Fuel Depletion & Lattice Reactivity in CANDU

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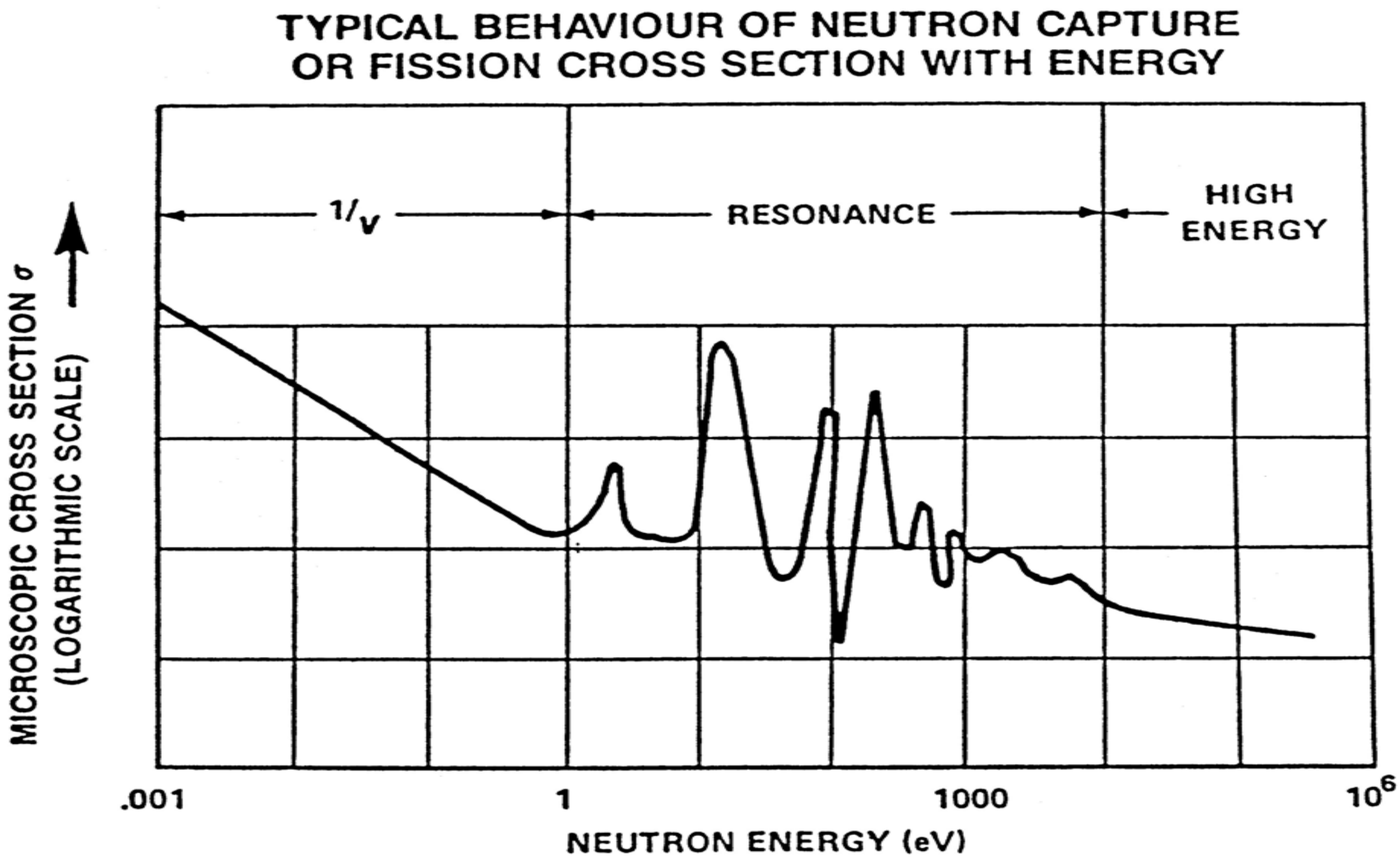
Canada

AECL Atomic Energy of Canada Limited

EACL Énergie atomique du Canada limitée

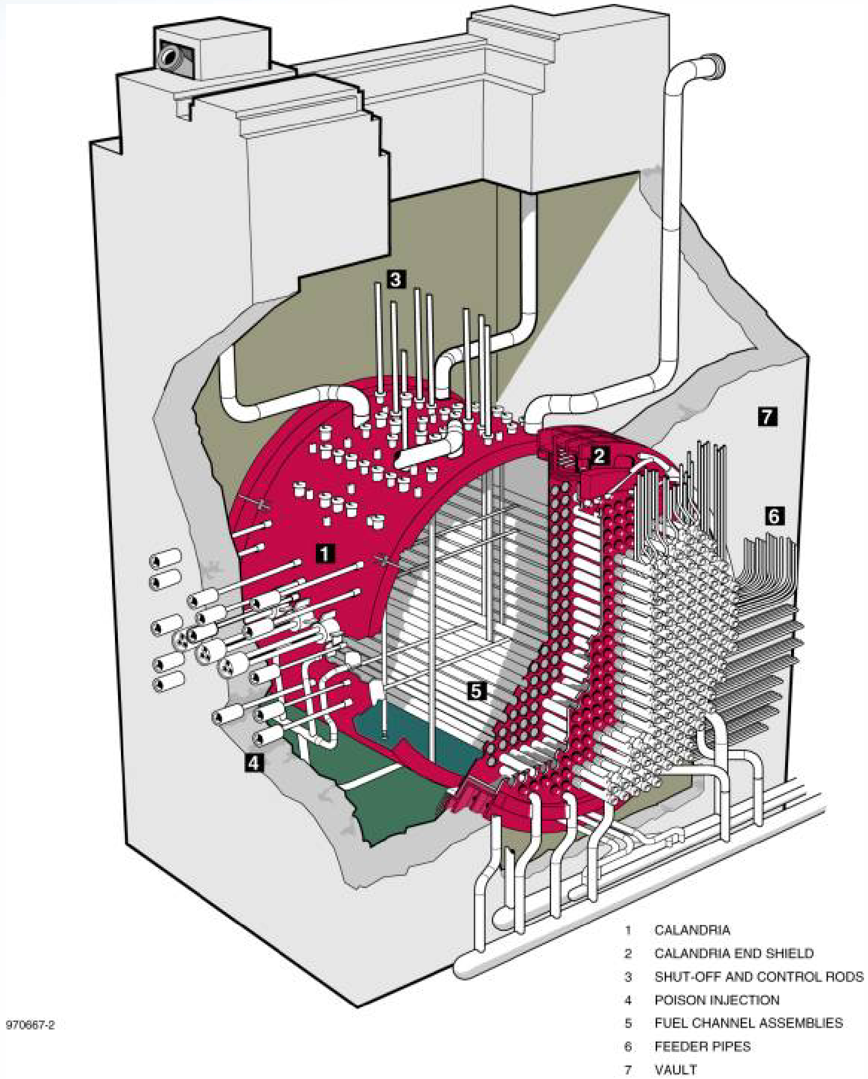
CANDU-6 Reactor

- The CANDU-6 reactor is a heavy-water-moderated and heavy-water cooled CANDU reactor using natural uranium as fuel.
- The CANDU design is modular, with fuel channels set on a square lattice of lattice pitch equal to 28.575 cm. Each fuel channel contains 12 fuel bundles.
- The CANDU 6 has 380 fuel channels.
- The next 3 slides show in turn:
  - The CANDU-6 Reactor Assembly
  - A sketch of the face view of the CANDU 6
  - A 37-element fuel bundle used in the CANDU 6.



