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NATURAL URANIUM METALLIC FUEL ELEMENTS
FABRICATION AND OPERATING EXPERIENCE

By
F.H. Hammad, A.A. Abou-Zahra
and S.W. Sharkawy

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ABSTRACT

The main reactor types based on natural uranium metallic fuel element, particularly the early types, are reviewed in this report. The reactor types are: graphite moderated air cooled, graphite moderated gas cooled and heavy water moderated reactors. The design features, fabrication technology of these reactor fuel elements and the operating experience gained during reactor operation are described and discussed. The interrelation between operating experience, fuel design and fabrication was also discussed, with emphasis on improving fuel performance.
INTRODUCTION

Natural uranium in metallic and ceramic forms were used as nuclear fuels in the CP-1, the world's first reactor that went critical in December 2, 1942. The success of that early pile had led to the building of series of reactors based on natural uranium metallic fuel elements in the US, UK, France, Canada, India, Italy, Japan and Spain. The early types of such reactors erected in the U.K. & France, known as the Magnox or Gas-Graphite reactors, were used as plutonium producers while the advanced types were used mainly for power production. The electricity produced from the Magnox reactors in UK represents about 35% of the total electricity generated from nuclear sources (1).

The number of Magnox power reactors (>150 MWe) in operation in the world are 26 reactors in 17 stations having installed capacity of 8816 MWe out of 100 stations (>150 MWe) in operation (in 1977) with total installed capacity 74189 MWe (2). The accumulated electricity produced from this reactor system reached 30.3% of the world total electricity generated from nuclear sources (1). However, no more reactors of the Magnox type were built after the Wylfa station in UK (1971), Bugey station in France (1972) and Vandellos in Spain (1972).

The ARE Atomic Energy Establishment (AEE) has operated since 1961 a 2 MW research reactor (UA-RR-I), subsequently designated (ET-RR-I). This reactor has limited research facilities in connection with materials and fuel irradiation to develop nuclear fuel technology in Egypt in conjunction with nuclear power programme experience in fuel fabrication and fuel performance. Thus there is a need for a high flux reactor in the 40-60 MW range or higher provided with facilities for fuel and material irradiation. This reactor could be designed to act as dual or multi-purpose, e.g., the reactor could serve as a prototype for water desalination purposes or providing.
power to a local community. The proposed reactor or modification of
the existing research reactor should be based on natural uranium fuel
either metallic or ceramic to avoid the highly expensive enriched
fuel.

Another possible approach is increasing the power of the exist-
ing HT-RR-1 reactor and modifying the core design to allow irradiation
of fuels of various designs. Furthermore, a core based on enriched
fuel acting as a driver fuel and neutral uranium fuel can be designed.

The design and fabrication of the fuel element must take into
consideration the fabrication and operating experience gained in
other countries. The technology of ceramic fuels and their operating
experience e.g., the CANDU type fuel elements are well developed and
will be the subject of another report. As for metallic fuels there is
a wide choice resulting from the numerous fuel element designs and
fabrication procedures and operating experience gained in various coun-
tries particularly for the early types of fuel elements.

In this report a review of the essential features of the metal-
llic fuel elements of the important natural uranium converter reactor
their fabrication technology and operating experience is presented.
This information is essential for the design and fabrication of a
research reactor based on metallic fuel elements in the Atomic Energy
Establishment. In an earlier report(8), the general features of
these reactors with particular emphasis on material selection has
been reviewed. These reactors were classified as: 1) graphite moder-
ated air cooled, 2) graphite moderated-gas cooled, and 3) heavy
water moderated reactors.
2. GRAPHITE-MODERATED AIR-COOLLED REACTORS

Those early reactors were built during and after the second world war. The Oak Ridge (X-10) and Brookhaven in USA, Windscale and RCP in UK and G-1 in France are examples of this type of reactors. More information concerning these reactors could be found in the Appendix. Both the X-10 and Brookhaven were research reactors while both the Windscale and G-1 were plutonium producers. However, the Brookhaven reactors fuel was replaced by highly enriched uranium in 1958.  

2.1 GENERAL FEATURES OF THE FUEL ELEMENTS: (4-7)

Fuel elements of these early reactors were composed of U slugs placed end to end and were arranged in horizontal channels. The slugs in the X-10 and Brookhaven reactor were 1.1" in diameter and 4.0" long. The fuel element of the X-10 had 33–54 slugs while that of the Brookhaven had only 33 slugs. In the Windscale reactor the fuel element has only 20 slugs each 0.9" in diameter and 12.6" long. As for the G-1 reactor, there were two fuel elements per channel, 37 slugs per element each was 1.62" in diameter and 3.94" long.

The fuel element features of the G-1 and the Brookhaven reactors are given in Figs. 1 and 2. The amounts of fuel used for the cores of Windscale, X-10, Brookhaven and G-1 reactors were 40.0, 34.0, 30.0 and 100.0 tons of natural metallic uranium, respectively.

The slugs of the X-10 and Brookhaven reactors were canned in 0.035" and 0.03" thick 2 S Al jackets, respectively while aluminium of high purity was used in case of Windscale and magnesium in the G-1. A layer of Al-Si eutectic was used to bond the Al jacket to uranium slug in X-10 and a graphite layer was used in the Windscale slugs. Anodized Al was used as a cladding for the fuel element of the Brookhaven reactor. Mechanical bonding was used for fuel element of G-1 reactor. No fins were required for the low power reactors of
Fig. 1. BNL FUEL ELEMENT

DIMS. IN in.
Fig. 2. THE FUEL ELEMENT OF G-1.

DIMS. IN mm.
Windscale and X-10. The fins of the G-1 acted as supports while those of Brookhaven acted as an extra heat transfer surface. The maximum can surface temperature in Windscale, X-10, G-1 and Brookhaven were 200, 250, 275 and 350°C, respectively. A helium tube was attached to one end of the Windscale and Brookhaven fuel element to monitor leakage. Loading and unloading in all mentioned reactors were performed during reactor shut-down.

