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Editorial

Alan Wyatt
Editor

With this issues of the *Journal* the first volume is complete. Although we have received many commendations about the content and quality of production of the *Journal*, for which latter we would like to express our appreciation to the University of Toronto Press, there are concerns on the financial side.

Total production costs for this first volume are expected to be close to $90,000 with revenue of about $30,000. A first year loss in the region of $60,000 is considerably higher than the original estimate due to production costs being higher and revenues lower than budgeted. Obviously a 600 member strong society cannot view such a loss with equanimity and certainly cannot afford to repeat it in the second year. The Journal Management Board has, therefore, instituted a major reassessment of all aspects of the operation of the *Journal*. The overall target is to reduce production costs by 25 per cent and increase revenues by a similar percentage so as to cut the second year deficit in at least half, compared to year one.

At the time of writing this editorial, in early September, no decisions had been made as to how these changes were to be achieved. However, many options were being examined. New quotations for printing the *Journal*, the major production cost item, were being solicited. Potential printers were being asked to recommend innovative ideas for cost cutting with as little sacrifice in quality as possible. This might involve going to some type of 'desk-top' publishing. The possible introduction of page charges was being considered. This is a common practice in scientific journals and can go a considerable way to offsetting the typesetting and other fixed costs of production. The imposition of such a charge is usually linked with the supply of a certain number of free offprints of the papers to the authors. In our first year not a single author bought a single offprint of a single paper. This item alone represented a $7,000 shortfall in estimated revenue. I sincerely hope that nobody breached our copyright by photocopying papers!

No advertising revenue was received. However, from experience, one can say that such revenue will only be received if it is aggressively pursued. The *Journal* will probably be soliciting informative and educational advertising. For the first year all members of the Canadian Nuclear Society received the *Journal*, and the *CNS Bulletin*, as part of their annual fees. This policy is being re-examined and members may be asked to subscribe to the *Journal* separately. There are a number of outside organizations, in the business of journal publication, that might be interested in assuming responsibility for the production and distribution of the *Journal*. As with so many other aspects of Canadian publishing the realities of a small market make profitable production difficult.

If anyone has suggestions or comments on this topic by all means write to me and I will see that the suggestion is passed on to the appropriate person.

Financial constraints are nothing new in the nuclear community and, by comparison, those facing the *Journal* are less significant. It was disturbing to hear that notice had been given of the possible loss of funding that could lead to the closure of the McMaster University research reactor within the next few years. Most politicians express support for the importance of developing a high-tech base in Canada, to ensure future prosperity. Such a base will only be developed if the actuality, in terms of funding and facilities, matches the words. Investing in and for the future requires leadership and a recognition that research and development take time and cost money. The time required for a new energy system to take even 10 per cent of the total market is of the order of 30 years and more. Unfortunately this is the life of about ten parliaments. Need one say more about the problems of getting steady political support.

Past support led to the recognition of the development of the CANDU system, as one of the ten exceptional engineering feats of the past century, in Canada's Engineering Centennial year. For so many bright possibilities in the years ahead one can but echo the wartime slogan of a British cabinet minister: 'Give us the tools and we will finish the job.'

The Outlook for Nuclear Power after Chernobyl

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Abstract

In light of the accident at the Chernobyl nuclear power station, 1986 represents a critical year for nuclear power programmes. The leaders of a large number of countries have declared their continued intention to rely on nuclear power, but some countries have decided to defer any decision on expansion of their nuclear power programmes (e.g. Finland and the Netherlands), or have reaffirmed earlier decisions to stop (e.g. Austria) or phase out (e.g. Sweden) nuclear power altogether. This paper reviews the nuclear power programmes in a number of countries, and gives an overview of the current status and future prospects of nuclear power for electricity generation in the world, with particular emphasis to developing countries. The paper also presents economic comparisons of nuclear and coal-fired stations, since the competitiveness of nuclear power is an important factor in nuclear power development. As reported by a recent NEA study, nuclear power has an economic advantage over coal for base-load electricity generation in many countries, except in situations where low-cost coal is readily available close to load centers. Conditions for nuclear power could be less favorable in countries with extensive infrastructure and technology transfer requirements if those additional investments are charged against the nuclear electricity generation costs. Finally, the paper presents projections of future nuclear electrical generating capacity. In the developing countries some 600–800 GW(e) of generating capacity will need to be added up to the year 2000. However, the IAEA estimates that only some 5% of these capacity additions would likely be with nuclear power plants. It is expected that by the year 2000 nuclear energy may supply about 20% of the world's electricity requirements (25% of requirements in industrialized countries; 10% of requirements in developing countries).

Résumé

Du fait de l'accident survenu à la centrale nucléaire de Tchernobyl, 1986 a été une année critique pour les programmes électronucléaires. Les dirigeants de nombreux pa fait part de leur intention de continuer à recourir à l'é d'origine nucléaire, mais certains Etats ont décidé de re toute décision relative à l'extension de leurs progra électronucléaires (cas de la Finlande et des Pays-Bas) confirmé celle qu'ils avaient prise antérieurement de r recourir au nucléaire (comme l'Autriche), ou d'y renonc gressivement (comme la Suède). Le présent mémoire en revue les programmes électronucléaires mis en c dans un certain nombre de pays et donne un aperçu situation actuelle et des perspectives d'avenir de l'é nucléaire dans le monde, et notamment dans les pa développement. Il établit en outre des comparaisons c économique entre les centrales nucléaires et les centra charbon, étant donné que la compétitivité de l'électronu est un facteur important de son développement. Ains est indiqué dans une récente étude de l'AEN, le nucléa plus avantageux du point de vue économique que le ch pour produire l'électricité servant à assurer la charge de dans de nombreux pays, sauf dans les cas où l'on peut ment obtenir du charbon bon marché à proximité des c de base. La situation pourrait être moins favorable au nu dans les pays ayant d'importants besoins en matière c structures et de transfert de technologie si les inves ments supplémentaires requis sont englobés dans les de production de l'électricité d'origine nucléaire. En mémoire contient des projections concernant la capac production électronucléaire. Les pays en développe devront se doter de quelque 600 à 800 GWe supplémer de puissance installée d'ici à l'an 2000. Cependant, estime que les centrales nucléaires ne représenteror bablement qu'environ 5% de ces capacités. On prévo d'ici à l'an 2000 l'énergie électronucléaire couvrira e 20% de la demande mondiale d'électricité (25% dai pays industrialisés et 10% dans les pays en développer

Current Status

The most significant event in nuclear power d 1986 was the Chernobyl accident. The overall effe this accident on the nuclear power programm member states have yet to be seen, but it ha caused the cancellation of any nuclear power gramme. The accident produced an immediate ups

Keywords: nuclear power, nuclear forecasts, energy demand, economics, costs, risks, Chernobyl.

Table 1: Connections to Grids during 1986

Country	*Number of units*	*Net capacity (MW(e))*
France	6	7,215
USA	6	7,034
Germany, Fed. Rep. of	2	2,534
Japan	2	2,182
Korea, Rep. of	2	1,800
Canada	2	1,353
Czechoslovakia	2	776
Hungary	1	410
Totals	23	23,304

in public and political concerns about nuclear power in many countries. However, a more accurate image of the accident is now becoming visible, and shows an accident comparable to some other grave industrial accidents, rather than an accident of unprecedented magnitude, as it was generally portrayed in media accounts in the period immediately after the accident.

With the significant exception of Chernobyl Unit 4, 1986 was another year of safe, reliable and economic operation for nuclear power plants.

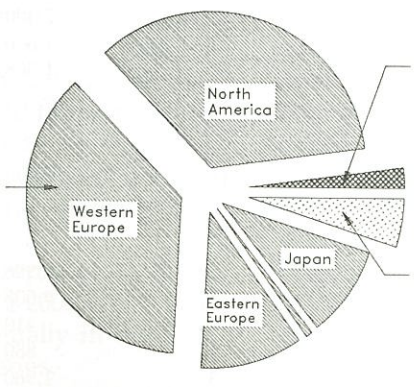
During 1986, total installed nuclear capacity increased by about 10%, with 23 new nuclear reactors, totalling more than 23,000 MW(e), connected to grids in eight countries (Table 1). The largest increase was in France, with 7,215 MW(e), followed by the United States, with 7,034 MW(e), the Federal Republic of Germany with 2,534 MW(e), Japan with 2,182 MW(e), the Republic of Korea with 1,800 MW(e), Canada with 1,353 MW(e), Czechoslovakia with 776 MW(e), and Hungary with 410 MW(e).

The only nuclear reactor which was shut down during 1986 was Unit 4 at Chernobyl. There were two cancellations in the US (MIDLAND-1, 491 MW(e) PWR, and MIDLAND-2, 816 MW(e) PWR) and one suspension in the Philippines (PNPP-1, 620 MW(e)PWR), of reactors under construction. By the end of 1986, there were a total of 31 shut-down nuclear reactors, with a total capacity of 4,328 MW(e). Most of these were reactors which began commercial operation in the 1960's, some even earlier, and which were shut down at the end of their economic life.

Table 2: Nuclear Power Reactors in Operation and Under Construction at the End of 1986

	Reactors in operation		*Reactors under construction*	
Country	*No. of units*	*Total MW(e)*	*No. of units*	*Total MW(e)*
Argentina	2	935	1	692
Belgium	8	5,486		
Brazil	1	626	1	1,245
Bulgaria	4	1,632	2	1,906
Canada	18	11,249	5	4,361
China			1	288
Cuba			2	816
Czechoslovakia	7	2,799	9	5,508
Finland	4	2,310		
France	49	44,693	14	17,809
German Dem. Rep.	5	1,694	6	3,432
Germany, Fed. Rep.	21	18,947	4	4,052
Hungary	3	1,235	1	410
India	6	1,154	4	880
Iran			2	2,400
Italy	3	1,273	3	1,999
Japan	35	25,821	10	8,431
Korea, Rep. of	7	5,380	2	1,800
Mexico			2	1,308
Netherlands	2	507		
Pakistan	1	125		
Poland			2	880
Romania			3	1,980
S. Africa	2	1,842		
Spain	8	5,599	2	1,920
Sweden	12	9,455		
Switzerland	5	2,932		
Taiwan (China)	6	4,918		
USSR	50	27,657	32	29,910
United Kingdom	38	10,222	4	2,520
USA	99	84,592	21	23,301
Yugoslavia	1	632		
Totals	397	273,715	133	117,848

At the end of 1986, worldwide, 26 countries were operating 397 nuclear power reactors (Table 2) with a total capacity of about 274 GW(e), accounting for about



Industrialized Countries	92.9%
USA	30.9%
France	16.3%
USSR	10.1%
Japan	9.4%
Germany, Fed. Rep. of	6.9%
Canada	4.1%
UK	3.7%
Sweden	3.5%
Spain	2.1%
Belgium	2.0%
Switzerland	1.1%
Finland	0.8%
South Africa*	0.7%
German Democratic Republic	0.6%
Italy	0.5%
Netherlands	0.2%
Developing Countries In CPE - Europe	2.1%
Czechoslovakia	1.0%
Bulgaria	0.6%
Hungary	0.5%
Developing Countries Outside CPE - Europe	5.0%
Korea, Rep. of	2.0%
Taiwan, China	1.8%
India	0.4%
Argentina	0.3%
Brazil	0.2%
Yugoslavia	0.2%
Pakistan	0.1%

Figure 1 Country distribution of installed nuclear generating capacity in the world, as of 31 December 1986. *Represented by the smallest slice. Source: IAEA Power Reactor Information System (IAEA-NENP-87-09).

Table 3: Nuclear Power Reactors by Reactor Type in Operation at End of 1986

Reactor type	*No. of units in operation*	*Net capacity (MW(e))*
PWR	207	164,890
BWR	83	64,782
LWGR	26	14,564
PHWR	26	13,792
GCR	33	6,692
AGR	10	5,736
FBR	7	2,380
HTGR	3	639
Other	2	240
Totals	397	273,715

AGR Advanced Gas-Cooled, Graphite-Moderated Reactor
BWR Boiling Light-Water-Cooled and Moderated Reactor
FBR Fast Breeder Reactor
GCR Gas-Cooled, Graphite-Moderated Reactor
HTGR High-Temperature Gas-Cooled, Graphite-Moderated Reactor
LWGR Light-Water-Cooled, Graphite-Moderated Reactor
PHWR Pressurized Heavy-Water-Moderated and Cooled Reactor
PWR Pressurized Light-Water-Moderated and Cooled Reactor

4,210 reactor-years of accumulated operating experience. There were also 133 nuclear power reactors under construction, totalling nearly 118 GW(e), mostly in the USSR (29,910 MW(e)), the USA (23,301 MW(e)), France (17,809 MW(e)), Japan (8,431 MW(e)), Czechoslovakia (5,508 MW(e)), Canada (4,361 MW(e)), and the Federal Republic of Germany (4,052 MW(e)).

Figure 1 shows the percentage distribution by country of the world's installed nuclear generating capacity as of 31 December 1986. As shown, 92.9% was in the industrialized countries, with 37.1% in W Europe, 35% in North America, 10.7% in E Europe (USSR and the German Democratic Rep and 9.4% in Japan. Developing countries outsi centrally planned economies of Europe (CPE-Eu accounted for 5%, and those in CPE-Europe for The distribution by type of reactor is given in Ta PWR's and BWR's account for more than 80% installed nuclear capacity to date.

The number of nuclear power reactors in ope and under construction in the developing count the end of 1986 is shown in Table 4. There w nuclear power reactors with a total capacity of MW(e) in operation in seven developing cou outside CPE-Europe, and 14 nuclear power re with a total capacity of 5,666 MW(e) in devel countries in CPE-Europe.

China, Cuba, Iran, Mexico, Poland, and Ro have their first units under construction. It is exp that LAGUNA VERDE-1 (654 MW(e) BWR) in M will be connected to the grid during 1987. connection for QINSHAN (288 MW(e) PWR) in Cl scheduled for 1989. In developing countries o CPE-Europe there were 15 units under constru with a total capacity of 9,429 MW(e); in devel countries in CPE-Europe there were 17 units construction, totalling 10,684 MW(e).

In energy terms, nuclear power plants gen about 1,515 TW.h(e) of electricity during 19 increase of 8% over 1985, and accounted for 15.5% of the world's electricity production in 1

Table 4: Nuclear Power in Developing Countries (as of 31 December 1986)

	Reactors in operation		*Reactors under construction*	
Country	*No. of units*	*Total net capacity MW(e)*	*No. of units*	*Total net capacity MW(e)*
Developing countries outside CPE-Europe				
Argentina	2	935	1	692
Brazil	1	626	1	1,245
China	–	–	1	288
Cuba	–	–	2	816
India	6	1,154	4	880
Iran, Isl. Rep. of	–	–	2	2,400
Korea, Rep. of	7	5,380	2	1,800
Mexico	–	–	2	1,308
Pakistan	1	125	–	–
Taiwan (China)	6	4,918	–	–
Yugoslavia	1	632	–	–
Sub-totals	24	13,770	15	9,429
Developing countries in CPE-Europe				
Bulgaria	4	1,632	2	1,906
Czechoslovakia	7	2,799	9	5,508
Hungary	3	1,235	1	410
Poland	–	–	2	880
Romania	–	–	3	1,980
Sub-totals	14	5,666	17	10,684

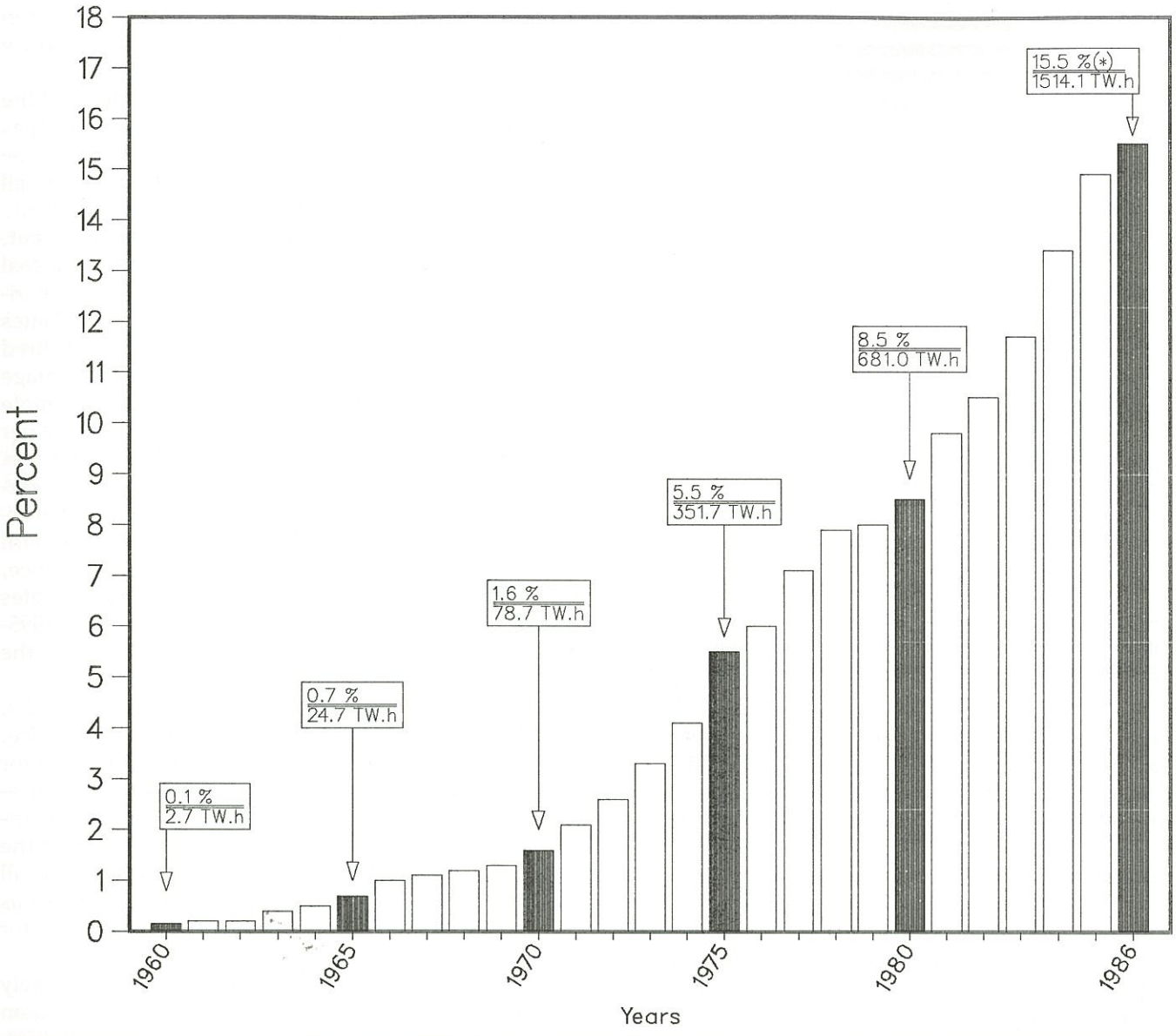


Figure 2 Nuclear electricity generation and share of total electrical energy for the period 1960 to 1986. *Total world electricity production in 1986 has been estimated. Source: IAEA Energy and Economic Data Bank (IAEA-NENP-87-18).

would take additional coal production equivalent to the current US production, or additional oil production equivalent to Saudi Arabia's production in 1982, to generate this amount of electricity by coal or oil, respectively. It is clear that if these additional amounts of fossil fuels were being required for electricity generation, the upward pressures on coal and oil prices could have a significant economic impact, particularly on developing countries.

The world's operating reactors represent a cumulative investment of well over US $ 200 billion, with an estimated $ 60 billion spent annually in building new plants and operating existing ones.

Figure 2 shows the growth of nuclear electricity generation and its contribution to total electricity production, since 1960. After more than 30 years of development, nuclear power is today providing a sizeable portion of the world's electricity. In the decade 1975–1985, nuclear-based electricity production quadrupled. Nuclear power plants in 1986 produced 10% or more of total electricity in 19 countries, and 25% or more in 12 of these countries (Figure 3). Three countries now produce more than 50% of their electricity from nuclear power plants (France – 70%; Belgium – 67%; Sweden – 50%). In the United States, where about 17% of total electrical energy was produced by nuclear power plants, the states of Vermont and North Carolina produced 65.4% and 63.2%, respectively, of their electricity from nuclear power plants; three other states, Connecticut, Maine, and

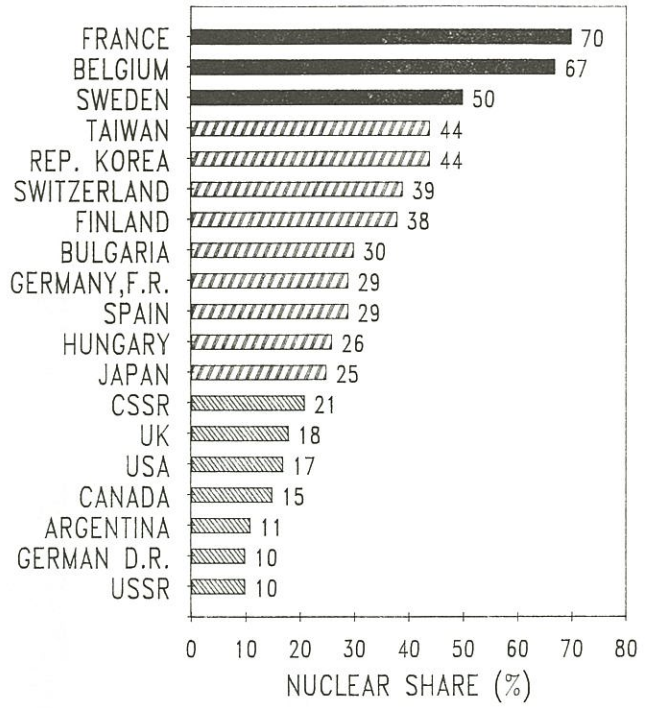


Figure 3 Countries with highest nuclear share of total electricity production in 1986.

New Jersey, produced over 50% of their electricity requirements from nuclear power plants. Similarly, in the province of Ontario in Canada, the nuclear share of electricity production was almost 50% in 1986.

Economic Viability

A study [1] published in 1986 by the OECD Nuclear Energy Agency shows that, in most countries, electricity generation with nuclear power plants is cheaper than with coal-fired plants. In some exceptional cases, such as in parts of the USA and Canada, coal-fired power plants located near the coal mines have lower generation costs than nuclear plants.

Some key results from the OECD (NEA) study are summarized in Table 5, showing the relative costs of nuclear and coal-fired generation in different European countries and in Japan. The comparison is expressed as a ratio of coal-fired to nuclear generation costs in each country, as it is judged not meaningful to compare the absolute values of generation costs in one country with those in another country.

It will be seen from Table 5 that nuclear electricity is expected to have a 20 to 80% economic advantage over coal-fired electricity, for stations commissioned in the mid-1990's in Europe and Japan.

It should be noted from Table 5 that a significant number of European countries expect the total costs for generation from nuclear plants to be less than the fuel costs alone for coal-fired plants operating in the post-1995 period. Under such circumstances it would be economic to construct new nuclear capac replace existing fossil-fuelled capacity, even additional capacity is not needed to meet elec demand.

Although no specific analysis has been made costs of base-load power generation using natu or oil fuels, it is expected that the latter w significantly more expensive sources than coal the countries over the lifetime of the reference

The situation in North America is less clea There are considerable regional variations i prices and this affects the coal / nuclear cost co son considerably. In some regions of the United with access to cheaper surface-mined coal, coa generation could have a 21 per cent cost adva over nuclear generation; in other regions more r from coal fields, nuclear plants could have a ni cent advantage. However, as explained in th report, the situation would be generally more fa able for nuclear plants in the United States if con tion times and costs could be brought into lin those experienced in, for example, Japan and F or the best industry experience in the United itself. While not considered achievable for a commissioned reactor, this is not unrealistic f longer term.

Data for Canada also show strong regional e and bring out an additional aspect of plant type c Ontario Hydro has been able to build multiple r stations and gain considerable benefits from re tion and scale. These advantages would not nec ly be available to smaller Canadian utilities, ar capital component of generation costs for such a utility, building only one reactor at a time on a sit a profound effect upon the unit production c nuclear electricity.

Clearly the significance of these two factors, n coal price and the ability to capitalize on repli and scale of operations in the nuclear case, will between countries and within continents, so th

Table 5: Nuclear vs. Coal Generation Costs* (1986 OECD–NEA

	Generating cost ratios	
Country	*Coal total / Nuclear total*	*Coal / Nucl*
Belgium	1.62	1.03
Finland	1.33	0.84
France	1.80	1.30
Germany, F.R.	1.68	1.20
Italy	1.41	1.01
Japan	1.37	0.82
Netherlands	1.31	0.88
Spain	1.19	0.83
UK (1) / (2)	1.40 / 1.71	1.15(

*Reference case. Plants for commissioning in 1995.
(1) Sizewell "B" station.
(2) Later repeat PWR station.

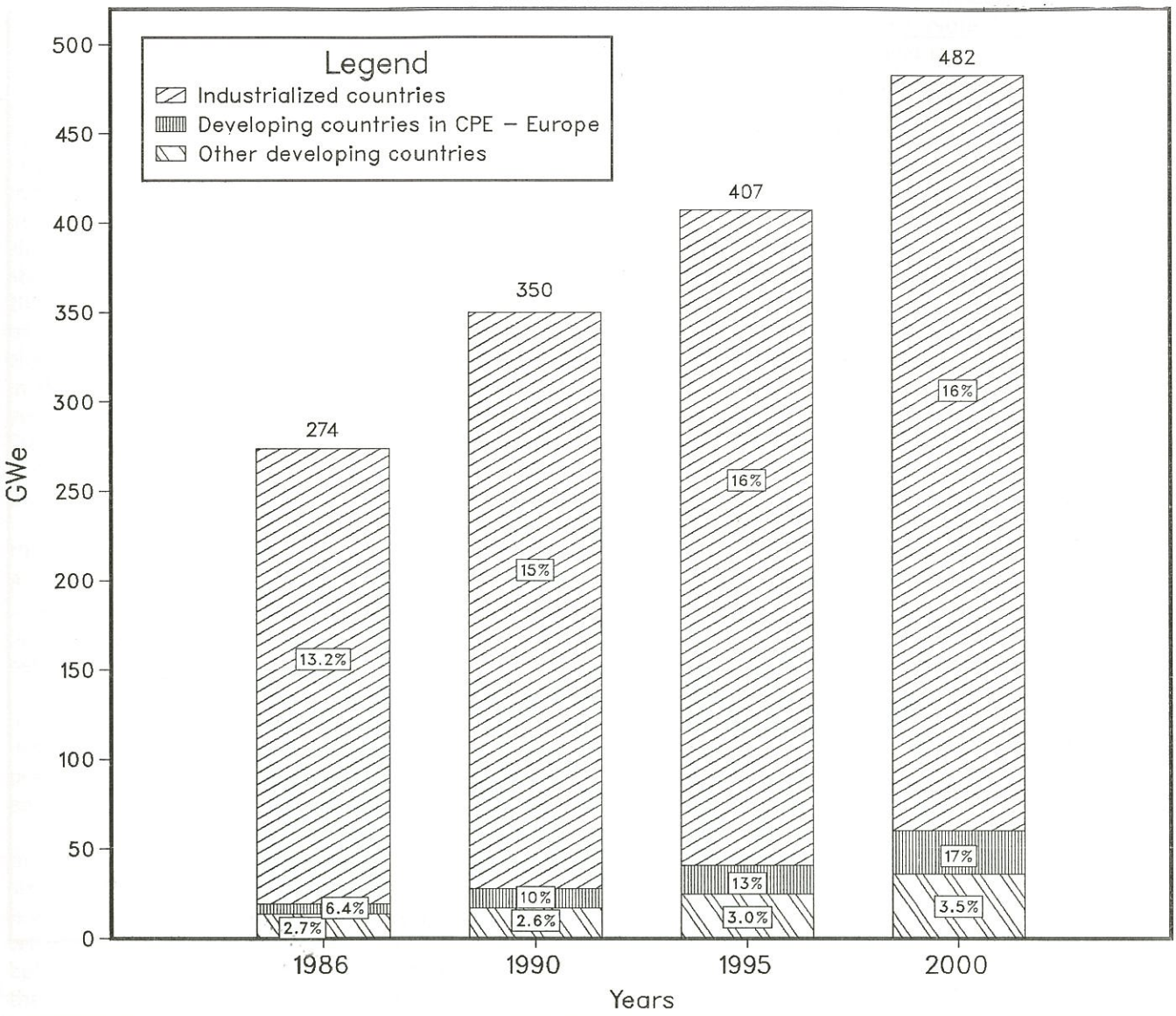


Figure 4 Growth in nuclear capacity up to 2000 (based on IAEA low estimates) and percentage nuclear contribution to total installed electrical capacity. Source: IAEA Energy and Economic Data Bank (IAEA-NENP-87-08).

relative attractiveness of nuclear and coal-based electricity has to be judged in relation to the local circumstances affecting the utility.

