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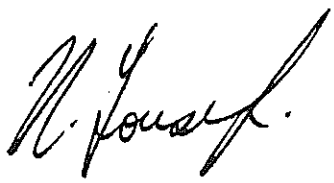
Message from the President

The Publishing of the Nuclear Journal of Canada is a major, yet a logical step in the progress of the Canadian Nuclear Society. After achieving world class status in all facets of nuclear related activities including the superior performance of the CANDU System, the Canadian Nuclear Society decided to publish a technical journal. The Nuclear Journal of Canada is yet another forum for sharing the research results and new applications of nuclear-related science, with your fellow men who are working in the nuclear field.

In the spirit of open communication, the successes and failures need to be analysed and told. This is your forum – your chance to participate.

Responsibility and technical excellence are necessary bed fellows in the nuclear field.

This is one major opportunity to demonstrate our strengths.



N. Yousef
President
Canadian Nuclear Society

Editorial

Alan Wyatt
Editor

The Canadian Nuclear Society is proud to launch the *Nuclear Journal of Canada*. This is a major undertaking and the culmination of several years of effort. The *Nuclear Journal of Canada* is a quarterly publication of international standard. It is being produced for the Canadian Nuclear Society by the University of Toronto Press, a leading publisher of scholarly journals.

The primary objective of the *Nuclear Journal of Canada* is to present papers on all aspects of nuclear science, engineering, and technology. Although many papers will naturally have a Canadian context, relevant papers from non-Canadian authors about overseas developments are encouraged. Papers published in the *Journal* will be grouped according to the eight areas of interest as defined by members of the Canadian Nuclear Society, namely:

1 nuclear science
2 design and materials
3 mining and manufacturing
4 environment
5 health and public affairs
6 waste management
7 nuclear plant operations
8 fusion

The language policy of the *Nuclear Journal of Canada* is to publish articles in either of Canada's official languages, as submitted by the author. Abstracts will normally be published in both English and French.

Many scientific and professional journals deal with a relatively narrow area of specialization. The general background to all the papers in those journals is understood by the great majority of those who receive them. In view of the breadth of coverage for the *Nuclear Journal of Canada*, a number of overview, or review papers is appropriate for each issue, averaging about 20 per cent of the papers published. In this issue the opening paper gives the historical background to the formation of the Canadian Nuclear Society.

After publication in the *Journal*, discussion of the published papers will be actively encouraged. Closing date for receipt of letters (maximum length 1000 words) is approximately six weeks after the relevant number's date of issue. For this first issue the closing date for communications concerning the papers is 5 May 1987. All such communications are to be addressed, in duplicate, to the editor, who will forward them to the author for his/her response. The letters and author's response will normally be published in the second issue following publication of the article; letters on this first issue will be published in the third. The editor reserves the right to edit letters and responses to fit available space.

Manuscripts of papers to be considered for publication in the *Nuclear Journal of Canada* should be sent to the editor at the address shown on page i. Instructions for authors are on pages 92 and 93 and, for the first few issues, will be included (in alternate issues) in English and French. No page charges are levied on authors. At time of acceptance of papers for publication, authors are advised of the standard optional charges for offprints and reprints, which are produced to the identical quality standard of the *Journal* itself.

Since its inception, the Canadian Nuclear Society has itself sponsored, or co-sponsored with sister societies, a number of conferences and seminars covering a wide range of topics. A summary of these meetings, a list of papers presented and their authors, and where copies of the proceedings may be obtained, will be published in the *Journal*. The proceedings of some conferences will be published as a supplement to the *Nuclear Journal of Canada*. Subscribers interested in acquiring these supplements will be advised of their cost and provided with an order form before the date of issue.

A key factor in establishing and maintaining the high standards of any journal is the review process for papers submitted. Peer-review, prior to publication, is essential,and all papers considered for publication, whether submitted individually or presented at Canadian Nuclear Society conferences, must follow the review procedures of the *Journal*. The members of the editorial board are the vital link in the review process,

acting in both the roles of selector of reviewers and reviewer themselves. In order to provide for uninhibited review, reviewers are selected from organizations other than those of the authors, and are guaranteed anonymity.

The *Nuclear Journal of Canada* will carry advertisements for technical and scientific books, conference notices, complementary journals, educational courses, employment opportunities, personnel availability, and educational materials. Advertisements from commercial organizations will only be accepted where the copy is primarily to inform of new developments, processes, and applications, and not for purely commercial promotional purposes. Advertising copy should be sent to the Journal Manager, University of Toronto Press, 63A St. George Street, Toronto, Ontario M5S 1A6.

The success of the *Nuclear Journal of Canada* depends on a steady flow of high quality original papers. The *Journal* is produced for the benefit of its readers. You can help all of them by stimulating the submission of papers, which will be promptly acknowledged and processed expeditiously. With your support I am confident that the *Nuclear Journal of Canada* will be of great value to the scientific and engineering community, both in Canada and internationally. Any comments or suggestions that any of our readers may have will be welcome.

The Formative Years of the Canadian Nuclear Society 1976–1984

G.R. Howey, P.A. Ross-Ross, and J.S. Hewitt.

Introduction

The Canadian Nuclear Society / Société Nucléaire Canadienne was born as the 'Technical Society of the Canadian Nuclear Association,' on 11 June 1979, at the Annual Business Meeting of the Canadian Nuclear Association. The Canadian Nuclear Society (CNS) was to be a learned society, with membership open to individuals. The CNS was to be distinct from the Canadian Nuclear Association (CNA) whose membership is composed primarily of industries and institutions, and thereby serves as an industrial forum.

In some respects it was an unlikely time for a learned nuclear society to be formed in Canada. The golden age of free exchange of nuclear information, best represented by the Geneva-type conferences, was over. The slowing of the economy on a worldwide basis was creating a bleak outlook for the nuclear power industry, and the media craved sensation and showed little regard for the real pros and cons of nuclear science and technology and their benefits to the public and the economy.

Two basic factors held the door open for the creation of a learned nuclear society in Canada. The first was CANDU. The second was the early existence of the CNA. Had it not been for CANDU and the CNA, the CNS might already have become another chapter of the American Nuclear Society (ANS). Britain had pursued the gas-cooled type of power reactor, Canada evolved CANDU, and the rest of the Western industrial world, following the American lead, pursued the 'pressure vessel' type. The ANS, one of the best-organized learned societies in the world, became the focal point of learned nuclear society activities in the US. The ANS has been a benevolent giant from which many overseas chapters were formed and from which, in many cases, national societies evolved.

The CNA has also enjoyed good relations with this benevolent giant, and a small chapter of the ANS has operated successfully in Canada for many years. However, the presence of the CNA (established in 1960), whose focal point was CANDU and which has played the dual role of industrial forum and technical society, countered any strong interest of the ANS to expand into Canada. But the CNA recognized its limitations in trying to serve both institutional interests and the interests of the individual seeking enrichment of his chosen pursuits, hence there was a small group within the CNA and elsewhere that promoted the formation of a learned nuclear society in Canada.

Given the continued existence of a niche for the CNS, a number of factors conspired in a timely way to finally propel it into existence, and later to nurture and sustain it. The story about to be told is one concerning the perception of a need and the identification and marshalling of forces, not all of which were fully evident at the beginning; a story of perseverance, hard work and enthusiasm, and, above all, patience, understanding, and trust on the part of the many individuals and institutions having diverse as well as common interests in the emerging organization.

The present story deals with the formation and the early years (to 1984) of the Canadian Nuclear Society. By 1984, only preliminary consideration had been given to establishing a most important enterprise, namely, the *Nuclear Journal of Canada*. It is apt that this early history of the CNS appear now in the first issue of the *Journal*. The intervening period, from 1984 to the present, during which the *Nuclear Journal of Canada* became a reality and a symbol of achievement by the CNS as a learned society, is left to future authors.

The Learned Society and the CNS in Perspective

The 'learned society' as an organization serving, through orderly deliberation, the intellectual pursuits of its members, and from thence the welfare of society at large, has survived many centuries. In each instance, the creation of a society depended on the motivation of the practitioners of the subject pursued, and its survival depended largely on whether society at large and the prevailing institutions were hostile or supportive concerning the perceived aims of the organization. Thus, in Galileo's time, the Accademia dei Lincei (1603), the first scientific society on record, was suspended because of 'ecclesiastic opposition,' while the Royal Society received external support through its charter, granted by Charles II in 1662.

The many thousands of scientific / technical socie-

ties, including the CNS, have been modelled on such early examples and have had the dual obligations of the advancement of their scientific or technical subject, on the one hand, and service to a larger society, to which they also owe their existence, on the other. An important background point for consideration, as we review the evolution of the CNS, is that the influence of the support group need not compromise the more altruistic aims of the learned society, namely, to advance the understanding of its subject area through the unfettered exchange of ideas among its members. The interest of the state, for example, in benefitting from the Royal Society may not have impeded its contributions to the progress of the scientific and industrial revolutions. Indeed, symbiotic relationships between science and society have historically augmented scientific and technological developments within the wider community.

As we shall see, the CNS evolved with a great deal of support at its foundation from existing bodies, particularly the CNA and the founders' employers. Receipt of this support did not compromise the freedom of the CNS to attain the state of autonomy appropriate to the good health of a learned society. Moreover, its ability to function in the interests of the broader community has been demonstrated.

The Gathering Support for a Canadian Nuclear Society

The earliest active support for a Canadian learned society devoted to the nuclear sciences and their practical applications came from individuals who would eventually become its members. Such individuals would likely already have been members of the American Nuclear Society, as well as of one or more societies for established disciplines based in Canada, the United States, or Europe. They may also have been active in the work of the CNA. For various reasons, relating to the uniqueness of Canadian nuclear development in energy and medicine and to the perceived benefits for the professional development of the subject area and its practitioners, the question of why there was no Canadian nuclear society was frequently asked, privately, in the course of everyday activity in industry, university, and college.

Such discussion, in the late 1950s, led to the formation, in 1960, of the CNA. [1] Although this organization was established as a corporate-based association of Canadian nuclear industries, with a primary objective of advancing nuclear energy based on the CANDU development, a number of features of its organization and its activities over the years betrayed the breadth of aspiration of its founders. One such activity was organizing the series of technical sessions held, on a fairly regular basis, at the annual CNA conferences in the years immediately before the CNS assumed, in 1980, the responsibility for such sessions. The first such technical programs were organized by A.A. Harms for the 1973 conference in Toronto. W.B. Lewis was the session chairman. Similar events, elaborated to include multiple parallel sessions, were organized in later years, by W. Paskievici (with A. Wyatt), J. Howieson (with L. Bennett), and D. Meneley, for the 'Third Day' at the CNA Annual Conferences in, respectively, Montreal (1977), Ottawa (1978), and Toronto (1979). Over the years since its founding in 1960, the CNA had also sponsored and organized numerous conferences and seminars on special topics, as well as the annual student conferences.

In the meantime, the American Nuclear Society had held its 1976 Summer Meeting in Toronto, jointly with the 1976 CNA International Annual Conference. The ANS had established, many years previously, the ANS Niagara-Finger Lakes Section, catering to ANS members in Southern Ontario, as well as ANS student chapters at the Ecole Polytechnique and the University of Toronto. At the time of the joint ANS / CNA conference in 1976, the possibility of promoting full-scale Canadian sections of the ANS was raised, particularly by the ANS, but Camille Dagenais, CNA President at that time, identified the importance of exploring the possibility of a Canadian-based nuclear society, rather than supporting development of a Canadian arm of a foreign-based organization. [2]

A number of individuals felt strongly about forming a Canadian nuclear society and at various times committed their thoughts to writing. [3] A letter, from G. Howey, Chairman of the CNA Education and Manpower Committee, soliciting the support of his sponsoring director in working towards an 'autonomous body either within the CNA or totally independent of it,' was particularly perceptive of the need for the new organization.

> For several years the Education & Manpower Committee of the CNA has been concerned about the need for a professional scientific / engineering society for the nuclear community in Canada. Such a society would provide an opportunity for the scientist at university or industrial R&D and the engineer associated in some way with the nuclear power program (mining, fuel processing, design, construction, manufacturing, operation or regulation) to share technical ideas, to deliver learned papers and to participate in professional argument and criticism ...
>
> The two former options [of participating in the CNA seminars / conferences or the technical / scientific sessions of CNA annual conferences] tend to be based on a broad range of participant interest and do not lend themselves to very detailed levels of scientific or technical debate. The latter [option of presenting or publishing through numerous existing learned societies in Canada] permits the desired level of detail and depth but does not provide the breadth and overlap needed to keep a subject in perspective. Indeed, dependence on these other Canadian professional associa-

tions tends to dilute the topic and the expertise and does not provide a single Canadian focal point. As a result, serious minded Canadian scientists and engineers who desire true professional recognition of their work in the nuclear power area are obliged, in the absence of a single Canadian agency, to seek some forum outside of Canada such as the American Nuclear Society. [4]

The sponsoring director, R. Langlois, Dean of the Ecole Polytechnique, was later to be appointed Chairman of the CNA Scientific / Technical Task Force, which would eventually make specific recommendations to the CNA Board of Directors on the creation of the new technical society. As it turned out, the enabling legislation was presented at the CNA annual general meeting two and a half years later, in 1979, at which time individuals were invited to become charter members. The course of events during this period is somewhat complex, but bears recounting, to illustrate the convergence of the various interests which propelled the CNS into existence.

The Langlois Task Force

The first formal meeting dedicated to considering the development of a Canadian nuclear society took place in July 1977. The ten-member *ad hoc* committee consisted mainly of members of the CNA Education and Manpower Committee and the CNA Technology Committee. A few weeks earlier, on 8 June 1977, at the CNA Annual Conference, the Third Day Program, under the direction of W. Paskievici and A. Wyatt, had been eminently successful, with 80 papers presented in five parallel sessions, drawing a total of 300 attendees. Soon after the close of the conference, a Technical Sessions Steering Committee was established to assure the continuance of the Third Day Program. The Steering Committee, which formed the nucleus of the subsequent ten-member committee, consisted of G. Howey, R. Harrison, and J. Hewitt of the CNA Education and Manpower Committee, and W. Tarasuk, D. Evans, and D. Meneley of the CNA Technical Committee. Among the additional persons attending the *ad hoc* meeting were A. Harms, an early advocate of the formation of a nuclear society and a former member of the Education and Manpower Committee, and J. Weller of the CNA.

The deliberations of the *ad hoc* committee led to the proposal that the question of a Canadian nuclear society be considered by an on-going task force under the chairmanship of R. Langlois. Members of the Task Force were identical to those of the Steering Committee mentioned above, with the exception that J. Howieson had replaced D. Evans.

At its first meeting in January 1978, the Task Force undertook the preparation of a Prospectus on a 'Professional Association of Individuals Working Within the Canadian Industry' that outlined the needs, objectives, and constraints for such an organization. The Prospectus stressed the need for improved communications at the scientific and technical level through publication, peer-review, public exchange of information, and debate. The Prospectus also included a recommendation for 'the society [to] be allowed to evolve naturally,' without predetermined or rigidly imposed notions of 'structure, objectives, roles, organization, etc.' The Task Force would 'provide guidance and leadership to the process.'

In keeping with its recommendations, the Prospectus included some modest proposals, such as (1) that an 'Individual Members Section' be formed under a standing committee of CNA; (2) that the Section be responsible for the Third Day Program at the CNA annual conferences; (3) that there be no plans, initially, for the publication of journals; and (4) that the Standing Committee of the Individual Members Section form the executive of the new organization.

Although it described an organization that, in profile, fell short of what the Society actually became a few years later, the Prospectus in many ways set the tone for the evolution of the Society. Specifically, the Society did build on its established base each step of the way and did respond to the various interests that influenced its development. More importantly, the Prospectus was circulated just before the 1978 CNA Annual Conference, with a questionnaire to middle management personnel in the nuclear industry. Of the 140 responses, 80% were in strong support of the proposals of the Prospectus. The results of the questionnaire were used as the basis of an open discussion after the Third Day luncheon at the 1978 Annual Conference in Ottawa. P. Ross-Ross, who had had many years of experience in the development of the Engineering Institute of Canada, and who would later play several key roles in CNS development, led the discussion following his luncheon address.

At the Langlois Task Force meeting of 17 August 1978, following a review of the questionnaire, many members expressed the view that, while there was an important role to be played by the new society in the area of 'authoritative' public relations with respect to nuclear technology, the greater emphasis should be placed initially on developing learned society traits. With the benefit of hindsight, this discussion must be considered ironic, because at the 26 February 1979 meeting of the Task Force, in which representatives of the CNA Public Affairs Committee were included, the case was made, that if the new society were to have local chapters or branches, individual members could play an important role in addressing local public nuclear issues. This activity would substitute for that of CNA Chapters, which were being proposed by the CNA Public Affairs Committee on the recommendation of its consultant Charles Yulish. Moreover, a particular individual member of the new society could be

perceived, by the public, as having less of a vested interest than the same individual might acting as a representative of the CNA!

The Langlois Task Force completed its work before adjournment of the 26 February meeting. It undertook to request, through its chairman, that the CNA Board of Directors establish a professional society (later referred to as a Technical Society) within the CNA. The Society was to be organized both regionally and nationally. [2] It was also agreed to ask the Board of Directors to nominate a small task force to develop the structure of the proposed society with a view to presenting the details to the CNA membership at the upcoming 1979 CNA Annual Meeting in Toronto. The CNA Board of Directors accepted these recommendations at its 13 March 1979 meeting, and promptly prepared to announce the existence of the 'Technical Society of the Canadian Nuclear Association' at the annual conference, to distribute early membership information, and to solicit individual memberships for the year 1980 at a $20.00 membership fee.

The membership information contained a statement of objectives and organization for the Society reflecting the recommendations of the Task Force. It also contained the names of the members of the Organizing Committee for the New Society. This committee would be charged with the task of completing the details of organizing the Society and to act as the Council, until such time as the Technical Society were able to hold elections, as set out in a constitution that it would prepare immediately.

The Langlois Task Force and the CNA Board of Directors had performed their respective tasks, and, as of 11 June 1979, the Canadian Nuclear Society at this point existed in all fundamental respects but that of its name.

The *Pro Tem* Council, 1979–80

Creation

The Organizing Committee of the Technical Society of the CNA, as named by the CNA Board at the time of the 1979 Annual Council, consisted of the following: W. MacOwan, CNA President; J.S. Foster, W. Bulger, N. Ediger, R. Langlois, and W.J. Smith of the CNA Board of Directors; J. Hewitt, G. Howey, and W. Paskievici of the CNA Education and Manpower Committee; D. Meneley of the CNA Technology Committee; and J.A. Weller, CNA General Manager.

The Organizing Committee met on 27 August 1979, under the chairmanship of W. MacOwan. The official name of the new Society was now to become: The Canadian Nuclear Society – The Technical Society of the Canadian Nuclear Association. The following were nominated and accepted as executive of the Organizing Committee, and therefore as officers of the *Pro Tem* Council of the new Society:

G. Howey, President
J. Hewitt, Vice President
D. Meneley, Secretary-Treasurer

It was understood that the *Pro Tem* Council would aim to have essential elements of the new Society in place (namely, its constitution, the assignment of its committee members, its membership eligibility, branch organization, and its 1980 budget) by 30 September 1979. The Organizing Committee then adjourned and did not subsequently need to reconvene as a full committee. Rather, the *Pro Tem* Council carried on the active business of the Society until the first Council was duly elected the following June.

The newly appointed executive met immediately in what was later referred to as the first meeting of the CNS Council. By the second meeting of the CNS Council on 11 September 1979, the membership consisted of: G. Howey, President; J. Hewitt, Vice President; D. Meneley, Secretary-Treasurer; W. Paskievici, Program Chairman; R. Harrison, Member-at-Large; and J. Weller, General Manager. P. Ross-Ross, who had participated in many of the earlier preparations, was added a short time later as Membership Chairman.

Accomplishments

The President and Vice President revised the earlier draft version of the Constitution and By-Laws to reflect the new circumstances. After approval by Council, the document was presented to the CNA Board on 25 October 1979. By 14 September 1979, Dan Meneley had prepared a budget for fiscal year 1980 (1 February 1980 to 31 January 1981). The budget was for $7,500. Membership fees for 1980 had already been set at $20. The need to keep separate CNS accounts was recognized, as there would be a finite period of CNA subsidy for the new Society.

In the early Council meetings, careful discussion took place in establishing necessities such as the logo and letterhead. Membership was the key item, and membership brochures, application forms, and a membership drive were needed. A branch structure was essential as a means of soliciting membership. Membership eligibility received careful attention. (The CNS is not a 'professional' society in the restricted sense; membership is not tied to educational qualifications. Rather, membership is open to individuals who are contributing, in a positive way, to the advancement of nuclear science and technology.) Those who joined before the end of 1980 are identified as 'Charter Members' of the Society. J. Hewitt took the lead role in preparing a membership brochure, and 2,000 were available for distribution in early 1980.

Policies, procedures, and guidelines were needed, and Phil Ross-Ross, with his wealth of experience from the EIC and CSME, took a lead role in preparing policy

statements. At this stage 'enabling legislation' was needed and duly prepared.

The evolution of the CNS would inevitably give rise to adjustments in the CNA. Jim Weller, in his dual role of General Manager of both the CNS and the CNA, kept the *Pro Tem* Council aware of CNA positions and interests. Lines of communication with the Board of Directors, however, were not always adequate in the circumstances. Although the Board's representation on the Organizing Committee, and hence on the Council, had discretely retired, there was some concern by the Board as to just what Council was doing. The one and only meeting at which the CNS found it necessary to explain its position to the CNA was held on 18 January 1980. Howey, Hewitt, Meneley, and Weller met with Bill MacOwan, Chairman of the Board of Directors. Concerns about membership, the relationship between the CNS and CNA, the independence of the CNS, conference responsibilities, and other miscellaneous matters were amicably resolved. To assure communications, and in keeping with the original intent, the CNS President was invited to attend *ex officio*, subsequent CNA Board of Directors meetings. Also, the newly appointed CNA President, Norm Aspin, attended Council in February 1980, and was constantly supportive of CNS affairs.

The 'Third Day', the day of specialized technical papers at the CNA Annual Conference, and one of the considerations in the formation of the Technical Society, had been placed by the CNA in the hands of the *Pro Tem* Council. Council was now faced with creating a set of new precedents. Decisions regarding calls for papers, instructions for authors, paper selection, camera-ready mats, and publication of proceedings would have to be made; such decisions would set many of the standards for CNS conferences in years to come. Council thus found itself heavily involved as 'the conference committee' for the first annual conference at a time when it really had expected to concentrate on organizing the Society. Fortunately, Wladimir Paskievici, who had been heavily involved in the 1977 CNA Annual Conference, did a noble job as the CNS's first conference chairman. Organizing the First Annual Conference put extreme demands on Wladimir's time and energy, particularly as he was also the Technical / Scientific Chairman of the CNA Conference. As there was initially some difficulty in having the Third Day actually identified as the CNS First Annual Conference, to don later a badge sporting both the CNS and CNA logos at the jointly arranged Annual Conference was a source of quiet pleasure to all those who had worked on the creation of the CNS and on the Conference.

Another major task of the *Pro Tem* Council was to produce a newsletter for disseminating information to members. John Hewitt was the key figure in creating the 'CNS-Bulletin-SNC.' The first 'Bulletin' was issued in May 1980. The cover page was printed on CNS letterhead; the contents were mostly oriented to the First Annual Conference of the CNS, to membership, election procedures, and general news on progress of the Society.

The CNS First Annual Conference was finally held in conjunction with the 20th Annual International Conference of the CNA in the Queen Elizabeth Hotel in Montreal on 18 June 1980. Forty-four papers were presented in eight sessions, and there was a closing panel session. 'The Three Mile Island Accident – The Canadian Perspective.' The guest speaker at the CNS luncheon was, fittingly, Prof. M. Tubiana, past-president of the Société Française de l'Énergie Nucléaire, a distinguished radiation scientist and physician.

The first Annual General Meeting of the CNS was held after the conference luncheon. George Howey, Interim President of the new society, could be justifiably proud of the achievements of the *Pro Tem* Council. The conference was a success, and the year had been very productive. Basic policies and procedures were in place, membership stood at 370, the first 'CNS-Bulletin-SNC' had been issued, and work was progressing to create branches to serve members locally and to establish divisions to meet the scientific and technical interests of the members. The nominating committee, chaired by Bob Harrison, presented a slate of officers selected to ensure reasonable continuity and also a reasonable geographic and technical representation. With no further nominations from the floor, the first Council, elected by acclamation, took office.

The Council consisted of George Howey, President; John Hewitt, Vice President; Bob James, Secretary-Treasurer; Ernie Card, Tony Colenbrander, Tom Gellatly, Joe Howieson, Wladimir Paskievici, Phil Ross-Ross and Tom Schur, Members-at-large; and Jim Weller, General Manager.

The First Elected Council, 1980–81

The first meeting of the new, duly elected Council, (also the 11th meeting of the CNS Council) was held on 18 July. One of the first items of business was the approval of the 'Guidelines for CNS Committees' prepared by Phil Ross-Ross. Five standing committees were formed: Communications, Program, Technical Divisions, Membership, and Finance and Administration. The chairmen of these new committees were Hewitt, Ross-Ross, Howieson, Colenbrander, and James, respectively. During the autumn of 1980, the creation of branches (under Tony Colenbrander) and divisions (under Joe Howieson) was high on the priority list.

Branch Steering Committees were formed in five locales: Ottawa (Joe Howieson), Manitoba (Ernie Card), Montreal (Michel Therrien), Chalk River (Ian Hastings), and Toronto (Arthur Guthrie). The Ottawa

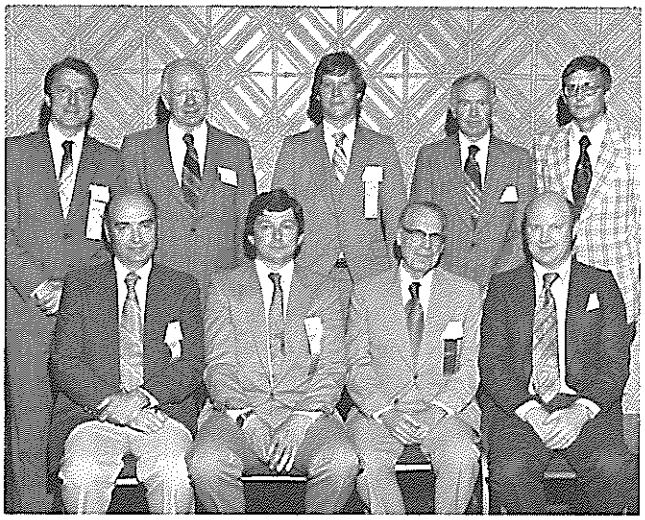


Figure 1 The first elected Council of the Canadian Nuclear Society, June 1980. Seated left to right: W. Paskievici; J.S. Hewitt, Vice President; G.R. Howey, President; R. James, Secretary-Treasurer. Standing left to right: J.A. Weller, CNA; T. Schur; E. Card; P. Ross-Ross; A.H. Colenbrander.

Branch, although not yet formally constituted, was first off the mark, holding a meeting on 4 February 1981, with Ron Hodge, as guest speaker, addressing a group of thirty on the 'Canadian Coast Guard's Nuclear Ice-Breaker Project.' The Manitoba Branch was the first constituted (approved by Council on 30 January 1981) and held its first meeting in Winnipeg on 23 February, with guest speaker John Boulton giving a talk on 'Nuclear Fuel Waste Management in Canada.' Requests for formation of branches in Ottawa (Joe Howieson), Chalk River (Ian Hastings), and Toronto (Arthur Guthrie) were approved by Council on 23 April 1981. The request from the Quebec Branch (Jan Charuk) was approved on 26 June 1981.

Branches are important to a society as a means of involving members in activities such as meeting with guest speakers, hosting technical conferences, field trips, and social events. In the early days of the CNS emphasis was placed on securing new members through the branches, and Tony Colenbrander contributed greatly in supplying the guidelines and background information to get branches operating and membership drives underway. As more members were enlisted into committees and activities, there was a need for a who's who (and where); the first issue of the booklet 'Membership List' was published in mid-1980.

The heart of a learned society is, of course, its scientific and technical structure. Joe Howieson was the key man. He presented a 'Policy Statement on the Formation of Technical Divisions' to Council in November 1980. The various areas of interest were grouped into four divisions:

Nuclear Science
Design and Materials
Mining, Manufacturing and Operations
Environment, Health and Public Affairs

Committees for each division were set up, and took lead roles in the development of the division. Questionnaires were sent out to establish the areas of interest of the members, and hence to complete the matrix approach of having each member identified with one branch and at least one division.

The division committees began to work on establishing how they wanted to govern themselves (within the guidelines provided by the society), how their interests would relate to those of the CNA Technology Committee, and how to get into the symposia, seminar, or conference arena. Some prepared 'Constitutions' designed specifically to govern the division's activities. Radioactive waste management was a topic that received much attention at this time, and this was a likely topic for the first major international conference of the new Society. In late May 1981, when Tom Drolet, Chairman of the Environment, Health and Public Affairs Division, announced plans to hold the conference in Winnipeg in 1982 in a convention centre still under construction, a few Councillors were apprehensive. The case was well presented, however, and Council, true to its policies on passing enabling legislation and planning for new activities, gave the organizing committee the go-ahead.

Although CNS membership was small, many of those who joined had contacts with other societies and specialist groups, and soon the CNS was being asked to co-sponsor (by lending its name) other conferences. The CNS immediately became a co-sponsor of the Eighth Annual Simulation Symposium in Toronto, March 1981; the Sixth International Conference on Modern Trends in Activation Analysis in Toronto, June 1981; 'Materials in Nuclear Energy' in Huntsville, September 1981 (Can. ASM); and 'Decontamination of Nuclear Facilities' in Niagara Falls, September 1982 (ANS, CNA).

The Program Committee (Ross-Ross) was putting together policy statements and guidelines related to planning, sponsoring, and running conferences. The CNS Second Annual Conference, held in Ottawa on 10 June 1981, in conjunction with the 21st Annual International Conference of the CNA, was organized under the able chairmanship of Frank McDonnell. He was well supported by members in the Ottawa Branch. They relieved Council of direct responsibility for the conference, and did much to develop the CNS's capability to run a first class conference. There were 65 papers in 13 sessions. The luncheon guest was Milt Levenson, speaking on the consequences of nuclear accidents. Levenson was later, as ANS President, to play a key role in formalizing CNS / ANS co-operation.

The Communications Committee (John Hewitt) edited and produced the CNS-Bulletin-SNC on a reasonably regular basis. It was proving to be extremely valuable as a means of informing members of CNS progress, and of upcoming conferences. In April of 1981, David Mosey relieved John Hewitt as editor of the Bulletin. Subsequently, Hugues Bonin became Associate Editor and Dave McArthur became Production Editor.

Bob James, as Secretary-Treasurer, worked with the staff of the CNA to establish procedures for handling the work of servicing the membership and the Council, and on the financial arrangements of the CNS. Headquarters was already well experienced in handling CNA activities and the CNS affairs presented new challenges which were met admirably by the staff of the CNA / CNS office.

By the end of the second year with George Howey as President, the CNS had the framework for the Society fairly well established. Branches and divisions had been formed, guidelines were in place, headquarters had been set up to handle a society of individuals, and the Bulletin was providing a means of communication between the society and its members. Membership was over 500, and over 50 were now serving on the various committees of the Society. The Society was now ready to start serving its members and the Canadian interest, both at home and abroad, in nuclear science and technology.

The Adolescence, 1981–1984

At the Second Annual General Meeting of the Society, the Council elected for 81/82 was Phil Ross-Ross, President; John Hewitt, Vice-President; Peter Stevens-Guille, Secretary-Treasurer; Ernie Card, Tony Colenbrander, Tom Drolet, Antoine Duchesne, Walter Harrison, Joe Howieson and Wladimir Paskievici, Members-at-Large; George Howey, Past-President; and Jim Weller, General Manager.

One should look at this period over the longer term. A great deal of stability had been achieved; there was a gradual turnover in Council membership, and the formation of branches and divisions had provided a source of dedicated members from which future councillors and presidents would emerge.

Although Ross-Ross and the Council were not generally in favour of a president serving two successive terms, the events of the time made his continuation highly desirable, and he agreed to stand for a second term.

There follows a record of Council membership for 1982–83 and 1983–84: for 1982–83, Phil Ross-Ross, President; John Hewitt, Vice President; Peter Stevens-Guille, Secretary-Treasurer; Ernie Card, Tony Colenbrander, Tom Drolet, Jan Charuk, George Bereznai, Joe Howieson, Irwin Itzkovitch, Members-at-Large; plus G. Howey and J. Weller. For 1983–84, John Hewitt, President; Peter Stevens-Guille, Vice President; Tony Colenbrander, Secretary-Treasurer; George Bereznai, Richard Bolton, Ernie Card, Jan Charuk, Irwin Itzkovitch, Gerry Lynch, Nabila Yousef, Members-at-Large; plus Ross-Ross and J. Weller.

Advances in the various areas of endeavour during the adolescence of the CNS will now be sketched under separate headings.

A Long-Term Plan

The first period of forward planning of the CNS occurred during the Task Force years. During those years, Phil Ross-Ross, John Hewitt and George Howey, all of them living in Deep River, formulated together the various guidelines and policy statements that had a fair influence on the organization of the CNS as it stood at the end of its second year of operation.

When Ross-Ross became President in June 1981, he prepared a document with the aid of Council entitled, 'The CNS of Tomorrow.' The document covered goals, organization, services to members, strengths and weaknesses of learned societies in Canada, strengths and weaknesses of CNS / CNA relations, and many other topics. It served as a guide for priorities and direction for the next few years. The key points of 'The CNS of Tomorrow' were as follows:

1 It is very difficult for any small Canadian learned society to work in the shadow of a corresponding giant to the south. That the ANS is a very well-organized society with well-developed conference and publication programs, heightens this concern.
2 The CNS needs visibility.
3 A conference program is good for visibility, and conferences should always be designed to earn revenue for the Society.
4 The CNS will have a journal one day, but not now. Journals can be a burden on a society in terms of resources and finances.
5 Someday the CNS may be an independent society. Meanwhile, the CNS should be run on an independent basis and should move towards paying for services received from the CNA.

Building strong membership in a technical society is a universal and difficult challenge. For CNS, superimposed on this challenge was an identity problem. The CNS was often inadvertently referred to as the CNA, and though one could point out the differences, the fact that the CNS was the Technical Society of the CNA still left some doubt and confusion. Identity was essential. To overcome these limitations the CNS would have to become well recognized within Canada, if only to facilitate building its program and membership, and also to become recognized internationally. The plan called for careful attention to be given to making the Society visible, and to earning the respect of the nuclear community everywhere. Three main avenues were open: (1) through a technical program of confer-

ences and seminars; (2) through first rate publications; (3) through active involvement in international affairs. As will be seen, progress was made on all counts over the next few years.

International CNS *Developments*

Establishing the CNS internationally was essential. The world was experiencing a proliferation of nuclear conferences, and the ANS and ENS (European Nuclear Society) were well entrenched. Lead times of three to five years on conferences are not unusual. The CNS would, in some sense, have to pry its way in if it were to run an international conference program without severe scheduling conflicts.

Canada has, for many years, contributed its fair share to the conferences and publications of other societies throughout the world, and the contributions have been of high quality. Canada has benefitted, and will continue to benefit from this exchange. Through an international conference program, the CNS hopes to make it easier for Canadian experts to exchange information with their peers from abroad. Through CNS proceedings, and through a journal, the CNS hoped not only to provide another vehicle for exchange, but to concentrate papers from Canadian programs under a Canadian label and thereby better identify Canadian contributions.

Although the CNA had carried the torch in these endeavours for many years, and could well have continued, many international technical conferences were run by the technical societies and the CNS was better suited to negotiate with its direct counterparts. So, in late 1981, riding on the coat-tails of the CNA and a well-established national nuclear program, the CNS began to move into the international scene. The move occurred at an opportune moment, owing to a combination of developments in the nuclear world at the time. The circumstances bear closer examination.

The ANS and ENS, the latter composed of a consortium of many European national societies, dominated the international scene, and the ANS had in addition very strong international committees. With the emergence of nuclear power and new national societies, the ANS was looking for some means of bringing the learned nuclear societies together to improve communication. Two main concerns were the proliferation of conferences and the world-wide spread of the anti-nuclear movement. In the late 70s, after a number of attempts, efforts were focused on the creation of an 'International Union of Nuclear Societies.' Rik Bonalumi and Jim Weller attended a formation meeting in Washington as CNS representatives in November 1980. Prior to this, W.B. Lewis, well known as the CANDU pioneer and promoter, and as a former ANS president, had been invited to a similar meeting a year earlier. Jim Weller had attended on behalf of the Canadian interest.

By 1981 the IUNS was developing into a cumbersome organization (in some ways paralleling the IAEA) and the promoters needed a means of easing into a less formal organization. In November of 1981 Weller and Ross-Ross attended a formation meeting in San Francisco and contributed significantly to the discussions that, essentially, abandoned the IUNS and led to the formation of the International Nuclear Societies Group (INSG). The INSG became a very loosely structured informal organization, composed of 12 delegates representing four areas: three from Europe (ENS), three from Asia (Japan, China, Korea), three from Latin America (Argentina, Brazil, Mexico) and three from North America (two from the ANS and one from the CNS). Although only delegates vote, observers from many other countries and institutions (e.g. IAEA) attend and participate in discussions. In 1982 Ross-Ross attended meetings held in conjunction with major conferences in Brussels (ENS as host) and Washington (ANS as host). The CNS was an accepted partner in these and subsequent meetings.

Ross-Ross later contributed to the progress of the INSG by chairing the third meeting of that organization in Vancouver, in September 1983, in conjunction with the fourth Pacific Basin Nuclear Conference. This was the first to be held outside the ANS / ENS jurisdiction. Canada and the CNS were in a unique position; although the CNS was a young emerging society, it was backed by a fully established national nuclear program. The CNS was in a good position to speak on behalf of the old established, as well as the new emerging societies, and the Canadian voice was heard. Ross-Ross later drafted the charter for the INSG. The INSG served well as a means of giving the CNS identity and visibility. Most nuclear societies, including the ANS and ENS, as well as Latin American and Asian societies, now knew there was a CNS as well as a CNA, both distinct from the ANS.

In early 1983, Muntzing of the ANS, supported by Zaleski of the ENS, placed before the INSG a proposal to form an International Institute on Nuclear Safety. The Institute would be composed of experts named by the learned societies (free of political and institutional impediments) who together could act as a prestigious group on a range of subjects, from a uniform nuclear safety philosophy to recommendations on codes, risk assessment methods, source terms, and other topics. A Special Committee of the INSG was formed under Prof. Zangger of Switzerland. Dan Meneley, with assistance from Gary Vivian and others, represented the CNS. The proposal was very ambitious, and there were many practical considerations needing attention. Many of the objectives overlapped with the IAEA's established role. The end result was a special advisory group to the Director General of the IAEA called the International Nuclear Safety Advisory Group (INSAG). Dan Meneley continues as the Canadian representative on INSAG.

The Third International Conference on Nuclear

Technology Transfer (ICONTT-III) is an excellent example of the CNS presence leading to a large and important Canadian participation in a conference, because it helped to plan the conference at a very early stage. The CNS was known to the other key partners in the ICONTT organization through INSG, and succeeded in obtaining CNS representation on the Program (T. Carter, N. Yousef) and Steering (J. Hewitt, J. Boulton) Committees. Not only did the CNS secure a large number of opportunities for paper contributions to the conference, it was recognized with the ANS and ENS as having had a special role in organizing the conference.

During this period Ed Hennelly, Manning Muntzing, and Milt Levenson (one-time presidents of the ANS) were the INSG delegates and key people in international affairs. Ross-Ross had the opportunity to talk with them about ANS/CNS co-ordination and co-operation, and he drafted the 'Agreement of Co-operation Between the Canadian Nuclear Society and the American Nuclear Society.' The agreement was approved by both societies in June 1983 and formally signed in Vancouver in September 1983. The agreement helped open the lines of communication between the two societies and included a number of benefits to members. The CNS now had more opportunity to participate on ANS committees, and vice versa. Of particular importance were the various committees dealing with technical programs. Rudi Sligl and Peter Stevens-Guille became the first CNS representatives on the ANS Program Committee and the ANS Power Reactor Division Committee, respectively.

As a result of INSG participation and the CNS/ANS Agreement, the CNS was now better positioned to co-ordinate its conference program with the international scene. Of importance was the potential for avoiding conflict through co-operation in establishing conference topics, dates, and locales at an early planning stage, and through the co-sponsoring of conferences to achieve broader publicity and attendance.

China, a country emerging from many years of isolation, was seeking to improve its participation in the world of technology transfer. Through the INSG, the Chinese Nuclear Society approached the CNS, in 1983, concerning an agreement of co-operation. Such an agreement between the two was ceremoniously signed in Toronto in November 1984, following a year of preparation by Presidents Hewitt and Stevens-Guille, with President Jang of the Chinese Nuclear Society.

On his retirement as CNS president in 1983, Phil Ross-Ross became the first incumbent appointed to the position of CNS International Delegate. The position later became an *ex officio* position on the CNS Council.

Conferences

As indicated in the description of the *Pro Tem* Council, the CNS did not delay getting its conference program underway. Thanks to the efforts of Tom Drolet, Eva Rosinger, Mel Feraday, Nabila Yousef, Ernie Card, and others, the first totally CNS-planned flagship event was an outstanding success. The International Conference on Radioactive Waste Management was held in September 1982 in Winnipeg, in conjunction with the Annual Information Meeting of the Canadian Nuclear Fuel Waste Management Program. About 240 attended the three-day conference for which 90 papers were selected. Of these, over one third were from abroad. The conference earned a surplus of over $20,000 – just what the Society needed. The Waste Management Conference set a standard for all future conferences.

The four Technical Divisions were effective in sustaining both the annual international-type of conference, and a series of seminars and symposia. The conferences organized by the CNS during its adolescence involved many individuals, all experts in the topic at the conference or seminar they organized.

With the limited space available here, only conference chairpersons are named, and events are listed only where the CNS was the principal organizer. Numerical Methods in Nuclear Engineering, September 1983 (Rik Bonalumi); Applications of Robotics and Remote Handling in the Nuclear Industry, September 1983 (Hugh Irvine); Containment Design, June 1984 (Nabila Yousef); Simulation Symposium, May 1982 (Rik Bonalumi); Commissioning Symposium, May 1983 (Brian Harling); CNA/CNS Student Conference, March 1984 (Eva Hampton, John Marczak, Archie Harms); Simulation Symposium, April 1984 (Neil Craik); CNS Annual Conference, June 1982 (D. Meneley); CNS Annual Conference, June 1983 (Jan Charuk); CNS Annual Conference, June 1984 (Irwin Itzkovitch).

Internal CNS Development

To help maintain a sense of common purpose among active members of the CNS, Ross-Ross introduced the concept of the Officers' Seminar, an idea borrowed from the ASME. The first such seminar for the CNS was held on 23 September 1982. All members of Council, Branch, Division, and other committees are invited to meet once a year to get up-to-date on the affairs of the Society. The seminar provides an opportunity to be reminded of who's who in the CNS; the aims, objectives, and services of the Society; and how the policies, procedures, and guidelines can help the officers in their specific roles. The Officers' Seminar has become a tradition.

By the end of the CNS's adolescence, the Technical Divisions had reached a degree of maturity, and constitutional changes were made so that, from July 1984, each Technical Division would be represented on the Council by its chairperson. Thus, the Technical Divisions Committee now became obsolete after having contributed much to the development of the Society. Also during this period, the position of Branch Activity Chairman was established on the

Council. The position was ably filled in the first instance by Ernie Card, the founder of the first CNS Branch.

CNS *Publications*

The CNS-Bulletin-SNC became a typeset newsletter in the fall of 1983. Also, a typeset technical supplement to the Bulletin was introduced at the suggestion of Joe Howieson in 1983–84. The feature of each supplement was a technical paper on a subject of current interest. At this point, the Bulletin had become an informative and entertaining newsletter with a serious technical dimension, thanks to the creative wit of its editors.

Also during 1983–84, the CNS undertook to develop an international market for its conference proceedings, both of the many conferences that had already taken place, and of future conferences organized by the Society. The favourable response to this offering of CNS publications was a good omen for the introduction of a CNS-published journal.

Early in the adolescent period, a segment of Council, including Hewitt, Howieson, Bereznai, and others, wrestled with the question of establishing the *Nuclear Journal of Canada*. Although most members considered a journal as a clearly desirable element of a learned society, it was difficult to put together a business plan to show that the journal would neither drain society funds nor burden the individual member through annual increases in membership fees to offset the conceivable losses.

In the autumn of 1983, the Council sought professional advice and commissioned Alan Wyatt to study the matter and report. Wyatt's proposal for a 'Canadian Nuclear Journal,' dated 18 February, was received by Council on 7 March 1984. Wyatt was authorized to recommend the next step.

By mid-1984, the CNS was a well-recognized society, the reserve fund was healthy, and it appeared as if the Canadian Nuclear Society could seriously consider launching the *Nuclear Journal of Canada*. At this writing, it is evident that the *Journal* is now launched, but the details are appropriately left to the sequel of the present historical account.

Concluding Remarks

In the present account, the evolution of the CNS has been traced from its early conception, through its formation, birth, and adolescence, to the stage of affirming many of the attributes of a mature scientific and technical society. That the CNS has achieved such a sound and visible stature, in so few years after the first co-ordinated steps were taken, is remarkable.

