

*Boiling-light-Water and
Organic-liquid-cooled
CANDU Reactors
(CANDU-BLW and
CANDU-OCR)*

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INTRODUCTION

When the Canadian nuclear power program was beginning to take shape the choice of the moderator, heavy water, was clear but the choice of the coolant to go with it was not. Consideration was given to pressurized heavy water, boiling light water, steam, fog (steam with entrained water droplets), organic liquid, carbon dioxide and helium. Even after pressurized heavy water was chosen there were reservations about its high cost and the ability to control leakage to a degree where heavy-water makeup costs would not be prohibitive. Accordingly, it was considered prudent to initiate work on alternative coolants as backup to heavy water. The alternative coolants chosen for backup study were boiling light water and organic liquid; reactors based on those coolants were designated CANDU-BLW and CANDU-OCR. Serious work on the concepts started in the early sixties and late fifties respectively.

CANDU-BLW

The conceptual natural-uranium-fueled CANDU-BLW power reactor (see figures 16.1 and 16.2) was similar to the CANDU-PHW (see chapter ten) in that the heavy water moderator was contained in a large, low-pressure calandria and the fuel and coolant were contained in pressure tubes. However, in the CANDU-BLW, the pressure tubes were oriented vertically in the core and the fuel bundles, which were strung on a central support tube, could be inserted and removed from the bottom only. As the light water coolant passed upward through the core it boiled and exited the core at about 290°C, 7.7 MPa¹ and 20 percent quality (i.e., with 20 percent of the water converted to steam). The coolant was then directed to a steam separator from which the steam passed directly to the turbine generator. The water from the separator was mixed with the steam condensate from the turbine generator and recirculated. To obtain a satisfactory energy output (burnup) from each fuel bundle, the fuel strings had to be removed periodically to the fuel bays, where the bundles would be rearranged on the central support tube and returned to the reactor for further irradiation.² The potential attractions of the CANDU-BLW (relative to the CANDU-PHW) were that it would have a lower capital cost because of less heavy water, no steam generators and a slightly higher heat-to-electricity conversion efficiency, and it would have lower operating costs because

¹ This is to be compared with 310°C, 10 MPa and close to 0 quality in the CANDU-PHW (see chapter ten).

² This is accomplished in the CANDU-PHW by bidirectional fueling.

it eliminated the problem of heavy-water leakage from the high-pressure coolant system. The main potential difficulties were that the coolant absorbed more neutrons than the pressurized heavy-water coolant, making it more difficult to achieve an economical burnup in natural uranium fuel, and that boiling led to variable neutron absorption in the coolant, making the reactor difficult to control.

Accordingly, development work focused on designs of fuel and fuel channels that would maximize neutron economy, on definition of the limits of steam quality that could be safely used and on methods of controlling the reactivity variations that would result from inevitable differences in boiling rates in the fuel channels. Methods also had to be developed to control coolant chemistry in the boiling regime, so that excessive corrosion and solids deposition on heat transfer surfaces could be avoided.

The fuel design chosen (see figure 16.3) was similar to the successful CANDU-PHW design (see chapter thirteen), except that the fuel pins were fatter, to improve the uranium dioxide to zirconium ratio, and the fuel pin spacing was less, to reduce the amount of coolant in the channel. Because of the fatter fuel pins, the maximum channel power had to be reduced by about 20 percent relative to the CANDU-PHW, to keep the uranium dioxide temperature near the centre of the pellets at acceptable levels. This meant that the CANDU-BLW had to have about 20 percent more channels than a CANDU-PHW with the same power output. Further, because of the close element-to-element spacing, the sheath strain generated by the volumetric expansion of the uranium dioxide and the pressure from released fission gases had to be severely limited. This was aided by using hollow end plugs to provide extra space for the fission gases. Irradiation testing

FIGURE 16.1
Schematic Illustration of a CANDU-BLW Power Station

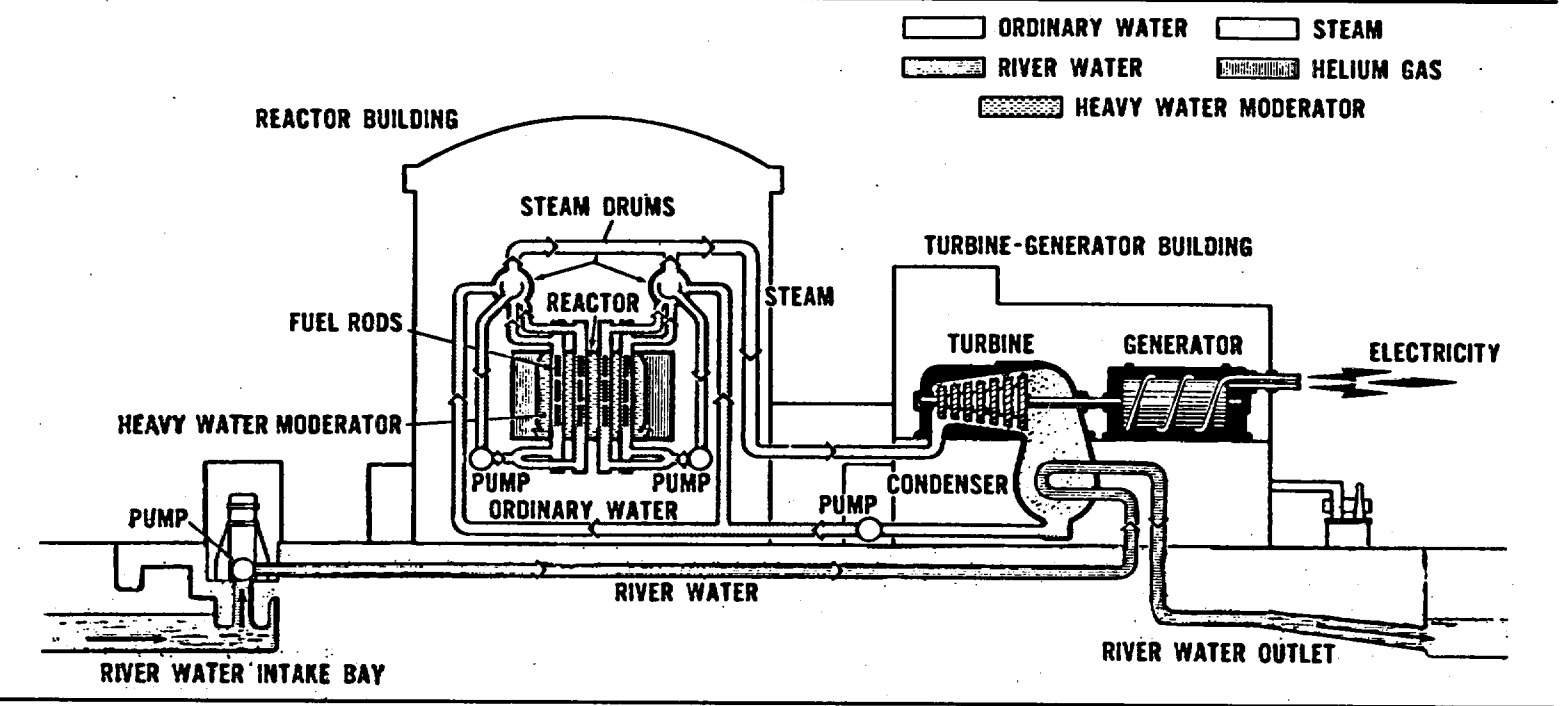
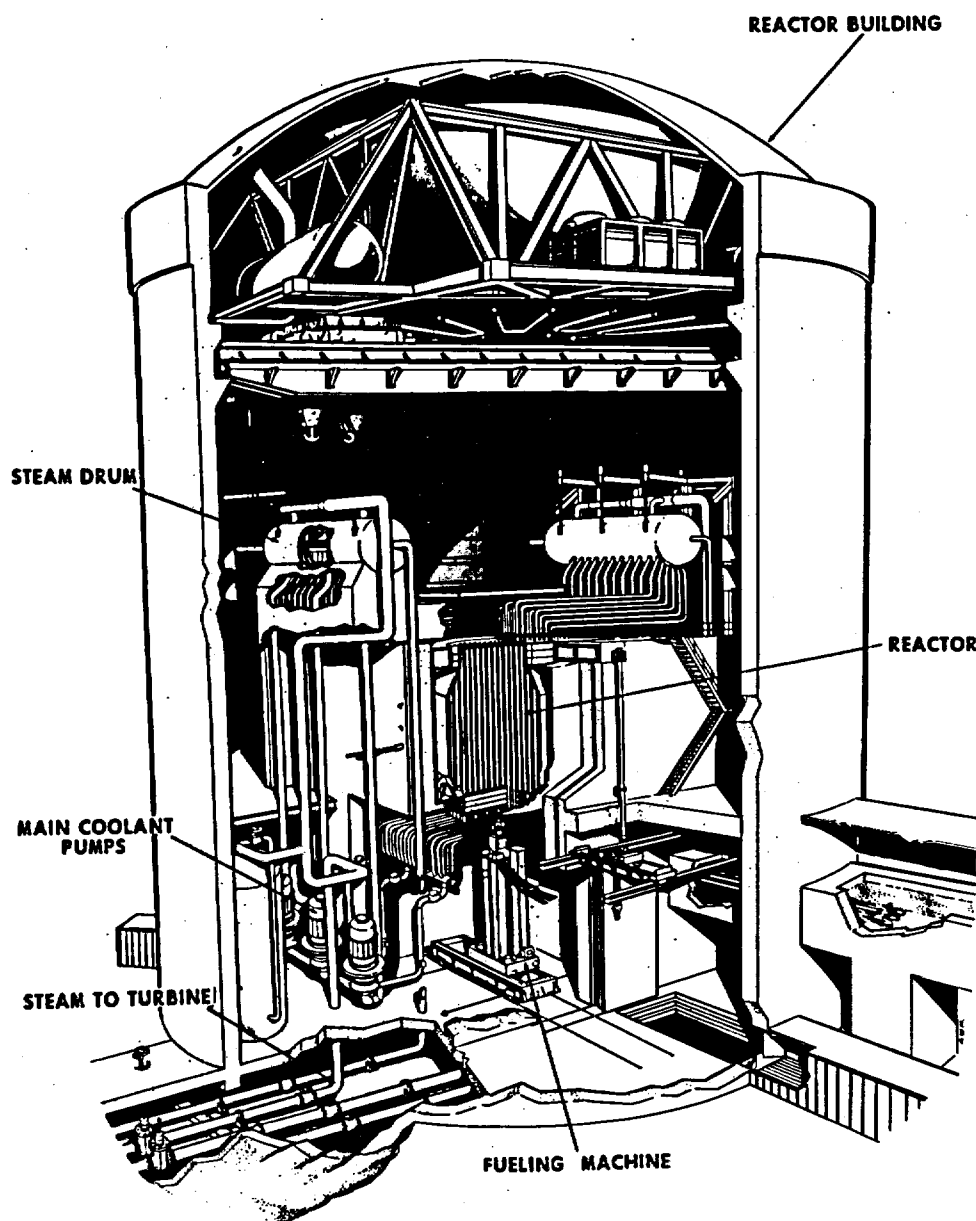


FIGURE 16.2
Cutaway of a CANDU-BLW Reactor Building



of fuel pins of this general design started in boiling light water loops in NRX in 1961, and by 1964 full bundles were being irradiated in a large boiling-light-water loop in NRU. In general, the fuel performed as expected and the tests verified that fuel of this design could give acceptable performance in a boiling-light-water-cooled reactor. Further, the fabrication of the test sections by the CANDU-PHW fuel suppliers confirmed that fuel for CANDU-BLWs could be manufactured on the same fuel fabrication lines as the CANDU-PHW fuels, with only minor alterations. This gave confidence that the fuel for CANDU-BLWs could be fabricated at the low costs that were already being achieved for the CANDU-PHW fuel. However, even with all the precautions taken to save neutrons, the average discharge burnup from a natural-uranium-fueled CANDU-BLW power reactor was expected to be only about 180 MWh/kgU, compared with 240 MWh/kgU for a CANDU-PHW. Taking into account slight differences in fabrication cost, the overall fueling cost was expected to be about 25 percent higher. This, at least partially, offset the expected lower capital and operating costs.

As for the fuel, the fuel channels for the CANDU-BLW (see figure 16.4) were similar to those already developed for the CANDU-PHW (see chapter twelve). The pressure tube material was Zr-2.5Nb, as chosen for the CANDU-PHW, but because of the difficulty in achieving acceptable neutron economy in the CANDU-BLW, it was used in the heat-treated rather than the cold-worked state. The increased tensile strength provided by the heat treatment allowed the thinnest possible tube to be used (chapter twelve). The

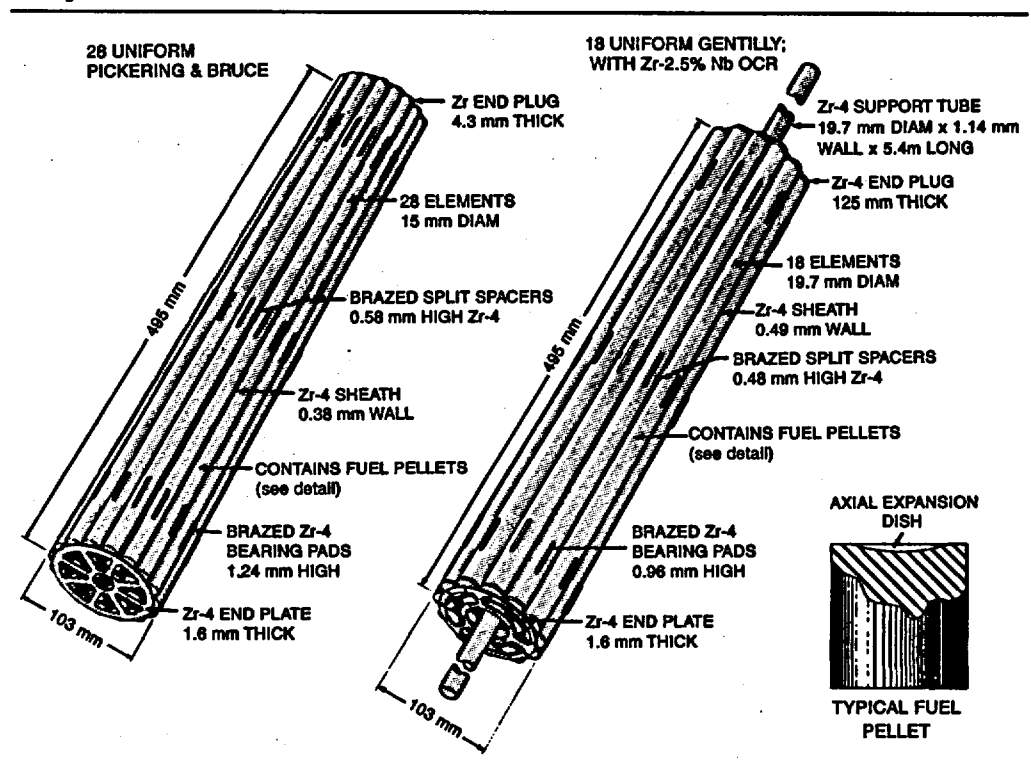
end fitting design was of course different, to accommodate the different design of fuel assembly and fueling from one end only. The single-ended fueling allowed the channel to be designed for easy replacement. This was a very attractive feature because it was expected that the fuel channels in all CANDU reactors would have to be replaced at least once during each reactor's projected life.

To define the steam quality operating limits, AECL funded the construction of out-reactor test loops, both in-house and at Westinghouse Canada, to determine under what fuel power and coolant flow conditions the fuel cladding would go into dryout. Dryout is a condition where the water film on a heated surface breaks down, causing a precipitous decrease in the local heat transfer coefficient and a resultant sharp increase in the surface temperature. It is a condition to be avoided in reactors, as the high temperature leads to an accelerated deterioration of the fuel cladding. In the out-reactor tests, the fuel pins were simulated by electrically heated tubes, and thermocouples were buried in the tube walls at strategic locations, to detect when the surfaces went into dryout. The results allowed the scientists to develop a generalized correlation that could be used to predict the conditions that would lead to dryout in a power reactor channel. The correlation was then confirmed in thermocoupled fuel experiments in the in-reactor loops. The predictions were expressed as a critical power ratio (CPR), which is the ratio of the fuel power at which dryout will occur to the design operating power in the highest powered channel. It was found that the CPR was 1.5 for operation at 20 percent outlet quality but only 1.2 for operation at 40 percent. Thus, at 40 percent with typical in-core power variations, periodic excursions into dryout on some fuel sheaths could be expected. Although experiments indicated that a fuel could be developed to

withstand these periodic excursions, it was considered prudent to avoid them, so the design outlet quality for CANDU-BLW power reactors was set at 20 percent.

The control problem in the CANDU-BLW resulted from its relatively high positive power coefficient. Even at constant reactor power, it was impossible to always keep a constant steam quality in each channel; when the local quality increased, the local neutron flux increased and the local fuel power increased, driving the local steam quality even higher. These local reactivity excursions would not necessarily lead to a loss of overall reactor control, as the changes in spatial flux shape would increase overall neutron leakage. Nevertheless, the resulting spatial power oscillations were expected to be intolerable. Thus it was deemed essential to have a spatial flux control system to

FIGURE 16.3
Comparison of Fuels for CANDU-PHW, CANDU-BLW and CANDU-OCR Reactors



limit them to acceptable levels. To design such a system, it was necessary to be able to calculate the magnitude of the flux perturbations to be expected, their location in the reactor and the rate of reactivity change. To do such calculations, the scientists at Chalk River developed a computer code that they were able to check against experiments in the ZED-2 reactor. Calculations showed that the reactivity excursions could be controlled by absorber rods, strategically located in the core, which would automatically move to counteract the flux perturbations when they occurred.

As in the CANDU-PHW, the coolant in the CANDU-BLW had to be kept alkaline, to prevent excessive corrosion of both the fuel cladding and the out-reactor circuit, and excessive deposition of corrosion products on the fuel. In the CANDU-PHW this

is done by adding lithium hydroxide, but lithium hydroxide could not be used in the CANDU-BLW because it would concentrate in the water phase, making the water more alkaline and the steam and steam condensate less alkaline than desired. Thus in the CANDU-BLW the coolant was kept alkaline by the addition of ammonia. Ammonia is volatile and would circulate with the steam phase, but it would also decompose with time, so a continual makeup was necessary. A series of tests in the in-reactor loops showed that if the concentration of ammonia in the exit water was maintained at seven parts per million (ppm), both fuel-cladding corrosion and deposition of out-reactor corrosion-product iron oxide on the fuel cladding remained at acceptable levels. The tests further showed that the growth of the iron oxide film on the cladding was reversible: if the

film grew thicker because of a temporary loss of control of the ammonia concentration, it would return to its former equilibrium thickness when the ammonia concentration was returned to its specified level. This gave added confidence that corrosion-product deposition on the fuel could be adequately controlled, but led to concerns that radioactive corrosion products would be transported to, and deposited in, the out-reactor circuit.

By 1965, Hydro-Quebec had become interested in gaining experience in nuclear power. Although Hydro-Quebec still had potential water power sources available, these sources were getting further and further from the load centres. In 1966 the federal and provincial governments reached an agreement whereby AECL would build a prototype 250 MWe natural-uranium-fueled CANDU-BLW power station to be known as Gentilly, on the south bank of the St. Lawrence River, near Trois Rivières. The arrangements were similar to those made with Ontario Hydro for Douglas Point, in that AECL would pay for the station while Hydro-Quebec would operate it and

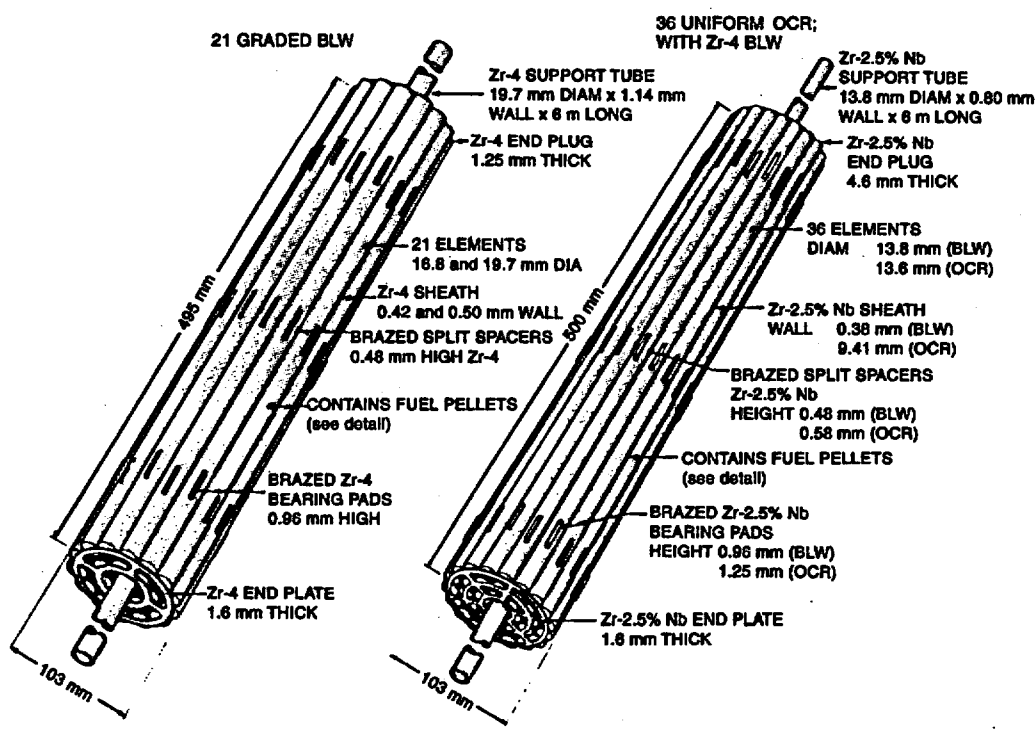
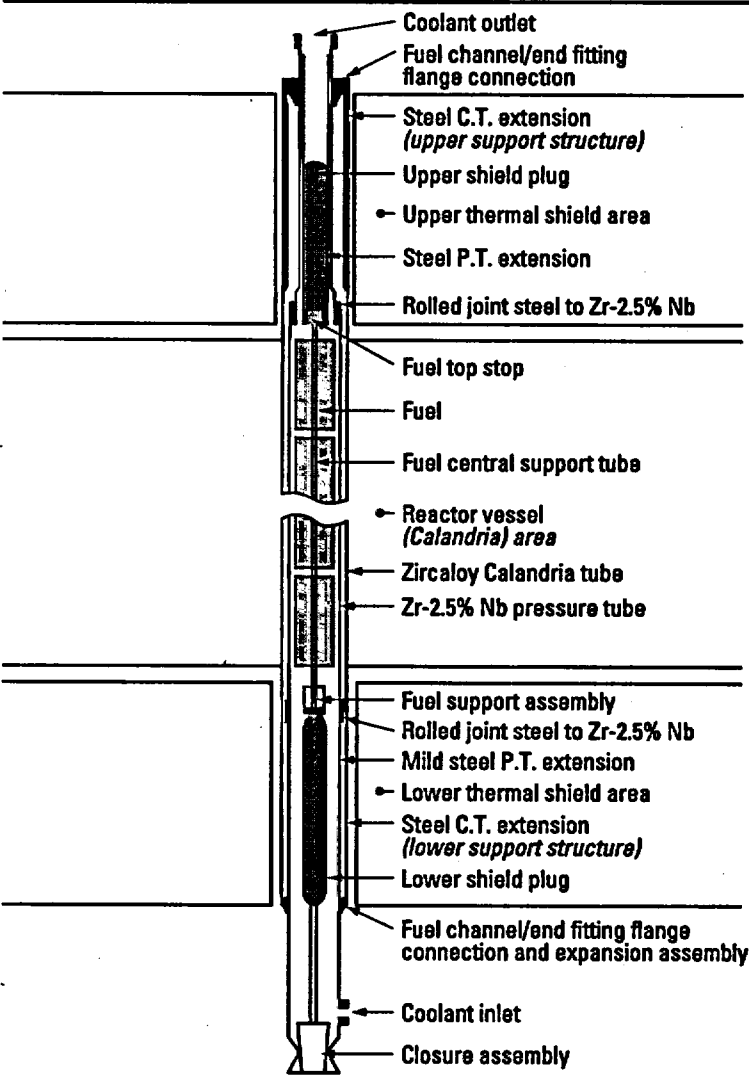


FIGURE 16.4
Typical Fuel Channel for the CANDU-BLW



pay for the power at an agreed rate consistent with the formula used in the Ontario Hydro/AECL agreement for Douglas Point. A number of engineers transferred from Chalk River to the AECL design offices in Toronto, to form the nucleus of a team to do the detailed design; by 1970, design and construction had

been completed. First power was produced in April 1971 and full power was reached in May 1972, but the station did not perform as well as expected. The absorber rods were located too close to the centre of the reactor, resulting in insufficient control in the outer regions, with consequent excessive flux tilts, particularly during booster rod insertions. In November 1972 there was insufficient heavy water available in Canada to start up Pickering 3, so it was decided that the heavy water from Gentilly-1 would be loaned to Pickering 3 and, while it was away, the absorber rods would be moved to correct the control problem. The absorber rods were moved but the heavy water was not returned until late 1974. When the reactor restarted it was plagued with corrosion problems in the condensers, apparently related to the start up, in the interim, of new industrial plants on the St. Lawrence river upstream of the cooling water intake. Further, even after the repositioning of the absorber rods, the reactor remained difficult to control. By 1977 it was clear that concern about control would preclude the commitment of any larger natural-uranium-fueled CANDU-BLWs. Also, the performance of Ontario Hydro's CANDU-PHW stations had been excellent, serious heavy-water leakage problems had not materialized, and Hydro-Quebec had committed to the construction of a 600 MWe CANDU-PHW power unit, Gentilly-2. The question now arose as to what to do with Gentilly-1. A decision was made to modify it to produce steam for the LaPrade heavy-water production facility then under construction at the Gentilly site. Design work on these modifications proceeded through 1977 and some modifications were started, but the work was terminated in early 1978 when Hydro-Quebec agreed to supply the steam from Gentilly-2 at a price that could not be matched by using Gentilly-1. Gentilly-1 continued to generate electricity through 1978, but its operation was sporadic and the information it was generating was not worth the cost of operation. Thus in April 1979 it was shut down and by 1984 it had been decommissioned.

In the laboratories it was always recognized that the CANDU-BLW would be a better concept with uranium-235- or plutonium-enriched fuel, rather than natural fuel. With

enrichment there wasn't such a high premium on neutron economy, so the fuel pins could be thinner, the channel power higher and the lattice pitch tighter, resulting in a considerably lower inventory of heavy water. Further, with enrichment, the fuel could be taken to about 580 MWh/kgU, vs 180 MWh/kgU for natural uranium fuel, mitigating any fueling cost penalty associated with enriched fuel fabrication. But most importantly, with enrichment, the positive power coefficient would be substantially reduced, making it much easier for the control rods to correct any reactivity excursions resulting from boiling imbalances. Accordingly, even after problems materialized with Gentilly-1, work continued on the CANDU-BLW concept, with emphasis on its use as a plutonium burner (CANDU-BLW-PB). The development program had two basic components: 1) to demonstrate the improved controllability, and 2) to demonstrate the fabricability and irradiation performance of plutonium-bearing fuels. Similar designs were being developed in the United Kingdom and Japan. To demonstrate improved controllability, AECL built a hybrid computer system that would simulate an operating CANDU-BLW-PB. This system had two state-of-the-art analogue computers that could be programmed to represent the reactor system behaviour in real time. This behaviour could be controlled by a digital computer in exactly the same way that the computer would control the reactor. By 1975, this system had confirmed that a large plutonium-burning CANDU-BLW power reactor could be controlled without difficulty. To demonstrate the fabricability and irradiation performance of the plutonium-bearing fuel, AECL planned to fabricate plutonium-bearing CANDU-BLW fuel in a recycle fuel fabrication facility (see chapter eighteen) and irradiate it to the target burnup in a boiling-light-water loop in the NRU reactor. The fuel was to be essentially identical to the Bruce 37-pin bundle (see chapter thirteen), with the central pin replaced by a central support tube. For various reasons, completion of the recycle fuel fabrication facility was seriously delayed, and by the time it was completed the time scale for recycle of plutonium in Canada had receded to the distant future. So work on the CANDU-BLW was stopped. Thirty-six-pin plutonium-burning fuel assemblies were later successfully irradiated to the target

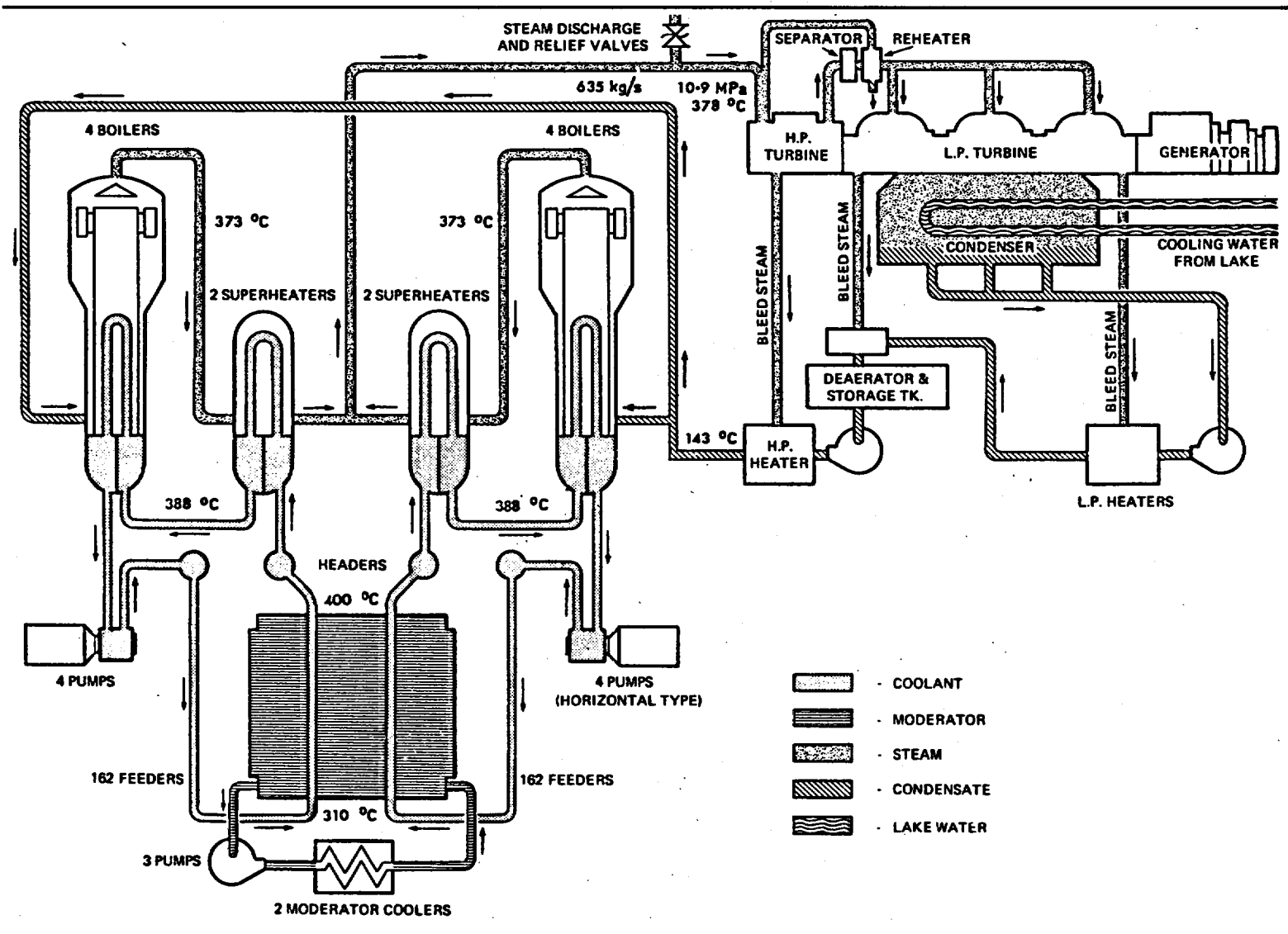
burnup in pressurized water coolant, but not in boiling light water. The United Kingdom's enriched-uranium-fueled equivalent of the CANDU-BLW, the Steam-Generating Heavy-Water Reactor, operated successfully until 1990, and the Japanese plutonium-burning equivalent of the CANDU-BLW, FUGEN, is still operating.

CANDU-OCR

In the mid-fifties the United States Atomic Energy Commission (USAEC) was funding a program on enriched-uranium-fueled, organic-cooled-and-moderated reactors managed by Atomics International, a division of General Atomics in Los Angeles. Engineers at Canadian General Electric who were following the progress of this program concluded that a reactor cooled with organic liquid and moderated with heavy water could be used with natural uranium, and would have some attractive features for power production. They suggested the concept to AECL and, in 1959, AECL agreed to help fund it; the CANDU-OCR program was born.

The conceptual natural-uranium-fueled CANDU-OCR (see figures 16.5 and 16.6) was similar to the CANDU-BLW, in that the pressure tubes (see figure 16.7) were oriented vertically in the calandria and the fuel bundles were strung on a central support tube (see figure 16.8). Like the BLW, fueling was from one end only and the fuel strings had to be periodically removed for bundle shuffling, but in the OCR fueling was from the top rather than the bottom. The steam generating system was, however, similar to the CANDU-PHW in that the steam to drive the turbine was generated on the secondary side of a heat exchanger. The potential attractions of the CANDU-OCR relative to the CANDU-PHW, were its lower capital cost, resulting from a lower heavy-water inventory, a lower coolant pressure and a higher coolant temperature (higher conversion efficiency), and its lower operating costs resulting from the elimination of heavy-water leakage from the primary circuit and the minimal transport of radioactivity by the coolant (ease of maintenance). As in the CANDU-BLW, these potential advantages were partially offset by higher fueling costs, but it

FIGURE 16.5
Schematic Diagram of a CANDU-OCR Power Station



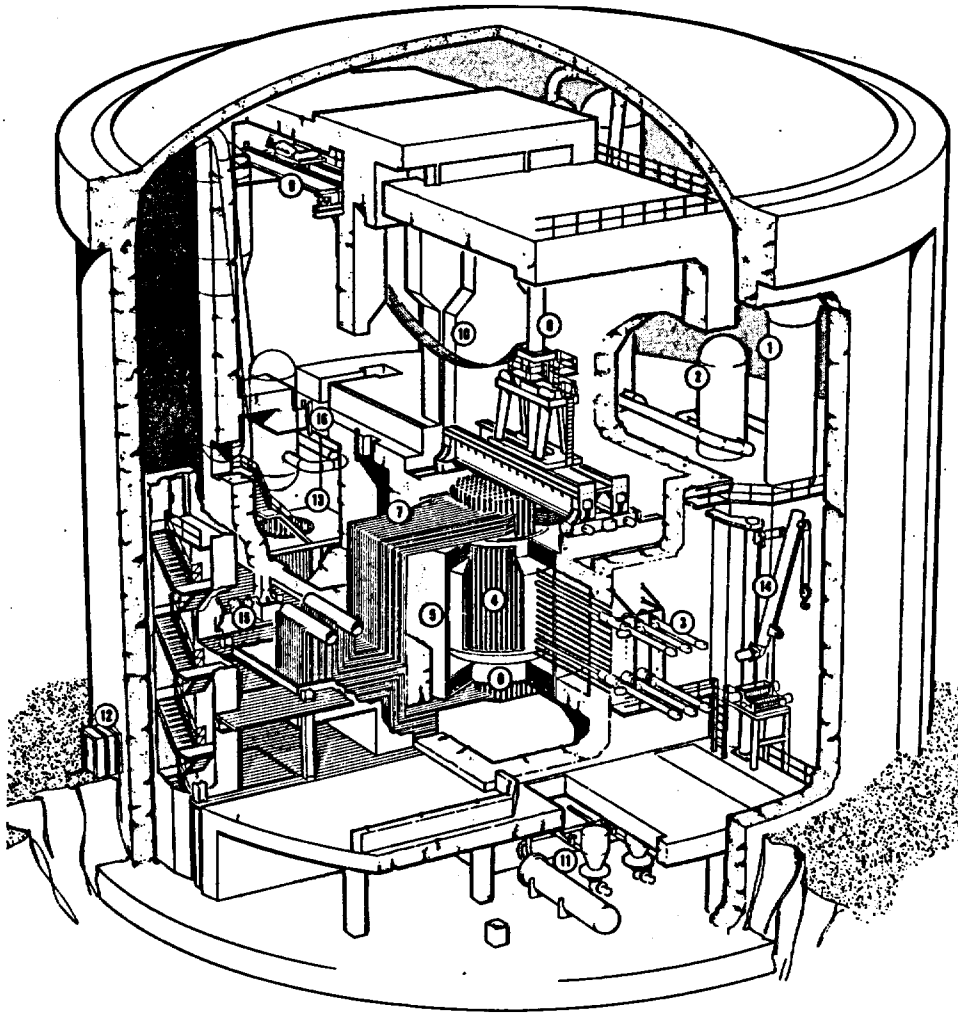
was felt that in the OCR this fueling cost disadvantage might disappear if uranium carbide or uranium metal, with their higher uranium densities and their better thermal conductivities, could be used as fuel. This seemed possible in an OCR,

because the corrosion problems that prohibited the use of uranium carbide or uranium metal with water coolants did not exist with organic coolants. The main potential difficulties of the OCR were that the coolants gradually decomposed and

polymerized under heat and radiation, that fuel surfaces could foul if the coolant chemistry was not right, that the fuel cladding and pressure tubes had to withstand higher coolant temperatures, and that the coolant was flammable. Development work was required to identify the most stable practical coolants, to determine the coolant treatment necessary to avoid fouling, to determine the heat transfer properties of the selected coolants, to develop suitable fuels and pressure tubes, and to determine the best methods of preventing and handling fires. The development work was focused on Whiteshell Reactor-1 (WR-1), a 40 MWt, heavy-water moderated, organic-cooled research reactor designed and built by CGE under contract to AECL.

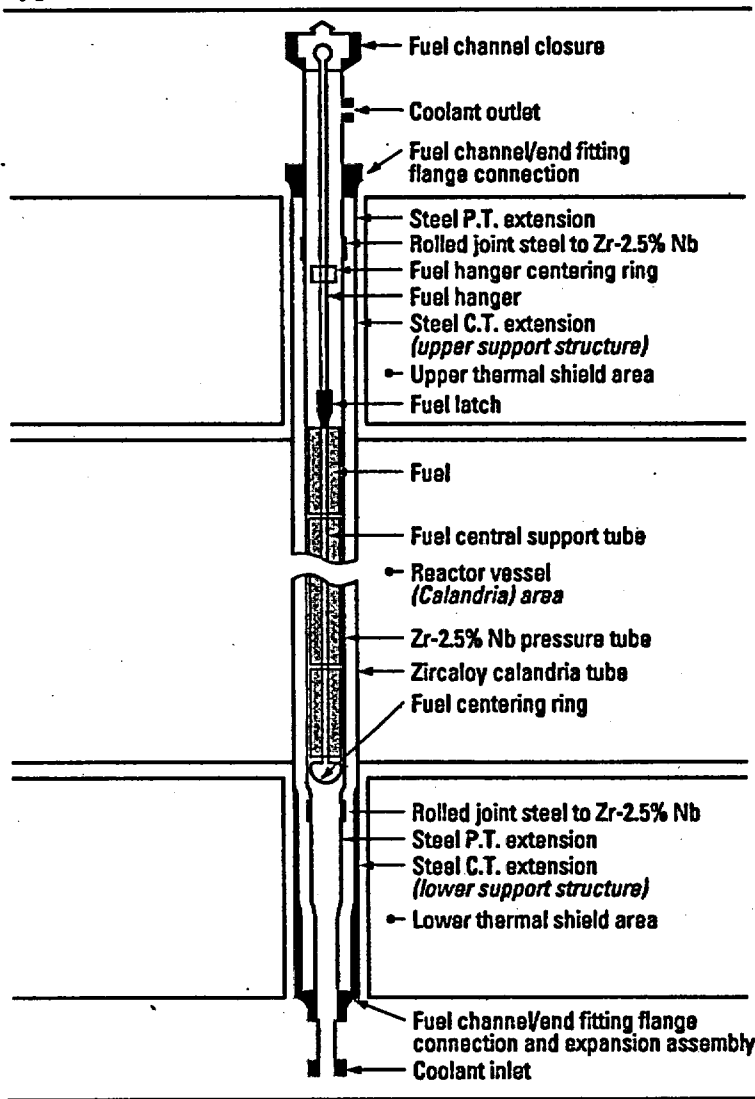
As a result of screening tests done in the United States, the initial reference coolant for WR-1 was a commercially available mixture of terphenyls known as Santowax-OM. It was produced by the Monsanto Chemical Co. as a by-product of the manufacture of biphenyl from benzene. The screening tests showed that, of organic compounds available at low cost, the terphenyl family offered the best radiation and thermal stability. Santowax-OM was chosen over other available terphenyl mixtures because it had the lowest melting range. However, even Santowax-OM was highly viscous at room temperature, and with it some trace heating of reactor circuits was deemed necessary. Thus one of the earliest tasks of AECL scientists assigned to the OCR program was to find a coolant that would not require trace heating. After some study they concluded that HB-40, a partially hydrogenated mixture of terphenyls marketed by Monsanto as a plasticizer and heat transfer

FIGURE 16.6
Cutaway of a CANDU-OCR Reactor Building



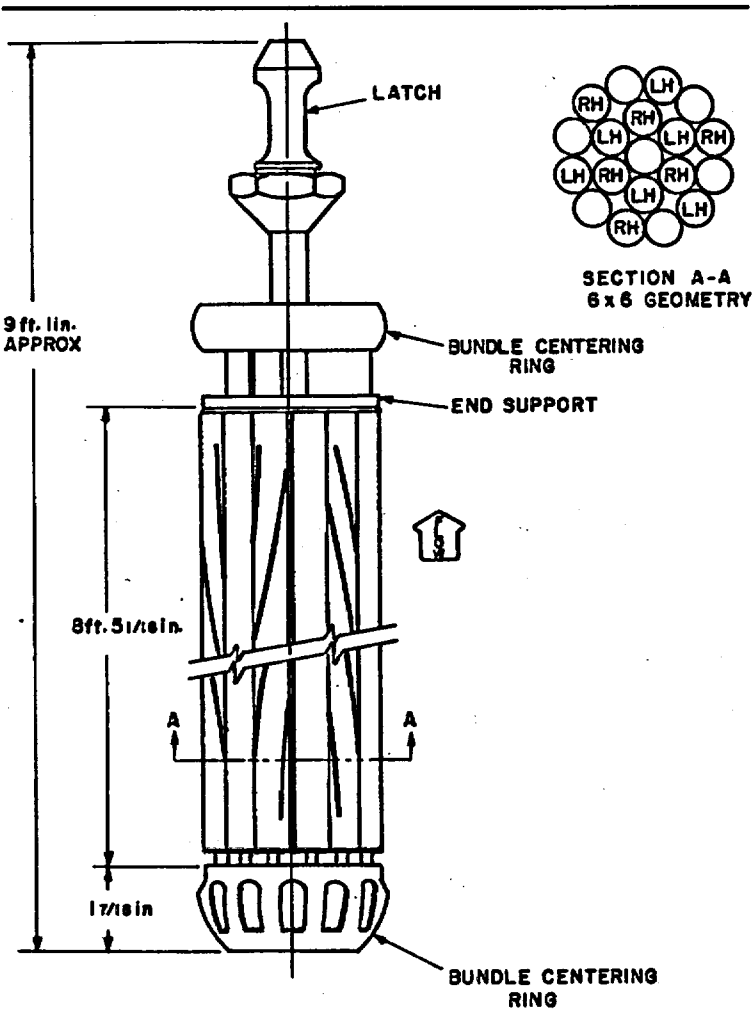
- | | |
|-----------------------|---------------------------|
| 1. Boilers (8) | 9. F/M Service Crane |
| 2. Superheaters (4) | 10. F/M Vault Door |
| 3. Booster Rods | 11. Moderator System |
| 4. Calandria Assembly | 12. Emergency Airlock |
| 5. Shield Tank | 13. Fuel Transfer Bay |
| 6. End Shield | 14. Booster Flask Crane |
| 7. Feeders | 15. Primary Pumps (4) |
| 8. Fueling Machine | 16. Fueling Machine Ports |

FIGURE 16.7
Typical Fuel Channel for the CANDU-OCR



agent, showed enough promise to be used at least as the startup coolant for WR-1. Initially it was thought that, because of the higher decomposition rates of HB-40, the outlet temperature of the reactor would have to be limited to 350°C. However, under heat and radiation the composition of the coolant changed and

FIGURE 16.8
Early Design of a WR-1 Fuel Bundle Assembly



the decomposition rate decreased to a point where, at equilibrium, operation at temperatures up to 400°C became routine. Thus HB-40 became the reference coolant for WR-1 and CANDU-OCR power reactors, and the need for trace heating was eliminated.

