Reactor Boiler and Auxiliaries - Course 133

THE CANADIAN DESIGNS AND THEIR PERFORMANCE

The success of the natural uranium heavy water reactor system, which has reached the point where it can compete in cost with conventional power stations, has demanded a high performance from the fuel elements and a considerable amount of development work has been concentrated on fuel element design. The fuel must be in a form which is stable under irradiation for long periods while immersed in heavy water at high temperature and pressure. Also, because natural uranium contains only a small proportion of fissile material, it is important to make the most efficient use of all available neutrons, and therefore, the fuel elements must contain a minimum of neutron-absorbing material.

The major step in satisfying these stringent requirements has been the development and manufacture of fuel elements consisting of sintered UO_2 pellets enclosed in zirconium-alloy sheaths. Processes for making the pellets, already described, have been developed which give a product of great stability. The problems of fabrication and quality control of zirconium-alloy sheaths with wall thicknesses as low as 0.38 mm have been solved. The complete fuel elements have been manufactured in large quantities by two manufacturers and are capable of burnups in excess of 10,000 Mwd/te U while operating at high heat ratings.

Performance of UO_2 Fuel Assemblies in Canada

Before summarizing the performance of the Canadian fuel design to date, it would seem appropriate to give a brief description of the different fuel designs produced by Canadian manufacturers. The vast majority of power reactor fuel under development and irradiation today in Canada consists of 50 cm long bundles of either 7- or 19-element bundles, which fit inside an 8 cm I.D. pressure tube. Pickering and Gentilly G.S.'s will utilize 28-element and 18-element bundles respectively, designed to fit within 10 cm I.D. pressure tubes. The 7-element bundles in NPD are no longer considered to be a potential future fuel design, but were an important intermediate step in the development of smaller diametered elements, now in use at Douglas Point and to be used at Pickering G.S. The 7-element bundles are held together by either rivetted or fusion welded end plates, with inter-element and bundle-pressure tube spacings maintained by spiralled, spot-welded wire wrap. The NPD 19-element bundles, initially conceived because early irradiation tests at CRNL indicated that a core fuelled entirely with 7-element fuel would lead to elements near the centre of the core being operated at ratings which at that time were believed to be too high, are of similar design, but the elements are of reduced diameter (1.5 cm vs 2.5 cm).

The Douglas Point fuel bundles, several of which are undergoing irradiation in NPD, evolved from the NPD 19-element bundle, and are remarkably similar except for different detailed design of wire-wrap and the addition of bearing pads to improve the wear properties between the bundle and the coolant tube. Other changes involved manufacturing methods (described previously) to reduce costs. The tungsten inert gas (TIG) end cap and element-to-end plate welds of NPD were replaced with resistance welding for Douglas Point. (See Fig. 1, lesson 133.71-2.)

The second generation of fuel for Douglas Point is still under development, but will have no wire wrap. Instead, element spacing is accomplished by the use of small spacers at the bundle mid-plane, and bundle-to-pressure tube spacing is accomplished by means of three planes of straight bearing pads. Fig. 1 shows the type of brazed spacer in plan view being developed; note that the spacers are at an angle of 30° to each other to prevent inter-locking of mating spacers. Fig. 2 shows an elevation and cross-sectional view of a typical straight bearing pad. The reason for the elimination of the wire-wrap from the Douglas Point design was that out-reactor flow endurance tests on wire-wrapped bundles indicated that at the high flow rates needed for Douglas Point fuel, (12.6 kg/s maximum) the elements would vibrate enough that in the life of the fuel, the wear on adjacent sheathing by the wire-wrap may be sufficient to penetrate the sheath (called "inter-element" fretting). The criss-crossed spacers allow one to tolerate a higher fretting rate, and fretting occurs on a raised surface.