Outlook for the Future

Obviously, the Chernobyl accident will affect, to varying degrees, the outlook for nuclear power programmes in different countries. Following the political debates that were precipitated by Chernobyl, some countries will postpone decisions and delay their plans, as well as lower their expectations, for nuclear power development. Indeed, a few governments have felt compelled to present policies promising the dismantling, freezing, or phasing out of nuclear power.

However, it now appears that most countries with well-established nuclear power programmes will continue with the planned further development of nuclear power, with only minor perturbations due to the Chernobyl accident.

Based on data collected by the IAEA, a 'low-case' projection of nuclear capacity up to the year 2000 is shown in Figure 4. Nuclear capacity is expected to increase by 28% in the period 1986–1990; in the period 1990–1995 the increase is projected to be only 16%, reflecting the generally low ordering rate. The situation after 1995 is less predictable, particularly after the Chernobyl accident, but under the IAEA low-case estimate, an increase of 18%, from 407 GW(e) in 1995 to 482 GW(e) in 2000, may be expected.

In the industrialized countries, the percentage share of nuclear capacity in the total installed electrical capacity is expected to increase from 13.2% in 1986 to

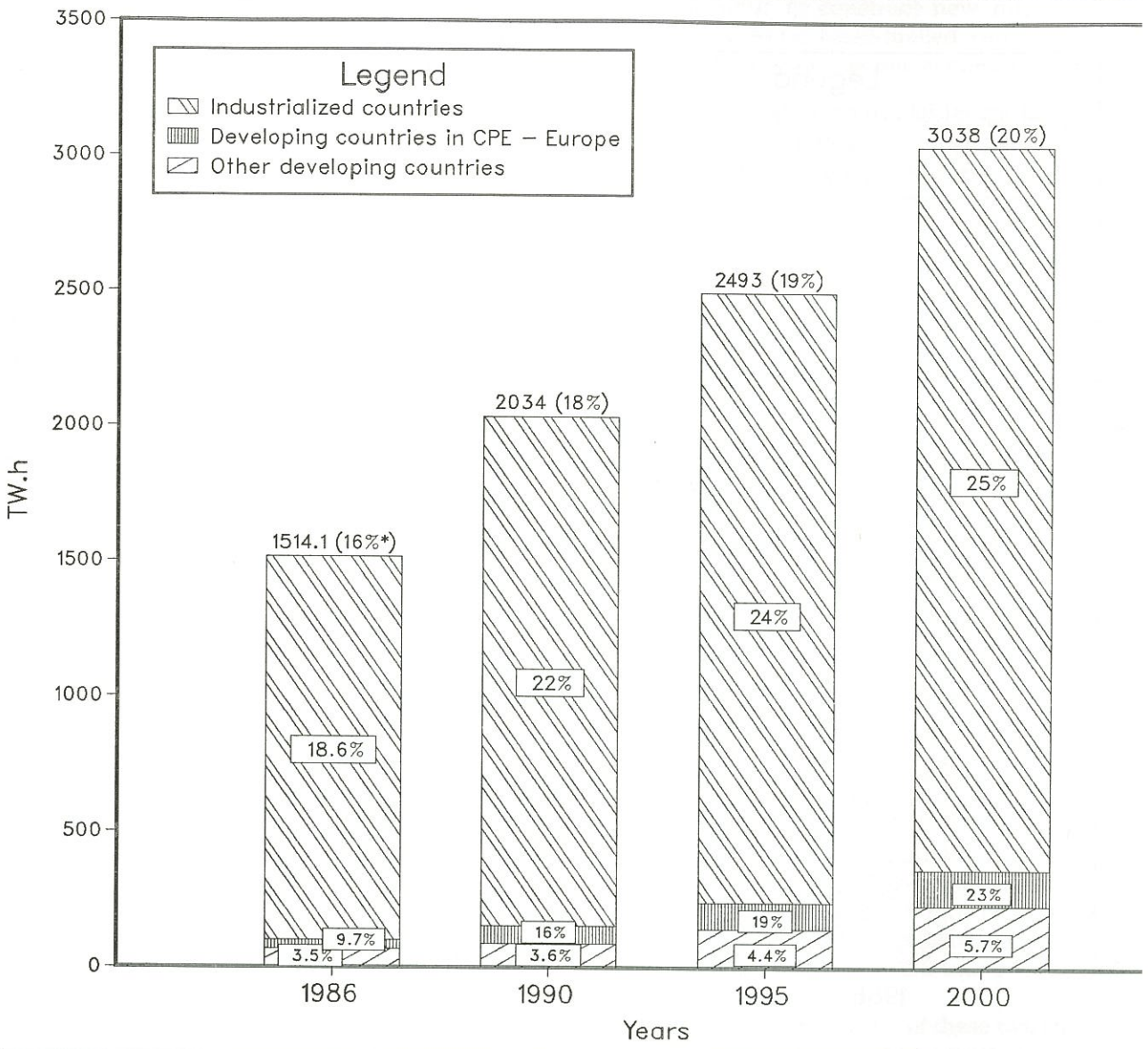


Figure 5 Growth in nuclear electricity generation up to 2000 (based on IAEA low estimates) and its corresponding shares in satisfyin electricity requirements. *Total world electricity production in 1986 has been estimated. Source: IAEA Energy and Economic Data (IAEA-NENP-87-07).

15% by 1990, with a further increase to 16% by 1995, and to remain at 16% up to the turn of the century, when the installed nuclear capacity is projected to be 423 GW(e). In the developing countries in CPE-Europe, the percentage nuclear contribution to installed electrical capacity is expected to increase from 6.4% in 1986 to 10% in 1990, 13% in 1995, and to reach 17%, corresponding to a nuclear installed electrical capacity of 24 GW(e), by the year 2000. In developing countries outside CPE-Europe the nuclear contribution to installed electrical capacity is expected to rise from a present 2.7%, to 3.5% by the year 2000, when nuclear installed electrical capacity is projected to be 36 GW(e).

The limited growth of nuclear power in developing countries is not due to a ready availability of alternative sources of energy, but rather to other consi tions, such as infrastructure requirements, econ viability, and acceptable financial arrangements. can also note that a few technologically adva developing countries are making excellent and e sive use of nuclear energy for electricity produ and have developed a broad research capacity i nuclear field.

However, the projections presented in Figure 4 to some important conclusions which are not de dent on numerical precision of the data:

- The share of world nuclear capacity locate developing countries is likely to remain at m levels for the foreseeable future;
- The IAEA estimates that nuclear power plant

provide only about 5% of the 600–800 GW(e) of total electrical capacity additions projected to be needed in developing countries up to the year 2000.

In terms of electrical energy, the projected growth of nuclear electricity generation up to the year 2000 is shown in Figure 5, for the IAEA low-growth estimates. In the 14-year period up to 2000, nuclear-based electricity generation is expected to more than double. The percentage share of nuclear electricity to the world's electricity generation is expected to increase from about 16% in 1986 to 18% in 1990, 19% by 1995, and 20% by the end of the century. In developing countries in CPE-Europe the growth of the share of nuclear electricity generation is more pronounced, from 9.7% in 1986 to 16% in 1990, 19% in 1995, and 23% in the year 2000. In the developing countries outside CPE-Europe, the share of nuclear electricity generation is expected to increase from 3.5% in 1986 to 5.7% by the turn of the century.

Challenges

As stated above, the nuclear power programmes in a number of countries are the subject of public and political debate. However, this was also the case before the Chernobyl accident.

The economic viability of nuclear power is certainly of great importance to industry and to the competitiveness of countries. However, public opinion on nuclear power reacts more to perceptions of accident risks and environmental impacts.

Public opinion was seen to react dramatically after the Three Mile Island accident and, again, even more strongly after the Chernobyl accident. However, there is already evidence in some countries that opinion is returning to the levels of acceptance that existed before Chernobyl. This trend needs to be reinforced by the dissemination of factual information on nuclear power.

In this regard, a sense of public and political reality needs to be promoted about the relative risks of nuclear power, alternative energy sources, and other industrial activities.

Reference is often made to environmentally benign, renewable sources like hydro, wind, and solar power. Of these, only hydro now makes a significant contribution (21%) to the world's electricity production. It is not without environmental consequences. Big hydro projects have major local impacts and we have learned through the years that they can also entail risks for major catastrophes. One dam accident in 1979 in Morvi in India caused an estimated 15,000 deaths. In Europe, the dam failure in Vaiont, Italy, in 1963, killed 3,000 people.

The chemical plant accident at Bhopal in India resulted in about 2,500 deaths and 150,000 injured, and also had long-term health effects.

This does not mean that one should draw comfort by comparing a disaster in the nuclear industry with disasters in other industries. Nonetheless, it must be made clear that while nuclear power has some unique features, the level of risk it poses to health and environment is not unique. Whether or not these risks are tolerable should be assessed in the same way for nuclear as for other energy sources, and for other industries.

It is a question of choice, and that choice should be made in full awareness of the possibilities for managing the risks for health and environment associated with the different energies. It would be paradoxical if nuclear power were rejected and phased out in some countries for environmental and security reasons, only to be replaced by an energy source that would jeopardize the health and lives of even more people, and that would subject the environment to much greater hazards.

Acknowledgements

The author wishes to thank Miss Raghida Chehade for preparing the illustrations used in this paper.

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Sea Dumping of Radioactive Wastes

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Abstract

This paper outlines the history and regulation of sea dumping of packaged low-level radioactive waste in the ocean. The procedures by which dumping limits are established and periodic safety evaluations conducted under international auspices are described. Particular reference is made to recent negotiations on the future of the practice within the London Dumping Convention and the role played by Canada in this forum, and in others relevant to this subject.

Résumé

Cet article esquisse l'histoire et la régimentation de la pratique du déversement en mer de déchet radioactif de faible activité. Les méthodes utilisées pour établir les règlements internationaux et les évaluations périodiques de sécurité sont décrites. Référence particulière est faite aux négotiations sur le futur de la pratique dans le forum de la Convention de Londres et le rôle du Canada ici et dans les autres organisations internationales pertinentes.

Introduction

The use of the ocean for waste disposal is a subject of some controversy. Some regard the ocean as a legitimate receptacle for wastes arising from human and industrial activities; others wish to preserve the ocean in as pristine a state as possible and therefore oppose any deliberate use of the oceans for waste disposal. Debates over the use of the ocean for waste disposal have intensified during the last four decades, i.e. since the end of the Second World War, both within national jurisdictions and in the international community. Increased public awareness of environmental damage, hazards to human health and the desirabi improving the level of environmental protectio the adverse effects of human and industrial ac on the one hand, and, on the other, of the n dispose of a variety of wastes arising from ant genic activities has contributed both to the pc tion and intensity of these debates. This pa concerned with one facet of this subject – that r to the sea dumping[1] of low-level radioactive wa the ocean that has been carried out under the sions of the London Dumping Convention an largely, under the auspices of the Nuclear E Agency of the Organization for Economic Coope and Development (OECD).

The two main routes of deliberate dispo radionuclides into the ocean being practiced n the direct discharge into the sea of low-level wastes from the reprocessing of nuclear fuels f recovery of plutonium, and the dumping of pac low-level radioactive waste into the deep oce third route of deliberate disposal being consider future use is the emplacement of high-level radic waste within, or on, the seabed. Use of this option currently seems unlikely and, in any ev is at least a decade distant. The word 'deliber used here to discriminate between these activiti the incidental introduction of radionuclides in ocean through fallout from nuclear weapons sions. This latter fallout has both increased the n concentrations of certain natural nuclides, su tritium and radiocarbon, and introduced a vari predominantly artificial (fission-product and a tion-product) nuclides into the marine environ The particular avenue of radioactive waste di that has been the subject of most international c is the dumping of packaged low-level radio waste into the deep ocean, which has been pra since the end of the Second World War. In this p the history of such dumping, the manner in wl has been regulated, and some aspects of the

Keywords: radwaste, sea dumping, ocean disposal, international regulation, oceanography, nuclear safety, rad protection, London Dumping Convention.

debate about its future within the London Dumping Convention are described.

The History of Radioactive Waste Dumping in the Ocean

Dumping of low-level radioactive waste in the ocean has been carried out since 1946. Between 1946 and 1967, the United States dumped approximately 4,000 TBq of radioactive waste into the Pacific and Atlantic Oceans and the Gulf of Mexico. This includes about 1,200 TBq of activation products in the reactor pressure vessel of the *Seawolf* submarine propulsion unit. About 90% of this total activity was dumped in the North Atlantic at the '2,800 m site' located at 38°30'N, 72°06'W. Packaged radioactive waste has also been dumped at ten sites in the northeast Atlantic in the vicinity of 46° N, 17° W by seven western European countries since World War II. The location of the most recently used NEA-approved dumpsite is shown in Figure 1. A summary of the recorded amounts of radioactive waste dumped in the northeast Atlantic between 1949 and 1982 is given in Table 1. The dumped low-level wastes come from nuclear power plants, other nuclear fuel cycle operations, medicine, research, industry and the decontamination and decommissioning of plant and equipment. The waste is of a similar nature to that arising from non-nuclear industrial, medical, and research facilities, except that

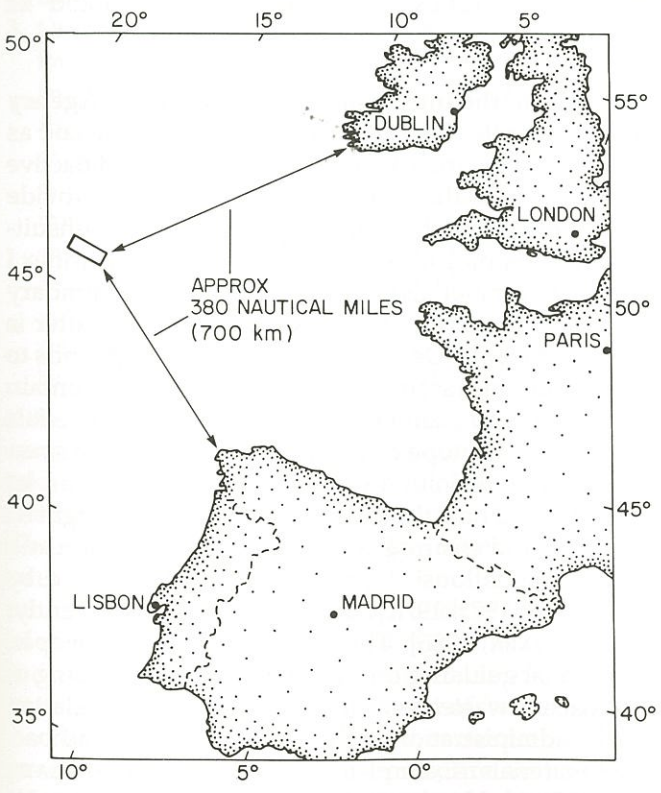

Figure 1: Location of the North-East Atlantic Dumpsite.

Table 1: Aggregate Radioactive Waste Dumping in the Northeast Atlantic Ocean 1948–1982 (Source: Templeton and Bewers, 1986)

Gross mass	142,275 tonnes	
Alpha activity	6.8×10^2 TBq	(18.4 kCi)
Beta / gamma activity	3.8×10^4 TBq	(1,027 kCi)
Tritium*	1.5×10^4 TBq	(405 kCi)

*1975–1982 only. Tritium in previous years included in beta / gamma activity.

Table 2: Average Rates of Individual Nuclide Dumping During the Period 1978–82 (Source: NEA, 1985b)

Nuclide	*Average activity dumping rate TBq per year*	*Nuclide*	*Average activity dumping rate TBq per year*
^{241}Am	$1.28\ 10^1$	^{210}Po	$4.09\ 10^{-1}$
^{14}C	$5.04\ 10^1$	^{238}Pu	6.23
^{244}Cm	$1.13\ 10^{-1}$	^{239}Pu	$2.48\ 10^1$
^{58}Co	$4.13\ 10^1$	^{240}Pu	$1.22\ 10^1$
^{60}Co	$2.54\ 10^2$	^{241}Pu	$1.49\ 10^3$
^{134}Cs	$3.27\ 10^1$	^{242}Pu	$3.40\ 10^{-1}$
^{137}Cs	$1.34\ 10^2$	^{226}Ra	1.25
^{55}Fe	$1.61\ 10^1$	^{35}S	6.09
^{3}H	$2.43\ 10^3$	^{90}Sr	$8.04\ 10^1$
^{125}I	$1.20\ 10^1$	^{234}U	$2.89\ 10^{-2}$
^{54}Mn	7.32	^{235}U	$5.72\ 10^{-1}$
^{237}Np	$1.95\ 10^{-1}$	^{238}U	$2.92\ 10^{-2}$

it includes items having radionuclide contamination in surficial and chemically-incorporated forms and induced radioactivity. Accordingly, this material requires a range of special handling, treatment, and disposal arrangements. The composition of the wastes dumped has varied year by year. Plutonium isotopes and ^{241}Am account for over 96% of the aggregate alpha activity and tritium and ^{241}Pu account for over 87% of the aggregate beta-gamma activity dumped. The remainder of the long-lived beta-gamma activity is composed principally of the fission products ^{90}Sr and ^{137}Cs and the activation product ^{60}Co. The average dumping rates of a number of individual nuclides during the period 1978–1982 are shown in Table 2. The waste packages are designed to provide shielding and containment of the waste during handling and transportation, and to ensure that the packages reach the seabed (at depths equal to, or greater than, 4000 metres) without losing their integrity. The integrity of the packages after descent to the seabed is not assumed or required in the development of regulations. However, some types of package can maintain their integrity, and restrict the release of contained radionuclides, for several decades after dumping. Figure 2 provides an example of the manner in which the waste packages appear on the ocean floor although, in this instance, the container is of U.S. origin and was dumped at the 2,800-metre site in the Atlantic Ocean in the early 1960s.

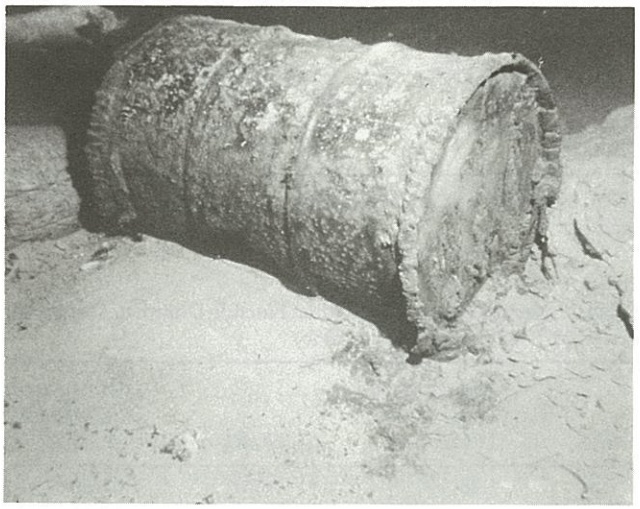

Figure 2: Radioactive waste container on the floor of the Atlantic Ocean. The United States Environmental Protection Agency is acknowledged for providing permission to reproduce this photograph.

Regulation of Sea Dumping of Radioactive Waste

The political and administrative framework within which sea dumping of radioactive waste is carried out involves two international bodies. The first of these is the London Dumping Convention (LDC) which was finalized in 1972 and entered into force in 1975 [IMO, 1982]. This is the major international instrument for the formulation of international regulations for sea dumping activities and has now been ratified by 61 States. The other international body is the Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD), within which data on the actual amounts dumped are collated and the safety of such disposals assessed multilaterally on a quinquennial basis. The NEA created, in 1977, a Multilateral Consultation and Surveillance Mechanism [NEA, 1977] for these purposes and, in all respects, the NEA activities are consistent with the intent and principles of the LDC. All countries involved in dumping in the Northeast Atlantic during the last decade (Belgium, the Netherlands, Switzerland, and the United Kingdom) are parties to this agreement, while other non-dumping NEA countries (Canada, Denmark, Finland, France, the Federal Republic of Germany, Ireland, Italy, Japan, Norway, Portugal, Spain, Sweden, and the United States) have been willing to participate in associated site-suitability reviews and safety assessments carried out under the auspices of the NEA. However, it is within the forum of the LDC, or in connection with this Convention, that the major international negotiations respecting radioactive waste dumping at sea have occurred. As will be shown, the debate on the future of this practice within the LDC has intensified since 1983.

The London Dumping Convention

The London Dumping Convention (formally re
to as the Convention for the Prevention of N
Pollution from Dumping of Wastes and Other N
London, 1972) was created following a recomn
tion of the First Stockholm Conference on the Er
ment and has as its objective the prevention of r
pollution through dumping at sea. The Conven
composed of a series of principles or articles anc
technical annexes. The first annex (Annex I) con
list of substances that are proscribed for dump
the ocean, except as 'trace amounts' in other mat
and includes *inter alia* organohalogen compc
mercury compounds, cadmium compounds, per
plastics, crude oil and associated wastes, and
level radioactive material – deemed unsuitat
dumping at sea because of the human health anc
hazards associated with such disposals. Annex
materials for which special care must be exerci
respect to their disposal into the marine enviro
and includes *inter alia* wastes containing sign
amounts of lead, copper, zinc, organosilicon
pounds,[2] cyanides and fluorides. It also in
radioactive wastes and all other radioactive mat
included in Annex I. Annex II also specifies th
International Atomic Energy Agency (IAEA)
'competent international authority' for specify
the LDC what types of radioactive materials fall
Annexes I and II. Annex III contains a list of c
upon which an evaluation of the effects and per
bility of a proposal to dump material shou
evaluated.

The Role of the International Atomic Energy A

The major role that the IAEA has played in the
the 'competent international authority' for radic
matters under the Convention has been to p
definitions of high-level radioactive wastes 'u
able for dumping at sea' (i.e. the definition of A
radioactive materials). This definition of the bou
between Annex I and Annex II radioactive ma
termed 'the IAEA Definition.' The IAEA also appe
the Definition, a set of 'Recommendations' that c
its advice as to manner in which radioactive ma
having radioisotope concentrations below those
ified in the Definition (i.e., Annex II wastes) c
dumped and how the safety of such dumping mi
assessed and ensured. These periodic 'Definitic
Recommendations' documents have been issu
the IAEA in 1975, 1978 [IAEA, 1978] and, most rec
in 1986 [IAEA, 1986]. The Agency has also deve
additional guidance on the subject of sea dump
radioactive wastes as well as ancillary material re
to the administration of the LDC in respect of ra
tive materials. Examples of such guidance are
Safety Series Nos. 61 and 65 which deal, respec
with the overall framework for the control of

disposal into the marine environment [IAEA, 1983] and environmental assessment methodologies that can be applied to sea dumping of radioactive wastes [IAEA, 1984]. This, then, describes the role and responsibilities of the IAEA under the LDC. Before going into a more detailed explanation of the contents of the 'Definition and Recommendations' documents, the relevant principles of radiological protection advanced by the International Commission on Radiological Protection (ICRP) are outlined briefly below.

Principles of the International Commission on Radiological Protection

The entire consequences of the generation, use, and disposal of radionuclides are regulated on a relatively simple set of principles devised by the International Commission on Radiological Protection [ICRP, 1977] and, in the main, adopted by national regulatory authorities such as the Atomic Energy Control Board of Canada (AECB). These principles are as follows:

1. *Justification*. No practice, involving potential exposures to radiation, should be adopted unless there exist clear net benefits to society, i.e. that the overall benefits outweigh the overall detriments (such as exposures to radiation) to the society affected. Justification applies to an entire practice (e.g. investment in the fission power industry) rather than to components of that practice such as uranium mining.
2. *Compliance with dose limits*. Limits of exposure to radiation for both radiation workers and members of the public are laid down.
3. *Optimization*. Exposures to radiation should be kept as low as reasonably achievable, taking technical, social, and economic factors into account. Thus, exposures should be reduced by technical means, or through the use of alternative options for the handling and disposal of radioactive wastes, so that the overall exposures resulting from the activity or sub-activity are as low as economically and socially justified. The application of this principle requires complex balancing of scientific, economic, and political factors but, in many cases – such as the comparison among options for radioactive waste disposal – these balances can be somewhat simplified.

In practice, justification, as a conscious and deliberate process, is seldom rigorously exercised because the need for regulatory activity has usually followed from investment in the practice, rather than preceded it. Therefore, for all intents and purposes, the major over-riding principles that must be adhered to in the authorization of any sub-practice, such as the disposal of radioactive wastes arising from the nuclear power industry, are compliance with dose limits and optimization. At this juncture, the concept of *'collective dose'* needs to be introduced. The dose consequences of a practice, for optimization purposes, are assessed on the basis of the summation of all individual exposures, referred to as the collective dose. Thus while the upper limit of individual dose is important for the protection of so-called 'critical groups' of individuals, the actual radiation detriment from a practice must consider the sum of all exposures to all exposed individuals.

The IAEA Definition of High-Level Radioactive Wastes under the LDC

The basic process by which the Definition of 'wastes unsuitable for dumping at sea' is derived is composed of an evaluation of the 'capacity' of a hypothetical ocean basin, about the size of the North Atlantic, to receive radionuclides without violating the appropriate dose limits established by the ICRP for members of the public. The introduction of radionuclides into the ocean through dumping is counterbalanced both by radioactive decay and by the removal of those radionuclides to ocean sediments within which the radionuclides eventually become isolated from the biosphere. These processes, namely introduction, removal, and decay, can be modelled in such a way as to relate the concentrations of individual radionuclides in various sectors of the ocean to the rate of waste dumping. The process of deriving the definition is thus one of calculating the rates of release (at the ocean floor) of each potential constituent radionuclide, which results in an equilibrium concentration field which, in turn, corresponds to a radiation exposure (dose) to individual members of critically exposed population groups (critical groups) equal to the ICRP individual dose limit. The oceanographic model relates the marine concentration fields to rates of release of individual nuclides. While equilibrium concentrations can be reached relatively quickly for short-lived nuclides, which decay before they can be transported great distances, such equilibria for the very long-lived nuclides are only obtained on time scales comparable with the half-life of the nuclides, which can be much longer than ocean mixing time scales. Therefore, the model has to predict maximum concentration fields that are obtained after some preconceived time of continued dumping practice, or assume that the practice continues indefinitely, and predict equilibrium fields that in some cases are only obtained on geological time scales. The choice between these options is referred to in the next paragraph of this paper. The oceanographic model is coupled to a radiological model that accounts for routes of human exposure from the marine environment, such as the consumption of seafood, recreational occupation of beaches, and the inhalation of marine aerosols (Table 3). Other potential exposures associated with future activities like deep-sea manganese nodule extraction are also considered. The limiting rates of release that correspond, for each constituent nuclide, to the dose limit are referred to as 'release rate limits' and these constitute the basic values for the establishment of a definition of high-level radioactive waste unsuitable for dumping at sea. For administrative

Table 3: Exposure Pathways Considered in the Derivation of the IAEA Definition (Source: IAEA, 1986)

Pathway	*Symbol*	*Intake rate or occupancy time*	*Other parameters*
Actual pathways		*Ingestion rate*	
Surface fish consumption	FISH-S	600 $g \cdot d^{-1}$ [a]	
Mid-depth fish consumption	FISH-M	600 $g \cdot d^{-1}$ [a]	
Crustacea consumption	CRUST	100 $g \cdot d^{-1}$	
Mollusc consumption	MOLL	100 $g \cdot d^{-1}$	
Seaweed consumption	WEED	100 $g \cdot d^{-1}$	
Salt consumption	SALT	3 $g \cdot d^{-1}$	
Desalinated water consumption	DESAL	2000 $g \cdot d^{-1}$	
		Inhalation rate	*Concentrations*
Suspended airborne sediments	SED	23 $m^3 \cdot d^{-1}$	10 $\mu g \cdot m^{-3}$ water, particles[b]
Marine aerosols	EVAP	23 $m^3 \cdot d^{-1}$	10 $g \cdot m^{-3}$ vapour
		Occupancy times	*Modifying factors*
Boating	BOAT	5000 $h \cdot a^{-1}$	γ 0.2 β 0
Swimming	SWIM	300 $h \cdot a^{-1}$	γ 1 β 0.5
Beach sediments	BEACH	2000 $h \cdot a^{-1}$	γ 0.5 β 0.5
Deep sea mining[c]	MINE	500 $h \cdot a^{-1}$	γ 1 β 0
Hypothetical pathways			
Deep fish consumption	FISH-D	60 $g \cdot d^{-1}$	
Plankton consumption	PLANK	3 $g \cdot d^{-1}$	

[a] Reduced to 300 $g \cdot d^{-1}$ when summing doses over actual pathways so that the total seafood intake is 600 $g \cdot d^{-1}$, made up of 300 g of mid-depth or surface fish, and 100 g each of crustacea, moluscs and seaweed.

