EXTENDED-BURNUP CANDU FUEL: DESCRIPTION OF DATABASE AND COMPARISONS WITH CODE PREDICTIONS

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ABSTRACT

Interest in the modelling of extended-burnup CANDU fuel has been prompted by the examination of such fuel from CANDU reactors and the development of extended-burnup CANDU fuel for the future. A study was done at AECL Research to compile a database on experimental irradiations whose outer-element burnups exceed 300 MW.h/kgU. This paper describes the elements that comprise the database and discusses analysis of fission-gas-release measurements against experimental parameters. Fission-gas-release measurements are also compared against predicted results from the MOD10 and MOD11 versions of the code ELESIM.

1. INTRODUCTION

The outer elements in CANDU power reactor fuel bundles are typically irradiated to burnups of approximately 250 MW.h/kgU (10.4 GW.d/tU). Because of operational requirements, a few fuel bundles have been irradiated for longer periods, resulting in outer-element burnups up to 780 MW.h/kgU. At the NPD reactor, a few bundles were intentionally irradiated to outer-element burnups up to 850 MW.h/kgU. Experimental fuel bundles have been irradiated in the NRU reactor to outer-element burnups as high as 640 MW.h/kgU.

Interest in the modelling of extended-burnup fuel, where the burnup exceeds 300 MW.h/kgU, has been prompted by the examination of extended-burnup fuel from CANDU reactors and the development of higher burnup fuels for the future.

A study was recently done at AECL Research to assess and qualify data from extended-burnup irradiations from experimental and power reactors. As a first step in the comparison of measured and predicted results, measured gas-release results from Post Irradiation Examinations (PIE) were compared to predicted gas releases from ELESIM(1), a FORTRAN computer program that models the behaviour of CANDU fuel under normal operating conditions. This paper describes the extended-burnup database and discusses the results of the comparison of code-calculated and measured fission-product release.

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2. ELEMENT SELECTION FOR THE DATABASE

Fuel elements from 19 CANDU-type fuel bundles with outer-element burnups greater than 300 MW.h/kgU, known power histories, measured Fission-Gas-Release (FGR), and fabrication information were included in the database. Outer elements were selected for the comparison, since these generally undergo more detailed PIE than the inner elements. Intermediate and inner elements will be included at a later stage of the development of the database.

Because of neutron flux variations within a reactor, most bundles experience circumferential, as well as longitudinal, flux variations. This variation in flux causes a corresponding variation in element burnup. Hence, to make use of measured element burnups, it was decided to use individual elements in the database, rather than bundles.

2.1 Bundle and Element Descriptions

Element powers, burnups, and FGR's are summarized in Table 1. Element powers are based on the predicted element powers from reactor-physics information.

Table 1

Summary of Bundle and Element Information

DESCRIPTION	MAXIMUM ELEMENT POWER (kW/m)	AVERAGE ELEMENT POWER (kW/m)	FISSION-GAS-RELEASE (mL @STP)	AVERAGE ELEMENT BURNUP (MW.h/kgU)
BDL-400 PY	63.7	48.8	49-58	403
BDL-403 PZ	61.0	52.5	57-67	366
BDL-411 GB	72.0	63.0	43-57	299
NPD-40/BDL-412 KE	70.0	33.6	130-143	366
NPD-40/BDL-412 KF	68.7	35.8	146-151	398
BDL-416 AAW	70.2	42.2	100-177	640
BRUCE F04857	49.4	35.1	10-30	547
BRUCE J24518	23.8	20.7	0.3-0.8	462
BRUCE J24533	50.2	37.0	111-120	778
BRUCE J24546	50.3	35.7	104	761
BRUCE J64703	40.8	26.0	33-40	442
DME-191 ELEMENT 9111	53.1	33.3	73	702
DME-195 ELEMENT 9507	56.8	40.2	36	417
DME-195 ELEMENT 9530	60.4	39.4	44	532
DME-195 ELEMENT 9549	60.4	39.4	52	532
NPD-56 DG035	36.3	16.7	57-84	806
NPD-56 DG063	32.7	18.2	1-2	832
NPD-56 DG111	34.4	17.9	1-2	848
NRU-229 JC	60.2	39.9	45-57	643

BDL-400 Bundle PY & BDL-403 Bundle PZ. These two bundles were 36-element, prototype BLW-PB (Boiling Light Water Plutonium Burner) bundles irradiated in NRU. Both were graded UO_2 enriched bundles with an enrichment of 1.8 wt% U-235 for the 18 outer elements, and 3.0 wt% U-235 for the twelve intermediate and six inner elements. Pellet density was 10.60 Mg/m^3. The pellets were clad in Zr-2.5 wt% Nb sheaths, containing a CANLUB DAG-154 coating on the inside surface. The bundles were irradiated under conditions typical of present CANDU reactor designs. Boiling light-water coolant was not used.

BDL-411 Bundle GB. This bundle was a Gentilly-II prototype. It consisted of Zircaloy-4 sheathed pellets enriched to 1.7 wt% U-235 with a UO_2 pellet density of approximately 10.64 Mg/m^3. The sheaths were coated with a thick, 6-13 μm, AQUADAG ES-242 CANLUB coating. Alternate outer elements contained graphite plenums to limit the internal pressure to within 10% of the coolant pressure. The other outer elements contained no plenums and reached an estimated internal pressure of about 1.6 times the coolant pressure.

