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Editorial

Alan Wyatt
Editor

In this, our second issue, the *Nuclear Journal of Canada* includes its first paper on the Chernobyl accident. Although the accident took place over a year ago, on 26 April 1986, we make no apology for scientific papers taking a considerable length of time to appear in print. The process of ascertaining the facts, checking them, analysing them for pertinent lessons, submitting a scientific paper for peer review, and journal production will take from 12 to 18 months and maybe more. This may not satisfy the desires of the mass media, but seeking the truth is more important, in the long run, than an eye-catching but wildly inaccurate headline.

The paper by Professor Rogers in this issue is a useful overview of many of the major implications of the Chernobyl accident. In our next issue there will be a more detailed technical evaluation by V.G. Snell and J.Q. Howieson. It is expected that these papers will generate an interesting discussion.

The Chernobyl accident also raises many interesting questions that are not strictly in the scientific areas. Since nuclear power is a global energy source, any problems with it have global implications. If there were major shortcomings in the RBMK design and operating procedures, why did the professional critics of nuclear power not draw proportionate attention to them – proportionate to the attention that they gave to their perception of the shortcomings of designs in the Western World?

The Three Mile Island Unit #2 accident in 1979 had negligible health implications, but a side-effect was a delay of six years in starting up the adjacent undamaged and uncontaminated Unit #1. At Chernobyl, Units #1 and 2 were decontaminated and restarted some six months after the accident to Unit #4. The adjacent Unit #3, which shared the control room with the destroyed reactor, is scheduled to go back into service about now, and a new fifth unit will be in service by year end. To what extent are these differences between TMI and Chernobyl due to differences in political systems, to the Soviet need for electric power, and to the actual level of hazard involved in working on the Chernobyl site?

About 30 lives were lost in the actual Chernobyl accident, mainly among firefighters. Best estimates are that, over the course of the next 70 years in a population of about 70 million, the number of additional cancer deaths resulting from the release of radioactivity is in the range of 2,000 to 5,000 people. In that same period, in that same population, many more people will die from tobacco and alcohol-related diseases, and more than one thousand times as many will die from 'natural' cancers. The Soviets have already announced a major program of health monitoring of the population of European Russia in order more accurately to assess the effects of the accident. From the evidence of similar programs carried out over the past 40 years on the survivors of the Hiroshima-Nagasaki bombs, the resulting early diagnosis of other medical problems will probably save more lives than will be lost from the effects of the radioactivity released. This should surely raise some pertinent questions of the value, or lack of value, that is placed on preventive medicine in countries that spend billions on weapons.

Steady day-by-day carnage on our highways does not seem to provoke much soul-searching for remedies. Even major single disasters that kill hundreds in a few minutes, such as aircraft crashes or ferry sinkings, rarely rate headlines for more than a few days. Even a nuclear accident that injures nobody often attracts more attention. Although many in the nuclear industry feel somewhat paranoid about this, my own opinion is that it is a reflection of the exceedingly high standards of safety set and practiced throughout the nuclear industry. When any accident occurs we try harder to learn more from it – and that is as it should be.

Since our first issue went to press we have been saddened to learn of the deaths of Dr W.B. Lewis, on January 10, at the age of 78, and of J.L. Gray, on March 2, at the age of 74. Both died in Deep River, close to the Chalk River Nuclear Laboratories that were the centrepiece of their professional lives. Their

departure marks the end of an era. Neither of them was involved with the Canadian work on the Manhattan Project at the end of the Second World War. Both were architects of 'atoms for peace,' nationally and internationally.

Both men were members of the Canadian Nuclear Society. Much will be written about them when the histories of Canada's involvement in the nuclear age are written. The highlights of their lives are given on the following pages; however, I would like to add my personal tribute to these two great Canadians.

My first contacts with Dr Lewis occurred in 1958. I was a recent immigrant from the UK, working, in the very early days of the Douglas Point Project, on trying to develop a new steam cycle suitable for the commercial CANDU reactors. I was surprised, and somewhat alarmed, to start receiving lengthy memos from a remote figure at Chalk River, whom I had never met, raising detailed questions about steam cycle thermodynamics and the design of a steam turbine plant. I was yet another recipient of the probes by Dr Lewis into every aspect of the plant design. He wanted the best and he was determined that you were going to produce it. Shortly afterwards he cornered me in my cubby-hole of an office in order to carry on his questioning at first hand. His keen interest certainly spurred me on to do my best, and for the next 15 years, up until his retirement, even though for much of that time I was pursuing a career outside the nuclear industry, he corresponded with me, forever seeking refinements and improvements in cycle efficiency. His grasp of the fundamentals of science and technology was awe-inspiring. He was a true scientific genius, in the full sense of the word.

Although I first met Lorne Gray in the discussions on the selection of the turbine generator for Douglas Point, it was not until the mid-sixties that I wa him in action at close quarters. At that tim working directly for AECL on the Gentilly-1 and had been the principal force behind recon ing the selection of a high-speed (3,600 rpm) generator instead of the almost universal lo (1,800 rpm) machines. In the usual fiercely c tive bid negotiations for these large contrac had become a political issue and had been rais ministers and deputy ministers in Ottawa. I wa to attend a meeting with Lorne Gray to settle the morning Lorne took me through every s every possible question on the entire contract afternoon a meeting was held with the DM o from every department in Ottawa evenly r connected with the contract – a horde of man Apart from my being asked to confirm so minor technical points, Lorne answered ever question concisely and accurately, and the award was approved as recommended. I was im that a topic that had involved my full atten several months, and had a host of nuances an cations, could be handled in such masterly fa

I would hope that these two anecdotes il how fortunate AECL and Canada were to ha outstanding scientific and administrative lead The conjunction of the unique talents of th men, over the quarter-century 1948–1973, was factor in establishing the CANDU system as a Canadian achievement on the world scene. T paper in this issue was prepared for the s sponsored by the Canadian Nuclear Society recent Engineering Centennial Conference in real. In it Lorne Gray details some of the key d in the development of the CANDU program.

Wilfred Bennett Lewis

1908 June 24	Born Castle Carrock, Cumberland, England
Education	Clare House School Haileybury College Cambridge University
1930–39	Cavendish Laboratory, Cambridge Worked on alpha radioactivity with Lord Rutherford Worked on nuclear disintegration by particles accelerated by high voltage and on the construction and operation of the Cambridge cyclotron
1939–46	On loan to the British Air Ministry. At end of the war was Chief Superintendent of the Telecommunications Research Establishment
1945	Fellow of the Royal Society (London)
1946	Appointed Director, Division of Atomic Energy Research, NRC at Chalk River
1952	On formation of Atomic Energy of Canada became Vice-President, Research and Development Fellow of the Royal Society of Canada
1955–64	Director of the American Nuclear Society (President 1961–62)
1963	Appointed Senior Vice-President (Science) of AECL
1966	First recipient Outstanding Achievement Award of the Public Service of Canada
1967	U.S. Atoms for Peace Award
1968	Companion of the Order of Canada
1971	Honorary Fellow of Gonville and Caius College, Cambridge University
1972	Royal Medal of the Royal Society of London
1973	Retired from AECL. Appointed Distinguished Professor of Science, Queen's University
1981	U.S. Department of Energy Enrico Fermi Award

In the international sphere Dr Lewis was a member of the Scientific Advisory Committee to the Director General of the International Atomic Energy Agency and Canadian delegate to the Scientific Advisory Committee to the Secretary General of the United Nations. He was also active in the organization of the various UN Geneva Conferences on the peaceful uses of nuclear energy.

James Lorne Gray

1913 March 2	Born Brandon, Manitoba
Education	Winnipeg Public School Saskatoon High School University of Saskatchewan B. Eng. 1935 M. Sc. (Mech. Eng.) 1938
1938	Canadian General Electric Test Cou
1939	University of Saskatchewan – Lectu Engineering
1939–45	RCAF (retired as Wing Commander)
1945–6	Associate Director-General, Researc Development Division, Departme Reconstruction and Supply, Otta
1946–8	Montreal Armature Works Limited, Montreal
1948	Scientific Assistant to the President, National Research Council
1949	Chief of Administration, National Re Council – Chalk River Project
1952	General Manager – Atomic Energy Canada Limited
1954	Vice-President, Administration and Operations, AECL
1958–74	President, AECL
1961	D. Sc. University of British Columbi LL. D. University of Saskatchewa
1962–73	Member, Board of Governors, Carle University (Chairman 1970–73)
1969	Appointed a Companion of the Ord Canada
1973	Awarded The Professional Engineer Medal by the Association of Profe Engineers of Ontario

Early Decisions in the Development of the CANDU Program

J.L. Gray
Atomic Energy of Canada Limited,
Ottawa, Ontario

Abstract
This paper identifies about twenty major decisions of the Canadian Nuclear Power Program from its inception in 1942 to maturity – the decision of Ontario Hydro to complete the Pickering Nuclear Power Station and proceed with a major nuclear power installation program. The decisions are discussed briefly.

This paper reflects my views of the early years of the CANDU program to the point of program maturity. I have drawn heavily on the research work of Professor Robert Bothwell – who is writing a book on Atomic Energy of Canada Limited (AECL) – particularly for the period up to the incorporation of AECL, when I became directly involved. I certainly learned much from having access to Bothwell's early drafts of the period 1942 to 1952. I have also found *A History of the Atomic Energy Control Board* (AECB) by Gordon H.E. Sims to be a valuable reference.

C.D. Howe

The first 'key decision' was taken by Prime Minister W.L. Mackenzie King when he appointed C.D. Howe as minister of the Department of Munitions and Supply and, in October 1944, minister of the Department of Reconstruction – he was also the chairman of the Privy Council Committee for Scientific and Industrial Research, which, incidentally, almost never met. Howe, a former professor of engineering and a successful business man, had no trouble understanding nuclear fission, and he was familiar with the management and promotion of large and expensive projects.

His stature in the government and his more-than-persuasive powers allowed the nuclear program to obtain support during its early years, and he was able to keep his finger on the three main areas of importance: uranium, nuclear research and development, and nuclear regulation and control. Eldorado, with its uranium operations; the National Research Council (NRC) and later AECL, with the nuclear development program; and the Atomic Energy Control Board (AECB), the regulatory agency – all reported to Howe.

I need say no more about Howe, but I suspect that without him the nuclear development program would have died in its first few years, and probably have been reborn in the 1950s with, hopefully, but not certainly, heavy water reactors in the program.

One of the earliest actions that produced a key decision resulted from the war in Europe and the fall of France. Both the British and the French were aware in 1939 of the possibility of a nuclear weapon as a result of the work of the Curies and others. Hans von Halban, an Austrian, was working with Frederic Joliot-Curie, along with Lew Kowarski, a Russian, and the French team had advanced to the point of outlining an arrangement of uranium and a moderator that could sustain a chain reaction – a reactor. The team recognized the worth of heavy water and cornered the world's supply that had been produced in Norway, 185.5 kg.

With the invasion of France some of the scientists decided to depart, taking with them their heavy water. Halban and Kowarski arrived in England in 1941, and they were soon part of the British nuclear program. They joined other scientific colleagues at Cambridge University's Cavendish Laboratory. There was no shortage of scientists, both British and from Europe, but Halban and Kowarski were the main proponents of a reactor system using heavy water and uranium. Others were more involved in separating the fissile atom of natural uranium, U-235. They knew about plutonium but seemed to feel that a nuclear weapon was more likely to be achieved, during World War II, with U-235 than with plutonium, and the reactor development program was not of the highest priority.

The entry, by December 1941, of the USA into the war, both in the Japanese and European theatres, changed the tempo of the US nuclear development program. General Leslie Groves led the Manhattan Engineering District, and by the winter of 1942 US nuclear research was leading the way for the allies.

Hans von Halban visited the US in 1942, and recognized that the effort and facilities in the US outclassed

those of his group at Cambridge. It was suggested that the British might send Halban's group to the US – or at least some individuals expert in heavy water developments. The British director of their program, Wallace Akers, was not too impressed with the US proposals, due partly to patent rights clauses, and Canada was considered as an alternative. But neither was Akers very attracted to locating the British team in Canada.

When details of the US offer were studied, Halban realized he would only be a part of a US laboratory, with no overall authority. Canada had a small program at NRC, with Dr G.C. Laurence experimenting with uranium and coke – a rudimentary pile – and Halban could expect to be placed in charge of a joint project; this was enough for him to press for a Canadian location, which would still have access to US information and supplies. The outcome was an agreement between the British and Canada, with the support of the US, that moved the British Heavy Water Project from Cambridge to Montreal in late 1942 – a key decision. It was to be headed by Halban, but without Kowarski, and included among its personnel several British, as well as Canadians and some 'refugees' from Europe, which posed a security question mark for the US; and rightly so, as we found out from the Gouzenko affair. This was the start of heavy water-moderated reactor research and development in Canada – an 'Anglo-Canadian Project' with support from the US.

Hans von Halban, as director of the Montreal nuclear laboratory, had some good points to offer the project, but he also had some deficiencies. He was an able scientist from an aristocratic family, and he had strong personal ambitions to lead an important nuclear program. He had a high opinion of his abilities, but it turned out that he lacked the personality and abilities of leadership and management.

Halban was certainly persistent and believed in the heavy water natural uranium system, and knew it was worth fighting for. The project faced severe difficulties in 1943 and 1944. The US and Britain were quarrelling, Groves was dissatisfied with the progress of work at the laboratory, the morale of the staff was very low, and the death of the lab a distinct possibility. Halban's work in France coupled with the work of Kowarski – the foundation for the reactor project – was being questioned by the US Manhattan Engineering District, but stood up under review. Halban's insistence on the merits of the heavy water-moderated natural uranium-fuelled reactor system resulted in its survival, but he was not around to see the first reactor project committed. Some feel that it was Halban's persistence that saved the project, and if so his appointment rates as a key decision. While he created the lab, he was almost the death of it.

The problems of management and program direction at the Montreal laboratory concerned n the British and Canadians, but the US as reactor design and construction program ha proposed with a price tag of about $50 mil would be at Canada's expense. The British anc icans were not getting on too well, partly bec Halban and others of the British team in M but also because of personality conflicts amon personnel on both sides of the Atlantic.

The British came close to abandoning the but instead of pulling out appointed a new d Professor John Cockcroft, who arrived in Apr and things began to change rapidly. This a ment, leading to Cockcroft's arrival in Montre certainly a key decision. He took command a gated Halban to head of physics. At the sam Groves gave his consent to building a reactor though he knew it was obviously not going tribute to the supply of plutonium for the war. So the military purpose of the Anglo-Ca project had disappeared. General relations w US improved – as they had to.

Halban eventually got himself into trouble by ing security procedures to report to Juliot-Curie progress of work in North America, both Ca and US. This action disturbed the British, the dians, and particularly the US. Halban was access to the labs and soon returned to Englan post at Oxford.

Cockcroft was a highly respected scientist able leader, quiet but firm. He brought Lew Kc to join him.

Following discussions with the US and the tory staff it was recommended that a 10 MWt geneous reactor moderated with heavy water, with natural uranium and cooled with light w built – NRX (National Research Experiment) – ce a key decision. The program proceeded with the engineer and construction manager and Brace as the contractor, in July 1944.

The selection of Chalk River as the reactor s a very good one, and some may view it as a ke sion – the only 'key' I can see is that it is in the p where Ontario Hydro is located, a utility that be interested in nuclear power, still a dream scientists' minds.

In November of 1945, much to the surprise Mackenzie, president of NRC, it was learne newspaper reports that Cockcroft was being ap to head the new British Atomic Energy Establis at Harwell. Attempts were made to keep him i ada, at least until NRX was operating. He eve left in September 1946.

The British government had taken a firm deci have a full-scale nuclear program of reactors, ch separation facilities, and isotope separation independent of Canada and the US and loca

Britain. This meant that the Anglo-Canadian agreement that was based primarily on the development of a means of producing plutonium for weapons was no longer valid, and that the Anglo-Canadian program would be terminated. Canada would have to proceed alone with minimal help from the British and with possible support from the US. Although this key decision was taken by the British Government, it resulted in the Canadian program for nuclear power development.

Many of the staff returned to the UK, but some stayed on at Chalk River and made major contributions to the Canadian program. Before Cockcroft left he helped arrange for a replacement from Britain, Dr W.B. Lewis, who had had an outstanding career in radar development. His confirmation as Head of the Scientific program at Chalk River was certainly a key decision. He arrived in Chalk River in September 1946 to run the research program.

Dr Wilfrid Bennett Lewis, C.C.

As I have said before, if I had to pick one person who contributed most to the success of the Canadian nuclear power program it would be W.B. Lewis – having said that, let me add that he could not have had the successes he did without the help of scientific colleagues and, more important, some very good engineers translating his ideas into practical plants.

Lewis was undoubtedly a brilliant scientist – he lived for his work, day and night; he carried a large briefcase to and from the workplace and worked late in his study at home and used his slide rule as well as any engineer. He was particularly noted for his reverence for neutrons – one wasted meant another fissile atom had to be added. The preservation of neutrons was a religion to him, and consequently he had many battles with the design engineers or materials development staff, insisting on minimum material in the reactor cores. He did not always win, but he certainly made everyone think seriously of possible ways to reduce neutron-capturing material in the core.

Lewis, as an outstanding scientist, had his own targets and pursued them intensely. I am sure he is considered as having been one of the world's leading nuclear scientists. He received many honorary degrees and awards – the Major Achievement Award in 1963 from the Public Service of Canada, the Atoms for Peace Award from the United Nations in 1967, the Royal Medal from the Royal Society in 1972, and the prestigious Fermi Award from the US Government in 1981 – all indicating the place he held in international science.

There is absolutely no doubt in my mind that Lewis' attitude towards the value of a neutron and his intense commitment to seeing they were not wasted has resulted in a natural uranium-fuelled reactor, moderated and cooled with heavy water, that is about as close to ideal as is possible in a practical reactor: CANDU. It has resulted in extremely low fuel costs for these reactors – an essential element in the success of the plants in operation.

Lewis spent a lot of effort developing an organic-cooled heavy water-moderated reactor system fuelled with thorium. He showed that in a well-designed thermal reactor with enriched fuel to start, he could approach breeding through the production and recycling of U-233. Theoretically, he was probably right, but the cost of a program to prove his case was estimated by some to be $500 million and there was no guarantee of success. CANDU units were performing very well, and Ontario Hydro was not interested in a major role in developing another system, although with thorium as the fuel it would extend fuel reserves for centuries.

Lewis called his system the 'Valubreeder.' He did his best to get the Valubreeder approved, but after much deliberation within the AECL Board and among various government departments in Ottawa, it was decided to terminate the project – certainly a key decision, and a negative one. Lewis was very disturbed, but he settled down to being concerned with the CANDU-BLW system for Gentilly I.

NRX-ZEEP-NRU

The decision taken in Cockcroft's time to build research reactors was a key one, and NRX, followed by NRU, became the backbone of the research and development programs at CRNL. The approval of ZEEP followed NRX, but it was such a simple reactor that it was operating by 22 July 1945. It is rumored that Cockcroft wanted it to keep Kowarski busy, and it was used extensively to check calculations and theories being used for NRX, particularly lattice arrangements. It was the first reactor to go critical outside the US.

NRX was a very solid accomplishment. It was by far the best R&D nuclear facility in the world, with a high neutron flux and a relatively large core. It had excellent neutron beams for basic research and very good facilities in core for fuel and materials research and development. The loops – a small reactor inside a reactor – were particularly valuable and were used extensively by the British and, more particularly, the USAEC and, of course, AECL, particularly for fuel, materials, and coolants development.

NRX went critical in 1947 at 10 MWt and was soon raised to 20 MWt and later to 30 MWt, and was used continuously to December 1952, when we did what we called a very advanced experiment that required a shutdown for about a year. We learned much from the accident, and the rebuilding program, and now have a 40 MWt reactor. It was just recently given an award by the American Nuclear Society as being the oldest reactor in operation today.

NRX produced Pu in the natural uranium metal fuel, and some was sold to the USAEC under contract, and

since the life of the reactor was in doubt – some thought perhaps five years – a proposal was put forward for a second reactor of greater power to make money selling Pu and have improved research and development facilities in the reactor. NRU (National Research Universal), a 200 MWt reactor, was approved on 20 December 1950. It went critical on 3 November 1957, and Canada once more had the best research and development reactor in the world. Unfortunately, it did not make much money selling Pu, but did sell isotopes that were distributed throughout the world by AECL, and it proved to be a superb reactor for research, and with its in-pile loops that could really test, at full scale power, reactor fuels and coolants, it advanced the fuel and materials program greatly and was of particular interest to the USAEC for joint programs.

The agreements to sell Pu to the USAEC, the sale of isotopes and the prospects of commercial arrangements with the USAEC to support joint programs centred at CRNL, and the general view that CRNL was getting more commercial led Howe to decide that the existing form of the organization – a division of NRC – was not appropriate for the future, and this resulted in the incorporation of AECL. Certainly a key decision, and a good one.

Atomic Energy Control Board (AECB)

One key decision – and in my view one of the more important ones – was that of the Government of Canada to enact the Atomic Energy Control Act of 31 August 1946, which set up the Atomic Energy Control Board (AECB) to assist the Government and '... to make provision for the control and supervision of development, application and use of atomic energy and to enable Canada to participate effectively in measures of international control of atomic energy.' It should be noted, particularly, that the AECB is a federal agency and its mandate covers the whole of Canada.

The mandate to the AECB to 'control' and to 'supervise development and application' was a very broad one that would require an amendment to the Atomic Energy Control Act within a few years.

In reading Gordon Sims *A History of the Atomic Energy Control Board*, which is recommended to those interested in atomic energy control procedures exercised by the AECB under the Act, I was surprised to learn that the Chalk River Nuclear Laboratories were to be 'controlled' and 'directed' by the AECB, through a Government directive, dated December 1946, to 'assume the control and direction of the Chalk River (NRX) Project.' In actual fact, the AECB requested the National Research Council to take over the operation and management of the project on behalf of the AECB – a sensible decision.

One should note that, during the early years of the AECB, there were only two full-time professional employees: a secretary and legal advisor, and a scientific advisor, both very able people, but hardly an organization to assume control of a project of the Chalk River, with major items of equipment an including two chemical separation facilities an important research and development program. ing to Sims, the AECB advised NRC of the policy 'as to objectives of operation priorities tion to research and development, capital e tures, communication of information to the B

The AECB had part-time Board members an time president, Dr C.J. Mackenzie – who v Acting President of NRC (confirmed in 1948) 1961. The fact that Mackenzie was the Chie tive officer of both NRC and AECB helped m arrangement possible. There was no doubt al wording of the Atomic Energy Control Act. T was required to control, to supervise devel application, and use of atomic energy, but structural organization really to undertake sponsibility. In fact, as Sims says, very little and direction was exercised over the Cha research project.

However, the AECB did an excellent job of d ing regulations relating particularly to health an Being short on staff they resorted to the A Committee formula, and internal staff stren capability gradually increased after the appo of Dr G.C. Laurence, with his serious person est in reactor safety, as president on a full-tim The great majority of the early work was don Committees made up of representatives of fed provincial departments, particularly Health a fare, along with experts in the fields of atomic from NRC, AECL, universities, industries, and the

There was a requirement for development lations and procedures for handling applicat licenses, ranging from radium treatment appl in medicine, through cobalt sixty beam therap to nuclear electric power stations of varying si complexity. In my view, the AECB have done a lent and thorough job in this very important their responsibility. I must say there were time those of us who were trying to get nuclear plants built and into operation wondered v AECB could be so stubborn, especially when us in the business had much more experien reflection, I am sure our plants have benefitt the input of the AECB committees and staff re our submissions, and, moreover, a safety revi qualified and responsible independent licensing is more acceptable to the public.

The decision to increase AECB staff, started rence and expanded by succeeding presider certainly a key decision, and a good one. As the power program expanded, the work load on t sory committees, particularly the Reactor Safe sory Committee, became intolerable and the A had to take on more and more work. By 1980, numbered about 200, with well-qualified, expe

personnel, quite capable of meeting the Board's responsibilities without the use of the RSAC.

Effective 1 April 1952, an agreement was reached *via* the Appropriations Act, authorizing the transfer of the Chalk River establishment from NRC to a crown company, Atomic Energy of Canada Limited (AECL). AECL had been incorporated under Part 1 of the Companies Act 1934, on 14 February 1952, but AECL was set up pursuant to the Atomic Energy Control Act, which allowed the AECB to procure the incorporation of companies to exercise on behalf of the Board such power that had been conferred on the Board.

Although I was appointed General Manager of AECL in 1952, with Chalk River the main project, and with Dr Mackenzie as President (and we had an independent and excellent board of directors), I just realized from reading Sims that AECL was 'set up to administer the project (CRNL) on behalf of the Atomic Energy Control Board.' I have no recollection of AECL reporting to the AECB, although financial estimates were submitted to the Government *via* the AECB. Once again, with a joint-president it was possible to operate with little, if any, friction.

There is no question that AECL was required to report to the Government through the AECB, and Mr Howe stated categorically that AECL '... is a creature of the AECB and AECB is in charge of atomic works in Canada.'

Howe's right-hand man and 'enforcer' for nuclear energy development in Canada was C.J. Mackenzie; as president of AECB, NRC and, for two years, AECL. It was undoubtedly a closed-shop organization, with Howe calling the plays and Mackenzie executing them up to 1953, when he retired as president of AECL. The supposed control and direction of the research and development program for nuclear energy by the AECB was really a myth. The AECB had virtually no staff and no organization to exercise the responsibilities conferred on them by the Act. The arrangement of Howe and Mackenzie, and the positions the latter held, represented one way to move forward with the nuclear program, though haltingly at times.

It became clear that this was not the best organizational structure, particularly when Mackenzie retired and was succeeded by W.J. Bennett, who had been executive assistant to Howe during the war. Some action was indicated.

On 1 April 1954 the Atomic Energy Control Board Act was amended to transfer research and production functions of the AECB to a minister (the chairman of the Privy Council Committee for Scientific and Industrial Research); certainly a key decision that should have been taken sooner. Both crown companies, Eldorado and AECL, were to report to Howe.

Nuclear Power Reactors

Early in the history of the program in Canada, the staff, and others, were discussing the possible use of nuclear energy for electric power production, but the first definitive decision came early in 1953, when C.D. Howe stated in the House of Commons: 'Canada should develop atomic power in this country and discussions are underway to bring about this development. Production of power should rest with the utilities such as Ontario Hydro.'

The Nuclear Power Branch of AECL

This quickly led to another key decision. The AECL Board agreed, late in 1953, to set up a study team in its Nuclear Power Branch to look at a small power reactor. Perhaps the real key decision was the appointment of H.A. Smith of Ontario Hydro in January 1954 as head of the team. The members of the team included not only AECL and Ontario Hydro staff, but representatives of other utilities and Canadian industry, and consulting engineers.

By August 1954 the studies indicated that the design, engineering, and construction of a prototype power reactor should be undertaken – size 10–20 000 kWe – D_2O moderator and coolant using natural uranium fuel with the possible use of some Pu for enrichment.

The AECL Board took a formal decision 30 November 1954 that although AECL was necessary in the role of research and development, AECL should not directly design and construct nuclear electric power plants – it should be the responsibility of the manufacturers and the utilities. Certainly a key decision but one that would not stand the test of time.

The Board also felt that, although the small prototype power reactor was an essential step, a larger unit should be studied to uncover the problems likely to be encountered in commercial-sized power units.

Nuclear Power Demonstration (NPD) Plant

When it was decided to proceed with a demonstration power reactor to be designed and constructed by a manufacturing company, AECL invited seven Canadian corporations to bid on a contract to design and construct the Nuclear Power Demonstration plant. Proposals were to be in by February 1955. AECL undertook several responsibilities, such as supply of D_2O, nuclear data, uranium fuel, and some key personnel to assist the contractor. AECL also assumed responsibility for the operation of the reactor from the nuclear standpoint, but the contractor was to be responsible for mechanical performance.

The AECL Board, at a meeting on 10 March 1955, reviewed the proposals received; they were carefully evaluated and shops inspected to check facilities, experience, and capacity to handle power plant design, supply, and construction. Participation by a utility was also reviewed; two had presented proposals – Ontario Hydro and Nova Scotia Light and Power Company.

The Board recommended that the Canadian General Electric Company Limited be selected to design and

construct the demonstration nuclear electric power plant, and that the Ontario Hydro proposal for participation be accepted. The participation of Ontario Hydro was probably the most important key decision.

CANDU – Douglas Point

Following the initial report of the group in the Nuclear Power Branch, which resulted in NPD I, it was decided to extend the work, again under Harold Smith, with seconded representatives from Ontario Hydro and other utilities, and manufacturing companies as the Nuclear Power Group, which relied heavily on AECL staff for advice and guidance in scientific considerations of nuclear systems.

The Nuclear Power Group reported in May 1957 (NPG 10) and recommended that a nuclear power reactor of about 200 MWe be considered, employing pressure tubes to contain the fuel and heavy water coolant. With short-bundle fuel fed from each end of the reactor – on power, bi-directional fuelling – adequate fuel burnup could be achieved with natural uranium. The Group recommended that a program of design, development, and construction of one or more such units could 'reasonably be expected to compete favourably on the basis of safety, reliability and economy – with conventional steam-electric plants for base load applications in Ontario.' With a proper, well-developed program this could be achieved in about ten years.

This report – NPG 10 – was not in itself a key decision, but it certainly led to key decisions taken within the next couple of years.

In February of 1958 AECL published *A Statement of the Nuclear Power Development Program* (AECL No. 561). It was a comprehensive statement covering Canada's need for nuclear power, the status of nuclear power development, AECL's basic policy for development work, the Ontario Hydro proposal for participation, a summary of the program, and the organization to carry out the program.

The statement included a detailed description of the Nuclear Power Plant Division that was to be set up by AECL in Toronto, to meet the requirements of Ontario Hydro for their serious participation in the Canadian program for nuclear power development.

The work of the NPG on CANDU resulted in a very key decision that was not easy to take – to change the basic design of NPD-1. It recommended horizontal pressure tubes in place of the 'pressure vessel' in the NPD I design, to contain the pressurized heavy water coolant and the fuel – which, among other things, eliminated the limitation of size that the pressure vessel design faced – and recommended a new method of fuel loading and unloading that was a major improvement over NPD I. The new design concept looked so promising that a key decision was taken to cancel the pressure vessel supply contract and redesign NPD I along the lines recommended in the CANDU study. This caused a delay in the completion of NPD of about two years, but the change was well warran fitted smoothly into the planned program of p sively developing a commercial nuclear pow that could compete with other sources of therr tric energy. NPD had some problems, but gen has been very successful and a major contrib the Canadian program.

Another key decision came in 1959, with th ment between the Federal Government, repr by AECL, and the Hydro Electric Power Commi Ontario (Ontario Hydro) to commit to a pro design, development, construction, and oper the 200 MWe CANDU power plant subsequently at Douglas Point.

AECL – Nuclear Power Plant Division (NPP

Coincidently, another key decision was taker up, in Toronto, a Nuclear Power Plant Divisior under the general direction of AECL, assisted s tially by Ontario Hydro staff for reactor design. effort was effectively integrated to cover all as a nuclear power plant from design, through d ment, construction, and operation – a basic ı ment of Ontario Hydro if they were to be a partner in Douglas Point.

The decision to set up NPPD requires specia tion as it was not universally popular at th since a manufacturing company had a nuclea plant design capacity and was actively engage design and supply of NPD, and contemplating sales and alternative designs. The utilities g preferred to acquire plant and equipment by c tive tendering, and this was not possible wi one qualified supplier. Some Canadian consul gineering organizations contemplated putting t a nuclear design team that could be develope fully rounded design group, when coupled wi experience in conventional power stations.

On looking at such an arrangement, and at t of the art of nuclear power, and the very pro and important position of AECL (with its labo and years of experience in the nuclear field fr damental research, through development anc tion of advanced research and development re it became obvious that AECL should take a ma in at least the CANDU 200 MWe Douglas Point

H.A. Smith, in a paper presented to the Eu Nuclear Conference on Nuclear Energy to M April 1975, stipulated clearly the principles most important in the overall organization and istration of a nuclear power project.

1) To insist that the client-owner of the ultimate fa heavily involved both financially and administr
2) To assign authority and responsibility in relatic financial risk accepted.
3) To encourage direct participation by consulti neering and industrial organizations.

4) To restrict the number of development options to be pursued.
5) To minimize duplication of effort, particularly avoiding commercial competition in experimental and developmental assignments.
6) To ensure continuity of, and maximum feedback in successive stages of the program.
7) To maintain strong liaison with other nations developing nuclear power plants, both for information exchange and monitoring.

With Harold Smith appointed General Manager of NPPD, you may be sure these principles were adopted, and the structure of the Agreement between AECL and Ontario Hydro included the NPPD organization, which was the most attractive route to the design and development of the Douglas Point Station.

It is clear to me that this joint arrangement between AECL and Ontario Hydro, with maximum use of consultants and industry was a key decision in Canada's nuclear power program, and an essential one. One can only wonder why other countries, particularly the UK, have not followed suit – except, perhaps, they are reluctant to accept a colonial program that has been successful in such an advanced energy field. A single industrial company as designer and supplier of a nuclear power plant did not fit the principles thought necessary, in a utility like Ontario Hydro, for a program.

AECL and Ontario Hydro reached a formal agreement that resulted in a NPPD of AECL located in Toronto, managed by an Ontario Hydro engineer, H.A. Smith, supported by a member of an engineering consulting firm, J.S. Foster, and surrounded with very capable staff from AECL, Ontario Hydro, and the general engineering community of Canada.

This group, which had in part been together from 1953 through the original NPD I designs and recommended the changes that resulted in NPD II, now started, in 1959, to design and supervise general development of the 200 MWe CANDU at Douglas Point.

The group, of course, continued to look at advances in system designs and to encourage Canadian industry to manufacture products to nuclear standards. They not only designed systems but operated extensive prototype test facilities to ensure satisfactory Canadian equipment to meet the rigorous standards of the heavy water-moderated and cooled power reactors.

Another very significant decision – and a 'gutsy' one – was taken in 1964, prior to the start-up of Douglas Point, to proceed with the first two 540 MWe units at Pickering. Again it was a joint program of AECL, Ontario Hydro, and the Province of Ontario, with AECL taking the responsibility for the supply and performance of the NSSS; Ontario Hydro was responsible for the general construction program and supply of the conventional part of the plant. Ontario Hydro agreed to own and operate the plant, and pay AECL and the Province, on an agreed formula, the difference between the cost of the electric energy sent out and that from a similar coal-fired station at Lambton of generally the same size and vintage. Incidentally, up to the time of the shutdown to change pressure tubes in Pickering I and II, I understand AECL had recovered all its costs including interest.

Following Pickering I and II, Ontario Hydro have proceeded on their own to meet their needs for nuclear electric capacity using the nuclear power plant group in Toronto for design and development. Their key decisions – Pickering completion, Bruce A and B, and Darlington – were major undertakings, and will contribute to continued relatively low-cost electric energy in Ontario for many years.

Nuclear Fuel Supply

The first real inkling we had that we might be able to replace the metal uranium fuel we had for NRX and NRU with an uranium oxide fuel for power reactors, and obtain much higher burn-up, came when Gib James X-rayed an experimental fuel assembly to be irradiated in an NRX loop for the US Nautilus fuel development program, and found, contrary to any information supplied by the USAEC, we were to irradiate enriched UO_2 fuel clad in Zr. This really started Canada on the road to UO_2 fuel for power reactors, probably a key decision that would allow the CANDU reactor system to perform so well. The USAEC claimed we had breached security, but Gib James said 'no one puts anything in the fuel channels of my reactor without an X-ray examination; we have done it from the start.' We survived the investigation and got busy looking at UO_2.

AECL and the Department of Mines and Technical Surveys developed a very good joint in-house program of development of a natural uranium oxide (UO_2) fuel, including manufacturing processes, but it became evident that this was one area where Canadian industry could take over the production and supply of power reactor fuels as a commercial venture and, presumably, develop a profitable line of business. The decision to place the responsibility for the manufacture of nuclear power reactor fuel in the private sector is one real success story and a key decision.

Proposals were invited from Canadian companies, and after some changes the private companies, particularly the Canadian General Electric Company Limited (CGE) and Westinghouse Canada Limited, with equal production capabilities and their own development facilities and competence, have become the main suppliers, on a commercial basis, of UO_2 fuel for all CANDU reactors. The quality of product is very high, with few failures. It is a simple fuel, and one of the keys to very low fuel costs in a neutron-efficient reactor.

Canadian Supply of Heavy Water

The first heavy water production in Canada was at

the Consolidated Mining and Smelting Company (CMS) plant at Trail, B.C. The US contracted with CMS to produce heavy water from the deuterium in the hydrogen stream they had for the production of fertilizer. Production started in 1943, and they produced about six tons per year to 1955, when the economics did not match the production costs from the large plants in the US.

As soon as Ontario Hydro and AECL decided, in 1964, to proceed with the Pickering Station, it became evident that the supply of heavy water from any source might be a holding item, and construction of large Canadian heavy water production plants was necessary to ensure an orderly program of nuclear power development.

A contract with Deuterium of Canada Limited (DCL) had been signed in late 1963, but with Pickering underway a second unit was approved to be located in Nova Scotia under a contract with CGE. Both plants had a design capacity of 400 tons / year and both, particularly the DCL plant, had their difficulties.

Although granting authority to enter into contracts for the units in Nova Scotia was a key decision and put Canada on the road to supplying an essential part of the CANDU power reactor program, the lack of early production and the small size of plants became particularly evident when Ontario Hydro announced its decision to proceed with the 3,000 MWe Bruce station. AECL was authorized to build a 400-tons-per-year facility at Bruce using nuclear steam for energy – increased to 800 t / yr in 1969, when it was realized that the DCL plant was not able to supply any product for Bruce. This AECL plant was purchased by Ontario Hydro after having operated successfully.

The first Bruce heavy water production plant went so well that Ontario Hydro took the initiative and proceeded on its own with a major program that led the way to major heavy water production facilities that ensured ample supply for the foreseeable future of this essential material from Canadian sources. The design and construction and early operation of the Bruce I plant, and subsequent plants, benefitted from the start-up and operation of the CGE plant, where many problems were solved and the information passed on to Bruce.

Incidentally, the DCL plant did not produce product, and in 1971 AECL took over the plant for major redesign and rehabilitation – full production was only achieved in 1979. CGE found they could not meet their contract prices, and AECL took over the plant. These earlier 'questionable decisions' were rather di to all concerned, but the plants at Bruce w successful. All plants are now 'mothballed,' orders from reactors yet to be committed.

In summary, the decision to contract wit manufacture heavy water and to locate the f Glace Bay – which AECL did *not* recommend disaster; the decision to contract with CGE for Hawkesbury plant production was a relative decision. The key decision was the decision the first large plant at Bruce, designed by L owned by AECL originally, but built and ope Ontario Hydro and powered with nuclear s was an essential decision that allowed the power program to proceed.

CANDU BLW and Organic-cooled Reactors

When the AECL directors decided not to proce an organic-cooled reactor system, but felt th back-up to the CANDU-PHW system was advisa selected the boiling light water-cooled versio DU. Other countries, particularly Japan, w considering the same basic system; only the J have continued in a serious manner.

The Gentilly station was committed as a BLW, and although it operated it did not pro an acceptable unit in the Hydro Quebec sys its operation was terminated.

L.R. Haywood, who was at the centre of e ing development at CRNL during this period, gested to me that, if AECL and Hydro Quebec h as much time and effort to ensure good initi tion of Gentilly I as we did for Douglas Point, v have had a successful unit. He may be right; the Japanese have shown the system will w economic assessments are not as yet complet

In the light of experience from the organi research reactor WR1 at the Whiteshell Nuclear Establishment, some of us feel we may have the boat in not developing the organic syste second string to the Canadian bow. So AECL' average of key decisions was very high, bu not bat 1,000.

Altogether, the nuclear power program d taken by the Canadian Government, Ontario and AECL have generally been successful and ing. They rested heavily on decisions taken the engineers' desks and in the scientists' lab of AECL, Ontario Hydro, and the suppliers – of many hundreds of man-years of good solid

Insights from Chernobyl on Severe Accident Assessment of CANDU Reactors

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Abstract

The accident at the Chernobyl-4 RBMK reactor near Kiev in the USSR on 26 April 1986 is described. The characteristics of the RBMK reactors are compared to those of CANDU reactors. Certain insights on the assessment of severe accidents in CANDU reactors are drawn from the Chernobyl-4 accident. In particular, the importance of the design of the safety shutdown systems in CANDU is recognized. The most significant lesson from the Chernobyl-4 accident is that primary responsibility for the safety of a nuclear power plant must lie with the operating utility itself, and all managers and operators must be fully conscious of their responsibility for worker and public safety.

Résumé

L'accident du 26 avril 1986 au réacteur RBMK Chernobyl-4, près de Kiev en URSS, est décrit. Les charactéristiques du réacteur RBMK sont comparées à celles du réacteur CANDU. Certaines conclusions sur l'évaluation d'incidents sérieux survenus à des réacteurs CANDU sont tirées de l'accident de Chernobyl, en particulier l'importance du système des mécanismes d'arrêt sécurité pour le CANDU. La plus importante leçon apprise de l'accident de Chernobyl est que la responsabilité première d'une station nucléaire doit reposer à la station elle-même et que tous les superviseurs et opérateurs doivent être conscients de leur responsabilités en ce qui a trait à la sécurité des travailleurs et à celle du publique en général.

Introduction

No one in the nuclear power field is ever likely to forget 26 April 1986, the date of the most serious accident ever at a nuclear power plant. The accident on that date to the Chernobyl-4 RBMK reactor near Kiev in the USSR resulted in the destruction of the reactor and building, the deaths of 31 plant and emergency workers, a major release of radioactivity, the evacuation of 135,000 people from a region within 30 km of the plant, and a significant collective dose of radiation to the population of the USSR and other parts of Europe.

The impact of the accident on world-wide public attitudes to nuclear power has been negative, as would be expected. Moreover, a number of nuclear power projects and commitments have been delayed or postponed and others threatened with cancellation.

In this situation, it is incumbent on those in various countries who recognize the present and potential future benefits of nuclear power to assess the accident at Chernobyl and to learn from it. In this way, they can ensure that their own reactor technologies and reactor safety practices provide high confidence that the benefits of nuclear power can continue to be gained at acceptably low risks to operators and to the public.

The purpose of this paper is to evaluate certain severe accident scenarios in CANDU reactors in the light of the Chernobyl reactor accident. No claim is made for completeness of the assessments presented here; rather they are representative of assessments in the areas of CANDU safety in which the author has some experience and with which he is familiar. It is hoped that the insights gained will contribute to the necessary on-going process of learning from the Chernobyl accident.

The Accident at Chernobyl-4

Information on the RBMK reactor and the accident was obtained chiefly from the report of the USSR State Committee on the Utilization of Atomic Energy to the IAEA in Vienna [1], but also from other sources [2, 3], including personal communications with personnel of Atomic Energy of Canada, Ltd., Ontario Hydro, and the Atomic Energy Control Board.

Comparison of RBMK and CANDU Reactors

The Chernobyl-4 reactor was one of four RBMK reactors

Keywords: reactor safety, CANDU reactors, Chernobyl accident.

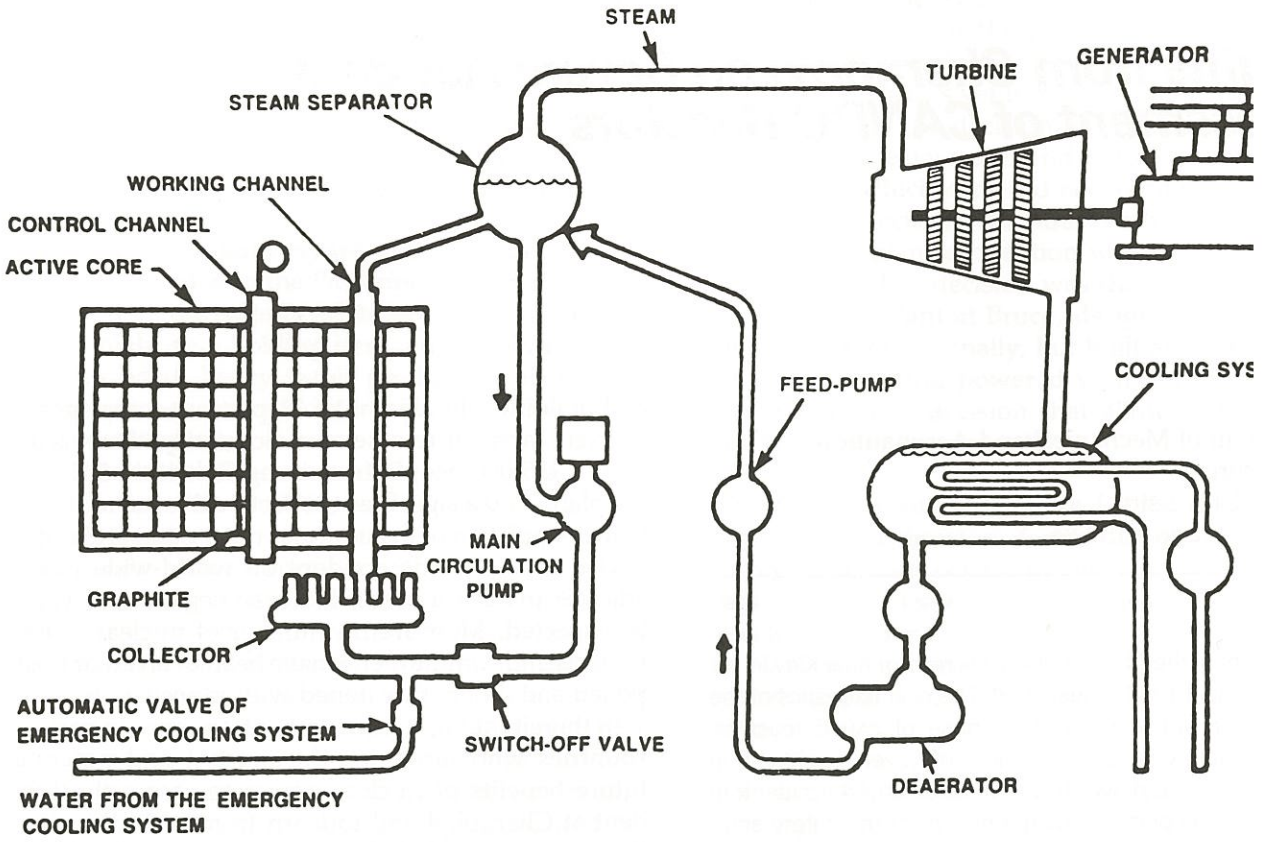

Figure 1. Schematic diagram of RBMK reactor.

at the site, each with a rated net output of 960 MW(e). The RBMK reactor is a graphite-moderated, boiling light water type, operated on a direct-cycle, as can be seen in Figure 1. Certain important characteristics of the RBMK reactor are given in Table 1, where they are compared to those for a typical CANDU reactor of somewhat lower thermal power: a Bruce-B reactor unit. Similarities of the RBMK to the CANDU reactor include the pressure-tube design, with Zircaloy-Niobium pressure tubes, and the use of on-power fuelling.