2.2. FUEL ELEMENT FABRICATION:

2.2.1 The Brookhaven National Laboratory (BNL); (5)

a) The Slugs:

The uranium slugs (Fig. 1) were fabricated from extruded rods as follows:

1. The extruded 1.2" diameter rods were straightened in a straightener and machined to 1.19" diameter by 4.0" long.
2. The machined slugs were annealed and degassed at 600°C for 12 hrs in an argon atmosphere to decrease the hydrogen content of the metal and to relieve machining and bending stresses.
3. Slugs were machined to the final diameter (1.1")
4. Slugs were degreased in a vapor phase degreaser at 400°C for 4 hrs in an argon atmosphere.
5. Tests were done to ensure the soundness of the slugs.

b) The Finned Tube:

1. The tube was made by extrusion followed by drawing to straighten fins and to give the exact size of the tube.
2. The interior of the tube was anodized in sulphuric acid at the proper voltage, current and time to yield a layer of 0.0008" thick.
3. The end caps were also anodized on the surface in contact with uranium.
4. The interior and exterior of the anodized film was stripped from an area extending 0.25" from the end of the tube to avoid porosity in welding with end caps.

c) The Fuel Element:

After fabrication of the slugs and the finned tubes, the fuel element (Fig.2) was fabricated as follows:

1. Trimming of the finned tube
2. Welding of the butt closure
3. Dehydrating the anodized film
4. Leak testing to determine the weld soundness
5. Loading of uranium slug into the tube. A high tolerance is required for uranium slugs and the finned tube to avoid rubbing during loading.
6. Trimming of the loaded tube leaving 0.725" to receive the brazing cap.
7. Brazing of the helium cap using a filler rod (95% Al - 5% Si)
8. Leak testing to check brazing and to recheck the weld and tube wall prior to hydrostatic compression.
9. Hydrostatic collapsing of the finned tube at 2500 psi followed by air drying.
10. Flame brazing of the anchor.
11. Flame brazing of the helium tube to the fuel element.
12. Testing to determine whether the helium tube and the cartridge were inter-connected, and to ensure the soundness of the helium tube, joints and finned tube.

A schematic flow sheet for the various fabrication process is shown in Fig. 3.

2.2.2. The X-10:

Originally the reactor was fuelled with unbonded Al-jacketed slugs of 5/8-extruded uranium. During the spring of 1952, it was
Fig. 3. A flow sheet for the fabrication of the "BNL" fuel element.
reloaded with rolled, B-treated uranium rods with Al-Si bonding.
The fabrication steps of the bonded fuel element were as follows:

1. The cast bars of uranium were rolled to the required diameter of the slugs and cut to length.
2. The slugs were B-heat treated in a salt bath at 700°C for 3 min, and then chemically cleaned.
3. Metallurgical bonds were made by immersing slugs in a duplex bath consisting of a lead layer in the bottom and Al-Si layer above with flux on top at a temperature of 500°C. Slugs were held for 35 sec. in the lead layer and only for 5 sec. in the Al-Si layer.
4. Following the duplex bath, slugs were immersed in Al-Si bath (11.2 wt. % Si) at the same temperature (500°C) for 13 sec.
5. Cans were fabricated by impact extrusion.
6. Slugs were then inserted in the preheated cans filled with Al-Si.
7. A preheated Al cap prewet with Al-Si is inserted into the end of the can which is then quenched in water.
8. End caps were cut to proper length and welded with an inert gas shield.
9. The finished fuel element is tested by visual inspection and autoclaving at 400°C to observe blisters or ruptures.
A flow sheet of these steps is given in Fig. 4.

2.2.3. The Windscale:
The early fuel charges consisted of cast uranium machined to final size. Later charges were made from uranium ingot billet by vacuum melting and casting into steel moulds washed with a protective refractory coating. Beta heat treatment followed this process to remove the coarse grained structure of the cast rods.
Fig 4. A flow sheet for the fabrication of the "X-10" bonded fuel element.
Bars were then cut to length, machined to size and an anti-diffusion barrier layer of graphite was applied to prevent the uranium diffusion into the Al can and the formation of the brittle compound UAl$_4$(8-10).

The can was made from super purity Al either by machining or extrusion. A hollow cup shaped cap was inserted into one end of the can and welded by an argon arc process. The weld was then rolled inwards by a press tool and is finally covered with Al-Si brazing alloy. After testing the can seal for leak tightness, the interior surface of the can was coated with graphite barrier layer and the slugs were then inserted into the can and the second end was sealed in a similar way as the first one. During the final sealing, helium was inserted through a tube in the cap which was then crimped off and welded. The helium was used to improve the heat transfer and for testing purposes. Tests included long heating, steam autoclaving and mass-spectrometry.

Fig. 5 illustrates the sequence of the fabrication techniques used for this type of fuel elements.

2.3. OPERATING EXPERIENCE:

Air cooled reactors were operated at moderate temperatures of 200°C or less to avoid excessive oxidation of graphite. However, at this low temperature, graphite is susceptible to radiation damage which is manifested chiefly as growth in dimension and in storage of the energy known as "Wigner Energy"(11).

2.3.1 The X-10 Fuel Elements:

Originally the fuel elements of X-10 were not bonded up to 1952. From 1943 to 1952 the percentage failure of slugs was 0.056%(4). Rupture of unbonded slugs began at the center rows of the reactor and proceeded to the periphery. The reasons for such rupture were U oxidation and U-Al diffusion. Operating experience showed that
Fig. 5. A flow sheet for fuel element fabrication of "Windscale Reactor".
Al cans should not be operated at an interface temperature above 225°C to avoid the increase in slug's volume due to the formation of the brittle compound $\text{UA}_{14}$.

From April 1952 up to the end of 1957, 124 bonded slugs has suffered from rupture. The various reasons were:

a) Incomplete 100% $\beta$-transformation during heat treatment

b) Circumferential cracking at one or both ends of the can due to higher stresses resulting from growth of the slugs by oxidation and swelling.

c) Fabrication defects such as variation of Al-Si layer thickness or the presence of some voids in this layer.

2.3.2 The Brookhaven Fuel Elements:

The experience gained in operating the reactor indicated that the occasional failure in the fuel element was primarily due to development of a leak in helium tube, usually in the region of the brazed joint. Both the end caps design and method of joining the helium cap to the tube were changed. The two ends were chemically stripped of their anodic films. Discs of anodized Al of various thickness were put between these caps and the uranium slugs. The helium tube was brazed by a brazing rod (88% Al-12% Si), and the end caps plus helium tube were then arc welded. Up to March 1955, 1499 cartridges had been discharged from the reactor. Although 771 cartridges developed leaks, only 28 cartridge were reuptirrad sufficiently to produce detectable exit air radioactivity.