The chemical burnup for this bundle's outer elements averaged 299 MW.h/kgU, which is at the borderline between normal and extended-burnup fuel. However, it was decided to include this bundle in the database because of the amount of available information, and the chemical burnups of some elements exceeded 300 MW.h/kgU.

NPD-40/BDL-412 Bundles KE and KF. These bundles were part of an experiment that started in the NPD reactor and ended in the NRU reactor. The bundles consisted of 19 elements fueled with $(U,Pu)O_2$ or MOX (Mixed OXide) pellets with an enrichment of 3.0 wt% Pu and a density of 10.19 Mg/m^3. The pellets were clad in a 0.65 mm thick Zircaloy-4 sheath, containing a CANLUB graphite layer on the inner surface. All elements contained Zircaloy-4 plenum inserts.

BDL-416 Bundle AAW. The bundle contained 1.7 wt% U-235 enriched UO_2 fuel with various combinations of double- and single-dished pellets sheathed in Zircaloy-4, containing a DAG-154 graphite CANLUB coating on the inner surface. Pellet density was 10.55 Mg/m^3.

Bruce Bundle F04857. This bundle had a predicted outer element burnup of approximately 507 MW.h/kgU, and was discharged from Bruce NGS Unit 1. Density of the natural UO_2 pellets was 10.60 Mg/m^3. Twelve chemical burnup samples were taken from four outer elements at the top, mid-plane, and bottom pellet locations. There is considerable axial variation in the burnup of the fuel pellets for each of these elements, as well as variations between elements.

Bruce Bundle J24518. Bundle J24518 was irradiated in Bruce NGS-A Unit and discharged with a predicted outer-element burnup of 415 MW.h/kgU in 1983 April. Density of the natural UO_2 pellets was

10.59 Mg/m^3. Because of the low bundle power (predicted outer element linear powers of approximately 20 kW/m), FGR's from the elements were very low (less than 2 mL).

Bruce Bundle J24533. Bundle J24533 was irradiated in Bruce NGS-A and discharged with a predicted outer-element burnup of approximately 700 MW.h/kgU in 1983 April. Density of the natural UO_2 pellets was 10.59 Mg/m^3.

Bruce Bundle J24546. Bundle J24546 was irradiated in Bruce NGS-A and discharged with a predicted outer-element burnup of 690 MW.h/kgU in 1983 April. Density of the natural UO_2 pellets was 10.59 Mg/m^3.

Bruce Bundle J64703. This bundle was discharged from Bruce NGS-A, Unit 3 in 1986 November. It had a predicted outer-element burnup of 430 MW.h/kgU. Density of the natural UO_2 pellets was 10.74 Mg/m^3.

DME-191 Element 9111. DME stands for DeMountable Element. The elements consisted of UO_2 pellets enriched to 1.38 wt% U-235 with a density of 10.67 Mg/m^3. The inside surfaces of the sheaths used were coated with siloxane CANLUB coatings with various thicknesses, and had different Zircaloy substrate treatments.

DME-195 Elements 9507, 9530, & 9549. Elements consisted of UO_2 pellets enriched to 1.38 wt% U-235 with a density of 10.52 Mg/m^3. The inside surfaces of the sheaths used were coated with standard siloxane CANLUB coatings. Element 9507 had a different irradiation history than elements 9530 and 9549, which had the same irradiation history.

NPD-56. The NPD-56 experiment consisted of 10 production bundles, fueled with UO_2 with an enrichment of 1.4 wt% U-235. Pellet density was 10.57 Mg/m^3. These bundles were used as driver fuel in the NPD reactor to compensate for the loss of reactivity caused by cobalt bundles. The elements had an outside diameter of 15.22 mm and were clad in Zircaloy 4. Spacing and pressure tube clearance were maintained by wire wraps of 1.25 mm and 1.68 mm diameter, respectively. Specific information on three bundles from this experiment is given below.

Bundle DG035. This bundle was power-ramped in the NRU reactor after irradiation in NPD. Its estimated outer element burnup is 829 MW.h/kgU.

Bundles DG063 and DG111. These bundles are similar to Bundle DG035, except they were not power-ramped after irradiation in NPD. Their estimated outer element burnups are 827 and 840 MW.h/kgU, respectively.

NRU-229 Bundle JC. Elements consisted of UO_2 pellets enriched to 1.55 wt% U-235. Element sheaths were not coated with CANLUB. Nominal pellet density was 10.6 Mg/m^3. Its estimated outer element burnup is 668 MW.h/kgU.

3. FISSION-GAS-RELEASE OBSERVATIONS

FGR from UO_2 fuel is affected by many parameters, two of which are burnup and linear power, which define the power history of the fuel. Power histories are generally categorized as steady or quasi-constant, declining, and ramped. Typical examples of different power histories are shown in Figures 1 to 3.

Figure 4 shows a graph of measured FGR (expressed as the percentage of that produced) as a function of the outer-element burnup. Percentage FGR has been used for comparison, since it normalizes the differences between elements that have been irradiated for differing time periods. In Figure 4, no burnup threshold for enhanced FGR, or burnup enhancement of FGR, can be seen for burnups up to 850 MW.h/kgU. This observation is supported in the literature (2,3). The scatter in the results of Figure 4 is due to the variation of power levels between the elements. The tendency for FGR to decrease is due to some of the very long burnup bundles having low powers.

Figure 5 shows a graph of the percentage of FGR as a function of the average element linear power. The correlation between FGR and average element power (Regression coefficient, r=0.7) is better than the correlation between FGR and burnup (r=0.3), when bundles that underwent large power ramps at the end of their irradiation are excluded (KE, KF, and DG035). The reasoning for this is discussed below. The relationship between higher FGR and higher element powers is supported in the literature (2,3) as a power or temperature threshold.

