Load-Following Performance and Assessment of CANDU Fuel ■

by M. Tayal*, M. Floyd*, D. Rattan*, Z. Xu*, A. Manzer*, J. Lau*, and E. Kohn‡

ABSTRACT

Load following of nuclear reactors is now becoming an economic necessity in some countries. When nuclear power stations are operated in a load-following mode, the reactor and the fuel may be subjected to step changes in power on a weekly, daily, or even hourly basis, depending on the grid's needs.

This paper updates the previous surveys of load-following capability of CANDU® fuel, focusing mainly on the successful experience at the Bruce B station. As well, initial analytical assessments are provided that illustrate the capability of CANDU fuel to survive conditions other than those for which direct in-reactor evidence is available.

1.0 INTRODUCTION

In a world of diminishing fossil fuel reserves and fluctuating energy costs, the nuclear component of electrical energy is increasing in several countries. Nuclear power plants not only cater to the base-load demand, but are also expected to contribute to meeting the peak power demands of individual grid systems. When the thermal output of a nuclear reactor is adjusted to match the grid's demand, the reactor is said to "load follow". When nuclear power stations are operated in a load-following mode, the reactor and the fuel may be subjected to step changes in power on a weekly, daily, or even hourly basis, depending on the grid's needs.

Previous experience with operating CANDU reactors in the load-following mode using 37-element bundles has been good, with the bulk of it coming from the Bruce B Nuclear Generating Station. This paper describes (a) the Bruce B experience, and (b) analytical assessments that show that CANDU fuel is also capable of withstanding load-following conditions other than those for which direct in-reactor evidence is available.

One consideration in a postulated load-following operation is its effect on the performance and integrity of the fuel bundle. There have been previous compilations of load-following performance of CANDU fuel [References 1 to 5]. This paper provides a current summary, primarily featuring information from the Bruce B power station and from initial analytical assessments that cover operational conditions that are beyond the existing database. For completeness, a summary of previous surveys is also given,

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* Atomic Energy of Canada Limited, Ontario, Canada.

‡ Ontario Power Generation Incorporated, Ontario, Canada.

covering KANUPP, Wolsong, Embalse and Point Lepreau stations as well as experimental irradiations at Chalk River Laboratories (CRL).

2.0 BRUCE B EXPERIENCE

For 9 months during 1986, the 3 commissioned reactors in the Bruce B station in Canada performed extensive load following. The station had recently been commissioned, with Unit 6 being the first to be declared "in-service" in 1984. The frequency, duration, and nature of these manoeuvres varied considerably. Typically, the frequency varied from 0 to 3 manoeuvres per week, with the duration of reduced power being approximately 8 h. The station was operated with 19 cycles of "deep" load following (power reduction ≥40%), plus up to 65 comparatively shallower cycles (between 0% and 40% reduction in power) and 11 trips. The trips occurred mainly during the start-up period of Unit 7. If the trips are counted, the above manoeuvres add up to a total of 95 cycles.

During the load-following period, the fission-product levels in the coolant increased in Bruce B Unit 6—see Figure 1. Seventeen "new" defects were detected in the three commissioned units of the station. In comparison, at any given time, 3 units of the Bruce B reactor contain about 19 000 fuel bundles and about 700 000 fuel elements. Thus the failures represent a very small fraction of the total bundles in the core. Each case of fuel failure was thoroughly followed up, and its root-cause was determined in the spent fuel bay—and in hot cells in some cases. The investigations revealed that fifteen of the failures were caused by debris fretting. The remaining 2 failures were caused by manufacturing flaws (porous end caps). Thus although fuel failures were detected during the load-following operation in Bruce B, their root causes were not directly related to the reactor power manoeuvres. The above operating experience from Bruce B is very positive.

3.0 OTHER IN-REACTOR EXPERIENCE [References 1 to 5]

Load-following operations have also been conducted at the KANUPP reactor in Karachi, Pakistan (about 90 cycles [1]), and at the Embalse reactor in Argentina (about 30 cycles [2]). In addition, in CANDU reactors such as the Point Lepreau reactor, fuel is frequently exposed to repeated power cycles in the vicinity of liquid zone controllers (LZCs) because of fluctuations in the LZC levels brought on by specific refuelling patterns. Some fuel failures have been reported; however, fuel performance under the above conditions has not been followed up by the same degree of post-irradiation fuel examinations as it has been for the Bruce B fuel, as noted above. Hence the effect of power cycles on detailed fuel integrity parameters in these reactors is not currently known with full confidence.

There have been 3 sets of power cycles in the Wolsong 1 reactor in South Korea [1]. In 1983, about 40 cycles of about ±4% were applied. In 1984, about 1200 cycles of 1% to 4% were applied. In 1986/87, about 30 cycles of up to 30% were applied. The above cycles did not cause any fuel failures. Similarly, the following experimental irradiations

[3] at the CRL showed no fuel failures during repeated power cycles: X-218 (490 cycles, 0-40 kW/m), X-411 (95 cycles, 0-70 kW/m), and U-900 (up to 103 cycles, power reduction >25%). The available reports do not indicate any refuelling power-ramps in these CRL irradiations. Reference 4 provides further details on previous load-following experience.