It will be evident from the foregoing account that the CNS was not simply 'willed' into existence. Rather, the founders took account of the prevailing circumstances, so that the CNS was indeed, in the words of the Langlois Task Force, 'allowed to evolve naturally.' Thus, the emerging Society was able to draw on the best of its environment to lay a sound foundation, to grow quickly, and to stand tall, strong, and self-directing among the nuclear societies serving the common good through the advancement of nuclear science and technology. Whether an earlier genesis would have produced such a viable creature is, at this point, immaterial. What is important is that the CNS remain self-sustaining and purposeful in the changing environment of the years to come. To succeed in the future, the Canadian Nuclear Society / Société Nucléaire Canadienne has only to remember and preserve its early genetic rapport with its environment as it tackles the many new challenges ahead.

Acknowledgements

During the first five years of its evolution, the CNS was able to plan and develop its future in an atmosphere free of the sporadic financing forced on societies primarily dependent on membership fees. Council was very conscious of financial matters and distributed funds with care. The Council's approach to finance was to keep membership fees low, put the profits from special events (conferences) into a reserve fund, steadily reduce the subsidy from the CNA, and launch new projects (e.g., the *Journal*) out of the reserve funds when the benefits and risks were properly defined. In this context, thanks are extended to President Norm Aspin, the Board of Directors, and Lloyd Secord, the long-time and capable treasurer of the CNA. Secord, more than anyone, was aware that, not only did the CNS receive a subsidy in the form of CNA headquarters support, but also had inherited responsibility for a conference program that had formerly been a source of funds for the CNA. Secord, annually (usually when discussing the sharing of the proceeds of the CNA / CNS annual conference) would give that friendly nudge to remind the CNS that the treasury of the CNA, as well as the CNS, looked forward to the day when the CNS would no longer be financially dependent on the CNA. One must assume that the Board was well satisfied with CNS progress, for they did not once interfere in Society affairs.

A special tribute is owed to Jim Weller and his staff. Jim, as General Manager of both the CNS and the CNA, was the Great Facilitator who somehow managed to get everything to work.

Another special tribute is paid to all who served on Branch, Division, and Conference Committees. Their diligence and eagerness to see the CNS succeed was indispensible to the well-being of the fledgling organization.

The support and patience of the first CNS members, including those who were unable to make an active contribution within the organization, was vital to the formation and development of the Society.

Finally, the authors, as the Society's first three

presidents, take this opportunity to thank the individuals of the CNS and its supporting institutions for the rewarding opportunity of serving a noble cause, and to offer best wishes for the success of the *Nuclear Journal of Canada*. May the *Journal* long serve to expand the objectives of the Canadian Nuclear Society / Société Nucléaire Canadienne.

Appendix: Chronology of Events Spanning the Formation and Early Years of the Canadian Nuclear Society

1960	Inauguration of the Canadian Nuclear Association (CNA)
1973–06	Technical Papers Program at the CNA Annual Conference
1976–06	Canadian Nuclear Association / American Nuclear Society Joint Conference in Toronto
1976–11–12	Letter from G. Howey to R. Langlois calling for the formation of a Canadian nuclear society
1977–06	First of a series of 'Third Day' Technical Programs organized by Standing Committees of the CNA for the CNA Annual Conferences in 1977, 1978, and 1979.
1977–07	Technical Society Steering Committee recommends ongoing Task Force.
1978–01	Langlois Task Force, first meeting
1978–06	Open discussion on a technical society at the Third Day Luncheon of the CNA Annual Conference
1979–02–26	Langlois Task Force makes recommendations to the CNA Board on the nature and establishment of the new Society, at its final meeting
1979–06–11	The Technical Society of the CNA formally established at CNA Annual Conference
1979–07–31	Meeting of the Technical Society Organizing Committee. The full name of the society as the Canadian Nuclear Society – The Technical Society of the Canadian Nuclear Association and the officers of the *Pro Tem* Council are confirmed
1979–07–31	First meeting of the CNS Council (*Pro Tem*)
1979–10–25	The CNS Council presents the draft CNS Constitution and By-Laws to the CNA Board
1980–05	CNS-Bulletin-SNC first issued
1980–06–18	First Annual CNS Conference, held in conjunction with the twentieth CNA Annual Conference in Montreal
1980–11–25	First technical publication of the CNS available as the full proceedings of the First CNS Annual Conference
1980–1981	CNS branches established for Manitoba, Ottawa, Chalk River, Toronto, and Quebec
1980	CNS Membership List first published
1981	Planning document 'The CNS of Tomorrow' tabled
1982–09	CNS holds its first major international conference, on Radioactive Waste Management, in Winnipeg
1983–09	CNS hosts meeting of the International Nuclear Societies Group (INSG) at the Pacific Basin Nuclear Conference in Vancouver and is heavily involved with the ANS and the ENS in INSG formation and development
1983–09	CNS / ANS Agreement of Co-operation formally signed at Vancouver
1983	CNS participates, through the INSG, in the formation of the International Nuclear Safety Advisory Committee (INSAG)
1984–10–29	CNS contributes a major formal submission to the Interfaith Program for Public Awareness of Nuclear Issues (IPPANI)
1984–11	CNS and the Chinese (PRC) Nuclear Society formally sign an Agreement of Co-operation in Toronto
1984	CNS commissions a study and receives a formal proposal for the *Nuclear Journal of Canada*

Notes and References

1. Minutes of the CNA Education and Manpower Committee, 2 March 1978. (Attributed to D.G. Andrews.)
2. *Weller JA*. Background paper of 24 July 1979. Attachment to agenda for the 27 August 1979 session of the Organizing Meeting of the Technical Society of the Canadian Nuclear Association.
3. Individuals who were early promoters of the concept of a Canadian nuclear society included D.G. Andrews, S. Banerjee, A.A. Harms, G.R. Howey, J.S. Hewitt, and W. Paskievici.
4. Letter from G.R. Howey to Roger Langlois, 12 November 1976. Appended to the minutes of the CNA Education and Manpower Committee, 13 January 1977.

A Review of the Health Effects of Energy Development

D.K. Myers and M.M. Werner
Radiation Biology Branch
Atomic Energy of Canada Limited
Research Company
Chalk River Nuclear Laboratories
Chalk River, Ontario, K0J 1J0

Abstract

Health risks associated with the commercial production of electrical energy from various sources have been estimated by various authors. Recent literature on this topic is reviewed. Commercial energy consumption per capita, a measure of a country's level of industrialisation and technological development, has increased several-fold over the past century. This development has been associated with an average increase of about 35 years in life expectancy in Canada and other industrialized countries. The average health benefits which accrue to the population of these countries outweigh the health risks of the required energy production by a very large factor. The statistical probabilities of mortality from various causes are reviewed.

Résumé

Les risques portés à la santé par la production d'énergie électrique par différentes sources ont été évalués par plusieurs auteurs. La littérature récente sur ce sujet est revue. Depuis le début de ce siècle la consommation énergétique commerciale par capita, une mesure du niveau de développement industriel et technologique, a augmenté de plusieurs fois. Au Canada et en d'autres pays industrialisés, ce développement a été associé à une augmentation moyenne de l'âge de survie d'environ 35 ans. Les bénifices moyens pour la santé résultants pour la population de ces pays dépassent de beaucoup les risques portés à la santé par la production énergétique requise. Les probabilités statistiques des différentes causes de mortalité sont revues.

Introduction

Remarkable social changes have resulted from technological development that has occurred in industrial societies over the past century [23, 40, 41]. Average income in constant dollars has increased markedly, even while the proportion of time spent at work has decreased and the amount of time available for creative and leisure activities has increased. The proportion of societal effort required to produce food for the population has decreased greatly due to the shift to energy-intensive industrial agriculture. Rapid communication and transportation systems have been introduced. Average life expectancy has increased (Figure 1), and the birthrate decreased as the ability of humans to control the circumstances of their life has grown.

Energy production plays a crucial role in technological development and industrial prosperity [5, 37, 41]. The present paper is concerned with the health costs and health benefits of this energy development.

Attributable Health Risks

Observed increases in life expectancy in industrialized societies (Figure 1) have been due largely to increased understanding of the causes of illness and death, and to increased ability to do something to minimize these causes. For example, the discovery that scurvy was attributable to the lack of some nutritional factor led to the use of citrus fruits and later of vitamin C, and to the rapid transportation of fresh produce to consumers, as successful preventive measures. Similarly, the discovery of bacteria and viruses as causes of various infectious diseases led to the application of disinfectants, chlorine in municipal water supplies, pasteurization of milk supplies, improved sanitary standards, and the introduction of vaccines and antibiotics, as successful preventive or therapeutic measures.

Immediate causes of death are currently classified into many different, internationally recommended categories [47]. The broad categories of immediate causes of death in Canada for 1983 are summarized in Table 1. Recent changes in death rates in these categories are indicated in Figure 2. It is apparent that death rates in most categories, except cancer and violent causes, are decreasing.

Keywords: health effects, energy production, nuclear power, energy consumption, life expectancy.

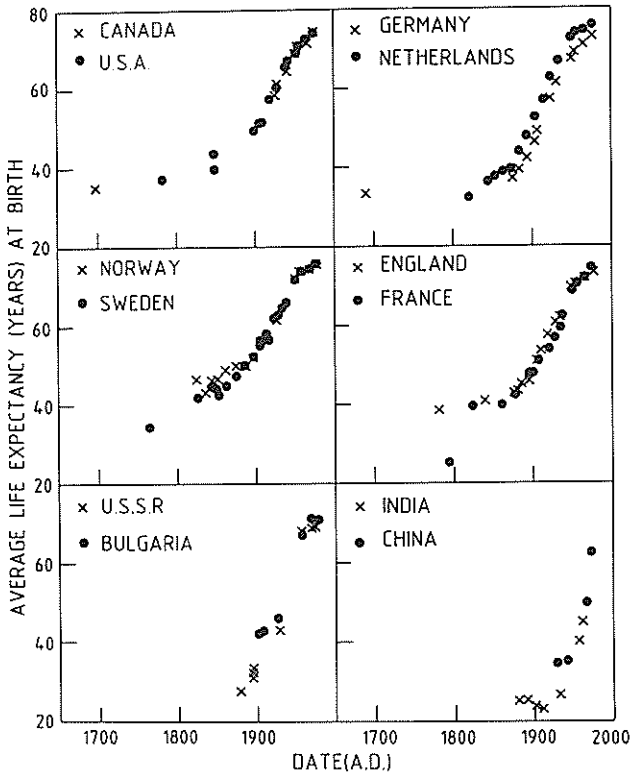


Figure 1 Average life expectancy at birth in various countries. Derivation of data given in Myers *et al.* 1984

For continued improvement in general health and longevity, it is essential to know more about the factors that contribute to these various categories of mortality. This understanding is also required for assessment of the health risks of energy production. Some comment

Table 1: Immediate Causes of Death in Canada, 1983 (Statistics Canada 1985; and Labour Canada 1982)

	Standardized death rates per 100,000			
Cause	*Male*	*Female*	*% of total*	
Pneumonia, influenza and tuberculosis	22.4	18.8	3.5	
Lung cancer	56.5	17.1	6.2	24.8
Other cancers	113.6	108.4	18.6	
External causes:				
occupational (1980)	6.1	0.2	0.5	
motor vehicle	24.2	9.1	2.8	8.7
suicide	21.1	6.2	2.3	
other	23.5	13.6	3.1	
Cardiovascular diseases	290.7	211.0	42.1	
All other causes	148.3	102.3	21.0	
Total	706.4	486.7	100.	

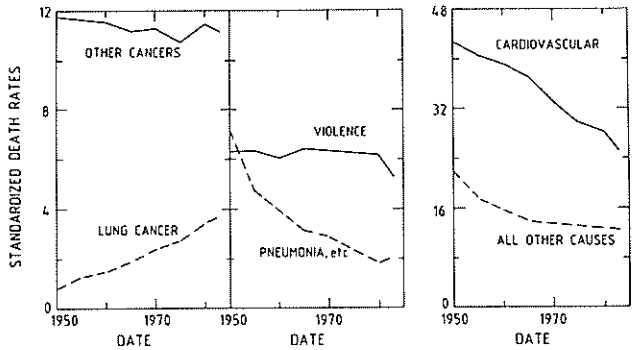


Figure 2 Standardized death rates for various causes of death, Canada 1950–1983. Data from General Mortality 1976, and Statistics Canada 1975, 1980, and 1983.

on factors that contribute to deaths in these categories would thus seem appropriate.

About five per cent of all deaths from violent causes [42] are due to occupational accidents [21]. A small proportion of other deaths from violent causes involve transportation of goods required for energy production. The immediate cause of a violent death is usually obvious and thus reliable statistics on known actuarial deaths attributable to various phases of energy production can be calculated.

In most other categories, the attributable cause of any specific death is less obvious and can only be derived on the basis of statistical probabilities. Considerable effort has been devoted to the quantitative assessment of the health effects of exposure to ionizing radiation [2, 18, 46]. The risk of fatal cancers induced by radiation is generally taken to be about 1.3×10^{-5} per mSv of whole body irradiation. On this basis, average exposures of 2 mSv per year for 75 years from natural sources of ionizing radiation would be responsible for about 0.8% of all fatal cancers in North America. Exposure to ionizing radiation is also believed to contribute to the genetic disorders that require medical attention at some time in a person's life and that can, on occasion, result in premature death; normal background levels of radiation are again thought to contribute about one per cent to the total of these genetic diseases. Estimates of the probability of induction of curable cancers by low doses of radiation are also available.

Combustion products were the first causes of cancer to be identified in the late eighteenth century and, in the form of cigarette smoke, are generally believed to be responsible for most of the recent increase in lung cancer deaths (Figure 2). Quantitative estimates of the risk of lung cancer following inhalation of combustion products have been derived [26, 28, 30, 32]. A small number of other types of fatal cancer can also be attributed to this cause [30]. Exposure to high concentrations of combustion products also contributes to

Table 2: Proportion of Cancer Deaths Attributable to Various Factors (Doll and Peto 1981; see also Wynder and Gori 1977; and Higginson and Muir 1979)

Factor	*Percent of all cancer deaths*
Diet	35
Tobacco	30
Viruses and other infections	10
Reproductive behaviour	7
Occupation	4
Alcohol	3
Pollution	2
Sunlight	2
Ionizing radiation	1
Medical procedures	1
Industrial products	< 1
Food additives	< 1 (a)
Unknown	5 ?

(a) Might also be less than zero allowing for protective effects of antioxidants and other preservatives. Note that most of the percentages given in this table are best estimates which are subject to considerable uncertainty.

premature death caused by non-malignant respiratory diseases [9], but it is uncertain whether or not the probability of these deleterious health effects is proportional to dose, at low levels of exposure. Since the urine of cigarette smokers is known to contain appreciable levels of mutagenic agents [20, 52], it is to be expected that inhalation of combustion products would increase the incidence of genetic disorders in the population; there are however no quantitative estimates of risk for this particular endpoint.

Cancer-causing agents have attracted considerable public attention in recent years. This may be due in part to the fact that more information on causes of cancer is becoming available (Table 2). Some 80–90% of all cancers can be correlated with differences in lifestyle in different societies and social groups, and are thus presumed to be preventable [6, 7]. Moreover, the proportion of all deaths in Canada due to cancer increased from 12% in 1941 to 25% in 1983 (Figure 3). This increase is due largely to successful reduction of other causes of premature death (Figure 2) and the resultant increase in life expectancy (Figure 1) in recent decades.

Health Hazards Associated with Energy Production

Considerable effort has been devoted in recent years to quantitative assessment of the health hazards of energy production, and there have been at least four international scientific conferences within the past eight years concerned with this particular topic. One of the earlier summaries of health hazards of four sources of electrical power is shown in Table 3. Although these assessments continue to become more sophisticated, the general magnitude of the resulting

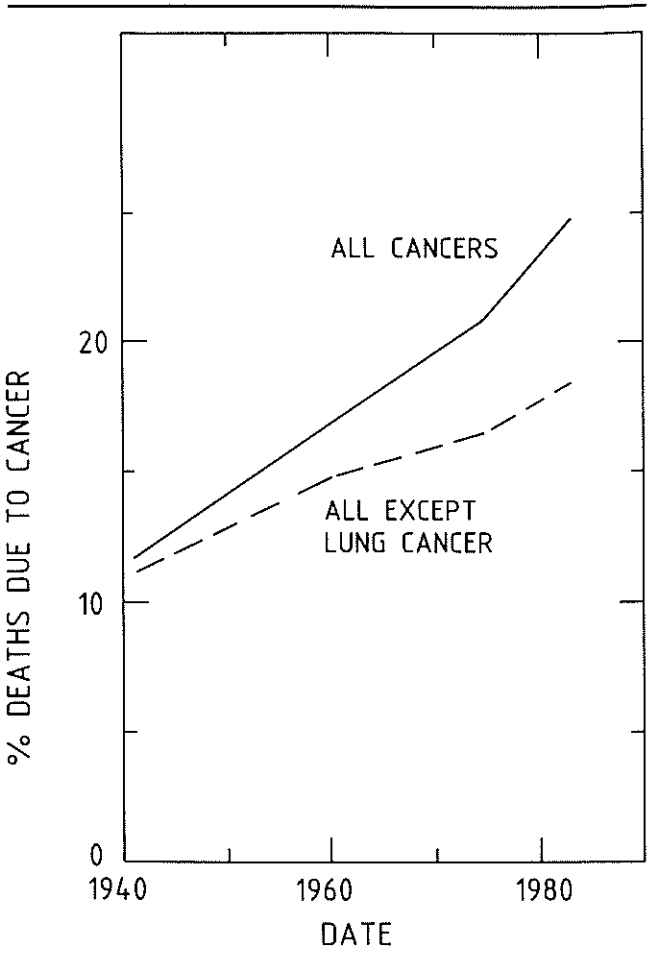


Figure 3 Percentage of deaths due to cancer, Canada 1940–1983. Data from Canada Year Book annual issues; see also references Figure 2.

data on hazards has not changed markedly (Tables 4 and 5). The predicted number of deaths attributable to production of electrical power varies from about 0.2–1 per GW·a for safer sources of energy such as nuclear power and natural gas, to about 10–20 per GW·a for more hazardous sources such as oil and coal [29]. These estimates include operations in all phases of energy production, from initial recovery of raw materials through refinement of these raw materials and production of useable energy in power plants, to disposal of waste products. Estimates of the statistical probability of catastrophic accidents, including those

Table 3: Estimated Deaths per GW(e)·a for Electricity Produced from Four Sources (American Medical Association 1978)

	Coal	*Oil*	*Natural gas*	*Nuclear*
Occupational	0.5–8	0.14–1.3	0.06–0.3	0.035–0.9
Public	1.6–306	1–100	—	0.01–0.2
Total (a)	25 (2.2–314)	10 (1.1–101)	0.13 (0.06–0.3)	0.2 (0.05–1.1)

(a) Total deaths given as the geometric mean of the range of values indicated in brackets.

Table 4: One Analysis of Estimated Deaths per GW(e)·a for Electricity Produced from Two Sources (Hamilton 1984)

	Coal			*Nuclear*		
Process	*Fatal accidents*	*Occupational disease*	*Public disease*	*Fatal accidents*	*Occupational disease*	*Public disease*
Underground mining (a)	1.2–1.5	0.6–1.5	—	0.4	0.2	0.05
Processing	0.03–0.1	—	—	0.006	0.05	0.01
Transport	0.2–4.8 (b)	—	—	0.01	0.001	0.0005
Electr. generation	0.1–0.2	—	15	0.01	0.13	0.12
Waste management	(not tabulated)			0.0001	0.005	0.000001
Total (a)	1.5–6.6	0.6–1.5	15	0.4	0.4	0.2
		17–22			1.0	

(a) Fatalities per unit energy are considerably smaller for surface mining; total deaths per GW(e)·a would become 15–19 for coal and 0.6 for nuclear if the fuel source were extracted by surface mining.
(b) Estimated fatalities depend on the method of transportation of coal, being lowest for transportation by pipeline and highest for transportation by truck.

Table 5: Estimated Deaths per GW·a for a Variety of Energy Sources (Inhaber 1982)

Energy source	*Occupational*		*Public*		*Total (approximate)*
	Accident	*Disease*	*Accident*	*Disease*	
Coal	2.5–6.7	0–0.8	0.8–1.9	17–60	20–70
Methanol	17–18	0.1	0.1–0.3	0.1–0.5	18
Oil	0.3–2	—	—	6–17	6–20
Solar space and water heating (a)	5.5–8.9	0.01–0.04	0.2–0.5	0.4–1.4	6–11
Solar photovoltaic	2–3.4	0.01–0.03	0.2–0.5	1.2–3.6	3–8
Wind	3.6–4.3	0.02–0.04	0.1–0.2	0.5–1.5	5
Hydroelectric	1.5–2.6	—	1.1–1.6	—	3
Solar thermal electric	0.4–0.5	0.03	0.15–0.3	0.6–1.8	1–3
Ocean thermal	1.7–2.3	—	0.03–0.07	0.04–0.1	2
Nuclear	0.3–0.8	0.2–0.8	0.01	0.07–0.5	0.6–2
Natural gas	0.2–0.5	—	0.01	—	0.2–0.5

(a) Values given for solar space heating are per GW thermal energy; all other values in the above table are per GW electrical energy.

that might occur in nuclear power stations with western standards of secondary safety features and containment, are usually included in these calculations. While Chernobyl has been the most noteworthy catastrophic accident in the energy production sector in the past few years, it should be noted that approximately 500 people died after an explosion in a gas storage complex in Mexico City in 1984, over 200 people died when a dam broke in Stava, Italy in 1985, and more than 200 coal miners have died in three accidents in as many years in Japan.

A more detailed comparison of coal and nuclear power is shown in Table 4. The phase of the fuel cycle responsible for the major portion of the health detriment depends upon the energy source. For coal, the major detriment appears to be public hazard from the effluents from coal-fired power stations; for nuclear, the occupational hazards associated with underground mining.

This type of assessment has been extended to a variety of other energy sources [10, 11, 19, 24, 33]. Many of these (hydroelectric and solar power, for example) appear to be intermediate in health hazards per GW·a between coal and nuclear power (Table 5), while

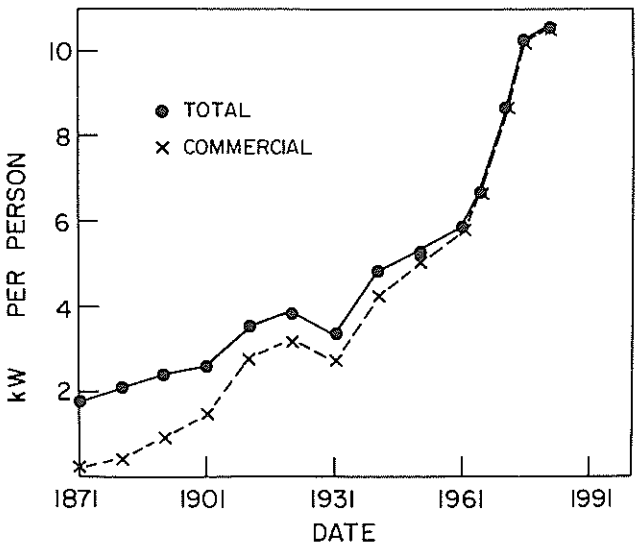


Figure 4 Total and commercial (fossil fuel, hydroelectric power, and nuclear power) energy consumption per capita in Canada, 1871–1981 (Steward 1978). Data for 1981 were extrapolated from United Nations Statistical Yearbook 1981.

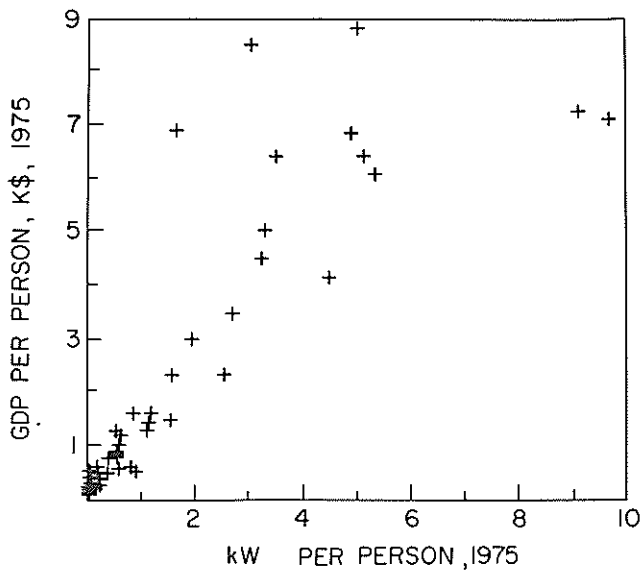


Figure 5 Relationship between commercial energy consumption and gross domestic product per person for various countries in 1975 (United Nations Statistical Yearbook 1981). Data have been converted to kW·a assuming 0.123 tonnes coal equivalent equals 1000 kWh (United Nations Statistical Yearbook 1981).

other traditional sources of heat such as fuel wood are apparently more hazardous than coal-fired power stations [10, 24].

Health Benefits of Energy Production

Technological development and industrial prosperity require affordable supplies of useable energy. In Canada in 1871, this energy was derived largely from fuel wood, animal work, and wind and water-driven mills, with a small contribution from human work. In recent decades, energy has been derived largely from fossil fuels, hydro-electricity, and nuclear power (Figure 4).

Energy consumption in different countries is closely correlated with industrial prosperity and the gross domestic product per capita (Figure 5). A close correlation with per capita purchasing power (income) based on actual costs in local currency of goods and services has also been demonstrated [4]. The plot of life expectancy in various countries in a given year versus commercial energy consumption per capita shows a striking increase in life expectancy, with increases in energy consumption up to about 0.5 kW per person, an intermediate phase in which life expectancy increases more slowly as energy consumption increases, and a mature phase above 2–3 kW per person in which there is little further increase in life expectancy (Figure 6). In this mature phase, nutritional, sanitary, and health care standards appropriate for that time have presumably been put into place in all industrialized countries. However, life expectancy has continued to increase with time in those countries where commercial energy consumption exceeds 2–3 kW per person (Figures 1 and 6). A large portion of this continued increase in life expectancy can be attributed to biomedical research in industrialized countries on causes of disease, and to application of methods to prevent or treat these diseases [25].

Reliable data on the relationship between total energy consumption and life expectancy during the earliest phases of technological development in human

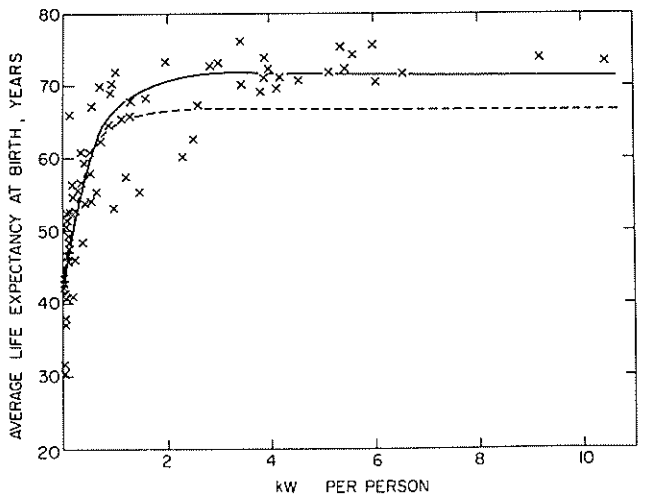


Figure 6 Effect of commercial energy consumption per capita on average life expectancy at birth in various countries. The data points represent the most recent values (usually about 1975) available from United Nations Statistical Yearbook 1981 for countries with more than five million inhabitants. The dashed and solid lines are the curves fitted to similar data for the years 1950 and 1975 respectively (Sagan and Afifi 1978).

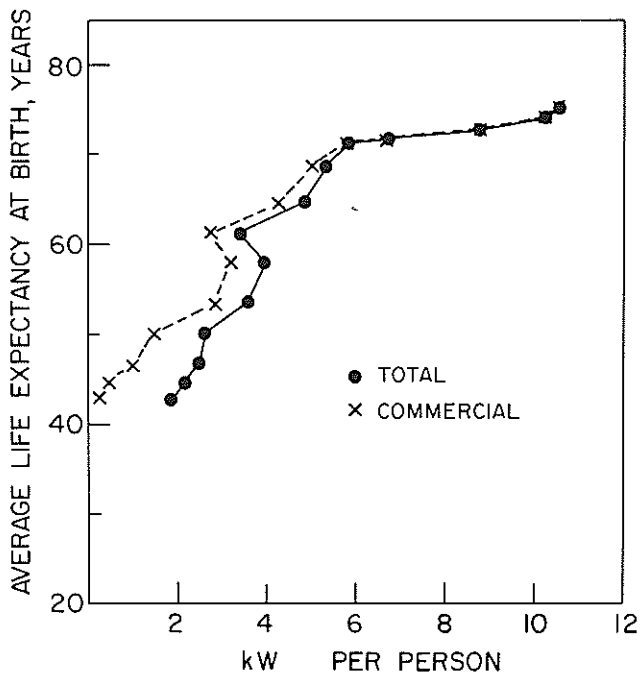


Figure 7 Effect of total and commercial energy consumption per capita on average life expectancy at birth, Canada, 1871–1981 (Steward 1978, and Historical Statistics of Canada 1983).

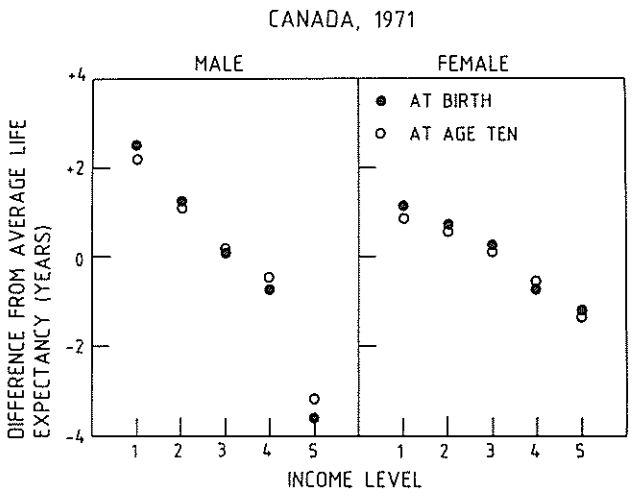


Figure 8 Effect of average household income on life expectancy at birth (Wigle and Mao 1980). Average income levels for groups 1–5 were taken to be 12.1 K, 10.6 K, 9.1 K, 8.0 K, and 6.7 K Canadian dollars, respectively. Each of the five groups contained 20% of the total urban population in Canada in 1971.

history are not available. We do however know that the average life expectancy in Ohio during the period 700–1100 A.D. was about 20 years [22], and that average life expectancy in the same part of the world was about 40 years by 1871 (Figure 1), at a time when total energy consumption was about 2 kW per person (Figure 4). Assuming that energy production under primitive conditions was limited to the work capacity of the human body (i.e., about 0.1 kW per person), the health benefits associated with the earliest stages of societal development would on average be somewhere in the region of 10 years increase in life expectancy per kW energy production per person.

When life expectancy is plotted against energy consumption in Canada over the past 110 years, the data indicate a rapid increase in life expectancy of about six years per kW per person at a time when total energy consumption was increasing from 2 to 6 kW per person; a slower increase, amounting to about 0.7 years per kW, is evident as total energy consumption increased above 6 kW per person (Figure 7). The difference in the slopes of the lines between the earlier and later portions of this century is probably related to two factors: recent increases in energy consumption for material comforts not directly essential for health, and the fact that most of the infectious diseases that can result in premature death of otherwise healthy individuals had been largely eliminated by the 1950s.

Another estimate of the effects of prosperity on average life expectancy in Canada can be derived from Figure 8, which shows deviations from average life expectancy for urban populations in 1971 with different average household incomes. Average energy consumption in Canada at this time was about 8.5 kW per person (Figure 4). Assuming that differences in average income levels will be reflected, directly or indirectly, in proportionate differences in average energy consumption, the data suggest an average increase in life expectancy of 0.9 years (1.2 for males, 0.6 for females) per kW per person between the low and high income urban groups in Canada. It is of some interest to note that this difference in life expectancy (Figure 8) was more closely correlated with income than with average educational levels [49], and continued to exist at a time when there was universal insurance for medical care costs, safe urban water supplies, and compulsory basic education. Similar variations in average life expectancy of high and low income groups in the same community have been demonstrated for people living in Montreal, Canada [50], and in other countries. The differences between males and females cannot be explained in terms of occupational health hazards [27] and are usually attributed to differences in lifestyle.

Ratio of Health Benefits and Costs of Energy Development

As indicated above (Figure 1), the major health benefit of recent technological development is an increase of 35–40 years in average life expectancy. Many factors contribute to this increase, notably adequate nutrition, safe water supplies, adequate housing, improved medical care, etc. It is difficult to separate out the contribution of any single factor in a complex industrial society to the increase in life expectancy. An

Table 6: Health Benefits and Detriments of Technological Development and Energy Production in Terms of Life Expectancy (LE)

	Effect per kW per capita	*Reference*
Health Benefit		
Increase in LE due to technological development as total energy production increased from 2 to 6 kW per capita	6 years	Figs 1, 6, & 7
(Potential increase in LE attributable to commercial energy production in above range)	(0.6 years)	(see text)
Increase in LE as total energy production increased above 6 kW per capita	0.8 years	Figs 7 & 8
(Potential increase in LE attributable to commercial energy production above 6 kW per capita)	(0.08 years)	(see text)
Health Detriment		
Total risk of fatalities for safer sources of commercial energy production such as nuclear power and natural gas	$0.2–1 \times 10^{-6}$ deaths per year	Tables 3–5
(Potential loss of LE attributable to safer sources of energy production)	(0.0005–0.002 years)	(see text)
Total risk of fatalities for more hazardous sources of commercial energy production such as oil, coal, and wood	$10–20 \times 10^{-6}$ deaths per year	Tables 3–5 and text
(Potential loss of LE attributable to more hazardous sources of energy production)	(0.02–0.04 years)	(see text)

analysis of the variations in average life expectancy among different countries and within given countries over a 25-year period from 1950 to 1975 (Figure 6), indicated consistent correlations only with energy consumption per person (an indicator of industrial development) and with literacy of the population [35, 36]. On the basis of this type of analysis, the proportion of the total increase in life expectancy that might be directly attributed to energy production has been taken to be about 10% [35]. The cost of energy in all forms delivered to the end-using device is also about 10% of the gross domestic product [41]. This value of 10% will be used as a working hypothesis for calculation of cost-benefit ratios in terms of life expectancy, even though it is obvious that life expectancy depends upon many other factors in a complex social system, and that the industrialized social systems that have led to marked increases in life expectancy cannot function in the absence of energy sources.

Assuming, then, that the observed increase of 35 years in life expectancy in industrialized countries (Figure 1) is achieved by the time that the required commercial energy production has reached 2–6 kW per person, or total energy production has reached 4–6 kW per person (Figures 6 and 7), and that 10% of this increase could be assigned to energy production, the total health benefit in the earlier stages of industrialization would thus be approximately 0.6 years increase in life expectancy per kW per person (Table 6). More recent increases in life expectancy in Canada have been considerably smaller per unit energy consumed (Figure 7); assuming again that 10% is assigned to energy production, the health benefit would appear to be about 0.08 years increase in life expectancy per kW per person (Table 6).

The health costs of energy production can be calculated in the same units (i.e., years of life expectancy per kW energy consumed per person) from data on the statistical probability of a premature death per kW·a of energy produced (Tables 3–5). For this purpose, it is assumed that each premature death would result in an individual loss of about 30 years in life expectancy. (This value is a composite of estimated individual losses of about 10–15 years for fatal cancers induced by occupational exposure to radiation, combustion products, and other cancer-causing agents; about 20–30 years for exposures of the public to the same agents; and about 35 years for fatal accidents, either to workers or the general public.) It is further assumed that energy consumption will continue at the same rate for a lifetime of 75 years. The average loss life expectancy per KW per person is thus in the region of 0.001

Table 7: Benefit/Cost Ratios for Effects of Energy Production on Life Expectancy (LE), Assuming 10% of Health Benefits of Technological Development Could be Attributed to Energy Production
(data derived from Table 6)

	Increase in LE / *Loss of* LE	
	Developing countries	*Industrialized countries*
Safer sources	300–1000	30–160
More hazardous sources	15–30	2–4

years for safer sources of commercial energy, such as nuclear power and natural gas, or of 0.02 years for more hazardous sources, such as coal-fired power stations (Table 6). The benefit: cost ratios for energy production in terms of life expectancy would thus appear to be high, and are still much greater than 1.0 even in the industrialized nations of North America, which have currently reached higher levels of energy consumption per person than any other country in the world (Table 7).

The average health detriment associated with commercial energy production is not uniformly distributed in the population but falls selectively upon certain groups of workers, notably those involved in underground mining of coal and uranium, in logging and in offshore exploration for gas and oil. This topic has been considered in more detail elsewhere [27] and is summarized briefly in Table 8 for countries, such as Canada, that utilize about 10 kW per person from a

Table 8: Effects of Technological Development and of Various Aspects of Nuclear Energy on Life Expectancy

	Change in life expectancy in years
Shared benefit of technological development	+35
Shared detriment of production of 10 kW from mixed energy sources	−0.1
Specific detriments:	
(a) occupational fatalities in safe industries	<−0.15
(b) average occupational fatalities for all workers in Canada	−0.15
(c) occupational fatalities for workers in CANDU stations in Ontario	−0.05
(d) radiation-induced fatalities for members of public living on boundary of a CANDU site	−0.0014
(e) radiation-induced fatalities for members of public living within 30 km of Chernobyl site 26 Apr.–5 May 1986	−0.05

(a–c) Assuming a working lifetime of 50 years at recent levels of occupational fatalities or radiation exposures.
(d) Assuming an incremental exposure of 0.05 mSv per year for 75 years.

Table 9: Deaths of Persons Age 20–64 in Canada, 1983 (Statistics Canada 1985)

Cause	*Total deaths* (a)	*% of total*	
Pneumonia, influenza and tuberculosis	610	1.3	
Lung cancer	4507	9.5	42.3
Other cancer	15611	32.8	
External violent causes:			
occupational	782 (b)	1.6	18.6
motor vehicle (non-occupational)	2584	5.4	
suicide	2966	6.2	
other	2515	5.3	
Cardiovascular diseases	15124	31.8	
All other causes	2812	5.9	
Total	47511	100.	

(a) The values given are not standardized for age distribution of the population.
(b) Data from 1980 (Labour Canada 1982). An additional 89 deaths were attributed to occupational diseases.

mixture of energy sources. In contrast to underground uranium miners, workers in nuclear power generating stations do not suffer from a relatively high rate of fatal occupational accidents [12]. Average occupational exposures of these workers to ionizing radiation [12] are also low enough to bring the predicted rates of radiation-induced fatal cancers well below the limit of occupational fatalities in other safe industries. About 80% of all workers in Canada are employed in safe industries where the probability of an occupational fatality does not exceed one per 10,000 workers per year [21, 27]. For workers in nuclear power generating stations as well as for workers in other safe industries, therefore, the probability of a premature death due to occupational circumstances does not exceed one-fiftieth of the probability of death due to all other causes between ages 18 and 65 (Table 9).

For the type of calculations given in Tables 3 to 6, the predicted health hazards of releases of cancer-causing materials from current energy production facilities are usually summed over periods of about 100 years into the future. A complete cost-benefit analysis on the health effects of current commercial energy production and technological development might be considered to require a summation of all health benefits extrapolated over all future generations, and of all health detriments extrapolated over all future generations. No reliable analysis of this kind can be carried out at present. Improvements in technology and health care during the past 100 years have been so dramatic that any attempts to extrapolate the future health costs and health benefits of current activities to times much greater than 100 years into the future are extremely unreliable.

Discussion

There has in recent years been appreciable discussion on the advisability of further development of various energy sources. Some of these discussions have centred on associated health risks and on issues such as voluntary versus involuntary hazards [38, 39]. From the perspective of general health of the public and of workers, a more balanced discussion would attempt to examine both the health benefits and health risks of energy production and would include the involuntary health hazards associated with poverty and lack of technological development. Data pertaining to this topic have been presented above. The general conclusion from these data is that industrialization and technological development have proven to be remarkably effective in improving public health. The health benefits vastly outweigh any health detriments associated with the required energy production.

Many other factors (for example, relative costs, security of future supplies of raw materials, long-term effects of acid rain, and long-term effects of atmospheric accumulation of carbon dioxide) must be considered in societal efforts to make rational choices concerning further development of various energy sources. In general, most of these factors would appear to favour nuclear power as one of the major sources of electrical power in the future.

Any consideration of the health effects of energy development would be incomplete without some consideration of the effects of the catastrophic accident at the Chernobyl nuclear power site in the Ukraine in April 1986. The health effects of the Chernobyl accident have been discussed in detail [16]. It has been reported that 31 of the workers and firefighters at the site died as a result of the skin burns, trauma, and high radiation doses received on 26 April 1986; another 170 workers received radiation doses high enough to cause temporary, non-fatal symptoms of radiation sickness [16]. The number of immediate, known deaths attributable to this accident is small compared with potential effects on the general public of radionuclides released into the biosphere. About 135,000 members of the general public were evacuated from a 30 km zone around the Chernobyl site within 10 days of the accident; the increase in radiation dose received by these persons is currently estimated to average about 0.14 Sv [16], or roughly the same dose that people normally receive from natural sources over a 70-year lifetime. On the basis of internationally accepted standards, about 200 additional fatal cancers might be anticipated in this group of 135,000 persons within several decades; the resulting average decrease in life expectancy for these persons would be about 0.05 years (Table 8). There is still considerable uncertainty concerning the average increase in radiation dose to persons living in the Ukraine and other parts of western Russia outside the 30 km zone; current estimates suggest that the average increment in effective dose equivalent summed over the next 50–70 years would be somewhere in the region of 0.01 Sv per person [16]. The effects of 0.01 Sv on mortality of the general population of western Russia are summarized in Table 10. At this dose, the predicted number of additional fatal cancers among 75 million persons would be about 10,000 (with a two-fold range of uncertainty) and the average loss of life expectancy for the whole population would be about 0.005 years or 2 days, assuming 20–30 years loss of life expectancy per radiation-induced fatal cancer. This value, while not insignificant, is small in comparison to the health benefits associated with technological development and energy production (Figure 1 and Table 6).

Table 10: Predicted effect of radionuclides from the Chernobyl accident on the mortality of 75 million inhabitants of the Ukraine and western Russia (IAEA 1986)

Cause of death	*Expected before Chernobyl*	*Expected after Chernobyl*	*Difference*
Cancer	9.5 million	9.51 million	+ 10 thousand
Other causes	65.5 million	65.49 million	− 10 thousand
Total	75 million	75 million	0

In this respect, several additional items might be noted. First, the Chernobyl power station was not designed to the same standards of safety as, for example, the CANDU system [34, 44]. Second, the normal variation in exposure to radiation from natural sources in different Canadian cities is considerably larger, approximately 0.1 Sv over 70 years, due mainly to differences in the average concentrations of radon daughters in houses [46]. No significant differences in cancer mortality attributable to variations in radiation doses from natural sources in different Canadian cities have been observed. Third, current vital statistics for the USSR suggest that about 13% of all deaths are due to cancer [16, 48; see Table 10]; these statistics are similar to those which were applicable in Canada about 40 years ago (Figure 3). And finally, the health effects of the Chernobyl accident decrease with increasing distance from the site of this release of radioactive materials into the biosphere. The predicted number of fatal cancers induced in the Canadian population as a result of miniscule increments in radiation exposure caused by the Chernobyl accident appears to be less than one [31]. Although the Department of National Health and Welfare in Canada did recommend a temporary ban on drinking rain-water, due to detectable increases in radioactive iodines in May 1986, the limit on which this recommendation was based [15] is roughly 100 times smaller than the limits recommended by most other organizations concerned with emergency planning [17].

Radionuclides released into the biosphere as a result

of the Chernobyl disaster are thus not expected to have significant adverse effects on the health of people in any country, including the USSR (Table 10). Assessments of the safety of nuclear reactors with North American standards for secondary protective features including containment are currently being re-examined. There does not as yet appear to be any reason to alter the conclusion that nuclear power produces fewer adverse health effects than most other sources of energy in western countries (Tables 3–5), and that the average health benefits associated with technological development and industrial prosperity outweigh any health detriments of the required energy production by a very large factor (Tables 6–8).

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Food Irradiation: Commercial and Scientific Review

B. K. Wilson
Director of Marketing
AECL-Radiochemical Company
Industrial Division
413 March Road
Kanata, Ontario K2K 1X8

Abstract
Radiation processing offers the food industry a 'new' process to control food spoilage and pathogenic organisms. The process is based on four decades of extensive research on the underlying aspects of radiation chemistry and radiobiology, concurrent with over 25 years of technical development and commercialization of radiation processing equipment. To date, many countries have approved a wide range of food irradiation applications, but commercial use of the process is just beginning to grow significantly. This paper reviews some aspects of the scientific and commercial background behind this process, focusing on the microbial and chemical safety, and the nutritional issues.

Résumé
Le traitement par radiation offre à l'industrie alimentaire un 'nouveau' procédé permettant de maîtriser les organismes pathogènes qui causent la pourriture. Ce procédé a fait l'objet de quatre décennies de recherche approfondie sur les aspects sous-jacents de la chimie de la radiation et de la radiobiologie. Depuis plus de 25 ans, on effectue des travaux de développement technique et on pratique la commercialisation d'appareils de traitement par radiation. Jusqu'à présent, de nombreux pays ont accepté une large gamme d'applications diverses pour le procédé d'irradiation alimentaire mais l'utilisation commerciale du procédé ne fait que commencer à démarrer. Le présent document fait l'analyse de certains aspects scientifiques et commerciaux reliés à ce procédé et met l'accent sur la sécurité microbienne et chimique ainsi que sur la question de qualité nutritive.