Among the RBMK characteristics listed in Table 1 which were of significance in the accident are the graphite moderator, the boiling coolant, and the very large core.

The moderator consists of a graphite block structure with the fuel channels running through the centres of the blocks. Heat generated in the graphite is removed by the primary coolant *via* graphite rings between the blocks and the pressure tubes. The resulting moderator temperatures at the design point range from about 270 degrees Celsius to about 700 degrees Celsius, so that there are no Wigner energy problems, which led to the Windscale reactor accident [4]. The graphite block structure is located within a thin-walled metallic container. The voids in the container are filled with a nitrogen-helium blanket to promote heat transfer and to prevent oxidation of the graphite. The moderator both enhanced and m the effects of the accident, as will be explaine

The boiling water coolant introduces voids core and results in positive reactivity feedb

Table 1: Comparison of Chernobyl Reactor with Bruce-E

	Chernobyl	*Bruc*
Type	RBMK	CANI
Thermal power, MW	3200	2852
Moderator	Graphite	Heav
Coolant	Water (boiling, $x_0 = 14\%$)	Heav
Cycle	Direct	Indir
Fuel	UO_2-2% enriched	UO_2
Orientation	Vertical	Hori
Core outlet pressure, MPa	7	9.3
Pressure containment	Pressure tubes (Zr Nb)	Press (Z
Number of fuel channels	1660	480
Core diameter, m	11.8	7.07
Core height or length, m	7.0	5.94
Re-fuelling	On power	On p

Table 2: Comparison of Chernobyl Reactor with Bruce-B Reactor: Reactivity Worths of Control and Safety Systems for Equilibrium Core Conditions

	Chernobyl	*Bruce-B*
Total of automatic control systems, mk	~6–8[1]	~33[2]
Safety systems, mk	Total: ~30[1]	SDS #1: 73.6[3] – 32 rods 53[3] – 30 rods[4] SDS #2: >300[5]

[1]Total worth of all automatic and manual control and protective system = 105 mk. Very slow emergency insertion rate (~0.4 m/s).
[2]Adjusters, zone controllers, control absorbers.
[3]Available within 2 seconds.
[4]With two most effective rods not available.
[5]55 mk available within 2 seconds.

Table 3: Comparison of Chernobyl Reactor with Bruce-B Reactor: Reactivity Coefficients

	Chernobyl	*Bruce-B*
Void coefficient at operating point	$+2.0 \times 10^{-4}$/ vol.% void	$+1.14 \times 10^{-4}$/ vol.% void
Power coefficient (fast) at operating point	-0.5×10^{-6}/MW	-0.73×10^{-6}/MW
Fuel temperature coefficient	-1.2×10^{-5}/K	-4.2×10^{-6}/K
Moderator temperature coefficient	$+6.0 \times 10^{5}$/K	$+6.0 \times 10^{-5}$/K

we will see. The very large core presents problems of spatial stability of power distribution.

Also of significance in the accident were the characteristics of the control and safety shut-down systems, as given in Table 2, which compares the reactivity worths of these systems for the RBMK and CANDU reactors,[1] and Table 3, which compares the reactivity coefficients for the two reactor types. The reactivity worths of the automatic control system and of the safety shut-down system of an RBMK are considerably less than those of a CANDU, and the reactivity insertion rates of the safety shut-down systems of an RBMK are also significantly lower than those of a CANDU. Furthermore, the control and safety shut-down systems are not independent in an RBMK, as they are in a CANDU, which, in addition, possesses two independent safety shut-down systems. Both the reactor types have positive void coefficients, but that of an RBMK is almost twice that of a Bruce-B unit.

The RBMK emergency coolant injection (ECI) system consists of two high-pressure accumulator-driven sub-systems, plus one pumped sub-system, to provide emergency cooling for the first one to two minutes. There is a separate pumped sub-system for the longer term. All sub-systems inject into the headers below the core.

Of importance in the accident was the lack of a containant over the reactor core in an RBMK unit. Most of the primary heat transport system is located in concrete compartments, called the accident localization system, but not the piping and other components above the core, which are located in the reactor building. The reactor building was not designed as a containment building. There is also a steam suppression pool below the reactor.

Description of the Accident

The accident occurred during a low-power test before a scheduled shut-down to demonstrate the ability of a turbine-generator, disconnected from the grid, to provide power for the short-term emergency-coolant pumped system during the turbine-generator rundown after interruption of the steam flow. This mode of power supply to the ECI is necessary, in the design-basis accident of a pipe break plus loss of offsite power, to run the pumps before the stand-by diesel generators can pick up the load. Such tests had been performed successfully and safely on other RBMK units, and a test had previously been done safely, but not successfully, in Chernobyl-4.

However, in this case, an operator error combined with a number of violations of procedures, and with the characteristics of the RBMK, to cause a disastrous accident. A very significant factor in the accident was pressure on the operators to complete the test successfully, since the next opportunity to undertake it would not occur until the next scheduled shut-down in a year's time. The following description of the accident, taken mainly from reference 1, is based on a reconstruction of the events by the Soviet authorities, using instrument charts and real-time analytical simulations of the reactor neutronics, thermohydraulics and control and safety shut-down system actions.

Prior to the test, in preparation for shut-down, the operators reduced reactor power to about 1,600 MW (half-power) and shut-down one of the two turbine generators supplied by the reactor. In accordance with the planned test procedure, the ECI system was blocked, to prevent spurious injection during the test. However, at this point, the grid demand resulted in the unit being required to continue to operate for about nine hours at 1,600 MW, still with the ECI system blocked-off in violation of operating rules.

Power reduction was then resumed, since the test was to be performed at an initial reactor power of 700 to 1,000 MW. However, when the operator switched from local automatic power control to bulk automatic power control, which was required for low-power operation, he failed to establish correctly the controller set-point, with the result that the power fell below 30 MW. Only after some time did the operator succeed in stabilizing the power at 200 MW. Power could not be raised higher because of the build-up of xenon during the long period at part load and the negative

effect of the increased water content in the core following power reduction.

Additional primary coolant pumps were started up so that the coolant flow rate would still be adequate to cool the core after the turbine-generator run-down following its isolation from the steam supply. The normal reactor trip which would shut down the reactor with both turbine-generators valved out was also blocked off. These steps were taken, under the pressure to complete the test successfully, so as to enable the test to be repeated with a different type of generator voltage control.[2]

The flow rate through the core was now much higher than pump cavitation limits would normally permit, not only because of the additional pumps but also because of the low power, which reduced steam generation rate, and hence void, and thus core hydraulic resistance. The steam pressure also was dropping because of the reduced steam generation rate. The operators, in attempting to stabilize the operating conditions without tripping the reactor, then blocked the reactor trips for low separator water-level and low separator pressure.

Because of the low core void and the continuing build-up of xenon, the core reactivity continued to drop, which resulted in the automatic control rods being withdrawn, and which also forced the operators to withdraw some of the manual absorber rods. The reactivity margin was now reduced below the level that required immediate shut-down of the reactor. Nevertheless, operation was continued. Just before the start of the test, the operators significantly decreased the feedwater flow rate in an attempt to stabilize the water level in the steam drums. This action resulted in an increasing inlet temperature to the core.

The core was now in a potentially very unstable condition with very little reactivity margin, and under power, flow-rate, and inlet temperature conditions such that there was only a low void near the core exit, giving a high sensitivity of void to power changes.

At this point, the test was begun by closing the stop valves of the operating turbine. This action resulted in the pressure in the steam drums increasing as the steam flow rate decreased, and the coolant flow rate decreasing as the turbine and pumps ran down. The core void fraction was now being influenced by the increasing pressure, the increasing core inlet temperature, and the decreasing core flow rate. The first factor tended to decrease void, the other two to increase it. The net result was a rapid increase in core void fraction, which caused a rapid increase in reactivity,[3] and therefore a rapid increase in power. The increase in power generated more void, which accelerated the power increase, a classic case of positive feedback. The control system could not respond rapidly enough to limit the power surge; the operator activated a manual trip 36 seconds after the t menced, but this was ineffective because of flux shape, the location of some of the abso and their slow emergency insertion rate. In a the safety absorber rods did not insert fully, ably because of damage to the core by this ti estimated that the power surged to 100 ti power in about four seconds.

The fuel overheated and disintegrated, st Zircaloy reacted to generate hydrogen, the fu nels ruptured, which permitted steam to re graphite to generate hydrogen and CO, and t erator container ruptured, permitting H_2 an mix with air. Two explosions in rapid sequer heard, the first apparently associated with t steam formation and resulting fuel channel r and the second possibly with a chemical e (CO, H_2, and air igniting).

Approximately 4% of the fuel was ejected core and the graphite moderator ignited, and ally about 10% of its 2,500 tonnes burned be fire was extinguished several days later.

The reactor building was destroyed and a of fires were started around the unit, whi extinguished in a few hours.

Estimated radioactive emissions from the d reactor reached about 96 million curies by Ma which the releases dropped to minor levels. the estimated releases are given in Table 4.

The adverse health effects of these emiss not germane to the topic of the paper and sc discussed here.

Certain potential severe accident sequen CANDU reactor will now be assessed in the lig accident to the Chernobyl reactor.

Table 4: Chernobyl Reactor Accident Estimated Radioactive Emissions[1]

	Emissions,[2] *Curies*	*Per cent of core inventory*
Noble gases	45×10^6	~100
Iodine	7.3×10^6	20
Tellurium		15
Cesium – 137		13
Cesium – 134		10
Total	96×10^6	~6–8

[1]Radioactivity emitted up to May 6 1986, calculated as of M minor emissions after May 6.
[2]Accuracy: ±50%.

Loss-of-Regulation Accidents in a CANDU R

The Chernobyl reactor accident was essentiall of-regulation accident in which the positive efficient of the RBMK reactor played a major ro the CANDU, being over-moderated like the RB has a positive void (and moderator temperatu

ficient, the question arises as to the susceptibility of the CANDU to a similar loss-of-regulation power excursion accident. Since recent CANDU reactors operate at low-quality conditions (~4%) at the core exit when at full power, some concern may exist on this point, although it is recognized that the 'stiff' CANDU heat transport system reduces the power-to-void feedback effect below that of the RBMK.

However, a major difference between typical CANDU and RBMK reactors in this respect is the much greater speed of insertion of negative reactivity by the safety shut-down systems of the CANDU compared to the RBMK, as can be seen from Table 2. Also, as shown in Table 2, the total reactivity worth of the two CANDU safety shut-down systems (SDS-1 and SDS-2) is much greater than that of the single RBMK emergency system.

To illustrate the importance of the speed of response and the worth of the CANDU safety shut-down systems, it has been calculated that, had the Chernobyl-4 reactor emergency protective system had the same worth and insertion rate as a Bruce reactor SDS-1, assuming that the manual trip occurred at the same instant, the reactor power would have been turned around at about 15% over-power, and probably no serious damage would have resulted [5].

Furthermore, the existence of the two separate, independent, completely redundant and diverse shut-down systems in CANDU, which are also independent of the automatic control system, would provide much greater emergency shut-down reliability for the CANDU than for the RBMK. Indeed, the main reason for the provision of the two independent shut-down systems in CANDU reactors is to ensure reliable emergency shut-down in all accident conditions, particularly considering the positive void coefficient of reactivity.

Therefore, it can be concluded that the rate of insertion and depth of the two independent shut-down systems in CANDU, and their independence from the automatic control system, would prevent, with a very high level of reliability, power excursion accidents similar to that at Chernobyl. The wisdom of providing two independent fast-acting, high-worth safety shut-down systems in CANDU reactors would seem to be vindicated by the Chernobyl accident.

Impairment or Blocking of ECI in a Large LOCA in a CANDU Reactor

As we have seen, the emergency coolant injection system of the Chernobyl-4 reactor was blocked to prevent spurious injection during the planned test.[4]

The question arises of the consequences of the blockage or impairment of the ECI system during a severe accident in a CANDU reactor.

The dual-failure accident of a large stagnation LOCA plus a loss of emergency coolant injection (LOECI) is a design-basis accident for a CANDU reactor in Canada. The AECB requires for this case, as for all dual-failure accidents, that the maximum dose to an individual in the public not exceed 0.25 Sv and that the collective public dose does not exceed 10^4 person-Sieverts [6].

Considerable analytical and experimental work has been expended over the years in Canada to demonstrate that these requirements can be met.

In such an accident, the fuel and pressure tubes overheat and the pressure tubes deform into contact with the calandria tubes. Depending on the size and location of the pipe break or other event causing the LOCA, the deformation of the pressure tube will consist of a uniform radial ballooning or an eccentric sagging, as shown in Figure 2. If pressure-tube overheating occurs early in the blowdown transient following LOCA, when internal pressure is high, it will deform by ballooning. If overheating occurs late in the transient, when internal pressures are low, it will deform by sagging. In either case, deformation of the pressure tube will provide a heat flow path of relatively low thermal resistance from the fuel to the separately cooled, low-temperature moderator, which thus provides a back-up heat sink for the stored and decay heat and heat generated by the exothermic Zircaloy-steam reaction.

The computer simulation codes, CHAN and CHAN-2, have been developed by AECL and Ontario Hydro to predict the thermal behaviour of a fuel channel for the case of a pressure tube ballooning or sagging into contact with a calandria tube [7, 8]. There is considerable experimental verification of the models used in these codes [9, 10]. Results obtained using the CHAN code for the thermal behaviour of the hottest point along a high-power fuel channel in a Bruce reactor, for pressure tube ballooning following a LOCA plus LOECI, are given in Figure 3, taken from reference 11. The results shown are for the worst residual steam flow conditions in the channel, considering the exothermic steam-Zircaloy reaction and cooling produced by the steam flow. Figure 3 shows that there will be no gross melting of the fuel in a CANDU reactor, in spite of the loss of ECI, and that the maximum pressure tube temperature remains low enough to ensure its integrity.[5]

A computer simulation code, IMPECC, has been developed at Carleton University, under contract to the AECB, to predict the thermal behaviour of a fuel channel for the case of a pressure tube sagging into contact with a calandria tube [12, 13]. The model used for the non-conforming contact thermal resistance between the pressure and calandria tubes in IMPECC has experimental confirmation [13, 14, 15]. Results obtained using IMPECC for the thermal behaviour of the hottest point (circumferentially and axially) along a high-power (7.5 MW) fuel channel in a Bruce reactor following a LOCA plus LOECI are given in Figure 4, taken from reference 16. Again, it can be seen that there will be no gross melting of the fuel and that the maximum

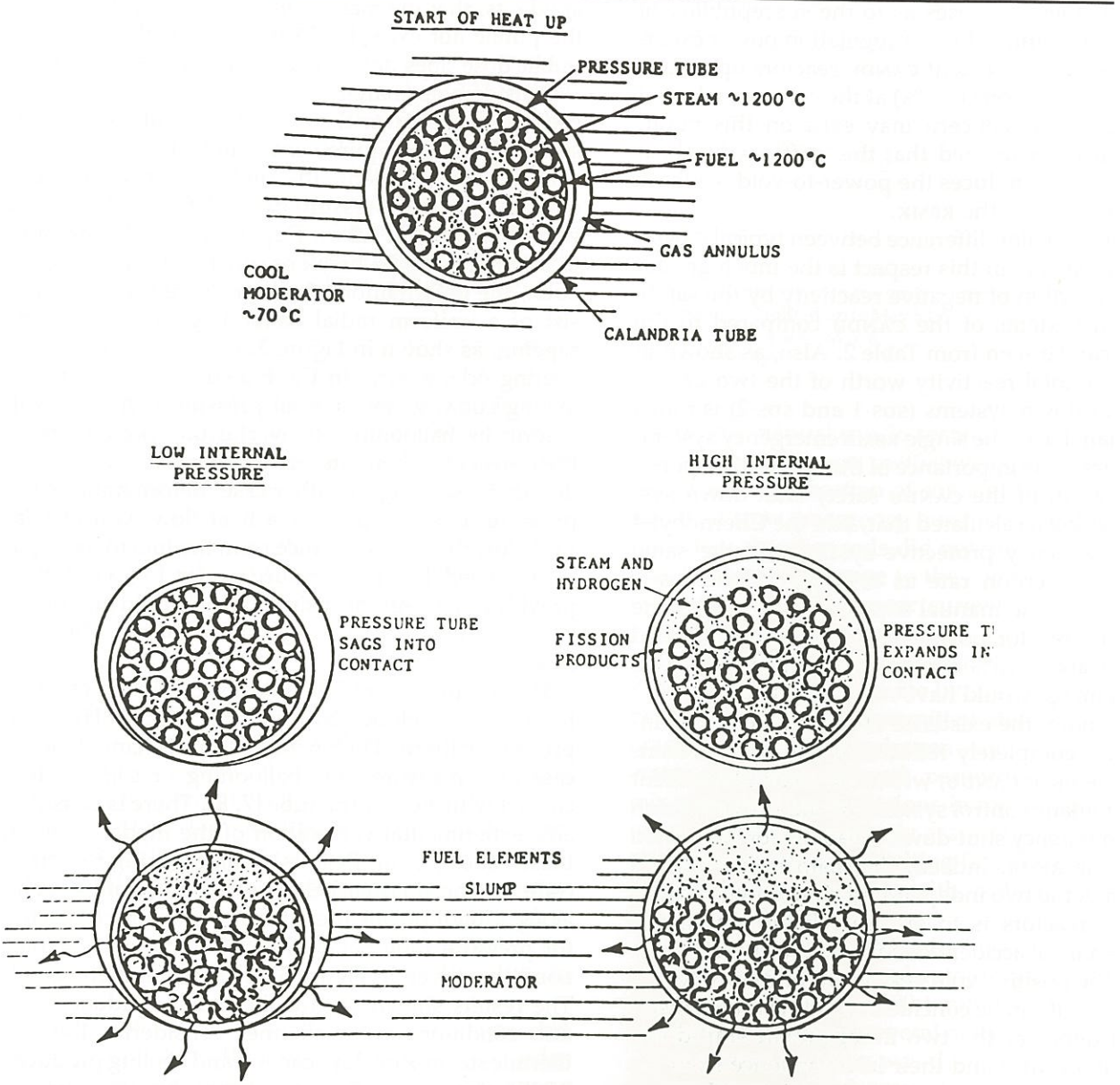

Figure 2. Deformation of CANDU pressure tubes in a LOCA plus failure of ECI.

temperature of the pressure tube is well below its melting point. Brown, *et al.*, using the code CHAN-2, also show that there will be no gross melting of the fuel and that pressure tube integrity will be maintained in the case of pressure tube sagging in a CANDU reactor fuel channel at an initial power of about 6 MW [17].

It has been concluded from these studies that there will be no gross melting of the fuel, although some fuel damage would certainly occur, and that pressure tube integrity will be maintained in CANDU reactors in the event of failure or blockage of emergency coolant injection following a loss-of-coolant accident, because of heat transfer to the low-temperature, independently cooled moderator.

The Chernobyl accident provides an intere
sight into the effectiveness of a separate mod
a heat sink in a severe reactor accident. Figure
the estimated fuel temperature in the Cherno
tor as a function of time after the accident. I
seen that, after an initial excursion, the tem
dropped to about 800 degrees Celsius and r
close to this value for a considerable time, tl
again to a peak of about 2,200 degrees Celsiu
dropping off. This behaviour is attributed by the
of reference 1 to the effect of the graphite m
and structure acting as a heat sink, as well as
of some fuel particles and fission products f
core. It is noteworthy that the moderator ac
heat sink, even though a significant portion w

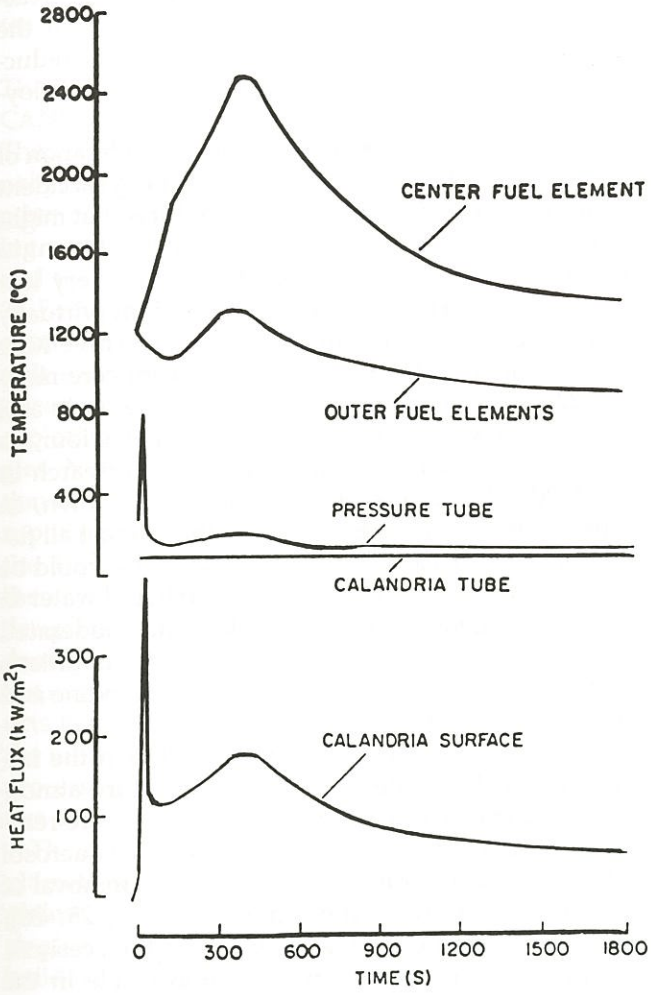

Figure 3. Thermal behaviour of a CANDU fuel channel in a LOCA plus failure of ECI. Pressure tube ballooning.

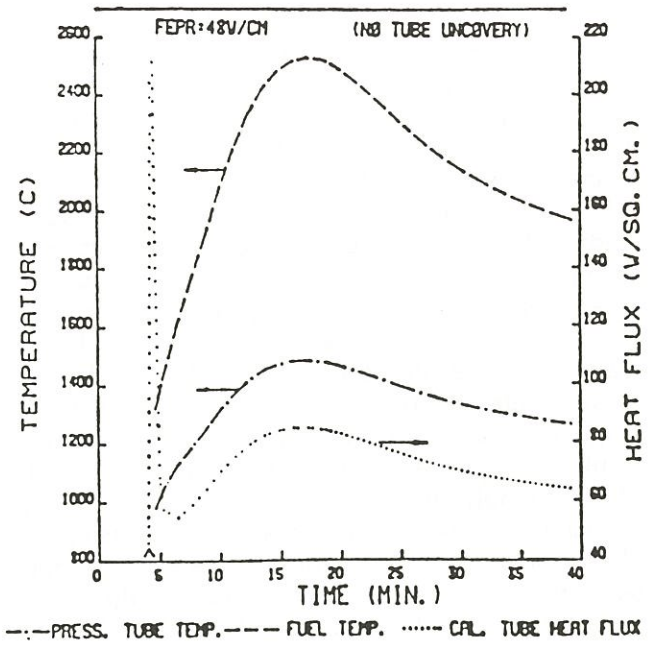

Figure 4. Thermal behaviour of a CANDU fuel channel in a LOCA plus failure of ECI. Pressure tube sagging.

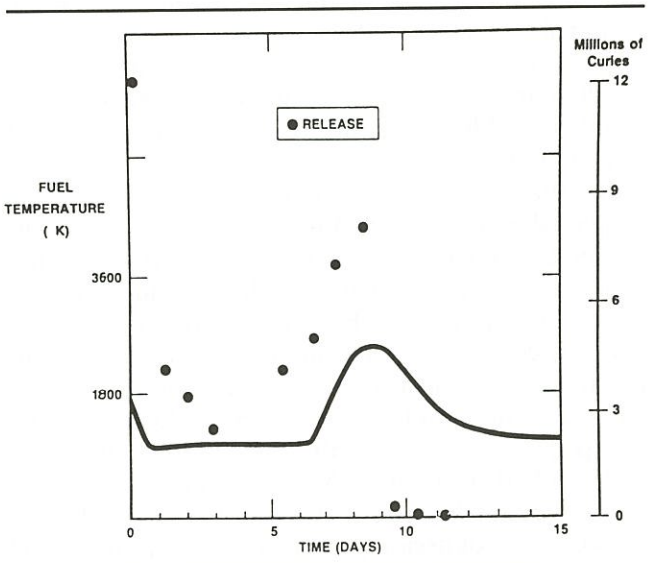

Figure 5. Estimated fuel temperature and radioactivity release in the Chernobyl reactor accident.

ing. The temperature rise from about day seven to day nine resulted from the average temperature of the moderator increasing slowly because of the heat being stored in it and because of the restriction of natural convection flow through the core by material dropped on the reactor from helicopters to reduce radioactive releases from the core. The ultimate temperature turn-around resulted from the effects of fission product decay and natural convection air cooling through the moderator blocks, as well as the introduction of liquid nitrogen below the core. It has been concluded that there was no gross melting of the fuel in the accident, except perhaps for some in the initial power surge.

While the conditions in the Chernobyl accident were greatly different from those for the hypothetical LOCA plus LOECI in a CANDU, the accident does demonstrate that a separate moderator can act as a heat sink in a very severe accident. It is quite probable that the effectiveness of the graphite moderator as a heat sink in the Chernobyl accident prevented even more serious releases of radioactivity than actually occurred, and eventually assisted in controlling the accident.

Fission Product Releases in a Severe Accident in a CANDU Reactor

As can be seen from Table 4, significant fractions of the inventories of the more volatile fission products were released to the environment in the Chernobyl accident. Lower fractions of the less volatile fission products and actinides were also released [1].

Two issues of significance in severe accidents in a CANDU reactor arise here, the first being the mechanism of large releases of fission products from the fuel under conditions of no fuel melting, and the second being the quantities of various fission products that would be released into and from containment in a severe accident in a CANDU.

It has been generally accepted since the Rasmussen study [18] that major releases of fission products other than noble gases, would not occur from the fuel in a severe reactor accident unless gross melting of the fuel occurred. Nevertheless, major fission-product releases from the fuel did occur at Chernobyl in the apparent absence of fuel melting. The initial power surge in the accident, to about 100 times full power, resulted in rapid energy deposition in the fuel to levels of considerably more than 300 cal / gm (1.25 J / kg) [1],[6] the value assumed in the USSR as that causing fuel element and pellet disintegration. The accident report states that fuel particles were carried into the coolant and embedded in the moderator [1]. With such significant disintegration of fuel, much of the grain-boundary inventory, as well as the free inventory of the high-volatile and medium-volatile fission product species (Xe, Kr, I, Cs, Te) would be rapidly released from the fuel, and the release of the grain-bound inventory would also be facilitated. Therefore, the initial release of radioactivity from the fuel was very high, which resulted in very high release to the environment, about 20–22 MCi at the time of release [1], in spite of little or no fuel melting.

For the first few days after the accident, finely dispersed fuel was carried from the reactor by graphite combustion products and hot air. This situation resulted from the oxidizing conditions caused by the air, and possibly CO, which, under the existing temperature conditions, brought about continuing disintegration of the matrix of the fuel elements, thus facilitating continuing release of fission products from the fuel [20]. In this period, the high releases from the fuel resulted in about 80 MCi of radioactivity, as of May 6, being released into the environment. The virtual cessation of fission product releases from the fuel into the environment after May 6 can be attributed to the introduction of liquid nitrogen below the core, which not only reduced fuel temperatures, but provided a nitrogen blanket for the core, which effectively stopped the oxidation process.

Therefore, significant fission product releases from UO_2 fuel can occur even in the absence of fuel melting, given a significant disintegration of a fuel matrix by oxidizing conditions following the accident. Such a situation would not be expected to occur in a CANDU reactor for two reasons. First, as we have already seen, the two fast-acting, high-worth, independent shut-down systems preclude power excursions of the magnitude experienced in the Chernobyl accident. Second, in the most severe accident in a CANDU reactor, a large LOCA plus LOECI, the conditions in the neighbourhood of any damaged fuel would be reducing rather than oxidizing, because of the Zircaloy-steam reaction, producing H_2 gas.

The second issue that arises from a consideration of the fission-product releases in the Chernobyl accident is the quantities of various fission products that might be released into and eventually from containment in a severe accident to a CANDU reactor. The very low levels of iodine (13 to 17 curies) and cesium (virtually none) released to the atmosphere in the Three Mile Island accident [21], in spite of significant core melting [22], led to re-assessment of past experience and existing knowledge of fission-product behaviour, as well as to stimulation of more intensive research in this field. The results of this work have shown, in general, that it would be expected that almost all fission product species except the noble gases would be retained to an overwhelming extent in liquid water in accident sequences in water-cooled and moderated reactors, considering the water-chemistry conditions in these reactors and the chemical forms of iodine and other fission products [11, 23, 24].

Also, experience and studies have shown the importance of the existence of a high-moisture atmosphere in the reactor building following a severe reactor accident in promoting such processes as aerosol formation, adsorption, and deposition for removal of fission products from the atmosphere [23, 25, 26]. Thus, we would expect that very little iodine, cesium, or other fission products would be available in the atmosphere for ready release to the environment in a severe accident in a reactor with large water inventory, as was observed in the Three Mile Island accident and in other accidents in water-cooled reactors [23].

Although the Chernobyl-4 reactor was cooled with boiling water, the water inventory would be relatively low compared to that of water-moderated reactors. The explosions at the moment of the accident, followed by the moderator fire, would have dispersed the water and ensured that the atmosphere around the reactor remained dry. Therefore, processes for fission product removal from the atmosphere would not have been effective. Thus, the high releases of iodine, cesium and tellurium that occurred would be expected. It should be noted that the previous reactor accident which released the greatest amounts of fission product species, other than noble gases, occurred in the Windscale reactor – graphite-moderated and gas-cooled, and thus with no water inventory [4, 23].

We would, therefore expect that, following a severe accident in a CANDU reactor, there would be very low concentrations of fission products, other than noble gases, in the atmosphere in the reactor building – and thus readily available for potential release to the envi-

ronment – unlike the situation in the Chernobyl reactor accident.

The Role of Containment in a Severe Accident in a CANDU Reactor

There was no containment over the top of the Chernobyl reactor, nor over the steam-water piping and other components above the core. The reactor building was of conventional industrial building design and was not designed as a containment building [1, 3]. Even if the reactor building had been designed as a containment building similar to those used for single-unit stations in other parts of the world [27], it is uncertain whether it would have survived the initial explosions intact, considering their very large energies. Nevertheless, especially since a significant fraction of the explosion energy must have been used in rupturing fuel channels and piping, as well as lifting the 1,000 tonne reactor cover plate several metres, there is at least some probability that a standard-design containment building, while suffering some damage, might not have failed catastrophically. Thus, not only might the initial large release of fission products been reduced significantly, but the presence of the building might have permitted inherent removal processes for air-borne fission products, plus radioactive decay, to reduce subsequent releases also.

The containment system in CANDU reactors is the ultimate line of defence in the defence-in-depth design philosophy. Whether it is of the high-pressure type for single-unit stations, or of the low-pressure, vacuum building type for the Ontario Hydro multi-unit stations, it is designed to cope with the maximum energy release and to prevent or minimize the release to the environment of fission products from a large stagnation LOCA combined with a LOECI. The performance of the containment in such an accident must limit the maximum individual and collective doses to the levels prescribed by the AECB's dual-failure criteria [6].

The containment in a CANDU need not be designed to resist a reactor-power excursion of the magnitude experienced in the Chernobyl accident, because the two independent, fast-acting, high-worth shut-down systems, coupled with the inherent characteristics of CANDU, virtually preclude such accidents under any foreseeable conditions in a CANDU. Nevertheless, the CANDU containment can survive accidents more severe than the above design-basis accident, and thus limit the release of fission products to the environment even in such cases. A study has been undertaken at Carleton University, under contract to the AECB, of severe accidents in which the moderator cooling system fails or the moderator heat sink is lost in a LOCA plus LOECI, both highly improbable sequences of events [28, 29]. The study shows that, although gross fuel melting would eventually occur after several hours in these cases, assuming no operator intervention, the molten core would be effectively contained in the calandria vessel, separately cooled by the shield tank cooling system, and would eventually solidify there. (The calandria vessel acts as an inherent core catcher.) Although fission product releases into containment would be very large in this case, there would probably be no consequent failure of containment, as can be seen from Figure 6. Figure 6 shows the estimated pressure transients in containment following a failure of the moderator cooling system during a LOCA plus LOECI in the Bruce-A station, assuming no dousing after the initial one. The peak pressure is seen to lie between 35 kpag and 55 kpag. The higher of these two values is about the same as the containment design pressure difference (~50 kPa) and is well below the test pressure difference (~80 kPa) [28]. This peak pressure difference would not cause any containment failure [30].

While, obviously, the fission product leak rate from containment in this case would eventually be higher than in the design-basis accident of LOCA plus LOECI, the intact containment would provide continuing effective conditions for natural removal processes for iodine, cesium, and other fission product aerosols from the high-moisture containment atmosphere, and thus would limit releases of these fission products from containment.

General Insight

When we examine the magnitude of the Chernobyl accident, with the complete destruction of the reactor and building, many fuel fragments ejected from the core, and great quantities of fission products released, it appears that this may have been the ultimate reactor accident. It is hard to visualize another situation in which the inherent characteristics of a reactor, coupled with human error and many violations of procedures, could combine in such a way as to produce a greater disaster. If so, even though different weather conditions might have resulted in greater predicted adverse health effects, the large worker death toll, the very high economic costs of population evacuation and foregone crop and land use, and the slightly increased risks of cancer to the general population of the USSR, may represent a real upper limit to the consequences of any power reactor accident. This possibility should be considered in future power reactor risk studies.

Conclusions

The insights in this paper into assessments of severe accidents in a CANDU reactor gained from a study of the accident to the Chernobyl-4 RBMK reactor are, of course, based on a first, rather rapid analysis of the accident. But, while some of the technical details may change on further study, the insights should probably remain valid.

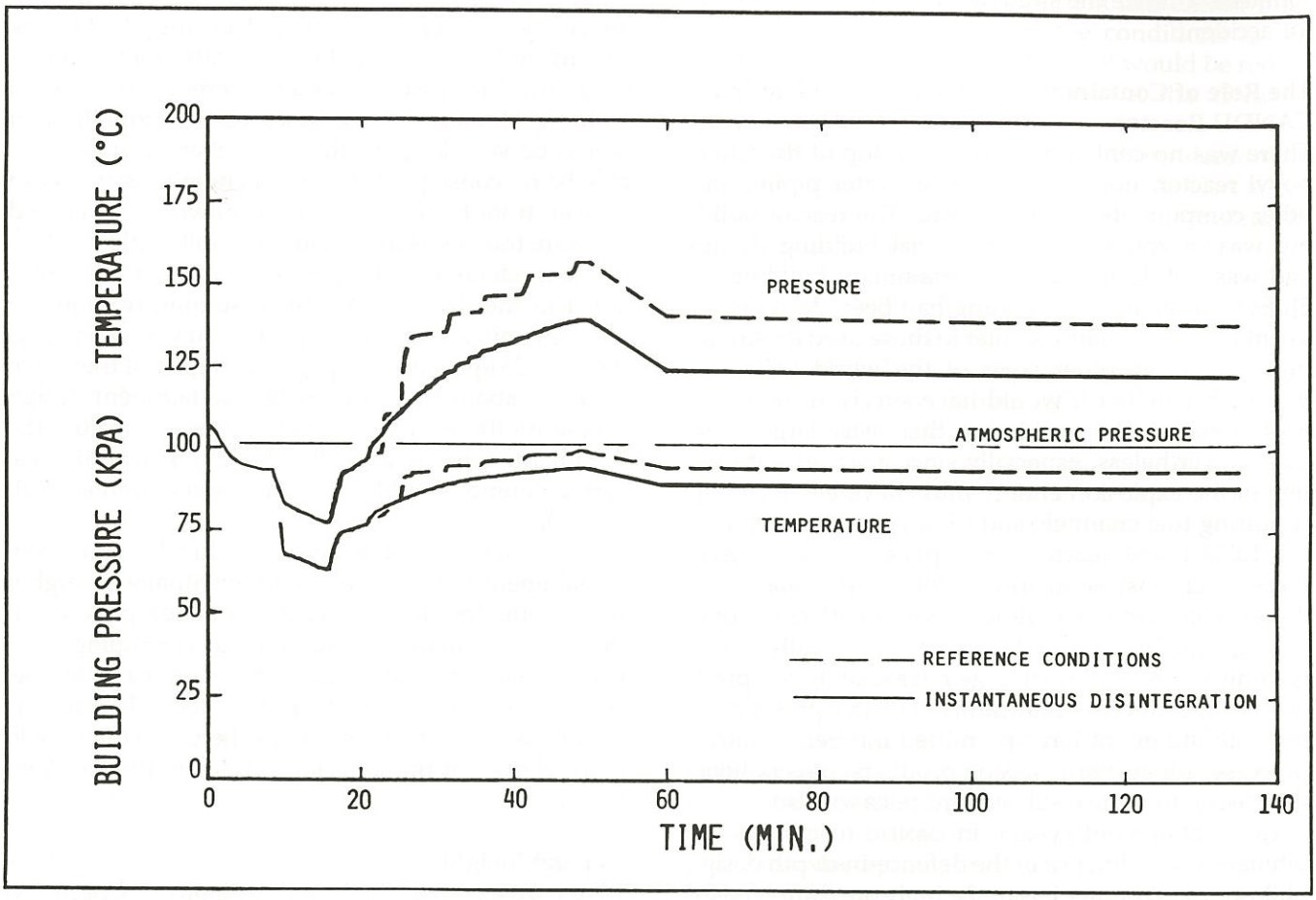

Figure 6. Containment pressure transient in the Bruce-A NGS in a LOCA plus failure of ECI plus failure of the moderator cooling system.

We may conclude that the inherent characteristics and designs of CANDU reactors would preclude accidents of the magnitude experienced at Chernobyl, and that these characteristics and designs, especially the shut-down systems and other special safety systems, would greatly mitigate the consequences of any accidents that might occur, as happened in the TMI-accident. The ability of a CANDU reactor to survive a serious accident with minimal damage and with no adverse health effects to the public was clearly demonstrated by the pressure-tube rupture accident to Pickering-2 in August 1983.

While these insights and conclusions about CANDU safety are heartening, we must recognize the significance of the human element in the Chernobyl accident, as in the TMI-2 accident. While the inherent characteristics of the RBMK resulted in a great disaster, the ultimate cause was a human error, coupled with a number of serious violations of procedures and common sense, as summarized in Table 5. While it would be more difficult, physically, to violate certain of these procedures (e.g., blocking of trip signals) in a CANDU station than it apparently was at Chernobyl, operators can still make mistakes under stress or under pressure, as was the case at Chernobyl. This accident emphasizes, once again, the need for the continuation of thorough operator training and the need for ensuring that nuclear utility employees and management must be very conscious of their responsibilities for public and worker safety at all times. Perhaps this is the most important lesson to be learned from the Chernobyl accident.

Table 5: Chernobyl Reactor Accident: Violations of Procedures before and during Planned Test

1. Inadequate attention to safety in written program for test.
2. Emergency coolant injection system blocked out.
3. Test conducted at 200 MW(t) instead of 700–1000 MW specified (Error in establishing control set-point).
4. Reactivity margin reduced below required level.
5. Total coolant flow rate through core higher than permitted.
6. Trip signals for low separator water level and low separator pressure blocked out.
7. Trip signal for closure of turbine stop valve blocked out.

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Notes

1. There is some uncertainty in the reactivity worths of the control and protective elements for the RBMK reactor as given here. The values given in Table 2 represent the author's interpretation of reference 1 after consulting with Canadians who attended the IAEA Experts' Meeting in Vienna, at which the information in reference 1 was presented.
2. Reference 1 states that the repeat test was to be performed in case the first test failed, not because two different voltage controls were to be tested. However, a Canadian delegate who remained after the official IAEA meeting was informed that the latter was the reason for ensuring two tests could be done. A repeat test required the reactor to continue to operate during the first test.
3. The positive void reactivity coefficient was about 50% greater than normal because of the particular core operating conditions.
4. It is unlikely that the accident consequences would have been mitigated to any significant extent had the ECI not been blocked. However, this is not the question that concerns us here.
5. The nominal melting point of UO_2 is about 2,800 degrees Celsius and that of Zircaloy is about 1,750 degrees Celsius.
6. For the estimated power excursion to 100 times full power in 4 seconds, assuming a linear power ramp, the average energy deposited in the fuel by this excursion was about 800 cal / gm. Note that the level of energy storage to cause fuel disintegration generally accepted in other countries, including Canada, is 200 cal / gm based on a conservative interpretation of TREAT and SPERT experiments [19].

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Whole-Body Retention of ^{99m}Tc-imidodiphosphate: A Measurement of Total Body Bone Turnover Rate

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Abstract

The 24-hour retention of ^{99m}Tc-imidodiphosphate has been measured in 39 subjects. The mean retention for five normal premenopausal women was 49 ± 5%. For postmenopausal women the mean retention was 57 ± 6%. The mean retention for eight patients suffering from Paget's disease was 75 ± 13%. Retention was reduced from 57 ± 8% to 52 ± 7% following estrogen replacement therapy. These results support the contention that ^{99m}Tc-imidodiphosphate is retained *in vivo* to a greater extent than other ^{99m}Tc-labelled compound phosphates, and that such retention reflects the rate of whole-body bone mineral turnover.

Résumé

Nous avons mesuré la rétention du ^{99m}Tc-imidophosphate durant 24 heures dans 39 sujets. La rétention moyenne dans cinq femmes normales n'ayant pas atteint leur ménopause est de 49 ± 5%. Dans les femmes ayant dépassé l'âge de la ménopause la rétention moyenne est de 57 ± 6%. La rétention moyenne dans huit malades souffrant de la maladie de Pajet est de 75 ± 13%. La rétention peut être réduite de 57 ± 8% à 52 ± 7% après un traitement de remplacement des oestrogènes. Ces résultats supportent la proposition que la rétention *in vivo* du ^{99m}Tc-imidophosphate est plus grande que celle d'autres composés de phosphate marqués au ^{99m}Tc et que cette rétention réflète le taux de renouvéllement du contenu minéral des os.

Introduction

The uptake of ^{99m}Tc-labelled condensed phosphates by bone is such that images of the skeleton can be obtained which are of sufficiently high quality to allow the detection of various bone disorders [Merrick 1975; Ram and Fordham 1979; Matin 1983]. The fractional retention of such radiopharmaceuticals 24 hours after injection is thought to be an indication of the rate of skeletal turnover. Condensed phosphates that have been used for this measurement include pyrophosphate [Martin *et al.* 1983] and various diphosphonates such as methylene diphosphonate [Hyldstrup *et al.* 1984] and hydroxyethylidene diphosphonate [Fogelman *et al.* 1978].

The fractional uptake of ^{99m}Tc-labelled imidodiphosphate (^{99m}Tc-IDP) into the skeleton is greater than that of either pyrophosphate or the diphosphonates [Brody *et al.* 1976] and has led to the claim that ^{99m}Tc-IDP is the optimum condensed phosphate for bone imaging. The purpose of the work reported here is to present the results of measurements of the 24-hour whole-body retention of ^{99m}Tc-IDP in specific groups of patients in whom bone turnover is expected to be different. To our knowledge, these are the first such measurements using ^{99m}Tc-IDP.

Materials and Methods

Diphosphate retention was measured in a total of 39 subjects divided into three groups: five premenopausal women, 26 postmenopausal women, and eight patients suffering from Paget's disease. The five premenopausal women who served as control subjects were university students with no evidence of bone disease, and were aged from 19 to 30 years.

The effect of age upon retention was assessed cross sectionally from measurements in 26 women attending a menopause clinic. Their ages ranged from 34 to 66 years, and the time since menopause varied from one to 26 years. At the time of the study, three of the 26 women were receiving estrogen replacement therapy.

Keywords: ^{99m}Tc-imidodiphosphate whole body retention, premenopause, postmenopause, Paget's disease, estrogen replacement therapy.

The effect of Paget's disease was assessed from retention measurements in six men and two women. These patients were aged between 55 and 84 and were referred from a metabolic bone disease clinic. The serum alkaline phosphatase in these patients ranged from 131 to 357 U / L (normal < 90).

The effect of estrogen replacement therapy upon retention was assessed using a longitudinal before-after study design. Retention measurements were repeated in eight of the 26 postmenopausal subjects after 6–10 months of estrogen replacement therapy (conjugated estrogen 0.625 mg daily with, in five of the eight women, the addition of 5 mg cyclic progestogen during the last seven days of each estrogen cycle). None of the eight had received supplements prior to the first retention measurement, their ages ranged from 36 to 57 years, and the number of years since menopause ranged from one to 27.

For each retention measurement, approximately 1 MBq of ^{99m}Tc-IDP containing 2 mg of imidodiphosphate is injected at zero time. Whole-body radioactivity is measured at 15 mins and 24 hours using a shadow shield counter containing a single 27.9 × 10.2 cm NaI (Tl) detector. For the first measurement the couch speed is 40 cm min^{-1}. At 24 hours the couch speed is 10 cm min^{-1}. Dead time effects are corrected with a pulser. The radiation dose associated with the retention measurement is about 0.03 mSv to the skeleton and about 0.1 mSv to the bladder [Graham *et al.* 1974].

The mean retentions for the three groups of subjects were compared by analysis of variance. Forward stepwise multiple regression was used to investigate the importance of other factors contributing to the variance of retention measurements in the patients with Paget's disease. The influence of estrogen replacement therapy was assessed using a one-way paired t-test.

Results

The means and standard deviations for the measured retentions in the five controls, the 26 postmenopausal women, and the eight patients with Paget's disease are compared in Figure 1. There is a statistically significant difference between these three groups (ANOVA, $F = 17.6$, $p < 0.05$). The means and standard deviations for the diphosphate retentions, body weights, ages, and heights for the three groups are given in Table 1. There was no difference between the mean retentions for the three postmenopausal women who had previously taken estrogen and the 23 women who had not (58 and 57%, respectively).

Diphosphate retention was lower in premenopausal women (49%) than in postmenopausal women (57%). This difference was statistically significant (Scheffé test, $p < 0.05$).

The mean retention for the patients with Paget's disease (75%) was significantly greater than that of the postmenopausal group (Scheffé test, $p < 0.05$). Forward stepwise multiple regression analysis indicated that sex and alkaline phosphatase levels were important independent variables that contributed to the variance of diphosphate retention in the patients with Paget's.