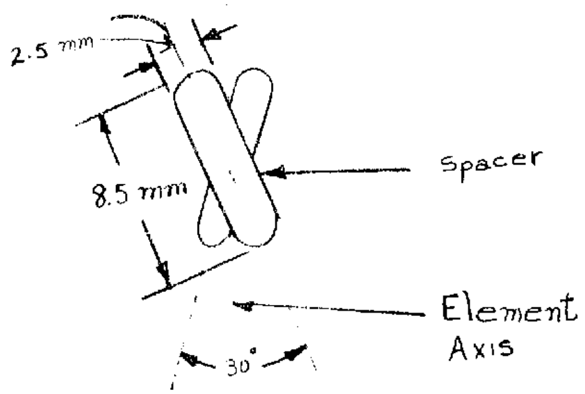


Fig. 1

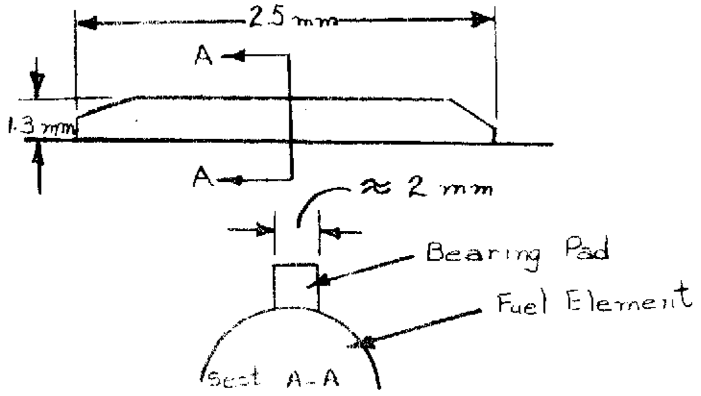


Fig. 2

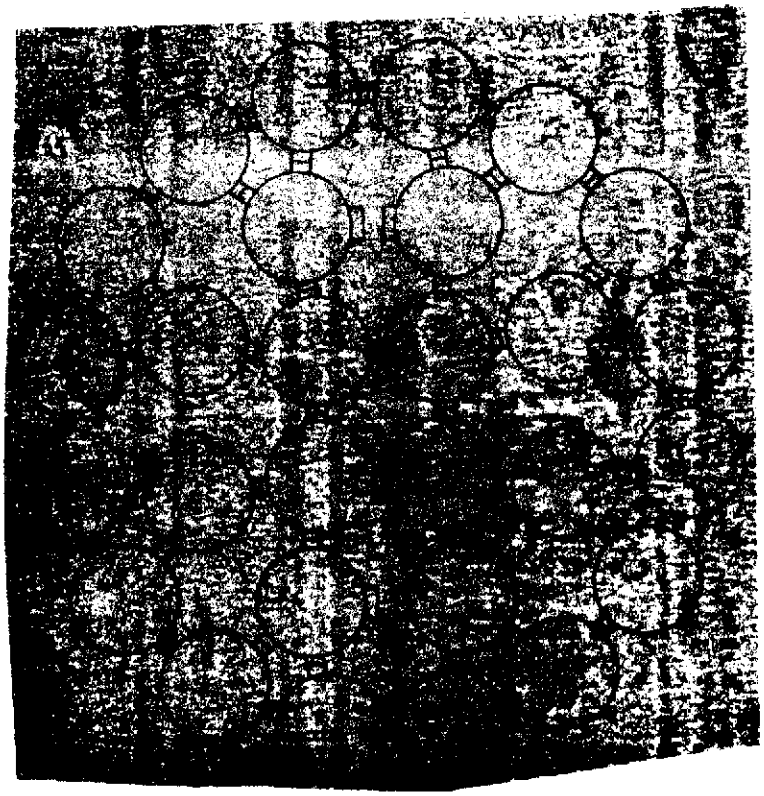


Fig. 3

The Pickering reactors will utilize 10 cm pressure tubes. In order to take advantage of the technology developed for the 1.5 cm elements in use at NPD and Douglas Point, the size of element will remain the same, but the number of elements increased from 19 to 28. Fig. 3 shows a transverse cross-section of such a bundle at the axial mid-plane; note the use of spacers the same as those for the Douglas Point replacement fuel. The bearing pads have been omitted from the sketch.

Extensive tests on the 19-element designs have been carried out in the loops of the NRU reactor at CRNL. In addition, the irradiation of statistically significant numbers of bundles of each of the 8 cm O.D. designs is continuing in NPD and commencing at Douglas Point. In NPD, the irradiation of the first charge of fuel has reached equilibrium burnup (6400 Mwd/te U); moreover, a large number of bundles have been replaced with fresh fuel while attaining burnups in excess of 10,000 Mwd/te U on certain development bundles, at Douglas Point ratings ($\int \lambda d\theta = 36$ w/cm). Post-irradiation examination of high burnup elements has revealed them to be in excellent condition. While the UO_2 generally was cracked and very brittle, there has been no evidence of any inter-action between the fuel and sheathing.

Bundles of the 19-element CANDU-PHW design have been successfully operated in the U-2 loop at CRNL in steam-water mixtures for prolonged periods, ie, in a boiling environment. Post-irradiation examinations have revealed that the UO_2 - Zr-2 fuel design can be successfully operated in a two-phase coolant, and tests are continuing. Dimensional stability of the bundles tested has shown a maximum length increase of less than 1/4% at $\int \lambda d\theta = 55$ w/cm, much higher than the maximum rated Pickering elements and high burnup bundles have been shown to be structurally sound enough to pass specified <u>pre-irradiation</u> strength tests.