Introduction

Food is essential for human survival, but unfortunately it begins to deteriorate soon after harvest. Much of the spoilage is due to microbial action and insect infestation. Radiation processing in combination with proper storage technology has proven to be effective in controlling microbial and insect causes of spoilage. It is also an effective method of reducing or eliminating pathogenic food-borne organisms such as *Salmonella*. Radiation processing technology, properly and appropriately applied, can contribute to increasing the availability of good-quality foods, and to improving their safety.

Food irradiation attracted many countries during the 1950s push to find peaceful uses for nuclear technology. Food spoilage and food-borne disease was, and remains, a universal problem. Post-harvest losses range from 5% to 50%, depending on the country. Economic and human losses due to pathogenic food-borne *Salmonella* organisms alone are enormous. A recent U.S. Department of Agriculture report [1] estimates economic loss at several billion dollars/year, and, in Canada, a Health Department report [2] estimates 763 deaths per year in Canada are attributable to salmonellosis. A survey in 1968 found that 76 countries had active programs on food irradiation. [3] The need then and now remains the same.

Amongst the most active countries in these early years were: Argentina, Belgium, Canada, Chile, Denmark, the Federal Republic of Germany, France, Greece, Hungary, India, Israel, Italy, Japan, the Netherlands, Pakistan, the Phillipines, South Korea, Thailand, the U.K., the U.S.A., and the U.S.S.R. These countries independently, and often in co-operative projects encouraged or sponsored by international organizations, such as the United Nations and the World Health Organization, generated a vast body of published scientific work. The results addressed the fundamental issues of the safety of irradiated foods and the benefits of using the technology.

Beginning in 1961 and concluding in 1980, five international Joint FA/IAEA/WHO Expert Committees (JEC) were convened to address the issue of irradiated food safety. Each committee, in turn, examined the ever-growing and extensive scientific information

Keywords: irradiation effects, irradiation machines, micro-organisms, foods, nutrition.

available, and recommended additional areas of study. The final JEC, in 1980, recommended that the irradiation of any food up to an overall average dose of 10 kGy is safe. [4] This recommendation was circulated to all 122 member countries of the Codex Alimentarius Commission (CAC), a joint FAO/WHO body established to develop international food standards. After two years of deliberation, the CAC adopted the 1980 JEC recommendation and promulgated a Codex General Standard for Irradiated Foods and a Recommended Code of Practice for the Operation of Radiation Facilities Used for the Treatment of Foods. [5]

Since 1983, many nations have issued first regulations permitting food irradiation; others have broadened existing rules (e.g. U.S. FDA, April 1986[6]), and others, like Canada, are in the process of modifying their regulations to be more consistent with the Codex General Standard. Presently, over 25 countries have approved a wide range of irradiated foods, and commercial implementation is progressing steadily.

Before addressing the commercial part of this subject, it is appropriate to review the principal technical and scientific aspects as they relate to the food irradiation process. These cover the microbiological aspects, chemical effects, and nutritional quality.

The process

Food irradiation consists of exposing foods to a source of ionizing radiation to obtain the dose necessary to achieve the desired effect. Different doses achieve different effects in different foods as can be seen in Table 1. The dose needed may also vary from product to product depending upon the organisms involved and the tolerance of the product. Establishing the correct dose for a given product is a routine matter of product testing with easily established protocols. However, not all foods respond favourably to irradiation. For example, most varieties of grapes will tolerate low doses used to kill or sterilize insects, but when higher doses are applied to prevent mould growth the grapes become softer than is desirable. Similarly, irradiated milk, although effectively pasteurized, tastes differently from heat-pasteurized milk.

In addition to establishing the correct dose, other factors must be addressed, such as the need for temperature control during irradiation, the selection of radiation-compatible packaging materials, and the definition of irradiation procedures that conform to approved food handling practices. These factors are not complex, but because food irradiation is a 'new' process, they are being addressed, in most cases, for the first time for specific products in the context of national requirements. In this regard, the Codex General Standard and Code of Practice provides a measure of standardization.

At this time there is only one source of ionizing radiation readily available, and technologically developed for food irradiation. This is the radioisotope cobalt-60. Cobalt-60 is a deliberately produced isotope that emits ionizing energy in the form of gamma rays; it is not a waste product. These rays are not capable of inducing radiation in the irradiated food. The rays pass through the product depositing energy. The deposited energy kills organisms by mechanisms which are discussed in the next section.

According to AECL marketing data there are, at present, over 125 large cobalt-60 sourced industrial irradiators, over 2000 cobalt-60 sourced cancer treatment machines, and over 300 cobalt-60 sourced research irradiators in use world-wide.

A second source of ionizing enery being developed for food applications uses a machine to produce an accelerated beam of electrons. These electrons can be applied directly, or can be used to generate high energy X-rays. The electrons are less penetrating than gamma rays, but the ionizing effects are essentially the same. The converted X-rays have good penetration, but the costs of conversion are significant in terms of power consumption/loss. Electron beam machines have been in use for cancer treatment and industrial processing of thin films for many years. However, the technology still needs to be developed further and adapted to meet some of the potential food irradiation applications.

Each source technology is best suited to different products, and each will complement the other in commercial application. Cobalt-60 irradiators do an efficient and cost-effective job on bulky, packaged foods, and moderate volumes of high-value granular products such as enzymes. Electron beam machines, when they are developed and commercially available, should be cost-effective for huge volumes of fluid and granular products that can be thinned, and may also be appropriate for the processing of thin packages in in-line configurations.

Presently, there are ten large food irradiators in use world-wide, and several more are under construction. There are about 25 other irradiators that process small quantities, from time to time, in addition to their normal throughput of medical disposable products. There are also plans for an additional six dedicated food irradiators that are likely to be realized in the next few years (AECL data). A large pallet irradiator, similar to AECL installations in Holland and the Federal Republic of Germany, is shown in Figure 1.

Microbiological Aspects

Ionizing radiation affects living cells by creating in the product ions, excited molecules, and free radicals that react with other molecules which, in turn, damage the cells. [8] When enough damage occurs, the cells die or cease to function. The sensitivity of an organism to radiation is usually expressed as a 'D-value.' This is the radiation dose required to inactivate 90% of the

Table 1: Some Examples of Irradiation Applications[1]
(Not Comprehensive – Examples Only)

Food	*Purpose of irradiation*	*Approximate dose needed (kGy)*	*Countries which permit application*
Potatoes and/or onions	Sprout inhibition	up to 0.15	Bangladesh, Brazil Canada, Chile, China Israel, Italy, Netherlands, South Africa, USSR, USA, Yugoslavia
Strawberries[2]	Shelflife extension/mould inhibition	1–4	Brazil, Chile, South Africa
Wheat[3]	Insect disinfestation	up to 0.15	Canada, Chile, USA, USSR
Chicken	Shelflife extension and Salmonella control/elimination	2–7	Bangladesh, Brazil, Chile, France, Israel Netherlands, South Africa
All fruits and vegetables	Disinfestation and shelflife extension	up to 1	USA
Fish (fresh)	Shelflife extension	up to 2	Bangladesh, Brazil
Shell fish (frozen)	*Salmonella* control/elimination	2–7	India
Spices[4]	Microbial reduction, insect disinfestation, sterilization	up to 10, 30 kGy in USA	Bangladesh, Brazil Canada, Chile, India, France, Israel Hungary, Norway South Africa, USA

[1] Data taken from 'Food Irradiation Newsletter' a Joint FAO/IAEA, publication, Vienna, August 1985, [7] and revised by author.
[2] Commercial operation in South Africa
[3] Commercial operation in USSR
[4] Commercial operation in Brazil, France, Israel, Hungary, South Africa, USA

population of that organism in the irradiated sample. Some representative D-values for a range of microorganisms are listed in Table 2. It can be noted that *Clostridium botulinum* Type F spores are the most resistant in that they require a 2.5 kGy dose to achieve a 90% reduction, whereas *Escherichia coli* vegetative organisms can be reduced the same amount with only a .09 kGy dose.

From a practical, processing viewpoint it is important to identify the types of organisms that are the target of irradiation, whether the damage they do is a matter of spoilage or is pathogenic in nature. This enables reasonable estimates to be made of the required doses to achieve the desired result. These estimates, in turn, can then be confirmed by product testing.

Two primary concerns have been raised over the years regarding the application of sub-lethal doses to organisms in food. The question of sub-lethal doses is a very real one, as the current international recommended dose maximum is only 10 kGy. For example, referring to Table 2 and assuming a dose of 10 kGy, it will be noted that the population of *Salmonella typhimurium* would be reduced by a logarithmic factor of 50 and thus effectively eliminated, whereas the population of *Clostridium botulinum* Types A, B, E, and F would only be reduced by logarithmic factors of three to ten. Given enough *Clostridium botulinum* contamina-

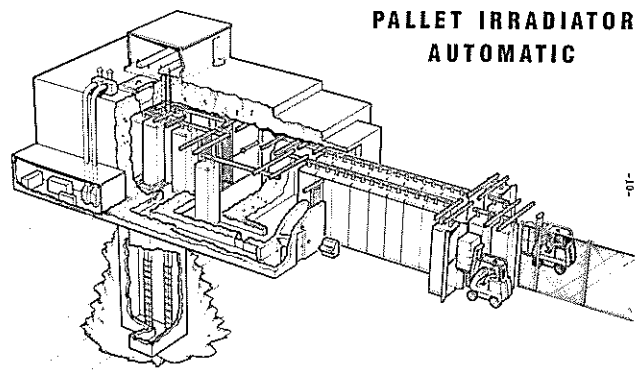

Figure 1 Automatic pallet irradiator

Table 2: D-Values for Irradiation of Some Microorganisms[1]

Microorganisms	*D-value (kGy)*
Vegetative Cells	
Escherichia coli	0.09
Pseudomenas	0.05–0.5
Salmonella typhimurium	0.2
Staphylococcus aureus	0.02
Micrococcus radiodurans	1.9–3.0
Vibrio parahaemolyticus	0.05–0.14
Spores	
Clostridium botulinum	
Type A	2.1–2.3
Type B	1.6–3.7
Type E	0.8–1.6
Type F	2.5

[1] Data taken from Banwart, 1979. [9]

tion in the product, it is quite possible that a number of *Clostridium botulinum* organisms would survive (i.e. a sub-lethal dose would have been applied).

The first concern is that the survivors might be mutant and might have increased resistance to radiation and/or would have more harmful characteristics. These mutant organisms could possibly multiply and be spread throughout the world. A second concern is that the survivors, whether mutated or not, would be able to thrive, unbridled by competition from the weaker organisms that were eliminated. If these thriving survivors happened to be pathogenic organisms, the product could appear to be unspoiled, having no deleterious (or undesirable) odours or taste, while in fact it would be microbially unsafe.

Both concerns were well addressed by the JECs and found to be unsupportable. In addition, other authoritative bodies have examined these concerns and supported the findings of the JECs. For example, the Board of the International Committee on Food Microbiology and Hygiene of the International Union of Microbiological Societies addressed these issues at its 1982 meeting and concluded that there was no cause for concern. An excellent summarizing statement, quoted in full below, is taken from the 1986 *Report on the Safety and Wholesomeness of Irradiated Foods*, [10] prepared by the U.K. Advisory Committee on Irradiated and Novel Foods:

5.14 The Panel considered the extensive data on this point and concluded that radiation-induced mutations were not likely to constitute a microbiological hazard. Most mutations in micro-organisms induced by ionizing radiation or by any other means are deleterious to the organism, and will be associated with a decrease in pathogenicity. Furthermore, there is no selective pressure to encourage the enhanced survival in foods of strains of increased virulence. There is no experimental evidence for irradiation producing an increase in pathogenicity, and there are no reports of any greater difficulty in identifying the surviving micro-organisms in irradiated food by standard microbiological techniques than in an unirradiated population. Radiation-resistant mutants of micro-organisms can be selected out under laboratory conditions involving many repeated doses slightly below the lethal dose, but such conditions would not occur in any foreseeable applications of food irradiation.

5.15 We consider that there are considerable benefits from the microbiological effects of food irradiation, both in terms of the extension in shelf life resulting from the reduced number of spoilage organisms in irradiated foods, and in terms of potential health benefits from the fact that food irradiation constitutes an effective addition to the methods of controlling pathogenic organisms in food. We also concur with the Panel that the use of food irradiation is not likely to present any microbiological hazard to the consumer. Although irradiation up to an overall dose of 10 kGy would not kill all pathogenic micro-organisms, and could allow continued growth of surviving pathogens, the same possibilities arise with all of the accepted non-sterilising methods of food processing, and we consider that standard techniques can be applied to determine microbiologically safe conditions for any particular application of food irradiation. Finally, we are satisfied that the ability of ionizing radiation to produce mutations in micro-organisms does not constitute a special microbiological hazard to the consumer.

Chemical Effects

In addition to controlling or eliminating micro-organisms, the ionizing energy is also absorbed by the food itself. The effect on the food is that a few of the molecules are split into new molecules. These new molecules are referred to as radiolytic products. The primary concern that arises from this fact is whether or not these radiolytic products are harmful.

The nature of the radiolytic products depends primarily on the chemical composition of the food itself. The quantity of such products generally increases with radiation dose, but can be modified by factors such as temperature, the presence or absence of air, and the water content of the food at the time of irradiation. However, the amount of energy absorbed by the food during irradiation is much less than that absorbed during heat processes such as canning. It is therefore not surprising that the amount of chemical change is smaller than in comparable heat processes. The 1976 JEC concluded that radiolytic products detected in the wide range of irradiated foods studied did not appear to pose any toxicological hazards in the concentrations at which they were detected [4].

Similarly, other studies of this issue by credible national authorities have reached the same conclusion. One of the most comprehensive of these studies was undertaken by the American Council for Agriculture Science and Technology (CAST), which reported in

July 1986 [11]. Among the many safety issues addressed in their report is the subject of Unique Radiolytic Products (URPS). Scientists in the field define URPS to be compounds that are formed by treating foods with ionizing energy, but which are *not* found normally in any untreated foods, and are *not* formed by other accepted methods of food processing. On the basis of this definition, the CAST report states: 'No unique radiolytic compounds have been found in 30 years of research. Compounds produced in specific foods by ionizing energy have *always* been found in the same foods when processed by other accepted methods, or in other foods.' [11]

The only concerns of substance with regard to radiation-induced chemical changes are *non-toxic* changes that affect the sensory attributes of the food and/or the nutritional quality. As previously discussed, some foods do not respond well to irradiation in that their organoleptic qualities change. This unwanted change is due in large part to the formation of non-toxic radiolytic products that impart different flavours and aromas. The key to avoiding these changes is to ensure that the appropriate radiation dose is applied under the correct conditions to a suitable product. Failure to do so may result in an unmarketable product, even though chemically safe. The second concern, the effect of radiolytic products/chemical change on nutritional quality, is addressed in the following section.

Nutritional Quality

This issue has also been thoroughly researched by reputable authorities, resulting in consistent findings. These can best be summarized by quoting the abstract of the 1978 paper by Dr E.S. Josephson *et al.* 'When foods are exposed to ionizing radiation under conditions envisioned for commercial application, no significant impairment in the nutritional quality of protein, lipid and carbohydrate constituents was observed. Irradiation was no more destructive to vitamins than other food preservation methods. Protection of nutrients is improved by holding the food at low temperature during irradiation and by reducing or excluding free oxygen from the radiation milieu.' [12]

When carbohydrates are irradiated, some splitting of complex compounds into simpler compounds occurs, the main effects of which are hydrolysis and oxidative degradation. Detailed studies have shown these changes to be of no nutritional significance.

The main reactions of irradiation of fats are oxidation, polymerization, decarboxylation, and dehydration. These changes are non-toxic and can be reduced, if desired, by various processing techniques. Regardless of the changes themselves, no significant effect on the digestibility of fat-containing foods has been found.

Proteins are affected similarly to carbohydrates in that complex protein molecules are broken into smaller protein molecules. However, these smaller protein molecules yield the same amino acids upon digestion as the larger unbroken molecules. No effects of significance with respect to nutritional quality have been found.

Vitamins are somewhat different in that some are virtually unaffected even by high radiation doses, whereas others are affected to the same degree as when other processing technologies are used. Two direct quotes from the 1986 CAST report provides a good summary of this aspect:

> Many experiments have been done on the effects of ionizing energy on vitamins. Some vitamins appear to be affected very little by ionizing energy. Vitamin K, for example, appears to be relatively stable. A significant proportion of the vitamin C may be changed to dehydroascorbic acid, but this compound has almost the same vitamin C value as ascorbic acid, which is vitamin C itself. Tocopherols, which are antioxidant compounds with vitamin E activity, seem to be especially sensitive to ionizing energy in the presence of oxygen, as would be expected from their antioxidant properties. Vitamins are sensitive also to processing by heat. Research on vitamin B6 has shown less destruction of this vitamin in products sterilized by ionizing energy than by heat. Vitamin retention in food is greatest when the processing with ionizing energy is carried out at low temperatures in the absence of oxygen.
>
> Exposure of food to ionizing energy is somewhat destructive of vitamins, but no more so than are other food preservation methods used commercially. [11]

A significant summary statement with respect to the absence of deleterious effects on nutritional quality of foods irradiated at low doses for insect disinfestation and shelf-life extension is given in the U.S. Federal Register, which contains the new F.D.A. regulation '21 CFR Part 179 – Irradiation in the production, processing and handling of food: Final Rule':

> 4. Destruction of Nutrients
>
> 12. Several comments stated that destruction of nutrients should be a concern in this rulemaking. The comments stated that many vitamins are light or heat sensitive, and that irradiation will destroy them. One comment stated that nutritional problems may develop for consumers because of nutrient loss when an entire class of foods is irradiated.
>
> 13. The proposal discussed this issue and explained that the available literature indicated that there are no nutritional differences between unirradiated food and food irradiated at levels below 1 kGy (100 krad). The minor ingredients allowed to be irradiated at higher doses are not sources of nutrients. Therefore, the agency believes it is appropriate to conclude that destruction of nutrients

is not an issue in this rulemaking. There have been no additional data submitted to change this conclusion. [6]

Commercialization

If it can be accepted, based on statements from respected scientific and government health authorities the world over who have examined them, that (1) irradiated foods are safe and wholesome, and that (2) facilitative regulations are steadily being developed and promulgated, what are the prospects for increased commercial use of food irradiation technology?

Today, more food is being irradiated and consumed than ever before. The quantity, however, is not huge; the IAEA estimate for 1985 is only 350,000 metric tons. Nevertheless, annual quantities are increasing significantly, as are the number of food irradiators. In the two-year (1986 and 1987) period AECL Radiochemical expects to install three full-scale and five pilot-scale, or small-upgradeable food irradiators. The U.S. Department of Energy has announced plans to construct six strategically located pilot-scale food irradiators. The purpose of these demonstration facilities is to provide the U.S. food industry with readily accessible test facilities that are able to process significant quantities of foods.

For the first time ever, in September 1986, an irradiated fruit was allowed into the U.S.A. for general public consumption. A Puerto Rican mango grower teamed up with the world's leading contract radiation processing firm, Isomedix Inc., to conduct a market trial of irradiated mangoes in Miami. The test was considered a success by all involved, including the retailer who sold the irradiated mangoes faster than locally grown, non-irradiated mangoes. The retailer stated that his customers are more interested in the appearance, quality, and taste of his products than in the way they were treated.

This positive result confirms several North American consumer attitude surveys conducted in the past few years. These studies report that 20–35% of consumers are ready to purchase irradiated foods, 60–70% want more information, and 5–15% are not going to purchase initially. Obviously, 20–35% of consumers is enough to result in good business for the grower, distributor, and retailer. Isomedex Inc. plans to process commercial quantities of mangoes for sale in the U.S. during the 1987 mango season.

Given the reality that very little co-ordinated consumer education efforts have been made yet in North America, and that an extremely vocal anti-food-irradiation lobby has been actively opposing the process for the past 18 months, the Miami mango test results appear even more positive. Pro food irradiation organizations in Canada and the U.S.A. are being organized, and are just beginning to distribute factual information to the consumer. The U.S. Institute of Food Technologists has included food irradiation in their active program of media/public relations. The U.S. Coalition for Food Irradiation, sponsored by the National Food Processors Association, includes membership from major U.S. food companies. This Coalition has hired a professional Public Relations firm (February, 1986) to assist in providing factual information to consumers, the media, and politicians. In Canada, the newly formed Canadian Advisory Committee on Food Irradiation (June, 1988), initiated by Agriculture Canada, will provide a standing forum for industry, government, and consumer representatives to discuss key issues. A spin-off of this forum will almost certainly be the formation of an industry-led organization to inform the Canadian consumer and the media.

Regardless of commercial progress elsewhere, in particular in large developing countries like China, the world, for the time being, still looks to North America for the lead. The North American food industry is, in general, well informed, and many major companies have active Research and Development programs on food irradiation. However, before these firms make significant moves toward using the technology, they want more proof of consumer acceptance. Over the next few years, a combination of factors will likely result in greater acceptance of food irradiation. The effective banning of ethylene dibromide fumigant in the U.S.A. and Canada, and the likelihood of more countries doing the same, has left many fruit and vegetable producers with no effective alternative except irradiation. Increased, co-ordinated, factual and authoritative consumer education will gradually have a positive effect on consumer attitudes. Additional market trials of significant quantities of irradiated foods will be conducted and will provide hard evidence that the consumer can differentiate between Chernobyl and food irradiation. Some firms (the pioneers) will want to gain a marketing advantage and will decide 'to go for it,' if for no other reason than to capture the business of the 20–35% of consumers that will buy irradiated food now.

The prospects for regulatory and consumer acceptance of irradiated foods world-wide are good, as are the prospects for commercial use. The technology itself is well understood. Good progress has been made in terms of the quantities and range of foods being irradiated. In short, the prospects for the commercialization of food irradiation technology have never been better.

Summary

The scientific work done since the early 1950s is well documented and comprehensive, as it relates to safety and nutritional issues. It has also resulted in a good definition of applications that work and those that don't. However, there are several areas that call for more attention by the scientific community, in particular the generation of additional data on the safety of

foods irradiated to doses higher than the 10 kGy Codex-recommended maximum average, and the generation of data on treatments involving irradiation in combination with other technologies and processes.

For the food industry and the consumer, the future commercialization of food irradiation technology is only a matter of how quickly it will come about. The benefits in terms of reduced food losses, improved maintenance of quality, and reduced incidence of pathogenic contamination are too compelling to be ignored or rejected.

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Technetium-Iron Oxide Reactions Under Anaerobic Conditions: A Fourier Transform Infrared, FTIR Study

R.I. Haines and D.G. Owen
Research Chemistry Branch

T.T. Vandergraaf
Geochemistry and Applied Chemistry Branch
Atomic Energy of Canada Limited
Whiteshell Nuclear Establishment
Pinawa, Manitoba
R0E 1L0

Abstract

The behaviour of technetium in the geosphere is important in nuclear fuel waste management studies because this man-made element has a long half-life and, under ambient conditions in the laboratory, is not readily sorbed on geological materials. Fourier transform infrared spectroscopy studies have shown conclusively that the reaction between TcO_4^- and Fe_3O_4 at room temperature occurs via a surface reduction of TcO_4^- to TcO_2, the latter being precipitated at the Fe_3O_4 surface. These results confirm that technetium can be contained effectively by magnetite in the geosphere, provided reducing conditions can be maintained.

Résumé

Le comportement du technetium dans la geosphere est d'une importance particuliere dans les études de gestion des dechets de combustible nucleaire du fait que cet element artificiel a une longue periode radioactive et que, dans les conditions ambiantes du laboratoire, il ne sorbe pas facilement sur les materiaux geologiques. Les études de spectroscopie de l'infra-rouge a transformation de Fourier ont montre d'une maniere concluante que la reaction entre TcO_4 et Fe_3O_4 se produit a la temperature ambiante par une reduction de surface de TcO_4^- a TcO_2, ce dernier precipitant a la surface de Fe_3O_4. Ces resultats confirment que la magnetite de la geosphere peut confiner efficacement le technetium a condition de pouvoir maintenir les conditions de reduction.

Introduction

The fission of ^{235}U and ^{239}Pu in a nuclear reactor produces a large number of radioactive products. Most of these decay to stable isotopes within a few years after the fuel has been discharged from the reactor and, therefore, pose no problem in the long-term management of nuclear fuel wastes. There are, however, a number of long-lived radionuclides that must be considered in assessing the environmental impact of any nuclear fuel waste disposal vault in the geosphere. For example, the fission products technetium and promethium are unique in that they do not have any stable isotopes and are expected to occur in nature only in minute amounts as a result of spontaneous fission of uranium. While promethium has a number of chemical analogues in the other rare earth elements, this is not the case for technetium, and therefore it is difficult to predict its behaviour in the geosphere.

The technetium isotope of interest for nuclear fuel waste disposal is ^{99}Tc. It is a pure β-emitter (E = 0.293 MeV) with a half-life of 2.13×10^5 a. Its high fission yield of 6% accounts for the relatively high concentration (approx. 0.02% by weight) [9] in fuel discharged from a CANDU reactor (burnup = approx. 650gJ/kg U).

Technetium is a Group VII B element. Its chemical behaviour is not well known, but it is expected to fall between that of manganese and rhenium. Figure 1 [25] summarizes the aqueous technetium species possible at 298 K under a wide range of redox potential (E_h) and pH values. Under oxidizing conditions, technetium exists in solution in the +7 oxidation state as the anionic species TcO_4^-, and shows little sorption by geological materials [3, 14, 18, 23]. For this reason, in previous safety and environmental assessments of geological disposal of nuclear fuel wastes, technetium had been assumed to travel at the same rate as moving groundwater [28].

Under reducing conditions, or in the absence of oxygen, sorption of technetium has been noted in some cases. Bondietti and Francis [4] reported the removal of technetium from a nitrogen-sparged solution by basalt and granite, and Allard *et al.* [3] reported

Keywords: technetium, FTIR, sorption, redox, radionuclide, magnetite, hematite, anaerobic.
Please address correspondence to R.I. Haines at the address above.

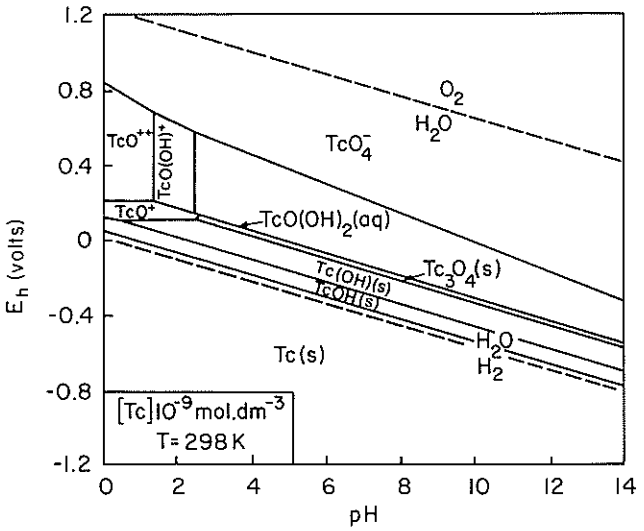

Figure 1 E_h–pH diagram for technetium speciation in aqueous solution at 298K. (E_h = equilibrium redox potential, in volts).

k_d values (defined as the ratio of sorbed to nonsorbed concentrations) well above zero, indicating sorption on Fe(II)-containing minerals in contact with deaerated solutions, and on granite in a deaerated solution containing Fe(II) ions. Vandergraaf *et al.* [25] showed that iron oxide fracture infilling materials in altered granite and iron oxide inclusions in biotite are responsible for the removal of technetium from solution under reducing conditions. Walton *et al.* [26] have reported that sorption of a reduced technetium carbonate complex occurs on iron oxyhydroxides. Meyer *et al.* [15] showed sorption of technetium under anoxic conditions from a 0.1 mol.dm^{-3} NaCl solution on hematite, ilmenite, and iron-containing microcline. The same authors [16] also reported sorption of technetium in recirculation column experiments using basalt, hematite, ilmenite, and mordenite, and stated that the reduction of TcO_4^- to Tc(IV) or other oxidation states is a complex process. Additional studies [19] have shown that Tc(IV) forms a complex in carbonate solution under reducing conditions, which has a much higher solubility than the sparingly soluble technetium(IV) oxide. Walton *et al.* [26] show that this technetium(IV)-carbonate complex is in turn strongly adsorbed by iron oxides. Allard [2] has suggested that under strongly reducing conditions, the formation of Tc(O) is possible.

In this paper, we describe a study of the interaction of pertechnetate ion with magnetite and hematite surfaces in aqueous anaerobic solution, using Fourier transform infrared spectroscopy to characterize the nature of the interactions of the radionuclide with the iron oxide surfaces. Evidence is presented for the formation of technetium(IV) oxide in the reaction of pertechnetate ion with magnetite, and a possible mechanism is discussed.

Experimental

Solutions of ammonium pertechnetate were prepared by dilution from a stock solution (Amersham-Searle) with doubly distilled de-ionized water. Technetium-(IV) oxide ($TcO_2.2H_2O$) was prepared by electrochemical reduction of the pertechnetate anion. Magnetite powder was synthesized under ambient conditions from ferrous solutions at pH 9 [10]. Hematite was prepared by the hydrothermal decomposition of ferric chloride [8].

Analysis for the TcO_4^- anion was performed by UV/visible spectrophotometry (λ_{max} (TcO_4^-) = 244 nm; ϵ_{max} = 6190 mol^{-1} dm^{-3}cm^{-1}) using a Hewlett-Packard model 8450A diode-array spectrophotometer. Trace concentrations were determined with a Tennelec LB-5100 β-counter, calibrated with a U.S. National Bureau of Standards ^{99}Tc standard source. The purity of the iron oxides was confirmed by X-ray diffraction analysis.

Fourier transform infrared, FTIR spectra were measured using a Nicolet 10 MX single-beam spectrometer under nitrogen purge, using a 0.2-g-KBr disc as a reference. A Globar (Reg. trademark) source, a room temperature deuterated triglycine sulphonate, DTGS, detector and CsI optics were used to collect data from 300 to 1500 cm^{-1} at 2 cm^{-1} resolution. Thirty-two scans were signal-averaged for most spectra, with no smoothing of data.

Because of the tendency of technetium(IV) oxide to build up static charge in the dry state, the infrared spectrum of the electrochemically reduced technetium was measured using a nujol mull between KBr plates. Transmission FTIR spectra of the TcO_4^- anion were measured using the KBr disc method. Since preliminary measurements of spectra of $KTcO_4$ using pressed CsI discs showed changes in the far infrared (300 to 400 cm^{-1}) lattice vibrations, which indicated the solid state metathesis of $CsTcO_4$, KBr discs were used for subsequent measurements. Aliquots of stock pertechnetate solution were added to accurately weighed, 0.2 g portions of anhydrous spectroscopic-grade potassium bromide (Aldrich). The resulting solutions were evaporated in an oven at 363K overnight and desiccated over phosphorus pentoxide at room temperature for 72 hours. The crystalline residues were ground in an agate mortar and pestle and pressed into 1-cm-diameter discs in a stainless steel die at 280 MPa for 2 minutes at room temperature.

The sample preparation for iron oxide–pertechnetate interactions followed that described above, except that, prior to the disc-pressing step, 1-mg amounts of magnetite or hematite were intimately mixed with the dry TcO_4^-/KBr mixture. Spectra of these samples represent physically admixed reagents. To study the interaction of TcO_4^- with the oxide surfaces in aqueous solution, the discs were taken up in 2 cm^3 of nitrogen purged water in an anaerobic chamber

(Vacuum Atmospheres DL 002-SP). The oxygen level in the atmosphere in the chamber was maintained below 2×10^{-5} % by volume, as determined with a Teledyne model 317-X trace oxygen analyzer. The resulting solutions were allowed to stand, with periodic agitation, for up to 14 days at room temperature. The solutions were centrifuged, and the iron oxide residues collected, dried and pressed with approx. 0.2 g KBr. To remove residual excess non-sorbed technetium from the samples, these discs were redissolved in distilled, deionised water, and three centrifugation/washing cycles were performed. The final residues, which contained only technetium species that were irreversibly adsorbed on the iron oxide surfaces, were pressed with a further 0.2 g KBr.

Results and Discussion

Figure 2 shows a typical FTIR spectrum of potassium pertechnetate (KBr disc) in the 1200 to 300 cm^{-1} range. The three bands between 300 and 400 cm^{-1} are due to lattice vibrations in the $KTcO_4$ crystals [21]. The absorbance at 912 cm^{-1} is a composite of ν_1 and ν_2 vibrations of the tetrahedral TcO_4^- anion of T_d symmetry [6]. This peak is indicative of the Tc(VII)-O stretching vibration, and may be used as a qualitative indicator of the presence of Tc(VII) adsorbed onto the iron oxides. Quantitative measurement of adsorbed technetium is not feasible, since interaction of the tetrahedral anion with the hydrous oxide surfaces (e.g. of hematite) would lower the symmetry. The resulting C_{3v} (for monodentate co-ordination) or C_{2v} system (for bidentate co-ordination) would show greater multiplicity in the Tc-O band [17], with different extinction coefficients [11, 13] from the original composite peak centred at 912 cm^{-1}. The electrochemically-reduced technetium(IV) oxide gave a spectrum with a Tc-O stretching vibration band at 896 cm^{-1}

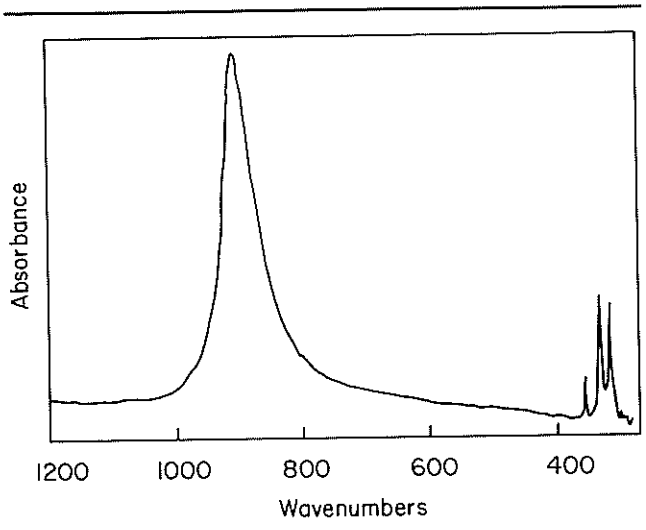

Figure 2 Typical FTIR spectrum of $KTcO_4$ (KBr disc).

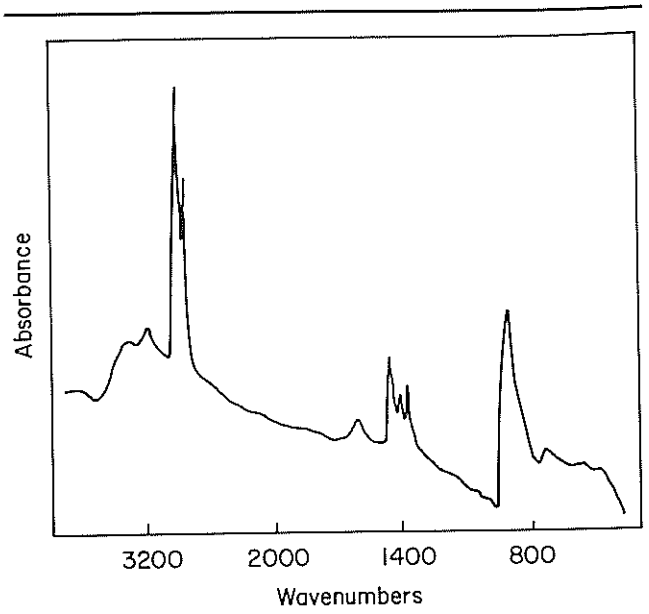

Figure 3 FTIR spectrum of technetium(IV) oxide (nujol mull).

(Figure 3). The mid-infrared spectrum of this oxide has not been reported previously.

Comparisons between the FTIR spectra of physical admixtures of pertechnetate and the iron oxides, and the oxides themselves are illustrated in Figures 4a and 4b. The spectral bands of the iron oxides do not mask those due to Tc-O stretching vibrations between 890 and 920 cm^{-1}. Subtraction of normalized digital spectra

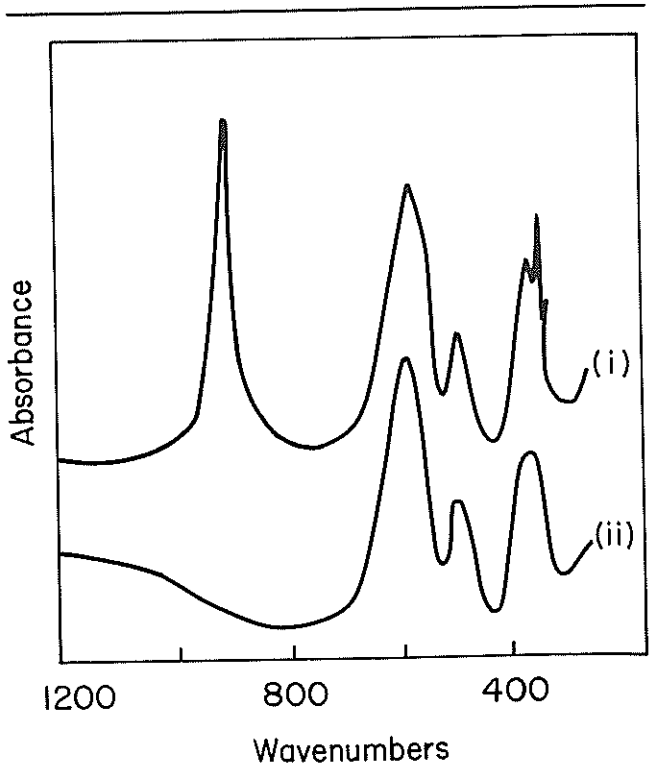

Figure 4a FTIR spectra of hematite (i) with and (ii) without 0.1 cm^3 of stock NH_4TcO_4 solution (see text).

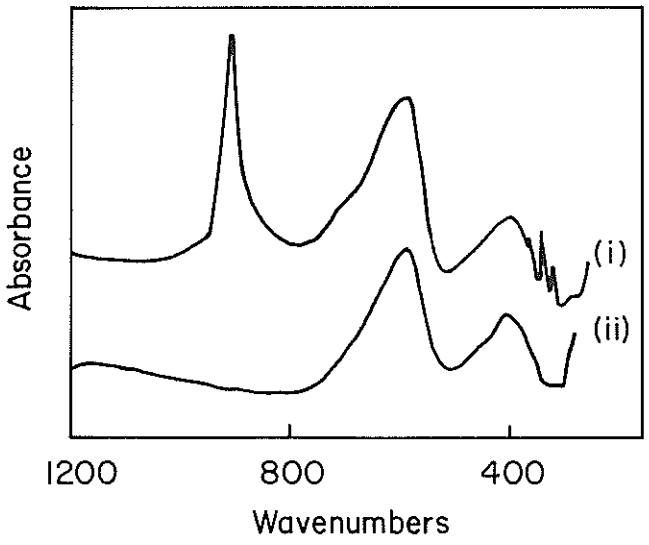

Figure 4b FTIR spectra of magnetite (i) with and (ii) without 0.05 cm^3 of stock NH_4TcO_4 solution (see text).

of the separate components from the mixtures show additive behaviour, indicating no chemical interaction between the reagents in the dry state.

To study the interaction of aqueous pertechnetate anion with the iron oxide surfaces, nitrogen-purged water was added to the physical admixtures, producing a technetium concentration of 10^{-3} mol.dm^{-3}. Infrared spectra of magnetite and hematite showed the presence of significant amounts of residual TcO_4^-, by the presence of the Tc-O vibration at 912 cm^{-1} (Figure 5a). Copious washing of the hematite residues completely removed the band at 912 cm^{-1} from the spectra. Hence, if physisorption of the pertechnetate ion onto hematite occurs, the process is readily reversible. Anions are known to adsorb onto hydrous oxide surfaces [12, 27]; here a ligand exchange mechanism occurs, with replacement of surface hydroxyl groups by the incoming anion. The pertechnetate anion appears to behave as a simple non-co-ordinating species. The development of surface charge on the iron oxides stems from surface hydration in aqueous suspension, followed by dissociation of the surface hydroxyls [20]. At the zero point of charge (zpc), the surface is neutral. At a pH < zpc, the surface becomes protonated, resulting in positively-charged particles. A negative charge is generated at pH > zpc, by acidic dissociation of surface hydroxyl groups [22]. The mean zpc of hematite samples with varying histories has been determined as 8.5 ± 0.5 [5]. The sorption experiments were performed using neutral water containing 10^{-3} mol.dm^{-3} NH_4^+ (as the pertechnetate salt). This will result in a solution buffered to a pH of 6. Under these conditions, the hematite suspension is expected to have a positive charge, facilitating reversible physisorption of the TcO_4^- anion.

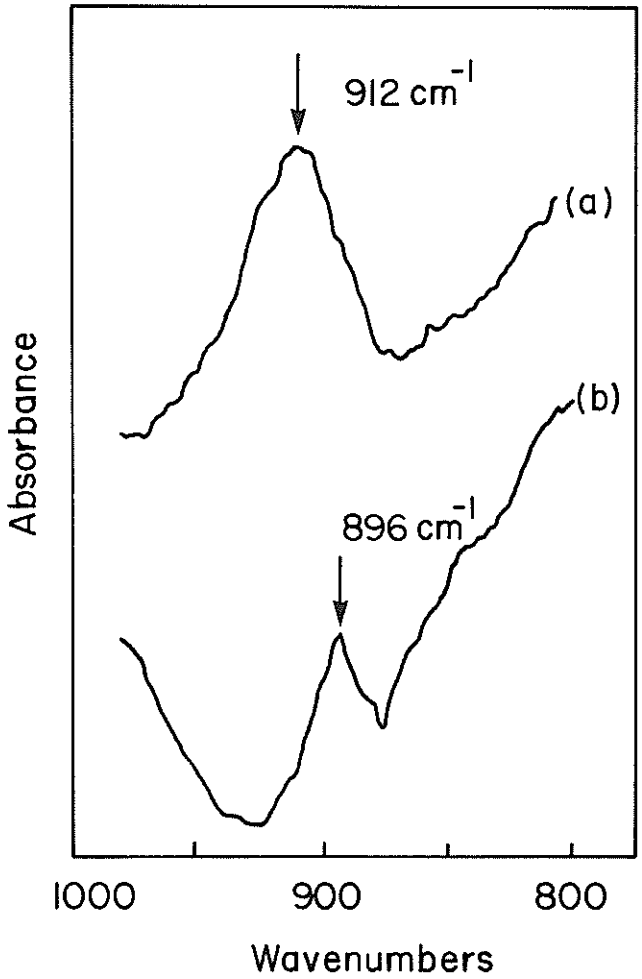

Figure 5 (a)Tc-O vibration region of the infrared spectrum of hematite after reaction with aqueous TcO_4^-, before removal of the excess NH_4TcO_4. (b)As Figure (a), for a magnetite sample after removal of the excess NH_4TcO_4.

In the reaction between pertechnetate anion and magnetite in aqueous solution under anaerobic conditions, similar residual TcO_4^- was found to be present on the centrifuged Fe_3O_4 particles. After repeated washing of the residue, the infrared spectrum of the dried magnetite was measured. Figure 5b shows that no TcO_4^- remained (indicated by the absence of a band at 912 cm^{-1}), but a band at 896 cm^{-1} was observed. This band coincides with that observed for technetium(IV) oxide (Figure 3). Literature values of the zpc for magnetite lie in the range 6.5 ± 0.2 [1, 22, 24]. Thus, the magnetite particles in the sorption experiments are expected to have a slightly positive charge. Some electrostatic attraction between the TcO_4^- anions and the magnetite surface is expected.

The relative redox potentials for the Fe(II)/(III) and the $TcO_4^-/TcO_2.2H_2O$ couples in aqueous solution [7] favour the reduction of Tc(VII) to Tc(IV) by Fe(II). The redox reaction in homogeneous solution is slow, but may be catalysed at the magnetite surface.

Conclusions

The results of this study indicate that under neutral anaerobic conditions, technetium undergoes irreversible adsorption at magnetite surfaces. The infrared observations confirm that the resulting technetium species is no longer the pertechnetate anion, but is, in fact, a Tc(IV) oxide. The mechanism of the reaction is envisaged as proceeding via a positive electrostatic attraction by the magnetite surface for the TcO_4^- anion at pH values below the zpc. A surface redox step between Fe(II) centres at the surface of the oxide and the Tc(VII) species, produced the sparingly soluble Tc(IV) oxide, which is precipitated at the magnetite surface. The buildup of a layer of the Tc(IV) species, coupled with the concomitant oxidation of the Fe_3O_4 surface to Fe_2O_3 serves to inhibit further reduction of TcO_4^-. Thus, the extent of reduction of Tc(VII) by Fe_3O_4 is expected to be dependent upon the surface area of the oxide. This work provides the first known spectroscopic evidence for the reduction of Tc(VII) to Tc(IV) by an Fe(II)-bearing oxide, and indicates that, under the conditions postulated to occur in and around a used-fuel disposal vault, the mobility of technetium may be severely restricted. Further work is in progress to clarify the effects of carbonate on the mobility of technetium in the geosphere.

