ASSESSMENTS OF SHEATH STRAIN AND FISSION GAS RELEASE DATA FROM 20 YEARS OF POWER REACTOR FUEL IRRADIATIONS

P.L. PURDY AND A.M. MANZER
Atomic Energy of Canada Limited
2251 Speakman Drive
Mississauga, Ontario, Canada L5K 1B2

R.H. HU
Hydro-Quebec
1155 Metcalfe Street, 8th Floor, Suite 880
Montreal, Quebec, Canada H3B 2V6

R.A. GIBB
New Brunswick Power
Point Lepreau Generating Station, P.O. Box 10
Lepreau, New Brunswick, Canada E0G 2H0

E. KOHN
Ontario Hydro
700 University Avenue
Toronto, Ontario, Canada M5G 1X6

ABSTRACT

Over the past 20 years, many fuel elements or bundles discharged from Canadian CANDU® power reactors have been examined in the AECL hot cells. The post-irradiation examination (PIE) database covers a wide range of operating conditions, from which fuel performance characteristics can be assessed. In the present analysis, a PIE database was compiled representing elements from a total of 129 fuel bundles, of which 26% (34 bundles) were confirmed to have one or more defective elements. This comprehensive database was assessed in terms of measured sheath strain and fission gas release (FGR) for intact elements, in an attempt to identify any changes in these parameters over the history of CANDU reactor operation. Results from this assessment indicate that, for the data that are typical of normal CANDU operating conditions, tensile sheath strain and FGR have remained within 0.5% and 8%, respectively. Those data beyond these ranges are from fuel operated under abnormal conditions, not representative of normal operation, and thus do not indicate a trend toward unexpected fuel behaviour. The distributions of the PIE measurements indicate that maximum expected sheath strains and FGR for normally operated fuel are 0.7% and 13%, respectively.

CANDU® is a registered trademark of Atomic Energy of Canada Limited (AECL).

1. INTRODUCTION

For many years, post-irradiation examination (PIE) has been performed on fuel elements or bundles discharged from Canadian CANDU power reactors. The examined fuel has been exposed to a wide range of operating conditions that is much larger than the range of conditions that CANDU fuel is normally exposed to [1-2]. The extensive PIE database, from the early 1970s to current reactor operation, can be analyzed to aid in the assessment of CANDU fuel performance characteristics.

In this analysis, a PIE database has been compiled, representing a total of 129 fuel bundles composed of thirty-four 28-element and ninety-five 37-element bundles, as shown in Table 1. These bundles were irradiated in the reactor units at either the Bruce-A, Bruce-B, Pickering-A, Pickering-B, Darlington or Point Lepreau nuclear generating stations. The bundles were fabricated by General Electric Canada (GEC), Westinghouse Canada Limited (WCL) - now Zircatec Precision Industries (ZPI), or Combustion Engineering Superheater (CES) - no longer a CANDU fuel manufacturer. Of the total 129 bundles, 26% (34 bundles) were confirmed to have one or more defective elements (see Table 1). Since measurements on defective elements are not representative of those that exist in the reactor core, only PIE data from intact (i.e., non-defective) elements were included in the database. A number of these intact elements, however, were from bundles in which other elements failed.

The compiled database includes measurements of sheath strain, ridge height, element bow, fission gas release (FGR), and appendage fretting and wear. The data were analyzed in order to identify any changes in these parameters over the past 20 years of Canadian CANDU power reactor operation.

It should be recognized that the fuel bundles in the database were often examined either as part of a surveillance program, or in order to determine the root cause of failures or because the fuel operated near some threshold. As such, the database is not a statistically random sample over time. The data do span the typical range of reactor operating conditions, however, and can still be used to examine general trends of power reactor fuel performance. In this paper, the mid-pellet circumferential sheath strain and FGR data and assessments are presented.

2. POST-IRRADIATION EXAMINATION DATABASE

In order to obtain a database that was within normal CANDU fuel operation, it was necessary to consider the irradiation history of each bundle. For instance, some bundles experienced burnups greater than 475 MW·h/kg U because of abnormal fuelling restrictions that led to those bundles remaining in the channel until a regular maintenance shutdown. In other cases, again because of abnormal fuelling restrictions, a small number of bundles were ramped to high powers after having resided in low-power positions for extended periods. Hence, the available PIE data that can be considered to be within normal CANDU operating conditions excludes end-of-life power-ramped bundles and bundles with burnups greater than 475 MW·h/kg U resulting from abnormal fuelling restrictions.

Of the 129 examined fuel bundles, many were not measured for sheath strain or for FGR. Hence for the analysis presented here, the data consist of sheath strain measurements on 44 bundles and FGR measurements on 70 bundles (see Table 1). Moreover, the available PIE database that can be considered applicable to normal CANDU operating conditions (i.e., not experiencing fuelling restrictions) contains intact elements from 39 bundles for the sheath strain assessment and intact elements from 63 bundles for the FGR assessment (Table 1). The remaining data, not typical of normal operation, are also included in this analysis, however, as a comparison to the data that are representative of normal CANDU operating conditions. These bundles, subjected to abnormal operation, have experienced average burnups as high as 686 MW·h/kg U, as illustrated in Figure 1. Normally operated bundles are discharged with burnups below 450 MW·h/kg U.

As shown in Figure 1, the database consists of peak outer-element average linear powers (i.e., the outer-element average linear powers corresponding to the peak bundle powers) and bundle average burnups that are quite evenly distributed over the 20-year period under investigation. Hence the distribution of the irradiation history data should not bias the data into artificial trends with time. It should be noted, however, that UO_2 density and bundle uranium mass have increased during this time frame [3], which may introduce some bias, particularly to the sheath strain results.

The PIE data were analyzed in terms of trends with time and of the dependence on power and burnup. The outer, intermediate and inner elements, where the central element was included as an inner element for the 37-element bundles, were divided into separate data subsets to provide a direct comparison of the sheath strains and gas releases as a function of element location within the bundle. For those bundles that had a number of elements examined, the data from elements of the same ring (i.e., outer, intermediate or inner elements) were averaged to represent that ring of elements for the power and burnup conditions experienced by the bundle.