Reference 5 reviews fission-gas releases in fuel that had load-followed in the Bruce B reactor. The survey concludes that load following does not affect fission-gas release. The raw data from Reference 5 has recently been independently reanalyzed, and the same conclusion was reached

4.0 ADDITIONAL DUTY CYCLES

The accumulated Bruce B experience is positive, but it does not include all load-following scenarios. For example, the average residence period for natural-uranium fuel in a CANDU 6 reactor is about 250 full-power days (FPDs) in the inner core and about 270 FPDs in the outer core. The corresponding maximum residence periods are about 330 and 700 FPDs respectively. Therefore, during daily load following, natural-uranium fuel in a CANDU 6 reactor can be expected to encounter a few hundred power cycles—considerably more than the Bruce B cycles, noted above. Hourly load following would subject the fuel to even larger number of cycles. Similarly, load following of higher-burnup fuel will result in a larger number of cycles. The capacity factor of the reactor also influences the number of power cycles that the fuel will be exposed to at a given burnup.

Further, the nature of the refuelling power ramps differs between Bruce B and CANDU 6 reactors because of differences in their fuelling schemes. Moreover, the operating power levels also differ between Bruce B and CANDU 6 reactors. All these parameters may influence the integrity of the fuel sheath during load following.

For a full duty-cycle consisting of power cycles and refuelling ramp(s), the loads and the defect thresholds are expected to be influenced by a variety of parameters. These include the following: the severity of the refuelling ramp(s); the size(s) and number of the power cycles; the hold period(s) at low power; the power level and its history; fuel burnup; details of the fuel design; manufacturing AQL (acceptable quality level); etc.

The above parameters interact in a complex manner; hence extrapolation of the available operating and design data to other operating and design conditions requires the assistance of suitable scientifically based computer codes. Further, the worst-case scenario is not obvious by inspection; hence a number of assessments are required to cover the various possibilities.

5.0 ILLUSTRATIVE STRAIN MEASUREMENTS

The expected degree of sheath damage can be gauged by the strain cycles imposed by the power-ramps and power-cycles. Illustrative measurements are available from the In-Reactor Diameter Measuring Rig (IRDMR) at CRL. This equipment was used to measure in-reactor on-power expansions and contractions of the sheath during power-ramps and power-cycles [6].

Figure 2 shows some typical IRDMR measurements. An experimental fuel element named ACK was ramped to about 60 kW/m at a burnup of 98 MW•h/kg U–see Figure 2(a). The figure shows that the ramp resulted in an incremental hoop strain of about 1.3% at the circumferential ridge. In another Canadian experiment using strain gauges, an on-power hoop strain of 1.8% was measured at 60 kW/m in fresh fuel [7]. We note that the strains are influenced by a number of parameters, such as size of the power-ramp, prior power history, burnup, ramp rate, diametral clearance, pellet density, pellet shape, etc.

Similarly, an experimental fuel element named ABH was power-cycled in IRDMR between 30 and 58 kW/m—see Figure 2(b). This test resulted in persistent hoop strain cycles of up to 0.4%—see Figure 2(b).

For comparison, the one-cycle ductility of unirradiated sheath in an inert environment is at least 20% at room temperature. In the reactor, the sheath is subjected to many strain cycles while being exposed to a corrosive environment created by fission products such as iodine, cesium and cadmium, and while undergoing microstructural damage from fast neutrons.

Curve (a) of Figure 3 illustrates how the strain required for failure decreases with the number of cycles in inert environment [8]. For about 200 cycles, the strain-to-failure is about one order of magnitude lower than the corresponding value for one cycle. Figure 4(a) shows that increasing levels of iodine result in progressively decreasing levels of fatigue life [9]; the effect tails off at either end. About 30 Pa of iodine can reduce the fatigue life by an order of magnitude; the operating pressure of fission gas in the fuel element is a few megapascals. Figure 4(b) shows that neutron damage, too, causes the sheath to fail at considerably lower strains [10]. At the average exit burnup of CANDU 6 outer elements, the ductility reduces by about 60% from the unirradiated, as-received condition.

In view of the above sources of reductions in strains to cause failure, the measured strain levels in CANDU fuel during experiments noted above are considered significant. The actual in-reactor strains and corrodant concentrations will differ from the above measurements because of variations in power histories and in the details of the internal designs of the fuel element (such as CANLUB, pellet shape, pellet to sheath clearance, pellet density, etc.). Therefore, initial analytical assessments were done to extrapolate the available experience to other possible scenarios of power cycles.

6.0 ANALYSIS METHODOLOGY

The analytical assessments, performed to date, have focused on determining the risk to fuel integrity caused by stress corrosion-fatigue (SCF) of the sheath, arising from expansions and contractions of the pellet. The driving force was considered to be the combined influence of stress corrosion-cracking (SCC) caused by the refuelling power-ramps, plus additional damage caused by corrosion-assisted fatigue (CAF) from the power cycles. The assessments concentrate on conditions at the circumferential ridge of the sheath.

The assessments consist of 3 steps: (i) determine the static and cyclic strains and the corrosive environment applied to the sheath; (ii) determine the strength of the sheath to resist the applied loads, considering the effects of neutron damage and of corrosive environment on the strength of Zircaloy, and (iii) compare loads with strength to reach conclusion about sheath integrity. These steps are explained below through an illustrative example for the situation of Bruce B 37-element fuel that is exposed to a refuelling ramp because of a 4-bundle shift, and to weekly cycles of load following in the 100%-60%-100% power range.

Figure 5(a) shows a typical stylized power history that was used in the simulation. The refuelling occurs at 80 MW•h/kg U. The pre-ramp power is 25 kW/m, and the ramped power is 55 kW/m, for a refuelling power-ramp of 30 kW/m. The scenario is designated as Case 1a in Table 1.