Power ramps early in life have a minimal effect on the total FGR at the end-of-life, since only a limited gas inventory has been built up until the power changes. Power ramps late in life, however, result in large FGR's, because of the large inventory of fission-gas and higher temperatures at higher powers, which accelerate FGR. This is seen in Table 1, which shows bundle KE having a high FGR compared to bundle PZ. These bundles have similar element burnups, but despite a lower average outer-element power over the irradiation, KE has a higher FGR. A similar comparison can be made between KF and PY. As well, a comparison between DG035, and DG063 and DG111, shows that, despite similar power histories up until the end of the irradiation, the power-ramped DG035 outer elements have more than an order of magnitude greater release of fission gas.

4. COMPARISONS WITH ELESIM

4.1 Comparison With MOD10 Version of ELESIM

Figure 6 compares the predicted gas release using the ELESIM code with the measured gas release for the elements in the present database. There is significant scatter in the results, as well as a strong tendency to underpredict the FGR. This is indicated by the number of points lying under the diagonal or exact agreement line.

Steep power ramps near the end of irradiation cause an enhanced release of fission-gas contained within the UO_2 fuel. Agreement is poor for those reactor bundles and elements where this occurred. This is, in part, the result of applying a steady-state code such as ELESIM to a high power level for a short time period.

4.2 Sensitivity Study of ELESIM To Power and Burnup Variations

A sensitivity study of the current version of the ELESIM (MOD10) to burnup and linear power was carried out. Using the extended-burnup database cases, power and burnup variations of ±5% and ±10% were applied to the power histories to determine the sensitivity of ELESIM to small changes in the power/burnup history. A ±5% variation in power and burnup in the database cases caused an average variation in gas release of +26% and -22%. A ±10% variation in power and burnup in the database cases caused an average variation of gas release of +56% and -41%. The range of values varied considerably, depending on the power history. The differences between the results are summarized in Table 2.

Table 2

Variation in Predicted Fission-Gas-Release
For Variations of Power and Burnup Using ELESIM

Category	Base vs.5% Increase	Base vs. 5% Decrease	Base vs. 10% Increase	Base vs. 10% Decrease
Range of Differences	10.6% to 65.5%	-10.1% to -56.3%	21.7% to 154.1%	-19.6% to -84.9%
Average Difference	25.8%	-22.1%	55.9%	-41.1%
Standard Deviation	15.2%	11.3%	32.0%	19.0%

The study confirms that FGR is very sensitive to element power and burnup, and the accuracy of FGR modelling predictions using ELESIM is heavily dependent on the accuracy of the power history used.

4.3 Comparison With MOD11 Version of ELESIM

Recent development work on ELESIM at AECL Research has resulted in significant improvements to the code(4).** Changes to the code include the following:

**Revised code version referred to as MOD11.

1) Improved calculation of the diffusion of fission-gas, present in the grain at the beginning of the time step, to the grain boundary.

2) A correction for the calculated grain-boundary area.

3) Change to a lenticular grain-bubble shape from a spherical shape and a modified venting criteria.

4) Gaseous swelling is translated into strain throughout the irradiation, rather than when it only exceeds the remaining porosity after densification.

5) Change to the calculation of hydrostatic stress in the UO_2 pellet and the use of a lower temperature of plasticity in extended-burnup fuel.

Figure 7 compares the predicted gas release using the revised version of ELESIM (MOD11) with the measured FGR in the database. Agreement between measured and predicted gas releases is significantly improved from Figure 6. The points are more grouped and tend to lie near the diagonal line. Also, there are no longer any points that lie directly along the x-axis. Table 3 compares the linear regression results for predicted FGR as a function of measured FGR (a slope of 1 and an intercept of 0.0 would indicate exact agreement). Outliers that exist far from the diagonal line are due principally to bundles KE and KF, which underwent large power ramps late in life. Some outliers are also from bundle AAW, which has a significant circumferential variation in measured burnup and FGR between elements. The scatter from bundle AAW is also due to the use of an average burnup for twelve elements of the bundle without measured burnups, and with significant variation in FGR. Table 3 shows that there is significant improvement in slope and y-intercept for MOD11 when these three bundles are not considered.

Table 3

Linear Regression Comparison of
Current and Revised Versions of ELESIM
Compared With The Extended-Burnup Database

	All Elements Included		Without Bundles KE,KF, & AAW	
	MOD10	MOD11	MOD10	MOD11
r	0.56	0.76	0.20	0.71
m	0.277	0.519	0.158	0.792
b	10.18	20.54	13.84	9.89

where

r = regression coefficient, m = slope, b = y-intercept.

5. SUMMARY OF RESULTS

This paper described an extended-burnup CANDU fuel database and has presented information on FGR. It also described a comparison between the current version, and a revised version, of ELESIM with measured FGR's.

The extended-burnup database presently consists of 112 outer elements from bundles having known power histories, fabrication data, and FGR. In addition, measured chemical burnups have been incorporated into the database.

Based on the extended-burnup database, there appears to be no burnup enhancement of, or burnup threshold for, FGR. However, there is, as expected, a correlation between the increasing percentage FGR with increasing average linear power.

FGR release is very sensitive to variations in power and burnup. Variations of 5% and 10% in power and burnup of the database cases caused ELESIM-predicted variations in FGR of approximately 25% and 50%, respectively. Accurate modelling of FGR requires an accurate estimate or measurement of element powers and burnups.