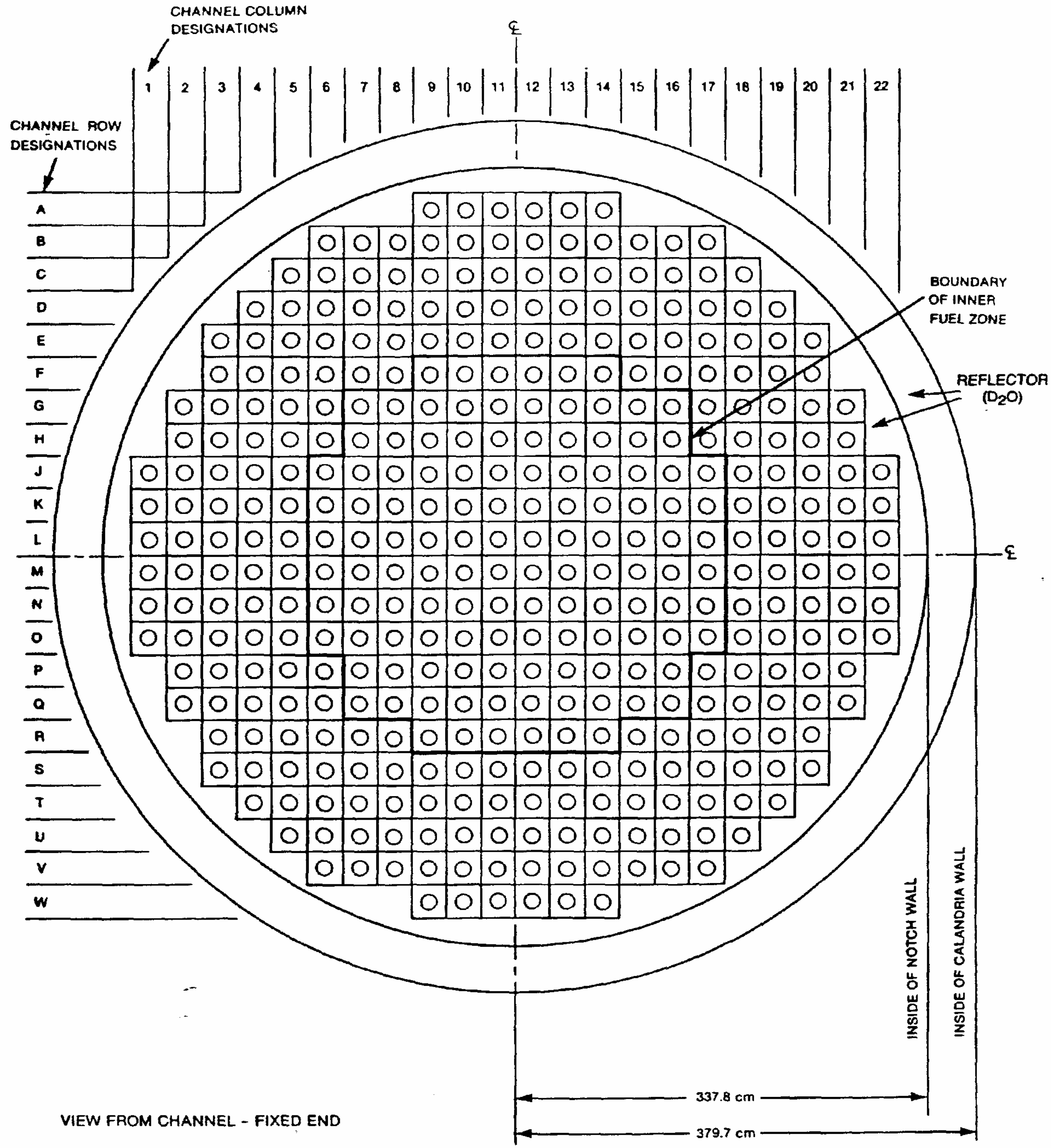
Schematic View of a Typical Cross Section as a Function of Neutron Energy (or Speed), Showing Resonances