It is possible that the compiled database does not contain all of the PIE that has been performed during the 20-year period. More data will also be available as future PIE is performed. The frequency distributions of the strain and FGR measurements from the current database were therefore used to predict the expected range of sheath strain and FGR data for bundles experiencing powers and burnups within those of the current database. Moreover, since the current data span the normal range of reactor operating conditions (see Tables 1 and 2), in-reactor plastic sheath strains and FGR can be expected to be within those given by the distribution results.

3. SHEATH STRAIN ASSESSMENT

Post-irradiation measurements of fuel element diameters were used to quantitatively assess circumferential plastic sheath strain by taking the ratio of the post-irradiation measurements to the as-fabricated nominal element diameters. Mid-pellet sheath strains were investigated in the present analysis, where the average strain for each element was used since, in many cases, the diameters were not measured along (or reported for) the entire element length.

Results from the mid-pellet sheath strain measurements are provided in Figures 2 and 3, where positive values represent tensile strains and negative values indicate compressive strains. Sheath strain is plotted over time in Figure 2a, where there is evidence that strains have increased slightly over time. This may be a result of the increased uranium mass, however, as discussed in Reference 3. Even so, outer-element tensile strains have remained below 0.5% for normal operation, and have remained below 1% for abnormal conditions. Normally operated intermediate and inner elements exhibited strains below 0.1%, indicating that element strains are not uniform throughout the bundle, but decrease towards the centre of the bundle.

The frequency distribution of the strain measurements is plotted in Figure 2b. For normal operation, the distributions of the strains are centred about an average value of 0.09% for outer elements, -0.26% for intermediate elements and -0.25% for inner elements (see Table 2). When the data from abnormally operated fuel are also included, the average is 0.11% for outer elements, 0.05% for intermediate elements and -0.03% for inner elements (Table 2). From the normal distribution of the data, the outer-element strains for normally operated bundles are expected to be less than 0.7%, to within 99% confidence (i.e., 3 standard deviations). Although there are only a few data for intermediate and inner elements, resulting in large standard deviations (Table 2), the intermediate and inner element strains should be less than 0.7%, since strain decreases towards the centre of the bundle where the element power ratings are lower (see Figure 3).

In Figure 3, the measured sheath strains are plotted versus peak outer-element average linear power (Figure 3a) and versus bundle discharge burnup (Figure 3b). For the envelope of the maximum strain values, a very mild trend of increasing strain with power and burnup is evident from Figure 3. The scatter of data within these envelopes indicates, however, that a more complex relationship exists with the power history (i.e., time at power, power at burnup and power ramps), bearing in mind that there may be some uncertainty in the power and burnup measurements. Since outer, intermediate and inner elements are identified in Figure 3, it is again evident that intermediate and inner elements exhibit lower strains than do the outer elements. This is because of the lower power ratings of the intermediate and inner elements.

4. FISSION GAS RELEASE ASSESSMENT

During irradiation, a fraction of the gases produced by fission eventually migrates to the internal void volume. If the pressure becomes too high, this released fission gas can reduce the heat transfer from the fuel to the sheath, resulting in increased fuel temperatures and additional gas release. As a consequence, the risk of failure due to stress corrosion cracking could increase due to the higher concentrations of corrosive fission gases in the gap. A measure of FGR into the fuel-to-sheath gap during irradiation can be obtained from that fraction of the total amount of xenon produced during irradiation that is released into the gap. This ratio is expressed as a percentage.

Results from the PIE FGR measurements are provided in Figures 4, 5 and 6. As can be seen in Figure 4a, FGR has been measured as high as 25%, but has consistently remained below 8% for outer elements operated under normal CANDU reactor conditions. Of the few measurements on normally operated intermediate and inner elements, FGR has remained below 1%, indicating that FGR decreases towards the centre of the bundle. None of the elements from the normally operated bundles were reported to exhibit evidence of abnormally high temperatures or high gas pressures (i.e., above coolant pressure) that could lead to sheath lift-off from the pellet.

The frequency distribution of the gas release measurements is plotted in Figure 4b. For normal operation, distributions of FGR are centred about an average value of 2.6% for outer elements, 0.2% for intermediate elements and 0.2% for inner elements (see Table 3). When abnormally operated fuel is included, the average is 3.6% for outer elements, 1.3% for intermediate elements and 0.2% for inner elements (Table 3). Unlike the sheath strains, the FGR data do not follow a normal distribution. Hence the standard deviations could not be determined for the FGR data, and are not included in Table 3.

The range of expected FGR data was determined by using the cumulative distribution of the data. Figure 5 illustrates the fraction of the measurements that are greater than the specified incremental gas releases. As shown in Figure 5, the data are represented by an exponential equation which asymptotically approaches 0. The equation, as given in Figure 5, was fitted to the data where it was constrained to a value of 1 for FGR greater than 0% (i.e., all of the data are greater than 0% gas release). The equation was then used to determine the FGR value of 13% at which only 1% of the data can be expected to exceed (i.e., 99% confidence interval). This method was not performed for the intermediate and inner elements because of the small number of data. Gas release from the intermediate and inner elements should also be less that 13%, however, since FGR decreases towards the centre of the bundle where the element power ratings are lower (see Figure 6).

Figure 6a illustrates how FGR, as a percentage, remains near zero for outer-element linear powers below 40 to 45 kW/m. An increase in gas release is then seen to increase with increasing linear power at around 40 to 45 kW/m. This behaviour is consistent with reduced fission product mobility in the UO_2 matrix at temperatures below about 1000°C [4]. Maximum FGR is seen to gradually increase with burnup in Figure 6b for all available PIE data. This is to be expected with the buildup of long-lived fission products as irradiation proceeds. For normally operated fuel, the maximum measured gas release remains at or below 8%. Intermediate and inner elements are identified in Figure 6, where the gas releases are lower than in the outer elements.

5. CONCLUSIONS

Based on a review of CANDU power reactor fuel PIE data, sheath strain and fission gas release have not increased significantly over a 20-year period. Hence in terms of these parameters, the performance of the fuel has not changed. Specifically, the trends in the data confirm:

- Circumferential sheath strain and FGR increase with increasing power and burnup. These parameters are not uniform throughout the bundle, where maximum values are found among outer elements that have the highest power ratings in the bundle.

- High strain and FGR have been predominantly found in high-power and high-burnup or end-of-life power-ramped fuel bundles. These bundles are outside the range of normal CANDU power reactor operation.