The Brookhaven fuel elements were replaced by highly enriched uranium in 1958; the fuel element design was also changed into 3 bent sandwiches of Al-8 wt.% U (93% enriched). The power was not changed, while neutron flux was raised.
2.3.3 The Windscale Fuel Elements:

On Oct. 10, 1957, a serious accident occurred in the Windscale reactor. A number of fuel elements in two fuel channels glow red hot and caught fire. The reason of this accident was the annealing operation carried out on graphite core as a routine maintenance to relieve the stored energy in graphite. The excessive temperature caused by the Wigner Energy was high enough to fire the uranium rods (12-15).

The failure of early fuel elements during normal operation was less than 0.1% of the total number of the fuel elements that have passed through the reactor. The various possible reasons for these failures were:

a) Changes at the bar ends due to increase in length up to 0.3" leading to circumferential tears around the end of the can.

b) Thinning of the can wall near the end-cap during manufacturing.

c) Defects in the anti-diffusion layer leading to UAl₄ formation as a result of the diffusion of uranium into the Al can.

d) Wrinkling in coarse grains of cast fuel which in turn caused some local distortion in the can.

However, later charges have been heat treated to remove the coarse grained structure of cast fuel to minimize wrinkling.

2.3.4 The C-1:

The reactor was in regular service for more than 12 years. It was shut down in Oct. 1958 for economical reasons as its design was too old to produce plutonium on an acceptable cost. The amount of plutonium produced per year was 15 kg (6, 7).
8. GRAPHITE MODERATED GAS-COOLED REACTORS

The graphite moderated gas cooled reactors (GCR) discussed here are those reactors which use natural uranium metal rods, sealed in finned magnesium cans, stacked in channels in the graphite core, and cooled by a flow of carbon dioxide. These were developed in the UK and France. For detailed information see the Appendix.

The UK has chosen the gas cooled reactors as the basis of its power program before 1972. It has operated the reactors at Calder Hall, Chapelcross, Bradwell, Berkeley, Hunterston, Hinkley, Trawsfynydd, Dungeness, Sizewell, Oldbury and Wylfa Head. The Calder Hall was a dual purpose plant designed to produce plutonium and generate electricity while all other stations were mainly power producers. The UK has exported two nuclear power stations of the gas cooled type, The first was erected at Latina in Italy, while the second was erected at Tokai-Mura in Japan.

France has designed and operated the gas cooled reactors G-2 and G-3 as plutonium producers for military purposes, and the reactors of EDF mainly for power production. Vandellos, in Spain is of the gas graphite type.

3.1 GENERAL FEATURES OF FUEL ELEMENTS

The general shape of the fuel used in most of these reactors was a solid cylindrical rod with a diameter of over than one inch and length ranging between 11.8" (as in G-2) to 42.0" (as in Sizewell). These of the EDF and Tokai-Mura were closed tubular rods with an inner and outer-diameters of less than 1.0" and 2.0", respectively. Those of Bugey were annular rods with an inner diameter of about 3.0" and an outer diameter of less than 4.0"
with internal and external cooling. However, the CEA in France has
developed a fuel element that was used in the EDF 2-4 and Vandellos
reactors\(^{(16)}\). In this new design a core of graphite of low porosity
was introduced to occupy the inside free volume of closed tubular
uranium rod.

The British adjust the composition of the uranium fuel to yield
a typical analysis of 800 ppm C, 650 ppm Al, 260 ppm Fe, 50 ppm Ni
and 20 ppm Si\(^{(17)}\). The presence of Fe and Al in the so called "adjusted
uranium" decreases the irradiation growth of the fuel, decreases the
mobility of xenon and form a fine dispersion of U-Al and U-Fe compounds
which nucleate the fission gases as very small bubbles in which the
surface tension restrain the swelling at the higher temperature. The
fine precipitates also influence the crystallographic orientation in
such a way that to decrease irradiation growth at lower temperature\(^{(18)}\).

The French fuel of the G-2 and G-3 is a metallic natural uranium
with 0.4% Al. The addition of Al to uranium minimizes irradiation
growth and thermal cycling distortion\(^{(8)}\). The U-0.5 wt.\% Mo alloy was
used in the fuel of EDF-1, the amount of Mo was increased to 1 wt. %
in the following EDF-fuel elements and that of Vandellos. It was
found that the U-1 wt. % Mo alloy creep resistance is one hundredth
less than the heat treated unalloyed uranium\(^{(19)}\). As for the Bagey-1
station the fuel used is Sial-71 alloy (120 ppm Si, 800 ppm Cr, 700
ppm Al and 300 ppm Fe). This alloy has a lower macroscopic absorption
cross section in comparison with that of U-Mo alloys\(^{(17)}\).

The average specific power in the fuel was increased from 1.7
KW/Kg of natural uranium in the Calder Hall to 3.19 KW/Kg of natural
U in the Wylfa Head, 5.54 KW/Kg of natural U in the EDF-3 and 6.0
KW/Kg of natural U in the EDF-3\(^{(17,20)}\). The average burnup of
these reactors varied from 3000 MWD/t up to 4500 MWD/t. An in-
crease in fuel element burn-up by 1000 MWD/t would reduce the fuel
price by 15-20% (e.g. the annual fuel required for the core of the
Hunterson is 50 tons at 3500 MWd/t and 40 tons at 4500 MWd/t). On the other hand the amount of plutonium produced is reduced by about 7% (for the same core, the amount of plutonium was reduced from 102.5 to only 95 Kg(21).

The early cladding material used in the British Reactors was a magnesium base alloy containing 0.8 wt% Al, 0.002-0.05 wt. % Be, 0.008 wt.% Ca and 0.006 wt. % Fe. This alloy was selected for its high oxidation resistance by the coolant Co up to 500°C, in addition to its compatibility with fuel and the good mechanical properties. Lately, the amount of Al in the alloy was reduced to avoid the occurrence of the marked grain growth in can walls. Calcium was eliminated due to observation of weld cracking which occurred during sealing end caps(22).

The cladding material used in some of the British reactors, like Sizewell and Oldbury, was Mg-1 wt% Al alloy known as the Magnox A-12 alloy. The French cladding material was Mg-0.8 wt.% Zn alloy which has good nuclear, mechanical and corrosion properties at the operating conditions as well as better grain size stability compared to the Magnox A-12 alloy(6,7,23).