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Progress of the Swedish Radioactive Waste Management Program

Hans G. Forsström
Swedish Nuclear Fuel and Waste
Management Company
Stockholm, Sweden

Abstract
The implementation of the Swedish radioactive waste management system is making good progress. A central interim storage facility for spent nuclear fuel and a sea-based transportation system for spent fuel and other radioactive residues are already in operation. An underground final disposal facility for low- and medium-level waste from reactor operation is under construction. For the remaining steps (final disposal of spent fuel and long-lived radioactive residues) one concept has been developed and approved. Research and development work is continuing, to provide the basis of a final decision on disposal method and site for around the year 2000. This decision will be based on a broad evaluation of available alternative designs.

Introduction

The safe and efficient management of radioactive waste from nuclear energy production in Sweden is the responsibility of the owners of its nuclear reactors. This responsibility also includes financing of the total costs. To fulfil their obligations, the four utilities that operate nuclear reactors in Sweden have formed a special company, the Swedish Nuclear Fuel and Waste Management Company (SKB). SKB is responsible for all handling, transportation, and storage (temporary and permanent) of spent fuel and radioactive waste from nuclear power plants. Furthermore, SKB is responsible for the planning and construction of all facilities required for the management of spent fuel and radioactive waste and for the comprehensive research and development work necessary to provide such facilities.

The Swedish Nuclear Power Program

Sweden's nuclear power program consists of 12 nuclear reactors located at four different sites, with a combined capacity of 9,650 MW of electric power. In 1985 the last two reactors of this program, Forsmark 3 and Oskarshamn 3, reached full power and were taken into commercial operation. The nuclear power plants generated 42% of Sweden's total electric power in 1985. In 1986 this figure is expected to increase to 45–50%, the remainder being hydro-power (Figure 1).

According to a decision by Parliament, no more

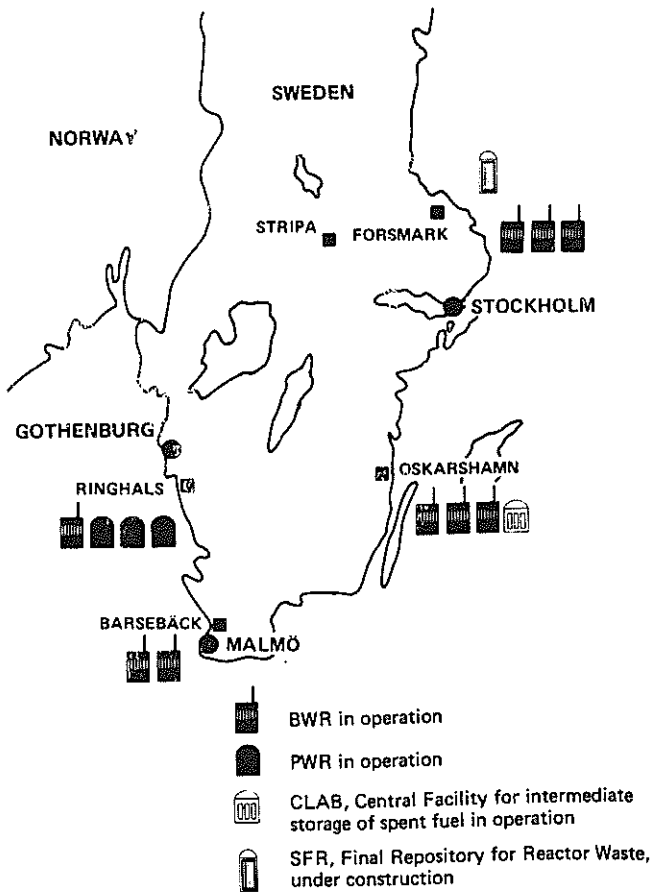

Figure 1 Swedish nuclear power reactors.

Keywords: nuclear waste management, storage, disposal, spent fuel management, Sweden, decommissioning, costs.

reactors are to be built in Sweden, and the existing ones shall not be operated beyond the year 2010. This decision was taken after an intense debate on the nuclear issue during the 70s and a referendum in 1980. The exact time-schedule for the phasing out of nuclear power has yet to be established. In the wake of Chernobyl, the debate on nuclear power has revived and the time-schedule is at present being evaluated by a government commission.

The influence of this evaluation on the work performed within the waste management program is, however, expected to be limited. It will only have an impact on the size of the facilities to be constructed.

Radioactive Waste Management in Sweden

A complete system has been planned for the management of all radioactive residues from the 12 nuclear reactors and from research facilities. The radioactive residues generated by the operation of the Swedish reactors consist of spent nuclear fuel and various kinds of low- and medium-level reactor waste. In addition, decommissioning waste will be generated in the future when all reactors are decommissioned and dismantled.

The accumulated amounts of waste from the operation of the 12 Swedish reactors up to the year 2010 have been estimated at about 7,800 metric tons (calculated as uranium) of spent fuel, 90,000 m^3 of low- and medium-level reactor waste, and around 115,000 m^3 of decommissioning waste. In addition to this, approximately 10,000 m^3 of low- and medium-level waste will be produced at the research facilities in Studsvik, and from the use of radio-elements in industry, medicine, and research.

An overview of the management scheme for these radioactive residues is shown in Figure 2. The basic strategy of the Swedish radioactive waste management

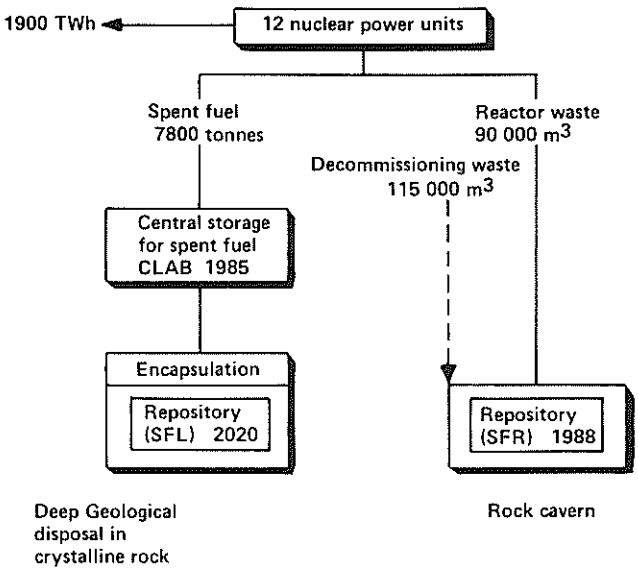

Figure 2 The Swedish waste management system.

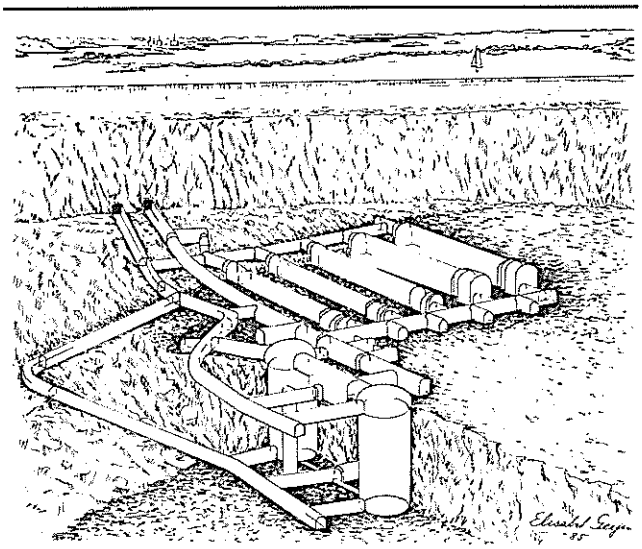

Figure 3 Overview of tunnels and storage chambers in SFR.

is that short-lived waste (< 500 years) should be deposited as soon as feasible without interim storage; whereas for spent fuel and other long-lived wastes, an interim storage of 30–40 years is foreseen, prior to final direct disposal in crystalline rock formations. A central facility for interim storage of spent fuel (CLAB) has been in operation since July 1985.

This once-through strategy has been judged to be the most rational and cost-effective solution in Sweden under the prevailing conditions. It is also, at present, the politically preferred option. In addition, the fairly long period of time that is assumed for interim storage of spent fuel provides freedom of choice among various management options up to the year 2000, when a final decision has to be made.

Management of Low and Medium Level Waste

Both the reactors and the CLAB facility will produce operational waste with a relatively short life, such as ion exchange resin, filter materials from water cleaning systems, etc. The low- and medium-level waste is conditioned at the reactor sites and at Studsvik. Ion exchange resins and filter materials are packaged and solidified, or dewatered, to facilitate further handling. Concrete and bitumen are currently employed for solidification. Most of the combustible waste is sent to Studsvik for incineration [1].

The operational waste is at present being stored in specially built facilities at the reactor sites and at Studsvik, but will, from 1988 onwards, be transported to the central repository for final disposal of low- and medium-level reactor waste, called the SFR (Figure 3). Some very low-level wastes will be disposed of by shallow land burial at the reactor sites.

Final Repository for Reactor Waste – SFR

The waste generated by reactor operation, some of

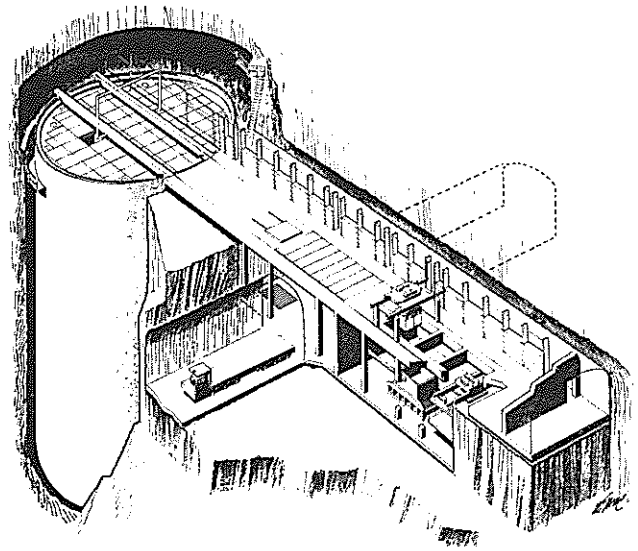

Figure 4 Handling of waste in SFR silo.

which has to be isolated for up to 500 years, will be disposed of in crystalline rock in a central underground facility known as the SFR. It is currently under construction near the Forsmark nuclear power plant, 200 km north of Stockholm [2].

The entire repository is situated in rock 50 metres below the sea bed, about 1 km outside the Forsmark harbour. The repository consists of rock caverns of various designs, depending on the type of waste to be disposed of (Figure 4). The medium-level waste, which contains most of the activity, will be deposited in a silo-like concrete structure cast inside a cylindrical rock cavern, and the waste will be isolated from the surrounding rock by concrete walls and a layer of clay backfill (bentonite) between the silo and the rock. The low-level waste will be deposited without extra barriers in the rock caverns designed for the particular type of containers being used for such waste.

When the repository is filled, the entrance tunnels will be plugged with concrete to seal the caverns. When the entire repository has been sealed, the transportation tunnels will also be sealed at ground level to prevent future access. After sealing of the repository, no further surveillance will be needed.

The SFR will accommodate all the 90,000 m^3 of low- and medium-level waste generated in Sweden up to the year 2010. The first phase, now under construction, will have a capacity of 60,000 m^3. Scheduled for commissioning in 1988, the repository is planned to be extended at the end of the 1990s. Later, when the reactors have been decommissioned, another extension will be made to accommodate the decommissioning waste.

The location of the SFR below the sea bed is advantageous from the point of view of safety. Under the horizontal seawater table, the driving forces of the ground water flow in the fissures of the rock are almost eliminated, and very little transportation of radioactive substances can take place in the almost stagnant water around the repository.

Management of Spent Fuel

The most important and crucial part of the waste management system is the handling and disposal of spent fuel. The main features of the Swedish once-through scheme for direct disposal are briefly as follows:

After discharge from the reactor, the spent nuclear fuel is stored in the storage pools at the power plants for at least a six-month period. The fuel is then transported to the central interim storage facility for spent fuel, the CLAB. Here the fuel will be stored for a period of 30–40 years, and thereafter transported to encapsulation and final disposal.

CLAB – Interim Storage Facility for Spent Fuel

In the CLAB, which was placed in operation in July 1985, the principle of wet storage in pools is applied. The CLAB facility (Figure 5) is situated on the east coast of Sweden, adjacent to the Oskarshamn power plant [3]. The facility constitutes a fundamental strategic element in the Swedish spent-fuel management scheme. It will ensure uninterrupted nuclear power production, and it will provide ample time for R&D work, site selection, system design, and optimization for the development of a permanent repository.

The facility consists of underground storage pools in a rock cavern and a receiving building on the surface. The storage pool, now built, can hold 3,000 tonnes of spent fuel, which will cover Swedish needs up to the mid-1990s. One or two additional expansions will be required to accommodate all the spent fuel and core components from the 12 Swedish reactors: about 8,000 tonnes. The total receiving capacity is 300 tonnes per year, or about 100 spent-fuel shipping casks (Figure 6).

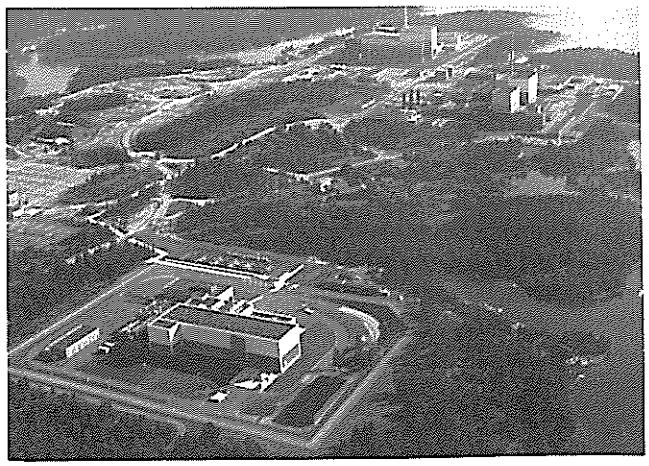

Figure 5 Aerial view of CLAB at Oskarshamn.

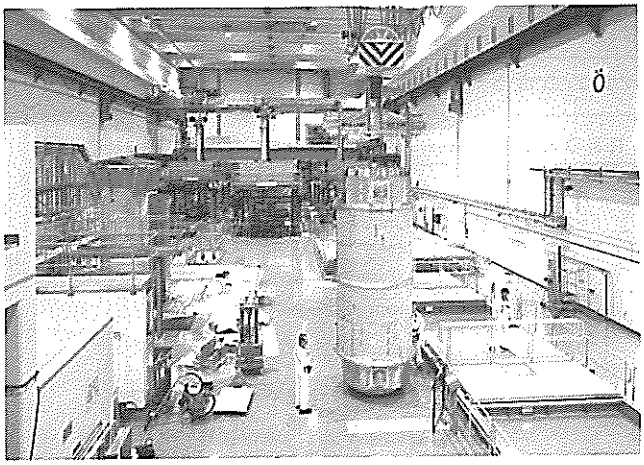

Figure 6 Handling of cask at CLAB.

Transport System for Spent Fuel and Radioactive Waste

All the Swedish nuclear power stations, as well as storage facilities, are located on the coast. It has therefore been deemed expedient to develop a sea-transportation system for the nuclear waste. A ship (Figure 7) was launched in 1982 that can take 10 fuel casks, each with a weight of about 80 tonnes. Each cask can hold 17 BWR or 7 PWR fuel assemblies. The fuel is transported dry and cooled by natural air convection. The casks are mounted on transport frames that are handled by a terminal vehicle. The same equipment will also be used to transport low- and medium-level waste in steel containers. Each container has a transport weight of up to about 120 tonnes.

Final Repository for Spent Fuel

In accordance with present plans, facilities for the final disposal of spent fuel and long-lived waste will not be required until the year 2020. Consequently, construction work has not yet commenced on these facilities. Extensive research on the matter has, however, been performed in Sweden since the mid-70s. It has resulted in concepts for the final disposal of waste from reprocessed fuel (KBS-1), as well as for the direct disposal of spent fuel without reprocessing (KBS-3) [4, 5]. These concepts have been evaluated by the government and have been found to be acceptable with regard to safety and radiation protection. Based on these evaluations, the government has granted loading permits for six reactors in 1979 and 1984.

However, before a decision regarding the construction of a repository is taken around the turn of the century, other concepts will also be explored. The purpose of this is to develop a disposal system that fulfills the requirements of safety and is optimized from a technical and economic point of view. The study of alternatives is thus in the interest of the utilities. It is also required by law.

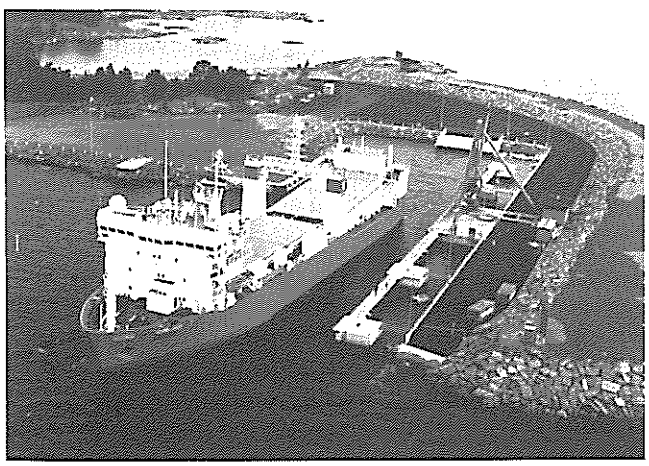

Figure 7 M/S Sigyn.

The KBS-3 Concept

Final disposal of spent nuclear fuel according to KBS-3 is shown schematically in Figure 8. The fuel is encapsulated in copper canisters, that are deposited 500 metres down in the crystalline bedrock. The deposition is made in bore holes at the bottom of tunnels, one canister in each bore hole. In the bore hole the canister is surrounded by a buffer of bentonite clay. When the repository is full, all shafts and tunnels are backfilled and sealed.

The safety of the KBS-3 disposal system is based on the multibarrier principle, by which the waste is

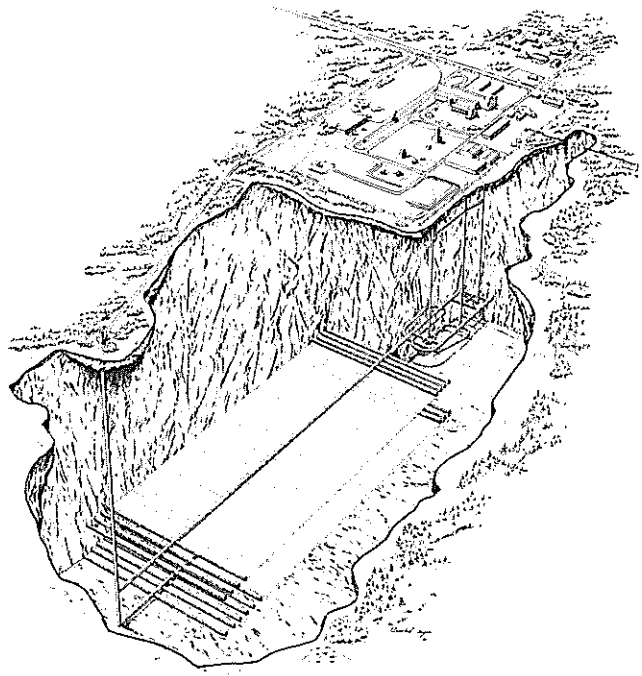

Figure 8 Final disposal according to KBS-3.

isolated from the environment by a number of mutually redundant barriers. The barriers are:

1 the rock mass, with its slow water transport and high capacity for absorption of radionuclides;
2 the almost impermeable bentonite buffer;
3 the highly corrosion-resistant copper canisters;
4 and the low solubility of the spent fuel.

Based on the results of the KSB-studies it may be concluded that:

1 Swedish bedrock is stable enough for a repository;
2 there exist in Sweden bedrock areas of acceptable size that are well suited for a repository;
3 the spent fuel can be surrounded by man-made barriers of long service life;
4 combinations of natural and man-made barriers can be designed to such a quality that the radiological impact on man will be insignificant.

Research and Development

In KBS-3, *one* feasible method has been demonstrated. Considerable work remains to be done in order to develop the optimal method, however. The future work will consist of:

1 continued research and development work, in order to deepen further the scientific knowledge that constitutes the base for the performance and safety assessment;
2 studies and evaluation of alternatives to the methods and concepts investigated so far;
3 optimization of systems in terms of technology, economy, and resource utilization, in view of the improved scientific knowledge base;
4 investigation for site selection

In September 1986 a comprehensive plan for future R&D work will be presented to the Swedish government. The basis for planning the R&D work is the overall timetable shown in Figure 9. This timetable is based on forty years interim storage of spent fuel in the CLAB. An application for licensing of a repository is foreseen around the year 2000.

Alternative Disposal Concepts

As a base for the optimization of the disposal system, a number of alternative disposal concepts will be studied and evaluated from the point of view of safety, technical feasibility, acceptability, and economy. All alternatives will, however, be based on disposal deep in Swedish crystalline bedrock. The word 'alternative' is used here in a broad sense. It includes changes in the materials and general layout envisioned by the existing concepts, as well as more profound changes in the concepts themselves. Also, alternative sites for the repository will be studied.

Examples of materials for which studies have been initiated during the last years are carbon steel for canisters and cement for grouting purposes, in the

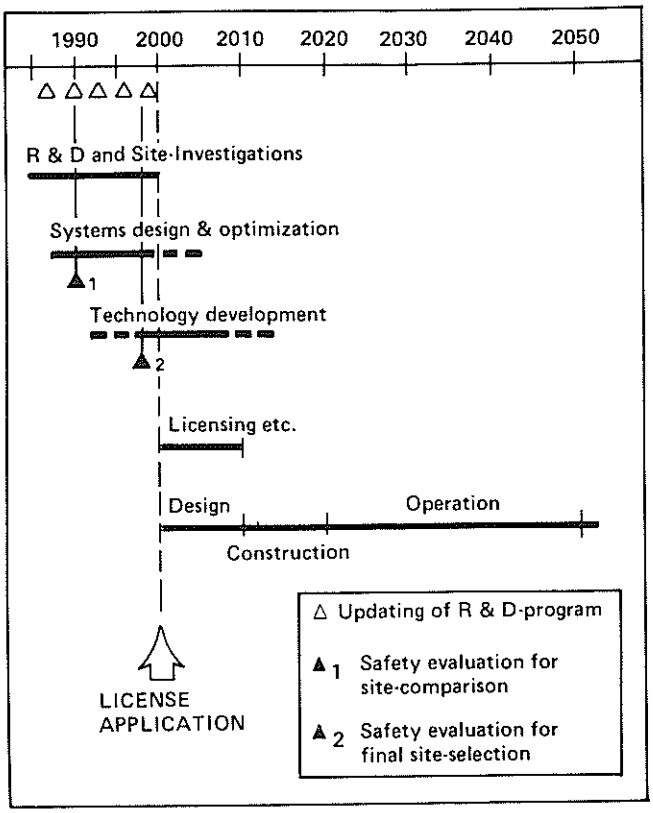

Figure 9 General timetable for realization of a final repository for spent fuel.

nearby rock. The layout of KBS-3 has been designed to keep the temperature increase low. The maximum temperature of the buffer has been limited to 100°C. If a higher temperature could be allowed, a higher waste load per canister could be used, or, alternatively, a more close-packed layout.

An example of a concept that is based on a somewhat different philosophy than that of KBS-3 is shown in Figure 10. This concept, the WP-Cave, is at present being evaluated by SKB [6]. The main features are:

1 a large amount of encapsulated fuel (> 1500 tonnes), concentrated in a central cavern that will be ventilated and cooled for about 100 years;
2 the cavern and surrounding rock surrounded by a five-metre-thick buffer layer of sand/bentonite;
3 this structure is surrounded by an hydraulic cage, consisting of a number of horizontal annular tunnels connected by drill holes.

The temperature in the cave will be limited to about 60°C during the ventilation period, and will rise to about 140°C after closure. The purpose of the buffer is to cut off the streaming of water between the inner and outer rock mass, and to delay the migration of radionuclides out of the repository. The hydraulic cage has the dual function of draining the inner rock mass during construction and operation, and of limiting the

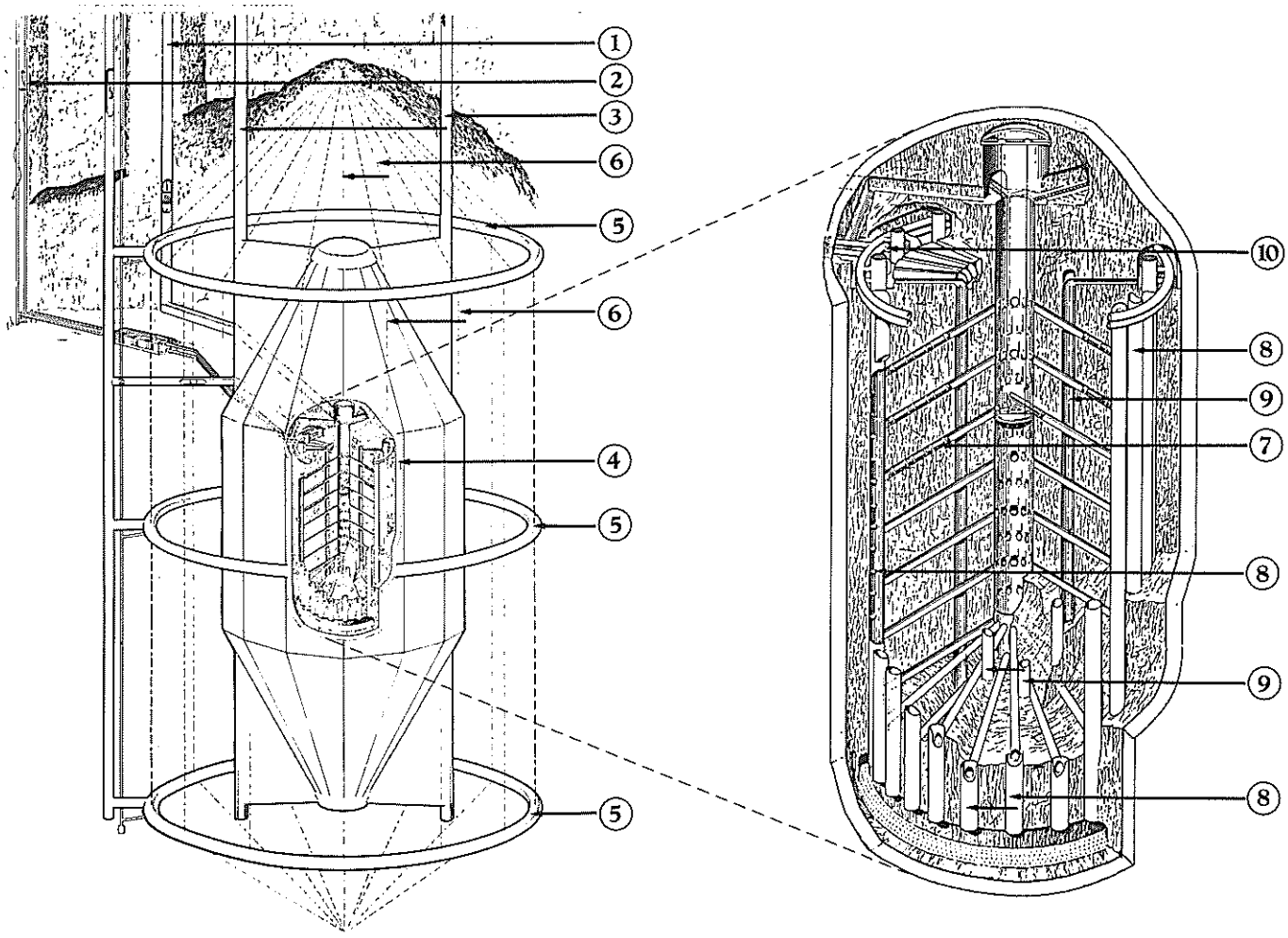

Figure 10 WP-Cave concept for final disposal.

hydraulic gradient over the repository after the central cavern has been water-filled.

Site Investigation and Selection

The investigation and selection of a site for the final repository is technically and economically very important, and of great political and public interest. The site will be chosen on geological, demographic, and economic grounds. Fourteen sites have been investigated since 1977 (Figure 11). On eight of these a full investigation program has been carried out.

The investigation program included 10–14 core bore holes, drilled to a depth of 500–1000 metres. Geophysical, hydrological, and geochemical measurements were made on the surface and in the bore holes. A program to improve measurement techniques is being pursued in parallel, and the results from this are being applied in the site investigations.

The investigations have confirmed that Sweden's bedrock, dominated by granite and gneissic formations of great age, provides acceptable conditions for safe disposal at many locations. In order to be able to choose among different sites, more studies at depth will be needed. Also, factors other than the geological conditions must be allowed for in the final decision.

At the beginning of the 1990s, a couple of sites will be chosen for more detailed study, including a pilot shaft down to the foreseen repository level. Final site-selection will be made at the end of the 1990s.

General R&D

The safety assessments performed until now have been based on several pessimistic assumptions. Circumstances, barriers, and factors that are insufficiently known are not accounted for if they work towards increased safety. Analysis methods and data are persistently chosen to give an upper limit for the calculated consequences.

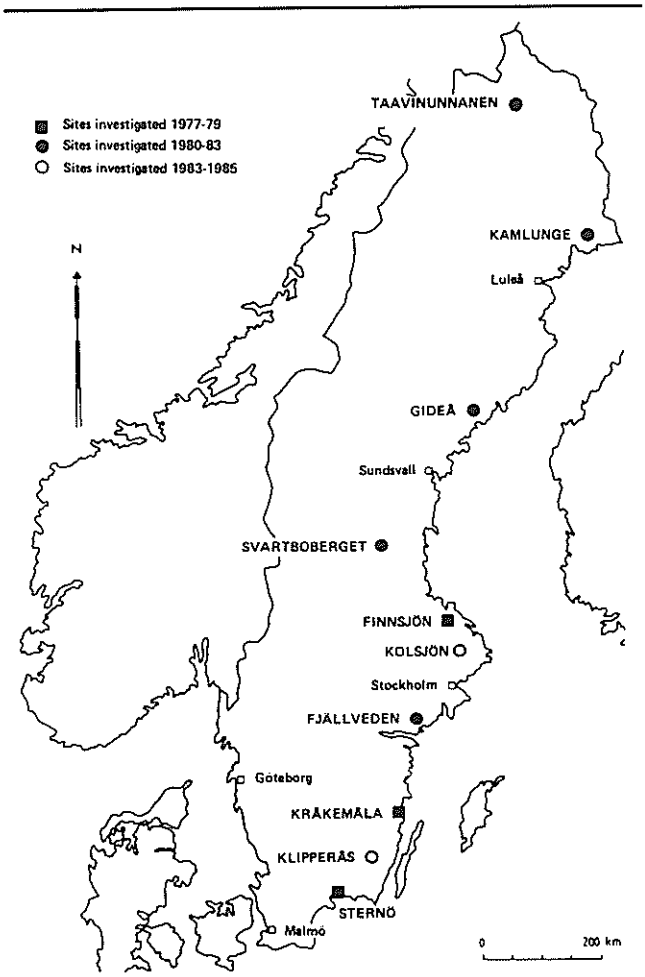

Figure 11 Sites investigated in Sweden.

It is important to improve knowledge in areas of key interest for the performance of the repository. Some important areas being studied [7] for a repository located in Swedish bedrock are:

1 leaching behaviour of spent nuclear fuel;
2 groundwater chemistry and its interaction with materials in the repository;
3 properties of materials used;
4 groundwater transport in the rock mass and in fracture zones;
5 radionuclide transport in fractured rock;
6 stability of the bedrock.

The results of these research areas are applicable, whatever alternative concept may be developed. The study of alternatives, however, provides input concerning environmental factors, etc. Of special interest are the studies of natural analogues that can provide a long-term proof of the validity of the models and data used.

Cooperation and exchange of information on an international or bilateral basis is an integrated part of the R&D-activities of SKB. As an example, an international research project is being executed in Stripa, an abandoned iron mine in central Sweden, with the participation of nine OECD countries. Different aspects of the geological and engineered barrier systems are being investigated on a large scale and in a realistic underground environment in Stripa.

Costs and Financing

According to Swedish law, all costs for the management of radioactive waste, including the decommissioning of the nuclear power plants, have to be borne by the owners of these plants. The costs are covered by a fee determined annually by the government. The basis for the fee is a cost calculation of all the activities for the back-end of the nuclear fuel cycle, which is carried out by SKB each year [8]. The cost calculation is based on a scenario for the back-end, including the construction and operation of all necessary facilities and equipment.

The experience gained from the construction of the CLAB and SFR facilities is very valuable as a background for the cost data. The construction cost of the CLAB was SEK 1.7 billion (Can $350 million) and is expected to be SEK 0.8 billion (Can $150 million) for SFR phase 1. The total cost incurred through 1986 for the back-end of the nuclear fuel cycle was SEK 5.3 billion (Can $ 1.1 billion). This figure includes, besides the CLAB and the SFR, costs for reprocessing services, the R&D program, and the transportation system. The estimated future costs (at the January 1986 price level) are about SEK 39 billion (Can $ 7.8 billion). The value of the corresponding electricity produced at the nuclear power plants is about SEK 500 billion.

Many of the costs will be incurred fairly far into the future. The total expenditure will be spread out over a period of more than 70 years. Figure 12 gives a rough account of the distribution in time. If the reprocessing costs are excluded, the breakdown of costs is roughly:

Interim storage of spent fuel	21%
Encapsulation and final disposal of spent fuel and long-lived wastes	39%
Final disposal of operation and decommissioning wastes	4%
Transportation of wastes	8%
Decommissioning and dismantling of nuclear power plants	19%
Miscellaneous including R&D and pilot facilities	9%

The fee for 1986 is SEK 0.019/kWh (Can $ 0.004/kWh), which corresponds to a total cost for the Swedish utilities of SEK 1.2 billion/year (Can $ 240 million/year). The fee is paid into funds, one for each utility, at the Bank of Sweden. The funds are administered by the state authority, SKN (The National Board for Spent Fuel), which also allocates money from the funds to the various waste management activities performed by SKB.

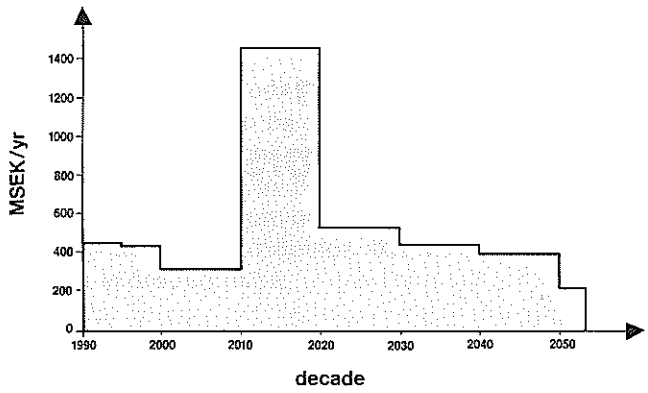

Figure 12 Approximate distribution in time of future costs for waste management system.

Concluding Remarks

The Swedish system for handling and disposal of radioactive residues is planned and designed to meet a very high standard in terms of occupational health and safety and environmental impact. It also satisfies the requirement that no undue burden should be laid on future generations and that only proven techniques should be used.

The implementation of all steps of the back-end of the nuclear fuel cycle is managed within Sweden. The progress of the work has been very satisfactory, with the start-up of the CLAB facility, the advanced construction schedule for the SFR, and the acceptance by the government of concepts presented by the power industry for final disposal of spent fuel.

A prerequisite for the successful implementation of the system is that it be accepted by the public. For the CLAB and SFR, very little public opposition was encountered. The final disposal of spent fuel is, however, a more difficult task, not least from the point of view of public acceptance. Recently a site investigation had to be stopped due to local opposition. According to recent polls, there is almost an equilibrium between those who believe that radioactive waste can be safely handled and disposed of and those who think it cannot. Although Sweden has more positive figures in this respect than most other countries, more work is needed.

Consequently, communication with the public concerning facilities and the safety of the systems is a very important part of the waste management program. In order to achieve acceptance of the site for a final repository at the end of the 1990s, extensive information will have to be furnished, both locally and on a national level.

For Sweden, as for most other countries with nuclear power, broad and open international cooperation in the field of radioactive waste management plays an important role, not only because of obvious gains in technical efficiency, but also from the point of view of public understanding and acceptance. Although radioactive waste management has to meet national demands and conditions, radioactive waste disposal is a truly international question, and it is particularly important that the international work include long-term radiation protection objectives as well as principles for evaluation and performance of various disposal systems.

It is obvious that work remains to be done in all the scientific and technical disciplines involved, and that balanced and extensive information needs to be made available, before a high-quality and properly situated repository can be commissioned in Sweden. Adequate time is available for this.

Acknowledgements

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A Non-Intrusive Neutron Method for Poison Concentration Monitoring in CANDU Reactors

E.M.A. Hussein, D.R. O'Connor and M.E. Mosher
Department of Chemical Engineering
University of New Brunswick
Fredericton, N.B. E3B 5A3

Abstract

A neutron reflection method is developed for measuring the concentration of the neutron absorbing gadolinium in the tanks of the liquid poison injection shutdown system of a CANDU reactor. The feasibility of the method is demonstrated experimentally, and Monte Carlo simulations are utilized to design and model the performance of a proposed device.

Résumé

On présente une méthode basée sur la réflection des neutrons pour mesurer la concentration du gadolinium, absorbeur de neutrons, dans les réservoirs du système d'arrêt à injection de poison liquide du réacteur CANDU. On démontre expérimentalement que la méthode est pratique et on utilise des simulations Monté Carlo afin de concevoir un dispositif et d'en modéliser la performance.

Introduction

The liquid poison injection shutdown system, the secondary shutdown system (SDS2), of a CANDU reactor utilizes a gadolinium nitrate, $Gd(No_3)_3 \cdot 6H_2O$, solution to neutron-poison the reactor following a trip. The solution is contained in several (six to eight) tanks that are connected via an open-line to the reactor calandria. The solution is injected, when needed, under pressure into the moderator system. In order for this shutdown system to be effective, the concentration of natural gadolinium in the solution is required to be no less than 4000 mg Gd/Kg solution.

The effectiveness of the SDS2 is presently insured by periodic measurement of the gadolinium concentration. Typically, a sample is taken weekly from one of the tanks and analyzed in the laboratory using atomic absorption. This is supplemented by mass spectroscopy analysis, usually twice a year, in order to ensure the adequacy of the isotopic content of the material used. Natural gadolinium, as shown in Table 1, contains about 30 per cent of the neutron absorbing isotopes, Gd-155 and Gd-157. To avoid the remote possibility of using gadolinium burned-out by a prior reactor irradiation, isotopic analysis is required.

Table 1: The Isotopes and Neutron Cross-Sections of Natural Gadolinium (a)

Isotope	*Abundance (per cent)*	*Thermal Absorption Cross-Section (barns)*
152	0.2	1.1×10^3
154	2.1	90
155	14.8	6.1×10^4
156	20.6	2
157	15.7	2.55×10^5
158	24.8	2.4
160	21.8	0.8
	100.0	4.9×10^4

(a) Lederer & Shirley 1978 [1].

The periodic analysis procedure, in addition to being manual and subject to administrative error, does not provide a continual indication of the effectiveness of the system. It is possible that one or more tanks can be downgraded in gadolinium solution and remain undetected until a sample is withdrawn from the tank a few weeks later. It is desirable, therefore, to have an on-line monitoring device that can continually provide an estimate of the poison concentration in each tank. Such a system would reduce the manual effort required for testing and would improve control room information.

In this paper, we present a new non-intrusive method for on-line monitoring of the neutron-poison solution in each of the tanks of the SDS2. The method is based on neutron reflection and utilizes external neutron sources and detectors. The details of this technique are given later. We begin, however, by

Keywords: fast neutrons, gadolinium, neutron absorbers, nuclear power plants, pressurized water reactors, radiation monitoring, radiation scattering, radiation shielding, radioisotopes, safety devices, safety measures, thermal neutrons.

reviewing some of the methods that can be utilized for gadolinium concentration monitoring.

Measurement Methods

Several methods can be considered for on-line monitoring of the poison concentration in SDS2 tanks. However, in order to be able to choose a suitable method for application, one must define the characteristics of the required device. An on-line monitoring system is desired to be:

1 capable of yielding continual measurement of the solution's neutron absorption ability in each of the SDS2 tanks;
2 non-intrusive to the existing piping system;
3 maintenance free;
4 inexpensive.

An on-line monitoring system is required to provide continual information that can be accessed from the control room. It is also desired that the system provide directly a measure of the amount of neutron-absorbing material available in each tank. This enables the detection of any changes in the isotopic content of the gadolinium. Since the piping network of the SDS2 is designed as a Nuclear Class I system, the monitoring device is preferred to be non-intrusive, to avoid the costly process of modifying the system to accommodate the measuring equipment. The monitoring system also needs to be easy to install, reliable, reasonably maintenance-free, and inexpensive. Some of the techniques that have been previously proposed for poison monitoring are discussed below and assessed against the above characteristics.

Mass Spectroscopy

Mass spectroscopy provides an accurate means of measuring the isotopic content of an ionized sample. The method is based on the deflection of ions in an electromagnetic field. This provides a direct estimate of the type and amount of isotopes in the sample. The device is delicate, expensive, and requires samples to be withdrawn from each tank.

Atomic Absorption

In this method, a light source of a frequency corresponding to the energy required to excite a gadolinium atom is applied to an atom vapour sample of the solution. The gadolinium present absorbs the incident radiation to a degree proportional to the concentration of gadolinium in the sample. The apparatus used in this method is a delicate one and is difficult to apply on-line. Moreover, the method is intrusive and does not directly measure the neutron absorption ability of the solution.

Electrical Conductivity

The conductivity of a gadolinium nitrate solution varies with its concentration. A conductivity meter can, therefore, be used for gadolinium concentration monitoring. This requires the introduction of a probe into the tank or the associated pipes. However, the conductivity measurement does not provide a direct indication of the neutron absorption ability of the solution. It requires frequent calibration and is easily affected by impurities and the degree of acidity of the solution.

Thermal Neutron Attenuation

This is a direct method for measuring the absorption ability, and consequently the concentration of the neutron-poisoning isotopes. A very strong neutron source would, however, be needed, if the method were to be applied directly to the solution contained in an SDS2 tank. This is due to the large distance neutrons have to travel in a highly absorbing medium. The method can, however, be applied to a small circulating volume of solution drawn from the tank. Since available isotopic neutron sources emit fast neutrons, a thermalization assembly around the source is required. The thermalized neutrons can be utilized to measure the concentration of more than one solution sample located around the neutron source. This method is simple, capable of providing continual measurements, and relatively maintenance free. However, an intrusive sampling system is needed.

Neutron Flux Depression

One may also consider using the flux depression caused by inserting a sample containing gadolinium solution into a neutron medium. The flux depression can be measured directly or by using neutron activation analysis of an internal flux monitor. The magnitude of the flux depression provides a direct measure of the solution neutron absorption ability. However, this is an intrusive method that requires the withdrawal of a solution sample from the tank or the use of a circulating sampling system.

Neutron Activation

Gadolinium produces gamma-active radionuclides under neutron irradiation. The activity of these nuclides can be used, at least in principle, to measure the gadolinium concentration. Activation analysis, however, requires a large neutron flux that can be only obtained inside a nuclear reactor, and therefore cannot be applied on-line.

All the systems discussed above are intrusive in nature, and only the conductivity and neutron attenuation or flux depression methods can be easily applied on-line. The conductivity method does not provide a measure of the neutron absorption ability of the solution, while the neutron attenuation and flux depression methods require a continuous circulating system to be installed on each tank. An alternative method that can overcome the above difficulties is,

therefore, needed. A new measurement technique is introduced below.

Neutron Reflection Method

The neutron appears to be the most appropriate probe for monitoring the concentration of poison solutions, since it measures directly the neutron absorption ability of the solution. However, the neutron attenuation, flux-depression or activation techniques, discussed above, require intrusion into the existing piping system. To avoid this problem we propose to utilize neutron scattering for measuring the poison concentration. If an epithermal neutron beam is directed towards a tank, the neutrons will be scattered and slowed-down, some to the thermal energy, by the heavy water contained within the tank. The slowed-down neutrons will then attempt to escape from the tank and can be recorded by a neutron detector located outside the tank. The scattered thermal neutrons will, however, be exposed to absorption by the gadolinium contained in the solution. The neutrons will be also subject to attenuation by the relatively thick (typically 9 mm) stainless steel walls of the tank. This, however, introduces a constant attenuation factor that is independent of the poison concentration. The degree to which thermal neutrons can escape from the tank is, therefore, inversely proportional to the concentration of the gadolinium-absorbing isotopes in the tank. As will be shown later, the maximum neutron scattered flux is obtained at 180 degrees with the incident beam. Neutron reflection is, therefore, proposed for gadolinium concentration monitoring. Figure 1 shows a schematic diagram of the proposed technique. The monitoring system consists mainly of a neutron source and a detector positioned on the side of each tank. The detector is surrounded with a cadmium sleeve, so that only thermal neutrons reflected from the tank are seen by the detector. Radiation shielding surrounds the source and the detector. No shielding is required on the opposite side of the tank, since the reactor wall can be used for this purpose.