6.1 Loads and Corrosive Environment

First, the ELESTRES code [11] was used to calculate the static and cyclic hoop strains. For example, Figure 5(b) shows that the refuelling ramp at 80 MW•h/kg U subjects the sheath to a hoop strain increment of about 1.25%. The power cycles subjected the sheath to varying levels of cyclic strain increments in the range of 0.1% to 0.4%. These levels of calculated hoop strains are consistent with on-power strains measured during in-reactor tests noted earlier.

Second, the FEAST code [12] was used to account for the effect of sheath bending at the circumferential ridges. For the above values of hoop strains, the FEAST code provided the values of radial, axial, and shear strains generated by bending near the circumferential ridges. A typical example is given in Reference 1. It shows that at the circumferential ridges, the sheath is subjected to significant levels of stresses in hoop, axial and radial directions. Thus the stress–strain system is highly multiaxial.

Third, to be able to compare calculated multiaxial strains to uniaxial test data, Sines' law [13] was used to combine the above multiaxial components of strains into an equivalent value of uniaxial strain. Typically, for the conditions analyzed, the equivalent uniaxial strains were about 30% higher than the hoop strains noted above.

Fourth, ELESTRES also yielded the fission-product concentration (FPC) in the pellet-sheath gap. Figure 5(c) shows that the FPC increases with burnup and power, and varies between 0 and 150 $\mu g/mm^2$ of the sheath surface. This quantifies the corrosive environment in the pellet-to-sheath gap.

Because of the CANLUB layer, the corrodant concentration at the inner surface of the sheath will be lower than the above value. The impact of this effect in elevating the defect threshold is covered in the next section.

6.2 Sheath Strength

The sheath strength was determined by starting with the fatigue curve of unirradiated Zircaloy [8], the "S-N" curve. This is shown as curve (a) in Figure 3. The portion of curve that is between 200 cycles and 100 000 cycles is based on measured data. To obtain the full range of required cycles, the curve was extrapolated to one cycle based on AECL's specifications for the ductility of as-fabricated Zircaloy. This yielded the point labelled "A" in Figure 3.

Next, the S-N curve was adjusted to account for the effects of corrosive environment [9] and neutron damage [10] at various burnups. To achieve this, the CAFÉ defect thresholds [14] for thick-CANLUB sheaths at various burnups were simulated using the ELESTRES and FEAST codes. For example, at 80 MW•h/kg U, the defect threshold is 23.5 kW/m for power-ramp (ΔP) and 59.2 kW/m for ramped power (P_{max}). The CAFÉ correlation requires both thresholds to be exceeded in order to cause SCC failure. For the pre-ramp power of 25 kW/m, both defect thresholds are reached when the power is increased to 59.2 kW/m. Thus for this initial power, failure occurs at a power-ramp of 34.2 kW/m, which is well in excess of the ΔP defect threshold of 23.5 kW/m noted earlier.

Accordingly, the ELESTRES code was used to simulate a power increase from 25 to 59.2 kW/m at 80 MW•h/kg U. The simulation showed that this ramp results in an incremental hoop strain of 1.65%. Using FEAST simulations of multiaxial elastic-plastic stresses/strains at the circumferential ridge, and Sines' law, the equivalent uniaxial strain increment is about 2%. This is plotted in Figure 3 as point "B", and represents the equivalent uniaxial strain that causes SCC failure in thick-CANLUB sheath in one cycle at 80 MW•h/kg U.

Next, the full S-N curve for the burnup of 80 MW•h/kg U was determined by drawing a curve parallel to curve (a), but starting at point "B". Thus the protection provided by thick CANLUB against SCC was carried through to CAF as well.

These steps were repeated for a number of burnups; for example, curve "b(ii)" represents a burnup of 100 MW•h/kg U. The above process gave a family of curves that define the fatigue behaviour of thick-CANLUB sheaths under conditions of corrosive fission products and neutron damage.

6.3 Comparison of Strength versus Load

The loads change with burnup (Figure 5b), as does the strength (Figure 3). To account for these variations, the Palmgren-Miner law [15] was used to calculate the cumulative damage from the refuelling power ramp as well as from the power cycles. Thus for any given strain cycle shown in Figure 5(b), the corresponding fatigue life (N) was obtained from curve (b) of Figure 3.

At any strain level, the ratio of applied number of cycles (n) to the life (N) at that level gives the relative damage at that strain level. The relative damages (n/N) are calculated for all applied strains. The sum of all the relative damages ($=\Sigma n/N$) gives the cumulative damage from all applied cycles. This is sometimes also called life fraction.

The operational limit is reached when the cumulative damage is equal to 1. This is also called the 100% level. Note that cumulative damage of 100% takes the fuel to the defect threshold, i.e. puts the fuel at 1% probability of defect.

For the above illustrative example of weekly load following in the Bruce B reactor, Figure 6(a) shows the load conditions and compares them with the SCC defect thresholds for power-ramp (ΔP) and ramped-power (P_{max}). Figure 6(b) shows that the refuelling ramp alone uses up 45% of the sheath life. The weekly cycles use up another 16% of sheath life, for a cumulative damage of 61%.

As a sensitivity study, Figure 6(b) also illustrates the situation when the refuelling power ramp is assumed to be coincident with reactor power increase from the load-following cycle. This scenario is labelled as Case 1b in Table 1. As noted earlier in Section 6.0, the nominal pre-ramp power at 80 MW•h/kg U is 25 kW/m. The low part of the load-following cycle reduces this by 40%, i.e. to 15 kW/m. In the sensitivity study, it is assumed that when the bundle is shifted to its high-power axial position in the channel, the reactor is also simultaneously returned to 100% power. Thus the bundle's outer element is ramped from 15 kW/m to 55 kW/m, for a net power-ramp of 40 kW/m. The cumulative damage from this power-ramp plus the power-cycles from load following is now 73%.