The current version of ELESIM (MOD10) tends to underpredict fission-gas release for extended-burnup CANDU fuel, with some database cases being significantly underpredicted. A revised version of ELESIM (MOD11) provides significantly better predictions of the FGR of extended-burnup fuel.

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7. REFERENCES

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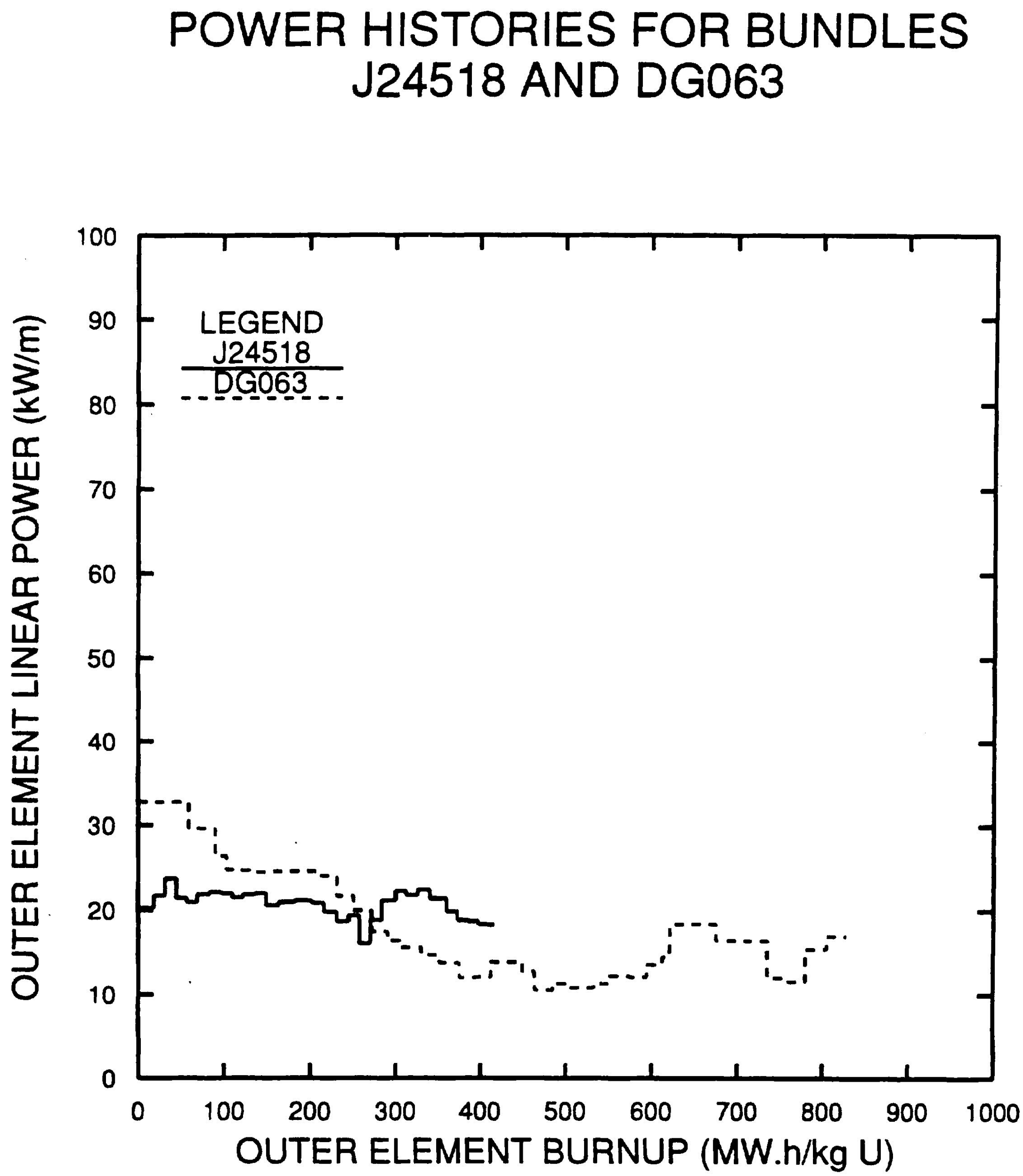

Figure 1 - Quasi-constant power history extended-burnup irradiation.