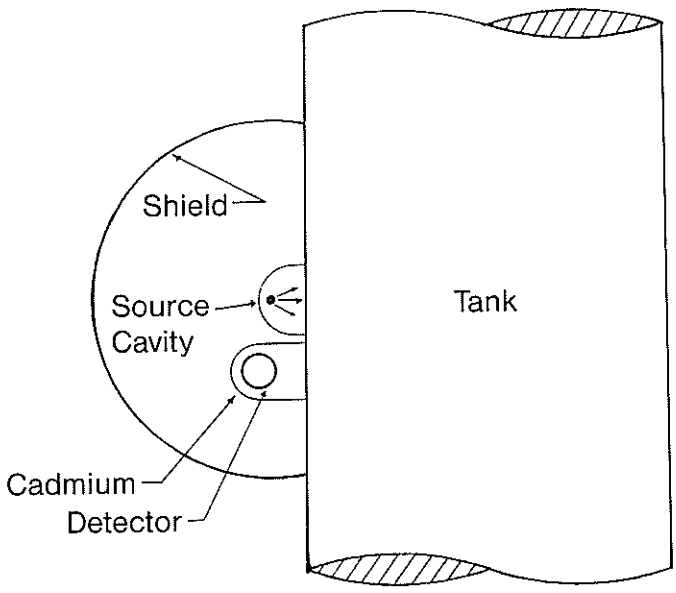

Figure 1 A neutron reflection gadolinium reflector.

The neutron reflection technique, as shown above, is non-intrusive and capable of providing continual measurement of the concentration of the neutron-absorbing isotopes in the solution. The simplicity of the technique makes it a relatively maintenance-free and inexpensive method for poison monitoring. The technique, therefore, meets all the desired characteristics defined earlier. In the following sections, the feasibility of the technique is demonstrated and a conceptual design is presented.

Experiments

In order to prove the feasibility of the neutron reflection technique, a set of experiments was carried out. A sheet-metal cylinder of a diameter equivalent to that of a typical SDS2 tank (236 mm) was used in the experiments. A neutron beam extracted from an americium-beryllium source (average energy of around 5 MeV) was employed. A BF_3 thermal neutron detector and standard counting electronics were utilized. The detector was surrounded with a cadmium sheet so that only neutrons scattered from the tank were detected. Virgin heavy water and gadolinium nitrate salt were supplied by the Point Lepreau Nuclear Generating Station (PLGS) for use in these experiments. Solutions of different concentrations were prepared and their concentrations were verified by atomic absorption analysis performed at PLGS. The gadolinium used was obtained from a shipment whose isotopic content had been previously confirmed by mass spectroscopy.

Results

As a first step, a set of experiments was carried out using light water, which has a slowing-down power much greater than that of heavy water. More thermal neutrons would, therefore, be produced inside a light water tank. Consequently, if one could not measure the gadolinium concentration in light water using neutron reflection, it would not be possible to measure it in heavy water. The results of the light water experiments are shown in Figure 2, on a semilog graph. The figure indicates that the detector response is approximately an exponential function of the gadolinium concentration.

The light water experiments were also used to obtain the optimum detector location. This location is defined as the position at which maximum foreground-to-background count ratio is obtained. The experiments showed that the optimum detector position is located a few centimeters directly beneath the neutron source. The same conclusion was obtained for heavy water.

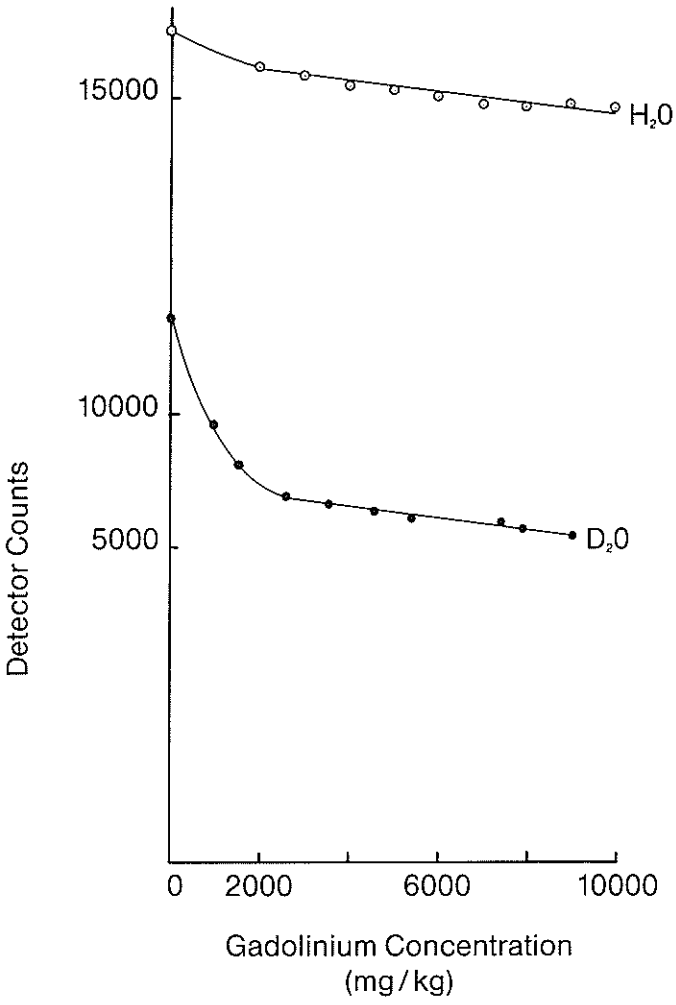

Figure 2 Experimental results.

The experimental results obtained using heavy water solutions are also shown in Figure 2. A nearly exponential relationship exists between the detector count rate and the gadolinium concentration, except at low concentrations. The behaviour of this detector response is explained later, when the physical theory behind the technique is presented.

The experiments presented above demonstrate the feasibility of using neutron reflection for gadolinium concentration monitoring. Although the detector response tends to saturate at high concentration, the logarithm of the signal provides a near-linear relationship that can be used to decide whether the gadolinium concentration has declined. It can be also noted that the reflected signal count rate is always larger than the background signal. This indicates that a useful signal can always be obtained even when the detector count rate is saturated. Given the success of the experiments, it was decided to perform some computer simulations in order to understand further the physics of the problem and to optimize further the performance of the techniques.

Monte Carlo Simulations

The Monte Carlo method is the most suitable approach to simulating the neutron reflection technique, because of the irregularity of the boundary conditions involved. A modified version of the COM [2] program was utilized for this problem. COM is a Center-of-Mass Monte Carlo program that was originally developed to simulate neutron scattering of a two-phase flow in a pipe. The program calculates the response of a thermal point detector located outside the pipe and provides a plot of the distribution of thermal neutrons generated inside the pipe.

The COM program was used to study the effect of the neutron source energy on the detector response. A set of hypothetical mono-energetic sources, and that of a californium-252 neutron source were considered. As it can be seen from Figure 3, the detector response is generally an exponential function of the gadolinium

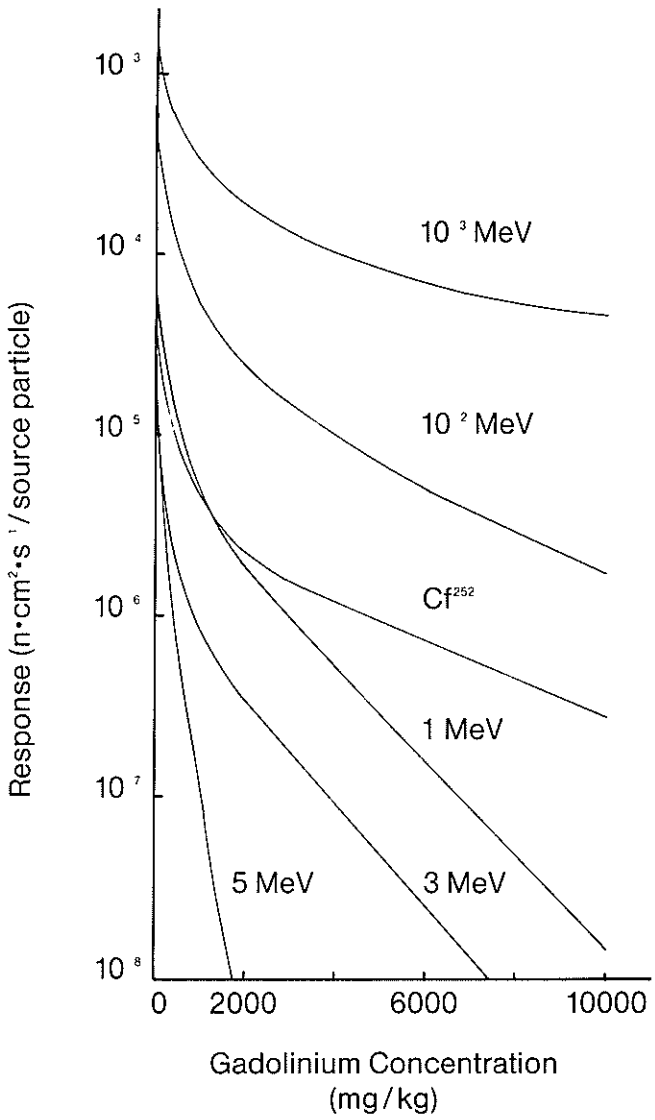

Figure 3 Monte Carlo results.

concentration, with some deviation at low concentrations. This deviation becomes more pronounced at lower energies. A theoretical explanation of this behaviour is given in the next section.

The measurement resolution, as indicated by the slope of the logarithm of the detector responses, increases with the source energy, as Figure 3 shows. The detector response, however, decreases in value as the energy increases. One must therefore choose an appropriate energy at which reasonable resolution and count rates can be obtained.

Mono-energetic isotopic sources are not readily available; therefore, polyenergetic sources must be used. The americium-beryllium source used in the experiments reported above emits neutrons of an average energy of about 5 MeV. This results, as Figure 3 indicates, in a low detector count rate per source neutron. A californium-252 neutron source was, therefore, considered. This is a spontaneous-fission source that emits neutrons of an average energy of about 2 MeV. The californium-252 source appears to provide a good compromise between resolution and count rate.

Empirical Model

For calibration purposes, and to understand better the system's behaviour, a simple empirical model is desired. We present here a model based on the observations obtained from the experimental and Monte Carlo results presented above. The model assumes that the distribution of the thermal neutron cloud generated inside the tank is independent of the gadolinium concentration. This is a reasonable assumption, since gadolinium is not a very effective neutron thermalizer. Gadolinium will, however, absorb the generated thermal neutrons as they diffuse throughout the tank. The amount of neutron absorption is assumed to be governed by the exponential relationship:

$$M_p = I_o \exp [-Kp] + V(p) \qquad [1]$$

where M_p is the detector response for a poison concentration p, I_o is a calibration constant, K is an attenuation parameter that incorporates the microscopic absorption cross section of gadolinium and the geometry of the thermal neutron cloud, and V(p) is a term that accounts for deviation from the exponential function. The exponential nature of the detector response is depicted in the semilog graphs of Figures 2 and 3, obtained from experiments and Monte Carlo simulations, respectively.

The deviation from the exponential function observed at low concentrations can be attributed to the fact that the effective area of the thermal neutron cloud contributing to the detector increases as the gadolinium concentration decreases. That is, for high gadolinium concentration, only those neutrons closest to the tank boundary near the detector contribute significantly to the detector counts, while neutrons near the centre of the tank have a very low probability of reaching the detector. On the other hand, at low gadolinium concentration, thermal neutrons generated near the centre of the tank have a higher chance of reaching the detector. In other words, the thermal cloud appears to the detector as more visible at low gadolinium concentrations. This effect is taken into account in equation (1) by introducing the visibility term V. This term is not only a function of the gadolinium concentration, but it is also a function of the incident neutron energy. This is due to the fact that the concentration of the thermal neutron cloud, and consequently its visibility, change with the incident neutron energy, as Figures 4 and 5 demonstrate.

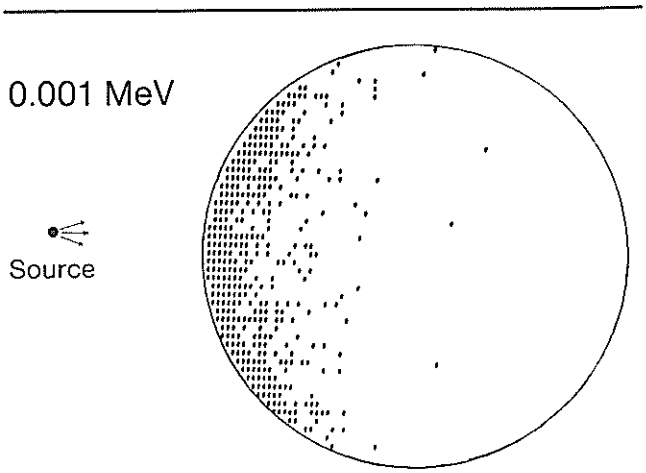

Figure 4 Neutron thermal cloud for a 0.001 Mev source.

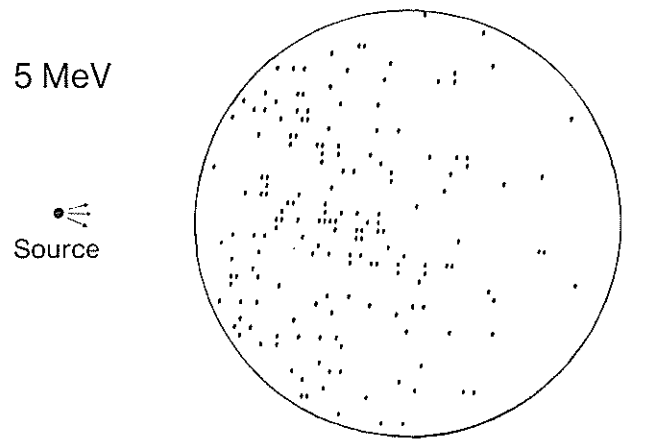

Figure 5 Neutron thermal cloud for a 5 Mev source.

The experimental and Monte Carlo results indicate that the visibility term can be represented by the equation:

$$V(p) = (M_o - I_o) \exp [-\beta p] \qquad [2]$$

where M_o is the detector response at zero poison

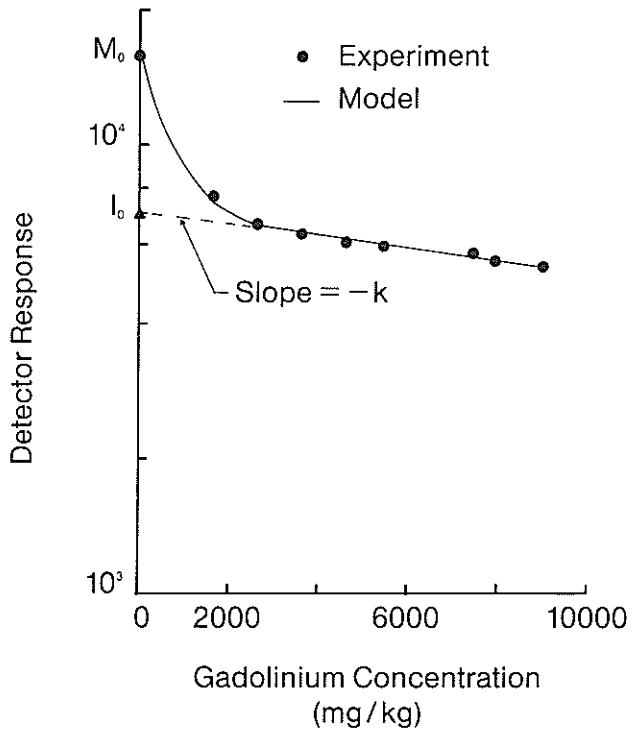

Figure 6 Theoretical model versus americium-beryllium heavy water experiments.

concentration, and β is the visibility coefficient at concentration p.

In order to demonstrate the validity of the above model, let us consider the americium-beryllium heavy water experimental measurements, shown in Figure 2. The slope of the line of the logarithm of the dominant exponential function at high concentration, shown in Figure 6, determines the value of the attenuation parameter K. The intersection of this line with the zero-concentration axis determines the constant I_0 of equation (1). The visibility coefficient β is determined by the slope of the line of the logarithm of equation (2). Figure 6 shows a good agreement between the model and the experimental measurements. Similar results were obtained for the californium-252 Monte Carlo results. This indicates that equation (1) is a valid representation of the detector response. The empirical model developed above can, therefore, be used to produce calibration curves for field-monitoring devices.

Preliminary Design

Based on the above experimental and theoretical evidence, one can now propose a preliminary design of a monitoring system. The device consists of a 2 μg californium-252 neutron source and a BF_3 detector enclosed inside a 200 mm-thick polyethylene shield. The basic configuration of the apparatus remains as shown in Figure 1.

The device should be located as close as physically possible to the tank and positioned so that use can be made of the reactor wall as a radiation shield. Vertical motion of the device can also permit the detection of gadolinium precipitation.

Californium-252 is chosen as the neutron source because, as indicated earlier, it provides a good compromise between resolution and detector count rate. Additional advantages of this source are its low heat generation, high specific activity, and low gamma radiation output. Therefore, no cooling of the device is required, and a minimum gamma ray shielding is needed. A 5 mm-radius sphere of lead surrounding the source is considered to provide sufficient gamma shielding for the proposed source strength.

The main disadvantage of a californium-252 source is its relatively short half-life (2.6 years). In order, however, to obtain the neutron yield of 4.6 million n/s of a 2 microgram (1.07 mCi) californium-252 source, 2.1 Ci of americium-beryllium, or 354 mCi of radium-beryllium would be required. The choice of the californium-252 source, therefore, results in a significant reduction in radioactivity, and consequently in the shielding requirements. A device based on a californium-252 source is, therefore, much more compact in size and of much less weight, compared to a device that uses any alternative isotopic neutron source. The short-half life problem of the californium-252 source is partially compensated for by the fact that the source is commercially widely available at a reasonably low cost. Therefore, the replacement of the source, expected to be required no more than three times during the life-time of the reactor, does not constitute a significant addition to the cost of the device. The effect over time of the change in the source strength can be accommodated for through software, or by providing a dummy reference system to which measured signals can be compared. The latter alternative is possible because of the low cost of the device. A BF_3 thermal neutron detector is proposed in the design for economical reasons. The more efficient, and more expensive He-3 detector is, however, preferred. In order to improve the signal-to-noise ratio, a cadmium sleeve should be used to cover the surface that is not directly exposed to neutrons reflected from the tank. A standard electronics counting system can be used with this device.

Conclusions

This paper has demonstrated the feasibility of a neutron reflection method for gadolinium concentration monitoring in the tanks of the liquid poison injection shutdown system of a CANDU reactor, and has presented a conceptual design of a suitable device. The method is non-intrusive and uses off-the-shelf equipment. The signal obtained from the device provides a direct measure of the neutron absorption

ability of the poison solution contained in the tank. The device can also scan the tank to monitor gadolinium precipitation. Further work is being undertaken to construct a prototype system suitable for installation at an operating CANDU reactor.

Acknowledgements

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Notes and References

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Containment Systems Capability

W.G. Morison and W. J. Penn
Ontario Hydro, Canada

K. Hassmann
Kraftwerk Union, West Germany

J.D. Stevenson
Stevenson and Associates, U.S.A.

K. Elisson
ASEA-ATOM, Sweden

Abstract

This paper provides a summary of the designs and capabilities of principal containment systems associated with BWR, PWR, and PHWR reactors in operation and under construction in the United States, Canada, West Germany, and Sweden. The many conceptual differences in design and modes of operation following accidents are briefly described, with commentary on their evolution and alternatives considered. Specific examples for each reactor system in operation in the four countries are detailed [1, 2, 3, 4]. The containment design differences and requirements are mainly attributable to the fundamental arrangements of the reactor and secondary side systems, and their demonstrated behaviour during normal operation and following accident conditions. However, two other important considerations that strongly influence design are national regulatory requirements [5, 6, 7, 8] and the number of generating units in a station. These broad issues, together with site conditions and proximity to population, dominate containment performance requirements for economic generation of electricity and public safety. Emphasis is focused on the capability of the various systems to meet design basis accidents. However, the TMI-2 incident has caused plant owners and regulators to examine the ultimate capability of containments, far beyond maximum credible accident bases. Postulated severe degraded core accidents, with a predicted frequency several orders of magnitude lower than other recognized world-wide hazards for which protection is provided, are currently under intense scrutiny [9, 10, 11]. This paper describes the status of some of these studies.

Résumé

Ce document passe en revue les principaux systèmes de confinement associés aux réacteurs à eau bouillante, aux réacteurs à eau sous pression et aux réacteurs à eau lourde sous pression en service ou en construction aux États-Unis, au Canada, en Allemagne de l'Ouest et en Suède. On y décrit brièvement les nombreuses différences existant au niveau de la conception et des modes de fonctionnement à la suite d'accidents tout en commentant l'évolution des systèmes et les solutions de rechange envisagées. S'ajoutent à cela des exemples précis pour chaque filière en service dans les quatre pays [1, 2, 3, 4]. Les différences au niveau de la conception et des exigences sont principalement attribuables au montage fondamental du réacteur et des systèmes du côté secondaire ainsi qu'au comportement démontré en cours de fonctionnement normal et lorsque placés dans des conditions d'accidents. Il est à noter que les exigences nationales de réglementation [5, 6, 7, 8] et le nombre de tranches dans la centrale sont deux autres considérations influant considérablement sur la conception. Ces facteurs importants, joints aux conditions du site et à la proximité de la population, viennent au premier rang des exigences d'efficacité des systèmes de confinement en ce qui concerne la production d'électricité et la sécurité de la population. Le document met l'accent sur l'efficacité des divers systèmes à faire face à des conditions possibles d'accidents. L'incident de TMI-2 a cependant amené les propriétaires de centrales et les organismes de réglementation à pousser leurs recherches au-delà des modèles d'accidents plausibles et à examiner l'efficacité ultime des systèmes de confinement. C'est ainsi qu'on étudie actuellement de façon rigoureuse des hypothèses d'accidents graves impliquant la dégradation du coeur du réacteur [9, 10, 11] et d'une fréquence prévisible de plusieurs fois moindre que celle d'autres dangers reconnus dans le monde entier et pour lesquels il existe un système de protection quelconque. Ce document décrit le stade actuel de certaines de ces études.

Keywords: nuclear safety, containment systems, international experience, design and performance of nuclear safety methods.

Containment Functional and Design Requirements

The universally accepted philosophy for providing assurance of nuclear safety in accidents is the adoption of the principle of 'defense-in-depth,' which prevents or limits the release of radioactive material for a wide range of circumstances. 'Defense-in-depth' embodies a multiplicity of physical and chemical actions attributable to station process systems, but specifically includes three often duplicated and diverse safety systems to effect prompt reactor shut-down, ensure continuing and controlled heat removal, and automatically minimize/prevent radioactive release to the environment.

Containment systems are the ultimate line of defence and safety barrier for preventing the escape of radionuclides to the environment. The functional requirements of containment do not differ in principle for BWR, PWR and PHWR systems, but design requirements are significantly different. The design requirements are not only set by the overall arrangement of the primary reactor and secondary systems, but also (importantly) by national code and regulatory differences. A major additional influence that determines containment system designs is the extent, rate, and duration of accident pressure and temperature transients.

During normal operation the function of all containments is to minimize the release of gaseous, liquid, and solid radioactive materials produced during electricity production, and which are not retained in process systems. The objective is to ensure that emissions are as low as reasonably achievable, economic and social factors taken into account. Following an accident, the objectives are to retain radioactive materials released as a result of process equipment failure.

Thus, containment system designs have evolved from the basis that they should provide radiation shielding and retain all of the steam and water discharged following an internal reactor system piping failure. The primary element of containment systems is a practical engineered and economic 'leak-tight' building that covers and encloses the reactor systems. Piping or ventilation systems that might convey radioactive material outside the containment boundary are isolated immediately after an abnormal condition is detected. Sub-systems to reduce pressure in the building also feature in the designs. In some designs these systems include venting to 'gravel' beds or the atmosphere in a controlled manner, to ensure safe regulatory releases are met.

In addition to the provision for internal containment loads, protection of containment, and hence reactor systems, against external loads (i.e., earthquakes, hurricanes, tornadoes, explosions, aircraft impacts, and plant-induced missiles), are also major design requirements.

Pressurized Heavy Water Reactor Design Requirements

In Canada, PHWR containment designs must adhere to the CSA N290.3 standard, which differs only in detail from similar ASME codes. The CANDU containment design requirements are uniquely influenced by the adoption of multi-unit stations (eight units in the case of Pickering NGS, four units per station in other plants) and on-power refuelling, where a single integrated containment system employing negative pressures, dousing water pressure suppression, and a vacuum building is deployed. Elsewhere in Canada and overseas, AECL-designed 600 MWe single CANDU units include similar dousing water pressure suppression systems and filtered air discharge.

The fundamental difference in design of PHWR's and LWR's, namely the physical separation of the primary coolant and moderator system within the PHW reactors, reduces the probability of core melt in postulated severe accidents by orders of magnitude. In essence, the large heat sink provided by the moderator system gives high assurance of fuel channel integrity and prevents gross fuel melting to the extent that meltdown sequences are not generally considered credible [12].

The design of CANDU containment features are influenced by the structure of the Canadian Atomic Energy Control Board (AECB) regulatory requirements. To provide understanding of this influence, the AECB Siting Guide [6] is briefly described. The logic of this guide is based on a two-tier radiation dose limit applied separately to the most exposed individual and to the population. Process failures are judged against a 'single failure' dose limit (e.g., 3 rem to the thyroid of the most exposed individual). In common with LWR systems, these single failures range up to a guillotine break of the largest-diameter heat transport system piping. The limiting frequency of serious process failures (those requiring intervention by a safety system in order to prevent fuel failures) is one per three years. It must be emphasized that 'single failure' in this context is a different concept than that used in LWR licensing logic. In this case it means total failure of a system with no mitigating action by other process systems; only the safety systems can be credited.

The second part of the AECB Guide requires the analysis of 'dual failures,' involving serious process failures with simultaneous failure of one of the safety systems (either the emergency coolant systems or a major containment subsystem) to perform its function. This particular requirement is unique, but not necessarily more demanding than those of other national regulatory jurisdictions. The thyroid dose limit to the most exposed individual from these 'dual failures' is 250 rem. Containment design is strongly influenced by the requirement to meet this dose limit.

Light Water Reactor Design Requirements

The majority of nuclear power reactors in operation and under construction in the world today are either Pressurized Water Reactors (PWR) or Boiling Water Reactors (BWR).

The design requirements of PWR and BWR Containment Systems must adhere to the national codes and regulatory licensing requirements in the country of plant siting. In the United States principles have been developed for steel and concrete structures by the American Society for Mechanical Engineers (ASME) and American Concrete Institute (ACI). Historically, these code requirements for containments and their subcomponents have developed over the thirty years, culminating in an ASME and a joint ACI-ASME Code that caters to the many different combinations of steel and concrete structures that constitute the containment of operating reactors and those under construction. The ACI-ASME code combines a factored load approach with allowable stress criteria for all internal and external load consequences.

The various national codes, in addition to consideration of the ASME (steel) and ACI-ASME (concrete) codes for containment design, performance, and serviceability, reflect geographical, social, political, and regulatory requirements in their own environment. Thus the German 'Kerntechnischer Ausschuss, the Swedish, and the Canadian codes show differences from those in the United States, as well as alternative requirements for external loads such as historical seismicity, siting conditions, and threats of local explosion and aircraft crashes.

The accident internal service load is historically associated with any single component failure in the generating plant, having a frequency typically greater than 10^{-7}, events/yr, which causes a maximum energy, pressure, temperature, and radioactive release. In this regard, the design basis accident for which most containment systems are conservatively designed is the largest double-ended primary pipe rupture, (predicted frequency of 10^{-4}/yr), recognizing that while continued operation of the core cooling system is likely, its full credit cannot be assured in all accident conditions.

Certain combinations of extreme internal/external loads are also typically used in design of containments. Perhaps the most famous one is the combination of LOCA with some level of earthquake. On this issue, there is no general worldwide agreement. In the United States, for example, the largest postulated LOCA has been combined with the largest Safe Shutdown Earthquake (SSE). In other countries, while the combination is considered, it is not necessarily assumed that the largest LOCA and the largest earthquake are coincident. The reasoning for this position is that the reactor coolant system is specifically designed to resist earthquakes; therefore earthquakes do not cause LOCA's, but such an independent event cannot be discounted immediately following. The impact of military aircraft, blast waves, and a turbine wheel rupture impact are also considered in many designs.

The following categories of loads are not normally considered in the design process but have received increasing attention in determining containment performance capability.

The first category includes those loads with a negligible frequency ($< 10^{-7}$ per yr). Such loads would typically include meteorites, large commercial aircraft impact, and volcanic eruption.

The second category involves extreme internal accidents. Most countries typically do not combine LOCA with a secondary system failure as a design basis, although analysis of this combination is often undertaken. Rotating equipment and pipe support failure within containment are also not typically considered. Also, major component rupture, including vessel, pump, steam generator, and pressurizer, are not typically a design basis.

The third category involves the question of the degraded core, the so-called 'Class 9' accident. There are three particular types of containment loads that *might* be associated with such a situation. These loads include degraded cores possibly leading to some melting of containment, steam explosion, and hydrogen generation if it results in deflagration. As a result, containment overpressurization at elevated temperatures due to postulated failure of mitigation systems is under study.

Changing Emphasis on Performance Requirements

Table 1 [10] summarizes the evolution of containment performance criteria. The order listed relates to the growing emphasis that each have received over the last forty years, culminating with containment capability for degraded core accidents.

The criteria for radiological releases were the first to be developed. For all but extremely remote sites, this led to the use of containment systems with acceptable leakage related to site-specific characteristics. Most often these pressure retention containments were freestanding steel or steel-lined concrete structures that, for LWR designs, could be demonstrated to leak considerably less than 1 per cent of the containment volume per day during accident conditions. For multi-unit CANDU systems, where accident source terms and

Table 1: Evolution of Performance Criteria

1	Criteria for radiological releases
2	Criteria for direct radiation doses
3	Protection against external missiles
4	Consideration of degraded cores

energy release into containment are lower, and filtered venting to control long-term releases is deployed, leakage rates of less than 1 per cent per hour have been adopted.

These criteria provided protection against leakage, but not from direct radiation due to radioactive material within the containment after accidents. It was initially assumed that people near the site could be evacuated to minimize their exposure from material inside containment if an accident occurred.

The next criterion added was the requirement of shielding from direct radiation at all but the most remote sites. This led to the widespread use of steel-lined, reinforced or prestressed concrete structures for containment that combined low leakage capability with shielding from possible radiation.

The next important criteria to be added were for protection against external phenomena, such as missiles resulting from tornadoes. Similar criteria were developed relating to aircraft crashes at sites, depending on the frequency of air traffic. These additional criteria made the use of reinforced or prestressed concrete containments, or the addition of a special concrete missile shield, essential.

The fourth set of criteria, associated with degraded cores (or more precisely the need for such criteria) have been under intense scrutiny and debate since the Three Mile Island Unit 2 accident on 28 March 1979. There are two major investigations addressing these issues in the United States. One is a series of programs funded by the NRC on containment integrity. In these programs the behaviour of isolation features, structural capacity of containment, leakage characteristics of mechanical and electrical penetrations, and behaviour of the base mat when subjected to a core melt are being investigated [13, 14, 15, 16]. The other major investigation is the extensive U.S. IDCOR Program [9], which is currently under discussion with the USNRC. Also, intensive studies [2], concentrating on the sequences of core meltdowns and the accompanying accident consequences, have been conducted in the Federal Republic of Germany during the past ten years to ascertain the ultimate capability of containment systems for their operating LWR's. In Canada, important fission product distribution studies concentrating on 'Lessons learnt from Three Mile Island' have resulted in containment design modifications. Also, heavy emphasis on dual failure accidents (e.g., a large LOCA resulting in stagnation cooling conditions plus assumed coincidental containment impairment) continues in that country.

Containment Systems and Component Function

The specific details of existing containment systems depend on the project commitment dates, but their generic nature are a function of reactor type, site location, utility preference, economic considerations, number of units per station, and national regulatory influence. Thus, there is considerable design diversity in existing structures, although basic concepts have not radically changed in more than twenty years.

In the 1940s, the control of public exposure following a design basis accident was provided by the use of large exclusion areas, rather than a containment structure. For example, the Clinton pile at Oak Ridge was associated with a 60,000 acre site, the Hanford production reactors with a larger area, and the U.S. National Reactor Testing Station was located in the Idaho desert. The need to locate nuclear power plants nearer the consumer resulted in containment systems. Early containments were static pressure envelopes with few penetrations. These were not practical for commercial electrical generating stations. Subsequently, active containment structures, with a multiplicity of penetrations designed to close on accident signals to form a leak-tight barrier, evolved. Later, systems were introduced to suppress pressure and temperature within containment following accidents, and also mitigate fission product transport to the environment either by chemical means, controlled filtered venting, or returning leakage to containment by the addition of an outer barrier and pumping circuits. Tables 2 and 3, list respectively, the principal containment systems that are in general use, and those that have seen less use, or have just been studied.

Figure 1 illustrates the many variations of PWR containment, either in operation or committed, by 1972, worldwide. The variations on the three basic systems, (i.e., dry pressure retention containment, ice condenser pressure suppression, and subatmospheric pressure suppression) include single versus multiple barriers, the geometry of the steel or concrete structures, and the nature of allowable structural stress. The dominant system is the medium pressure dry containment with a single pre-stressed concrete cylinder. The majority of these containments are in the United States.

Today, the tendency for PWR containments is towards two dry barriers to fission product release, with provision to filter and vent the annular separation space.

All modern BWR containments are of the pressure suppression type (wet well and dry well) in order to reduce containment volume. This is because, in a design basis accident, BWR's would blow down by far the largest volume of high energy fluids of all water reactor systems. There are three variants (General Electric Company, Mark I, II, and III) of this basic system, with specific differences adopted in West Germany and Sweden. The Canadian PHWR System requires the lowest demand for design basis accident energy containment, due to the physical separation of primary, secondary, and moderator systems.

In more recent times there has been a trend towards

Table 2: Principal Containment Systems

Confinement:	Reactor systems enclosed in a low-leakage building, filtered discharge and negative pressure
Low pressure:	Large diameter hemispherical dome, 35 kPa
Medium pressure:	Low-leakage PWR or steel-lined concrete structure (0.2 to 0.5 MPa); variants in France and U.S.
High pressure:	Low leakage, PWR for pressures 0.5 MPa, steel vessel (FRG, U.S. and France)
Pressure suppression:	BWR system within compact low-leakage steel or steel-lined concrete structure, water and drywell energy suppression
Ice condenser:	A PWR energy-suppression system
CANDU pressure suppression:	Reactor and primary systems within steel-lined pre-stressed concrete containment at negative pressure; pressure suppression by dousing
CANDU shared containment:	Large pre-stressed concrete containment at negative pressure surrounding multi-units connected to vacuum building; pressure control via dousing and filtered venting

Table 3: Other Systems in Use or Studied

Multiple containment:	Two pressure-retaining low-leakage barriers
Pressure release:	Controlled filtered venting and scrubbing
Stronger Containment:	Increased wall thickness for 0.85 MPa
Shallow underground:	Standard containment with 10 m overburden
Deep underground:	Containment 30 m underground, turbine at grade
Increased volume:	Double normal volume, 0.42 MPa pressure
Compartment venting:	Vented to high-pressure structure with douse
Thinned base mat:	Permits core melt to inert gravel bed
Evacuated containment:	Operates at 35 kPa or less

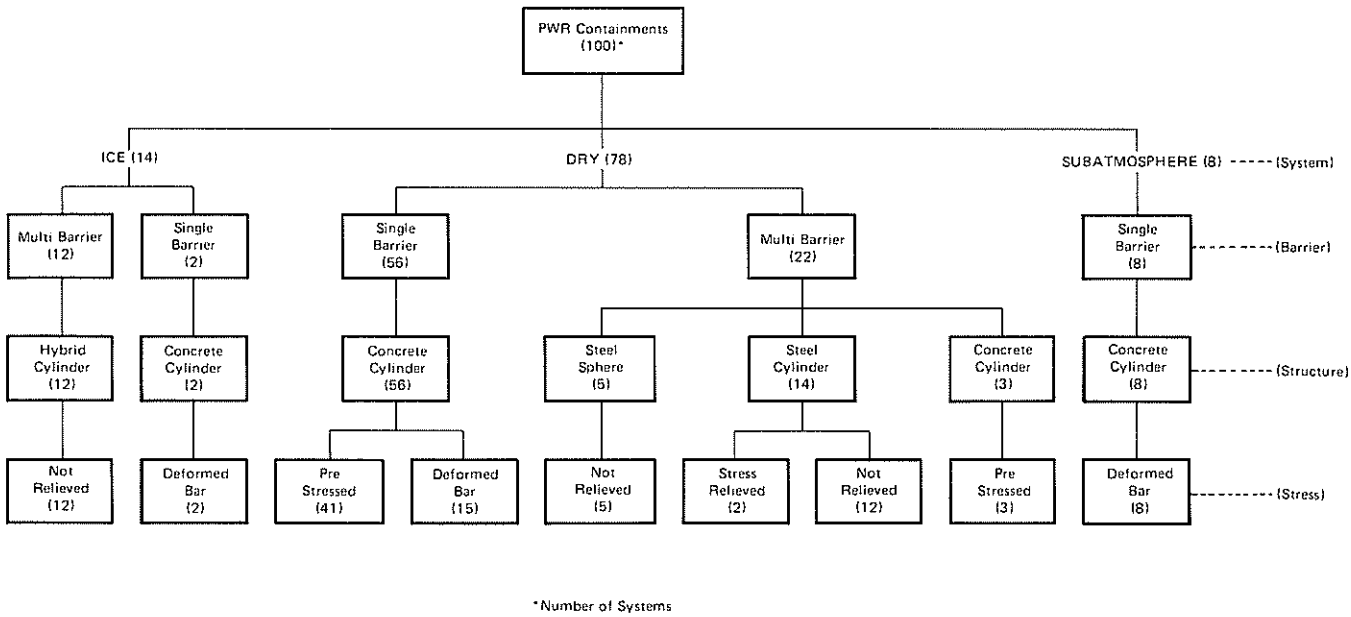

Figure 1 PWR containment designs (committed by 1972).

standardization for PWR, BWR, and PHWR containments, with differences in detail only dependent on the country of siting. Selected designs for the United States, West Germany, Sweden, and Canada follow.

Pressurized Water Reactor Containments

Figure 2 shows a low-leakage, pressure retention design, consisting of a pre-stressed concrete cylinder with a steel liner. A vertical buttress system together with a horizontal ring at the spring line, is used to anchor pre-stressing tendons. The dome and cylinder are separately pre-stressed. This design is widely used in the United States. More recent modifications to the design eliminates the dome ring, introduces partial buttresses in a hemispherical dome, and anchors the wall and some dome tendons at the base mat. As noted in Figure 1, this type of single-barrier containment is the most widely used in PWR stations operating today. Another version of this type of containment is the deformed bar-reinforced concrete cylinder and dome.

Steel containments, either cylindrical or spherical, are widely used in U.S., West Germany and Japan. In these double-barrier designs a concrete biological shield, which also serves to protect against external loads, surrounds the steel containment. The cylindrical design shown schematically in Figure 3 has wide application in the United States and Japan.

A common form of double-barrier containment in the future is expected to be the steel sphere surrounded by a concrete shield building, as developed in West Germany, and also applied to some plants designed in the U.S. Figure 4 shows a sectional view of the German, 1300 MWe Biblis B plant [2]. The inner detached steel shell of the containment (wall thickness 29 mm) constitutes a passive pressure-tight barrier. The containment sphere has a free volume of 70,000 m^3. The concrete structures within the steel containment (about

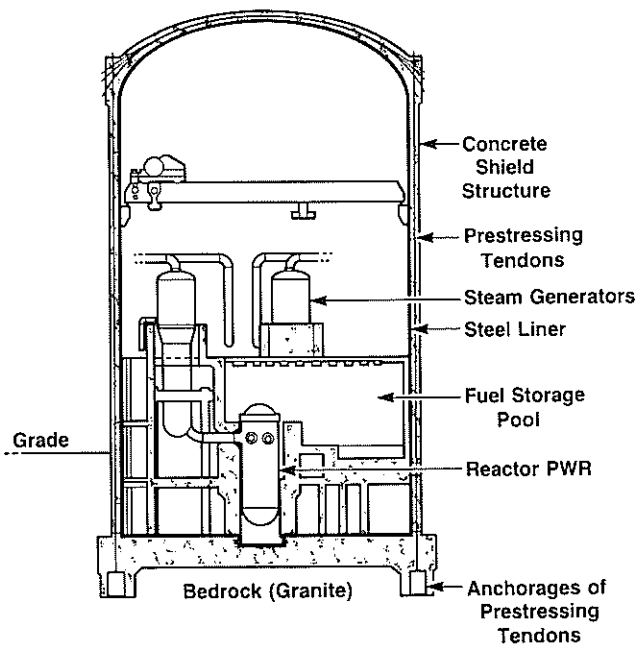

Figure 2 Dry containment: Steel-lined pre-stressed structure.

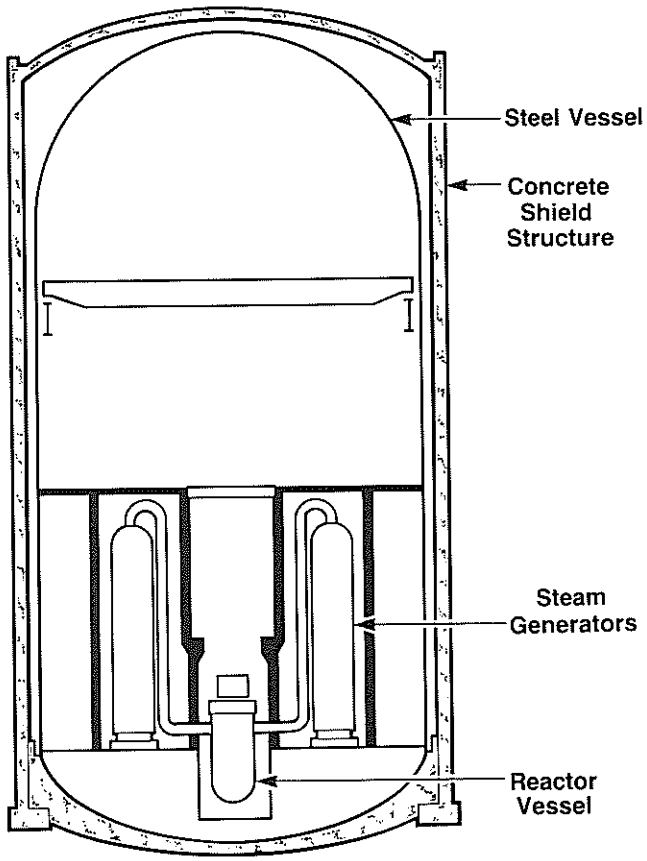

Figure 3 Dry containment: Steel vessel within concrete shield structure.

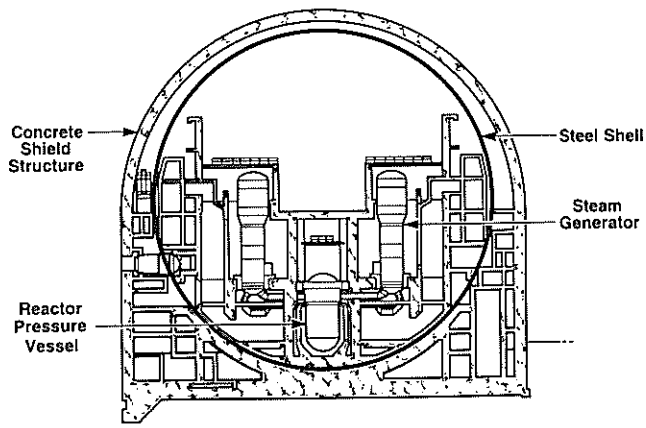

Figure 4 Dry containment: Spherical steel vessel within concrete shield structure.

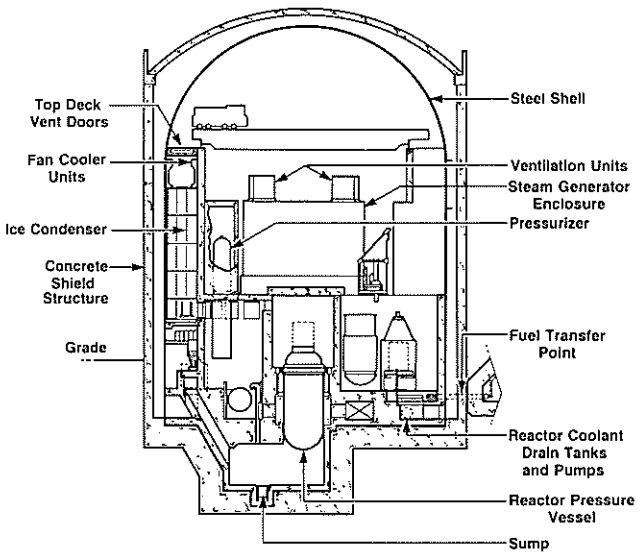

Figure 5 Ice condenser: Steel shell within concrete shield structure.

15,000 m^3) also reduce long-term pressurization by their heat storage capacity, and physically separate safety systems and the irradiated fuel storage pool.

The annulus between the steel containment and the outer concrete shielding (1.8 m thick), which is exhausted through a qualified filter system and stack, provides for additional deposition of radioactive products in the event of containment impairment. A subatmospheric pressure system is designed to direct flows from compartments having lower activity to those with higher activity following any accident.

Another double-barrier annulus concept, developed in France, includes a cylindrical concrete containment lined with steel and an outer concrete shield. Recently, France has developed a design for 1300 MWe plants which does not require the steel liner.

Two types of pressure control containments have been developed for PWR's, the subatmosphere containment (−5.0 psig operating pressure), and the ice condenser. A typical ice condenser containment is shown in Figure 5. Steam and air resulting from an accident is forced by the pressure from the lower compartment through the ice beds where the steam is condensed. The design pressure for this containment is one bar, whereas a PWR dry containment for the same rating would range from three to five bar. However, current economic considerations have limited this design to 1000 MWe units and larger.