Various forms of fins (helical, spiral, polyzonal and longitudinal) were used to maximize the heat transfer in those power reactors. In the fuel element of G-2 the fins acted as supports as well as an extra heat transfer area. The maximum cladding temperature varied between 400°C (as in Hinkely point) and 495°C (as in Chapel Cross). A splitter was added to the can in some of the Magnox reactors in order to avoid bowing due to transferred load. This method was used with the cans of Calder Hall, Bradwell, Dungeness, Latina, Trasby, Hinkley, Oldbury and EDF-1. If a small amount of fuel element bow develops, the splitter contacts the channel wall and in turn reduces the bowing rate. The fuel elements of Hunterston, Tokai-Mura, Vandellos and
EDF-1 were supported by horizontal supports linked to graphite sleeves, so the load was transferred to the sleeves and in turn to the channel wall. The fuel element of Berkeley was designed to be supported also in a frame of graphite struts linked by an end piece of Zr-alloy, the upper stud was subsequently replaced by a sliding nose piece located in the graphite struts. The sliding nose allows for differential thermal expansion, due to the growth of uranium under irradiation, and in turn mitigates the bowing problem.

The fuel elements in the G-2, G-3 and EDF-1 were arranged in horizontal channels while in the other reactors were arranged in vertical channels. The number of fuel elements per channel varied from one station to another. The Clader Hall used 8 elements one on top of the other, while G-2 and G-3 used 28 elements end to end in horizontal channel. Loading and unloading was carried out during operation in all mentioned reactors except in the Calder Hall and EDF-1 where it was carried out during reactor shut-down.

Figs. 6-8 illustrate the various types of fuel elements used in some of the gas cooled graphite moderated reactors.

3.2 FUEL ELEMENT FABRICATION:

The fuel element cores of the Magnox reactors were fabricated by the casting techniques and machined to the final size. Initially casting was done in cold moulds, but recently hot moulds were preferred as it minimize piping of the product. Examples are given below:

3.2.1 Calder Hall:

In case of the British Calder Hall reactors the rod of metallic cast uranium was machined to the required diameter with circumferential grooves to assist in locking the fuel and cladding together. Heat treatment was applied to improve the irradiation stability and mechanical properties. Heat treatment consists of B-quenching from 700°C in a water
Fig. 6. THE FUEL ELEMENTS OF G-2, BRADWELL AND EDF-1
Fig. 7. THE FUEL ELEMENTS OF EDF-2, HUNTERSTON, AND BERKELY REACTORS
Fig. 8. THE FUEL ELEMENTS OF DUNGENESS AND CALDER HALL REACTORS

(a) DUNGENESS FUEL ELEMENT

(b) CALDER HALL FUEL ELEMENT

DIM. IN in.
spray, annealing at 500°C vertically in an argon atmosphere, removed at 150°C and supported on "Vee" block in order to reduce the small amount of residual distortion. The grain size of the manufactured bar is very fine (\(<0.01\) mm) within 2-4 mm of the rim and much coarser (1.0 mm) in a core about 8 mm diameter \((8)\). End caps were then edge welded to the clad. A good contact between the can and the fuel was achieved by means of isostatic pressure with \(\text{CO}_2\) at 500°C \((17)\).

The grain size of the cladding material was controlled to meet the requirements of various temperature regions. Cladding material with fine grain size was used in regions having temperature below 350°C, while materials with coarse grain size was used in regions above 250°C. The fine grained structure clad has higher ductility at the lower temperature since this was required in order to accommodate the low strain rate conditions which may lead to cavitation failure. The coarse grained structure clad was used above 350°C where high creep strength is required to resist the gas pressure in the high temperature regions \((17,24)\). Following the can grain size control operation, end fittings were screwed on and a braze consisting of three longitudinal struts was attached to the fine.

Fig. 9 shows a flow sheet of fuel elements fabrication of the British Calder Hall reactor.

3.2.2 G-2 and G-3:

The French technique used for fuel fabrication of G-2 and G-3 fuel element did not differ much from that of the Calder Hall. Fig. 10 shows a flow sheet of fuel element fabrication of the G-2 and G-3 reactors \((17)\).

3.2.3 EDF Reactors:

In case of the EDF, the tubular U-Mo rod was grooved, forced into the can in a gas autoclave. The element was then vacuum annealed at
Fig. 9. A flow sheet for the fabrication of the "Calder Hall" fuel element.
Fig. 10. A flow sheet for the fabrication of the “G-2 & G-3” Fuel elements.
570°C to restore cladding ductility. The new fuel element used in the EDF reactors, with the exception of Bugey 1, was formed by casting U-Sicral F-1 alloy in vacuum around a graphite core of low open porosity. Then it was heat treated and machined before putting in a machined sheath with herring-bone fins, which was closed by two welded end caps. Bonding was achieved by hydraulic and hot pneumatic application of high pressure.

3.3 OPERATING EXPERIENCE

The Magnox reactors were operated above the temperature of the peak stored energy release (approximately 200°C) and below the Magnox alloy upper limiting oxidation temperature (approximately 500°C). There was no fear of firing of uranium rods as in the Windscale as there was neither significant nor effective stored energy. It could be concluded that uranium rod swelling was the major life limiting factor of the Magnox fuel element. The failure rate of the Magnox reactor in UK has been about 10⁻³ of the fuel elements loaded. Fin deformation, as a result of the stretching action of the growing oxide film, represents a short fall of the fuel element in the Magnox reactor.

3.3.1 Calder Hall

The first Calder Hall fuel element had a can with fins which was in the form of a single start helix. These fins were replaced by multi-start helical fins divided into quadrants by a full length splitter, when these elements were loaded into the reactor, failure of the cans occurred in the hottest reactor region. The failure was attributed to fatigue arising from fuel element vibration in the gas stream. The solution to that problem was by the addition of a Nimonic spring to the top end fitting.