For the current PIE database of bundles operated under typical CANDU conditions, tensile sheath strains and FGR have remained within 0.5 and 8%, respectively. Those data beyond these ranges are from fuel operated under abnormal conditions, and thus do not indicate a trend toward unexpected fuel behaviour.

The frequency distributions of the strain data indicate that, for normally operated fuel, the average outer-element sheath strain is 0.09%. Within 99% confidence, the maximum sheath strain that can be expected is 0.7%, for fuel operated under conditions that are within the range experienced by the bundles in the database.

The frequency distributions of the FGR data indicate that, for normally operated fuel, the average outer-element FGR is 2.7%. Within 99% confidence, the maximum that can be expected is 13%, for fuel operated under conditions that are within the range experienced by the bundles in the database.

Since the range of powers and burnups experienced by the bundles in the database span that which is typical of normal reactor operation, the plastic sheath strain and FGR of 0.7% and 13%, respectively, represent the maximum that can be expected in-reactor under normal operating conditions.

ACKNOWLEDGEMENTS

This work was jointly funded by Hydro Quebec and New Brunswick Power.

The authors would like to acknowledge the time and effort of the many individuals who performed the PIE in the AECL hot cells located at the Chalk River Laboratories (CRL) and the Whiteshell Laboratories, without whom this work could not have been accomplished. The PIE was funded by Atomic Energy of Canada Limited, Ontario Hydro, Hydro Quebec and New Brunswick Power.

The authors would also like to thank Ontario Hydro for permission to use their data in formulating the PIE database.

REFERENCES

[1] A.J. Hains and J. Novak, "Ontario Hydro High Burnup Power Reactor Fuel Performance", in Proc. 2nd International Conference on CANDU Fuel, Pembroke, Canada, 1-5 October 1989.

[2] Y.N. Zhou, M.R. Floyd and M.A. Ryz, "Performance of Bruce Natural UO_2 Fuel Irradiated to Extended Burnups", in Proc. 4th International Conference on CANDU Fuel, Toronto, Canada, 21-25 September 1995.

[3] S.J. Palleck and R. Sejnoha, "Bundle Uranium Content and Performance of CANDU Fuel", to be presented at the 5th International Conference on CANDU Fuel, Toronto, Canada, 21-25 September 1997.

[4] D.R. Olander, "Fundamental Aspects of Nuclear Reactor Fuel Elements", ERDA Technical Information Center, Oak Ridge, Tennessee, T1D-26711-P1, April 1976.

TABLE 1: Summary of CANDU Power Reactor Fuel Bundle PIE Database.

Database:	**28-Element**	**37-Element**	**Total**
Total Number of Bundles	34	95	129
Number of Intact Bundles	26	69	95
Number of Bundles Having Defective Elements	8	26	34
Sheath Strain Assessment:	**28-Element**	**37-Element**	**Total**
Total Number of Bundles	22	22	44
Number of Bundles Representative of Normal Operation *	22	17	39
Number of Bundles Not Representative of Normal Operation	0	5	5
Fission Gas Release Assessment:	**28-Element**	**37-Element**	**Total**
Total Number of Bundles	23	47	70
Number of Bundles Representative of Normal Operation *	23	40	63
Number of Bundles Not Representative of Normal Operation	0	7	7

* Excludes end-of-life power ramped bundles and bundles with burnups greater than 475 MWh/kgU resulting from abnormal fuelling restrictions.

TABLE 2: Average Mid-Pellet Sheath Strains.

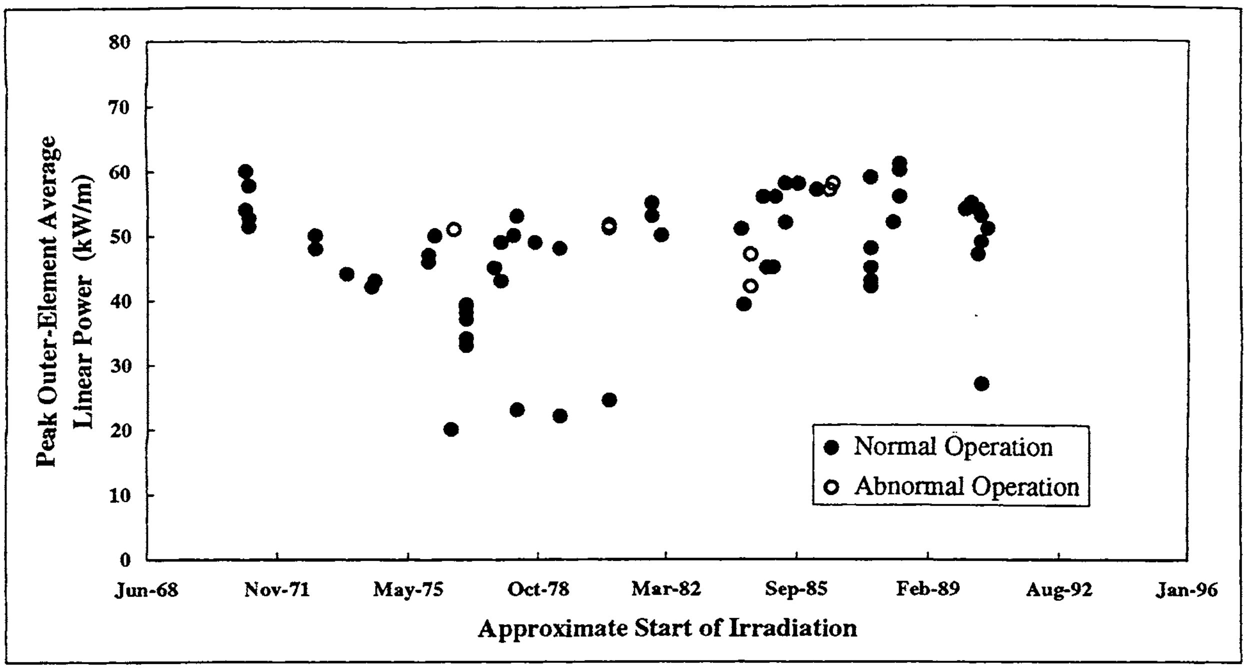
Sheath Strain Assessment	**Sample Size** (# of bundles)	**Bundle Discharge Burnup (MWh/kgU)**			**Peak Outer-Element Average Linear Power (kW/m)***			**Sheath Strain (%)**	
		High	Low	Average	High	Low	Average	Average	Std Dev
Normal Operation Only:									
Outer Elements	40	409	74	194	61	22	46	0.09	0.19
Intermediate Elements	6	409	74	149	53	24	42	-0.26	0.32
Inner Elements	5	409	74	160	53	24	41	-0.25	0.13
Normal & Abnormal Operation:									
Outer Elements	41	477	74	201	61	22	46	0.11	0.23
Intermediate Elements	10	686	74	324	58	24	47	0.05	0.48
Inner Elements	8	686	74	333	57	24	46	-0.03	0.37

* Represents the outer-element average linear power corresponding to the peak bundle power.