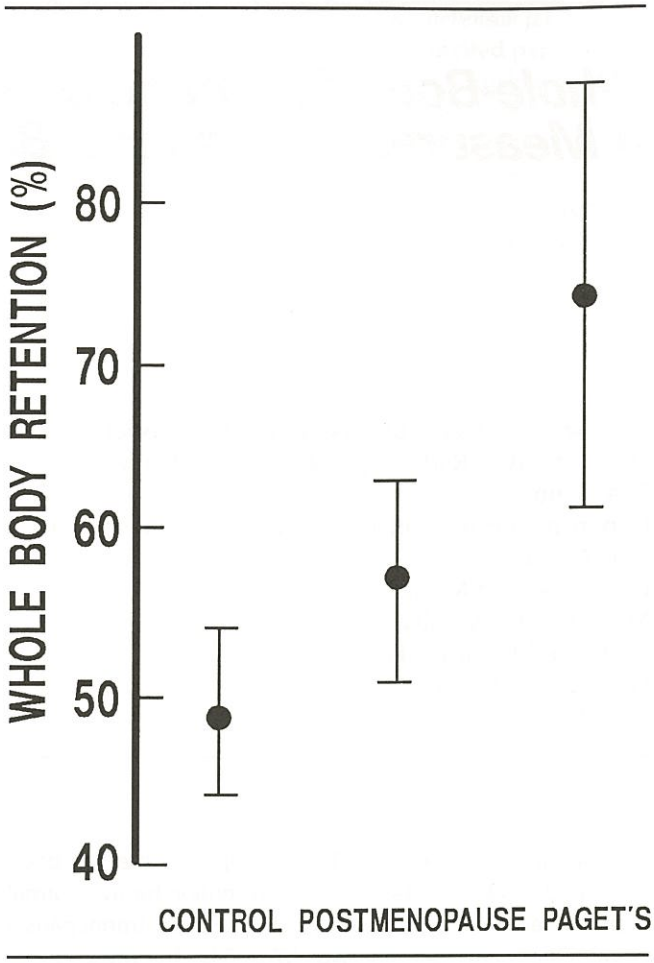

Figure 1: Mean whole body retention in each subject group. The vertical bars equal two standard deviations.

For the eight women placed on estrogen therapy there was an inverse correlation ($r = 0.60$) between the initial retention measurement and the number of years since menopause. With estrogen therapy, the mean retention in these eight women fell from 57 ± 8% to 52 ± 7%. This reduction was statistically significant (paired $t = 2.1$, $p < 0.05$). There was no relation ($r = 0.09$)

Table 1: The Means (Standard Deviations) for the Ages, Heights, Weights and Imidodiphosphate Retentions in Each Subject Group

	Controls	*Postmenopause*	*Paget's*
n	5	26	8
Age	23 (4)	56 (10)	67 (10)
Ht (cm)	161 (4)	162 (6)	167 (9)
Wt (Kg)	60 (7)	63 (10)	75 (14)
^{99m}Tc-IDP Retention (%)	49 (5)	57 (6)	75 (13)
Range	41–54	46–68	58–94

between the change in retention and the number of years since menopause. However, the change in retention was correlated with the initial retention value ($r = 0.53$).

Discussion

Our results show that young adults retain about one half of a dose of ^{99m}Tc-IDP, 24 hours after injection. This is about 10% greater than the mean retention of ^{99m}Tc-pyrophosphate measured in 17 women between the ages of 26 and 40 [Martin *et al.* 1983]. However, these same workers found a mean retention of 48% in three women between the ages of 20 and 25. The 24-hour retention of the diphosphonates is considerably smaller than that of both imidodiphosphate and pyrophosphate. For normal adults of various ages, Caniggia and Vattimo (1980) found a mean retention of 33% for ^{99m}Tc-methylenediphosphonate. Hyldstrup *et al.* (1984) measured a mean value of 30% for the same diphosphonate in women aged between 20 and 29 years. Again, in women between 20 and 29 years of age the mean retention of ^{99m}Tc-hydroxyethylidenediphosphonate was only 18% [Fogelman and Bessent 1982]. These observations suggest that the simpler the condensed phosphate the greater will be the retention of the ^{99m}Tc-labelled compound.

The increase in ^{99m}Tc-IDP retention following the menopause is similar to the increases observed with pyrophosphate [Martin *et al.* 1983] and the diphosphonates [Fogelman and Bessent 1982; Caniggia and Vattimo 1980] and probably reflects an increase in bone resorption due to the reduction of endogenous estrogen [Gallagher 1981]. The different retention values between pre- and postmenopausal women are unlikely to be due to age differences, since it has been shown that, at least for pyrophosphate and the diphosphonates, an age-dependent increase in retention is not observed until the eighth decade [Martin *et al.* 1983; Hyldstrup *et al.* 1984]. This age-dependent increase appears to be related to both increased resorption of bone and a reduction in glomerular filtration rate.

As expected, the patients with Paget's disease showed a higher mean retention than both the control and postmenopause groups. Multiple regression revealed that sex and the serum alkaline phosphatase levels were associated with the 24-hour fractional retention of ^{99m}Tc-IDP. The significance of the former correlation is not clear, although it is known that retention is normally greater in males than in females [Fogelman and Bessent 1982] and six of our eight patients were male. The correlation with the serum alkaline phosphatase indicated that both parameters reflect the severity of the disease.

In the small number of women selected for estrogen replacement therapy, an inverse correlation was detected between the initial retention measurement and the number of years since menopause. This suggests that in these patients there was an increased rate of bone turnover during the early postmenopausal years. After six to 10 months of replacement therapy, there was a significant reduction in ^{99m}Tc-IDP retention. This retention was greatest in those subjects with the highest initial retention values, suggesting that the effectiveness of exogenous estrogen was greatest in those with the highest resorption. Since the change in retention was not related to the number of years since menopause, the effectiveness of exogenous estrogen is not restricted to the early postmenopausal years. These observations are consistent with the concept that exogenous estrogen reduces bone resorption, and the coupling of resorption to formation decreases osteoblastic cellular activity, with a consequent reduction in whole-body diphosphate retention. Although the number of subjects studied is small, and only limited confidence can be placed in these conclusions, our results do support the suggestion of Fogelman *et al.* (1980) that the retention of ^{99m}Tc-labelled condensed phosphates may be used to titrate the effect of a specific dose of estrogen in each subject and to select those who may or may not benefit from replacement therapy.

The data reported here suggest that ^{99m}Tc-IDP is another labelled condensed phosphate, the 24-hour retention of which can be used as an indicator of the rate of whole-body bone turnover. It remains to be shown whether or not ^{99m}Tc-IDP has any advantages in discriminating among various forms of metabolic bone disease.

Acknowledgements

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Corrosion of Heat Exchanger Materials under Heat Transfer Conditions

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Abstract

Severe pitting has occurred in moderator heat exchangers tubed with Incoloy-800 in Pickering Nuclear Generating Station. The pitting originated on the cooling water side (outside) of the tubes and perforation occurred in less than two years. It was known from corrosion testing at CRNL that Incoloy-800 was not susceptible to pitting in Lake Ontario water under isothermal conditions. Corrosion testing with heat transfer across the tube wall was carried out, and it was noted that severe pitting could occur under deposits formed on the tubes in silty Lake Ontario water. Subsequent testing, carried out in co-operation with Ontario Hydro Research Division, investigated the pitting resistance of other candidate tubing alloys: Incoloy-825, 904 L stainless steel, AL-6X, Inconel-625, 70:30 Cu:Ni, titanium, Sanicro-30 and Sanicro-28.[1] Of these, only titanium and Sanicro-28 have not suffered some degree of pitting attack in silt-containing Lake Ontario water. In the absence of silt, and hence deposits, no pitting took place on any of the alloys tested.

Résumé

De fortes piqûres de corrosion se sont produites dans les échangeurs de chaleur à tubes d'Incoloy-800 du circuit de modérateur de la centrale électronucléaire de Pickering. Les piqûres ont pris naissance du côté eau de refroidissement (extérieur) des tubes et il y a eu perforation en moins de deux ans. On sait, d'après les essais de corrosion effectués aux LNCR, que l'Incoloy-800 n'est pas susceptible aux piqûres de corrosion dans de l'eau provenant du Lac Ontario dans des conditions isothermiques. On a effectué des essais de corrosion sous transfert de chaleur à travers la paroi de tubes et remarqué que de fortes piqûres de corrosion peuvent se produire sous les dépôts formés sur les tubes dans de l'eau vaseuse provenant du Lac Ontario. Lors d'essais ultérieurs, effectués en collaboration avec la division de recherche de l'Ontario Hydro, on a étudié la résistance aux piqûres de corrosion d'autres alliages pour tubes à retenir, à savoir: l'Incoloy-825, l'acier inoxydable 904 L, l'Al-6X, l'Inconel-625, le Cu:70–Ni:30, le titane, le Sanicro-30 et le Sanicro-28.[1] De tous ces alliages, il n'y a que le titane et le Sanicro-28 qui n'ont pas subi un certain degré de corrosion par piqûres dans de l'eau vaseuse provenant du Lac Ontario. En l'absence de vase et donc de dépôts, aucunes piqûres de corrosion ne se sont produites sur aucun des alliages soumis aux essais.

Introduction

Until recently it was considered that copper-based alloys, such as admiralty brass, aluminum brass, or the copper-nickel alloys, provided adequate corrosion resistance to fresh waters in heat exchanger service. Alloys such as the austenitic stainless steels or nickel-chrome-iron family were regarded as necessary only when brackish or sulphide-polluted water was used, and too expensive for most typical freshwater applications. Indeed, isothermal corrosion testing in freshwaters, in the presence of a crevice, indicated that alloys such as Incoloy-800, Inconel-600, and type 304 stainless steel were not susceptible to freshwater corrosion at temperatures up to 70°C [1].

It was somewhat surprising, therefore, to find that I-800 moderator heat exchanger tubes, installed in the Ontario Hydro Pickering Nuclear Generating Station (PNGS) (Unit 3) following fretting failure of 70:30 copper:nickel tubes, pitted to failure in approximately two years. This premature failure, the result of under-deposit crevice corrosion [2], raised concern because plans for subsequent nuclear reactors using freshwater cooling called for I-800 heat exchanger tubes.

The work to be described here is a program of corrosion testing, with and without heat transfer across the tube wall, carried out on various candidate heat exchanger tubing alloys used in low temperature (100°C) service. The tests with heat transfer were designed to reproduce the pitting observed in PNGS and appeared to accelerate the pitting process in I-800 by about 50%, relative to in-service failure rates. It is important to remember, in assessing the corrosion susceptibility of the austenitic alloys tested, that heat transfer criteria

Keywords: heat exchanger tube corrosion, alloy 800, alloy 825, Sanicro-28, pitting under heat transfer, fouling.

Table 1: Nominal Compositions of Alloys Tested

		Composition (wt%)					
Alloy	*UNS No.*	*Fe*	*Ni*	*Cr*	*Mo*	*Cu*	*Other*
Admiralty Brass	C44300					72	1 Sn, bal. Zn
90:10 Cu:Ni	C70600	1	10			88	
70:30 Cu:Ni	C71500	0.5	30			69	0.6 Mn
Monel	N04400	2.5	66			29	1.8 Mn, 0.5 Si
Type 304 SS	S30400	71	8	18			2 Mn, 1 Si
Inconel 690	N06690	9.5	60	30			
Inconel 600*	N06600	8	76	15			
Inconel 625*	N06625	2.5	61	21.5	9		
Incoloy 800*	N08800	46	33	21			
Incoloy 825*	N08825	30	42	22	3	2	1 Ti
Sanicro 30*	N08830	46	33	21			0.03 C
Sanicro 28*	N08028	38	31	27	3.5		
Titanium Gr 2							100 Ti
AL-6X	N08366	48	24	20			2 Mn
904L SS	N08904	44	25	22			2 Mn

*Nuclear grade materials were used.

require the use of thin-walled tubing for efficient heat exchanger design. Thus, simple remedies, such as increasing tube wall thickness, are not really practical. Similarly, copper-containing alloys are not used for newer nuclear power stations because of the low pH which can be produced as a result of the use of gadolinium nitrate for reactivity control during outages. These considerations imply that only two alternatives exist:

1) select a copper-free alloy with sufficient resistance to under-deposit corrosion under heat transfer conditions to provide 40 years' service;
2) provide an environment which does not lead to the buildup of silt or other deposits under which corrosion may occur.

In is known [3] that austenitic alloys, such as type 304 and 316, can tolerate quite high chloride levels (up to 200 ppm) in the absence of deposits. I-800 would generally be expected to have better corrosion resistance than 316 under these conditions. The choice between (1) and (2) is thus one of cost effectiveness; the balance being between the cost of a truly corrosion-resistant alloy that is approved for use in nuclear power stations, and the cost of a high-quality water cooling system.

Experimental Details

Isothermal Tests

Samples of various heat exchanger tubing materials (see Table 1 for nominal compositions) were exposed to cooling waters taken from Lake Ontario (see Table 4) and the St Lawrence River (Gentilly Site Water, 20 ppm Cl^-, 26 ppm $SO_4^=$) for times up to two years. Water temperatures were maintained at room temperature (20°C average), 45°C, and 70°C. Silt and chloride ion were added to some of the waters and samples were held in a polyethylene 'tubesheet' to simulate a typical in-service crevice. The degree of corrosion was assessed by visual and metallographic examination, as necessary, following sample washing and brushing.

Heat Transfer Tests

Figure 1 shows the experimental arrangement used to conduct the tests with heat transfer. These tests were carried out primarily using Lake Ontario water. The alloys used are listed in Table 1, along with the nominal compositions. The tube inner wall temperature was controlled at 60 or 70°C. The mean moderator water temperature in the moderator heat exchanger was approximately 60°C. The silt used for the crevices was

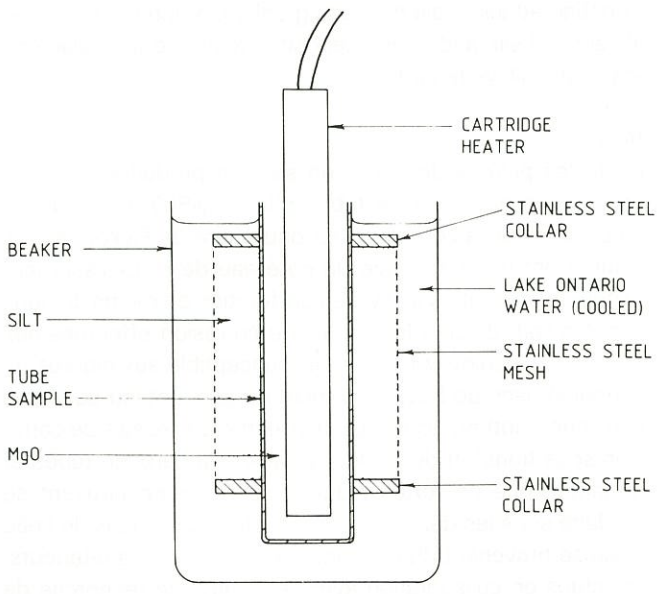

Figure 1: Schematic of test apparatus used for corrosion tests with heat transfer and silt crevices. The stainless steel used for collars and mesh was type 304.

Table 2: Summary of Long-Term Isothermal Corrosion Test Results*

Alloy	*St Lawrence River, 20°C Silt + 1000 mg / kg NaCl*	*St Lawrence River, 45°C*	*St Lawrence River, 70°C*	*Lake Ontario, 45°C*
Incoloy 800	crevice attack 0.1 mm / a	NA	NA	NA
Inconel 600	crevice attack 0.04 mm / a	NA	NA	NA
Inconel 690	crevice attack 0.09 mm / a	NA	NA	NA
Type 304 SS	NA	NA	NA	NA
Monel 400	crevice attack 0.05 mm / a	NA	NA	NA
90:10 Cu:Ni	patchy attack 0.01 mm / a	etch 0.01 mm / a	NA	NA
70:30 Cu:Ni	patchy attack 0.06 mm / a	etch 0.01 mm / a	NA	NA
Admiralty brass	–	crevice attack 0.07 mm / a	crevice attack 0.2 mm / a	crevice attack 0.03 mm / a
	pH 7.7	pH 7.8	pH 9.1	

*Corrosion rates are average of 3 deepest portions of pit or crevice.
'–' = no data.
NA = not attacked.

obtained from the PNGS forebay, and used as received. Lake Ontario water, again obtained from PNGS, was cooled with a simple cooling coil using tap water, which maintained the Lake Ontario water at 16–22°C, depending on the season.

Testing times varied from four weeks to six months, and after exposure samples were washed and brushed. Pitting was generally only observed under very adherent deposits, and these were removed either chemically (Clarke's solution) or mechanically in order to locate the pitted areas. Pit depths were measured using metallography. Chemical analyses were carried out using X-ray diffraction (XRD) on a Scanning Electron Microscope (SEM) equipped with energy-dispersive analysis of X-rays (EDX). Surface analyses were carried out using Scanning Auger Microscopy.

Results and Discussion

Isothermal Tests

Selected results are shown in Table 2. None of the alloys tested at room temperature in natural freshwaters showed any evidence of attack after two years' exposure. Those alloys which suffered some degree of corrosion at 45 and 70°C are shown in Table 2. This was invariably crevice attack. The copper-based alloys, specifically admiralty brass, 70:30 and 90:10 copper: nickel, were slightly attacked in St Lawrence River water, the severity increasing with temperature.

In the presence of silt and chloride contamination (1,000 mg / kg Cl^- in H_2O; during spring run-off the near-shore concentrations of Cl^- in Lake Ontario water in the vicinity of PNGS frequently reach 200 mg / kg or more), both the nickel-based and copper-based alloys suffered some attack at room temperature. The degree of attack appeared to increase with temperature, but this observation is based on visual examination only.

As indicated in Table 2, I-800 was susceptible to crevice attack in room-temperature chloride-contaminated St Lawrence River water, the rate being 0.1 mm / a. No direct comparison of the pitting rate can be made with either the in-service data for I-800, 0.5 mm / a, or the heat transfer test results, because the isothermal tests at 70°C for contaminated Lake Ontario water were not carried out. However, it is clear that chloride contamination appears to be necessary in these isothermal tests to induce pitting attack of I-800 and the nickel-based alloys.

The comparisons of the admiralty brass results with in-service data are quite good. In-service admiralty brass condenser tube pitting rates of 0.13 mm / a over a ten-year period appear to be typical, and this value is in good agreement with the isothermal laboratory test results ranging from 0.03 to 0.2 mm / a, depending on temperature and water chemistry. Similarly, 70:30 copper:nickel in service in PNGS has shown onlt slight corrosion, usually an insignificant surface etching and dealloying, which is consistent with the results reported here. In both cases, 70:30 copper:nickel and admiralty brass, it is likely that heat transfer conditions would increase the corrosion rate.

Heat Transfer Tests

The results presented here are limited to alloys suitable for moderator heat exchanger service, and thus include only those alloys listed in Table 3. Where no silt crevices were used, no pitting was observed on any of

Table 3: Under-Deposit Pitting Depths for Various Alloys in Heat Transfer Tests*

	Exposure time (weeks)				
Alloy	*6*	*8*	*12*	*24*	*26*
Incoloy 800		0.45 mm			0.75 mm
Sanicro 30					0.4 mm
Incoloy 825	no pits	0.15 mm	no pits	0.2 mm†	0.7 mm (few pits)
Sanicro 28	no pits		no pits	no pits	
904L SS	no pits	0.3 mm	no pits	no pits	0.3 mm
Al-6X	no pits		no pits	no pits	0.9 mm (few pits)
Titanium Gr 2	no pits		no pits		no pits
Inconel 625	no pits		no pits	no pits	no pits
70:30 Cu:Ni	no pits		slight etch		slight etch

*Tube inner wall temperature 60°C.
†One measurement was at 70°C inner wall temperature; pitting depth there was 0.3 mm.
200 ppm Cl^- in water.

these materials. The results in Table 3 represent an average, at each temperature, of several tests. In the case of I-800, I-825, and Sanicro-28, four or five tests were carried out, particularly at the longer exposure times. The statistical distribution of the pitting was such that results ranged from no pitting to severe pitting on materials such as I-825, and I-800 always suffered some pitting; Sanicro-28 never did show any evidence of pitting attack. Figure 2 presents the pitting data in terms of pit depth as a function of time.

The results for the other alloys tested, AL-6X, type 904L stainless steel, titanium grade 2, Inconel-625, and 70:30 Cu:Ni are an average of only two tests at each temperature. In these cases, the results were very reproducible within the duplicates, but insufficient sampling was carried out to provide statistically-significant predictability. The results in these cases do provide, however, a clear guide as to the type of alloy that would be suitable for corrosion-free service as tubing in moderator heat exchangers.

As shown in Table 3, I-800 is quite susceptible to pitting, I-825 somewhat less so but still susceptible, and AL-6X and type 904L stainless steel subject to occasional deep pitting. Figures 3 to 6 show some maximum depth pits for I-800, I-825, AL-6X, and type 904L stainless steel. The pit shapes are characteristic of under-deposit chloride attack. Sanicro-30 was less sus-

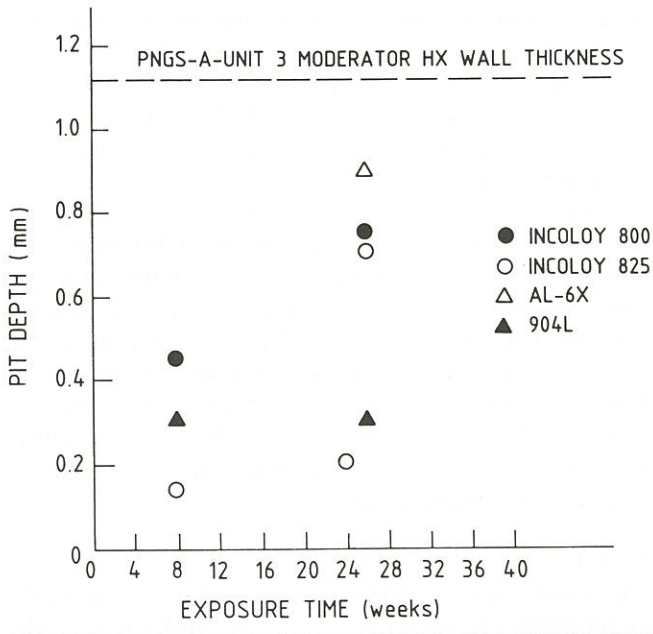

Figure 2: Average maximum pit depths for moderator heat exchanger tube materials subjected to laboratory testing in Lake Ontario water.

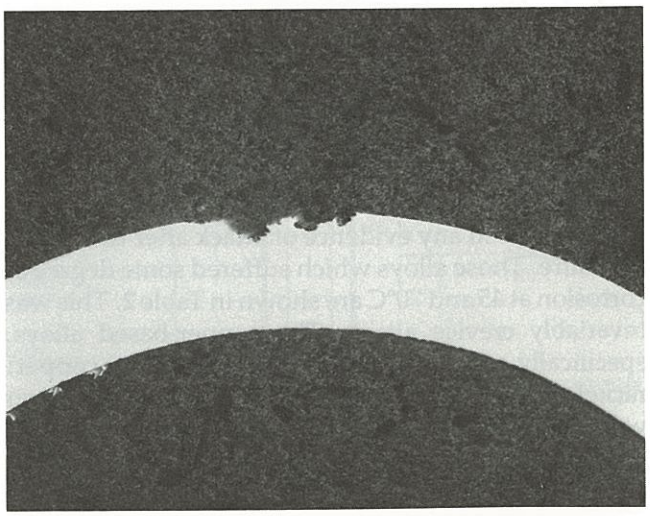

Figure 3: Typical deep pit in Incoloy 800 after two months' exposure to Lake Ontario water (with a crevice).

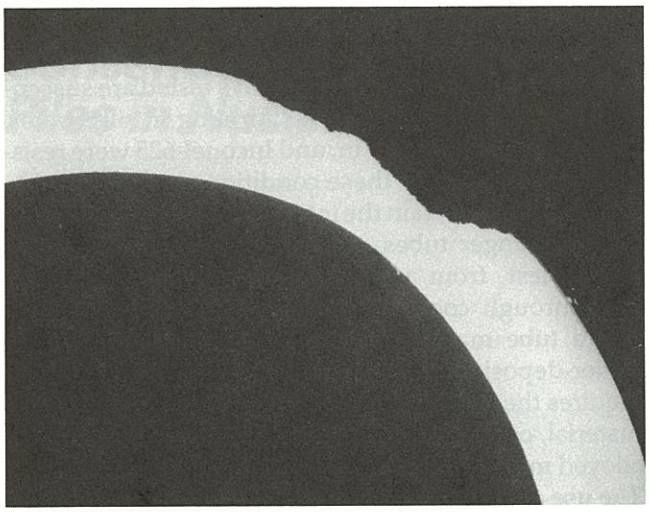

Figure 4: Typical deep pit in Incoloy 825 after 26 weeks' exposure to Lake Ontario water (with a crevice).

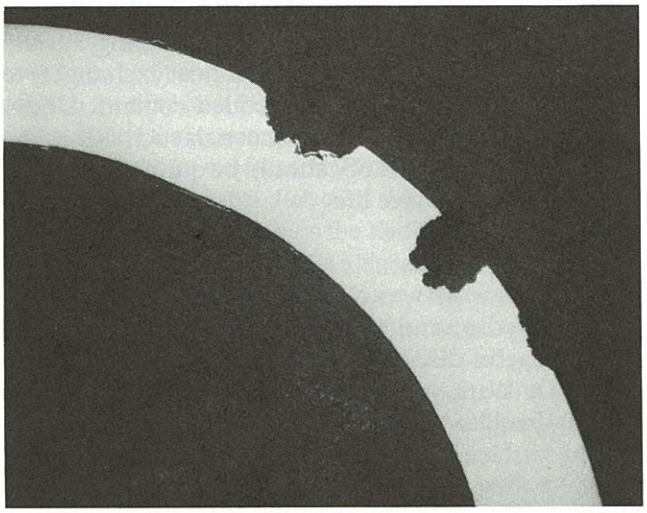

Figure 5: Typical deep pits found on AL-6X sample after 26 weeks' exposure to Lake Ontario water (with a crevice).

ceptible to pitting than I-800, but insufficient testing was carried out to confirm this conclusion. Sanicro-28 was completely resistant to pitting over the range of tests carried out here. This was also true of grade 2 titanium and Inconel-625.

PNGS has operated sucessfully with 70:30 Cu:Ni tubes for some years, and the results of the experiments reported here are in agreement with this observation. The only sign of corrosion on the material was under very adherent deposits, where redeposition of leached copper was noted. The corrosion in this area was limited to slight etching which had penetrated less than 1 μm in six months.

In most instances, the observed corrosion on alloys suffering attack was associated with dark green or

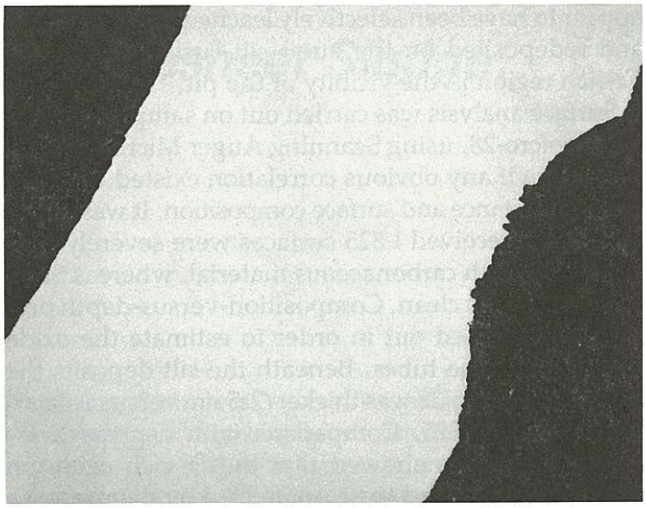

Figure 6: Pit noted on 904L sample after 26 weeks' exposure to Lake Ontario water (with a crevice).

black deposits, which had formed at the tube-silt deposit interface. Analyses of these dark areas showed them to be rich in Ca, S, and Cl, as well as containing significant amounts of chromium. The outer portion of the overall silt deposit (i.e. at the water interface) was found to contain Ni, Fe, Si, Ca, and Cl. Before the test the silt was found to be composed mostly of Ca and Si, with small amounts of S (probably as sulphate) and Cl (present as 30 mg/kg Cl^-). Table 4 shows the typical Lake Ontario water chemistry. Because EDX can detect elements such as Cl and S only at concentrations greater than 0.1 atomic %, it is apparent that considerable concentration of these elements has occurred in the dark deposits found on the tube surface. The concentration mechanism is diffusion of chloride ion to the acidic environment that is found in crevices formed

Table 4: Typical Lake Ontario Water Chemistry

pH	8.7
Conductivity	365
Cl^- (mg/kg H_2O)	30
$SO_4^=$ (mg/kg H_2O)	75
Ca^{2+} (mg/kg H_2O)	38–41
Mg^{2+} (mg/kg H_2O)	7.5–8
Na^+ (mg/kg H_2O)	10–14
Alkalinity (mg/kg H_2O)	115
Hardness (mg/kg H_2O)	126
Silt* (turbidity units) (surface)	350–400
Suspended solids (mg/kg) (5 foot length)	50
Total dissolved solids (mg/kg)	309

*Samples taken 1973.04.06 at various depths of intake channel.

in aerated water. Under these deposits, Fe and Ni appear to have been selectively leached from the alloy, and redeposited on the outer silt surface, leaving a Cr-rich region in the vicinity of the pit.

Surface analysis was carried out on samples of I-825 and Sanicro-28, using Scanning Auger Microscopy, to determine if any obvious correlation existed between pitting resistance and surface composition. It was found that the as-received I-825 surfaces were severely contaminated with carbonaceous material, whereas Sanicro-28 was very clean. Composition-versus-depth profiles were carried out in order to estimate the oxide thickness on the tubes. Beneath the silt deposits the oxide on Sanicro-28 was thicker (2.5 nm versus 1.0 nm) than that on I-825. Comparison with degreased as-received samples showed that the Fe-rich oxide on Sanicro-28 appeared to be unaffected by the presence of silt deposits, whereas that on I-825, which was Ni-rich, was considerably thinned. Sulphur contamination was found in the sub-silt I-825 oxide, whereas none was noted on the Sanicro-28 surface. These results suggest that the initial surface condition of the tubing material is of some importance. This is clearly related to the composition of the passive film, but more work needs to be carried out to substantiate this. The role of biological agents also needs to be clarified, and work of this nature is underway at Ontario Hydro Research Division.

Summary and Conclusions

Laboratory studies have shown that the under-deposit pitting attack observed on I-800 tubes in PNGS moderator heat exchangers may be simulated with a simple corrosion test incorporating heat transfer across the tube wall. The laboratory test may be somewhat more aggressive than in-service conditions, and the data summary presented in Figure 2 indicates that the time to failure for I-800 in laboratory tests would be approximately one year, and actual in-service failures were occurring in approximately one to two years. [2]. This small discrepancy may be at least partially the result of temperature differences. Indications from this work are that an increase in inside-wall temperature from 60 to 70°C results in a 50% increase in pitting rate. These data have not been published here (there is one point in Table 2) because variations in pitting susceptibility are too high to demonstrate effectively the temperature effect. Almost all these laboratory studies were carried out using Lake Ontario water taken from one sampling (approx. 30 mg / kg chloride), whereas in-service impurity concentrations fluctuate from less than those shown in Table 4 to considerably more. Biological activity is also seasonal and may prove to have a significant effect on variations in pitting rate.

On the basis of results given here, isothermal corrosion testing is inadequate to reproduce and / or predict in-service failure of heat exchangers, where heat transfer and deposit formation is occurring. The heat transfer tests show that most of the alloys tested are susceptible to pitting under deposits forming in silty water. Only Sanicro-28, titanium, and Inconel-625 were resistant to pitting under these conditions.

In order to maintain the integrity of low-temperature heat exchanger tubes, two approaches seem viable. The easiest, from a design point of view, is to use once-through coolant, such as Lake Ontario water, and a tube material that is completely resistant to under-deposit corrosion in that water. This choice requires the use of alloys such as titanium for the tube material, or, based on the tests carried out here, highly alloyed materials, such as Sanicro-28 and Inconel-625. The use of such materials will involve regulatory code approvals and, often, special fabrication techniques, in addition to high raw materials costs. Titanium is being used, however, in increasing amounts where seawater cooling is used, and other materials are simply not adequate to guarantee long-term service.

Another approach, that was used by several CANDU 600 stations, is to use a recirculating cooling system (which itself uses a titanium-tubed heat exchanger) to provide clean, chemically controlled coolant. Under these circumstances, the use of materials such as Incoloy-800, and even type 304 SS, should be quite adequate to provide a long service life.

Acknowledgements

Some of this work was carried out in conjunction with Ontario Hydro, and we acknowledge their support. Several useful discussions took place with R.G. Barton, B.D. Warr, and P.J. King of Ontario Hydro. I would also like to thank R.D. Davidson for carrying out the surface analysis measurements.

Notes

1. Incoloy and Inconel are trade names of the International Nickel Company of Canada, AL-6X is a trade name of Allegheny-Ludlum, and Sanicro is a trade name of Sandvik Steel.

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Concentration Processes under Tubesheet Sludge Piles in Nuclear Steam Generators

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Abstract

The process by which bulk water solutes are concentrated under tubesheet sludge piles in nuclear steam generators was investigated in the laboratory under simulated CANDU operating conditions. Concentration rates were found to depend on the tube heat flux and pile depth, although beyond a critical depth the concentration efficiency decreased. This efficiency could be expressed by a concentration coefficient, and was found to depend also on the sludge pile porosity. Solute concentration profiles in the sludge pile suggested that the concentration mechanism in a high-porosity / permeability pile is characterized by boiling mainly near or at the tube surface, while in low-porosity piles, the change of phase may also become important in the body of the sludge pile. In all cases, the full depth of the pile was active to some extent in the concentration process. As long as the heat transfer under the pile was continued, the solute remained under the pile and slowly migrated toward the bottom. When the heat transfer was stopped, the solute diffused back into the bulk solution at a rate slower than that of the concentration process.

Résumé

L'accumulation des espèces chimiques solubles sous les dépôts de produits de corrosion sur la plaque tubulaire des générateurs de vapeur dans les centrales nucléaires CANDU a été étudié au laboratoire. Le taux de concentration dépend du flux thermique ainsi que de l'épaisseur des dépôts. Au dela d'une profondeur critique, l'efficacité de concentration diminue. Cette efficacité peut être exprimée par un Coefficient de Concentration, qui depend aussi de la porosité des dépôts. La distribution des solutés dans les dépôts indique que dans le cas des dépôts à porosité élevée, l'ébullition a lieu près de la surface des tubes, tandis qu'avec des dépôts de porosité basse, l'ébullition dans la masse des dépôts devient plus importante. Dans tous les cas, la profondeur entière des dépôts est plus ou moins active dans l'accumulation des impuretés. Tant que le transport de chaleur continue, les solutés restent sous les dépôts et sont transportés de plus en plus profondément. Quand le transport thermique cesse, les solutés retournent par diffusion vers l'eau du générateur de vapeur. Ce processus de diffusion est beaucoup plus lent que celui de concentration.

Introduction

Steam generators are crucial components of pressurized water reactors. Steam generator failure as a result of tube degradation has been a major cause of PWR plant unavailability world-wide. Steam generator problems have caused major economic losses in terms of lost electricity production through forced unit outages, and, in cases of extreme damage, as additional direct costs for large-scale repair or replacement of steam generators.

Steam generator tubes are susceptible to failure by a variety of mechanisms, the vast majority of which are related to corrosion [1]. Although in some cases this attack has occurred on the inside of the tubes (i.e. in the primary coolant), it has more usually originated on the outside of the tubes or on the tube support structures (support plates, lattice bars, tubesheets, scallop bars) on the secondary side of the steam generator.

The feedwater that enters the steam generators under normal operating conditions is extremely pure, but nevertheless contains low levels (generally in the μg / L concentration range) of impurities such as iron, copper, chloride, sulphate, etc. When water is converted to steam and exits the steam generator, the non-volatile impurities are left behind. As a result, their concentrations in the bulk steam generator water are considerably higher than those in the feedwater. Nevertheless, the concentrations of corrosive impurities are still generally sufficiently low that the bulk water is not significantly aggressive towards steam generator materials (in exceptional circumstances, such as massive condenser failures or major make-up water treatment plant

Keywords: steam generator, sludge piles, tubesheet, hideout, concentration processes, dry-out, corrosion product deposits, secondary side, water chemistry, feedwater, thermalhydraulics, boiling, modelling.

excursions, bulk steam generator water chemistry may deteriorate sufficiently to become directly aggressive, but such occurrences are relatively rare). However, in certain restricted-flow areas at the heat transfer surface, boiling causes further localized increases in impurity concentrations, and conditions may develop that are aggressive to steam generator materials.

The performance of the recirculating steam generators in Ontario Hydro CANDU plants has been outstanding. The percentage of tubes plugged (~0.02%) is well below the industry average of ~2%. Steam generator tube integrity is particularly important in a CANDU reactor because of the severe economic penalty of losing heavy water from the primary system. The excellent performance to date of CANDU steam generators can be attributed, in part, to their design and performance characteristics, which typically involve higher recirculation ratios and lower heat fluxes and temperatures than do other PWR steam generators. Tube support structures are either tri-foil broached plates or lattice bar supports, thus minimizing the likelihood of 'denting' attack. There are no deep tube/tubesheet crevices because of a roll near the secondary face of the tubesheet.

Tubesheet Sludge Piles

Sludge piles have accumulated on the tubesheets in CANDU units, as in most steam generators world-wide. In this region, the thermalhydraulic conditions are favorable for particle formation and deposition. The sludge piles are kidney-shaped and are deepest in the middle of the hot leg bundle. At the Bruce 'A' plant, the piles are estimated to be up to 12 cm deep, while in the Pickering 'A' steam generators they are up to 50 cm in depth. The sludge piles have been determined to be rock-hard, despite the fact that the units have always used AVT chemical control. Analysis of the sludge samples indicates that their main constituents are magnetite and elemental copper. In the Pickering samples, high concentrations of silicates and sulphates of calcium and magnesium have been detected. These are the result of chronic condenser leakage experienced in the Pickering 'A' units. The Pickering sludge samples were found to have high compressive strength (~17 MPa) and relatively low porosity (between 20 and 35%). Some of the samples showed a layered structure, with platelets of magnetite embedded in a matrix of calcium sulphate/silicate. The composition and structure of the samples are indicative of considerable *in situ* precipitation and recrystallization of the constituents, presumably driven by temperature and concentration gradients in the porous sludge pile [2].

Ontario Hydro is conducting a research program to investigate corrosion processes under tubesheet sludge piles, since this is the area where our steam generators are judged to be at greatest risk of widespread degradation. Although the build-up of concentrated solutions in sludge piles and crevices is qualitatively understood, there are a number of aspects of the concentration process where more detailed information is required in order to predict the effect of operating events on materials integrity. Most of the detailed studies reported to date have been carried out with tube/support plate crevices and tube/tubesheet crevices. While indicative of the overall processes that take place in confined regions of steam generators, these results can not be directly applied to describe the behaviour under sludge piles. This paper describes the results of experiments carried out at Ontario Hydro to study the influence of some physical characteristics of sludge piles (depth, porosity) on the rate of accumulation of contaminants (so-called 'hideout'), on the de-concentration ('hideout return') of the contaminants when heat transfer stops, and on the distribution of the contaminants in the sludge pile.

Solute Concentration under Sludge Piles

On the free, unobstructed tube surfaces of a nuclear steam generator, heat is transferred by nucleate boiling. In this process, the phase change occurs on the tube surface. The bubbles generated move away from the surface due to buoyancy forces and large quantities of liquid are also moved because of the turbulence generated, thus effectively preventing the accumulation of solutes which may be present in the bulk water. Concentration factors of only five or less have been measured in this type of boiling [3].

The mechanisms of the boiling process in confined regions can be drastically different. The flow of liquid to the heat transfer surface may be insufficient compared to the amount required to sustain nucleate boiling. Vapour tends to stay near the heat transfer surface, since it cannot easily escape. The remaining liquid is not replaced as effectively as in the nucleate boiling process, and as a consequence it may become enriched in the bulk water solutes.

The accumulation of vapour near the heat transfer surface increases the resistance of the medium to heat transfer. Consequently, monitoring the temperature of the tube wall can provide an indication of the heat transfer regime. This approach has been used and results are reported in the literature [4].

Although the heat transfer process determines the vaporization rate, and this in turn the concentration rate under the pile of the bulk water solutes, certain aspects of the mass transfer process cannot be well understood from the heat transfer data alone. For intance, in an alternate wetting and drying condition, the tube surface temperature changes with variable frequency, depending on the degree of dry-out in the region. To relate the wetting and drying frequency with the concentration rate is quite complex.

The flow of liquid in a boiling process under a porous body is determined by capillary forces. Several

flow configurations have been postulated. The liquid may flow towards the heat transfer surface along small pores, where capillary forces are important, while the steam flows away from the heat transfer surface through the larger pores. Alternatively, the liquid may be flowing along the periphery of the pores while escaping steam flows through the centre [4]. In either case, the concentration process under the sludge pile is strongly dependent on the structure of the porous body and on heat transfer parameters that determine the flow pattern. While wall temperature measurements can be indicative of the heat transfer process, they fail to explain fully the mass transfer into the pile. Therefore, it is necessary to measure directly solute concentrations to characterize the process. This was the approach followed in the present study.

Experimental

Two series of experiments were carried out (series A and B, respectively). Each series consisted of tests to determine concentration / deconcentration rates, and tests to characterize the concentration profiles in the sludge.

Autoclave Vessels

The experimental apparatus consisted of a four-litre Inconel 600 autoclave, equipped with a temperature-control system and a small volume (<10 mL) sampling condenser. The autoclave lid had several access ports, three of which allowed the insertion of the test probes. The remainder were connected with the control instrumentation. The autoclave was part of a high-pressure loop and could be operated in a refreshed mode or in a static mode.

Test Probes

The two series of experiments used test probes of different designs. The probes used in the series A tests were constructed by welding a length of steam generator alloy tube (Inconel 600, 13 mm o.d.) to a stainless steel fitting that could be attached to the autoclave lid. Electric cartridge heaters (9.5 mm diameter) were inserted into the probes, and the gap between the heater and the tube was pressurized with helium at a pressure of 9 MPa (see Figure 1). Helium was chosen for its high thermal conductivity and chemical inertness.

The probes for the series B tests used a 6.4 mm cartridge heater, and a copper sleeve placed between the heater and the tube (see Figure 2). Two 1 mm diameter sheathed thermocouples were positioned in a groove machined in the copper sleeve and the tips were insulated from the copper by means of a ceramic cement bead. The thermocouples could thus monitor the temperature of the inner wall of the tube. The presence of the copper sleeves permitted effective axial heat transfer and provided a closer simulation of the temperature-controlled heat transfer situation that occurs in a steam generator.

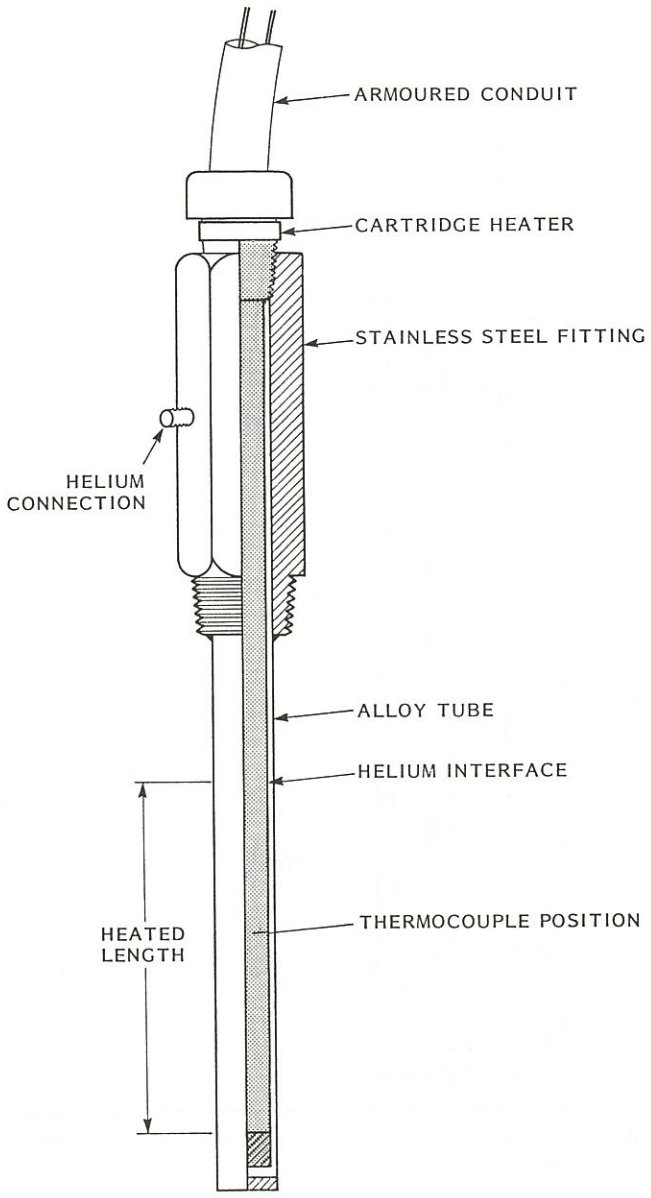

Figure 1: Test probe used in series A experiments.

Sludge Piles

To simulate a tubesheet sludge pile, the series A tests used a cylindrical carbon-steel holder clamped to the lower section of the tube probe (see Figure 3). The simulated sludge was obtained from the mud drum of a coal-fired station. This sludge and the deposits obtained from the U-bend and tubesheet regions of the Bruce NGS steam generators had similar chemical composition [2]. The sludge was compacted into the holder by tapping and vibrating. The final porosity of the pile was 0.57. Three sludge pile heights were used in the experiments: 50, 100, and 150 mm. Several thermocouples were fixed to the outside of the tube.

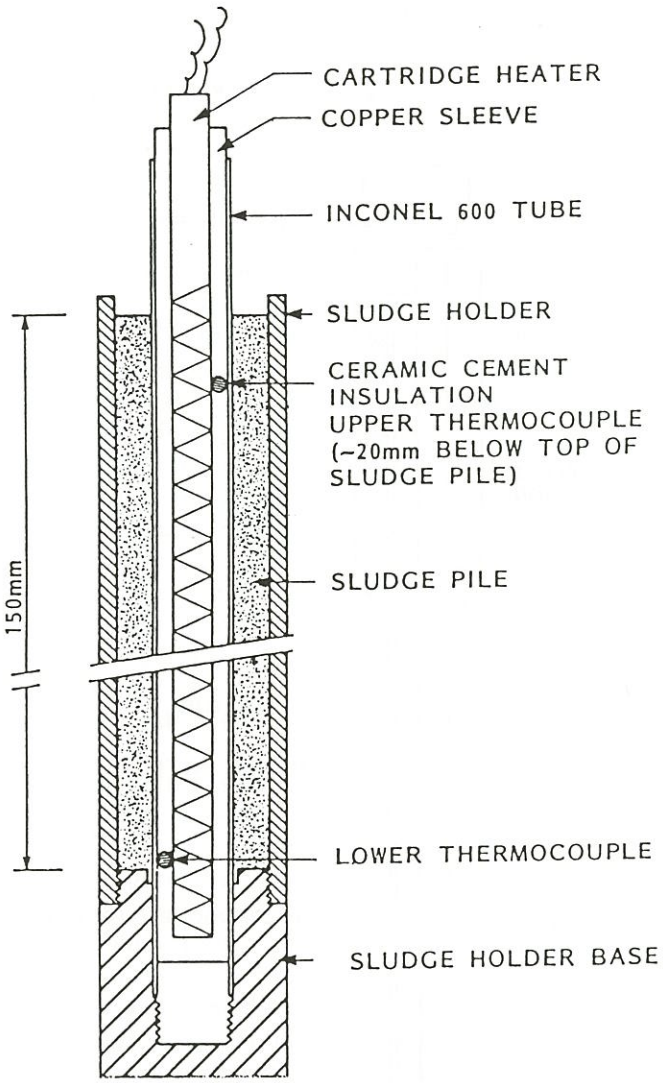

Figure 2: Test probe and sludge pile simulant used in series B experiments.