At the start of the Calder Hall operation, supports to maintain the rigidity of rods were used in the end fittings of each rod. Discharged
rods have showed that these supports were insufficient to prevent bowing at the center of each rod. The reasons for such bowing were:

a) load transferred from one rod to the other below
b) total irradiation
c) initial bow
d) temperature. Bowing was found to be proportional to the first three reasons but the variation of bowing with temperature is not precisely known. Operating experience showed that the second rod from the bottom suffered the greatest bowing. Bowing problems were solved by fitting a magnesium braze to the mid section of each fuel element. Thus the middle part of the fuel element was given lateral supports and hence halving the length. Discharging of fuel with lateral supports was more easy and no jam has been reported.

Up to 1958, only 13 fuel elements have failed out of 30,000 elements passed through the reactor. Possible reasons for such failure were:

1. Poor welds at the end caps enabling fuel elements to pass test and fail after a slight straining.
2. Leaks in welds during manufacturing.
3. Small void accompanied by small cracks along some grain boundaries in the can wall due to radiation induced growth in the uranium.

The flow of coolant past the fuel element causes drag force which may deform the can. However, deflection at 400°C was not more than 0.025" during 300 days working life of the fuel element.

3.3.2 Berkeley:

The most common modes of deformation observed in fuel elements of the station were fin waving, splitter movement and brace lift. Long leak paths were also observed in damaged end cap welds which led to the oxidation of the fuel and in turn can rupture. The total number of failed elements during operation up to 28 March 1968 were 117 elements out of 183154 charged elements giving a failure rate of 6x10^-4. The maximum increase in diameter of the element due to irradiation at
4000 MWd/t was about 1% and in length about 0.4% at 1000 MWd/t. Surface wrinkling detected was less than 0.005" and bowing was about 0.08" (17).

3.3.3 Bradwell:

The main problem of the fuel elements used in this station was the distortion of the heat transfer surface; maximum distortion being in the lowest three positions in the channel. The bottom three elements deformed by irradiation creep combined with the deformation caused by texture. Uranium rods suffered from bowing which made the axial shuffling difficult. The total number of failed elements during operation up to 28 March 1968 was 25 elements out of 83219 charged elements giving a failure rate $\leq 3 \times 10^{-3}$ (17).

3.3.4 Dungeness:

The problem of low ductility of Magnox cans at low temperature was solved in that station by raising the inlet gas temperature to 250°C. This remedy has improved the life time of fuel elements in the bottom half of the core due to reduction of irradiation growth of uranium and the increased ductility of the clad. The total number of failed elements during operation up to 28 March 1968 were 27 elements out of 61128 charged elements (17).

3.3.5 The G-2 and G-3:

Little information are available about these reactors. However, it was reported that the resulting Pu diffusion through the can varied between 3000 ppm at the fuel clad interface to 200 ppm approximately at a depth of 1.2 m at a burn-up of 3777 MWd/t and a can temperature of 450°C (18).
3.3.6 The EDF 1, 2, 3 and 4:

The clad of EDF's 1-4 fuel element has suffered from Pu diffusion at about 500°C. This problem was attributed to the fairly high diffusion coefficient of Pu in the cladding material, which is greater than 200 times the U-diffusion coefficient. However, this problem has been mitigated by putting a layer of 5 µm of graphite between the clad and the fuel for all French reactors of the Magnox type except the Bugey-1\(^{(17)}\). As for the Bugey-1, a metallurgical diffusion barrier of an intermediate layer of Al deposited by flame spraying was used\(^{(18)}\). 20,000 units have been loaded in the core of this reactor with no sign of troubles\(^{(16)}\).

The EDF-1 reactor was shut down in 1973 for fuel replenishment, while the other French gas-graphite reactors are still loaded and unloaded continuously during operation\(^{(2,16)}\). With the exception of Bugey-1, the fuel renewals included axial rearrangement of the least irradiated element achieving a fuel economy of about 12%. The same procedure was applied to Bugey-1\(^{(18)}\).

The lugs incorporated in the can to fix the cartridge in its graphite sleeve have suffered from fracture at the root of the joint between the can and the lug. This was due to the disturbance created in the flow of heat transfer leading to fatigue failure of the lugs. The solution to that problem consisted of i) blocking the lugs in the jacked sleeve groove by undulating the top of the lug to prevent the play in the sleeve groove, and ii) reducing the distances between the cartridges and in turn reducing the disturbances and stresses which arise in the space between them\(^{(25)}\). Up to March 1968 it was reported that 12 failures have occurred in the EDF-1 due to fuel handling, 2 failures only have occurred in the EDF-2, 10 failures in the EDF-3, and 8 failures in the EDF-4. In case of EDF-4, 2 of
the failures has resulted from damage in handling during loading of
the first core, 2 failures has appeared in elements equipped with
cladding temperature measuring devices, and the remaining 4 failures
in tubular element burned beyond 4500 MWd/t and were probably due
to the use of this fuel near its limit capabilities. However, the
tubular element was replaced by the new designed element with gra­
phite core in the reactors of EDF-2; 3 & 4 and Vandellós. At early
Sept. 1974 no significant fuel failure was reported (19).

3.4 GENERAL PERFORMANCES

The Magnox reactor system has the largest cumulative reactor
experience out of 1000 reactor-year experience of all reactor types.
The Magnox reactors have achieved a record of 410 reactor-year.

The load factor (the ratio of gross electricity generated in a
plant to the gross design output figures) is a good parameter for
measuring the performance. The average annual load factor for 1976(2)
was 84.94% for the Magnox reactors which was higher than that of
pressurized water reactor (61.38 %) and the Boiling water reactors
(59.20%). It is only superceded by the pressurized heavy water reactor
which had a load factor of 77.6%. The cumulative load factor,
calculated over the entire period of operation of the station reported
for 1976(2) showed that the Magnox reactors had the highest cumula­
tive load factor (61.01%) compared to PHWR (54.73%) and BWR(51.40).
In all reactor systems, cumulative load factors are less than their
annual load factor indicating that all system current achievement exc­
eds their long term achievement.
4. HEAVY WATER ACCELERATED RESEARCH REACTORS

Countries with limited sources of uranium has paid great attention to heavy water reactors which operate with minimum initial fuel inventory as well as minimum annual fuel consumption per every MW(t) installed. Particular attention to this kind of reactors has been paid mainly by Canada and India (25,27). Canada constructed a high flux research reactor (40 MWt) in 1949. This was followed by a more powerful reactor, NHR (200 MWt) which was completed in 1957. Under the Canadian-Indian cooperation, India has operated the CIR (40 MWt) of the NRX type in 1961 and a recent reactor is under construction of this system NHR (100 MWe). Both the NRX and CIR used light water as coolant while the NRU uses heavy water as coolant as well as a moderator (28).