It was recognized early that control of fouling – the formation of low thermal conductivity deposits on fuel heat transfer

surfaces – would be one of the key feasibility questions to be answered if the CANDU-OCR was to become a viable concept. The ultimate result of severe fouling would be an increase in fuel cladding temperature and fuel failure. Research work identified two main types of fouling: mass transfer fouling which led to inorganic deposits, and particulate fouling which led to carbonaceous deposits. The mass transfer fouling was believed to be caused by impurities in the coolant that dissolved iron from the out-reactor circuit and then precipitated it as iron carbide or oxide on the hot fuel cladding. The mechanism for the particulate type of fouling was never fully explained, but it produced deposits that were mainly organic in composition. The rates of both types of fouling increased with increasing fuel cladding temperature, but that of mass transfer fouling increased with increasing coolant velocity, while that of particulate fouling decreased. If either type of fouling increased to a point where flow stagnation occurred, a third phenomenon took place, called coking. Coking resulted when high molecular weight compounds, produced by coolant polymerization, exceeded their solubility limit in the stagnation areas and precipitated as a hard carbonaceous solid. It was coking in the stagnant areas of the moderator that eventually led to the demise of the organic-cooled and -moderated concepts in the United States, but this would be a problem in the heavy-water moderated CANDU-OCR only if fouling could not be controlled on the fuel surfaces. Over time, a set of coolant specifications was developed that led to acceptable fouling performance. These specifications were: maintain the chlorine content below $50\mu\text{g/kg}$, maintain the combined oxygen below $200\mu\text{g/g}$, maintain the water content at $200\text{--}600\mu\text{g/g}$, maintain the iron content below $500\mu\text{g/kg}$, maintain the particulate content below $1,000\mu\text{g/kg}$ and maintain the coolant velocity above 5 m/s . To meet these specifications, all feed coolant had to be processed to remove dissolved oxygen, chlorine and particulate matter, a by-pass stream of coolant had to be continuously passed through attapulugus clay and glass spool filters for purification, chloride compounds had to be rigorously excluded from the reactor area, water had to be continuously added as it was removed in the degassing circuit, and coolant degassing and

recycling systems had to be fabricated from stainless steel, to minimize the return of iron to the core. Because the mechanisms of fouling were so complex and the consequences of fouling so severe, it was deemed necessary to develop an on-line measurement technique to warn when the fouling potential of the coolant was increasing. The technique developed was the small-probe fouling test (SPFT). In this test a small flow from the reactor coolant system was passed over an electrically heated stainless-steel probe for a period of twenty-four hours. At the end of the period the probe was washed and the film was scraped from a fixed length of the hottest region. The weight loss on scraping was a measure of the fouling potential of the coolant. It was found that if the fouling potential was less than $5\text{ mg/m}^2\text{h}$ the films on the fuel surfaces would remain thin and would have no detrimental effects on operation. During the entire operation of WR-1 there was only one serious fouling incident. This occurred in 1968 when an experiment to determine the minimum amount of coolant chemistry control required coincided with the first attempt to raise the coolant temperature from 345°C to 400°C . Although the coolant chemistry was apparently adequate for successful operation at 345°C it was not adequate for operation at 400°C . By the time this was recognized, about 40 percent of the fuel from the coolant circuit involved³ was rendered unusable, because of heavy fouling. When the circuit was subsequently taken to 400°C with better coolant chemistry control there was no problem. Eventually, one of the circuits was operated at a coolant temperature of 425°C , again with no fouling problems.

The heat transfer properties that are significant in discussing coolants for nuclear reactors are: the interface heat transfer coefficient, which determines the temperature difference between the bulk coolant and the fuel cladding; the heat capacity, which determines the flow of coolant required and the temperature rise across the core; the onset of nucleate boiling (ONB), which describes the conditions at which the coolant begins to boil; and the departure from nucleate boiling (DNB), which describes the

³ The WR-1 reactor had three separate coolant circuits, and different experiments could be done in the different circuits simultaneously.

conditions at which the fuel surface temperature rises sharply, leading to rapid fuel failure. The heat transfer properties of organic coolants are very different from water and the significance of these differences are discussed below. Firstly, the heat transfer coefficients of organic coolants are much poorer than those for water, thus for equal surface heat fluxes⁴ the difference between the fuel-cladding temperature and the bulk coolant temperature is much greater. In an organically cooled power reactor, where the outlet temperature will approach 400°C, the peak fuel cladding temperature will approach 500°C, as opposed to perhaps 315°C in a water-cooled power reactor with a coolant outlet temperature of 300°C. Secondly, the heat capacity of organic liquids is considerably less than that for water. Thus, to remove the same amount of heat, more organic coolant has to be moved through the core, or the temperature rise across the core has to be higher. Since there is a premium on neutron economy, this means, in practice, that in organically cooled systems both the coolant linear velocity and the temperature rise across the core are higher. Thirdly, in organically cooled systems, nucleate boiling results from the volatilization at the heated surface of low boiling fractions of the multicomponent coolant, while in water-cooled systems nucleate boiling results from the formation of steam bubbles at the heated surface. In both cases the bubble formation at the surface disrupts the surface film and improves heat transfer, but in the organic system the effect on sheath temperature is more pronounced, because of the poorer heat transfer coefficient. In the organic system the phenomenon is called simmering, and it can reduce sheath temperatures by 25-30°C, compared with a non-simmering system. Fourthly, in water-cooled systems departure from nucleate boiling is well defined, because the single-component coolant all boils at the same temperature. However, in organically cooled systems, each of the many components of the coolant mixture has its own boiling point and, after the low boiling components are boiled away, high boiling components remain to cool the fuel. Thus in organic coolants there is no precipitous increase in sheath temperatures at any given surface heat flux, but rather a continuously increasing temperature as the heat flux

is increased and the more volatile components of the coolant are removed. This makes the organically cooled systems more tolerant to accidental overpower situations, and has a significant impact on safety analysis.

When the natural uranium CANDU-OCR power reactor was first conceived, the reference fuel was uranium carbide clad in sintered aluminum product (SAP), and the reference pressure tubes were made from SAP. SAP is a mixture of 15 percent aluminum oxide in aluminum metal, the aluminum oxide being added to give high-temperature strength. Zirconium alloys were not chosen as fuel cladding or pressure tube material, because it was felt that at the high temperature demanded, about 500°C for fuel cladding and 400-425°C for pressure tubes, excessive hydriding would make their use impractical. It was decided that neither uranium carbide fuel nor SAP pressure tubes could be developed in time for WR-1, so its fuel was to be enriched uranium dioxide clad in SAP and its pressure tubes were to be stainless steel. The plan was to change to uranium carbide fuel and SAP pressure tubes when they were developed. However, in irradiation testing, no matter what variables were tried, all SAP-clad UO_2 fuel failed, because of the low ductility of the SAP and, since time for fabricating the first fuel charge was growing short, the designers felt they would have to revert to stainless steel for cladding as well as pressure tubing. While the SAP-clad fuel was being tested, an AECL scientist was investigating the hydriding of zirconium alloys in organic coolant, and concluded that alloys of zirconium and niobium could be used as fuel cladding, provided that they were properly handled and the coolant chemistry was properly controlled. He prevailed upon the designers to try zirconium-2.5 percent niobium-clad UO_2 of the Douglas Point wire-wrap design (see chapter thirteen). From the first test these irradiations were successful. It turned out that, not only was the hydriding rate low enough in its own right to allow Zr-2.5Nb to be used as cladding, but, as an added bonus, practically all the hydrogen that was picked up

⁴ Surface heat flux is directly related to fuel power; the higher the power the higher the surface heat flux.

migrated to the cooler wire wrap, where it did no structural damage. After only one test, a decision had to be made regarding the cladding for the first charge of WR-1, and Zr-2.5Nb was chosen. In WR-1 it proved highly successful and it soon became the reference cladding for the proposed CANDU-OCR power reactors. Further, this success refocused scientists' attention on the use of zirconium alloys rather than SAP as a material for pressure tubes, with the result that by 1971 all WR-1 stainless-steel pressure tubes had been replaced by tubes made from either Zr-2.5Nb or Ozhennite 0.5,⁵ a zirconium alloy developed in the USSR that had a lower out-reactor hydriding rate than Zr-2.5Nb. It was later found that the in-reactor hydriding rates were virtually identical.

With the successful development of Zr-2.5Nb-clad UO_2 as fuel for WR-1, fuel effort now turned to the development of uranium carbide, which, it was expected, would give lower fueling costs in a natural-uranium-fueled CANDU-OCR power reactor. After an extended development program, a satisfactory fabrication technology, involving arc melting and casting, was developed at Eldorado Nuclear Limited, and irradiation testing started. It was determined that slightly hyperstoichiometric material (4.8-5.0 wt. percent C) performed best and could be irradiated to at least 400 MWh/kgU at linear heat ratings up to 100 kW/m without any problem. By 1971, the technology had reached the stage where Zr-2.5Nb-clad UC fuel could be chosen with confidence in CANDU-OCR power reactor studies, and by 1973 all necessary steps had been taken to allow UC fuel to replace UO_2 as the fuel for WR-1. This change took place in 1974, and from then until the reactor was shut down in 1985 (because it was no longer required in AECL's experimental program), the UC fuel performed very well.

One of the main differences between organic and water coolants is that the organic coolants are flammable. This feature threatened to eliminate the organic coolant program just as it was getting started. The initial organically cooled test facilities at Chalk River were plagued with spontaneous fires in the pipe lagging and it was felt that if this incidence of fires was the norm, the concept was not viable. However, it was discovered that the problem was caused by coolant leakage

into the lagging, which provided enough insulation to keep the coolant hot and enough air ingress to allow it to ignite. The solution was good maintenance to eliminate leaks, good house-keeping to remove coolant-soaked lagging, and air-resistant paints on the lagging to restrict the access of air. With these precautions, the problem was almost eliminated from the test facilities and subsequently from WR-1. WR-1 was equipped with non-incendive⁶ equipment and wiring, smoke detectors, temperature sensors and automatic sprinkler systems in potentially hazardous areas, and the occasional small fires that occurred were easily controlled. Even though the WR-1 operating experience with respect to fire was excellent, it was recognized that it would not be sufficient to obtain an operating licence for a CANDU-OCR power reactor. To obtain such a licence it would have to be proven that the consequences of even the most severe combustion event that could possibly happen would be acceptable. The two events of most concern were coolant autoignition and coolant vapour explosion. Work showed that the possibility of autoignition could be eliminated simply by keeping the ambient air temperature below 350°C, but that vapour explosion could occur if the coolant was dispersed in air to a concentration between 0.5 and 5.0 volume percent and an external source of ignition was present. Thus it was concluded that the power station would have to have a high-speed fire-suppression system if it was to be licensed.

By 1969, W.B. Lewis, who had started out only lukewarm to the CANDU-OCR concept, had become its strongest supporter. He was attracted by the good performance of WR-1, the demonstrated high coolant temperature and high power density capability, and the low radioactivity of the out-reactor circuits.

⁵ Ozhennite 0.5 has a nominal composition by weight: zirconium-99.5 percent, tin-0.2 percent, iron-0.1 percent, nickel-0.1 percent, niobium-0.1 percent.

⁶ Non-incendive equipment is designed to minimize the possibility of fires, but it is not as complex or expensive as the more familiar explosion-proof equipment. It was felt that explosion-proof equipment was not required in WR-1.

He felt that the CANDU-OCR was the ideal reactor for his high-burnup thorium/uranium-233 cycle (see chapter eighteen), and he developed a concept that would produce 1,500 MWe from a core the size of the 250 MWe Gentilly-1.

By the end of 1970, it was considered that all the feasibility aspects of a natural-uranium-fueled CANDU-OCR had been demonstrated and that a decision was required as to its future. Accordingly, in March 1971 the AECL board of directors authorized a detailed design and cost study of a 500 MWe CANDU-OCR generating station, to be done by the design team at AECL Power Projects, assisted by staff from the research laboratories. The AECL board of directors specified that the fuel was to be natural uranium, feeling that a design based on the thorium fuel cycle was premature. In the design, the fuel was Zr-2.5Nb-clad uranium carbide, the pressure tubes were Zr-2.5Nb and the coolant was HB-40. The coolant outlet temperature was 400°C, which led to a cost-optimized steam-cycle efficiency of 34 percent. The station had some attractive safety features relative to water-cooled reactors, including low radiation fields, low stored energy, low coolant pressure, no fuel ballooning and no dryout, but it had the unattractive feature of

a flammable coolant. The study concluded that the CANDU-OCR would have a 10 percent advantage over the CANDU-PHW in both capital and unit energy costs, and the AECL board of directors had to decide whether that potential advantage was sufficient to continue the program. In January 1973, the board of directors decided not to continue the program, the main reasons being that the CANDU-PHW was working well, there was little utility interest in pursuing another concept, and there was little prospect for raising the \$250 million required for a CANDU-OCR prototype. The CANDU-OCR technology was to continue to be demonstrated through the operation of the WR-1 reactor, but the development effort was to be switched to improving the CANDU-PHW.

It is perhaps unfortunate that the CANDU-OCR technology was not developed at the time that opportunities for prototype stations were available. In the mid-eighties an Advanced CANDU study team, headed by the AECL design organization, set out the properties of the ideal coolant for a CANDU reactor, and found that the organic coolant used in WR-1 approached the ideal much more closely than any other coolant they could find.

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Chapter Seventeen

Heavy Water

H. K. RAE

THE NEED FOR R&D

Heavy water has been produced in nine countries using seven different processes in twenty-two plants. But heavy water has always been expensive, as the following examples illustrate:

LOCATION OF HEAVY WATER PLANT	YEAR	\$ OF THE YEAR/KG	1985\$/KG
Rjukan, Norway	1935	500	4,300
Trail, British Columbia, Canada	1945	130 ¹	900
Savannah River, South Carolina, U.S.A.	1955	62	275
Port Hawkesbury, Nova Scotia, Canada	1970	48	150
Bruce, Ontario, Canada	1985	300	300 ²

The range in constant 1985 dollars is \$4,300/kg to \$150/kg. Contrast this range with the price of some Scotch whiskys – \$35/kg in 1985 dollars, or of silver – \$160/kg, and one can appreciate that this is a very costly form of water. Even a small heavy-water research reactor requires about 10 Mg of heavy water, while a CANDU 6 power reactor requires 450 Mg, worth about \$135 million in 1985.

Thus, there has been, and continues to be, a large incentive to develop ever-cheaper processes for heavy-water production. This goal has been sought by many researchers in many countries over the last five decades, since the first heavy-water reactor operated in Chicago in 1944. This chapter will describe AECL's part in this worldwide search.

Deuterium occurs in nature with hydrogen at a concentration varying from about 80 to 160 parts per million atom ratio. All the world's heavy water has been produced from water or industrial hydrogen, although natural gas, petroleum and industrial ammonia are other large-scale potential sources. The high cost of heavy water stems from this low natural abundance of deuterium, requiring separating equipment with both a very large flow capacity and a high degree of enriching capability.

Having a compelling need for a reliable and economical supply of heavy water for its power reactor program, Canada produced more than 80 percent of the world's heavy water between 1970 and 1985. These heavy water plants were based on earlier American technology, and benefited from their operating experience.

¹ \$60/kg operating cost plus 15 percent of the investment per annual kg of heavy water.

² The higher cost of heavy water in constant dollars at Bruce, relative to Port Hawkesbury, reflects the fact that construction costs, and especially energy costs, have risen from 1970 faster than the Canadian Consumer Price Index.

However, unlike reactor development, there has been no large technical effort elsewhere during the past two decades that could provide a basis for collaboration³ in further development of heavy-water production technology. So Canada has been largely alone in this field.

Even more important than the search for a better heavy-water process was the major development program in the seventies, led by AECL, to assist the staff of Canada's four large heavy-water plants to achieve reliable and successful operation. This was a multidisciplinary team effort over ten years involving, at its peak in 1974, well over 100 scientists, engineers and technologists from AECL, Ontario Hydro, Canadian General Electric (CGE), and twenty other industrial organizations, government laboratories and university departments. It is a good example of the ability of a large mission-oriented research organization to develop the detailed scientific and technical understanding necessary to solve unexpected and complex operational problems.

HEAVY-WATER PRODUCTION IN CANADA

By the end of 1943, a small heavy-water plant (6 Mg/a) had begun production at Trail, British Columbia. It had been built by the Consolidated Mining and Smelting Company of Canada Limited (Cominco) to an American design under the direction of, and financed by, the United States War Department's Manhattan Project. It extracted deuterium from electrolytic hydrogen by a catalyzed vapour-phase exchange reaction between steam and hydrogen. By October 1945, Trail had produced 8 Mg of heavy water. The United States had built three other heavy-water plants, using the water distillation process, which had also begun production at the end of 1943; they provided a total of 21 Mg of heavy water by October 1945, when they were shut down because of their high operating costs. Of this total stock of 29 Mg, the United States transferred 20 Mg to Canada for use in ZEEP and NRX.

The Trail plant continued to be operated by Cominco for the United States Atomic Energy Commission (USAEC) until 1956, when Cominco's contract expired. They offered to produce

heavy water for AECL at \$76/kg, but the offer was not accepted and the Trail plant was closed. AECL had assurance of much larger quantities of heavy water from the USAEC at \$62/kg. The small electrolytic cells used at Trail for final enrichment of heavy water (2-99.8 percent) were the basis for the design of a unit for reconcentrating (upgrading) recovered heavy water at Chalk River (see chapters two and fourteen).

In 1952, E.I. DuPont de Nemours and Co. Inc. began operating two large heavy-water plants at Dana, IN (410 Mg/a) and Savannah River, SC (450 Mg/a). The Girdler-Sulphide (GS) process (see Box A) was used. In 1955, an exchange agreement on atomic energy was signed by Canada and the United States, extending the collaboration that had begun in 1944. This agreement provided Canada with a heavy-water supply, at least for the initial phases of the CANDU program, and access to all the American heavy-water production technology.

This agreement turned the issue of securing a long-term, large-scale supply of heavy water at a reasonable cost for the CANDU program from process development to technology transfer. As a result, the small effort on heavy-water process development underway at Chalk River in the early fifties was terminated.

Once the large reactors at Savannah River were supplied with heavy water, the Dana plant and two-thirds of the Savannah River plant (SRP) were shut down over the 1957-59 period. This left Savannah River with an output of 180 Mg/a and it continued to operate until 1981.

Between 1955 and 1967, AECL purchased about 1,000 Mg of heavy water from the USAEC for ZEEP, NRU, ZED-2, WR-1, NPD, Douglas Point and the first Pickering unit.

A Canadian heavy-water production capability would be needed to sustain a major CANDU program, and government policy in the sixties required that this activity should be in the private sector. In 1963, AECL sought bids for the supply of 900 Mg over a five-year period at a price below \$49/kg. The successful bidder would have a royalty-free license to the basic GS

³ Collaboration with India, the only other significant heavy water producer in the eighties, was ruled out under Canada's nuclear non-proliferation policies.

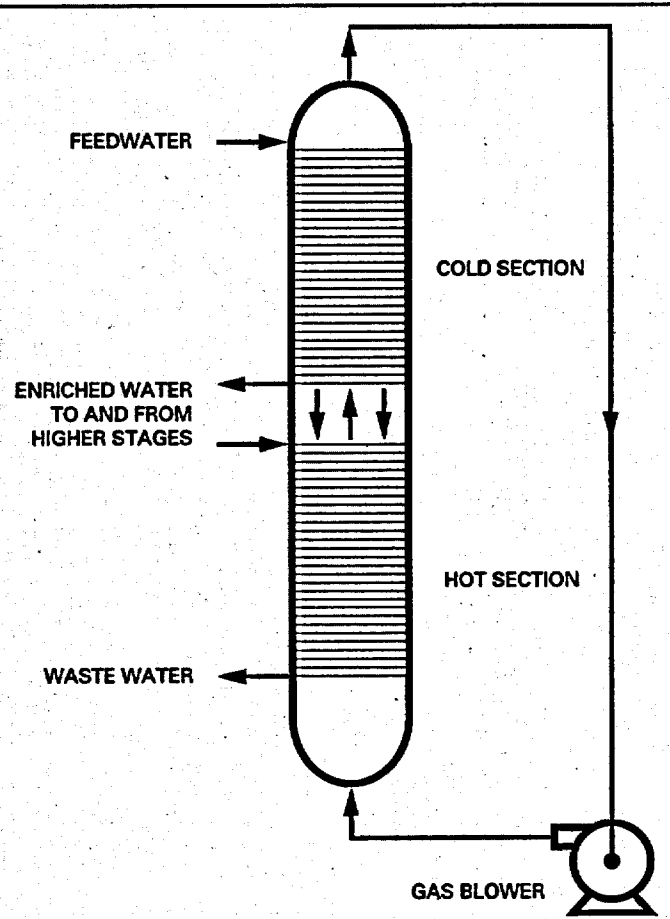
BOX A THE GS PROCESS

When the USAEC selected the water-hydrogen sulfide exchange process to make heavy water for the Savannah River reactors, they asked the Girdler Corporation to operate a pilot plant and to design the first production plant, located at Dana, Indiana. The process became known as the Girdler-Sulfide, or GS.

The basis for this process is the exchange reaction between liquid water and gaseous hydrogen sulfide, $\text{H}_2\text{O} + \text{HDS} \rightleftharpoons \text{HDO} + \text{H}_2\text{S}$. The reaction occurs spontaneously in the liquid phase between the water molecules and dissolved hydrogen sulfide molecules, which are themselves in equilibrium with the gas phase. Deuterium transfers between the gas and the liquid until the two phases are in equilibrium with each other.

The process is carried out in large towers, analogous to distillation towers, containing many sieve trays (see later) on which the gas and water are intimately mixed so that their deuterium contents approach equilibrium. The two phases separate and the gas flows upward from tray to tray countercurrent to the water flowing down from tray to tray. This repeated countercurrent contacting leads to enrichment of the deuterium in one phase relative to that in the other. The tower (figure 17.A1) is divided into two temperature regions. In the upper, cold section, where feed water enters, deuterium is transferred from the gas to the water. At the bottom of this section some enriched water is withdrawn to feed a smaller, higher enrichment tower. The majority of the enriched water passes to the lower hot section where the direction of deuterium transfer is reversed. This hot section provides the enriched gas going to the cold section by stripping deuterium from the water. The gas-to-water flow ratio is carefully controlled (at about 2) so the gas leaving the top of the tower is close to equilibrium with the feedwater, and the waste water leaving the bottom of the hot section is close to equilibrium with this same gas that has been recycled to bottom of the tower. The deuterium concentration in the waste water, relative to that in the feed, thus approaches the ratio of the equilibrium constant in the hot section (130°C) to that in the cold (30°C), 1.8/2.3 or 0.8. Therefore, for these

FIGURE 17.A1
GS Process Tower



temperature conditions, deuterium recovery from the feed is limited to 20 percent. These temperatures and a pressure of 2000 kPa are close to the optimum process conditions which give a reasonable recovery without excessive energy requirements to maintain the hot section temperature; a lower cold

section temperature is prevented by the formation of an ice-like hydrate of water and hydrogen sulfide.

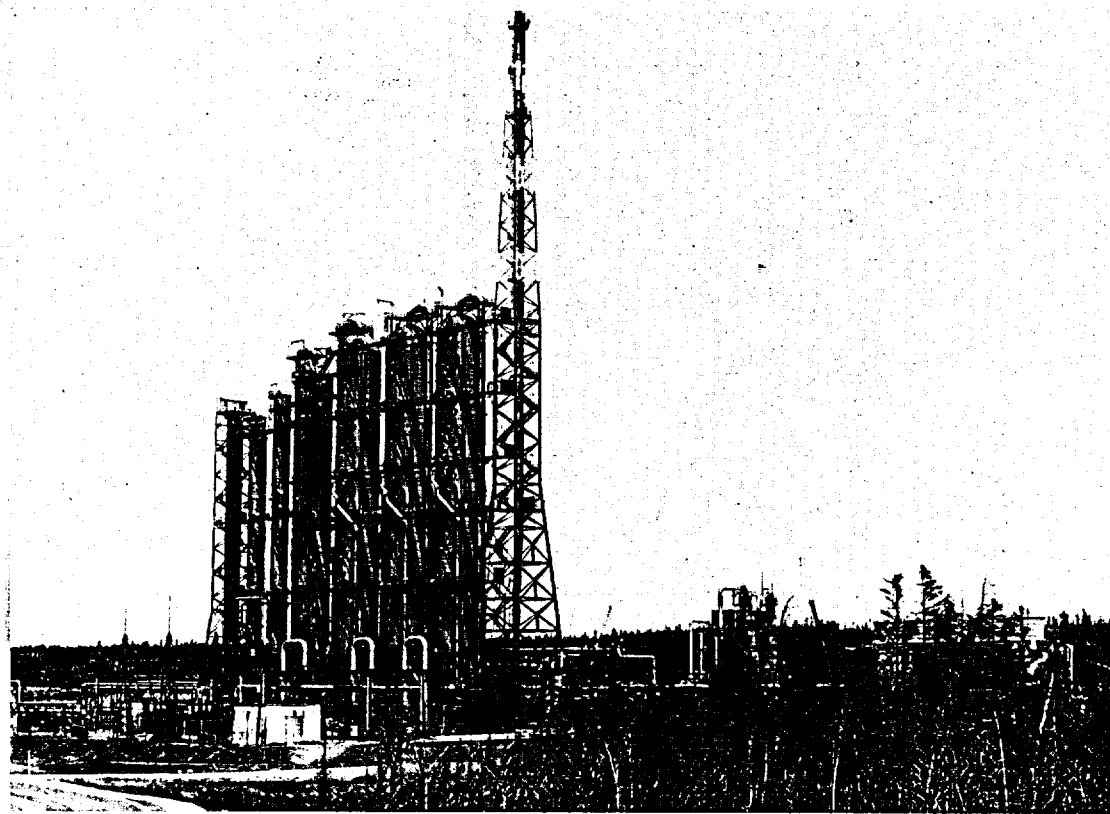
A 400 Mg/a unit, such as the PHHWP shown in figure 17.A2 or one half of each BHWP, has three parallel first stage towers each 8.5 m in diameter and 75 m high to enrich the water by a factor of 4. A 7 m diameter second stage tower brings the enrichment to 24 times natural and two 3 m diameter third stage towers in series (one hot, one cold) produce 30 mole percent D_2O .

The criteria for the final enrichment process are simplicity, reliability and lower pressure. The holdup (inventory)

of heavy water in this part of the plant can be worth millions of dollars so that the risk of any loss must be kept very small. For these reasons water distillation is the preferred process to provide reactor grade product (99.75 mole percent).

The key features of the GS process include a large flow of treated feedwater, large exchange towers with many sieve trays to provide intimate countercurrent contact of gas and water, substantial supplies of steam and cooling water, efficient heat exchange, materials suitable for the corrosive nature of aqueous hydrogen sulfide, and systems to handle toxic hydrogen sulfide safely and to minimize releases of it to the environment.

FIGURE 17.A2
Port Hawkesbury Heavy Water Plant



process patent, which had been assigned to AECL in Canada by the USAEC under the exchange agreement, and would have access to the American technology at Savannah River.

Inherent in AECL's status as a crown corporation, the federal government played an over-riding role in decisions on major expenditures, and these could be used to reinforce other government priorities. In Canada's first heavy-water plant, the government saw an opportunity to foster its new regional economic development policy. The result was that the contract was awarded to Deuterium of Canada Limited (DCL), once they had obtained strong financial support from the Nova Scotia government and revised their bid to propose a location at Glace Bay in Cape Breton, rather than in Alberta. The energy source for the plant would thus become subsidized coal, rather than cheap natural gas, but badly needed jobs would be created in an underdeveloped area. The president of DCL, J.S. Spevack, had invented the GS process during the Manhattan Project. As a result, the USAEC had the basic patent, but Spevack controlled several process improvement patents. Although his technical knowledge of the process was considerable, his company had little experience in managing the design, construction or operation of a large chemical plant. Despite strong reservations expressed both by AECL and the USAEC to granting the contract to DCL, AECL was instructed to do so. It was not an auspicious beginning, and it led ultimately to failure.

The Glace Bay project was plagued by problems – managerial, labour, financial, technical and organizational. It proceeded in isolation from AECL, and even contact with Savannah River was rather limited. The design was more complex than the American plants, in part because of the decision to use sea water both as feed and for cooling.

Meanwhile, in 1964 AECL invited another round of bids to establish a second heavy-water plant, since more capacity would clearly be needed. Once again, as in the first round of bids, the only company with chemical or petrochemical plant experience that bid was Imperial Oil Limited and, for a second time, their bid was rejected as being too high. All the other bids were found to be inadequate. In 1965, AECL obtained approval to negotiate with CGE for the supply of 4,500 Mg of

heavy water over a twelve-year period; the contract was signed in 1966. CGE chose Port Hawkesbury in Cape Breton for their plant site, where steam from an adjacent oil-fired power plant was available, and a federal capital grant was provided for locating in a "designated area". The site had adequate fresh water both as a source of deuterium and for cooling.

Coincident with the CGE contract, AECL was instructed by the government early in 1966 to negotiate a new contract with DCL, to double the capacity of the Glace Bay plant to 360 Mg/a. The new contract called for DCL to deliver 4,500 Mg over twelve years at \$41/kg. However, by the end of 1968, attempts to commission the plant at Glace Bay were abandoned. Spevack had left and Nova Scotia became the sole owner of DCL. An assessment of the plant by SRP personnel was arranged by AECL. Extensive corrosion by sea water, equipment failures and technical errors in design meant that substantial modifications were needed. After prolonged negotiations between the Nova Scotia and federal governments, in 1971 AECL was directed to oversee the rehabilitation of the plant and then operate it.

The Port Hawkesbury Heavy Water Plant (PHHWP) was designed and built by the Lummus Company of Canada Limited, the Canadian subsidiary of the designer of SRP. Where SRP had consisted of twenty-four parallel units, PHHWP had three, each with a six-fold larger output. This scale-up seemed justified in light of the progress in chemical plant experience between 1950 and 1966. Heavy-water production began in 1970. In the next few years a number of serious problems were encountered, which limited output to less than half the rated capacity until 1974.

With the decision to build the Bruce reactors, more heavy-water production capacity was required. In 1969, AECL obtained approval to build the Bruce Heavy Water Plant (BHWP), which Ontario Hydro would operate and have an option to purchase. AECL contracted with Lummus to design and build a plant essentially identical to the PHHWP. During the design phase the capacity was increased from 400 to 800 Mg/a, when it became clear that the Glace Bay plant would require extensive rehabilitation before it could operate. Steam was to be supplied from the Douglas Point reactor, backed up by oil-fired boilers,

and later from the Bruce reactors. Production began in 1973, but was restricted during the first year of operation by some of the same problems encountered at the PHHWP. In June 1973, Ontario Hydro purchased the plant and declared it in-service.

Late in 1973, Ontario Hydro committed three additional 800 Mg/a heavy-water plants for the Bruce site, in anticipation of a continued high growth rate in the demand for electricity. By 1976 Bruce Heavy-Water Plant C (BHWP C) was cancelled and the schedule on BHWP D was extended; it was stopped in 1978 at 70 percent completion. BHWP B was completed and reliable production began in 1981; it incorporated many improvements based on the experience with the first plant, now called BHWP A.

In 1971, AECL awarded a contract to CANATOM MHG⁴ for redesign and rehabilitation of the Glace Bay Heavy Water Plant (GBHWP). A new process arrangement was adopted that would optimize production from the existing towers. A reservoir for fresh feedwater was established, steam from a nearby power station was available, and heat rejection through cooling towers was adopted. The large towers, most heat exchanger shells, many pumps and some piping from the original plant were retained; the remainder was scrapped or sold. Production began in 1976, but satisfactory operation was not achieved until 1978.

AECL's Heavy-Water Projects Division was formed in 1971, with responsibility for the Glace Bay Heavy-Water Plant (GBHWP) and the completion of the BHWP A. AECL in 1973 shared Ontario Hydro's rosy expectations of future heavy-water demand. An 800 Mg/a plant was committed in 1974 for construction by CANATOM at LaPrade in Quebec. This plant, along with the PHHWP and the GBHWP, was to service anticipated sales of CANDU reactors in Canada outside Ontario and in the export market. Construction of LaPrade was slowed in 1976 and stopped in 1978 at 40 percent completion.

In 1975, CGE sold the PHHWP to AECL; they could no longer meet the heavy-water contract price as energy costs rose. Since AECL would soon be operating the GBHWP and was building LaPrade, the sale made better sense than renegotiating the contract. Thus, by 1975 the two organizations responsible for Canadian heavy-water production were Ontario Hydro and AECL.

The failure of the original Glace Bay plant and the production delays at the PHHWP meant that AECL had to scour the world for heavy water for Pickering units two, three and four. About 1,500 Mg were purchased or leased from other countries; e.g., Argentina, Sweden, the United Kingdom, the USSR and the United States. To maintain the Pickering schedule, it was also necessary to temporarily shut down other Canadian reactors, so that their heavy water could be loaned to Pickering; e.g., NRU, Douglas Point, Gentilly-1 (CANDU-BLW). The pressure to get the PHHWP and the BHWP A into full production was intense.

By the end of 1974, Canadian production began to exceed demand, and has done so ever since. Over the three decades from 1945 to 1974, Canada obtained close to 2,500 Mg of heavy water from abroad, about 80 percent of it from the United States. A total of 18,000 Mg had been produced in Canada to the end of 1990. With limited demand for heavy water outside Ontario in the eighties, AECL recommended that the PHHWP and the GBHWP be shut down in 1982; however, the government delayed their approval until 1985, when these plants were finally shut down and subsequently were dismantled. By 1985, AECL had a substantial stock of heavy water available to service future CANDU sales. Once sufficient heavy water for the four Bruce reactors was available, Ontario Hydro shut down BHWP A in 1984, and it has been dismantled. The BHWP B continued to operate, supplying heavy water for Darlington and as make-up to replace losses at all the Ontario Hydro nuclear stations.

THE AMERICAN CONNECTION

Canada had available the United States experience at Savannah River. To achieve as low a heavy-water cost as possible, the throughput per first stage tower in the Canadian plants was increased by a factor of eight over that at the SRP. Tower diameter was increased by a factor of 2.5 and gas velocity by 1.3. Stable operation at the higher gas velocity was demonstrated in 1966 in tests in one Savannah River tower, as part of a broad Canada-United

⁴ A consulting engineering company located in Montreal, originally called CANATOM-MONMAX.

States cooperative development program on heavy-water reactor technology. The extraction of heavy water per first stage tower at the PHHWP and the BHWP was designed to be six times that at the SRP. That it was not eight times higher (directly proportional to throughput) was because the optimum hot tower temperature was found to be lower than used at the SRP, and the number of trays was reduced so that hot and cold contacting could be fitted into a single tower, which provided savings in piping, valving, hemispherical tower ends and tower supports.

Key staff members of the PHHWP and especially of the BHWP were attached to SRP to become familiar with the details of GS process operation and the various problems that had been encountered.

GS PROCESS DEVELOPMENT IN CANADA

Scope of the Development Program

In 1969, with the commitment of the BHWP, AECL began a small effort on GS process development at Chalk River. By 1971, CGE was experiencing hydraulic instability and foaming problems at the PHHWP which were severely limiting production. CGE began a development program to study the flow characteristics of sieve trays when foamy water was present, and to identify foaming agents present in the process; this work was done at the central laboratories of General Electric at Schenectady, NY. The AECL effort was expanded and was soon given a high priority when the extent of the difficulties being experienced became evident. By early 1973, with the BHWP also severely limited in its production, the AECL effort was expanded substantially, technical groups at each heavy-water plant were increased and other organizations were involved on a contract basis.

The total program was coordinated jointly by representatives of AECL, Ontario Hydro and CGE at the senior management level. Many collaborative working groups were formed to exchange information and plan experimental programs. There was good communication between the plants and the research sites, which proved to be an important element in the overall success of the heavy-water production program.

At Chalk River, a small pilot plant was built, with water treatment facilities and a 0.3-metre-diameter column that could operate at either hot or cold tower conditions. It was used for extensive studies of tray performance, of the effects of water quality and to screen a wide range of antifoaming agents. Experiments were done in the production plants to define detailed process behaviour. To obtain adequate information from these plant experiments, additional instruments and sample points were added to one of the towers at the BHWP, and later a tower at the GBHWP was similarly fitted. At the BHWP, AECL supplied and staffed a temporary special analytical laboratory.

The development program was wide ranging, including process analysis and control, process chemistry, analytical chemistry, materials behaviour, mechanical equipment, sieve tray hydraulics and efficiency, gamma scanning and water distillation.

All the Canadian heavy-water plants have experienced equipment, materials and process problems that, by and large, had already been encountered at Dana or the SRP; however, in the American plants these problems had not occurred to the same degree. The American plants had been designed very conservatively, because at that time the GS process was unproven and reliable production on a tight schedule was deemed to be far more important than reducing the capital costs. There was a large element of over-design and the cost of the plants was high – the cost of the SRP (\$130 million in 1951) was about twice that of the PHHWP or the BHWP in constant dollars per annual Mg of production capacity.

The very much larger equipment in the Canadian plants, the higher gas velocities in the towers, and different feedwater chemistry, all contributed to an enhanced effect of plant problems. While the Dana and SRP experience was invaluable in providing a sound foundation for the Canadian plant design and operation, it could not provide the depth of process understanding required to extrapolate reliably to the new conditions. However, this was not appreciated when the PHHWP and the BHWP were being designed, or even when the GBHWP was being redesigned. The large development program was necessary to gain this understanding, which helped to overcome the operating problems, and ultimately to fine-tune plant operation and develop process improvements.

BOX B SIEVE TRAY OPERATION

A sieve tray consists of thin metal sheets perforated by many small holes; the sheets are supported on a set of beams attached to the tower wall. The total area of the holes is about ten per cent of the tray area. Hole size has little effect on tray performance – diameters of 6, 10 and 13 mm have been used in the Canadian plants.

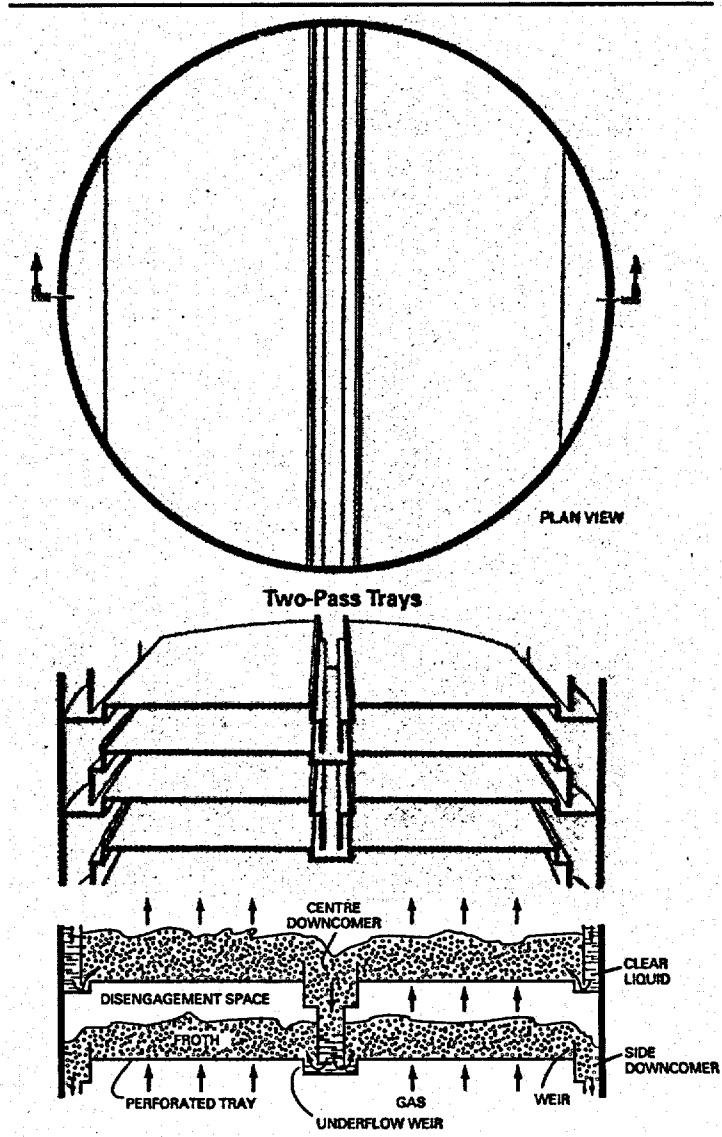
The kinetic energy of the gas passing through the holes, as it flows up the tower from tray to tray, maintains the liquid on the tray. The liquid enters one tray through two downcomers from the tray above located on chords at opposite sides of the tower, and flows inward across each half of the tray and over weirs into a downcomer located on the diameter perpendicular to the flow. This downcomer carries the liquid to the tray below where the horizontal flows across the tray are reversed. This arrangement is called a two-pass tray.