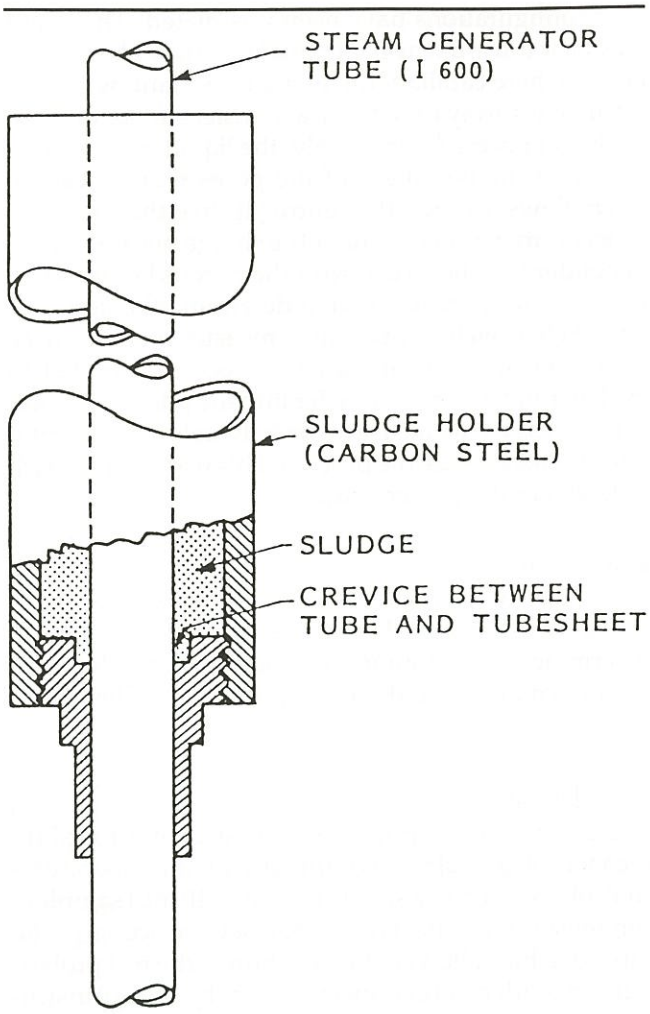

Figure 3: Sludge pile simulant used in series A experiments.

The sludge pile simulant for the series B experiments consisted of a holder, which was also made of carbon steel but was screwed onto the base that fit the bottom part of the steam generator alloy tube (see Figure 2). The same sludge simulant was used, but it was compacted into the holder using a plunger and a press. Compacting pressures of 1.2, 23, and 59 MPa produced sludge pile porosities of 0.57, 0.50, and 0.43, respectively.

Procedure

In both series of experiments, the concentration tests involved the operation of the autoclave in the static mode. The test solutions consisted of sodium chloride in deionized water (chloride concentration in the range 30–100 mg / L). The solution was adjusted to pH 8.5 with morpholine and dosed with 1 mg / L hydrazine. The autoclave vessel was charged with 2.7 L of solution and brought to the test temperature (256°C). Only one probe was tested in each experiment. The concentration process was started by supplying power to the heater. It was stopped by terminating the power supply. The concentrated solution under the pile was then allowed to diffuse into the bulk water. Samples were collected periodically during the experiment and analyzed for sodium, chloride, and conductivity. The concentration and de-concentration phases of an experiment typically lasted several days.

The heat flux was controlled differently in the two series of experiments. In series A, the desired heat flux was obtained by setting the voltage of the cartridge heaters to a prefixed value. In series B, the voltage was adjusted to obtain a temperature between 310 and 315°C at the inner wall of the tube. The resulting heat fluxes ranged between 20 and 25 kW / m^2.

The procedure used in the tests to determine the concentration profile under the sludge pile was also different in the two experimental series. In series A, the vessel was quickly emptied after a concentration period. The test probe was removed and the sludge in

the holder was extracted in 25-mm sections. The sodium and chloride content of each section was then determined. In series B, the vessel was operated in the refreshed mode to maintain a constant solution concentration of 100 mg / L chloride in the bulk water. The heat transfer process in the three probes carried on for eight days. The system was then shut down and the vessel quickly emptied. The sludge pile simulants were removed and cut radially into 12-mm sections that included the tube and carbon steel holder. These sections were mounted in epoxy resin and examined in a Scanning Electron Microscope (SEM) equipped with an Energy Dispersive X-ray attachment (EDXA) with mapping capabilities.

Results

Dynamics of Contaminant Concentration

Figure 4 shows the variation of the bulk water chloride concentration with time for a typical experiment from series A (100 mm sludge pile, 40 kW / m^2). The solute concentration in the bulk water continuously decreased after time zero (when the heat transfer through the tube was started). After approximately 50 hours, the solute concentration reached a value below the detectability limit of the analytic technique used (0.1 mg / L). As long as the heat transfer was maintained, the solute hideout in the sludge pile persisted. As soon as the heat transfer was stopped, the solute began to diffuse into the bulk water. The slightly altered probe design used in the series B tests showed the same behaviour, as shown in Figure 5. This test was carried out with a 150 mm sludge pile of porosity 0.50. In this experiment, as in others, the hideout return process was significantly slower than the concentration process. The same behaviour as that illustrated in Figures 4 and 5 was observed in all the concentration tests with different combinations of pile heights, porosities and heat fluxes.

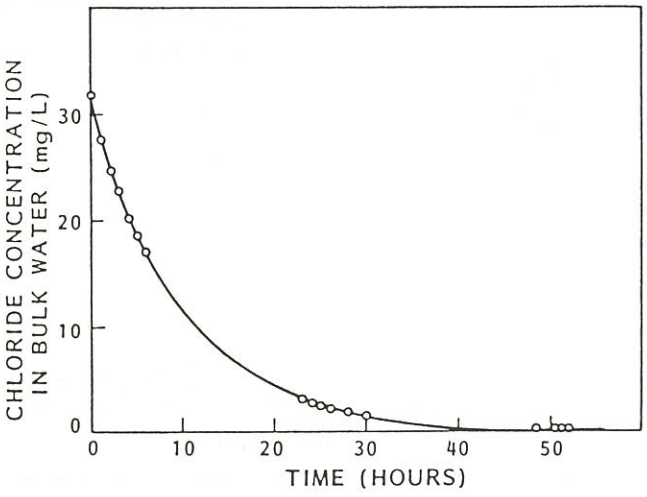

Figure 4: Bulk water chloride concentration versus time for a typical series A experiment (sludge pile depth: 100 mm; heat flux: 40 kW / m^2).

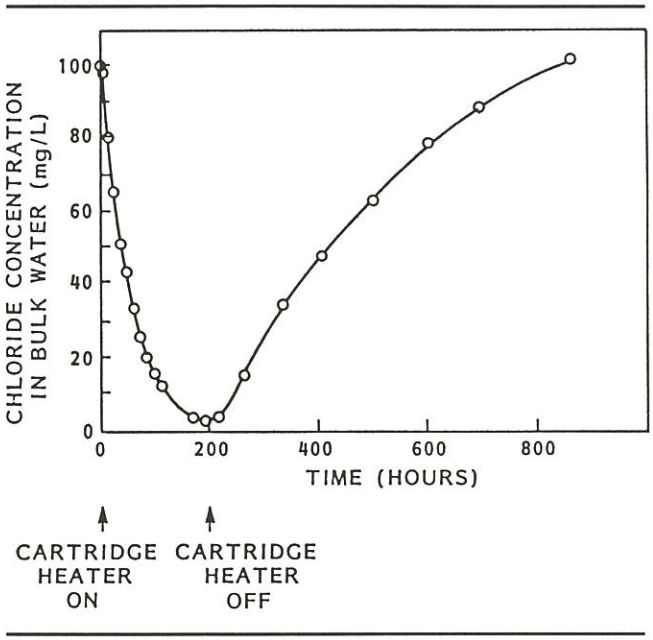

Figure 5: Bulk water chloride concentration versus time for a typical series B experiment (sludge pile porosity: 0.50; sludge pile depth: 150 mm; inner wall tube temperature: 310°C; heat flux: 20 kW / m^2).

Most of the concentration tests were carried out in duplicate. The reproducibility was satisfactory.

Influence of Pile Depth and Heat Flux

In the series A tests, the rate of concentration of sodium chloride under the sludge pile was measured at three heat fluxes (40, 75, and 110 kW / m^2) for the 50- and 100-mm-deep sludge piles and at 40 kW / m^2 for the 150-mm-deep sludge pile. The concentration rates increased when the heat flux was increased. This is shown in Figure 6, where the sludge pile concentration factor is plotted versus time for a 50-mm-deep sludge pile at three heat fluxes. The concentration factor is defined here as the solute concentration in the pile over the solute concentration in the bulk water. The solute concentration in the pile was obtained by calculating the solute hideout from a mass balance, and the void volume in the sludge pile from the initial porosity. The concentration factor calculated in this manner is an average value; actual factors are likely to be much higher in localized regions. Figure 6 shows that the concentration factor after 30 h can be greater than 10^5.

The influence of sludge pile depth on the solute concentration process is shown in Figure 7, in which the concentration factors are plotted vs time for three different pile depths at a heat flux of 40 kW / m^2. The 50-mm-deep pile showed lower concentration rates than the two deeper piles, whose concentration rate profiles were similar.

Influence of Sludge Pile Porosity

In the series B tests, the influence of sludge pile

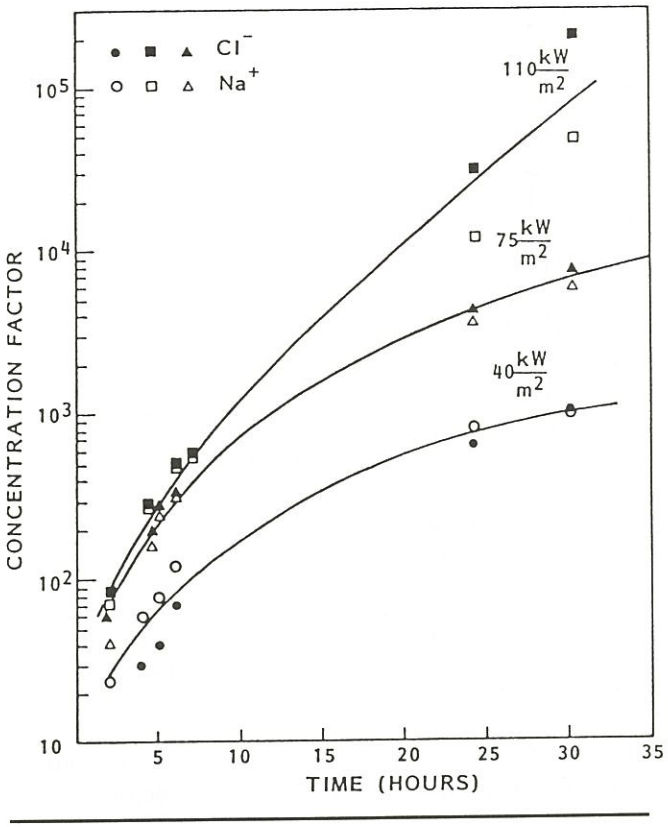

Figure 6: Concentration factors versus time at different heat fluxes (sludge pile depth: 50 mm).

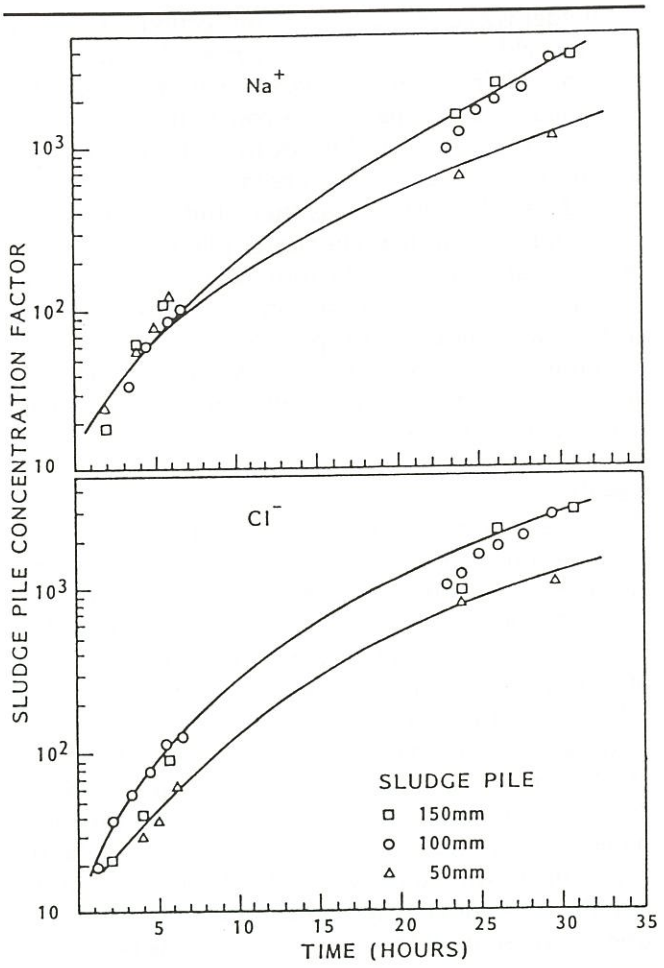

Figure 7: Concentration factors versus time for different sludge pile depths (heat flux: 40 kW / m²).

porosity on the contaminant concentration rate was investigated. Figure 8 shows the bulk water chloride concentration / time profiles for sludge piles with porosities of 0.43, 0.50, and 0.57. Heat fluxes during the experiment were 25, 20 and 20 kW / m², respectively. The concentration rate decreased with decreasing pile porosity.

Concentration Profiles Under Sludge Piles

Influence of Pile Depth

Figure 9 shows the sodium chloride concentration profiles for 150-, 100-, and 50-mm sludge piles after a concentration period of 100 h at 110 kW / m². In the 50- and 150-mm piles, the solute concentration increases exponentially with depth. The bottom of the pile had solute concentrations greater than 1,000 mg / kg. The trend observed in the 100-mm piles was similar, although less pronounced. At the end of each of the experiments, some sludge adhered very strongly to the tube and could not be dislodged, making the recovery of the sodium chloride solute incomplete. The most adherent scale was formed at the bottom of the sludge piles, and the 100-mm pile was more severely affected than the other two. This greater amount of unrecovered sodium chloride at the bottom of the probe may account for the

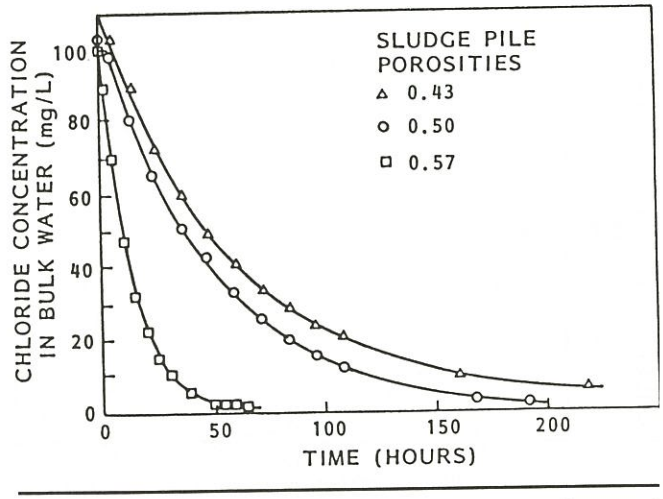

Figure 8: Bulk water chloride concentration versus time for several sludge pile porosities (pile depth: 150 mm; heat flux: 20–25 kW / m²).

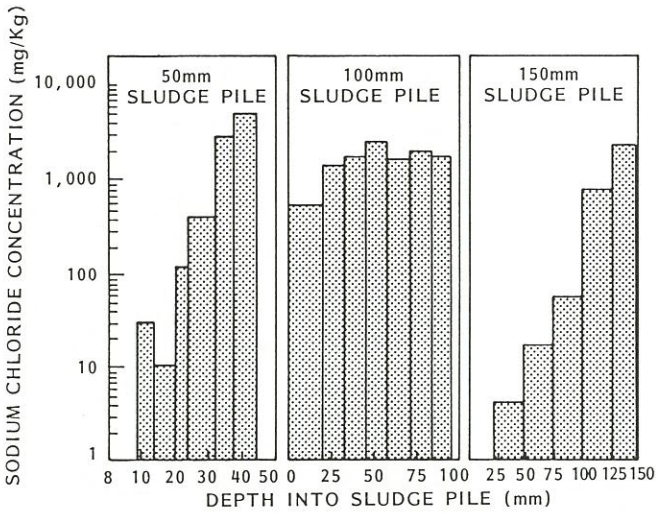

Figure 9: Sodium chloride concentration in sludge versus depth (50, 100 and 150 mm sludge piles; porosity: 0.57).

apparently flat concentration profile of the 100-mm pile in Figure 9.

Effect of Porosity

Figure 10 shows profiles of chloride concentration in the sludge versus depth for the three porosities investigated. The figure contains two sets of EDXA data. The first shows the composition of the sludge near the tube wall, and the second of the sludge approximately 3 mm away from the tube wall. Each EDX analysis corresponds to an area of approximately 2 × 1.5 mm.

The chloride concentrations near the tube wall in the 0.57-porosity pile decreased at increasing depths. The concentrations up to 20 mm in depth near the wall were quite high, approximately 20% of the detected elements. The average in the tube region along the full depth was 9%. The chloride concentration in the region 3 mm from the tube is also maximum near the top of the pile. The average concentration in this region along the total depth (2%) was significantly lower than the average near the tube wall.

In the 0.50-porosity pile, the solute concentration increased with depth. The highest concentration in the region near the tube wall was 4%, and the average along the full depth was 1%. The region away from the tube showed a similar trend, and the average concentration was 0.8%. The region that had the highest concentration was between 110 and 140 mm. The radial concentration gradient was less pronounced than in the higher-porosity pile, but showed an increasing solute concentration near the tube wall. Figure 11 shows an EDXA map of the chloride distribution in the region situated near the tube wall. The solute concentration increases near the tube wall, but it can be still significant at a certain distance. The points of maximum concentration appear to be localized in some of the 'voids' in the sludge pile, which appear as dark areas in Figure 11a.

The concentration of chloride in the 0.43-porosity pile was generally low (Figure 10), and no significant radial gradient was observed. The average of the chloride concentration along the full depth in the region near the tube wall was 0.36%, and in the region away from the tube wall 0.33%

Discussion

Dynamics of Solute Concentration Under the Sludge Pile

The dependence of the solute concentration rates on the different experimental parameters was not simple. A model was developed to explain the results of the concentration experiments.

The heat transfer through the tube was heat-flux controlled. This parameter was thus known and fixed in each test. Steam is produced at or near the tube wall and leaves the pile after travelling through the porous sludge. Water from the bulk flows through the sludge to satisfy the boiling requirements.

If the assumption is made that all the heat that is transferred under the pile is spent in steaming, the outflow of steam from the pile can be expressed by

$$\dot{m}_s = \frac{\phi A}{\Delta H} \quad (1)$$

where

ϕ: tube heat flux (kW/m^2)
A: heat transfer area under the sludge pile (m^2)
ΔH: latent heat (J/kg)
$\dot{m}_s$: rate of steam generation (kg/s)

and a mass balance for the water/steam around the sludge holder generates the expression

$$\dot{m}_i = \dot{m}_L + \dot{m}_s \quad (2)$$

where

$\dot{m}_i$: inflow of bulk water into the pile (kg/s)
$\dot{m}_L$: outflow of liquid water carried as droplets by the steam leaving the sludge pile.

At steady state, and if it is assumed that the solute partition coefficient is much smaller than one, a solute mass balance produces the equation

$$\dot{m}_i C_B^{ss} = \dot{m}_L C_H^{ss} \quad (3)$$

where

C_B^{ss}: concentration of solute in bulk water at steady state
C_H^{ss}: concentration of solute in the water leaving the sludge holder.

Experimental evidence shows that $C_H^{ss} \gg C_B^{ss}$; consequently

$$\dot{m}_i \gg \dot{m}_L \quad (4)$$

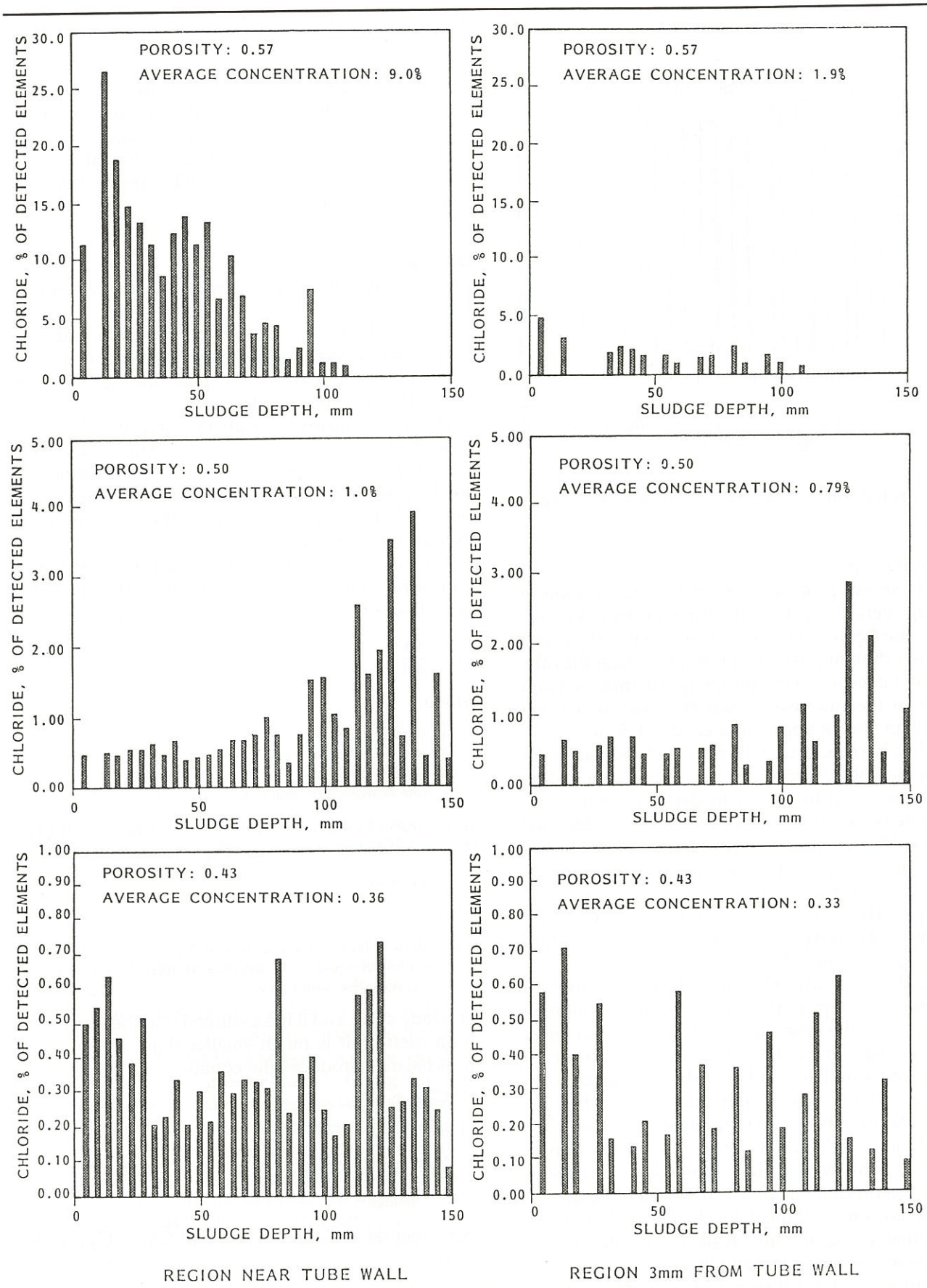

Figure 10: Chloride concentration in sludge versus depth for 150 mm sludge piles of different porosities.

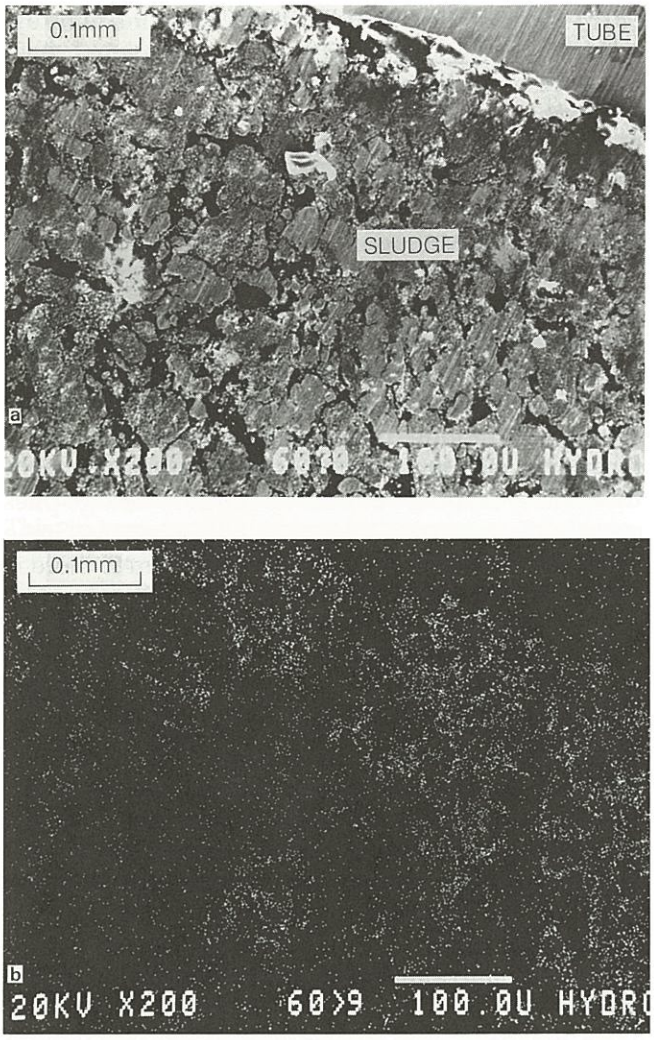

Figure 11: SEM image (a) and chloride EDXA map (b) of sludge region near the tube wall (top right corner). Point density is proportional to chloride concentration. The sludge pile had a 0.50 porosity and the region shown in the micrograph was at a depth of 125 mm.

and therefore

$$\dot{m}_i \simeq \dot{m}_s. \tag{5}$$

In other words, virtually all the bulk water that enters the sludge holder leaves it as steam.

A mass balance of the solute around the sludge holder during the concentration process can be expressed as

$$C_B \frac{\dot{m}_i}{\rho} = C_H \frac{\dot{m}_L}{\rho} + V_H \frac{dC_H}{dt} \tag{6}$$

where

C_B: concentration of solute in bulk water (mg/L)
C_H: concentration of solute in the water in the sludge holder (mg/L)
ρ: density of water (kg/L)
V_H: volume of water in sludge holder (L)
t: time (s).

The total mass of solute in the vessel can be expressed by

$$C_B V_v + C_H V_H = M \tag{7}$$

where

V_v: Volume of water in vessel (L)† (See note, p. 140)
M: Total mass of solute in vessel (mg)

Derivating expression 7 with respect to time gives

$$V_v \frac{dC_B}{dt} = -V_H \frac{dC_H}{dt} \tag{8}$$

and substituting 7 and 8 into 6, the following expression is obtained:

$$\frac{dC_B}{dt} + C_B \left(\frac{\dot{m}_i}{\rho V_v} + \frac{\dot{m}_L}{\rho V_H} \right) = \frac{M}{V_v V_H} \frac{\dot{m}_L}{\rho}. \tag{9}$$

This equation can be easily integrated with the following boundary conditions:

$$t = 0,\ C_B = C_I \quad \text{and} \quad t \to \infty,\ C_B = C_B^{ss}$$

where

C_I: initial concentration of solute in bulk water.

The resulting expression is

$$C_B = C_B^{ss} + (C_I - C_B^{ss}) \exp - \left[\frac{\dot{m}_i}{\rho V_v} + \frac{\dot{m}_L}{\rho V_H} \right] t. \tag{10}$$

C_B^{ss} should be zero if the assumption that $\dot{m}_i \gg \dot{m}_L$ is rigorously followed; however, it was left in the expression since it could easily be estimated from the experimental data by the numerical technique used in the data analysis. From equations 1 and 5, and ignoring the term containing $\dot{m}_L$ because of equation 4, expression 10 becomes:

$$C_B = C_B^{ss} + (C_I - C_B^{ss}) \exp - \left[\frac{A\phi}{\Delta H \rho V_v} \right] t \tag{11}$$

This expression implies that the concentration of solute in the bulk water decreases exponentially with time, and that the rate of concentration change is also an exponential function of the heat transfer area and of the heat flux.

A parameter K, called the 'concentration coefficient,' is required to relate expression 11 with the experimental data. K can be described as the fraction of heat transferred under the sludge pile that effectively contributes to the concentration of solute. This could be interpreted as the fraction of the heat flux that is converted to steam or, alternatively, the fraction of the heat transfer area under the pile that is actively generating steam. Revising expression 11 to include K gives

$$C_B = C_B^{ss} + (C_I - C_B^{ss}) \exp - \left[\frac{KA\phi}{\Delta H \rho V_v} \right] t. \tag{12}$$

The numerical technique used in the data reduction consisted of an unweighted, least-squares regression,

fit to a function of the type

$$Y = P(2) + (P(1) - P(2)) \exp [-P(3) X]$$

in which P(1), P(2), and P(3) are unknown parameters that are resolved by the regression analysis. These parameters corresponded to

$$C_I,\ C_B^{ss},\ \text{and}\ \frac{KA\phi}{\Delta H \rho V_v},$$

respectively, in expression 12.

The regression analysis produced a satisfactory correlation between expression 12 and the experimental data. The calculated values for C_I and C_B^{ss} were almost identical to the ones observed experimentally.

Effect of Heat Flux and Pile Depth on Concentration Coefficient

The concentration coefficient K obtained in each of the series A tests is listed in Table 1. The K values for the 50 mm pile are approximately unity, which indicates that all the heat transferred under the sludge pile is effective in concentrating the solute. This implies that the other heat losses are not significant compared to the heat transfer invested in steaming. The permeability of the pile is sufficiently high to satisfy the water-replenishment requirements for steaming at all the heat fluxes investigated. Another implication of this result is that the assumption of negligible carry-over of water droplets by the steam leaving the pile is valid.

The K coefficients for the 100-mm pile are less than unity and decrease when the heat flux increases. Since the physical characteristics of this pile are similar to those in the 50-mm pile, the increase in height seems to be the only cause for the reduced K factor. The increase in height may reduce the overall permeability of the pile, thus hindering the flow of water to the bottom. A steam-blanketed region may develop there, and the efficiency of steam generation may be diminished. A local increase in temperature can make the heat losses through conduction a significant portion of all the heat transferred. When the heat flux increases, the extent of the liquid-deficient region also increases, and the value for the K coefficient diminishes. For a given heat flux, the K coefficient will become less than unity when the pile depth is such that liquid-starved zones develop; this is not likely to occur abruptly, but rather in a progressive manner. It will affect increasing portions of the heat transfer surface with increasing depth. However, a 'critical depth' can be artificially defined below which the heat transferred is totally ineffective in the concentration of solute, and above which steam is 100% efficiently generated. This critical depth can be calculated by multiplying the K coefficient by the total depth of the sludge pile. The critical depth should depend only on the heat flux. Only one experiment at low heat flux could be carried out with the 150 mm pile, but the critical depth obtained (150 × 0.59) is similar to the critical depth of the 100-mm pile (100 × 0.80) at the same heat flux (see Table 1).

Table 1: Sludge Pile Concentration Coefficients for Several Sludge Pile Depths and Heat Fluxes (Pile Porosity 0.57)

Pile height (mm)	*Heat flux (kW / m²)*		
	40	*75*	*110*
50	1.08	1.04	1.01
100	0.80	0.70	0.51
150	0.59	–	–

The concentration profiles in Figure 9, which show the highest chloride concentrations near the bottom of the sludge pile, are not necessarily inconsistent with the finding that the bottoms of the deep sludge piles are not fully active in the concentration process. Some residual steam generation may occur even if there is steam-blanketing, since the surface is likely to be periodically rewetted. Water will flow through the sludge to replace the steam. This water will become enriched in solutes as it migrates through the regions of active boiling near the top of the pile. In the series A experiments, the bulk water became depleted of solutes; therefore, when entering the sludge pile, it did not contribute to the solute enrichment of the top of the sludge pile, but rather it leached the solute from this area and transported it to the deeper regions in the pile.

Effect of Sludge Pile Porosity on Pile Concentration Coefficient

Table 2 shows the concentration coefficient for 150 mm high sludge piles with porosities of 0.43, 0.50 and 0.57, respectively.

The 0.43-porosity pile had a coefficient of only 0.14, while that for the 0.57-porosity pile was 0.75. The low concentration coefficients for the low-porosity pile can be attributed to two factors: the lower permeability of the porous structure, which hinders the water flow, and an increased thermal conductivity of the sludge medium due to the higher degree of compaction of the porous structure. The heat that is not invested in generating steam on the tube surface is conducted out into the sludge.

The axial concentration profile for the 0.57-porosity pile (Figure 10) appears to agree with the concept of 'critical depth' proposed above. The top of the pile is more active at concentrating the solute than the bot-

Table 2: Sludge Pile Concentration Coefficients for 150 mm Sludge Piles with Varying Porosities

Pile porosity	0.43	0.50	0.57
Concentration coefficient	0.14	0.22	0.75

tom. However, the profiles for the lower-porosity piles are quite different. This discrepancy may be due to the high permeability of the 0.57-porosity pile. This permeability was enhanced by channels that were formed near the top surface of the sludge during the tests. The presence of these channels, which generally extended to the tube surface, and the higher concentrations near the tube wall, suggest that the boiling mainly took place near or at the tube surface, and that a wick boiling process was prevalent with high heat-transfer rates. These rates were lower deeper into the pile, and consequently the resulting solute concentrations were also lower. Nevertheless, the full depth of the pile was still active at concentrating the solute.

The sludge piles with lower porosities did not show the presence of channels. Their permeability, and hence the boiling heat transfer were significantly smaller. This, in turn, would produce lower average solute concentrations in the lower-porosity piles, as observed in Figure 10. Boiling in these piles likely occurred with comparable rates on the tube surface and in the sludge body. This could explain why the radial concentration gradients become flatter with decreasing porosities.

The difference in the axial concentration profiles between the piles of similar porosities (0.57) in the series A and series B experiments (Figures 9 and 10, respectively) originates in the experimental procedure used in each case. In the first series, carried out in static autoclaves, the bulk water became depleted of solute during the concentration process. As explained above, the movement of depleted water through the sludge transports the solute to the bottom of the pile. In the B series of tests, carried out in refreshed autoclaves, the bulk water always contained the same solute concentration, and the concentration profile would be expected to be fairly constant with time, with the upper areas of the sludge pile becoming progressively enriched in the solute.

Hideout Return

No attempt was made to correlate the hideout return rates to operating and physical parameters; however, it was consistently observed that the hideout return took place at a much slower rate than the concentration process. The lower rate of hideout return is not unexpected, since the solute mass transfer in this process is driven by diffusion forces. The concentration process, on the other hand, depends on boiling, which is a very efficient mass transfer phenomenon even when hindered by the presence of a porous body.

Concentration Processes in Operating Units

If concentration processes occurring in the tubesheet sludge pile of a recirculating steam generator are similar to those observed in these experiments, the rate of concentration of the bulk water solutes under the sludge should depend on the porosity / permeability properties of the sludge file. For high porosities / permeabilities, the concentration rate will be directly proportional to the local heat flux, and consequently will be maximum in the hot leg. Deep sludge piles would tend to concentrate solutes at a higher rate, since there is a larger heat transfer area active in the concentration process. High local solute concentrations may have an impact on changing the porosity of the pile if the solubility product for some species is exceeded. Low porosity / permeability piles will concentrate solutes at a reduced rate. If the bulk steam generator water has a high solute concentration for a limited period of time, the solute will concentrate under the pile at a rate determined by the heat transfer parameters of the pile. When the bulk concentration returns to a low value, the solute concentrated near the top of the pile will be slowly transferred deeper into the sludge pile. The experience accumulated in these experiments shows that the sludge pile behaves as a solute trap as long as the heat transfer process is active. The release of the concentrated solute takes place slowly, and only when the heat transfer is fully stopped.

The controlling variable in the heat-transfer process in these experiments was the heat flux. This is different from a recirculating steam generator in which the wall superheat is the controlling variable. However, the experimental procedure, which emphasized obtaining wall superheats similar to those in a CANDU system (especially in the second series of tests) should make the conclusions of this work valid for an actual operating system. Further experiments using a temperature-controlled heat transfer system more closely simulating actual steam generator heat-transfer conditions are currently underway.

Conclusions

- Bulk water solutes concentrate under sludge piles at a rate proportional to the heat flux. The rate is also affected by the physical characteristics of the sludge pile.
- For a given heat flux and sludge pile characteristics, the concentration rate is proportional to the pile depth until a critical depth is reached. Piles that exceed the critical depth showed a decreased concentration efficiency, which can be represented by a concentration coefficient.
- Concentration coefficients are a direct function of the porosity of the sludge pile.
- High-porosity piles show a concentrating behaviour that is most consistent with boiling near the tube surface and high heat-transfer rates. In less porous piles, heat may be transferred through conduction through the sludge, and boiling may take place in the sludge body. In all cases, the full depth of the pile is active to some extent at concentrating the solute, since high concentrations of solute were observed even below the critical depth.
- Sludge piles behave as solute traps as long as the heat transfer is maintained. Solutes in the pile are transferred

towards the bottom of the pile during the heat transfer process. Hideout return rates are slower than concentration rates.

Notes

† During a typical experiment, several samples were taken from the vessel and the water volume in the vessel, V_v, was not constant. This was included in calculations by computing a corrected concentration which took into account the sampled volume.

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Simulation Methodology for Pressure Tube Integrity Analysis and Comparison with Experiments

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Abstract

The computer code SMARTT (Simulation Method for Azimuthal and Radial Temperature Transients) is used in safety analysis of Ontario Hydro's CANDU reactors to predict fuel and pressure tube thermal and mechanical behaviour under asymmetric coolant conditions, such as stratified flow. This paper presents comparisons of SMARTT predictions, with preliminary results of two experiments in which large temperature non-uniformities developed on pressure tubes undergoing heatup and transverse strain at 1.0 MPa internal pressure. Temperature asymmetries developed as a result of slow boil-off of coolant in the channel. The SMARTT temperature predictions are shown to agree well with the experimental results. SMARTT accurately predicts the time at which the pressure tube balloons into contact with the calandria tube. The transient liquid level in the channel can also be accurately predicted using a simple venting / boil-off model.

Résumé

Le code de calcul SMARTT (Simulated Method for Azimuthal and Radial Temperature Transients – méthode de simulation pour les variations de température azimutale et radiale) est utilisé dans l'analyse de la sûreté des réacteurs CANDU d'Ontario Hydro pour prédire le comportement mécanique et thermique des tubes de force et des crayons de combustible soumis à des conditions assymétriques du caloporteur, comme dans le cas d'un régime stratifié. Ce document compare les résultats obtenus avec le code SMARTT avec les résultats préliminaires de deux expériences ayant enregistré des écarts importants de température sur les tubes de force causés par l'ébullition lente du caloporteur dans le canal de combustible et une tension transversale causée par une pression interne de 1 MPa. Les températures calculées par le code SMARTT sont en bon accord avec les résultats expérimentaux. Le code SMARTT calcule avec exactitude le moment où le tube de force flue et entre en contact avec le tube de calandre. Il est également possible de calculer avec précision le niveau de liquide dans le canal en utilisant un modèle simple d'éventage / ébullition.

Introduction

One of the objectives of safety analysis of CANDU reactors is to demonstrate that a postulated accident does not lead to rupture of the fuel channels. The computer code SMARTT (Simulation Method for Azimuthal and Radial Temperature Transients) [1] is one of the analytical tools used in the analysis of fuel channel integrity. SMARTT models fuel and pressure tube thermal behaviour under asymmetric fuel cooling conditions, such as stratified coolant flow. Such conditions can lead to non-uniform pressure tube heatup in the circumferential direction. If there is a highly localized hot region on the pressure tube circumference while the pressure tube is undergoing transverse strain (ballooning), the pressure tube may fail prior to contacting its calandria tube. SMARTT predicts the pressure tube circumferential temperature distribution and its associated effects on pressure tube ballooning, and whether the pressure tube will fail, or whether it will contact the calandria tube.

This paper describes comparisons between SMARTT predictions and preliminary results of two experiments in which pressure tubes were heated up to initiate ballooning under the influence of non-uniform circumferential temperature distributions. The temperature non-uniformities arose as a result of slow boil-off of coolant in the channel. The pressure tube ballooned into contact with the calandria tube, without failure, in both experiments.

The experimental apparatus and results are described first, followed by a discussion of the modification implemented in SMARTT in order to simulate the experiments. The treatment of the thermal-hydraulic boundary conditions required as input data to SMARTT is then

Keywords: nuclear safety, pressure tubes, LOCA.

described. SMARTT predictions are presented and compared to the experimental results. Areas of disagreement and agreement are discussed, along with their implications for the validity of the models used in the simulations.

Description of Experimental Apparatus and Results

A series of four pressure tube circumferential temperature distribution experiments is being performed under CANDEV at Whiteshell Nuclear Research Establishment. Two of these experiments have been completed [2]. Preliminary results from these two tests are described below. A detailed description of the experiments and analysis of the results will be available when the series of experiments is completed.

The apparatus consisted of a 2.3-metre-long segment of a CANDU-type fuel channel, as shown in Figure 1. The pressure tube was closed at one end and open to a vertical pipe (1.16 cm ID) at the other. In the channel, 36 indirect heaters were grouped into three different rings. These heaters, together with a supporting central tube, formed the CANDU-type 37-element fuel bundle configuration. The power distribution to the three rings of heaters is shown in Figure 2.

Thermocouples were placed on the outside of the pressure tube to monitor its temperature distribution during the experiments. Their locations were different in the two tests (see Figure 1). In the second test, thermocouples were also placed on the heater sheaths. The temperature of the fluid at the exit of the channel was also measured. The vertical pipe was connected to a surge tank so that the pressure in the channel could be kept relatively constant at 1 MPa during the experiment. The channel was immersed in a pool of water (23°C at 1 atmosphere) to simulate the moderator.

At the start of each test, the water in the pressure tube was heated slowly from room temperature. When the thermocouples on the top of the pressure tube registered saturation temperature (181°C), the power to the heaters was raised to a preset level, as shown in Figure 2. The experiment terminated when the heaters failed. In both tests the pressure tube was dry fully around its circumference when this happened.

Figure 3 shows an example of the circumferential temperature distribution measured around the outside of the pressure tube. Just before the power was raised, the temperature distribution varied from saturation at the top to subcooled at the bottom. The water inside the pressure tube should have had a similar distribution, as the heat transfer across the pressure tube was low. As the power was increased, water boiled off gradually. The inside of the pressure tube became

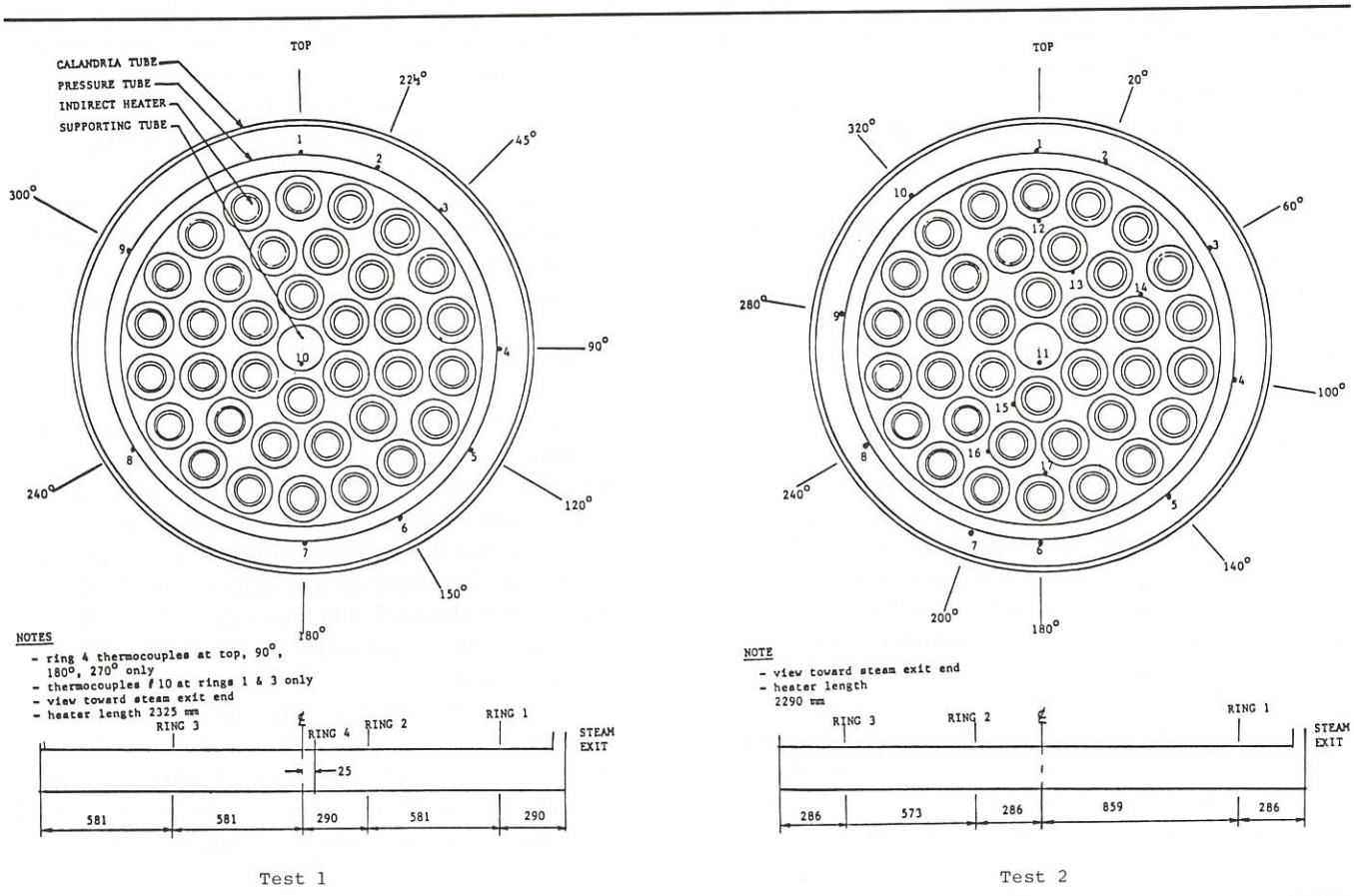

Figure 1: Schematic of the pressure tube circumferential temperature distribution experiment.