From the above calculational results, high-power Bruce B fuel is expected to survive SCF from weekly load-following, until the discharge burnup of natural-uranium 37-element fuel is reached. In comparison, the Bruce B station was operated in the load-following mode for 9 months in 1986, and no failures were attributed to SCF.

7.0 ASSESSMENT RESULTS

Six scenarios were analyzed using the above methodology, as listed in Table 1. They cover situations postulated for Bruce B and CANDU 6 reactors, and include refuelling power ramps, envelope power operation, weekly cycles, and daily cycles. All cycles were considered to be in the 100%-60%-100% power range. Cases 1b and 3b refer to the "sensitivity studies" of cases 1a and 3a respectively, as described above in Section 6.3.

The detailed results are given in Table 1. Depending on the loading cycle, cumulative damage varies from 50% to 73% in the Bruce B reactor, and from 48% to 89% in the CANDU 6 reactor. The highest damage (89%) corresponds to daily load following under conditions of envelope power; this subjects the fuel to most cycles among the cases analyzed. The cumulative damage is less than 100% for all 6 cases analyzed.

8.0 DISCUSSION

The above results suggest that fuel in Bruce B and CANDU 6 reactors would survive the analyzed combinations of power histories and cycles. When combined with refuelling power ramps, load following can significantly reduce operating margins. If needed, the performance margins can be increased by a variety of options such as reduction of sheath stresses by using improved shapes of pellets and modified clearances, and reduction of corrodant concentration by employing the CANFLEX® fuel design to reduce the element ratings.

Furthermore, CANDU stations are equipped to deal with fuel failures should they arise. First, different regions of the core impose different loads on the fuel bundle, and at the same time, one would also expect a distribution of strength in as-built fuel. Therefore, only some fuel, if any, would initially exceed the defect threshold. The few initial fuel failures will give the operator advance warning, and time to react and contain the situation. Second, most CANDU stations are equipped with gaseous fission-product (GFP) and delayed neutron (DN) systems to detect and locate failed fuel. Monitoring will help detect early indications of impending problems, if any, and help prevent the situation from overwhelming the GFP–DN systems. Third, the suspect and failed fuel, if any, can then be removed promptly by the on-power fuelling system. Fourth, the coolant purification system can be used to clean up the released activity, if any. The degree of these capabilities differs from one CANDU station to another, e.g. DN vs. GFP, differences in fuel operating margins, etc. Thus, even if load following beyond the existing database should lead to erosion of fuel performance margins, the above features can help deal with the consequences.

9.0 CONCLUSIONS

Operational feedback from 3 Bruce B reactors shows no evidence of fuel failure from SCF for up to 3 reactor power manoeuvres per week for 9 months. Fuel irradiation experience from the research reactors is also encouraging. Initial analytical assessments for SCF show that fuel would survive more frequent load-following operation, albeit with reduced margins to failure. Thus CANDU fuel continues to show good performance in base-load and in load-following modes of operation.

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Table 1: Cumulative Damage for the Analyzed Cases

Case	Description	Burnup at Ramp (MW•h/kg U)	Power Ramp (kW/m)	Ramped Power (kW/m)	Number of Cycles	Cumulative Damage
1a	Bruce B: regular refuelling (bundle shift) power ramp + weekly cycles.	80	30	55	51	0.61
1b	Same as case 1a above, plus the refuelling power ramp is coincident with reactor power increase from the load-following cycle.	80	40	55	51	0.73
2	Bruce B: nominal design envelope power + weekly cycles.	Not Applicable	Not Applicable	57	35	0.50
3a	CANDU 6: regular refuelling (bundle shift 1-9) power ramp + weekly cycles.	41	38	49	60	0.48
3b	Same as case 3a above, plus the refuelling power ramp is coincident with reactor power increase from the load-following cycle.	41	42	49	60	0.48
4	CANDU 6: refuelling power ramp (bundle shift 2-10) + weekly cycles.	100	14	40	62	0.57
5	CANDU 6: envelope power + weekly cycles.	Not Applicable	Not Applicable	50	40	0.50
6	CANDU 6: envelope power + daily cycles.	Not Applicable	Not Applicable	50	282	0.89