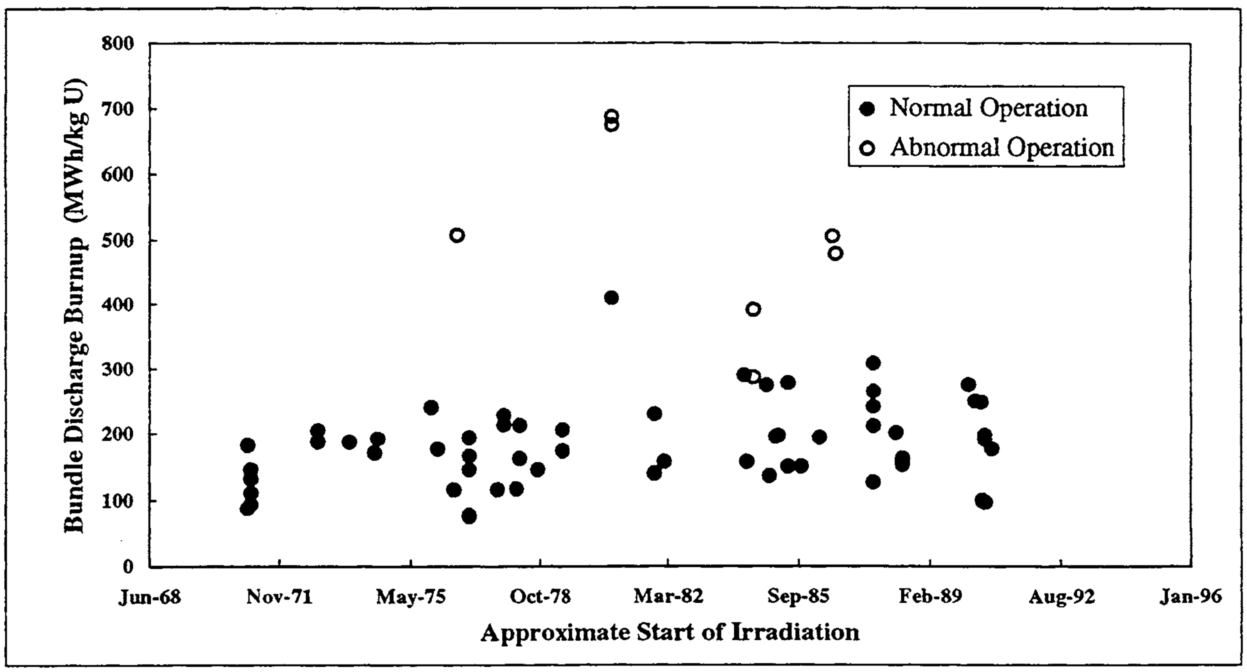
TABLE 3: Average Fission Gas Releases.

Fission Gas Release Assessment	**Sample Size** (# of bundles)	**Bundle Discharge Burnup (MWh/kgU)**			**Peak Outer-Element Average Linear Power (kW/m)***			**Fission Gas Release (%)**
		High	Low	Average	High	Low	Average	Average
Normal Operation Only:								
Outer Elements	65	409	74	184	61	22	48	2.7
Intermediate Elements	3	390	99	228	47	42	45	0.2
Inner Elements	2	390	194	292	45	42	44	0.2
Normal & Abnormal Operation:								
Outer Elements	69	686	74	207	61	22	48	3.6
Intermediate Elements	8	686	99	441	58	42	50	1.3
Inner Elements	5	673	194	453	57	42	49	0.2

* Represents the outer-element average linear power corresponding to the peak bundle power.

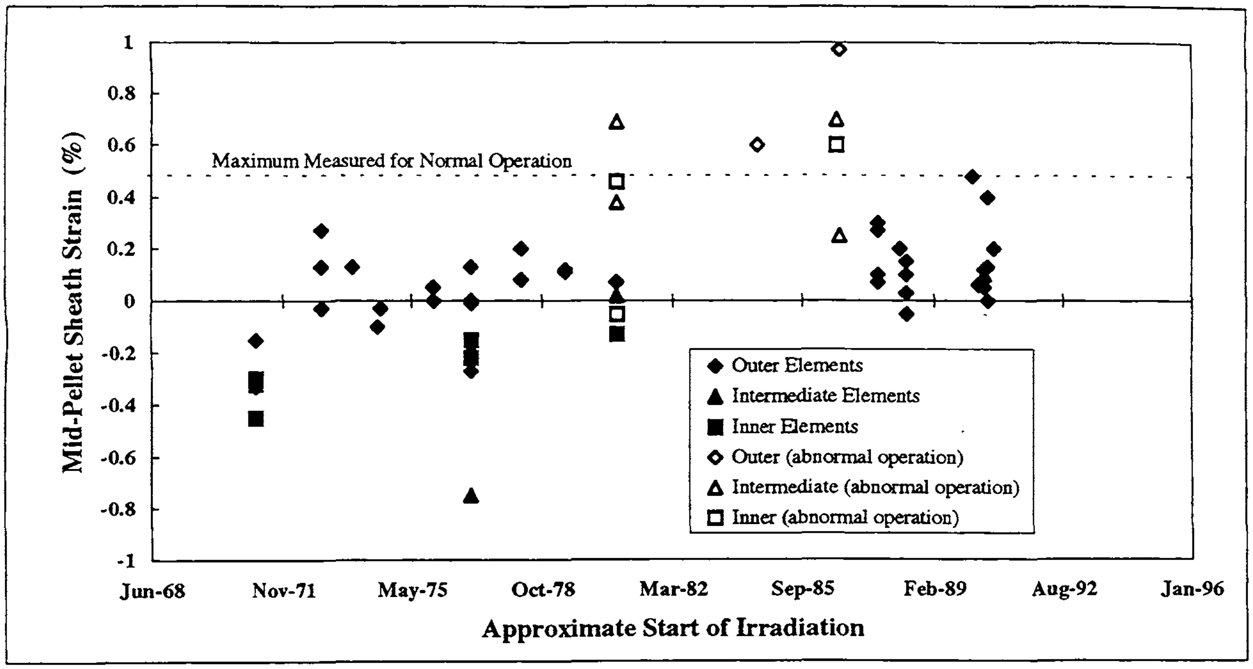


(a) Peak Outer-Element Average Linear Power.