The velocity of the gas leaving the holes (4 m/s) creates a turbulent mixture of gas bubbles and liquid droplets having a large gas-liquid interfacial area. This mixture is called a froth. The amount of deuterium transferred between the phases depends on the area created and the length of time the gas and liquid are in contact – typically about a half a second for the 0.45 m tray spacing used. The tray efficiency, the ratio of the actual amount of deuterium transfer that occurs to the maximum possible if gas and liquid reach equilibrium, is a convenient measure of tray performance.

The froth height on the tray depends on the weir height and the head of froth flowing over the weir necessary to handle the total liquid flow rate in the tower. Obviously, froth density will also be a factor.

The disengagement space above the froth must be sufficient for most liquid droplets to settle out of the rising gas so that the amount carried to the tray above as entrainment is no more than a few percent. This entrainment represents backmixing of the downflow of liquid in the tower and reduces tray efficiency. The overall time for the gas to pass through the froth and then separate from it is about one second (see Figure 17B).

FIGURE 17.B
Schematic of an Operating Two-Pass Tray



Froth entering the downcomer must also separate so that clear liquid enters the tray below, and backmixing of gas is avoided. Therefore, adequate downcomer area is needed and the flow resistance at the underflow weir must provide an appropriate depth of liquid and froth in the downcomer.

If the froth height on the tray is increased, the overall gas pressure drop will rise requiring a larger head in the downcomer. For a foamy system with a low froth density,

this could lead to downcomer backup and the tower would flood. High froth height will also increase entrainment.

Stable tray operation can occur over a range of gas flow rates. The minimum is set by excessive weeping through holes ultimately dumping the liquid from the tray. The maximum occurs when the rise in pressure drop causes the tray to flood due to froth filling the disengagement space.

In general, mechanical equipment performance at the Canadian heavy-water plants has been very satisfactory. Valves, rotating equipment and instrumentation have all performed with a reliability better than or equivalent to that which is typical of the petrochemical industry. Heat exchangers ranked high as a cause of lost production at the PHHWP and the GBHWP, because of the time required to disassemble, repair and re-install.

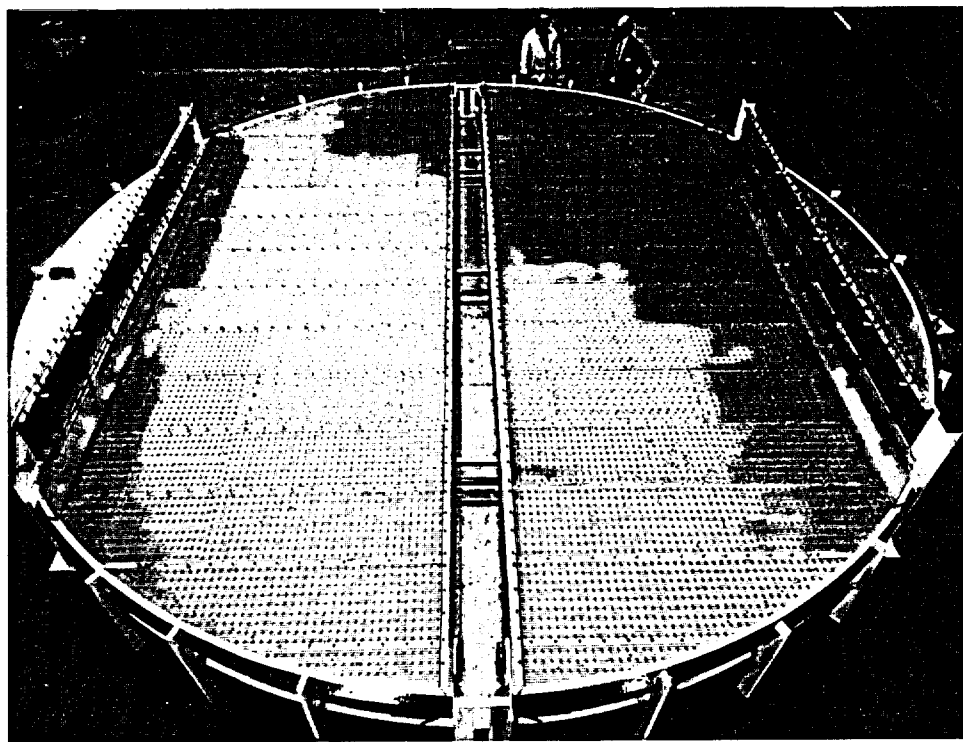
Gas-Liquid Contacting and the Need to Control Foaming

The key to success in GS process performance lies in the design and operation of the sieve trays (see Box B and figure 17.1). These trays must satisfy conflicting requirements: intimate mixing of the gas and liquid on the tray as a froth with adequate residence time for the exchange reaction to occur, and essentially complete separation of the liquid and the gas as they flow to the tray below and the tray above, respectively.

These requirements can be met if the bubbles in the gas-liquid froth have a short lifetime, considerably less than a second. However, if materials are present that increase this lifetime, then the froth is said to be foamy, and the tray performance may suffer.

Foaming had been observed from time to time at the SRP and appeared to be associated with feedwater treatment problems or periods of high turbidity in the raw water. This foaming was adequately controlled by the addition of a silicone-based

FIGURE 17.1
Pre-installation Trial Assembly of a Bruce Heavy Water Plant 8.5 Metre Diameter Tray



antifoaming agent. In the design of the trays for the Canadian plants, it was assumed that the froth could be kept in a non-foamy condition with antifoamers or high-quality water treatment. This design was based on a large body of experience in the chemical and petroleum refining industries with sieve trays, as well as the data from SRP. A froth density equivalent to a non-foaming system was assumed which led to the choice of 100-mm-high weirs, to give sufficient gas-liquid contact time for deuterium exchange.

At both the PHHWP and the BHWP, early operation was plagued by hydraulic instability. This was manifested by a high and fluctuating pressure drop, which led to flooding and dumping of the trays. Stable operation was possible only at about 70 percent of the design flow rate, and deuterium extraction was less than half of the design rate. Foaminess was causing downcomer backup when flows were increased, and high froth height even at the reduced flow rate.

These tray problems were difficult to diagnose. Individual tray operation could not be observed or monitored by instruments. In the early operation of Port Hawkesbury, there were other problems – such as the plugging of the tray perforations by iron sulfide due to iron in the feedwater, necessitating a new water treatment system, and heat exchanger corrosion – which compounded the situation.

The R&D groups that were building up at this time soon found themselves in a crisis situation, responding to a variety of problems, and evaluating many possibilities in parallel to solve major problems, such as the hydraulic instability of the trays. When a small viewing port was installed on one tower at the PHHWP, foaming was observed. The first use of antifoam was successful only for a short time, and then foaming became even worse. It took some time to discover that emulsifiers present in the silicone antifoamer were hydrolyzed in the hot section of the tower to produce an accumulation of fatty acids that are foam promoters. The silicone antifoam specification was changed.

Adding this new silicone-based antifoamer to the feed water at a concentration of a few parts per million allowed stable operation at about 90 percent of the design flow, resulting in 70

percent of the design deuterium extraction rate. Using activated carbon at the PHHWP to remove trace organics from the feedwater was beneficial, but expensive. It was used for only a short trial period.

Before the second enriching unit at the BHWP A was put into operation in 1973, the weirs on the cold tower trays were reduced in height to 40 mm, so that froth height would be lowered. With antifoamer addition to this unit, stable operation at 110 percent of design flow and 93 percent of design extraction was soon demonstrated. The cold tower weirs in the first unit at the BHWP A and at the PHHWP were also reduced in height later in 1973. Once other equipment that was limiting water flow was modified in 1975, it was possible to operate the BHWP A towers routinely at 120 percent of design flow.

Surface chemistry studies demonstrated that the water-hydrogen sulfide system was inherently foamy at cold tower conditions, where the pressure is close to the liquefaction point. It was found that multilayer adsorption of the gas molecules onto the water surface could have the considerable effect of increasing bubble stability by several orders of magnitude relative to that for conditions far from liquefaction; e.g., in the hot tower or at low pressure. This inherent foaminess, apart from any added effects of foam-promoting impurities, meant that froth density was lower than assumed and froth height higher than expected. This did not impede stable hydraulic operation as long as antifoamers were used, but it had an important influence on tray efficiency, as discussed below.

Foaming has caused problems in other areas, such as the dehumidifier,⁵ particularly at the GBHWP, and in the second or third stages. This was due to the accumulation of impurities that were carried down the cold tower, but had sufficient volatility in the hot tower that they did not leave in the effluent. They may originate in the feedwater, in the hydrogen sulfide gas supply, in oils that enter the process from compressor and pump shaft seals, or in the antifoamers added to the feedwater.

⁵ The dehumidifier is the region of the tower where the gas is cooled before it enters the cold section.

A great deal of laboratory effort was devoted to identifying impurities that increased foaming, and this led to improving the specifications for the purity of feedwater, gas supply, seal oil, etc. The Nova Scotia Research Foundation, in collaboration with AECL, played an important role in this work.

A large number of antifoamers were tested in the laboratory and the pilot plant. This work showed that several non-ionic surfactants were excellent antifoaming agents at process conditions. In the plants, these new antifoamers have further improved the stability of the operation and thereby increased production, and are also considerably cheaper than the silicone-based agent that had been used.

Sieve Tray Efficiency

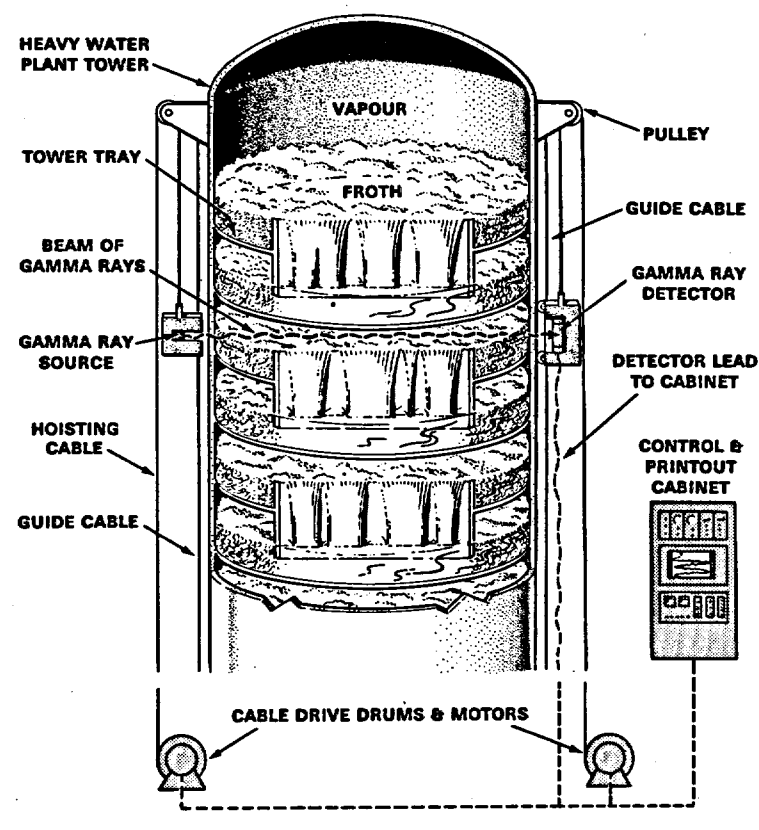
Once the plants were operating stably at near design flow rates, it was clear that the tray efficiency was lower than design. Tray efficiency is the ratio of the actual exchange occurring to the maximum possible if the gas and liquid reached equilibrium before each left the tray. The design had been based on measurements of tray efficiency made at Dana. The technique used was to measure the change in the deuterium concentration profile in the towers that resulted from making small changes in the ratio of gas to liquid flows. It was not until the mid-seventies, when similar measurements were made in the Canadian plants and thoroughly analyzed, that the degree of uncertainty in the Dana results was fully appreciated. However, when the Canadian plants were being designed, the Dana values of 65 percent in the cold tower and 70 percent in the hot tower agreed with the sieve-tray efficiency correlations then available, and were accepted as reasonable. We now know that these correlations do not apply to the GS process cold tower trays, with their low froth densities.

The low efficiency was shown to be in the cold tower once pilot plant measurements were made, and this was confirmed by plant experiments which showed that the cold tower tray efficiency was only 50 percent. The effect at the PHHWP and the BHWP was that the recovery of deuterium from the feedwater was reduced to 90 percent of design. However, this was offset by operating at above the design flow-rate. The GBHWP design was

more sensitive to reduced tray efficiency, and in this plant both cold and hot tower tray efficiencies were low. Thus, the design production rate could not be achieved with the existing trays.

A development program was required to improve tray performance. This program relied on the Chalk River pilot plant, where a wide range of tray operating parameters could be measured, and on plant experiments. The most important tool for obtaining plant data was the gamma-scanning equipment illustrated in figure 17.2. Originally used at the SRP to locate damaged trays, this technique was developed further at Chalk

FIGURE 17.2
Simplified Diagram of Gamma Scanner (installed on a third stage tower with single-pass trays)



River to increase its sensitivity. The attenuation of gamma rays passing through the tower was measured as a function of height. Comparing attenuation through the froth with that through the disengagement space gave the froth density, and as the detector and source were raised could determine froth height on any tray during operation. The technique was also useful for locating regions prone to flooding, and had been useful in diagnosing some of the early problems at the PHHWP and the BHWP A.

To correlate all these data and apply them to making tray improvements, a fundamental mathematical model of the mass transfer and hydraulics of an operating tray was developed by AECL. This model predicted tray efficiency for a given set of conditions defined by pressure, temperature, gas and liquid flows and tray geometry. It took into account the effects of entrainment of liquid in gas rising to the tray above, weeping of liquid through the sieve tray holes, the degree of mixing of gas across the tower cross-section, the degree of liquid mixing in the downcomers and the two-dimensional liquid flow distribution across the tray. Predictions by the model have been in good agreement with direct plant measurements, except for the high cold tower efficiency obtained at Dana.

As a result of modeling, tray modifications were made at the PHHWP in 1977. Certain trays in the dehumidifier were redesigned and the height of the weirs on alternate cold tower trays was increased. These changes increased production by at least 5 percent.

The GBHWP was being re-designed at the time of the major uncertainties about tray performance in the other plants. As a result, the trays were designed conservatively. Low weir heights in the third stage, together with relatively low gas and liquid flows, resulted in low froth height and density, causing low efficiency in both hot and cold towers. When higher weirs were installed in 1977, the gain in efficiency was 23 percent resulting in 12 percent more production. The first-stage hot tower trays also had low efficiency. With the tray model and extensive tray performance data available, it was possible to design a new tray that gave a 12 percent gain in tray efficiency. These new trays were installed in some of the first-stage hot towers, along with higher cold tower weirs. However, by this time a surplus

of heavy water was beginning to accumulate and further retraying could not be justified. If all the first- and second-stage towers had been modified it is likely that the GBHWP would have been able to reach its design output.

Process Analysis

A steady-state computer simulation model of an entire heavy-water plant was developed by AECL to assist in diagnosing problems and optimizing production. This was important, because a GS plant operates as an integrated whole, where all the towers are closely coupled and a number of interconnecting feedback loops exist; it was very difficult to analyze any one part in isolation. The models described the whole plant in detail, and were used to assess the effect of the variation of a local parameter at any point in the process on the overall operation, as well as to define local conditions at a point where no instruments existed. These simulation models were based on methods evolved at McMaster University.

Enthalpy, water, hydrogen sulfide and deuterium balances were calculated for approximately 1,000 trays by iteration. There were typically 2,200 streams to be defined, including a number of large, interacting recycle flows, which made the computation complex and the iteration lengthy.

Extensive plant data available from data loggers and numerous in-plant experiments enabled the simulation model to be tuned to the observed plant behaviour. Close collaboration between the process analysts and the plant operators was essential to ensure that the predictions were realistic.

The model showed how process control could be improved. Small perturbations from ideal operation have significant effects on production, particularly with the reduced tray efficiency. However, the effect of the perturbation of an individual process variable on plant output is very difficult to observe. The model was used to define the ideal settings of process variables and the precision with which they should be maintained. At the GBHWP, the role of the model in process control was more important, because of the complex interconnection of flows amongst towers with this design.

An early application of the simulation was the optimization of the temperature profile in the cold section of the towers. While they are nominally isothermal, there is a cooling effect due to the expansion of the gas from tray to tray. Simulation determined the correct gas and liquid inlet temperatures needed to locate the minimum temperature at its optimum position; experiments in the instrumented tower at the BHWP demonstrated that stable operation was possible within a degree of the hydrate temperature. Thereafter, all the cold tower sections at the BHWP were operated with this optimum temperature profile and the minimum temperature 2°C lower than design, which increased production by 3 percent (worth \$12 million a year in 1985 dollars). Optimizing various temperatures and flows in the second and third stages, based on the simulation model, led to another 3 percent increase in output.

The simulation model has been used extensively to optimize steam consumption for a variety of situations, and to re-optimize process conditions to maintain an acceptable cost of production as steam costs have risen. The model played an essential part in assessing the value of the various tray modifications already discussed.

The detailed understanding of the process that has been gained through the simulation work has led to several potential improvements for future plants, and appropriate process patents have been obtained. The most important of these involves introducing the feedwater at the bottom of the hot section of the tower, and arranging for the heat transfer trays in the humidifier⁶ to also provide deuterium transfer. With this design change, it should be possible to produce 10 percent more heavy water per unit tower volume and unit of process heat than in the best of the existing plants.

Materials

Carbon steel was used wherever possible in the GS plants. Its performance has been good, provided quality assurance specifications and welding procedures were rigorously followed. Blistering, due to hydrogen accumulation at voids or inclusions, has been rare. Following the instances of blistering found early

in the operation of the BHWP B, specifications and inspection techniques were tightened, and would apply to any new plant.

Erosion-corrosion of carbon steel was generally avoided by using stainless steel in regions of high turbulence; however, a few unexpected failures did occur, requiring replacement or cladding by stainless steel. More troublesome was general corrosion of carbon steel piping in the BHWP A downstream of coolers, and the deposition of the iron so dissolved on hot tower trays and in process heaters. This iron transport was also a function of water chemistry and was not a problem at the Nova Scotia plants. A large laboratory effort by AECL to study the corrosion mechanism and the properties of the various sulfides of iron (see chapter seven) showed that the iron dissolution was due to the limited protection provided by the initial corrosion product on steel (mackinawite), and its very slow transformation in these cold regions at the BHWP to the protective sulfides (pyrrhotite and pyrite). A pre-operational conditioning treatment to passivate carbon steel surfaces was developed, and used successfully in the BHWP B; in this case, deposition on hot tower trays has been minimal.

Experience with stainless steel and Inconel in the GS process has been generally good. The use of low-carbon stainless steels minimized the risk of stress-corrosion cracking in sensitized regions, and type 316 was used to avoid pitting in regions where deposits may occur. Polythionic acids, formed by the interaction of sulfides, moisture and oxygen, were found by laboratory studies to be particularly aggressive. It was therefore recommended that all stainless-steel equipment should be flushed with a neutralizing solution before being opened to the air.

Development Program Costs and Benefits

The major benefits of the development program were increased heavy-water production, lower heavy-water cost and better process understanding. Three main contributions which R&D made to increased production were assessed in 1981: helping

⁶ The humidifier is the region at the bottom of the tower where the incoming gas is heated and saturated with water vapour.

the plants achieve mature production rates sooner, defining optimum conditions that gave higher mature production rates, and determining process and equipment modifications that increased production, such as new antifoamers and tray changes.

The initial period of operation, when process instabilities and equipment shortcomings were overcome, was not included in the assessment. The increased production throughout this period was the result of the increasing experience of operators and designers.

Another important contribution from the R&D program, which was difficult to quantify, was the additional technical support available as the expertise of the R&D staff developed. The quick response that the R&D staff made in helping to solve urgent process and equipment problems at the plants contributed to higher capacity factors.

The total cost of the GS process development work to AECL and Ontario Hydro over the period from 1969 to 1980 was \$64 million (1981 dollars), including overheads. This represented slightly less than 3 percent of the value of the 7,810 megagrams of heavy water that was produced during that period. A value of \$300 (1981 dollars) per kilogram was used in this analysis.

The increased production up to 1980 due to the R&D was estimated to total 800 Mg, or about 10 percent additional heavy water. Of this amount, 320 Mg were ascribed to process simulation, 140 Mg to sieve tray improvements and 70 Mg to better antifoamers. The benefit-to-cost ratios for these specific activities were estimated to be 12, 6 and 20, respectively, while this ratio for the overall program was 4. These are conservative values, since no allowance was made to include ongoing benefits.

Although the simulation models cost several million dollars to develop, their benefits have repaid this investment handsomely. However, when the model development was committed, it was much more as an act of faith than a decision based on estimated payback.

The decision to commit a large R&D program to support the heavy-water plants was based on the need to ensure the security of a heavy-water supply. No payback analysis was made beforehand, although an adequate payback was expected. That judgement by AECL and Ontario Hydro management was based

on many years of experience with mission-oriented R&D. It has been confirmed by the results described above and the success of the plants in meeting all heavy-water demands after 1973.

The severity of the difficulties in bringing the PHHWP, BHWP, and the redesigned GBHWP to mature operation was unexpected. The difficulties were certainly more severe than normally encountered in scaling up a chemical process in size and throughput. In retrospect, the effects of foaming and of low tray efficiency stand out clearly. At the time, it took a great deal of effort to disentangle these effects from a myriad of other possible causes of poor operation.

ALTERNATIVE PROCESSES

Early Work

In the late forties and early fifties, AECL had a small R&D effort to develop heavy-water production processes. Variants of the steam-hydrogen exchange process that could be operated independently of a hydrogen production facility were considered. While offering larger-scale production than was possible at Trail, such processes were found to be too costly.

AECL assessed a number of heavy-water processes in 1955, and concluded that the most attractive were likely to be water distillation, hydrogen distillation and several chemical exchange processes. Of the latter, exchange between water and hydrogen sulfide was mentioned, but not discussed at any length. This may seem surprising, considering that the two large American plants that had been started up in 1952 used this method, and that within a decade it was the technology to be adopted for Canada's new heavy-water plants. However, in 1955 the process was still classified as secret, and its details were not known by AECL.

When the scope of the American heavy-water production capacity became known in the mid-fifties, and with the signing of a Canada-United States exchange agreement on atomic energy in 1955, further heavy-water process development by AECL was halted in 1956.

One interesting project during the period of early process studies was the development of a parallel-plate, wetted-wall

packing⁷ for water distillation. A very low pressure drop was achieved, which could significantly reduce the energy required for a distillation system for heavy-water production, using a heat pump to recover the heat from the condenser at the top of the column and transfer it to the boiler at the bottom. However, this could not compete with the GS process. Although this packing was not pursued beyond the pilot scale by AECL, a similar wetted-wall principle was developed successfully into a more practical design by Sulzer Bros. in Switzerland. Their distillation units are used at many heavy-water plants and heavy-water nuclear power stations for concentrating heavy water from about 15 mole percent heavy water to reactor grade, 99.75 mole percent.

It might be concluded that AECL missed an opportunity to develop a product important to the nuclear program, and with applications to other chemical separations. However, in 1956 AECL could not divert scarce resources to such an undertaking and no other organization was interested in licensing the wetted-wall concept.

Process Evaluation

Over the years, AECL has screened a large number of possible methods for heavy water production. Most have been judged to be uneconomical. Several merited further study as alternatives to the GS process. Amine-hydrogen exchange and water-hydrogen exchange were both developed to the pilot-plant stage. During the early seventies, Polysar participated in these evaluations and related process development. Work continues on the water-hydrogen process, since it offers the greatest potential for cheaper heavy water. Laser isotope separation of deuterium was studied extensively by AECL and Ontario Hydro research division in the eighties, but does not appear promising.

Ammonia-Hydrogen and Amine-Hydrogen Exchange

Starting in 1966, a detailed process evaluation was made of the dual-temperature ammonia-hydrogen exchange process, by a group drawn from AECL and several chemical and engineering

design companies. A large-scale plant based on water as feed was studied. High-temperature water-hydrogen exchange transferred deuterium from the feed to hydrogen, which then entered the ammonia-hydrogen dual-temperature enriching process. Exchange rates and other process parameters were based on laboratory results in the published literature. It was concluded that this process was unlikely to be cheaper than the GS process. However, the use of dual-temperature ammonia-hydrogen exchange to extract deuterium from ammonia synthesis gas was also studied, and appeared to be more attractive even though plant capacity would be limited to about 60 Mg/a by the size of a typical ammonia plant (900 Mg NH₃/d).

A small plant using a monothermal version of ammonia-hydrogen exchange⁸ was being built in France at the time, and operated well for four years, starting in 1968. Since 1970, five ammonia-hydrogen plants have been built in India.

The use of amine in place of ammonia in a dual-temperature process appeared to offer advantages, and an experimental program to develop the dual-temperature amine-hydrogen process began at Chalk River in 1967. Methylamine was soon shown to be the best amine. The exchange reaction between the amino group of the methylamine (CH₃NH₂) and dissolved hydrogen is catalyzed homogeneously by potassium methyl amide dissolved in methylamine. However, at a process pressure of about 7 MPa (the pressure for ammonia synthesis), potassium hydride was found to precipitate, thus seriously reducing the concentration of the catalyst. This was also discovered in France, where amine-hydrogen exchange was also being studied. Scientists at AECL discovered that changing the catalyst to a mixture of lithium and potassium methyl amides avoided this problem, and gave AECL a key process patent. The catalyst is extremely reactive. Studies at Chalk River and on contract by Raylo Chemicals

⁷ A distillation column packing in which vapour flowed upward between many parallel, vertical plates, spaced 5-15 mm apart, on which thin films of liquid flowed downward.

⁸ In the monothermal process, the enriched gas that enters the bottom of the cold tower section (Box A) is produced by decomposing enriched ammonia, rather than by exchange in a hot tower section.

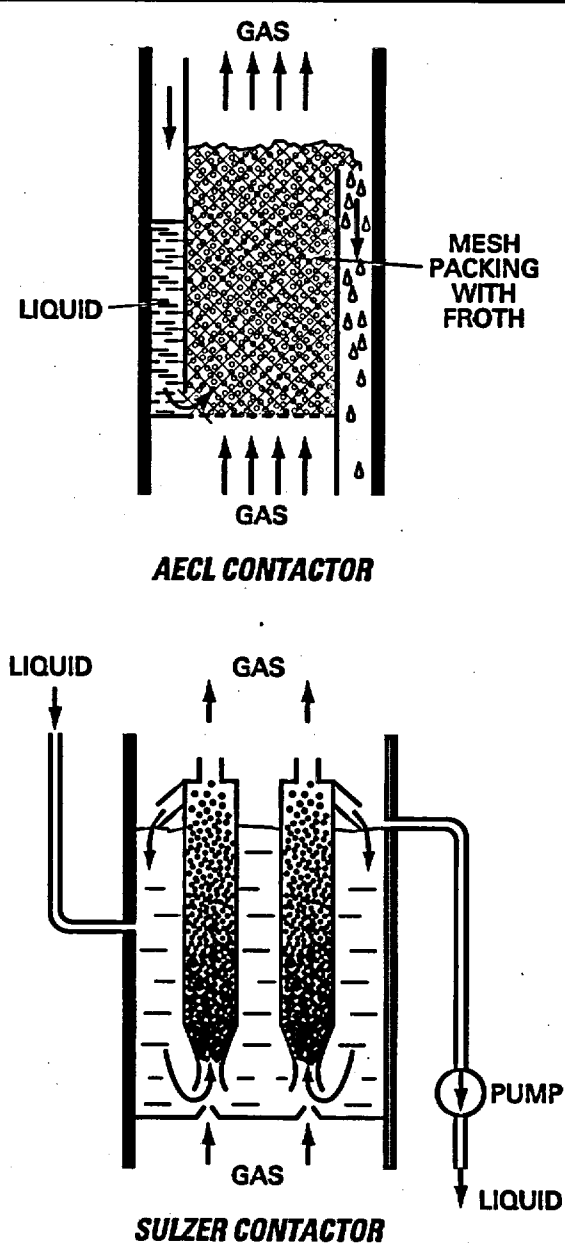
Limited, of Edmonton, defined the reactions with impurities such as water, oxides of carbon and oxygen. Raylo also measured rates of thermal degradation, which set the upper limit for hot tower temperature at 40°C.

Because of the low solubility of hydrogen in methylamine, the mass transfer rate at cold tower conditions is very much lower than that for the GS process. A gas-liquid contactor was developed by AECL, in which a gas-liquid froth flowed up through a deep (1 m) bed of knitted mesh packing on a sieve tray, and then the gas continued to the tray above and liquid flowed over a high weir to the tray below (figure 17.3). This arrangement gives a long gas-liquid contact time. Tests in a 25-cm-diameter contactor of this design demonstrated a tray efficiency of about 20 percent at -50°C. Sulzer had developed a contactor in which the gas and liquid were mixed by nozzles, to give a fine dispersion of liquid droplets in the gas having a very high interfacial area; the mixture was separated centrifugally and the gas went on to the next tray, while the liquid had to be pumped to the tray below, because of the high pressure drop in the nozzles. Some liquid recirculated to the nozzles (figure 17.3). This design, which had been used in ammonia-hydrogen plants in France and India, was tested by Sulzer with the amine system as part of a collaborative program with AECL. A tray efficiency of 70 percent at -50°C was obtained. Thus, the Sulzer contactor required a considerably smaller tower than the AECL contactor, and because of the high pressure and thick vessel walls, this saving more than offset the extra energy and complex tower internals required with the Sulzer contactor.

Sulzer prepared preliminary engineering designs for an amine plant to be attached to an ammonia plant in Alberta. Estimates of heavy-water production costs from this plant favoured the exploitation of the process. This was in 1976, just when the demand for heavy water was being revised downward, and concern about an excess of plant commitments was beginning to appear. As a result, work on the project and development of the process were halted and have not been revived.

The major advantages that the amine process offers over the ammonia-hydrogen process are a significantly higher rate of exchange and a lower cold tower temperature, so that the

FIGURE 17.3
AECL and Sulzer Contactors



equilibrium constant is higher and fewer gas-liquid contacting steps are needed. The result is the potential for a smaller tower volume and lower capital cost. The disadvantage is a more complex catalyst chemistry with an increased operating cost and perhaps lower reliability. The amine process does not seem to have a clear advantage over the GS process.

Water-Hydrogen Exchange

The potential of the direct exchange between water and hydrogen has been recognized since early in the Manhattan Project. A bithermal arrangement similar to the GS process would have the advantages of larger equilibrium constants, less corrosive fluids and much lower safety concerns. However, until 1970, there was no suitable catalyst, and even by 1990 catalyst activity was not high enough for an economic bithermal process. Nonetheless, there are other, more attractive applications of water-hydrogen exchange.

Most catalysts for water-hydrogen exchange fail because they are wetted by water which prevents ready access of the gas to the catalytic sites. An AECL chemist invented a "wetproofed", or hydrophobic platinum catalyst in 1969.⁹ The catalyst, platinum on carbon, is placed on the surface of a tower packing material, such as small alumina spheres. A coating of Teflon makes the catalytic surface hydrophobic. The platinum is thus accessible to the hydrogen and water vapour, so that deuterium can transfer from one to the other. Bare alumina spheres mixed in with the catalytic spheres provide a surface for water to spread on, where exchange between water vapour and liquid water can occur to complete the overall reaction. The catalyst performance has been steadily improved over the past two decades by extensive laboratory and pilot-scale programs. A number of different physical arrangements of the original concept are now available. All involve the countercurrent flow of hydrogen and water through a tower over the catalyst and an adjacent hydrophilic surface.

This exchange reaction can be coupled with electrolytic cells in a flowsheet similar to the Trail plant, to recover heavy water as a by-product of electrolytic hydrogen production. The exchange

towers would be an order of magnitude smaller than those required at Trail, where the gas flowed alternately through dry catalyst beds and sieve trays to achieve transfer of deuterium from gas to water vapour to water. This application of water-hydrogen exchange, known as the CECE process (combined electrolysis and catalytic exchange), is undoubtedly the cheapest method to produce heavy water, provided that the cost of electrolysis is included in the value of the hydrogen produced. Thus, heavy water is a potential by-product of water electrolysis and a Canadian supplier of electrolytic units, Noranda-Electrolyser, has participated in the CECE process development. However, the potential production is small. Large electrolytic hydrogen plants (for fertilizer manufacture) are attractive only in a few locations, with cheap electricity, remote from markets. For example, the 180 MW Kima electrolytic plant at Aswan in Egypt could produce 20 Mg of heavy water per annum. In the early nineties joint studies of this possibility were being conducted by Canada and Egypt.

A somewhat similar combination of water-hydrogen exchange with a conventional hydrogen-producing steam-methane reformer¹⁰ appears attractive. This is called the CIRCE process, for combined industrially reformed hydrogen and catalytic exchange. The process can best be applied to extracting deuterium from the large hydrogen streams produced for upgrading heavy oils in the tar sands, since here the hydrogen is not diluted with nitrogen. Process evaluation in collaboration with a

⁹ He got the idea for the "wetproofed" catalyst while treating his basement walls with a silicone formulation, which, according to the advertisement, repelled liquid water but allowed water vapour to diffuse through it in either direction. He had been interested in heavy-water processes for many years and knew that platinum was a good hydrogen-water isotope exchange catalyst when the water was in the vapour phase, but that with liquid water the rate was low. The nub of his idea was to keep liquid water out but to let the hydrogen and water vapour reach the active sites on the surface of the catalyst.

¹⁰ A reformer is a chemical plant unit in which steam and methane are converted to hydrogen and carbon dioxide, and the latter is removed to produce relatively pure hydrogen.

hydrogen producer has begun, and since 1992 CIRCE development has been given a higher priority so that the process can be demonstrated before the BHWB reaches the end of its life.

The wetproofed catalyst can also be used for upgrading heavy water (a CECE process application,) and in processes to separate deuterium and tritium.

Laser Isotope Separation

When a compound of hydrogen is irradiated with photons of a precise energy that excites only the molecules containing deuterium, they can decompose to give an enriched product. An enrichment of 5,000 or more in a single step has been observed in the laboratory, and the production of a few micrograms of 30 mole percent heavy water from natural fluorine (CF_3H) was demonstrated by AECL researchers. However, the development remains a long way from a practical process. Two key requirements for success are efficient use of the costly laser energy and finding a molecule that works well in the laser separation step and can be redeuterated by exchange with water or hydrogen. Unfortunately, common molecules like water, hydrogen or ammonia cannot be used directly in the process.

HEAVY WATER ANALYSIS

The two most sensitive methods for analyzing water for its deuterium content depend directly on the mass difference between hydrogen and deuterium atoms, rather than on the much smaller variation in a physical property with deuterium/hydrogen ratio. Obviously, one is mass spectrometry, where an ion beam is separated into components of different mass by a magnetic field; this method can be applied to any hydrogen compound. The other is infrared spectroscopy, based on the fact that the three molecular forms of water, H_2O , HDO and D_2O have different molecular vibration frequencies, and thus different absorption spectra, with the most useful absorption peaks occurring in the infrared region. Since deuterium is heavier, the molecules containing oxygen-deuterium bonds vibrate at lower frequencies than those containing oxygen-hydrogen bonds.

Deuterium concentration is usually expressed as atom fraction, $\text{D}/(\text{H}+\text{D})$, or atomic percent, $100 \text{ D}/(\text{H}+\text{D})$. In the case of water, the latter is often referred to as mole percent heavy water, even though the deuterium is present as both HDO and D_2O molecules, and at low concentration is almost all HDO.

Dr. H.G. Thode, at McMaster University, working in collaboration with the Montreal Laboratory in the mid-forties, developed a mass spectrometer for deuterium analysis and provided early heavy-water standards. Mass spectrometry was used routinely by AECL from the beginning to measure heavy water concentration. A small sample of water is injected into a heated volume, where it is totally evaporated, and the vapour is passed over uranium metal at 600°C to produce hydrogen, which then flows to the spectrometer where the mass 2 and 3 or mass 3 and 4 ratios are measured. In the sixties, simpler mass spectrometers became available that could be readily adapted for routine H_2 -HD analysis. Chalk River developed several automated sample systems for these spectrometers, which circumvented the need for special sample preparation or purification, and provided rapid conversion of liquid to water vapour with negligible change in concentration. Modifying the new spectrometers for the requirements of heavy-water analysis in the 0 to 0.1 mole percent range, and combining them with an automated sampler, resulted in an overall analysis system that was reliable, precise and rapid. This overall system has been the standard for our heavy-water plants and research laboratories, and has given a routine precision of ± 0.00003 mole percent heavy water (0.3 parts per million) at natural concentrations.

In the early fifties, infrared spectroscopy methods were developed at Chalk River. Detailed spectra for H_2O and D_2O were determined and then, using samples of intermediate mole percent heavy water, the spectra for HDO was deduced by subtracting the contributions of H_2O and D_2O from the overall observed spectra. A double-beam differential technique capable of good precision was defined for the concentration ranges below 1 percent and above 99.5 percent. In this method, an HDO absorption peak of the sample is compared

with that of a standard in a two-compartment cell. Later, a two-frequency technique was developed, in which absorption at an HDO peak was compared with absorption at a frequency where there is no discrimination amongst the three water molecules. The ratio of these two absorption measurements depends only on heavy-water concentration, and is independent of the effects of temperature drift, sample impurities or instrument instability.

The absorption peaks that enable good precision to be achieved at low concentration are very intense, and this limits their range of application to a narrow region at each end of the overall concentration range. However, a less precise infrared procedure was developed to cover the 1 to 99 mole percent range.

Several on-line instruments for heavy-water analysis in the range 99.0 to 99.9 mole percent were developed by AECL based on infrared absorption. Commercial models were produced by Barringer Research Limited and they have been installed in nuclear power stations and heavy-water plants. As with the infrared heavy-water leakage monitors discussed in chapter fourteen, it was found that the total system – the instrument, sample purification and conditioning equipment, and maintenance strategy – must be specifically engineered for each installation to achieve reliable operation.

The infrared spectroscopy methods were developed originally to be cheaper, faster and more precise than the early mass spectrometry. This was achieved for intermediate and high deuterium concentrations. However, the simple mass 2-3 spectrometer and its automated sampler displaced infrared spectroscopy for low deuterium concentrations.

In the late sixties, a systematic survey of the heavy-water concentration in Canadian waters was made. Ocean samples

were found to contain 0.0155 mole percent heavy water. Values for fresh water varied from 0.013 to 0.015 percent. High values were found in coastal regions, or where enrichment by evaporation can occur, as in the shallower Great Lakes. The low values occurred in northern rivers and in the Rocky Mountain region, where rainfall derives from clouds already depleted in heavy water by previous precipitations. Natural gas in Alberta was found to vary from 0.011 to 0.0135 percent, while industrial hydrogen produced by steam reforming varied from 0.009 to 0.012 percent D/(H+D).

None of the analytical procedures measures absolute D/(H+D) ratios – all rely on calibration against standards. However, various methods have been evolved at AECL and elsewhere to determine the absolute concentration of some of the standards. For those in the concentration range near natural the absolute value has been determined at Chalk River to within ± 0.00005 percent. For example, Chalk River's value for Standard Mean Ocean Water (an international standard) is 0.01576 ± 0.00005 , in agreement with many other laboratories.

The primary Canadian heavy-water standard is maintained at Chalk River. There is no recognized international standard. From 1963 to 1984, its concentration was defined as 99.95 ± 0.03 weight percent heavy water. However, in intercomparisons with other laboratories in the seventies, AECL values tended to be low. Then, in the early eighties, several AECL experimenters measuring various equilibria in near-pure heavy water found that they could achieve consistent results only if deuterium concentrations were adjusted upward. A new technique to measure the absolute concentration of HDO in heavy water was developed, and the Canadian standard was redefined as 99.9720 ± 0.0006 weight percent heavy water.

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Chapter Eighteen

Advanced Fuel Cycles

R. G. HART

INTRODUCTION

Nuclear power reactors produce electricity by harnessing the tremendous amount of energy released when unstable heavy atoms split to form two lighter atoms, the phenomenon known as fission. The only naturally occurring fissionable material is uranium-235, which exists at a concentration of 0.7 percent in uranium. However, nature also contains two materials, uranium-238 and thorium-232, which are converted to fissile material when they absorb neutrons. These are known as fertile materials. When uranium-238 absorbs a neutron it transforms to uranium-239 which decays first to neptunium-239 and then to the fissile material, plutonium-239. When thorium-232 absorbs a neutron it transforms to thorium-233 which decays first to protactinium-233 and then to the fissile material, uranium-233. The recycle of fissile, or fissile and fertile, material from used fuel is known as advanced fuel cycles. The fertile materials uranium-238 and thorium-232 are about 7,000 times more abundant in the earth's crust than is uranium-235, so it is possible to greatly increase the world's supply of nuclear fuel by using advanced fuel cycles. Further, recycling the fissile and fertile material can make the cost of fueling insensitive to the market price of uranium and thorium, allowing lower-grade ores to be used, thus further increasing the nuclear fuel supply.

In the early years of nuclear power it was universally accepted that the supply of economically recoverable uranium in the world was severely limited. Consequently, it was universally accepted that advanced fuel cycles would be a necessity. It was envisaged that the first generation of power reactors would be thermal neutron reactors fueled with either natural or slightly enriched uranium and that the plutonium produced in those reactors would be separated and recycled with fertile material to extend the available fuel supply. Accordingly, in most countries with nuclear energy programs early research programs included components on plutonium separation, the physics of reactors in which to use it, and the development of fuels in which it could be used. In many countries, early advanced-fuel-cycle work focused on fast breeder reactors, which it was thought could use all of the world's uranium. In Canada, early thinking on advanced fuel cycles was focused on thorium breeder fuel cycles, which it was thought could use all of the world's thorium. As early as 1950, W.B. Lewis was writing papers in which he envisaged low-cost electricity for thousands of years through the use of all the world's uranium in fast reactors and all the world's thorium in heavy-water reactors.

As the demand for uranium increased, more was found, and by the mid-sixties it was clear that enough economically recoverable uranium was available to fuel the world's reactors on once-through cycles for many decades. Advanced

fuel cycle work now had to be justified more on economic grounds than resource grounds. In countries with adequate uranium reserves, work was phased down, commensurate with the longer-term nature of the need. In countries with poor uranium reserves, work continued with some priority, on arguments of security of fuel supply and possible foreign price gouging, but the emphasis shifted to recycle in thermal reactors. Fast-reactor programs continued in some countries where the perceived need was the greatest, most notably France and Japan, but with diminished priority. In Canada, there was an early flurry of activity on plutonium and uranium-233 separation and on recycling plutonium in the NRX reactor. Subsequently, the work was confined to the laboratory scale, with the objective of verifying the capability of the CANDU reactors to economically use advanced fuel cycles when that became necessary.

