusion at high-temperature (which is an Arrhenius function of temperature T), (ii) irradiation-enhanced vacancy production at intermediate temperatures (where the uranium vacancy concentration is also a function of the deviation from stoichiometry x in UO_{2+x}), and (iii) irradiation-enhanced (athermal) diffusion at low temperature (which depends on the fission rate density F_v):

$$D(x,T) = D(T) + D(x,T) + a \cdot F_v \quad \textbf{(20)}$$

Specifically, the fission product diffusion coefficient D ($m^2\ s^{-1}$) at a temperature T (in K) and fission rate density F_v (in fission $m^{-3}\ s^{-1}$) for hyperstoichiometric UO_{2+x} fuel is [9]:

$$D(x,T) = 7.6x10^{-10}\exp\left\{\frac{-35230}{T}\right\} + \left[\sqrt{2F_v}\,x10^{-25}\exp\left\{\frac{-13890}{T}\right\}\right] + x^2\,2.22x10^{-8}\exp\left\{\frac{-20230}{T}\right\} + 2x10^{-40}F_v \quad \textbf{(21)}$$

LWR's operate at much lower linear ratings and to higher burnups as compared to CANDU fuel. A power dependence of $LHGR^{1.5}$ for the release rate strictly implies athermal diffusion for the lower-powered fuel as follows from Eqs. (4) and (20) for Booth diffusion given the square-root dependence on D' and the fact that the birth rate B is also proportional to the linear heat rating [16]. On the other hand, defective fuel in CANDU reactors operate at much higher ratings where diffusion occurs as an Arrhenius function of temperature and on the stoichiometry deviation (see Eq. (20)).

Although WANO used a linear power normalization of L_n = 18 kW/m, a more appropriate reference value for CANDU fuel is: L_n = 38 kW/m as shown in Table 1 based on median bundle powers/element power ratings for various reactors (i.e., assuming 28 elements/bundle for Pickering and 37 elements/bundle for the others with a fuel stack length of 0.48 m) [17]. An extrapolation from the low linear power in the PWR to typical CANDU values would result in a significant source of error.

As shown from sweep gas experiments and the multi-slit defective fuel loop test, the empirical diffusion coefficients for iodine and noble gas are equal [3,4,6,18,19] with an $R/B \propto \lambda^{-1/2}$. This work is also consistent with that observed by Appelhans and Turnbull [7] with swept fuel assemblies irradiated in the Halden reactor. Hence, to evaluate the power dependence of the empirical diffusion coefficient D' in Eq. (9), one can use the steady-state release rate of ^{133}Xe from commercial reactor experience. Due to the relatively long half-life ($t_{1/2}$ = 5 days), and chemically inert nature, ^{133}Xe is the least affected by defect size. This approximation is illustrated in Figure 2(a) which shows ^{133}Xe release rates versus linear power *LHGR* (in kW m^{-1}) for defects of various sizes and burnups for different reactors [20], with fitted coefficients a =

9.86 ± 0.06 and b = 0.057 ± 0.003. For a steady-state diffusional release, the release-to-birth ratio (R/B) of ^{133}Xe is obtained from Eq. (4), where $\lambda \ll \nu$ for the long-lived ^{133}Xe isotope so that

Table 1: Median Bundle Powers and Burnup and Average Element Rating for CANDU Reactors

Reactor	Start of Decade	Median Burnup (MWh/kgU)	Median Bundle Power Rating (kW)	Average Element Rating (kW/m) (*LHGR*)
Bruce A	1970	133	566	
	1980	203	703	
	1990	192	604	
	2000	206	678	
	2010	218	719	
	All	195	658	**37.0**
Bruce B	1980	179	670	
	1990	190	633	
	2000	190	633	
	2010	189	652	
	All	188	638	**35.9**
Pickering A	1980	198	499	
	1990	204	556	
	2000	198	554	
	2010	210	564	
	All	202	547	**40.7**
Pickering B	1980	188	550	
	1990	193	554	
	2000	190	547	
	2010	195	548	
	All	191	550	**40.9**
Darlington	1990	195	674	
	2000	204	700	
	2010	204	709	
	All	201	690	**38.9**
Gentilly-2		174	653	**36.8**
Point Lepreau		170	653	**36.8**
Aggregated		192	624	**38.1 (= L_n)**

$$\frac{R}{B} = \frac{10^{a+b\cdot LHGR}}{f_f Y \cdot LHGR} = 3\sqrt{\frac{D'}{\lambda}} \qquad \textbf{(22)}$$