CANDU-6 Reactor Assembly

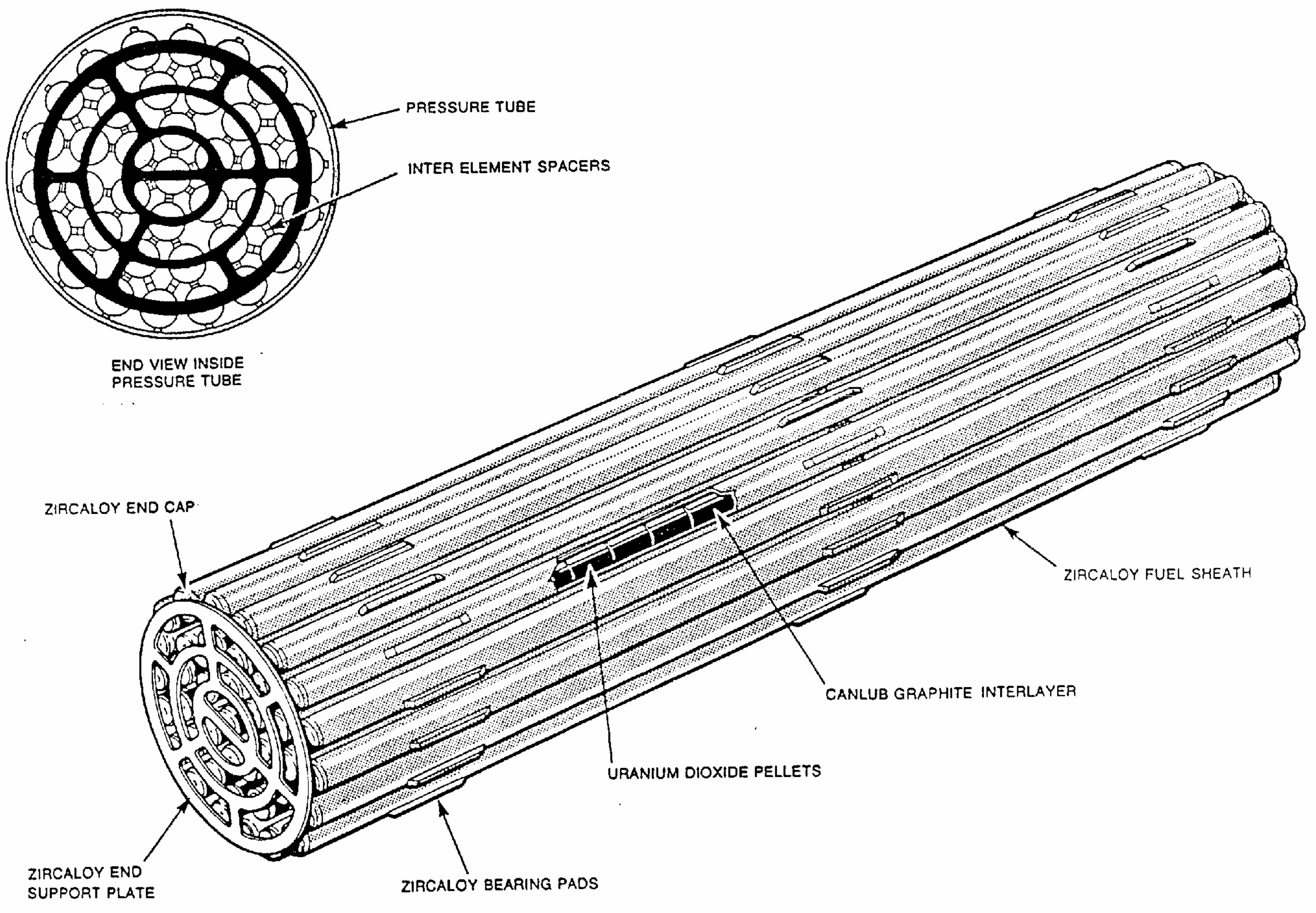


CANDU 6 Reactor Assembly

Face View of CANDU-6 Reactor



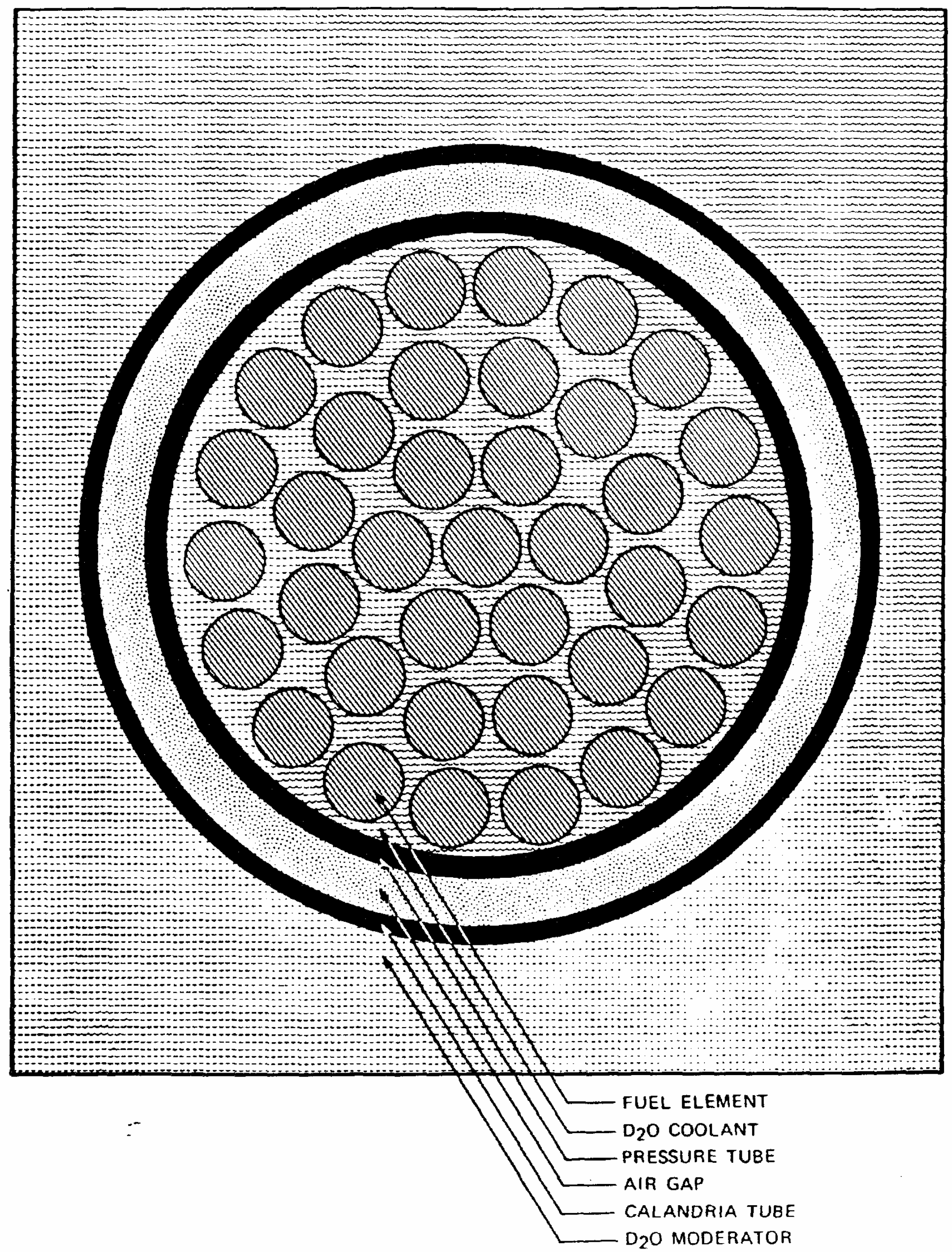
CANDU 37-Element Fuel Bundle



CANDU Basic Lattice Cell

- The basic building block of the CANDU design is the basic lattice cell, of dimensions 1 lattice pitch by 1 lattice pitch by 1 fuel-bundle length (28.575 cm x 28.575 cm x 49.53 cm).
- This basic lattice cell, shown in face view in the next Figure, has as components:
  - a 1-bundle length of pressure tube, with fuel bundle and heavy-water coolant,
  - the calandria tube, concentric with the pressure tube and separated from it by an isolating gas gap, and
  - the associated volume of heavy-water moderator.

CANDU Basic Lattice Cell with 37-Element Fuel (not to scale)

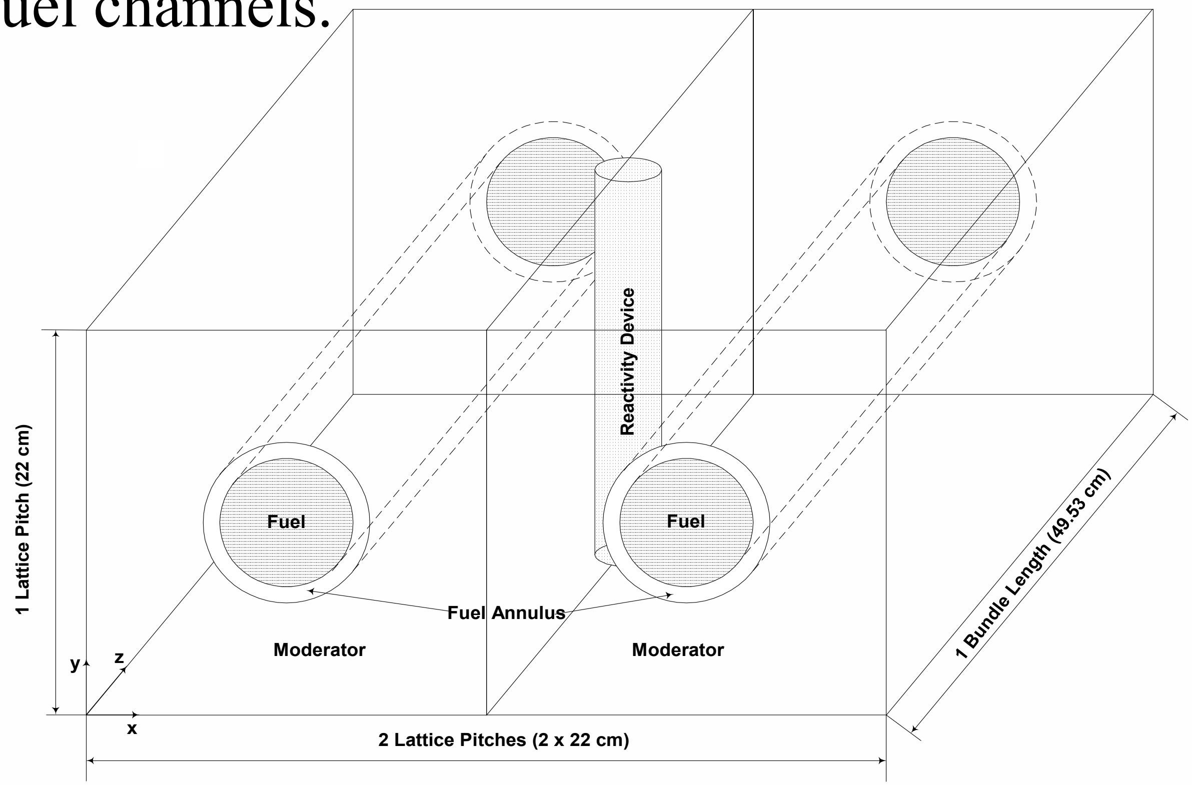


CANDU Reactivity Devices

- In the reactor core, there are also reactivity devices, such as zone controllers and adjusters, which are positioned interstitially between fuel channels (see example in next Figure.
- These devices perturb the nuclear properties of the lattice in their vicinity, and this effect has to be taken into account in the reactor model.

Interstitial Reactivity Devices

Here, the vertical device is inserted interstitially between the horizontal fuel channels.