In 1968, a government/industry study on the Canadian incentive for fuel reprocessing and plutonium recycle concluded that plutonium recycle could be marginally attractive in the decade 1975-85. However, the attraction was not sufficient to change the overall thrust of the fuel-recycle program. Again in 1976, when the pressure for work on disposal of used nuclear fuel was mounting and the first oil shocks were reverberating through the economy, AECL asserted to the government that it would be irresponsible to dispose of the fuel without recovery of the tremendous energy value contained in the plutonium, and that the recycle program should be expanded. Reprocessing had, however, by that time, become publicly unpopular, so the only concession to the argument was that the waste management program could contain a component on the disposal of reprocessing waste.

By the mid-eighties the main interest in CANDU advanced fuel cycles was from foreign owners and potential foreign customers, and AECL was searching for a way to demonstrate the technology on a large scale. To do this, they turned to the slightly enriched uranium (SEU)¹ fuel cycle, which could provide a 20 percent savings in fueling costs over the natural-uranium fuel cycle in Canadian reactors. Although not an advanced fuel cycle, the SEU cycle could demonstrate at least the fuel behaviour and fuel management aspects of advanced fuel cycles on the scale required.

ADVANCED FUEL CYCLE PHYSICS

Lewis's early prediction of almost unlimited energy from the thorium-uranium-233 fuel cycle was based on relatively rudimentary hand calculations of neutron balances using the best cross-section data then available. However, by the mid-sixties these predictions were being made using the computer code LATREP (see chapter eleven), and LATREP had been combined with the power-reactor-cost estimating code, CANCAP, to allow various reactor designs and fuel cycles to be compared for cost and natural-uranium utilization. The predictions were, of course, more certain for the natural uranium cycle than for the others, as the physics could be confirmed by experiment (see chapter eleven) and the costs were better known. Nevertheless, the predictions were considered adequate to demonstrate the development potential of the CANDU system and its ability to respond to any future uranium shortage without any major change in reactor design. At the 1971 Geneva Conference, AECL presented papers emphasizing that natural uranium, plutonium-uranium and uranium-233-thorium fuel cycles could all be used in CANDU-PHW, CANDU-BLW and CANDU-OCR reactors with very little difference in fueling costs, but with substantial savings in uranium utilization.

In the plutonium-uranium cycles, irradiated natural-uranium fuels would be reprocessed to recover the plutonium and depleted uranium. The plutonium would be sent to a recycle fuel fabrication facility, where it would be mixed with either natural or depleted uranium to a fissile concentration of about 1.2 percent, converted to conventional oxide pellets, and fabricated into conventional CANDU bundles. In the reactor, the mixed oxide (MOX) fuel would produce about 480 MWh/kg of heavy elements (HE) before it would have to be removed for further recycle, as opposed to about 170 MWh/kg for natural uranium, and would give a saving of about a factor of two in uranium utilization. In the uranium-233-thorium cycle, the plutonium recovered from irradiated uranium would be mixed

¹ Slightly enriched uranium is the fuel of choice for conventional light-water-cooled and -moderated power reactors (LWRs).

with thorium to a fissile concentration of about 2 percent, converted to oxide pellets and made into CANDU bundles. After an irradiation of 240-900 MWh/kgHE it would be removed and reprocessed to recover the residual plutonium, the uranium-233 produced and the thorium. The uranium-233, plutonium and thorium would then be recycled in mixed-oxide form, with just enough extra plutonium to keep the cycle going. The amount of extra plutonium and thus the amount of natural uranium required to keep the cycle going was dependent on the frequency of reprocessing, with the high-burnup cycles giving about a factor of four saving in uranium utilization, and the low burnup cycles giving factors as high as 100. Lewis was particularly enthusiastic about the high-burnup thorium cycle in combination with the CANDU-OCR. He claimed that, because of the high allowable power rating of the fuel (100 kW/m), the high net conversion efficiency (37-39 percent) and the low heavy-water inventory (150 g/kWe), this combination would produce much lower power costs than any other combination possible.²

As better information became available throughout the seventies, it became clear that the costs of both reprocessing and recycle fuel fabrication had been grossly underestimated in the earlier predictions, and that fueling costs with advanced fuel cycles would be at least twice those of once-through cycles at then-current uranium prices. Thus Canadian utilities had no immediate interest in introducing them into Canadian reactors. Canadian utilities did, however, retain a long-term interest, and there was a more immediate interest in some foreign countries who were not blessed with indigenous natural uranium. Potential customers in those countries wanted to know that there were resource-conserving fuel cycles available if the cost of imported uranium became unacceptable. Thus the program continued. Further, as an aid to marketing CANDUs to utilities who already owned light-water reactors (LWRs), the scientists now conceived two fuel cycles that used the fuel discharged from LWRs as the feed fuel for CANDU reactors. The uranium discharged from LWRs contains about 0.9 percent uranium-235, and this can produce an additional 300 MWh/kgU in CANDU reactors. It can also be re-enriched and used again in LWRs, but this is complicated by the presence of uranium-236 and gamma

activity³ in the product, and recycle in CANDUs is simpler and gives better uranium utilization. The recycling in either reactor type involves reprocessing, and consequently higher costs than once-through fuel cycles, but it provides an insurance that has been attractive to some countries, most notably Korea. This cycle has been designated CANFURL, an acronym for CANDU Fueled with Uranium Recycled from LWR. An alternative to this cycle is to recycle both the plutonium and uranium separated from LWR fuel, in CANDU reactors. This material, which is about 1.5 percent fissile, would likely be diluted by about one-third with natural uranium, to be used in a CANDU reactor. This fuel, which would have a fissile content of about 1.2 percent, would produce an energy output of about 530 MWh/kgHE. This cycle has been designated the TANDEM fuel cycle, and again has been of interest to utilities owning both LWRs and CANDUs.

In the late seventies, in the United States, President Carter became very concerned that the spread of advanced fuel cycle technology – in particular, reprocessing technology – around the world would lead to nuclear weapons proliferation. He therefore persuaded the International Atomic Energy Agency (IAEA) to organize an International Nuclear Fuel Cycle Evaluation (INFCE), the objective of which was to develop resource-conserving fuel cycles that would avoid the need to separate fissile material in pure form. Preferred cycles were to avoid reprocessing completely. AECL put a lot of work into the study and, since CANDU is a good neutron-conserving reactor, developed several fuel cycles that gave improved uranium utilization relative to those currently in use. However, none produced by Canada or anyone else could approach the fuel utilization

² For comparison, the natural-uranium-fueled CANDU-PHW power reactors have an established maximum fuel-element rating of 65 kW/m, a net conversion efficiency of just under 30 percent and a heavy-water inventory of 600 g/kWe.

³ Uranium-236 is produced by an n, gamma reaction with uranium-235. It is a strong neutron absorber and concentrates in the fuel during re-enrichment. The irradiated fuel also contains other uranium isotopes produced during irradiation that have gamma-active daughter products.

obtainable through complete fissionable material separation, and after a couple of years the study died a natural death.

By the eighties, AECL had conceived and developed the technology for several resource-conserving fuel cycles that were attractive to CANDU customers, but had been unable to demonstrate them on a large scale in Canadian reactors, because their costs were substantially higher than the once-through natural uranium cycle. However, one fuel cycle – SEU – had a lower cost than the natural-uranium fuel cycle and, although SEU was not an advanced fuel cycle, it would demonstrate the fuel performance characteristics of the advanced fuel cycles, and the reactor and fuel management technology for transition from the conventional to the advanced fuel cycle. Reactor analysis showed that a CANDU fuel enriched to 1.2 percent uranium-235 in uranium could produce 530 MWh/kgU in a CANDU-PHW reactor, with a 20 percent saving in fueling costs and a 30 percent saving in uranium utilization, and that the enriched fuel could be introduced to the reactor in a radial pattern, working from the outside in, without loss of reactor power. The analysis showed that fuel management, both during transition and at equilibrium, would be aided by the use of a fuel bundle that had smaller-diameter elements in the outer two rings and larger-diameter elements in the centre, which gave it more tolerance to neutron flux variations in the reactor.

Early in the advanced fuel-cycle program, the LATREP code used in the reactor physics calculations was verified against experiments with natural uranium and plutonium-depleted uranium fuel in ZED-2 (see chapter eleven), but there was a continuing desire to obtain experimental verification with all the other fuels proposed. This was one of the major rationales for the commitment in the late sixties to a recycle fuel fabrication laboratory (RFFL), which was to be used to produce multi-bundle-sized lots of plutonium-natural uranium, plutonium-thorium, uranium-235-thorium and uranium-233-thorium fuels in oxide form. However, because of the low priority of the advanced fuel cycle program, financing was limited and the RFFL did not produce its first fuel until 1978, and its first recycle fuel until 1979. Once fabrication started it went well, but the reactor physicists did not have all the fuel they required for all of their ZED-2 experiments until

1988. To attempt to ameliorate the delays, the physicists, around 1975, negotiated an agreement with CNEN (the Italian nuclear research organization) to fabricate (plutonium, uranium) O_2 fuel for ZED-2 experiments from plutonium separated from CANDU fuel in a cooperative reprocessing experiment. Fabrication of this fuel was also substantially delayed, and it was not available for experiments until 1985. Eventually, however, experiments were completed on (plutonium, uranium) O_2 , (thorium, plutonium) O_2 and (thorium, uranium-233) O_2 , and checked against code predictions. By this time, the type of calculation being done had changed from a cost comparison of different fuel cycles in different reactor types to a more detailed optimization of lattice parameters and fuel utilization with different fuel cycles in the CANDU-PHW. Accordingly, LATREP had been replaced by WIMS-AECL, a more sophisticated code originally developed at the Winfrith laboratory of the United Kingdom Atomic Energy Authority (Winfrith Improved Multigroup Scheme), which had been substantially modified by AECL to adapt it to CANDU lattices. In general, the agreement between WIMS-AECL predictions and the experiments was good, with the only apparent problem being the inability of the code to accurately predict the change in reactivity with temperature of fuels containing plutonium.

ADVANCED FUEL CYCLE FUELS

The reference fuels for advanced fuel cycles in CANDU reactors were plutonium-uranium, plutonium-thorium and uranium-233-thorium, all in oxide form. Until the mid-eighties, the reference bundle geometries were identical to those for natural uranium. Thus, all the development work for natural uranium fuels (see chapter thirteen) was directly relevant to fuels for advanced fuel cycles. The additional work required was to verify that the fuel could survive the higher burnups required in the advanced fuel cycles (500 vs 200 MWh/kgHE), to adapt the manufacturing techniques to accommodate the alpha activity of plutonium and the gamma activity of the uranium-233 feed,⁴ to ensure that these

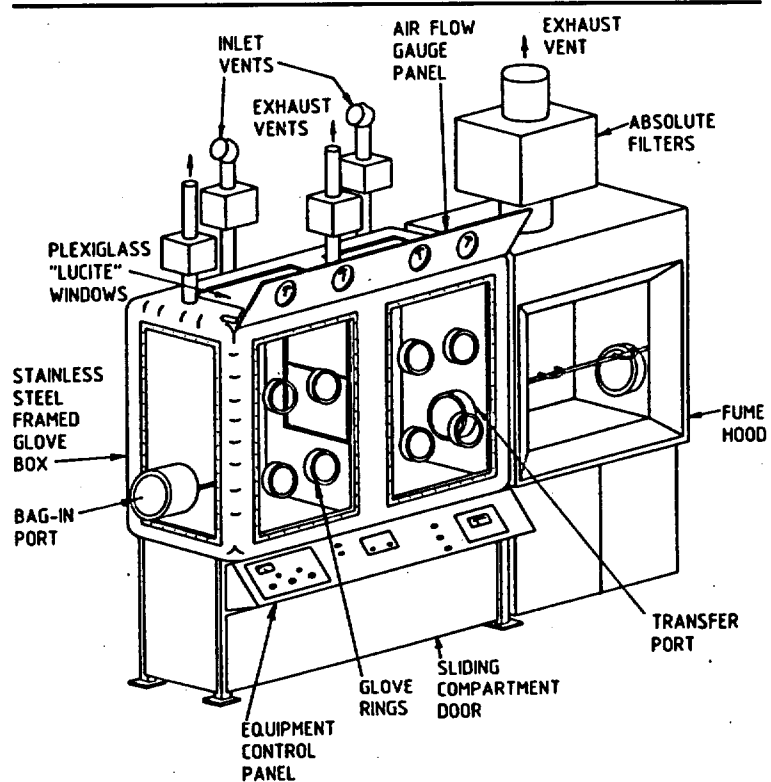
⁴ The gamma activity comes from daughter products of uranium-232, which is produced by an n,2n reaction with uranium-233.

adaptations did not negatively impact on fuel performance, and to ensure that the behaviour of thorium-based fuels was at least as good as uranium-based fuels.

In 1965, work commenced on the fabrication of single-element (uranium-plutonium) O_2 and (thorium-plutonium) O_2 fuels for irradiation testing. It was expected that fuel pellets could be made by mixing the oxide powders, then pressing them into pellets and sintering in cracked ammonia, as was then done for uranium-dioxide fuel pellets. However, cracked ammonia is highly flammable, and there was concern about using it for large-scale production of pellets containing toxic plutonium. There was a desire, therefore, to determine whether pellets produced by sintering in a dilute hydrogen atmosphere (10 percent H_2 in argon) would give acceptable irradiation performance. Accordingly, pellets were made by both methods for comparison. The other variable tested was powder particle size. The fuel elements were irradiated in the X-2 loop in NRX in 1966. The results showed that there was no significant difference in performance between pellets made by the different sintering methods, or from the different particle sized powders, that there was no significant difference between the behaviour of (uranium-plutonium) O_2 and uranium dioxide irradiated at the same ratings, and that (thorium-plutonium) O_2 was somewhat more resistant to structural change than uranium dioxide irradiated at the same ratings. Subsequent tests showed that acceptable fuel could be made from powders arising from co-precipitation, and that oxygen/metal ratios should be kept close to stoichiometric.

The RFFL at Chalk River comprised three lines of glove boxes and fume hoods connected by appropriate conveyers for the product (see figure 18.1). Sintered oxide pellets were made, ground, sized and stacked (see chapter thirteen), and were loaded into commercially fabricated Zircaloy sheaths suitably equipped with brazed spacers and bearing pads, and with the first end-plug resistance welded in position. After loading and filling with helium, the second end-plug was inserted and welded in place using tungsten inert-gas welding. Bundles were then assembled by welding the elements to Zircaloy end plates using a commercially made resistance welding and assembly device. Product quality was assured by stringent control of

FIGURE 18.1
Glovebox and Fumehood, RFFL



incoming powders for blending, along with careful control of process parameters to ensure consistency. The laboratory had its own dedicated analytical section, with the capability to do all the required quality assurance work. There was also a dedicated metallographic/ceramographic line for examination of fuel microstructure. Finished fuel elements were nondestructively assayed for fissile content using a commercial neutron interrogation device, and neutron radiography was available to provide additional information if other inspection techniques showed anomalies. Protection against the toxicological hazards of plutonium and uranium-233 was assured by handling them in dry-boxes until they were sealed in the Zircaloy sheaths, and by using personal air samplers to monitor the air in the vicinity

of each worker. Protection against the gamma radiation emitted by the daughters of uranium-232 present in the uranium-233 feed was assured by processing the uranium-233 shortly after purification, and by the use of thermoluminescent dosimeters (see chapter three) worn on both the chest and the fingers. Protection against potential criticality incidents was assured by dividing the laboratory into fifteen zones and controlling the amount of fissile material in each zone to well below its critical mass.

As mentioned earlier, the RFFL was delayed in its construction and commissioning, and was not ready for operation on recycle fuel until 1978, but once operation started, fabrication went very well. From 1979 through 1980, fifteen 36-element bundles of (uranium-0.5 wt. percent plutonium) O_2 were completed for irradiation in NRU. Thorium was introduced into the line in 1981 and from 1981 through 1982 production included six bundles of (thorium-1.75 wt. percent, plutonium) O_2 for irradiation in NRU, one bundle of natural ThO_2 for irradiation in NRU (to study the distribution of uranium-233 produced), two bundles of (thorium-1.8 wt. percent uranium-235) O_2 for irradiation in WR-1, and two bundles of (thorium-2.3 wt. percent plutonium) O_2 for irradiation in WR-1. From 1982 through 1988, 1,332 elements of (thorium, plutonium) O_2 fuel and 1,350 elements of (thorium/uranium-233) O_2 fuel were fabricated for reactor physics experiments in ZED-2.

During the seventies some work was done on the fabrication of thorium/uranium-233 fuels by alternative methods that might be more compatible with remote fabrication. This would eliminate the need to fabricate shortly after purification. The methods studied were: 1) impregnation of porous pellets of thorium with a nitrate solution of uranium-233, 2) vibratory compaction of fully dense, mixed-oxide microspheres into the fuel cladding (the microspheres would be produced by an already proven process known as the sol-gel process), and 3) extrusion of a mixed-oxide clay, made by a process similar to the sol-gel process, into a mixed-oxide rod, which could then be cut into pellet-length slugs and sintered. After testing, the impregnation process was eliminated because it was incompatible with co-processing, vibratory compaction was eliminated because of

poor irradiation performance, and extrusion was eliminated because experience showed it to be no easier to adapt to remote fabrication than the conventional pellet route.

Because of the delay in the RFFL, advanced fuel-cycle fuel bundles were not available for irradiation testing to the high-burnup required until the early eighties. However, relevant information could be obtained by taking SEU fuels to higher burnups than those required for natural uranium; by irradiating thorium mixed with uranium-235, and by following the fate of natural-uranium fuel that for various reasons was delayed in discharge from the power reactors. Since the large water loops at Chalk River were tied up with priority work on natural uranium fuel, irradiations were done in NPD and WR-1. In the seventies, the irradiation of four 19-element bundles of (thorium, uranium-235) O_2 was started in NPD and the irradiation of twenty-four 22-element bundles of (thorium, uranium-235) O_2 was started in WR-1. These bundles performed well and had reached burnups of 900 and 450 MWh/kgHE respectively, when the reactors were shut down in 1985 (WR-1) and 1987 (NPD). Unfortunately, because of the characteristics of these reactors, the fuel could be operated only at modest power ratings (36-48 kW/m). Ten 19-element bundles of SEU fuel were also irradiated in NPD, and they also worked well to 700 MWh/kgU, but again at low rating. A study of the history of the natural-uranium fuel in the power reactors showed that by 1985 over 3,000 bundles had been irradiated to burnups between 270 and 700 MWh/kgU, with many of them exceeding 500. All had been discharged and transferred to storage in the normal course of fuel handling with no problems. Unfortunately, the ratings of these are indeterminate, but most are likely low.

The plan for the irradiation of fuel fabricated in the RFFL was to put it into the NRU loops whenever there was appropriate space available, so that eventually it would achieve high burnup at high rating and, along the way, some bundles could be power ramped⁵ as desired. Irradiation of (uranium, plutonium) O_2 and

⁵ Moving quickly from low power to high power. This is an action that will occur in CANDU reactors and it stresses the fuel more than uniform power operation.

(thorium, plutonium) O_2 bundles started in 1981 and 1984, respectively. In addition, irradiation of three bundles of SEU and two bundles of (thorium, uranium-235) O_2 fabricated in industry started in 1980 and 1982, respectively, and in 1983 AECL was able to take over responsibility for an irradiation of five elements of (thorium, uranium-235) O_2 and nine elements of (thorium, uranium-233) O_2 started by Westinghouse Power Division (United States) in NRX in 1981. By 1985 SEU had been taken to over 700 MWh/kgU, (uranium, plutonium) O_2 , (thorium, uranium-235) O_2 and (thorium, uranium-233) O_2 to over 500 MWh/kgHE, and (thorium, plutonium) O_2 to 115 MWh/kgHE, with only one failure, that in a SEU bundle that was power-ramped to a very high rating (70 kW/m). Most tests were to be continued to a target burnup of 1,000 MWh/kg.

These tests were augmented by special-effects tests in smaller loops, to study diametral changes, internal operating temperature, fission-gas release, defect behaviour and accident conditions, and all tests were compared against predictions made by computer codes developed for this purpose (see chapter thirteen). In most cases the predictions were accurate, with the only major discrepancy being in the predictions of the behaviour of (thorium, uranium-235) O_2 fuel made by the co-milling route. This fabrication route produced pellets with a coarser microstructure and a lower thermal conductivity than those made by the conventional method, so they operated at a higher central temperature. This fabrication route was discontinued. Subsequently, hot-cell examination of high-burnup fuel from a Bruce reactor identified a second area of discrepancy between prediction and performance, in that the fission-gas release was considerably higher than predicted. This was attributed to a burnup-dependent decrease in thermal conductivity, believed to result, at least in part, from the buildup of solid fission products in the uranium-dioxide matrix. The increase in fission-gas release was not enough to cause the fuel to fail, but it did introduce a factor that had not before been appreciated.

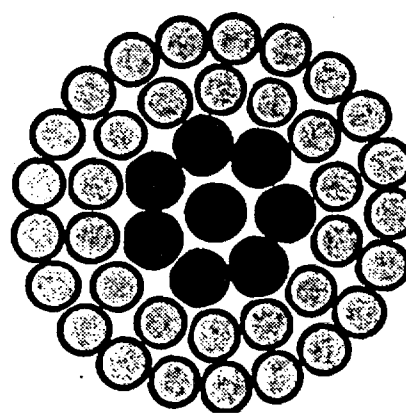
The excellent behaviour of natural uranium taken to high burnup in power reactors and the behaviour of SEU and (thorium, uranium-235) O_2 in NPD and WR-1, reinforced by the excellent behaviour of SEU, (uranium, plutonium) O_2 , (thorium,

uranium-235) O_2 , (thorium, uranium-233) O_2 and (thorium, plutonium) O_2 in the loops, led, by 1985, to a high degree of confidence that fuel of the conventional CANDU design would give good performance in any of the advanced fuel cycles envisaged.

Although it was clear that fuel of conventional CANDU design could be used with confidence in advanced fuel cycles, it was also clear that this was not the optimum bundle design for these cycles. A better design would be one that tended to equalize the linear power rating of elements across the bundle. AECL therefore developed a new design that it called the CANFLEX bundle. The bundle was more subdivided than the conventional 37-element design, as it contained 43 elements, with the elements in the outer two rings being of smaller diameter than those in the centre. (see figure 18.2). The smaller outer elements allowed the bundle to operate at 1,250 kW without exceeding the 65 kW/m peak element linear rating that had been established as the maximum for CANDU fuel. Alternatively, if the CANFLEX bundle was used at the power limit of 1,035 kW, set for 37-element bundles, the peak element rating would be reduced by about 20 percent. This would provide increased flexibility to operate to

FIGURE 18.2
Canflex Fuel Bundle

CANDU FLEXIBLE FUELING



higher burnups, to withstand greater power ramping and to accommodate greater flux variations during transition from a natural-uranium fuel cycle to an advanced one. The advantages were such that the CANFLEX bundle was accepted as the reference design for all advanced fuel cycles.

Over the period, irradiation testing was also done on fuel types that, it was thought, could operate at the very high power ratings proposed by W.B. Lewis in some of his thorium reactor concepts. These fuels had either annular pellets or thin graphite discs between pellets, or a combination of both, to keep the temperature of uranium dioxide at acceptable levels. The fuels were irradiated in the NRX and WR-1 loops at powers up to 70 kW/m to burnups up to 800 MWh/kgU without any problems. In fact, two of the elements were inadvertently ramped to 90 kW/m for short periods without problem. The tests showed that CANDU fuels capable of operating at much higher power densities than that possible with the CANFLEX fuel were practicable and easy to fabricate, if required.

FUEL REPROCESSING

The wartime task given to the Canadian-United Kingdom team at the Montreal Laboratory was the pursuit of the heavy-water reactor route to plutonium and uranium-233 production. As a result, the early Chalk River laboratory included not only the NRX reactor but chemical plants for the separation of the plutonium and uranium-233 produced in it. These plants operated from 1948 until 1954. Final purification facilities were added in 1951 and 1952, and the plutonium separation plant was replaced by a much simpler one, based on ion exchange, in 1955. All facilities were shut down in 1957, when arrangements were made to sell used NRX and NRU fuel to the United States. Over the period 1948 to 1957, these facilities produced about 17 kg of plutonium and about 500 g of uranium-233. Most of the plutonium was recycled in NRX in plutonium-aluminum alloy fuel rods. None of the processes used was suitable for use in advanced fuel cycles so they are not described in this chapter. They are described briefly in chapter two and details can be found in the references given in the footnote.⁶

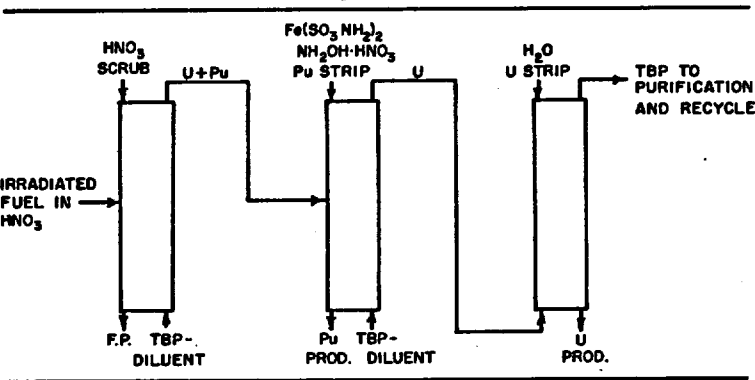
By 1950, W.B. Lewis was beginning to develop his first concept for a natural-uranium-fueled, heavy-water-moderated power reactor. In this concept, the fuel was to be uranium metal, which would have to be recycled with its contained plutonium after about 70 MWh/kgU, because of the dimensional instability of the uranium metal under irradiation. Lewis assigned the task of developing a process that could economically recover both uranium and plutonium to the chemical engineering branch at Chalk River.

At the time, laboratories in the United States had begun to work on a solvent extraction process that used tri-butyl phosphate (TBP) as the extractant, and this process was beginning to look very promising. Thus the TBP process was adopted by the chemical engineers as their reference process (see figure 18.3). In this process a 20 percent solution of TBP in a diluent such as kerosene was counter-currently mixed with the nitric acid dissolver solution from which it extracted both the uranium and the plutonium. The plutonium was then preferentially backwashed from the solvent using a dilute aqueous solution of a reducing agent such as ferrous sulphamate, following which the uranium was backwashed from the solvent with a very dilute aqueous solution of nitric acid. After a number of laboratory studies to optimize the process parameters, the chemical engineers designed and built a pilot plant to test the process on NRX fuel.

The heart of the plant was a series of mixer-settlers (see figure 18.4), in which the solvent and aqueous fluids were mixed and separated in counter-current flow. In theory the number of mixing and settling stages determined the extraction efficiency and product purity, but in practice these were determined by the presence of minute amounts of complexes that behaved differently than the bulk materials. Thus in practice it was necessary to have three extraction/backwash cycles, with conditioning between cycles, to obtain the product purity required. The pilot plant worked well and confirmed that uranium and

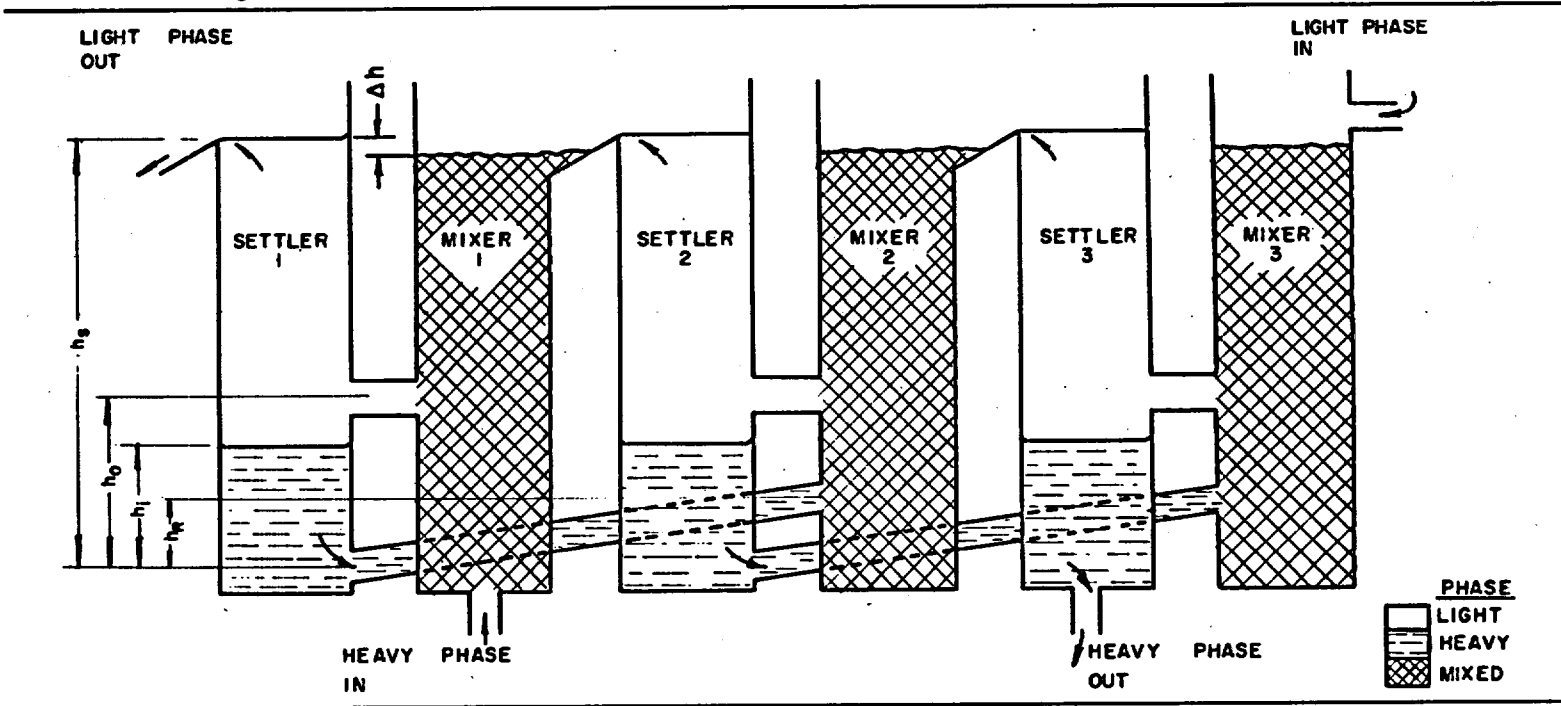
⁶ W.M. Campbell, Canadian Chemical Processing, Nucleonics, Vol. No. 9, September 1956. Also, O.J.C. Runnalls, Irradiation Experience with Rods of Plutonium/Aluminum Alloy, AECL-601, September 1958.

FIGURE 18.3
TBP Process Flowsheet – Single Cycle



plutonium could be effectively and probably economically recovered for recycle, as required in the power reactor concepts

FIGURE 18.4
Schematic Flow Diagram for the Three Stage Mixer Settler



being proposed. However, by this time, development work in other areas had indicated that uranium dioxide fuel could produce close to 240 MWh/kgU in a heavy-water-moderated reactor without recycling, and that this was sufficient to make a once-through fuel cycle economical. Thus the immediate need for recycle was removed and the TBP pilot plant was shut down. The focus of AECL's reprocessing work now changed to that of putting on the shelf all the technology required to allow the rapid design and construction of a versatile commercial reprocessing plant when such a plant became necessary. The spectrum of fuels to be considered was broad, including uranium metal, uranium oxide, thorium metal, thorium oxide and plutonium-aluminum alloy, all clad in aluminum or zirconium. Since the TBP process offered the promise of being able to reprocess all of the above fuels in a single plant, it was adopted as the reference process. The work, which was done on a laboratory scale,

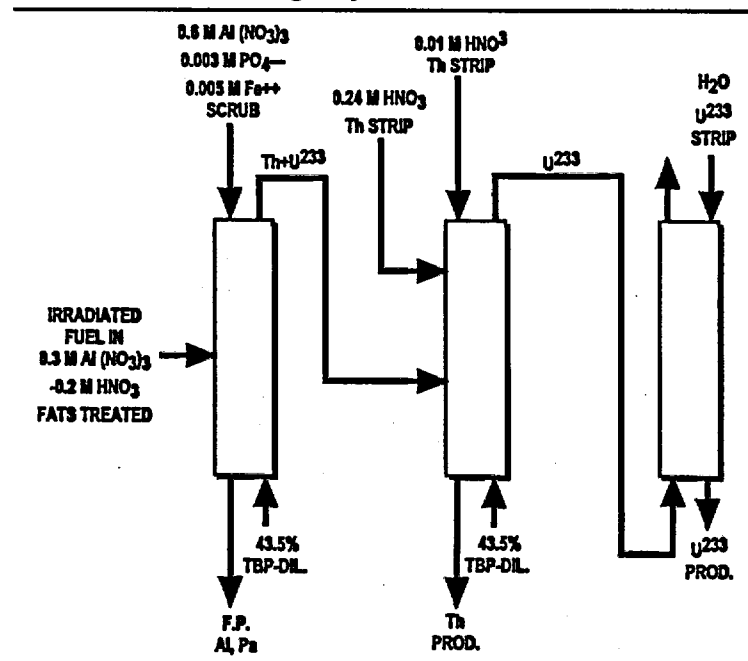
covered all aspects of the process, including decladding and dissolution of the fuels, optimization of the extraction and backwash conditions for each of the fuels, concentration of the final products, and recovery and reconditioning of the solvents for re-use.

Generally, the preferred methods for decladding were dissolution of aluminum in caustic soda and chopping the zirconium cladding into short lengths to expose the fuel. The alternative for aluminum was to dissolve the cladding in nitric acid, along with the fuel, and this was done with the addition of mercuric nitrate, to speed up the reaction. The alternatives for zirconium were dissolution in ammonium fluoride, dissolution in hydrofluoric acid, volatilization of the cladding as $ZrCl_4$ (hydrochlorination) and disintegration of the cladding by high-temperature oxidation. Reaction rates were obtained for all systems, and information was obtained on the problems with each method, and ways of controlling those problems.

For the TBP processes, all fuels had to be dissolved in nitric acid, and reaction rates were determined for all fuels of interest. In the case of the thorium metal and ThO_2 fuels, hydrofluoric acid had to be added to increase the dissolution rates to practical levels and, in the case of plutonium-aluminum fuels, mercuric nitrate had to be added for the same purpose. Phase diagrams were developed for the relevant systems, so that solution volumes could be minimized without the risk of precipitation. The advantages of batch vs continuous dissolution were evaluated, and methods of saving acid using the appropriate treatment of the off-gas were developed.

For extraction and separation, partition coefficients were developed for all product and relevant contaminant materials over a range of conditions, and these were used in the conventional way (McCabe-Thiele diagrams) to establish the optimum conditions for product recovery and product purity. For irradiated uranium fuels, the optimum feed solution was 2-3 M HNO_3 , while for irradiated thorium fuels the optimum feed was acid-deficient⁷ (-0.2M HNO_3 , and 0.6M $Al(NO_3)_3$) (see figure 18.5). For irradiated plutonium-aluminum fuels, it was considered best to add uranium and process in the same manner as the irradiated uranium fuels. This alleviated criticality concerns and

FIGURE 18.5
Thorex Flowsheet - Single Cycle



also was thought to improve fission-product decontamination. In uranium or plutonium-aluminum processing, the plutonium was backwashed from the solvent first with a reducing agent, either ferrous sulphamate or hydroxylamine nitrate, then the uranium was backwashed with 0.01M nitric acid. In thorium processing, the thorium was backwashed from the solvent first with 0.24M nitric acid, then the uranium-233 was backwashed with 0.01M nitric acid.

The products from solvent extraction existed in quite dilute solutions, and they had to be concentrated before they could be stored or used. At the time, three concentration methods - evaporation, cation exchange and anion exchange - were in general use and these were evaluated. Evaporation following steam stripping of the solvent was quite straightforward for all products, and

⁷ The solution was made acid deficient by evaporating to molten thorium nitrate and decomposing some of the nitrate ions.

phase diagrams for product/nitric acid/water were developed for all products, to avoid overconcentration. The only significant problem was the formation of large amounts of hexavalent plutonium, which was undesirable in subsequent fuel fabrication steps. Some work was done on reducing agents that would put all the product in the desired tetravalent state, but the best solution turned out to be to hold the product in 7-8M nitric acid for a couple of months, during which time the product would reportion to over 99 percent tetravalent. Both cation exchange and anion exchange would provide additional purification if used for concentration, with anion exchange giving the better purification and cation exchange the greater concentration. Auxiliary purifications, such as silica-gel removal of fission product zirconium and niobium, and thiocarbanilide-thioacetamide removal of ruthenium, were available to complement the ion exchange purifications if they were required.

Tributyl phosphate in use undergoes hydrolysis to form dibutyl phosphate, monobutyl phosphate and phosphoric acid, all of which can have a detrimental effect on process decontaminations, so it must be treated periodically to remove these impurities. It was found that a simple caustic wash was insufficient and that this would have to be followed by steam stripping. Work, both at Chalk River and elsewhere, showed also that the choice of diluent was quite important, as some were much more stable than others and their decomposition products affected process decontamination. Olefin-free normal dodecane and C12n-paraffin were specified as preferred diluents, and the specification called for them to be vacuum distilled, to remove impurities before use.

It was mentioned earlier that the microcomponents, not the macrocomponents, of the system governed the product decontamination obtained in each cycle, and that in practice three extraction/backwash cycles with intermediate feed conditioning were needed to give the product purity required. Some work was undertaken to identify these microcomponents, to determine whether they could be altered so that the number of extraction/backwash cycles, and thus the number of contactors in the reprocessing plant, could be reduced with resulting cost savings. Fission-product ruthenium and plutonium were the

problem contaminants in the uranium product stream, while fission-product zirconium and its daughter niobium were the problem contaminants in the plutonium stream. It was discovered in other laboratories that ruthenium nitrosyl nitrate and colloidal zirconium were the problem fission-product species, and it was discovered at Chalk River that plutonium dibutylphosphate was the problem plutonium species. Work on the ruthenium and zirconium problems failed to produce any practical method of reducing the number of purification cycles required, but work on the plutonium problem at Chalk River produced a practical method of reducing the number of plutonium/uranium separation cycles from two to one, and the total number of contactors in the reprocessing plant from nine to seven.

By about 1960, the technology required to design and construct a multi-purpose reprocessing plant was reasonably in place and, since there was a need for all the effort that could be made available on problems related to the development of the once-through cycle CANDU, work on reprocessing was suspended.

In 1968, W.B. Lewis became a champion of the CANDU-OCR concept, which he felt was the best reactor for his Valubreeder fuel cycle, a fuel cycle based on the recycle of uranium-233 and thorium topped up with plutonium from uranium-fueled reactors. Since organic reactor technology was primarily a Whiteshell responsibility, Whiteshell also undertook to verify the technical aspects of the Valubreeder fuel cycle – in particular, the reprocessing aspects of that cycle. As mentioned above, technology had been developed for the recovery of plutonium and uranium from irradiated uranium fuels and uranium-233 and thorium from thorium fuels, but the Valubreeder fuel cycle required recovery of only plutonium from the uranium fuels, and plutonium, uranium-233 and thorium from the thorium fuels. Since there was concern about the hydrolysis and polymerization of plutonium in the acid-deficient thorium reprocessing system used at that time, it was felt that an acid flowsheet would have to be developed. Further, it was felt that the economics of the cycle would be improved if only plutonium was removed from the irradiated uranium fuel, and the uranium and fission products were denitrated and calcined to a form suitable for disposal.

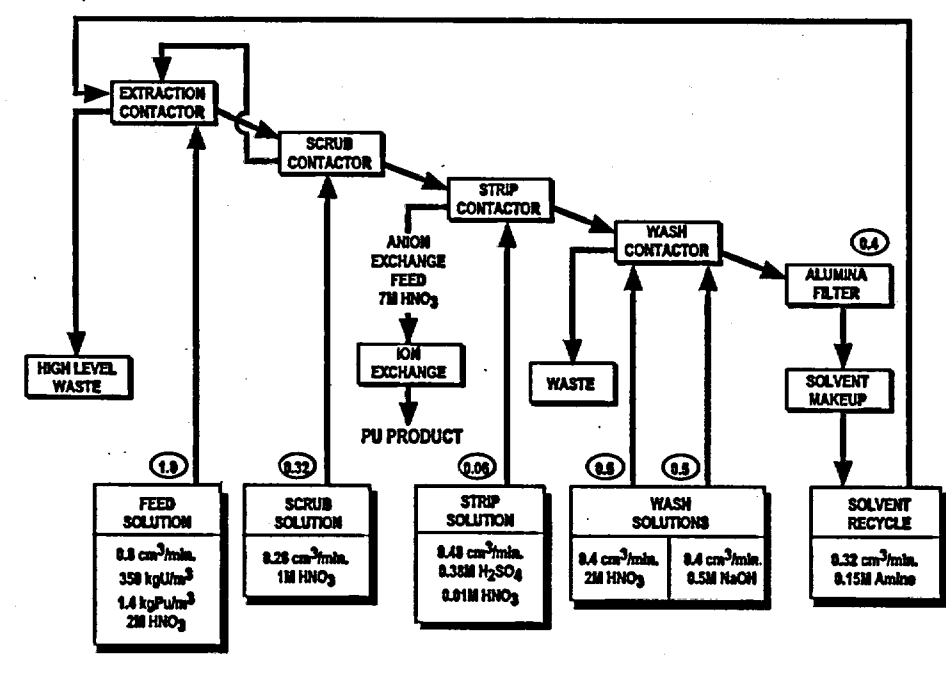
Plutonium had already been effectively separated from uranium and fission products using anion exchange (see chapter two), and anion exchange resins were now available in liquid form. Thus it was felt that the old anion exchange process could be used, but now in a more convenient liquid/liquid extraction form. These "liquid ion exchange resins" were amines, and the process became known as the amine extraction process (see figure 18.6). The amine, diluted with diethylbenzene, was contacted in conventional counter-current solvent extraction equipment, with the dissolver solution conditioned to 2M nitric acid. The plutonium was extracted into the solvent, leaving behind the uranium and fission products. After scrubbing, to improve the decontamination, the plutonium was backwashed with dilute sulphuric and nitric acid, and then concentrated by anion exchange. The process was tested in a mini pilot plant constructed in the Whiteshell hot cells, and it worked well. It was found that

two cycles with intermediate conditioning were sufficient to give the product purity required. In the early seventies, the opportunity arose to demonstrate the amine process on a larger scale in a versatile pilot facility in Italy. Fifty bundles of discharged Pickering fuel were acquired from Ontario Hydro and shipped to Italy, where the technical viability of the process was verified on the larger scale. Concurrent with the amine process tests, work was initiated on the denitration and calcination of the uranium/fission-product stream. This work showed that the uranium/fission-product waste could be conditioned for disposal using fluidized bed denitration and a rotary calciner. However, work was done only on inactive synthetic solutions of the appropriate composition, and the process could not be considered proven until demonstrated on the radioactive solution itself.