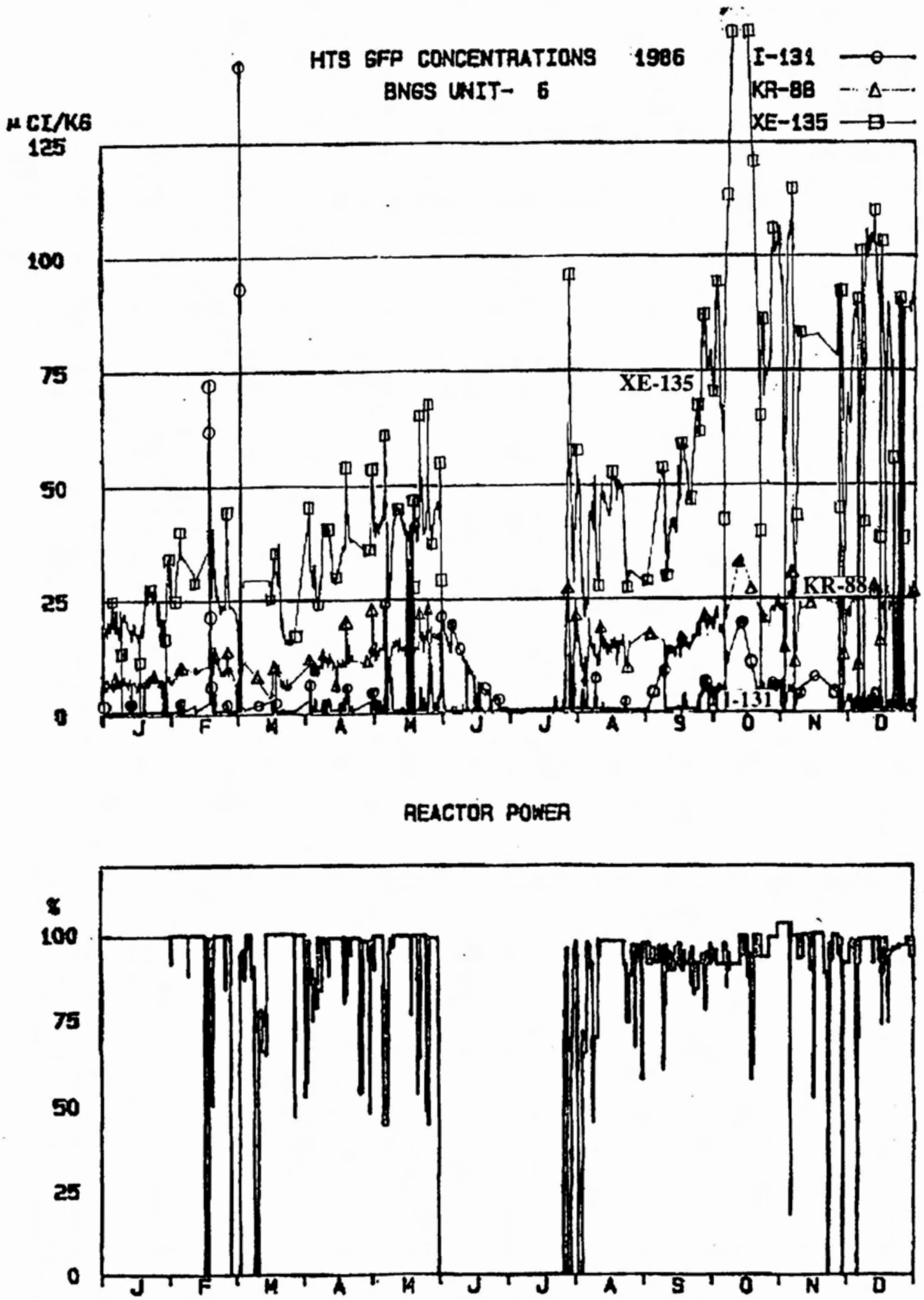
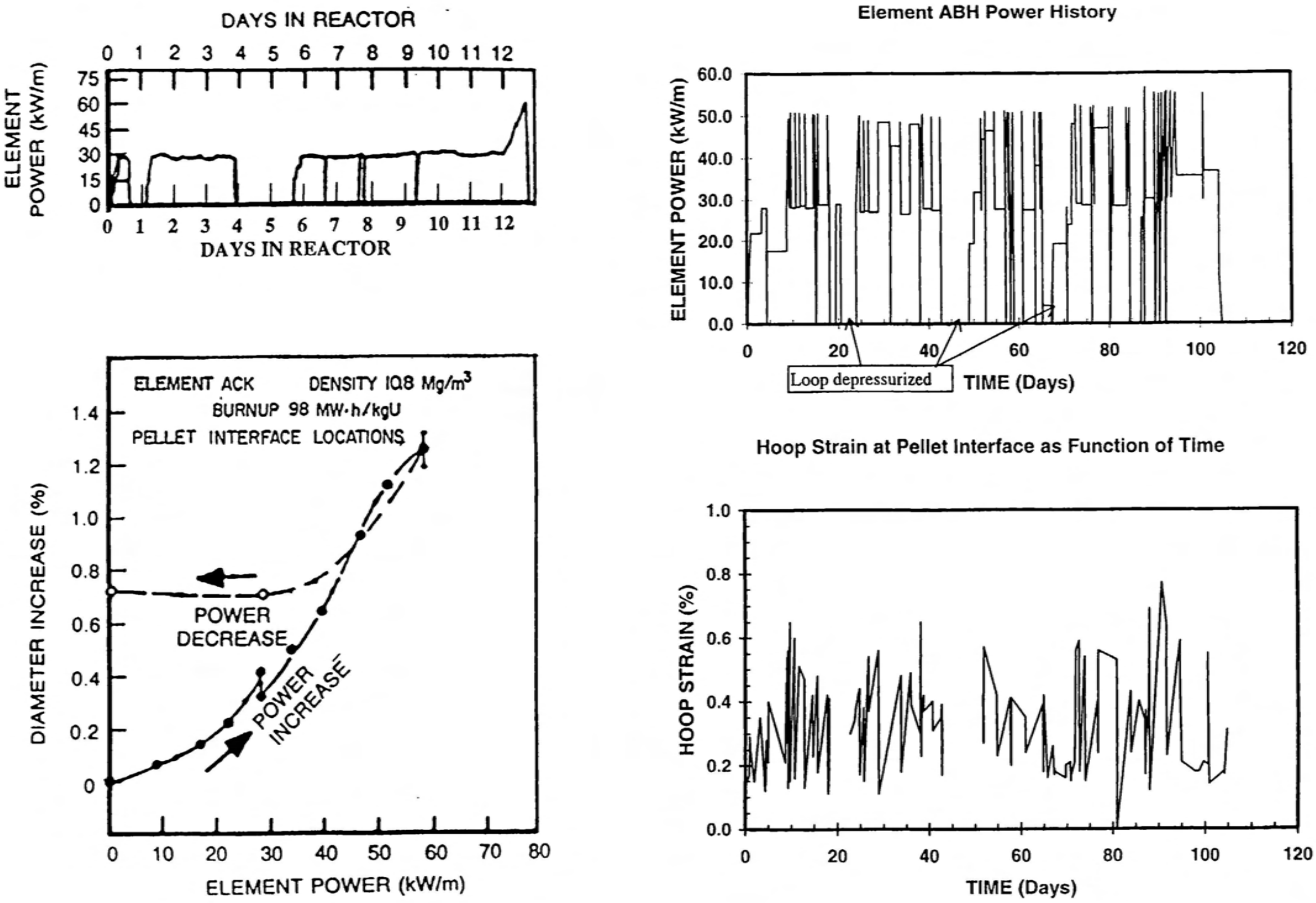