Boiling Water Reactor Containments

All modern BWR containments are of the pressure-suppression type, incorporating drywells and wetwells as pressure-suppression chambers. Following a LOCA, the steam/water flow causes a rapid increase of pressure and temperature in the drywell. The pressure difference between the dry and wet wells forces the contained water out of the blowdown pipes, and high pressure steam then flows to the wetwell pool. Steam condensation occurs and non-condensible gases collect in the wetwell airspace or compression chamber. Given the relatively small containment volume of BWR's compared with other reactor systems, this condensation process is the key element in limiting maximum pressures to 3 bar or less.

During the last thirty years there have been progressive changes to the shape, geometry, size, and location of the various suppression chambers relative to the reactor core within containment. The latest Mark III General Electric design is shown in Figure 6. The quenching pool has been moved to the side, whereas in the previous Mark II design it was underneath the reactor vessel. This made it possible to reduce the elevation of the reactor vessel, and created the best compromise with regard to the height of the vessel, its accessibility, and construction of containment. The design shown in Figure 6 uses a steel containment within a concrete shield. However, because of localized dynamic loading from the wetwell during LOCA and Safety Relief Valve discharge, the steel containment was replaced by a hybrid shell in later designs. This hybrid employs a concrete base mat, a concrete shell in the pool region with a steel containment shell above the pool. In future applications of U.S.-built BWR's, it is anticipated that a full-height reinforced concrete shell would be the preferred arrangement, which is also the practice in other countries.

Modern large BWR's typified by the U.S. General Electric Mark III design, the Gundremmingen KRB-2 1300 MWe units in West Germany, and the Swedish BWR-75 1000 MWe units, have steel liners and cylindrical pre-stressed concrete containment structures. The

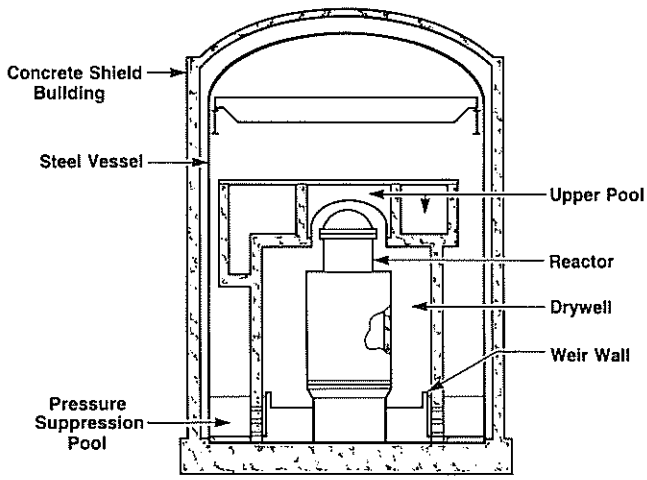

Figure 6 Pressure-suppression containment: Steel vessel within concrete shield building (BWR Mark III containment).

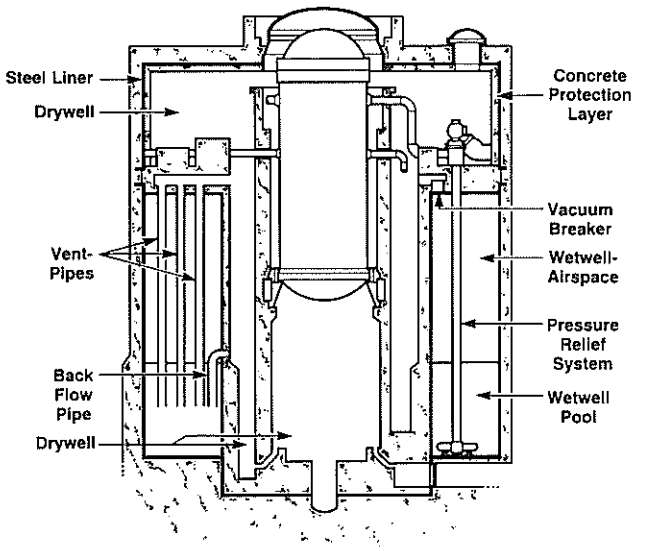

Figure 7 KRB-2 pressure-suppression containment (BWR concept).

German and Swedish designs, however, have retained features of the Mark II concept, where the drywell is both under and over the reactor vessel. The overall objective of these systems is to maintain low design pressure with relatively small containment volumes, and to provide for an emergency condenser during plant transients and accidents.

In West Germany, earlier BWR containments were a spherical steel shell. The current KRB-2 design is changed as shown in Figure 7. It consists of a cylindrical pre-stressed concrete structure with an embedded steel liner that is protected by additional concrete. The drywell space surrounds the reactor vessel and heat transport piping extending to the second isolation valve. Many large-diameter vent pipes from the drywell extending into the pool provide the path to condense LOCA-induced steam/water mixtures. A separate pressure-relief system provides for coolant pressure control. The containment is protected from large wetwell overpressures relative to that in the drywell during LOCA by vacuum breaker swing check valves that allow pressure equalization in the two chambers. The suppression system design pressure is typically 4 bar compared with maximum expected LOCA pressures of less than 3 bar. The wall of the reactor building serves as a secondary containment, and the annular space between it and containment is sub-atmospheric, to prevent leakage to the environment. German regulatory authorities require the reactor building walls to withstand an external blast wave of 0.45 bar, a site-dependent earthquake, and the crash impact of military aircraft. To provide further assurance of containment integrity from external events, the reactor buildings are not rigidly joined, apart from the common foundation.

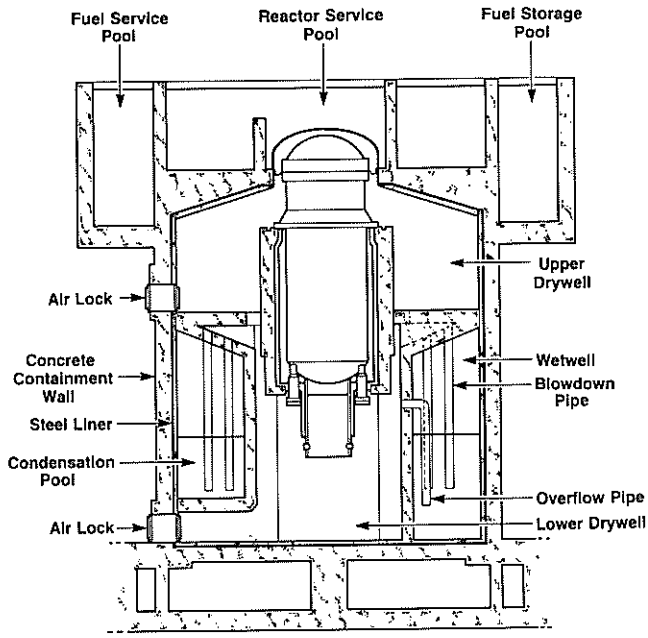

Figure 8 Swedish BWR 75 containment: Steel-lined concrete shield structure.

Figure 8 shows a sectional view of the Swedish BWR 75 containment [4], which is a reinforced, partly pre-stressed concrete cylinder provided with an embedded liner of carbon steel. The drywell, wetwell, and blowdown pipes are similarly arranged to the German KRB design, and the entire containment is totally steel lined. A different labyrinth arrangement exists between the upper drywell and wetwell than in the West German design. The containment and reactor building basement structure is different, but each design has no structural tie (other than expansion joints) between containment and adjacent buildings. The steel liner embedment of between 20 to 30 cms within the concrete is deeper than the KRB containment. The upper drywell contains primary and secondary reactor process systems, including main steam, feed water, and containment cooling systems. The lower drywell contains systems such as the control rod drives and recirculation pump motors. The wetwell is an annular enclosure. Blow-out panels in the lower part of the reactor concrete shield provide a path to the lower drywell in the event of a LOCA within the reactor compartment.

Pressurized Heavy Water Reactor Containment

This section concentrates on the CANDU containment system associated with the multi-unit stations in Canada [3]. The single 600 MWe units designed by AECL use similar negative pressure containment (NPC systems, with the omission of a vacuum building.

The NPC1 design concept, (where reactor units are

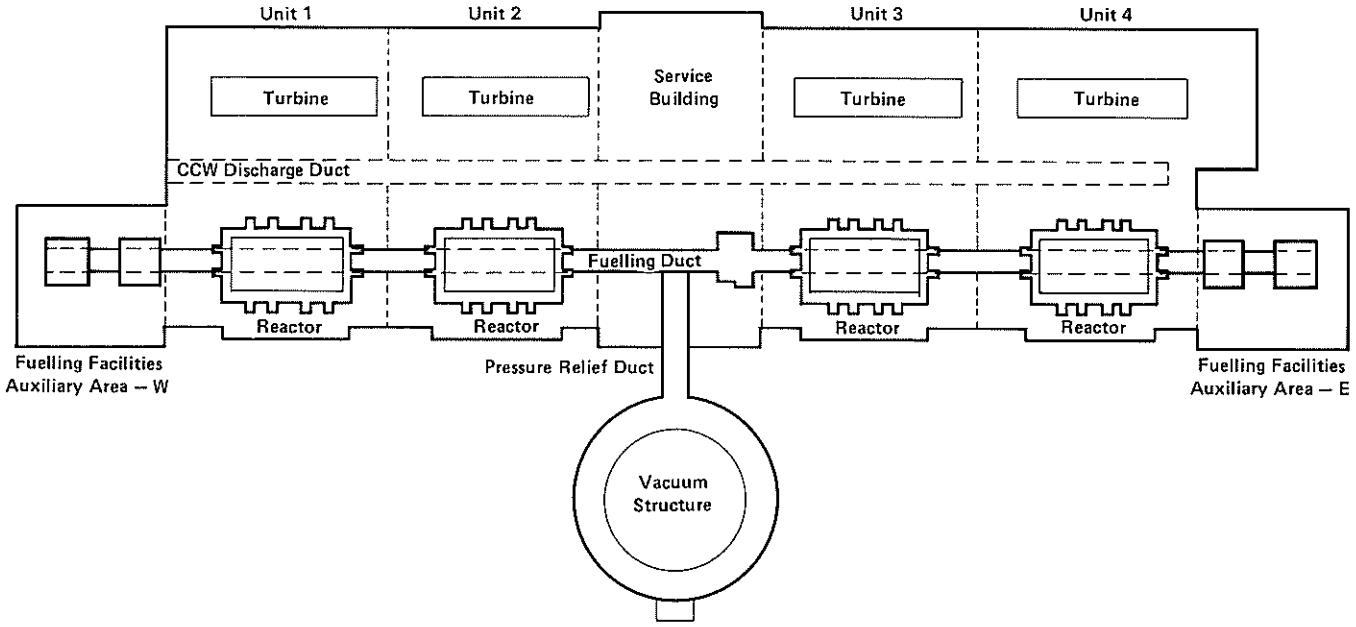

Figure 9 Multi-unit CANDU containment.

isolated from one another), is used in the eight-unit Pickering NGS, which came into service in the period 1971–1985. The second major type (NPC2), was used in the four-unit Bruce NGS A, which came into service in 1977–1979, and in all subsequent four-unit stations.

The prime difference between the NPC1 and NPC2 concepts is that the latter locates most of the supporting process equipment outside the primary containment envelope, although it follows that some equipment must be in secondary confinement areas. Another feature of NPC2 is that the four reactor vaults are interconnected during normal operation due to the choice of common on-power fuelling systems for all units.

The main reason for adoption of the NPC containment concept was increased effectiveness required to satisfy concerns for relative close population siting that existed at the time of the Pickering NGS A project commitment. The NPC2 design was developed primarily to improve maintenance access to process equipment during operation.

The basic operating principle of negative pressure containment is to maintain a negative pressure such that air leakage through the structure is inward. Any discharge required to maintain this negative pressure differential is along defined pathways that can be filtered, treated, and monitored to control releases to the environment.

Figure 9 shows the NPC2 containment envelope, which is normally at sub-atmospheric pressure. In the event of a LOCA, various systems act to provide for short- and long-term pressure and effluent control. The short term period extends from the LOCA, when very fast pressure transients are experienced with possible 'puff' releases of radioactivity, to the re-establishment of sub-atmospheric pressure within containment. The long term period is associated with the initial activation of the Emergency Filtered Air Discharge System (EFADS) until cleanup operations are complete. EFADS is manually activated when containment pressure approaches atmospheric several days after the event. Figure 10 lists the systems that collectively perform the containment function in the two time frames.

The principles of pressure control used in the CANDU NPC2 containment in the short term are 'pressure relief' followed by 'steam-suppression' as depicted in Figure 11. Following LOCA, the reactor vaults and fuelling duct connecting the multi-unit station are pressurized by the resulting high-enthalpy fluid flashing to steam. The extent of pressure rise is limited by the very large volume of the containment envelope. The increase in pressure, acting across the Pressure Relief Valve (PRV) pistons, automatically opens the valves and releases the air-steam mixture into the vacuum building (VB).

The steam suppression function is carried out by a dousing system located in the vacuum building. When the PRV's open and VB pressure rises, water is forced over a weir structure and into spray headers located under the dousing tank. The spray water falls through the steam-air mixture, reduces pressure, and provides for soluble fission-product retention.

The principle of effluent control used in the short

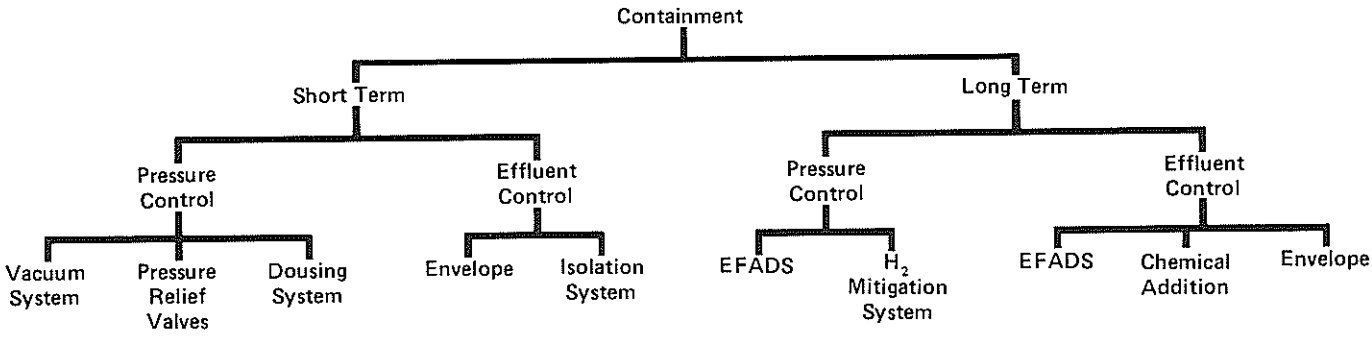

Figure 10 CANDU containment systems.

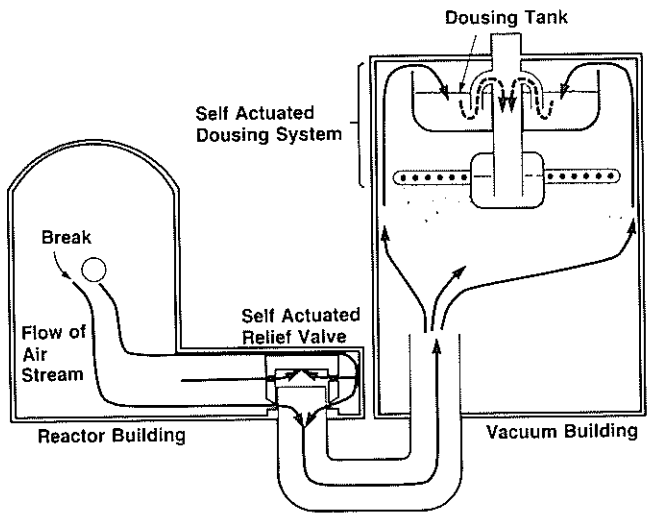

Figure 11 Operation–Negative-pressure containment.

term is isolation by physical barriers. Containment operates at 98 kPa (− .5 psig) and the vacuum building at 7 kPa (− 13.7 psig). Typical design pressures for containment are 170–200 kPa (10–14 psig) and 50 kPa (− 7 psig) for the VB.

Containment Capability Studies

Over many years, there have been numerous containment studies [2, 3, 11, 13] performed with the aim of establishing design parameters, proving that regulatory limits for design basis events are met, and identifying ultimate capability to withstand severe postulated accidents. Given that current research to provide 'best estimate' source terms and fission-product transport is also important to demonstrating containment capability, there is no doubt that high emphasis of nuclear reactor safety R&D today, is on containment systems.

This section summarizes this containment R&D, and provides a few examples of the many studies performed in West Germany, Sweden, the U.S.A., and Canada to demonstrate containment capability for LWR and PHWR nuclear stations. Present studies are largely associated with very low-frequency ($\leq 10^{-7}$ events per year), high-consequence events, since it is generally recognized that all containment systems are adequately designed for likely accidents.

Containment Research

Tables 4 and 5 provide a synopsis of typical integral containment tests for PWR's and BWR's to verify containment analysis codes and assure adequate designs. Many experiments have been performed elsewhere, notably in Japan.

In addition to those integral tests, there have also been numerous separate-effects tests performed in all countries (often involving international collaboration, as at Marviken) to understand jet impingement loads, vent flows, and condensation heat transfer. Experiments [17, 18] to determine the effects of external missiles (including large steel piping and segments of a turbine rotor) impacting on containment have been performed in the U.S. and elsewhere.

In Canada, as elsewhere, there were a number of on-site containment tests during the period 1970–1983, conducted by AECL and Ontario Hydro, to determine leakage rates and the thermal utilization of dousing flow in the Vacuum Buildings, and/or containment. In addition separate-effects tests of all containments were performed over the period 1960–1984, to understand transient compressible flow in interconnected volumes, jet loading, tee-junction losses, vessel-pipe fluid mixing, and liquid-steam phase separation at tee junctions.

The majority of current containment research is centred on the ultimate capability of LWR systems when subjected to severe accidents in the Class 9 category, as typified by the *idcor* program. The Industry Degraded Core Rulemaking (IDCOR) Program in the U.S. is supported by 62 nuclear utilities, architect-engineers, LWR vendors in the United States, and by Japan and Sweden. The IDCOR mission was to develop

Table 4: Integral PWR Containment Experiments

Year	*Facility*	*Measurement purpose*	*Specific information*
1965	CSE, U.S.A.	Vessel blowdown	Fission product transport and removal
1970	CVTR, INEL, U.S.A.	Peak pressure and temperature effects	Axial wall temperature distribution, heat transfer coefficients
1975	Battelle, Frankfurt	Pressure and temperature measurements during blowdown	Pressure waves, wall temperatures, H/T coefficients, jet impingement and hydrogen distribution
1981	Lucal Heights	Pressure/temperature response, small steel containment	Compartment pressure/temperature and heat transfer
1982	HDR, Karlstein	Blowdown for different break sizes/locations	Wall temperatures, steam-air concentrations, jet impingement, strains accelerations
1983–6	Sandia Nat. Labs., NM	Failure conditions modes beyond DBA	Structural failure mode, leakage paths, penetrations behaviour, base mat melt, bypass, margins

Table 5: BWR Containment Experiments

Year	*Facility*	*Measurement, purpose*
1960s	Humboldt Bay Bodega Bay	Drywell, wetwell pressure transients
1972/73	Marviken, Sweden	Full scale containment tests
1972,75	GKM 1, KKB	Vent pipe loads, full-scale
1975,77	Karlstein large tank and concrete cells	Multivent pipe tests
1976/77	GKM 2S	Vent pipe and pool wall loads, condensation, transient and static tests
1978/80	Studsvik, Sweden	Pool swell in different geometries
1984	GKSS	Vent clearing, pool swell and fall back
1983,86	Sandia Nat. Labs, NM	Large scale, Mk I, II, III overpressure tests, failure mode/timing, and design margins

a comprehensive, technically sound position on the issues related to potential severe accidents in light water power reactors.

IDCOR resulted from the USNRC's evaluation of the TMI-2 degraded core condition, which was more severe than that previously assumed in a design-basis accident. In October 1980, the NRC initiated a 'long-term rulemaking to consider to what extent, if any, nuclear power plants should be designed to deal effectively with degraded core and core melt accidents.' The NRC's rulemaking proposed to address the objectives and content of a degraded core-related regulation, the related design and operational improvements under consideration, their effects on other safety considerations, and the costs and benefits of design and operational improvements.

Subsequently, the NRC issued a proposed Commission Policy Statement [19], to implement the 2 October 1980, 'Advance Notice of Rulemaking,' and identify the severe accident decision process on specific stan-

Table 6: Severe Accident Phenomena Addressed by IDCOR to Establish Ultimate Containment Capability

1	Steam explosions causing pressure pulses; liquid slugs or missiles
2	Overpressure due to rapid steam generation
3	Overpressure due to hydrogen generation combustion
4	Containment by-pass via interface systems to environment
5	Overpressure due to noncondensable gases
6	Melt-through of containment base mat
7	Overpressure due to loss of containment heat removal
8	Containment failure modes
9	Radionuclide release and transport

dard plant designs, and on other classes of existing plants, which may or may not include rulemaking.

IDCOR identified key issues and phenomena, developed analytical methods, analyzed the severe accident behaviour of four representative plants, and extended the results as generically as possible. The methods used in the study were 'best-estimate,' rather than the conservative engineering approaches in technical analysis usually characteristic of licensing submissions. Existing methods and experimental data were thoroughly reviewed and new programs were undertaken where confident support of prior positions was uncertain. In general, IDCOR has demonstrated that consequences of dominant severe-accident sequences are significantly less than previously anticipated. Most accident sequences require long times to progress, allowing time to achieve safe stable states. Table 6 lists the reactor safety phenomena considered in reaching these conclusions.

While studies continue, the most important results to date are: containment overpressure capability is several times the pressure associated with the design basis accident; limited impairments of the containment envelope would likely occur on failure, thus stabilizing or gradually reducing pressures, which would limit the rate of radioactive release; hydrogen-related concerns can be mitigated or do not exist; and early failures of containment due to all causes are most unlikely, thus permitting sufficient time for interdictory actions.

Corium and fissium experiments [20, 21] are also on-going in the U.S., West Germany, and Sweden. Concrete-corium interaction tests to determine the extent of base-mat erosion are continuing in West Germany and U.S.A. Current experiments at Sandia National Laboratories [13, 14, 15, 16] sponsored by USNRC are addressing the issue of 'when, where and how' various steel and concrete containments will fail, and the resultant extent of radioactive release. Large-scale models have been or are being constructed to identify containment safety margins, and the integrity of containment pipe and electrical penetration assemblies when subject to overpressure loads. This large program is scheduled for completion by the end of 1986. In West Germany [2], studies suggest that the

Table 7: IAEA Review of Hydrogen Studies

1	Hydrogen distribution in containment
2	Lower flammability limits
3	Combustion limits of H_2-air-steam-CO_2 mixtures
4	Available hydrogen and oxygen detectors
5	Pre-inerting as a mitigation scheme
6	Effectiveness of various ignition sources
7	Controlled burning and extinguishing systems
8	Fog/spray suppression
9	Minimum equipment to survive degraded-core accident.

weak point of their PWR containment is associated with the sealing box which is part of the main airlock, in the event of overpressure accompanying a Class 9 accident. It is considered that the failure mode will be 'leak instead of break,' which will either result in a maximum stabilized containment pressure below ultimate capability, or reducing pressure. In other words, containment-pressure relief will occur rather than gross containment failure. Experiments to prove this engineering assessment are now being planned [2].

Another area of research of importance to all nuclear power systems, and prompted by TMI-2, is that of ensuring control of hydrogen generation in severe accidents. This subject is the focus of attention of a current IAEA working group who are reviewing the issues identified in Table 7, using information from the major investigations already carried out by EPRI, Sandia Labs, and WNRE.

These studies are confirming that hydrogen recombiners or igniters for controlled burning, will prevent large containment overpressures. In many containments, the predicted volumetric concentration of hydrogen is far too low for combustion to occur.

Containment Response Analyses

The objectives of containment analysis are to establish design parameters and to verify that regulatory dose limits are not exceeded following any process system failure that leads to a release of radioactive material within the containment envelope. Design and regulatory processes require that containment response be analysed for a large number and variety of postulated system pipe failures ranging from a small leak up to a guillotine failure of the largest piping in the heat transport system.

PHWR Analyses

For CANDU reactors, accidents are characterized according to the postulated LOCA break-discharge rate, since this parameter has the dominant effect on subsequent containment response. A coolant channel end-fitting failure is used to bound the radiological consequences of small breaks in the heat transport system piping. The accident sequence postulated is an instantaneous maximum opening break, with the resultant ejection of all 13 fuel bundles from the channel.

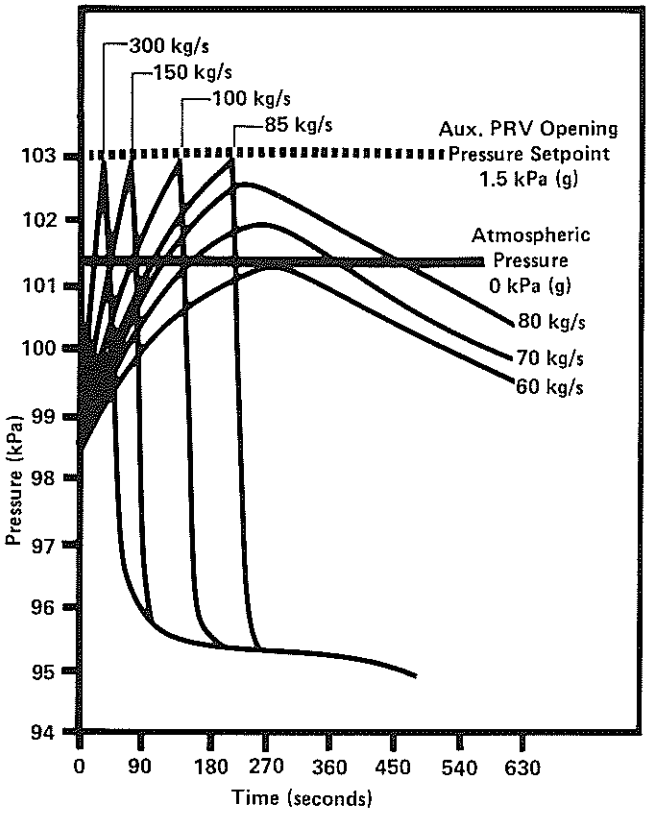

Figure 12 Negative-pressure containment response (small break).

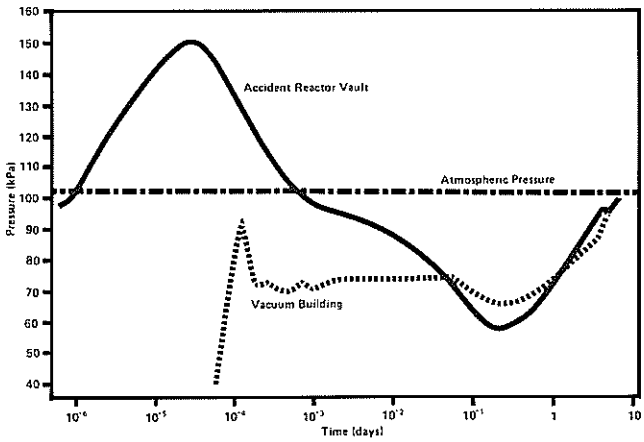

Figure 13 Negative-pressure containment response (large break).

Severance of an end fitting results in an initial coolant discharge rate up to 200 kg / s. The ejected fuel bundles will likely be damaged on impact with the reactor vault, and will release fission products into containment at a rate dependent on the extent of fuel cooling.

The containment pressure due to small breaks is strongly affected by containment heat sinks and, in particular, by the number of vault air-cooling units assumed operational at the time of the break. Figure 12 shows containment-pressure transients for various initial-break discharge rates. For small breaks above 80 kg / s, the duration of the overpressure period is determined by the time for the pressure relief manifold to pressurize to the setpoint of the pressure relief valves (PRV's). Below this discharge rate, energy removal due to air coolers and condensation on cold surfaces is sufficient to offset the energy addition from the break, with the result that the containment pressure can remain slightly above atmospheric without initiating PRV opening. The containment overpressure period will then last until either the break-energy discharge rate decreases sufficiently that the heat sinks are able to reduce the pressure to sub-atmospheric by steam condensation, or the operator manually intervenes by switching the PRV's to control mode. In spite of the potentially extended containment overpressure period for certain small breaks, releases into the environment are very small, since fuel damage is limited to a fraction of the core inventory.

Certain large breaks in the heat transport system, which could result in coolant stagnation within fuel channels, are capable of producing extensive fuel failures throughout the core. In addition, the initial pressure excursion presents a challenge to containment integrity. Figure 13 shows the estimated pressure transients in the accident vault and vacuum building following a postulated guillotine break in a pump suction line.

In this event, a peak pressure of 150 kPa occurs in the accident vault at less than 3 seconds. This is substantially below the containment design pressure. One minute after the break, the accident reactor vault becomes sub-atmospheric. From this time onward, the heat-removal rate exceeds the steaming rate at the break. The containment atmosphere continues to cool down and depressurize, until in the long term it becomes repressurized by air in-leakage, instrument air, and any gas evolution within the containment envelope.

Even with the fuel cladding damaged, the fission product release from the fuel is initially limited to a gradual escape of the 'free' inventory of volatiles. Only when the fuel heats up to high temperatures (well in excess of 1000°C) can a significant amount of volatiles start escaping from the 'bound' inventory. Thus, the concentration of activity in containment takes some time to build up to appreciable levels. With the effective pressure suppression provided by the NPC system, the amount of activity escaping to the environment by pressure-driven leakage is correspondingly small.

The bulk of activity enters the containment during the sub-atmospheric holdup period. These fission products then experience decay and undergo numerous interactions before a small portion is gradually

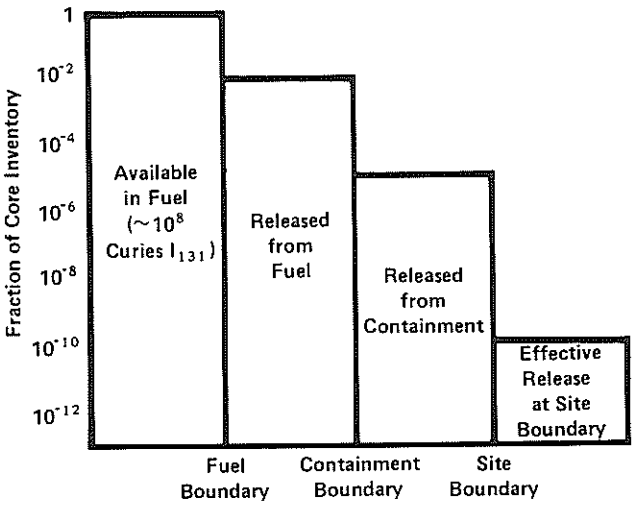

Figure 14 Typical attenuation CANDU (large break).

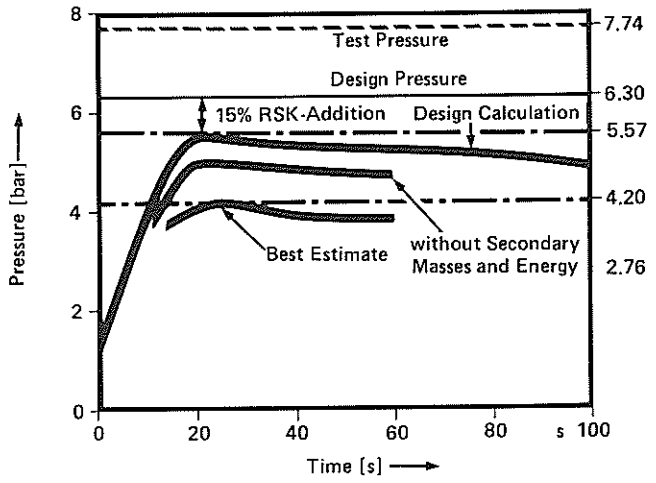

Figure 15 DBA pressure transients for standard 1300 MWE German PWR.

released by (EFAD) filtered venting. With the exception of noble gases, the fission products become trapped in water within containment, either by dissolution in liquid droplets or by becoming nucleation centres for liquid aerosols. Eventually, the airborne activity consists of only the noble gases and a small amount of volatile chemical compounds (e.g. organic iodides) in equilibrium with the solution on the floor.

Figure 14 illustrates the mitigating processes of CANDU containment systems in terms of I^{131}- attenuation resulting from the largest LOCA. Assuming the most adverse weather conditions, the 'effective release' is an equivalent amount of I^{131} that an individual could receive if present at the exclusion boundary for several months. The total attenuation for this severe accident is at least ten orders of magnitude, and the resultant dose, if individuals remained indefinitely at the site boundary, is within regulatory requirements.

In Canada, the current emphasis is on studies to delay or reduce the extent of containment venting even though regulatory limits are met. Regulatory requirements demand that dual-failure dose limits not be exceeded for LOCA's coincident with various containment impairments, including failure of isolation dampers, simultaneous deflation of four airlock seals on a double-door system, failure of pressure-relief valves, and loss of reactor vault air-cooling units.

PWR Analyses

As an example, the containment-response analysis performed in West Germany [2] to establish DBA parameters, and capability in severely degraded core accidents for the standard 1300 MWe PWR, is next described. While assumptions required by the German Advisory Committee on Reactor Safeguards (RSK) are not the same as those in the United States, the analysis results are generally typical of most PWR's. Also, while the extent and timing of BWR severe accidents is different from PWR's, the questions on ultimate containment capability are not dissimilar. The German containment design-basis accident (DBA) is a double-ended break in a main coolant pipe. For containment design purposes, RSK also requires a number of conservative assumptions which include:

(a) decay heat according to ANS Standard plus 20 per cent;
(b) maximum LOCA pressure assuming a 2 per cent decrease in containment volume, and a 2 per cent increase in primary and secondary circuit volumes (blowdown mass and energy include one secondary steam generator content);
(c) a 15 per cent safety margin applied to calculated maximum LOCA pressure;
(d) the steel containment shell to be designed for maximum containment temperature (145°C), rather than its expected temperature (60°C).

Figure 15 shows that the 'best estimate' of the maximum LOCA pressure will be 4.2 bar. Also shown, are the design calculation results, assuming (a), (b), and (d) above, for the pressure transients when the additional energy from the assumed secondary break is either excluded or included. The containment design pressure of 6.3 bar includes the additional assumption (c). The containment test pressure of 7.74 bar, prior to reactor criticality, is set by the difference in yield at the testing and LOCA temperatures. Thus, there is a substantial margin (up to 84 per cent) between the expected LOCA pressure and the demonstrated test capability.

In the event of a core melt-down, there would be a substantial release of fission products and steam/water to containment. The extent of release to the environment is highly dependent on the containment isolation

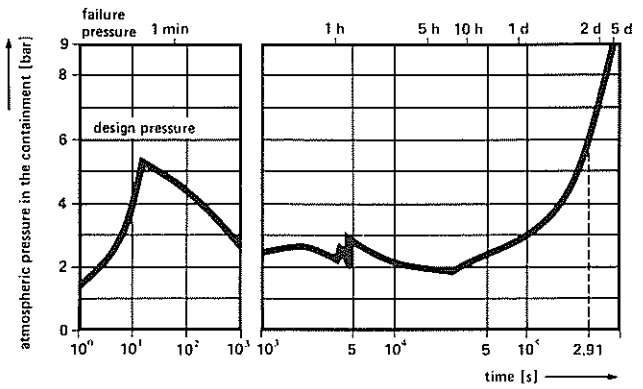

Figure 16 German PWR containment in core melt accompanied by failure of the sump cooling system.

time, the extent and nature of any containment leakage, and the transport and driving force paths from the annular space between containment and the reactor building. A very important mitigating process in this regard is the finding [22] that all radioactive substances, with the exception of the noble gases and airborne gaseous iodine, are bound to aerosol particles and subject to highly effective removal mechanisms. These removal mechanisms, involving plate-out and absorption on containment structures, reduce aerosol mass concentrations by five to six orders of magnitude within five days. Containment overpressure failure at the weakest point is not expected during this period.

Figure 16 shows the predicted pressure variation in containment for a core melt-down sequence. The maximum transient pressure during blowdown of 5.3 bar is reached at 17 seconds after LOCA. The transient LOCA pressure, which is relieved by the containment volume and condensation, reduces to 2.5 bar during the next 10^3 seconds. The core melt-down process due to the onset of evaporation of the moderator water and assumed complete absence of emergency forced cooling, commences at about 20 minutes.

Core degradation then proceeds, accompanied by hydrogen production due to steam/zirconium reactions. Shortly after one hour, the core structure is predicted to fail, allowing significant amounts of core material to drop into the water contained in the lower plenum of the reactor vessel, with the resultant violent evolution of steam.

Subsequently, at about 1.9 hours after blowdown, reactor vessel failure is predicted to occur, and core melt interaction with the concrete basemat begins. Given that 80 tonnes of metallic melt and 130 tonnes of oxide melt at a temperature of 2400°C are assumed available, it is predicted that the concrete shielding surrounding the reactor vessel will be eroded within 7 hours, causing the containment sump water to contact the melt. Violent evaporation of the sump water in the isolated containment subsequently results in its pressurization to design pressure after 3 days, and to 9 bar after 5 days, as shown in Figure 16. This sequence of events raises the question of ultimate containment overpressure capability, the mode and extent of containment failure, and subsequent extent of radioactive release from the annulus between containment and the reactor building, via filtration to the environment. These questions also highlight the 'defense-in-depth' provided in LWR stations for public protection, and the extended time available for any necessary emergency evacuation.

Recent studies in West Germany [2], and those of IDCOR [9] in the United States, are showing that containment overpressure capability before failure is up to 2–3 times design pressure for the undisturbed steel shell. As an example, West German experts predict overpressure failure of the 1300 MWe standard PWR containment to be above 14 bar in this case, and that the mode of failure will be 'leak instead of break.'

It is considered that containment overpressure will result in a leak at weak points, such as in main airlock components or at electrical/pipe penetrations, and will either permit a stabilization or reduction of containment pressures. The net result is the maintenance of major containment integrity, and only gradual activity release (likely after filtration), to the environment. Also, recent West German studies and those in the United States have demonstrated that previously assessed source terms are too high by several orders of magnitude.

Summary

This paper has discussed the functional requirements, the evolution of designs, and the influence of national regulatory requirements on containments for PWR, BWR, and PHWR reactors. Particular containment designs are not only a function of national siting requirements in the United States, West Germany, Sweden, and Canada, but also relate to specific reactor system performance in perceived accident conditions, and the number of reactors constituting the generating station. In all cases, it is evident that the various containment systems easily meet their design-basis accidents. Since the TMI-2 accident, all jurisdictions have examined the need for design changes to meet post-accident scenarios.

In Canada, increasing attention has been paid to large stagnation LOCA's with assumed coincidental containment impairments. In this regard, methods to delay or reduce the extent of atmospheric venting of containment to relieve pressure are under active study, even though regulatory limits are met.

In West Germany, the United States, and Sweden, emphasis is on the ultimate capability of PWR and BWR containments to withstand overpressures and evaluate environmental releases for class 9 core melt acci-

dents, which are beyond, or bordering on the range of credible frequency. Current information from the U.S. based IDCOR study and the independent West German and Swedish research work, indicates that LWR containment designs are capable of withstanding overpressures up to three, and possibly four times their design pressure. In the event of containment failure, it is predicted that radioactive release will likely result from gradual leakage from weak points (rather than from a gross containment break) to the reactor building, resulting in a slow and delayed discharge to the environment. Experiments and scaled tests of containments have been performed, or are currently underway in many countries to determine ultimate containment ability and failure modes. These tests, together with allied fission-product source-term and transport tests, and comprehensive experiments on hydrogen generation and mitigation, are an important area of reactor safety research today. In addition, research in the U.S. is directed at decoupling LOCA plus SSE as a design basis, and reducing postulated high-energy system pipe breaks and loading phenomena.

If containment failure were to occur, in either the CANDU, PWR, or BWR reactor systems, it is predicted to do so many days following the most severe postulated accident, permitting adequate time for assurance of public safety.

Acknowledgements

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Chemistry Control at Bruce NGS 'B' from Construction to Commercial Operation

J.G. Roberts
Ontario Hydro
Bruce Nuclear Generating Station 'B'
Tiverton, Ontario
N0G 2T0

Abstract

Pre-operational storage chemistry and flushing of the secondary side is described. The approach devised for Bruce NGS 'B' Unit 6 was unique for an Ontario Hydro Nuclear Unit. The significance of the improved Construction installation and Quality Assurance procedures, combined with those of Operations is identified. Secondary side chemistry during both commissioning and later operation is reported. It will be shown that the application of ALARA (As Low As is Reasonably Achievable) concept has resulted in tighter chemical specifications being met.

Résumé

Ce document décrit le contrôle de la corrosion et l'enlèvement des débris de corrosin du côté secondaire avant la mise en exploitation commerciale. On y traite de la nouvelle approche adoptée par Ontario Hydro pour la tranche 6 de la centrale nucléaire Bruce 'B' ainsi que le rôle important joué par l'utilisation de procédures d'installation, d'assurance-qualité et d'exploitation améliorées. On y parle de la chimie secondaire à la fois pendant les étapes de mise en service et d'exploitation. Ce document démontre aussi comment l'application du concept ALARMA (niveau aussi bas qu'il est raisonnablement possible d'atteindre) a permis de satisfaire des normes chimiques plus sévères.

Introduction

Bruce NGS 'B' is a four-unit CANDU (CANadian Deuterium Uranium) Nuclear Generating Station (NGS) situated on the eastern shore of Lake Huron. This NGS was designed as a repeat of Bruce NGS 'A' with, among others, major changes to secondary side components. The condensate / feedwater train was designed to be all ferrous, and different turbine generators were supplied. Construction started in 1977 and commissioning of the last unit is presently under way. The first three units went critical in May 1984, November 1984, and January 1986. This paper will address secondary side chemical control from construction through to commercial operation for Unit 6, the first unit.

Construction Phase

All major secondary side components – feedwater heaters, de-aerators / storage tanks, external preheaters, boilers (steam generators), moisture separator / reheaters (MSRs), and steam turbines – were supplied, as specified, with controlled environments to minimize corrosion. All vessels were received with nitrogen blankets. The nitrogen blanket was monitored and maintained by our Construction forces until such time as the vessel was to be incorporated into the system [1]. All the steam turbine components arrived with protective coatings. All pipework for the secondary side is carbon steel. The designers had specified that all secondary side pipework must be pickled. The process employed, in sequence, baths of caustic, rinse water, phosphoric acid, rinse water and hot chromate. Following this treatment the pipework was allowed to dry, and was then capped. The resultant oxide was protective, providing that the pipework remained dry. Indoor storage of the treated capped pipework became the norm in 1981, following Operations' input. Routine inspection of the pipework showed the occasional presence of a white deposit. Analysis of this deposit showed it to be an iron phosphate (33.7% Iron, 33.9% Phosphate) [2]. Further investigation showed that there was inadequate chemical monitoring / control of the pickling plant. The deposit appeared only after the iron level in the phosphoric acid bath had increased to approximately 1.5%. As a result, the Operations' Chemical Laboratory supplied an analytical service; the phosphoric acid bath was changed after the iron

Keywords: chemistry, nuclear, boiler, ALARA, condensate, pickling, conditioning, steam generator, control, storage, commissioning, feedwater, debris, flushing, specification.

Table 1: Typical Hydrostatic Test Solution Analyses – Unit 6 Condensate Storage Tank [2]

Date	*Time*	*pH*	N_2H_4 *mg/kg*
82-01-13	0840	9.8	118
82-01-14	0950	10.0	150
82-01-18	1515	9.0	150
82-04-05	1000	10.3	120
82-04-19	1310	10.4	127

level reached 1.0%. Subsequent pipework inspections showed no sign of the white deposit previously observed. Later, an inhibited phosphoric acid was used in order to minimize both chemical and disposal costs.

Prior to installation, and after weld preparation, each section of pipe was blown out and all debris, including welding rods, removed. These steps were part of the excellent Quality Assurance program instituted by Bruce Construction [1]. The hydrostatic test of each section of the secondary side was performed with treated demineralized water. This was a major change from Bruce NGS 'A'. As these systems would remain wet stored for up to one year, without any recirculation, high pH and hydrazine levels were required to minimize corrosion. The treatment consisted of 150 mg/kg hydrazine and pH 10–10.5 with ammonia. The solutions were made in batches by Construction forces, checked by the Operation's Chemical Laboratory, and then added to the system under test. See Table 1 for typical analyses.

Whenever possible the hydrostatic test solution would remain in the system until commissioning got under way. Whenever wet storage was not possible the pipework was drained, low points tapped, and dried.

All boilers and preheaters were hydrostatically tested with wet lay-up solutions and subsequently wet stored. After much discussion with Design, valves were added to the boiler steam and blowdown systems to allow recirculation of the boiler wet store solutions. This was achieved via temporary pipework and pumps supplied by Operations. (See Figure 1.) Each boiler's solution was recirculated for four hours, then sampled, analysed, and dosed as required. Once the lay-up solution was within specification, the next boiler's solution would be recirculated. Two boilers, one from each bank, could have their solutions recirculated at any one time. Table 2 shows typical results. Recently, a permanently piped system has been installed to allow wet storage of boilers during outages.