After the mini pilot-plant tests were conducted on the amine process, attention turned to reprocessing the thorium-uranium-plutonium fuels. Conventional McCabe-Thiele

calculations using partition data developed in the laboratories indicated that thorium, uranium-233 and plutonium could be extracted with 30 percent TBP from a 2M nitric acid feed solution and backwashed with 0.01M nitric acid containing 0.3M hydroxylamine nitrate and 0.1M hydrazine, to reduce the plutonium to the trivalent state (see figure 18.7). The backwash solution was then to be adjusted to 1.5M nitric acid and contacted again with 30 percent TBP, which would extract the thorium and uranium, but leave the plutonium in the trivalent state behind. The thorium could then be backwashed from the solvent with 0.5M nitric acid leaving the uranium behind, then the uranium could be backwashed with 0.01M nitric acid. In tests in the mini pilot plant, it was found that the hydroxylamine nitrate could not hold the plutonium in the aqueous phase in the plutonium separation step, so it had to be replaced by 0.05M ferrous sulphamate. With this change, the process worked well, but the

FIGURE 18.6
Amine Process Flowsheet



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WASTE MANAGEMENT

Chapter Nineteen

Waste Management

R. G. HART

LABORATORY WASTE HANDLING

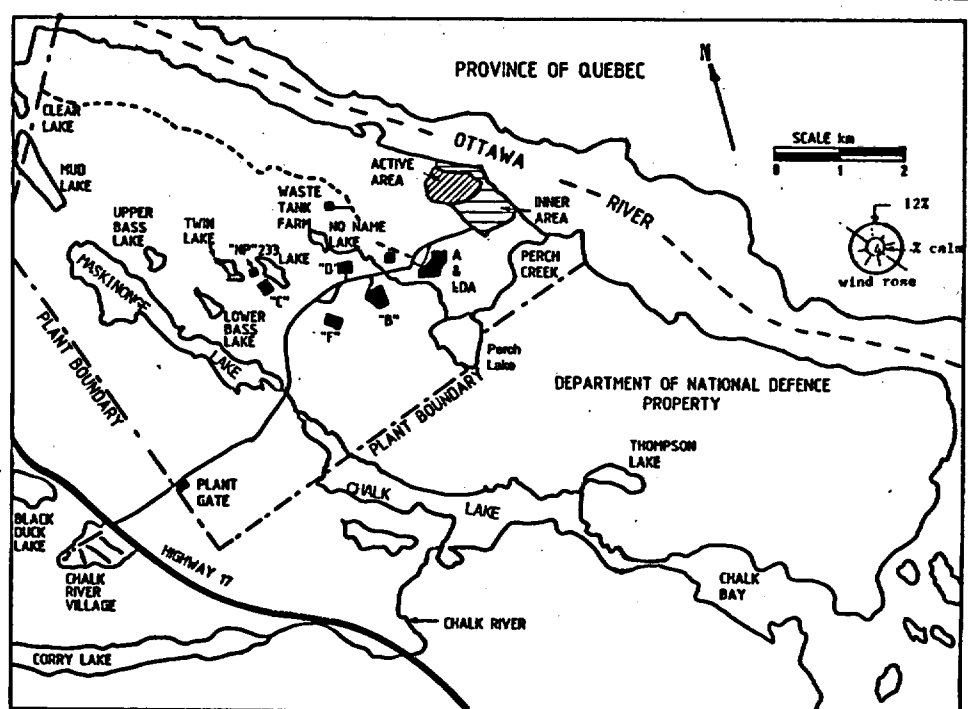
By the time Canada's fledgling nuclear program moved to the Chalk River Laboratories in 1946, scientists were well aware of the hazards of radioactivity and knew that the wastes from the laboratory would have to be handled with special care if harmful effects to people and the environment were to be avoided. The wastes the scientists had to deal with were as varied as the operations of the laboratory itself, ranging from highly radioactive irradiated fuel and radioisotopes and highly radioactive solutions from fuel reprocessing and isotope separation, to slightly radioactive liquids from rod bay cooling, active-area process drains, and equipment and clothing cleaning, to radioactive paper, rags, glassware, rubber gloves and other solids arising from general laboratory operations and experimentation. The highly radioactive fuel reprocessing and isotope separation wastes could be stored for a time while processes for treating them were developed, but the lower activity wastes had to be handled immediately. The volume was simply too large to afford delay.

In specifying the methods of handling the wastes, the early scientists had to be aware of the radioactivity released to the environment from the remainder of the laboratory, in order that the total release would be well within the safety guidelines of the day. Forty years later it would be necessary to perform complex computer calculations to prove "beyond reasonable doubt" that the selected method for handling the wastes would not have deleterious effects on people and the environment. However, in 1946 only sound logic and good judgement was required. That the waste management facilities at Chalk River have withstood the test of over forty years of operation without even approaching any releases of concern is testimony to the soundness of judgement exercised by those early scientists.

The Chalk River property extends over a total area of approximately 37 km². The topography is typical of the Canadian Shield, consisting of gently rolling hills (mainly eroded bedrock outcrops) interspersed with several small lakes and marshes (see figure 19.1). In many areas between bedrock outcrops, pockets of glacial till and other sediments are covered by extensive sand deposits. The water table in those sandy areas is usually several metres below the surface. These well-drained sandy uplands were chosen as the best locations for the waste repositories. The good drainage would preclude the wastes sitting in water, thus minimizing leaching, while the sand would adsorb material that was leached, retarding its movement by groundwater flow. In the centre of the 37 km² property is a depression in the bedrock, from which all water drains into a small lake, called Perch Lake. A stream, known as Perch Creek, drains the lake to the Ottawa River. The early scientists decided to locate the waste management areas in the sandy uplands of the Perch Lake basin. Thus they had a situation where any radionuclides leached from the waste had to collect in Perch Lake, where the water could be monitored and if necessary processed. No such processing has ever been necessary.

The first waste management area to be brought into service was a 1.2 hectare area located 750 metres from Perch Lake. It is now designated Waste Management Area A (WM Area A) (see figure 19.1). It was fenced-in in 1946 to restrict access by people and large animals, and was used from 1946 until 1952 for emplacement of low-level solid wastes, which were transported to the area in plastic or paper bags and buried directly in the ground in sand trenches. During this period, the amount of radionuclides going to the process sewers was limited by collecting them in glass bottles in each laboratory. The type and scale of laboratory activity at that time made this practical, and

FIGURE 19.1
Map of CRL Property Showing Waste Management Areas



allowed most of the process sewer water to be discharged directly into the Ottawa River. These glass bottles were taken to WM Area A and buried in a large concrete tank, which contained sufficient limestone to neutralize any acid in the bottles if they broke. The area also contained a ceramic "tomb," which was used for the emplacement of cobalt-60 as metal. It was subsequently sealed all round with asphalt.

The 1952 accident to NRX prompted a drastic change in the use of WM Area A. After the accident, about 4,500 m³ of contaminated water collected in the NRX basement, and it had to be removed before cleanup and rehabilitation could commence. It was decided to pump the contaminated cooling water into WM Area A. A series of interconnecting ditches was dug over the existing sand trenches, and the contaminated water was pumped into these ditches via a temporary steel

pipeline. Subsequently, the damaged NRX calandria and a miscellaneous collection of contaminated pipes, equipment, shielding and other material associated with the reactor dismantling and repair were also buried in WM Area A.

The use of the area for the NRX accident waste essentially terminated its use for routine emplacement of low-level solid waste, although bottles of laboratory liquid waste continued to be buried in the large concrete tank in the area until 1955. In 1954 and again in 1955, experimental quantities of fuel-reprocessing waste were released into the area to determine the hazards that could arise if a reprocessing-waste storage tank were to fail and allow its strong acid contents to penetrate the soil to the water table. It was found that the high-density acid solution sank rapidly through the soil to the water table, carrying most of the radionuclides with it. However, as soon as the acid was neutralized by the soil, the movement of radionuclides slowed to a small fraction of ground water velocity, and at that rate of movement would decay to an insignificant level by the time they reached Perch Lake. After 1955, all activities in WM Area A, with the exception of routine monitoring, ceased.

With the rebuilding of the NRX reactor in 1953, and the increased emphasis on nuclear power in the research program, the waste management requirements changed significantly. More experimental loops in NRX, more fuel examinations in reactor bays and hot cells and a general change from laboratory-scale to engineering-scale activities led to increased levels of radioactivity in both the process sewers and the solid wastes. As a result, it was necessary to include in the waste management area facilities for handling large volumes of slightly radioactive process sewer wastes and facilities for handling more highly radioactive solid wastes.

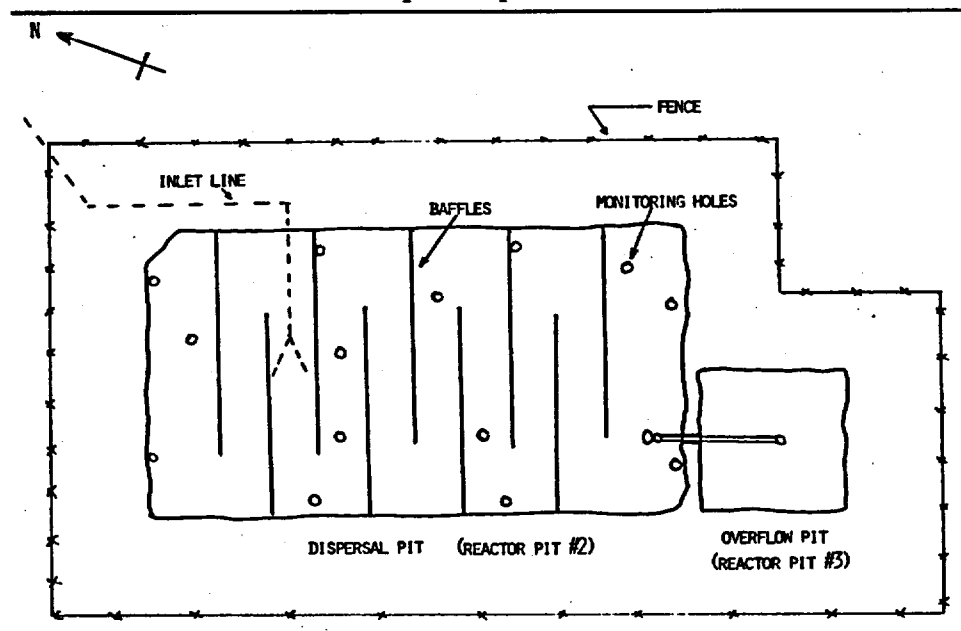
The process sewer waste was separated into two categories: (1) reactor process sewer waste, comprising water from the fuel rod bays, the reactor loops and other reactor systems, which

contained few chemical contaminants, and (2) chemical process sewer waste, comprising liquid from the chemical plants and the chemical and metallurgical laboratories, which could contain acids, alkalis, complexing agents or salts in low concentration. Solutions containing large quantities of chemicals were not permitted in the process sewers.

The solid wastes were now also separated into two categories based on the amount of contained radioactivity. Solid waste that measured less than 0.01 mSv/h gamma 30 centimetres from the surface, or less than 10 mSv/h beta at the surface, and which contained no alpha contamination, was designated as low-level waste. All other solid waste was designated as high-level waste. Alpha-contaminated waste was included in the high-level category because it had to be assumed that the alpha irradiation came from toxic plutonium.

In 1953, a liquid dispersal area (LDA, see figure 19.1) was opened in the sandy uplands of the Perch Lake basin, close to WM Area A. The first pit in the area was a natural depression

FIGURE 19.2
Plan View of Reactor Pits and Liquid Dispersal Area



in the sand, approximately 45 metres in diameter and 3 metres deep. It was designated Reactor Pit #1 and was used between 1953 and 1956 to disperse 230,000 m³ of low-level liquid waste from the reactor process sewer, mainly from the fuel-rod bays. When taken out of service in 1956 the pit was filled and covered with clean sand. It was replaced by a pit of improved design, designated as Reactor Pit #2. This was a man-made pit, baffled with asbestos board and filled with coarse aggregate (see figure 19.2). The liquid waste entered the pit about 1.2 metres below the surface and the waste seeped into the sandy soil. The aggregate provided means of preventing the sand walls from crumbling, acted as a gamma-radiation shield, controlled the spread of radionuclides by either wind or biota, and prevented freezing. In 1960 an overflow pit, Reactor Pit #3, was constructed to the same design, because it was felt that a buildup of sludge in reactor Pit #2 was going to limit its life. However, about that time, improvements in the systems feeding the reactor process sewer made Reactor Pit #3 redundant. From 1956 to the end of 1984, a total of 1,467,000 m³ of reactor process sewer waste was dispersed in this area, but the annual volume decreased from about 210,000 m³ per annum in 1960 to about 15,500 m³ per annum in 1984.

In 1956, a circular pit 25 metres in diameter, filled to a depth of 4 metres with aggregate, designated Chemical Pit #1, was built for disposal of wastes from the chemical process sewer. This was followed in 1958 by a second pit of the same design for overflow. From 1956 to 1984, a total of about 315,000 m³ of chemical process sewer waste was dispersed in the area, but the volumes decreased from about 18,700 m³ per annum in 1960 to about 3,400 m³ per annum in 1984.

Since 1984, the volume of liquid wastes and the amount of radioactivity dispersed in this area have been reduced even further, as a result of the operation of the Waste Treatment Centre, which will be discussed later.

In 1953, a new area was put into service for the emplacement of solid wastes. It was located in another sandy upland of the Perch Lake basin, about 600 metres removed from WM Area A and again about 750 metres from Perch Lake itself. It was designated WM Area B (see figure 19.1). It is a much larger area than

Area A, comprising about 14 hectares, 8 hectares of which have to date been fenced in for use. Initially, all waste was buried in unlined sand trenches, as in Area A. However, as the operations of the laboratory changed and the waste was separated into high-level and low-level categories, new methods were developed for handling the high-level segments. The first method was to bury the high-level waste in asphalt-lined trenches in the sand. Once full, the trenches were covered with a layer of asphalt, and then covered with sand. This method was abandoned in 1959, when severe cracking of the asphalt covers was noted and water was found inside one of the trenches. A reinforced polythene sheet was installed above the asphalt to keep the water out, and the polythene was covered with sand to protect it. The asphalt trenches were followed in 1959 by reinforced concrete bunkers. Initially, a unit consisted of a rectangular bunker 60 metres long, 1.8 metres wide and 1.8 metres deep, divided into 12-metre long bays by cross walls. When in use, the structure was covered with a temporary roof. When full, the contents were covered with sand and a concrete roof. The structure was then covered with sand and graded (see figure 19.3). In 1963 the design was modified to a double bunker with cross and centre walls. The top 30 centimetres of the bunker were left above ground, and the cap was made removable to facilitate eventual recovery of the waste. In 1979 the design of the concrete bunkers was changed from rectangular to cylindrical because of concern regarding the strength of the unsupported rectangular bunker walls while the bunkers were empty (see figure 19.4). The cylindrical bunker is still in use.

As the experimental program, particularly the power reactor program, progressed, it became necessary to include in WM Area B storage facilities for intensely radioactive material such as irradiated fuel, hot-cell wastes, experimental fuel bundles, unusable radioisotopes, spent-ion exchange resin columns, active exhaust-system waste and fission-product waste from medical isotope production. These materials were stored in facilities known as tile holes which were concrete drain pipes set vertically on a poured concrete base at the bottom of a trench a minimum of 1.2 metres above the water table. The pipes were 4.9 metres deep and of varied diameters, up to 2.45 metres (see

FIGURE 19.3
Typical Single Concrete Bunker, Waste Management Area 'B'

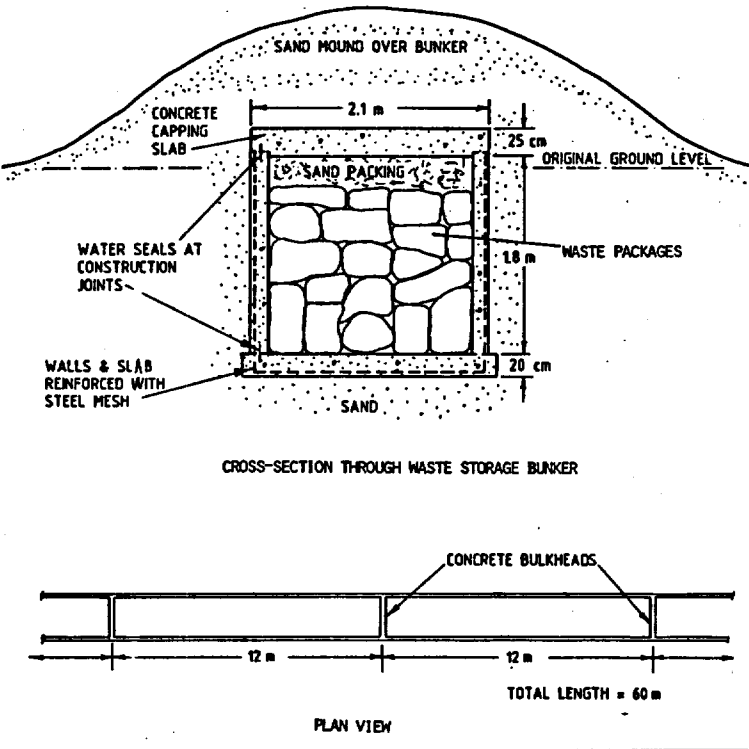
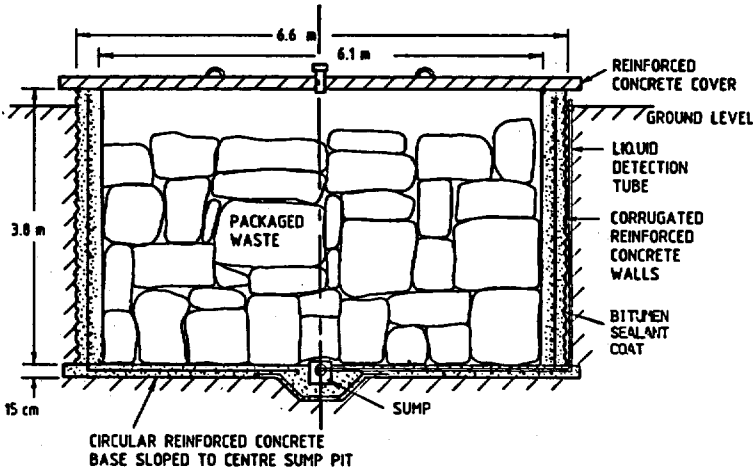


figure 19.5). The outer surface was waterproofed and the joint at the base was sealed. When the array of tile holes was assembled in the trench, the tops were closed with steel-and-concrete caps and shield plugs, and the trench was backfilled with sand to within a few centimetres of the top. The tile holes were then filled from transport flasks as required. For irradiated fuel, special tile holes were developed, with steel pipes within the tile and vented closures, to aid in the dissipation of heat and prevent a buildup of internal pressure (see figure 19.6).

Over the years, a number of special burials took place in WM Area B, including the second NRX reactor vessel in 1970, the first NRU reactor vessel in 1973, an evaporator used to reduce the volume of highly active reprocessing wastes in 1958, about 300 kg of natural-uranium scrap and turnings from fuel manufacture

FIGURE 19.4
Typical Cylindrical Concrete Bunker – Waste Management Area 'B'

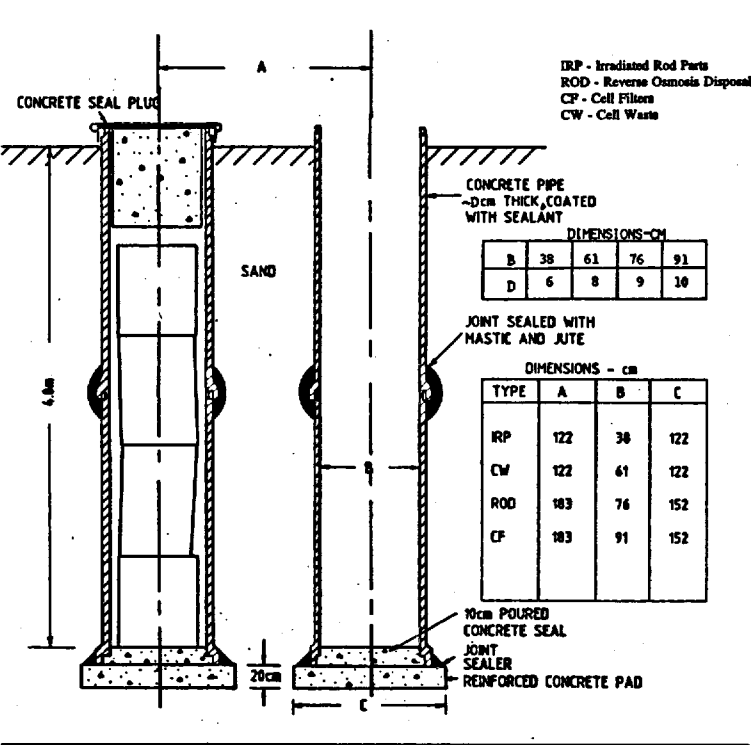


dating from 1962 and 1963, concrete made from mildly radioactive reprocessing wastes in 1955, and spent cobalt-60 sources from radiotherapy units returned between 1957 and 1960. WM Area B is still in service.

One of the earlier activities at the Chalk River laboratory was the experimental reprocessing of the irradiated fuel from the NRX reactor to recover the plutonium (see chapter two). Initially the plutonium was recovered by extraction with triethylene-glycol dichloride (the trigly process), which left as waste a highly radioactive solution of nitric acid, ammonium nitrate and uranium. In 1955 this process was replaced by an anion exchange process which left as waste a highly radioactive solution of nitric acid and uranium. Subsequently, the uranium was recovered from these solutions using tributyl phosphate, leaving two types of waste: one containing radionuclides, ammonium nitrate and nitric acid, and the other containing only radionuclides and nitric acid.

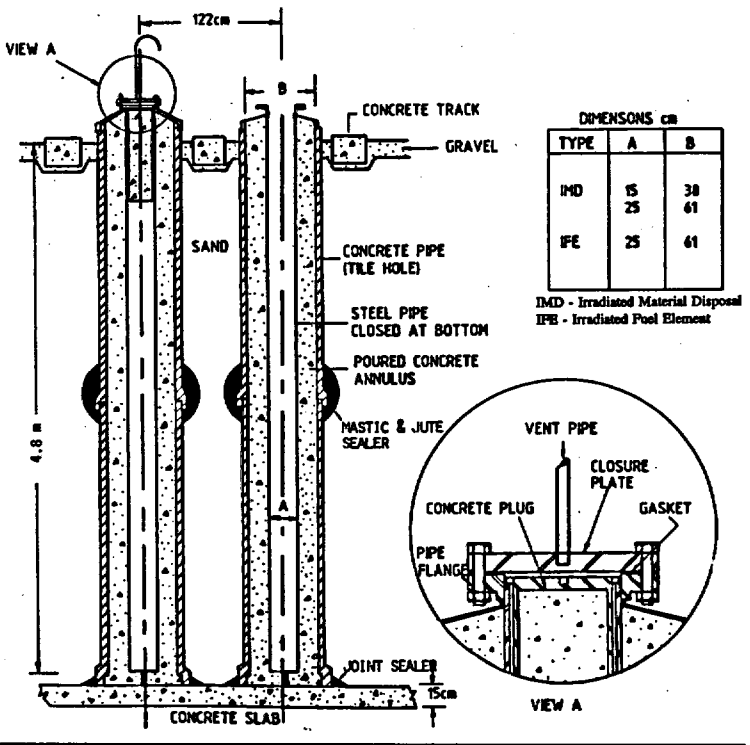
At the time, it was anticipated that reprocessing waste would be handled by evaporating off the liquid and immobilizing the remaining radionuclides in a leach-resistant glass, and processes to do this were being developed at Chalk River. The waste solutions

FIGURE 19.5
Typical Current Concrete Tile Hole in Waste Management Area 'B'



containing only radionuclides and nitric acid could be concentrated and glassified directly, but those containing ammonium nitrate could not. A process had to be developed for eliminating the ammonium nitrate before the trigly process wastes could be glassified. The development of this process led to the only work-related fatality to an AECL employee in the history of the laboratories. Ammonium nitrate will decompose in solution at the boiling point if chloride ion is added to catalyse the decomposition, but it cannot be allowed to collect in solid form anywhere in the reaction vessel without a risk of explosion. During the testing of the process on trigly wastes, an explosion occurred that killed a pipe-fitter who, at the time, was working in an unprotected area near the vessel. A plant was eventually constructed

FIGURE 19.6
Typical Current Irradiated Material and Fuel Storage Tile Holes in Waste Management Area 'B'



with appropriate blast protection in an isolated area of the laboratory property (see figure 19.1, N.P. 233), and the ammonium nitrate in the trigly process waste was successfully decomposed. The waste solutions from both processes were then concentrated. However, only a small fraction of these wastes was converted to glass. In 1957 reprocessing of NRX fuel ceased as the result of an agreement to sell all irradiated NRX and NRU fuel to the United States, and it was felt that the small amount of reprocessing waste accumulated to that time could be stored until commercial conversion facilities became available. A test set of twenty-five nepheline syenite glass blocks was made in 1958, and these were buried in a 5x5 block grid below the water table in an unused area of the Perch Lake basin about 400 metres from

the lake. When early results showed leaching rates to be too low to allow leaching patterns to be studied, a second set of twenty-five blocks was made of purposely poorer glass, and buried in a 5×5 block grid in a second unused area about 300 metres from the first site. After an initial high leaching rate, the leach rate decreased dramatically to an insignificant level. The initial higher leach rate was shown to be due to leaching from the near surface of the poorer glass. To date, no leaching has been detected from the first high-quality set of blocks. The remainder of the concentrated reprocessing waste was stored in double-lined tanks in a tank farm located in a sandy upland of the Perch Lake basin, about 2.5 kilometres from the lake (see figure 19.1), where it will remain until facilities are available to convert it to glass. Also in the tank farm are regeneration solutions from fuel-rod bay ion-exchange resins that have been concentrated tenfold and then neutralized. These will remain in the tank farm until they are incorporated into bitumen.

In 1963, for the first time, waste was buried outside the Perch Lake basin. In order to preserve the area in WM Area B for high-level solid and special wastes, low-level solid waste burial grounds were located on a sandy upland in the Maskinonge Lake basin about three kilometres from the laboratories and about one kilometre from Maskinonge Lake. This area was designated Waste Management Area C (see figure 19.1). Maskinonge Lake is another lake on the Chalk River property, somewhat larger than Perch Lake. It has an outlet to the Ottawa River via Chalk Lake, Chalk River and Chalk Bay. As in Perch Lake, the release to the Ottawa River could be monitored in a stream connecting Maskinonge and Chalk lakes. The fenced-in area covers 4.2 hectares. As in Area B, the low-level solid waste was buried in unlined sand trenches. In addition to low-level solid waste, inactive acid, solvent and organic liquid waste was placed in specific sections of the trenches, or in special pits located along one edge of the area. In 1982, to conserve space, the method of burial was changed from parallel trenches separated by an intervening wedge of sand to a continuous trench similar to municipal landfills.

Over the years, surface and sub-surface monitoring has been done on a regular basis to study the migration of radionuclides in the waste management areas and assess the hazard they represent.

It has been found that tritium travels at the rate of groundwater, that strontium-90 and cobalt-60 travel at about 3 percent of the rate of ground water, and that other radionuclides travel much more slowly. At this rate of travel, all will have decayed to insignificant levels by the time they reach Perch or Maskinonge lakes. The annual tritium, strontium-90 and cobalt-60 releases to Perch and Maskinonge lakes have never exceeded, respectively, 0.015, 0.001 and 0.001 percent of Chalk River's derived release limits for these species. The derived release limit is the calculated release of radionuclides from the plant boundary, which will ensure that the most exposed member of the general public will not receive a radiation exposure in excess of the Control Board regulatory limits.

When the Whiteshell Laboratories started up in 1961 in Manitoba, it was not blessed with sandy uplands well above the water table. Its property consisted of about 35 km² of flat grassland near the north-east boundary of the plains area of Manitoba, on the east bank of the Winnipeg River. The regional overburden is clay, deposited during the glacial lake period, overlying a compact layer of glacial till and bedrock. The water table is generally within one to two metres of the surface, so it was not possible to find an area where the waste could be buried above the water table. However, on the property, about four kilometres from both the laboratory buildings and the Winnipeg River, is a hydrogeologic discharge area; i.e., an area where the groundwater flow is to the surface. This area was chosen for the waste management area because if water flow was to the surface it could be channelled into a discharge ditch and monitored to ensure that excessive levels of radionuclides were not released to the Winnipeg River. An area of 4.6 hectares was fenced off for this purpose and no other area has subsequently been required.

The methods of handling the different categories of waste at Whiteshell were similar to those used at Chalk River, including unlined trenches for low-level wastes, concrete bunkers and concrete stand-pipes for medium-level wastes¹

¹ Whiteshell nomenclature. This is waste that at Chalk River is designated as high-level waste, except that it cannot contain significant quantities of experimental irradiated fuel.

and stainless-steel tanks for high-level waste liquids, although there were minor differences in design to reflect the different environment. To minimize weather exposure and problems related to the high water table, low-level wastes were buried in the spring and in the fall. In the interim periods they were held in a steel Butler building with a storage volume of 300 m³. The Whiteshell waste area also was equipped with an incinerator to burn organic liquids, primarily unusable fractions of the organic coolant used in the WR-1 reactor. The incinerator used a vortex burner, which allowed incineration without visible smoke or ash. Radioactive emissions were controlled by limiting the radioactivity in the feed to less than 15 Bq/mL. Organic liquids with radioactivity above that level were stored in stainless-steel drums on a concrete pad near the low-level waste building, until the radioactivity decayed to the required level. As at Chalk River, the waste handling facilities were modified with time to reflect experience. For example, concrete bunkers and stand-pipes are now surrounded by bentonite clay to keep water out and adsorb any radionuclides that might be released; concrete stand-pipes are now lined with galvanized steel to further protect against water ingress; and scrap and waste from irradiated fuel examinations are now stored in concrete canisters (to be described later) rather than concrete stand-pipes.

Surface, sub-surface and incinerator stack monitoring indicates that the only significant release of radionuclides from the waste handling facilities has been from the incinerator. That release has been less than 0.005 percent of the derived release limit for the laboratory, assuming that all the radioactivity was due to strontium-90, the most hazardous isotope of those potentially present.

All of these laboratory wastes at both Chalk River and Whiteshell are considered to be in safe storage; that is, they will present no hazard to the public as long as qualified people are present to look after them. However, safe storage cannot be equated with safe disposal. Plans for moving from storage to disposal will be discussed later in this chapter.

USED-FUEL STORAGE

From the earliest days of nuclear-reactor operations the accepted method of handling used fuel has been storage in water-filled storage bays. The water provides both cooling and shielding and its transparency permits visual examination of the fuel and the operations in the bay. The NRX storage bay consists of a long, narrow trench 37 × 1.35 × 3.75 m deep connected to a cut rod vertical storage bay, and a fuel-rod examination bay. The walls are thick concrete, painted white in some areas to improve visibility. Fuel is removed from the reactor in a lead shielded flask and is deposited in a storage block located over one end of the bay. After a short period of forced cooling in the storage block, the rod is discharged into the bay on a carriage mechanism, which orients it from vertical to horizontal and transports it to the cutting area of the trench. The inlet and outlet hardware is sawn off for reuse and a monorail transports the aluminum-clad fuel section to the storage area. Staff working with long tools from a bridge above the water surface can assist the operations and correct any malfunctions. Initially, the rods were held in the bay for a minimum six-month cooling period and were then transferred through a transfer chute to a dissolver, where they were dissolved for plutonium recovery. In the early years of operation the water in the bay could become quite radioactive, particularly if a lot of damaged fuel was being handled, and the water would have to be periodically replaced by feed and bleed. However, after side stream purification by ion exchange was introduced, this problem was eliminated. By 1957, reprocessing at Chalk River had ceased: arrangements had been made to sell the used fuel from NRX and NRU to the United States and ship it to the United States Atomic Energy Commission Savannah River facility for reprocessing. The rod bay was now used to load the shipping flasks. The flask was lowered into the bay with an overhead crane, its lid was removed, the rods were loaded, the lid was replaced, the flask was removed and was decontaminated before shipment.

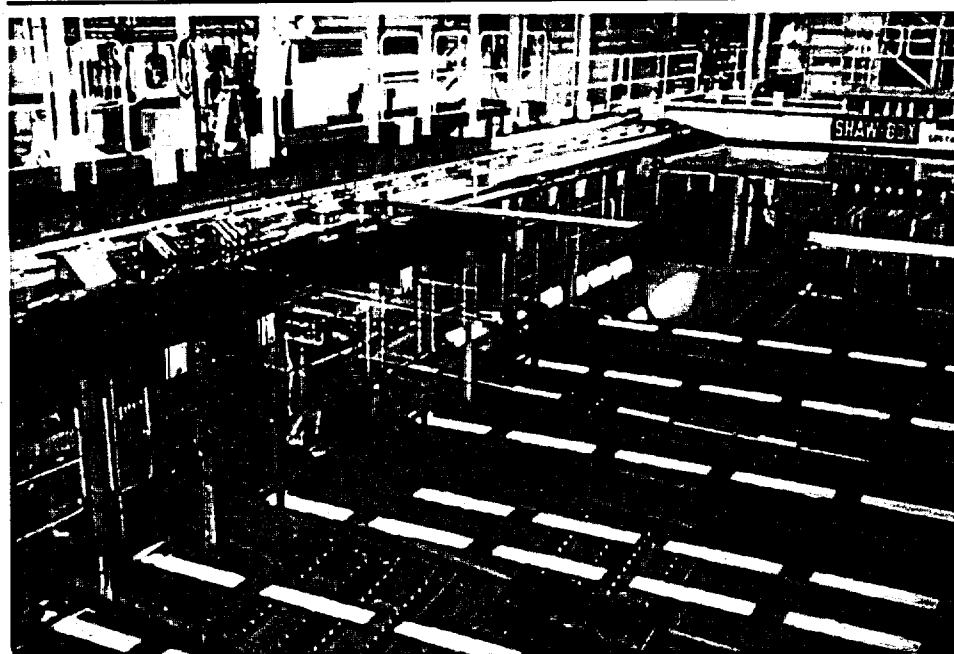
The NRU bay is similar to the NRX bay, but it is more square and deeper. Some areas are painted white and others are surfaced with white tile. The fuel is removed from the reactor

on-power and transferred to the storage block where it is force cooled by the heavy-water reactor coolant. After a suitable cooling period it is transported by flask to the bay where it is suspended in a vertical orientation. After the top is sawn off for reuse, the aluminum-clad fuel section is stored in a vertical orientation in the reactor bay. The NRU fuel was never reprocessed on site. It was sold to the United States. The fuel was loaded into the shipping flasks in the NRU bay.

When Canadian power reactors were built, they basically adopted NRX and NRU technology for used-fuel management. Typically, a CANDU 600 reactor fuel bay (see figure 19.7) has a storage area $21 \times 12 \times 8.4$ metres deep and will hold ten to fifteen years' production of used fuel.² The walls of the bays are reinforced concrete, approximately 2 metres thick, and are lined with stainless steel or fibreglass-reinforced epoxy paint. The bay water is cooled through a heat exchanger and is kept clean by continuous circulation of a side stream through ion-exchange resins. The fuel in the bay is held in either large baskets or stacks of trays. When each basket is filled or each stack is completed, it is sealed with a zirconium alloy seal, the crystal pattern of which is unique. This seal was developed by AECL scientists to allow International Atomic Energy Agency inspectors to ensure that fuel from CANDU reactors, Canadian or foreign, was not being clandestinely diverted to weapons production. The inspector keeps a record of the signatures of the seals, and since no two crystal patterns are the same, the seals cannot be broken and replaced without the inspector's knowledge.

The WR-1 reactor, built at Whiteshell in 1965, had a very small used-fuel bay, and by about 1972 the bay was filling up. Whiteshell had to either build a new bay or devise another method of storing fuel. They decided to experiment with dry storage of used fuel in reinforced concrete canisters, basically

FIGURE 19.7
Typical CANDU Power Reactor Fuel Bay



a concrete can with walls thick enough to provide the required shielding (see figure 19.8). The attraction of the canisters was that they could be built as needed and thus would avoid the large up-front expenditure that would be required for a new bay. The concern was that the thermal gradient across the concrete (hot on the inside, cold on the outside) would cause the inner concrete to expand and crack the outer concrete causing the exterior surface to weather rapidly or, in the extreme, lose shielding because of a crack through the wall. Fine exterior cracks were expected to relieve stresses, but it was the size and behaviour of the cracks that was important. The objective of the program was not only to demonstrate that used fuel could be stored in this way, but to develop and verify prediction methods that would allow future

² Storage bays in the multi-unit stations are larger, typically $32 \times 17 \times 9$ m deep, and auxiliary bays have been built at Pickering and Bruce.

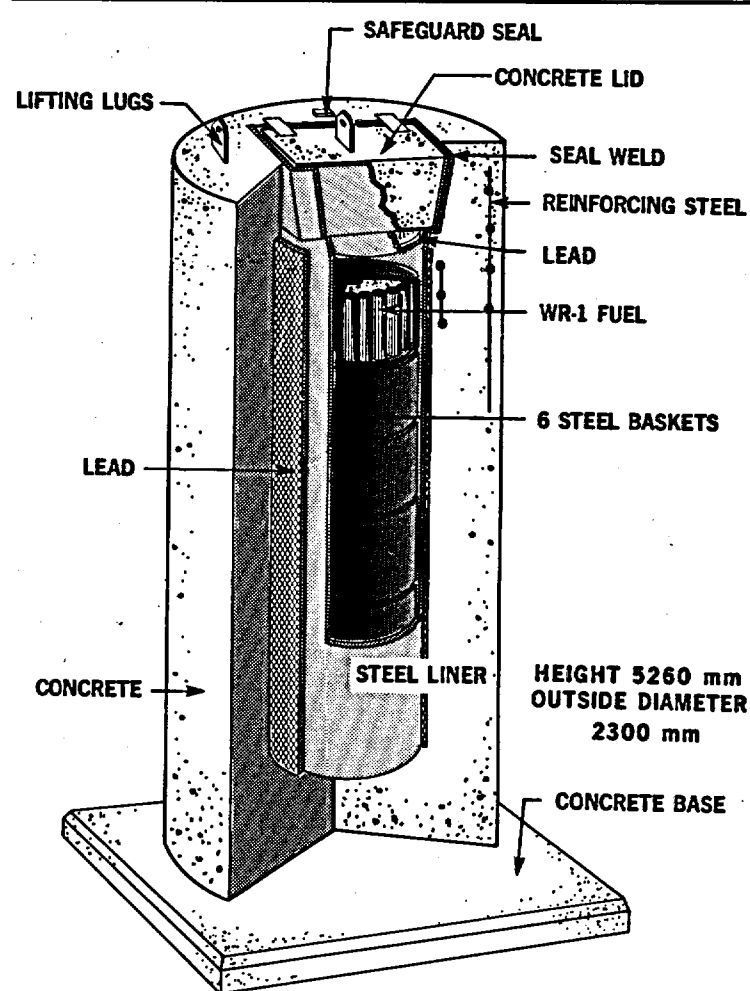
designs to be optimized. It was decided to build four experimental canisters: two cylindrical and two square, with beveled corners (square prismatic). Although it was clear that cylinders would be most efficient in terms of heat transfer and utilization of material, all power-reactor fuel-handling equipment was based on a rectangular configuration, so it was deemed prudent to test a square design as well. One canister of each design was to hold fuel and the other was to be equipped with electric heaters to test the limits of applicability of the concept. The canisters were 5.26 metres high by 2.36 metres diameter or square, with interior mild steel liners 0.8 metre diameter or square. The fuel was to be loaded in a hot cell, into mild steel baskets 0.74 metre diameter or square, and a lid was to be welded on. The baskets were to be filled with inert gas, to prevent oxidation of the fuel should a fuel pin fail. Six baskets were to be placed in each canister. The design heat load was 2.0 kilowatts, or 4.4 tonnes of five-year cooled fuel.

Tests on the electrically heated canisters showed that both designs were practical and conservative. In fact, tests at 11 kilowatts, including cycling tests to represent summer and winter extremes, showed that cracks did not penetrate deeply and would close up as the heat decayed. Accordingly, the second canisters of each type were filled with fuel: the cylindrical one with WR-1 fuel and the square, prismatic one with Douglas Point fuel. These tests confirmed the viability of the concept and it was quickly adopted for the storage of excess WR-1 fuel. Optimized designs have since been used for the storage of Gentilly-1, Douglas Point and NPD fuel when those reactors were shut down. The first experimental concrete canisters at Whiteshell were left in an area of the property immediately adjacent to the parking lot, so that visitors could see for themselves that used fuel from nuclear reactors could be safely and economically stored for long periods.

USED-FUEL DISPOSAL

In the early years of nuclear power development it was universally accepted that the world's supply of economically available uranium was limited and that used fuel would have to be

FIGURE 19.8
Cylindrical Canister #2 Containing WR-1 Fuel



reprocessed to recover the contained uranium and plutonium for reuse. In fact, in the economics of early United States power stations, a credit of \$10/gm was allowed for the plutonium contained in the used fuel. Accordingly, it was generally accepted that the waste that would have to be handled was a nitric acid solution of fission products from which the plutonium and uranium had been extracted, and that the handling method would

be to incorporate the fission products into leach-resistant glass blocks for burial in a suitable geologic formation. In the early to mid-fifties AECL scientists developed a process for incorporating the fission product waste into a glass comprising primarily nepheline syenite, a relatively abundant mineral widely used in the glass industry. The concentrated nitric acid solution containing the waste radionuclides was mixed with nepheline syenite powder in a ceramic crucible in proportions that would cause the mixture to form a gel. The crucible was then placed in a high-temperature furnace and the temperature was brought to about 1,350°C. At this temperature, the contained nitrates decomposed, leaving behind a relatively clear and highly leach-resistant block of glass. The off-gases passed through a liquid scrubber which removed entrained material and oxides of nitrogen, and an iron oxide bed which removed the volatile radionuclide ruthenium. As mentioned earlier, twenty-five such blocks were buried in sand below the water table and found to be so leach resistant that a second set of deliberately poorer quality blocks had to be made to allow leaching patterns to be studied. It may not be surprising that scientists at the time felt that the waste problem was solved in principle, and that there would be little difficulty when the time came to select a suitable site for final burial. History would show the naivete of this assumption but work on final disposal ceased and was not resurrected until the mid-seventies.