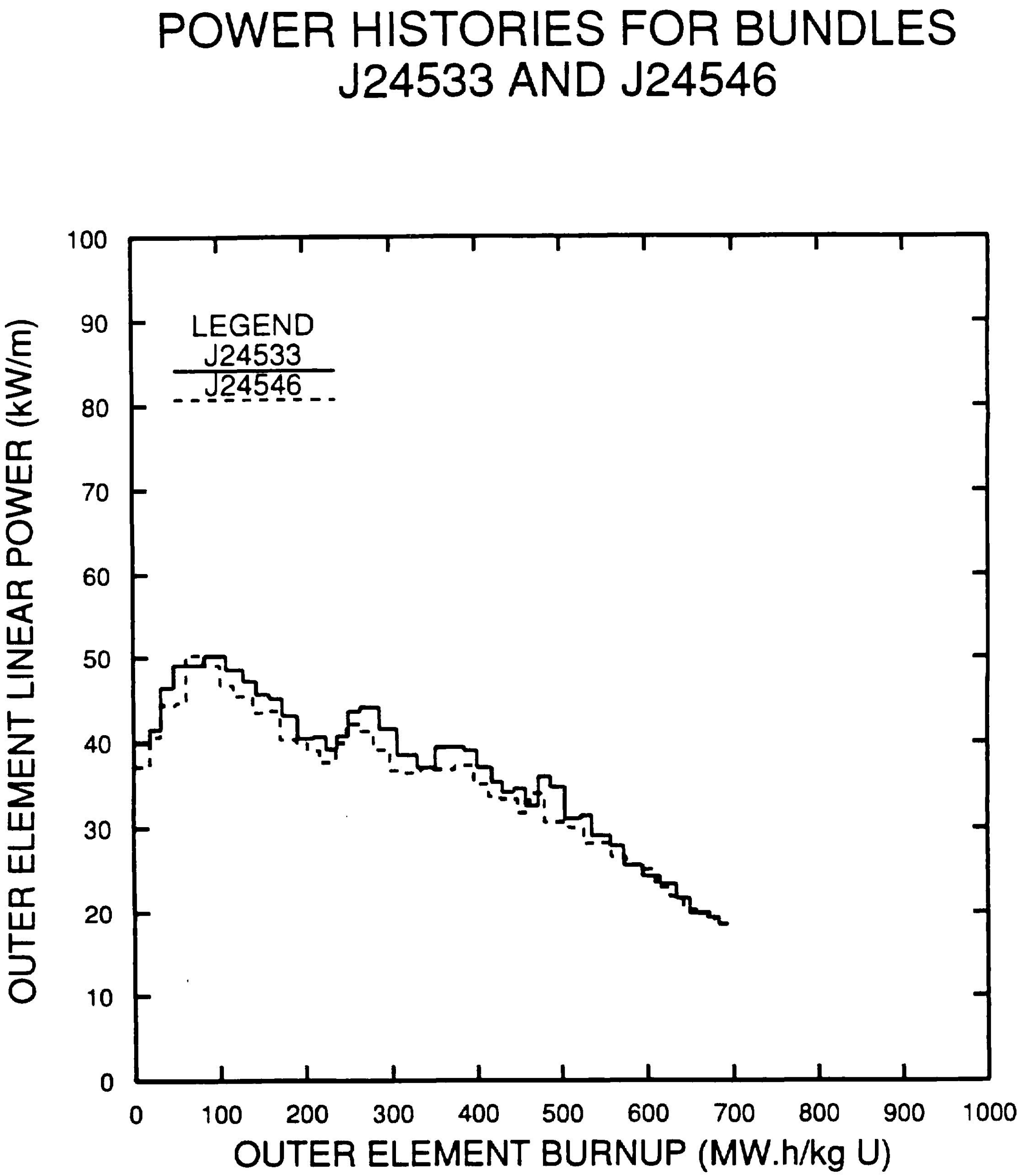

Figure 2 - Declining power history extended-burnup irradiation.