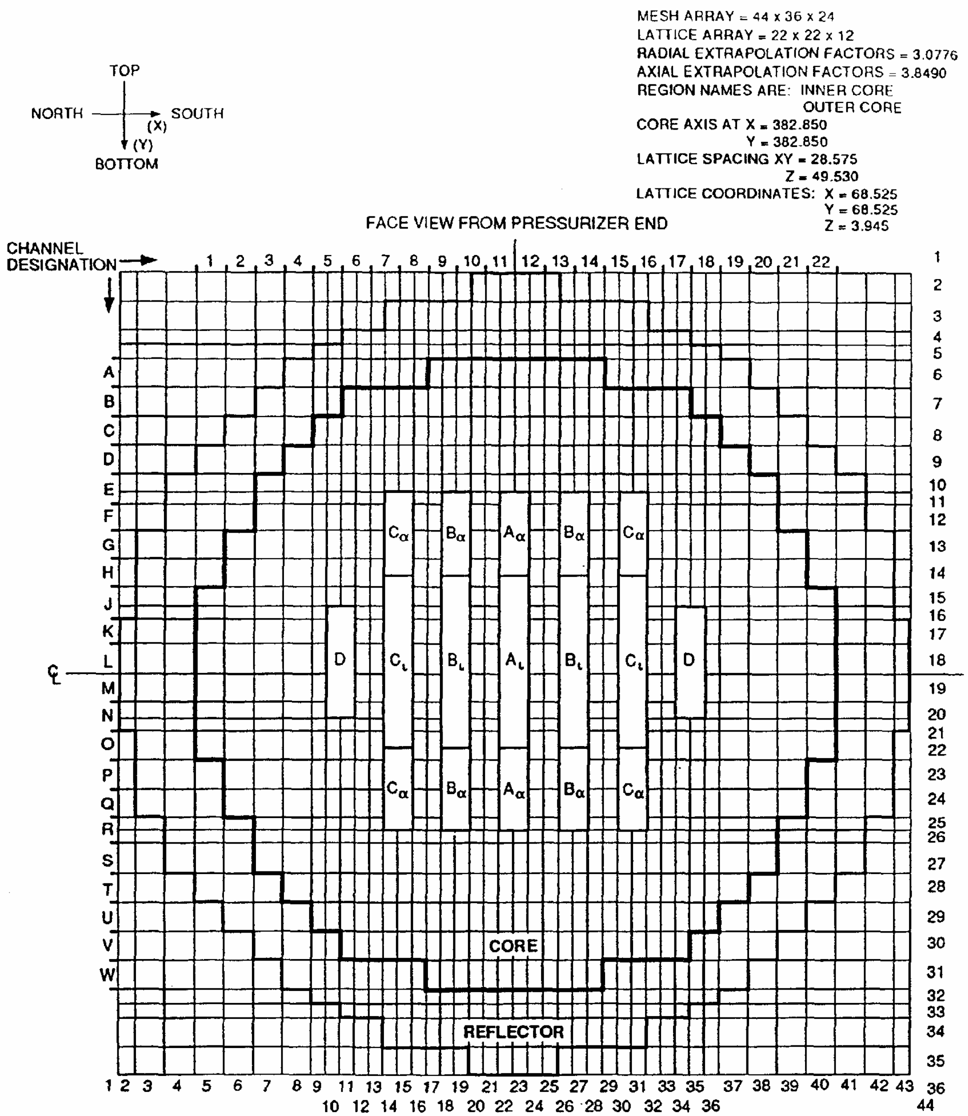
Boundary Conditions:

X_- Reflection	X_+ Symmetry
Y_- Reflection	Y_+ Symmetry
Z_- Reflection	Z_+ Symmetry

CANDU Reactor-Physics Computational Scheme

- **The computational scheme for CANDU neutronics consists of three stages. Computer programs have been developed to perform the calculations corresponding to each stage.**
- **The three stages are:**
  - **Lattice-Cell Calculation (POWDERPUFS-V or WIMS-IST)**
  - **Reactivity-Device (Supercell) Calculation (DRAGON)**
  - **Finite-Core Calculation (RFSP-IST)**
- **The next Figure shows the geometry of a typical RFSP-IST finite-difference diffusion model.**

Typical RFSP-IST Full-Core Model (Face View)



NOTE: ADJUSTERS ARE SHOWN IN THE ASYMMETRIC VERTICAL POSITION IN THE ABOVE DRAWING
ACTUAL VERTICAL POSITION IN MODEL IS DIFFERENT FOR EACH ADJUSTER (SEE TABLE 5.1A)
MESH SPACINGS ARE GIVEN IN TABLE 2.2A
A,B,C AND D ARE ADJUSTER ROD SEGMENTS AND SUBSCRIPTS ι AND α STAND FOR
INNER AND OUTER SEGMENTS RESPECTIVELY

NOT TO SCALE

Lattice Calculation

- A lot can be learned from the first stage of the scheme, the lattice calculation.
- This involves calculating the average nuclear properties ("homogenized-cell nuclear cross sections" for absorption, moderation, fission, etc.) for the basic lattice cell consisting of the fuel, coolant, pressure and calandria tubes, and moderator.
- Quantitatively, the lattice properties govern the neutron-multiplying behaviour of the reactor lattice.
- We will be focusing mainly on the lattice calculation.
- We start by considering the changes which occur in the fuel as it "burns" in the reactor.

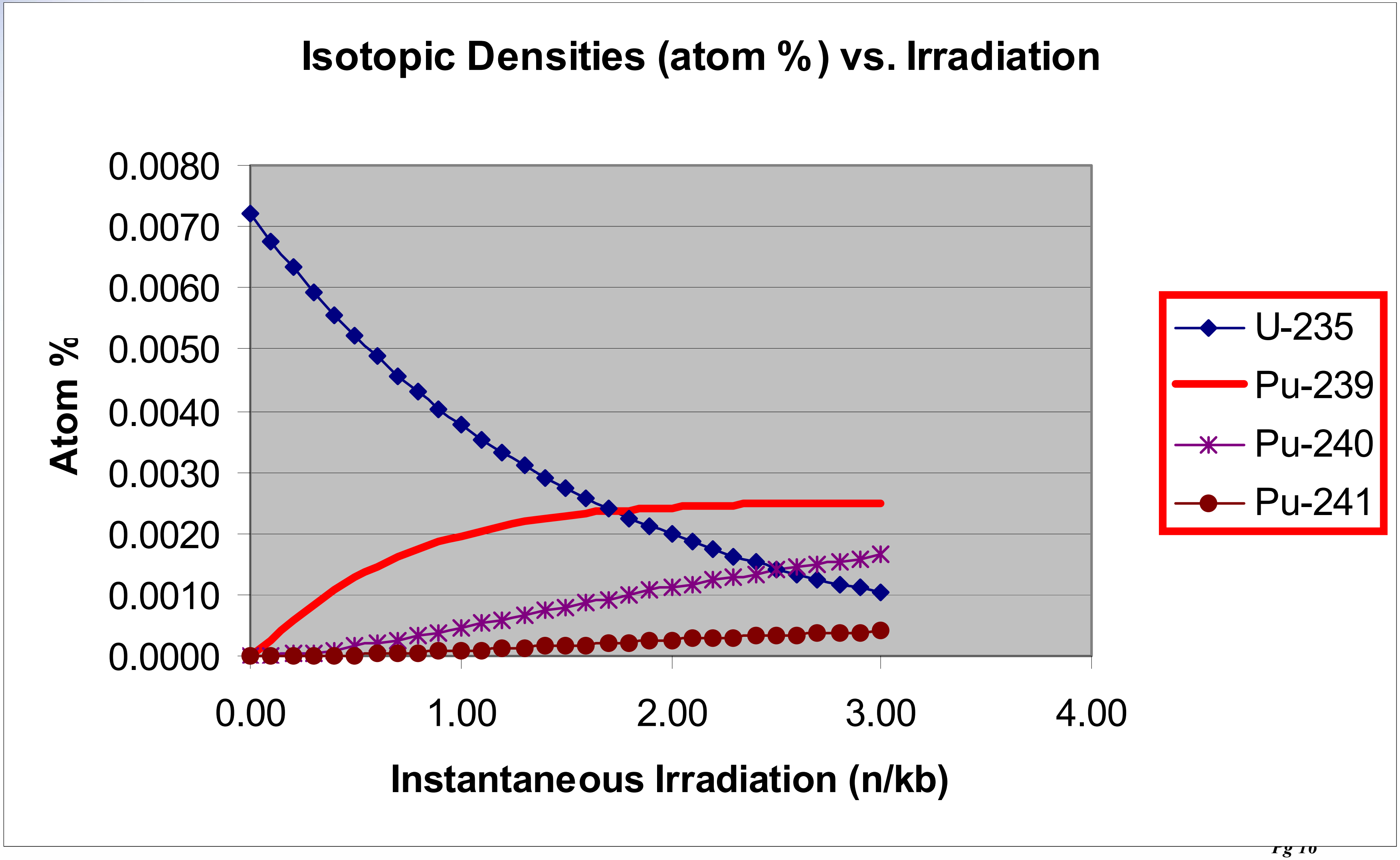
Changes in Fuel as it Burns

- Natural-uranium fuel: initially 0.72% ^{235}U (0.71% by weight).
- As fuel is irradiated in reactor, energy is generated by fissions.
- In fresh fuel, 95% of fissions occur in ^{235}U, and only 5% in ^{238}U.
- The ^{235}U depletes steadily as it fissions.
- Fission products build up; these increase the non-productive absorption of neutrons, i.e., they “rob” neutrons from the chain reaction
- (Looking ahead) At discharge of fuel from the reactor, ^{235}U concentration ~0.22%.