In the early seventies nuclear power stations were operating well in Canada but the social climate had changed. There was significant public opposition to nuclear power and one of the major focuses of this opposition was nuclear wastes. Critics claimed that there was no safe way of disposing of the wastes from the used fuel; the early glass block experiments were deemed insufficient and irrelevant because there had been essentially no accompanying geologic work and there were now no plans to reprocess the used fuel. Accordingly, in 1974, AECL, with strong government support, initiated a major program to prove "beyond reasonable doubt" that wastes from the used fuel could be managed safely. It would be necessary to shield the fuel for several hundred years, while the bulk of the gamma-emitting radionuclides decayed, and to isolate the fuel for a

much longer period, to ensure that long-lived, toxic, beta- and alpha-emitting radionuclides did not escape in significant concentrations to drinking water. The challenge was to prove that long-term isolation was possible. The program was centred at Whiteshell and was to occupy a major portion of that laboratory's efforts for the next fifteen years.

Two objectives were established at the outset: one a safety objective and the other a responsibility objective. The first was that "the wastes should be managed in such a way that the potential hazards are minimized to the greatest extent practical."³ The second was that "they should be managed in such a way that the trouble and concern to future generations in keeping them safely will be minimized."⁴ A number of potential methods of meeting these objectives were reviewed, including transmutation to short-lived radionuclides, extra-terrestrial disposal, ice-sheet disposal, sub-seabed disposal and underground disposal. It was decided that underground disposal was the best option for Canada; it was deemed the method whose safety could most readily be proven "beyond reasonable doubt." AECL enlisted the help of the Geological Survey of Canada (GSC) to determine the type of geologic formation best suited for underground disposal in Canada. Around the world, a number of geologic media were being considered for disposal of nuclear waste, including clay, shale, tuff (volcanic ash), basalt (volcanic rock), salt, and hard rock. However, the scientists at GSC felt that a particular type of hard rock formation, a pluton, would be the most suitable formation for nuclear waste disposal in Canada. Plutons occur in abundance in the Canadian Shield, where they have been stable for hundreds of millions of years and are likely to remain stable for millions more. Their stability, uniformity, resistance to erosion, and abundance in areas of little commercial value and low population led to their selection by the GSC. The GSC recommendation was accepted by AECL and endorsed by an independent group of experts that had been

³ The objective was to limit the risk to the most exposed individual to a small fraction of that from natural background.

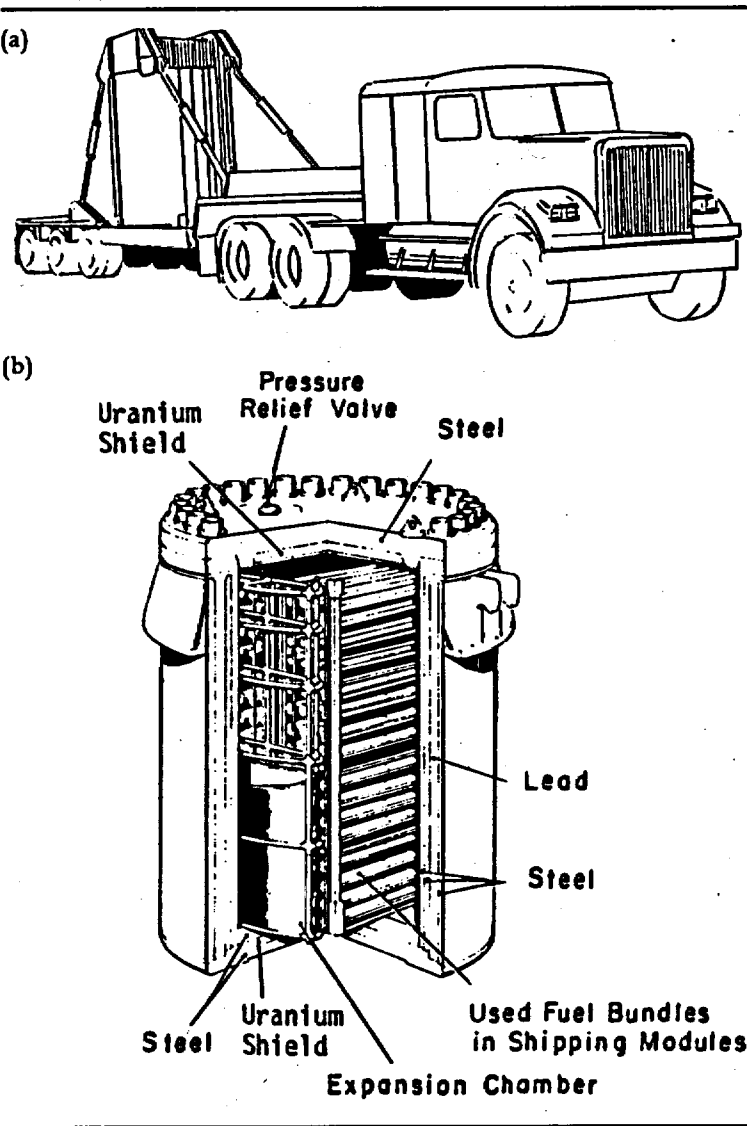
⁴ The objective was to preclude the need for any long-term care or monitoring.

commissioned by the Department of Energy Mines and Resources and headed by Dr. F.K. Hare, then Head of the Department of Environmental Studies at the University of Toronto.

AECL, with the assistance of Acres Consulting Services and Associates, next developed two reference systems for disposal of fuel wastes from CANDU power reactors: one assuming that the fuel would not be reprocessed, the other assuming that it would.

In the reference system for unprocessed used fuel, the fuel bundles would be allowed to decay and cool for a minimum of ten years after removal from the reactor before transportation to the disposal facility. Transportation would be in casks designed to protect against radiation and the release of radionuclides, even if an accident resulted in severe stresses due to impact, fire or immersion in water (see figure 19.9). The casks could be transported by truck, train or barge. At the disposal facility, the bundles would be loaded on end into 1.15-metre-high x 0.91-metre-diameter titanium disposal containers in two layers of thirty-six bundles each. After welding on a lid with fill and vent holes, the free space in the containers would be filled with a molten lead-antimony alloy, which on solidification would immobilize the bundles and support the container walls. The vertical orientation of the bundles would facilitate complete filling of the free space. The fill and vent holes would then be welded shut and the filled container would be transported to the disposal vault. The vault would be built on a single level 500 to 1,000 metres below the surface. Exhaust shafts would be sunk at one end and a waste-lowering shaft, intake ventilation shaft and service shaft would be sunk at the other. Underground passageways, known as drifts, would be excavated to connect these shafts and to permit removal of excavated rock and the transfer of the waste and backfill materials. Further excavating and the emplacement of wastes would be carried out on a modular basis, starting at the end near the exhaust shafts and working toward the waste-bearing shaft (see figure 19.10). Excavation and waste emplacement would use different drifts, to separate construction and waste traffic. The finished vault would contain sixteen panels, each containing fifty-two rooms, each of which

FIGURE 19.9
Used Fuel Transportation System: (a) Transport Truck with Cask, (b) Currently Used Fuel Shipping Cask



would hold 300 disposal containers in seventy-five rows of four. In each room the disposal containers would be placed on a one-metre layer of buffer material, and as the room filled,

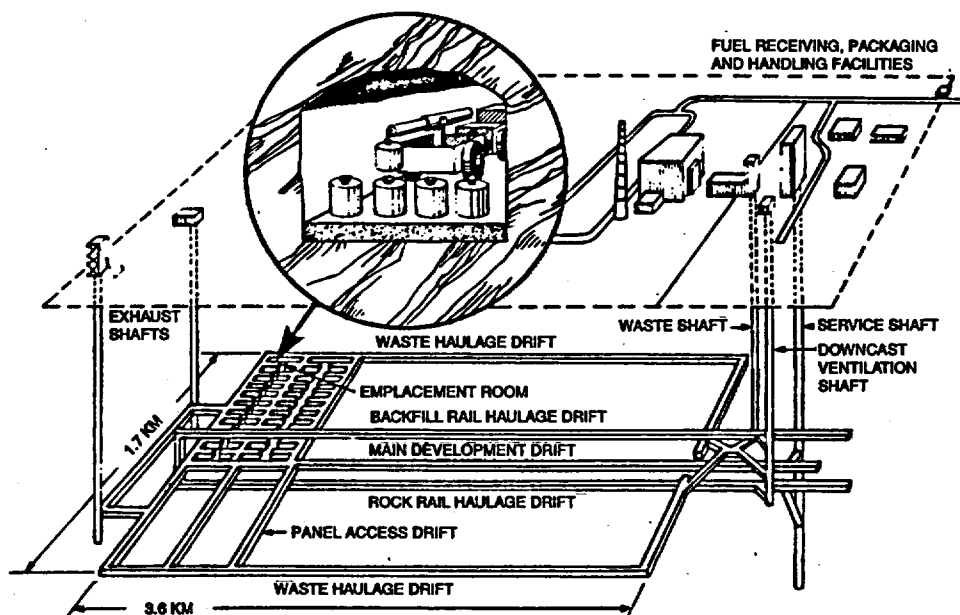
buffer material would be packed around and above the containers to a width and depth of at least one metre. The buffer material would consist of a sand-clay mixture designed to retard the movement of groundwater and radionuclides, and possibly to modify the chemical conditions in the vault to retard corrosion. Backfill material would fill the remainder of the rooms, the drifts and the shafts. The major component of the backfill would be crushed rock or sand, or a mixture of the two, with some clay added to achieve the desired low permeability. The reference unprocessed fuel disposal vault would be 3,600 metres long and 1,700 metres wide, and it would hold 18 million fuel bundles. This was the then-projected total fuel arising

in Canada up to 2055. After the vault was sealed, the surface facilities could be decommissioned and the site could be returned to close to its original state or used for another purpose.

In the reference system for the reprocessed fuel, the wastes would be incorporated into glass and poured into titanium containers 3.275 metres high \times 0.457 metres diameter at the reprocessing facility. The containers would be sealed and then shipped to the disposal facility in transfer casks of the quality mentioned earlier. At the disposal facility, the containers would be transferred to a vault of similar design to the unprocessed fuel vault, but the reprocessed waste containers would be placed in boreholes in the floor of the disposal rooms, rather than placed

on the floor. Each room would have 480 boreholes, arranged in 120 rows of four, and one waste container would be placed in each borehole. The container in the borehole would be surrounded with buffer material to a thickness of at least 0.1 metre and then the room would be backfilled. The reference reprocessing waste vault would be 3,600 metres long and 900 metres wide, with fifteen panels, each having twenty-five rooms. This single vault could contain the waste from about 13 million fuel bundles, the projected total fuel arisings in Canada until 2050.⁵

FIGURE 19.10
Underground Layout of the Conceptual Disposal Facility



During vault construction, about 9 000 000 m³ of rock would be excavated, half of which would be returned to the vault as part of the backfill. An area at the surface would be required for storage of rock and other materials. An additional area would be designated as an exclusion zone, so the total surface area required for the disposal centre would be about 27 km². The area of the vault itself would be about 7 km².

⁵ An amusing incident occurred when the author presented the AECL program, at about this stage of development, to the Ontario Government's Royal Commission on Electric Power Planning (The Porter Commission). To ensure that the proceedings were seen to be scrupulously fair, Dr. Porter tended to let questioning of witnesses take its course without much restriction. Late one night, after what seemed like interminable and repetitive questioning from anti-nuclear intervenors, the author was asked: "How long do you think civilization will last?" Rather taken aback, the author paused for a moment and in that pause, the Commission Counsel was heard to mutter "Not as long as this commission."

Developing the reference system was relatively easy; the major challenge facing the scientists was to prove beyond reasonable doubt that the "radioactive material would be isolated such that the potential hazards to man and his environment would be negligible for all time." How could one prove in a few years that the system would present negligible risks for hundreds of thousands of years? To answer this challenge, the scientists devised a method they called a Systems Variability Analysis: it provides a probabilistic estimate of the radiation dose to people caused by radionuclide migration from the disposal vault. The exposure mechanism of concern is that in which groundwater penetrates the vault, corrodes the containers, dissolves the radionuclides in the waste and carries them to the surface environment, where they could disperse or concentrate and deliver via the food chain a dose of ionizing radiation to flora, fauna and humans. That radiation dose is dependent on a number of phenomena, including the rate of movement of groundwater, the rate of corrosion of the containers, the rate of leaching of radionuclides from the waste, the rate of decay of the radionuclides, the rates of sorption and desorption in the buffer material, the backfill and the surrounding rock mass, the rate of dispersion when the radionuclides reach the surface, and the rate of concentration in plants and animals consumed by people. Values of the parameters required to quantitatively describe these phenomena could be obtained from laboratory and field research, theoretical considerations, historical data, the opinions of experts, and simulation studies, and this information could then be fed into a mathematical model that represented the relevant physical processes and calculated the maximum radiation dose to people.

There is nothing unique about such an approach. Computer modeling to handle complex phenomena such as this is today the norm. What was unique about the Systems Variability Analysis was its method of handling the inevitable uncertainty in the values for the system parameters. To account for the natural variability within a pluton and probable errors in measurement, the values of the system parameters were expressed as probability distributions. To obtain parameter values from the probability distributions, the code randomly selected a single

value from each distribution for each computer simulation. Many simulations were performed, to ensure a statistically significant use of the parameter distributions, and the results were expressed as a radiation dose versus probability plot. From such a plot one could determine, for example, the probability of the radiation dose to people ever exceeding 1 percent of natural background. From the analysis one could also determine the radionuclides causing the exposure and their rate of migration through each barrier, allowing modification of the system to possibly retard that particular radionuclide if it was considered a problem. To simplify the handling of the data and to optimize the value of the results, the system was divided into three physical regions: the vault, the geosphere and the biosphere. Sub-models were developed for each region, and then linked by an executive program called SYVAC (Systems Variability Analysis Code). The approach has proved very effective and is now widely used in nuclear waste management programs around the world.

The original concept was to have GSC identify 1,000 plutons in the Ontario portion of the Canadian Shield, select from maps and existing information the 100 best prospects from those and then select by surface investigation the ten best prospects for sub-surface study. The parameters from the best two or three of those sites would be used in the safety analysis. To do these sub-surface investigations, GSC required some rather complex equipment, some of which they had to develop, and they wanted a pluton near their laboratories in Ottawa in which to test it. The site they selected was the Mount Maria pluton near Madoc, Ontario. At a public meeting to explain the program, held in Madoc on 16 March 1977, AECL representatives faced a hostile, placard-waving audience of about 1,100, which included many of the anti-nuclear leaders from across the country. The AECL representatives presented their case and answered questions, and explained the 1,000, 100, 10 scenario, but if they convinced anyone that Mount Maria hadn't been already chosen as the Canadian nuclear waste vault site it wasn't apparent. The local MP and MPPs convinced the Minister of Energy, Mines and Resources to ban AECL and GSC from working in the area. Ironically, a group of local citizens, including

some councillors, subsequently approached AECL and the minister to try to get the ban lifted, because they were interested in the jobs a disposal vault would create, and they felt they could turn public opinion around. However, too much damage had been done and the minister remained adamant.

The Madoc incident prompted a complete rethinking of the approach to site selection and the field research program. It was decided that it was both impractical and unfair to ask communities to consider hosting a waste disposal facility before they could be assured beyond reasonable doubt that it was safe. Accordingly, AECL decided that they would do the work necessary to generically prove beyond reasonable doubt the safety of the concept, and that they would submit that proof to the Control Board for review. Only after the Control Board agreed that the safety of the concept was proven beyond reasonable doubt would they ask communities to consider hosting a disposal site. It was recognized that this generic approach would increase the variability of the system parameters, because now many sites would have to be considered, but it was felt that the probabilistic treatment of the parameters could handle it. It was still necessary to gain access to appropriate plutons for field research, but it was to be made clear to neighbouring communities that these sites were for research purposes only, and that selection of a site for the disposal vault would not proceed until the concept had been accepted as safe by the Control Board. Further, it was deemed essential to ensure that local governments, regional politicians and the local public were fully aware of the research work that was to be done in the area, and were reasonably supportive before any work was done. At this time, AECL also decided that its work must have independent and formal peer review, beyond what it would get through publication, if its work on such a sensitive subject was to have public credibility. It therefore established a technical advisory committee, made up of independent experts in the appropriate key disciplines, recommended by the appropriate Canadian scientific societies. This group was to review the program and the work as it progressed, and publish an annual report on their findings. The committee is still active and their advice and criticism have been invaluable.

Atikokan, Ontario was the first site where the revised approach was used, and it met with enthusiastic response. The local government and population seemed to feel that if they supported the research work, they would have an advantage over other communities at the time of site selection, with resulting major economic benefits. The Atikokan site became one of AECL's main field research sites and contributed a wealth of information to the program.

The second site where the revised approach was used was Massey, Ontario, and the response was less than enthusiastic. The Atikokan pluton is made of granite and is representative of about 80 percent of the plutons in the Canadian Shield. The other 20 percent are made of gabbro, and it was felt that gabbroic plutons should be investigated to determine whether they had better properties for waste disposal than granite. A gabbroic pluton exists near Massey. Activists were able to convince many Massey citizens that AECL had given up on Atikokan because it was unsuitable and had selected the pluton near Massey as the Canadian waste vault site. However, with the help of strong support from regional politicians, the necessary work was done. The work showed that there was no significant difference between granite and gabbro for waste disposal; after about three years the Massey field research area was closed.

The third site of extensive field research was Whiteshell itself. The site is located on a granite pluton, the Lac du Bonnet batholith, which is about seventy kilometres long and twenty kilometres wide. Access to most areas of that pluton was obtained without major difficulty. An area about fifteen kilometres from the site and ten kilometres from Lac du Bonnet was selected for extensive study, and eventually became the site of an Underground Research Laboratory (to be discussed later). Although the Chalk River site was not on plutonic rock, some work was done there as well.

With the acquisition of research areas it was possible to proceed with the field research program that had been envisaged. The objectives of the program were twofold. The first was to develop non-intrusive methods of characterizing plutons; that is, to be able to predict from existing information, aerial surveys

and surface observations the likely structure, fracture patterns, permeabilities and water flows beneath the surface. This was to aid in site selection. All such information would have to be verified by sub-surface investigation (drilling), but it was important from both a cost and a public relations viewpoint that approved sites not have to be subsequently abandoned for technical reasons. The second objective was to obtain accurate information on parameters such as water chemistry, fracture patterns and porosity, rock composition, water flow and rock uniformity, which were required in the analysis.

The non-intrusive investigation started with airborne measurements of magnetic and density variations to define the boundaries of the pluton. These were followed by careful examination of the surface within those boundaries, to identify any features that were likely to be the surface expression of faults and fracture zones in the underlying rock. Then at selected locations seismic and electromagnetic waves were transmitted from the surface into the underlying rock to determine the potential presence and orientation of fracture zones. The waves reflect differently from fractures than from a coherent rock mass. Core samples were then bored out of the rock mass at selected locations to confirm the picture of the underground structure predicted from the non-intrusive investigations. It was found that the non-intrusive methods gave a three-dimensional picture of the underground structure sufficiently accurate to screen potential sites, but not accurate enough for site selection without borehole confirmation.

The Systems Variability Analysis required methods for calculating the rate of failure of the containers, the rate of release of radionuclides from the waste forms, the rate of movement of radionuclides through the buffer, the rate of transport of radionuclides through the rock mass of the pluton (the geosphere), the buildup of radionuclide concentrations in water, food and air, and the consequent uptake of these radionuclides by people. It was the task of the researchers to provide the information required for these calculations. It was found that the variability of the values of all the parameters required could be adequately described using the six probability functions shown in figure 19.11.

To describe the rate of failure of the waste containers, a container failure function was used that gave the fraction of the containers failing as a function of time. The container failure function was based on the rates of uniform corrosion of the thin-walled titanium container, and had to take into account the space and time variations of temperature in the waste vault. It was found that the failure rate could be expressed as a truncated normal-probability distribution with a start time of about 16,500 years and a mean value of about 17,000 years. The corrosion rate as a function of temperature was determined in the laboratory using water of chemical compositions covering the range found in field research boreholes. The water from deep in the boreholes was reducing and strongly saline. This removed from consideration some of the lower-cost options for container materials, such as stainless steel and nickel alloys, which corroded relatively rapidly in such water.

To describe the rate of release of radionuclides from the unreprocessed used fuel, it was necessary to use a two-stage release model, to account for an instantaneous release and a long-term release. In the analysis it was assumed that container and fuel-sheath failure was simultaneous, and experiment showed that there was an "instant" release of iodine, cesium and selenium when the fuel sheath failed. This resulted from fractions of these elements that evaporated from the hot centre of the fuel pellet during operation and condensed in cracks in the fuel, gaps between the fuel pellets, and gaps between the fuel pellets and the sheath. The "instant release fraction" was defined by a lognormal probability distribution with a range of from 1.2 to 10 percent and a mean of 3.6 percent, and was based on results from a study on dissolution of CANDU fuel. Long-term radionuclide release was based on the assumption of congruent dissolution; that is, that the radionuclides are released as the uranium dioxide containing them is dissolved. It was found that the rate of dissolution of uranium dioxide was controlled by its solubility, which could be described by a lognormal probability distribution with a wide range and a low mean value. The wide range reflected the wide range of water chemistry that could possibly contact the fuel, and the low mean reflected the fact that the most

probable water chemistry was the reducing saline composition found at depth.

To describe the rate of release of radionuclides from the reprocessing waste, researchers had to first specify the composition of the glass that would be used to immobilize the bulk of the waste and the other waste forms that would be produced in the reprocessing and glassification processes. Experiments were

done to optimize glass stability and leach rate against ease of manufacture (low melting point), and it was decided that the optimum was achieved with a sodium calcium aluminosilicate glass. In the glass manufacturing process, radioactive iodine and carbon-14 are volatilized, and they have to be collected and immobilized and put in the waste vault as well. It was determined that bismuth oxyiodide and barium iodate could be used for iodine, and barium carbonate for carbon.

Finally, in reprocessing, zirconium fuel sheaths, containing radioactive activation products, are left behind, and these have to be compacted and disposed of as well. The release of radionuclides from the sodium calcium aluminosilicate glass was described by a model conceptually similar to that for used fuel, with an "instant release" followed by a congruent release determined by the glass solubility. The instant release fraction was represented by a uniform distribution and the solubility was represented by a lognormal distribution with a low mean value. The release rates from the other waste forms – bismuth oxyiodide, barium iodate, barium carbonate and zirconium – were all controlled by their solubilities, which were represented by lognormal distributions.

Possible ways for the radionuclides released from either waste form to move through the buffer are convection and diffusion. Convection refers to the transport through the buffer by groundwater movement. Diffusion refers to the movement through ground water to regions of lower concentration, and is represented by the molecular diffusion coefficient for ions in water. Although both mechanisms were included in the model for transport through the buffer, experiment and analysis showed that migration was dominated by diffusion. This was because groundwater flow

FIGURE 19.11
Probability Density Functions Recognized by SYVAC

CLASS OF PDF	SAMPLE PLOT	REQUIRED DISTRIBUTION PARAMETERS
Constant		Constant value of x.
Uniform		Lower limit of x. Upper limit of x.
Piecewise uniform		Number of ranges. for each range { Lower limit of x. Upper limit of x. Fraction of values of x that lie within the range.
Log uniform		Lower limit of y, where y = log10x. Upper limit of y, where y = log10x.
Normal		Mean, μ, of values of x. Standard deviation, σ, of values of x.
Lognormal		Mean, μ, of values of y, where y = log10x. Standard deviation, σ, of values of y, where y = log10x.

through the buffer was extremely slow. The molecular diffusion coefficient increased with temperature, and the temperature in the buffer would vary with time and with position over a range of 150-15°C. Values of the molecular diffusion coefficient were calculated over this range of temperature, and their probability was weighted by estimating the durations of temperature regimes after container failure. This resulted in their representation by a piecewise uniform distribution. Calculations showed that buffer layers as thin as only 25 centimetres would delay movement of dissolved or suspended radionuclides for thousands of years. The model also included a method for calculating retardation by sorption, which required the determination of partition coefficients for each radionuclide and their expression as appropriate lognormal distributions.

To describe the rate at which radionuclides leaving the buffer would reach the surface, it was necessary to determine the rate of movement of groundwater through the intervening rock (convection transport), the distance it would have to travel to reach the surface, the fraction that would be diverted from the main flow path by cross flow and diffusion (hydrodynamic dispersion), the fraction that would disappear by radioactive decay, the new radionuclides that would be produced as daughter products in the decay process, and the retardation that would occur by sorption and desorption on the walls of fractures in the rock. It was found that these phenomena could all be quantified using experimental observations made in the field research areas or the laboratories, and that each of the experimentally derived parameters required could be adequately described by one of the six probability distributions shown in figure 19.11.

The flux of radionuclides leaving the geosphere was used in a biosphere sub-model, to calculate the maximum annual radiation dose to an individual in the reference group. The reference group comprised the people who would live their entire lives in the area where radionuclides from the vault would reach the surface. It was assumed that a member of this group would eat locally grown food and that the source of household water would be either a well (well scenario) or a local lake (lake scenario). The biosphere sub-model consisted of transport and

dose components. The transport component was used to calculate time-dependent concentrations of radionuclides in air, soil, lake and well compartments. The dose component was used to calculate the radiation dose from ingestion and inhalation of radionuclides and from external exposure. Variables affecting the radionuclide concentrations in the various compartments included size of the soil discharge area, soil depth, degree of soil saturation, net precipitation, soil type, soil density, soil infiltration function, soil distribution coefficient, lake area, lake water depth, ratio of lake drainage basin to lake depth, sediment depth, sediment density, sediment distribution coefficient, atmospheric dust load and deposition velocity from air. Variables affecting the dose calculations included the per capita water demand by both people and animals, peoples' total energy intake rate, the radionuclide holdup time in food products or drinking water, the energy provided by each food type in the diet, and transfer coefficients from soil to plants, from plants to animals and people, from animal products to people, from freshwater to aquatic food and from aquatic food to people. Values for all of these parameters were described by appropriate distribution functions based on experiment or available information.

The results of a SYVAC assessment of the safety of the reference disposal system were published in 1985. It was an interim assessment based on more than 1,000 scenarios each for used fuel and reprocessing wastes. These results were fed back into the system to improve both the analysis and the design, but the interim results gave confidence that the wastes could be safely handled. No consequences were predicted for tens of thousands of years and there was a 90 percent probability that no radiation dose would exceed 1 percent of the annual dose from natural background for ten million years. For used fuel, iodine-129 was the only radionuclide that produced radiation exposures even approaching 1 percent of natural background, and it was only the iodine-129 from the instant release fraction that produced the exposures. For fuel reprocessing wastes with iodine incorporated in bismuth oxyiodide, technetium-99 produced the highest doses, but when iodine was incorporated in barium iodate both iodine-129 and technetium-99 were contributors. No other radionuclides contributed as much as 0.1 percent of the

dose from natural background. The main barriers limiting the dose were the waste forms, uranium dioxide, sodium calcium aluminosilicate glass and bismuth oxyiodide. In the biosphere the only significant contributor to dose was ingestion; the other sources were insignificant. For used fuel the only significant contributor was the household water from the well. For reprocessing waste, ingestion of technetium-99 in terrestrial plants also contributed.

The results of the analysis are consistent with analogues available in nature. For example, a few billion years ago uranium in the earth's crust was sufficiently enriched in the fissile isotope uranium-235 that natural fission reactors existed where conditions were optimum. One such location was Oklo, in Gabon, Africa, and scientists have studied that location to determine the fate of the wastes from those naturally occurring reactors. Their studies show that the plutonium and some of the most hazardous fission products had hardly moved before they decayed to insignificant amounts, even though the location is far from optimum for a disposal site. Closer to home, the stability of uranium oxide in groundwater is being studied at the Cigar Lake uranium deposit in northern Saskatchewan. The ore body is 2,000 metres across and 20 metres thick, and contains 150,000 tonnes of high-grade ore with an average uranium oxide concentration of 12 percent and peak concentrations as high as 60 percent. The uranium/oxygen ratio is close to that of the uranium dioxide in used fuel and the groundwater chemistry is reducing, similar to that found deep in plutons. Over the last 1.3 billion years the deposit has been in contact with groundwater and it is still intact. In fact, field measurements show that the uranium concentration in water only five metres away from the ore bodies is below the level specified for drinking water. These natural analogues give added confidence to the very low radionuclide release levels predicted by the SYVAC analyses.

It was estimated that the capital cost of a waste repository of the reference design would be \$470 million (1979): \$237 million for the surface facilities and \$233 million for the underground facilities. The facility was projected to operate for thirty-five years to clear the backlog and the fuel arisings to 2050, and have an annual operating cost, including transportation, of \$140-400

million (1979). The biggest variable was the cost of transportation, which was highly dependent on the location of the repository. Using these numbers, the cost of disposal to the electricity customer would be 0.01 to 0.03 cents/kWh (1979). The facility would employ a permanent labour force of about 480 and a peak force of about 800 during construction.

LOW- AND INTERMEDIATE-LEVEL WASTE DISPOSAL

Concurrent with its decision to initiate an extensive program on final disposal of used fuel and reprocessing waste, AECL decided to initiate a program on final disposal of low- and intermediate-level wastes. Until that time, all such wastes at AECL laboratories had been managed carefully, as discussed earlier, but they had not been, and were not being, managed in a way that would leave no burden of responsibility to future generations. Similarly, at power reactor sites across the country, such wastes were being compacted and stored in engineered facilities, but they were not being disposed of. The program AECL now initiated was to develop methods for volume reduction and final disposal that could be used not only for the wastes produced at or collected by its sites, but for similar wastes collecting at power stations. This program was centred at Chalk River.

Work initially concentrated on volume reduction of low-level wastes and subsequent immobilization of the concentrate for disposal, and this work focused on a new Waste Treatment Centre started at Chalk River in 1977. In this facility, combustible solid waste, which had previously been buried in sand trenches, was burned in a two-chamber incinerator to minimize radioactivity in the off-gases. In the first chamber, waste was pyrolyzed in a reduced supply of oxygen to minimize turbulence and the entrainment of radionuclides. The off-gases, mainly hydrogen and methane, were then burned in the second chamber, the afterburner, to produce carbon dioxide and water. Similar non-combustible solid waste was compacted in a fifty-tonne press to give a volume reduction of a factor of 7. The dilute liquid waste, which was previously dispersed in the reactor and chemical pits, was concentrated by ultrafiltration/reverse

osmosis to 2-5 percent solids, and was then concentrated further in an evaporator. The ash from the combustion and the liquid concentrate were then combined with bitumen in the evaporator, to form a leach-resistant bitumen product for disposal. Experiments showed that volume reductions of about 20 and 2,500 could be achieved in the incinerator and the reverse osmosis units, respectively.

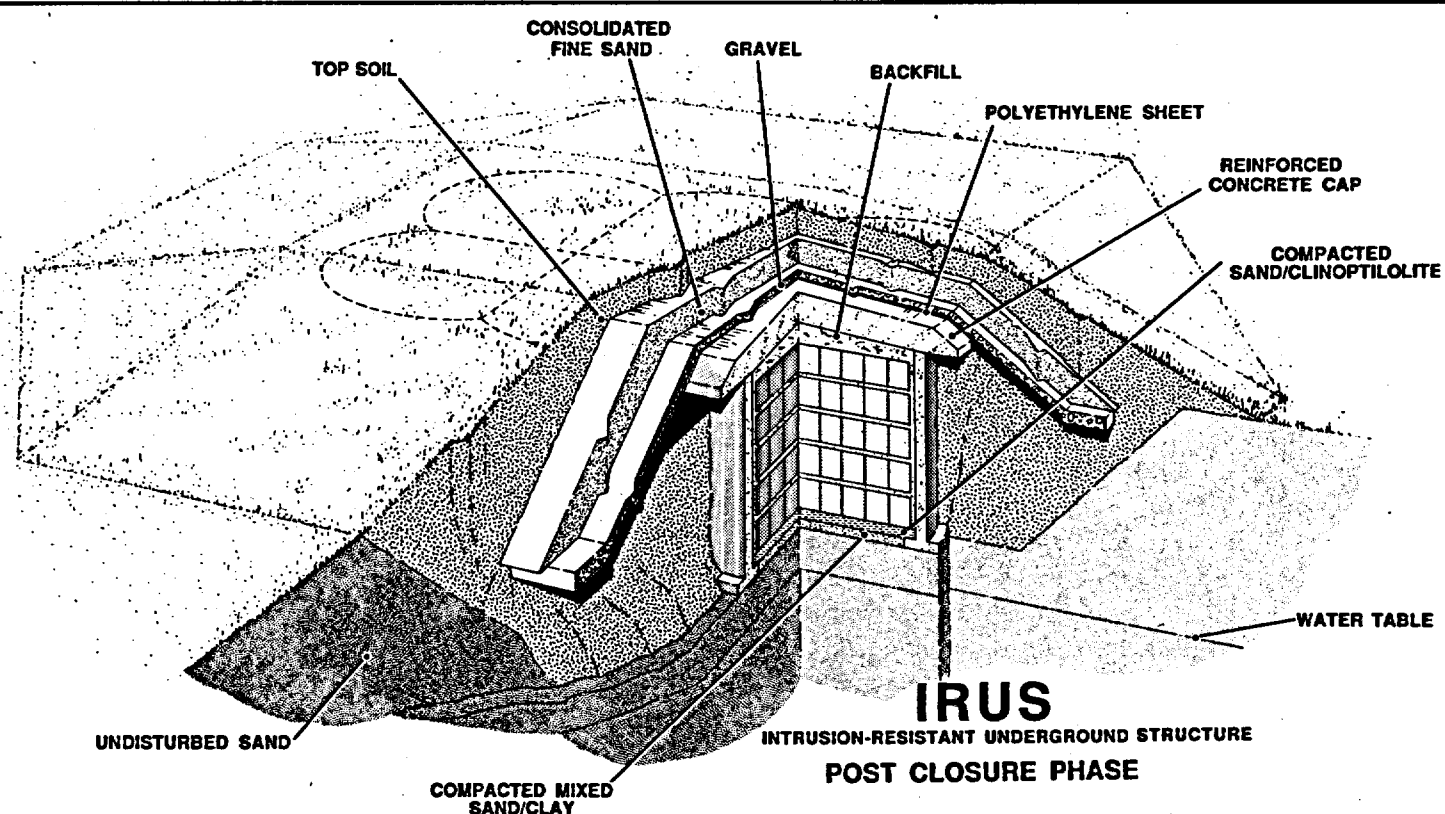
Once the Waste Treatment Centre was operational, attention turned to disposal. The plan was to develop methods for permanent disposal of the wastes being produced currently, and then to determine which of the wastes already in the waste management areas had to be recovered for further treatment, and which could be considered to be in permanent disposal in their existing state. The first thing required was a reclassification of the wastes. For waste management, the waste had been categorized primarily for handling, and primarily on the basis of external radiation fields. For waste disposal, it was necessary to categorize the wastes on the basis of the length of time the material would remain hazardous. In the new classifications, category 1 wastes would have radionuclide concentrations that would decay to innocuous levels within about 150 years, category 2 wastes would have radionuclide concentrations that would decay to innocuous levels within about 500 years, and category 3 wastes would have radionuclide concentrations that would remain hazardous for very long periods. It was decided that it was reasonable to consider disposing of category 1 and category 2 wastes at Chalk River, but that category 3 wastes would probably have to be sent to the fuel waste repository when one was built in Canada.

Three methods were considered for disposal at Chalk River: (1) an intrusion-resistant underground structure, (2) an improved sand trench, and (3) a shallow rock cavity. The intrusion-resistant underground structure was essentially a concrete box with an open bottom installed above the water table. In the open bottom was a layer of compacted sand/clay mixture that would pass water but adsorb many of the radionuclides. Thus if water gained access to the vault it would drain out but not carry radionuclides with it. The waste would be stacked in the structure and after it was full it would be capped with backfill,

concrete, a polythene sheet, gravel, sand and topsoil. The cap and walls would be thick enough to discourage intrusion. The improved sand trench would be similar, except that the walls and cap would be concrete rip-rap or large boulders. It would be cheaper, but not as resistant to water ingress. The shallow rock cavity would be a vault excavated in the rock below the weathered zone (about 50 metres) but above about 200 metres. The vault would be below the water table, but the path for water would be tortuous enough that the water from it would not reach the surface for many centuries. It would be very intrusion resistant but much more expensive than the other options. Since it was considered that the improved sand trench would only provide adequate containment for category 1 wastes, and that the intrusion-resistant underground structure gave adequate security at a lower cost than the shallow rock cavity, it was decided to proceed with a prototype intrusion-resistant underground structure (see figure 19.12). The design was examined in more detail and a three-module arched-wall concept was found to be most cost-effective. The design was completed and a safety assessment was done in a similar manner to the SYVAC assessment for used fuel waste. It was submitted to the Control Board and at the time of writing was waiting for the Board's approval.

One of the major aspects of the research program associated with the disposal of laboratory and similar wastes is the development of methods of sorting them into categories 1, 2 and 3. To attempt to develop such methods, the wastes arriving at the Waste Treatment Centre are being analyzed to determine whether pre-sorting is necessary, and to develop economical methods of analysis to sort the wastes at the treatment centre, if that is necessary. Once such methods are available, they can be used to determine the concentrations of long-lived hazardous radionuclides in the various wastes already in the waste management areas. This information can then be used in a SYVAC type of analysis to determine what further treatment, if any, is required to accomplish disposal. Other elements of the research program are the characterization of possible components of the engineered containment system, such as immobilizing matrix, packaging and backfill materials, improving the knowledge of

FIGURE 19.12
IRUS (Intrusion Resistant Underground Structure) – Post Closure Phase



the characteristics of the radionuclide pathways from the disposal facility to people and establishing baseline data for the environs of the disposal area for comparison with data obtained after disposal has taken place.

In 1975, it was discovered that in the forties radioactively contaminated soil from the Eldorado Mining and Refining uranium refinery in Port Hope, Ontario, which was destined for the company's disposal sites, had been diverted for fill in the Port Hope area, and that significantly contaminated soil was widespread in that area. The Control Board, which in 1975 had responsibility for cleanup, asked AECL whether this soil could be disposed of on the Chalk River property, and AECL agreed.

AECL subsequently also agreed to accept contaminated soil from niobium smelting operations in Ottawa, and radium dial-painting operations in Toronto. The site selected for the waste was a 1.5 hectare site located in a small valley between a sand ridge and a hard rock outcrop near the divide between the Maskinonge and Chalk Lake basins. It was designated WM Area F (see figure 19.1). The site was lumbered and cleared of slash, and bulldozers were used to spread and compact the contaminated soil in layers about 0.7 metres deep, to a total depth of about 12 metres. When the waste reached a level close to that of the adjacent sand ridge, it was covered with almost 30 centimetres of packed clayey silt to reduce penetration of water,

70 centimetres of sand to prevent erosion of the clayey silt, and 15 centimetres of sandy topsoil. It was then ditched to direct water into the sandy ridge, fertilized and seeded with a mixture of birds-foot trefoil and rye. In 1979, the area was becoming full and no end to the soil transport appeared to be in sight, so AECL informed the Control Board that it could accept no more and the transfer stopped. Investigations in 1980 and 1981 showed that the clay/silt barrier had become relatively ineffective as a rainwater shed because of drying and cracking. Further investigation showed that, while the radium had not moved, a significant quantity of arsenic, which was abundant in the waste, had moved. Fortunately, it was found that it had been collected by iron oxyhydroxide coatings on the underlying sand, and had not contaminated the local groundwater.

In total, over 118,700 tonnes of contaminated soil containing about 14 grams of radium were transferred to Chalk River; that total comprises 95,050 tonnes containing about 3 grams of radium from Port Hope, 22,180 tonnes containing about 10 grams of radium from Ottawa, and 1,530 tonnes containing about 1 gram of radium from Toronto. About 900,000 tonnes remain in Port Hope, and pockets of lesser contaminated soil have subsequently been discovered in Scarborough and Vancouver. Chalk River probably could have found other areas on its property where, with the addition of some engineered barriers, more contaminated soil could have been accepted. However, potential alternative solutions, such as transporting the soil to existing Eldorado waste sites in the Port Hope area – where it would have added only infinitesimally to the radioactive contaminants already present – seemed much more logical and cost-effective, and AECL's decision to accept no more Port Hope soil was at least partly motivated by a desire to force a more logical and cost-effective solution. This was probably a mistake. Shortly after the transfer of contaminated soil to Chalk River stopped, the government transferred responsibility for cleaning up these "historic wastes" from the Control Board to AECL, and AECL quickly found that the alternative "logical and cost-effective" solutions foundered on the emotional and highly political issue of one jurisdiction accepting

another jurisdiction's wastes. Since that time, millions of dollars have been spent on study commissions, public participation and temporary solutions, but a publicly acceptable resting place for this soil has not been found.

URANIUM MINING WASTES

With programs on used fuel and low-level wastes in hand, AECL, in concert with the federal Department of Energy Mines and Resources (EMR), tried to encourage the Canadian mining companies to undertake programs to verify that uranium mine tailings were being responsibly managed. In 1981, EMR established a committee, in which AECL participated – along with representatives of EMR, the Ontario and Saskatchewan governments, and the mining companies – to outline action required on mine wastes.

This committee concluded that there was not enough scientific data available to confidently predict the long-term effects of uranium tailings on the environment, and recommended that a research program be conducted to fill the knowledge gap. The Government of Canada responded to this recommendation in 1982 by announcing a National Uranium Tailings Program. This was a five-year program funded by the Federal Government at about \$8.5 million for the five years. Attempts to get shared funding from the provinces and mining companies failed, so the resources were limited to those provided by the Federal Government. A small group was set up within EMR to direct the work, which was all done on contract. Throughout the program, plans and progress were reviewed by two groups: a technical advisory committee (TAC), with representatives from five uranium mining concerns, three provincial governments and five federal institutions (AECL being one of the five), and a senior review board, with representatives from the Ontario Ministry of Environment, the Saskatchewan Ministry of the Environment, the Control Board and EMR.

The focus of the work was the development of a probabilistic predictive model, using the SYVAC approach, that would allow a comparison of the long-term implications

of various tailings close-out strategies. The development of this model required a fundamental understanding of the scientific and engineering principles involved in contaminant migration, and some initial research work was done to provide this basic information. A preliminary model was then developed and tried for a hypothetical reference tailings site. Output from this was then used to further define information required, and the iterative process continued until the end of the program. Model development was the responsibility of SENES Consultants, to whom AECL acted as a sub-contractor. The model was designated UTAP, for Uranium Tailings Assessment Program. A particular concern of the TAC, particularly the industry members, was that the outputs from the predictive model would not be accepted by the regulatory authorities for licensing purposes. With the agreement of the provincial regulatory agencies, the Control Board was asked to represent the regulatory agencies on a modelling review group that reviewed and contributed input into UTAP, to ensure its understanding and acceptance by the regulatory agencies.