[b] Made up of 0.25 $\mu g \cdot m^{-3}$ fine coastal sediment particles, 3.3 $\mu g \cdot m^{-3}$ dried sea salt particles and 6.6 $\mu g \cdot m^{-3}$ particle associated water.

[c] Not included in reference sets of calculations.

purposes, some simplification of the manner of presentation has been made by the Agency such that the Definition takes the form both of release rate limits, and of derived activity concentrations for a small group of classes of radionuclides, with the class limit defined by the most critical or limiting nuclide in each class, based on assumptions as to the limiting mass of radioactive waste to be dumped in any year (Figure 3).

Both the oceanographic and radiological bases of these models have been revised since the formulation of the previous Definition and Recommendations in 1978. In particular, the oceanographic models have been quite substantially refined. The Definition produced in 1978 took account only of radioactive decay as a means of radionuclide removal and ignored particle scavenging and settling processes that result in the eventual association of nuclides with deep ocean (pelagic) sediments. It was previously assumed that, for exposure pathways originating in water, nuclides remained wholly within the water phase; while for pathways originating in pelagic sediments, it was assumed that radionuclides released from dumped containers became wholly associated with pelagic sediments. The model's results were, consequently, somewhat conservative or pessimistic. The revised oceanographic models used for the derivation of the revised definition in 1986 included, for the first time, representations of particle scavenging and sedimentation processes for the removal of radionuclides from the ocean. Additional nuclides to those considered in the 1978 Definition, and a more realistic time sc the practice, were also introduced. Whereas, oceanographic model used for the derivation previous Definition, the practice had previousl

DEFINITION OF HIGH-LEVEL RADIOACTIVE WAST OTHER HIGH-LEVEL RADIOACTIVE MATTER UNSUI FOR DUMPING AT SEA

For the purposes of Annex I to the Convention, high-level tive waste or other high-level radioactive matter unsuit dumping at sea is defined as follows:

(1) Irradiated reactor fuel; liquid wastes from the first extraction cycle of chemical reprocessing of irradiated fuel, or equivalent processes; and solidified forms of suc and

(2) any other waste or matter of activity concentration exc
   (a) 5×10^{-5} TBq kg^{-1} for alpha-emitters;
   (b) 2×10^{-2} TBq kg^{-1} for beta / gamma-emitters with hal greater than 1 year (excluding tritium); and
   (c) 3 TBq kg^{-1} for tritium and beta / gamma-emitters w lives of 1 year or less.

The above activity concentrations shall be averaged over mass not exceeding 1,000 tonnes.

Materials of activity concentration less than those in (2) s be dumped except in accordance with the provisions Convention (Annexes II and III thereto) and the Recommer set out in this document. The maximum dumping rate into ocean basin of volume at least $10^{17} m^3$ shall not exceed 10^8 kg p

Figure 3: The 1986 revision of the IAEA definition and recom tion. Source: IAEA, 1986.

assumed to continue unaltered for 40,000 years (with very long-lived nuclides not attaining steady-state distributions in that time), the most recent Definition was based upon models that invoke equilibrium for nuclides of half-life less than about 100 years, and separately predict maximum concentrations achieved by longer-lived nuclides following 1,000 years of continuous dumping practice. The choice of 1,000 years for the length of practice results from evaluations of the probable life of the nuclear fission process as a means of obtaining electrical power which is estimated to be *ca.* 500 years [UNSCEAR, 1982]. In the radiological models, revised values for the annual limits of intake (ALI) of nuclides, and changes to the recommended methods of calculating exposures to populations, have been introduced, and this has resulted in some changes in the details of the calculations. An additional and important change in the radiological models was the imposition of a reduced value for the individual dose limit. In the Definitions produced by the IAEA up to and including 1978, this limit was set at 5 mSv (milli-Sieverts), but the most recent definition used 1 mSv on the basis that prudence dictated that, for exposures received over long periods (as could be the case for exposures resulting from sea dumping activities) from a single practice (ocean dumping), a lower individual dose limit was more appropriate as a limit to exposures from the practice. To a large extent this latter change has been allowed for by an amendment to the manner in which the Definition is expressed. In the 1978 Definition the release rate limits were defined as those applicable to the introduction of nuclides from *all sources* excepting natural background and fallout from nuclear weapons tests. In the most recent (1986) Definition (Figure 3) the release rate limits apply specifically to ocean dumping.

The Definitions produced by the IAEA become an integral part of the LDC requirements upon its Contracting Parties and define what radioactive materials cannot be dumped in the ocean under its provisions. As already stated, the Definition is accompanied by a series of recommendations that spell out under what conditions radioactive materials can be considered for ocean dumping and how the safety of such practices may be assured. Thus, the Recommendations apply to the manner in which the 'special care' provisions of Annex II of the LDC are to be satisfied. These Recommendations, which are also based upon the principles of the ICRP, include sections on environmental assessment procedures; selection of dumping sites; packaging and transport procedures; control, surveillance, and monitoring of dumping activities, and their consequences. While these recommendations do not take the form of binding conditions under the LDC, Contracting Parties to the Convention have agreed to satisfy the Recommendations to the extent possible. It must be stressed that the LDC merely lays down requirements to which the competent national licensing authority of a Contracting Party shall adhere. Therefore, in practice, the actual licensing of a dumping operation is provided by national authorities, although there is an after-the-fact notification procedure within the provisions of the Convention. It is worthwhile now to provide some more detailed explanation of the Recommendations before dealing with the manner in which the consequences of ocean dumping practices are assessed and the safety of such practices assured.

The IAEA Recommendations Pertaining to the Dumping of Radioactive Wastes Required by Annex II of the LDC

The main features of the Recommendations are dictated by the need to ensure that dose limits are not exceeded and that optimization is carried out adequately for both individual and aggregate sea dumps of radioactive materials. The Recommendations first define appropriate individual dose limits for the practice. They stress that, since members of the public will be receiving doses from other sources and activities, it cannot be assumed that a dose limit of 1 mSv is intrinsically acceptable and that, for actual ocean dumping activities *per se*, an upper bound to the dose should be established. Since, however, no such bound has yet been internationally established, individual national authorities should use a dose limit that is 'substantially less than 1 mSv.' The Recommendations also note that implementation of the optimization principle should ensure that doses actually received from the practice will be only a small fraction of 1 mSv. (As will be seen, the subject of dose upper bounds has also been a topic in recent LDC deliberations on the future of low-level radioactive waste dumping practices.) The recommendations then outline the criteria relevant to environmental assessment and safety assurance of both individual and aggregate dumping operations. They define exclusionary criteria relating to the selection of dumping sites – these must be situated between latitudes 50°N and 50°S and have average water depths greater than 4,000 metres, be clear of continental margins, islands, mid-ocean ridges, ocean trenches, fracture zones, plate boundaries, and areas of volcanic activity. In addition, the use of dumping sites must not interfere with, or prejudice, other legitimate uses of the sea. Sites should therefore be situated away from spawning areas, fishing grounds, the paths of submarine communication cables, and potential ocean mining sites (for the recovery of mineral deposits). Finally, the number of sites should be minimized and their location strictly defined. Each site should be as small as practicable (and no greater than 10^4 km^2 in area) and should not be subject to undue navigational hazards during dumping (i.e. coverage by satellite navigation should be available and the site should not be situated in

shipping lanes). The forms in which waste may be dumped and the packaging requirements are then specified and the manner in which transport and handling activities are made consistent with IAEA Transport Regulations [IAEA, 1973; 1985a] are described. This is followed by a short section on monitoring of the condition of the seas with respect to dumped radioactive wastes in the vicinity of the dumping site. This section refers to relevant guidance provided in previous IAEA [IAEA, 1983] and ICRP [ICRP, 1985] publications. The final sections of the Recommendations deal with approval of the dumping ship and its equipment; the role, duties, and authority of escorting officers (who act as representatives of the national authorities responsible for granting the dumping permit(s) and ensure that the holder of the dumping permit complies with its requirements and conditions); and international cooperation and observation. The text of these latter sections has remained largely unaltered from those in previous IAEA Recommendations. The section on international cooperation and observation is relevant to the actual procedures of multilateral consultation and mutual safety assurance. The Recommendations also advocate that dumping be carried out within the framework of regional cooperation agreements, as provided for by Article VIII of the Convention which states:

> Furthermore, international cooperation in the selection and use of dumping sites and multilateral or international observation of loading and disposal operations is encouraged in order to provide greater assurance that dumping operations are carried out in accordance with the Convention and the IAEA recommendations.

It is in this latter context that the NEA plays an important coordinating role in appointing escorting officers to sea dumping operations, and in conducting site suitability reviews and safety assessments associated with sea dumping of radioactive wastes in the Northeast Atlantic, which has been the only continuous activity of this kind in recent years.

Assessment of Consequences and Safety Assurance of Radioactive Waste Dumping

While the IAEA Definition and Recommendations set the limits for the amounts of material that can be dumped and impose constraints as to the methods and locations for sea dumping, safety assessments of sea dumping operations are normally the responsibility of individual national regulatory authorities. Safety assessments for sea dumping of radioactive wastes in the Northeastern Atlantic Ocean are carried out multilaterally under the NEA Multilateral Consultation Mechanism [NEA, 1977] as part of the quinquennial site suitability review process. These reviews are intended to outline the features of the site that make it acceptable for dumping under the LDC / IAEA criteria, define the nature and composition of the wastes du cover the process of optimization (both in te comparisons between sea dumping and other o and optimization specific to the sea dumping r disposal), provide estimates of current dos predictions of future doses resulting from the p and demonstrate compliance with the provision LDC (e.g., the provisions of Annex III) and th Recommendations under the Convention. Dur process of site suitability review carried out i [NEA, 1980], it was recognized that there ex number of deficiencies in the information perti the dumping site, which had the effect of limit degree to which predictions of consequences c made. Estimates of dose consequences of ag dumping were based upon the same models u the basis for the 1978 IAEA Definition, wh already noted, were conservative. No attem made to estimate collective doses associate dumping because of a conviction that such es would be subject to extremely large (several or magnitude) uncertainties. This meant (and was that the ICRP principle of optimization could c applied to comparison among options, or wit dumping option itself, on a wholly qualitative b individual national authorities. It was noted would be desirable if this deficiency was co through the acquisition of more specific informa the consequences of dumping, and of other d options, for optimization purposes. Conseque the conclusions of the review, it was stated that: is a need to develop a site-specific model transfers of radionuclides, particularly on sh medium time-scales, from the dump area to populations. Therefore, there is clearly a n continue investigations presently aimed at imp our knowledge of transport processes in the East Atlantic. It is recommended that a well c programme plan be developed over the next 12 r within the appropriate international framew meet this objective.' It was further conclude 'although the next assessment of the suitability present dump site will normally take place years, it is recommended that a review of the sc basis for making the assessment and of the g body of knowledge about radionuclide transpo cesses in the North-East Atlantic be undertaken that time.'

It was for these reasons that the NEA establis 1981, a Coordinated Research and Environ Surveillance Program (CRESP) [NEA, 1981]. The b this program was, predominantly, research to in the quality of the site suitability review and assurance – particularly optimization – proce Safety assurance procedures used for sea dum radioactive waste are based upon the use of pre models to describe the results of various scena

ocean disposal of radioactive wastes. Since, in the main, the radionuclides released from previously dumped wastes are not detectable, even within the area of the present dumpsite, heavy reliance has to be placed upon the use of models that depict the processes controlling the transport and behaviour of analogue stable elements. In fact, the weakest aspect of the most recent predictive models is the reliability of representations of bio-accumulation and sediment-water partitioning processes for radionuclides that are vitally important to an appreciation of the rates at which radionuclides are able to enter exposure pathways for man, and the likely effects upon populations of organisms. Significant individual exposures are many decades, perhaps centuries, distant, but the scenarios used for safety evaluation conceive of ocean dumping and direct discharges continuing for the life of the nuclear fission industry, currently projected to be 500 years. Early and reliable prediction of consequences is important if the ocean's resources are to be continuously protected and the assimilative capacity of the ocean is not be be exceeded. The process of refining both the models for, and the process of, safety assurance is not only dependent upon the willingness of nations to be involved and to contribute to this kind of work but also on the acquisition of better understanding of the processes of transport, behaviour, and bio-accumulation of radionuclides and their analogues in the marine environment.

Oceanographic scientists and health physicists, respectively, have played a very important role in the development of 1) oceanographic models that take account of physical, biological, and geochemical processes in the ocean; and 2) radiological models that ensure that all important routes of humane exposure have been identified and considered in establishing the suitability and safety of this practice. It must be remembered that the population potentially exposed to radiation resulting from this practice is extremely widespread. Indeed, for the longer-lived nuclides, it is the group containing heavy consumers of seafood in areas very remote from the northeast Atlantic that may be potentially the most exposed. Several non-dumping nations, including Canada, have adopted the stance that they should participate in the assessment of such practices that have potential effects on widespread populations, not only to ensure that their own populations are adequately protected but also to satisfy international obligations, such as those under the London Dumping Convention. Such involvement has had a very significant impact on the nature of negotiations on the subject and has given these countries an enhanced reputation for objective assessment and as sources of sound scientific advice. In the OECD / NEA forum, countries such as Canada and the United States have been very successful in stimulating substantial improvements in the nature and quality of the safety assessment process and have been willing to contribute to the acquisition of scientific information that is required to improve the technical aspects of these assessments. An example of these improvements has been the CRESP program [NEA, 1981], which was instituted to improve the data and comprehensiveness of safety-assessment modelling. Within the research aspects of this program particular attention has been, and continues to be, paid to the study of potential vertical transport mechanisms that might significantly short-circuit the physical oceanographic transport of dumped radionuclides to the ocean surface. The fact that there are a number of non-dumping countries that wish to ensure that the consequences of sea dumping are acceptable in terms of hazards to their own and other populations, and that, in some cases, might eventually want to evaluate the sea dumping disposal as an option for their own waste management purposes, provides considerable incentive to dumping countries to do the best they can in safety assurance and environmental surveillance. CRESP serves as one of the best and most scientifically rewarding multilateral programs related to waste disposal anywhere in the world. In the first four years of its existence it has produced extremely valuable information that has enabled safety assessment and site suitability reviews to be greatly improved. Details of the results of this program can be found in recent NEA publications [NEA, 1983, 1985a].

By the time of the 1985 Site Suitability Review the state of knowledge regarding conditions at the North-East Atlantic dumpsite had improved considerably, thanks largely to data acquired through the CRESP program. Furthermore, the development of models for radiological assessment purposes enabled better estimates of maximum individual dose to be made, and collective doses to be estimated. The record of this review [NEA, 1985b] contains chapters on the international framework for control of sea dumping, the quantity and composition of the waste dumped, a description of the oceanography and biology of the site, the results of monitoring around the site carried out largely under CRESP, radiological assessment of the site and, finally, a discussion of compliance with the international agreements governing sea dumping. From the results of surveillance work, it was concluded that the incidence of radionuclides in biological samples obtained from the dumping site were generally consistent with those expected from fallout and could not be attributed to radionuclides released from dumped wastes. The major improvement in the radiological assessment aspects of the review were: 1) the use of new models to take into account the rates of nuclide release from waste packages; 2) substantial refinement of oceanographic transport models, partly as a result of work carried out by a Working Group [GESAMP, 1983] of the United Nations Joint Group of Experts on the

Scientific Aspects of Marine Pollution, which had also formulated the models adapted for use by the IAEA in formulating the 1986 Definition; 3) the inclusion of particle-water exchanges; and 4) the use of revised data on sediment-water partition coefficients and biological concentration factors developed jointly by the IAEA and NEA (IAEA, 1985b). Previous assessments had assumed instantaneous release as soon as the packages arrived at the seabed following dumping. The new waste package model introduces release rates from five types of packages representing the range of waste packaging used in the past. The new radiological assessment includes realistic modelling of radioactive decay chains and detailed evaluation of the sensitivity of the models to uncertainties in the parameterization. It was possible to improve the estimation of peak individual dose rates, and to calculate collective doses and collective dose commitments, from past practices and for continued dumping, for a further five years at rates ten times those of recent years.

The radiological impact of dumping activities is predicted to be very low. The peak individual dose from past dumping is calculated to be 20 $nSv.a^{-1}$ (nanoSieverts per year). It arises 200 years after dumping starts and occurs by way of ^{239}Pu and ^{241}Am accumulation in molluscs. Moreover, this peak individual dose involves the assumption that molluscs from the Antarctic might be exploited for human consumption, which is currently not the case. Even if dumping is continued for a further five years at ten times the rates of previous years, the peak individual dose is only 100 $nSv.a^{-1}$ occurring 200 years after the commencement of dumping. The corresponding peak collective dose rates are predicted to be 4.2 man Sv per year for aggregate past dumping, and 42 man Sv per year for past dumping combined with continued dumping for five years at ten times previous rates. These collective dose rates are dominated by the radionuclide ^{14}C which, because of its long half-life, would need to be isolated and contained for very long periods in order to reduce the collective dose from this or other disposal practices. Sensitivity analyses indicated that the peak individual dose rates were most sensitive to changes in the numerical representations of particle scavenging of radionuclides, which confirms previous conclusions that this aspect of the modelling, namely the representation of particle scavenging processes and the manner in which water / particle partitioning is parameterized and numerically represented, is the most important for continued investigation and improvement.

The most recent NEA review [NEA, 1985b] concludes that the site is suitable for continued dumping for a further five years at rates up to ten times those dumped in recent years. If rates of dumping are proposed that would exceed ten times previous rates, the suitability of the site should be reconsidered before approval for these increased rates of dumping is given. I recognised that, before further dumping per issued, certain other aspects of the suitabilit dumping practice need to be considered, par how sea disposal compares, on environmenta economic, and technical grounds, with alt disposal options. This latter point stresses a need for greater attention to be paid to the o tion process and the requirement to demonst exposures are maintained at levels 'as low as ably achievable' – the so-called ALARA princi bodied in the optimization process laid dow ICRP.

Recent Developments within the Forum of t London Dumping Convention

In 1983, at the Seventh Consultative Meetin LDC, Kiribati and Nauru, Pacific island Con Parties to the LDC, proposed an outright ban dumping at sea of any radioactive waste. Afte sion, the meeting adopted a moratorium on dumping pending a review, by an independe of experts, of the scientific and technical bas which dumping practices were regulated a safety assessed. This panel, composed of nominated by the IAEA and the International of Scientific Unions (ICSU), subsequently subm report to the Ninth Consultative Meeting in Se 1985. The main conclusions of this report summarized as follows:

1. The present and future risk to individuals from pa dumping of radioactive waste is extremely small. of developing a fatal cancer or severe hereditary predicted to peak about 200 years in the future at less than 10^{-9} per annum. The most potentially individuals would be those consuming shellfish h in Antarctic waters.
2. Notwithstanding the very small risk to individ aggregate exposure to the global population fr lived components of the dumped waste imply total casualties resulting from past dumping may about 1,000 spread over the next 10,000 years o dominant isotope responsible for this collecti commitment is ^{14}C, with ^{239}Pu being the ne important isotope, giving rise to a few per cent of collective dose. If the radiocarbon, and a fe long-lived radionuclides, were to be removed f waste before disposal in the ocean, the collecti commitment from future dumping operations w very much reduced. However, other means of di these nuclides, other than very long-term conta would result in comparable collective dose comm
3. The incremental dose from past dumping to in marine organisms on the sea-floor at the dum nearby, will be significantly less than the dose organisms receive from naturally-occurring radic

Table 4: Voting Pattern on Resolution LDC 21(9) at the Ninth Consultative Meeting

For	*Against*	*Abstain*
Australia	Canada*	Argentina
Brazil	France	Belgium
Chile	South Africa	Greece
Cuba	Switzerland	Italy
Denmark	United Kingdom	Japan
Dominican Republic	USA	Portugal
Federal Republic of Germany		USSR
Finland		
Haiti		
Honduras		
Iceland		
Ireland		
Kiribati		
Mexico		
Nauru		
Netherlands		
New Zealand		
Norway		
Oman		
Panama		
Papua New Guinea		
Philippines		
Saint Lucia		
Spain		
Sweden		

* In October, 1985, following the Ninth Consultative Meeting, Canada advised the LDC Secretariat that its vote against Resolution LDC.21(9) had been an error and that Canada supported the Resolution.

and hence is not expected to cause any detectable effects on populations of organisms. A resumption of dumping at a rate an order of magnitude higher than previously might cause damage to individual organisms, but would still not be expected to affect an entire population significantly.

The Panel's overall conclusion, acceptable to the experts, but not to national representatives present at its final, expanded, session, was that *'No scientific or technical grounds could be found to treat the option of sea dumping differently from other available options when applying internationally accepted principles of radioprotection to radioactive waste disposal.'* However, the Ninth Consultative Meeting of the LDC, after considering the Expert Panel's report, and after much debate, adopted by a vote (the voting pattern is shown in Table 4) a Resolution [LDC.21(9)] of which the operative part states that the LDC:

1. Agrees to a suspension of all dumping at sea of radioactive wastes and other radioactive matter to permit time for the further consideration of issues which would provide a broader basis for an informed judgement on proposals for the amendment of the Annexes to the Convention. This suspension will continue pending the completion of studies and assessments referred to in paragraphs 2 to 5 hereunder;
2. Requests that additional studies and assessments of the wider political, legal, economic and social aspects of radioactive waste dumping at sea be undertaken by a panel of experts to complement the existing Expanded Panel Report;
3. Requests that further assessments examine the issue of comparative land-based options and the costs and risks associated with these options;
4. Requests that studies and assessments examine the question of whether it can be proven that any dumping of radioactive wastes and other radioactive matter at sea will not harm human life and / or cause significant damage to the marine environment;
5. Requests the IAEA to advise Contracting Parties with respect to certain outstanding scientific and technical issues relating to the sea dumping of radioactive wastes; specifically:
   (a) To determine whether additional risks to those considered in the revised IAEA Definition and Recommendations justify re-examination of the definition of radioactive wastes and other radioactive matter unsuitable for dumping at sea for certain individual radionuclides;
   (b) To establish source (dose) upper bounds appropriate to the practice of radioactive waste dumping under the Convention;
   (c) To define quantitatively the exempt levels of radionuclides for the purposes of the Convention,
6. Requests the Organization to approach international agencies to establish and maintain an inventory of radioactive wastes from all sources entering the marine environment;
7. Calls upon Contracting Parties to develop, as envisaged in Article X, procedures for the assessment of liability in accordance with the principles of international law regarding state responsibility for damage to the environment of other States or to any other area of the environment resulting from dumping.

Some aspects (Paragraph 5) of the work proposed here are being dealt with as part of the ongoing work program of the IAEA, others (Paragraph 3) are an existing requirement on national authorities under the IAEA Recommendations, and others (Paragraph 6) are being considered in the context of future UNSCEAR or IAEA work plans. None of these items is in any way urgent at the current rates of dumping. There is considerable merit in considering assessments of liability under international law (Paragraph 7) since this is likely to prove to be a very time-consuming task and one that might subsequently be applied not only to other ocean waste dumping practices but to a whole range of practices that potentially affect other nations. It is difficult, however, to see how the harmlessness of dumped waste (Paragraph 4) can be established when risk assessment is at the heart of the entire radiological protection process. It is implicitly assumed that risk of radiological harm increases in proportion to the dose

received without threshold, and therefore non-zero risk is intrinsically assumed in any practice that involves dissemination of radionuclides to the environment. The major subject of controversy within the LDC has been Paragraph 2, to which some Contracting Parties have strenuously objected on the grounds that the LDC has no right to deal with political and social issues within the purview of sovereign state jurisdictions. Nevertheless, a procedure has been devised to form a new Contracting Party expert panel on issues raised in Paragraph 2 during 1987. It remains to be seen how these activities will impinge upon the future of ocean dumping of radioactive waste. It should be noted that the Canadian delegation proposed amendments to delete Paragraphs 2 and 4 from the resolution prior to the vote and, when these amendments were not adopted, voted against the resolution (Table 4). Subsequently, however, the Secretary of State for External Affairs, in a letter to the Secretary-General of the International Maritime Organization, pointed out that the Canadian delegation had erred and that Canada supported the resolution.

In view of the strong international controversy surrounding low-level radioactive waste disposal in the ocean it is difficult to predict the future of this practice. The current moratorium on such dumping which has been in effect since 1983 will continue, and probably be observed, until at least 1988, when the next Consultative Meeting of the London Dumping Convention is to take place. Progress made in respect to the further evaluations proposed in the most recent LDC resolution on this issue will be reviewed at that time. Meanwhile the United Kingdom, one of the countries most affected by the moratorium, has carried out and published an evaluation of various options for the disposal of low- and intermediate-level solid radioactive waste [HMSO, 1986a] that would seem to point to continued U.K. interest in the sea dumping option for some kinds of radioactive waste, although the practice of waste dissemination to the environment has been recently criticized by the British House of Commons Environment Committee [HMSO, 1986b]. Nevertheless, the strong front against sea disposal is bound to stimulate increased efforts to develop and prove alternative, land-based, methods of disposal. Clearly, the final verdict on sea dumping of low-level radioactive wastes under international law is not yet in. The developments within the London Dumping Convention during the next two years will probably decide both the fate of this practice and the future of the Convention itself.

If sea dumping of radioactive waste is eventually proscribed through, for example, the allocation of *all* radioactive materials to Annex I of the Convention, some urgency will be placed upon the definition of so-called *de minimis* amounts of radioactivity. Unless such a definition can be formulated it can be argued that, since all substances, natural and anthrop contain radionuclides, albeit of natural origin ing of any material should be prevented beca intrinsically radioactive. In such an event, the f tion of exemption rules to permit materia considered for ocean dumping without consi of their radioactive character would need to b lated. This subject is also one that has beer work program of the IAEA for several years. Wh relatively simple matter to define levels of triv for the development of such exemption ru rather more difficult to convert dose to concentration that would be needed for application. Such conversions would probabl be site-specific and, since most ocean dumpin in the coastal zone, the variety of conversions to account for the extremely large heteroge conditions in inshore areas will be large. Furth the application of exemption rules in the cc collective dose requires more study.