Ontario Hydro Nuclear Generating units under construction prior to Bruce NGS 'B' had flushes of the secondary side to remove debris and form a protective oxide. These flushes, part of the construction process, bypassed all feedheaters, boiler feed pumps, and preheaters by the use of temporary pipework. In addition, the flow path was reversed for part of the system. This required non-return valves to be stripped to allow flow in the reverse direction. Unfortunately, the removal of this temporary pipework took a great deal of time. This negated any benefit as the drained,

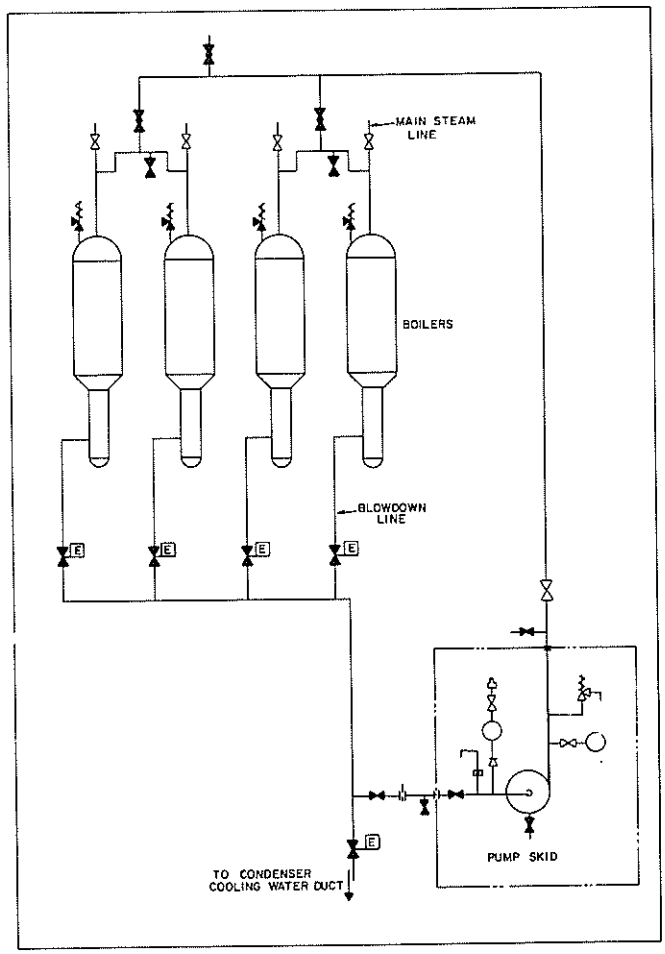

Figure 1 Boiler Wet Storage Recirculation Schematic

Table 2: Typical Wet Storage Chemical Data [3]

Date	*Time*	*Boiler*	*pH*	N_2H_4	*Action*
84-01-25	0130	1	10.1	80	
		5	9.9	104	Add 20 l NH_4OH
	0530	2	10.0	42	
		5	10.2		
	0830	2	10.0	49	
		6	10.0	51	
	1300	3	10.1	13	Add 1 pail N_2H_4
		7	9.7	16	Add 1 pail N_2H_4
84-01-26	0030	3	10.2	95	
		7	9.9	78	Add 20 l NH_4OH
	0330	4	9.9	53	
		7	10.1	66	
	0530	8	9.8	50	Add 20 l NH_4OH
	1100	8	10.2		
	2100	1	10.0	53	
		5	10.0	77	

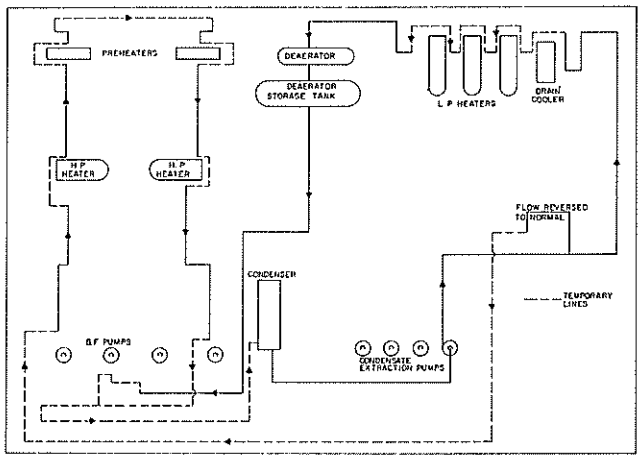

Figure 2 Original Alkaline Flush Flowpath

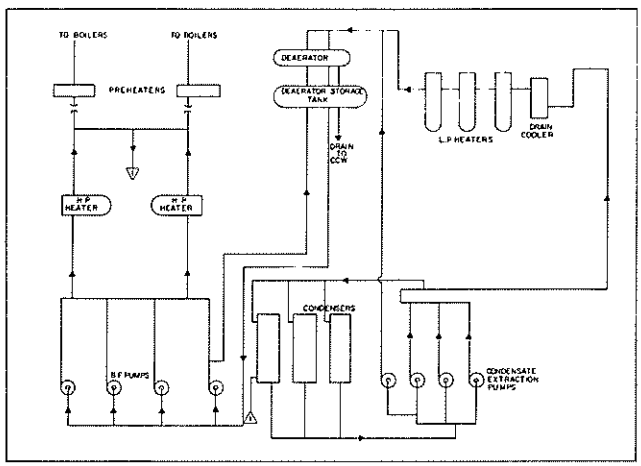

Figure 3 Flowpath Used for Alkaline Flush

wet pipework was exposed to the atmosphere for approximately three months. At Bruce NGS 'B' an alternative was proposed by Operations that was accepted, and the process redesigned.

The alternative flushes eliminated virtually all temporary pipework and allowed forward flow through feed heaters and boiler feed pumps. In addition, maximum velocities could be used, non-return valves would remain intact, and the Construction schedule would be shortened. The effectiveness of the oxide-formation flush was questioned, particularly due to the low maximum temperature allowed.

Figure 2 outlines the originally specified flow path and temporary pipework, whereas Figure 3 illustrates the flow path used.

Tests carried out at Ontario Hydro Research [4] demonstrated that at 60°C, the maximum allowed temperature for the Condensate Extraction Pump (CEP) glands, little benefit would be gained. It was agreed to test the process on the system and monitor the process via corrosion coupons.

Two flushes were to be carried out; the first to flush out debris and the second to attempt to form a protective oxide. The first flush basically consisted of a bleed-and-feed operation. The system would be recirculated continuously while being bled via the de-aerator (D/A) storage tank drain line, and fed via the normal demineralized water makeup routes to the condensers. Each condenser, and the D/A storage tank, had weirs installed to assist with removal of suspended solids.

Environmental Considerations

The licence to discharge cooling water allowed solutions containing up to 0.5 mg/kg hydrazine to be discharged into the condenser cooling water (CCW). Negotiations had previously been conducted with the Ministry of the Environment to raise the allowable concentration of hydrazine to 50 mg/kg. These solutions would be discharged into CCW, providing the limit of 0.5 mg/kg hydrazine in the total discharge was not exceeded. This would allow the alkaline flush to proceed as perceived; also, it would permit wet storage of boilers and secondary-side pipework as a routine during commissioning and operation.

Commissioning

Alkaline Flush

The hydrazine level fell rapidly as the solution was recirculated. The pH was maintained > 9.5 by ammonia addition. After six days the system was declared clean when the suspended solids in each were < 10 μg/kg.

At this stage the system was drained. All the vessels in the loop were opened up, photographed, cleaned out and parts were installed as required and photographed again. Then the system was refilled with dosed demineralized water. This took seven days. Now the second stage of the flush began. The maximum temperature, 60°C, as dictated by the CEP glands, was maintained throughout this flush. The hydrazine level was maintained with considerable difficulty – 5,700 kg of 35% hydrazine were used.

After 96 hours the process was considered complete. Evaluation of corrosion coupons [5] confirmed that the process was not successful in forming a protective oxide layer. As a result, a second cold flush is now carried out, in place of the original hot flush, to remove any corrosion products formed during the drain-down period.

The result of the flushes was that the system was cleaned and 70 kg (⅓ of a 200 litre drum) of material removed [6]. All the equipment that had been flushed was now wet stored. This lay-up solution was recirculated on a regular basis and redosed as necessary.

Table 3: Typical Boiler and Condensate Chemistry During Hot Conditioning of Unit 6 (27 May 1983) [3]

Sample Point	pH	Conductivity mS/m	Hydrazine m/kg	Sodium µg kg	Chloride µg kg	Nitrite µg kg	Nitrate µg kg	Sulphate µg kg	CRUD µg kg
Boiler 1	9.4	5.0	11	370	355	< 20	< 20	555	150
2	9.2	5.1	12	290	117	33	66	286	100
3	9.2	3.4	27	340	521	< 10	76	388	
4	9.3	4.9	29	515	224	< 20	128	670	150
5	9.1	4.9	11	590	254	< 20	26	189	200
6	9.5	4.5	13	500	253	< 20	50	531	200
7	9.5	4.8	26	570	252	< 20	87	586	200
8	9.5	5.2	27	405	202	< 20	74	499	200
CEP	9.8	6.3	72	< 10					
BFP	9.7	8.5	19	< 10					

Hot Conditioning

Hot Conditioning of the Heat Transport System was the next event that required the secondary side to be in operation. This would involve running the secondary side continually for 5+ days. During this time there would be no vacuum on the condenser. The heat input to the boilers would be insufficient to generate enough steam to pull, and maintain, condenser vacuum. Besides, the vapour barrier between the condenser and turbine to prevent moisture rising to the turbine would have been compromised.

During hot conditioning the secondary side pH and hydrazine specifications were > 9.3 and > 10 mg/kg, respectively. In effect, the hydrazine ranged between 0.6 and 150 mg/kg at the boiler feed pump. Bulk dosing of the system was essential in order to maintain > 10 mg/kg. The dosing pumps were incapable of maintaining these levels, which were 1,000 times greater than normal operational levels.

Table 3 shows typical values for secondary side chemical parameters during hot conditioning. This was the first time that the secondary side had flow to the boilers, and that steam was generated.

Following Hot Conditioning, a boiler inspection was carried out. It showed that the internals were in excellent condition. Subsequently the boilers, condensate, and boiler feed systems were wet stored.

At this time the temporary boiler recirculation equipment came into its own – a problem with the fuel channel assemblies was identified that resulted in a considerable delay to the commissioning program. During this time the boilers were wet stored and the solutions frequently recirculated.

Prior to criticality all systems were given a final rundown called Hot Performance. Secondary side chemistry control was identical to that for Hot Conditioning. This chemical status remained through criticality and Phase 'B' commissioning. During Phase 'C' commissioning reactor power was raised above 1%. At approximately 5% power, steam was put to the turbine and vacuum drawn on the condensers. During this period the maximum sodium and silica levels experienced were 500 µg/kg and 5000 µg/kg, respectively.

Some sulphate was expected to be found in the boilers as a result of hydrolysis of Gulf Coat VT. Gulf Coat VT is a light oil rust preventative that was applied to the turbine rotors by the supplier, in order to minimize corrosion during the period between installation and first operation. This substance contained approximately 200 mg/kg sulphur [7]. The highest sulphate level recorded in boiler water was 1,500 µg/kg. Subsequent sulphate levels were much lower.

Silica proved to be an unusual problem. The sodium and silica levels were not consistent. The water treatment plant had continued to produce demineralized water of exceptional quality. The silica was coming from a protective paint that had been applied to the turbine cylinders and connecting parts. The paint was peeling off, and some was getting through pump strainers. The paint was General Electric Material A50A307 – a beige, water-reducable, alkyd rust preventative primer, known as C.A. Reeve Paint Co., Inc., Beige Alkyd Primer No. 2733 (Turbine Grade). The GE specification [8] quotes acceptable limits for chlorine, cobalt, fluorine, lead, and sulphur – no mention is made of silica! This paint had a calcium/magnesium silicate filler. This was the source of the silica in the boilers. Allowing this paint to remain on the turbine components was possibly an error. The amount of additional blowdown required to control the bulk concentration of silica was 60 minutes per day per boiler for the first three months of operation [9]. Table 4 lists some of the specifications for boiler chemistry. Table 5 summarizes secondary side chemistry for the first three months of operation.

Table 4: Boiler Chemistry Specification – Unit 6, 1984

pH	9.3–9.6
Sodium µg/kg	< 100
Chloride µg/kg	< 100
Nitrite µg/kg	< 100
Nitrate µg/kg	< 100
Sulphate µg/kg	< 100
Phosphate µg/kg	< 100
Silica µg/kg	< 500

Table 5: Summary of First Three Months Operational Boiler Chemistry

	Bank A			*Bank B*		
	Av	*Max*	*Min*	*Av*	*Max*	*Min*
pH	9.6	10.4	8.9	9.6	10.5	9.0
Sodium μg/kg	120	2400	<2	120	2800	<2
Chloride μg/kg	34	250	<20	40	160	<20
Nitrate μg/kg	26	220	<20	35	1300	<20
Nitrite μg/kg	20	100	<20	20	190	<20
Sulphate μg/kg	66	530	<20	110	3300	<20
Phosphate μg/kg	21	180	<20	20	990	<20
Silica μg/kg	600	2500	<10	650	2700	<10

Later, the requirement for a turbine warranty (heat rate) test was identified. In order to determine the steam quality, caustic soda was introduced into the boilers on three occasions by way of the CEPS. The caustic added gave a concentration of 1,300 μg/kg, with all blowdown isolated. Following each test all boilers were blown down at maximum rate, and all were within specification within eight hours. This confirmed the high blowdown rate and makeup water quality.

Even after all the load rejections, the condensers contained no additional debris [10] compared with approximately 1,000 kg at Bruce NGS 'A' [11]. Condensate and boiler feed pump strainers had been cleaned as required without Unit 6 shutting down. These facts demonstrate that the improved Construction QA program and the wet storage regimes were effective.

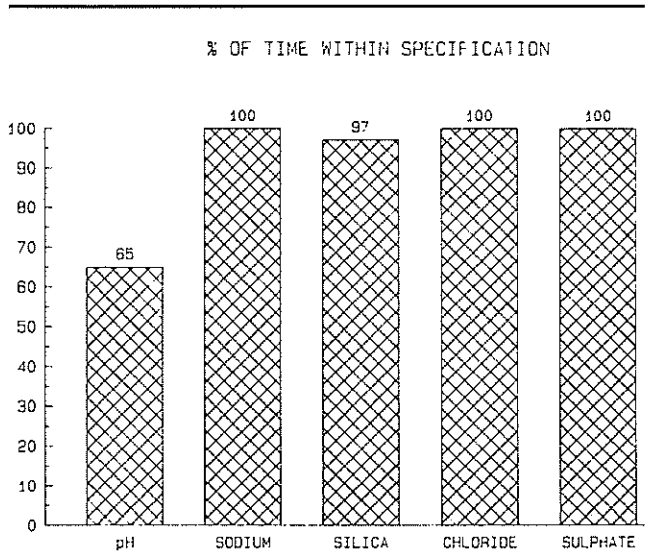

Figure 4 Summary of Unit 6 Boiler Chemistry Performance, October–December 1985

Table 6: New Boiler Chemistry Specifications 1986

Unit 6	
pH	9.3–9.8
Sodium μg/kg	<20
Chloride μg/kg	<20
Sulphate μg/kg	<20
Silica μg/kg	<200
Other anions μg/kg	<20

Commercial Operation

The unit was declared in service on 14 September 1984. Since March 1985, chemical control has been such that the original specifications can be maintained with very little effort. Figure 4 shows a summary of secondary side chemical control for the last three months of 1985. During this time it was very unusual for a boiler in Unit 6 to have a blowdown requested due to a chemical parameter being out of specification. As a result of the ease of maintaining chemical control, the specifications for boiler chemistry have been tightened, in accordance with the ALARA (AS LOW AS is Reasonably Achievable) principle. Table 6 lists some of the revised specifications.

As Nuclear Steam Generators, in general, behave as giant garbage cans for secondary side corrosion products, it was decided to attempt to minimize such accumulations at BNGS 'B'. To continue striving for excellence in chemical control, a series of blowdown requests have been initiated. One of these is an operator routine that requests that the diametrical (tube-free lane) blowdown header be opened up for one hour on each boiler in turn. The second is requested if, during the week, a major blowdown has not been requested for chemical control; in this case, the peripheral blowdown header is opened up for half an hour per boiler. The purpose of these exercises is to attempt to suspend matter that has accumulated in the boiler, and get rid of it!

During the recent Unit 6 warranty outage, one boiler, the condensers, de-aerator, MSRS, and the turbine were inspected. The MSRS showed no lack of passivation, as had been feared due to the partition coefficient of ammonia. The other components appeared to be in good condition. However, some corrosion was apparent under the low-pressure turbine hoods where the paint had peeled off. The boiler tubes were not shiny, but the carbon steel surfaces were well passivated.

Future Direction

Future work required to confirm the excellence of the program at BNGS 'B' includes mass balances and identification of sludge piles. The design of the startup sampling systems does not allow for mass balance studies, so equipment is being procured to enable these investigations. During the next major outage of Unit 6, one boiler will be inspected to determine the sludge pile depth.

Improvements can be made to the Chemical Control Program at Bruce NGS 'B'. The importance of controlling compatible materials became painfully apparent during the recent turbine warranty outage. The mechanical group had used a copper-containing anti-seizure compound over casing bolts and stationary stages – in an all-ferrous system!

Conclusion

The approach to chemical control during the construction and commissioning phases has significantly contributed to the high standard of boiler chemistry at Bruce NGS 'B.' In addition, Unit 6 operation has not been compromised as a result of secondary side chemical control.

Acknowledgements

The author wishes to acknowledge the contribution of the Chemical Unit Staff to the excellent chemical control at Bruce NGS 'B.'

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Gestion du cycle de combustible à la Centrale Nucléaire Gentilly-2

G. Hotte
Hydro-Québec, Service technique, Gentilly-2
6600 Côte-des-Neiges, suite 210
Montréal, Québec
H3S 2A9

G. Parent, L. Cloutier, A. Aubin, and N. Macici
Hydro-Québec, Service technique, Gentilly-2
C.P. 360
Gentilly, Québec
G0X 1G0

Résumé

Hydro-Québec, la principale compagnie responsable de la production et de la distribution d'énergie électrique au Québec (Canada), exploite un seul réacteur nucléaire sur un réseau essentiellement hydro-électrique d'une puissance d'environ 25,000 MWe. La centrale nucléaire Gentilly-2, un réacteur à tubes de force préssurisés et modéré à l'eau lourde de la série CANDU-600, ne produit que 685 MWe de la capacité totale du réseau. Cet article fait une revue générale des principales activités de gestion du combustible nucléaire et présente les particularités de leurs applications à Gentilly-2. Les sujets suivants sont traités:

- le contrôle et l'assurance de la qualité du processus de fabrication du combustible;
- l'approvisionnement en combustible;
- la gestion du combustible en pile, incluant le suivi du cœur et la stratégie de rechargement;
- la détection et le remplacement du combustible défectueux;
- la géstion du combustible usé.

Abstract

Hydro-Québec, a major electrical public utility in Canada, operates a single nuclear reactor on a grid which is mainly composed of hydroelectric plants. The Gentilly-2 pressurized heavy water reactor of the CANDU-600 series produces 685 MWe of Hydro-Québec's 25,000 MWe total capacity. The organization of nuclear-related activities is greatly conditioned by the small fraction of nuclear power on the grid. This paper gives an overview of the Gentilly-2 experience in managing the CANDU nuclear fuel cycle in an efficient and secure manner. The following areas of fuel management are covered:

- quality assurance in fuel design and fabrication;
- fuel supply;
- in-core fuel management including core follow-up and fuel replacement strategies;
- detection and replacement of defective fuel;
- spent fuel storage.

Introduction

La centrale nucléaire Gentilly-2 est dotée d'un réacteur à tubes de force pressurisés, modéré à l'eau lourde de la série CANDU-600. Sa puissance (685 MWe) ne représente que 2.7% des 25,000 MWe installés sur le réseau de Hydro-Québec, composé essentiellement de centrales hydro-électriques situées à grande distance des centres de consommation. Bien que conçu comme centrale de base, sa production est présentement limitée pour suivre les fortes variations saisonnières de production et de consommation. Ainsi Gentilly-2 est actuellement maintenue à pleine puissance au cours d'une période hivernale d'environ 5 mois pour satisfaire les besoins du réseau. La puissance doit toutefois être réduite à 50% de sa capacité (50% PP) lorsque les crues printanières augmentent le potentiel de production du réseau hydro-électrique.

Dans ce contexte, Hydro-Québec traite dans ses services internes les principaux aspects techniques et commerciaux de la gestion du combustible nucléaire, soit:

- l'approvisionnement en concentré d'uranium;
- l'achat de services pour le traitement du concentré et la fabrication des grappes;
- la gestion du combustible en pile (le rechargement du réacteur);
- le développement de logiciels informatisés, incluant les modèles mathématiques et physiques utilisés;
- les activités spécifiques à l'exploitation du réacteur, incluant la production de cobalt, les études de sûreté ainsi

Mots-clés: réacteurs nucléaires, physique du réacteur, gestion du combustible, stratégie de rechargement, assurance qualité.

que l'analyse de la performance du réacteur et de ses systèmes de régulation.

Description du combustible

L'utilisation d'uranium naturel et le rechargement en marche déterminent deux caractéristiques importantes de la filière CANDU: un haut niveau d'économie de neutrons et un assemblage de combustible dépourvu d'éléments de contrôle ou de mesure. La grappe de combustible des CANDU-600 est constituée, tel qu'illustré à la figure 1, de 37 crayons contenant des pastilles de bioxyde d'uranium naturel enfermées dans une gaine mince de ZIRCALLOY-4 (un alliage de zirconium, étain, fer et chrome utilisé pour ses faibles propriétés d'absorption neutronique). La gaine est séparée de la pastille par une interface lubrifiante à base de graphite (CANLUB) introduite pour augmenter les tolérances mécaniques du crayon aux interactions entre les éléments. La fiche technique de la grappe utilisée à Gentilly-2 est donnée au tableau 1.

La pastille de bioxyde d'uranium et la gaine de combustible constituent les deux premières barrières de confinement des produits de fissions radioactifs. Chaque étape de fabrication d'une grappe de combustible, soit l'affinage du concentré d'uranium en poudre de UO_2, la fabrication de la pastille, des crayons de combustible et l'assemblage en une grappe, est soumise aux normes d'assurance et de contrôle de la qualité régissant les systèmes et articles reliés à la sûreté. Le détail des procédés utilisés est bien documenté [9]. Mentionnons simplement que pour maintenir l'intégrité du confinement assuré par la gaine, la grappe de combustible doit résister aux contraintes neutroniques, mécaniques et thermiques qu'elle subit à partir de son introduction dans le réacteur jusqu'à son transfert à l'aire de stockage permanent du combustible usé.

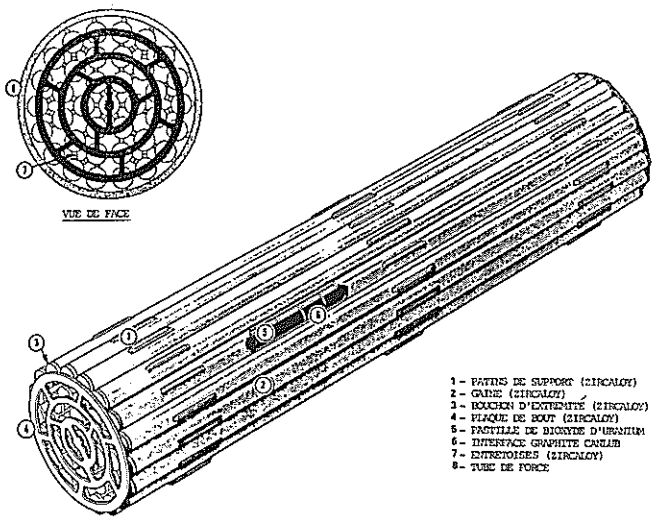

Figure 1 Grappe de Combustible à 37 Éléments

Tableau 1: Fiche Technique de la Grappe de Combustible Utilisée à la Centrale nucléaire Gentilly-2

Grappe	
–Longueur	495.30 + 0.75 mm
–Diamètre (maximum)	102.50 mm
–Grappes par canal	12
–Poids de la grappe	24.08 kg
–Poids d'uranium par grappe (minimum)	19.25 kg
Éléments	
–Nombre d'éléments par grappe	37
–Matériau de la grappe	Zircaloy 4
–Diamètre extérieur (maximum)	13.10 mm
–Diamètre intérieur (minimum)	11.45 mm
–Épaisseur de la gaine (minimum)	0.38 mm
Pastilles (UO_2 fritte)	
–Densité	10.50 ... 10.80 Mg / m^3
–Rapport O / U	1.995 ... 2.015
–Rapportlongueur / diamètre(maximum)	1.41
–Espace libre pastille-gaine (maximum)	0.15 mm

La performance du combustible à Gentilly-2 se compare à celle observée dans les autres réacteurs CANDU. Les taux de combustion (burnup) moyen à la décharge des grappes de combustible se maintient couramment au-delà de 170 MWh / kgU (7083 MWj / tonne U), résultant en un taux de rechargement d'environ 15 grappes par jour équivalent de fonctionnement à pleine puissance (JEPP). Certaines grappes de combustible atteignent régulièrement un taux de combustion de 260 MWh / kgU (10833 MWj / Tonne U). Si ces valeurs peuvent sembler faibles par rapport à celles obtenues dans les réacteurs préssurisés à eau légère des filières américaine ou française (PWR), il est important de noter que le coût de l'élement CANDU est de trois à quatre fois inférieur à celui du combustible PWR. Les études indiquent que la grappe de combustible actuelle pourrait atteindre, si le combustible était enrichi, un taux de combustion substantiellement plus élevé (600 MWh / kgU–25,000 MWj / tonne U) sans modification majeure à sa conception [2].

Le taux de défectuosité du combustible à Gentilly-2 résume bien la performance du combustible CANLUB. Des 14,618 grappes irradiées à la fin juin 1986, seulement 11 grappes se sont avérées défectueuses. Les défectuosités ont toutes été attribuées [5, 6] soit à des vices de fabrication ou à l'usure causée par la présence de débris dans le circuit caloporteur. Ces résultats ne sont pas étrangers au suivi des programmes d'assurance qualité implantés à chaque étape du processus de fabrication du combustible jusqu'à sa réception en centrale. À Hydro-Québec, le programme d'assurance qualité et les plans d'inspection et d'essai d'un fournisseur sont évalués avant l'adjudication du contrat ou avant le début de la fabrication et révisés lors de modifications aux procédures ou à une fréquence dépendant du rendement du fournisseur. L'application du programme d'assurance qualité soumis pour satisfaire nos clauses contractuelles est vérifiée en

Tableau 2: Performance du Combustible A G-2 (en date du 1er Juillet 1986)

Nombre de Grappes Irradiées	*Nombre de Défectuosités*			
	Fabrication	*Usure par Débris*	*Rampe de Puissance*	*Total*
14,618	8	3	0	11

usine chez le fournisseur et ses sous-traitants. Hydro-Québec participe aussi aux activités du groupe conjoint des producteurs d'électricité canadiens (Joint Audit Group) qui effectue des vérifications régulières auprès des affineurs de UO_2 et des fabricants de grappes. Finalement, sur réception du combustible en centrale, un échantillonnage représentatif est mis à l'épreuve dans des condition typiques d'exploitation afin de prévenir tout problème systématique de fabrication avant que le combustible soit chargé dans le réacteur.

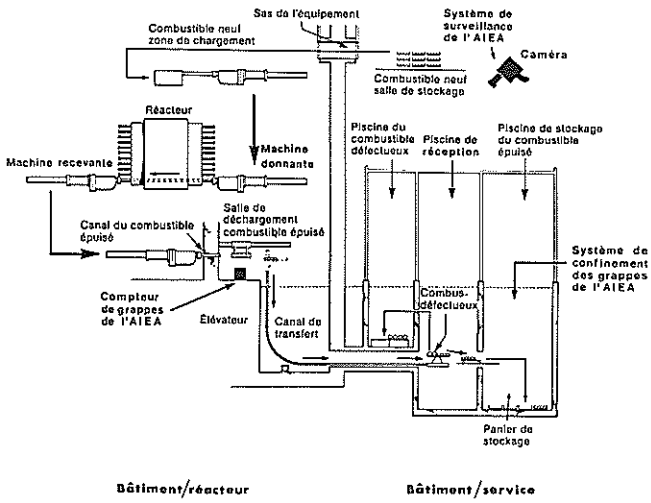

Figure 2 Ordre des Opérations de Manutention du Combustible dans un Réacteur CANDU-600

Gestion du combustible en pile

Le principal objectif de la gestion du combustible en pile d'un réacteur CANDU est de déterminer les rechargments qui minimisent le coût unitaire de l'énergie produite tout en respectant les limites dictées par la sûreté du réacteur et l'intégrité du combustible. Ce but général se traduit par les objectifs opérationnels suivants:

- maximiser le taux de combustion du combustible;
- maintenir les puissance de grappe et de canal ainsi que les pics de puissance à un niveau qui assure une marge de manœuvre suffisante pour garantir le fonctionnement de la centrale à pleine puissance;
- s'assurer que le combustible n'est soumis à aucune rampe de puissance qui puisse causer des ruptures de gaines.

On distingue trois phases à la gestion du combustible d'un réacteur CANDU: la détermination de la charge initiale, l'approche à l'équilibre et le maintien de l'équilibre du rechargement. La charge initiale doit être choisie pour satisfaire les objectifs et contraintes tout au long de la période d'environ 100 JEPP qui précédera le premier rechargement. L'accumulation simultanée de plutonium fissile dans le combustible produira, au cours de cette période, une augmentation substantielle des puissances dans la région à haut flux neutronique. L'aplatissement de la distribution de puissance à cet important pic plutonium est obtenu en introduisant du combustible appauvri en U-235 (160 des 4,560 grapes de la première charge), dans cette région du cœur. La période d'approche à l'équilibre débute avec le premier rechargement. L'introduction du combustible neuf produit alors d'importantes variations des puissances zonales et locales causées par le changement non-uniforme de la composition du combustible. Ces distorsions auront tendance à s'atténuer au fur et à mesure que les rechargements seront dispersés dans le cœur, réduisant l'écart entre le taux de combustion des différentes zones. À l'équilibre du rechargement, le taux de combustion moyen à la décharge du combustible se maintiendra à un niveau dicté par le mode de rechargement axial et la configuration nominale des mécanismes de régulation.

Un réacteur CANDU est muni de deux machines de rechargement montées sur pont mobile. Chaque machine peut contenir jusqu'à 12 grappes permettant de recharger un canal complet ou plus d'un canal lors d'une même visite. Un rechargement consiste à remplacer un certain nombre de grappes d'un canal. Les séquences de manutention du combustible sont illustrées à la figure 2. Les grappes neuves contenues dans les barillets de la machine donnante sont insérées à l'aide d'un ringard qui pousse la chaîne de combustible, déchargeant ainsi les grappes usées dans les barillets de la machine recevante. Lors d'un rechargement normal, le combustible est introduit dans le sens du débit. Exceptionnellement, les grappes peuvent être introduites à contre courant pour recycler du combustible qui n'aurait pas atteint un taux de combustion suffisant ou pour éviter que les grappes soient endommagées par l'impact qui résulterait de leur accélération dans le débit d'un canal partiellement ou complètement vide. Toutes les opérations de transfert sont commandées à distance par ordinateur de façon à minimiser les risques d'incidents au cours des manœuvres délicates d'ouverture d'un canal pressurisé et de transfert du combustible irradié vers son aire de stockage. Le nombre de grappes introduites définit le mode de rechargement du réacteur. Le rechargement de canaux adjacents se fait en sens opposé de façon à produire une distribution de puissance symétrique dans la direction axiale. A Gentilly-2, huit grappes

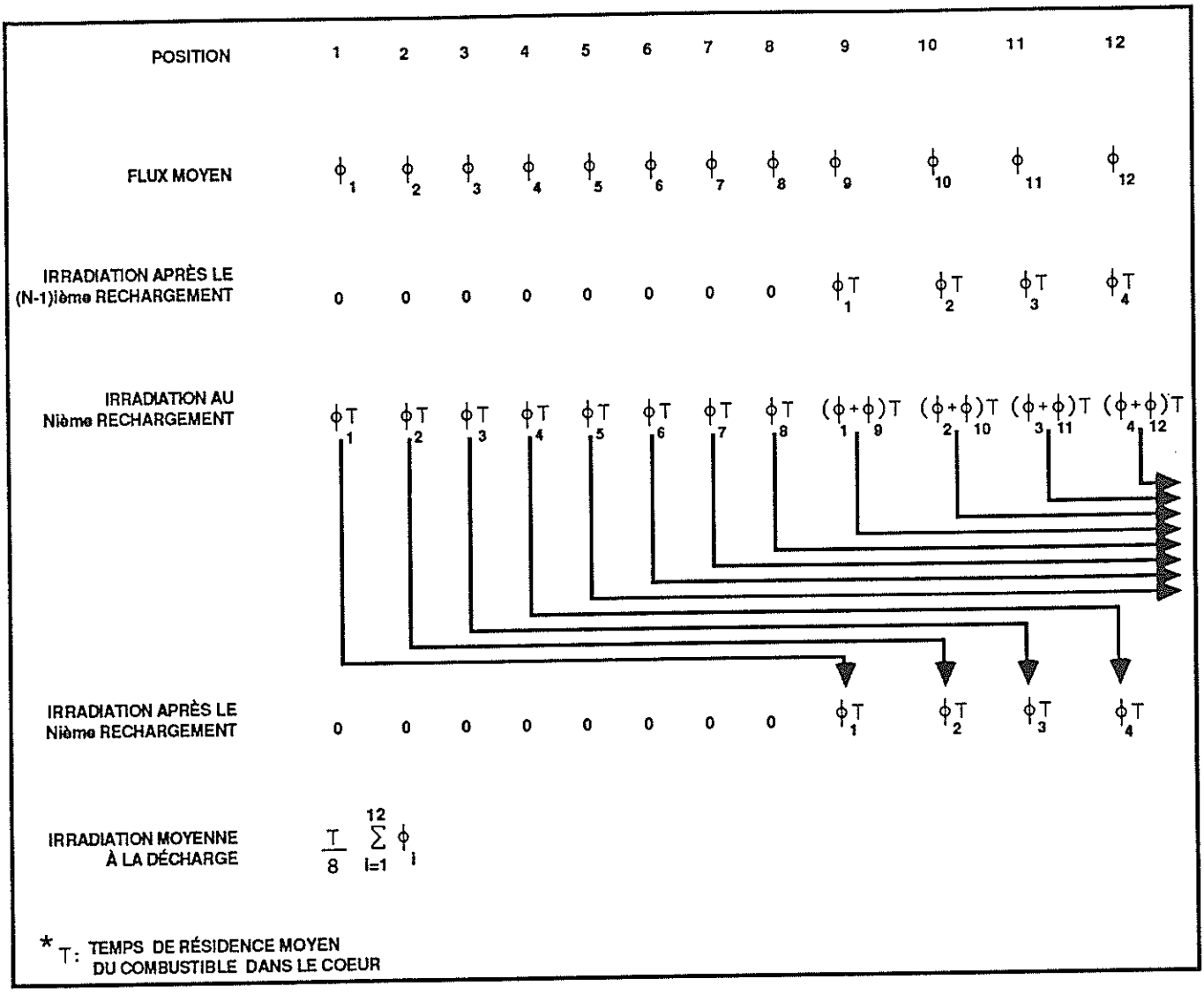

Figure 3 Rechargement à Huit Grappes (Moyenne dans le Temps)

neuves d'uranium naturel sont introduites à chaque rechargement tel qu'illustré à la figure 3.

Des modes de rechargement mixtes ou avec recyclage du combustible pourraient améliorer le taux de combustion. À titre d'exemple de recyclage, lors de la première visite aux canaux de Gentilly-2, les grappes 3 et 4, qui se seraient normalement retrouvées aux positions 11 et 12, ont été retirées du cœur et les grappes 11 et 12 remises à leur position avant rechargement. Cette variante du mode de rechargement à huit grappes nous a permis d'augmenter l'irradiation moyenne à la décharge du combustible retiré à la première visite et de profiter du taux de combustion réduit au moment du rechargement des grappes 11 et 12.

Suivi du cœur

Le rechargement en marche d'un réacteur CANDU constitue un processus continuel qui exige un suivi de l'irradiation et de la puissance de chaque grappe dans le cœur. En l'absence de mesures directes de la puissance produite par chaque grappe, l'évolution du cœur doit être simulée à l'aide d'un modèle mathématique approprié. La précision avec laquelle les puissances de grappe sont ainsi calculées a une incidence directe sur le niveau de puissance globale du réacteur et les marges de manœuvre permises: une précision insuffisante forcerait éventuellement une réduction inutile de la puissance globale. Le modèle utilisé doit aussi être suffisamment simple pour générer ces données sur une base régulière à partir de capacités de calcul accessibles de la centrale.

Une méthode particulière de calcul a été adoptée à Gentilly-2. Notre programme de simulation, SIMEX [8], utilise les signaux de détecteurs de flux neutronique situés dans le cœur pour améliorer la précision des

calculs des modèles théoriques normalement utilisés pour le suivi du cœur des centrales CANDU.

Par définition, le taux de variation de l'irradiation ($\omega(t)$) d'une grappe de combustible sur un intervalle de temps Δt est donné par

$$\omega(t_o + \Delta t) - \omega(t_o) = \int_{t_0}^{t_0+\Delta t} F_p\,[\omega(t)]\,\phi_p(t)\,dt \qquad [1]$$

$F_p[\omega(t)]$ représente le rapport du flux de neutrons dans le combustible au flux de neutrons dans la cellule unitaire de calcul ($\phi_p(t)$). Ce rapport est fonction de l'irradiation du combustible alors que flux de neutrons $\phi_p(t)$ dépend de l'irradiation et de l'interaction du combustible avec les matériaux de structure et de contrôle situé dans le cœur du réacteur. Dans l'approche traditionnelle, ces quantités sont considérées constantes sur l'intervalle de simulation. L'irradiation est alors donnée par

$$\omega(t_o + \Delta t) = \omega(t_o) + F_p\,[\omega(t_o)]\,\phi_p\,[\omega(t_o); R(t_o)] \qquad [2]$$

dans laquelle le flux de neutrons dans la cellule unitaire est obtenue d'un calcul de diffusion statique à deux groupes d'énergie pour l'irradiation $\omega(t_o)$ du combustible et la position $R(t_o)$ des mécanismes de régulation au début de l'intervalle de simulation.

L'approche utilisée par SIMEX consiste à exprimer le flux neutronique comme la somme pondérée d'un certain nombre de fonctions représentatives (ϕ_i), appelées modes:

$$\phi_p(t) = \sum_{i=1}^{N} A_i(t)\phi_i \qquad [3]$$

L'amplitude $A_i(t)$ de chaque mode est calculée de façon à minimiser l'écart quadratique moyen entre le flux calculé et celui indiqué par les signaux de 102 détecteurs Vanadium disposés à cette fin dans le cœur [11]. Le mode fondamental ($i = 1$) utilisé par SIMEX est le résultat du calcul tridimensionnel de diffusion de l'approche traditionnelle. Les modes supérieurs ($i > 1$) incluent des harmoniques naturelles ainsi que des solutions perturbées de l'équation de diffusion.

La figure 4 illustre le processus de calcul de SIMEX et ses interactions avec les ordinateurs de commande de la centrale.

L'originalité de notre méthode de calcul réside dans le fait que l'expansion modale apporte une correction aux flux et puissances calculés par le modèle de diffusion. À cet effet, l'approche SIMEX présente les avantages suivants [4]:

- l'intégration dans le temps des signaux des détecteurs Vanadium corrige l'approximation du flux neutronique constant dans le calcul d'irradiation:

$$\omega(t_o + \Delta t) = \omega(t_o) + F_p(\omega) \sum_{i=1}^{N} \phi_i \int_{t_0}^{t_0+\Delta t} A_i(t)\,dt; \qquad [4]$$

- les effets de combustion sur le facteur $F_p(\omega)$ sont simulés en calculant une moyenne sur l'intervalle Δt:

$$F_p(\omega) = \{F_p\,[\omega(t_o)] + F_p\,[\omega(t_o) + \phi\,[\omega(t_o); R(t_o)]\Delta t]\}/2; \qquad [5]$$

- les effets dynamiques vus par les détecteurs sont reflétés dans les valeurs de flux et de puissances calculées par expansion modale;
- les erreurs systématiques de modélisation sont corrigées par les mesures du flux de neutrons.

Prédiction de l'évolution du cœur et sélection des rechargements

En plus de fournir la réactivité requise, les rechargements contribuent à la régulation de la distribution spatiale de puissance. L'ingénieur responsable de la gestion du combustible doit donc connaître l'état du réacteur au moment où il désire déterminer les rechargements et être en mesure de prévoir le résultat de sa stratégie de sélection pour différentes configurations des mécanismes de régulation. Hydro-Québec a développé à cette fin un logiciel informatisé, SRG2 [3], qui simule les rechargements désirés en utilisant l'approximation traditionnelle du flux neutronique constant sur l'intervalle de combustion. Ce logiciel contient un algorithme de sélection automatique des rechargements, et peut effectuer ses calculs de flux neutronique en deux ou trois dimensions. Les calculs bidimensionnels (2D) ont été introduits pour les raisons suivantes:

- le problème de sélection des rechargements est défini dans un plan bidimensionnel;
- les calculs 2D nous permettent de simuler en détail les perturbations de chaque rechargement à un faible coût;
- l'approximation d'une forme axiale constante, utilisée dans l'approche 2D, nous permet de réduire les coûts de calcul tout en maintenant une précision satisfaisante pour plusieurs analyses.

La stratégie de sélection des rechargements est fondée sur une approche de façonnement de puissance: le cœur y est subdivisé en régions et zones de combustion qui sont rechargées de façon à maintenir la distribution de puissance simulée aussi près que possible d'une distribution cible. Chaque site disponible au rechargement est examiné de façon à déterminer les canaux qui ont un taux de combustion maximum tout en respectant les contraintes suivantes:

- le nombre de rechargements dans chaque direction doit garantir le maintien d'une distribution axiale de puissance symétrique;
- la puissance du canal et de ses voisins immédiats doivent être inférieures à des limites choisies pour minimiser les pics de puissance à court terme après rechargement;
- les taux de combustion des canaux voisins doivent être supérieurs à des limites choisies pour éviter le regroupement de rechargements qui pourrait éventuellement mener à des pics de puissance inacceptables.

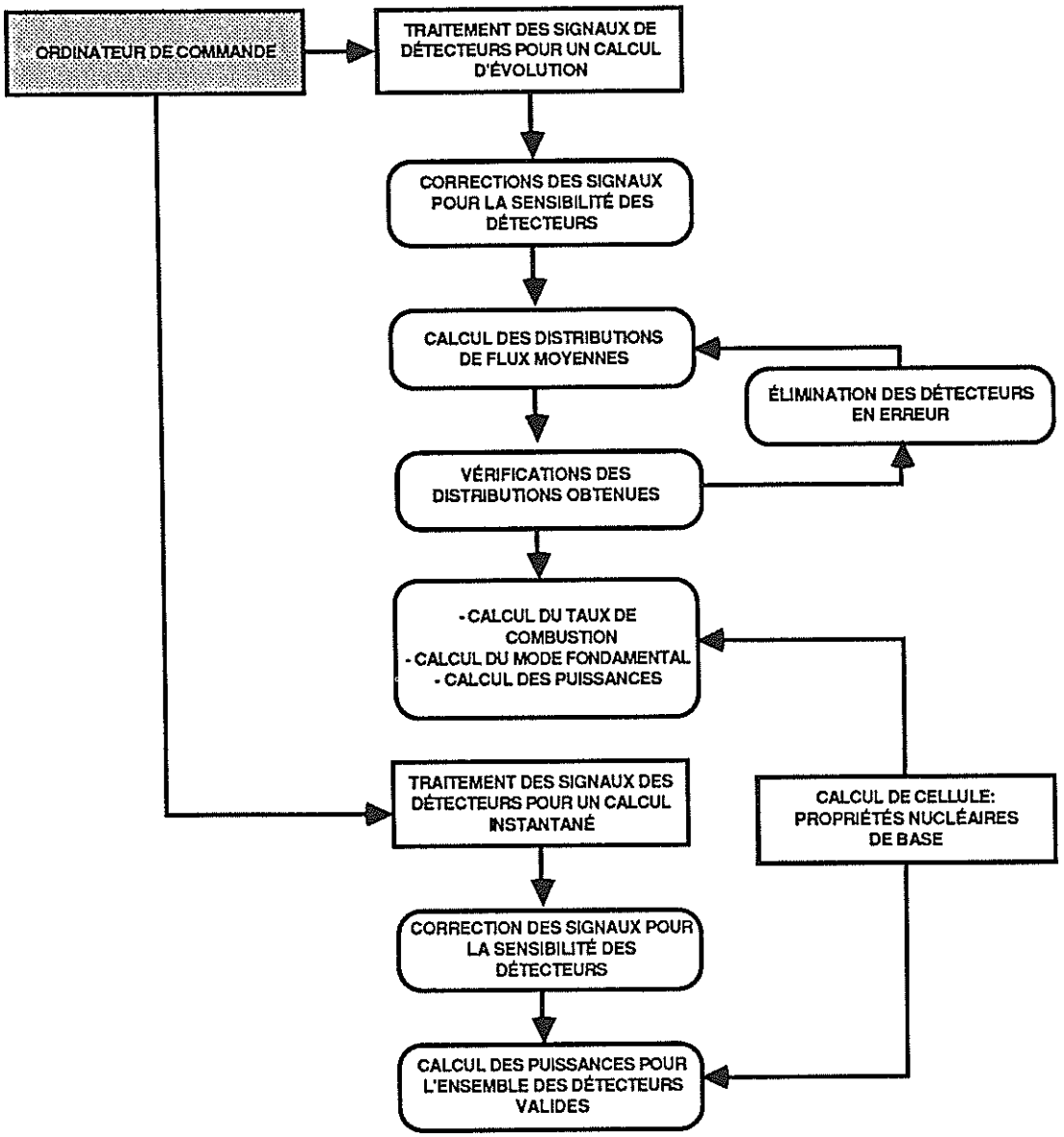

Figure 4 Le Processus de Calcul du Programme SIMEX

Chaque rechargement choisi est simulé individuellement et les puissances résultantes vérifiées pour s'assurer que les limites permises ne sont pas dépassées.