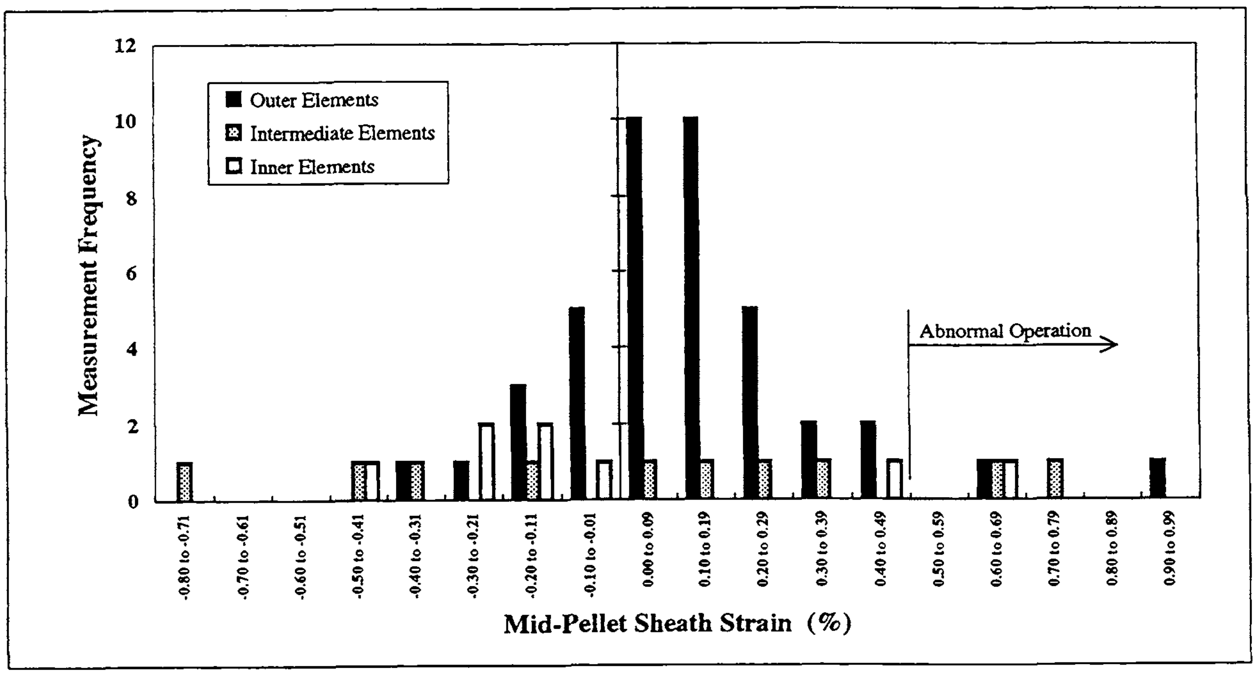


(b) Bundle Discharge Burnup.

FIGURE 1: (a) Peak Outer-Element Average Linear Power and (b) Discharge Burnup of Post-Irradiation Examined Fuel Bundles.

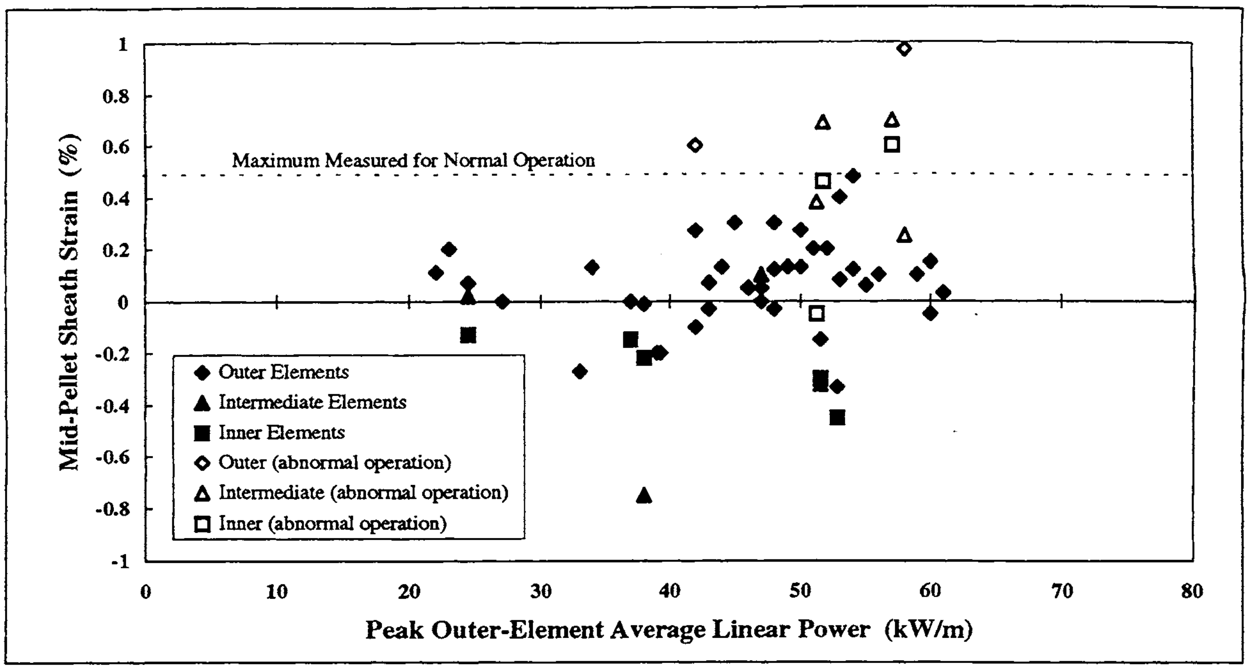


(a) Mid-Pellet Sheath Strain Measurements.