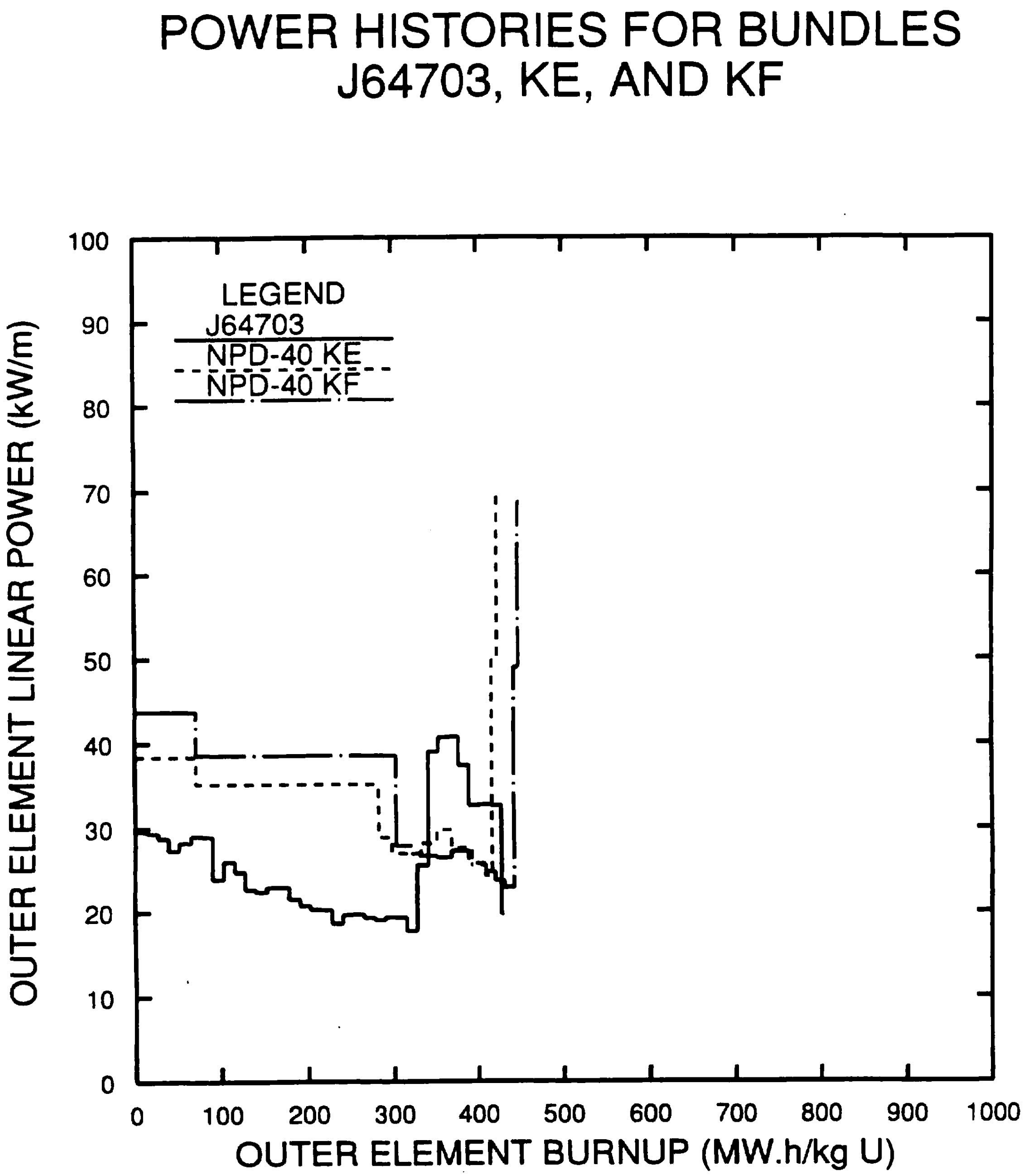

Figure 3 - Quasi-constant power history extended-burnup irradiation followed by a power ramp.

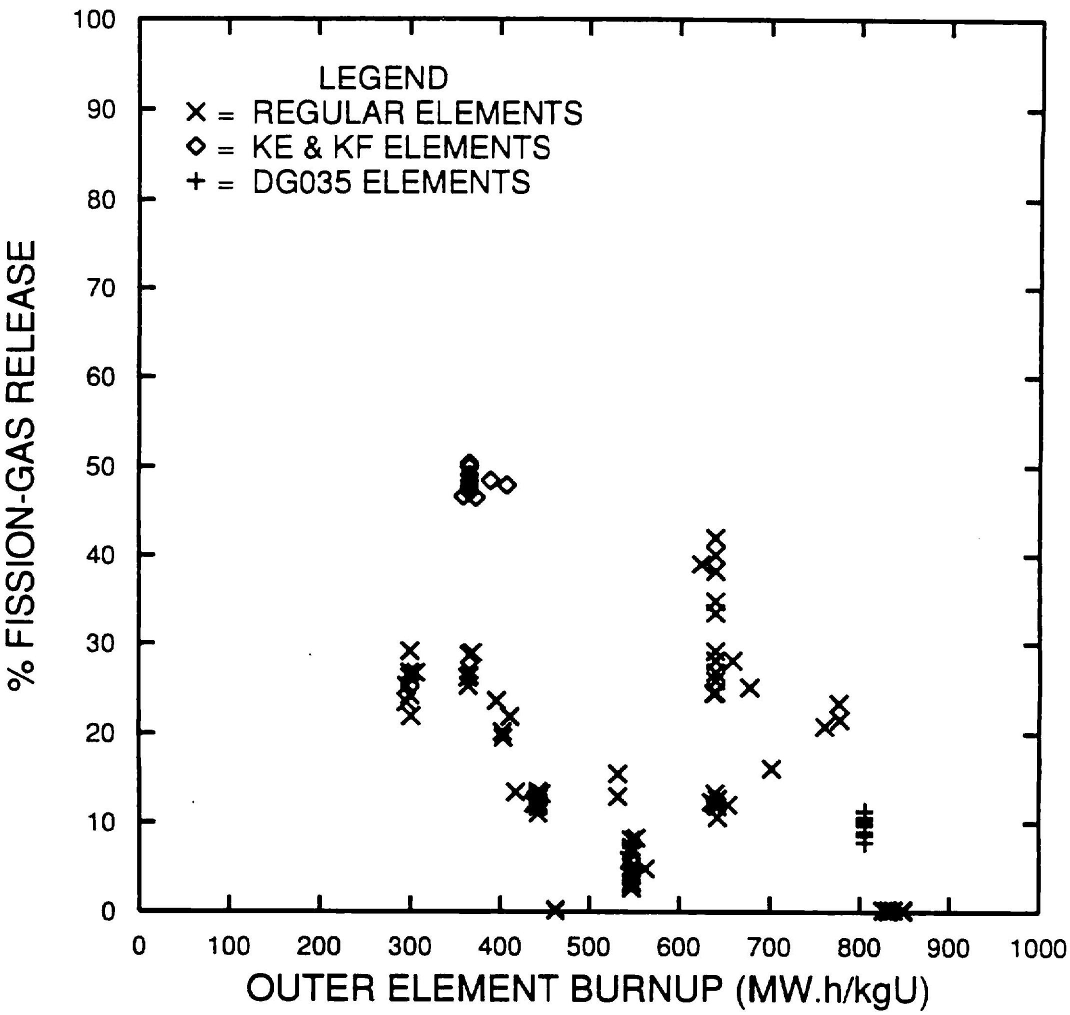

Figure 4 - Percentage fission-gas release as a function of burnup.

(Note: REGULAR ELEMENTS in legend refers to all elements except those from bundles KE, KF and DG035.)