Here f_f = 1.489×10^{13} [fission.m.kW^{-1}] (i.e., $F = f_f \cdot LHGR$) , Y = 0.066991 [atom.fission^{-1}] is the ^{133}Xe cumulative yield, and λ = 1.53×10^{-6} [s^{-1}] is the ^{133}Xe decay constant. Using Eq. (22), the empirical diffusion coefficient D'(in s^{-1}) is given by:

$$D' = 8.6\times10^{-12}\left[\frac{10^{(0.114\cdot LHGR)}}{LHGR^2}\right] \qquad \textbf{(23)}$$

Eq. (23**)** is based on CANDU operational data in the power range of 30 to 50 kW.m^{-1}. In comparison, for PWR rods that operate typically below ~22 kW/m, D' is directly proportional to the linear power rating *LHGR* [1,2,16]. For CANDU fuel, L_n = 38 kW/m, where a scaling can be performed for a given plant for the given *LHGR* value listed in Table 1. For higher-powered

fuel (i.e., > 30 kW/m), as shown in Fig, 2(a), the release rate is proportional to $10^{0.057 \cdot LGHR}$ for the power scaling. The data in Fig. 2(a) are for the CANDU 37-element fuel-bundle design used in the Darlington, Bruce and CANDU-6 reactors. Data are further shown for the Douglas Point reactor that employs a 19-element bundle as well as the Pickering reactor with a 28-element