The early experience in those reactors in Canada led to the development of NRU reactors with oxide fuel elements. These were the base of the development of the successful commercial CANDU reactors which are in operation or under construction in Canada, India, Korea, Pakistan, Argentina. The report deals with early research reactors and the CANDU system will be treated in a separate report.

Czechoslovakia and USA has operated heavy water moderated reactors named Bohunic A1 and Russian Organic reactor, respectively. The Bohunic A1 reactor and the Russian reactor used heavy water as a moderator but CO₂ as a coolant for the fast and an organic coolant for the second (30). USA has given little attention to this kind of reactors. It has operated the Savannah River reactors for Pu production with D₂O as both coolant and moderator.
4.1 GENERAL FEATURES OF FUEL ELEMENTS:

The design features of fuel elements for the above mentioned reactors depend on the average specific power in the fuel. Due to the high average specific power of the 110 MW(e) gas cooled heavy water moderated reactor (GCHWR) of Czechoslovakia, 32.8 Kw/KgU, the fuel has been made in form of pins, 4 mm diameter (30). The early fuel element of the NRU was a flat bar of natural metallic uranium up-till Nov. 1963 where it was converted to highly enriched fuel (98% U^{235}). The shape of the fuel elements was changed into tabular and pencil rods. As for the NRX the original fuel element was a solid rod similar to that of the CIR, 3.45 cm diameter and 305.0 cm long. The fuel was changed in the spring of 1962 from metallic into natural ceramic UO_2 rods in the core outer zone, annular UO_2 rods plus enriched rods in the intermediate zone and enriched rods in the central zone (28). Table (1) gives the details of the fuel rods currently used in both the NRX and NRU reactors. Fig. 11 gives the shape of the fuel element of the Czechoslovakian HWGCR. The Russian organic reactor had a fuel of thick walled annular uranium core (29).

In the NRU reactor, 200 MW(t), the core loading was only 11 tons of natural metallic uranium while that of the early air cooled graphite moderated X-10 reactor, 4 MW(t), was 54 tons natural metallic uranium. The average burn up is 900 MWD/t for the NRX, 3000 MWD/t for the Czechoslovakian (Bohunic A1) and 8000 MWD/t for the Organic Russian Reactor (R-1) (30, 31).
Fig. 11. FUEL ELEMENT OF THE CZECHOSLOVAKIAN HWGCR.
### Table (1)
The Details of Fuel Rods of Both NRX and NRU\(^{(28)}\)

<table>
<thead>
<tr>
<th>Material</th>
<th>NRX</th>
<th>NRU</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Solid</td>
<td>Annular</td>
</tr>
<tr>
<td>UO(_2)</td>
<td>UO(_2) rods</td>
<td>pencil rods rods</td>
</tr>
<tr>
<td>U(_{285}) content</td>
<td>Nat.</td>
<td>Nat.</td>
</tr>
<tr>
<td>Outside diameter, cm</td>
<td>3.58</td>
<td>3.58</td>
</tr>
<tr>
<td>Inside diameter, cm</td>
<td>---</td>
<td>1.53</td>
</tr>
<tr>
<td>Density, g/cm(^3)</td>
<td>10.4</td>
<td>10.4</td>
</tr>
<tr>
<td>Thickness, cm</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td>Cladding:</td>
<td>Al</td>
<td>Al</td>
</tr>
<tr>
<td>Thickness, cm</td>
<td>0.127</td>
<td>0.127</td>
</tr>
<tr>
<td>Coolant:</td>
<td>H(_2)O</td>
<td>H(_2)O</td>
</tr>
</tbody>
</table>

The R-l and Czechoslovakian fuel was clad with Mg-Be powder\(^{(9)}\). A layer of Ni is used as a metallurgical bond in case of the early NRU fuel element to prevent U diffusion and the formation of the brittle UAl\(_4\) compound at the working temperature\(^{(28)}\). The high absorption cross section of Ni compared to Al (\(\sigma_{\text{a}}\text{Ni} = 4.8 \text{ b}, \sigma_{\text{a}}\text{Al} = 0.23 \text{ b}\)) limited the thickness of this layer to about 20 \(\mu\text{m}\) only. Deposition of this thin layer was made by electropolating\(^{(31)}\).
The Savana River Reactor fuel elements were in the form of slugs 1.0" in diameter and 8.0" long, clad in 2S Al of 0.04" thick. 

4.2 FUEL ELEMENT FABRICATION 

4.2.1 The CIR 

The main steps of fabrication of this type of fuel element are as follows: 

1. Melting and Casting: The size selected for the bars to feed the rolling mill was 3.0" in diameter and 36.0" in length. Melting was carried out by induction furnace under vacuum. Molds were coated by thorium to reduce carbon pick-up which was found to be about 500 ppm. 

2. Alpha Rolling: Cast billets were rol-I-rolled in a salt bath at a temperature above 450°C. The diameter was reduced in two stages from 3.0" to the 1.45" in diameter. The rod was then quenched and cut to the proper length and examined for any surface defects. 

3. Heat Treatment: The structure of uranium resulting from an-rolling was fine but with preferred orientation which had a low irradiation stability. In order to improve irradiation stability the structure of the rod must be fine and also free from preferred orientation. This was achieved by B-heat treatment at 730°C. 

4. Machining: The rod was straightened on a straightener and rough ground on centerless grinder; Both ends were than threaded and the rod finish-ground to 1.36 in diameter. 

5. Canning: The finished rod undergoes series of cleaning and inspection. The Al end plugs were screwed on its two threaded ends, and the rod was then put into finned 1S Al tube, and draw through a die to secure good heat transfer. The tube was then cut to approximately 0.3" longer at each end than the Al end plugs, each end was then rolled over the end plugs and the two outer Al plugs were screwed
one at each end. End closure was carried out by argon welding. Fig. 12 represents the fabrication flow sheet of the CTR fuel element.