Conclusions

As with other nuclear industrial activities, rac waste dumping has aroused strong feelings a potential dangers to human and environmenta All recent assessments of the consequences ous dumping, and of continued dumping for decade at similar rates, yield very small in doses to members of critical populations. Th on the safety of the practice has consequentl around the collective dose commitment and th ated casualties worldwide. Certainly, the leve associated with the practice are small compa levels of involuntary risk currently assumed bers of a wide variety of societies [Allman, 198 is, however, some merit in the argument that of certain societies not currently enjoying the of peaceful nuclear energy (and indeed choc so to do) are potentially at risk in the sense th casualties in such societies are statistically based upon the assumption of radiological ha a function of dose without threshold. The ments are in many ways analogous to the com in my backyard (NIMBY) syndrome,' but at ar tional level. Unless the current differences the dumping and objecting countries can be through the establishment of some mutua factory compensation mechanism, the sub continue to place considerable strains on Continued lack of consensus or mutually a agreement might result in the impasse being certain nations unilaterally resuming sea du some time in the future. Whether this enda continued viability of the London Conver matter of considerable concern to Canada a countries.

Notes

1. The use of the term 'dumping' in this paper is consistent with that in the London Dumping Convention, wherein it is defined *inter alia* as meaning 'any deliberate disposal at sea of wastes or other matter from vessels, aircraft, platforms or other man-made structures at sea.' Dumping therefore covers the introduction of waste material into the sea from ships in packaged or unpackaged forms. The term 'sea disposal' is more generic and is used to cover both dumping, as defined above, and discharges to the ocean from land.
2. A recommendation to delete organosilicon compounds from Annex II was adopted by the LDC in 1986.

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The Effect of Steam Generator Tube Temperature on the Stress Corrosion Cracking of Alloy 600

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Abstract

This review of the stress corrosion cracking (SCC) of Alloy 600 tubing assesses the relative importance of steam generator operating temperature in contrast to other factors. Although temperature is the most important environmental factor affecting primary side high-purity water SCC, the individual effect of stress and the combination of stress and microstructure are far more important. In regards to secondary side crevice-related SCC, other environmental factors are of greater importance than temperature. The use of stress-relieved tubing, low design stresses, stringent water chemistry control, and non-stagnant working fluids are the major reasons for the absence of SCC in many nuclear steam generators (SG). This includes the Babcock & Wilcox Canada (B&WC) recirculating steam generator (RSG) designs and the B&W USA once-through steam generator (OTSG) designs. This is in contrast to other SGs that have experienced SCC. The afflicted units usually have the following characteristics: low-temperature mill-annealed tubing, highly stressed areas in the tubing, poor water chemistry control, stagnant regions, and low recirculating ratios. An analysis of laboratory data demonstrates that the use of stress-relieved (621°C (1,150°F)/10 hrs) or thermally-treated (704°C (1,300°F)/16 hrs) tubing stressed (total stress, i.e., applied plus residual) to no higher than 70% of the yield strength will survive 33 times longer at 343°C (650°F) than mill-annealed tubing stressed to 125% of the yield strength. This improvement is at least an order of magnitude larger than the increase in mill-annealed tubing life obtainable by reducing the operating temperature from 327°C (620°F) to 304°C (580°F).

Résumé

Cette étude de la fissuration par corrosion sous tension (FCT) des tuyaux en Alliage 600 évalue l'importance rela température de fonctionnement du générateur de v rapport à d'autres facteurs. Bien que la températu plus important des facteurs de milieu qui affectent la l'eau très pure du côté primaire, l'effet particulier de l et la combinaison de la tension et de la microstruc beaucoup plus importants. En ce qui concerne la FC secondaire liés à des crevasses, d'autres facteurs ont une plus grande importance que la température. de tuyaux soumis à un traitement de détente des des tensions nominales faibles, un contrôle rigour chimie de l'eau, et des liquides de travail non stagn les principales raisons de l'absence de FCT dans de r générateurs de vapeur (GV) nucléaires. Ceci inclut cepts de générateur de vapeur à recirculation de B Wilcox Canada et les concepts de générateur de vap recyclage de Babcock & Wilcox USA. Ceci s'oppose a GV qui ont été atteinte de FCT. Les unités affec généralement les caractéristiques suivantes: tuyau en usine à basse température, zones de tension éle les tuyaux, mauvais contrôle de la chimie de l'eau stagnantes, faibles taux de recirculation. Une ana données de laboratoire démontre que l'emploi d ayant subi un traitement de détente des tension (1,150° F)/10 hr) ou traités thermiquement (704° C (10 hr) soumis à des tensions (tension totale, c. à d. a plus résiduelle) maximum de 70% du seuil de dureront 33 fois plus longtemps à 343° C (650° F) tuyaux recuits en usine soumis à des tensions de seuil de plasticité. Cette amélioration est supérie moins un ordre de grandeur à l'augmentation de la d tuyaux recuits en usine qu'on peut obtenir en réd température de fonctionnement de 327° C (620° F) (580° F).

Introduction

Previously published work showed that the t ture of the environment is a factor in the scc 600 tubing on the primary side of U-bends an primary and secondary sides in tubesheet reg

Keywords: stress corrosion cracking, steam generator designs, Alloy 600 tubing, heat treatment, activation energy, c experience, residual stress, water chemistry.

Table 1: PWR and PHWR Steam Generator Operating Experience through December, 1984[a]

Manufacturer	Number of plants	Number of tubes in operation	Tube years accumulated ($\times 10^6$)	Total tubes removed from service	Defects per tube – year ($\times 10^4$)
B&W Canada[b]	15	405,930	2.195	33	0.15
B&W USA (OTSGS)[c]	8	248,052	1.319	820	6.22
Mfr C	29	294,756	0.713	613	8.59
Mfr D	11	124,376	0.806	832	10.32
Mfr E	6	50,099	0.282	293	10.38
Mfr F	9	77,668	0.347	2,206	63.49
Mfr G	13	211,485	1.000	8,204	82.01
Mfr H	49	527,758	2.481	21,729	87.59
World totals[f]	147	2,083,156	9.691	34,730	33.59 (Avg)

[a]Based on data from AECL 8268 (1984) and AECL 9107 (1986).
[b]Excluding NPD reactor – 1 horizontal SG for 22 MWe, used mainly for research.
[c]Excluding number of tubes plugged at TMI-1, per US NRC NUREG-1063.

This current report reviews the effect of temperature on SCC in relation to the effects of other important variables such as stress state, material condition, alloy type, water chemistry, and SG designs.

Stress corrosion cracking in SG tubing is a complex function of the following parameters: temperature, time, tube material composition, tube material microstructure, total stress (applied plus residual), environment and interactions of these factors. Sufficient data and information are available to separate semi-quantitatively these interactive effects and to demonstrate their relative importance.

Both operating experience and laboratory results were used to evaluate the relative importance of each variable. Factors other than temperature should be considered in ascertaining the reasons for the excellent performance of some SGS as opposed to the relatively poor performance experienced by others. As Table 1 and Figures 1 and 2 show, the improved performance of some of these SGS is not restricted to the absence of SCC but includes an extremely low incidence of other tube degradation problems. A more detailed categorization of reasons for tube plugging in OTSGS is presented in Table 2. At present there is a remarkable accumulation of over 3×10^6 effective full-power tube years (EFPTY) in B&W RSGS[1] and over 1.5×10^6 EFPTY in B&W OTSGS without in-service SCC.

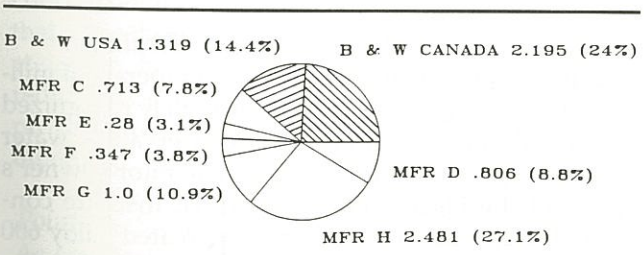

Figure 1: PWR and PHWR operating experience. Accumulated tube-years ($\times 10^6$).

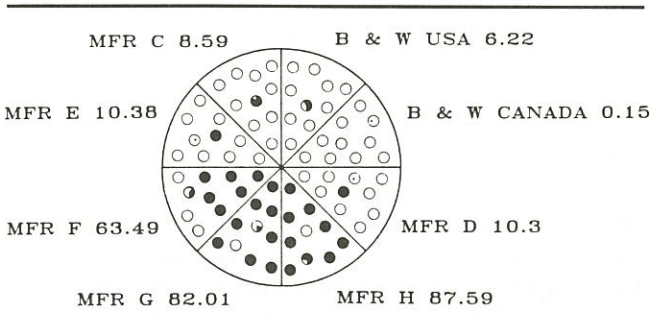

Figure 2: PWR and PHWR operating experience. Number of tubes plugged/10^4 tube-years. Each tube represents 10 tubes.

Operating Experience

There are several examples from operating experience which demonstrate that other factors are more important than temperature in controlling the incidence and severity of SCC.

Point 1 – Lack of Correlation of SCC with Temperature in Individual Steam Generators

For SG designs susceptible to SCC there exists a higher incidence of primary side cracking in the U-bend regions than in the appreciably hotter straight tube hot leg regions. This is an example where tube material condition in combination with the stress state, and not temperature, are the primary controlling factors of SCC. Two examples are Cook 2 and Farley 1, which are of similar SG design (W-51). These two units reported U-bend cracking at 302°C (575°F) in 140 and 30 weeks, respectively, but no hot leg cracking at 318°C (605°F) and 322°C (611°F), even after 360 weeks of operation [2, 3, 4].

Point 2 – Lack of Correlation of SCC with Temperature for Equivalent Steam Generator Designs

Not all SGS of similar design, operating characteristics, and time in operation are susceptible to SCC, although temperatures are essentially equivalent. In some cases,

Table 2: Secondary Side Damage Mechanisms Requiring Tube Pluggage or Removal (through 11/20/85)

	Maximum one OTSG		*All OTSGs*[a,b]	
	Number of tubes	*Percent*	*Number of tubes*	*Percent*
Corrosion fatigue – lane region	27	0.17	111	0.05
IGA – upper span/tubesheet	184	1.18	257	0.12
Corrosion/erosion – peripheral tubes	218	1.40	287	0.13
Fretting	11	0.07	28	0.01
Waterhammer[c]	10	0.06	26	0.01
Unknown	25	0.16	106	0.05
Not Service Related	79	0.51	278	0.13
Total			1,098	0.50
Total service-related			820	0.37

[a]Includes seven operating plants, excludes TMI-1 and TMI-2.
[b]Based upon approximately 217,000 tubes.
[c]Only three operating plants contained internal auxiliary feedwater headers which were associated with waterhammer damage to the tubes because of the deformed header.

some units free from SCC operate at even higher temperatures than those which are experiencing SCC. Such an example is Cook 2 compared to Salem 1, both of which are W-51 SG designs. Cook 2 has experienced U-bend cracking in 140 weeks, whereas Salem 1 has been operational for over 400 weeks with no observed U-bend problems. The temperature profiles are similar in that the primary inlet temperatures are 319°C (607°F) and 321°C (609°F), respectively [2, 3, 4].

Point 3 – The Importance of Stress and Deformation

In-service primary side SCC is invariably associated with areas characterized by high plastic deformations and high residual or applied stresses [5, 6]. These include U-bend, transition zones, and denting affected areas (i.e., U-bend and crevice areas). These deformed areas are undoubtedly under high residual stress, which is believed to approach the yield strength of the deformed material. Furthermore, dynamic straining occurs during denting and is probably producing strain rates known to cause more severe tendencies for SCC than equivalent, but constant stress conditions.

Non-deformed Alloy 600 tube sections adjacent to U-bend and transition areas displaying SCC have performed well in some cases for over 20 years. For these cases, where SCC has occurred in less than 1–2 years in deformed regions, it may be concluded that deformation produces an effect that is at least 20 times that of temperature. Furthermore, the temperature differentials between the non-deformed hot leg region and the U-bend area, and between the non-deformed hot leg region and the cold leg transition region are typically 22°C (40°F) and 28°C (50°F), respectively. This translates into an even greater difference between the effect of deformation and stress versus temperature on SCC.

Point 4 – The Importance of Water Chemistr

In many steam generators experiencing secon
SCC, crack propagation was arrested and init
SCC reduced after improved water chemistry
were introduced. These practices included co
from phosphate chemistry (PO_4) to all-volatil
(AVT) secondary water, the avoidance of cc
leaks, the use of demineralizers, and the re
copper alloys from feed streams. These succes
obtained for many cases in which the c
temperatures remained unchanged [2, 3]. Co
the sodium to phosphate ratio to between 2.
reduced the incidence of SCC in phosphat
generators. Unfortunately, the conversion fr
phate to AVT secondary water chemistry als
SCC to occur as a result of denting [7]. The sec
events leading to this SCC involved chlorid
from condenser leaks, which concentrates
support structure crevices, causing accelerate
of non-protective magnetite. The resultant
were in some cases sufficient to cause both
and secondary side SCC in the tube/support
areas, and U-bend primary side SCC. One ame
practice includes adding boric acid to re
corrosion rates of the carbon steel support p

Point 5 – The Importance of Thermal Treatm

The increased susceptibility of low-temperat
annealed Alloy 600 tubing to SCC is widely re
[8]. In addition to suggesting conservati
chemistry practices, the Steam Generator
Group of the Electric Power Research Insti
siders the substitution of thermally treated
for mill annealed as another means of redu
likelihood of SCC [2]. This recommendation
without corresponding requirements for redu

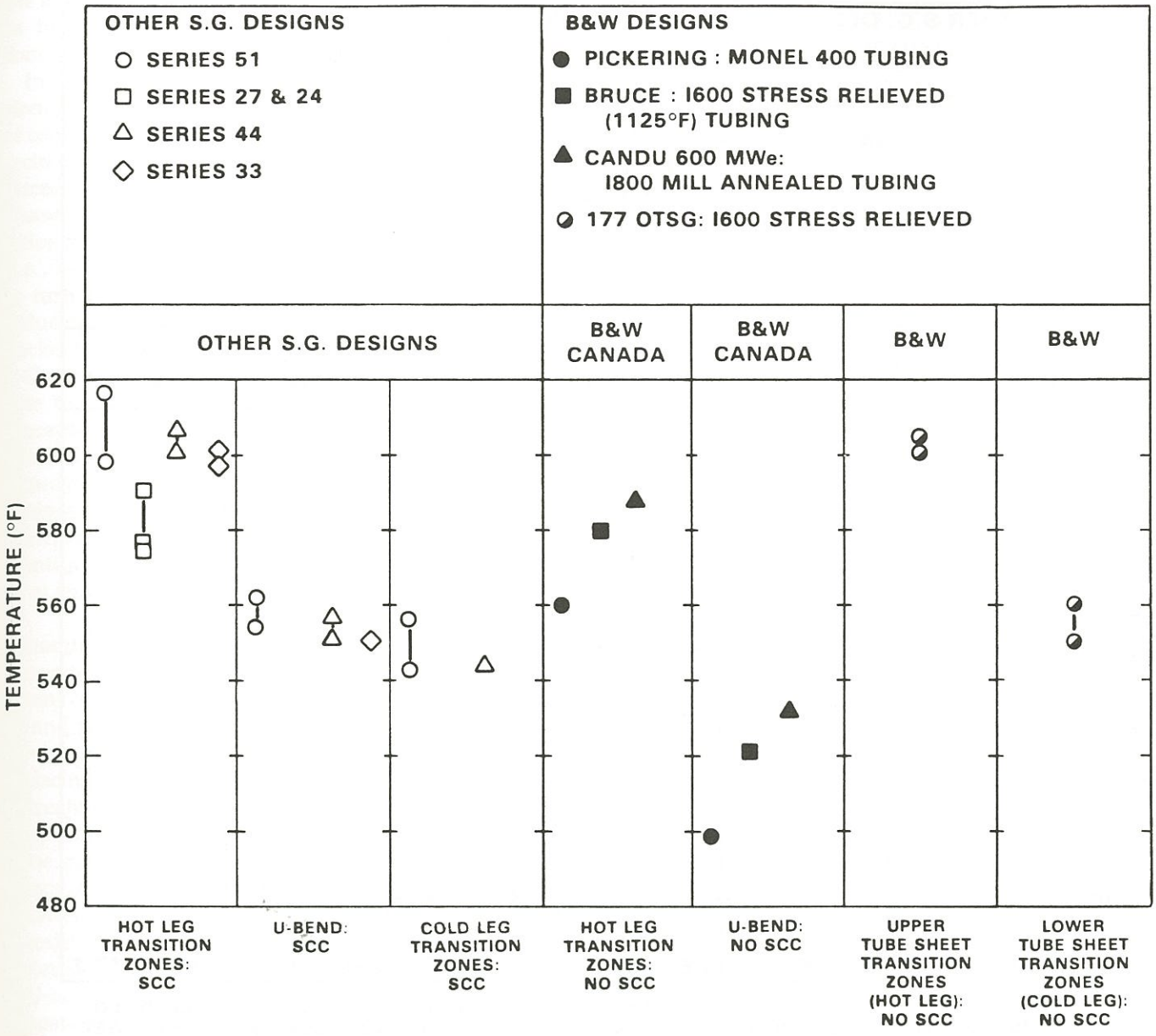

Figure 3: Temperature profiles and primary side SCC incidence.

erating temperatures. Furthermore, thermally treated base Alloy 600 tubing is used in nickel-clad sleeving operations to reduce the risk of SCC for members known to be placed in regions already possessing critical conditions [9]. It should be noted, however, that these thermal treatments must produce appropriate grain boundary carbide structures as well as reductions in residual stress to fully obtain the improved properties relative to the mill-annealed condition.

Point 6 – Steam Generator Design Effects

As emphasized earlier, stress-relieved or alternative-material SG tubing in both recirculating and once-through designs has performed extremely well in terms of high-temperature water SCC. A significant portion of this tubing operates at temperatures equivalent to or in excess of those encountered in failed tubing of other designs, as seen in Figures 3 and 4. These figures illustrate, for example, that the primary side operating temperatures of hot leg transitions in non-susceptible designs are typically 11°C to 17°C (20°F to 30°F) higher than U-bend and cold leg transition temperatures for designs susceptible to SCC. Furthermore, there is as little as a 11°C (20°F) differential in hot leg transition temperatures between non-susceptible recirculating designs and those experiencing extensive SCC. Similarly, there is only a 17°C (30°F) differential in U-bend temperatures [2, 3]. Activation energy predictions of failure times indicate that this

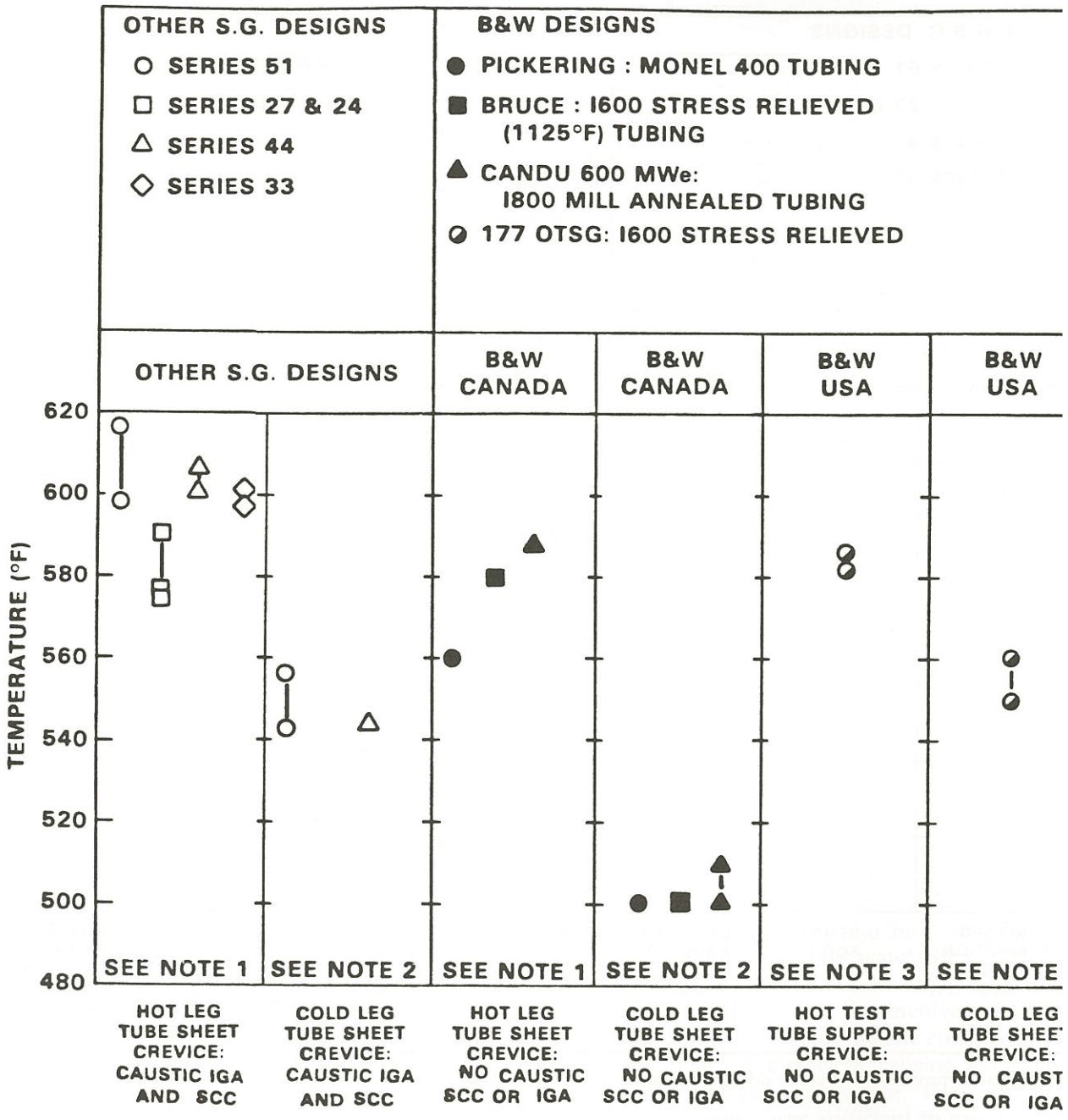

Figure 4: Temperature profiles and secondary side scc incidence.

large divergence in performance cannot be attributed to this small variation in operating temperatures.

Another important feature of SG design affecting secondary side SCC is tube support crevice design. The use of open flow tube support designs elimin tight crevice conditions which create a vulnera the buildup of contaminants and sludges. more, the improved fluid flow precludes cha

the local secondary side crevice environments, which in turn can lead to either high acidic or caustic conditions conducive to denting and SCC.

In conjuntions with tube support crevice design, circulation ratios[2] can greatly affect the concentration of contaminants. Because OTSG designs do not include recirculation of secondary water, there exists a reduced tendency to build up contaminants in the liquid phase. In addition, the typically higher circulation ratios for some RSG designs (i.e., 5–6) versus others (i.e., 2–4) results in higher fluid flow velocities, which in turn improves the stripping action of the fluid, reducing the tendency for crud deposition and vapor pocket formation.

On the basis of these observations, it seems reasonable that the improved performance of these non-susceptible SGS can be attributed to the use of stress-relieved (or thermally-treated) tubing, better water chemistry control, lower applied and residual stress designs, and better fluid flow conditions. Indeed, incorporation of some, if not all, of these improvements may be necessary to guarantee that SG lives will be at least 30 years.

Laboratory Experience

Laboratory studies show thermal treatment to be a means of markedly improving resistance to both caustic and primary water SCC [1, 2, 4, 10–14]. Although service experience is scarce concerning the success of replacing damaged mill-annealed tubing with thermally treated material, laboratory results are abundant. They conclusively demonstrate the importance of these metallurgical factors in reducing the tendency for SCC, to the extent that this replacement practice was followed in a number of cases [15]. This improvement in resistance to SCC is greatest when thermally-treated material is substituted for cold-worked material typical of U-bend or roll transition regions. Furthermore, it is almost certain that future steam generator designs will avoid recommending temperature reductions, but will vary other design characteristics to avoid SCC, and yet maintain or even increase operating temperatures to produce equivalent or enhanced power ratings and efficiencies.

There is only one incidence of SCC of non-deformed mill-annealed Alloy 600 in high-purity water 365°C (689°F) when loaded to below the room temperature 0.2% offset yield strength [16]. Indeed, this one sample was loaded to 90% of the room-temperature yield strength of the non-deformed material, which will produce a substantial plastic strain at test temperature (i.e., near yield). Failure occurred in only 90 days. Two other SCC failures occurred in high-purity water at 343°C (650°F) in the laboratory in deformed Alloy 600 at stresses below the room-temperature yield strength of the deformed material [17]. One specimen prestrained 5% failed in 19,700 hours (2.24 years) when tensile

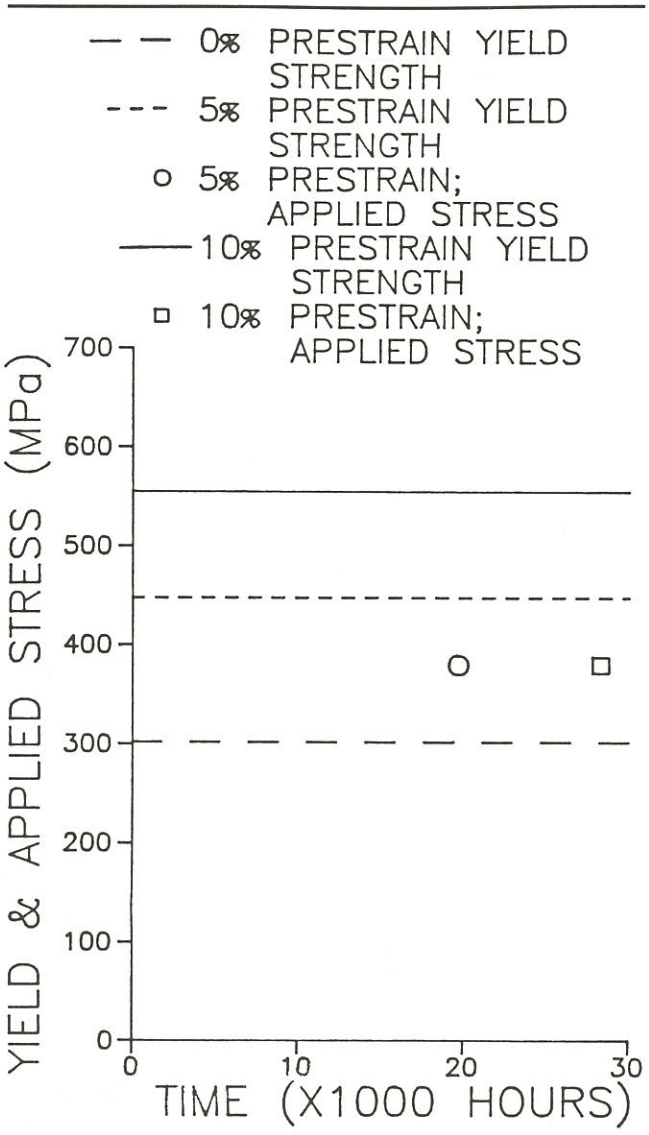

Figure 5: High-purity water tests 343°C (650°F). Applied stress must be near to or grater than the 0% prestrain yield strength to cause SCC.

loaded to 84% of the prestrained material yield strength (i.e., 379 MPa (55 ksi), YS = 448 MPa (64.9 ksi)). The second specimen prestrained 10% failed in 28,300 hours (3.23 years) when loaded to 67% of the prestrained yield strength (i.e., 379 MPa (55 ksi), YS = 555 MPa (80.5 ksi)). Evaluating these results, as shown in Figure 5, demonstrates that stresses must be near if not above the mill-annealed material yield strength (determined at operating temperature) in order to produce SCC in high-purity water.