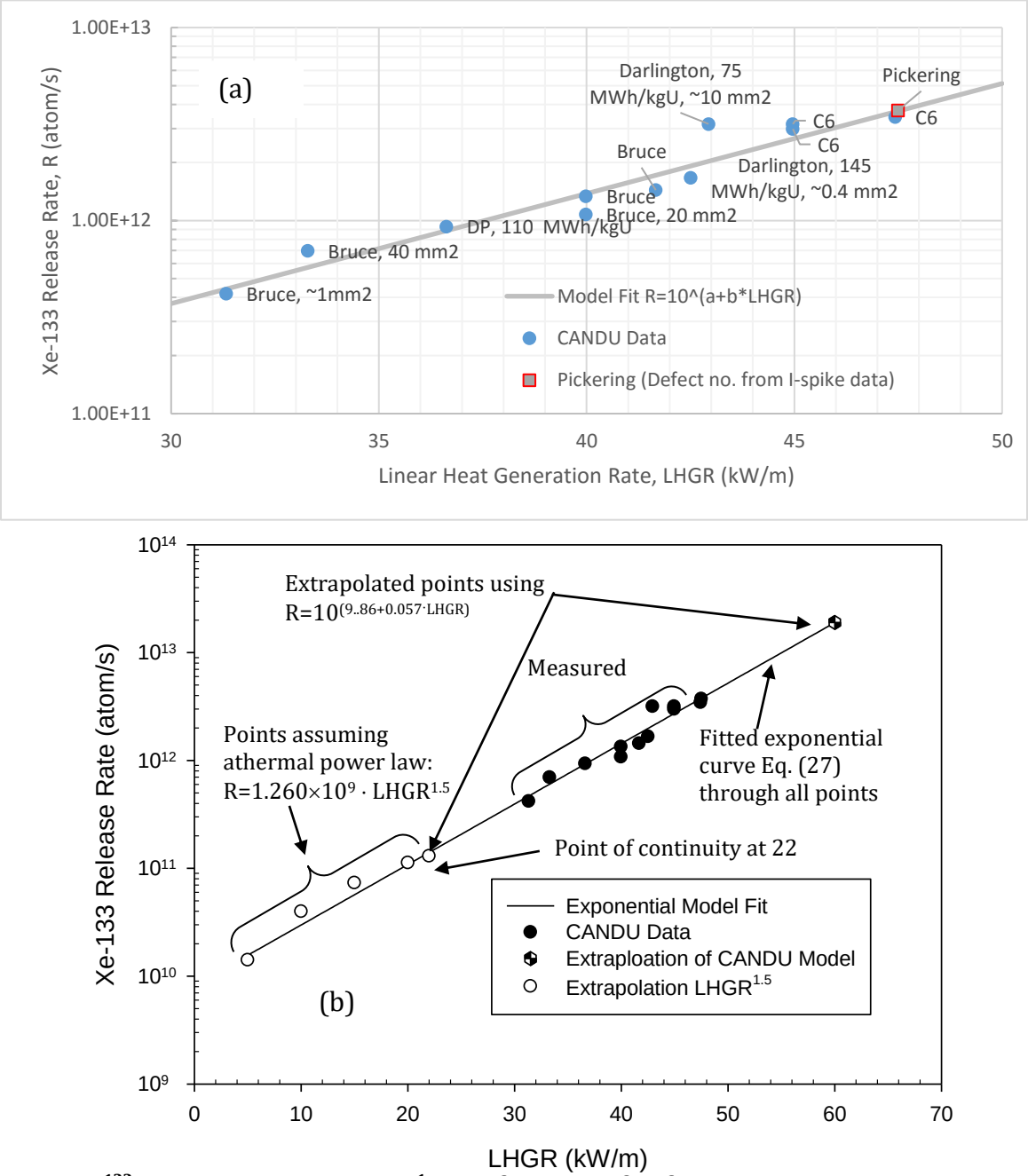

Figure 2. (a) ^{133}Xe release rate [atom·s^{-1}] as a function of defective element linear power [kW·m^{-1}]. Defect size and element burnup shown if known (C6 = CANDU-6, DP = Douglas Point) (Taken from Ref. [20].) (b) Exponential scaling model for extrapolated reactor power analysis.

bundle. The Pickering data point is specifically derived from the Xe-133 steady-state coolant activity as measured as a result of power-ramp failures during a defect excursion from November 1971 to February 1972.

Thus, since the WANO power scaling is not appropriate for CANDU fuel, it would affect the WANO scoring. Hence, using a power dependence of $R \propto 10^{b \cdot P}$ as a general function of the rod power P (kW/m) gives:

$$FRI_C = A_{D,C} \times 10^{0.057 \times \left[L_n - LHGR\left(\frac{P_o}{100}\right)\right]} = \left[\left(A_{131}\right)_N - k_C \left[\left(A_{134}\right)_N \right] \right] \times 10^{0.057 \times \left[L_n - LHGR\left(\frac{P_o}{100}\right)\right]} \quad \textbf{(24)}$$

$$A_N = A_M \left[\frac{\lambda + B_a}{\lambda + B_n} \right] \quad \textbf{(25)}$$

$$k_C = \frac{\left[Y\lambda / \left(\lambda + B_n\right) \right]_{I-131}}{\left[Y\lambda / \left(\lambda + B_n\right) \right]_{I-134}} = 0.00900 \quad \textbf{(26)}$$

with reference values: L_n = 38 kW/m and $B_n = 5 \times 10^{-5}$ s^{-1}. The FRI_C is reported in units of Bq/g or μCi/g when the measured coolant activity A_M is expressed in these respective units (where the conversion factor is: 1 μCi/g = 3.7×10^4 Bq/g). Values for the linear heat generation rate (*LHGR*) at 100% power can be obtained from Table 1 for the respective CANDU station. The values in Table 1 are scaled accordingly depending on the percent of reactor power P_o. *One must be careful on extrapolation to low linear powers with Eq. (24) given that data are only available down to about 30 kW/m in Figure 2 (a).*