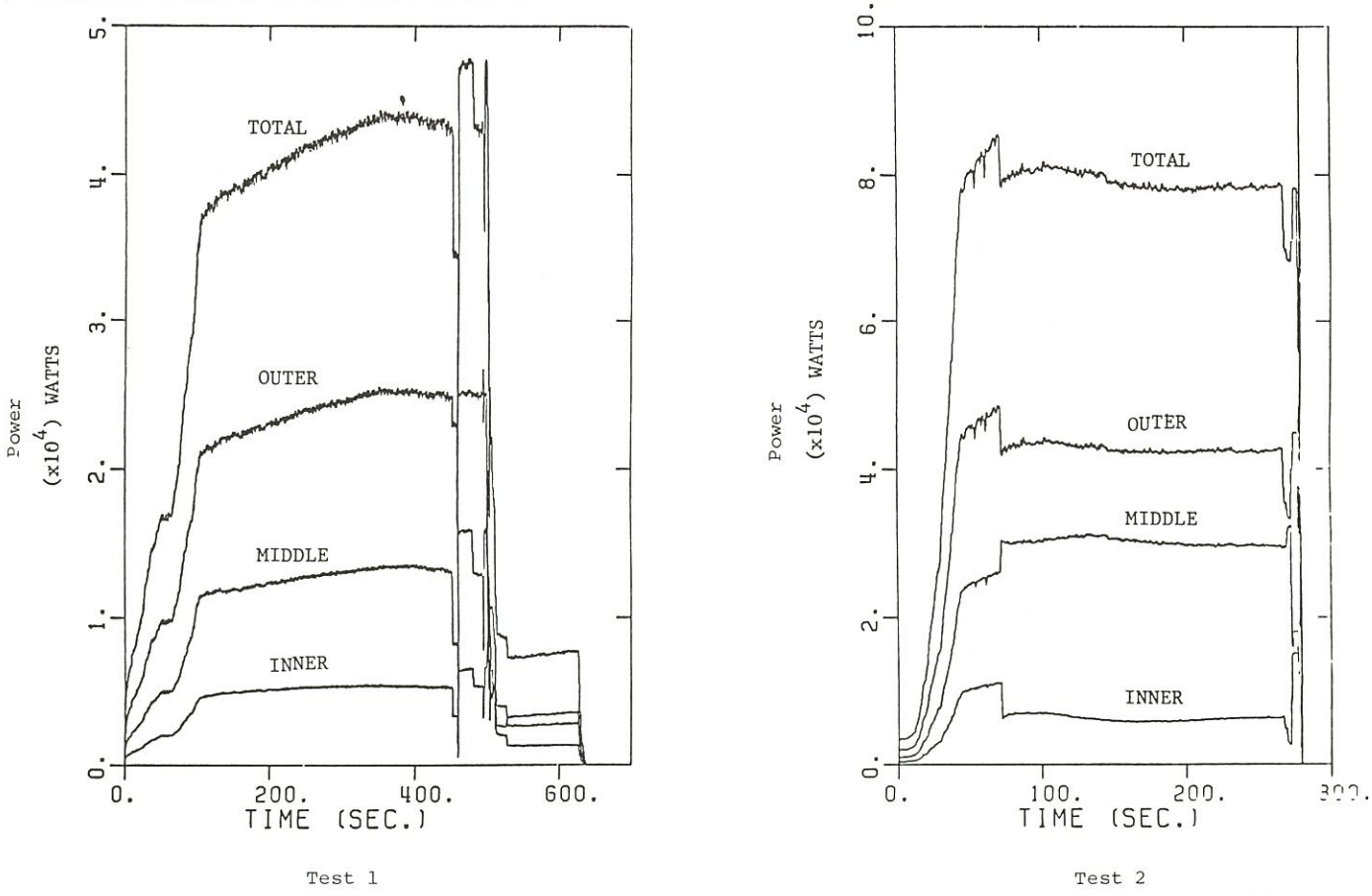

Figure 2: Power input to the indirect heaters.

exposed to steam, in succession from top to bottom. This is reflected by the sequence of sudden rises in temperature above saturation on the thermocouple readings in Figure 3. The pressure tube strained as the temperature increased. When the pressure tube came into contact with the calandria tube, heat transfer to the moderator water increased, which caused a decrease in the pressure tube temperature. The decrease in pressure tube temperature was limited by an apparently low contact conductance between the pressure tube and calandria tube. This low contact conductance was likely a result of interference by thermocouple cables located between the two tubes.

The axial temperature gradients were relatively small when compared to the circumferential temperature gradients. In general, the axial temperature distribution had a maximum in the middle of the channel.

SMARTT Model

The 37-element SMARTT model is described in detail in Reference 1. Although the geometry of the experimental rig is identical to that of a segment of a fuel channel containing a 37-element bundle, the fuel element simulator internal geometry and materials are different from those of a CANDU fuel bundle. Each simulator consists of a pressurized water reactor (PWR) fuel sheath, contained concentrically within a Bruce-size CANDU sheath (as shown in Figure 1). Helium gas fills the internal voids. Power is applied directly to the PWR sheath, which heats the CANDU sheath *via* radiation and conduction through the helium. SMARTT was modified to model this heater geometry.

Figure 4 shows the SMARTT model of a standard CANDU fuel element, and the SMARTT model of the fuel element simulators used in the experiments. The fuel element simulator is modelled exactly, with the two innermost radial nodes treated as helium. The third and fourth nodes are Zircaloy, representing the PWR sheath heater. The fifth node is helium, and the sixth node is Zircaloy, representing the CANDU fuel sheath. This was the only modification made to the models in SMARTT for the simulation of the experiments.

Boundary Conditions

The important input data required by SMARTT are the power transient, the pressure transient, the coolant temperature transient in each subchannel, and the transient convective heat transfer coefficient on each sheath and pressure tube surface node. This information was obtained either directly from experimental

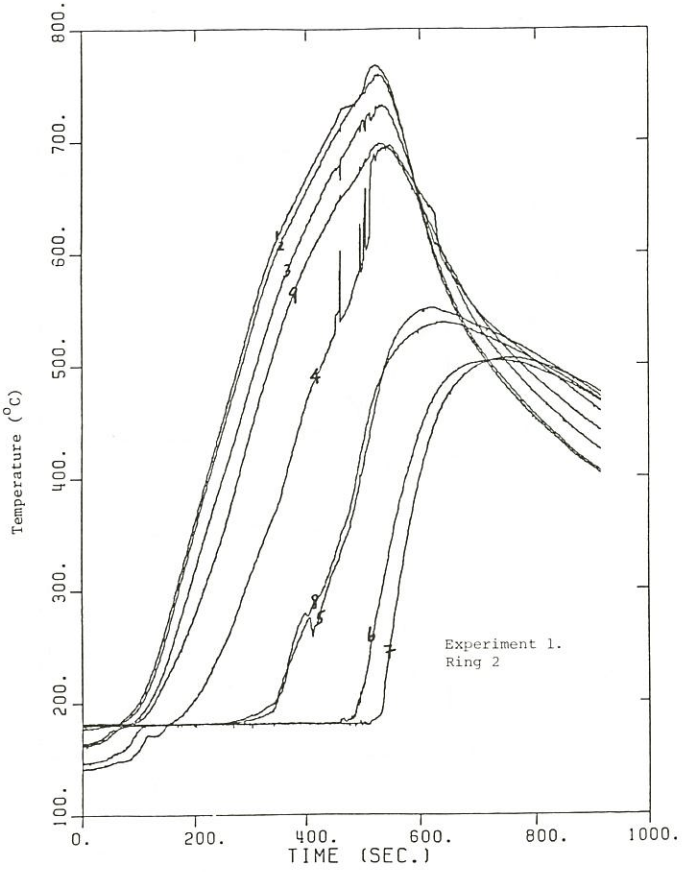

Figure 3: Measured pressure tube circumferential temperature distribution.

measurements (power and pressure), or was derived using simple models or approximations (thermal-hydraulic boundary conditions).

The power and pressure transients were obtained directly from measured values. The pressure in both tests was approximately constant at 1.0 MPa. The power in the first test was about 40 kW, which corresponds to about 1% of full power for a 7.5 MW channel. The power in the second test was 80 kW.

For simplicity in modelling, two types of thermal-hydraulic subchannels are defined. The first type is characterized by good convective heat transfer to saturated liquid. The heat transfer coefficient is assumed to be 50 kW / m^{2}K, a value which is sufficient to provide an adequate heat sink for the power generated in the fuel. This type of subchannel represents portions of the bundle covered by liquid and cooled by boiling heat transfer. The second type of subchannel represents portions of the bundle exposed to steam only, and is characterized by significantly lower convective heat transfer to steam.

The convective heat transfer coefficient for the steam filled subchannels is derived from

$$Nu = \frac{hd}{k} = 4 \qquad (1)$$

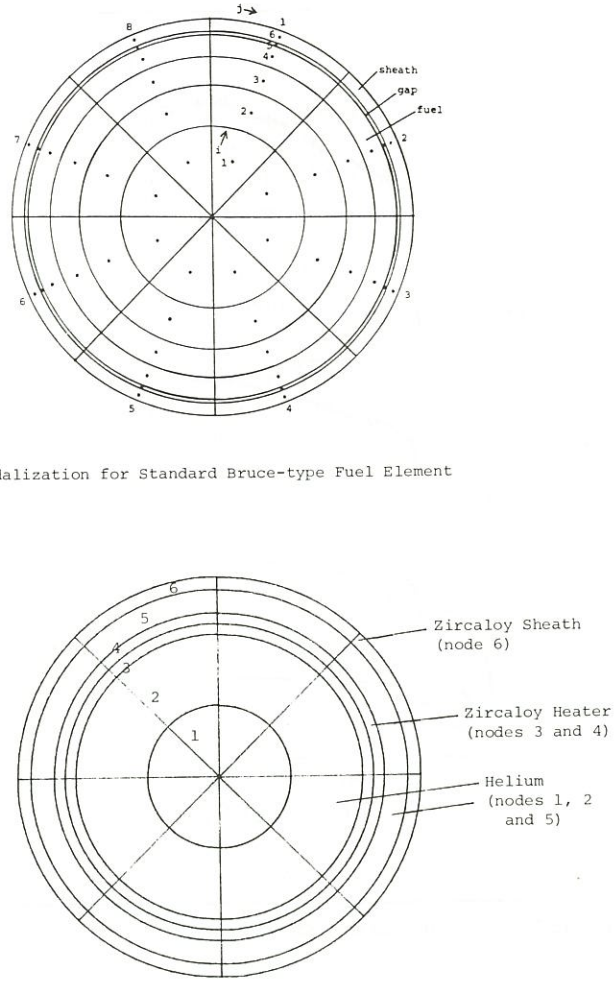

Figure 4: SMARTT fuel element nodalization.

where

Nu = Nusselt number
d = hydraulic diameter
k = thermal conductivity of steam
h = heat transfer coefficient.

A Nusselt number equal to 4 approximates laminar cooling. This was judged to be a reasonable assumption, since the mass flow rate of steam at the centre of the channel is estimated to be typically less than 10 g / s based on the observed boil-off rate. This results in Reynolds numbers within the laminar regime.

Coolant temperatures in the steam-filled subchannels are estimated by setting them equal to the average of the temperatures of the sheath surfaces encompassing each subchannel, weighted by the arc length of each surface. This approximation is also consistent with low flowrates of steam, and implies that steam cooling or heating is not a dominant factor in these experiments. This approach permits heat transfer in the vertical direction, since lower fuel elements may b

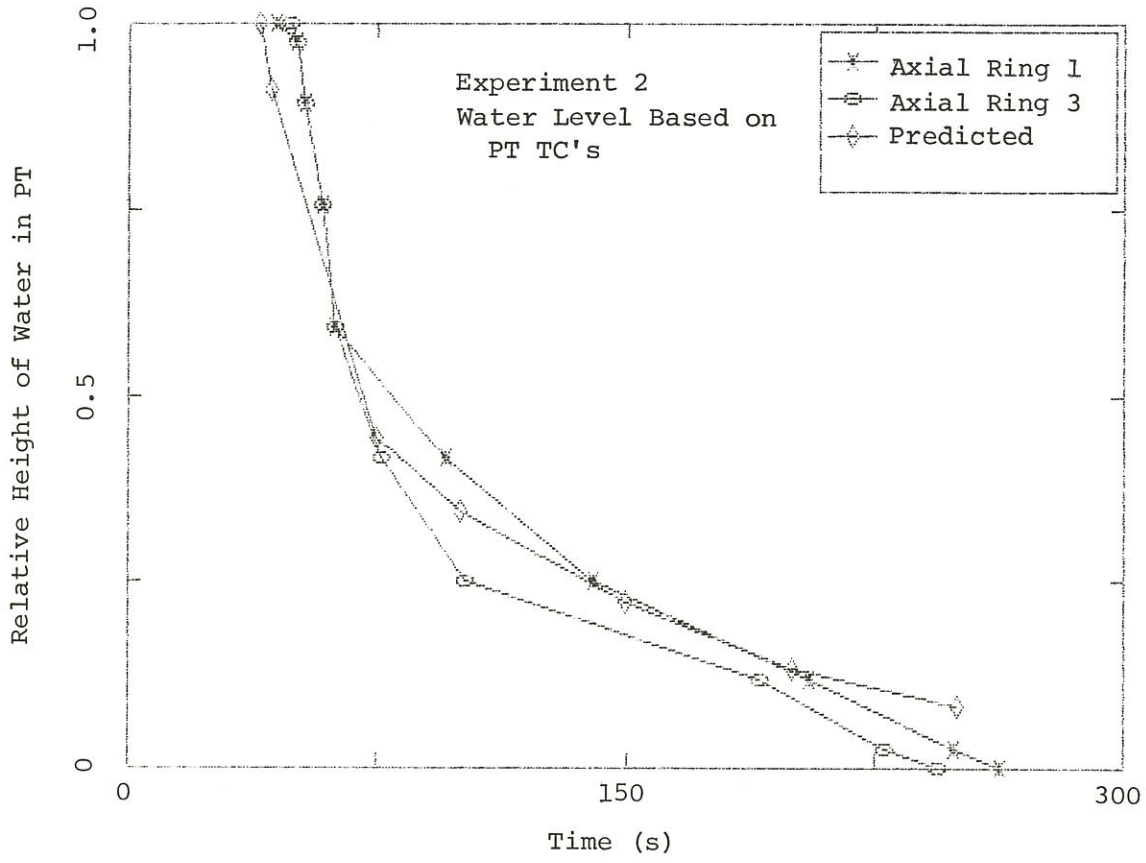

Figure 5: Predicted and inferred water level transients in test 2.

covered by liquid while upper elements are exposed to steam. Heat transfer from hot, upper elements in a subchannel to the steam in the subchannel, and then to the cooler lower elements, simulates the actual effect of the liquid heat sink at the bottom of the channel.

The timing of the transition from water-filled to steam-filled in any subchannel is derived using the venting / boil-off model described in Reference 3. This model predicts the average steam fill fraction in the channel using the measured power transient and the hydraulic resistances associated with the experimental rig. Minor modifications were made to the model to account for venting from the test section with one end closed.

In the experiments, the vertical portion of the piping between the channel exit and the condenser is initially liquid filled. Steam generated in the channel must overcome the static head of the liquid in the exit piping prior to venting from the channel. Subsequently, the rate of venting from the channel depends on the pressure difference between the channel and the condenser, the steam generation rate, and the hydraulic losses in the exit piping. When the liquid level drops to the point where the steam generation rate is not sufficient to maintain an excess pressure in the channel, the subsequent liquid level transient is governed by simple boil-off. The venting / boil-off model accounts for these processes, and predicts an average liquid level transient.

This information is used in SMARTT by defining subchannels above the calculated liquid level to be steam filled, and the subchannels below the calculated liquid level to be liquid filled. SMARTT considers seven possible liquid levels, ranging from a liquid-filled channel to a steam-filled channel. The timing of the transition from one level to the next is based on the level transient derived using the venting / boil-off model.

Comparison with Experiments

The second experiment was conducted with thermocouples monitoring fuel sheath temperatures as well as pressure tube temperatures, whereas in the first test only pressure tube thermocouples were used. Because of this, and because some heater elements burned out prior to the pressure tube reaching temperatures at which it would balloon in the first test, the emphasis of the comparison of predictions with measurements is on the second test.

Figure 5 compares the predicted average water level transient with the transient inferred from measurements of pressure tube thermocouples in Test 2. The inferred transient is derived by assuming that the tim-

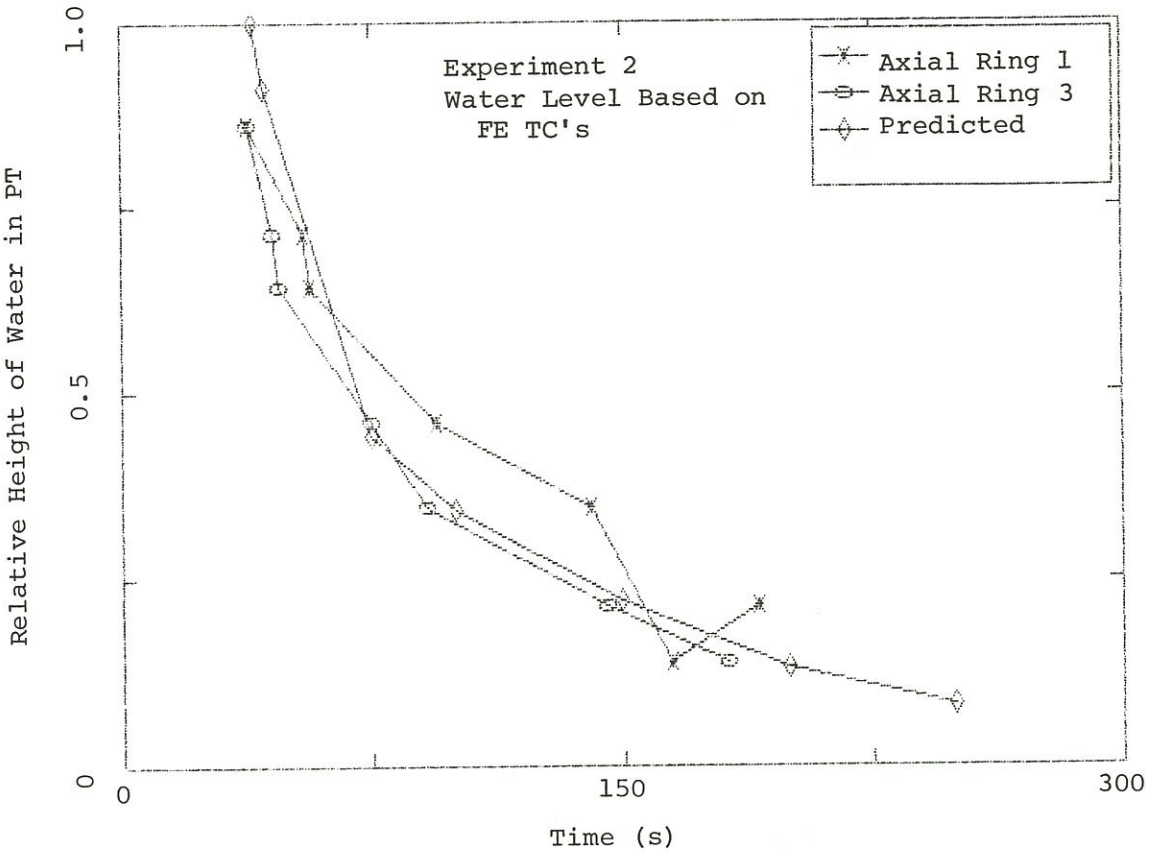

Figure 6: Predicted and inferred water level transients in test 2.

ing of the first indication of dryout on a thermocouple is the time the water level reaches the height of that thermocouple. Thus, two inferred water level transients are shown in Figure 5, one derived from the set of pressure tube thermocouples in Ring 1 in Figure 1, the other based on thermocouples in Ring 3. Similarly, Figure 6 compares the predicted water level transient with transients inferred from fuel element thermocouples in Rings 1 and 3, respectively.

Figures 5 and 6 show that the predictions of the simple venting / boil-off model are in good agreement with water level transients inferred from thermocouple measurements. In particular, the initial rapid drop in level, corresponding to venting from the channel, is very well predicted. The figures also show that, according to the thermocouple indications, the fuel elements in the upper portion of the channel experience dryout before the pressure tube, while the converse is true in the bottom portion of the channel.

Figure 7 compares SMARTT predictions with measurements of temperature at the top of the pressure tube (thermocouple 1 in Rings 1, 2, and 3 in Figure 1). SMARTT slightly underpredicts the temperatures initially, however the agreement towards the end of the transient is excellent. (The comparisons are terminated at the time of predicted pressure tube ballooning contact with the calandria tube, which, as discussed later, is in good agreement with observed time of ballooning contact.)

Figure 8 compares SMARTT predictions of temperatures at the side of the pressure tube (100 degrees from the top) with measurements from thermocouple 4 in Rings 1, 2, and 3 (see Figure 1). The SMARTT predictions fall within the range of the experimentally observed temperatures.

Figure 9 compares the pressure tube circumferential temperature profile predicted by SMARTT at 225 seconds (10 seconds prior to ballooning contact), with the experimentally observed profile at the same time at axial Rings 1, 2, and 3. Again, the SMARTT predictions are in good agreement with the experimental observations falling within the range of measured temperatures.

Figures 10 and 11 compare the SMARTT predictions of fuel sheath temperature with results obtained from thermocouples 12 and 14 in Rings 1 and 3. These thermocouples are located on the underside of the top fuel element in the outer ring of elements, and on the top of the fuel element located at 60 degrees in the intermiediate ring, respectively (see Figure 1). The thermocouple transients are terminated at the time when the readings become irrational, indicating that thermocouples have failed.

Figures 10 and 11 show a significant axial variation in the measured fuel temperature, with the fuel at the

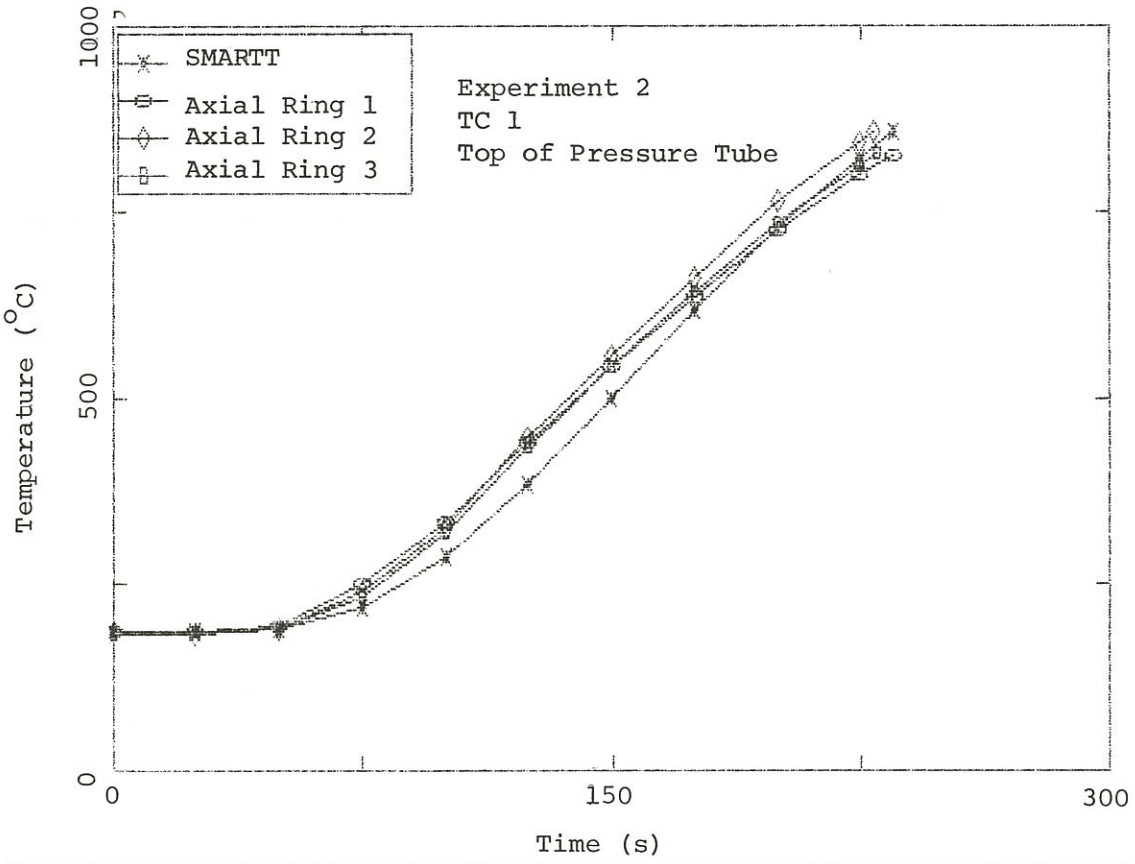

Figure 7: Comparison of measured and predicted temperatures at top of pressure tube in test 2.

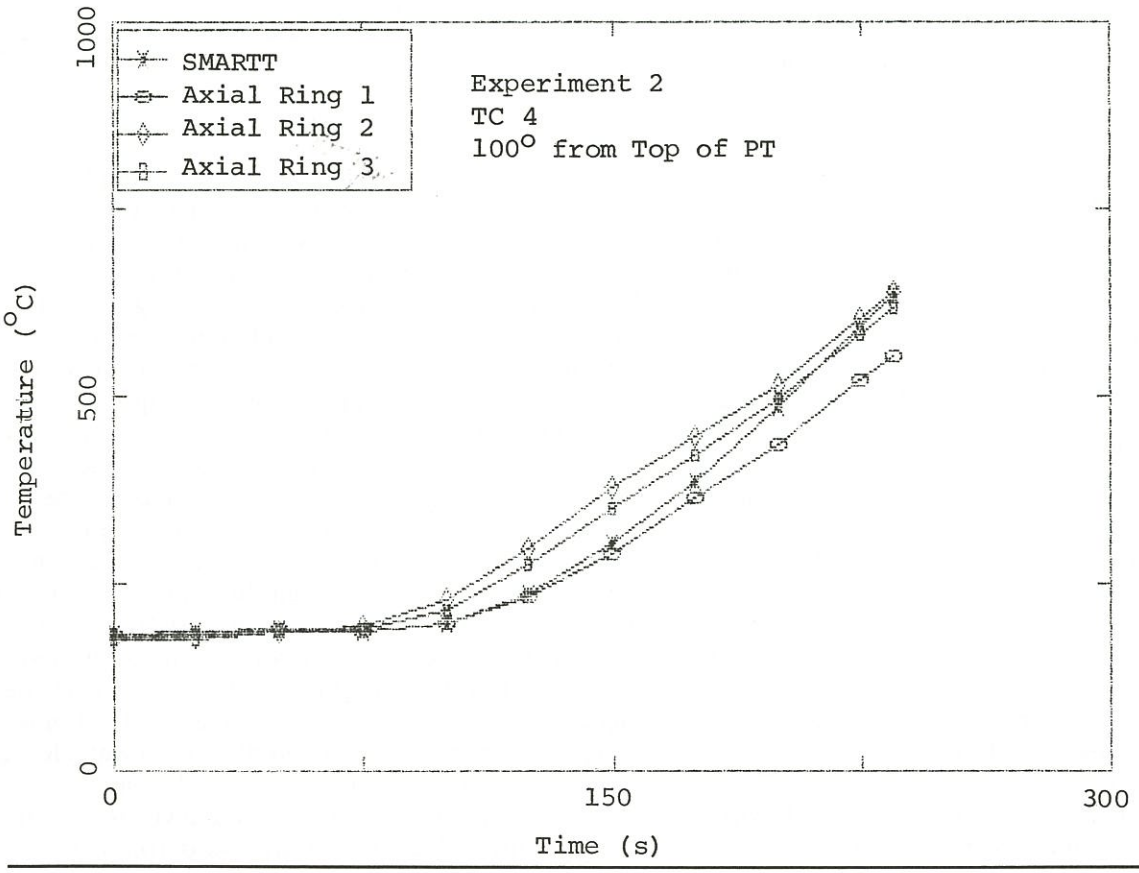

Figure 8: Comparison of measured and predicted temperatures at side of pressure tube in test 2.

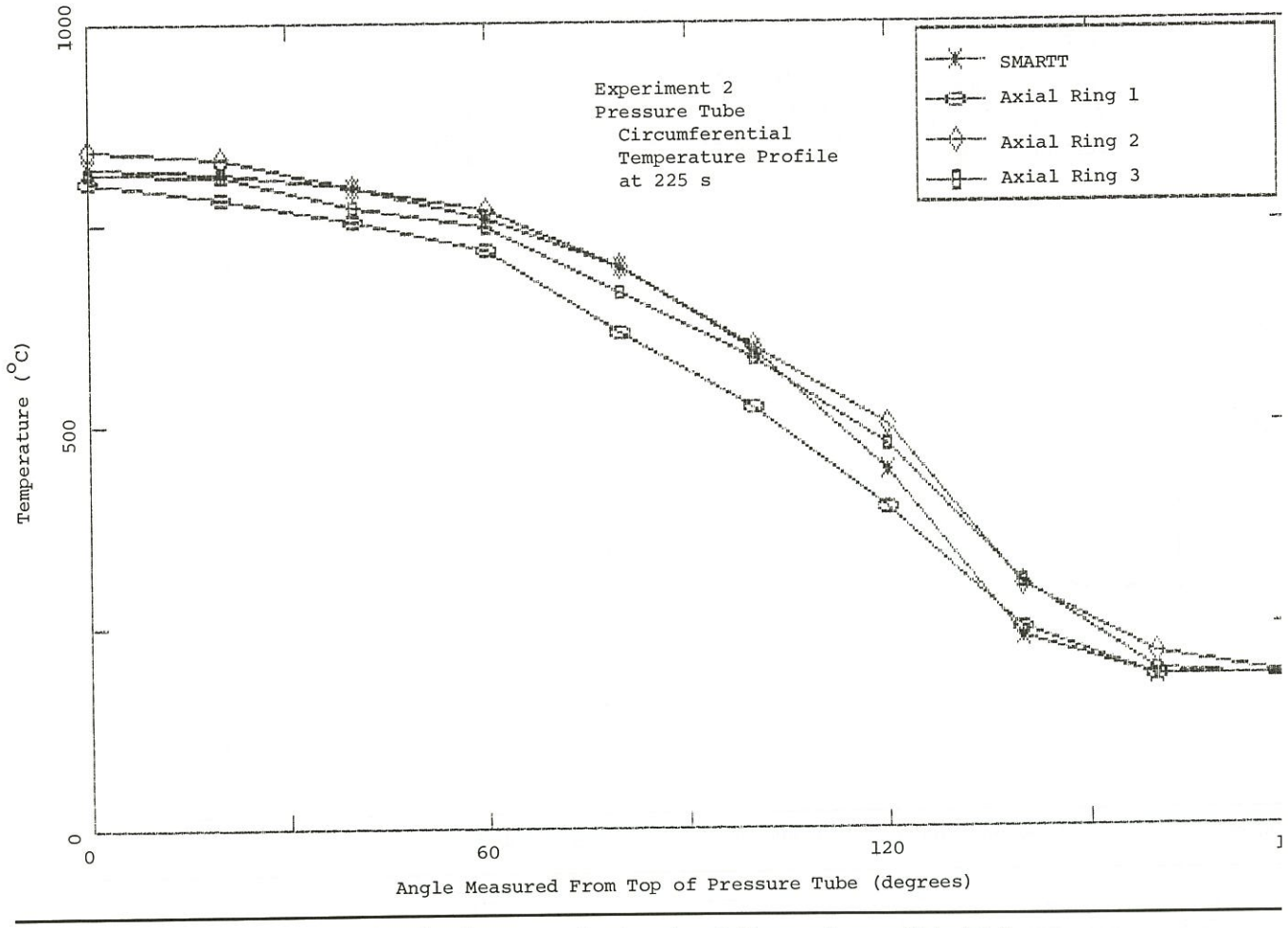

Figure 9: Comparison of measured and predicted pressure tube circumferential temperature profile in test 2.

outlet end exhibiting lower temperatures. The variation ranges from 50–100°C on the top fuel element, to 100–200°C on the intermediate ring element. This axial variation is likely due to convective cooling by steam, which would have a greater effect toward the channel outlet end, as observed. The SMARTT predictions generally fall between the two thermocouple transients on each figure, but are closer to the transients recorded at axial Ring 1.

The final comparisons presented for test 2 are of the observed and predicted times of pressure tube ballooning, and the observed and predicted local pressure tube strains following ballooning. Table 1 compares the time of pressure tube / calandria tube contact predicted by SMARTT with the contact inferred from pressure tube thermocouple measurements (indicated by a sharp decrease in temperature). SMARTT predicts a single time of contact for all points on the pressure tube circumference, because the tube is assumed to remain circular and concentric within the calandria tube. Thus, the pressure tube is assumed to touch the calandria tube at the same time at all locations on its circumference. Based on the pressure tube thermocouple mea surements, however, the pressure tube contacts th calandria tube at the top first, followed by a gradua spreading of contact in the circumferential direction This phenomenon is accentuated at Rings 1 and 3 which are near the closed and outlet ends of th channel, and may be influenced by end effects. Ba looning appears to be more uniform at Ring 2, near th centre of the channel. The SMARTT-predicted contac time is in good agreement with the contact time at th top of the pressure tube at all three axial positions.

Table 2 shows the observed and predicted pressur tube wall thickness at the end of the test. SMART overpredicts the degree of wall thinning at the top c the pressure tube. This is consistent with the apparer non-circular ballooning behaviour in the experimen which would cause the pressure tube to contact th calandria tube with less-than-predicted wall thinning

Figures 12 and 13 compare predictions of water lev and top pressure tube temperature with results of te 1. The initial rapid drop in water level is slight underpredicted, leading to a slight underprediction

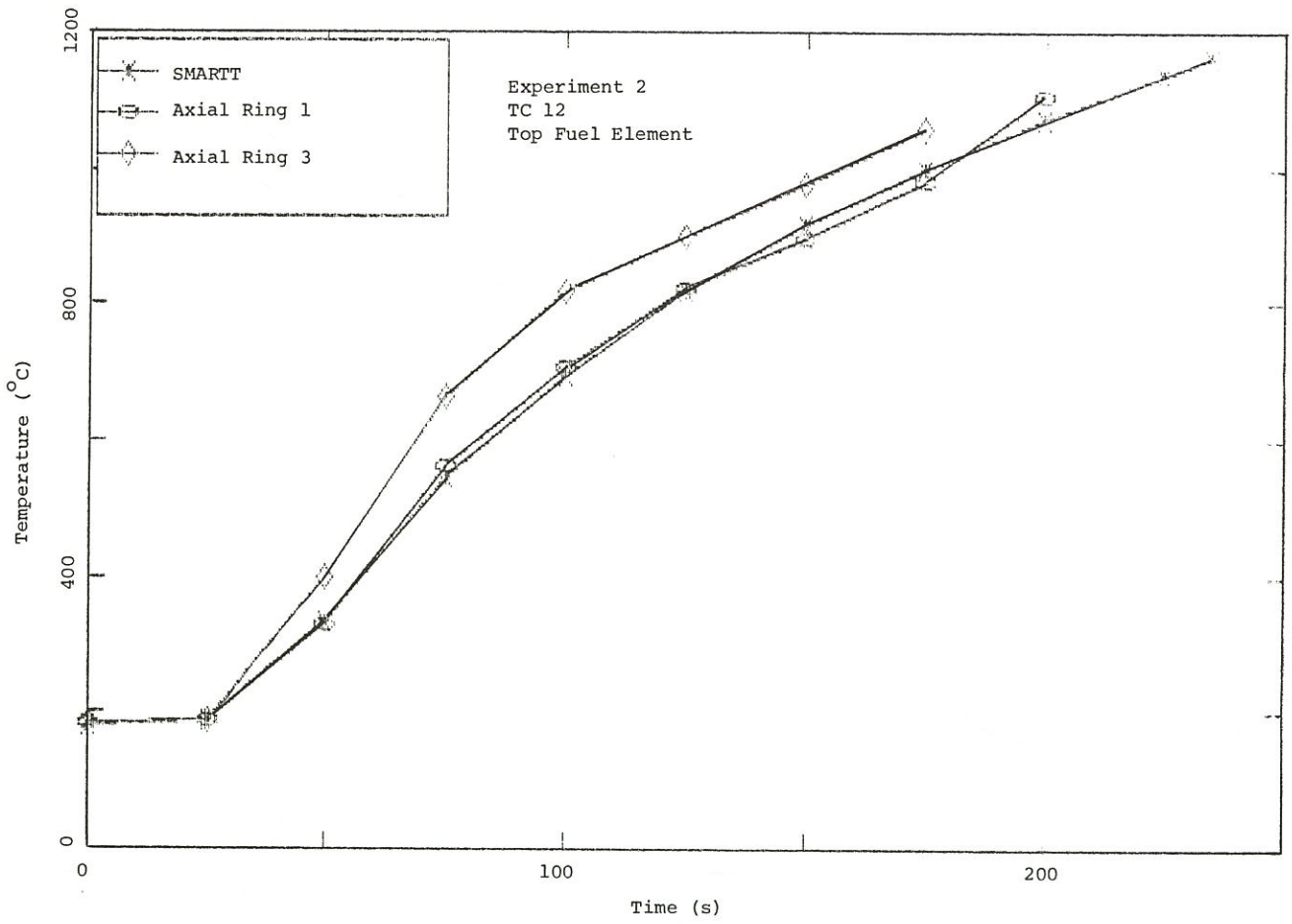

Figure 10: Comparison of measured and predicted fuel sheath temperatures in test 2.

pressure tube temperature in the early part of the transient. Toward the end of the transient, however, the pressure tube temperature is accurately predicted.

Discussion

The comparisons in Figures 5–13 show that the combination of the venting / boil-off model described in Reference 3, and SMARTT, using simple thermal-hydraulic boundary conditions, predicts the outcome of the experiments with good accuracy. Pressure tube temperatures as well as fuel temperatures are well predicted, as is the time of pressure tube / calandria tube contact.

Additional improvements in accuracy can likely be obtained by making further use of the venting / boil-off model. In addition to predicting the transient steam fill fraction used in the simulations described herein, this model also predicts steam temperatures and convective heat transfer coefficients as functions of axial position in the channel. This thermal-hydraulic information can be used in SMARTT instead of the simple assumption described earlier. The model can also be explicitly discretized in the axial direction, allowing the prediction of stem fill fraction to be dependent on axial position. With these refinements the venting / boil-off model – SMARTT combination can be used to predict fuel and pressure tube response at any axial position in the channel.

The comparisons in this paper show that the venting / boil-off model – SMARTT combination can be used with confidence in reactor safety analysis, at least when conditions are similar to those of the experiments. The combination of these two models is sufficient to carry out a complete analysis of pressure tube integrity, since the only required input conditions are the power, header pressures, and initial void and temperature distributions, and this information is typically generated by network thermal-hydraulic codes. The range of validity of the models will be expanded as more experiments in this series are completed.

Conclusions

This paper has presented comparisons of predictions with initial measurements obtained from the first two pressure tube temperature gradient tests performed under CANDEV at WNRE. The comparisons show that

1. the transient steam fill fraction in the channel is well

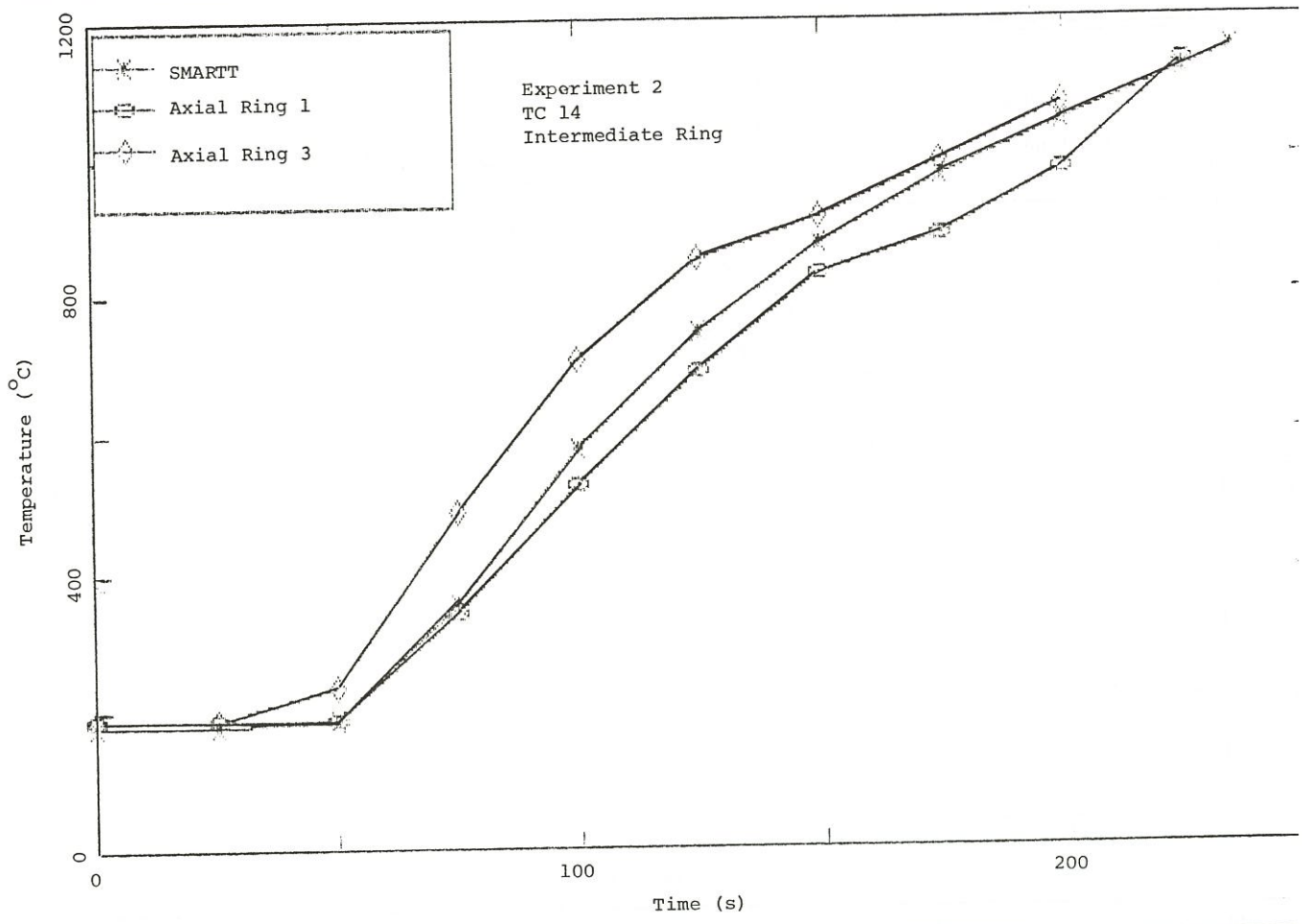

Figure 11: Comparison of measured and predicted fuel sheath temperatures in test 2.

predicted using the venting / boil-off model described in Reference 3;

2. fuel and pressure tube temperatures are accurately predicted by SMARTT when the transient steam fill fraction is obtained from the venting / boil-off model and simple thermal-hydraulic boundary conditions are used;
3. the time of pressure tube / calandria tube contact at the top of the pressure tube is accurately predicted by SMARTT; and

Table 1: Time of PT / CT Contact Test 2

SMARTT prediction *Circumferential contact position (degrees)*	*235 s*		
	Ring 1	*Ring 2*	*Ring 3*
0	241 s	229 s	230 s
20	238	232	238
40	247	235	250
60	254	238	253
80	283	247	263
100	293	254	271

Table 2: Post-Test PT Wall Thickness Test 2

Circumferential angle (degrees)	*Measured thickness (mm)*			*SMART predictic (mm)*
	Ring 1	*Ring 2*	*Ring 3*	
0	3.23	3.20	3.33	1.87
top				
20	3.26	3.12	3.18	2.25
60	3.70	3.40	3.48	3.77
100	3.76	4.04	4.01	4.15
180	4.08	4.15	4.15	4.15
bottom				

4. the venting / boil-off model – SMARTT combinati can be used for reactor analysis with confidence the accuracy of the results, when the accident co ditions are similar to those of the experiments.

Acknowledgements

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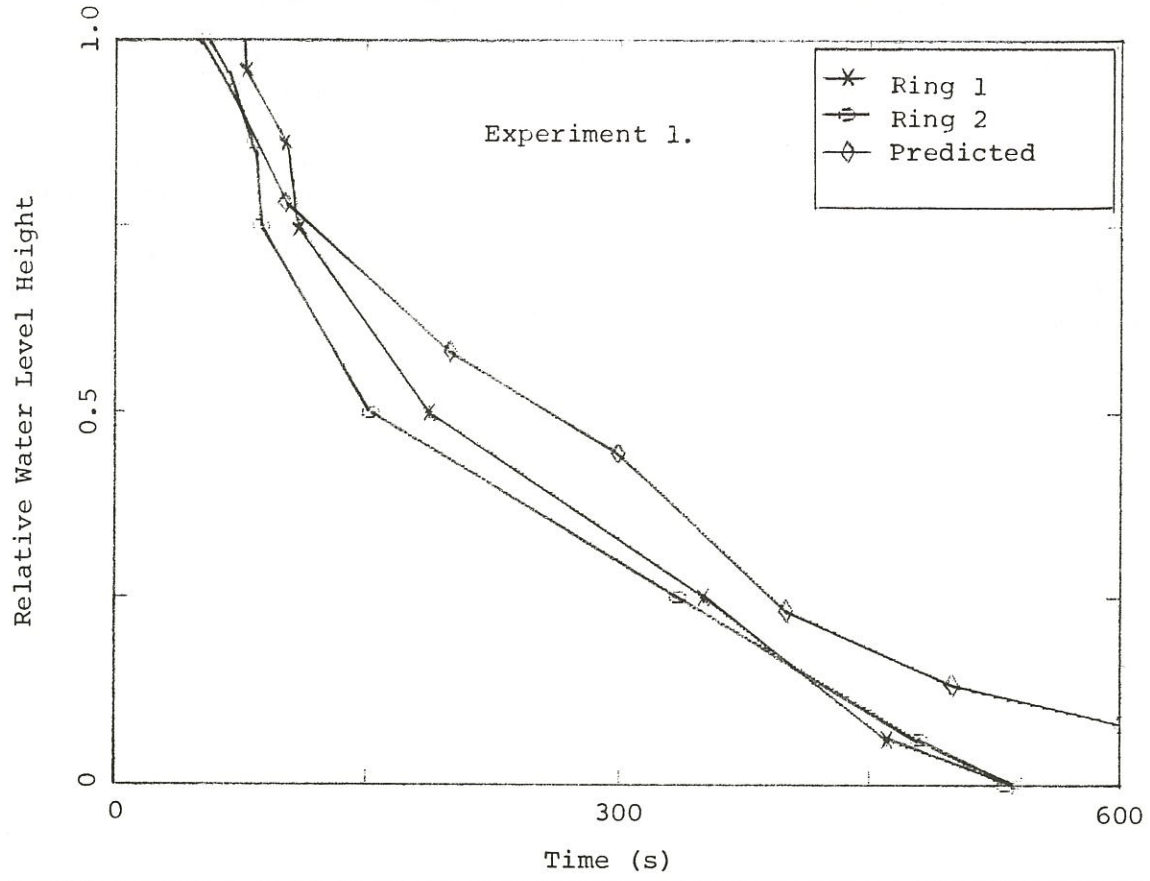

Figure 12: Comparison of predicted water level transient with transient inferred from PT thermocouples in test 1.