Recently, attempts have been made to estimate activation energies for initiation of SCC based upon laboratory studies [4, 18, 19]. The activation energy is determined from the slope of an Arrhenius curve, which is a plot of the logarithm of time to failure as a function of the reciprocal temperature. These estima-

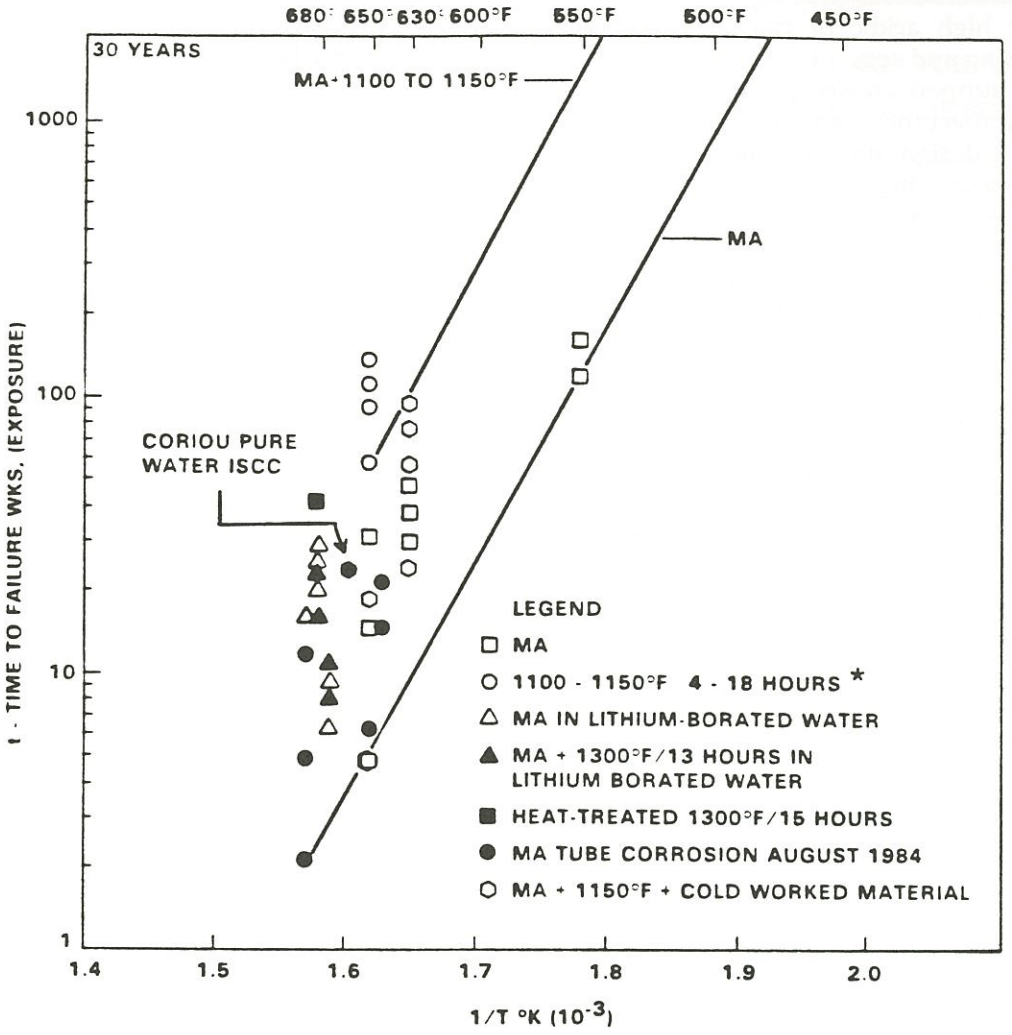

Figure 6: scc of Alloy 600 U-bends in high-temperature water (from reference 4).

tions in turn are compared to actual service data to develop models for remaining life prediction. In one such effort [4] it is assumed that the activation energy for initiation of scc is about 40 Kcal/mole – K°, and is basically independent of tube material condition (i.e., whether it is low- or high-temperature mill-annealed, thermally treated, stress-relieved, or even highly cold-worked). This estimation is made for the minimum failure times as a function of temperature for the most sensitive material condition, i.e., low-temperature mill-annealed. Therefore, the Arrhenius plot for the more resistant material condition should be of equal slope to the reference curve for low-temperature mill-annealed material, but displaced to longer times to failure, as shown in Figure 6.

It should be noted, however, that thermal treatments (i.e., 621°C (1,150°F) for 10 hours a
(1,300°F) for 15 to 16 hours) produce grain b
decorated microstructures similar to those
from stress relief treatments (i.e., 593°C (1,
621°C (1,150°F) for 10 to 18 hours) which are e
resistant to high-purity water scc. Indeed,
inspection of laboratory results reveals that
scc failures occurred out of a total of 11
relieved tests. Many stress-relieved specim
tested for as long as 93,750 hours (10.7 years)
[20]. Furthermore, failures occurred in stress
samples only for loads well in excess of t
temperature yield strength, and only for tem
of at least 343°C (650°F). Therefore, the res
sented in Figure 6 must be carefully int
recognizing that the vast majority of stress-re

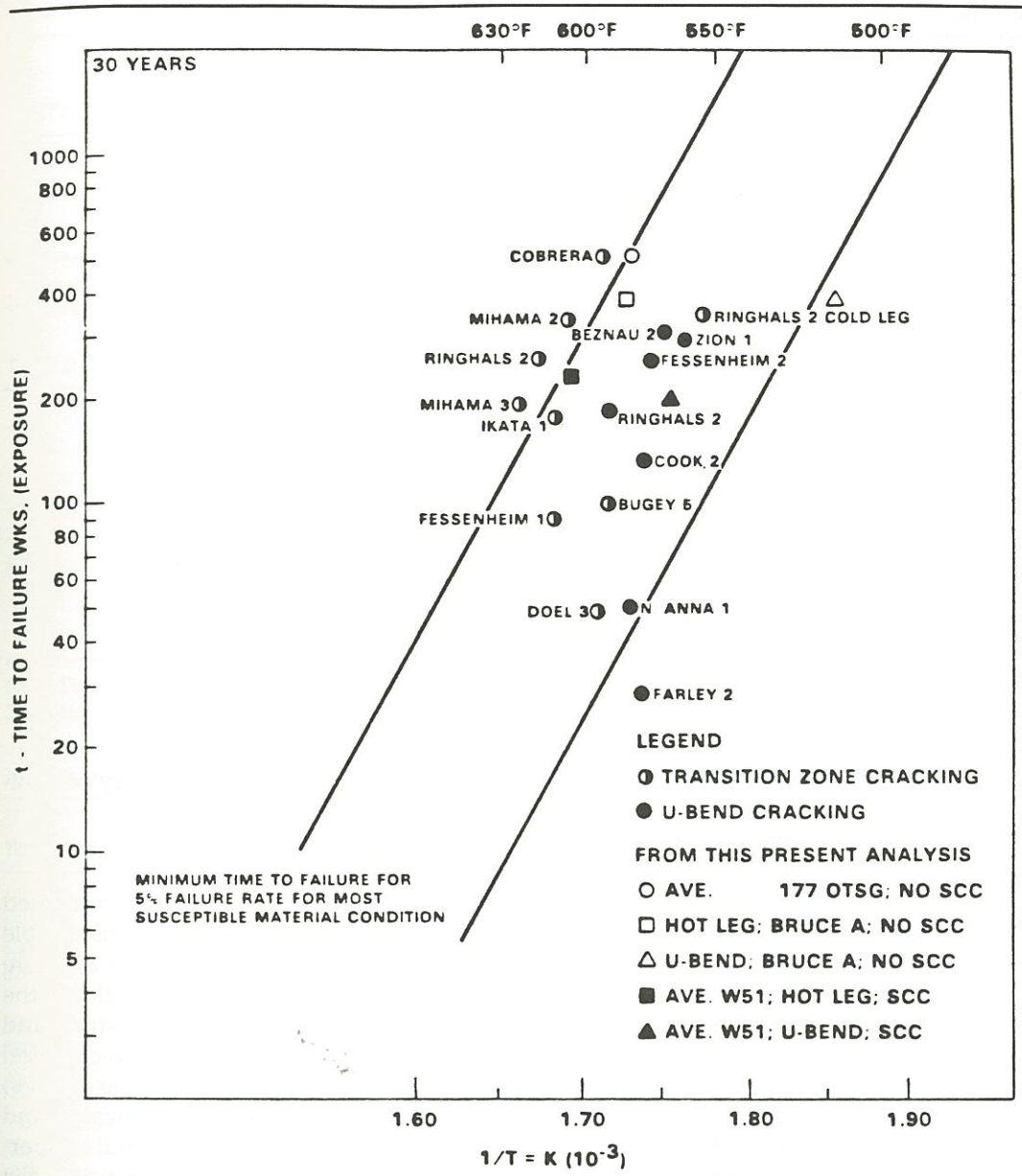

Figure 7: Steam generator operating experience. Note: No scc for B&W designs (from reference 4).

comparably heat-treated test samples lasted more than approximately five years, and some for as long as about 11 years, or until the program was terminated, without incidence of scc. These laboratory tests are being confirmed even today with operating plants that use stress-relieved Alloy 600 tubing, in that there are no reported high-purity water scc problems.

One conclusion reached in the cited activation energy analysis [4] was that thermal treatments for 593°C (1,100°F) to 621°C (1,150°F), for as short a time as four hours to more typically 18 hours, or 704°C (1,300°F) for 15 to 16 hours, greatly improved potential life, whereas other treatments (i.e., low-temperature mill-annealed or cold-worked) reduced life expectancy. However, another study showed that 10 hours of heat treatment at 621°C (1,150°F) will insure the formation of microstructures extremely resistant to caustic and high-purity water scc [10]. Nevertheless, the former analysis [4] showed that a 12-fold improvement in life can be obtained for thermally treated material versus annealed, even when heat treated for only 4 hours (Figure 6).

Another conclusion which can be drawn from the data is that there exists extensive scatter at any one specific temperature, which strongly suggests effects of other uncontrolled but important parameters. Higher temperature tests for thermally activated phenomena are normally prescribed to accelerate the processes, to allow for practical laboratory investigation. However, an additional effect often desired when testing at higher temperatures is a reduction in the scatter of the data. That this effect of scatter is probably occurring in the scc of steam generator tubing is apparent from the data presented in Figure 7. The scatter in times-to-fail

Table 3: Activation Energy Predictions of Time to Fail Compared to Actual Experience

Location	Temperature °C	Temperature (°F)	Predicted minimum time to fail using activation energy approach (ΔE = 40 KCAL/mole-°K) (weeks)* 50% failure rate	5% failure rate	Observed Minimum time to fail (weeks)
Hot leg transition zones					
B&W RSGS (Bruce A)	304	(597)	155	42	No failure (364)
B&W OTSGS	318	(604)	75	21	No failure (501)
Cobrera	313	(595)	95	29	500
Mihama 2	318	(605)	75	21	320
U-bends					
B&W RSGS (Bruce A)	272	(522)	1100	375	No failure (364)
Ringhals 2	294	(562)	270	90	190
Cold leg transition zones					
B&W RSGS (Bruce A)	263	(506)	2000	750	No failure (364)
B&W OTSGS	314	(598)	90	27	No failure (501)
Ringhals 2	293	(560)	270	90	350

*Note: These predictions are based upon the data plotted in Figure 6 and on the assumption that SCC is strictly a function of temperature and independent of material condition, water chemistry, etc. It should be emphasized that there are no known high-purity water SCC failures in service in stress-relieved 600 tubing.

for operational steam generators is appreciably greater than the scatter in laboratory data, which were generated at higher temperatures and plotted in Figure 6. It is recognized that an appreciable amount of this increased scatter may be attributed to the greater number of variables present in service vs laboratory conditions, or perhaps to the thoroughness of the eddy current inspections. However, it was suggested that increased scatter in failure times will occur with decreased temperature [1].

The implications of this scatter upon the present issue are noteworthy. The fact that some SGS are SCC-free while operating within the temperature ranges characterized by a high incidence of SCC and a large scatter in failure times for other designs, is extremely significant. If it is assumed there will also be a large scatter band for failure for the SG's designs that are presently performing without SCC, it may be concluded that the lower end of the scatter band for failure has not been reached as yet. Therefore, it is likely that the vast majority of these SGS will continue to perform without the development of SCC.

It was previously pointed out that activation energy analyses are being used to predict initial design and remaining SG lives. Applying this approach to service data further demonstrates that temperature is only one of several factors affecting SCC. Table 3 contains temperature profile data and predicted times-to-fail for different SG designs, compared with actual field experience. For these examples, there is an obvious discrepancy between predicted values and actual experience. The reasons for this discrepancy is that the method assumes temperature is the primar ling factor for SCC.

Empirical Correlation Life Predictions

An empirical life prediction technique was fo by performing tensile tests on highly su material (i.e., low-temperature anneal and pickle) in AVT water at 343°C (650°F) to eva effect of temperature [1, 17]. The effects of s heat treatment were determined by electro caustic tests at 288°C (550°F). As a result, the failure can be predicted for both mill-anne heat-treated material as functions of stress an ature. This correlation method was used t successfully service failure times for mini-s Doel 3.

Table 4 presents predicted lives and mu factors for mill-annealed and heat-treated ring tensile specimens in 343°C (650°F) AVT v function of load (in terms of per cent room tem yield strength). As Table 4 shows, the p technique is highly accurate at the one con which failure occurred (i.e., as received, 1 Due to the absence of failures of non-deform mens stressed below the yield in high-pur tests, the accuracy of the correlation metho been assessed at these lower stresses. Althou purity water tests were performed for exp long as 93,750 hours (10.7 years), the prog unfortunately terminated before predicted failure were achieved. However, it is believe estimations are conservative.

Table 4: Predicted Failure Times in High-Purity Water at 343°C (650°F) and Multiplying Factors from References 1 and 17

	As-received		*Heat treated*		*Heat treatment multiplying factor***	*Stress multiplying factor***	
% RTYS	*Hours*	*Years*	*Hours*	*Years*		*As-received*	*Heat-treated*
125	35,000*	4.0	136,000	15.5	3.9	1.0	1.0
90	102,500	11.6	451,000	51.5	4.4	2.9	3.3
70	185,000	21.1	1,171,000	133.7	6.3	5.3	8.6
60	245,000	28.0	–	–	–	7.0	–

*Two of three specimens have failed at 36,300 and 39,900 hours. The rest have not failed after 93,750 hours of testing.
**Derived from electrochemical caustic tests.

Table 5: Predicted Lives (Years) and Stress Multiplying Factors for Mill-Annealed and Heat-Treated Alloy 600

	Stress multiplying factors		*Temperature*					
			650°FF/343°C		*630°F/332°C*		*550°F/288°C*	
% RTYS	*As-recd*	*Heat treat*	*As-recd*	*Heat treat*	*As-recd*	*Heat treat*	*As-recd*	*Heat treat*
125	1.0	1.0	4.0	15.5	8.0	31	32	124
90	2.9	3.3	11.6	51.5	23.2	103	93	–
70	5.3	8.6	21.1	133.7	42.2	266	169	–
60	7.0	–	28.0	–	56.0	–	224	–

Note: Times to fail at 332°C (630°F) should be 2 times those at 343°C (650°F). Times to fail at 288°C (550°F) should be 8 times those at 343°C (650°F). From AVT water tests on highly susceptible material, i.e., low-temperature anneal with a severe pickle [1, 17].

Table 6: Temperature Multiplying Factors[a]

Temperature, °F/°C	*Empirical correlation method [1, 17]**	*Activation energy method [4]***
680/360	0.27	0.4
650/343	1.0	1.0
630/332	3.6	1.8
620/327	4.8[b]	3.0
610/321	5.9[b]	3.8
600/315.6	7.0[b]	5.0
580/304	9.5[b]	10.0
550/288	13.0	24.0

[a]Normalized relative to 650°F.
[b]Estimated.
*Factor = (avg. time to fail at T_1)/(avg. time to fail at 650°F).
**Reference 4 assumed thermal treated and annealed material had same activation energy (40 Kcal/mole-°K) for scc.

This predictive technique incudes the effect of stress, microstructure (i.e., heat treatment), as well as temperature. Table 5 presents the predicted lives and multiplying factors of mill-annealed and heat-treated Alloy 600 ring tensile specimens as functions of load and temperature. Similarly, Table 6 compares the temperature multiplying factors from the empirical correlation method [1, 17] to those from the activation energy method [4]. It should be noted that the correlation method temperature multiplying factors were derived from failure data of highly susceptible material (i.e., low-temperature (927°C/1,700°F) anneal with a severe pickle). However, multiplying factors from the activation energy approach [4] on mill-annealed material are not appreciably different, as is shown in Table 6. Both methods assume that the temperature multiplying factors are the same for heat-treated and mill-annealed material. In fact, the largest variation in temperature multiplying factors was at 288°C (550°F), where the value from the correlation method was 55% of the value from the activation energy method. It is emphasized that the stress multiplying factors for thermally treated and stress-relieved material are estimations, and are not based upon actual failures. Only a small per cent of test samples failed in high-purity water, and only when loads were appreciably higher than the yield strength.

Table 7 contains a comparison between temperature multiplying factors and those for the combined effects of stress and heat treatment (i.e., microstructure). Whereas only a 2 (9.5/4.8) [1, 17] to 3.3 (10/3) [4] increase in steam generator tube life is predicted if the operating temperature is decreased from 327°C (620°F) to 304°C (580°F), an improvement of at least 33 times is predicted if stress-relieved or thermally treated tubing is stressed (total stress = applied plus residual) to no more than 70% of the yield strength. Therefore, the combined effects of stress and heat treatment are 10 to 16.5 times that of temperature. Even individually the effects of stress and microstructure are larger than the effect of temperature. The effect of microstructure (i.e., heat treatment excluding stress relaxation) ranges from 3.9 to 6.3 for stresses at 125% to 70% of yield. The effect of stress for mill-annealed material ranges from 2.9 at 90% yield through 5.3 at 70% yield to 7.0 at 60% yield. These individual factors are equivalent or slightly higher than the factors for temperature (i.e., 2.0 to 3.3).

Table 7: Comparison of Temperature Multiplying Factor vs Stress, Heat Treatment (Microstructure) and Combined Stress Plus Microstructure

% YS	*B&W temperature factor* [1, 17]*	*Stein temperature factor* [4]*	*Stress multiplying factor mill-annealed***	*Heat treatment multiplying factor*	*Combined stress plus heat treatment multiplying factor*
125	2 (9.5/4.8)	3.3 (10/3)	1.0	3.9	3.9
90	–	–	2.9	4.4	12.8
70	–	–	5.3	6.3	33.4
60	–	–	7.0	–	–

*For temperature reduction from 327°C (620°F) to 304°C (580°F).
**343°C (650°F).

Conclusions

It may be concluded that the use of alternative alloys (e.g., titanium-stabilized Alloy 800 or Alloy 690), thermally treated or stress-relieved Alloy 600 tubing, with low stress and low cold work designs, stringent water chemistry control, and high fluid flow, in concert, far outweighs the influence of temperature on SCC. The combined effects of these other more important factors, and not lower operating temperatures, are the primary reasons for the improved resistance of some SGs to SCC. Controlling these design variables is a preferred approach for avoiding SCC to reducing operating temperatures, in that it allows for maximizing operating temperatures (which is desirable for reasons of efficiency) without the incidence of SCC.

Acknowledgement

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Notes

1. Canadian Deuterium Uranium Plants.
2. Definition: Circulation ratio – ratio of total mass flow in riser/steam flow (as opposed to recirculation ratio, which is the ratio of water in the downcomer/steam flow).

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Suitability Study of On-Line Leak Tests for CANDU Single-Unit Containment Buildings

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Abstract

CANDU reactor containment buildings are checked for integrity every five years. An on-line test could be performed more often, thereby reducing long-term unavailability of the containment system. A test procedure which accounts for operational factors is proposed. A theoretical error analysis is performed to study the effect of test parameters on the accuracy and sensitivity of the test. It is found that leak rates greater than 5% per day can be detected in less than one day. The target 0.5% per day requires tests longer than five days and the leak detection is complicated by operational factors. A test of two to three days will allow the detection of a 1% leak. It is also shown that a system modification which draws instrument air from inside the reactor building would allow the detection of a 0.5% leak rate in less than 12 hours.

Résumé

L'étanchéite de l'édifice de confinement des centrales de type CANDU est testée à tous les cinq ans. Un test pouvant être conduit durand l'opération de la centrale permettrait une vérification plus fréquente de l'étanchéité et reduirait le risque de défaillance du système de confinement. On propose une procédure d'un test qui tiend compte des facteurs opérationels. On présente une analyse théorique de l'erreur afin d'étudier l'effet des paramètres de contrôle sur la précision et la sensibilité du dit test. On démontre que des taux de fuite supérieurs à 5% par jour peuvent être détectés en moins d'un jour. L'objectif de 0.5% par jour ne peut être mesuré en moins de cinq jours et le calcul est compliqué par la presence des facteurs dûs à l'opération du réacteur. Un test d'une duré de deux à trois jours permettrait la détection d'un taux de 1% par jour. On démontre également qu'il est possible de détecter un taux de fuite de l'ordre de 0.5% par jour en moins d'un jour si le système d'air d'instrumentation est modifié de façon à tirer sa source de l'intérieur du bâtiment.

Introduction

The nuclear industry recognises safety as a top priority. One of the safety features of a CANDU reactor is the containment building in which it is housed.

These concrete containment buildings are checked for integrity at the commissioning of the plant and subsequently every five years. During these tests, the reactor is shutdown and the building pressurized to design pressure. A single-unit building must hold this pressure with an allowable leakage of 0.5% of the building volume per day.

Since these tests are only done every five years, it is possible that a leak could develop and go undetected until the next test. A leak test that could be performed during plant operation would allow more frequent tests. Frequent tests would ensure the continued reliability of the containment system.

The objective of this work is to investigate the suitability and feasibility of such tests for a CANDU single-unit containment building. A review of past efforts shows the importance of operational factors. This report presents a proposed test procedure which considers these factors, and gives a theoretical study of test performance. The parameters affecting leak tests are given and data is provided to allow the choice of a set of test conditions based on the objectives of the utility.

Theory of Leakage

Leak tests done below design pressure will show a lower leak rate. This result may, however, be used to determine the leakage rate at design pressure. The relationship between leak rate and pressure is needed. This relationship depends on whether the flow is laminar or turbulent. The flow regime is dependent on the leak path geometry, which is never known *a priori*.

Leak rate is also affected by air ingestion [Toossi, 1981], temperature, and other factors. For these reasons a relationship between the leak rate at low pressure and that at design pressure must be customized to a particular containment building. The rela-

Keywords: containment, leakage, tests.

Table 1: Summary of Previous Leak Rate Test Results

Station	*Type of containment*	*Test pressure (kPag)*	*Model used*	*% Lr and error (95% conf)*
Pickering[a]	multi-unit	+41.4	$L \alpha \Delta P$	0.18–0.54% / hr
Gentilly-1[a]	single	+117	$L \alpha \Delta P^{1/2}$	0.28 ± 16% / day
		+41	$L \alpha \Delta P^{1/2}$	0.25 ± 22% / day
USA's[b]		345–415	$L \alpha \Delta P^{1/2}$	0.02–0.16% / day
Pickering A[c]	multi-unit	+41	$L \alpha P_u^2 - P_d^2$	0.44% / hr
Bruce A[c]	multi-unit	+69	$L \alpha P_u^2 - P_d^2$	0.25% / hr
Pickering A[d]	multi-unit	+13.8	n.a.	1.2 ± 11.5% / hr
Bruce B[e]	multi-unit	+69	$L \alpha P_u^2 - P_d^2$	0.20 ± 10% / hr
Lepreau[f]	single	+124	n.a.	0.225 ± 0.8%/day

[a]Smith, 1987; [b]Brown, 1975; [c]Zakaib, 1982; [d]Zakaib, 1984; [e]Zakaib, 1985; [f]Harvey, 1982.

tionship can only be found if a low-pressure test immediately precedes or follows one at design pressure.

Review of Test Methods

Many methods have been proposed for testing containment integrity. Some tests have been performed and some are in the development stage. They may be grouped into the following categories:

1. External test methods such as external tracer detection, which can only give an indication that a leak is present [Spletzer, 1986].
2. Design-pressure leak rate tests [Zakaib, 1982; Smith, 1975; Whyle, 1984; Zakaib, 1985].
3. Positive lower-pressure leak rate tests, performed at different pressures and for different periods of time [Spletzer, 1986; Zakaib, 1984].
4. Tests at vacuum [Zakaib, 1984].
5. Continuous on-line monitoring, which is performed during normal operation [Zakaib, 1984].
6. Tracer gas monitoring and a mass balance over the reactor building [Spletzer, 1986; Zakaib, 1984; Boyd, 1986].

All of the above methods except 1 calculate leakage based on a mass balance using the ideal gas law. Leak rates are calculated using a linear regression analysis of the data [Zakaib, 1982; Smith, 1975; Zakaib, 1985; Zakaib, 1984; Brown, 1975]. Some results of these tests, which have been performed at various conditions, are summarized in Table 1. As can be seen from this table, none of the stations have attempted leak rate tests to detect small leaks (0.5% per day) at low pressure (less than 20 kPag).

Suitability of Previous On-Line Test Methods

For obvious reasons, full-pressure tests cannot be performed on line. External detection techniques are not suitable, since they are not quantitative.

Tests at reduced negative or positive pressures, with or without a tracer gas, can give acceptable results in a relatively short time for large le
For leaks on the order of 0.5% of the contain
day, much longer tests are required, and op
factors may significantly affect the calculated
Therefore a valid test for single-unit con
systems must take into account the error
system operation, as well as aim at redu
instrument error. On-line monitoring will be c
later in this paper.

Leakage testing for a single-unit CANDU bu
then narrowed to a mass balance-based te
with or without a tracer gas.

Operational Considerations

Safety

For an operating plant, there are factors w
affect any mass balance-based leakage test.
most of these factors is how the test will a
safety systems inside the reactor building.

A leak test will most likely involve either a
or negative change in the pressure inside th
building. If the building pressure rises above
then the high reactor building pressure trip p
be reached. It is important that the reactor t
the pressure inside the bulding rises by 3.45

It follows that if the building pressure is ch
ΔP, then the reactor trip point must be change
To ensure that this safety feature is at
bypassed, it will be necessary to change the
pressure slightly by dP, and then change the
inside the reactor building by dP, thus brin
building and trip point to the desired pre
steps and together. The implications of this c
the trip point on the safety analysis would h
studied.

This procedure will take time. As an ex
should take 16 hours to increase the pressu
building and the trip point to 10 kPa, and ano
reduce the pressure to normal operation at th

the test (NBEPC staff estimate). The same procedure must be followed at the end of the test so that at no time is the difference between the containment building pressure and the trip point set value greater than 3.45 kPa.

The dousing system is activated at 14 kPa, and it would be best to keep the test below this pressure. This should not be a difficult constraint to meet. It should be noted that any change in pressure in the building may change the relief pressure of the moderator relief devices and the pressure of the moderator system. Small changes in building pressure will not significantly affect moderator operation.

Air Flows

Air flow is difficult to measure accurately, particularly in large-diameter piping systems. It is therefore best to shut off all non-critical air flows into and out of the reactor building. These include the ventilation system inlet and outlet, emergency air, breathing air, and drier purges. The instrument air flow cannot be shut off, as it is required to operate. A short description of this system will be helpful.

Air is drawn from the turbine building, compressed, dried, and distributed through an instrument air header. One arm of this header penetrates the containment boundary to supply all of the instrument air requirements for the reactor building. This pipe discharges into three holding tanks from which air is drawn as required.

The instrument air flow can be measured easily with the installation of a vortex meter in the feed line. The pressure in this line is known to vary, but it must be maintained constant for a leak test. This can be achieved with the installation of a pressure regulator with the vortex meter.