Changes in Fuel as it Burns (cont'd)

- Simultaneously, small fraction of ^{238}U is transformed into ^{239}Pu: $^{238}U + n \Rightarrow {}^{239}U$, then double beta decay leads to ^{239}Pu.
- ^{239}Pu is fissile, it participates in fission chain reaction as well: ^{239}Pu builds up in the fuel and is also burned "in situ".
- At discharge of fuel from reactor, ^{239}Pu concentration in the "heavy element" (i.e., as a fraction of the total U + Pu) is about 0.23%.
- In fuel near end of its residence in reactor, about 70% of fissions occur in plutonium.
- Total $^{235}U + {}^{239}Pu$ at discharge ~0.45%, compared to initial fissile content of 0.72%.

Evolution of Isotopic Densities of Fuel Nuclides vs. Irradiation (Related to Burnup)



Changes in Fuel as it Burns (cont'd)

Additional changes occur in fuel:

- Buildup of fission products, leading to increased neutron absorption
- Formation of ^{240}Pu (absorber) and ^{241}Pu (fissile) by successive neutron absorption in ^{239}Pu
- Formation of higher actinides by successive neutron absorptions in U and Pu.

Burnup and Irradiation

- Fuel burnup is defined as the amount of energy per unit mass of uranium which the fuel has produced at any time since its entry into the reactor. Typical units MW.h/kg(U).
- Fuel burnup therefore increases with fuel residence time in the reactor.
- Fuel exit burnup $\equiv$ burnup of fuel at discharge from reactor.
- High value of exit burnup $\Rightarrow$ fuel has produced lots of energy, cost of energy low.
- Low value of exit burnup $\Rightarrow$ cost of energy high.
- Therefore, aim of nuclear operator is to maximize exit burnup, everything else being equal.
- Fuel irradiation is the product of neutron flux in the fuel with time:

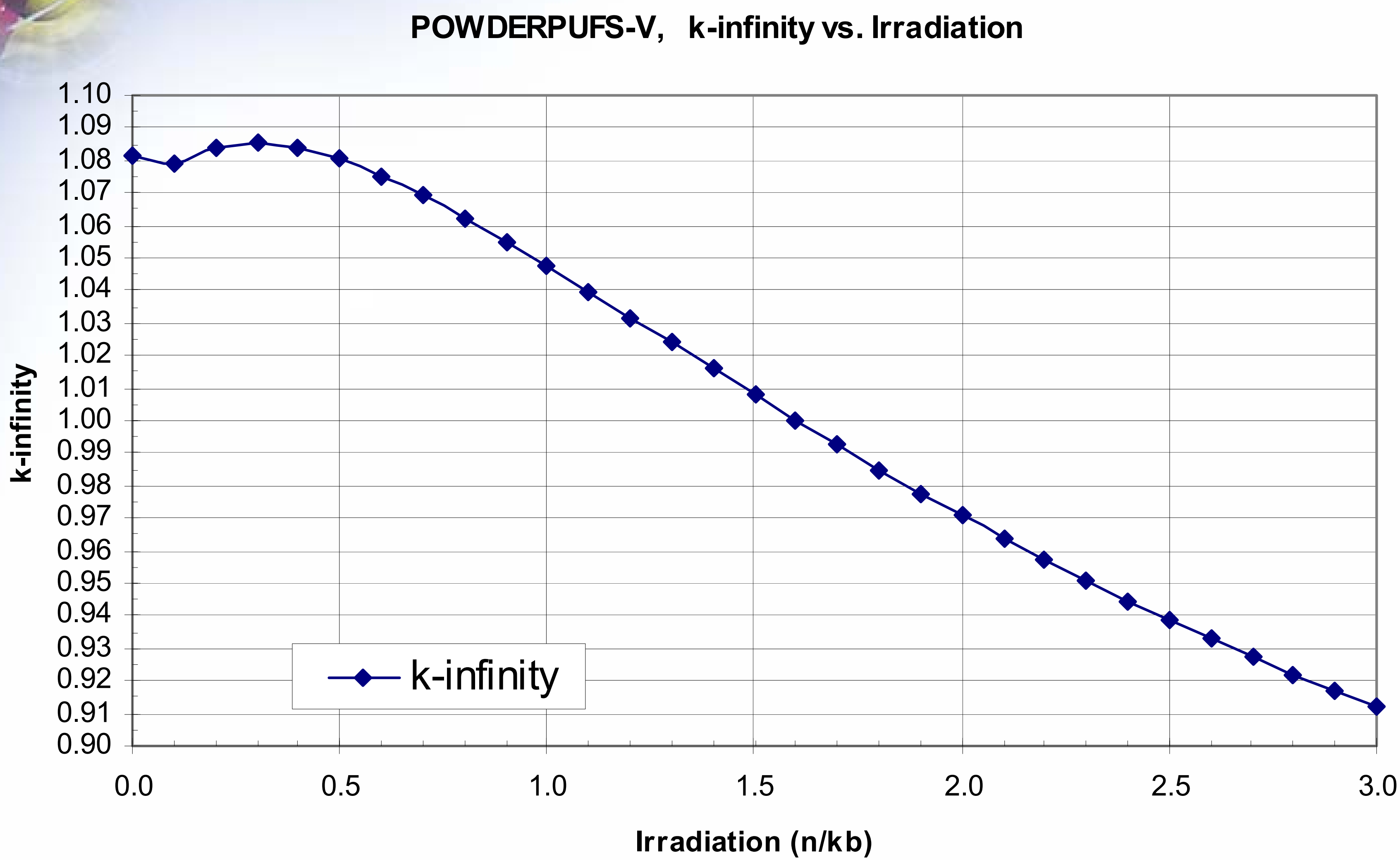
$$\omega \equiv \phi \cdot t$$

- Units of irradiation are n/cm^2, or, more conveniently n/kb.
- The POWDERPUFS-V value of fuel irradiation corresponds closely to 1% of the fuel burnup in MW.h/kg(U).

  For example, an irradiation of 2 n/kb corresponds to a burnup of 200 MW.h/kg(U).