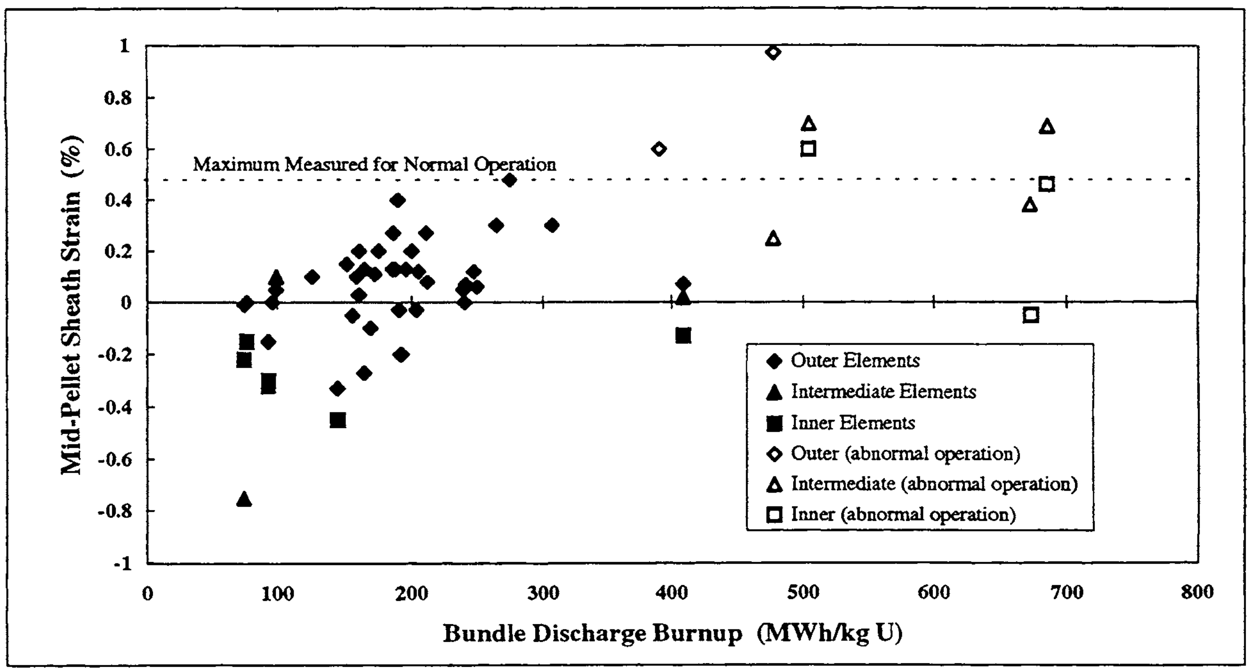


(b) Mid-Pellet Sheath Strain Measurement Frequency.

FIGURE 2: Mid-Pellet Sheath Strain (a) Measurements and (b) Measurement Frequency from Post-Irradiation Examined Fuel Bundles.

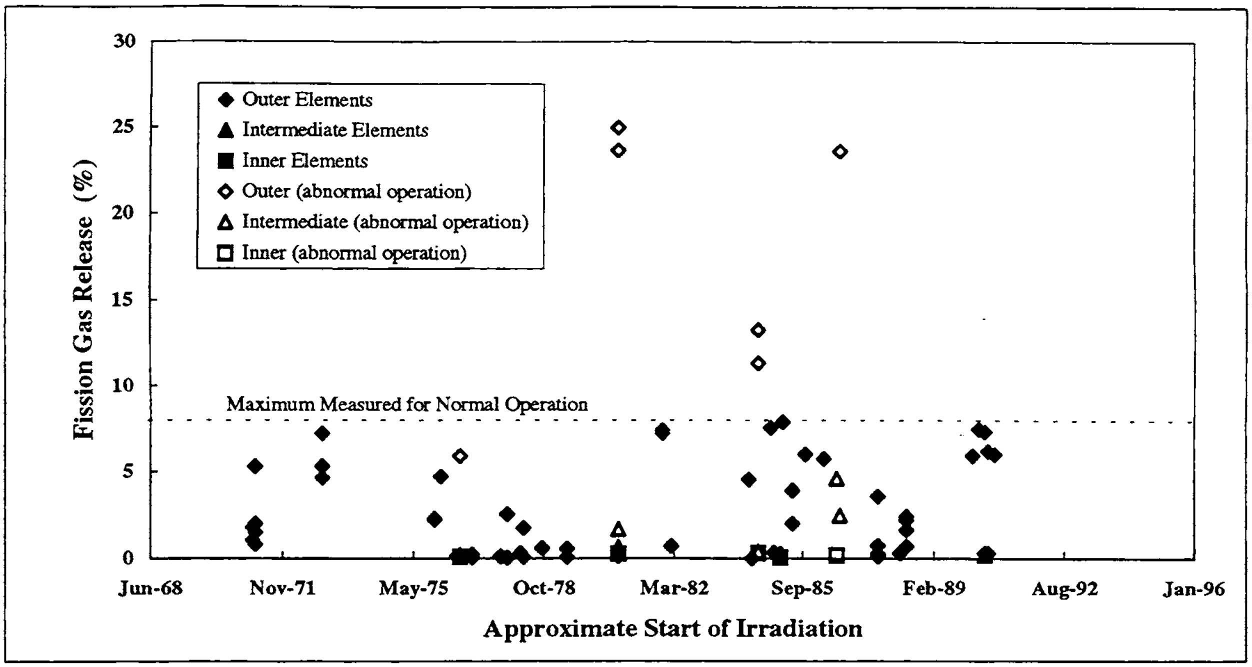


(a) Mid-Pellet Sheath Strain vs. Peak Outer-Element Average Linear Power.