4.2.2. The Czechoslovakian GCHWR (Echunic Al Reactor)

The fuel elements of the Czechoslovakian reactor were fabricated by forging, or by vacuum rolling, or by pressing the billets in Cu envelopes with subsequent drawing through a die until the necessary dimensions were reached. The rods were sheathed by a protective can of Mg-Be alloy 0.245 mm thick by vacuum distillation. The Mg-Be alloy condensed on the surface of the wire in the form of a thin layer.

4.2.3 The N3X

The rods were χ-rolled at 538 to 649°C on continuous mills. The rods were then subjected to a randomizing E-treatment by heating at 733°C in salt or lead bath followed by quenching in water. The 15 Al finned tubes were made by extrusion. Sheathing was done by inserting the fuel into the tubes and drawing the tubes through a circular sinking die to press the Al into contact with the U-cores. End caps were fixed by welding.

4.2.4 The NRU

The early fuel element of the NRU was fabricated by rolling in the χ-range, heat treated to achieve fine grained structure. As for the can it was fabricated by extrusion and machined to the final size.

4.3 OPERATING EXPERIENCE

4.3.1 The NRX Reactor

The NRX was the oldest high flux research reactor still in regular operation although its flux is now surpassed by other reactors such as NRU. It was badly damaged in Nov. 1951. The flow of coolant to about 1/10 the fuel rods has been reduced for experimental purposes.
Uranium Ingots
Vacuum Melting and Casting
Ultrasonic Testing
Salt Bath Preheating
Rolling
Inspection
β. Heat Treatment
Straightening
Rough Centreless Grinding
Threading
Final Centreless Grinding
Cleaning
Inspection

Al Block Machining
Threading of Al Inner Plugs
Screwing
U Rod in Al Sheath
Drawing
Ends Cutting
Ends Rolling

Threading the Outer Al Plug
Screwing on the Outer Plug
Plugs Welding
Radiography
Glycol Test
Cleaning & Storing

Cleaning & Storing

Fig. 12. Fabrication flow sheet of “CIR” fuel element.
Removal of samples being irradiated caused a great rise in reactivity. The resulting rapid temperature rise melted the double wall tubing around some of the fuel elements as well as portion of the rods themselves. 14 months were required to decontaminate the building (15). Operation was restored in Feb. 1955. An accident in 1955 involving Pu-Al diffusion and in turn the heavy water was degraded to about 97% D₂O (29).

Since 1947 up to 1958, 66 fuel elements failures were observed due to splitting of the inner sheath. These failures have occurred in the central portion of the calandria where higher flux and temperature exists (4). B-heat treatment eliminated preferred orientation and the rod has a greater dimensional and irradiation stability. However, during B-treatment considerable surface roughening has occurred. Irradiation conditions of U rods in the NEX from 1947 to 1958 are given in Table (a).

Table (a)

Irradiation Conditions of Natural Uranium Rod in the NEX Reaction (1947-1958) (4)

<table>
<thead>
<tr>
<th>Number of rods</th>
<th>Metallurgical condition</th>
<th>MWD output</th>
<th>Peak irradiation</th>
</tr>
</thead>
<tbody>
<tr>
<td>1037</td>
<td>As rolled</td>
<td>148.7 max.</td>
<td>616</td>
</tr>
<tr>
<td></td>
<td></td>
<td>36.5 av.</td>
<td></td>
</tr>
<tr>
<td>392</td>
<td>B-heat treated</td>
<td>414.0 max.</td>
<td>760</td>
</tr>
<tr>
<td></td>
<td></td>
<td>48.3 av.</td>
<td></td>
</tr>
</tbody>
</table>

From the previous table it can be seen that B-heat treated rods have higher MWD output compared to rolled rods. However,
operating policy has required replacement of rods at specific irradiation such as 50 MWd, however, rods have been allowed to remain longer.

It was possible to reduce the thickness of Al cladding from 0.08" to only 0.04" and of the cooling tube from 0.1" to only 0.07". Experience has indicated that failure has occurred in both thick and thin sheath(4).

4.3.2. The NRU Reactor:

The NRU reactor was originally fueled with natural uranium up to Nov. 1963 where it was converted to highly enriched U-Al fuel. The major problem in the early operating of NRU was the failure of fuel rod sheath due to the diffusion of Al through U. However, this problem was eliminated by the use of an intermediate Ni bond(28).

5. DISCUSSION

Natural uranium metallic fuel elements used in graphite moderated (gas cooled) or heavy water moderated reactors are of two type:

a) Pure uranium clad in high purity Al as in the early air cooled graphite moderated reactors. The fuel was made in the form of slugs or rods.

b) Dilute uranium alloys or adjusted uranium clad in Mg alloys, as in the gas-graphite (Magnox Reactors) in UK and France. The fuel was made in the form of solid or hollow rods.

The first reactors types were used for research reactors and for Pu production, while the second types were used mainly for power reactors.

The Mg alloys clad used in the Magnox or gas-graphite reactors fuel elements have some advantages over that of Al alloys clad. From
the nuclear point of view, the microscopic absorption cross section of 
Mg is about 40 times less than that of Al. Furthermore these fuel 
element types can be utilized at clad-fuel interface temperature of 
about 500°C. On the other hand the Al clad fuel element in air cooled 
reactors should not be operated at an interface temperature above 
225°C to avoid the formation of brittle U-Al intermetallic compound 
UA1. Fuel slug clad with anodized Al can be used at higher tem­
perature (350°C); however, these are not commonly used. In water­
cooled reactors the maximum clad surface temperature was lower (135°C) 
than air cooled reactors due to corrosion.

In case of Al clad U fuels, fuel clad metallurgical bonding 
was used between U and Al (in air cooled reactors); Al-Si eutectic 
was used in the X-10 to prevent oxidation and U-Al diffusion.

Control of thickness of the bonding layer is important, variations in 
the thickness can cause clad failure. Fineline fuel element had a 
graphite layer between U and Al to act as a diffusion barrier to 
prevent, also, U-Al interdiffusion. Defects in this layer lead to 
UA1 formation. In case of using anodized Al as clad, mechanical 
bonding (hydrostatic bonding) was used. In case of water cooled fuel 
(NRX) mechanical bonding by drawing was used. Nothing was mentioned 
in the literature about the presence of an intermediate layer. The 
early NRU fuel had, also no metallurgical bond; however, a thin Ni 
layer (20 μm) was used to prevent U diffusion and the formation 
of intermetallic compounds. Published data did not mention the maximum 
temperature of the clad fuel interface with this Ni bond.