Lattice Reactivity as Function of Irradiation

- Next Figure shows infinite-lattice multiplication constant k_∞ vs. fuel irradiation for basic CANDU-6 lattice with 37-element natural fuel.
- The immediate quick drop at zero irradiation (fresh fuel) is due to the buildup of saturating fission products.
- There is then an increase in k-infinity as plutonium starts to be produced and builds in – this is called the plutonium peak (the peak is in reactivity, not in Pu concentration)
- Following the plutonium peak, there is a steady decrease as fuel ages further (Pu buildup slows, fission-product load increases).



k_∞ vs. Fuel Irradiation for 37-Element Natural-Uranium Fuel

Lattice Reactivity as Function of Irradiation

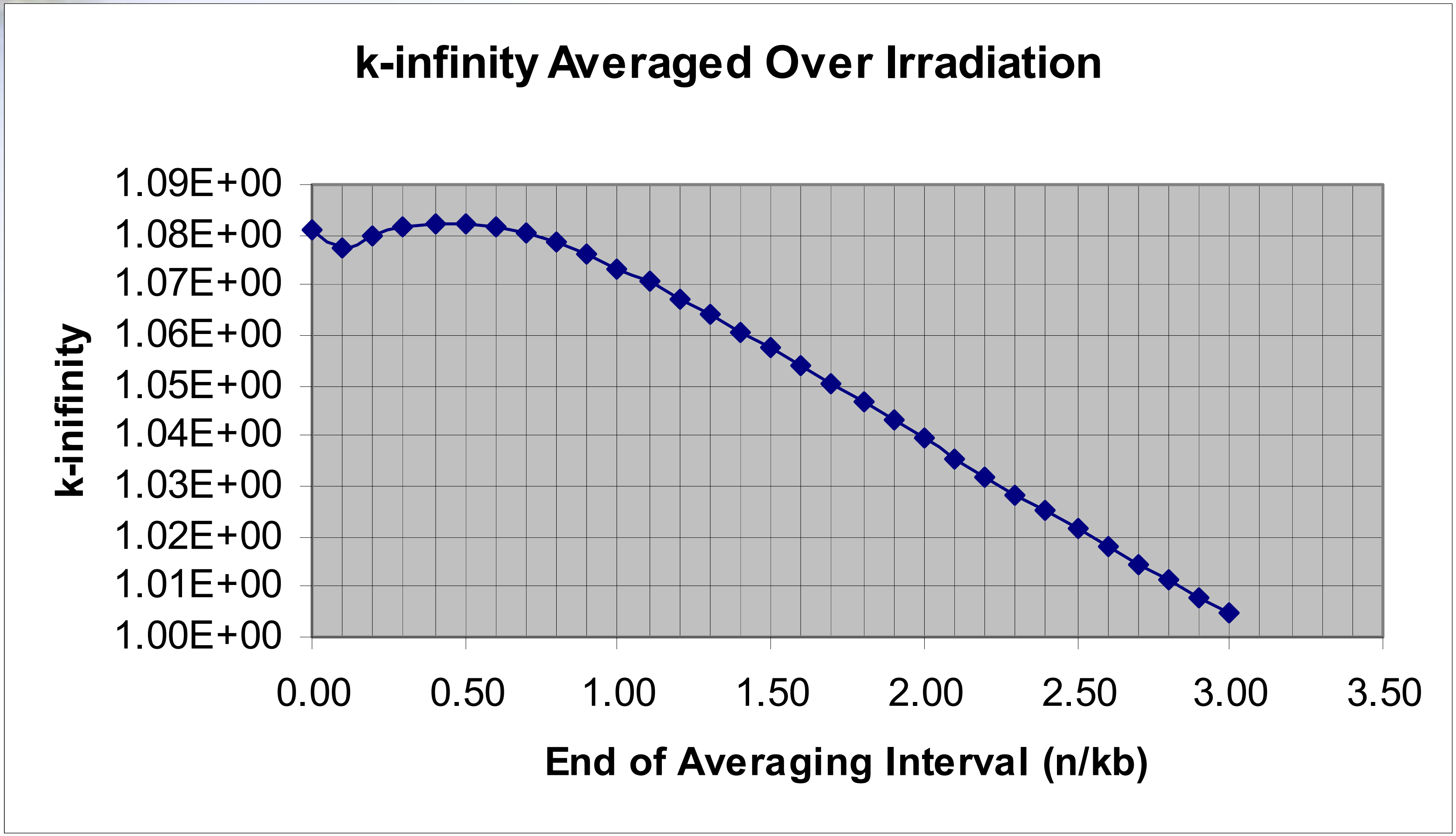
- As the irradiation (or burnup) increases to high values, k_∞ eventually drops to a value which cannot sustain the chain reaction.
- This value is not unity, because k_∞ measures the reactivity of the infinite lattice without reactivity devices, not of the finite reactor, it does not account for neutron leakage and absorption in devices.
- In the CANDU 6, leakage and device absorption amount to ~50 milli-k (0.050).
- Therefore, a k-infinity value below ~1.050 indicates that the chain reaction cannot be sustained.

Exit Irradiation and Burnup

- From the Figure, we see that k_∞ drops to 1.050 at an irradiation of ~0.9 n/kb. This corresponds to a burnup of 90 MW.h/kg(U) [in other units, ~3,750 MW.d/Te(U)].
- However, because CANDU is fuelled on power, the core contains a mixture of fuel with different values of burnup, ranging from 0 (fresh fuel) to high values.
- The net average k_∞ of the fuel in the core is 1.050, to yield a just-critical reactor with $k_{eff} = 1$.
- Therefore, the core-average irradiation & burnup are ~0.9 n/kb and ~90 MW.h/kg(U) .
- Because there is always some fresh fuel in the core, the average exit irradiation and burnup can be much higher.
- It turns out that fuel can remain in the core until it reaches about twice this burnup.
- In fact, typical fuel average exit irradiation and burnup in the CANDU 6 are ~1.75-1.80 n/kb and ~7,300-7,500 MW.d/Te(U).

Exit Irradiation and Burnup

- Because the core contains fuel with all irradiations from 0 to the exit irradiation, we can also estimate the latter to a good approximation by looking at a "running average" of the k_∞ as irradiation increases.
- The average exit irradiation should be the value at which this running average reaches 1.050.
- The next slide shows the running average plotted against irradiation.
- We can see that this procedure estimates the average exit irradiation at ~1.7 n/kb [average exit burnup ~7,100 MW.d/Te(U)], a fairly good estimate.