The empirical diffusion coefficient in Eq. (23) becomes unphysical at low linear power rating. The power-law correlation $R = 10^{9.86+0.057 \cdot LHGR}$ as originally proposed in Ref. [20] (see Fig. 2(b)) can be reasonably extrapolated to a higher rating of 60 kW m^{-1} as well as to a lower limit of 22 kW m^{-1} (i.e., typical of the upper rating for a PWR rod) for reduced reactor power analysis (20). The lower rating is demonstrated where linear powers predicted using Eq. (22), based on observed ^{133}Xe release rates for an element power-cycled between 23 to 39 kW/m in the X-2 loop experiment FFO-109-2, were in excellent agreement with the actual power ratings. Thus, the PWR scaling law of $R = a_1 \cdot LHGR^{1.5}$ (in accordance with athermal diffusion in Eq. (20)) can be applied with continuity at 22 kW/m in Fig. 2(b), yielding a fitted coefficient of $a_1 = 1.260 \times 10^9$ atom s^{-1} (kW·m^{-1})$^{-1.5}$. Hence, additional points can be derived for linear ratings below 22 kW/m. Moreover, as follows from Eq. (21), an *exponential* fitting of all data over the full range of linear element powers from ~5 to 60 kW/m in Fig. 2(b) yields a more physically-based scaling law:

$$R = a_2 \cdot e^{b_2 \cdot LHGR} \quad \textbf{(27)}$$

Here, the fitted coefficients based on a Marquardt-Levenberg non-linear regression analysis are: $a_2 = 8.195 \times 10^9 \pm 4.82 \times 10^8$ and $b_2 = 0.1291 \pm 0.0041$. Equivalently, Eq. (27) can be used to replace Eq. (24) such that:

$$FRI_C = \left[\left(A_{131}\right)_N - k_C \left[\left(A_{134}\right)_N \right] \right] \times e^{0.1291 \times \left[L_n - LHGR\left(\frac{P_o}{100}\right)\right]} \quad \textbf{(28)}$$

As expected, the use of Eq. (28) makes little difference (typically less than ~1%) from the calculated values with Eq. (24) over the power range of the measured data in Fig. 2(b).

4. *FRI* Example Calculation

Sample calculations can be performed with Eq. (24) with a comparison to the following model employed by WANO:

$$FRI_P = \left[\left(A_{131} \right)_N - k \left[\left(A_{134} \right)_N \right] \right] \times \left[\left(L_n / LHGR \right) \times \left(100 / P_o \right) \right]^{1.5} \quad \textbf{(29)}$$

$$k = \frac{\left(Y \times \lambda^{0.955} / \left[\lambda + B_n \right] \right)_{131}}{\left(Y \times \lambda^{0.955} / \left[\lambda + B_n \right] \right)_{134}} \quad \textbf{(30)}$$

As listed in Table 2A, specific CANDU historical cases include: (i) three cases with no defects (i.e., only fuel debris) (Cases 1A, 1B and 1C) after which there is an occurrence of a single defect in two cases with the previous amount of fuel debris (Cases 1D and 1E), and (ii) a defect excursion with many defective fuel rods and a high tramp uranium level (Case 2). Table 2B shows the corresponding coolant activity data and reactor parameters for these cases. The calculation of the *FRI* is specifically detailed in Table 3.

Table 2A. Sample Defective Fuel Experience in Commercial CANDU Cases

Bundle ID & Reactor	Case	Sample Date	Element Linear Power (kW/m)	Final Bundle Burnup (MWh/kgU)	Defect Description	Number of Failed Elements/Tramp U (g)
Case 1 XC4251C Bruce B, Unit 6 Full Analysis of ^{131}I, ^{132}I, ^{133}I, ^{134}I, and ^{135}I	1A	October 1, 2002	-	210	Coolant activity analysis of ^{133}Xe suggests the defect first occurred after a bundle shift on February 4, 2003. Defected outer element #13, fretted hole, and other fretting marks on element #14, total sheath breach ~ 10 mm^2.	0
	1B	December 12, 2002	-	210		0
	1C	December 28, 2002	-	210		0
	1D	March 24, 2003	32.0	210		1
	1E	March 31, 2003	31.1	210		1
Case 2 Douglas Point	-	March 21-21, 1982	40	150	End-weld defects and secondary hydriding	18/600 – 1000