In case of Mg clad fuel in the gas-graphite or Magnox reactors 
no bonding was required since Mg does not interact with U. However, 
a graphite layer (5 μm) between fuel and clad was used in the 
French reactors (G-1, 2, 3 and EDF-1, 2, 3, 4) to prevent Pu and U
diffusion through the cladding material which is considerably at operating temperature (500 °C). In one of the French reactors (EDF-G), an intermediate layer of Al deposited by flame spraying was used. Tests have shown the effectiveness of this layer at a burn-up of 600 MWD/t and a temperature 500 °C. It is interesting to note that information available on the UK MAINE reactors fuel does not mention the use of an intermediate graphite layer between fuel and the clad.

Fabrication of early fuel elements in slug forms was made by extrusion of U (ENL and early X-10). Later on, the fabrication technology adopted was either rolling of cast billet or direct casting into fuel rods. The rolling was used in later charges of X-10, X-12 and NRX. Direct casting was used in the Windscale fuel (pure U), the fuel (dilute U alloys) of the Calder Hall and MAINE reactors in UK and in the C-1 and following gas-graphite reactor in France. Casting was undertaken in metallic or graphite moulds (cold or hot). Little machining is needed to get the required rod diameter. If circumferential grooves are needed to assist in locking fuel and clad together, as in the EDF-fuel, it is done in the machining stage.

The presence of texture of preferred orientation resulting from hot working in the γ-range cause dimensional instability due to irradiation growth arising from the anisotropic nature of U. Coarse grained structure such as produced after working in theγ-range or by casting lead to surface wrinkling (surface roughening and pimpling) by irradiation. This creates large local strains which can cause clad failure.

Beta heat treatment is essential for controlling the microstructure. B-treatment followed by water quenching reduce the preferred orientation. Subsequent α-annealing cause nucleation with the formation of many fine grains which are randomly oriented. Beta treatment
...should be complete as insufficient B-treatment leads to fuel failures.

The microstructure of U can be further controlled or adjusted by introducing some additives in the ppm range such as Si, Cr, Al, Fe, Mo, and Ni.

These additives cause the formation of finely dispersed precipitates in the matrix. Fission gas produced during irradiation are nucleated as very small bubbles around these dispersions. Reasonable swelling was observed in the adjusted U metal with the addition of 300 ppm C, 400 ppm Al and 300 ppm Fe, followed B-quenching and \( \alpha \)-annealing. The increase in volume for this alloy was related to the burn-up by the relation:

\[
\frac{\Delta V}{V} = 7 \times 10^{-8} B^{3/2} + 2.3 \times 10^{-6} B + 0.0018
\]

where \( \Delta V \) is the change in volume, \( V \)

\( B \) is the burn-up MWd/t.

Natural U metallic fuel under irradiation has suffered from the anisotropic growth, wrinkling, swelling and thermal creep leading to bowing in the fuel elements. Great improvements were obtained in this fuel by suitable heat treatment and alloying.

In the early air cooled reactors, incomplete B-treatment of U (in the X-10) led to failure of cans. Defects in the Al-Si bonding layer during fabrication (in X-10 and Windscale) resulted in rupture of some cans. Defects in welding (in ENL and Windscale) led to leaks and in turn U oxidation. Changing the design and method of joining (in ENL) improved the quality of welding. Careful testing after the welding operation is necessary to reduce the failure rate due to welding defects.

The fuel elements of the British Magnox reactors has suffered from the distortion and waving of the fins. It was found also, that the creep
The problem leading to failure of some cans could be improved by reducing grain size in the can. At present all elements which are supplied for these reactors have cans with a relatively fine grained structure for the lower temperature positions in the reactor, while all higher temperature elements have a relatively coarse grained structure.

The French fuel elements of the Magnox type has suffered from Pu diffusion through the cladding, but this problem was overcome by spraying an Al layer on U rod, and putting a graphite layer between the cladding and U rod. It is interesting that this problem, of Pu diffusion, did not attract the British attention.
<table>
<thead>
<tr>
<th>Plant Name</th>
<th>Reactor Name</th>
<th>Out put</th>
<th>Date of operation</th>
<th>Fuel type</th>
<th>Pin Dims</th>
<th>Max Can/Fuel</th>
<th>Cladding material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Japan</td>
<td>Tokai Mura</td>
<td>587</td>
<td>1960</td>
<td>C . G</td>
<td>9.2 x 2.93</td>
<td>473/573</td>
<td>Magnex, 2.0</td>
</tr>
<tr>
<td>Spain</td>
<td>Vandellos</td>
<td>1750</td>
<td>1972</td>
<td>G . G</td>
<td>5.9 x 2.0</td>
<td>540/690</td>
<td>Magnex, 1.0</td>
</tr>
<tr>
<td>Czechoslovakia</td>
<td>Bohunic A-I</td>
<td>560</td>
<td>1971</td>
<td>G-HW(c)</td>
<td>4.1</td>
<td>0.63</td>
<td>Mg - Be, 0.5</td>
</tr>
<tr>
<td>Canada</td>
<td>N A X</td>
<td>1947</td>
<td>1957</td>
<td>W-HW(d)</td>
<td>10.5</td>
<td>130/545</td>
<td>AL, 1.0</td>
</tr>
<tr>
<td>India</td>
<td>C I R</td>
<td>40</td>
<td>1965</td>
<td>W-HW(e)</td>
<td>10.5</td>
<td>130/595</td>
<td>AL, 1.0</td>
</tr>
<tr>
<td>USSR</td>
<td>0-HW(f)</td>
<td>100</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Mg - Be</td>
</tr>
</tbody>
</table>

(a) Air cooled - Graphite Moderated  
(b) Gas cooled - Graphite Moderated  
(c) Gas cooled - Heavy Water Moderated  
(d) Water cooled - Heavy Water Moderated  
(e) Heavy Water Cooled and Moderated  
(f) Organic Liquid cooled - Heavy Water Moderated  
* The Reactor was shut down  
** The Reactors are cooled EDF4.
REFERENCES


(7) Center de Production de Plutonium de Marcoule, C.E.A.


(14) Accident at Windscale No. 1 pile on October 10, 1957, Nucleonics, Vol. 15 No. 12, (1957).


(20) Directory of Nuclear Reactors, Vol. 4, Vienna.