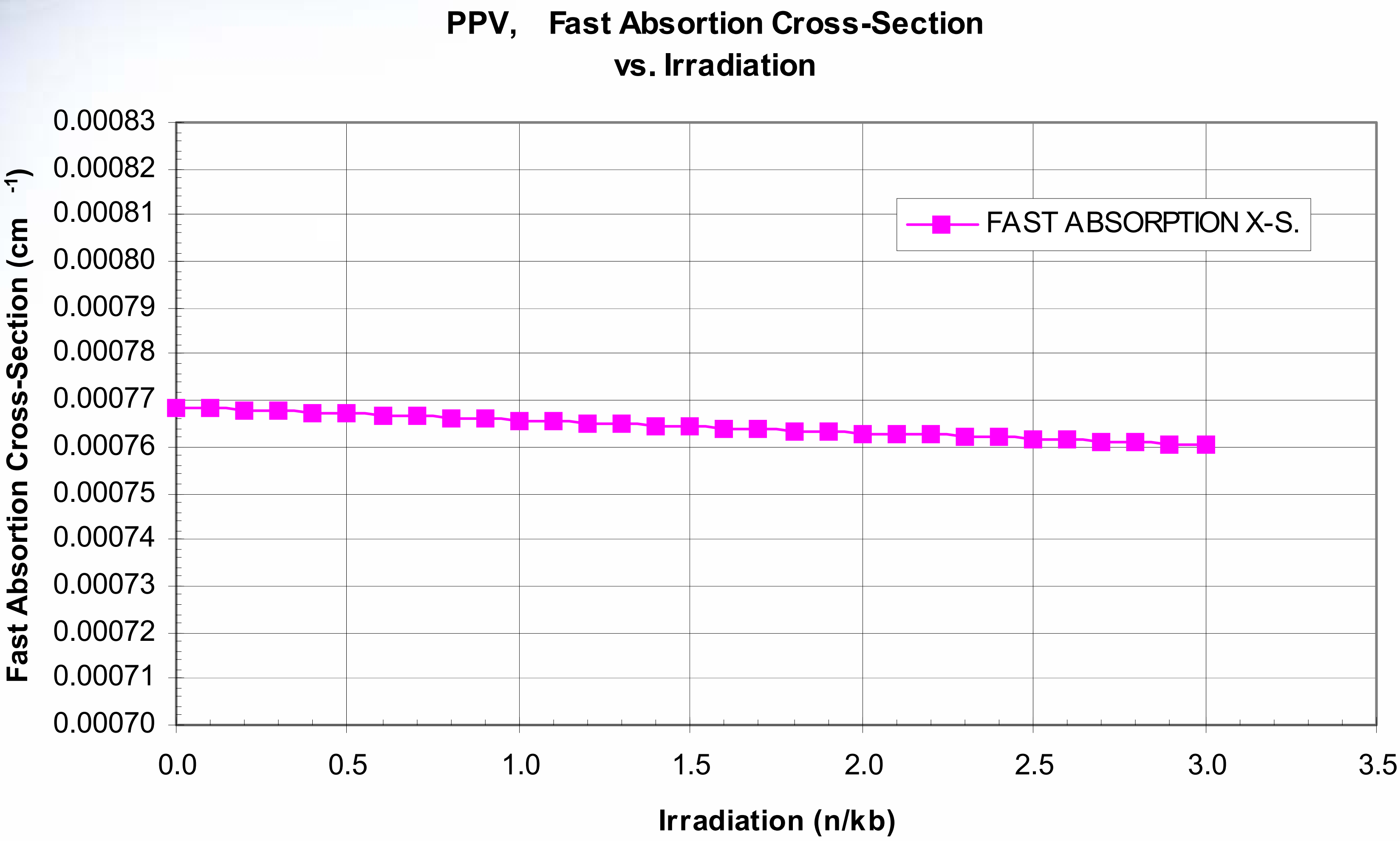


Running Average of k_∞ vs. Fuel Irradiation – Estimated Average Exit Irradiation ~1.7 n/kb

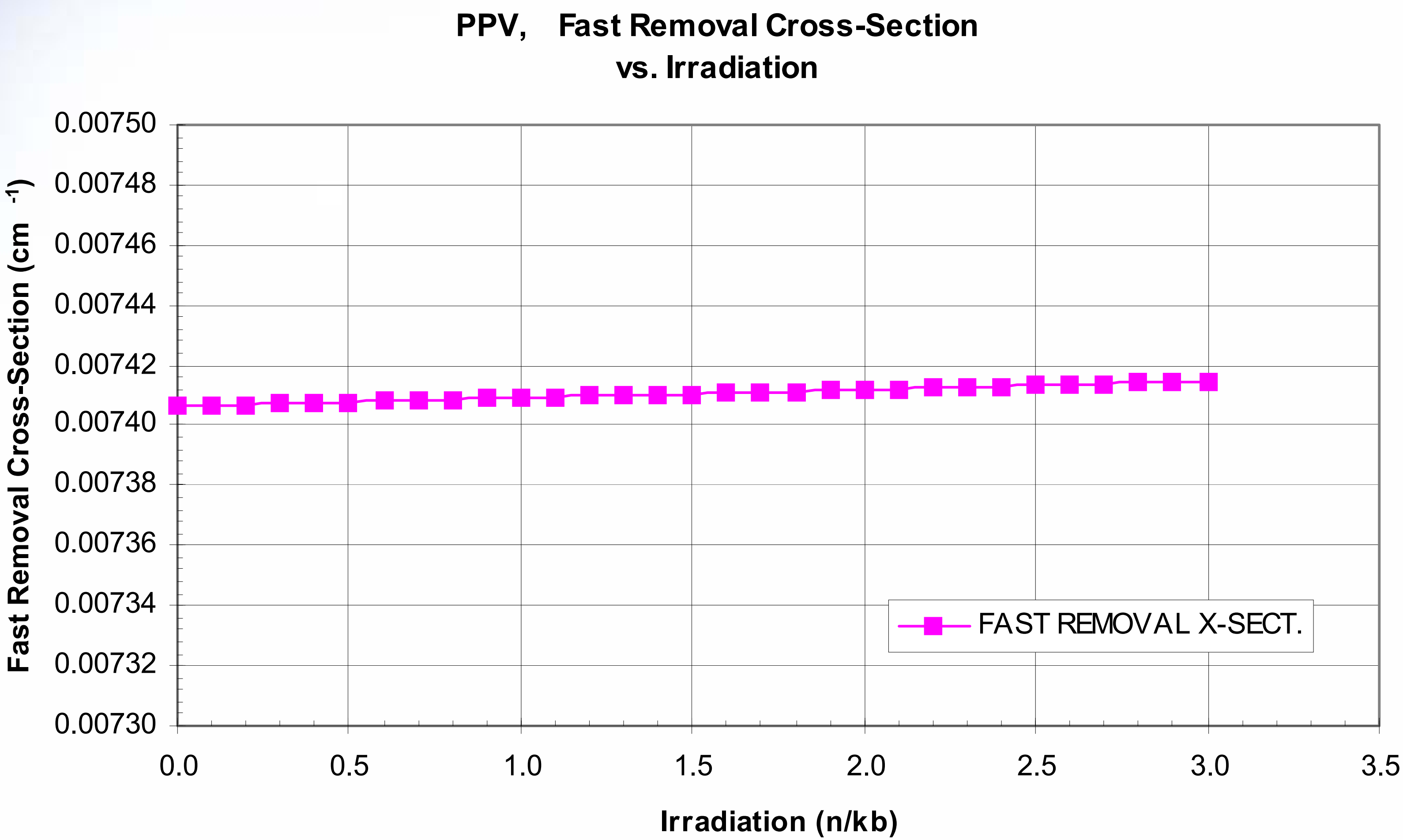
Cell-Averaged Cross Sections

- **The next 3 Figures show the basic-lattice cross sections, which “make up” k_∞, as a function of irradiation.**

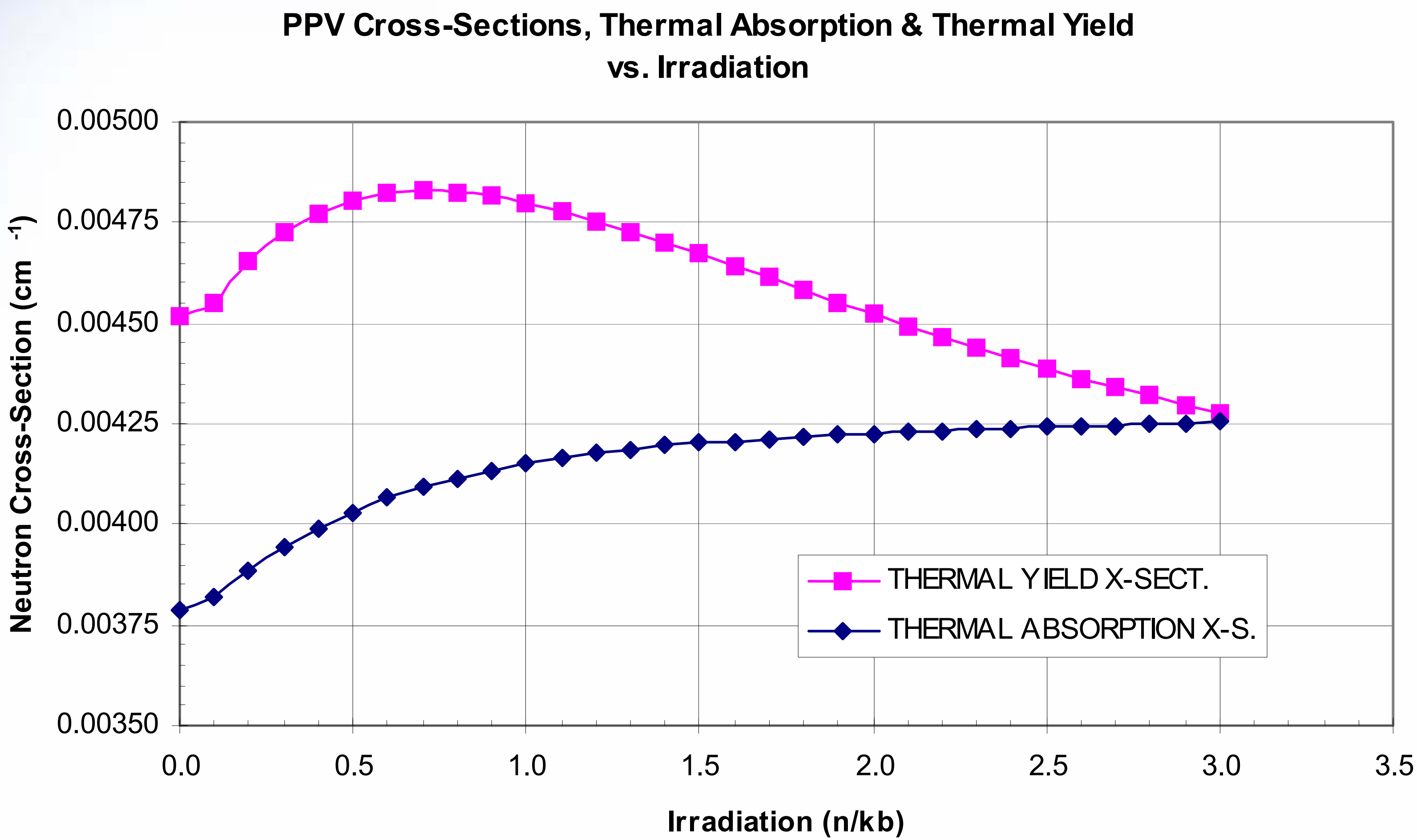
Fast-Neutron Absorption Cross Section vs. Fuel Irradiation