There is a difference in the purification flow rates between the Bruce and Douglas Point reactors (i.e., for Douglas Point the observed flow rate is greater than a factor of four of the assumed reference flow rate of $B_n = 5 \times 10^{-5}$ s^{-1}). In particular, if a reference value of $B_n = 2 \times 10^{-4}$ s^{-1} were used instead in the FRI_C calculation (which is more typical of the Douglas Point and CANDU 6 reactors), the FRI_C values in Table 3 would drop by a similar factor of ~4. The value of the tramp correction factor k_c would also change from 0.0090 to 0.0036. Thus, more operational data from the various types of CANDU reactors are needed to establish an appropriate reference value for the coolant purification flow rate.

As seen in Table 3, the CANDU FRI_C increases with the occurrence of a fuel failure and significantly more when many defects are present in the core. In fact, the data in the table are consistent with the suggestion of WANO that the presence of a defect corresponds to an *FRI* greater than ~19 Bq/g (~5 x 10^{-4} μCi/g) [1,2]. The difference in the *FRI*'s for CANDU

Table 2B. Coolant Activity and Reactor Data for Defective Fuel Cases in Table 2A

Case	Date	Radioisotope Data (Grab Sample Analysis)					Reactor Data		
		^{131}I µCi/g	^{132}I µCi/g	^{133}I µCi/g	^{134}I µCi/g	^{135}I µCi/g	Element Power (kW/m)	PHTS Purification Flow (kg/s)	Ba Purification Rate Constant (s^{-1})
1A[1]	01-Oct-2002	**1.331E-4**	4.6220E-3	1.6030E-3	**8.0430E-3**	4.1510E-3	-	9.92	4.13×10^{-5}
1B[1]	12-Dec-2002	**3.181E-4**	4.7680E-3	1.3130E-3	**8.8640E-3**	3.4190E-3	-	11.0	4.58×10^{-5}
1C[1]	28-Dec-2002	**1.010E-4**	4.9370E-3	1.4520E-3	**9.0530E-3**	3.7210E-3	-	8.87	3.70×10^{-5}
1D[1]	24-Mar20-03	**7.104E-4**	6.2380E-3	2.5320E-3	**6.9680E-3**	4.6530E-3	31.8	10.2	4.25×10^{-5}
1E[1]	31-Mar-2003	**6.736E-4**	6.4160E-3	2.7860E-3	**1.0200E-2**	4.3810E-3	31.1	10.3	4.29×10^{-5}
2[2]	21-Mar-1982	**5.615E-2**	0.3698	0.1704	**1.128**	0.3154	40	8.65	$2.66\ 10^{-4}$

1. Assumes a PHTS mass: 2.40×10^5 kg for Bruce B Unit 6.
2. Assumes a PHTS mass: 3.25×10^4 kg for a single loop for the double loop configuration in Douglas Point since the loops are isolated by the ion-exchange columns.

Table 3. *FRI* Calculation Using the CANDU [Eq. (24)] and PWR [Eq. (29)] Model

Reactor/Case	Coolant Activity Concentration (Bq/g)*		Purification Rate Constant (s^{-1})		Linear Power (kW/m)		Reactor Power %, P_o	Fuel Reliability Indicator Bq/g (or µCi/g)		No. of observed defects
	I-131	I-134	B_a	Reference B_n (CANDU/PWR)	*LHGR*	Reference L_n (CANDU/ PWR)		FRI_C	FRI_P	
Bruce B/Case 1A	4.92	298	4.13E-05	5.00E-05/2.00E-05	35.9	38/18	100	2.0 (5.3E-5)	-0.2 (-4E-6)	0
Bruce B/Case 1B	11.8	328	4.58E-05	5.00E-05/2.00E-05	35.9	38/18	100	10.4 (2.82E-4)	5.2 (1.4E-4)	0
Bruce B/Case 1C	3.74	335	3.70E-05	5.00E-05/2.00E-05	35.9	38/18	100	-0.1 (-3E-6)	-1.7 (-4.5E-5)	0
Bruce B/Case 1D	26.3	258	4.25E-05	5.00E-05/2.00E-05	35.9	38/18	100	26.6 (7.18E-4)	16.1 (4.36E-4)	1
Bruce B/Case 1E	24.9	377	4.29E-05	5.00E-05/2.00E-05	35.9	38/18	100	23.9 (6.45E-4)	13.8 (3.73E-4)	1
Douglas Pt/Case 2	2078	41740	2.66E-04	5.00E-05/2.00E-05	40.7	38/18	100	7158 (0.193)	6979 (0.189)	18