Figure 1: 1986 Bruce B Unit 6 Concentrations of ^{131}I, ^{88}Kr and ^{135}Xe in Coolant (μCi/kg, top) and Reactor Power (bottom)



(a) Power Changes

(b) Power Cycles

Figure 2: Incremental and Cyclic Sheath Strains during Power Changes
(Source: Fehrenbach et. al, 1982)

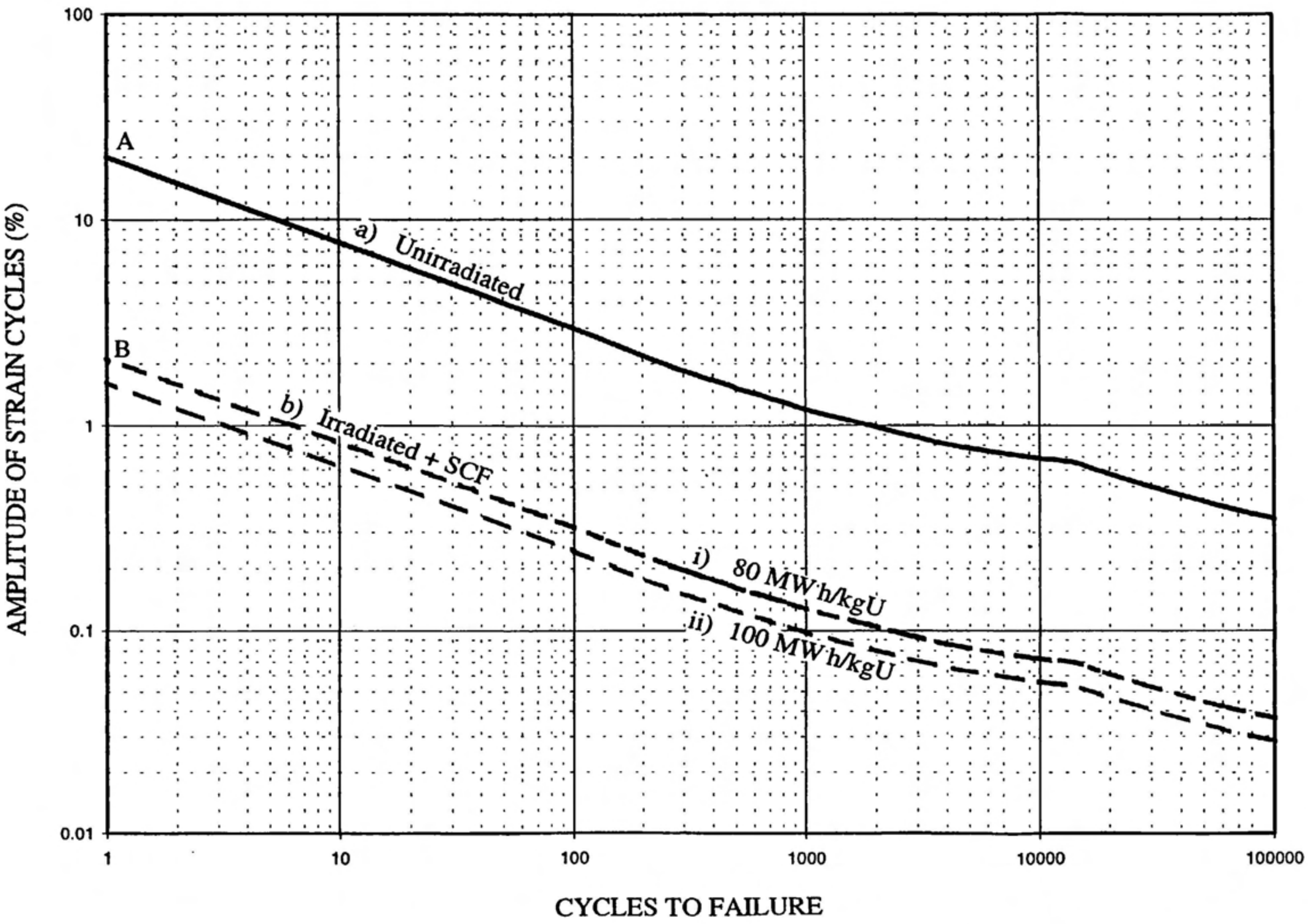
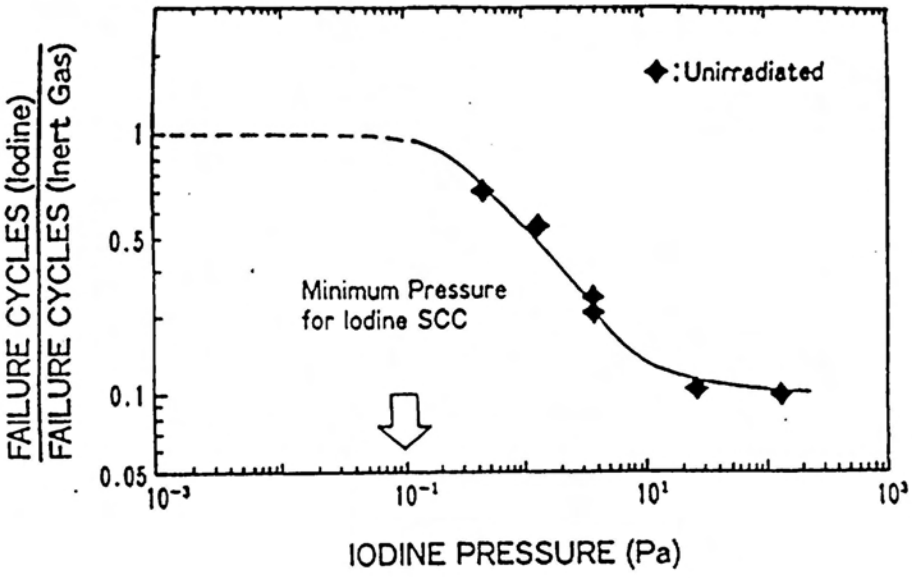
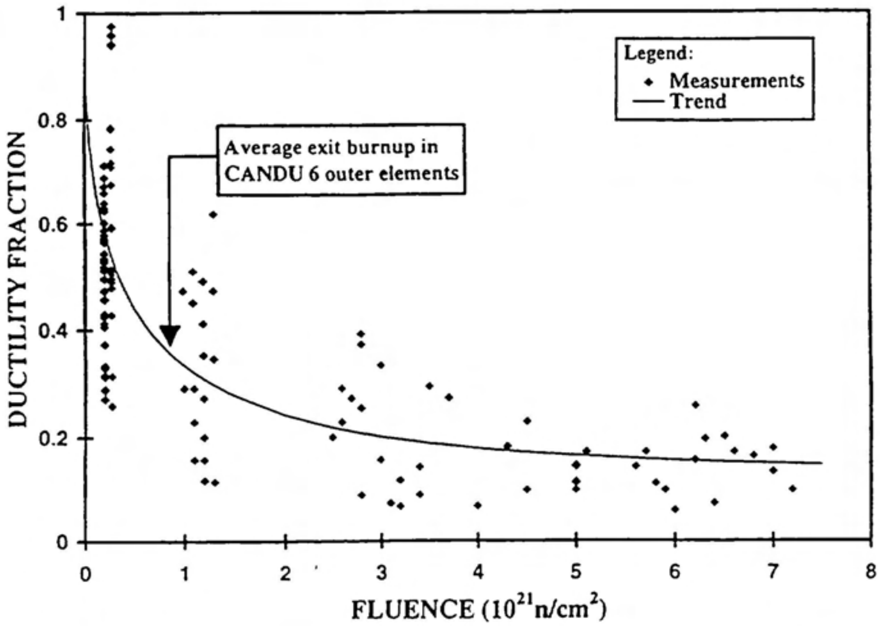