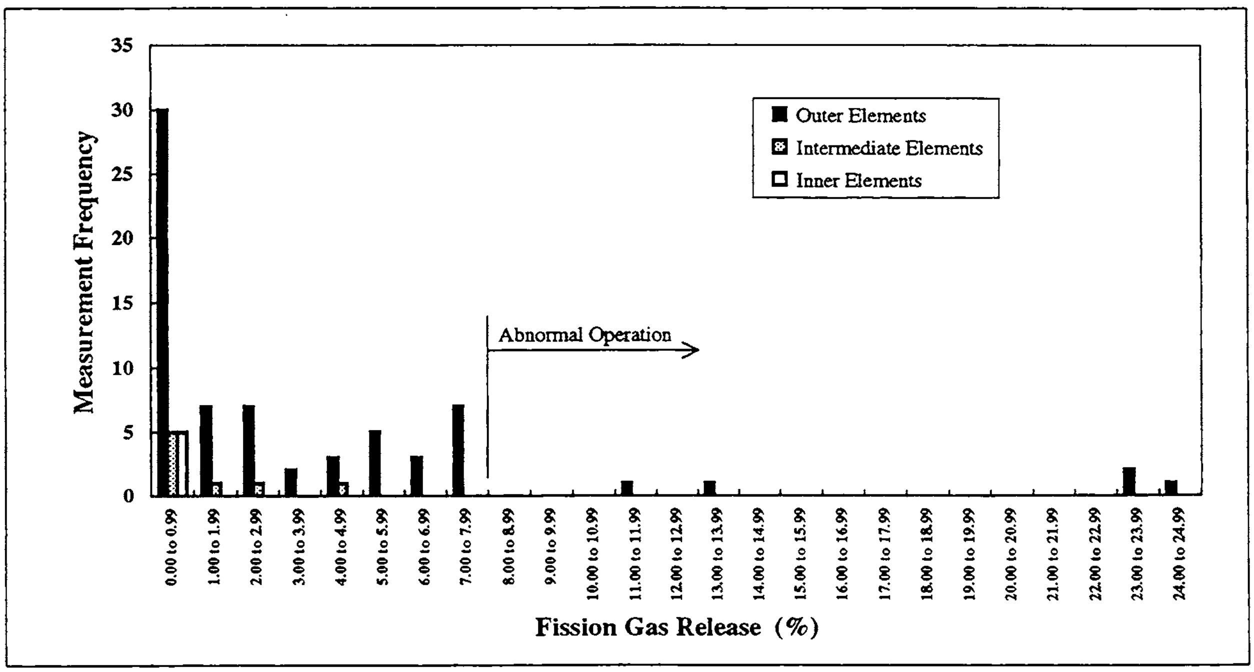


(b) Mid-Pellet Sheath Strain vs. Bundle Discharge Burnup.

FIGURE 3: Mid-Pellet Sheath Strain vs. (a) Peak Outer-Element Average Linear Power and (b) Bundle Discharge Burnup.



(a) Fission Gas Release Measurements.



(b) Fission Gas Release Measurement Frequency.

FIGURE 4: Fission Gas Release (a) Measurements and (b) Measurement Frequency from Post-Irradiation Examined Fuel Bundles.