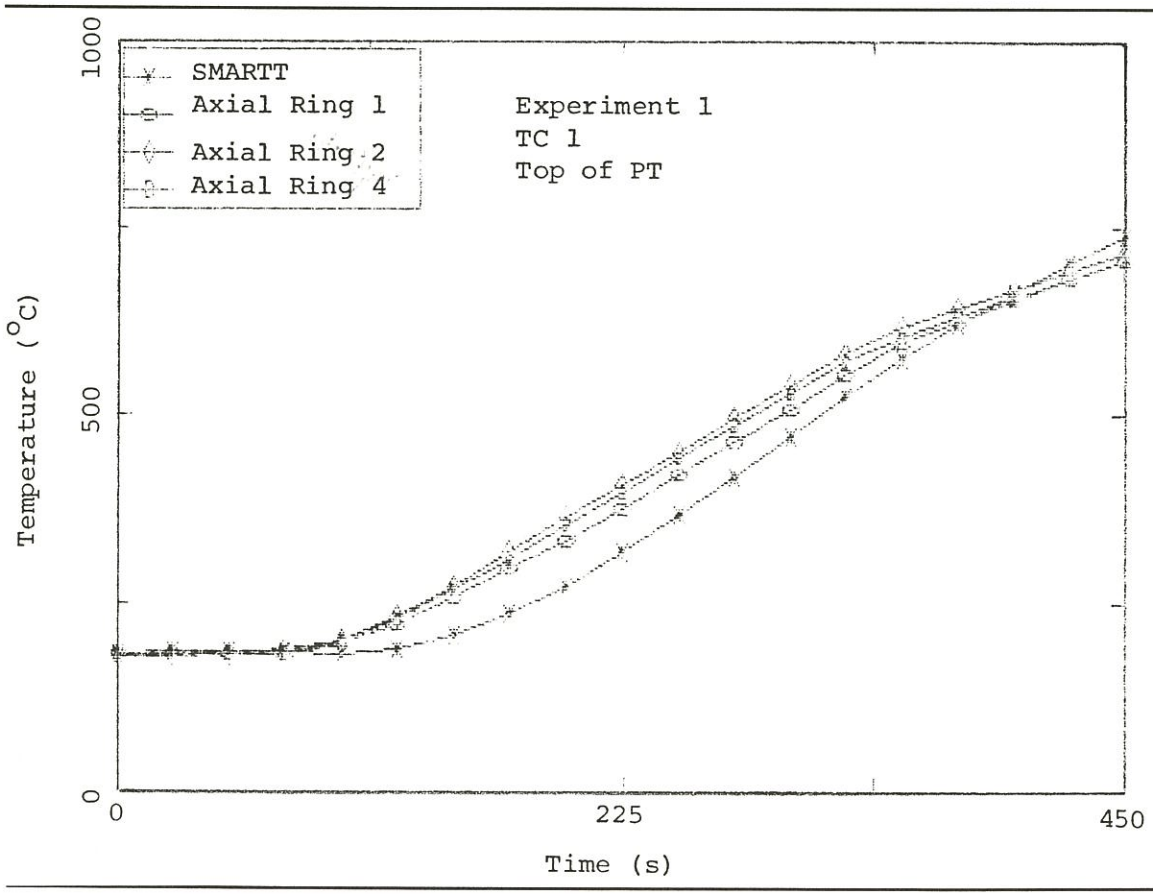

Figure 13: Comparison of measured and predicted PT temperatures in test 1.

information be released without the written consent of the COG–CANDEV Program Manager.

The experiments described in this paper were performed at Atomic Energy of Canada Limited's Whiteshell Nuclear Research Establishment. The experiments were performed by G.E. Gillespie, C.B. So, and R.G. Moyer.

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The Drift Flux Model in the ASSERT Subchannel Code

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Abstract

The ASSERT subchannel code has been developed specifically to model flow and phase distributions within CANDU fuel bundles. ASSERT uses a drift-flux model that permits the phases to have unequal velocities, and can thus model phase separation tendencies that may occur in horizontal flow. The basic principles of ASSERT are outlined, and computed results are compared against data from various experiments for validation purposes. The paper concludes with an example of the use of the code to predict critical heat flux in CANDU geometries.

Résumé

Le logiciel de sous-canaux ASSERT a été développé dans le but de modeliser l'écoulement diphasique du caloporteur dans les grappes de combustible nucléaire du type CANDU. Basé sur un modèle à écart de vitesses, ASSERT permet de prédire les vitesses inégales des phases et par conséquent de modéliser leur séparation dans un écoulement horizontal. On discute brievement le modèle physique d'ASSERT et les résultats numériques sont comparés aux résultats expérimentaux. On conclue avec un exemple d'application d'ASSERT pour la prédiction du flux de chaleur critique dans un canal de réacteur CANDU.

Introduction

As a nuclear reactor system relies entirely on fluid circuits for energy transport, mathematical modelling of thermalhydraulic phenomena plays an important role in reactor design and development, and methods of improving the accuracy and efficiency of thermalhydraulic computations are sought continually. In a CANDU reactor for example, the fluid behaviour may be adequately described by one-dimensional (cross-sectional averaged) models throughout most of the piping network. However, in the reactor fuel channel, flow must distribute itself amongst the intricate flow passages of the fuel bundle. One-dimensional analysis is adequate here to simulate overall or bulk energy transfer, but multi-dimensional analysis is necessary to model detailed local distribution of flows and temperatures inside this complex geometry. In particular, the conventional method of predicting critical heat flux (CHF) for natural-uranium-fuelled CANDU bundles is by applying a CHF correlation based on one-dimensional (cross-sectional-averaged) flow parameters. Such correlations are derived from curve fits to CHF data measured in experiments designed to simulate closely the geometry and heat flux distribution in CANDU bundles. These correlations are adequate to predict behaviour of current reactors, but would not be applicable to any future designs that have radical changes in radial flux distribution or geometry, as such changes may significantly alter the distribution of flow within the fuel bundle. These variations can be assessed only by another experiment, or by introducing a multi-dimensional flow calculation, for example in a subchannel code. In such a code, local flow and void distributions are calculated within the individual 'subchannels' or passages between fuel rods, and a local CHF correlation is then applied to each subchannel to assess when these local conditions will generate CHF.

The ASSERT subchannel code [1] has been developed to address the computation of flow and phase distribution within the subchannels of CANDU bundles, which are horizontal. Unlike conventional subchannel codes such as COBRA [2], which are designed primarily to model flow in vertical fuel bundles, and use a homogeneous mixture model of two-phase flow, ASSERT uses a drift-flux model that permits the phases to have unequal velocities, and includes gravity terms that make it possible to analyze separation tendencies that may occur in horizontal flow.

The development of ASSERT has included validation by comparison of computed results to data from a number of experiments involving two-phase flow in horizontal channels [3] and vertical bundles [4, 5]. Recently, ASSERT predictions of CHF have also been compared to the U1 horizontal 37-rod bundle experiments.

Keywords: subchannel, drift-flux, two-fluid, two-phase, critical heat flux, void fraction, separation.

Thermalhydraulic Model

The thermalhydraulic model equations used in ASSERT-4 (Version 1) are derived from the two-fluid formulation. The two-fluid equations are combined to obtain the ASSERT model equations. The transportive form is obtained from the conservative form merely by subtracting the identity expressed by the mass equations. ASSERT has options to solve either of the two-fluid equations, using either the drift-flux or the homogeneous mixture model.

Conservation Equations

Mixture mass (conservative form) is

$$\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho V) = 0 \tag{1}$$

where

$$\rho = (\alpha\rho)_g + (\alpha\rho)_f = \alpha_g^+ \rho_g + \alpha_f^+ \rho_f$$
$$(\rho V) = (\alpha\rho V)_g + (\alpha\rho V)_f = \rho V.$$

Mixture momentum (conservative form) is

$$\frac{\partial(\rho V)}{\partial t} + \nabla \cdot \left(\rho VV + \frac{(\alpha\rho)_g (\alpha\rho)_f}{\rho} V_r V_r^\dagger \right) + \nabla P = -F_w^\dagger + \rho g. \tag{2}$$

Mixture energy (transportive form) is

$$\rho \frac{\partial h}{\partial t} + \rho V \cdot \nabla h + \nabla \cdot \left(\frac{(\alpha\rho)_g (\alpha\rho)_f}{\rho} (h_g - h_f) V_r^\dagger \right) =$$
$$q_w'''^\dagger - \nabla \cdot ((\alpha q'')_g + (\alpha q'')_f)^\dagger. \tag{3}$$

Phasic energy (transportive form) liquid is

$$(\alpha\rho)_f \frac{\partial h_f}{\partial t} + (\alpha\rho V)_f \cdot \nabla h_f = q_{wf}'''^\dagger - \nabla \cdot (\alpha q'')_f^\dagger + q_{if}'''^\dagger. \tag{4}$$

Phasic energy (transportative form) vapour is

$$(\alpha\rho)_g \frac{\partial h_g}{\partial t} + (\alpha\rho V)_g \cdot \nabla h_g = q_{wg}'''^\dagger - \nabla \cdot (\alpha q'')_g^\dagger + q_{ig}'''^\dagger \tag{5}$$

where

$$\alpha_g + \alpha_f = 1.$$

Henceforth, for simplicity we will use α to denote α_g.

$^+$ – denotes variables that must be defined by state relationships; and

† – denotes variables that must be defined by constitutive relationships.

Subchannel Equations

ASSERT uses the subchannel approach used in the development of the COBRA-IV computer code [2]. Subchannels are defined as the flow areas between rods, bounded by the rods themselves and imaginary lines linking adjacent rod centres. Subchannels are divided axially into a number of control volumes that communicate axially with neighbours in the same subchannel and laterally across fictitious boundaries (gaps) with control volumes in neighbouring subchannels. The relationship between the reactor core, a fuel channel,

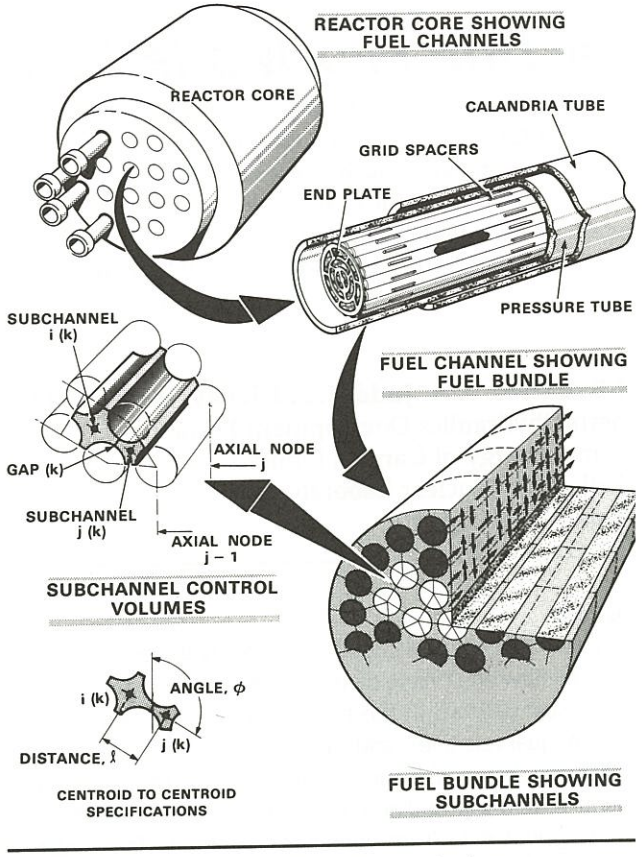


Figure 1: Formulation of subchannel control volume.

and the definition of particular subchannels is given i Figure 1.

The development of finite difference analogs to e press the equations (1–5) with respect to subchann control volumes follows the approach used in COBR but unlike COBRA, the transverse gravity terms a retained, making it possible to use ASSERT-4 to mod the effect of gravity on horizontal two-phase flo even if the homogeneous option is used. Spatial-di ferenced versions of the model equations are derive by applying the conservation equations to a repres tative control volume taken from subchannel i(k) whi shares gap k with an adjacent control volume in su channel j(k) between axial nodes j-1 and j. Details a given in the user's manual [1].

Closure Relationships

The required closure relationships, as indicated abov are the equations of state and constitutive relationshi relating relative velocity, fluid friction, wall heat tran fer, and thermal mixing to primary variables, pha flow velocities, densities, enthalpies, and pressure.

The relative velocity is the heart of the success application of the ASSERT model to horizontal bundl and channels. It comprises several effects, includin

1. relative velocity due to cross-section averaging;
2. local relative velocity due to gravity separation;

3. turbulent diffusion of void, both between neighbouring channels and towards a preferred phase distribution pattern.

The vapour phase velocity V_g is therefore considered to depart from the mixture volumetric flux, j, due to these effects, which are incorporated using, respectively, the Zuber-Findlay distribution parameter C_0, and the weighted mean velocity V_{gj} [6] and the void diffusion term [7].

$$V_g = C_0 j + V_{gj} - \frac{\varepsilon}{l\bar{\alpha}} \Delta_{ij}(\bar{\alpha} - \alpha_0) \quad (6)$$

Noting that

$$j = \bar{\alpha} V_g + (1 - \bar{\alpha}) V_f \quad (7)$$

and

$$V_r = V_g - V_f$$

yields the expression for relative velocity required in equation 2.

$$V_r = \left[(C_0 - 1)j + V_{gj} - \frac{\varepsilon}{\bar{\alpha} l} \Delta_{ij}(\bar{\alpha} - \alpha_0) \right] / (1 - \bar{\alpha}) \quad (8)$$

In the axial momentum equation only the first two terms in (8) are considered important. However, in the lateral momentum equation, according to the usual practice in subchannel analysis, void diffusion is considered dominant, and only the last two terms are used, as C_0 is also taken as unity.

$$U_r = \{(C_0 - 1)j + u_{gj}\} / (1 - \bar{\alpha}) \text{ (axial relative velocity)} \quad (9)$$

$$V_r = \{(v_{gj} - (\varepsilon / \bar{\alpha} l)\Delta_{ij}(\bar{\alpha} - \alpha_0)\} / (1 - \bar{\alpha}) \text{ (lateral relative velocity)} \quad (10)$$

The drift velocity of the bubbles, u_{gj}, is expressed in terms of terminal bubble rise velocity, V_∞, as follows:

$$V_{gj} = (1 - \bar{\alpha})^n V_\infty \cos \phi, \quad (11)$$

where n may vary from zero to 3. V_∞ is the terminal bubble rise velocity, which can be expressed as

$$V_\infty = k_1 \left[\frac{(\rho_f - \rho_g)}{\rho_f^2} \sigma g \right]^{k_2} \quad (12)$$

where ϕ is the direction angle of the connection between subchannels. This term models the gravity separation of the phases for horizontal flow.

The void diffusion term includes two effects: the turbulent diffusion of void fraction and the diffusion toward a nonuniform void. The parameter α_0 is an equilibrium void fraction included to account for the experimentally observed tendency of void to migrate toward larger subchannels, and is defined following Lahey [7]. The diffusion equation is expressed, as in the case of single-phase, in terms of the Peclet number ε_α / VD, from the work reported by Rudzinski [8] for void ranging between 0.30 and 0.6.

$$\varepsilon_\alpha / VD = C_\alpha (\alpha / 0.6)^6, \; C_\alpha = 0.075 \quad (13)$$

The equations 9–12 have a number of free parameters, for which constant values were used in earlier work with ASSERT [3]. Recently, however, relationships proposed by Ohkawa and Lahey [9] have been incorporated. These give compatible definitions of u_{gj} and C_0 in terms of characteristic properties of two-phase flow. The parameters n, k_1, k_2, and C_0 are thus varied, appropriately, with void fraction and the ratio of phase densities, thus extending the applicable range of the drift-flux model. These relationships produced satisfactory results over a range of conditions except at high void fractions. It is clear from equation 8 that the relative velocity will become infinite as α approaches unity, and in fact when the Ohkawa-Lahey relationships are used to define C_0 and u_{gj}, equation 8 became indeterminate (0 / 0). A smoothing factor compatible with the Ohkawa-Lahey equations was introduced to ensure correct asymptotic behaviour. The resulting equations are summarized in Reference 1.

Solution Procedure

The numerical solution over the bundle cross-section at each axial position is split into two parts. The first part solves the energy and state equations, using a block iterative method to calculate the mixture and phasic enthalpies for all subchannels, where current flow estimates are used as parameters. Once the energy equation solution inner iteration converges, the second part calculates the flows and pressure gradients at that axial position. This is done by the direct matrix solution of the crossflow equations, from which it is possible to calculate axial flows and pressure gradients. Both parts are repeated once to ensure a higher level of convergence of both energy and flow solutions prior to moving to the next axial position. The channel is successively swept from the inlet to the exit. This outer iteration continues until convergence is achieved, or until an iteration limit is reached. Successful completion would yield a steady-state solution, or one time-step of a transient solution.

The ASSERT code can be run with either flow, or header-to-header pressure drop specified, and has been written in a plane-by-plane solution mode that eliminates any restriction on the number of axial nodes used.

CHF Methodology in ASSERT

The probability of making a successful prediction of local CHF obviously is a direct function of the success of predicting local flow and phase distribution. In ASSERT, the CHF prediction is performed subsequent to calculation of flow distribution.

Whalley *et al.* [10] developed a film boiling model to calculate CHF for vertical annular upflow in round tubes. The method is based on the assumption that the flow regime is annular. The essential features of annular flow are that the gas travels in the centre of the channel, a liquid film travels on the channel walls, and liquid drops are carried along with the gas flow. The

continuity equation for the film flow in the liquid annulus is written in terms of evaporation, entrainment, and deposition, and solved for the dryout point, at which film thickness is effectively zero. In ASSERT the equation is solved for each subchannel to predict the rod, subchannel, and axial location where dry-out occurs.

Validation Studies

The early part of ASSERT development concentrated on the development of a suitable thermalhydraulic model [3] and then on validation, which, of course, involves continuous development.

Comparison with Air / Water Twin Channel Experiments

The work of Tapucu on exchange of air-water mixtures flowing in two parallel square communicating channels was used for initial testing of the ASSERT code.

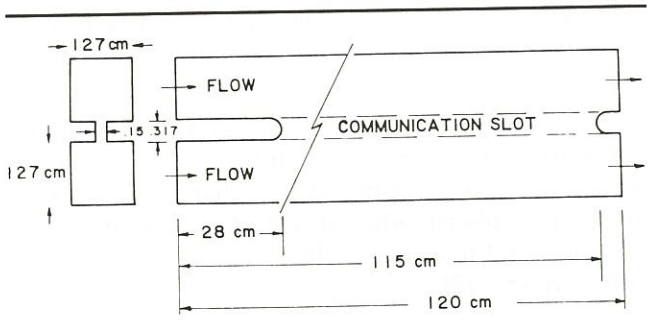


Figure 2: Geometry of the Tapucu experiment square subchannels.

Details of these experiments and the experimental technique are reported in reference [11]. Channel dimensions are given in Figure 2. The experiments were run at the same initial nominal mass flux of 3,060 kg m^{-2} s^{-1} in both subchannels, but for different initial voids and different orientations. The key parameters – pressure, void fractions, and liquid and gas flow rates in both

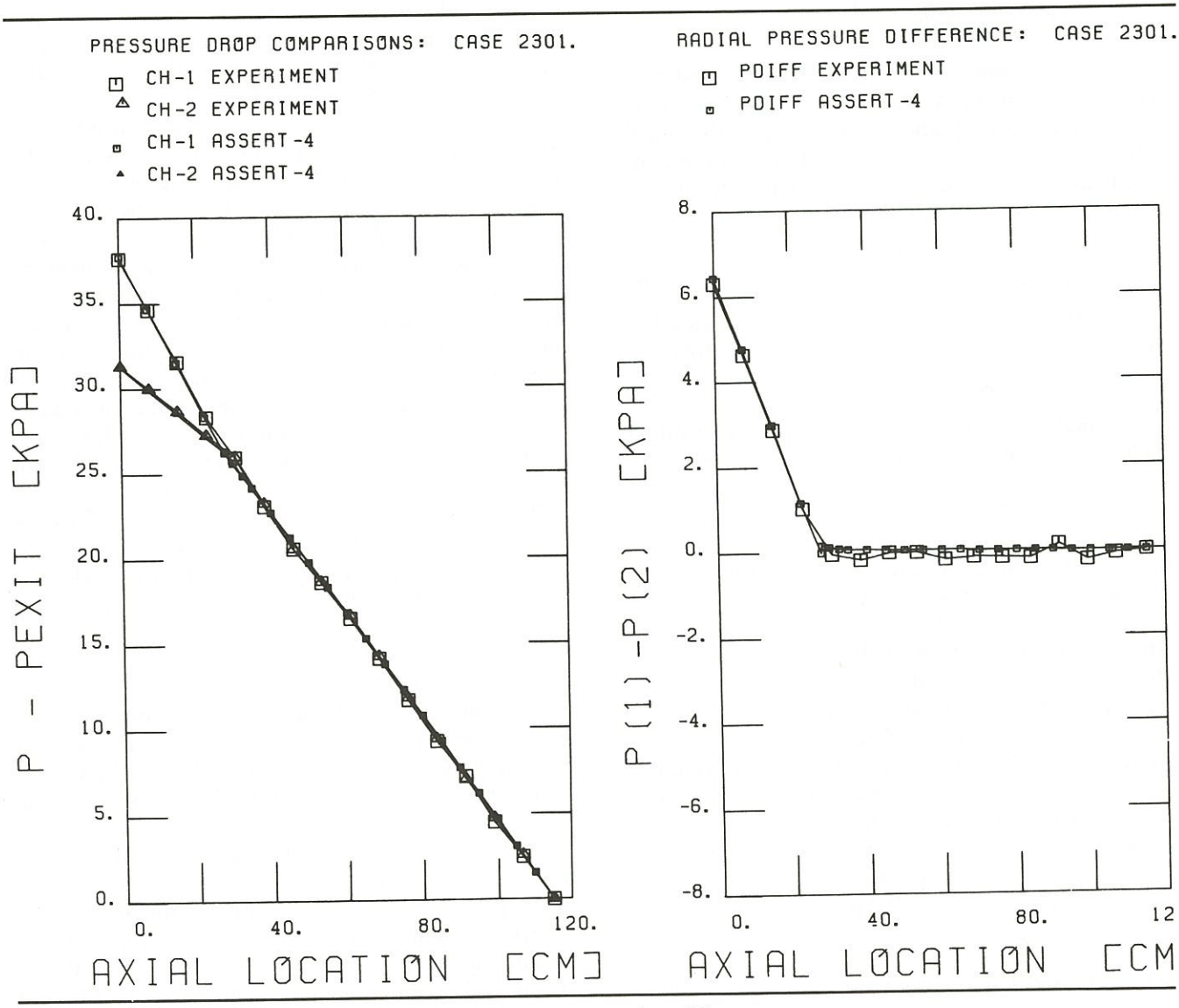


Figure 3: Typical experimental and computed pressure profile for vertical case V1.

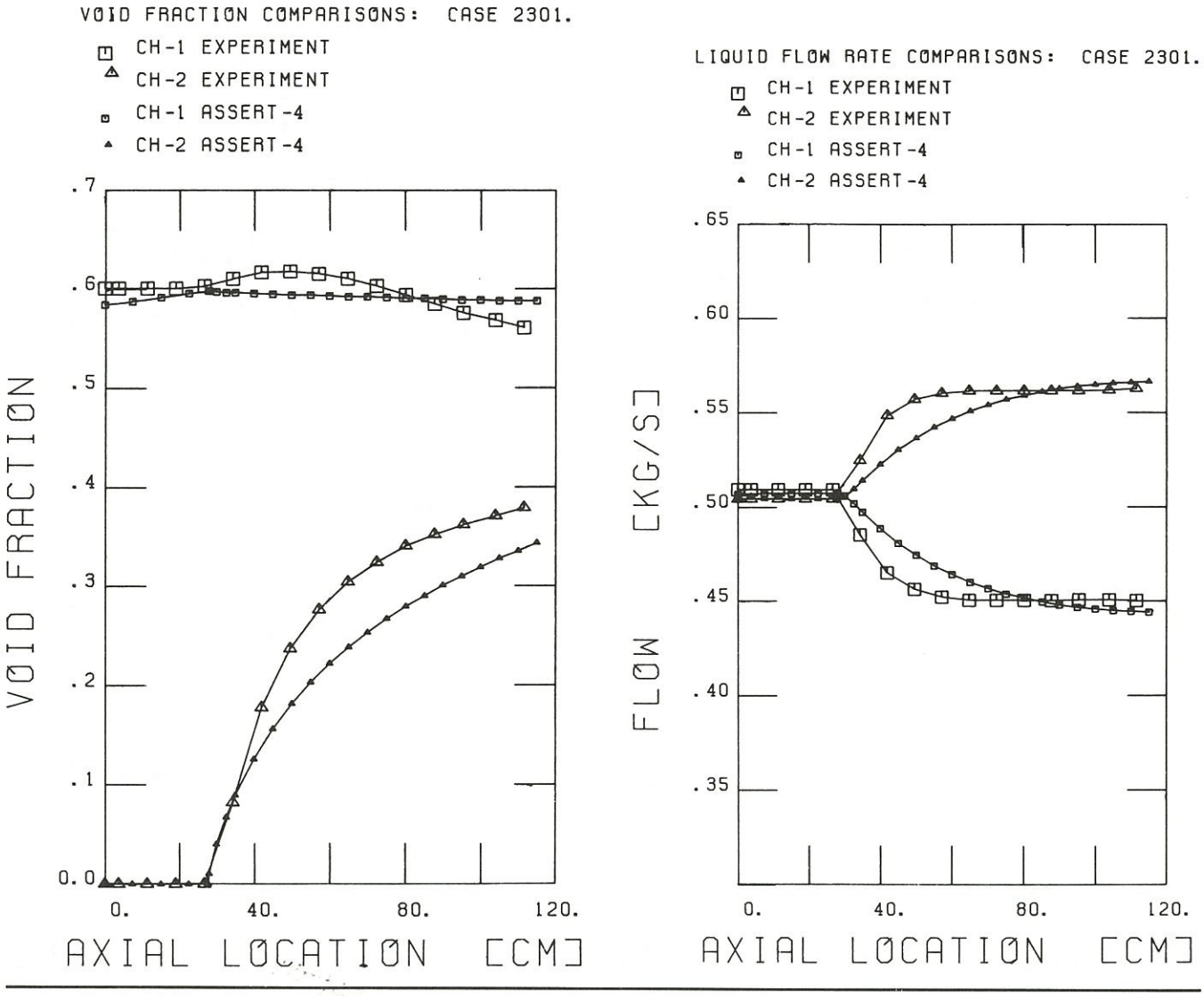


Figure 4: Computed and measured profiles of void fraction and mass flux for vertical case V1 (square).

channels along the interconnection – were measured at several axial locations.

Until recently, code comparisons with these experiments had concentrated on modelling void fraction in particular cases [3], and on exhibiting correct qualitative trends. A sufficiently general model, however, should be able to reproduce trends quantitatively throughout the entire spectrum of experimental conditions. This is not an unrealistic demand in this case, as the data base and range of parameters are quite limited. The current research has modelled both void fraction and mass flux for the entire range of 17 experiments.

In the vertical orientation, only two mechanisms are active, the diversion cross-flow and the turbulent exchange. The experiments are well documented and good agreement with measured pressure drops was first obtained by a single-phase friction factor and Armand two-phase multiplier. Together with a good estimate of form loss, these closely determine the pressure-driven diversion cross-flow. Pressure-driven cross-flow induces a co-current flow of air and water to flow from the higher pressure channel. In the experiments, pressure is quickly equalized in the slot, but the high void channel requires a higher initial pressure to overcome the two-phase pressure drop. A typical pressure profile is shown in Figure 3, along with computed pressures. The initial tendency in the vertical experiments is therefore for the recipient or low void channel to gain air and water from the donor. However, when the void fraction in the donor is high, the tendency towards turbulent exchange increases. This results in some counter-current flow, in which some liquid returns from the recipient to the donor and is replaced by air. This tendency is readily simulated by the diffusion model. Increased diffusion augments the tendency towards counter-current exchange.

Typical results showing void fraction and mass flux profiles are given in Figure 4. Note that in the void

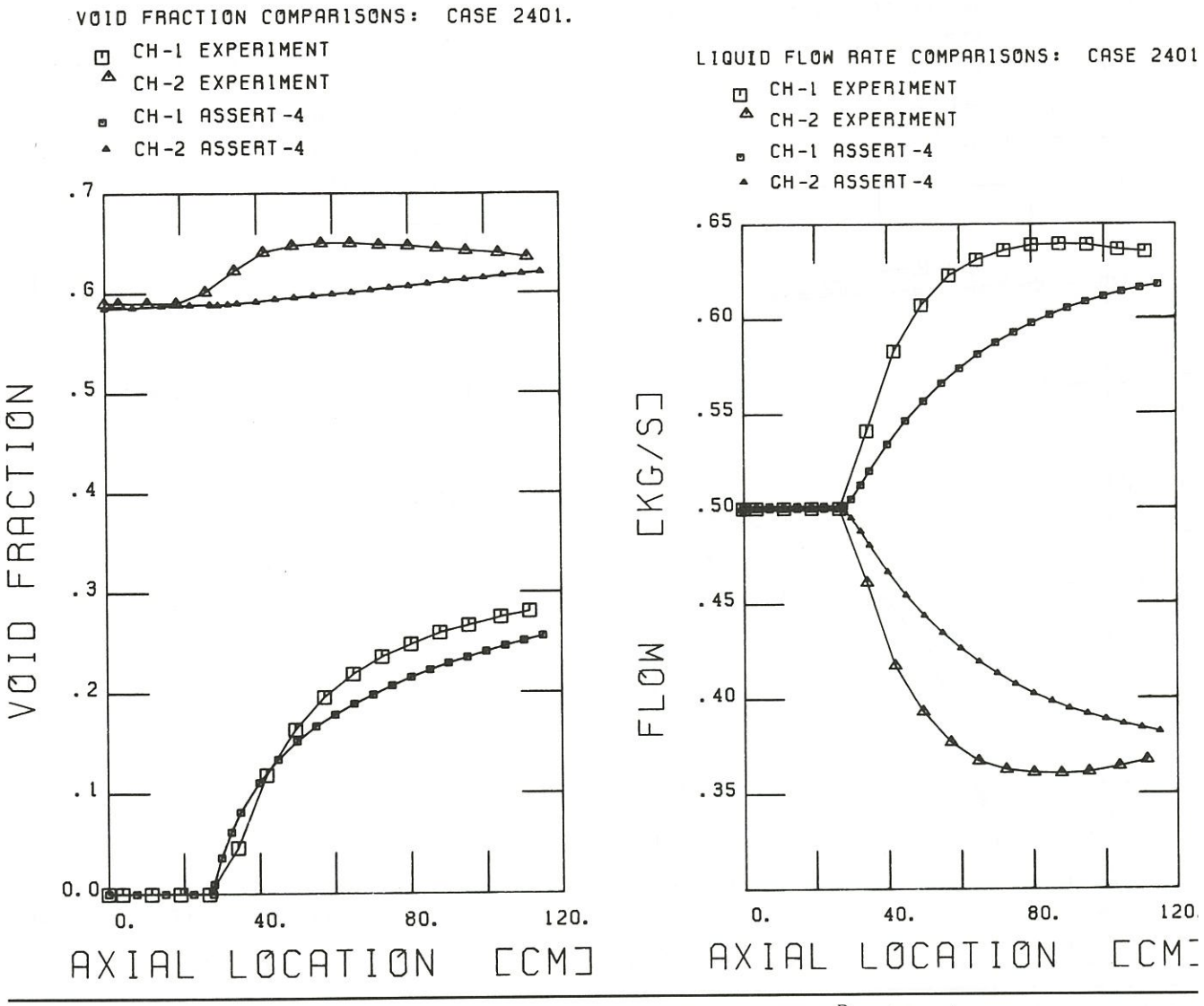


Figure 5: Computed and measured profiles of void fraction and mass flux for horizontal case $H_R^D - 1$ (square).

fraction profiles, donor void fraction initially increases somewhat, although air transfer is taking place to the receiver. This is typical of the experiments and is due to the fact that at the low experimental pressures (nominally 0.15 MN / m^2) the air expands significantly as it descends the pressure gradient.

In the horizontal orientation, with one channel above the other, cross-flow is now driven by gravity as well as pressure and diffusion. In simulating the experiments, it was postulated that although the internal distributions would be different from the vertical case, the turbulent exchange would be of the same magnitude. Attention was therefore turned to the formulation of the relative velocity due to gravity.

Two orientations were examined in the experiments, with the donor or high void fraction channel above and below the recipient, denoted by H_R^D and H_D^R, respectively. In the former case, gravity drift does not cause exchange, and diversion cross-flow and turbulent exchange dominate. In the latter, gravity drift is significant, but the initial interchange is a pressure driven diversion cross-flow during which both air and water flow from the donor into the recipient above. Eventually gravity forces tend to become dominant and counter-current lateral flow is set up.

Figures 5 and 6 show typical comparisons for the horizontal case $H_R^D - 1$ and its inverse $H_D^R - 1$. The particular pair of experiments was chosen because of the interesting behaviour of mass flux. This clearly illustrates that the initial tendency towards pressure driven co-current exchange is eventually overcome by gravity-driven counter-current exchange. Further details of the comparisons are given in References 3 and 12.

Similar experiments have recently been completed by Tapucu [12] in which the channels were fabricated to a form that simulates the shape of neighbouring subchannels in a rod bundle, as shown in Figure 7.

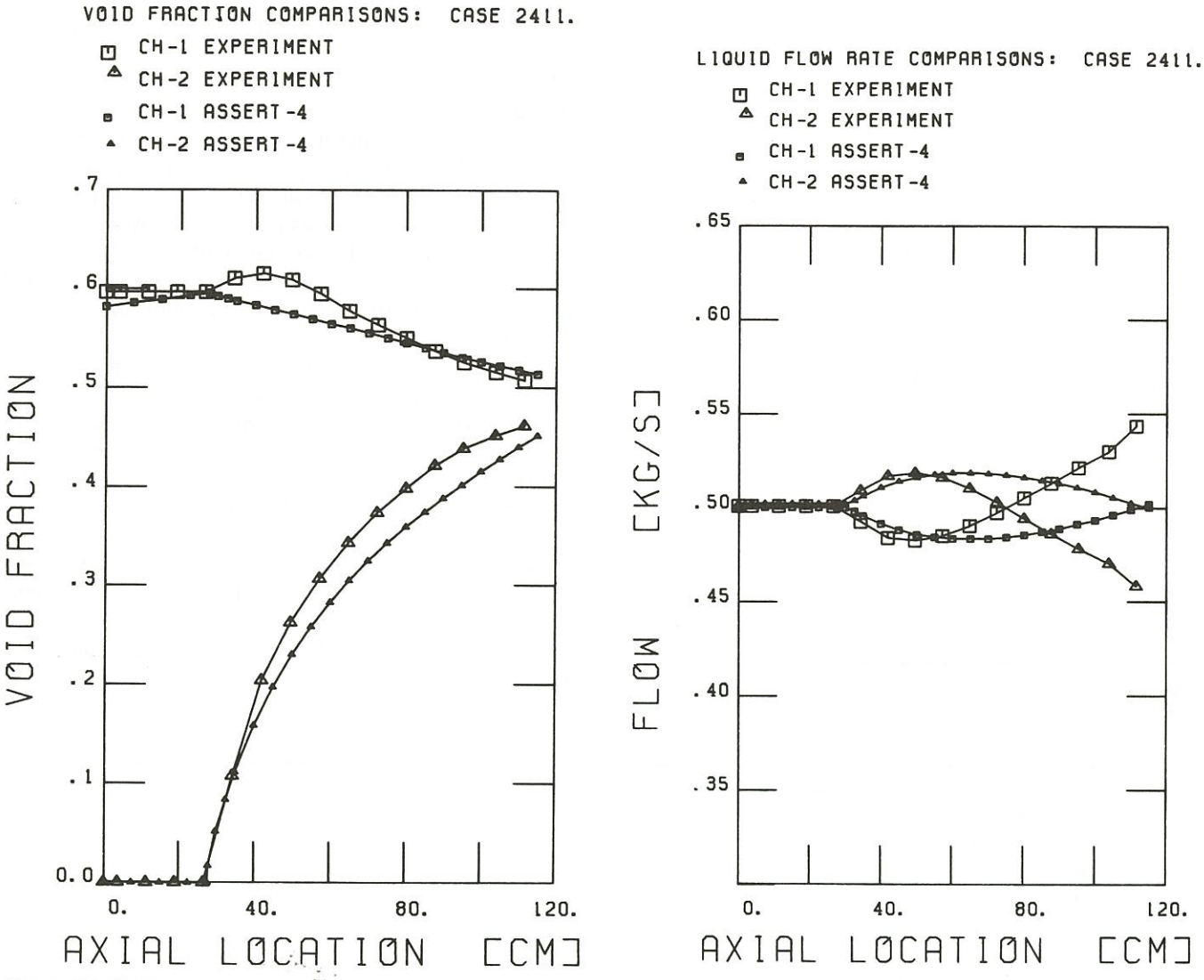


Figure 6: Computed and measured profiles of void fraction and mass flux for horizontal case $H_D^R - 1$ (square).

this case, the channels were run in five orientations, vertical and both horizontal in positions mentioned above, plus horizontal equal elevation and inclined orientations. All 24 of these experiments were simulated using ASSERT, but only representative cases are shown here.

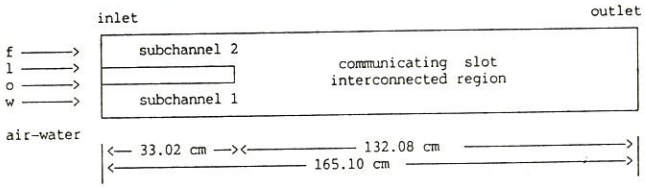
Computed and experimental results are shown for two reciprocal cases in Figures 8 and 9. Again, the cross-over tendency is apparent in the mass flux profile and is simulated quite well by the program. Full details of the comparisons are given in Reference 12.

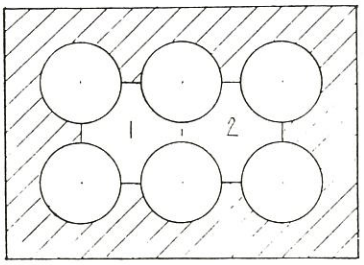
Comparison with Experiments in Vertical Bundles

The first bundle experiments used for validation were conducted by Bosio and Imset, using a vertical 7-rod bundle [14]. The bundle consisted of one electrically heated 3.6-m-long centre-rod, with a uniform axial power and six unheated peripheral rods (Figure 10). Simultaneous measurements of subchannel void fractions were performed in the marked zones of Figure 10 by means of an impedance probe at three axial levels. The average void in the subchannels was calculated by integrating the local values. Measurements were obtained at the various mass fluxes around 1,500 kg m^{-2} s^{-2}, 16 and 30 bars pressure, and about 5°C inlet subcooling.

Both ASSERT and COBRA-IV were used to simulate the complete repertoire of six experiments, without tuning to fit any particular experiment. ASSERT successfully simulates the experiments, and the difference between the ASSERT predictions and the experimental results averages less than 10% as typically shown in Figure 10. COBRA does not predict sufficient void migration into the unheated channel, as the COBRA mixing model is inadequate for two-phase flow. Further comparisons are discussed in Reference 4.

The experiments used in a second validation exercise were conducted by Nylund *et al.* Two bundles





- two rod-bundle subchannels on a square pitch
- adiabatic, air-water
- 7 vertical flow cases
- 17 horizontal flow cases

Experiment Geometry :		Subchannel Data : (per subchannel)	
channel length	= 165.10 cm	flow area	= 1.1663 cm^2
slot length	= 132.08 cm	hydraulic diameter	= 0.7620 cm
gap clearance	= 0.1660 cm	wetted perimeter	= 6.1223 cm
rod diameter	= 1.7580 cm	centroid distance	= 1.8660 cm

Figure 7: Geometry of the Tapucu experiment bundle-type subchannels.

were tested, one with 6 rods and one with 36 + 1 rods [15]. Both were uniformly heated with rods of 13.8 mm outer diameter and 4.4 m heated length. In the 36 + 1-bundle, an unheated centre rod of 20 mm diameter was used. The void was measured with a multi-beam gamma ray densitometer. By manipulating the beam in the radial direction it was possible to measure the void in different zones of the bundle. The 36 + 1 bundle was divided in three zones (rings) as shown in Figure 11. The measurements were taken at several axial locations in the bundles. Mass flux was again about 1,000 $kg\,m^{-2}\,s^{-1}$, pressure 50 bars; subcooling varied from 1° to 20°C.

For the 36 + 1-rod bundle, the scatter in the measurements of zones 1 and 2 was too large for meaningful comparison; the measurement in the remaining zones are quite consistent, as shown for zone 4 in Figure 11a. Comparisons were made only for bundle average, and zones 3 and 4 because of the lower scatter in the experimental results.

Typical ASSERT and COBRA-IV predictions for zone 4 are shown in Figure 11b, and are in good agreement with experimental results. In all of the cases, the COBRA or ASSERT predictions showed reasonable agreement. Comparisons for all the experiments are given in Reference 5.

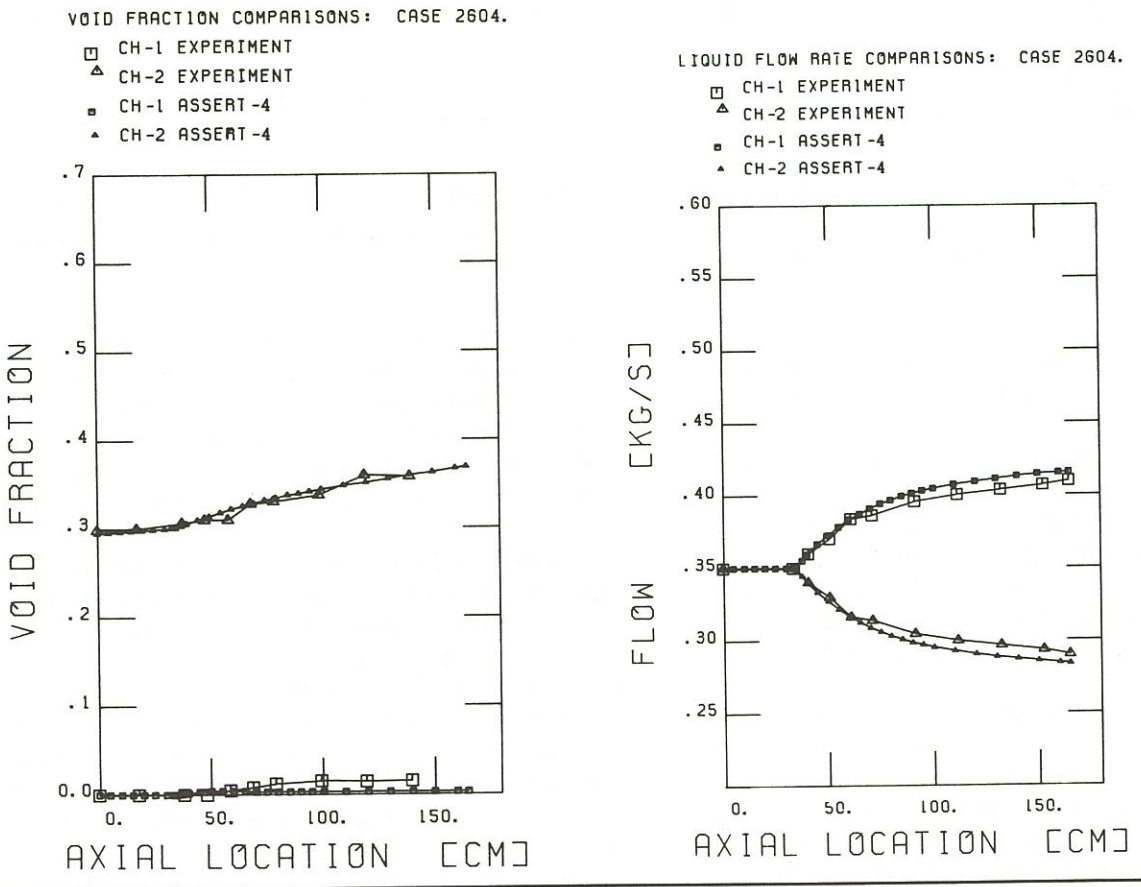


Figure 8: Computed and measured profiles of void fraction and mass flux for horizontal case $H_R^D - 4$ (bundle).