A small instrument air system inside the reactor building would also solve the problem of measuring this air flow. A compressor and a drier would be required. No measurement of the air flow would be needed since no air would be added to the building.

D_2O Recovery System

If all of the air flow out of the building is stopped, the relative humidity inside the building will rise due to leaks, water vapour in the air inlet, and various other sources. Some of this water may be D_2O. There are now eight fans which draw air from inside the building through driers and discharge the dry air back into the building. The continuous purge in this system draws air from the discharge of the other driers through a ninth drier, DR 5A/B, and exhausts to the atmosphere. This is the D_2O recovery system.

This is a closed loop system with the exception of the purge. Since no air is added, it will not affect the calculations if it is left running while a test is being done. This will help to keep the tritium levels low inside the reactor building. One possible concern is that in a vacuum test the fans may be starved. Account can be made of the change in moisture content between the air inlet and discharge by close monitoring of the relative humidity inside the building.

If a tracer gas is used, then its absorption properties in the drier bed must be considered. If it is absorbed, then the driers would have to be stopped for the duration of the test, or tracer concentration measured at the inlet and discharge of the driers. A further concern would be the effect of absorbed tracer gas on the operation of the driers.

Air Lock

Access to the reactor building may be required through one of two air locks. Entry is made through one door, and then that door is closed and the pressure equalized. Exit is through a second door. This process allows escape/ingress of air. The volume of the air lock is known and the pressure on each side of it is measured, which allows a calculation of the volume or weight of air change.

Fuelling Machine

If the test is to take more than a few days, the reactor may need to be refueled. During the refuelling procedure, there is a time when the refuelling machine is connected to the spent fuel discharge bay. At this time, there is a connection between the two buildings through a 0.635-cm orifice. Air will flow between the two buildings at a rate determined by the difference in pressure of the two rooms, and it will flow all the time that the machine is connected. This amounts to about 10 minutes for each refuelling.

Since the pressure in both the reactor building and the spent fuel discharge bay are measured, and the exact total time that the machine was connecting the two buildings can be recorded, then the amount of air added/lost can be calculated.

When the fuelling machine removes a fuel bundle from the reactor, about one litre of D_2O is spilled on the floor. This is one source of water ingress to the reactor building. Some of the water will be removed as liquid (a negligible volume relative to the volume of the reactor building) and some of it will vaporize. The vaporized D_2O is monitored by relative humidity measurements.

Other Considerations

Pressure gradients inside the building exist. These should be minimized, particularly for those rooms which have the containment boundary as one of their walls. Local pressure gradients will cause an error in leakage rate calculation. All doors that can be left open should be opened, to reduce these gradients and allow for mixing.

Temperature, too, will vary with location inside the

building. It is important, then, to measure temperature in as many areas as feasible. The thermocouples that are used for the full pressure leak test are sufficient for this measurement. Proper mixing is also required, so all of the fans inside the building should be on.

Changing the pressure inside the building may affect some of the equipment. The pressure and vacuum ratings for that equipment would have to be checked.

Proposed Test Procedure

The proposed test procedure is based on a mass balance done at a relatively constant pressure. The building is brought to a specified pressure, either positive or negative, and kept at that pressure by removing the same amount of air through the drier purge system as is added by the instrument air system. As an example, consider a test done at a positive pressure of 10 kPa. Tests at negative pressures and with a tracer gas are similar (see Figure 1).

Before the test begins, drier DR 5B should be shut off and the automatic swing mechanism on DR 5A/B stopped. This will allow air to purge only through DR 5A. Allow the packing inside the drier to become saturated such that the dew point of the air entering and leaving the drier is the same. This is to ensure that no water is either added or removed by this drier.

As many of the interior doors as possible should be opened and all of the fans turned on. This will allow for proper mixing and minimize local gradients.

The ventilation outlet and inlet valves should be closed. The ventilation system must be shut off as it is not feasible to measure flows in this system with any degree of accuracy. The breathing and emergency air systems should also be shut off at this time.

Driers DR 1–4, DR 7–8, and DR 9–10 all have purges

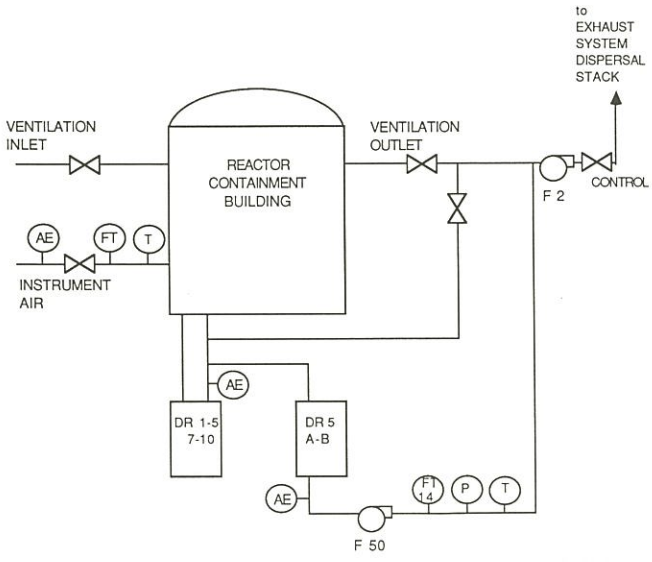

Figure 1 Leak test set up.

from the D_2O recovery system. These must l so that the only purge from these driers is th 5A.

The purge through drier DR 5A should b This now stops all flows out of the building will rise as the instrument air is still flowin building. A rise of about 1 kPa every thre expected [Ventzek, 1985] if the instrument the only air into the building and is operatir ly. It is possible to increase this rate by load air compressor or adding a portable air co Once the pressure has increased about reactor trip point will have to be raised by amount. The increase in pressure must be nied by increases in the trip point.

Once the building pressure is 10 kPa, and t trip point is set at 13.45 kPa, the discharge va F2 is opened. This valve is used to control th inside the reactor building at the 10 kPa pre only air flow into the building now is the i air, and the only out flow is through the dr Fan F2 is used to draw the vacuum for a vac

The test is now ready, and data acquisiti Entry into the building should be minimize entries logged. Fuelling should be delayed possible, and also logged.

A similar procedure using a tracer gas has ly been proposed [Boyd, 1986]. In this building is pressurized. The gas is then inje transient mass balance is done on the tra determine the leak rate.

Data Acquisition

Leak rate is calculated from a mass balance a containment building. The basic mass bala tion is:

$$Lr \times \Delta t = (m_{in} - m_{out})\,\Delta t + \Delta M + M_{fm} + M_{al} = y,$$

where Lr is the leakage rate in kg/unit time the mass flow of air in or out of the building i units. M is the mass in kg of air add containment building by either the connect fuelling machine to the spent fuel disch (subscript fm), or due to entries through th (subscript al), both since the time t. ΔM is the the mass of the air inside the building and the mass of the air at time t minus the mass c the start of the test ($M-M_0$).

Measurements to be taken are: temperat sure, and relative humidity inside the reacto temperature, pressure, and flow rate of the i air inlet and the drier purge outlet; and the on either side of DR 5B. If the test uses a t instrument air flow will not be measured b tration of the gas must be recorded. Fuelli lock openings must be recorded so that corr air ingress/outflow may be made.

Temperature, pressure, and relative humidity inside the reactor building should be measured as in the commissioning test, and averaged to give a T, Pt, an RH for each time interval. This will allow the calculation of the mass of air inside the containment building at each time interval, using the ideal gas law.

The partial pressure of the water vapour inside the building is found from:

$$P_w = P_{wv} \times RH/100, \quad (2)$$

where P_w is the partial pressure of water in the air, P_{wv} is the vapour pressure of water at the average temperature in the containment building, and RH is the average relative humidity inside the containment building.

The partial pressure P of the air in the building is:

$$P = Pt - P_w. \quad (3)$$

The mass of air inside the containment building is then calculated from:

$$M = \frac{P \times V \times MW}{R \times T}, \quad (4)$$

where M is the mass of air in the containment building in kilograms, P is the partial pressure of the air in the building in kPa, R is the universal gas constant, T is the average temperature in the building in degrees Kelvin, and MW is the molecular weight of the air.

Pressure, temperature, flow rate, and relative humidity of the instrument air inlet and the purge outflow measurements allow for the calculation of the mass inflow and the mass outflow. These data are taken frequently and integrated over the chosen time interval to give a total change in mass/time interval. The mass flow of air into the building is calculated as:

$$m_{in} = Q_{in} \times \frac{P_{in} \times 298.15}{101.325 \times T_{in}} \times 1.185, \quad (5)$$

where Q_{in} is the air flow rate in cubic metres per unit time and P_{in} and T_{in} are the absolute air pressure and temperature of the instrument air. 1.185 is the standard density of air in kilograms per cubic metre.

The air flow out is calculated as:

$$m_{out} = Q_{out} \times \frac{298.15 \times P_{out}}{101.325 \times T_{out}} \times 1.185, \quad (6)$$

where P_{out} is the pressure of the air out through the drier purge. If the fuelling machine or airlock have been operated in the given time interval, the air lost or added due to this procedure can be calculated. For the air lock, the calculations are as follows:

$$M_{al} = \frac{V_{al} \times P \times MW}{R \times T}. \quad (7)$$

V_{al} is the volume of the air lock, MW is the molecular weight of the air, and P is the partial pressure of the air. The air partial pressure calculation depends on whether the building is being entered or exited. If the containment is being entered then the air has a partial pressure associated with the ambient conditions outside, as outside air is being added. If the internal door is opened first, then air is leaving the building and the partial pressure is that of the air inside the containment. Calculations for the fuelling machine are basic fluid dynamics, based on flow through an orifice.

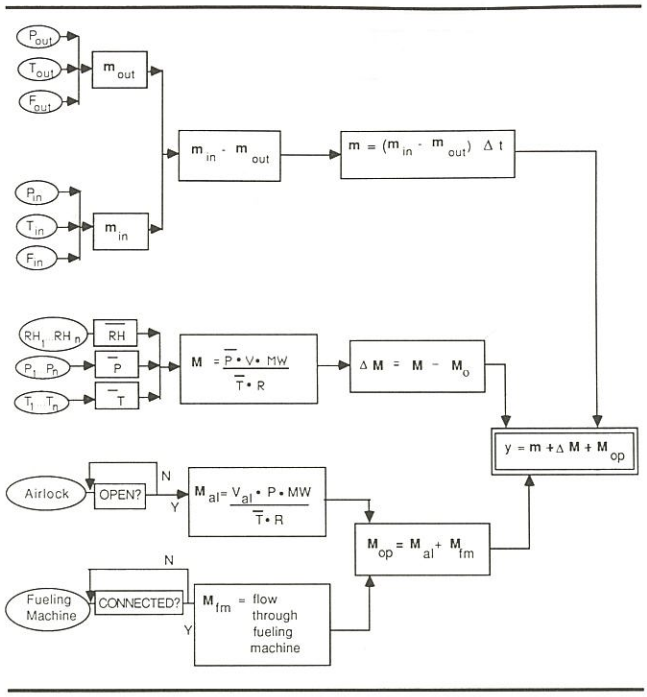

Figure 2 Flow chart of leak rate calculations.

The air added to the building, the air removed from the building, and the weight inside are calculated. By mass balance then, a leakage rate can be calculated. One leakage rate is calculated for each time interval, and the 'true' leakage rate is known from a linear regression done over the period of the test (see Figure 2).

Error Analysis and Results

A mass balance performed on the reactor building uses the measurement of flows and contained mass to determine a leak rate. Because this leak rate may be very small (0.5% of the contained air per day, at 124 kPag), its detection is limited primarily by the accuracy of the measurements. Therefore, the first step in determining the suitability of a particular test is to examine the error obtained under various test and operating conditions.

In this section, two tests are considered: the tracer gas and the absolute mass method. A theoretical error analysis is performed in an attempt to predict the effect of test pressure and duration on the sensitivity of the techniques.

Sources of Error

As mentioned earlier in this report, there are two main

Table 2: Test Instrumentation

Instrument name and brand	*Error*
Mensor quartz manometer	±0.010% R ±0.002% FS
Digital humidity analyzer	±0.3 °C
RTD thermometer	±0.025 °C
Gas chromatograph	±0.1 %
Vortex meter	±1.0% R
Flow meter (exhaust)	±3.0% R

types of errors which affect measurements of containment leak rates: operational and instrument errors. Operational factors that are important to consider are those which lead to mass transfer in or out of the building in a way which is not accounted for by flow measurement. Important factors include the air locks, the fuelling machine and the instrument air storage tanks. A secondary but significant factor is the fluctuation of atmospheric conditions. This question will be considered later.

In the test procedure proposed, an operational transfer term can be calculated, then applied to the mass balance equation. The quantity calculated enters the mass balance as a correction term which must be added, unless there is no entry of the reactor building and no refuelling is done. Because it is a correction factor, errors on the operational terms are of second order and they are difficult to evaluate.

The other source of error, the accuracy of the instruments, can be estimated from the known calibrated error of the component. In our theoretical analysis, only this source will be considered. During the actual test, errors can be continuously monitored by calculating the standard deviation of the accumulated data. Similarly, the test reproducibility cannot be accurately estimated except by repeating the actual test.

The instruments used and their associated specified accuracy are reported in Table 2.

Mathematical Expressions

For both methods of leak rate measurement, the expression relating the physical measurement to the leak rate stems from the basic mass balance equation:

$$m_{in} - m_{out} + \frac{dM_{acc}}{dt} = Lr, \tag{8}$$

which, integrated over time, becomes

$$y = (m_{in} - m_{out})t + M_{acc} = Lr \times t, \tag{9}$$

where M_{acc} is an accumulation term which accounts for the operational factors. M_{acc} is the sum of the operational terms M_{fm} and M_{al}, and the term ΔM in equation 1.

A regression analysis can be performed using equation 9, with y plotted against time. This method appears to be the most accurate way to an data [Koegh, 1985]. Lr, the leak rate, bec slope of the y versus t plot, which can be by the least square method:

$$Lr = \frac{n\Sigma(y \times t) - \Sigma y \times \Sigma t}{n\Sigma(t^2) - (\Sigma t)^2},$$

where n is the number of data taken. The the slope is then given by:

$$\sigma = (\alpha)\, S_{yt} \left[\frac{1}{\Sigma(t - t_{ave})^2}\right]^{1/2},$$

where S_{yt} is the standard error of estimate 1973], and α is a constant depending on the c interval desired; $\alpha = 1$ or ≈ 2 for 68% and fidence intervals, respectively.

In fact, σ is data-dependent, and it can be r calculated only from the actual experiment. I using the known accuracy of the instrument possible to estimate the overall experimenta 1977 treats this question in more detail. For pose, the instrumentation specifications wil to predict the actual experimental error.

In this case, the standard error of estima evaluated by the expression:

$$S_{yt}^2 = \sigma_y^2 = \sigma_{in}^2\,(t)^2 + \sigma_{out}^2\,(t)^2 + \sigma_{Macc}^2$$

where σ is the standard deviation of the meas M_{acc} contains a change in the contained m where the mass is calculated using the ideal

$$M_{acc} = a \times P / T + M_{op},$$

where a is a constant. Therefore,

$$\sigma_{Macc}^2 = 2 \times a^2 \times [(\sigma_P / P)^2 + (\sigma_T / T)^2].$$

Equation 12 contains a time dependent ter can be averaged over the time of the test, to

$$(\sigma_{in}^2 + \sigma_{out}^2) \times \Sigma t_i^2 / n,$$

where t_i is the time at which sample i is take

The theoretical standard error of estimate c be written as

$$S_{yt}^2 = (\sigma_{in}^2 + \sigma_{out}^2)\Sigma t_i^2 / n + 2M_{acc}^2[(\sigma_P / P^2) + (\sigma_T / T$$

The expression is similar for the tracer gas m long as the concentration is maintained appr constant. In that case, flows are those of t and the pressure and temperature terms of eq are replaced by σ_c^2, where c is the concen tracer gas in the reactor building.

Assumptions

Since the results of the test at reduced pressu be known in advance, it is necessary to acc assumptions in order to predict the error.

As discussed previously, the error on t tional correction terms is unpredictable. T

for the theoretical analysis, instruments are considered to be the only source of error.

Furthermore, in order to calculate the error, one must know the leak rate at any given pressure. For that purpose, and in light of the many correlations mentioned in a previous section, it is assumed that leak rates vary linearly with pressure.

Similarly, leak rates at negative and positive pressures are assumed to be identical. This is not entirely true; however, it is considered to be satisfactory for the purpose of error prediction.

Range of Test Parameters

Three main parameters are considered to affect the theoretical error prediction: the test pressure, duration, and the actual leak rate being measured. Our study includes test pressures of up to ±20 kPag, and durations of up to five days. Target leak rates of 0.5% to 24% per contained volume per day are used; they range from single-unit to multi-unit containment design leak rates.

The instrument air-flow rate used in this study is 150 m^3 per hour, and is matched by the exhaust flow rate. This represents an average of the values reported in Boyd, 1986.

Results

Results of the theoretical error analysis are reported in Figures 3 to 8. In Figures 3 to 7, the error is shown as a function of test duration for various test pressures. It should be noted that, for the absolute mass method, pressures indicated can be taken as negative as well as positive. In Figure 8, the sensitivity of the absolute mass method is shown as a function of pressure, for various durations. The minimum detectable leak rate is taken to be the one for which a 50% error at 95% confidence is obtained. All errors shown were calculated assuming that one data point was taken every 15 minutes.

As it can be seen in Figures 3 to 7, the error of a given test decreases with time and pressure. For a given pressure, the minimum error is reached at very long test durations. This error constitutes a limit for a test carried out at that pressure. This is mostly true for low pressures (≤3 kPag) and small leak rates. In all cases, the major contributor of inaccuracy is the flow measurements. They account for up to 95% of the total error, and they are most important for small leaks. For this reason, continuous on-line monitoring at normal operating conditions is impractical. Because of the very small pressure differential across the containment

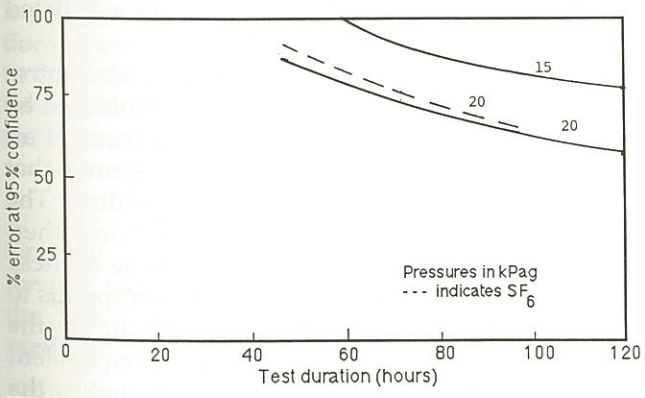

Figure 3 Test error for a target leak rate of 0.5% per day.

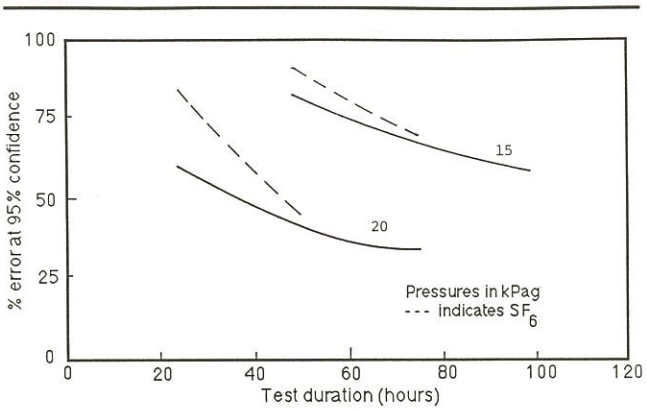

Figure 4 Test error for a target leak rate of 1% per day.

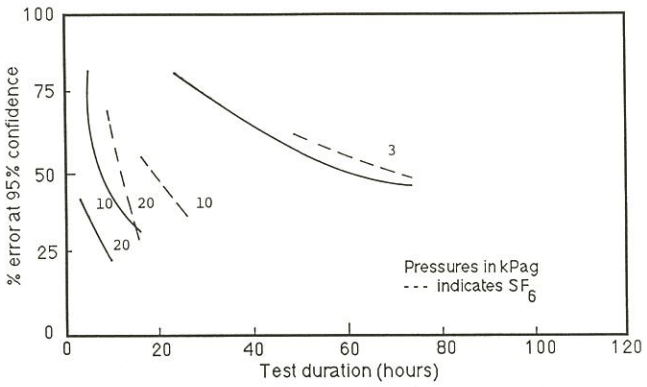

Figure 5 Test error for a target leak rate of 5% per day.

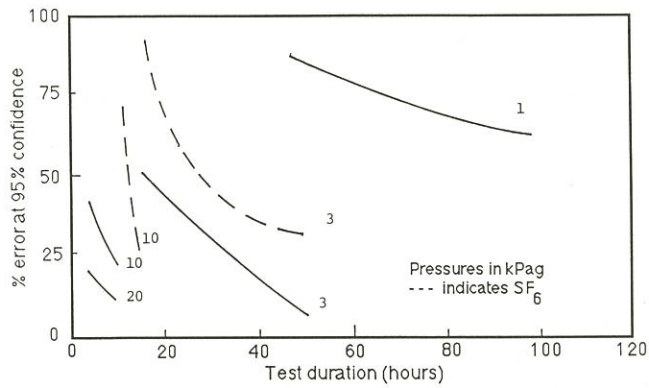

Figure 6 Test error for a target leak rate of 10% per day.

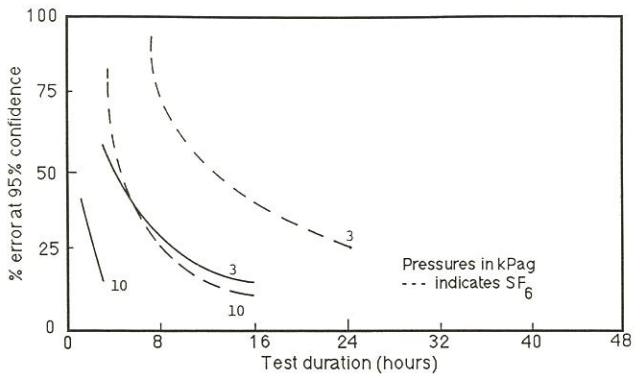

Figure 7 Test error for a target leak rate of 24% per day.

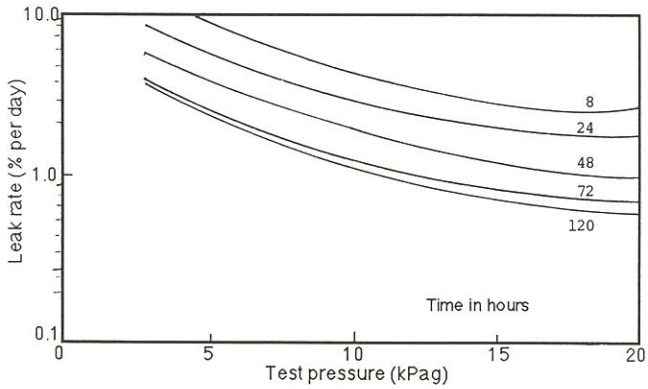

Figure 8 Minimum detectable leak rate at 50% error.

boundary, the actual leakage is too small to be detected at the desired accuracy.

The tracer gas method was originally proposed to eliminate the need for measurement of the instrument air flow rate, and the error associated with this measurement. Since there would be no tracer gas in this air, the flow rate would not be needed for a tracer gas mass balance. However, errors with the tracer gas method are generally worse than those with the absolute mass method.

Figure 8 yields additional information on the sensitivity of the test. Only the absolute mass method results are shown, since the tracer gas method would yield similar data. From this representation, it appears possible to classify the test parameters into three categories according to their ability to detect leak rates within a certain range. This result is summarized in Table 3. A leak rate of 0.5% per contained volume per day can be detected, but only after a long tes days. However, slightly higher leak rate detected within three days by carrying ou high pressure (±15–20 kPag). A rough eva the containment integrity may be obtained ir a day, but leak rates of less than approximate day would not be adequately detected.

Table 3: Classification of Test Parameters

Sensitivity (at 50%)	*Test duration*
high (~0.5% per day)	>5 days
medium (~1.0% per day)	2–3 days
low (~5.0% per day)	<1 day

Discussion

Reliability of the Error Analysis

Operational Factors

The error analysis carried out in the last meant to give an indication of the real exp error which would be obtained should th done. However, certain assumptions had to in order to predict the error. Some operation which were not considered may affect the pr Moreover, the errors were calculated from t instrument accuracies and may not be a representative of the actual experimental deviation.

The main operational factors which were sidered in the calculations, but which direc the mass balance equation, are the air l fuelling machine and the instrument air accumulation in the storage tanks, inside the

If the equipment airlock is opened at least o shift (eight hours), the air transferred via t corresponds to approximately 0.23% of the air per day if the building is pressurized to

The mass transfer due to the operatic fuelling machine and its temporary connect spent fuel discharge bay is more difficult to with a 10 kPa pressure differential between th building and the discharge bay, the flow t 0.635-cm orifice would be equivalent to 0.0 contained air for every hour during which th machine is connected.

Obviously, a test which is done with no and with no access to the reactor buildi eliminate the above two operational errors. T be the best circumstance. However if this is n for a given utility, the corrections must be and incorporated in the mass balance.

The instrument air pressure variation a duces an error which can, however, be con installing a pressure regulator on the instr line, or by modifying the instrument air syst air is drawn directly from within the buil pressure fluctuations in the storage tanks wise 875 to 910 kPa. Given the total volum tanks to be 25.5 m^3, the possible error corre approximately 0.02% of the contained a containment building. Assuming that the e design leak rate at 10 kPag is linearly relat 0.5% of the contained air per day, at full pre

possible air mass accumulation provided by the storage tanks is at least ten times greater than the mass of air leaked. This would seriously affect the leakage calculation if it were neglected.

There are other factors which may affect the error analysis. They have been mentioned in the section on operational considerations. The D_2O recovery system may leak; however, assuming the leaks to be consistent from test to test, this factor may be considered as a systematic error. Perfect mixing of the air was assumed in the calculations, as temperature, humidity, and pressure data were assumed to be unique. The multiplication of instrumentation measurements to obtain one single datum (the contained mass) will also certainly affect the error, and so will the local averaging.

The hysteresis effect is another important aspect; absorption by the concrete may lead to false readings. Results obtained after raising the pressure higher than the pressure test prior to the test will be different than if the test had commenced as soon as the test pressure was reached [Toossi, 1981]. However, because of air ingestion into the concrete and subsequent release as the pressure is reduced, American National Standard recommends that the pressure be held at 85% (or lower) of the test pressure, in order to insure conservative estimates [Koegh, 1985]. Because of the lack of data in the latter case, there is no consensus about which procedure should be followed. The alternative would be to maintain the pressure prior to the test in order to saturate the concrete and open as many pathways as possible.

Finally, variations in weather conditions and daily cycles were not considered in the calculations, but may significantly affect the measurements. Typical daily pressure fluctuations of 1 to 2 kPa [Dick, 1984] would lead to an additional error in the determination of the leak rate of 5 to 33% for test pressures of 20 and 3 kPag, assuming that pressure inside the building is unaffected by atmospheric pressure. However, it can be shown that inside pressure follows to some extent outside variations [Dick, 1984], thereby reducing these errors. Therefore, for reasons of consistency, it may be important to repeat the test over similar conditions. It is possible to eliminate the error due to daily fluctuations by performing a test over several days. However, in that case, weather stability and seasonal changes become important, though less significant because of the increased total leakage.