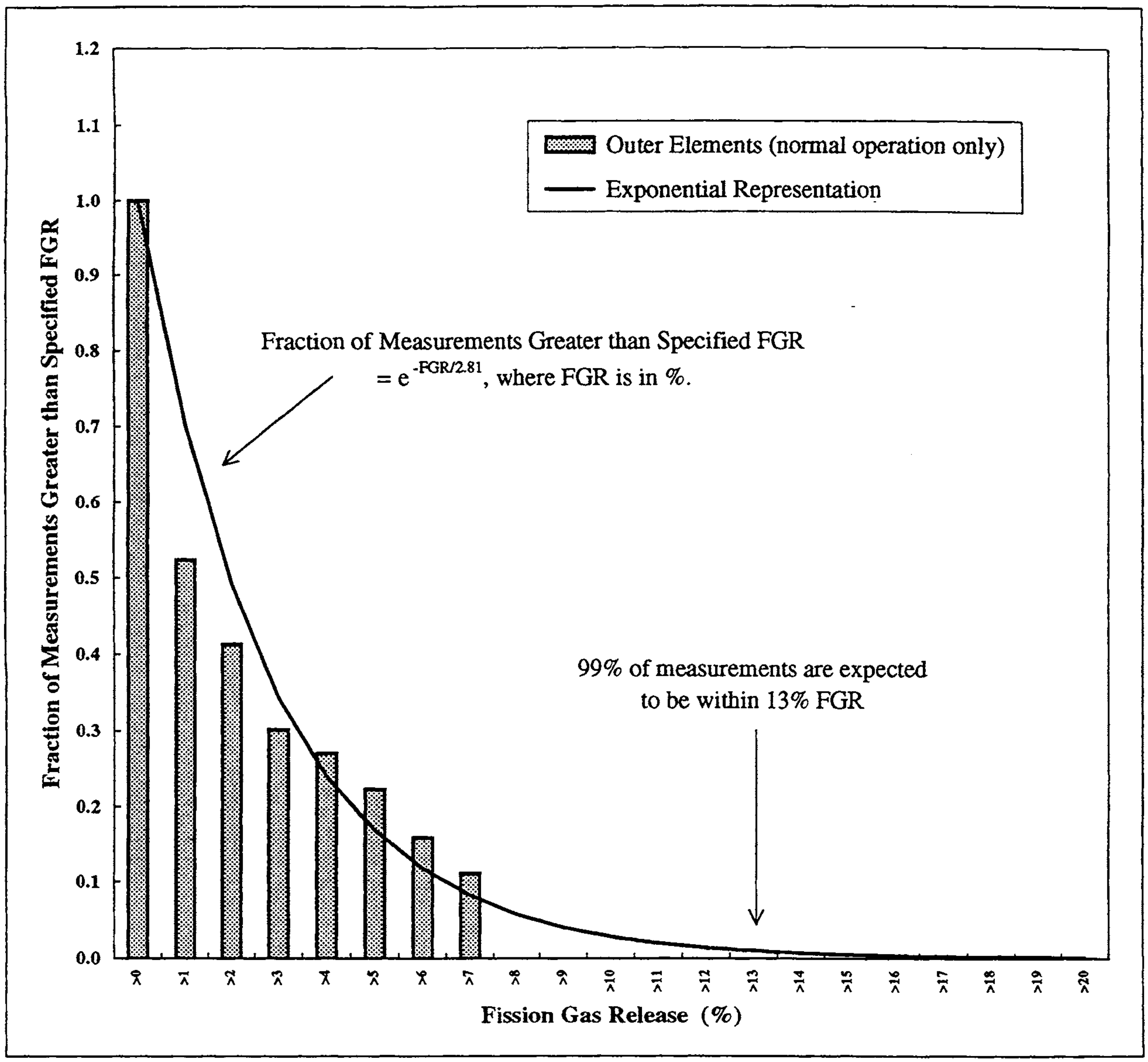
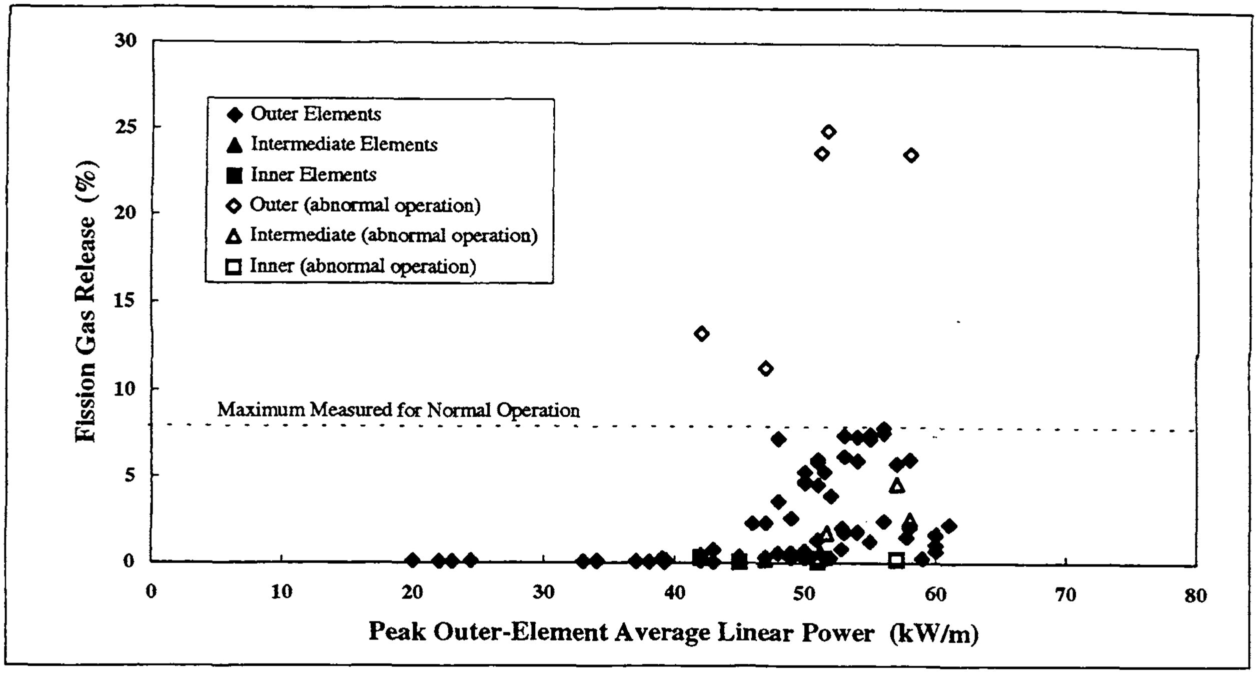
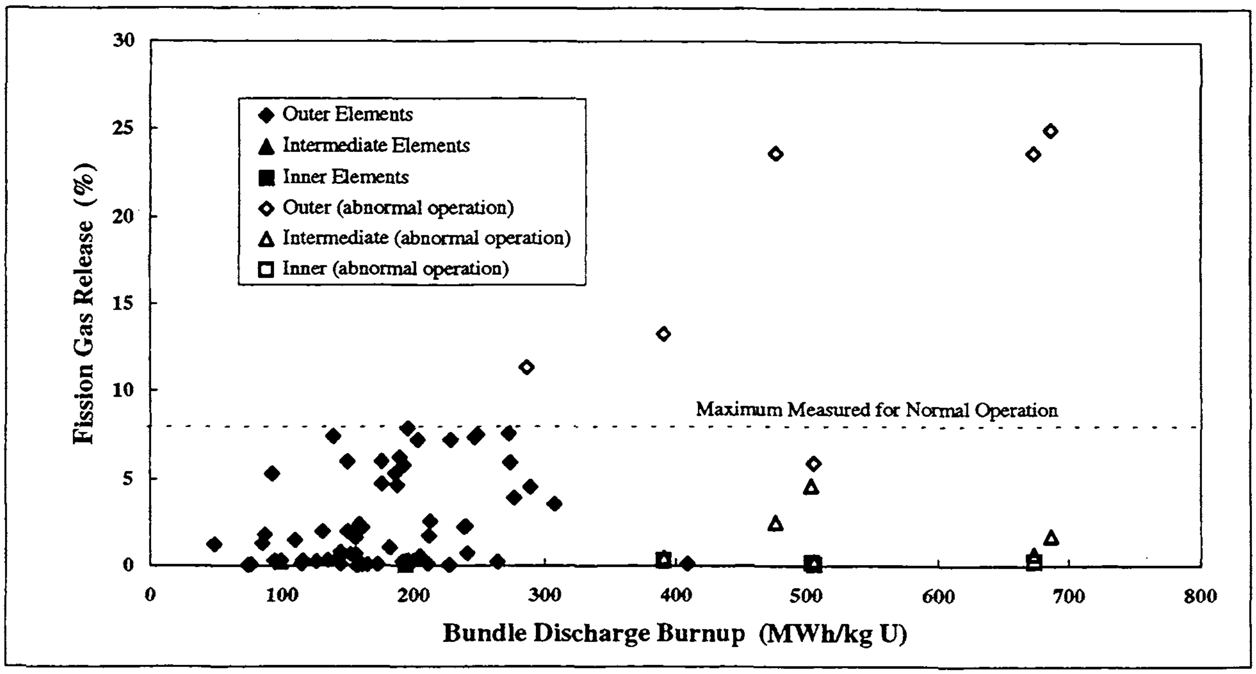


FIGURE 5: Estimation of the 99% Confidence Interval for FGR Measurements based on the Cumulative Distribution of the Measurements.



(a) Fission Gas Release vs. Peak Outer-Element Average Linear Power.



(b) Fission Gas Release vs. Bundle Discharge Burnup.

FIGURE 6: Fission Gas Release vs. (a) Peak Outer-Element Average Linear Power and (b) Bundle Discharge Burnup.