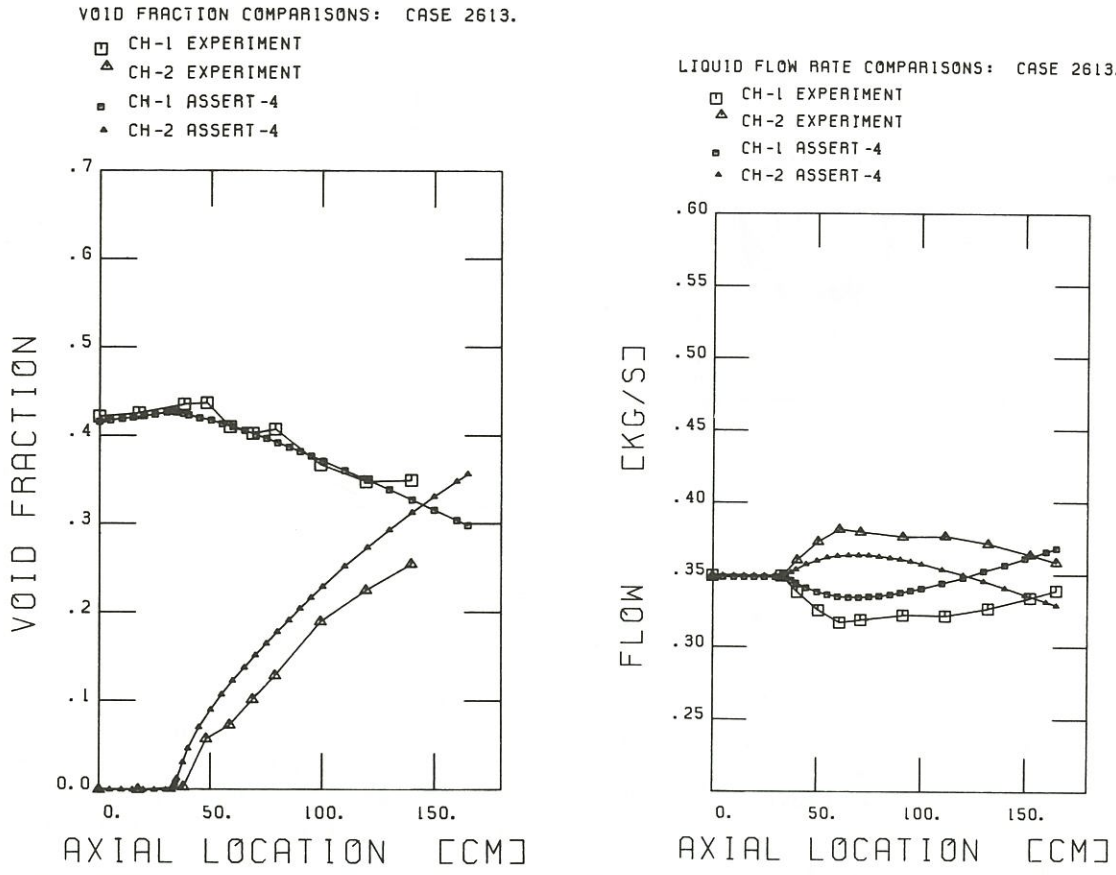


Figure 9: Computed and measured profiles of void fraction and mass flux for horizontal case $H_D^R - 3$ (bundle).

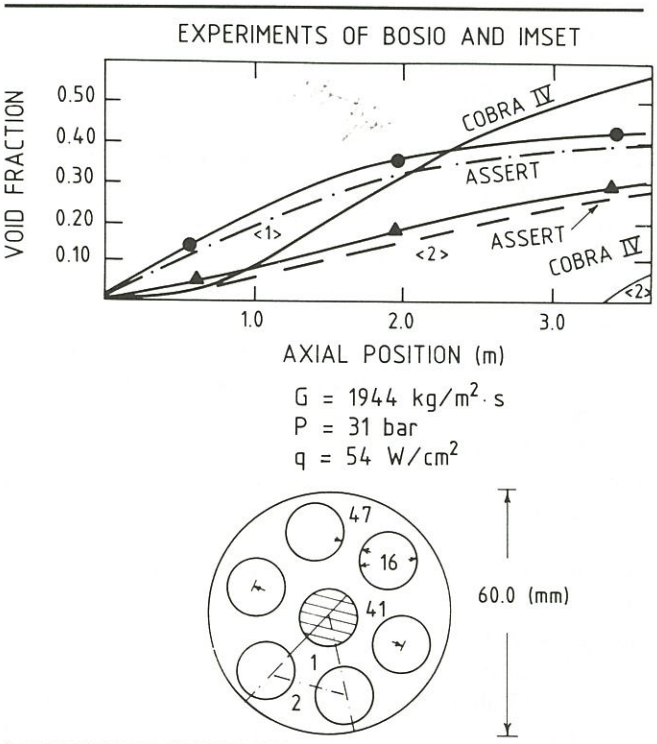


Figure 10: Bosio and Imset experiments: geometry and a typical comparison of computed and measured void profiles.

Validation for CANDU 37 Rod Bundle Geometries

A series of experiments, sponsored by the Atomic Energy of Canada – Ontario Hydro CANDEV co-operative agreement, have been completed in the Chalk River Nuclear Laboratories U1 experimental facility to measure critical heat flux (CHF) in a full-size 6 m horizontal channel, containing an electrically heated simulated string of CANDU 37-rod fuel bundles [16]. The experiments used for the comparisons had a non-uniform exit-biased cosine axial heat flux profile. The bundles were tested in the CRNL-U1 loop, which supplied light water coolant at flows, temperatures, and pressures covering the range of interest to CANDU operation and subject to the following limits: pressure 13.9 MPa, power 12.25 MW, and flow 17.0 kg / s. Tests were completed at four nominal pressure levels, six nominal flows, and seven nominal values of inlet subcooling, and results included pressure, temperature, and conditions corresponding to the first detectable occurrence of CHF on the available instruments, as evidenced by a surface temperature rise of at least 2°C, associated with a small increment in electrical power to the bundle. Resistance temperature devices (RTD) and sliding thermocouples were used to measure rod temperatures.

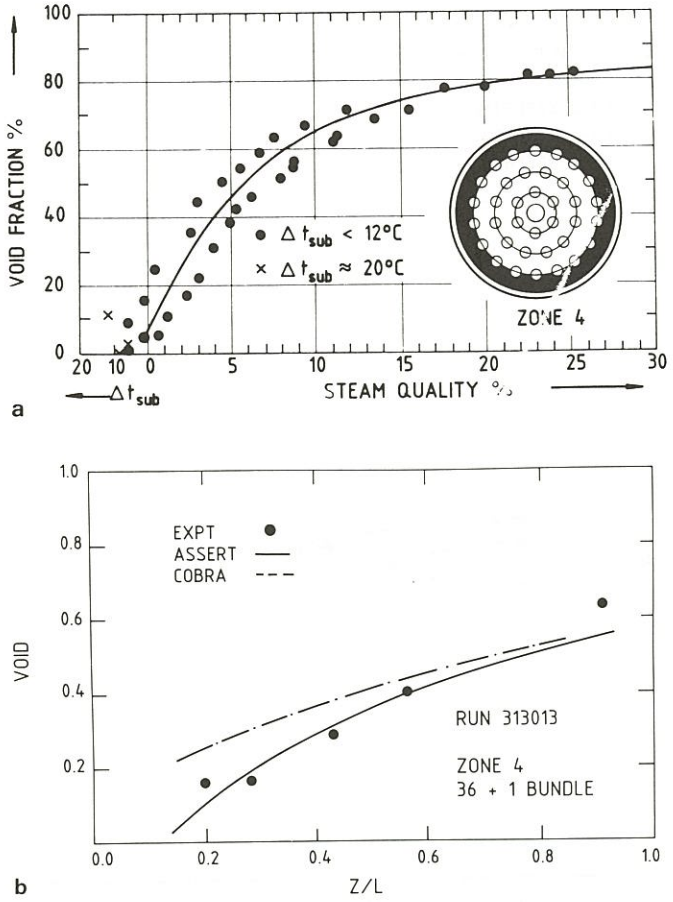


Figure 11: Marviken experiments: a) typical data, and b) typical comparison of computed and measured void profiles.

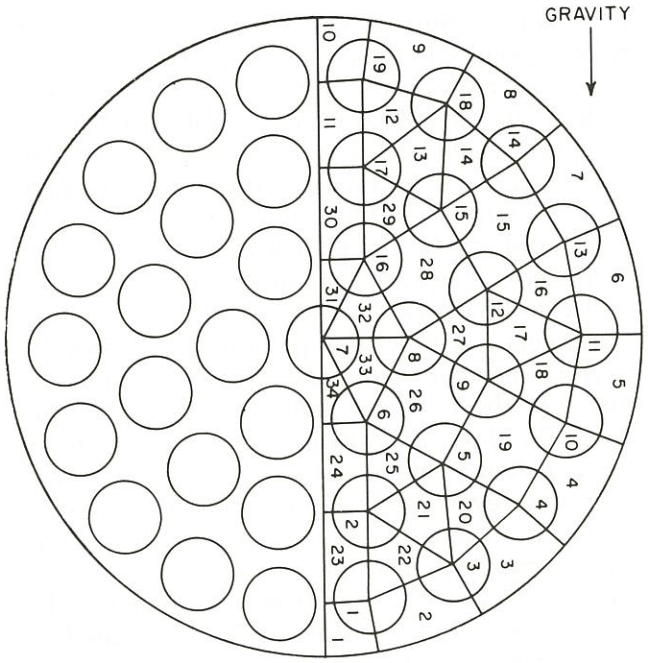


Figure 12: Subchannel and rod numbering in ASSERT for a 37-rod bundle simulation.

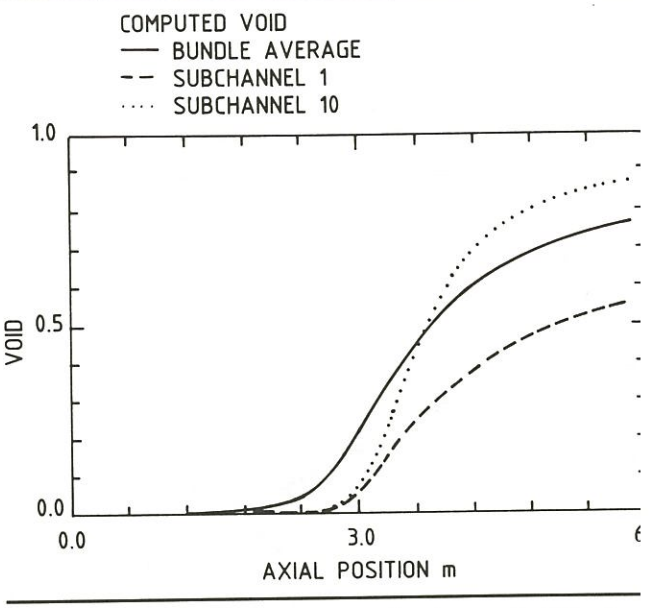


Figure 13: Pressure drop and void profiles for a typical 37-rod ex perimental case.

Prediction of Flow Distribution

Symmetry is used to advantage in modelling the bun dle in ASSERT, and rod and subchannel numbers are given in Figure 12. The axial layout included end plates and spacer planes. The first task is to check tha flow distribution is calculated adequately. Unfortun ately, there are no detailed experimental data that can be used directly to check distribution. The overal pressure drop is one criterion that can be used, and equally important, the overall pressure profile can b used to check single- and two-phase pressure drop calculations and the onset of local boiling.

Pressure profile predictions: A comparison of ASSER and measured pressure profiles is given for a typica case in Figure 13; the pressure drop profile is in goo agreement with the experimental measurements. Th point of change in slope of the pressure drop profil from a linear to non-linear relationship with axia position, indicates the axial location of the onset c significant void in the bundle.

This additional validation of the ASSERT results i important, as the onset of void occurs significantl upstream of the point at which bulk boiling would b computed to commence on a cross-sectional averag basis. In one-dimensional calculations, this effect usually accounted for by incorporating a subcoole boiling correlation, but may, in fact, be primarily due t the onset of boiling in the hotter subchannels. The fa that ASSERT predicts the apparent location of this poi suggests that calculation of flow distribution withi the subchannels is adequate.

Figure 13 also shows predictions of bundle averag void and the void in subchannels 1 and 10. Figure 1 shows that these are equivalent subchannels at th bottom and top of the bundle, respectively. The void

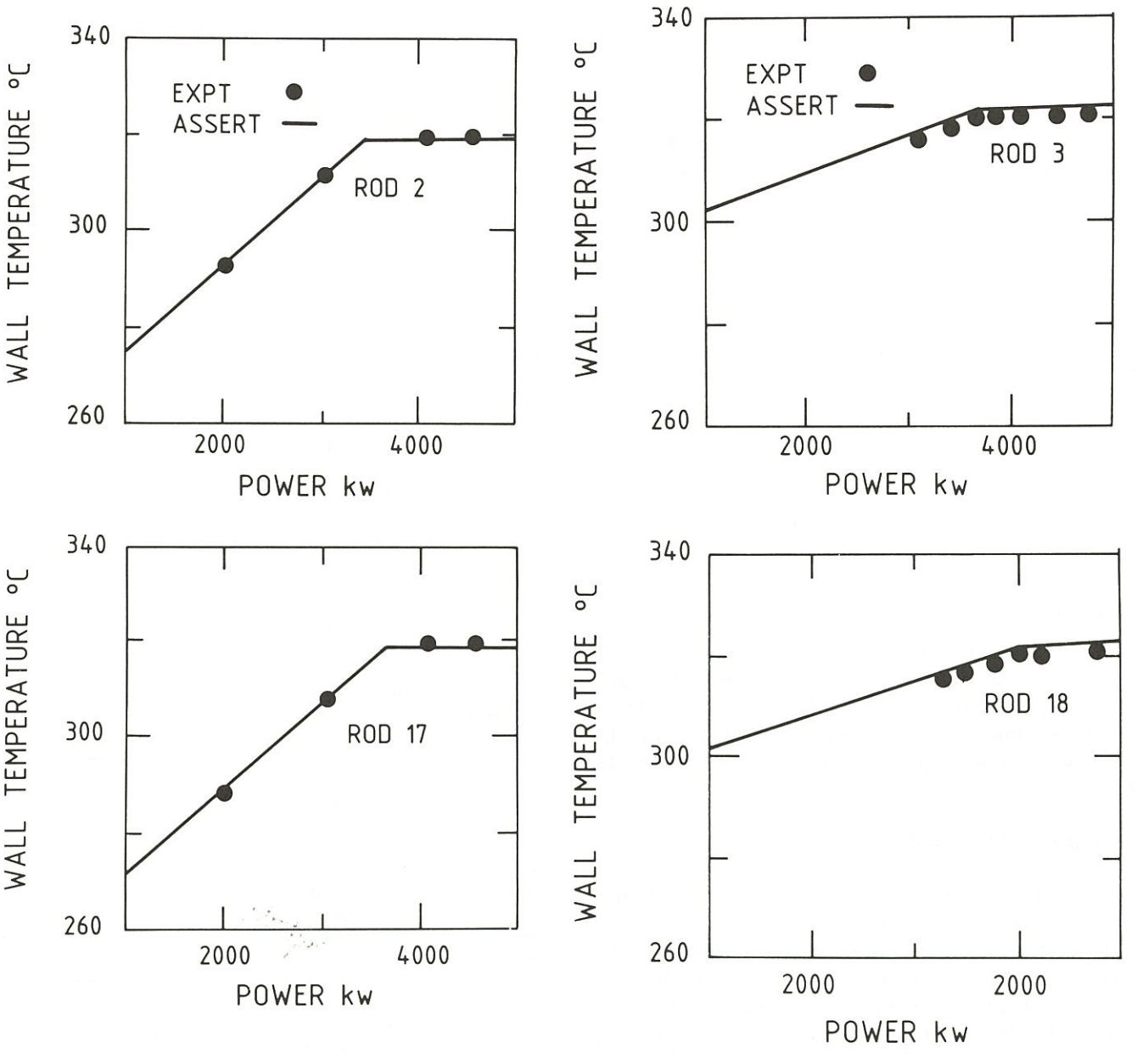


Figure 14: Measured and computed rod surface temperatures for a typical 37-rod case, showing two sets of upper and lower rods, for inner and outer subchannels.

the upper channel is predicted to be significantly higher in all cases. Comparisons with the experimental voids are not possible, as void was not measured.

Rod temperature predictions

A further means of checking the calculation of flow distribution is to calculate rod temperatures in ASSERT. This has been done for several representative experiments. The fuel model in ASSERT does not address electrical heaters, so the surface temperature of a rod section was calculated from the predicted fluid conditions in the subchannel facing the appropriate thermocouple, using boiling heat transfer correlations.

Comparisons between rod temperatures computed by ASSERT and the thermocouple measurements are given for two representative situations in Figure 14. The onset of nucleate boiling is shown clearly by the change in slope of the temperature curve, and the predictions agree well with the experiment. Furthermore, two interesting observations can be made. For the subchannels inside the bundle, it appears that buoyancy effects in the liquid are significant, as the upper subchannel of any geometrically similar pair reaches incipient boiling ahead of the lower one, as shown in Figure 14. However, in the outer subchannels this buoyancy effect is overridden by the stronger effect of eccentricity. The lower outer subchannels are smaller in size than the upper ones; this makes the

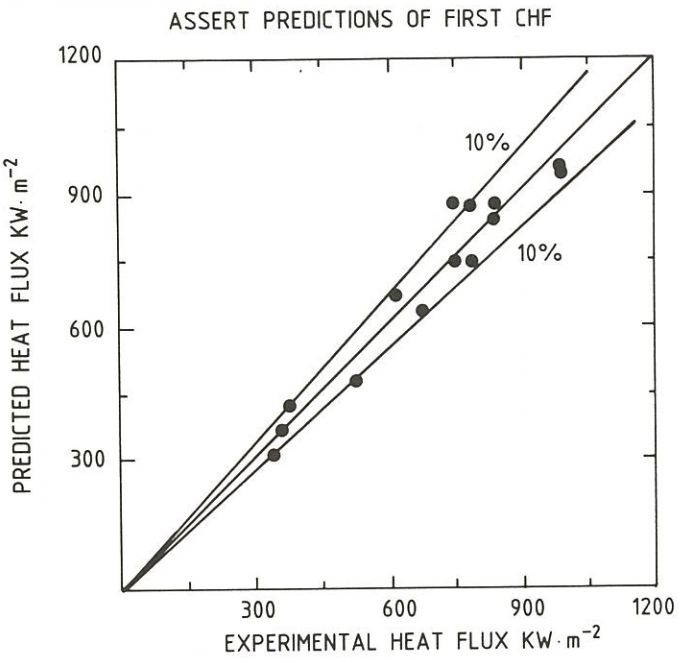


Figure 15: Measured and computed CHF for a 37-rod typical experimental case.

volumetric heat input higher and causes them to boil first, also as shown in Figure 14. Further details are given in Reference 17. Clearly, the onset of nucleate boiling is influenced by single-phase turbulent mixing in the liquid. The single-phase mixing model of Rogers and Rosehart [18] is used in ASSERT.

Critical Heat Flux Predictions

Having established that ASSERT adequately reproduces the measured pressure and temperature profiles, the next task is to compare CHF predictions.

Because of the non-uniform heat flux distribution, CHF occurs upstream of the end of the bundle, and not at the exit as in the case of uniform heat flux distribution. In this comparison, the total power was increased until the code predicted CHF somewhere in the bundle. Since the predicted location of CHF in the bundle is not necessarily that observed in the experiment, the heat flux is further increased until the predicted CHF location is at the same axial position as that observed in the experiment. The values of heat flux that first gave rise to a computed CHF anywhere in the bundle are referred to as 'first CHF predictions.'

In assessing these comparisons, it is also important to realize that the experimental data are not absolute, but have an associated RMS error. As discussed in Reference 16, the experimental CHF RMS error is about 5% for a tube. CHF measurements in a bundle having a non-uniform axial heat flux profile are more complicated than those of the tube experiments, so an RMS error greater than 5% is probable. The first CHF predictions, using the Whalley model in ASSERT, are given in Figure 15 for 14 experimental cases chosen to cover the range of variables.

Table 1: Nomenclature

Variable	*Description*
C_0	Phase distribution parameter
F	Wall friction
g	Acceleration due to gravity
h	Mixture enthalpy
h_g, h_f	Phasic enthalpy
j	Mixture volumetric flux, $(aV)^g + (\alpha V)^f$
P	Pressure
q	Heat transfer rate
t	Time
U	Mixture axial velocity
U_r	Relative axial velocity, $U_g - U_f$
V	Mixture velocity
V_r	Relative velocity, $V_g - V_f$
v_{gj}	Drift velocity lateral
V_r	Lateral relative velocity, $V_g - V_f$
$V\infty$	Bubble rise velocity
V_{gj}	Drift velocity
U_{gj}	Drift velocity axial
V_g, V_f	Phasic lateral velocity
α	Void fraction, $\alpha = \alpha_g$
α_0	Equilibrium void fraction
α_g, α_f	Phasic void fraction, $\alpha_g + \alpha_f = 1$
l	Centroid-to-centroid distance between subchannels
ϕ	Centroid-to-centroid angle
ρ	Mixture density
ρ_g, ρ_f	Phasic density
σ	Surface tension
ε	Diffusion parameter
Subscripts	
if	Interface-to-liquid
ig	Interface-to-vapour
f	Liquid
g	Vapour
w	Wall
wf	Wall-to-liquid
wg	Wall-to-vapour
Superscripts	
–	Average
→	Vector
′	Per unit length
″	Per unit area
‴	Per unit volume

In all cases, ASSERT predicts CHF to occur first in th top rod, number 19, of the outer ring in the bundle subchannel 10. This was also observed experimentall for low-flow cases, while for the rest of the cases, CH occurs in the top rod, number 17, in the second ring. I all cases, ASSERT predicts CHF will occur in this secon subchannel ring (subchannels 12 and 13) with a sligh increase in power. As discussed previously, thes computations represent a first application of ASSERT t CHF prediction, and it is clear that the results ar reasonable.

More recent work with ASSERT has been directe towards improving the CHF methodology, and extenc ing the ASSERT CHF repertoire, in particular to includ the CHF table method [18]. These extensions are d

scribed in Reference 19, and extensive comparisons have been made with CHF data for a number of different experiments in horizontal rod bundles. A detailed report on these comparisons is in press. Comparisons of ASSERT predictions to CHF experiments in a horizontal 28-rod bundle have been completed by Ontario Hydro [20].

Conclusions

The first phase of the ASSERT, advanced subchannel code development has been completed, illustrating that the code is capable of computing flow and phase distribution effects in horizontal channels and fuel bundles. In the U1 experiment, there were no direct measurements of flow distribution, so indirect indications of distribution were used for comparison. The code was able to match pressure profiles and rod temperatures quite closely. Finally, a first attempt at computing local CHF was made, and the results were encouraging.

Acknowledgements

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A Review of Low-Level Radioactive Waste Management Technology in the Canadian Nuclear Industry

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Abstract

Canada's low-level radioactive wastes (LLW) (generated primarily from the CANDU nuclear fuel cycle, radioisotope research, and industry) have a wide range of physical forms and radionuclides, and are currently managed either by producers or by the Atomic Energy of Canada's Chalk River Nuclear Laboratories (CRNL), which operates a national collection and management service for small producers. The processing and storage methods are generally well established. Substantial research and development is in progress for a gradual transition to disposal methods, including a shallow land burial (SLB) demonstration facility at CRNL. With a federal policy that encourages producers to propose disposal methods, the stage is now set for a transition from the current interim methods to long-term methods of LLW management.

Résumé

Au Canada, les déchets faiblement radioactifs sont surtout des sous-produits du cycle du combustible nucléaire CANDU ainsi que des activités de recherche et de fabrication de radio-isotopes. Ces déchets se présentent sous une grande variété de formes physiques et de radionucléides. Leur gestion est actuellement assurée par les producteurs eux-mêmes ou encore par les Laboratoires nucléaires de L'Energie Atomique du Canada à Chalk River, qui offrent un service de cueillette et de gestion aux petits producteurs de tout le pays. Les méthodes de traitement et de stockage sont généralement bien établies. Des travaux importants de recherche et de mise au point sont actuellement en cours afin d'assurer une transition graduelle vers des méthodes d'évacuation, y compris une installation de démonstration d'enfouissement à faible profondeur aux Laboratoires nucléaires de Chalk River. Grâce aussi à la politique du gouvernement fédéral encourageant les producteurs à proposer des méthodes d'évacuation, le Canada est maintenant prêt pour une transition sans heurts des méthodes actuelles provisoires à des méthodes à long terme de gestion des déchets faiblement radioactifs.

Introduction

Low-level radioactive wastes (LLW) generated in Canada broadly fall into one of the following categories:

a) those produced by the Canadian nuclear industry (e.g., in the uranium fuel production and power generating stages of the nuclear fuel cycle, and nuclear research and radioisotope processing facilities); and
b) those produced by a large number of licensed radio-isotope users (such as hospitals and laboratories) and a number of non-nuclear industries dealing with naturally radioactive feedstocks in their operations.

Electric utilities with nuclear generating stations in Ontario, Quebec and New Brunswick; uranium refiners; fuel fabricators; and Atomic Energy of Canada Limited (AECL) produce category a) wastes, which account for the major portion of the low-level wastes in Canada. AECL's Chalk River Nuclear Laboratories (CRNL) provide a national fee-based radioactive waste collection and storage service for those institutions that produce only small volumes of wastes, such as the over 5,000 licensed users of radioisotopes (category b). The wastes from industrial generators arising from processes that use raw materials containing naturally occurring radionuclides (abrasives manufacturing, specialty metal alloy production, etc.) make up the rest of category b). Not included here, are the uranium mine and mill tailings, which are locally managed by the mining industry. Table 1 summarizes the LLW arising in Canada.

There are no licensed low-level radioactive waste disposal facilities in Canada, although studies are in progress, and long-term plans are likely to be implemented in several organizations over the coming decades. The Low Level Radioactive Waste Management Office (LLRWMO) of AECL is spearheading analysis of the need and alternatives for establishing disposal facilities in Canada.

Low-level waste management in the Canadian nuclear industry has reached maturity in two important phases:

Keywords: radioactive wastes, processing, storage, disposal.

Table 1: Canada's Low-Level Waste Volume Projections (Based on Ref. 1)

	LLW projections (m^3) to year 2025
Category (a)	
Canadian nuclear industry	
Refining	65,000
Fuel fabrication	14,800
Utilities	156,500
Isotopes and research	61,200
Category (b)	
Other producers (institutional / industrial)	
Licensed users	12,900
Industries using naturally radioactive feedstocks	57,100
Total	367,500

These exclude about 1.2 million m^3 of wastes, primarily contaminated soils at several 'historic' sites, CRNL site and waste management sites of Eldorado Resources Limited at Welcome and Port Granby, Ontario.

Some compaction of the wastes at the source is assumed, as is carried out by the producers normally.

a) in the interim management of the diverse waste sources; and
b) in the technological research and development in support of plans for disposal of LLW.

This paper will review Canadian low-level radioactive waste management technology and outline Canadian efforts in developing low-level waste disposal systems.

Sources and Character of Low-Level Wastes

Technologies used in the various phases of LLW management share the common objective of safe containment of radioactivity. Waste properties differ widely across the industry and generally have been well characterized. Table 2 summarizes LLW characteristics.

The fuel production stages of the nuclear fuel cycle, which include uranium refining and fuel fabrication processes, yield uranium-contaminated materials and residues. Eldorado Resources Limited, the federally owned refiner, produces the major component of these wastes, which are currently managed in storage facilities near the plant at Port Hope, Ontario. Wastes from Canada's two fuel fabricators, Canadian General Electric and Westinghouse Canada, are sent to CRNL for storage.

Wastes in the power generating stages of the nuclear fuel cycle make up the major ongoing volume component of nuclear industry wastes. Ontario Hydro, which has a committed nuclear program of 13,600 MWe, is by far the major producer of these wastes; the other

Table 2: Characterization of Canada's LLW (1984 Statistics Compiled by Ontario Hydro for LLRWMO, Ref. 2)

Source / physical classification	*Major radionuclides (and half-life (a))*	*Typical radioactivity concentration Ci / m^3*
Category (a) – Canadian nuclear industry		
1 Refining	U-natural U-depleted	up to 5×10^{-2}
2 Fuel fabrication	U-natural U-238	up to 0.00013
3 Utilities		
Low-level (non-incinerable)	H-3 (12.3)	1×10^{1}
	C-14 (5730)	5×10^{-6}
	Co-60 (5.3)	1×10^{-2}
	Sr-90 (28.6)	1×10^{-3}
	Cs-137 (30.2)	3×10^{-3}
Low-level (incinerable)	Co-60 (5.3)	2×10^{-3}
	Sr-90 (28.6)	8×10^{-4}
	Cs-134 (2.1)	5×10^{-4}
	Cs-137 (30.2)	2×10^{-3}
Intermediate-level resins	H-3 (12.3)	2×10^{0}
	C-14 (5730)	2×10^{2}
	Co-60 (5.3)	4×10^{0}
	Sr-90 (28.6)	1×10^{-1}
Intermediate-level filters	Co-60 (5.3) Cs-137 (30.2)	Direct measurements not yet available
Irradiated core components (retubing)	Fe-55 (2.7)	3×10^{4}
	Co-60 (5.3)	6×10^{3}
4 AECL research	Co-60 (5.3)	9×10^{-2}
	Cs-134 (2)	15×10^{-3}
	Cs-137 (30.2)	2×10^{-2}
Low-level (non-incinerated)	H-3 (12.3)	5×10^{-3}
	C-14 (5730)	2×10^{-5}
	Sr-90 (28.6)	1×10^{-6}
	Tc-99 (2)	2×10^{-5}
	Cs-137 (30.2)	1×10^{-6}
	Mixed fission products (MFP)	7×10^{-2}
	Activation products (MAP)	17
Low-level (incinerated)	Co-60 (5.3)	7×10^{-1}
	Cs-134 (2)	1×10^{-1}
	Cs-137 (30.2)	4×10^{-1}
Intermediate-level	H-3 (12.3)	12×10^{-2}
	Co-60 (5.3)	134
	Sr-90 (28.6)	14×10^{-3}
	Cs-137 (30.2)	3×10^{-2}
	Ra-226 (1600)	2×10^{-2}
	Am-241(432)	1×10^{-2}
	MFP	5×10^{3}
	MAP	8×10^{2}
Sealed sources	Sr-90 (28.6) Cs-137 (30.2) Am-241(432)	

Table 2 (Continued)

Source / physical classification	*Major radionuclides (and half-life (a))*	*Typical radioactivity concentration Ci / m³*
Category (b) – Institutional / industrial		
Institutional	H-3 (12.3)	<100
	C-14 (5730)	3×10^{-3}
	Sr-90 (28.6)	5×10^{-4}
	Cs-137 (30.2)	0.3
	Ra-226 (1600)	9×10^{-5}
	Am-241(432)	0.2
	Co-60 (5.3)	1.3
Industrial	Ra-226 (1600)	1.5×10^{-4}
	Th-230 (7.7×10^4)	1.5×10^{-4}
	Th-232 (1.4×10^{10})	1×10^{-3}
	U-natural	2.1×10^{-4}
	H-3 (12.3)	<0.1
	C-14 (5730)	0.2
	Co-60 (5.3)	<0.1
	Cs-137 (30.2)	<0.1
	Ra-226 (1600)	<0.1

contributors are the provincial electric utilities of Quebec and New Brunswick. The wastes are classified as 1) low- and 2) intermediate-level wastes. Both these subcategories are non-heat-generating, and are hence 'low-level,' although intermediate-level wastes require shielding. Low- and intermediate-level wastes consist, essentially, of all radioactive wastes produced in CANDU nuclear generating stations (NGS), other than those contained in the irradiated fuel. These wastes primarily consist of

a) housekeeping wastes, such as paper and plastic sheeting, temporary floor coverings, used protective clothing, rubber gloves and plastic suits, mopheads, rags and other cleaning materials, and contaminated hardware;
b) spent ion exchange resins and filters from purification systems; and
c) large irradiated and contaminated core components, arising from rehabilitation and retubing of reactors.

These wastes are mostly contaminated with short-lived radionuclides, such as Co-60, Cs-137, Sr-90, and H-3, with a particular segment of the waste (resins) containing C-14, a radionuclide with a half-life of 5,730 years.

The nuclear research laboratories at the Chalk River Nuclear Laboratories in Ontario, the Whiteshell Nuclear Research Establishment in Manitoba, and AECL's radioisotope processing facility in Ottawa are the major contributors of the remaining wastes from the Canadian Nuclear Industry. These consist of contaminated materials from laboratories, maintenance and purification wastes from research reactors, and wastes from isotope processing. These are not altogether different from the utility wastes in radiological character.

Institutional and industrial (category b) wastes consist of a wide range of radionuclide materials, such as sealed sources used in industrial equipment such as gauges, industrial radiography cameras, and static electricity eliminators; contaminated materials (i.e., animal carcasses, scintillation vials, liquids, filters, syringes, wipes and gloves from medical applications of radioisotopes); and residues from abrasives manufacturing or speciality metal alloy industries, which process raw materials containing naturally occurring (incidental) radionuclides. While the institutional wastes are handled by CRNL's national collection and storage service, incidental wastes from the industries are generally managed by the producers themselves.

Waste Management Technology

The major technologies in the management of LLW include

a) processing;
b) transportation;
c) storage; and
d) disposal.

Producers segregate wastes 'at the source,' taking into consideration the physical / radiological properties of the waste, to facilitate the application of the above technologies.

Processing

Processing of wastes is undertaken to reduce the volume and / or produce a wasteform more suitable for packaging, storage, and eventual disposal. For example, some 90 per cent of low-level utility waste is processible, either by mechanical compaction or incineration. Compaction results in a volume reduction ratio of about six, while incineration provides a volume reduction ratio of about 75.

Processing of LLW by incineration and baling has been adopted by Ontario Hydro [3] and Chalk River Nuclear Laboratories [4], the two major producers in the Canadian Nuclear Industry. Ontario Hydro has been operating a Waste Volume Reduction Facility (WVRF) at the Bruce Nuclear Power Development (BNPD) since 1977.

With waste sources that rapidly increased in number in the 1970s, due to an expanding nuclear program Ontario Hydro put into service in-station waste management systems for collection, segregation, and packaging of wastes, as well as a centralized waste management site at the BNPD consisting of an incinerator baler / compactor system, and a central maintenance facility that carries out laundering, decontamination and other 'active' maintenance operations in support of nuclear stations [5]. AECL has constructed a Waste Treatment Centre (WTC) to process and condition CRNL LLW. The WTC is composed of an incinerator and baler for solid wastes, an ultrafiltration and reverse-osmosis system for the concentration of aqueous wastes, and equipment for immobilizing the ash and solids from the

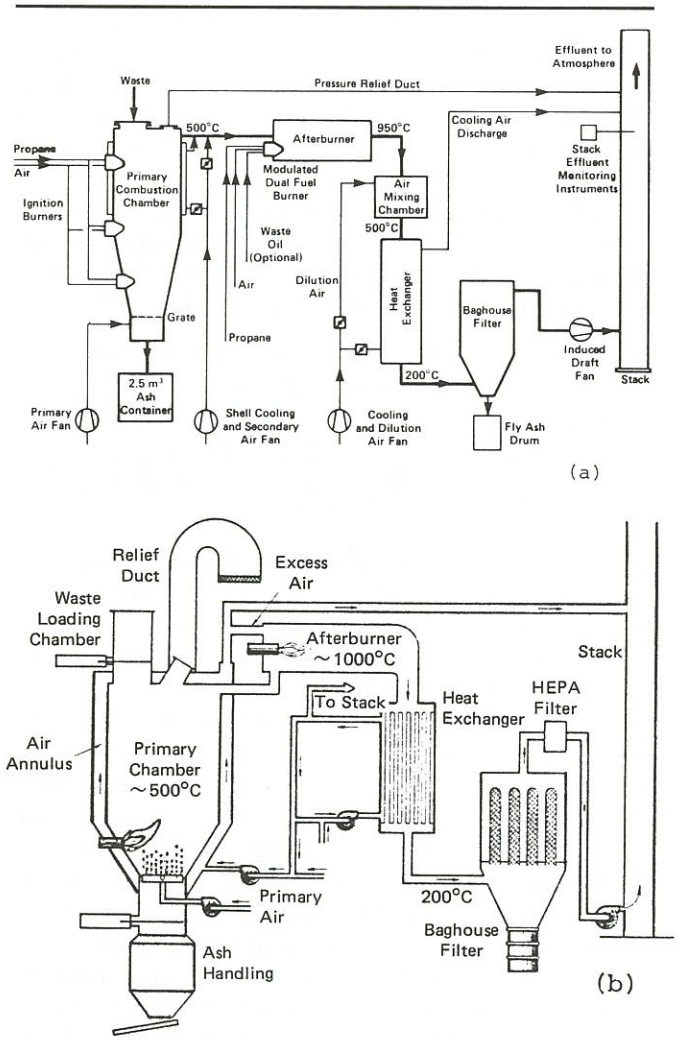

Figure 1: LLW Incinerators in Canada (a) Ontario Hydro (b) CRNL.

waste concentrates into a bitumen matrix. The goal is to produce a final-conditioned waste which is in a stable, compact, and leach-resistant form suitable for both storage and disposal. By combining several processes in a full-scale integrated system, the WTC serves to develop waste conditioning methods, improve the management of CRNL site wastes, demonstrate waste processing technologies, and generate performance and cost data for other Canadian nuclear facility owners.

Incineration Technology

Ontario Hydro's nuclear program currently generates about 6,000 m^3 unprocessed low-level waste per year, and this quantity is expected to increase to over 8,500 m^3/yr by 1992. Approximately 65 per cent of this volume is classified as incinerable. The Ontario Hydro system, like the CRNL system, utilizes a controlled air batch-pyrolysis technique (Figure 1), in which the combustion air quantity is starved in the primary chamber to about 30 to 50 per cent stoichiometric. The pyrolysis effluent from the combustion chamber is then fully oxidized in an afterburner. The dry off-gas clean-up system consists of an off-gas cooling stage and a one-step filtration stage in a baghouse; no polishing filtration is employed [3].

Although the Ontario Hydro incinerator is a working prototype that has required modifications during its operating life, it has, nevertheless, become one of the most productive incineration systems in the nuclear industry. To the end of 1985, over 20,000 m^3 of low-level waste has been processed in over 55,000 operating hours. Waste with a contact dose rate of up to 0.6 mSv/h is incinerated. Typically, solid waste with a specific gross gamma activity of 0.02 to 0.08 GBq/m^3 has been processed. Incinerator ash, which has a specific activity ranging from 0.08 to 8 GBq/m^3 is 'dumped' into 2.5 m^3, rectangular galvanized steel containers, which are then placed in the storage structures. Contact fields on most of the ash containers are between 0.1 to 0.2 mSv/h. Radioactive emission experience with the incinerator has been very satisfactory, with particulate gamma activity on the order of 70 KBq released through the stack for each m^3 of waste burned.

CRNL's incinerator, which also uses a starved-air batch pyrolysis process, is a more advanced version of the production unit operated by Ontario Hydro. It has improvements in control, process versatility, and the use of corrosion-resistant materials. It is designed to process batches of up to about 1,300 kg of solid waste in a nominal 24-h cycle. Particulate beta-gamma stack releases have remained less than 37 KBq per burn.

Transportation

Transportation of low-level waste is carried out in accordance with IAEA transportation regulations enforced by the Atomic Energy Control Board. Most wastes, such as the bulk LLW, contaminated soils, etc., qualify – depending on their radioactivity – either as LSA (low specific activity) or type A wastes. Waste materials with higher concentration of radioactive contaminants, such as intermediate-level wastes, require transportation in accident-resistant type B packages. The classification of transportation packages (as LSA, type A, or type B) is carried out in accordance with transportation regulations.

The infrastructure is now available in the Canadian nuclear industry to design, test, and commission transportation packages for low-level wastes, and for radioactive materials with higher levels of radioactivity – such as irradiated fuel and cobalt-60.

Storage

Ontario Hydro Experience

Currently, all radioactive waste materials are stored at BNPD, in a retrievable manner, in facilities having design lifetime of 50 years [6]. No radioactive materials are placed directly in soil; either in-ground or above-ground engineered structures are used.

The storage site consists of 19 acres (0.8 km^2) and a variety of storage facilities built on relatively impermeable glacial till deposits. Ontario Hydro has been developing the BNPD Radioactive Waste Operations Site for the last fifteen years [7]. To date, 37,000 curies (as stored) of radioactive wastes are estimated to be stored at the site. Among the storage facilities are

a) reinforced concrete trenches used for the storage of the low-level wastes;
b) in-ground structures, called 'tile holes' (used to store filters and ion-exchange resins that contain a higher level of radioactivity), including newer versions that employ borehole augering technology to allow faster construction, lower costs, and greater depths;
c) two above-ground prefabricated, prestressed concrete superstructures, called 'low-level storage buildings' (LLSB's), now being used for storage of low-level wastes with radiation fields less than 10 mSv / h;
d) double-walled, above-ground reinforced concrete structures, called 'quadricells,' used primarily to store intermediate-level resins, with a secondary role of storing highly radioactive core components.

AECL Experience

The CRNL facilities are located in elevated and well-drained deposits of sand [8]. The radioactive waste is generally placed above the water table, to reduce the likelihood of contact with water. Close to 100,000 m^3 of solid radioactive wastes are stored or buried at the CRNL property. Eighty per cent is LLW, 15 per cent is intermediate-level, and five per cent is high-level waste. The LLW is generally buried unprotected in sand trenches, well above the water table. Solid wastes with higher radioactivity are stored, retrievably, above the water table in engineered concrete structures, ranging in diameter from 0.15 to 6.0 m, and in depths of up to 5 m. Each structure is fitted with a removable, weatherproof shielding cap, and protrudes less than a metre above grade.

Others

Two other Canadian utilities (Hydro Quebec and New Brunswick Power) have local sites for management of low-level wastes. These utilities employ designs similar to the engineered storage facilities of Ontario Hydro and CRNL. Eldorado Resources Limited, the major refining industry, operates its own storage facilities a few miles from its Port Hope plants. These facilities primarily consist of above ground waste emplacement schemes or shallow burial. Industries using materials in production processes that are incidentally radioactive (e.g., abrasives industry) generally store the waste materials at the plant sites.

Present Research and Development into Disposal

The above methods of storage are considered interim in that at least some of the wastes will be radioactive beyond the timeframe of storage and will require disposal. Disposal, by definition, is a permanent method of management, without the intention of retrieval, and does not rely for its success on perpetual institutional controls and monitoring.

The Chalk River Nuclear Laboratories have taken the lead in developing and demonstrating a disposal capability for low-level wastes in Canada [9]. Three concepts selected for study by CRNL include

a) 'improved sand trench' (IST) for wastes that need isolation up to about 150 years;
b) intrusion-resistant 'shallow land burial' (SLB) for wastes that require isolation up to about 500 years;
c) 'shallow rock cavity' (SRC) for wastes that need isolation for more than 500 years.

Based on knowledge of the radiological characteristics of the stored wastes, it is anticipated that the bulk of the waste could be disposed of in the SLB Facility (Figure 2). The other two concepts are considered potential complements to SLB.

The SLB is about 100 m long by 20 m wide by less than 10 m deep, with the top of the wall near the surface and the bottom above the water table. Once filled it will be covered with a self-supporting, water-shedding, concrete roof (and perhaps other water-shedding barriers), then buried under a relatively thick ground-cover to prevent erosion, and thus stabilize the topography.

Continued engineered storage of LLW wastes is considered the essential ingredient in Ontario Hydro's plans. Programs are well advanced in the research and development of disposal technologies for those long-lived or higher-radioactivity wastes that will require disposal. Decisions on acquisition and proponency for a disposal facility have not been made up to this stage [10].

Eldorado Resources Limited (ERL) have been evaluating disposal facilities for their currently stored refinery wastes and for their ongoing production of LLW. Near-surface burial in glacial till, and intermediate-depth burial concepts in the local limestone geology, have been researched for application in the regions surrounding their Port Hope refining operations.

Responsibilities and Other Issues in LLW Management

Although the responsibilities of the provincial and federal governments in the area of low level-waste management is still a subject for discussion, some of the jurisdictional aspects are becoming clearer in Canada. Of importance are the following:

a) The federal government has established the Low-Level Radioactive Waste Management Office (LLRWMO) of Atomic Energy of Canada Ltd, in Ottawa, as the agency to discharge federal responsibilities in the area.

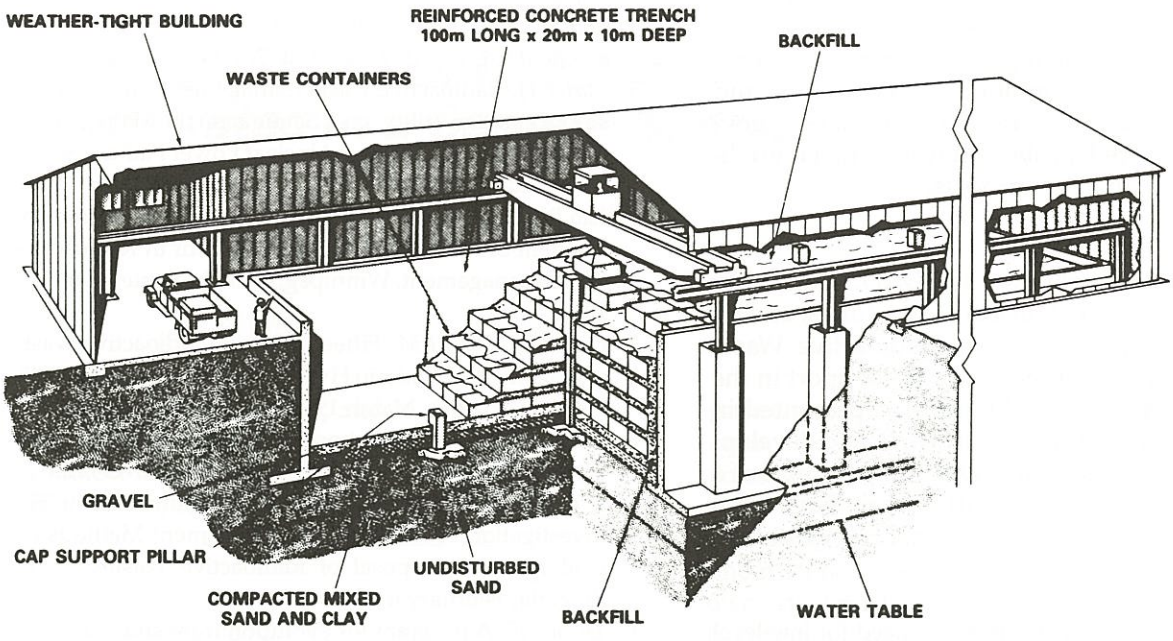

Figure 2: CRNL shallow land burial (SLB) facility.

b) The federal government accepts residual responsibility for LLW, i.e., responsibility for the wastes for which no person or company can be held responsible.
c) The federal government has adopted the principle (Federal Policy on LLW, 1986) that the primary responsibility for the management of radioactive wastes, including disposal, must rest with the producers of such wastes, and that the costs of waste management should be borne by those benefitting from the activities responsible for the generation of wastes [11].

The producers are accountable for ensuring that the wastes are properly isolated over their hazardous lifetime. This could include the development of sole- or joint-use disposal facilities and sites. The federal government may accept residual responsibility,

1. as in the case of cleanup and disposal of historic wastes, wastes from small producers, or companies no longer in business; and
2. as in the long-term stewardship of disposal sites after they have been closed and the producer's responsibility has been terminated.

One of the tasks undertaken by the LLRWMO is to establish, or to ensure the establishment of, low-level radioactive waste disposal facilities that could be used by institutions, such as universities and hospitals (small producers), on an ongoing basis. These low-volume producers are those who would otherwise be unable to establish their own facilities.

The benefits from the nuclear industry are diffused throughout society, while the perceived detriments from waste facilities are local to host communities. The Federal Policy on LLW management recognizes that the ideal democratic principle – that preference should be given to courses of action resulting in greater good for the greater number of people – is not widely accepted by residents who live near a proposed waste facility. Recent opposition from potential recipient (host) communities to relocation of contaminated materials / soils from past operations are cases in point. Although many factors (such as human health and safety, environmental protection, and general societal concerns) are taken into consideration by any proponent, it is absolutely essential that co-operation and participation of the public, and local and senior levels of government be sought in the necessary decision-making processes. In some cases, it is anticipated that an area that hosts a disposal facility may obtain 'offsets' for accommodating the facilities.