Pressure Correlations

Since the test is to be performed at reduced pressure, the leak rate measured will not be directly related to the leak rate at full pressure. This can affect the predicted error in three ways:

1. The theoretical error depends on the leak rate, and the estimated leak rate used in the error analysis assumed a linear pressure relationship; should this relationship be different, the actual measured leak rate at low pressure for a given leak rate at full pressure will not be that used.
2. For a given measured leak rate at low pressure, the extrapolation to a higher pressure will introduce an error.
3. Leak rates at negative and positive pressures are not identical, but they were assumed to be the same in the analysis.

Since it appears that pressure correlations are system-dependent, the error due to pressure extrapolation may be reduced by establishing a 'custom-made' correlation for the particular reactor building under investigation.

Comparison with the Previous Test

In order to compare errors predicted to the actual experimental errors, the parameters from the commissioning test [Harvey, 1982] were used in the error model. The calculated relative error was found to be 2.6%. The relative error given in Harvey is 0.8%.

This discrepancy does not discredit the theoretical analysis, but introduces the consideration of reproducibility. In practice, the standard deviation of the data collected may be more related to the reproducibility of the instrument, rather than to its accuracy, which is a deviation from a calibrated measurement. As long as the instrument calibration remains stable, and if the reproducibility does not vary with the scale, this comparison with the actual data from the commissioning test indicates that the error analysis results are conservative. However, one must be careful in interpreting this result, since operational factors did not interfere with the commissioning test.

Test Selection

Although the results thus far are not completely sufficient to allow a final choice of the most suitable method, enough information is available to provide some clue as to which methods may or may not be favorable.

The Tracer Gas Method

Injection of a tracer gas such as SF6 in the reactor building, and monitoring of its concentration in various relevant locations, appears to be a very attractive solution to test containment integrity. Calculated errors are found to be acceptable, if slightly higher than those obtained using the other method. The fact that the contained mass of SF6 can be obtained from one single measurement, the concentration, certainly favors this method.

However, there are some serious unknowns which may have an important impact on the validity of this technique. It is not known what the effect of small quantities of SF6 would have on the equipment and the structures. It is also difficult to predict the possible

Table 4: Test Options

Test	*Parameters*	*Sensitivity*	*Comments*
Tracer gas			Single measurement Simple procedure Unknown behavior
Absolute mass	≥5 days 20 kPag	0.5% / day	Low instrument error Chance of large operational error High theoretical sensitivity
	2–3 days 20 kPag	1% / day	Low instrument error Low operational error Moderate sensitivity
	≤1 day 10–20 kPag	≥5% / day	Minimum operational error Low sensitivity
1/2 Design pressure test	5 hours 62 kPag	0.5% / day	Performed at shutdown Instrument air off
Modified instrument air intake	12 hours 20 kPag	0.5% / day	Instrument air intake within reactor building

absorption of the tracer gas by concrete, or its behaviour in the D_2O recovery system. Furthermore, the tracer gas method restricts the possible test pressures to positive differentials.

The Absolute Mass Method

As was observed in the results from the error analysis, the absolute mass method can be used with a wide range of test parameters. As determined, three options are available for this test (see Table 4). Each option imposes different constraints on the operating system, and each focuses on a particular range of detectable leak rates. However, the technical feasibility of the test is also affected by two factors: time and pressure.

The major error source (flow measurements) cannot be eliminated for an on-line test, unless the instrument air intake is temporarily or permanently installed within the building. If this is done, it is conceivable that the overall errors reported in the error analysis could be reduced by up to 95%. This would make a leak rate of 0.5% of the contained air per day detectable in 12 hours at 20 kPag (with a 50% error at 95% confidence). Otherwise, optimization of the test parameters is restricted to the other factors.

A test which could be completed in less than eight hours would eliminate the need to apply any correction to the mass balance equation, since the air locks and the fuelling machine can easily be kept inoperative over such a short period. This leads to a minimum operational error, but because of the instrument accuracy, this test would be limited to detecting leak rates in excess of 5% of the contained volume per day.

A longer test (over five days) would theoretically allow the detection of the target leak rate of 0.5% per day. However, the reactor may have to be refuelled, and operation of the fuelling machine introduces the most important operational error. A solution might be to cease refuelling operations over an extended length of time. In that case, cost penalties due to fuel management may have to be considere

Another important consideration with lo the buildup of tritium in the reactor buil though it is difficult to estimate this quai hoped that the continued operation of th covery system will help maintain it at an a concentration.

Since short tests appear to have a low s and since long ones (over a period in which is required) introduce large errors due to the operation, the compromise is a test of mode tion, during which there is no refuellir burn-up penalty). With such a test, it s possible to detect a leak rate as low as 1 contained volume per day. Although this is current design target, it could be consid reasonable compromise for an on-line test.

The other factor which may affect the feas given test is test pressure. Ideally, higher should be used, because they yield higher and potentially lower errors. However, the major problems with raising the pressure di the equipment and the trip set points. Similai pressures take longer to reach, and requi adjustment of the reactor trip set point + 14 kPag, the dousing system set point wou be adjusted. This increases the time necessa preparation of the test, and it introduces s cerns.

There is a third option, along with the trac absolute mass methods, which may have so tive features: an annual test, at shutdown, d pressure. Such a test would yield an a estimate of a leak rate of the order of the des in a matter of 5 hours (with a 50% erro confidence), and would eliminate most of the associated with operational errors. Because t

would then be shut down, the test pressure could be reached without any concern for the trip set points. The whole test, including preparation, could be done in less than half a day, provided that the instrumentation was prepared before shut-down. This avenue has not been seriously considered, for a cost comparison would be required to assess its merits. However, it remains a possible solution.

A summary of the test options is given in Table 4.

Reliability of the Test Results

One of the main questions when performing an on-line test at reduced pressure will be: how far can the results be trusted? In fact, this is a very serious concern, since a wrong decision could financially penalize the utility.

The answer depends on many factors. The first one is the error expected, or predicted, for a 95% confidence interval. This should determine the interval of acceptability of the measured leak rate. However, this is not sufficient. Adequate control of the operating and the test conditions, along with good instrument calibration and reproducibility from test to test, is absolutely crucial to the reliability of the result. The accuracy of the pressure and temperature correlation is also important in order to account for slightly different conditions at the time of the test. Finally, the ability of the data acquisition system to detect transients such as those resulting from a change in the system is instrumental in the adequate and reliable determination of leak rates.

Reliability concerns introduce the concept of a criterion for test failures. Obviously, a test should fail if the measured value is above the 95% confidence interval. However, in such a case, it may be wise to repeat the test and to verify that no sporadic event occurred which led to an unaccounted mass transfer.

Conclusions

Based on the requirements for an on-line measurement technique, a test procedure is proposed. It is based on a mass balance over the reactor building. Integrated air flows and mass changes are measured, while data acquisition keeps track of the air locks and fuelling machine operation, in order to apply a calculated correction term to the mass balance equation.

An error analysis is used to determine the suitability of the test procedures and parameters. It was shown that it is possible to reduce the instrument error by performing a long test, but that short ones eliminate the need to apply operational corrections. Moreover, the overall test accuracy is increased by increasing test pressure differentials.

Finally, it was shown that the target leak rate of 0.5% per day could be measured at 50% error with 95% confidence in 12 hours at 20 kPag if the instrument air intake can be installed inside the reactor building.

While the most attractive test conditions seem to be moderate pressures (10–20 kPag) and a moderately long test (3 days) with no refuelling, the test parameters for a utility are best determined by site management and the appropriate regulatory agency. They will depend on the exact objective of the test, which may vary between accurate determination of the leak rate to monitoring of status changes and detection of gross leaks.

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Trivac: A Modular Diffusion Code for Fuel Management and Design Applications

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Abstract

A new three-dimensional neutron diffusion code named TRIVAC was set up using advanced discretization algorithms and improved iteration strategies. The two variable order discretization algorithms used in TRIVAC will be presented. These are based, respectively, on the variational and nodal collocation techniques. These algorithms will be shown to produce reconstructible solutions which are upper and lower limits of the exact solution. The eigenvalue matrix system is solved using an ADI preconditioning of the power method in conjunction with a symmetric variational acceleration technique. Validation results are reported for the IAEA two- and three-dimensional benchmarks, and for a two-dimensional PWR.

Résumé

Un nouveau logiciel tridimensionnel de diffusion neutronique a été conçu en utilisant des algorithmes de discrétisation modernes et des stratégies itératives améliorées. Nous allons présenter les deux algorithmes de discrétisation d'ordre variable utilisées par TRIVAC. Celles-ci sont basées respectivement sur les techniques de collocation variationnelle et nodale. Nous allons montrer que ces algorithmes produisent des solutions interpolables et qui correspondent à des limites supérieure et inférieure de la solution exacte. Le système matriciel aux valeurs propres est résolu à l'aide d'un préconditionnement ADI de la méthode des puissances, en conjonction avec une technique d'accélération variationnelle symétrique. Des résultats de validation sont rapportés pour les cas tests IAEA à deux et trois dimensions ainsi que pour une représentation bidimensionnelle d'un PWR.

Introduction

The primary goal of a discretization algorithm is to transform the differential operators of the diffusion equation into real number matrices adapted to an efficient numerical solution. A review of most common algorithms is presented in reference 1. This list should be updated by appending the variational and nodal discretization techniques presented in this paper [2, 3].

There is no such thing as an 'ideal' discretization algorithm and any choice is the result of a compromise. Some techniques, such as the analytical nodal method (ANM), are very efficient to compute a power map defined over coarse elements, but they lack a straightforward reconstruction technique for the solution [4]. Other algorithms, like the classical finite element approximations, are handicapped by difficulties such as their incompatibility with an ADI preconditioning.

The variational and nodal collocation techniques used in TRIVAC appear to include most desirable properties in spite of the fact that they cannot compete with the efficiency of nodal schemes based on the quadratic leakage approximation. Let us mention the four most interesting properties of the collocation techniques:

1. The numerical solution is reconstructible over each element of the domain. The polynomial nature of the solution greatly simplifies the integration of TRIVAC in applications using the generalized perturbation theory or the quasistatic algorithm of space-time kinetics.
2. The discretization order is variable, being a function of the degree of the polynomials used to represent the neutron flux over each element. High order polynomials (cubic or quartic) are used to model PWR while linear polynomials are used for CANDU reactors. In fact, the linear variational and nodal collocation techniques are respectively equivalent to mesh-corner and mesh-centered finite difference approximations.
3. Collocation techniques are compatible with an ADI preconditioning of the power method [5].
4. Matrices produced as a result of the discretization are real and independent of the eigenvalue. Matrices corresponding to the leakage terms are symmetric, positive definite, and diagonally dominant. Other matrices are diagonal.

The two types of collocation techniques available in TRIVAC will now be presented.

Keywords: reactor physics, diffusion equation, collocation techniques, finite elements, nodal methods.

Definition of the Polynomial Basis

Both variational and nodal collocation techniques rely on a polynomial representation of trial functions. A weighted residual approach is used to find the approximate solution of the diffusion equation over the reactor domain. There is an important distinction in the way the weighted residual formalism is applied to each type of collocation technique, and this distinction affects the choice of the polynomial basis. The variational collocation technique is based on a finite element formalism where the residue, defined over the entire domain, is orthogonal to the entire set of trial functions. In addition, the interface conditions are treated as natural conditions. With the nodal collocation technique, however, the residues are cancelled element by element and the interface conditions are taken into account *a posteriori*. Consequently, the variational collocation technique is constrained to use a polynomial basis with continuous trial functions over the element boundaries. More flexibility is left to the nodal collocation basis, which permits piecewise continuous polynomials to be used.

We will now present the collocation techniques in the case of a one-speed formalism. Changing over to the multigroup formalism presents no additional difficulty. Moreover, we will limit this study to three dimensional Cartesian domains composed of an assembly of homogeneous parallelepipeds. Under these conditions, the neutron diffusion equation is written

$$-\frac{\partial}{\partial x}D_x(x,y,z)\frac{\partial\phi}{\partial x}-\frac{\partial}{\partial y}D_y(x,y,z)\frac{\partial\phi}{\partial y}-\frac{\partial}{\partial z}D_z(x,y,z)\frac{\partial\phi}{\partial z}+\Sigma_r(x,y,z)\,\phi(x,y,z)=S(x,y,z), \qquad (1)$$

with $\phi(x, y, z)$ continuous everywhere in the domain. Currents $-D_x(x, y, z)\,\partial\phi/\partial x$, $-D_y(x, y, z)\,\partial\phi/\partial y$ and $-D_z(x, y, z)\,\partial\phi/\partial z$ are almost continuous, but may present localized discontinuities over lines of singularity [1]. The boundary conditions are either zero flux ($\phi(x, y, z) = 0$) or positive albedo:

$$D_x(x,y,z)\frac{\partial\phi}{\partial x}\pm\frac{1}{2}\frac{1-\beta(x,y,z)}{1+\beta(x,y,z)}\phi(x,y,z)=0; \qquad (2a)$$

$$D_y(x,y,z)\frac{\partial\phi}{\partial y}\pm\frac{1}{2}\frac{1-\beta(x,y,z)}{1+\beta(x,y,z)}\phi(x,y,z)=0; \qquad (2b)$$

$$D_z(x,y,z)\frac{\partial\phi}{\partial z}\pm\frac{1}{2}\frac{1-\beta(x,y,z)}{1+\beta(x,y,z)}\phi(x,y,z)=0; \qquad (2c)$$

where the "−" or "+" sign is used, depending on whether the boundary is to the left of or to the right of the domain in relation to the direction of each axis.

We will now assume that the nuclear properties are uniform over each parallelepiped composing the domain. These parallelepipeds will then be used to support one or more elements. We will designate the value of each nuclear property over element e by D_{xe}, D_{ye}, D_{ze}, and Σ_{re}.

Before introducing the trial functions defined over

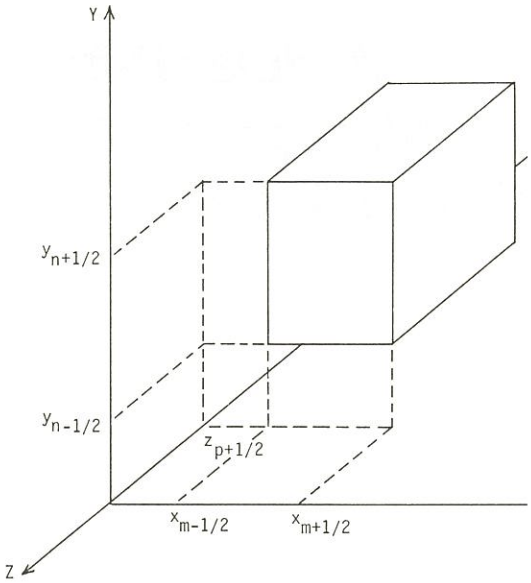

Figure 1: Discretization of a Cartesian domain.

element e, we will transform Cartesian co (x, y, z) of the element into local coordinate corresponding to a unitary cube of refere Cartesian coordinates are described in Figu

The following variable transformations wi

$$u=\frac{1}{\Delta x_e}\left[x-\tfrac{1}{2}(x_{m-\frac{1}{2}}+x_{m+\frac{1}{2}})\right]$$

$$v=\frac{1}{\Delta y_e}\left[y-\tfrac{1}{2}(y_{n-\frac{1}{2}}+y_{n+\frac{1}{2}})\right]$$

$$w=\frac{1}{\Delta z_e}\left[z-\tfrac{1}{2}(z_{p-\frac{1}{2}}+z_{p+\frac{1}{2}})\right],$$

with

$$\Delta x_e = x_{m+\frac{1}{2}}-x_{m-\frac{1}{2}}$$

$$\Delta y_e = y_{n+\frac{1}{2}}-y_{n-\frac{1}{2}}$$

$$\Delta z_e = z_{p+\frac{1}{2}}-z_{p-\frac{1}{2}}.$$

Over each element we will assume a tensor sion of solution $\phi(x, y, z)$ and of source term with the help of low order polynomials, denoted $\{P_k(u); k = 0, K\}$. In cases where poi belongs to element V_e, these two expansion form

$$\phi(x,y,z)=\phi_e(u,v,w)=\sum_{k_1=0}^{K}\sum_{k_2=0}^{K}\sum_{k_3=0}^{K}\Phi_e^{k_1,k_2,k_3}P_{k_1}(u)P_{k_2}(v)P_{k_3}(w)$$

and

$$S(x,y,z)=S_e(u,v,w)=\sum_{k_1=0}^{K}\sum_{k_2=0}^{K}\sum_{k_3=0}^{K}S_e^{k_1,k_2,k_3}P_{k_1}(u)P_{k_2}(v)P_{k_3}(w)$$

The Variational Collocation Method

The classical formalism of the finite element method involves the analytical integration of the terms composing the mass and stiffness matrices. The integrations are trivial insofar as the trial functions are simple polynomials. The mass matrix thus obtained is non-diagonal, which makes the classical approximations incompatible with the ADI preconditioning.

The variational collocation method corresponds to the approximations of the finite element method when the mass and stiffness matrices are numerically integrated in order to diagonalize the mass matrix [2]. This is made possible by expanding the solution in terms of Lagrangian polynomials, whose collocation points are identical to the base points of a Lobatto's quadrature formula. This type of quadrature makes possible the exact integration of the stiffness matrix, which is a necessary condition to ensure the convergence of the variational collocation method. Finally, we note that the linear variational collocation method is identical to the mesh corner finite difference method.

We have already tested the variational collocation method based on linear, quadratic, cubic, and quartic polynomials defined over two- and three-dimensional domains [5]. These approximations use the following trial polynomials:

1. Linear Lagrangian polynomials (K = 1)
   In this case, the collocation points are chosen at abcissa $u_0 = -1/2$ and $u_1 = 1/2$ in local coordinates. The trial polynomials are therefore given as:

$$P_0(u) = -u + \tfrac{1}{2} \tag{6a}$$

   and

$$P_1(u) = u + \tfrac{1}{2}. \tag{6b}$$

2. Quadratic Lagrangian polynomials (K = 2)
   The collocation points are now chosen at abscissa $u_0 = -1/2$, $u_1 = 0$ and $u_2 = 1/2$. The corresponding trial polynomials are

$$P_0(u) = 2u^2 - u, \tag{7a}$$

$$P_1(u) = -4u^2 + 1, \tag{7b}$$

   and

$$P_2(u) = 2u^2 + u. \tag{7c}$$

3. Cubic Lagrangian polynomials (K = 3)
   The collocation points are now chosen at abscissa $u_0 = -1/2$, $u_1 = -1/(2\sqrt{5})$, $u_2 = 1/(2\sqrt{5})$ and $u_3 = 1/2$. Note that the position of points u_1 and u_2 is imposed by the requirement that the set $(u_k;\ k = 0, K)$ correspond to the base points of a Lobatto's quadrature formula. The corresponding trial polynomials are

$$P_0(u) = -5u^3 + \tfrac{5}{2}u^2 + \tfrac{1}{4}u - \tfrac{1}{8}, \tag{8a}$$

$$P_1(u) = 5\sqrt{5}\,u^3 - \tfrac{5}{2}u^2 - \frac{5\sqrt{5}}{4}u + \tfrac{5}{8}, \tag{8b}$$

$$P_2(u) = -5\sqrt{5}\,u^3 - \tfrac{5}{2}u^2 + \frac{5\sqrt{5}}{4}u + \tfrac{5}{8} \tag{8c}$$

   and

$$P_3(u) = 5u^3 + \tfrac{5}{2}u^2 - \tfrac{1}{4}u - \tfrac{1}{8}. \tag{8d}$$

4. Quartic Lagrangian polynomials (K = 4)
   The collocation points are now chosen at abscissa $u_0 = -1/2$, $u_1 = -\sqrt{3/28}$, $u_2 = 0$, $u_3 = \sqrt{3/28}$ and $u_4 = 1/2$. The corresponding trial polynomials are

$$P_0(u) = 14u^4 - 7u^3 - \tfrac{3}{2}u^2 + \tfrac{3}{4}u, \tag{9a}$$

$$P_1(u) = 49\left\{-\frac{2}{3}u^4 + \frac{1}{\sqrt{21}}u^3 + \frac{1}{6}u^2 - \frac{1}{4\sqrt{21}}u\right\}, \tag{9b}$$

$$P_2(u) = \frac{112}{3}u^4 - \frac{40}{3}u^2 + 1, \tag{9c}$$

$$P_3(u) = 49\left\{-\frac{2}{3}u^4 - \frac{1}{\sqrt{21}}u^3 + \frac{1}{6}u^2 + \frac{1}{4\sqrt{21}}u\right\}, \tag{9d}$$

   and

$$P_4(u) = 14u^4 + 7u^3 - \tfrac{3}{2}u^2 - \tfrac{3}{4}u. \tag{9e}$$

Once the polynomial basis has been defined, the variational collocation method proceeds as any finite element formalism. However, all the linear or bilinear products involving the trial functions should be carried out using a Lobatto's quadrature formula in order to diagonalize the mass matrix.

The Nodal Collocation Method

We have developed a second family of collocation techniques using a nodal formalism. These techniques assume an expansion in Legendre polynomials of the neutron flux over each element, without imposing a C_0 continuity on the interfaces. The absence of a C_0 continuity precludes the use of a variational formulation based on the classical functional of the finite element method [1]. We will show that a nodal formalism makes it possible to bypass this restriction, while satisfying the four conditions previously stated and without having to resort to a numerical integration of the kind used in the variational collocation method.

The nodal collocation method has two features which distinguish it from other polynomial nodal approaches [6]. First, the nodal collocation method does not require the quadratic transverse leakage approximation to generalize in two or three dimensions. Instead, it uses a tensorial expansion of Legendre polynomials, which introduces no further approximation to the one-dimensional case. Second, we will show that the linear nodal collocation method is identical to the mesh centered finite difference method. Until now, only the analytical nodal method (ANM) appeared to be linked to this type of finite difference [4].

The Legendre polynomials used in the nodal collocation method differ somewhat from their classical definition because of the following two constraints:

1. Polynomials $P_k(u)$ must be defined over the interval $(-1/2,\ 1/2)$ to ensure that the reference cube has a unitary volume.

2. Polynomials $P_k(u)$ must be mutually orthonormal over the reference interval; that is,

$$\int_{-\frac{1}{2}}^{\frac{1}{2}} du\, P_k(u)\, P_l(u) = \delta_{kl} \tag{10}$$

where δ_{kl} is the Kronecker delta function.

We will therefore use the following Legendre polynomials:

$$P_0(u) = 1, \tag{11a}$$

$$P_1(u) = 2\sqrt{3}\,u, \tag{11b}$$

$$P_2(u) = \frac{\sqrt{5}}{2}(12\,u^2 - 1), \tag{11c}$$

and, in general:

$$P_{k+1}(u) = 2\sqrt{\frac{2k+3}{2k+1}}\frac{2k+1}{k+1}\,u\,P_k(u) - \sqrt{\frac{2k+3}{2k-1}}\frac{k}{k+1}P_{k-1}(u), \tag{11d}$$

if $k \geqq 1$.

A weighted residual approach is then applied to these trial functions in order to transform equation (1) into a constant matrix system. The algebra involved with the nodal collocation method is tedious and will be omitted here. A complete description of the method can be found in reference 3.

Matrix Storage Schemes and Resolution Techniques

With a two-group energy formalism, the variational or nodal collocation method is applied group by group to generate the following matrix system:

$$\mathbf{A}_{gg}\vec{\phi}_g = \vec{S}_g \tag{12}$$

with

$$\vec{S}_1 = \frac{1}{K_{eff}}\{\mathbf{B}_{11}\vec{\phi}_1 + \mathbf{B}_{12}\vec{\phi}_2\} \tag{13a}$$

and

$$\vec{S}_2 = \mathbf{A}_{21}\vec{\phi}_1. \tag{13b}$$

Groupwise values of the neutron flux are therefore represented by the polynomial coefficients associated to all the elements. The solution of the overall eigenvalue problem can be found using the preconditioning power method as presented in the reference 5. TRIVAC also offers the possibility to solve a fixed-source eigenvalue problem, which is useful in applications involving the generalized perturbation theory [7, 8, 9] and the improved quasistatic approach in space-time kinetics.

Convergence of the preconditioned power method is very slow in cases where the solution corresponds to a flattened neutron flux. This difficulty was resolved in TRIVAC using the symmetric variational acceleration technique (SVAT), as pointed out in reference 10. This approach offers some similarities with a gradient technique applied to the eigenvalue

The preconditioning matrix used in TRIV/ valent to an ADI splitting of the matrices approach will be effective only insofar as th $\mathbf{A}_{gg}$ may be split according to the equation

$$\mathbf{A}_{gg} = \mathbf{U}_{gg} + \mathbf{X}_{gg} + \mathbf{P}_y\mathbf{Y}_{gg}\mathbf{P}_y^T + \mathbf{P}_z\mathbf{Z}_{gg}\mathbf{P}_z^T,$$

where $\mathbf{U}_{gg}$= matrix containing the diagonal of $\mathbf{A}_{gg}$; $\mathbf{X}_{gg}$, $\mathbf{Y}_{gg}$, $\mathbf{Z}_{gg}$ = symmetrical matrices c the non-diagonal elements of $\mathbf{A}_{gg}$ correspor y, and z couplings, respectively; $\mathbf{P}_y$, $\mathbf{P}_z$ = pe matrices which ensure a minimum band matrices $\mathbf{Y}_{gg}$ and $\mathbf{Z}_{gg}$.

This type of splitting capitalizes on the po numbering the unknowns in such a way tha $\mathbf{X}_{gg}$, $\mathbf{Y}_{gg}$, and $\mathbf{Z}_{gg}$ appear with a diagona structure. An important characteristic of the and nodal collocation methods allows a maxi width, respectively equal to K + 1 and 2K f order of discretization.

Numerical Results

The variational and nodal collocation met programmed and included in the TRIVAC code. The two kinds of discretization share computer environment and use the same analysis techniques for the solution of matri

The TRIVAC computer code is written in FC and compiled by FORTRAN-VS (IBM) at level tv zation. All vectors and matrices are declared precision (1 word = 32 bits) and are dy allocated by a subroutine written in assembl accumulators assigned to bilinear product ca are declared in double-precision, in order to round-off error. Numerical tests were carrie IBM-4381 (group two) computer and are typical of a scalar computer.

It is useful to mention the main numerical t and the calculation options used in this stu

1. The neutron diffusion equation (1) is discre the variational or nodal collocation method. T of discretization is equal to three (cubic poly four (quartic polynomials). In all cases, the re plane is partitioned using one and only one e assembly.
2. The fundamental solution of the eigenvalue and (13) is obtained by the preconditioned pow with a two-parameter variational acceleratior
3. A preconditioning is applied by carrying out ADI iterations per outer iteration of the power
4. The power method is initialized by a uniform the solution (ϕ_i = 1.0). The iterations are when the following convergence criterion is

$$\frac{\max_i |\phi_i^{(k-1)} - \Phi_i^{(k)}|}{\max_i |\phi_i^{(k)}|} \leq 10^{-4},$$

Table 1: IAEA-2D Benchmark Calculations

	Polynomial order	Bandwidth	K^a_{tot}	K^b_{eff}	ϵ_{max} (%)	$\bar{\epsilon}$ (%)	CPU time (s)	Outer iterations	Inner / outer
Variational collocation	3[c]	4	568	1.029786	5.8	2.2	4.8	51	1
	3	4	568	1.029785	5.8	2.2	6.4	39	2
	4	5	1033	1.029596	0.93	0.34	15.9	87	1
	4	5	1033	1.029592	0.93	0.34	14.3	45	2
Nodal collocation	3[d]	6	621	1.029370	4.9	1.8	4.8	39	1
	3	6	621	1.029374	5.0	1.8	7.1	33	2
	4	8	1104	1.029590	0.78	0.26	15.1	63	1
	4	8	1104	1.029582	0.77	0.26	16.7	39	2

[a]The number of unknowns per energy group.
[b]The reference solution was obtained by a nodal analytic calculation with a mesh of 34 × 34. The corresponding effective multiplication factor is $K_{eff} = 1.029585$.
[c]See Figure 2 for an illustration of thermal flux distribution.
[d]See Figure 3 for an illustration of thermal flux distribution.