*1µCi/g = 3.7×10^4 Bq/g.

versus PWR fuel is due principally to the difference in the reference values for the linear heat generation and purification constant. The effect of scaling of the release rate dependence on linear power in the CANDU *FRI* is not as significant since L_n and *LHGR* are comparable using the revised L_n value chosen from the CANDU experience in Table 3. Note that for Case 1C in Table 3, the FRI_C for CANDU fuel is negative which can occur with a predominant tramp contribution given the uncertainty in the coolant activity measurement. However, in accordance with recoil theory in Figure 2, one can simply set any negative *FRI* to zero.

In order to fully qualify the current methodology for each CANDU station involved with 37-element and 28-element fuel bundle designs, it is recognized that additional iodine coolant activity data are needed as on-going operational experience is gained. Here information is needed on the number of known failures (along with purification flow rates and element linear power ratings). Further data are required to benchmark the model considering that only a limited preliminary benchmark has been possible with available data presented in Table 3.

5. Conclusions

1. Using a calculation of the release rate from measured coolant activity data, and normalizing this quantity by the fission yield, one can distinguish between defective fuel and a tramp fuel contribution. This approach has been employed by WANO in the development of their fuel reliability indicator (*FRI*). This analysis specifically accounts for different mechanisms of fission product release based on a Booth diffusion model and recoil theory. In this treatment, the coolant activity concentration of I-131 provides for an assessment of defective fuel in the reactor. This quantity is corrected for tramp uranium effects using a measured coolant activity concentration of I-134.

2. In the WANO methodology for development of a common fuel reliability indicator for PWR's (FRI_P), the coolant activity is corrected for coolant purification effects and normalized to a reference purification rate constant (i.e., $B_n = 2 \times 10^{-5}$ s^{-1}). The FRI_P is further scaled to a reference linear heat rating (i.e., $L_n = 18$ kW/m) employing a scaling law of $(LHGR)^{1.5}$ for the release rate. This latter correlation also depends on the average linear heat output for a given reactor unit at 100% full power, which is further adjusted by the percentage of reactor power at the time of measurement. However, because of the higher linear powers experienced in the CANDU reactor, the PWR power correlation needs to be modified (see item 3).

3. The new power scaling relation for CANDU fuel is $(10^{0.057 \cdot LHGR})$. This law replaces the PWR one that implicitly assumed athermal diffusion. In contrast, the new correlation takes into account intrinsic and vacancy-enhanced diffusion in oxidized fuel at the higher fuel ratings. This correlation is specifically developed from operational defect experience in various CANDU reactors including 28- and 37-element bundle designs. Different reference values from CANDU experience are further employed for the model normalization ($B_n = 5 \times 10^{-5}$ s^{-1} and $L_n = 38$ kW/m). This study therefore yields a new FRI_C for defective fuel assessment in CANDU reactors (i.e., Eqs. (24) and (26) or equivalently Eqs. (29) and (30)) based on fundamental fission product release theory. A FRI_C value of 19 Bq/g or ~5×10^{-4} µCi/g typically indicates the presence of a fuel failure (i.e., based on the available defect cases at Bruce Power and Douglas Point for a specific reference purification flow rate of 5×10^{-5} s^{-1}). This threshold value would change if a different reference value is deemed more appropriate with additional benchmarking data for the purification flow rate representative of all CANDU reactors. Additional data are needed which include several defect cases from each involved station where further comparisons can be made to show differences obtained when using the new methodology versus the current WANO *FRI*.