The first test of Pickering fuel has been completed; the test consisted of the irradiation of six 28-element bundles, four of which were operating at average outer element ratings comparable to the highest rated Pickering elements ($\int \lambda d\theta$ = 42 w/cm). Some bundles operated in a boiling environment with up to 6 wt % exit steam quality. Examination of the bundles, irradiated to a maximum burnup of 3000 Mwd/te U, showed them to be in excellent condition. Further Pickering tests are continuing.

Hydrogen Pick-up in Zircaloy-2

The deuterium released in the dissociation of heavy water can be picked up by the sheathing, to form ZrD_2, and the oxygen forms an oxide coating on the sheathing, according to the equations:

$$Zr + 2\ D_2O = ZrO_2 + 2\ D_2\uparrow$$

$$D_2 + Zr = ZrD_2$$

Excessive ZrO_2 thickness could adversely affect the heat transfer properties of the sheathing. The ZrD_2 does not present a problem when the sheathing is at high temperatures; when cooled, however, the ZrD_2 platelets precipitate out, and appear as short dark lines when viewed at high magnification. As long as the platelets are oriented circumferentially, they do not present a problem; if they precipitate preferentially in the radial direction, the decrease in local density could lead to cracking of the sheathing. Fig. 4 shows a cross-section of sheathing, with ZrD_2 platelets predominately circumferential, magnified to 250X. The vertical distance is 0.38 mm in the figure. A criterion has been developed, called the F_n number, which is the proportion of platelets oriented within 30° of the radial direction, for comparison of samples.

Fig. 4

Radial hydrides may develop in areas of stress concentration and they tend to migrate down temperature gradients to the coldest parts of the bundle (eg, end plates, end caps). Embrittlement of the sheathing results. In view of the hydrogen pick-up in Zircaloy-2, future Canadian fuels will be clad in Zircaloy-4 which is an alloy similar to Zircaloy-2, but has a slightly lower nickel content, which evidently leads to a reduced tendency to absorb hydrogen.

CANDU-BLW Fuel

The fuel for BLW is still under development, and is generally similar to that employed in the NPD and Douglas Point reactors. It consists of natural uranium dioxide pellets sheathed in Zircaloy tubes to form fuel elements. Eighteen of these elements are assembled into a fuel bundle configuration. As shown in Fig. 5, these eighteen elements are arranged in two concentric rings, the inner ring containing six elements and the outer ring twelve elements. The inner ring surrounds an unfuelled central position which is occupied by a tubular gas-filled Zircaloy tie-rod. This tie-rod is used to assemble ten bundles into a fuel assembly. Each fuel channel in the reactor contains one of these assemblies.

Zircaloy-4 is used to sheath the UO_2 pellets, which are larger in diameter than the Douglas Point or Pickering elements (2.0 cm vs 1.5 cm). The nominal maximum rating is $\int \lambda d\theta = 48$ w/cm; but the nominal maximum specific power output is lower than Douglas Point (24 w/gm U vs 34 w/gm U), due to the larger element diameter. Like the CANDU-PHW pellets, the BLW pellets have a shallow spherical dish at one end, the purpose of which is two-fold:

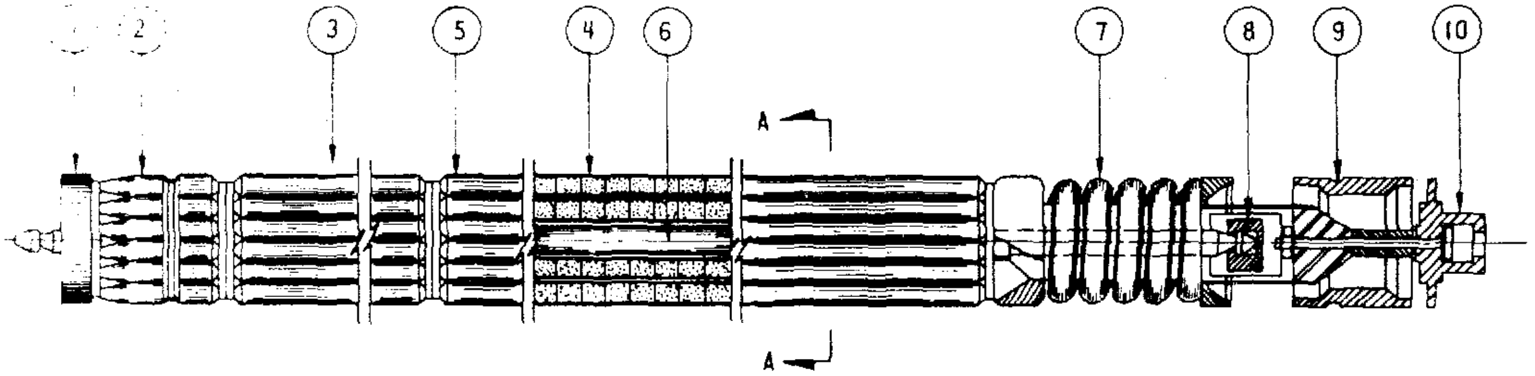
1. To increase the void volume in the element to accommodate the fission product gases, and reduce the internal pressure.