Following its development, UTAP was used to predict exposures to five different receptors for five different close-out alternatives at a reference site in the Elliot Lake area of Ontario, and used for two different receptors for four different close-out alternatives at a reference site in the Rabbit Lake area of Saskatchewan. The two reference sites were significantly different, in that the Elliot Lake tailings contain large quantities of pyrite, which over time oxidizes and produces sulphuric acid, while the Rabbit Lake tailings are primarily silica, gypsum and clays. The close-out alternatives considered for the Elliot Lake tailings were: (1) no remedial action, (2) waste-rock cover (0.6 m of crushed rock), (3) pyrite removal (cover

by 2.0 m of depyritized tailings plus calcite), (4) glacial till cover (2.0 m), (5) water cover. The close-out alternatives for the Rabbit Lake tailings were: (1) no remedial action, (2) sandy till, top soil and vegetation covers, (3) soil, bentonite barrier, top soil and vegetation covers, and (4) waste rock cover. The results showed that the only treatment that reduced exposures relative to no treatment was water cover at the Elliot Lake site. However, in no case did the exposures exceed 1 percent of that from natural background.

At the termination of the program, the UTAP model was turned over to the Control Board for maintenance and general use by both the regulatory bodies and the industry.

SIGNIFICANCE OF THE WASTE MANAGEMENT TECHNOLOGY

Radioactive wastes must be safely managed. This imperative has resulted in the development of a technology and methodology which, if applied to other hazardous materials produced by industries, could provide significant financial return on the investment that has been made. The SYVAC type of analysis is not only applicable to radionuclides; it can be used to predict the environmental consequences of any disposal method for any contaminant from any industrial process. It forces one to quantify the biological hazard of the contaminant, consider all potential pathways for exposure, quantify the parameters affecting release, distribution and uptake and quantify the potential environmental effects. It allows the hazard to be quantified prior to process implementation, so that attention can be paid to contaminants that have a potential significant environmental effect.

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ADAPTING TO A CHANGING ENVIRONMENT

Chapter Twenty

Business Development, Revenue Generation and the Impact of a Changing Environment

R. G. HART

OVERVIEW

At the time the fledgling nuclear research division of the National Research Council moved from Montreal to Chalk River in 1946 it did so in an atmosphere of enthusiastic support for science and technology in general and nuclear technology in particular. People had seen the atomic bomb bring an abrupt end to the second world war and, spurred by some rather science-fiction-like stories, were eager to see what applications might be developed for use in a peaceful world. The Chalk River laboratory was given no specific mandate to pursue any particular application but it was clear that C.D. Howe, the federal minister responsible for NRC, expected that practical peaceful applications would be developed and that he wanted such applications to be commercialized by private industry. In his view it was entirely appropriate that government should fund research and development and make any commercial opportunities which arose freely available to private industry for exploitation. To this end, when the crown corporation, AECL, was formed in 1952, he made sure that its board of directors was largely drawn from potential users; private business and the provincial utilities. Howe's views prevailed for the next two decades which saw the development of a flourishing radioisotope business, the development of the CANDU nuclear electric generating system and its adoption by Ontario Hydro, the development of a comprehensive nuclear power station design capability and the development of a self-regulating nuclear research reactor (SLOWPOKE) and its adoption by several Canadian universities. AECL made no attempt to negotiate royalties for these developments.

In the early seventies, the atmosphere began to change. Spurred by environmental and peace activism, a substantial fraction of the public became suspicious of science and technology in general and nuclear technology in particular. Their concerns focused on the safety of nuclear power stations, nuclear waste management and the potential for nuclear weapons proliferation through the export of power reactor and fuel cycle technology. With public disenchantment came political disenchantment and the expected expansion of the CANDU power system both within Canada and abroad was curtailed.¹ At the same time, the federal deficit was increasing and the user-pay principle was coming into favour. The Pickering generating station was working well and AECL began to get political feedback that the reactor development job was completed and that the bulk of government funding should be shifted to the areas of public concern – safety, waste management and non-proliferation. Further, the argument went, if more reactor development work was needed it should be paid for by the nuclear utilities whose customers were then benefiting from the low cost of nuclear electricity relative to that generated from coal. The utility executives on the AECL board, who in Howe's time were considered essential, were now considered to be in conflict of interest and were replaced by senior government bureaucrats. AECL knew that the reactor development job was not done and that technical problems could be expected down the line; hence, while it transferred a substantial amount of effort to the areas of public concern, it retained a substantial effort on reactor development and tried to supplement the government funds retained in this area with revenue from other sources. Since it shared the view that the beneficiaries of nuclear power should now be bearing more of the development costs, it pursued Ontario Hydro for some of this revenue. As a result, Ontario Hydro started to share, on a fifty-fifty basis, the cost of specific parts of the program that it thought were underfunded. The first programs to benefit from this arrangement were those on verification of reactor safety codes, long-term behaviour of zirconium alloy pressure tubes and behaviour of CANDU fuel.

In 1978 AECL was reorganized and the need for increased reliance on commercial revenue was reflected in the new structure. The various divisions of AECL were organized into five more or less independently operating companies, AECL Radiochemical (formerly Commercial Products), AECL Engineering (formerly Power Projects), AECL Chemical (formerly Heavy Water Projects), AECL International (formerly International Marketing) and AECL Research (formerly Chalk River Nuclear Laboratories and Whiteshell Nuclear Research Establishment), all reporting to an AECL Corporate Office. In AECL Research a business development office was established at its head office in Ottawa and commercial operations offices were established at each site. The mandate of these offices was to obtain commercial revenue that would assist in support of AECL's program objectives. A restriction was that they were not to accept work in direct competition with Canadian industry. In 1982 the AECL International, AECL Chemical and AECL Engineering companies were combined to form CANDU Operations. Over the decade following the reorganization AECL Radiochemical prospered with gross revenues increasing from \$29 million in 1977/78 to \$111 million in 1986/87. CANDU Operations fell on hard times with gross revenues falling from a peak of \$448 million in 1980/81 to \$88 million in 1986/87. AECL's two heavy water plants at Glace Bay and Port Hawkesbury had to be shut down and the staff level had to be reduced from about 3800 to 1400. This downsizing was done in such a way as to maintain the core capability to design CANDU stations when orders picked up. At the time of writing orders for three new CANDU units had been received from the Republic of Korea and CANDU Operations' (now AECL CANDU) fortunes were improving. In the Research company commercial revenue

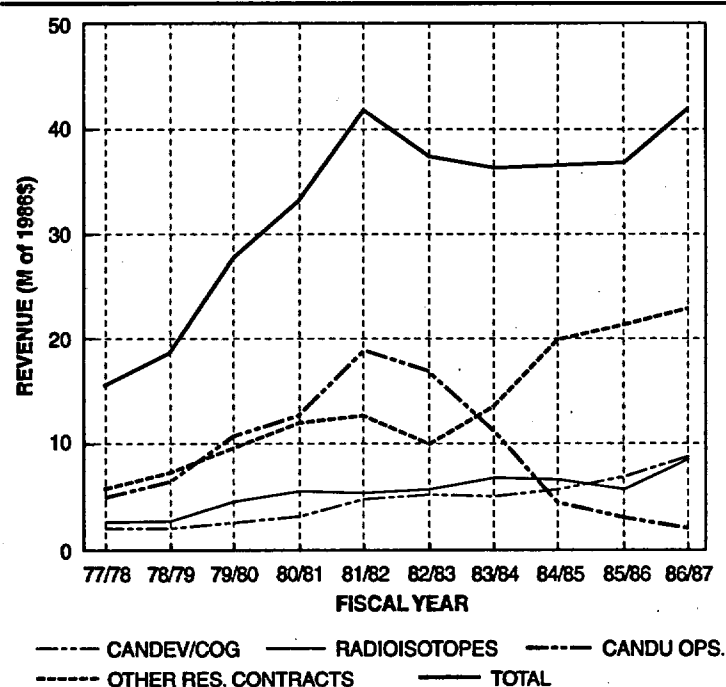
¹ In the early seventies both BC Hydro and Manitoba Hydro seriously considered nuclear power; BC Hydro for Vancouver Island and Manitoba Hydro as an alternative to further development of the Nelson River. In both cases, public opinion played a significant role in its rejection. Also, AECL was restricted in some promising foreign markets because of Canadian government proliferation concerns.

increased from \$8 million or 7 percent of total budget in 1977/78 to \$41 million or 18 percent of total budget in 1986/87.

By 1979 Ontario Hydro's contribution to shared-cost research programs was approaching \$2 million and the partners felt that some mechanism should be put in place to insure that the shared-cost funds were used to further the programs with the highest priority. Accordingly an agreement called CANDEV (for CANDU Development) was negotiated. A series of jointly staffed working parties, covering the spectrum of CANDU technologies, recommended programs to a steering committee composed of senior AECL and Ontario Hydro technical staff and chaired by a senior Ontario Hydro manager. The available funds were allocated by the steering committee and a full-time CANDEV manager coordinated the input, followed the progress of the programs and reported back to the steering committee. The arrangement was useful, not only in administering the shared funding, but also in ensuring user input into all of AECL's CANDU development programs and user access to the results. In 1983 AECL and Ontario Hydro succeeded in bringing the other Canadian nuclear utilities, Hydro-Quebec and New Brunswick Power, into the shared-cost program and the program was re-named COG (for CANDU Owners Group). Hydro-Quebec and New Brunswick Power contributed to programs in which they had an interest, in proportion to the number of CANDU reactors in their system relative to the total number in Canada. By 1986/87 the utilities were contributing about \$8 million in shared-cost funding and about another \$3 million in fully commercial contracts to the AECL CANDU development programs.

Canadian utility contracts were not the only, or even the major, source of AECL Research's commercial revenue. The growth of the various components of this revenue is shown in figure 20.1. In the early eighties much of the revenue came from CANDU Operations, primarily for computer rental and heavy water upgrading. However, by 1986/87, this had been largely replaced by revenue from research contracts with Canadian and foreign utilities, industries and institutions.

FIGURE 20.1
AECL Research Commercial Revenues; Constant 1986 \$s



Thus by 1986 a significant fraction of AECL's capability to respond to problems in the Canadian nuclear industry was being maintained by contracts outside that industry.

When AECL was reorganized in 1978, senior managers felt that, to respond to the new environment in Ottawa, they should not only attempt to increase the company's reliance on commercial revenue but should also try to produce new industries from the company's stock of technology. They sensed that, although there was still a good deal of support resulting from the development of the radioisotope business and the CANDU reactor, there was beginning to be something of a feeling "but what have you done for me lately?" Management felt that new industries could be developed from the wealth of technology available within the company if they could only encourage the staff to look for such opportunities.

Accordingly they instituted a program of incentives for the identification of ideas which could lead to new industries and they set up a process for evaluating ideas and supporting those judged to have the most promise. The effort was spearheaded by the new Business Development Office in the Research Company head office in Ottawa. The business development process selected was based on one used successfully by the 3M company, but within AECL it met with limited success. Over an eight-year period many ideas were put forward and nine new products were developed. However, at the time of writing, none of these had led to the major new product line that the proponents of the initiative were hoping for.

Although the Canadian nuclear utilities' contribution to AECL's R&D programs increased from 1977 to 1986 it was still, in 1986/87, only about \$11 million or about 5 percent of the federal government's contribution and the federal government felt that this was an unfair share. Accordingly, in 1986 it announced a phased reduction in its appropriation for AECL and initiated negotiations involving the federal government, the appropriate provincial governments, AECL, and the appropriate utilities to attempt to get the provincial utilities to provide replacement funding. These negotiations eventually culminated in agreements whereby AECL's base funding for R&D would remain at about \$220 million (1990), escalated according to the consumer price index, through fiscal 1996/97 with the federal government providing 65 percent of the funding and the nuclear utilities 35 percent. The utilities contribution was to be made through COG and COG ground rules were to apply to its use. This meant that AECL had to match COG dollars fifty-fifty and that the COG steering committee effectively controlled 70 percent of AECL's program. Some AECL programs, including the new industries initiative, were casualties of this arrangement but adjustments were made and the arrangement worked reasonably well. AECL was of course free to increase its outside commercial revenue as it was able and, as a result, it had one of the largest commercial R&D operations in Canada.

The events outlined in this overview provide a wealth of interesting stories, a few of which are related in the remainder of this chapter.

BUILDING THE RADIOISOTOPE INDUSTRY

With the startup of NRX in 1947, scientists at Chalk River had the best vehicle in the world for large-scale production of artificial radioisotopes and they soon developed methods for their production and processing into forms that could be used for radiation therapy, radiation sterilization and a variety of tracer experiments, particularly in the medical research and life sciences fields. By 1951, the radioisotope business had grown to a size where transfer to private industry, according to Howe's dictates, could be considered. However, at the time the only Canadian industry with any experience in the radioisotope business was the crown corporation Eldorado Mining and Refining which had established a Commercial Products division to sell its by-product radium-226. Thus in 1951 an arrangement was made to have this division take over the marketing and distribution of the radioisotopes being produced at Chalk River (see chapter five). When AECL also became a crown corporation in 1952, AECL and Eldorado agreed that in the future the Commercial Products division would have much more in common with AECL than with Eldorado so the division transferred to AECL as the Commercial Products division of AECL. Over the period 1952 to 1955, the division established facilities in Ottawa for processing most of the radioisotopes to their final product form and the technology for this processing was transferred from Chalk River. Only the production, the more highly radioactive processing, and the waste handling remained at Chalk River. Chalk River was paid for its part of the work on a marginal cost basis,² while the profits accrued to the new division. This caused some considerable chagrin to some of the Chalk River scientists involved who were concerned that the R&D budget would end up subsidizing a major commercial activity. Later events would show that their concern was justified.

² Marginal costing is a practice often used in industry whereby the main product, in this case R&D, bears the total overhead cost of the operation while the by-product, in this case radioisotope production, bears only its own direct cost.

In the early years of isotope production, when the demand was small, the production could be accommodated in the Chalk River reactors without difficulty. However, as isotope sales increased, isotope production increasingly dictated not only the use of research reactors but certain waste management activities as well. Accordingly the Research Company sought to change the basis of payment from marginal costing to full cost recovery and the urgency of this increased when initiatives were taken to privatize the Radiochemical Company (formerly the Commercial Products division). Subsequently, when the Radiochemical Company was privatized, an agreement based on full cost recovery for production and separation was achieved. However, the Research Company did not succeed in obtaining agreement on full cost recovery for waste handling and is facing substantial costs for waste treatment facilities, which it cannot recover.

BUILDING THE CANDU POWER STATION INDUSTRY

In the mid-fifties, when AECL and Ontario Hydro wanted to build a demonstration CANDU power station, the 20 MWe NPD station, the commercialization-by-industry philosophy was again attempted. The arrangement made was that Canadian General Electric Company (CGE) would supply the Nuclear Steam Supply System (NSSS) on a cost-plus, turnkey basis³ and would make a \$2 million contribution toward the cost; AECL would pay for and own the NSSS, and Ontario Hydro would pay for and own the balance of plant, operate the plant and purchase the steam from the NSSS at a price equal to the cost of steam at the J.L. Hearn coal-fired plant in Toronto. Any profits or losses on the NSSS part of the plant were to accrue to AECL. The idea was to have CGE become a turnkey supplier of CANDU generating stations, or at least the NSSS part of them, to the Canadian and foreign market. However, after NPD, Ontario Hydro made it clear that, for their stations, they did not want to be tied to a single supplier; hence, when the 200 MWe prototype station, Douglas Point, was built the arrangements were substantially different. AECL was persuaded

to establish its own engineering division, Power Projects, to design the NSSS while the design of the balance of plant was contracted to Ontario Hydro. Further, AECL contracted construction management, construction, and subsequent operation to Ontario Hydro. AECL owned the station and Ontario Hydro agreed to purchase electricity from it at a price equal to the cost of generation at its Lakeview coal-fired plant in Toronto. Any profits or losses accrued to AECL. In subsequent CANDU stations built by the utilities, this design and construction pattern was generally followed except that, for stations outside Ontario, the balance of plant was generally subcontracted to CANATOM,⁴ and the construction management of those stations was done by AECL. For a time CGE attempted to develop a business as a turnkey supplier of CANDU stations in the foreign market but without ongoing at-home business this proved impossible. Eventually they evolved into a role as a special systems designer and special components and fuel manufacturer and in these roles made a major contribution to the developing industry.

Making arrangements for design, construction and operation of prototype CANDU stations was only the start of the work involved in industrializing CANDU technology. The station is made up of a myriad of pumps, valves, pipes, heat exchangers, seals, fittings, instruments, all of which had to be made to a quality that was essentially foreign to Canadian manufacturers and critical components such as fuel, pressure tubes and calandria tubes had to be made from materials Canadian manufacturers had never even seen. To complicate matters the utilities wanted to have at least two qualified suppliers for each component so that they could have competitive bidding and the market for products of the quality required was, for the most part, not large. Industries were prepared to

³ A turnkey supplier is one who takes a contract for supply of the complete system including design, construction and construction management. In the case of NPD, CGE did the design and construction management but contracted the construction to Canadian Bechtel.

⁴ CANATOM is a private engineering company based in Montreal and jointly owned by several large Canadian engineering firms.

get involved but were not prepared to put out much of their own money to develop the new capabilities. Thus much of this capability had to be developed through development contracts. For example, a development contract would be let to a manufacturer to fabricate an experimental fuel assembly for testing in an NRX loop and laboratory scientists would work with the manufacturer to develop the technology required. These development contracts were often complemented by the attachment of manufacturer's staff to the laboratories to follow performance first-hand. In this way the capability to produce nuclear quality fuel, coolant tubes, calandria tubes, end fittings, valves, pumps, instruments, and a myriad of other components was developed in Canadian industries. That the method was successful is attested by the fact that over 90 percent of the cost of Canadian CANDU power stations is Canadian content.

THE SLOWPOKE RESEARCH REACTOR

Because of its large area and sparse population, Canada has a need for economical off-grid electricity and remote steam production. In 1964 AECL established a small group at Whiteshell to see if a small nuclear reactor could be developed to meet this need. The competition was packaged oil-fired boilers and the cost of the competition was determined by the cost of transporting the oil. Early in the study it became apparent that every practical innovative idea would have to be used if nuclear reactors were to meet the competition, with the most radical departure from conventional practise being completely automatic operation. At the size of reactor required to meet the bulk of the need, even one operator per shift would make the cost of the nuclear generated energy uncompetitive. After about eighteen months of study it was concluded that reactors could be developed to meet the competition in the more remote areas of the country but that the size of the market did not justify the development costs that would be required. Thus the group was disbanded.

However, some of the concepts lived on. A Chalk River physicist who had been attached to the group, returned to

Chalk River with an idea for a research reactor which could operate unattended and be used by universities, hospitals and research institutes for teaching, activation analysis⁵ and the production of small quantities of short-lived radioisotopes.

In early 1967, experiments at Los Alamos on critical masses had shown that when a uranium-235 polyethylene mixture was surrounded by a thick beryllium reflector the critical mass was surprisingly low. It was reasoned that if a beryllium reflector could be used to control the excess reactivity in a water-cooled core, a reactor could be designed in which any conceivable power transient would be self-limited by the negative temperature and void coefficients of the core. A reactor with such intrinsic safety features could be licensed for unattended operation even in large population centres where the universities, hospitals and research institutes were located. The design was developed through 1968 and 1969, and in 1970 the first prototype commenced operation at Chalk River. The reactor was named SLOWPOKE for Safe LOW POWER Critical(K) Experiment (see figure 20.2). At the maximum rated power of 20 kW, the reactor provided thermal neutron fluxes of 1×10^{12} and 4×10^{11} n/cm²/sec to sample sites in, and just outside, the beryllium reflector respectively. During 1970, the prototype reactor was put through its paces at Chalk River and it performed as expected. Then in early 1971, the reactor was dismantled and transferred to the Environmental Science and Engineering Institute of the University of Toronto where it was reassembled in a renovated building adjacent to offices, laboratories and a library. On 4 June 1971 the University of Toronto received AECB operating licences which subsequently included approval to operate unattended for up to 18 hours. Remote surveillance was provided by an alarm in the building security console. Because of the intrinsic safety features of the system, i.e., no fuel failures even with complete loss of control, licensed operators with formal reactor training were not required. Instead "licensed

⁵ The sample is irradiated with neutrons and the radioactive products are measured by beta- or gamma-ray counting. High sensitivity is possible for many elements (see chapter seven).

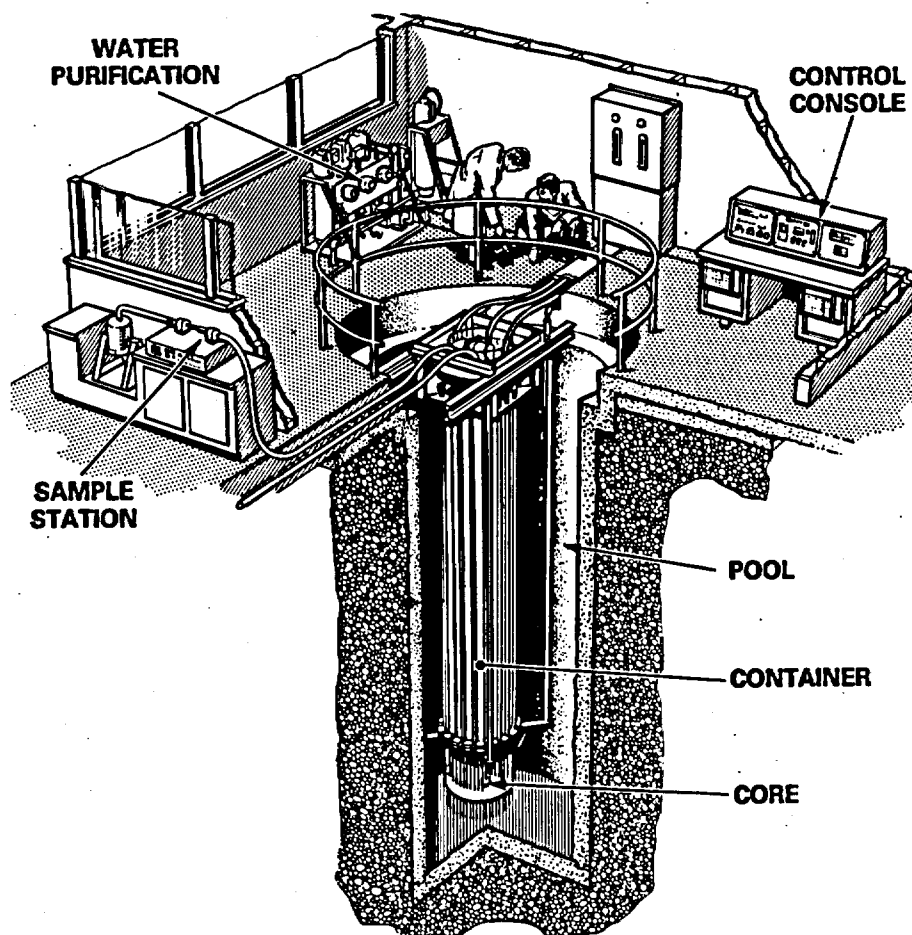
users" were allowed to start up and shut down the reactor, operate the water purification and gas purging systems and irradiate samples. However, only an authorized reactor engineer was allowed to adjust the beryllium reflector and one from AECL visited the University of Toronto every six to twelve months to perform this task.

Within AECL, the right to sell SLOWPOKE research reactors was granted to the Commercial Products group and in May 1971 a second SLOWPOKE reactor went into operation at their facilities in Ottawa. Commercial Products used their SLOWPOKE to provide a commercial irradiation service and to demonstrate the reactor to potential customers. A total of seven SLOWPOKE research reactors were eventually sold, five to Canadian universities, one to the University of the West Indies in Jamaica, and one to the Saskatchewan Research Council in Saskatoon.

SOME INTERESTING SHARED-COST PROGRAMS

One of the first shared-cost programs initiated by AECL and Ontario Hydro was aimed at improving the computer codes used to predict the behaviour of the primary and emergency coolant in the event of a pipe break in the primary heat transport system of a CANDU power station (see chapter fifteen). In the early codes used to analyze such accidents the simplifying assumption was made that the steam and water produced in the accidents were uniformly distributed in the fuel channels and piping, moved at the same velocity and were of equal temperature, with the result that all fuel elements in the fuel channel were cooled at the same rate. To a number of scientists in both AECL and

FIGURE 20.2
A SLOWPOKE-2 Reactor Installation



Ontario Hydro this assumption seemed to defy the laws of gravity, so when the CANDU-OCR program was terminated, Whiteshell was encouraged to establish a small group to investigate the problem. When experiments commissioned at Westinghouse Canada in 1973 verified that the steam and water would indeed separate in the channel causing uneven

cooling of the fuel, the group at Whiteshell embarked on a program to develop a computer code that would take into account the fact that the water would settle to the bottom of the channel with the steam at the top, that the steam and water would move at different velocities and that they could be at different temperatures. The program was complex, involving the need for a number of experimental facilities of increasing size and complexity and it soon became clear that Whiteshell could not, on its own, release sufficient effort to advance the program at the rate Ontario Hydro felt was necessary. Accordingly an agreement was negotiated whereby the effort would be increased and Ontario Hydro would share the cost on a fifty-fifty basis. Included in the agreement was a rental charge for the experimental facilities required to verify that the mathematical representations of the physical phenomena observed were accurate. Indeed when Whiteshell was unable to raise the capital for a large experimental facility which would better represent behaviour in the power reactors, Ontario Hydro contributed half the capital to ensure that the facility would be built. In the end, a new code that realistically described the LOCA sequence was developed and verified. Subsequently the cooperative program in the safety analysis area was expanded to obtain better models for the behaviour of fuel channels and calandria tubes in loss of coolant/loss of emergency coolant situations, the behaviour of fission products released from the fuel in accident situations and the behaviour of hydrogen released to containment in accident situations.

At approximately the same time as the shared-cost program on safety analysis codes was getting under way at Whiteshell, Ontario Hydro experienced cracks near the end fittings in the Zr-2.5Nb pressure tubes in Pickering 3 and Pickering 4, which eventually led to the need to replace sixty-nine tubes in these reactors (see chapter twelve). This was a serious economic problem for Ontario Hydro so, in order to see the cause and solution determined as quickly as possible, they shared in the cost of the investigative program. Although the cause and a relatively easy solution were determined fairly quickly, the incident left a nascent worry

in the minds of both AECL and Ontario Hydro people that perhaps our understanding of the long-term behaviour of this critical component was not as good as it should be. In fact at a meeting between senior AECL Research and utility staff in 1979, when asked what the three major concerns of the operating staff were, the Ontario Hydro representative answered, "#1, pressure tubes, #2, pressure tubes and #3, pressure tubes".⁶ With this kind of feedback, AECL substantially increased its allocation to the fuel channel program and continuing financial support was provided by Ontario Hydro. The objective of the program was to put in place the technology that would allow the behaviour of the pressure tubes to be confidently predicted for the projected life of the reactor.

Then on 1 August 1983, a major rupture occurred in a Zircaloy-2 pressure tube in Pickering 2 (see chapter twelve). Again the cause was quickly traced, this time to hydride blisters which had formed at a line of contact between the pressure tube and the calandria tube, the contact resulting from out-of-place garter springs. In this case, however, the solution was not easy. It involved the complete retubing of the reactor, and nagging questions remained as to why the hydrogen concentration in the tube was higher than expected, whether the concentration would be high enough in the Zr-2.5Nb tubes to cause the same problem,⁷ whether the garter springs in other reactors were out of place, if so how they could be relocated and how the tubes could be monitored to determine if they needed to be replaced. This resulted in an increase in shared-cost funding from Ontario Hydro and prompted the other Canadian nuclear utilities to get involved as well. After intensive investigation it was determined that many garter springs were out of place in all reactors and that the Zr-2.5Nb pressure tubes were susceptible to the same problem, although much later in life because

⁶ L. Woodhead, Private communication.

⁷ Only two of the commercial CANDU reactors built had Zircaloy-2 pressure tubes. All the rest had tubes made from Zr-2.5 percent Nb.

of the lower rate of hydriding. A method was developed for non-destructively monitoring the hydrogen concentration in the tubes so that it could be determined when they were approaching the danger point. Further, a joint CANDU Operations/Ontario Hydro Research program resulted in the development of equipment that could relocate the garter springs to their proper positions in existing reactors. For new reactors a design change was made to prevent the garter springs from moving.

SOME INTERESTING COMMERCIAL CONTRACTS

One of the critical parts of the CANDU program was the development of quality pump seals for CANDU reactors (see chapter fourteen). Because of the high cost of down-time for seal replacement the target was set that the seal should have an operating life of five years although existing seals were lucky to have a life of one year. The challenge was to maintain a constant, very small (0.25-2.5 micron) gap between a rotating face on the shaft and a fixed face on the housing for the five-year period. Using computer modeling to determine how the geometry could change with time as a function of temperature, pressure and material properties Chalk River engineers were able to develop a seal for CANDU pumps that met the target life. More importantly, since they had developed an understanding of why seals failed and why they did not, they were able to apply their technology to other pumps in other applications. By the late seventies many American utilities were having serious problems with pump seals and in response to the commercial initiatives of AECL Research the Chalk River team moved to sell their expertise in the United States. In the next few years, they obtained commercial contracts for over \$18 million and resolved pump problems for Niagara Mohawk, General Public Utilities and Louisiana Power and Light. In 1986, the United States space shuttle Challenger blew up shortly after launch with the loss of seven lives. The problem was diagnosed as leakage of hydrogen fuel from an O-ring type seal at the base of the booster rocket. The

development and testing of a new O-ring was contracted by NASA to the American company Morton Thiokol. As part of the pump seal development program Chalk River engineers had developed a considerable expertise in rubber O-ring performance and Morton Thiokol engaged Chalk River as a subcontractor to advise on the re-design and perform tests in their test rigs. In recognition of their contribution, two Chalk River scientists were invited to the first space shuttle launch in which the new O-ring was used. NASA has experienced no further difficulties from this source.

Over the period 1982 to 1985, the Chalk River biology division received grants totalling \$1.4 million from the United States National Institute of Health to assist in the study of families having genetic disorders which made them abnormally sensitive to carcinogens (see chapter three). It was found that in people with a sunlight-sensitive inherited disorder (named xeroderma pigmentosum) the ability of cultured skin cells to repair sunlight damage to DNA was impaired and that the incidence of skin cancer in these people bore a direct relationship to the degree of impairment. In patients whose cultured skin cells were ten-fold more sensitive than normal to the lethal effects of the ultraviolet component of sunlight, the probability of contracting skin cancer at an early age is approximately 10,000 times normal. Subsequent testing showed that even in populations considered to be normal, the ability to repair DNA damage caused by ionizing radiation and certain other carcinogens varied by as much as a factor of five. This indicated that the radiation dose given to people in radiation therapy for cancer should be varied depending on their DNA repair capability and suggested that perhaps certain members of the population should take more care than others to avoid carcinogens. In 1987, a contract for \$700 thousand was let by the AECB to the Chalk River biology division to assist in the development of a simple and economical test for DNA repair capability. The work was largely completed by 1989. The test involved taking a blood sample, separating and culturing the white blood cells, exposing a known number to a fixed dose of radiation, then periodically counting the number of cells to

determine the rate of recovery. At the time of writing, it had not been decided how, or if, the test would be used in the general population, but considerable effort in other laboratories in the United Kingdom and the United States is being devoted to its applicability to radiation therapy for cancer.

In the early eighties the German particle physics community was attempting to obtain international participation in the construction and utilization of a large \$0.5 billion, electron-proton colliding beam facility (HERA). The Canadian Institute for Particle Physics was interested in participating and, after some persuasion, the Canadian Natural Sciences and Engineering Council (NSERC) agreed to contribute funding. The arrangement made was that NSERC would pay for certain parts of the facility which could be built in Canada. After protracted negotiations, complicated not a little by a design change to HERA, AECL signed a contract to have its Accelerator Physics Branch at Chalk River provide some of these parts. HERA has three accelerating rings, each one taking the electrons and protons to higher energy, and AECL was to provide the 52 MHz radio frequency (RF) accelerating system for the second proton ring.⁸ The value of the contract was \$2.1 million. Subsequently AECL accepted a contract for the supply of the 52 MHz RF accelerating system for the third proton ring as well.⁹ The value of this second contract was \$2.6 million. The systems for the two rings were not identical and each had its unique problems. The basic system design was taken from an existing accelerator operating at similar frequency at the Fermi National Accelerator Laboratory (United States) but for the second ring at HERA it had to be adapted for a larger frequency swing and this took time. Further it was decided to incorporate a high quality ceramic pipe within the cavities to eliminate the need to evacuate the whole cavity. Unfortunately the quality of the first ceramic pipe used was not high enough and delays occurred while a replacement supply of higher quality was located. For the third ring the frequency swing was smaller so this was not a problem but the gap voltage had to be higher making the use of a ceramic pipe impractical and requiring evacuation of the entire cavity with attendant complications.

Thus the work took longer than expected, each system taking over two years to complete. However, they did not hold up the commissioning of HERA and at time of writing both were performing to expectation. Because of the delays AECL did not make a profit on the HERA contracts. However, AECL developed technology and contacts that subsequently led to other substantial contracts in both accelerator hardware and other industrial RF applications.

In the mid-seventies, the United States Department of Energy wanted to obtain experimental confirmation of the computer codes used to predict the behaviour of American light-water reactor fuel under accident conditions and they contracted this work to Battelle Northwest Laboratories in Hanford, Washington. Battelle needed to do a number of in-reactor tests on instrumented fuel assemblies under increasingly damaging conditions for comparison against code predictions and NRU was one of the few reactors in the world where such tests could be safely done. Accordingly, a sub-contract was let to Chalk River for the use of the U-2 loop in NRU for these tests. In total eleven tests were to be done over a period of about ten years. In the most severe tests, the fuel was starved of coolant and the reactor was shut down. The decay heat was then allowed to drive the temperature to the point where it peaked (around 2,400°C) and started to decrease. This indicated that the chemical reactions taking place had reached a point where they were limiting themselves without manual interference. Provisions were made to collect the hydrogen and fission products released in the test in order to check the code predictions of the mix of material released to power reactor containment after such an accident. After the

⁸ The system includes accelerating cavities, RF supplies and controls.

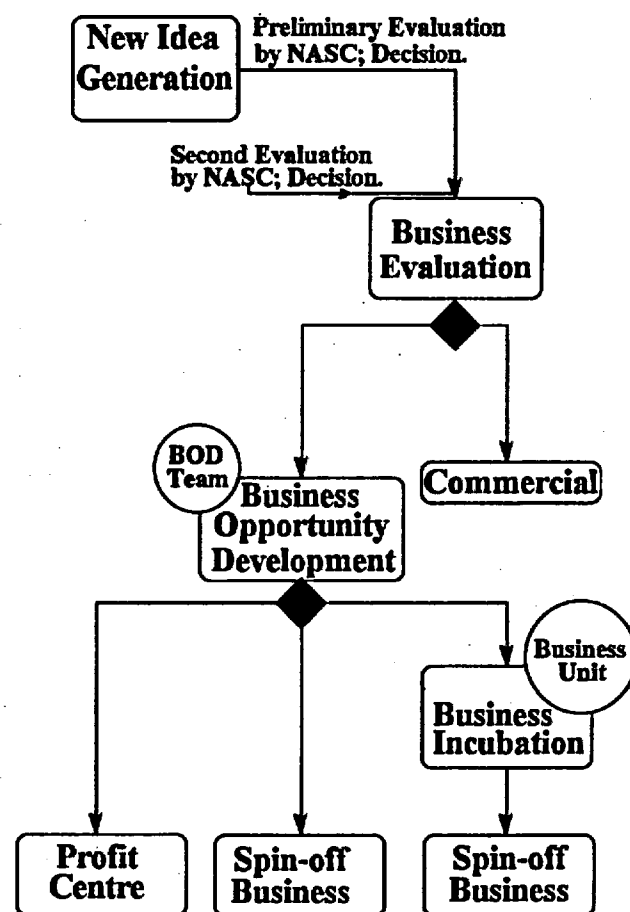
⁹ The third proton ring also includes a 208 MHz accelerating system provided by another contractor. The 52 MHz system in this ring is used to "bunch" the protons properly for injection into the 208 MHz system where they are accelerated to 820 GeV. The 52 MHz system can and, for a year before the 208 MHz system was ready, did accelerate the protons to their full 820 GeV but in this mode the bunch shape is not optimum.

tests, the severely damaged fuel was immobilized in an epoxy resin and examined in the Chalk River hot cells to determine the extent of damage and the extent and products of the reactions between coolant, cladding, fuel, control rod, and support grid materials. At time of writing, ten of the eleven tests had been completed without incident and the eleventh remained to be done. The contract value to AECL was about \$14 million plus access to the test information, which was valuable to the Canadian program.

THE NEW BUSINESS DEVELOPMENT INITIATIVE

In 1978, a new business development process was initiated within AECL based on one used successfully by the 3M company (see figure 20.3). Ideas for new businesses were encouraged from anyone within the organization. These ideas were submitted to a New Applications Steering Committee (NASC) and, if the committee considered an idea had promise, it was sent back for business evaluation. This evaluation was done by the proponent and a business development officer and involved an estimate of the potential market and market share and, if necessary, some testing of the technology. After this evaluation, the idea was resubmitted to the Steering Committee where a decision was made as to whether or not the idea still had promise, and if it had promise, whether it could best be exploited as a service to be offered from the laboratory to increase commercial revenue, as a technology to be licensed to an existing firm or as a new business. If the decision was that it should be exploited as a new business, a business opportunity development (BOD) team was established to further develop the technology, confirm the market size, share and entry requirements and prepare a business plan for furthering the opportunity. The business plan could suggest operation as an internal project centre within the laboratory, immediate launch as a spin-off business, or incubation within a business unit. If the suggestion was business incubation and that suggestion was accepted, an AECL-funded business unit would be established which would have to

FIGURE 20.3
The Business Development Process



confirm the technology by prototype construction, obtain initial orders from clients and establish business operations. Experience at 3M indicated that only about one idea in 500 would result in a successful new product line but this rate of return had been enough to keep 3M well ahead of its competitors in new product development.

In the eight-year period the initiative was in place several hundred ideas were put forward and nine new products were

developed. The nine new products were a personal neutron dosimeter, a line of eddy current probes for boiler and heat exchanger inspection, a new type of research reactor, a self-regulating space heating reactor, an engine wear meter, a steam quality meter, a wetproofed catalyst, a line of internally cooled clothing and a family of high-power industrial electron accelerators. These new products are described below.

The personal neutron dosimeter is a small cylinder of clear plastic supersaturated with Freon gas (see chapter three). When a neutron passes through the plastic, gas bubbles are precipitated and the extent of bubble precipitation is proportional to the neutron dose. Prior to this development, neutron monitors were too large and heavy to be conveniently worn by an individual. The development resulted in the incorporation, by the inventor, of a new business, Bubble Technology Industries, located in the village of Chalk River; it employs about fifteen people. In hindsight the dosimeter was not nearly ready for commercial application at the time the company was formed. Fortunately the new company was able to obtain contracts to complete the development and to produce other products which enabled it to survive its early years.

By the seventies remote-field eddy current inspection had become the preferred method for determining the condition of in-service boiler and heat exchanger tubes, with major applications, in particular, in the nuclear and air-conditioning industries. However a great deal of work remained to be done to develop inspection probes and signal processing systems that could separate the defect signals from extraneous signals. Defect conditions included weld defects, pitting, axial and circumferential cracks and tube thinning caused by wear against the support plates. Extraneous signals could come from tube sheets, support plates, corrosion products or simply probe "lift-off" from the surface. Complications were also introduced by the tube materials themselves, with ferromagnetic materials causing particularly difficult problems. By the mid-eighties the Chalk River group had developed a series of probes (including one for ferromagnetic materials) that could separate the defect signals from the extraneous signals and a business unit within the laboratory was marketing them. The

quality of the probes attracted the attention of the United States Westinghouse non-destructive examination unit, ECHORAM, who were looking for a line of probes to complement their signal processing equipment. An agreement was reached whereby AECL would license its probe technology to Westinghouse and Westinghouse would establish a probe manufacturing facility in the local area. This facility was established in the town of Deep River as the Nuclear Services Division of Westinghouse Canada; it employs about fifteen people.

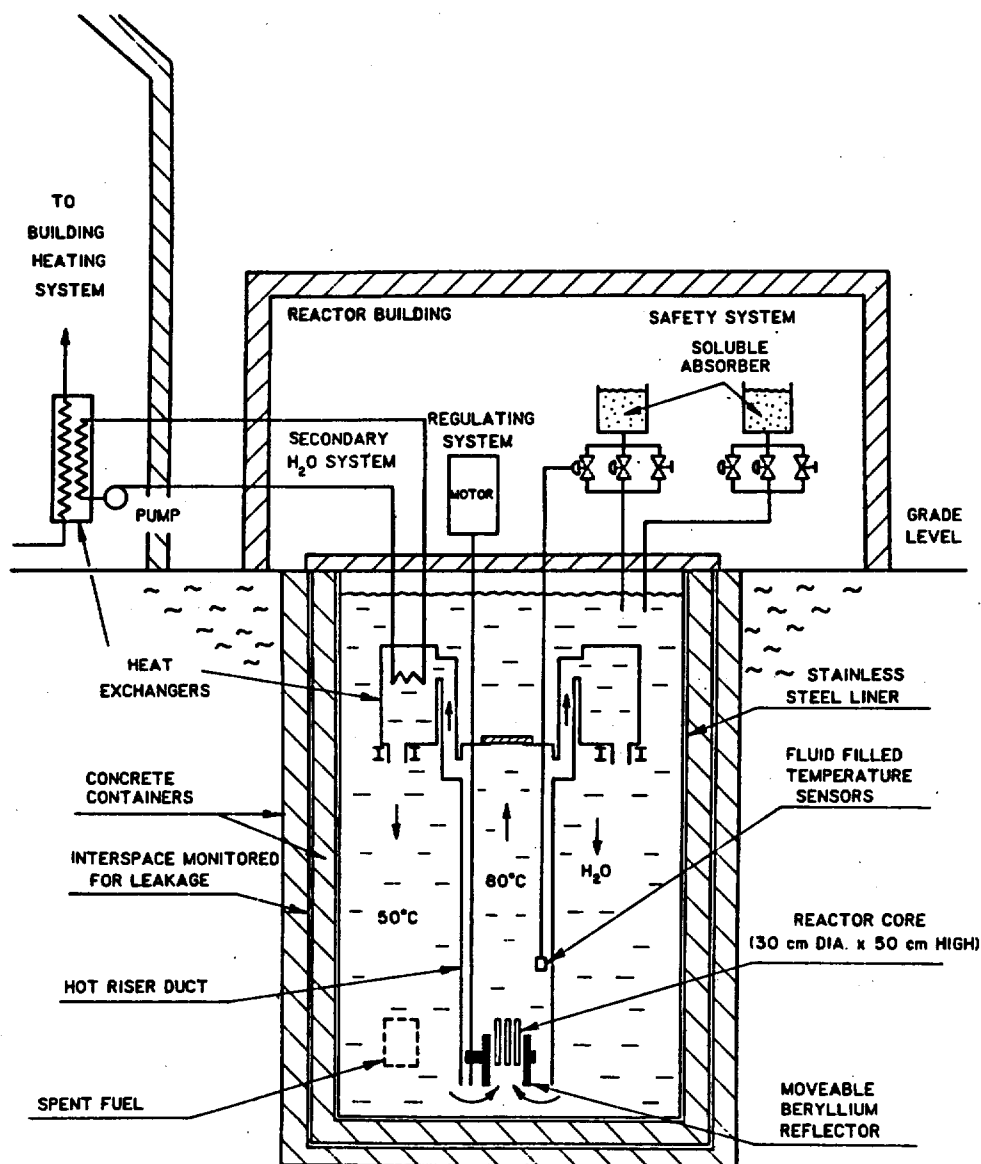
The new type of Research Reactor developed is called MAPLE (for Multi-purpose Applied Physics Lattice Experiment). It can be built in sizes from 1 to about 30 MW (thermal) to meet the program requirements of the customer. A 1 MW reactor could meet the major artificial radioisotope needs of many countries while a 20 MW reactor could perform most of the functions of NRX and be built and operated at a much lower cost than an NRX replacement. One was sold to the Korean Advanced Energy Research Institute but a second, planned for construction at Chalk River, has been cancelled.¹⁰

The self-regulating space-heating reactor resulted from the upgrading of the 20 kW (thermal) SLOWPOKE research reactor to 10 MW. Although the reactors operate on essentially the same principles, the excess reactivity required to over-ride xenon poisoning in the higher powered heating reactors resulted in a demand for safety systems that were more complex than had been anticipated. This made the reactors uncompetitive with oil and gas at today's prices. The technology has been documented so that it can be resurrected when it is needed (see figure 20.4). A 2 MWt prototype was built at Whiteshell and operated at low power for some time, to confirm the design parameters.