6. Acknowledgement

The authors would like to acknowledge funding from the CANDU Owners Group (COG) for this study. They would also like to thank R. Ham-Su (CNL), D. Law (NBPN), M. Dobrean (OPG-Pickering), R. Scrannage (BP), D. Park and S. Jun (KHNP), M. Chibzuloiu (SNN) and H. Anderson (COG) for assistance and guidance of this project.

7. References

[1] Chapter 5: "Fuel Reliability," Report from the World Association of Nuclear Operators (WANO), October 2001, pages 223-239.
[2] R. Chiarelli, "WANO Performance Indicator Programme Reference Manual," WANO MN 2014-2, May 2014.
[3] B.J. Lewis, *J. Nucl. Mater.* **160** (1988) 201.
[4] B.J. Lewis, R.D. MacDonald, N.V. Ivanoff, F.C. Iglesias, *Nucl. Technol.* **103** (1993) 220.
[5] A.H. Booth, "A Suggested Method for Calculating the Diffusion of Radioactive Rare Gas Fission Products from UO_2 Fuel Elements and a Discussion of Proposed In-Reactor Experiments that may be used to Test its Validity," AECL-700, Atomic Energy of Canada Limited (1957).
[6] I.J. Hastings, C.E.L. Hunt, J.J. Lipsett, *J. Nucl. Mater.* **130** (1985) 407.
[7] A.D. Appelhans and J.A. Turnbull, "Measured release of radioactive xenon, krypton and iodine from UO_2 during nuclear operation and a comparison with release models," in: Proc. 8th Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, Oct. 27-31, 1980.
[8] B.J. Lewis, R.J. Green, W.T. Che, *Nucl. Technol.* **98** (1991) 307.
[9] B.J. Lewis, A. El-Jaby, J. Higgs, W.T. Thompson, F.C. Iglesias, R. Laidler, J. Armstrong, R. Stone, R. Odunton, *J. Nucl. Mater.* **366** (2007) 37.
[10] A. El-Jaby, B.J. Lewis, W.T. Thompson, F. Iglesias, M. Ip, *J. Nucl. Mater.* **399** (2010) 87.
[11] J. A. Turnbull, J. R. Friskney, F. A. Findlay, F. A. Johnson, A. J. Walter, *J. Nucl. Mater.* **107** (1982) 168.
[12] B.J. Lewis, *J. Nucl. Mater.* **148** (1987) 28.
[13] K.S. Folk, "Nuclear fuel performance at Hatch-1 and 2" in: Proceedings of the 17th International Utility Nuclear Fuel Performance Conference, Charlotte, North Carolina (1986).
[14] F.W. Walker, J.R. Parrington and F. Feiner, "Nuclides and Isotopes", 14th Edition, General Electric Company, 1989.
[15] J.C. Killeen and J.A. Tumbull, "An experimental and theoretical treatment of the release of ^{85}Kr from hyper- stoichiometric uranium dioxide," in: Proc. Workshop on Chemical Reactivity of Oxide Fuel and Fission Product Release, Berkeley, Gloucestershire, England, April 7-9, 1987.
[16] C.E. Beyer, "Methodology Estimating Number of Failed Fuel Rods and Defect Size," Electric Power Research Institute, EPRI NP-6554, September 1989.
[17] L. Wilk, "CANDU Fuel Burnup and Power Rating 2012 Update," NWMO TR-2013-02, February 2013.
[18] B.J. Lewis, C.R. Phillips, M.J.F. Notley, *Nucl. Technol.* **73** (1986) 72.
[19] B.J. Lewis, C.E.L. Hunt, F.C. Iglesias, *J. Nucl. Mater.* **172** (1990) 197.
[20] S. Livingstone, B.J. Lewis, M. Ip, F. Iglesias and A. Fitchett, "Progress in developing an on-line fuel-failure monitoring tool for Candu reactors," 11th International Conference on CANDU Fuel, Niagara Falls, Ontario, Canada, October 17-20, 2010.