Fast-Neutron Moderation Cross Section vs. Fuel Irradiation



Thermal-Neutron Absorption and Fission Cross Sections vs. Fuel Irradiation



Cell-Averaged Cross Sections

- From these Figures it is clear that only two cross sections show significant variation with irradiation:
  - the thermal fission cross section Σ_{f2}(on account of the depletion of fissile material), and
  - the thermal absorption cross section Σ_{a2} (on account of the accumulation of fission products).
- When preparing a full-core RFSP calculation, the cross sections of each individual bundle must be input into the RFSP model according to the specific irradiation or burnup of the bundle.
- Thus, a lot of careful "bookkeeping" is required in setting up full-core calculations.

Factors Affecting Lattice Reactivity

- A number of operating parameters have an effect on lattice reactivity, sometimes a significant effect.
- Factors which increase lattice reactivity essentially move the entire k_∞-vs.-irradiation curve up (to higher values). Then k_∞ drops to 1.050 at a higher value of irradiation; thus, the average exit irradiation or burnup increases.
- Factors which decrease reactivity decrease exit burnup.
- Factors which increase reactivity:
  - Higher D_2O-moderator purity (even a very small increase has a significant impact on burnup)
  - Higher D_2O-coolant purity
  - Lower fuel temperature
- Other factors, which decrease reactivity:
  - Soluble poison (e.g., B or Gd) in moderator
  - Thicker pressure or calandria tubes
  - More water in the zone controllers.

Keeping the Reactor Critical Long-Term

- It is refuelling which keeps the reactor critical in the long term.
- In reactors which are not fuelled on-power, all (or most) of the fuel follows the k_∞-vs-burnup curve together.
- At low burnup, the fuel has very high reactivity. To achieve criticality, use is made of burnable "poisons" in the fuel or soluble poisons in the moderator. The poison burns off or is removed as the burnup proceeds and the k_∞ drops.
- In CANDU, this occurs only at the beginning of life, when all fuel is fresh; ~2 ppm of boron poison is used in the initial core, dropping to 0 ppm by ~100 days, and then refuelling begins.
- In light-water reactors, which have highly enriched fuel and are "batch" refuelled, boron concentration can be in the hundreds or thousands of ppm at the beginning of each cycle, and/or burnable poisons are used in the fuel.

On-Power Refuelling

- When refuelling starts, a number of fuel channels are refuelled per day.
- The amount of fuel replaced per day is determined by the average loss of reactivity from 1 day of burnup, and which must be made up.
- In the CANDU 6, 2 fuel channels are refuelled per day on average, using an 8-bundle-shift scheme (8 of the 12 bundles in the channel are replaced); i.e., replacing ~16 fuel bundles recovers the reactivity lost due to the daily increase in burnup of the core.
- The selection of channels to be refuelled each day is made on the basis of the distribution of burnup and power in the core on that day.
- The next slide shows the typical refuelling frequency of channels in the CANDU 6.

Typical Refuelling Frequency of Channels in CANDU-6 (in Full-Power Days)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									310	300	294	294	300	310								
B						356	300	253	237	227	226	226	227	237	253	300	356					
C					317	270	233	204	193	188	190	190	188	193	204	233	270	317				
D				304	258	222	198	177	171	169	173	173	169	171	177	198	222	258	304			
E			316	257	220	195	179	163	160	160	164	164	160	160	163	178	195	220	257	316		
F			265	222	197	180	169	158	173	174	177	177	174	173	158	169	180	197	222	265		
G		293	236	201	186	174	174	170	172	173	173	173	173	172	170	174	174	186	201	235	286	
H		258	213	187	178	170	172	170	171	172	172	172	172	171	170	172	169	178	187	212	252	
J	319	239	198	178	173	177	173	184	185	187	187	187	186	185	184	173	177	172	177	197	233	311
K	298	224	188	171	169	175	172	185	187	189	192	192	189	187	184	172	175	169	170	187	218	290
L	286	217	183	166	163	170	170	184	187	191	196	196	191	187	184	170	170	162	165	182	210	278
M	286	216	181	164	160	167	168	183	186	191	196	196	190	186	182	168	167	159	163	180	210	278
N	299	223	185	166	160	166	167	181	185	188	192	192	188	185	181	167	165	159	165	184	216	291
O	315	237	195	172	163	167	167	180	183	185	186	186	185	183	179	166	167	163	172	194	230	306
P		259	211	184	173	165	168	167	170	170	170	170	170	169	167	168	165	173	184	211	252	
Q		294	237	203	188	175	174	169	170	170	170	170	170	170	169	173	174	187	203	237	287	
R			274	232	208	188	173	165	173	172	173	173	172	173	165	172	188	208	231	273		
S			339	275	237	207	184	172	167	166	169	169	166	167	172	184	207	237	274	338		
T				336	281	238	208	191	182	179	183	183	179	182	191	208	237	281	336			
U					339	282	239	213	199	193	195	195	193	199	213	239	281	339				
V						350	283	241	221	209	206	206	209	221	241	283	350					
W									290	271	260	260	271	290								

Boundaries Of Radial Zones

Boundary Of Inner Burnup Region

Neutron Balance in CANDU

- The following slide shows the neutron balance in the CANDU-6 lattice, i.e., the quantitative balance between neutron production and neutron losses.

Production

56.5 Neutrons From U238 Fast Fission
491.9 Neutrons From U235 Thermal Fission
438.4 Neutrons From Pu239 Thermal Fission
13.2 Neutrons From Pu241 Thermal Fission

Total = 1000 Neutrons

Fast Leakage

6.0 Neutrons

Fast Absorp. In Fuel

31.7 Neutrons

Slowing Down

Resonance Absorp. in U238

89.4 Neutrons

Thermal Leakage

23.0 Neutrons

Thermal Absorption

849.9 Neutrons

Thermal Absorption In Non Fuel Components

14.4 Neutrons In Moderator
0.3 Neutrons In Coolant
19.0 Neutrons In PT
8.5 Neutrons In CT
6.2 Neutrons In Sheath
15.0 Neutrons In Adjusters Zone Controllers and Parasitic Absorbers

Total = 63.4 Neutrons

Thermal Absorp. In Fuel

242.5 Neutrons In U235
238.2 Neutrons In U238
228.1 Neutrons In Pu239
15.6 Neutrons In Pu240
6.2 Neutrons In Pu241
0.1 Neutrons In Pu242
0.6 Neutrons In Np
7.7 Neutrons In "Sm"
25.2 Neutrons In Xe
2.6 Neutrons In "Rh"
19.9 Neutrons In PFP

Total = 786.5 Neutrons

Notes:
Fast group denotes 10 MeV > E > 100 keV

Resonance group denotes 100 keV > E > 10 eV
Thermal group denotes sum of
Thermal group E < 0.625 eV and
Epithermal group 10 eV > E > 0.625 eV

Neutron balance in the CANDU-6 Equilibrium Core

AECL
EACL