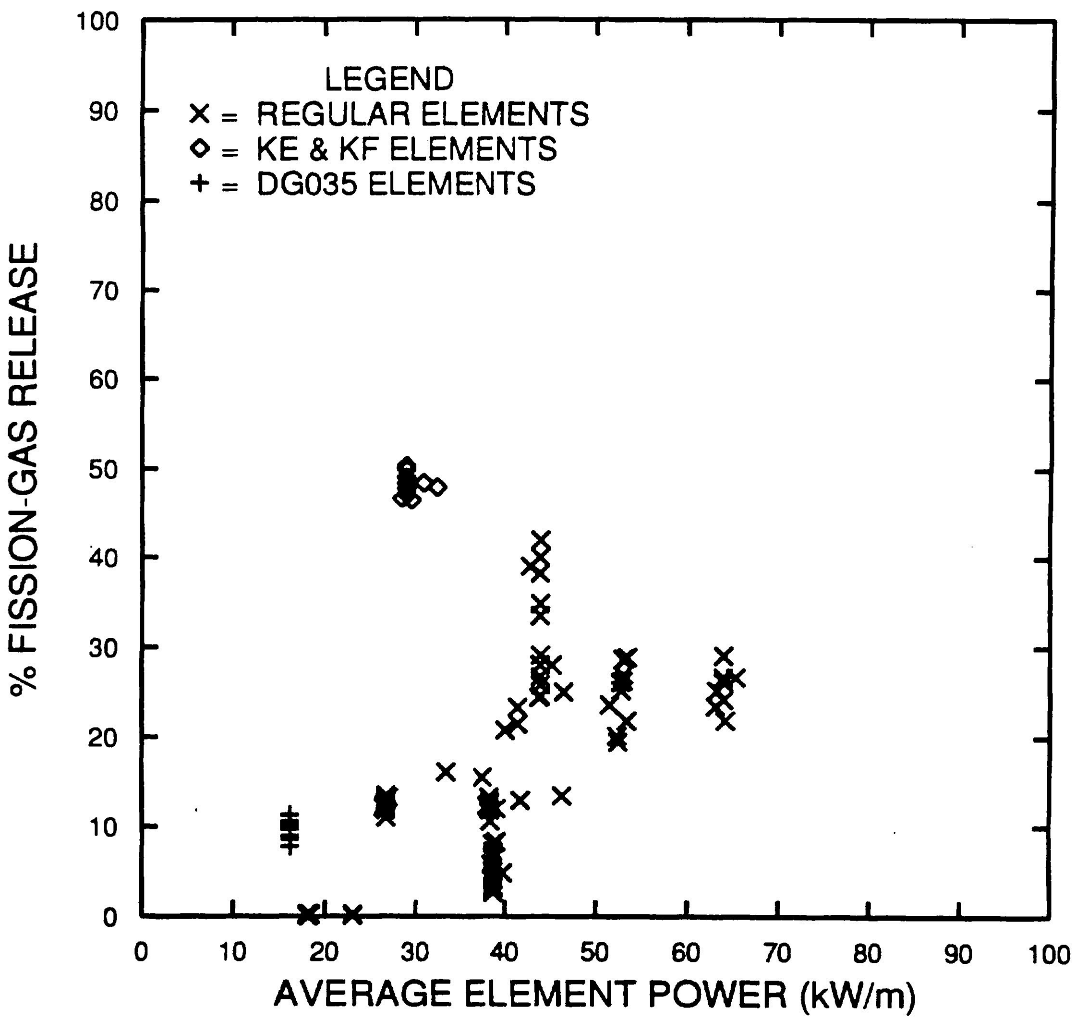

Figure 5 - Percentage fission-gas release as a function of average linear power.
(Note: REGULAR ELEMENTS in legend refers to all elements except those from bundles KE, KF and DG035.)