Summary

The Canadian nuclear industry has reached maturity in the interim management of all the waste segments produced not only by the nuclear industry, but by over 5,000 of the nation's institutions for which the Chalk River Nuclear Laboratories provide a service in collection and storage of their LLW. With the ultimate aim of providing for safe isolation of radioactivity over the hazardous life of LLW, the industry is involved in comprehensive technological research and development for disposal systems. CRNL has taken the lead in demonstrating the disposal capability in Canada, by commitment to a shallow burial facility (SLB) at the CRNL site.

Lastly, policies and regulations critical to acquisition

of new waste management sites are emerging in Canada. These should facilitate transition from the current interim waste management practices to methods providing permanent isolation of LLW. Co-operation and participation by the public and different tiers of government is considered an absolutely essential ingredient in the decision-making process.

Acknowledgements

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Characterization of Radioactive Wastes Incorporated in a Cement Matrix

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Abstract

The incorporation of radioactive material in cement mixtures is a commonly used procedure for conditioning various kinds of low- and medium-active waste streams. The resulting solid products are non-combustible, sufficiently radiation resistant, and exhibit reasonable mechanical strength as well as chemical and thermal stability. In order to learn more about the underlying mechanisms responsible for the fixation of the radio-elements, phase characterization studies have been performed. Main interest was focussed on the question of whether cesium and strontium show mineralization, and thus improve their leachability resistance. It could be shown that both elements appear to be just more or less uniformly distributed in the gelatinous filled pore spaces. Improvement of Cs and Sr leach rates is mostly due to sorption phenomena on selected additives, but more important, it depends essentially on the degree of porosity of the cementitious products.

Résumé

L'incorporation de matériel radioactif aux mélanges de ciment est une méthode utilisée couramment pour le conditionnement de divers types d'effluents de faible et de moyenne activité. Les produits solides qui en résultent présentent une tenue suffisante aux rayonnements, une résistance mécanique raisonnable ainsi qu'une stabilité chimique et thermique. Des études de caractérisation des phases ont été effectuées afin de mieux comprendre les mécanismes sous-jacents responsables de la fixation des radioéléments. L'intérêt principal a porté sur la question à savoir si le césium et le strontium présentent un état de minéralisation, ce qui améliorerait leur résistance à la lexiviabilité. Il a pu être prouvé que les deux éléments semblent distribués plus ou moins uniformément dans les pores remplis de matière gélatineuse. L'amélioration des taux de lixiviation du césium et du strontium est principalement due au phénomène de sorption se manifestant chez certains additifs, et dépend essentiellement du degré de porosité des produits de cimentation.

Introduction

The method generally considered for the permanent disposal of nuclear waste is to incorporate the radioactive material into a solid, which is then placed in a geological repository. The solid is made of waste and nonradioactive additives, with the formulation selected to produce a durable solid that will minimize the potential for dispersal of the radionuclides.

Cement mixtures are widely used as matrices for the immobilization of low- and medium-active wastes. Cement has many favourable characteristics that have contributed to its popularity as a radioactive waste fixation matrix [1]. The required solidification technology is rather simple. The resulting solid products are noncombustible, strong, radiation-resistant, and have reasonable chemical and thermal stability.

A disadvantage of concrete is its low thermal conductivity, which makes it unsuitable for high-activity wastes, where the heat deposition is appreciable. Also gas generation due to radiolysis may cause problems.

The waste form must be considered as one part of a total nuclear waste disposal system. Its primary role within this system is to provide the initial barrier against radionuclide release. Emphasis is focused on reducing the availability of the waste radionuclides to water leaching when the wastes are disposed of in a repository. Current policy on long-term waste isolation is that the waste package, which includes the waste form and the canister, should provide a complement to the natural geological barrier until the waste hazard has been greatly reduced by radioactive decay. Most important with regard to safety aspects is the operational time of a repository. In the post-closure phase, product integrity is of less importance, since in the long term the natural geological barriers will provide radionuclide retention.

Keywords: radioactive waste management, leaching, cesium, strontium, cement, radiation stability, mechanical stability, morphology, fission products.

The major issues regarding the application of cement-based waste forms to radioactive waste management problems are

1. leachability;
2. radiation stability;
3. mechanical stability;
4. phase complexity of the matrix; and
5. effects of the waste stream composition.

A cursory state of the art review is given for items 1 and 2, whereas a detailed presentation of experimental results is included for the other issues.

Leachability

Leach rates of radionuclides from the solid cementitious waste product indicate the important parameter: quantity of radioactivity as a function of time that is available for dispersal. However, one has to distinguish between two different mechanisms:

1. corrosion of the cementitious waste product, and thus uniform disintegration of the solid; and
2. element-specific dissolution of radioelements by selective leaching.

Considering a simple corrosion attack in salt brine, different processes are involved:

- the intrusion of chloride ions into the pore system of the cement stone;
- leaching-out of nitrate ions;
- interchange of Ca^{++} for Mg^{++};
- corrosive attack due to the formation of $CaSO_4$ phases;
- true dissolution of radionuclides.

This whole process does not necessarily involve a complete dissolution, and thus a mobilization of the disintegration material. Part of it may be truly dissolved, whereas the rest may just exist in a dispersed state. Because of the prevailing fairly high pH value, many ions form insoluble oxyhydrates. Only the radionuclides present as dissolved iogenic species are mobilized by liquid transport phenomena. The dispersed fraction is much less mobile.

The corrosion process obeys a $\sqrt{t}$-law. An important parameter affecting the corrosion rate is the water to cement value (w / c). The lower the ratio w / c, the better the corrosion stability. The parameter most likely responsible for the observed effect is the increasing porosity with increasing w / c-value, measured between 0.27 and 0.50. This observation is in agreement with the finding that low-porosity cementitious products obtained using blast-furnace or high-alumina cements exhibit a higher corrosion stability.

In the presence of brines containing sulphate (e.g., Q-brine), the cement products swell under the formation of gypsum phases or double salts like $Ca_3Al_2O_6 \cdot CaCl_2 \cdot 10H_2O$, $Mg_2(OH)_3Cl \cdot 4H_2O$. However, no ettringite formation, $Al_2(SO_4)_3 \cdot 6\,Ca(OH)_2 \cdot 26\,H_2O$, has been observed, which normally adversely affects the compressive strength of the product.

Table 1: Leach Rates of Different Nuclides from Portland Cement Products in Q-Brine Applying the IAEA Test at 20°C [2]

Nuclide	$R_L[g \cdot cm^{-2} \cdot d^{-1}]$
Cs	$1 \cdot 10^{-2} - 1 \cdot 10^{-3}$
Sr	$5 \cdot 10^{-3} - 5 \cdot 10^{-4}$
Co	$2 \cdot 10^{-4} - 2 \cdot 10^{-5}$
Ru	$1 \cdot 10^{-4} - 2 \cdot 10^{-5}$
Pu	$1 \cdot 10^{-5} - 1 \cdot 10^{-6}$

Q-Brine Composition: 63% H_2O, 34% $MgCl_2$, 2% $MgSO_4$, 0.6% KCl, 0.2% NaCl.

The concept of leach rates quantifies the rate a which water or brine attacks the matrix of the cemen stone matrix. The leach rate (R_L), which is a function o time, is defined as

$$R_L = \frac{m_1 \cdot w}{m_t \cdot S \cdot t} \; [g \cdot cm^{-2} \cdot d^{-1}] \qquad (1$$

where

m_1 = amount of a specified nuclide leached during the test duration;

m_t = amount of the specific nuclide initially present in the test specimen;

S = geometrical surface area of test specimen for cement produc

w = weight of test specimen;

t = duration of test.

Selective leaching experiments yield element-specifi leach rates. The incremental leach rate decreases rap idly with $1/\sqrt{t}$ at the beginning. This phase may b described by a diffusion-controlled mechanism. After period of a few hundred days the leach rate become constant. Typical leach rate values for Q-brine ar listed in Table 1.

Radiation Stability

Cementitious waste products contain a fairly hig amount (20–30%) of water, which is either distribute in pores or bound in the various hydrated phases. Ga production by radiolysis, and thus pressure build-u may therefore become an important safety conside ation. The problem could even be enhanced by th decomposition of nitrates present in most of the soli ified materials.

Although some experimental results for hydroge oxygen, and nitrous oxide formation are available [4], the necessary information is still insufficient. Mo data are required to evaluate the consequences arisi from radiolytic interactions. It seems, however, that harmful nitrous oxides, especially N_2O, are formed even nominal quantities. Mostly hydrogen is produce and also, in the presence of nitrate, smaller amounts

oxygen. Nitrate salts reduce the hydrogen production by factors of 3 to 10, depending on the nitrate percentage in the product due to secondary reactions consuming hydrogen atoms. The radiolysis gas is easily released from the cementitious waste product. Therefore, no serious structural damage (e.g., by crack formation, etc.) should take place as a consequence of internal gas pressure build-up.

Typical values for gas formation (10–30% $NaNO_3$ content) are

H_2:0.3 – 0.8 ml/kg·10^6 rad; and

O_2:0.1 – 0.2 ml/kg·10^6 rad.

As one would expect from theoretical considerations, alpha radiolysis is more effective, by a factor of 2 to 3, than gamma radiolysis.

Mechanical Stability

Mechanical stability is primarily concerned with compressive strength of the cementitious products. The surface area of the waste form is also an important parameter in minimizing the leach rate. Measurement of compressive strength can provide important additional information about the quality of the immobilized waste conditioned for disposal. A minimum compressive strength of 2.5 N / mm^2 (= 25 bar) is recommended in Germany. The values obtained are dependent on several factors, such as cement formula, salt content, and composition, additives, and water-to-cement ratio. The dependence of the compressive strength on the water / cement ratio can be deduced from the values in Table 2.

Increasing nitrate contents reduce the compressive strength of the cementitious waste forms. However, these products can tolerate quite high nitrate loadings, as can be deduced from Figure 1.

A drastic reduction of the bending strength of the cementitious waste products occurs if they are exposed to brine, particularly Q-brine. The effect is already observed after several days to a few weeks. Although no alteration is visible on the face of the product body, bending strength may have decreased below the limiting value of about 2 N / m^2. At the fractured edges liquid drips out.

The effects of addition of detergents on hydration

Table 2: Compressive Strength of Immobilized Medium-Active Waste Concentrate as a Function of the Water-to-Cement Ratio (Portland Cement, PZ 55), Curing Time 20–28 Days, Dry Salt Loading ≃ 25%

Water / cement ratio	*Compressive strength N / mm^2*
0.35	14.5
0.40	10.8
0.45	7.2
0.50	5.4

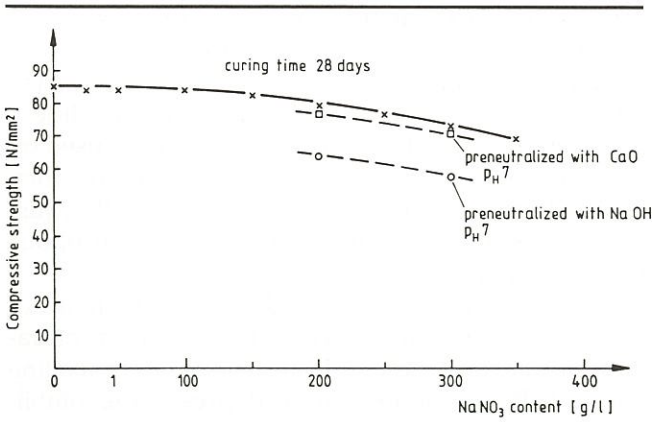

Figure 1: Dependence of compressive strength of MAW cement products on sodium nitrate content. Standard samples prepared according to DIN 1164, PZ 55; water / cement ratio = 0.45.

capability have been assessed, since waste waters to be treated very often contain such chemicals in considerable quantities. Only a minor influence of C_3S degradation has been observed, but a remarkable retardation occurs in the case of C_3A. Because the leach rates are related by the degree of hydration of the cement, longer curing times are required. However, the longer-term leach rates remain relatively constant, supposing the porosity is not changed considerably by the action of the detergents.

The results obtained may be summarized as follows:

1. The compressive strength decreases with increasing water-to-cement ratio.
2. Also, the compressive strength decreases with increasing salt loading of the product.
3. The leach rates increase slightly with decreasing compressive strength.
4. Increasing porosity of the cementitious waste form has a strong effect on its compressive strength; increase of porosity from 3 to 10% reduces the compressive strength to approximately 50% of its original value.

An effective way to improve products is by polymer impregnation [5]. It not only increases mechanical stability by approximately two to three orders of magnitude, but, still more advantageously, improves leaching resistancy. Leach rates are improved by factors of 10^2–10^3, at least in the short term.

The procedure consists of mixing the waste with cement and allowing it to solidify at ambient temperature. Afterwards, a catalyzed organic monomer, like styrene, is poured over the surface of the concrete block and allowed to soak in. The pores are thus filled up with the plastic, yielding a low-porosity product. In the course of heating at low temperatures (50–70°C), a polymer-impregnated concrete body is formed. In addition, the polymer forms a thin surface over the top of the material.

Morphology of Cementitious Waste Products

A question of utmost importance with regard to product quality is: What chemical compounds are formed in cement products containing the radioactive nuclides? In the present investigation, attention was focused on the radio-elements strontium and cesium. Are they built in to new stable mineral phases, or are they just more or less uniformly distributed in the amorphous gelatinous pore spaces?

It is known that the pore fluid in a set cement is not pure water, but a solution containing high concentrations of alkaline compounds [6]. The strongly alkaline nature of the micropore solution depresses the solubility of calcium and strontium but does not markedly affect the solubility of cesium, which remains high. A large number of fission product elements (e.g., rare earths, zirconium, actinides, as well as iron and cobalt) form insoluble oxides or hydrous oxides, which remain comparatively insoluble in an alkaline environment. The measured leach rates (see Table 1) are in agreement with these facts.

Cement matrices characteristically exhibit poor retentivity for species that are soluble in alkaline environments. A typical example of this kind is the alkaline element cesium, whereas strontium shows a distinctly stronger retention. Despite their high specific surface area, the hydration products of cement have virtually no sorptive capacity for cesium, and only a poor one for strontium.

In order to learn more about the underlying mechanisms responsible for the fixation of the radio-elements, phase characterization studies have been performed. The influence of the medium-active waste composition (e.g., salt load, complexing agents, etc.) was investigated. Important parameters with regard to cement hydration and reactivity are the degradation rates of the phases:

Ca_3Al (tricalcium aluminate = responsible for settling of the cementitious paste), and
Ca_3Si (tricalcium silicate = responsible for the product strength).

They have been measured by applying X-ray diffraction analysis. Some of the results are shown in Figures 2 and 3.

It is obvious that CaO is a much more suitable reagent for the degradation / hydration of Ca_3Si and Ca_3Al. Since sodium acts as a cement poison, which becomes evident by the reduced hydration ability, its addition should be minimized or, even better, totally omitted. A considerable increase of Cs and Sr leachability is the logical consequence. It appears that the immobilization of the radionuclides inversely correlates with the calcium hydroxide present in the hydrated cement matrix.

Main interest was focused on the question of whether cesium and strontium show mineralization, and thus improve their leachability resistance by a molecular-

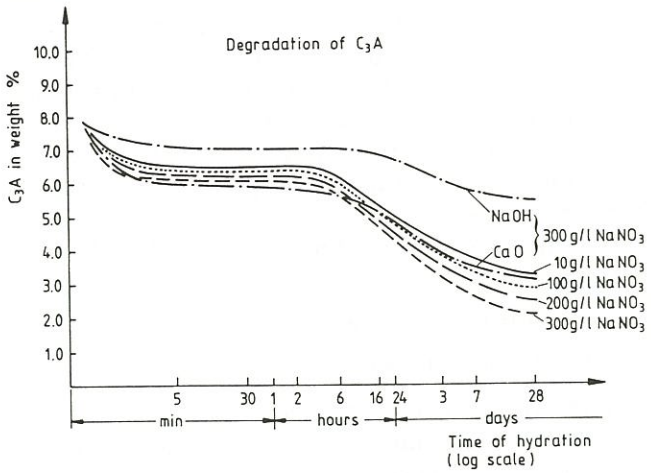

Figure 2: Influence of sodium nitrate content in MAW on hydration of cement; degradation of C_3A phases. Influence of pre-neutralization, either with NaOH or CaO.

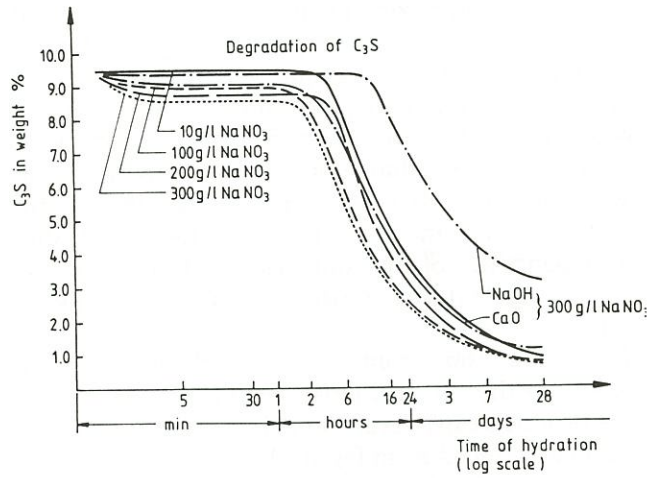

Figure 3: Influence of sodium nitrate on cement (e.g., degradation of C_3S phases. Pre-neutralization, either with NaOH or CaO.

disperse fixation. The microstructure investigations were carried out with a scanning electron microscope (SEM), coupled with X-ray disperse spectrometry (EDS). In addition, transmission electron microscopy (TEM), coupled with EDS and electron microprobe analysis (EMP), were employed.

Neither method succeeded in identifying a mineralized Cs or Sr phase. All Cs and Sr are retained in the gelatinous pore spaces. Chemical analyses of the pore water, and in addition X-ray diffraction investigations revealed that strontium is solely present in the form of $SrSO_4$. For cesium, no separate Cs-phases were traceable. Pore water analyses indicated that this element only subsides as $CsNO_3$.

Figure 4 shows a SEM picture of the pore space matrix. The images reveal the interior of the monolith

Figure 4: Inner pore space of a cementitious product; w/c ratio 0.34; $CsNO_3$ 50 g/1.

Figure 5: Globular $CsNO_3$ precipitation in the pore space matrix.

form, but no new crystalline phases of Sr or Cs. The bright parts indicate a Cs enrichment. A non-uniform Cs enrichment is typical if either a K or Na enrichment is also observed. It is well known that these two elements can substitute Ca in the pore spaces, as can Cs. In a few cases, a globular Cs enrichment could be detected, as is shown in Figure 5. It indicates a $CsNO_3$ precipitation. The existence of the Cs enrichment was proved by applying X-ray backscattering (Figure 6).

Conclusions

It turns out that cements and concrete can serve merely as physical barriers for cesium and strontium. No new crystalline phases, besides the plain carbonate, nitrate, and sulphate compounds, could be detected.

It could be shown that the improvement of cesium and strontium leach rates is mostly due to sorption

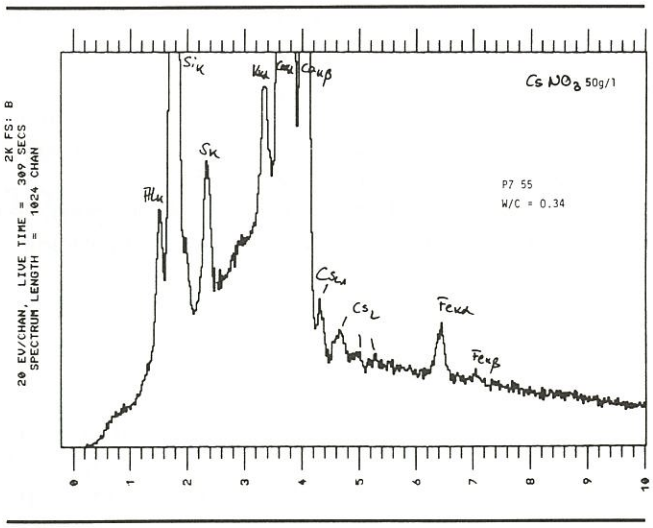

Figure 6: X-ray backscattering diagram of pore space matrix; $L_{\alpha 1}$ line at 4.3 kV.

phenomena on selected additives, but, more important, on the degree of porosity of the cementitious products. In fact, the leachabilities of the nuclides are essentially diffusion-controlled.

Improvement of cesium and strontium retention may be achieved in two ways:

1. Increase of the sorption characteristics of the matrix by incorporating substances that possess significant
   - sorption surface-active material (e.g., from blast furnace slag, natural pozzolan, active SiO_2, coal combustion fly ashes);
   - structure-active material capable of forming mineral phases. However, so far no really effective substance has been found.
2. Alteration of the physical properties of the cementitious waste form by the addition of cement modifiers, or utilization of alumina-rich cements. For example, the addition of high surface SiO_2 leads to extensive changes in the microstructure. The most important matrix parameters with respect to leaching are total porosity, pore size and distribution, and pore interconnectivity.

An improvement in the pore size distribution and a decrease of open porosity is best achieved by the employment of modified cement formulations with a high alumina content. Increasing the sorptive capacity by adding surface-active material like bentonite can be counterproductive in that it causes higher porosity.

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Spent Fuel Storage and Transportation Experience for the Idaho National Engineering Laboratory

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Abstract
Spent-fuel research and development demonstrations and associated transportation activities are being performed for the Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) as a part of the storage cask performance testing programs at the Idaho National Engineering Laboratory (INEL). These spent-fuel programs support the Nuclear Waste Policy Act (NWPA) and DOE objectives for cooperative demonstrations with the utilities, testing at federal sites, and alternatives for viable transportation systems. The shipment of spent fuel to the INEL from the Surry Power Station and the Nevada Test Site (NTS) required shipping plans and co-ordination between DOE, EG&G Idaho, and Virginia Power (VP) transportation personnel, as well as extensive communication with the corridor states.

Résumé
Les démonstrations development des récherches des combustibles épuisant et activities des transportation associé exéuter pour le Department d'Energie, le Bureau de l'Usure Radioactif Direction [DOE Office of Radioactive Waste Management (OCRWM)] au parti des programmes du tests regardant le fonctionnement des caisses d'emmagasinage au Laboratoire Nationale d'Idaho [Idaho National Engineering Laboratory (INEL)]. Ces programmes s'appuyent l'Acte pour les Politiques des Usures Nucléaire [Nuclear Waste Policy Act (NWPA)] et les objectifs DOE pour les démonstrations cooperatifs avec les utilités, les tests aux facilités fédérales, et les alternatives pour les systèmes du transport démontré. Le transport des combustibles épuisant au INEL, de Surry Power Station, et le Facilité des Testes au Nevada [Nevada Test Site (NTS)] se réquirent les plans du transport et la coordination entre DOE, EG&G Idaho, et Virginia Power (VP) et aussi les rapports extensifs avec les états affectés.

Program Information

Introduction

The spent-fuel programs being performed and planned at the INEL for DOE are composed of

- spent-fuel storage casks performance testing that involves performance of fuel storage casks with intact or consolidated fuel;
- fuel assembly rod consolidation projects, to develop dry-rod consolidation technology and prototypical equipment, as well as perform testing of storage casks containing consolidated fuel rods in canisters;
- the Nuclear Fuel Services, Inc. (NFS) spent-fuel transportable storage casks project, which involves the licensing and shipping of two loaded, transportable storage casks from West Valley, New York to the INEL;
- the cask systems acquisition (CSA) project to develop a complement of Nuclear Regulatory Commission (NRC)-certified prototype casks for shipment of spent fuel from reactor facilities to future repository sites.

The first two programs, cask performance testing and dry consolidation of spent fuel rods, provide engineering data from loaded-cask testing containing both intact and consolidated fuel. Dry consolidation of spent fuel rods will occur during the DOE-sponsored small-scale rod consolidation activities, and on a production-oriented scale during the prototypical rod consolidation project. The NFS transportable storage cask project will provide additional cask performance and surveillance data at the INEL. The CSA project will involve procurement of both transportation casks as well as transportable storage casks for DOE.

The Test Area North (TAN) facility located at the INEL was determined to be the appropriate federal facility in which to conduct these activities because of the availability of experienced staff, hot and cold test development areas, and the support needed to receive and store commercial spent-fuel assemblies.

These dry storage cask demonstrations support OCRWM and NWPA objectives and will establish a data base that

Keywords: nuclear, spent fuel, spent fuel transportation, spent fuel storage, cask performance testing, storage, consolidation, shipping.

can be used for NRC licensing by generic rule of at-reactor dry storage cask installations.

The transportation experience includes the handling of large empty metal spent-fuel storage casks, received by rail, and the shipment of a large number of spent fuel assemblies in NRC-certified shipping casks, transported over highways in transport vehicles having state-issued special permits, and meeting applicable Department of Transportation (DOT) requirements, as stated in 49CFR (Code of Federal Regulations 49, revised annually) and in accordance with specific DOE policies (DOE Order ID-1540.1, 1982).

Spent-Fuel Storage Cask Performance Testing

The VP / DOE cask testing project is a co-operative effort involving Virginia Power, DOE, and the Electrical Power Research Institute (EPRI). The co-operative agreement was established in 1984. The testing at the INEL will include casks containing both consolidated and intact fuel. The spent-fuel assemblies are shipped from the Virginia Power Surry Power Station to the INEL, using conventional licensed shipping methods.

The purpose of VP / DOE storage cask performance testing at the INEL is to remove conservatism from the licensing of dry storage casks and provide a storage technology that is generically applicable, so that the NRC can license dry storage of spent fuel by rule. Decay heat generation rates for selected fuel assemblies will total near the cask design limits. Thus, the casks may be tested above the NRC licensing limit. Data from INEL testing will confirm storage cask performance, predictive modelling capabilities, and fuel integrity at prototypical storage conditions; and will provide operational data to support the life-cycle cost studies of dry fuel storage.

The cask performance testing technical objectives are developed by Pacific Northwest Laboratories (PNL), reviewed by the co-operative agreement participants, and implemented by EG&G Idaho at the INEL.

INEL site preparations, fuel receipt, and the start of testing occurred in 1985. Storage cask testing with intact fuel was concluded in 1986, and cask testing with consolidated fuel will be completed in 1987 / 1988. Completion of testing reports and termination of the project is scheduled for mid-1988.

Fuel Assembly Rod Consolidation Projects

The rod consolidation projects consist of DOE dry rod consolidation technology (DRCT) and the prototypical consolidation demonstration project (PCDP). The small scale DRCT project will consolidate 48 fuel assemblies into 24 canisters with a consolidation ratio goal of 2:1. Furthermore, consolidation system characteristics and technical data (such as rod pulling forces, rod diameter measurements, and crud collection) will be obtained that can be utilized by the PCDP project and future DOE rod-consolidation facilities or programs.

The PCDP will demonstrate production-scale spent fuel rod consolidation in a dry environment at th Idaho National Engineering Laboratory (INEL) Tes Area North (TAN) facility. The consolidation equip ment developed during this project will provide th design basis for future equipment to be used at high level waste repositories, or the Monitored Retrievabl Storage (MRS) facility. This project will expedite th engineering development and demonstration of protc typical dry rod consolidation and associated handlin equipment. It will be developed under a competitiv design effort by the private sector and tested at th INEL using spent-fuel assemblies acquired for the dem onstration.

To obtain private-sector involvement, a four-phase single request for proposal (RFP) has been developed covering the phases of this project: a) preliminar design competition; b) detailed (final) design compet tion; c) equipment fabrication, installation, and col checkout; and d) hot demonstration and qualificatio of equipment at the TAN facility. The hot demonstra tion will be performed at the INEL using approximatel 100 PWR and 100 BWR spent-fuel assemblies typical c the light water reactor industry. The competitive desig with private sector demonstration is to be complete by mid-1989. The selection and shipment of spent-fu assemblies, and procurement and receipt of the cask that will later store the consolidated canisters, a being evaluated.

Spent-Fuel Transportable Storage Cask Project

The Nuclear Fuel Services, Inc. (NFS) spent-fuel tran portable storage cask project will demonstrate th feasibility of packaging, transporting, and storin aged spent fuel in two large dry storage casks, desi nated the Transnuclear, Inc. Big Rock Point (TN-BR and the Transnuclear, Inc. R.E. Ginna (TN-REG) cask The project was initiated in early 1984, when DC contracted with NFS for removal of spent fuel from th West Valley pool. NFS in turn contracted with Tran nuclear, Inc. for two large transportable storage cask one to hold 85 BWR assemblies stacked in two layer and one to hold 40 BWR assemblies. Applications f both casks are presently being processed at the NRC b Transnuclear, Inc., the manufacturer of the cask West Valley Nuclear Services (WVNS) and the WV Der onstration Project Office (WVDPO) will prepare th West Valley facility and supervise the fuel loadi operations in the WV pool gfor this project.

The project will provide data for railroad transport tion of loaded, spent-fuel dry storage casks, utilizi NRC licensing for the one-time shipment of each cas Demonstration of transportation and dry storage spent nuclear fuel will involve the shipment of tv casks loaded with fuel from West Valley, New York, the INEL for cask testing and monitoring under stora conditions. After arrival at the INEL, the TN-BRP a

Table 1: Information Summary for TN-BRP and TN-REG Casks

Features	*TN-REG*	*TN-BRP*
Fuel assembly capacity	40 PWR	85 BWR
Material	Forged steel	Forged steel
Nominal weight loaded (kilograms)	90,718 kg (100 tons)	90,718 kg (100 tons)
Nominal length (metres)	5.03 m (16.5 ft)	5.03 m (16.5 ft)
Nominal diameter (metres)	2.59 m (8.5 ft)	2.59 m (8.5 ft)
Maximum dose rate at 2 m (Sv / h)	1.0 E-4 (10 mR / h)	1.0 E-4 (10 mR / h)
Heat load (kW) (total)	Less than 5	Less than 5
Cover gas	Nitrogen	Nitrogen

TN-REG casks will be placed directly on the TAN storage cask test pad for long-term monitoring and surveillance. A summary of cask information is provided in Table 1.

Cask System Acquisition

The CSA project will review and evaluate proposals from private industry and eventually place several contracts for transportation casks as well as for at least two transportable storage casks. The casks may include innovative designs and materials. The CSA project will not be discussed further in this paper.

Project Status

The progress of the first three program elements (VP / DOE cask testing, PCDP, and NFS project) are discussed in reverse order in this section, because most of the activity to date has been with the cask testing project.

Progress in the NFS project included completion of manufacture of the TN-BRP and TN-REG casks in 1985, and delivery of the casks to West Valley, NY. Processing of the applications for NRC licensing of the casks for one-time shipments is also in progress. Current plans are to load the fuel into the casks at West Valley while the licensing activities are in progress, and make the shipments at a later date when the casks are approved.

The RFP for the PCDP was issued January 13, 1986. Appropriate evaluation criteria have been established, and completion of the private-sector competitive preliminary designs is planned by late 1986. This will be followed by a review of the proposals and selection of the competitors for the final design.

VP / DOE spent-fuel storage cask testing and demonstrations are progressing on schedule. The INEL TAN facilities were modified to accommodate the remote transfer of spent fuel in the TAN 607 Hot Shop, from the shipping casks to the storage casks, and to permit cask testing in the TAN 607 Warm Shop. In addition to the completion of all preparations for receiving and handling the casks and spent fuel, a concrete pad was constructed near the Hot Shop for temporary storage of loaded fuel casks. Storage will be in an open environment, with appropriate monitoring and surveillance.

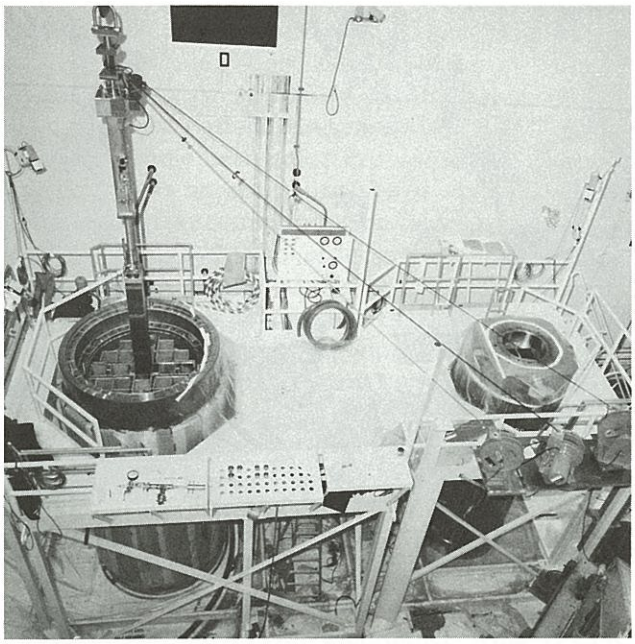

Figure 1: Hot Shop dual-cask work-stand for remote in-air transfer of fuel from shipping casks to storage casks (Spring 1986).

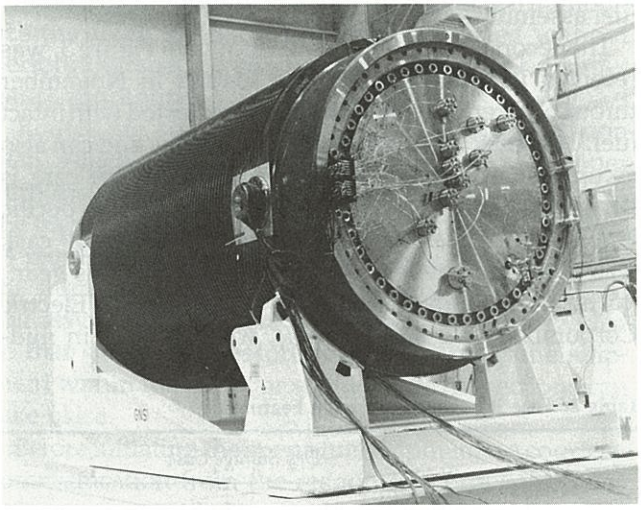

Figure 2: Horizontal testing of the first storage cask (GNS Castor V / 21) built by General Nuclear Services.

The first storage cask, GNS Castor V / 21, arrived at the INEL by rail in December 1984, was moved by heavy haul transporter from the INEL railhead to the TAN facility in February 1985, and loaded with spent fuel in July and August 1985. The dual-cask work-stand in the TAN 607 facility Hot Shop is shown in Figure 1. The first storage cask is shown (Figure 2) during testing in the horizontal position in the TAN Warm Shop. Short-

Figure 3: The second storage cask (TN-24P) built by Transnuclear, Inc.

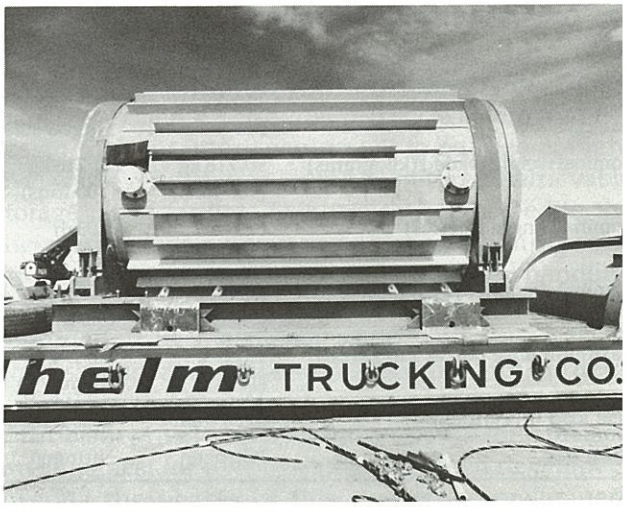

Figure 4: The third storage cask (MC-10) built by Westinghou Electric Corporation.

term monitoring and testing of this cask was completed in September 1985.

The General Nuclear Services, GNS Castor V/21, cask is designed for a heat load of 21 kW and will hold 21 fuel assemblies. The cask can accommodate intact fuel assemblies or consolidated rod canisters.

The second cask, the TN-24P, shown in Figure 3, was received in October 1985 and loaded in November through December 1985, and similarly tested with intact fuel. This cask will hold 24 fuel assemblies and can also accommodate consolidated rod canisters. The cask is designed to dissipate a 24 kW heat load under specific ambient and solar input conditions. The TN-24P cask was placed on the test pad in the spring of this year.

The third storage cask, the Westinghouse Electric Corporation MC-10 (Figure 4), was received in mid-March 1986. This cask is designed for a heat load 24 kW and will hold 24 fuel assemblies or consolidat rod canisters. Completion of loading and testing of th cask was accomplished in June 1986.

The fourth storage cask has not yet been designat for the VP/DOE co-operative agreement. Procureme of the fourth cask is in progress for acquisition by la 1987. Dry storage cask features for the GNS Cast V/21, Transnuclear, Inc. TN-24P, and Westinghou MC-10 storage casks are shown in Table 2.

Cask Handling Experience

Handling of the approximately 100-ton storage cask as well as numerous spent-fuel shipments, provid valuable cask-handling experience at the INEL.

The precise alignment of shipping/storage cask li

Table 2: Dry-Storage Cask Design Features

	Dry Storage Cask		
Design features	*GNS Castor V/21*	*TN-24P*	*Westinghouse MC-10*
1. Maximum weight on crane hook	104,326 kg (115 tons)	90,718 kg (100 tons)	90,718 kg (100 tons)
2. Capacity	21 PWR	24 PWR	24 PWR
3. Proposed licensed heat generation capacity	21 kW	24 kW	24 kW
4. Overall length	4.88 m (16 ft)	5.03 m (16.5 ft)	4.88 m (16 ft)
5. Outside diameter	2.44 m (8 ft)	2.44 m (8 ft)	2.44 m (8 ft)
6. Materials of construction	Nodular cast iron	Forged steel	Forged steel
7. Neutron shielding	Polyethylene	Borated plastic	Borated plastic

and fixtures must be evaluated, and alignment and handling of the lid systems verified prior to in-air remote fuel transfer operations. Likewise, the seal surface protectors that protect the cask sealing surfaces from damage must be checked out prior to initiating remote fuel-transfer activities in the Hot Shop.

High-quality metal or elastomer gaskets are essential for repetitive use where incremental fuel-loading or testing activities cause the lid to be removed and replaced several times. The metallic gasket on the GNS Castor V/21 primary lid was subjected to at least twelve such use cycles, and continues to function within the specified leak-rate limits.

The cover gas system used to evacuate, backfill, monitor, and obtain gas samples should be carefully designed, fabricated, and tested. High-quality fittings, quick disconnects, and valves should be utilized. A quick disconnect should not be utilized as the final barrier for air ingress, but should be backed by a block valve. The difficulty associated with backfilling the cask with a pure (>99%) cover gas, and obtaining gas samples without introducing air should not be underestimated. Procedures should specify that the cask be pumped down and backfilled at least twice, to ensure purity (>99%) of the final cover gas.

The information required prior to handling a cask should include design drawings and specifications, operating and maintenance manuals, procedures, and spare parts. The dry-run operational checkout of the cask and associated equipment should be performed for all phases, including handling, loading and backfilling the cask with a cover gas, and gas sampling. Cask vendor representatives should be in attendance and provide initial equipment training.

The preliminary training and operational checkout support provided by the cask vendors was very beneficial. Eighty-six fuel assemblies were transferred within the Hot Shop and all cask-handling activities were performed without incident.

The spent fuel cask testing and storage demonstrations at the INEL will enhance the overall dry-storage technology data base. The initial results indicate that the shielding and thermal performance are good. Testing with the casks loaded with consolidated rod canisters will further support the verification of the performance modelling.

Transportation Experience

The shipment of DOE-owned spent fuel is a well-structured activity, and is performed in accordance with certain DOE and DOT policies. Specific guidelines are established for the carrier, the originating facility, and the terminal point.

In addition to invoking all applicable DOT requirements, the shipping costs and schedules were evaluated for both DOE commercial spent-fuel shipments and the NRC shipping requirements. Applicable shipping forms are specified by the DOT per 49CFR. Along with the 69 fuel assemblies moved in 23 trips from Surry to the INEL, there were 17 fuel assemblies moved from the NTS to the INEL. An additional two assemblies will be moved from the Battelle, Columbus facility to the INEL. For these shipments, a policy to provide notification to state governors or representatives in advance of shipments (a courtesy communications system) was developed and implemented in order to keep the participating states fully informed.

Communications equipment is a vital part of the transportation planning. The communications network requires a central control point and the capability to contact local emergency response organizations. Call-ins by the carrier at specified intervals, normally every two hours, assures that key personnel are cognizant of the location and status of each shipment and can initiate an immediate response to any unusual situation.

The drivers are DOT-licensed and trained in the transporting of highway-route-controlled quantities of radioactive material in accordance with 49 CFR. Escort drivers or personnel are provided as required.

Constant surveillance of each shipment is provided by one of the drivers or escort personnel. The potential for acts of terrorism exists, and a vehicle immobilization system is required whereby the driver can immobilize the truck so that it cannot be restarted without performing specific mechanical or electrical maintenance procedures.

Routes are specified in the written transportation plan, which is reviewed and approved by the proper authorities. Following route approval and issuance of the transportation plan, the state overweight permits, as required, are obtained, and travel restrictions for the shipments provided to the drivers. In some cases, courtesy communications for each shipment is provided so that state patrol escorts can accompany the shipment within the state, or so that state inspections can take place.

Before initiating the spent-fuel shipments, a courtesy communication from the responsible DOE field office manager is provided to the designated representative for each affected state. The notification is provided in keeping with the spirit of consultation and co-operation described in the Nuclear Waste Policy Act of 1982. The approximate dates, duration, and number of shipments are provided. Also, the preferred routes are specified, but may be renegotiated by state officials to be in agreement with local preferences.

Major procurement activities were initiated by Virginia Power at the start of the DOE / VP / EPRI co-operative agreement, in order to supply the storage casks for the program. Three storage casks were procured from different vendors and transported empty to the DOE at the INEL. Also, it was necessary to lease shipping casks for the proposed spent-fuel shipments. Two NRC-licensed spent-fuel shipping casks, as specified in Transnuclear

(TN) TN-8L Certificate of Compliance USA/9015B( )F were utilized to transport the spent fuel. Each cask, with a capacity of three PWR fuel assemblies, was hauled on TN-supplied trailers.

The spent-fuel shipping casks, transported by highway from both loading sites to the INEL, utilized diesel units supplied by an Interstate Commerce Commission (ICC)-licensed shipping contractor, Tri-State Motor Transit Company. Overweight permits for the 50,802.35 kg (112,000 pound) combined units were required. The shipments followed the state-preferred highway routes as specified in the transportation plan.

Cask loading and transporter activities at both the Surry Power Station (SPS) and Nevada Test Site (NTS) facility were performed in accordance with applicable transportation and safety regulations. The documentation for each shipment was prepared, reviewed, and approved in accordance with 49CFR. Prior to each shipment leaving a loading site, a DOE-ID/EG&G Idaho, Inc. representative reviewed and signed the paperwork, accepting title to the fuel and shipment responsibility.

Notifications for each shipment were made by the DOE-ID/EG&G Idaho, Inc. traffic representative to the INEL receiving facility and the DOE Transportation Manager in Washington, D.C.

Meetings with state officials in the originating states were conducted to apprise them of intended shipments and key facts associated with the shipments. Periodic review meetings were also held with the carrier, shipping cask representatives, and DOE/EG&G Idaho/VP traffic personnel to discuss overall transport equipment readiness, procedures for driver activities, tractor/trailer permits, backup equipment, route conditions, and handling of shipping papers/permits.

For shipments from the SPS to the INEL, estimates of average round-trip times ranged from 12 to 14 days, depending upon the travel route used. Actual experience was 17 days for campaign #1, 13 days for campaign #2, and 11 days for campaign #3.

The reduction in round-trip times resulted from optimization of the travel route; changing to back-to-back shipments, so that crews and shipping casks were available for continuous loading and unloadin to provide a more cost-effective and efficient ope tion; and reduced handling times at SPS and the IN as workers became more familiar with their tasks. SPS, in particular, cask loading times were reduc from 30 hours to 15 hours. The trip from SPS to the IN averaged three days during campaign #3.

A substantial effort was made to ensure tractor a trailer readiness prior to each shipment from the SPS. local contractor was retained to provide inspectio maintenance, and repairs, if necessary. SPS qual control personnel performed incoming and outgoi inspections, and Virginia State Police hazardous ma rial personnel performed outgoing inspections. Simi arrangements were established for the shipments fr the NTS.

Comprehensive planning, attention to details, co munications networks, and compliance with establi ing requirements resulted in a relatively trouble-f series of shipping campaigns, with 86 spent-fuel asse blies received at the INEL.

Conclusion

The cask performance and testing demonstrations, alo with the long-distance transportation of a large numb of spent fuel assemblies are considered a success sto The evaluation and implementation of applicable quirements, industry perspective, and extensive pla ning all contributed to this achievement.

Acknowledgements

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Summaries of CNS Conferences, Seminars, and Workshops

Second International Conference on Radioactive Waste Management

The Second International Conference on Radioactive Waste Management was sponsored by the Canadian Nuclear Society and cosponsored by ANS.

The conference was held in Winnipeg, 7–11 September 1986, and attracted an attendance of nearly 300; about 180 from Canada, 60 from the U.S., and 50 from other countries.

During the three-day conference, 128 technical papers were presented in twenty-four technical sessions covering topics of transportation, storage and disposal of radioactive wastes, geoscience research, waste forms characterization, environmental assessment, facility design, uranium tailings management, regulatory affairs, and public attitudes. Many countries have made considerable progress in resolving scientific, technological, and socio-political issues since the first CNS-sponsored international conference on radioactive waste management held in Winnipeg in September 1982.

The theme for this year's conference was 'Demonstrated Results Providing Direction for the Future.' A special emphasis was placed on large-scale *in situ* experimental facilities able to demonstrate the concepts of safe management of radioactive waste and to provide confidence in our ability to predict the long-term performance of waste management systems. A technical tour to the Underground Research Laboratory and Whiteshell Nuclear Research Establishment was conducted at the conclusion of the Conference.

The following papers were presented:

'The Hades Demonstration Project for Radwaste Disposal in Deep Clay.' L.H. Baetsle, A. Bonne, P. Dejonghe (SCK/CEN Mol, Belgium)

'Disposal of Low and Intermediate Level Waste in Rock Caverns at Shallow to Intermediate Depth.' N.L. Mansson, M. Bechai, P.K.M. Rao (Ontario Hydro, Toronto, Ontario)

'Design and Performance Assessment of an Above Grade Disposal Structure.' R.R. Blickwedehl (Dames & Moore, West Valley, New York)

'Inground Containers at BNPD.' P.J. Armstrong (Ontario Hydro, Toronto,Ontario)

'Geological Investigation for High Level Radioactive Waste Isolation in Japan.' K. Doi, M. Yamakawa (Power Reactor & Nuclear Fuel Development Corporation, Tokyo, Japan)

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