The engine-wear meter monitors iron in oil on-line and can be used to determine when most kinds of rotary machinery require maintenance. This eliminates the need to base maintenance on mileage or hours of usage. The potential market for such an instrument is large, and AECL set up an internal

¹⁰ Subsequently the construction of two MAPLE reactors at Chalk River was commissioned by Nordion International for radioisotope production.

FIGURE 20.4
Proposed 2 MWt SLOWPOKE Heating Reactor



business unit, SENSYS, to try to tap that market. SENSYS succeeded in getting prototype wear meters installed in several applications notably aircraft engines, helicopter rotors, and natural gas pipeline pumps but the operation of the prototypes was not trouble-free and the problems took time to resolve. Thus, in the three years that SENSYS operated, it never obtained the major sale it required to make it profitable. Its assets were sold and the technology was licensed to an Ottawa engineering company, Gastops Inc.

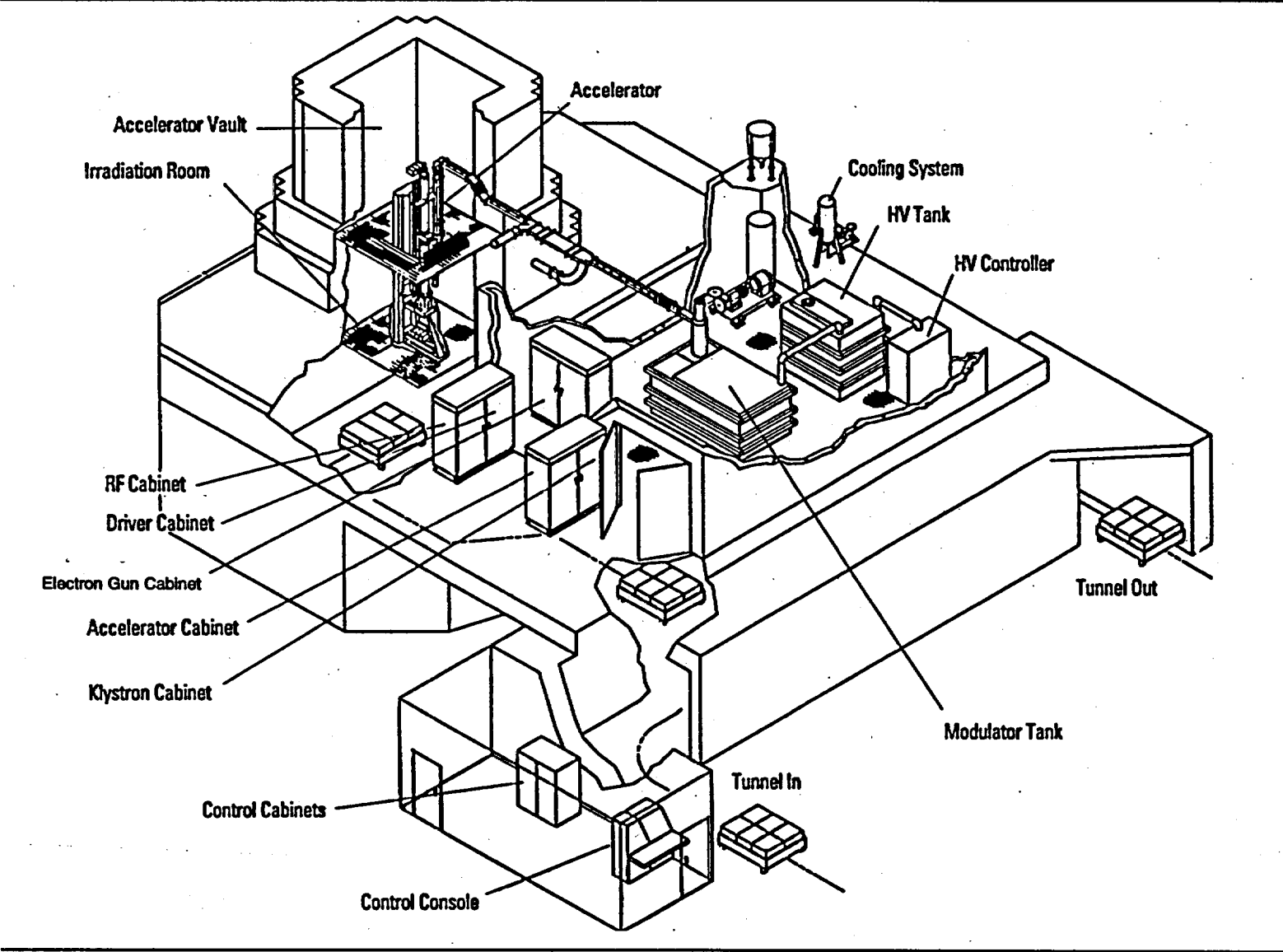
The steam-quality meter measures the amount of entrained water in steam by neutron attenuation. The meters were developed to measure steam quality in reactor safety experiments, but also have application in enhanced recovery operations in the petroleum industry. AECL set up sales offices in both Calgary, Alberta, and Bakersfield, California to market this product but field tests showed that the electronics developed for the reactor-safety application were not suitable for the petroleum application. The sales offices were therefore closed while new electronics were developed.

The wetproofed catalyst was developed to promote hydrogen/deuterium exchange in the liquid phase for heavy-water production but it can also be used to catalyze both oxidation and reduction reactions at low temperatures. Several applications of this property were studied including the recombination of hydrogen and oxygen to prevent the buildup of explosive mixtures in oxygen generators and aluminum batteries, the reduction of nitric oxide in fertilizer manufacturing-plant off-gases and the oxidation of carbon monoxide in a carbon monoxide-monitoring

instrument. Batches of catalyst are sold on a continuing basis for these applications and the carbon monoxide-monitoring instrument has been licensed to the Asahi Co. for marketing.

The internally cooled clothing product arose from a search for applications of the wetproofed catalyst. The recombination of hydrogen and oxygen on the catalyst produces heat

FIGURE 20.5
Impela Electron Beam Facility



and the original concept was to produce a line of internally heated clothing for underwater and arctic applications. Methods were developed to incorporate small-diameter flexible tubing into clothing to carry the heat transport fluid, and small pumps were developed to circulate it. However, after a time, the two people working on the program recognized that there was likely a larger, and certainly a more immediate, market for internally cooled clothing and that in many applications the cooling could be conveniently provided by an external source such as dry ice or an ice bath. Accordingly they incorporated a new business, EXOTEMP Ltd., in the city of Pembroke to produce internally cooled clothing. Originally they thought the main users might be people in motor sports and people who had medical disorders that prevented them from perspiring, but with the advent of the Gulf war their main market became the military, for personnel who had to wear chemical-warfare-resistant clothing in a hot climate. To meet the demand their staff level increased to about 35 but it has since decreased to less than half that number.

The family of high power industrial electron accelerators is an out-growth of the post-ING transition (see chapter nine) and has been given the name IMPELA (for Industrial Materials Processing Electron Accelerator) (see figure 20.5). The

family includes machines with beam powers from 25 to 250 kW and electron energies from 5 to 18 MeV. The power determines the amount of material that can be processed in a given time and the energy determines the penetration of the electrons. The accelerators have advantages over cobalt-60 for industrial irradiation in that they can be turned off and become non-radioactive when they are not in use and they can process much more material per unit time. Applications include existing ones such as sterilization of medical supplies and polymerization of plastics; possible future applications include food irradiation, the breakdown of lignin in paper manufacture, manufacture of composite materials, and sterilization of hospital and airport wastes. The accelerators are being marketed by an AECL business unit located in Kanata, Ontario. The business unit employs about twenty-five people who do the marketing, design and assembly and oversee the installation. The components are all fabricated on contract, mostly to local suppliers.

The 3M process applied within AECL did not produce the major impact product or products its proponents were looking for. Thus when programs had to be rearranged to reflect the new shared-funding agreement with the provincial utilities the initiative was terminated.

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A RETROSPECTIVE

Chapter Twenty-one *Learning from* *History*

J.A.L. ROBERTSON

INTRODUCTION

Canada's emergence as a major industrialized country coincided with the early period of this history. In 1950, North America was beginning to enjoy the fruits of massive wartime industrialization converted to peacetime purposes. For the next two decades Canada's booming economy allowed impressive expansion of the social programs, notably health-care and education. However, by 1970 the manufacturing sector was in relative decline, with exports of raw materials increasingly paying for imports of manufactured goods. Now, in the nineties, Canada is facing economic competition not just from the United States and Europe but even more from the Pacific Rim countries: virtually all consumer electronics and many automobiles are imported, while the aviation industry is in serious trouble.

Canada's future will be crucially dependent on developing and nurturing technically sophisticated industry, but there is little agreement on how this can be achieved. Many people believe that science and technology (S&T) is an essential ingredient, but again there is little agreement on what should actually be done. There are interminable declarations and conferences on S&T policy, but no effective S&T policy, and no evidence that relevant lessons are being learned from history.

Perhaps the most important lesson to be learned is that Canada is capable of fostering technically sophisticated industry, from hydroelectricity generation to telecommunications technology. At the Canadian Engineering Centennial in 1987, the top ten achievements of Canadian engineering during the previous hundred years were honoured. One of those achievements was the CANDU reactor. AECL's experiences in directing and managing its activities, especially the

R&D leading to the CANDU reactor, offer many useful lessons for S&T policy in Canada.

The CANDU reactor is one of only two internationally competitive nuclear power systems. It was developed at a fraction of the cost spent by other industrialized countries to develop or adapt the other system (see Table 21.1). How could a country of only 14 million people (in 1950), with a resource-based economy, pioneer a high-technology industry?

A retrospective of the Canadian program, comparing it with other national programs, identifies several factors contributing to this achievement, some external and some internal.

EXTERNAL FACTORS

The first factor that contributed to the unique Canadian program for nuclear energy was the wartime decision that Canada would investigate the potential of heavy-water-moderated reactors. This alone did not determine the future design of CANDU reactors, but it did give Canada a head start in the technology and, in NRX, a facility that allowed Canada to become pre-eminent in some areas of nuclear research and in the supply of radioisotopes.

The next important factor was the Canadian government's post-war decision not to develop nuclear weapons, and the derivative policy not to develop nuclear propulsion units for military purposes. In the United States, the nuclear-weapons program continued at high priority, while in the United Kingdom and France programs to develop their own independent nuclear weapons were initiated. In those two countries, graphite-moderated, gas-cooled reactors were used to produce the plutonium for the nuclear weapons; it was the design needing least development.

World War II left Europe desperately short of all forms of energy. The United Kingdom, before the discovery of North Sea oil and natural gas, was critically dependent on dwindling coal resources, not only as a fuel in its own right, but also for electricity and coal-gas. Coal shortages, exacerbated by strikes at the mines and in the distribution systems, were causing cuts in all three fuels, and three-day weeks for industry. Under these

TABLE 21.1
Total Nuclear Energy R&D Expenditures to 1985 (Billion \$US)

U.S.	31.6
France	12.2
West Germany	11.2
Japan	9.6
U.K.	6.0
Italy	3.4
Canada	2.5

circumstances, nuclear energy was seen as a boon that had arrived at just the right time to be exploited as quickly as possible. To do so, the plutonium-producing design was modified to allow the heat produced to generate electricity. France followed a similar path to develop its own design of gas-graphite power reactors.

The United States suffered no such energy shortages. Instead, the imperative there was to exploit nuclear energy for submarines with greatly extended range and quiet engines. For this application, a small reactor core is desirable and the gas-graphite design is too large. Thus the light-water-reactor (LWR) was developed, with its enriched-uranium fuel cooled by ordinary water in a compact pressure vessel. This design was feasible for the United States with its wartime uranium-enriching technology and an industry capable of manufacturing the pressure vessels

In the fifties and sixties, the United States Atomic Energy Commission investigated the potential of a wide variety of power-reactor designs, carrying several to the prototype stage. However, when it was time to offer reactors in the marketplace, both domestically and for export, it was the multinational corporations such as Westinghouse and General Electric that decided the reactor type. Westinghouse, which had developed the LWR for the United States navy, chose to stick with that type for civilian applications: General Electric, which had developed an unsuccessful, liquid-metal-cooled type for the navy, marketed boiling water reactors, a variant on the LWR in which the coolant boils in the core. Both the United Kingdom and France developed and built gas-graphite

reactors for civilian applications, but subsequently turned to the LWRs that were dominating the international market.

The way in which reactor types were developed in the three countries was an example of "industrial inertia." Once a technology is established in industry, it is easier to expand and extend that technology than to introduce a new one: there is a huge financial investment, facilities exist, expertise and experience have been gained, and there is a natural tendency to stick with the familiar and not to strike out into the unknown. This is not necessarily bad. The industry's customers – electrical utilities – are conservative, and rightly so, since they have an obligation to provide a reliable service.

INTERNAL FACTORS

In Canada there was no urgency to develop nuclear energy for either military or civilian applications. Ontario was running out of its traditional source of electricity, hydroelectric sites, but it could import plentiful coal from the United States. Consequently, nuclear energy was viewed as desirable, to reduce dependence on a foreign energy source, but only if it were economically competitive. The Canadian pioneers faced a tougher challenge than their colleagues elsewhere.

It was at this stage that individuals became important, and one in particular, W. Bennett Lewis, known as "WB" throughout AECL, was pre-eminent in defining and developing what became the CANDU reactor. In 1951, he put together three pieces of information to argue that a heavy-water-moderated reactor (HWR) could conceivably produce economically competitive power (chapter one). One of the pieces of information, the practicability of Zircaloy-clad fuel elements (containing uranium then, uranium-dioxide later), was largely due to two individuals in the United States, Admiral Hyman Rickover, who directed their navy's nuclear program and Ben Lustman, of Westinghouse's Bettis laboratory. However, the availability of this information to Canada was largely the result of the excellence of the NRX reactor in which Bettis was testing its fuel.

Lewis argued from basic principles that the capital cost of a nuclear-electric plant could never be low, so that the fueling

cost would have to be kept very low if the electricity produced was to be competitive. He identified neutron economy as the key criterion to achieving this target, and this provided the direction and the discipline for the Canadian program. Emphasis on neutron economy led to the selection of heavy water as moderator: a recognized merit rating for moderators is some thirty times better for heavy water than for light water.

It was not that Canada was alone in appreciating the intrinsic advantage of HWRs. The United States, United Kingdom, France and other countries all developed HWRs at one time or another, but subsequently dropped them, largely due to industrial inertia. The subsequent adoption of LWRs by the United Kingdom and France exemplified international industrial inertia supplanting the domestic variety. Canada, coming later, had the luxury of being able to choose what appeared to be the best design on technical and economic grounds, without the influence of extraneous political factors.

Other individuals vital to the success of the Canadian program besides Lewis were Harold Smith and John Foster, who led the engineering team that converted the scientific concept into safe and reliable CANDU reactors; and Lorne Gray, who managed the engineering support as well as providing the administrative and political expertise discussed by Robert Bothwell in "Nucleus".

Thousands more, scientists, engineers and others, worked in a highly cooperative environment at the Chalk River site.¹ This spirit of cooperation can be partly attributed to the site's geography. Although it occupies a total area of 40 square kilometres, most of the buildings are crowded into a relatively small space between the river and surrounding rocky slopes. Employees' cars had to be parked outside this inner area. The result was that buildings were close together, making visits to other divisions easy. Also, when expansion of the laboratories was agreed upon in 1960, a deliberate decision was taken to

¹ When I moved to Chalk River from the United Kingdom's equivalent, Harwell, in 1957, I was struck by the ease with which one could work with those in other disciplines, and how the reactor operators went out of their way to help experimenters.

establish a separate site in Manitoba, rather than make the Chalk River Laboratories larger than was thought to be ideal.

Individual laboratories in the United States and the United Kingdom had complements more than twice the size of Chalk River's, which was about 2,500. At the Argonne National Laboratory in the United States, one would normally drive between buildings, each of which had its own parking lot; at the Hanford site many of the buildings were out of sight from each other. The Harwell laboratory was less dispersed than those in the United States, but still less compact than the one at Chalk River: the divisions were more independent than at Chalk River, and individual divisions within Harwell tended to promote different reactor types.

There is another way in which the relatively small size of the Canadian program was beneficial. With only one company designing power reactors for non-competing electric utilities, and all of these federal or provincial crown corporations, there was not the commercial competition seen in the United States. There, if something went wrong, the first reaction by both vendor and utility would be to shut down communications between them, to protect their positions in case of legal action. In Canada, on the other hand, not only AECL and the utilities but also the component suppliers would cooperate openly in solving problems that arose. A good example was the rapid response by AECL, Ontario Hydro and the fuel fabricators to the detectable increase in fuel failures at Douglas Point in 1970, leading to the introduction of CANLUB fuel (chapter thirteen).

The same non-confrontational atmosphere existed between the regulator, the Atomic Energy Control Board, and its licensees. Most disputes were resolved at the technical level between the appropriate staffs of the regulator and the licensees, assisted by AECL as a consultant: when agreement was not reached, the five-person board acted as the final arbiter. The legal court system is not involved in licensing in Canada, as it is in the United States. For most of the period covered by this history, the Control Board employed only one lawyer. In contrast, the United States, with ten times the population, employed about one hundred lawyers in its nuclear regulating agency.

An important benefit of the nonconfrontational approach is an openness that encourages reporting of failings, both equipment failures and human errors, within the licensee's organization and by the licensee to the regulator. An open admission of failings, followed by their analysis and corrective measures, results in a continuous improvement in true safety. This mechanism for effective feedback of operating experience, combined with the defence-in-depth principle decreases the probability of a serious reactor accident.

Concentrating on a single reactor type, the HWR, was obviously one reason why Canada could achieve so much with so little. Also, because CANDU reactors have the potential to extend nuclear fuel resources by operating on advanced fuel cycles (chapter eighteen), Canada did not have to develop fast-breeder reactors with their challenging technology, considered necessary by countries with LWRs.

Yet another reason was to be found in the nature of the development programs. In several disciplines, e.g., reactor physics, thermalhydraulics, fuel development, and even Lewis's original concept assessment, AECL's design process was apparently less sophisticated than equivalents elsewhere. This was because AECL's approach was more empirical, less theoretical. With the advantage of excellent facilities, particularly the test reactors, AECL developers were able to derive simple empirical correlations from experimental results, drawing on an understanding of the physical phenomena to determine which variables would be important. The outcome was simpler and better rooted in reality.

Over the decades covered here, the design models have become much more sophisticated, so that they now differ little from those for other reactor types. The principal benefit is the ability of the designers to reduce the margins that had to be assumed to allow for uncertainties in various physical limits, e.g., critical heat flux and maximum power per unit length of element (chapter thirteen). However, much of the increased sophistication has been required to convince the regulator that various assumptions and engineering judgements were justified, largely in the area of predicting the consequences of postulated accidents (chapter fifteen). This has resulted in a

few instances where the design should be modified, but generally the increased sophistication has served only to confirm that even a very severe reactor accident would not be expected to cause serious harm to the surrounding population. Unfortunately, none of these technical assessments seems to have reassured the public.

The excellence of the NRX and NRU reactors provided another benefit, by attracting researchers from many other countries to perform experiments at Chalk River. As a result, AECL had access to detailed information on other national programs that would not otherwise have been available. The most obvious example was AECL's access to Bettis' development of uranium dioxide and zirconium alloys just in time for the redesign of the Nuclear Power Demonstration reactor.

Most of the lessons from AECL's experience are based on successful developments but it is just as important to learn from one's failures. Despite AECL's relatively small size and generally good cooperation, its most expensive failures can be at least partly attributed to poor communications and coordination. It is arguable that the pressure-tube failures due to delayed hydride embrittlement (chapter twelve) would never have occurred had the designers, the metallurgists and the commissioning engineers been fully conversant with what the others were doing. Similarly, the recent fuel failures at Ontario Hydro's Darlington Station (chapter thirteen) might have been avoided had researchers familiar with hydraulic vibrations been more involved in the design of the primary coolant circuit. In each case, no single individual had put together the various pieces of relevant information, and the design audit had failed to involve effectively all those disciplines that could potentially contribute to a complex technology.

Drawing a more constructive lesson from the same experience, one appreciates the importance of a competent champion to any project. The champion must not only be an enthusiastic leader capable of convincing others of the rightness of the project, but must also be right in the scientific direction of the project. CANDU and Lewis are the example par excellence. Had Lewis not been scientifically sound in his analyses, no amount of enthusiasm on his part would have made CANDU

successful; but it is unlikely that any design, however excellent technically, would have succeeded without his omnipresent and omniscient leadership. A disadvantage in having such an outstanding individual directing a project is that his lieutenants may rely on him too much for overall coordination, and not institute the necessary procedural checks, such as adequate design audits.

An example from AECL experience of the lack of an effective champion is the failure to develop commercially successful medical accelerators (chapter nine). Several desirable criteria for success were present, including a strong R&D team thoroughly experienced in accelerator technology, an identified market need and a well-defined mission. Competent champions for different approaches existed at both Commercial Products and Chalk River, but neither had the authority within AECL or the outstanding abilities of Lewis to bridge the division between the two sites. There was never the close collaboration between Commercial Products and Chalk River at the management level that there was between Power Projects and Chalk River, possibly because Commercial Products management personnel, unlike those of Power Projects, had never been located at Chalk River.

AECL's attempts from 1978 to produce new businesses from its technology (chapter twenty), as distinct from obtaining commercial revenues for its nuclear work and other R&D contracts, have to be judged unsuccessful in that very few new businesses were generated. Seen from the perspective of S&T policy, however, it can be regarded as a valuable experiment on a very large scale. Despite the best efforts of a successful R&D organization, and with enthusiastic support of senior management, no new means of generating industry from research were discovered. AECL's experience here deserves more analysis. Some believe that greater involvement by people with business experience would have made the difference, while others believe that trying to push out research into the marketplace is ineffective so one must look to market-pull to define the R&D needed.

This same experience illustrates another lesson important to S&T policy: failures are an integral and inevitable part of

research. If the results of a research program could be predicted, there would be no point in doing the research. In this respect, research can be compared to geological exploration, where hundreds of holes have to be drilled before one profitable mine or oil-well can be developed.

Some of AECL's programs cannot yet be assigned to either the success or failure categories. Even the CANDU program can be compared to a glass that is half-full or half-empty. CANDU reactors represent about a 5-percent share of the world's power reactors, and 10 percent of those under construction in 1993, but this is substantial compared with Canada's share for other high-technology industries. The major, decades-long program on waste management has resulted in a concept for safe and responsible disposal of fuel wastes (chapter nineteen) that is currently being assessed by a Federal Environmental Assessment Review Process Panel. Advanced fuel cycles (chapter eighteen) and small reactors (chapter twenty) have been developed to the point that they could be made commercially available quickly when the need arises: their development costs to date represent premiums paid on an insurance policy against uranium shortages and high oil prices. As for industrial irradiation, public acceptance of food irradiation could result in a rapid expansion of the market, for which AECL is well positioned with its accelerator development (chapter twenty).

Nuclear energy has acted as a pathfinder through some of the social problems of our technological world, with AECL playing a key role. Nuclear energy defined the health risks and proposed means of protection before the industry was established. The ICRP (chapter three) pioneered the simple principle, that these health risks should be kept as low as reasonably achievable (ALARA), social and economic factors being taken into account, and recommended that ionizing radiation should be assumed to be potentially harmful at any level, however low, as a prudent measure in framing regulations. Even today, guidelines for some other environmental contaminants assume that there is a threshold level below which they are harmless. Nuclear energy introduced the concept that its hazardous wastes should be contained from the start, and held in supervised storage until means for safe disposal can be agreed

upon: the industry proposed means on its own initiative. Other industries could benefit from two innovative approaches to the disposal of nuclear wastes (chapter nineteen):

- a deliberate separation of approval of the disposal concept from the selection of a disposal site, to reduce the socially paralyzing effects of the "NIMBY (Not In My Back-Yard) Syndrome."
- a siting process that does not impose a waste-disposal facility on a community, but allows potential host communities to volunteer, stating the conditions under which they would accept the facility.

In what follows, specific lessons derived from AECL's experience and relevant to S&T policy are itemized and explained. Those concerned with funding may have restricted relevance in the private sector, since AECL's R&D was very largely funded directly by the federal government. However, in Canada and elsewhere, much R&D is indirectly funded by the government, by tax provisions, military contracts, etc.

GENERAL LESSONS

Lesson #1. R&D that is intended to provide economic, industrial or social benefits (i.e., mission-oriented R&D as opposed to basic research) should be determined by the mission's requirements; it should be planned, directed and rigorously selective, with the selection based on sound principles.

In its early years, Chalk River was fully occupied operating the NRX reactor and the chemical extraction plants – activities arising from its wartime mission to demonstrate the feasibility of a heavy-water-moderated reactor – and conducting basic research. From the start, there had been an interest in the possible generation of economical nuclear electricity. However, it was not until the early fifties that NRX experience, both the behaviour of the reactor itself and of tests being performed in it, pointed the way to a practicable approach. Without disrupting the basic research, an R&D program was established with the clearly defined mission of developing an economical

nuclear-power plant. The R&D program, small relative to others, was narrowly channeled towards this mission, occasionally being redirected and modified to maintain its overall aim.

Throughout AECL's history the provision of radioisotopes for medicine, industry and research – for both the domestic and export markets – has been another example of a well-defined mission leading to a commercial success.

A more laissez-faire model for R&D management (such as that employed by the federal government's Natural Sciences and Engineering Research Council, and the Panel on Energy R&D of the seventies) makes funds available to researchers proposing peer-endorsed projects with only general relevance to stated objectives (e.g., economical energy conservation and supply). This approach may produce useful results, but lacks the drive and coherence needed to ensure that all the necessary R&D and associated activities are performed and completed to meet the overall schedule. AECL's unsuccessful attempts in the eighties to seek applications for R&D outside its specific mission demonstrated the difficulties of this approach.

Lesson #2. Potential users of the R&D should participate in formulating the broad policy for the R&D program.

In the formative days of the CANDU R&D program, the involvement of representatives of electrical utilities on the board of directors of AECL, hitherto essentially a research organization, helped to focus the program and keep it down to earth. More recently, the CANDU Owners Group (COG) has played a similar role. Greater involvement by potential users might have made more productive the search for spin-off businesses (chapter twenty).

Lesson #3. Managers of R&D programs must be vigilant in assessing alternatives and terminating those that prove disappointing.

The Canadian program explored alternatives, but on a smaller scale than other countries, and always within the program's general direction. Examples that were deliberately terminated

include the pressure-vessel design for the NPD reactor, uranium-metal fuel, aluminum-alloy sheathing, and the boiling-light-water and organic-cooled reactors. The fusion alternative was repeatedly visited, but assessments based on good knowledge of international progress concluded that a major program could not be justified.

Lesson #4. The development of modern technologies requires large interdisciplinary programs.

AECL's early strength in science continued to serve it well, particularly in providing sound guidance for the mission-oriented programs. With commitment of the first power reactor (NPD) came recognition of the need for more engineering skills: the CANDU concept would stand or fall on the performance of the fueling machines, sophisticated engineering devices in advance of anything elsewhere; excellence in engineering is the first line of defence in nuclear safety; metallurgical engineering was vital for the fuel-channel components, as was chemical engineering for heavy-water production.

AECL's program, unlike many other research programs, was constantly reviewed for economic feasibility; technical feasibility alone was not sufficient. As part of the concern for economics, technology transfer was integrated into the program from the start, with the result that researchers had to include in their jobs the training of staff attached from industry, the supervising of industrial contracts and the importation of industrial know-how, when appropriate. All these skills were integrated into a coherent program, making the AECL of the sixties unlike the popular idea of a scientific laboratory.

Most modern technologies require large, interdisciplinary programs with expensive, efficient and helpful support services, combining underlying and applied science; engineering development and design; safety, economic and market analysis; and the operation of test facilities as well as pilot and prototype plants. In Canada, universities and industry can rarely fill this role, and some excellent government laboratories have filled it. A major factor in CANDU's success has been the integration in a single program of all necessary components from basic science

through marketing. An incidental benefit of such integrated programs is that they provide the sort of "one-stop shopping" for Canadian technologies that is attractive to newly industrialized countries interested in importing new technologies.

Lesson #5. Even in a mission-oriented program there is need for research at a fundamental level.

AECL found its underlying-research component invaluable. Initially, it helped attract suitable researchers. Later, it provided a sound understanding of relevant phenomena, a "mobile reserve" of people capable of tackling from first principles the inevitable problems that arise in a major endeavour, intelligence about discoveries elsewhere through the international science subculture, a means of evaluating potential programs, and the basis for future advances.

Lesson #6. Applied research is an essential step in going from the international pool of basic research to the development of technologies for society's benefit.

Despite AECL's strength in basic research, most of the basic discoveries leading to the CANDU reactor were made elsewhere. They include nuclear fission, neutron moderation, heavy water, zirconium alloys and uranium dioxide. AECL was weak in basic patents but strong in adopting and adapting others' discoveries. This was achieved through an informed awareness of international developments and by application of the knowledge to Canadian objectives through applied research.

Lesson #7. A substantial R&D effort is required to absorb and maintain imported technology.

At first, the decision to produce heavy water in Canada seemed to involve no more than importing an already developed technology from the United States. In practice, however, this proved to be a very expensive oversimplification. The design of the plants, and the conditions under which they were to operate, differed in significant respects from those in the United States.

The problem is exacerbated when the technology has been established empirically, without a broad understanding of the underlying phenomena. Reliable heavy-water production was eventually achieved, largely through competent management of the plants, supported by an R&D strength in Canada (chapter seventeen).

Uranium-dioxide fuel in zirconium alloys is a more positive example, for which the original technology was imported at an earlier stage and into a strong R&D capability. This led to the development of CANDU fuel, with its excellent performance. AECL and its associated industrial companies (chapter thirteen) eventually contributed much more to both these technologies than had been imported.

Lesson #8. There is an ongoing need for R&D far beyond the innovation stage.

The unforeseen in-service rupture of pressure tubes is the single biggest failure of the CANDU program. While every effort must be made to avoid or mitigate such failures – and several detailed lessons have been learned from the incidents – the fact is that in any multi-billion dollar endeavour, such as a major energy system, some such failures must be expected. Their likelihood can be minimized by sound engineering, the application of familiar materials wherever possible, well-established, reputable manufacturers, quality assurance programs, etc., but provision must be made for their occasional occurrence.

In the response to the pressure-tube failures, AECL's multi-disciplinary laboratories proved invaluable: had they not been able to diagnose the fault rapidly, it might have been necessary to shut down all CANDU reactors as a precautionary measure.

Any technology that is stagnant will soon become obsolete. To remain competitive, a technology must continue to develop and improve itself, thus requiring ongoing R&D. When Ontario Hydro's Pickering nuclear station proved so successful, it was difficult to convince the funding authorities and the public that much R&D remained to be done, not only in waste management but also in continuing development of the CANDU design, to reduce costs and accommodate changes in the market and in

regulatory requirements. CANDU's foreign competitors were continuing to devote large sums to the development of their products.

Lesson #9. An R&D program has to plan from the start for the transfer of technology to industry.

Laboratories may produce the ideas for new products, but most do not produce the products themselves. AECL developed and designed the CANDU reactor, but Canadian industry manufactured the hardware. The transition from laboratory to industry is a constant subject for debate in discussions of S&T policy. AECL established several techniques to ensure an efficient transfer of the CANDU technology to industry, including development contracts, attachment of industry's staff to the laboratories and cooperative work on standards. CANDU fuel provides an excellent example of efficient technology transfer.

However, technology once transferred cannot be recovered, so financial and other arrangements must be agreed upon before the transfer. The development of CANDU fuel took place in the sixties when government policy was that government laboratories, such as AECL's, should simply assist in the development of Canadian industry, without obtaining any return of public funds. Since then, the policy has changed to one of user-pay. Although some hundreds of millions of dollars worth of fuel based on AECL developments is used each year, AECL derives no licensing fees to reinvest into further development.

Lesson #10. Some R&D is needed for the development of S&T policy.

While a major purpose of S&T policy is to determine what R&D should be done, R&D has an important role to play in determining S&T policy. Should the government devote substantial resources to developing a new technology? Usually, an informed decision requires the performance of at least exploratory R&D. Often, the open literature will give an over-optimistic impression of the new technology's potential,

not revealing the important problems or the probable costs. A large R&D program by AECL in the seventies on nuclear-waste disposal provided the basis for a decision by the federal and Ontario governments to pursue deep geological burial for the disposal of nuclear-fuel wastes. Outside AECL's sphere, renewable energy sources are a good example: much exploratory R&D is needed to establish technical feasibility and economic promise before making these technologies part of an S&T policy.

FUNDING LESSONS

Lesson #11. Even when all foreseen components of an R&D program have been allowed for in the budget, a substantial contingency fund is essential.

The nature and magnitude of the R&D needed for something like CANDU is generally not well understood; even the researchers most involved have habitually underestimated requirements. To provide the necessary heat-transfer data for CANDU required several multi-million dollar engineering rigs not foreseen in the early days. If public trust is to be deserved, R&D managers must provide more realistic estimates. The public sees gross cost overruns as evidence of incompetence or even dishonesty.

Uncertainties in cost estimates are inherent in the research process and must be recognized. However, past experience over many industries provides some guidance in what R&D expenditures have to be expected, and therefore what contingencies are reasonable. Low-technology industries, with which Canada is most familiar, devote only a few percent of product cost to R&D, while for high-technology industries, such as electronics and aerospace, the R&D expenditures may be around 20 percent.

Lesson #12. Funding for mission-oriented R&D should be determined by the mission's objectives: that for basic research should be a percentage of Gross Domestic Product (GDP) concordant with that devoted to basic research by comparable countries.

The S&T debate has been oversimplified by a demand that the government increase spending on R&D to some arbitrary percentage of GDP, as if that alone would ensure that S&T provided all the benefits expected of them. Government-funded applied R&D is wasted unless it is performed as an integral part of the government's industrial and social policies that are prepared to exploit successful R&D. A defensible S&T policy presupposes clearly formulated industrial and social policies. AECL experience suggests that finding an efficacious S&T policy will require changes in attitude at all levels.

Part of the trouble is the habit of talking of "S&T" and "R&D" as if these were homogeneous topics. It is true that one can trace a continuous path from pure scientific research to the final product, but the nature of the activities and their needs vary greatly along the way. The greatest distinction along the path is determined by the motivation for the work, between pure, or curiosity-driven, research on the one hand, and mission-oriented programs on the other. For the latter the mission should be defined by industrial and social policies, so that objectives and beneficiaries are clearly identified: the R&D activities are a necessary part of the program, so the mission manager should determine the funding out of the program budget which should contain an appropriate allowance. R&D is one means, not the end.

Pure research, while vital for the advancement of mankind in the long term, rarely benefits directly those who fund it. European research of the nineteenth century provided the basis for United States industrialization in the twentieth century, and Japanese industry is now benefiting from the science the United States performed earlier in this century. Canada, as an affluent country, has an obligation to support curiosity-driven pure research in proportion to its relative wealth, as an altruistic activity.

Most of AECL's R&D was mission-oriented, to apply nuclear energy to benefit Canadians. Some fundamental research was performed to support these missions, to provide a better understanding of the underlying phenomena (e.g., radiation-induced creep), and for other reasons explained under Lesson #5. AECL, acting within the mandate under which it was

established, deliberately devoted a fraction of its budget to curiosity-driven pure research in areas where it had unique resources; e.g., the NRX and NRU reactors for neutron diffraction.

Lesson #13. Accountability is necessary, but over-accountability is counterproductive.

An increasing recognition over recent decades that researchers must be accountable to those who provide their resources, ultimately the Canadian public, is a healthy trend. However, AECL experience in the eighties was that requirements for planning and accountability reached a level where a sizeable fraction of the total effort was devoted to the activities involved. The climate tended to discourage unplanned initiatives and cooperation. A judicious three-way balance must be struck between accountability, responsibility and trust.

Lesson #14. Program commitments should match the resources available: when funding is cut, programs may have to be eliminated, not schedules simply extended.

R&D programs have to be stopped as well as started. How this is done is important to staff morale. AECL found that displacing a low-priority program by a high-priority one was more effective than simple cancellation. This approach helped to maintain staff enthusiasm and support, something that is vital to a healthy R&D program.

Not all AECL experience was so satisfactory, however: when funding was reduced, cuts were sometimes imposed across the board; more selective suspension of whole programs would have resulted in the remaining programs being more productive. Also, during some cuts, a reluctance to reduce personnel levels resulted in too small a fraction of the budget being devoted to new equipment and capital facilities.

Lesson #15. The source of R&D funding needs careful and explicit consideration.

Some R&D activities may be appropriately provided from a levy on the beneficiary, while others may be a legitimate charge against the public treasury. These latter can include activities where the potential beneficiaries cannot be identified, and where work is needed to determine government policies.

LESSONS ON PUBLIC PERCEPTION AND ACCEPTANCE

Lesson #16. A research institute and its staff must be prepared to explain and defend its work to the public.

AECL, in common with the rest of the science and engineering community, was slow to recognize that a new technology, however sound technically, is not acceptable until the public accepts it. Even before the CANDU program started, AECL was doing research to protect people and the environment, simply regarding this as an obvious responsibility, but failing to publicize the fact. At that time, the public generally regarded technology as beneficial, and was content to leave its approval and control to anonymous government officials and elected representatives.

Since then, the public has become suspicious of technology and wants a say in important decisions. Thus, to be accepted, any new technology must not only be fully defensible, it must be defended and even promoted. Furthermore, the advocacy should not be left solely to public relations people who usually lack an exhaustive understanding of the subject. Their work must be complemented by articulate scientists and engineers

who are willing to listen as well as talk. Public relations must be a two-way process.

Lesson #17. Researchers must identify those who will benefit from their research, then sell themselves, either direct or through others.

R&D practitioners, both institutions and individuals, have to accept that the world does not owe them a living, paying them for what they enjoy doing. They have a responsibility to explain themselves to their patrons, to show why they should be supported, and to answer concerns. This attitude was foreign to most R&D institutions in the fifties. It is now an integral part of AECL's corporate culture.

Lesson #18. An R&D program for a new technology should include the necessary research on health and environmental effects.

The scope and character of many technologies are such that, without proper safeguards, they can produce serious effects on health, or major disruptions of the environment. Because of the known effects of radiation, from the earliest days of nuclear research, care was taken to protect the public and workers from undue exposure to radiation. The lack of a similar appreciation of the hazards of other technologies has resulted in serious impacts on the global environment, so that technology in general has got a bad reputation.

ACKNOWLEDGMENTS

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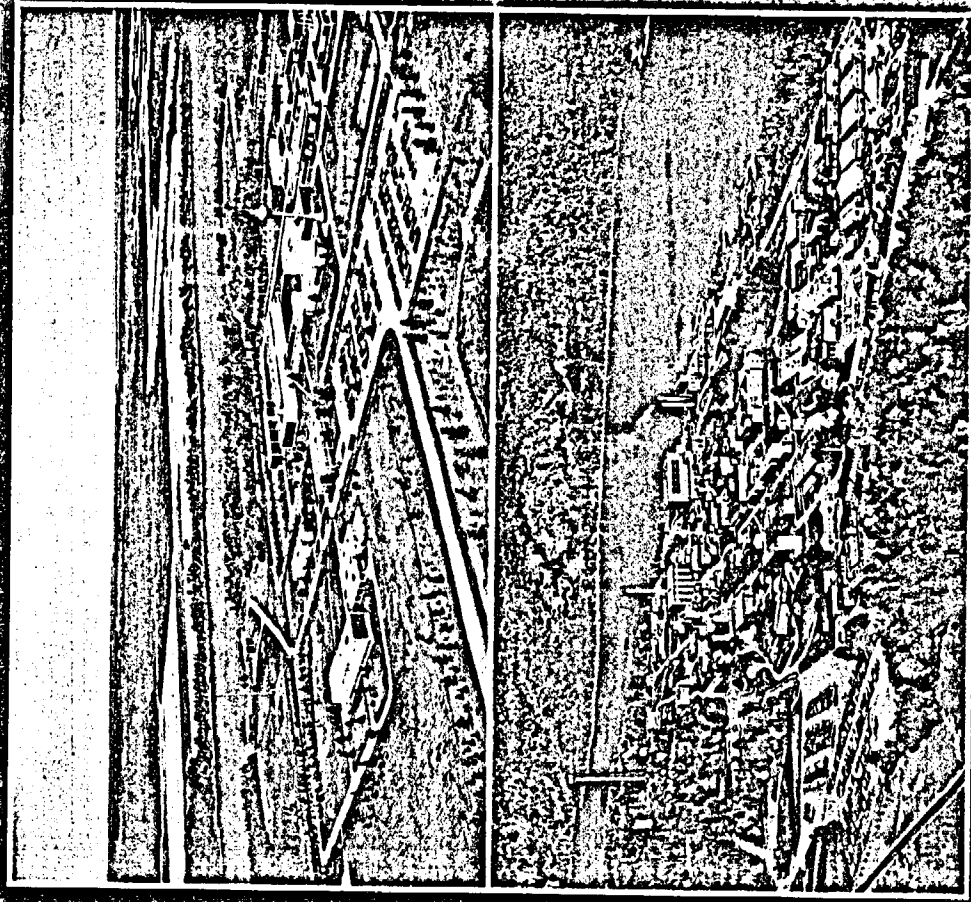
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