Figure 3: Effect of Corrosive Environment and Irradiation on the Fatigue Life of Zircaloy

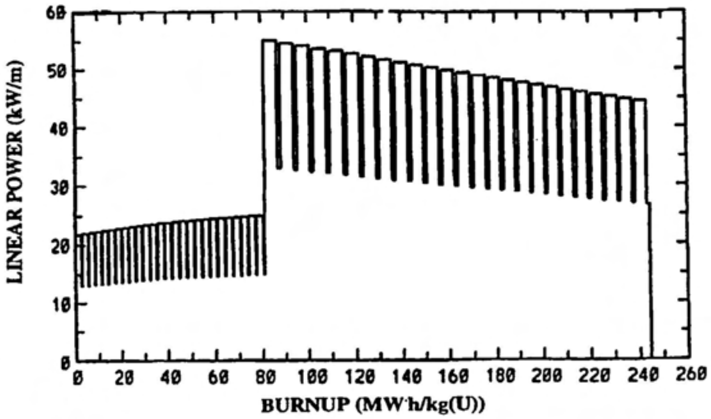


(a) Effect of Iodine Concentration
(Source: Nakatsuka et. al, 1991)

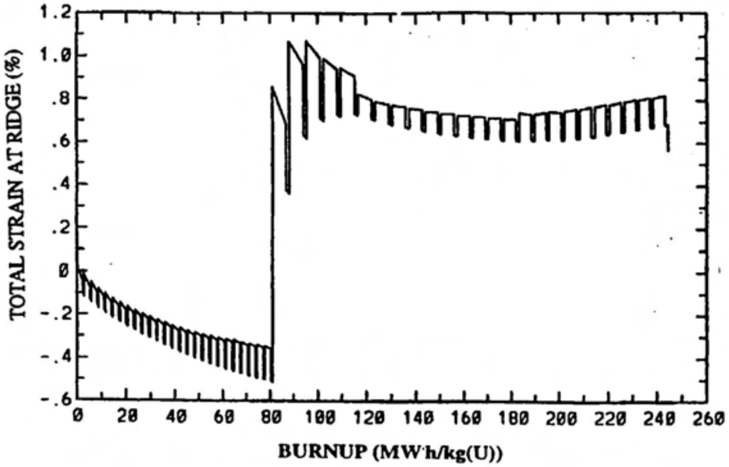


(b) Effect of Neutron Damage
(Source: Tayal et. al, 1995)