Bien que l'algorithme de sélection ne tienne pas compte présentement de tous les facteurs inhérents à la sélection des rechargements d'un réacteur CANDU, le programme SRG2 demeure un outil très utile pour l'ingénieur de gestion du combustible. Un certain nombre de modifications, dont le remplacement des calculs 2D par un calcul de perturbations à l'aide d'adjoints généralisés [10], sont envisagées pour en faire un programme de sélection et de simulation interactif muni d'une interface homme/machine adéquate.

Gestion de la réactivité excédentaire

Le rechargement en marche des CANDU permet de minimiser le réactivité excédentaire maintenue dans le réacteur. Tout surplus de réactivité doit être compensé par l'addition d'absorbeurs de neutrons, limitant le taux de combustion moyen à la décharge du combustible. La consommation actuelle de combustible à Gentilly-2 est environ 15% inférieure à celle qui avait été prévue dans les études de conception. Parmi les facteurs responsables de cette réduction, on note:

- le rehaussement de la pureté isotopique du modérateur de 99.79% à 99.965% massique;
- l'opération prolongée du réacteur à puissance réduite (50% PP);
- une gestion efficace de la réactivité excédentaire.

Tableau 3: Consommation et Taux de Combustion à la Decharge du Combustible

Mois	*Puissance % PP*	*Grappes Rechargées*	*Taux de Combustion Moyen à la Décharge (MWh/kg)*
Février 1985	100	470	162.2
Mars 1985	100	432	164.8
Mai 1985	50	104	184.9
Juin 1985	50	208	182.2
Février 1986	100	312	177.4
Mars 1986	100	480	179.5
Août 1986	50	152	183.4
Septembre 1986	50	152	191.8

La gestion de la réactivité excédentaire dépend en grande partie de l'organisation des activités de rechargement incluant les périodes d'entretien ou d'indisponibilité de la machine de rechargement. La politique d'exploitation établie à Gentilly-2 vise à garantir une réserve minimale de 2 JEPP de fonctionnement à 100% pleine puissance. Cette réserve, compensée par l'addition d'environ 0.1 ppm de bore soluble dans le modérateur, est jugée suffisante pour couvrir les indisponibilités prévisibles de la machine de rechargement. La fiabilité du calcul de la réactivité est assurée par une mesure périodique des concentrations totale et isotopique de bore dans le cœur. À puissance réduite, la plus faible charge de produits de fission ainsi que le coefficient négatif de puissance des CANDU augmentent la réactivité excédentaire disponible. Cette réserve supplémentaire nous permet de réduire la concentration de bore maintenue dans le cœur lors des périodes de fonctionnement prolongé à 50% PP. Tel qu'illustré au tableau 3, on a ainsi pu améliorer de 6% à 13% le taux de combustion moyen à la décharge mensuel par rapport à des périodes équivalentes d'opération à pleine puissance. Ce cœfficient négatif de puissance peut également servir à maintenir le réacteur en opération lors d'indisponibilité majeure de la machine de rechargement. À titre d'exemple, le réacteur pourrait être maintenu en marche pendant environ 41 jours sans rechargement à 50% PP.

Toutefois, un réacteur CANDU est généralement maintenu en marche sans rechargement en retirant les barres absorbantes de compensation (BC) normalement maintenues dans le cœur pour produire un aplatissement de flux à pleine puissance et assurer une réserve de réactivité suffisante pour les manœuvres de puissance. Les distorsions de flux produites par leur retrait commandent toutefois des réductions de la puissance globale pour assurer le respect des limites de puissance locales et d'inventaire de produit de fission dans le combustible et pour maintenir une marge de manœuvre adéquate. Selon les procédures actuelles à Gentilly-2, ce mode de fonctionnement (connu sous le nom de mode d'ajustement de la réactivité ou SHIM) fournit une réserve supplémentaire de 15 jours à des puissances variant de 94% PP pour le retrait du premier banc de BC à 82% PP pour le retrait des trois premiers bancs. Les distorsions de flux résultant d'un fonctionnement prolongé en mode SHIM doivent être ultérieurement corrigées par les rechargements dont la détermination est alors compliquée par la détérioration de la régulation spatiale. D'un point de vue opérationnel, nous préférons donc éviter d'avoir recours au quatre derniers bancs de BC qui peuvent toutefois donner plus de 40 jours additionnels de fonctionnement à des niveaux de puissance variant de 79% PP à 52% PP.

Détection et localisation des ruptures de gaines

La possibilité de retirer du cœur le combustible défectueux constitue un autre avantage du rechargement en marche des CANDU. L'exploitant peut ainsi maintenir à un minimum la quantité d'éléments radioactifs présents dans le circuit caloporteur réduisant d'autant les risques d'exposition du personnel aux rayonnements. Les ruptures de gaines sont détectées et localisées dans un CANDU-600 par deux systèmes indépendants opérés lorsque le réacteur est en marche [7]: le système de Détection de Ruptures de Gaines (DRG) et le système de Localisation des Ruptures de Gaines (LRG).

Le système DRG mesure, à l'aide d'un spectromètre à haute résolution, le niveau d'activité gamma de produits de fissions significatifs (Xenon-133, Krypton-88, Xenon-135 et Iode-131) présents dans chaque boucle du circuit caloporteur. L'interprétation des données obtenues permet d'identifier la boucle contenant le combustible défectueux. L'opérateur peut échantillonner chacune des boucles individuellement ou un mélange des deux boucles à partir de contrôles installés en salle de commande. La valeur et l'évolution du rapport des concentrations de Xenon-133 au Krypton-88 peut permettre d'évaluer l'importance des ruptures.

Le système LRG sert à localiser le canal contenant le combustible défectueux et à identifier la grappe défectueuse retirée du canal. Six détecteurs BF_3 permettent de mesurer les neutrons retardés émis par l'Iode-137 et le Brome-87 présents dans les lignes d'échantillonnage de chacun des canaux du cœur. L'analyse des résultats ainsi que le suivi du rapport des taux de comptage d'un canal au taux de comptage moyens des canaux de cette boucle (rapport de discrimination) permet de localiser le canal d'où provient les relâches. L'expérience montre qu'un canal, dont le rapport de discrimination est supérieur de trois déviations standards à la moyenne des valeurs pour les canaux de la boucle, est susceptible de contenir du combustible défectueux. L'identification de la grappe défectueuse est faite en suivant l'évolution du taux de comptage lors du rechargement du canal suspect.

Protection contre les pertes de régulation

Les réacteurs CANDU sont munis d'un système de détection de surpuissances locales (SDSL) conçu pour assurer une protection contre les pertes de régulation à haute puissance. Ce système est constitué de deux ensembles de détecteurs platines affectés indépendamment l'un de l'autre à chaque système d'arrêt d'urgence (SAU) du réacteur. Les seuils de déclenchement des SAU sont définis de façon à assurer qu'aucun canal de combustible n'atteigne, pour une multitude de formes de flux possible, une puissance qui pourrait causer un assèchement local et une surchauffe de la gaine de combustible dans les conditions thermohydrauliques normales du caloporteur. Le facteur de calibration utilisé pour corriger la dérive des signaux des détecteurs doit donc refléter:

- l'état réel des conditions d'écoulement du caloporteur;
- les distorsions de flux neutronique qui ne sont pas représentées dans le modèle utilisé dans les calculs de conception.

Le facteur de calibration est calculée à Gentilly-2, à l'aide d'un programme (PMCR) qui utilise une approche similaire à celle de SIMEX pour calculer les distributions de puissance à un instant donné dans le cœur. À la différence de SIMEX, l'expansion modale de PMCR génère des flux de canal qui ne reproduisent pas les variations locales causées par l'épuisement non-uniforme des grappes de combustible. On tient compte des effets de combustion en utilisant une corrélation déterminée à partir des résultats de simulations de périodes représentatives de l'historique de rechargement d'un CANDU-600. L'utilisation de cette approche simplifiée, mais suffisamment précise, est justifiée par le souci de s'assurer que le calcul du facteur de calibration puisse être fait sur des petits ordinateurs disponibles directement à la centrale. En l'absence de révision du facteur de calibration, les écarts entre les conditions d'exploitation et celles utilisées dans les études de conception du SDSL doivent être prises en considération dans la marge d'incertitude des seuils réduisant d'autant la marge au déclenchement disponible.

D'un point de vue opérationnel, il est important de maintenir une marge au déclenchement suffisante pour permettre l'exécution des manœuvres de rechargement du réacteur. En fait, la réduction des écarts entre les conditions réelles d'exploitation et les conditions de référence utilisées dans les calculs de conception du SDSL constitue une contrainte de sélection des rechargements qui peut limiter l'irradiation moyenne à la décharge du combustible ou encore nécessiter l'introduction de combustible appauvri. Une marge au déclenchement insuffisante peut provoquer dans certaines circonstances un déclenchement intempestif des systèmes d'arrêt d'urgence. À titre d'exemple, des procédures de rechargement particulières ont dû être adoptées pour éviter que le rechargement de canaux, situés près de détecteurs SDSL ou susceptibles de produire de fortes variations de la commande spatiale lors du déplacement du combustible, ne provoque le déclenchement d'une ou plusieurs chaînes des systèmes d'arrêt.

Protection contre les blocages de canal

Lors d'un rechargement, il est important de s'assurer que les manœuvres n'ont occasionné aucune restriction de débit. Les machines de rechargement sont instrumentées pour mesurer à cette fin la perte de pression dans le canal, avant et après le rechargement. L'instrumentation de mesure de température à la sortie du canal est aussi utilisée pour s'assurer qu'aucune obstruction n'est produite en cours de rechargement. Toute variation inopportune de cette empreinte thermique ou tout changement important de la perte de pression requiert une action immédiate de l'opérateur. Une vérification supplémentaire est effectuée régulièrement en mesurant les températures à la sortie de chaque canal à un niveau de puissance inférieur au seuil d'ébullition. Une vérification immédiate est alors faite en salle de commande et suivie d'une analyse détaillée par les ingénieurs de gestion du combustible.

Validation des modèles de calcul et vérification de la performance des systèmes de régulation

Les mesures de température prises pour la vérification de blocages de canal servent aussi à valider les calculs des programmes SIMEX et PMCR [4]. Les débits de chaque canal dans le cœur ont été mesurés par ultrasons lors des essais de mise en service à froid du circuit caloporteur. Les mesures de 12 canaux instrumentés pour le débit à Gentilly-2 sont utilisées pour extrapoler, par un calcul de débit dans des conduites parallèles, les données de mise en service aux niveaux de puissance désirés. La puissance générée par chaque canal (Pi) peut ainsi être calculée à l'aide de la relation générale,

$$P_i = G_i\,(h_i^{\text{sortie}} - h_i^{\text{entrée}}) \qquad [6]$$

dans laquelle G_i est le débit massique du canal et l'enthalpie (h_i) est déduite, à des températures inférieures à la température de saturation, des tables de propriétés thermodynamiques de l'eau lourde.

Un écart entre les données ainsi calculées et les simulations indique une possibilité d'erreur dans les données de simulation ou un dérèglement de l'instrumentation de mesures ou des systèmes de régulation. Toute indication à cet effet est suivie d'actions appropriées pour identifier la source du problème et apporter les correctifs nécessaires.

Gestion du combustible hors-pile

Le mécanisme de transfert du combustible usé est illustré à la figure 2 présentée plus haut. La machine

recevante se déplace de la voûte du réacteur pour se diriger vers l'aire d'entretien, située de chaque côté du réacteur, dans laquelle elle s'enclenchera sur un canal de combustible épuisé pour y décharger le contenu de ses barillets. Le mécanisme de transfert consiste en un godet élévateur mû par un moteur électrique installé à l'extérieur de la salle de transfert. Le godet peut recevoir deux grappes qu'il descend sur un panier convoyeur situé dans le fond de la piscine de déchargement. L'assemblage du godet est suffisamment lourd pour prévenir tout blocage du mécanisme et assurer que le combustible soit descendu par gravité dans la piscine si le câble, qui le retient, devait se rompre.

Le transfert du combustible à la piscine de déchargement constitue une étape critique au cours de laquelle les grappes font un trajet vertical d'environ 2 mètres à l'air libre. Toutes les manœuvres sont commandées à distance par ordinateur et une logique de contrôle est prévue pour s'assurer que le transfert se fasse dans les délais prescrits et qu'un refroidissement d'urgence adéquat soit mis en marche dans le cas contraire. En plus des mécanismes de transfert, la piscine de déchargement contient l'outillage requis pour isoler le combustible défectueux.

Le combustible usé est dirigé sur convoyeur à la piscine de réception qui contient un mécanisme semi-automatique destiné au transfert du combustible dans un panier de stockage. Ces paniers sont disposés dans la piscine de stockage à l'aide d'un treuil électrique installé sur un pont roulant situé au-dessus des piscines. Le combustible défectueux est entreposé dans une piscine particulière qui peut être isolée des autres en cas de contamination. Les piscines sont munies de double parois séparés d'un espace d'air surveillé pour détecter et récupérer toutes fuites. Des systèmes de purification de l'eau et de ventilation sont prévus pour permettre un accès sécuritaire aux aires de stockage.

À Gentilly-2, la piscine du combustible défectueux peut contenir jusqu'à 300 grappes et la piscine de stockage, jusqu'à 45,600 grappes de combustible. Au taux de consommation couramment observé, nous disposons donc d'une capacité d'environ 10 années de fonctionnement à pleine puissance avec un facteur d'utilisation de la centrale de 80%. Différentes options sont présentement à l'étude pour combler tout besoin additionnel d'entreposage à moyen terme du combustible usé. Parmi les solutions envisagées, nous considérons la construction d'une nouvelle piscine de stockage et l'entreposage à sec dans des enceintes de béton et d'acier.

Le processus de transfert du combustible usé à l'aire de stockage à Gentilly-2 est en grande partie déterminé par le souci de satisfaire les exigences du traité de non-prolifération des armes nucléaires. L'Agence Internationale de l'Énergie Atomique (AIEA), responsable de la vérification du respect de ce traité chez les pays signataires, requiert à cet effet que la quantité exacte de plutonium en piscine soit connue en tout temps et rapportée mensuellement à la Commission de Contrôle de l'Énergie Atomique (CCEA). En plus de nécessiter l'installation d'un système élaboré de surveillance, cette obligation exige un suivi de l'inventaire et de la position de chaque grappe entreposée dans les piscines de stockage. Un logiciel particulier développé à Hydro-Québec pour le suivi du combustible hors pile, INPHYCOM – (INventaire PHYsique du COMbustible), calcule la composition isotopique de chaque grappe en piscine à partir de données fournies par le code SIMEX [1]. Les données de base exigées des fournisseurs de combustible y sont enregistrées et mises à jour tout au long de son cheminement en centrale, soit de l'entrepôt au cœur et jusqu'à la piscine de stockage. Ces données sont aussi utilisées à plusieurs étapes de la gestion du combustible, tels l'approvisionnement en combustible, le suivi de la performance, la gestion et comptabilisation des stocks, la prévision des besoins, la rédaction des rapports à la CCEA et l'AIEA.

Hydro-Québec entretient une position de collaboration aussi étendue que possible avec l'AIEA et la CCEA en matière de non-prolifération. Nous avons, à cet effet, participé à l'installation et au développement de plusieurs prototypes conçus par l'Énergie Atomique du Canada Ltée pour le programme canadien de support à la non-prolifération. Ces équipements (enregistreurs et compteurs de grappes à l'entrée de la piscine, systèmes autonomes de surveillance ...) ont grandement marqué non seulement la gestion du combustible mais aussi les procédures d'exploitation de la centrale. En 1986, Hydro-Québec participe à une expérience d'une durée d'un an qui a pour but de vérifier et développer un système de confinement du combustible en piscine.

Conclusion

La gestion du cycle de combustible a une incidence déterminante sur le coût marginal de l'énergie produite par une centrale nucléaire. Nous avons décrit les différentes activités du programme de gestion du combustible mis en place à la centrale nucléaire Gentilly-2. En ce qui a trait à l'approvisionnement et la fabrication du combustible, une attention particulière est portée au suivi des programmes d'assurance qualité implantés chez nos fournisseurs et leurs sous-traitants. La situation du Canada comme principal producteur d'uranium du monde occidental nous rend relativement sûrs de nos approvisionnements en concentré d'uranium. Hydro-Québec essaiera d'obtenir dans ses prochains contrats une clause de flexibilité dans les livraisons annuelles de combustible et négociera un prix de base relié au marché 'Spot.' Nous croyons toutefois devoir maintenir un stock minimal de combustible couvrant six mois d'exploitation à pleine

puissance pour garantir le fonctionnement de la centrale dans l'éventualité d'une indisponibilité majeure de la seule usine d'affinage de concentré opérée par la compagnie Eldorado Resources Limited, qui détient un monopole au Canada.

L'objectif premier de nos activités de gestion du combustible en pile est de maximiser le taux de combustion à la décharge du combustible. Malgré les excellents résultats obtenus à ce jour, Hydro-Québec continue à déployer les ressources requises pour maintenir et améliorer le rendement de notre combustible. À court terme, la minimisation de la réactivité excédentaire nous permettra de réduire notre consommation de combustible. À plus long terme, l'utilisation de modes mixtes de rechargement pourrait réduire la consommation bien que le principal attrait de ces modes de rechargement soit de réduire les pics de puissance. Nous prévoyons aussi profiter des capacités de calcul que procurent les nouvelles générations de mini- et micro-ordinateurs disponibles sur le marché pour faciliter l'utilisation de nos logiciels de simulation, automatiser les interfaces de transfert de données entre les programmes et augmenter nos capacités de traitement des données d'exploitation. Nous espérons ainsi réduire le temps actuellement requis pour traiter les données produites pour fins d'analyse, de validation de calculs, de formation et générer plus efficacement l'information requise pour l'exploitation du réacteur.

Finalement, la piscine de stockage du combustible usé doit suffire à nos besoins d'entreposage à moyen terme pour les dix prochaines années. Deux alternatives sont evisagées pour combler nos besoins au-delà de cette période, soit l'addition d'une nouvelle piscine de stockage ou l'utilisation d'enceintes d'entreposage à sec. En ce qui concerne la disposition ultime de notre combustible usé, nous suivons de près le programme canadien d'étude sur l'enfouissement permanent dans des couches géologiques stables.

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Summaries of CNS Conferences, Seminars and Workshops

International Conference on CANDU Fuel

Delegates from the international nuclear fuel community met at Chalk River 6–8 October 1986 to discuss CANDU fuel technology. Sponsored by the Canadian Nuclear Society and the American Nuclear Society, and utilizing the facilities of Atomic Energy of Canada Ltd. Research Company, the first International Conference on CANDU fuel attracted about 100 participants from Argentina, India, Japan, Korea, Pakistan, the U.S., and West Germany, as well as Canada. Canadian representatives were affiliated with the provincial utilities supporting nuclear programs, with fuel fabricators, with the AECB and AECL. The opening keynote address was given by J.B. Slater, Director of Planning, Reactor Development, at Chalk River, who stressed the importance of co-operation in all phases of the fuel cycle in the success of CANDU. Dr R.E. Green, Vice-President, Reactor Development, at Chalk River, made the conference banquet presentation. Green described the newly restructured AECL Research Company, its mission, and highlights in the research and commercial areas.

The 45 technical papers contributed fell into the categories of international programs and experience, production and fabrication, performance and modelling, fuel handling and storage, fission product behaviour and fuel cycles. Technical highlights of these sessions were:

International Programs and Experience

Argentina (R. Cirimello, CNEA) has a well-equipped CANDU fuel fabrication plant, though the indigenous fuel has suffered defect problems. Redesign is thought to have eliminated the cause, and the plant is gearing up for full-core capability. Canadian fuel is performing well in Embalse, with a bundle defect rate around 0.1%. The Indian program (G.V.S.R.K. Somayajulu) is moving ahead ambitiously. About 42,000 fuel bundles have been fabricated for the current program, which is planned to expand to 10,000 MWe by the year 2000. The Indian fuel design has evolved to an all-welded 37-element bundle for their projected 500 MWe reactor series.

In the CANDU 600s, Korea (M.S. Yang) and New Brunswick (M. Shad) detailed the excellent performance of their respective units. Yang highlighted the QA/QC aspects of Korean fuel technology. P.T. Truant, A.J. Hains, and H.J. Underhill spoke on Ontario Hydro experiences: more than 145 reactor-years, 420,000 bundles irradiated with station incapability charged to fuel, less than 0.1% over the stations' lifetimes. Most impressive was the significant performance database that Ontario Hydro has generated, particularly for fuel power ramp performance up to 200 Mw.h/kgU. Truant also discussed the CANDU Owner's Group (COG), a cooperative venture to further improve CANDU fuel technology. M.J.F. Notley (AECL, CANDU Operations) looked at the CANDU of the future, showing the potential for further uprating and cost reduction, and emphasizing the potential for fuel cycles other than natural uranium. This latter topic was also taken up later in the session on Fuel Cycles.

Production and Fabrication

R.T. Tanaka and T.W. Kennedy traced the development of UO_2 powder production at Eldorado's Port Hope facilities. Pelleting experience with that powder was outlined by M.R. Hoare (Westinghouse Canada Inc.), in particular the effects of powder agglomerates and chemical impurities. T.J. Carter described Chalk River's glove-box fabrication facility and its capability to produce alpha-active fuel $(U,Pu)O_2$, $(Th,Pu)O_2$, and $(Th,^{235}U)O_2$. In normal fabrication techniques, C.W. Turner (AECL, Chalk River) reported preparation of thoria microspheres via external gelation of sols.

Performance and Modelling

Advances in modelling CANDU behaviour under normal operating conditions were presented by M. Tayal (AECL, CANDU Operations) and W. Hwang (KAERI, Korea). ELESIM and ELESTRES are the performance codes that form the basis for these studies. ELOCA is the code describing accident conditions; J.A. Walsworth (AECL, Chalk River) and P. Richinson (AECL, CANDU Operations) described the model and showed that its calculations compared well with measured data from a LOCA test on CANDU fuel performed at the Power Burst Facility, Idaho Falls. M. Tayal also used ELOCA to

estimate axial variation in sheath strain during a hypothetical large-break LOCA; maximum plastic loop strain was 2.5% for a maximum sheath temperature of 1050°C. H. Sills (AECL, Chalk River) showed how fuel and thermal-hydraulics codes are being integrated into the CANSIM code system, which allows easy substitution of improved versions of either code type. P.J. Fehrenbach (AECL, Chalk River) reviewed CANDU fuel behaviour during LOCA transients. In a series of in-reactor LOCA tests with sheath temperatures up to 1050°C, fuel behaved in a manner predicted by existing codes. Sheath strains were less than 5% and release of short-lived fission products via on-line monitoring was acceptably small. In an associated paper, L.R. Lupton (AECL, Chalk River) described a tomographic technique for non-destructive examination of fuel assemblies. This capability is particularly valuable in characterizing the distribution of previous molten U-Zr-O alloys in fuel assemblies subjected to accident-simulation tests.

Handling and Storage

Fuel handling at the generating stations and during the back-end of the fuel cycle is attracting current attention. J. Novak and G. Miller (Ontario Hydro) described fundamental studies on the oxidation behaviour of defected fuel, and experience during actual reactor situations, in particular of a bundle without sufficient cooling in the fuel transfer mechanism, two hours after discharge. K.M. Wasywich (AECL Whiteshell) gave the current status of the Canadian experimental dry storage program. Defected and undefected, irradiated CANDU bundles have been stored in concrete canisters at 150°C up to four years in a variety of environments without significant degradation of fuel or sheath. P. Sermer (Ontario Hydro) gave an analytical model for heat transfer of a CANDU bundle residing in air, while R. Beaudoin (Hydro Quebec) described the technical and cost aspects of storing 67 Mg of spent fuel at Gentilly I. The fuel is stored in concrete canisters, and a similar approach will be applied in the decommissioning of Douglas Point. Ontario Hydro's discharged fuel is currently stored in water pools at the reactor sites. C.R. Frost discussed the 85 water-pool-years experience; data suggest the irradiated fuel will maintain integrity underwater for at least 50 years.

Fission Product Behaviour

Two major topics were addressed in this session: fission product release from defected fuel and behaviour under accident conditions. B.J. Lewis (AECL, Chalk River) presented two papers. One showed how the magnitude of the iodine 'spike' on reactor shutdown can provide information on fuel failures in the core; the second presented a model for release behaviour of depositing fission products (cesium, iodine) and noble gases, from defected fuel. J. Judah (Ontario Hydro) and A.M. Manzer (AECL, CANDU Operations) gave practical examples of defected fuel behaviour. Judah discussed the application of dry sipping to the 1984 Bruce NGS-A Unit 3 fuel defect excursion involving 41 defected bundles. Manzer outlined transport mechanisms of uranium released from fuel defects to the coolant, with examples from CANDU 600 reactors. Iodine-134 coolant activity data is a good indicator of uranium release.

C.E.L. Hunt and D.S. Cox (AECL, Chalk River) gave results of laboratory experiments on fission product release from UO_2 under a variety of oxidizing conditions, the data being used for model verification. At 400–1100°C, in air, UO_2 oxidizes to U_3O_8, and xenon, iodine and cesium releases were measured. The lowest test temperature at which cesium was observed was 750°C. Papers by T. Ogawa (JAERI, Japan) and R.A. Verrall (AECL, Chalk River) outlined experiments on fission product behaviour in oxide fuel utilizing ion-implantation techniques. Ogawa showed the 'trapping' effect on noble gas diffusion is a complex function of dose. An interesting feature of Verrall's paper was the observation that iodine forms bubbles in oxide fuel, the first time this has been reported.

Fuel Cycles

Apart from one paper on the prospects of a once-through thorium cycle in CANDU (M. Milgram, AECL Chalk River), this session was devoted to studies on slightly-enriched uranium (SEU) fuel in CANDU reactors at Chalk River. This is generally seen as the next logical fuel cycle for CANDU. A.D. Lane summarized the incentives for SEU, leading to improved economics and performance in existing and future CANDUS and the establishment of the technology for future fuel cycles. G.M. MacGillivray followed up with SEU performance experience and the existing database, and M. Younis (AECL, CANDU Operations) outlined a novel 'checkerboard' fuel management scheme for SEU in CANDUS. P.G. Boczar described an advanced CANDU core with improved location of reactivity devices, applicable to natural or enriched fuelling.

The following papers were presented:

Programs And Experience

'Developments in CANDU Fuel Technology in Argentina.' R.O. Cirimello, R. Olezza (CNEA, Argentina)

'Development in Design and Engineering of KWU Fuel for Light and Heavy Water Reactors.' R. Holzer, H. Stehle (KWU, West Germany)

'PHWR Fuel Fabrication – Indian Experience.' G.V.S.R.K. Somayajulu, T.S . Krishnan, A. Singh, V.A. Chandramouli, K. Balaramamoorthy, N.P.S. Katiyar, R. Rajendran, A.K. Sridharan, A. Suryaprakash (India)

'Development of QA/QC Technology in Korea.' M.S. Yang, H.S. Kim, H.I. Kwon, Y.W. Lee, K.S. Suh (KAERI, Korea)

'Fuel For Advanced CANDU Reactors.' J.T. Dunn, J.J. Lipsett, M.J.F. Notley, N.J. Spinks (AECL – Chalk River)

'Ontario Hydro Fuel Experience.' P.T. Truant, A.J. Hains (Ontario Hydro)

'Pickering Operations Fuel Performance Review.' H. Underhill (Ontario Hydro)

'Fuel Management and Fuel Performance at Point Lepreau NGS.' M. Shad (New Brunswick Power)

Production and Fabrication

'Ceramic UO_2 Powder Production at Eldorado Resources Ltd. Port Hope Conversion Facility.' R.T. Tanaka, T.W. Kennedy, (ERL, Port Hope)

'UO_2 Pellet Manufacture for CANDU Fuel.' M.R. Hoare (WCI, Port Hope)

'Preparation of Thoria Microspheres by the External Gelation of Low Viscosity Sols.' C.W. Turner, B. Clatworthy, A. Celli (AECL, Chalk River)

'The Recycle Fuel Fabrication Laboratories at Chalk River.' T.J. Carter (AECL, Chalk River)

Performance and Modelling

'Modelling CANDU Fuel Behaviour.' H.E. Sills, F.C. Iglesias, J.A. Walsworth, B.J. Lewis (AECL, Chalk River)

'Improvement of CANDU Fuel Performance Analysis Code ELESIM MOD 10.' H.C. Suk, W. Hwang, J.H. Park, B.G. Kim, K.S. Sim, C.J. Jeong (KAERI, Korea)

'Experience with FRAPCON-2 for the Analysis of CANDU Fuel Performance Under Normal Operating Conditions.' M.J. Richards (Hydro Quebec)

'ELESTRES: Multiaxial Sheath Stresses Near Circumferential Ridges.' M. Tayal (AECL – CANDU Operations)

'ELOCA – A Model for CANDU Fuel Performance During Loss-of-Coolant Accident Conditions.' J.A. Walsworth, H.E. Sills, S. Sagat (AECL, Chalk River); J.D. Allan (Scientific Software Inc., Peterborough)

'CANDU Fuel Behaviour During LOCA: ELOCA-A Predictions.' E. Kohn, V.I. Nath, M. Tayal (AECL, CANDU Operations)

'Estimated Axial Variations in CANDU Sheath Strains During a Hypothetical Large-Break LOCA.' M. Tayal, E. Kohn (AECL, CANDU Operations)

'CANDU Fuel Behaviour Under LOCA Conditions.' P.J. Fehrenbach, J.A. Walsworth, I.J. Hastings, C.E.L. Hunt, J.J. Lipsett, R.D. Delaney (AECL, Chalk River); E. Kohn (AECL, CANDU Operations); J. Lau (Ontario Hydro)

'The Effect of CANLUB Graphite and Siloxane Coatings on UO_2/Zircaloy-4 Interactions.' C.S. Lim, D.J. Wren (AECL, Whiteshell)

'Tomographic Examination of a Section From the HTBS-004 Bundle.' L. Lupton, P.F. Fehrenbach (AECL, Chalk River)

'The Rating of Element Closure Welds and Its Influence on Fuel Reliability.' R. Sejnoha (AECL, CANDU Operations); G. Valli, G. Ceccotti, L. Magnoli (ENEA, Italy)

Fuel Handling and Storage

'Dry Fuel Handling: Station Experience and Ontario Hydro/CNS Programs.' J. Novak, G. Miller (Ontario Hydro)

'Design Considerations for Water Pool Storage of Irradiated CANDU Fuel.' C.R. Frost, S.J. Naqvi (Ontario Hydro)

'Current Status of the Canadian Experimental Dry Storage Program.' K.M. Wasywich (AECL – Whiteshell); C.R. Frost (Ontario Hydro)

'Commercial Applications of Dry Storage Technology.' D. See Hoye, P. Pattantyus, R. Beaudoin (AECL, CANDU Operations)

'An Analytical Model for Heat Transfer Within a CANDU Fuel Bundle Residing in Air.' P. Sermer (Ontario Hydro)

'The Effect of Fuel Power on The Leaching of Cs and I from Used Fuel.' S. Stroes-Gascoyne, L.H. Johnson, E.R. Vance, D.M. Sellinger, A.P. Snaglewski (AECL, Whiteshell)

'Characterization of Used CANDU Fuel By Non-Destructive Radiochemical Analysis.' J.D. Chen, R. Taylor, H.G. Delaney, D. Bell, D.G. Hartrick, N.L. Pshyshlak, K.M. Wasywich, A. Rochon, D.G. Boase (AECL, Whiteshell); K.I. Burns, J.J. Leppinen (AECL, Chalk River); C.R. Frost (Ontario Hydro); P.A. Beeley (Queens University, Kingston)

Fission Product Behaviour

'Iodine Release From Defective CANDU Fuel After Reactor Shutdown.' B.J. Lewis, D.B. Duncan (AECL, Chalk River); C.R. Phillips (University of Toronto)

'Determination of Release Rate From Measured Activity of Depositing Fission Products.' B.J. Lewis, D.B. Duncan (AECL, Chalk River)

'Defected Fuel Location by Dry Sipping.' J. Judah (Ontario Hydro)

'Transport Mechanisms of Uranium Released to the Coolant From Fuel Defects.' A.M. Manzer (AECL, CANDU Operations)

'Fission Product Release During UO_2 Oxidation.' C.E.L. Hunt, F. Iglesias, D.S. Cox, N.A. Keller, R.D. Barrand, R.F. O'Connor, J.R. Mitchell (AECL, Chalk River)

'UO_2 Oxidization Behaviour in Air and Steam With Relevance to Fission Product Release.' D.S. Cox, F.C. Iglesias, C.E.L. Hunt, R.F. O'Connor (AECL, Chalk River)

'Dose Effect on Release of Ion-Implanted Noble Gases From Oxide Fuel.' T. Ogawa (JAERI, Japan); R.A. Verrall, D.M. Schreiter, O.M. Westcott (AECL, Chalk River)

'Iodine Release and Bubble Formation in Oxide Fuel.' R.A. Verrall (AECL, Chalk River); T. Ogawa (JAERI, Japan); Hj. Matzke (Karlsruhe, West Germany)

'Gamma Scanning of CANDU Fuel.' H.G. Delaney, J.D. Chen (AECL, Whiteshell)

Fuel Cycles

'Once-Through Thorium Cycles in CANDU Reactors – A Review.' M.S. Milgram (AECL, Chalk River)

'Improved Location of Reactivity Devices in Future CANDUs Fuelled With Natural or Enriched Uranium Fuel.' P.G. Boczar, M.T. vanDyk (AECL, Chalk River)

'Predicted Performance of High Burnup CANDU Fuel Using the Checkerboard Fuel Management Scheme.' M.H. Younis, A.R. Dastur, P.S.W. Chan (AECL, CANDU Operations); I.J. Hastings (AECL, Chalk River)

'Incentives for Slightly-Enriched Uranium in CANDU.' A.D. Lane, F.N. McDonnell (AECL, Chalk River)

'Performance Experience With Slightly-Enriched Uranium Fuel.' G.M. MacGillivray, I.J. Hastings (AECL, Chalk River)

Water Chemistry and Materials Performance Conference

The Water Chemistry and Materials Performance Conference, sponsored by the Canadian Nuclear Society, was held in Toronto on 21 October 1986.

The papers presented at this conference addressed the interactions that take place between water and system construction materials during the operation of pressurized water cooled nuclear power plants, and the importance and impact of these effects on system design and developmental research. The papers covered aspects of both primary and secondary chemistry during commissioning, operation, and shutdown, and of maintenance, decontamination, and radiation control. Each of these topics has an impact on the day-to-day operation of equipment, and so affects the long term reliability, performance, and life span of the plant.

The conference was attended by 85 people drawn primarily from those Canadian utilities that have operating nuclear power plants, and Atomic Energy of Canada Limited. The keynote address was given by Dr D.A. Meneley, Professor of Nuclear Engineering, University of New Brunswick, and the luncheon address was given by Dr O.J.C. Runnalls, Centre for Nuclear Engineering, University of Toronto.

Significant points from the various papers included the importance of interpreting the results of water chemistry analyses; the effects of boiling primary coolant on both system materials and fuel sheaths; the need for station chemists to be intimately involved with system chemistry during all periods of operation, including construction pressure tests and long shutdowns; and the need for laboratory programs to reproduce as many aspects of the actual station conditions as is possible.

The following papers were presented:

'Water Chemistry – The Key Operating Variable.' D. Barber (AECL, CANDU Operations)

'Chemistry Conditions and their Effects On Boiling CANDU-600 Heat Transport Systems.' S.H. Groom, C. MacNeil (Point Lepreau NGS, New Brunswick Electric Power Commission)

'Chemistry in Nuclear Steam Generators.' P.V. Balakrishnan (AECL Chalk River Nuclear Laboratories)

'Concentration Processes Under Tubesheet Sludge Piles in Nuclear Steam Generators.' F. Gonzalez, P. Spekkens (Ontario Hydro Research Division)

'Chemistry Control at Bruce NGS "B" – From Construction to Commercial Operation.' J.G. Roberts (Bruce NGS, Ontario Hydro)

'Studies of Activity Transport and Fission Product Behaviour in Water-Cooled Nuclear Generating Stations and Consequences for Defective Fuel Removal.' D.R. McCracken (AECL Chalk River Nuclear Laboratories); M.R. Floyd (Central Nuclear Services, Ontario Hydro)

'Corrosion of Heat Exchanger Materials Under Heat Transfer Conditions.' R.L. Tapping, P.A. Lavoie, D.J. Disney (AECL Chalk River Nuclear Laboratories)

'Decontamination of the NPD Fuelling Machine.' B.M. Mitchell, R.A. Speranzini, J. Torok, (AECL Chalk River Nuclear Laboratories); M.S. Hubert (NPD, Ontario Hydro)

'The Effects of Steam Generator Tube Temperature on

the Stress Corrosion Cracking of Alloy 600.' F.P. Vaccaro, G.J. Theus, B.P. Miglin (The Babcock & Wilcox Company); S. Roy (Babcock & Wilcox Canada)

'Lay-up Chemistry during Pickering NGS Units 1 and 2 Fuel Channel Replacement.' G.C. Gillies, M.W. Hersey, D.C. McCool (Pickering NGS, Ontario Hydro)

'Corrosion Product Transport Studies at Point Lapreau.' G. Plume (Point Lepreau, NGS New Brunswick Electric Power Commission); W. Schneider (Babcock & Wilcox Canada); C. Stauffer (The Babcock & Wilcox Company)

Second International Conference on Simulation Methods in Nuclear Engineering

The Second International Conference on Simulation Methods in Nuclear Engineering was sponsored by the Nuclear Science and Engineering Division of the Canadian Nuclear Society.

The conference was held in Montreal, 14–16 October 1986, and attracted an attendance of 85. Fifty papers were presented covering thermalhydraulics, fluid dynamics, reactor physics, control and operation, fuel behaviour, small reactors, maxi and mini computers, and interdisciplinary studies.

The following papers were presented:

Thermalhydraulics 1: System Transients

'Some New Directions in System Transient Simulation.' V.H. Ransom

'The Development and Benchmarking of Integral Thermal-hydraulics Codes.' R.E. Henry

'Thermal Hydraulics of LOFT Large Break Experiments and its Simulations with DRUFAN-02.' A.B. Wahba, W. Pointner

'CATHENA Simulation of Thermosiphoning in a Pressurized-Water Test Facility.' J.P. Mallory, P.J. Ingham

'The GINNA Tube-Rupture Event: A Plant-Transient Benchmark for TRAC' J.F. Lime, R.P. Jenks

Interdisciplinary

'Statistical Approach to Sensitivity Analysis of the Computer Simulation Codes.' S.M. Aceil

'RETACT Models in Link's TMI-1 Advanced Simulator.' K.C. Lea, D.J. Boltz

'Development of a Three-Dimensional Simulator of a DIDO-Type Reactor.' E. Nonbol

'EXPRIME: An Export System to Analyse Primary Heat Transport System Accidents.' P. Tye, J.C. Amrouni

'Scaling Laws for Simulating the CANDU Heat Transport System.' P.J. Ingham, V.S. Krishnan, P. Sergejewich, K.H. Ardron

'PREMIERE: A Model to Calculate the Transfer of Deuterium Between the Moderator Fluid and the Cover Gas Volume in Candu Reactors.' K.K. Fung, M. Garceau, W.I. Midvidy

Reactor Physics Simulation

'A Nuclear Criticality Simulator.' R.E. Anderson, W.L. Lesko, R.D. Sachs, H.C. Bachman

'Kinetic Modeling for On-Line Advanced Plant Simulator.' J. Samuels, S. Kaya, S. Anghaie

'Development of the Modal Kinetics Model for the Gentilly-2 G.S.' J. Koclas, S. Alaoui, M.A. Petrilli

'TRIVAC: A Modular Diffusion Code for Fuel Management and Design Applications.' A. Hébert

'Semi-Analytical and Empirical Methods for Local Reactor Reactivity Estimation.' A.P. Muzumdar, R.A. Bonalumi, D.G. Andrews

'Producing Three-Dimensional Nodal Coupling Coefficients.' E.G. Derrick, G.V. Guardalben, A.L. Wight

Thermalhydraulics 2: Complex Geometries

'Fluid Flow Simulations in Complex Geometries.' P.F. Galpin, R.G. Huget, G.D. Raithby

'The Drift Flux Model in the ASSERT Sub-Channel Code.' M.B. Carver, R.A. Judd, J.C. Kiteley, A. Tahir

'Comparison of ASSERT-4 CHF Predictions with Measurements in 28-Element Simulated Fuel String.' J.K. Szymanski, G. Nurnberg, W.I. Midvidy

'The Canonical Form of Turbomachine Characteristics (Four-Quadrants) and Other Three Port Devices.' R.J. Schneck

Developments in Transport and Reactor Calculations

'Multilevel Transport Calculations.' R. Sanchez, J. Mondot

'Acceleration Strategies for the Interface Current Method.' G. Marleau, A. Hébert, R. Roy

'RAVE – A View Factors Code for Arbitrary Geometry with Shadowing.' M.S. Milgram, S.L. Thompson

'Fuel Management Simulation for an Uprated CANDU-600.' D.A. Jenkins

'The Optimization of Cobalt Adjusters with OPTEX-3.' D. Rozon, M. Beaudet

'Poison Injection Model for a CANDU Reactor.' S.G. Lie, G.V. Guardalben, E.G. Derrick, H.G. Austman, A.L. Wight

Fuel and Fuel Channel Behaviour

'Simulation Methodology for Pressure Tube Integrity Analysis and Comparison with Experiments.' G.H. Archinoff, J.C. Luxat, P.D. Lowe, K.E. Locke, A.P. Muzumdar

'Development of the MINI-START Code for Fuel Element/Pressure Tube Contact.' D.B. Reeves, G.H. Archinoff, K.E. Locke, A.P. Muzumdar

'Validation of the Computer Code NUBALL for Simulation of Pressure Tube Asymmetric Ballooning Behaviour.' P.S. Kundurpi

'AMPTRACT: An Algebraic Model for Computing Pressure Tube Circumferential and Steam Temperature Transients Under Stratified Channel Coolant Conditions.' P. Gulshani, C.B. So

'FAIG – A Code for Analysis of Irradiation-Induced Axial Dimensional Change in PWR Fuel Assemblies.' A. Iisuka

Applications of Advanced Computers

'The Use of Super-Computers in University Research.' R.E. George

'Monte Carlo Neutron Transport on the CYBER-205 Supercomputer.' T.A. Tassou, E.M.A. Hussein, V.C. Bhavsar, U.G. Gujar

'Solution of the Few Group Neutron Diffusion Equations on Parallel Architecture Computers.' S.K. Zee, P.J. Turinsky

'Reactor-Physics Computer Codes to Microcomputers.' B. Rouben, K.S. Brunner, D.A. Jenkins

'TOMCAT: A Transient and Operational Micro-Computer Based Analysis Tool.' K.S. Howe, P.E. Meyer

'Evaluation of Algorithms for the Solution of the Drift Flux Equations on Advanced Computers.' E.D. Sills, J.M. Doster

Thermalhydraulics 3: General

'TRAC A Thermalhydraulics Code – Past, Present and Future.' R.A. Nelson, Jr, R.P. Jenks, D.R. Liles, J.W. Spore, J.F. Dearing, R.G. Steinke, M.W. Cappiello, J.H. Mahaffy, Susan B. Woodruff, Manjit S. Sahota

'Best Estimate Analysis of Prairie Island Plant Transients Using Dynode-P.' J.K. Kapitz, D.A. Rautmann, R.C. Kern

'A Dynamic Model of A U-Tube Steam Generator for Real Time Simulation.' B.W. Rhee

'Multi-Rate Integration Methods and Table Lookup Techniques Used in High Speed Dynamic Simulation of Nuclear Power Plants.' H.C. Yeh, W.E. Kastenberg, W.J. Karplus

'The Rate Form of the Equation of State for Thermalhydraulic Systems: Numerical Considerations.' Wm.J. Garland, R. Sollychin

'An Algorithm to Estimate Initial Discharge Rates for Stagnation Feeder Breaks.' C. Blahnik, S. Nijhawan

Thermalhydraulics 4: Multi-Fluid

'Multifield Methods for Nuclear Thermalhydraulics Problems.' S. Banerjee

'Validation of the TREAT Code Against Actual Plant Event Data.' D.F. Holderbaum, R.P. Ofstun

'Calculation of a Steam Generator Tube Rupture Accident Using THERMIT-UTSG.' S.T. Free, I. Schor

'Prediction of Steam Generator Depressurization Transients with the CATHENA Code.' B.N. Hanna, G.R. McGee, T.E. MacDonald

'ATHENA Model for 4 × 350 MW(T) HTRG Plant Side-by-Side Steel Vessel Prismatic Core Concept.' R.G. Ambrosek

'Capabilities of System Codes in Simulating Small Break LOCAS in PWRS.' F. D'Auria, G.M. Galassi, F. Oriolo

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Prices of the proceedings are:

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