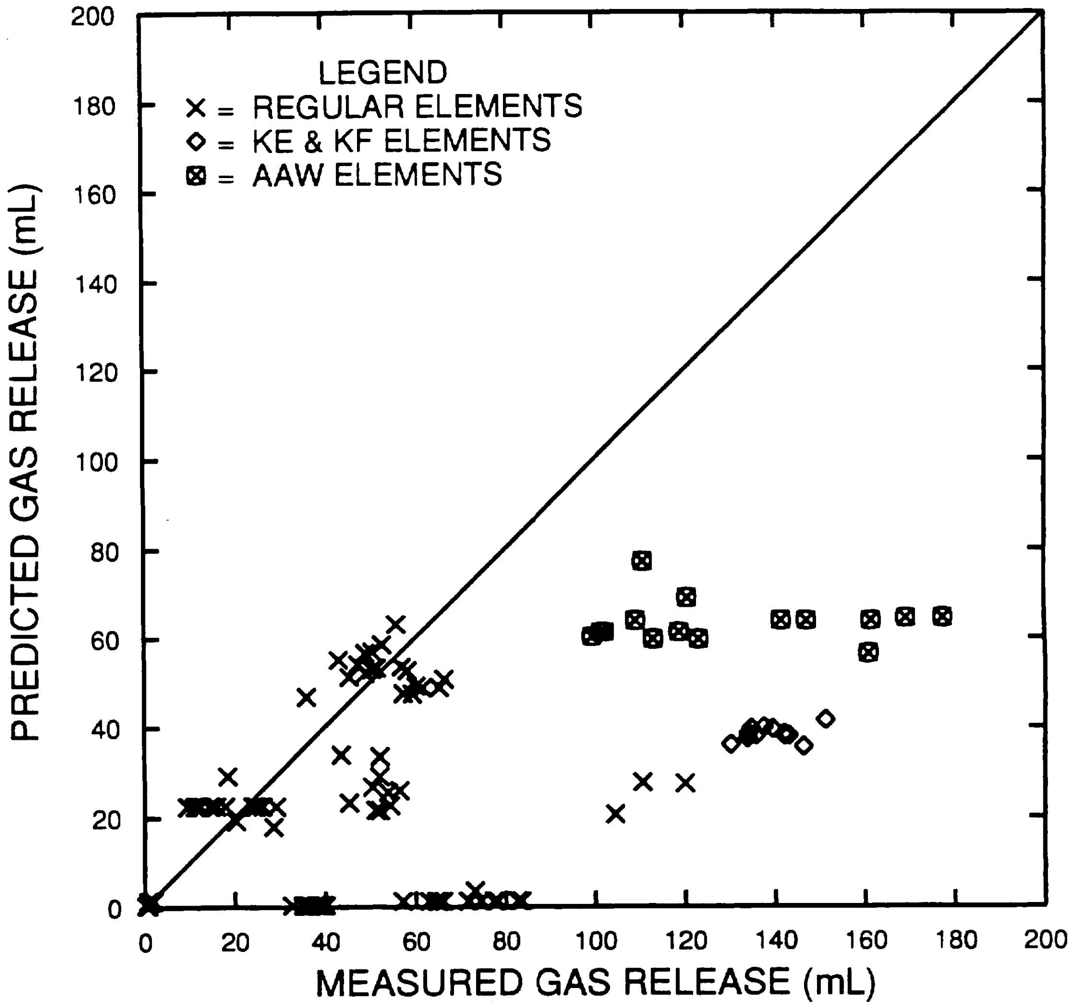

Figure 6 - Comparison of the current version of ELESIM prediction of fission-gas release against the extended burnup data.
(Note: REGULAR ELEMENTS in legend refers to all elements except those from bundles KE, KF and AAW.)

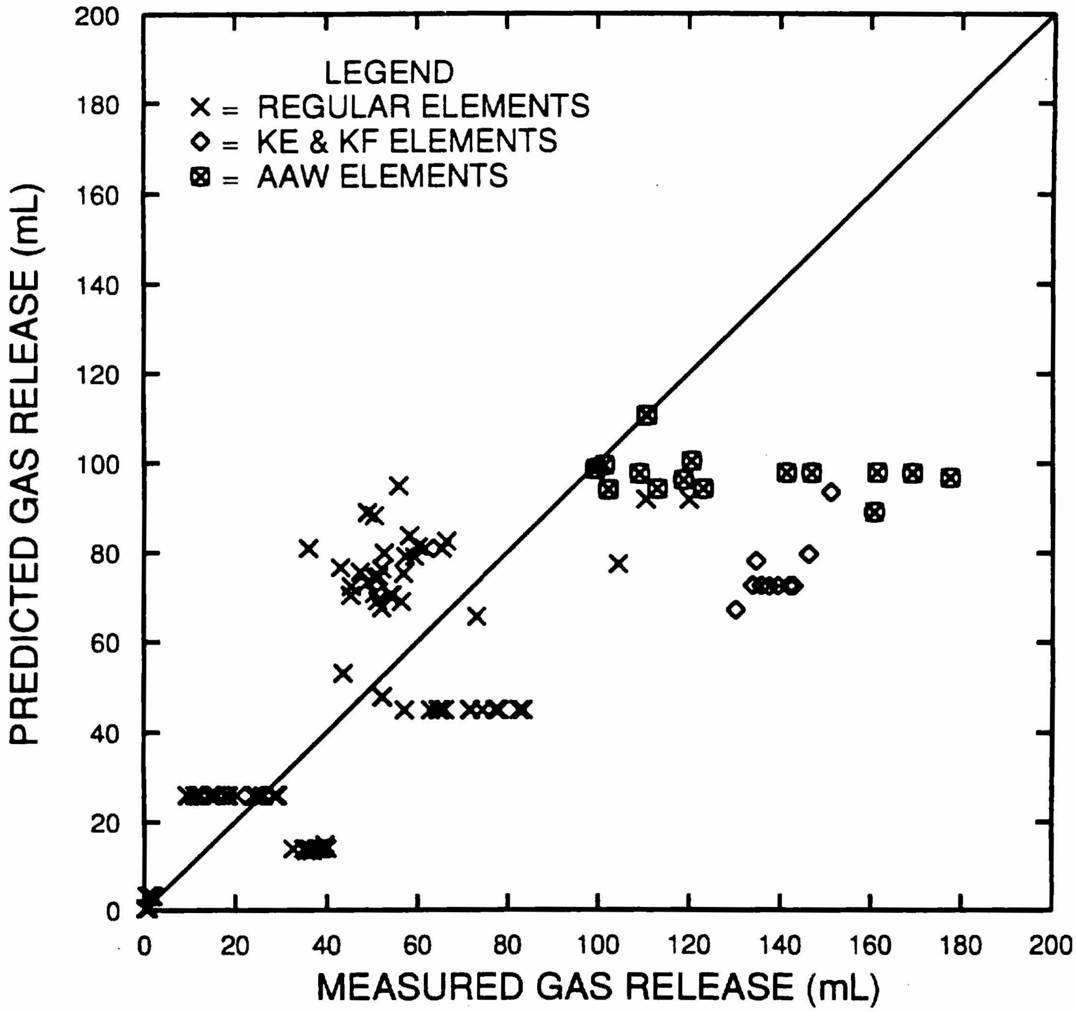

Figure 7 - Comparison of the revised version of ELESIM prediction of fission-gas release against the extended-burnup data.
(Note: REGULAR ELEMENTS in legend refers to all elements except those from bundles KE, KF and AAW.)