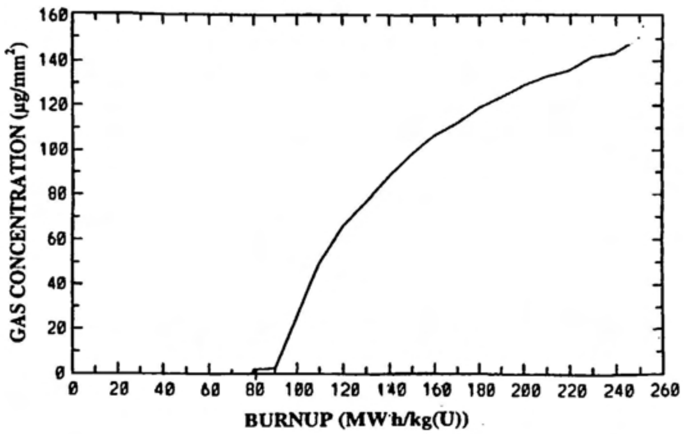
Figure 4: Effects of Corrodants and of Neutron Damage on Strength of Zircaloy



(a) Power History



(b) Strain Cycles



(c) Fission Gas

Figure 5: Refuelling Ramp and Weekly Cycles in Bruce B

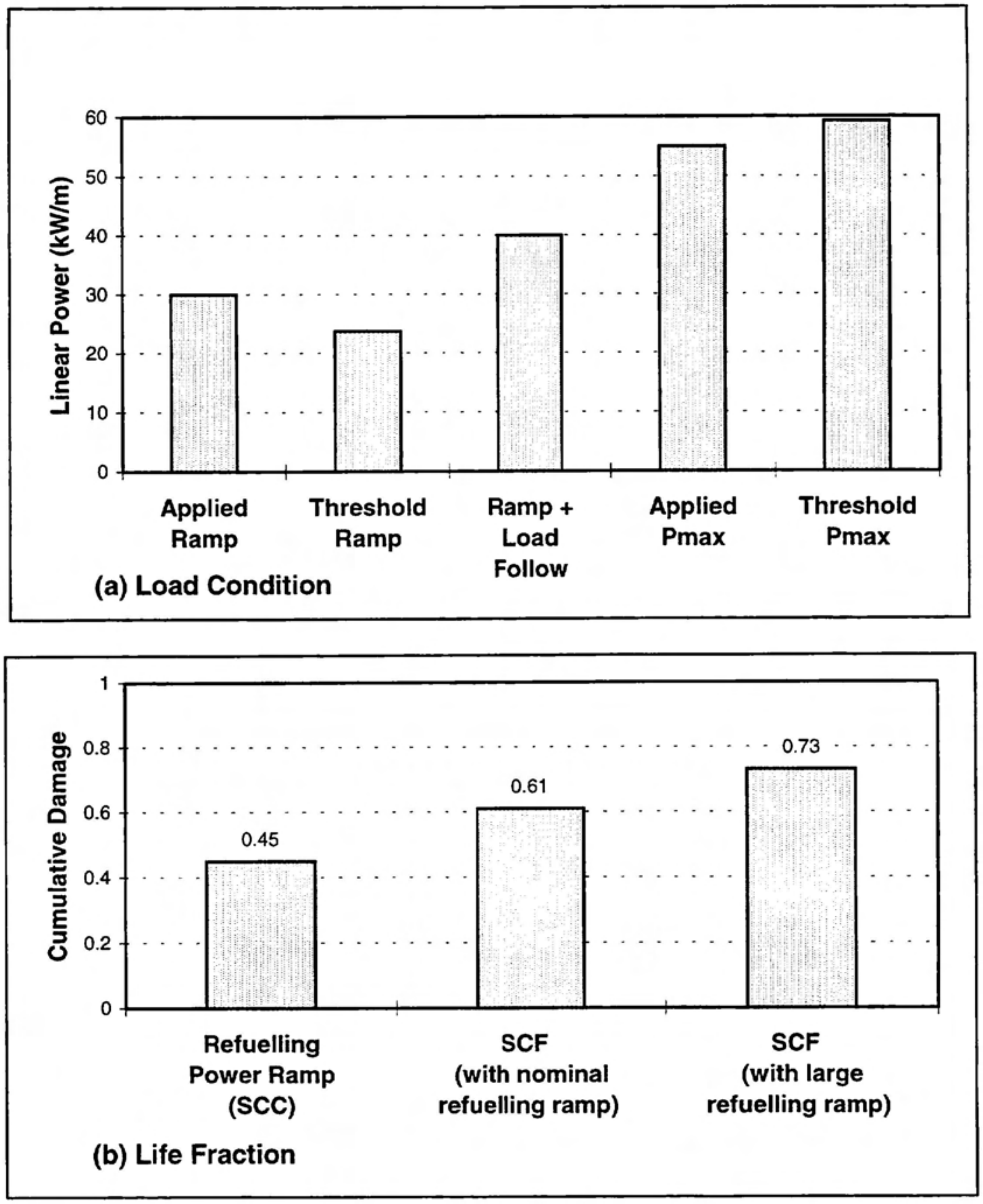


Figure 6: Fatigue Life of Bruce Fuel: Weekly Cycles