2. To allow some expansion of the pellets.

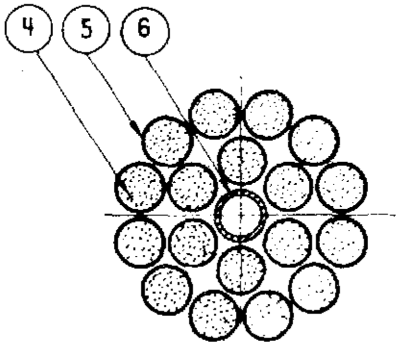
Zircaloy-4 was chosen as a sheathing material because of its lower rate of uptake of corrosion product hydrogen, as outlined previously. The table below gives a comparison of some of the relevent properties of these two materials; the figures are in weight %. The capture cross-sections of the two materials are about the same, with that of Zircaloy-2 slightly lower.

Element	Zircaloy-2	Zircaloy-4
Sn	1.20 - 1.70	1.20 - 1.70
Fe	0.07 - 0.20	0.18 - 0.24
Cr	0.05 - 0.15	0.07 - 0.13
Ni	0.03 - 0.08	0.007 maximum

Sum of Fe, Ni and Cr in Zr-2 must range between 0.18 and 0.38
Sum of Fe, Ni and Cr in Zr-4 must be 0.28 minimum



1. UPPER KEY~ VARIABLE THICKNESS
2. FLUX SUPPRESSOR
3. FUEL BUNDLE~ 10 PER ASSEMBLY
4. UO_2 FUEL PELLETS
5. SHEATH
6. TUBULAR TIE ROD
7. SPRING
8. LOWER KEY
9. FUELLING MACHINE LOCATING FINGER GROOVE
10. UPPER HALF OF T-SLOT
    CONNECTOR TO SHIELD PLUG



SECTION A-A

Fuel Assembly

Fig. 5

Other Advanced Fuel Designs

Other designs which are in the process of development involve fundamentals changes in the geometry of the fuel bundle. One of these is an annular fuel consisting of several concentric cylinders of fuel material with spaces between them to allow for the flow of coolant, surrounded by a tubular sheath. Another form of fuel rod, known as the tube-in-shell assembly (Fig. 6), consists of a large-diameter tubular sheath, filled with uranium dioxide, through which pass several small diameter coolant tubes. The ends of the sheath are closed by end-plates with holes into which the coolant tubes are sealed. The main advantage of the tube-in-shell bundle is the high fuel-to-coolant ratio offered, meaning that it could be used with a light water coolant. Testing to date has indicated that it does not offer any advantage when operated at high exit steam qualities; however, manufacture and production costs could be reduced with further development, since sintering, grinding, etc, of pellets would be eliminated. Vibratory compaction is used to pack the UO_2 in the shell. Two in-reactor tests have been tried to date, both ending with sheathing failures.

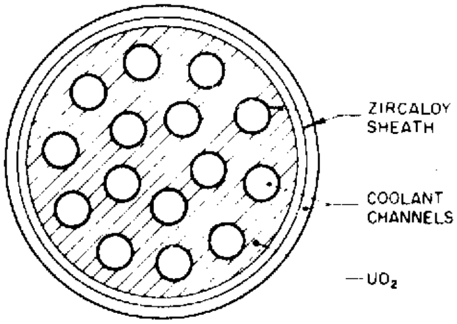


Fig. 6

ASSIGNMENT

1. Why must parasitic neutron absorption in the fuel be minimal with the CANDU-PHW and CANDU-BLW reactor designs?
2. Why was the wire-wrap eliminated from the design of the Douglas Point reference fuel design? What replaces it?
3. Summarize the operation to date of the Canadian power reactor fuel, to illustrate the compatibility of UO_2 and Zr-2.
4. How does corrosion hydrogen affect Zr-2 sheathing? What alloy offers reduced hydrogen pick-up rates?
5. What is the main advantage of the tube-in-shell fuel?

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