Table 2: Tihange Test Problem Calculations

	Polynomial order	Bandwidth	K_{tot}	K^a_{eff}	ϵ_{max} (%)	$\bar{\epsilon}$ (%)	CPU time (s)	Outer iterations	Inner / outer
Variational collocation	3	4	1888	1.000312	4.2	1.9	18.8	45	1
	3	4	1888	1.000312	4.3	1.9	24.7	33	2
	4	5	3401	1.000707	1.1	0.45	55.6	69	1
	4	5	3401	1.000704	1.1	0.46	57.9	39	2
Nodal collocation	3	6	1989	1.001300	4.1	1.7	35.4	63	1
	3	6	1989	1.001303	4.1	1.7	52.2	51	2
	4	8	3536	1.000971	1.1	0.43	128.1	111	1
	4	8	3536	1.000961	1.1	0.43	162.9	75	2

[a]The reference solution was obtained by a nodal analytic calculation with a mesh of 51 × 51. The corresponding effective multiplication factor is $K_{eff} = 1.000823$.

Table 3: IAEA-3D[a] Benchmark Calculations

	Polynomial order	Bandwidth	K_{tot}	K^b_{eff}	ϵ_{max} (%)	$\bar{\epsilon}$ (%)	CPU time (s)	Outer iterations	Inner / outer
Variational collocation	3	4	9088	1.029313	6.8	2.3	240.8	87	1
	3	4	9088	1.029315	6.8	2.3	258.2	51	2
	4	5	21693	1.029117	1.2	0.36	1012.0	141	1
	4	5	21693	1.029113	1.2	0.36	1128.0	87	2
Nodal collocation	3	6	9315	1.028810	5.3	2.0	223.9	75	1
	3	6	9315	1.028812	5.2	2.0	247.3	45	2
	4	8	22080	1.029045	0.97	0.34	815.5	99	1
	4	8	22080	1.029037	0.97	0.34	1144.0	75	2

[a]All TRIVAC calculations are based on a mesh of 9 × 9 × 5 with axial mesh lines at 0., 20., 150., 280., 360. and 380. cm.
[b]The reference solution was obtained by a nodal analytic calculation with a mesh 26 × 26 × 18. The corresponding effective multiplication factor is $K_{eff} = 1.029060$.

where $\phi_i^{(k)}$ is the i-th flux component after k iterations. This criterion makes possible a convergence precision better than 0.05% over the zonal powers.

5. The solutions thus obtained are compared to the reference calculations carried out with the analytic nodal method [4]. The maximum and average errors over zonal powers ϵ_{max} and $\bar{\epsilon}$ are calculated as in reference 1.

Tables 1 to 3 give the numerical results for three specific cases: the two-dimensional (2D) and three-dimensional (3D) IAEA benchmarks [11] and the two-dimensional Tihange test problem, which represents a complete configuration of a pressurized water reactor (PWR) at the beginning of the second cycle [12].

The overall numerical results reveal two interesting

Table 4: Bounding Effect of Zonal Powers[a]

e	*Variational collocation (%)*	*Nodal collocation (%)*
1	−0.09	0.12
2	−0.51	0.30
3	−0.36	0.25
4	−0.42	0.34
5	0.14	−0.05
6	−0.14	0.16
7	0.18	−0.09
8	0.56	−0.42
9	−0.39	0.25
10	−0.34	0.23
11	−0.30	0.23
12	−0.30	0.21
13	−0.02	0.05
14	0.21	−0.14
15	0.57	−0.46
16	−0.29	0.22
17	−0.24	0.19
18	−0.13	0.14
19	0.05	−0.01
20	0.23	−0.17
21	0.93	−0.77
22	−0.21	0.18
23	−0.19	0.21
24	0.11	−0.07
25	0.61	−0.48
26	0.37	−0.32
27	0.06	−0.04
28	0.89	−0.77
29	0.91	−0.77

[a]IAEA-2D benchmark calculations discretized by a quartic collocation method.

features, which are clearly seen in Table 4 for a specific case:

1. For a given order of discretization, the variational and nodal collocation methods offer similar accuracy.
2. The reference solution, obtained through the analytic nodal method, is bounded by the variational and nodal collocation solutions, respectively. This bounding effect is observed for the eigenvalue and for most zonal powers.

It is worth mentioning that the band width of matrices $\mathbf{X}_{gg'}$ $\mathbf{Y}_{gg'}$ and $\mathbf{Z}_{gg}$ is equal to K + 1 with the variational collocation method, while it is equal to 2K with the nodal collocation method. This feature tends to penalize the nodal collocation method in terms of calculation efficiency and memory utilization.

It is not possible to conclude that one discretization method is numerically more stable than the other. The variational collocation method is more stable for solving the Tihange test problem while the nodal collocation method seems to be preferable for the IAEA benchmarks.

Figures 2 and 3 illustrate the thermal flux distributions obtained for the IAEA-2D benchmark when a third order discretization is used. The distribution corresponding to the variational collocation method

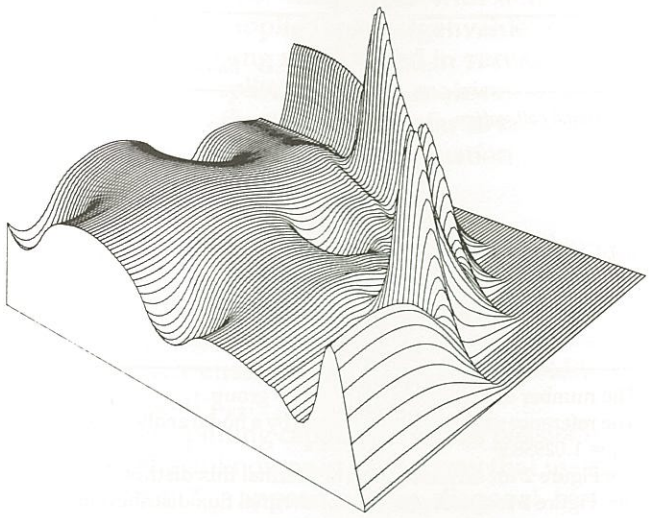

Figure 2: Distribution of thermal flux for a discretization of the IAEA-2D benchmark by the variational collocation method.

is formed from the assembly of bicubic polynomials and does not have any discontinuities at element boundaries. On the other hand, the distribution corresponding to the nodal collocation method consists of the assembly of piecewise continuous biparabolic polynomials. The discontinuities, located on element boundaries, are responsible for the incompatibility of a Legendre representation with the variational formalism presented in reference 1. While reconstructing the neutron flux obtained by the nodal collocation method, we have lost one order of representation on the trial functions. However, this causes no practical handicap during the later stages of calculation requiring such a reconstruction.

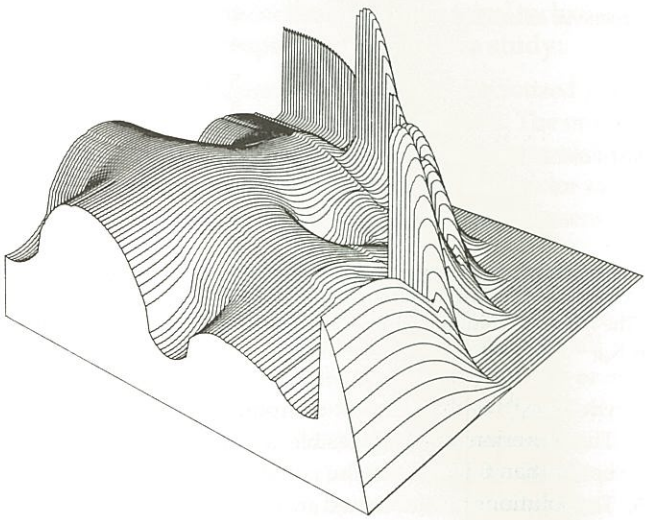

Figure 3: Distribution of thermal flux for a discretization of the IAEA-2D benchmark by the nodal collocation method.

Conclusions

We have described a new family of numerical techniques for discretizing the neutron diffusion equation. These collocation methods makes it possible to increase the order of discretization by varying the degree of polynomials used as trial functions. The minimum order, corresponding to a linear representation, is equivalent to the mesh-corner or mesh-centered finite difference method.

The variational and nodal collocation methods appear to be linked, in so far as they provide lower and upper limits for the exact solution of the diffusion equation. This property has not been proven analytically, but it has nevertheless been observed in the majority of numerical tests carried out so far.

The variational and nodal collocation methods also share a certain number of interesting properties for numerical applications: they are compatible with an ADI preconditioning and permit reconstruction of flux after convergence. They therefore share the main advantages of the finite element method, without being committed by its main drawback.

The main criticism that could be made of collocation methods is their reliance on tensorial expansions of trial polynomials. This means that when going from one to three dimensions, for a given order of discretization, the number of unknowns associated with each element is cubed. This approach, while mathematically coherent, requires more computer resources than methods based on the quadratic transverse leakage approximation [4].

Work is now underway to use this new diffusion module in fuel management and design applications, such as:

1. OPTEX-4 for the 3-D optimization of fuel enrichment, burnup, and adjuster grading in a CANDU reactor, using generalized perturbation theory [8, 9];
2. XSTATIC for the solution of the space-time kinetics equations using the generalized quasistatic algorithm;
3. a new diffusion module for the FMDP family of codes [13].

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Fuel for Thought

J.A.L. Robertson

Abstract

The outstanding performance of CANDU fuel is attributed to a well-rounded development program that combined sound science with excellent engineering. Economic analysis and technology transfer were integrated into the program to ensure successful commercialization. The CANDU fuel program provides useful lessons that are being ignored by policy-makers in their desire to exploit our world-class resources in science and technology to benefit the Canadian economy.

Résumé

La performance remarquable du combustible CANDU est le résultat d'un programme de développement très complet qui combine la science rigoureuse au génie par excellence. On a incorporé au programme une analyse économique et un procédé de transfert de technologie qui ont assuré une commercialisation réussie. Nos politiciens qui veulent exploiter au profit de l'économie canadienne nos ressources de classe mondiale en science et en technologie ne tiennent toutefois pas compte des leçons valables fournies par ce programme.

Purposes

The fuel for CANDU reactors represents a magnificent achievement of Canadian engineering. The first purpose of this paper is to trace the development of the fuel, identifying significant contributions. An occasion such as this Engineering Centennial is a legitimate reason for pride and self-congratualtion, and I cannot recall the development without reliving the excitement and satisfactions of that time. However, anyone who expects just nostalgic reminiscences and a eulogy to the good old days is going to be disappointed. I refuse to rest on our laurels, believing them to make an uncomfortable bed. I am much more interested in learning from the past how to improve the future.

The purpose of learning from our successes, just as much as from our failures, is particularly important and urgent now when many people with no experience in managing a successful technological development talk interminably about a policy for science and technology. Thus, this paper may be seen as a technical report leading to a political tract, if discussing a policy vacuum is political. Between the technical and political parts is an examination of future directions for the CANDU fuel program.

But first I wish to bring the acknowledgements up front, to emphasize that they are no ritual afterthoughts.

Acknowledgements

As always, in recalling the past human memory mercifully mists over periods of distress in favour of pleasant experiences. I still remember the anxious days of the mid-1960s when seemingly endless problems in getting the Douglas Point reactor to work properly caused concern that CANDU might be canned. However, happy memories of exciting and satisfying experiences predominate.

In discussing the subjects with erstwhile colleagues I was reminded of the times when a simple, but ingenious, experiment gave a 'Yes' or 'No' answer, and not just a revision of the last decimal place; when even an in-reactor experiment could be suggested, agreed upon, performed, and the results published in a matter of weeks; when we discussed and interpreted each others results without regard to organizational hierarchies; and when anyone returning from a conference seemed to bring back new results, and new controversies, leading to new experiments.

Some examples of simple, but definitive, experiments are given in the following sections.

Much of the credit for the success of the program is due to the excellent facilities that were available. In its early life the NRX reactor had the highest neutron flux of any research reactor. This fact attracted to Chalk River fuel researchers from other countries, notably those from the U.S. Bettis Laboratory who installed the in-reactor 'loops' that were to prove so valuable to our own fuel program. The NRU reactor, with its much larger loops, subsequently complemented NRX, particularly for engineering tests and now, nearly thirty

Keywords: nuclear fuel, science, technology transfer CANDU, engineering, public perception, science and technology policy.

years later, it is still one of the world's best for this purpose. Other lesser facilities were particularly useful to the fuel program in providing valid comparative irradiations quickly and at remarkably low cost: the innovative 'Hydraulic Rabbit' in the NRX reactor, largely due to John Melvin, and the 'Demountable Bundle'[1] in the NRU reactor loops, due to Jim Christie. Other facilities deserving special mention were the laboratory for fabricating experimental fuel specimens and the various shielded 'caves' in which the irradiated specimens were examined.

As so often happens, the whole was greater than the parts. The location of all these facilities within a radius of about one hundred metres gave us a tremendous advantage. Those running an experiment would be consulted by reactor operators when something unforeseen arose; those who had fabricated a specimen would be consulted during post-irradiation examination.

Even more of the credit is due to the people involved, several of whom are recognized in the next section. What made the Chalk River Laboratory so productive, and such a pleasant and inspiring place to work, was the competence and friendly cooperation of many anonymous individuals: Those who operated the reactors and loops; those who prepared the specimens and test equipment, and those who conducted the post-irradiation examination; chemical analysts and metallographers; designers and secretaries; reactor physicists and health physicists; and so many more whose help we took for granted.

No account of CANDU fuel is possible without recognition of Bennett ('W.B.') Lewis' responsibility for every aspect of the program. He epitomized in one person all that was best in the Canadian approach. W.B., although nominally a scientist rather than an engineeer, represented my ideal engineer in directing the fuel program: he had a fundamental and catholic grasp of scientific principles; he continually strove for excellence in all things; he was motivated to see his work applied for the benefit of society; he had a keen appreciation of what was practicable; he always kept in mind the economic implications; and he demonstrated a willingness to make engineering decisions based on existing knowledge, without waiting for the discovery of ultimate scientific truth.

Steps in Developing CANDU Fuel

The fuel is literally and metaphorically at the core of the reactor. The outstanding performance of the fuel, for reliability, economy, and safety, reflects that of the reactor. For both fuel and reactor the success of the product can be largely attributed to the soundly based, well-rounded engineering program for its development.

In my perception, with all the benefits of hindsight, the development of CANDU fuel proceeded in five distinct steps. The relevance to policy formulation lies in the fact that our program included all the elements essential to any technological development. Developing an engineering product can be visualized as an athlete ascending a flight of steps to light the Olympic flame. Each step, resting on the previous one, is essential and, despite some overlap, there is a general progression from one to the next until the goal is reached. When we took the first step we did not realize how far we would have to climb, but nowadays anyone responsible for technological development should plan for all five steps.

These five steps provide the outline for the technical part of the paper:

- sound science,
- excellent engineering,
- integrated economics,
- technology transfer, and
- public perception.

Sound Science

The first paradox of CANDU fuel, shared with the CANDU system as a whole, is that although it is a unique and highly successful product, most of the original basic research was done elsewhere. For the CANDU system, nuclear fission, fission reactors, heavy water, heavy-water reactors, pressure tubes, and zirconium-niobium alloys were all discovered elsewhere: for CANDU fuel, uranium dioxide (UO_2) and Zircaloy, its two major constituents.

The second paradox, which explains the first, is that the Canadian program was unusually strong in underlying science. (I distinguish between basic science, which is entirely curiosity driven without regard to any ultimate application, and underlying science, which is deliberately performed in areas of applied relevance to provide greater understanding of critical phenomena.)

The first benefit of this strength in underlying science was that the Canadian team was aware of international developments in nuclear science, and had the necessary expertise to select the best from what was available to synthesize a solution applicable to Canadian conditions.

The second benefit was that the excellence of the research, and of the research facilities, brought to Chalk River scientists from all other countries with significant nuclear-energy programs. Consequently, we had access to new results at an earlier stage and in greater detail than would otherwise have been possible, and we were therefore able to undertake critical assessments of new developments.

When Canada committed its first power reactor, the Nuclear Power Demonstration (NPD) reactor, in 1954, the reference fuel material was metallic uranium with only minor alloying additions, based on the successful operation of similar fuel in the NRX reactor. However,

both the U.S. (Bettis Laboratory) and the U.K. (Harwell Laboratory) programs had been experiencing difficulties with this material for their proposed power reactors, and were investigating UO_2. Although this had a lower density than uranium metal, making it less attractive for neutron economy, it was vastly superior in corrosion resistance and dimensional stability under irradiation – the two areas giving problems.

That we obtained access to the promising results on UO_2, and similarly on Zircaloy as the sheathing material, in time to select them for the NPD reactor was due to our having a tripartite exchange to which we were making substantial contributions. Even so, we might not have had the courage to take such a large step had we gained the information only by reading reports. In fact, we participated fully in the key irradiation tests, including the post-irradiation examination, conducted at Chalk River. It was this intimate experience that provided the necessary confidence.

Selecting these two materials was, however, only the start. In the design of CANDU fuel they were used in ways that had not been attempted elsewhere. We did not just adopt, but adapted, these materials to our own requirements in a unique design. The three most important differences were the use of thin collapsible sheathing, the operation at higher central temperatures, and the use of short bundles involving the ends of fuel stacks in positions of maximum power; all of which were required to enhance neutron economy. In each case the bold decision was based largely on favourable experience from empirical testing in the loops of the NRX reactor. The confidence in these decisions, however, was greatly increased by a sound understanding of the underlying phenomena.

The Chalk River team provided international leadership in understanding nuclear fuel behaviour, as illustrated by the following highlights:

–*Alan Ross* determined the effects on the thermal conductivity of UO_2 of porosity and excess oxygen, and discovered irradiation-enhanced elimination of fine porosity; he elucidated the factors controlling fuel-to-sheath heat-transfer, hence explaining the relatively small effect of helium-filling observed by other Chalk River researchers.

–*Ross MacEwan* demonstrated that the appearance of central melting in the fuel could result from migration up a thermal gradient of lenticular pores in the UO_2, and provided much of the information on grain growth in UO_2 needed to interpret structures seen in post-irradiation examination.

–*Mike Notley*, by first irradiating fuel elements to produce large grains in the centre of the UO_2, and then re-irradiating them at a higher power, disproved the claim of the U.S. Hanford Laboratory that these large grains had a much higher thermal conductivity.

–*John May* (on attachment from the U.K.), with others, confirmed this conclusion in a simple laboratory comparator and went on to show that the enhanced conductivity was due to a composition shift to UO_{2-x}.

–*Al Bain* demonstrated the in-reactor healing of cracks in the UO_2 at relatively low temperatures, and showed how dishing in the end-faces of the fuel pellets (a Canadian innovation) could be used to control the longitudinal and diametral strains in the sheath. With *Mike Notley* and others, he defined the fuel-sheath clearances that allowed the use of thin collapsible sheaths, having a thickness-to-diameter ratio one-quarter of that considered necessary in Light Water Reactors to avoid the formation of severe deformations leading to failure.

–*Denis Hardy* defined a stress-relief treatment for the Zircaloy, well short of the full anneal practised elsewhere, that combined adequate strength for the thin sheath with adequate in-service ductility.

–*Mike Notley* developed computer codes for a simplified elastic-plastic model that predicted the releases of fission-product gases and sheath deformations; and later validated the model by ingenious in-reactor experiments to measure internal gas pressure, fuel-stack elongation, and sheath strain. This model, by combining empirical results with an understanding of the phenomena, was simpler than other contemporary models, but at least as reliable.

–*Ross MacEwan* and *Bill Stevens* found that gaseous fission products were immobilized in very fine pores, and hence elucidated the diffusion of rare gases in solids.

–*W.B. Lewis* interpreted experimental results by *Roger Kelly, Bill Stevens, Bob Hawkings*, and *Bob Hart*, to establish an irradiation-induced mechanism for re-entry of these gases.

–*Brian Cox* and *Eric LeSurf* contributed much to the understanding of the factors controlling the corrosion and hydriding of zirconium alloys, particularly under irradiation.

–*Tony Sawatzky*, and later *Roger Dutton*, provided mathematical analyses that allowed prediction of how hydrogen would migrate under thermal and stress gradients.

–*Chuck Ells* and *Brian Cheadle*, in work primarily aimed at pressure tubes, helped to establish the metallurgical structures to minimize hydriding damage.

–*Gareth Parry* and *Windsor Evans* demonstrated that otherwise brittle hydrides could deform plastically under in-service conditions.

The introduction of UO_2-in-Zircaloy fuel worldwide was not without its setbacks. Here too Canada was well served by a program with a sound scientific foundation.

In the late 1950s researchers at the Bettis Laboratory attributed dramatic fuel failures, including one of theirs irradiated in the NRX reactor, to central melting of the UO_2. This interpretation could have necessitated a reduction in the power output intended for the

CANDU fuel, with a consequent economic penalty. The Chalk River team was able to prove that the structure thought to define melting could be produced by pore migration in solid UO_2, and that the actual temperatures were hundreds of degrees lower than believed.

Early in the 1960s, similarly dramatic failures of Zircaloy-sheathed UO_2 fuel rods that had been test-irradiated with deliberate holes in their sheaths were reported by the U.S. General Electric's San Jose Laboratory. This news was undermining international confidence in the still-new zirconium alloys as sheathing, and designers of Light Water Reactors were contemplating a return to stainless steel. This course was possible for them with enriched uranium, but would have been disastrous for the Canadian program, and we would have been severely taxed to have had to go it alone in developing zirconium technology. Fortunately, however, Mike Notley and others were able to demonstrate that the cause was probably a very high level of fluoride impurities in the San Jose enriched fuel, and that the low levels associated with natural UO_2 resulted in much stabler behaviour.

Late in the 1960s, a worrisome number of fuel failures in Light Water Reactors was attributed to hydriding of the Zircaloy sheaths. Our investigations proved that the hydrogen was being introduced as water adsorbed on the UO_2 fuel, and that this failure mechanism could be avoided by proper drying of the fuel and by good quality control.

In the 1970s, there was some concern in the Light Water Reactor community over observations of fuel densification in service, resulting in large gaps developing within the sheaths. CANDU designers and operators remained unconcerned, largely because of our higher as-fabricated fuel density, our shorter fuel bundles, our horizontal orientation, and because we had not encountered any such problem in our power-reactor fuel. This empirical confidence was reinforced by a good basic understanding of the densification phenomenon from studies by Alan Ross, Mike Notley, and Ian Hastings.

Excellent Engineering

The engineering design of CANDU fuel proceeded in parallel with the scientific studies on which the design was based. The essential design of rodded bundles was selected by the Nuclear Power Group, headed by Ontario Hydro's Harold Smith, working at Chalk River in the mid-1950s on the original design for the NPD reactor. The most important change in the fuel occurred in 1957, when NPD was changed from a pressure-vessel reactor with full-length vertical fuel to a pressure-tube reactor with short horizontal fuel capable of being changed on-power. Since then, CANDU fuel has consisted of 0.5 m-long rodded bundles.

Subsequent to the NPD reactor's startup, the maximum power that could be extracted from a CANDU fuel bundle was increased fourfold, as a result of three distinct factors. When, on going from the Douglas Point to the Pickering reactors, the diameter of the pressure tubes was increased, the bundle diameter increased from eight to ten centimetres, requiring more rods or elements. On going from the Pickering-A to the Bruce-A reactors, the number of elements in each bundle of 10 cm diameter was increased from 28 to 37. The third factor, an increase in the permissible power to be extracted from any individual element, was a direct result of the scientific studies on fuel behaviour. The current maximum power of one megawatt per bundle means that a single bundle, weighing only about 20 kg, generates enough heat to keep a hundred homes warm.

Developing means of fabricating the bundles constituted a major part of the engineering teamwork led by Ara Mooradian and Ron Page. The scientific studies showed that the use of thin collapsible sheaths was possible, but these could not have been incorporated in the design had it not been for the development of greatly improved methods for the non-destructive testing of thin Zircaloy tubes. The early spot-welded wire-wrap, used to separate the elements and to centre the bundle in its tube, was replaced by spacer- and bearing-pads, thanks to the development of a reliable method for beryllium-brazing these pads. Another development allowed the sealing of individual elements to be done by resistance-force welding, instead of the less efficient argon-arc method. This was important in minimizing the amount of non-productive Zircaloy at bundle ends, and in minimizing the separation between fuel stacks in adjacent bundles that causes power-peaking near stack ends. The *absence* of a gas plenum, another unique feature of the CANDU fuel design that contributes to both economy and safety, was attributable to the thorough understanding of the release phenomena from the scientific studies.

Concurrently, a large program of engineering testing at the Sheridan Park Engineering Laboratory established that the bundles had adequate strength to withstand mechanical and hydraulic loads, provided adequate coolant mixing, and would not subject the pressure tubes to excessive wear. Another major contribution by the group at Sheridan Park in cooperation with Ontario Hydro staff was the development of a quality assurance program for CANDU fuel, associated primarily with Milan Gacesa.

Thermalhydraulics constituted another essential element of the design process. Even today, the application of this subject to fuel design is still largely empirical but, just as in other aspects of the program, based on a very thorough understanding of relevant results and phenomena. Through an awareness and critical assessment of international research in this

area, Dé Groeneveld, Don McPherson, Joe Ahmad, and their colleagues were able to provide the designers with critical-heat-flux correlations from which to set maximum permissible power outputs from any fuel channel. Here too, the CANDU team fully exploited discoveries made elsewhere, but added significantly to the international fund of knowledge on the subject through both unique experiments and analysis. Particularly difficult was the analysis of phase separation in a boiling coolant for horizontal fuel channels.

The same thermalhydraulics researchers collaborated with the metallurgical engineers in another large program, to assure the safety of CANDU fuel even in the event of a serious reactor accident. The inclusion of this work in the overall program for developing CANDU fuel followed a long tradition of engineering that safety analysis is an integral part of any engineering design. This simple fact is rarely recognized by our critics or the media. Early work was spearheaded by Denis Hardy, with input from Mike Notley on fuel behaviour. The experiments progressed from laboratory simulations to in-reactor tests on full-size bundles until, in 1982, Dan Meneley and Bill Hancox were able to announce at an international conference that even in the extreme case of a loss of coolant and complete failure of the emergency injection system there would be no fuel melting and fuel-channel integrity would be maintained.

Besides integrating safety into the design, CANDU fuel development adopted two other sound engineering principles: 'Keep it simple,' and 'If it ain't broken, don't fix it.' The most significant change in the very simple CANDU fuel bundle since increasing its diameter was to add a very thin layer of graphite to the inner surface of the Zircaloy sheath, to produce what is termed 'CANLUB' fuel. This modification was introduced to provide greater resistance to fuel failures. In 1970 an increase in the failure rate in the Douglas Point reactor was detected, while the actual level remained at a very low value of under one per cent of all bundles. The threefold response to the problem, and its rapidity, were a tribute to the engineering excellence of a large cooperative team drawn from Atomic Energy of Canada Limited, Ontario Hydro, and the two commercial fuel fabricators, Canadian General Electric and Canadian Westinghouse.

First, the cause of the failures was shown to be a significant increase in fuel power after a prolonged period at relatively low power. Second, Ontario Hydro, from an understanding of the cause, was able to introduce restrictions on the magnitude and rate of power changes during refuelling that reduced the failure rate at the cost of some loss of operational flexibility. Third, CANLUB fuel, with improved tolerance to power increases, was selected from 17 potential solutions on the basis of an extensive reactor-testing program, and developed to the point of being introduced into a major power reactor within two years of the problem appearing.

This successful joint operation, directed in succession by Ron Page, Roy Thomas, and Al Bain, although essentially engineering in nature, again drew upon Chalk River's fund of scientific understanding. Some of the potential failure mechanisms were rejected as a result of this, while others were favoured, notably stress-corrosion cracking of the Zircaloy sheath by volatile fission products at stress concentrations over cracks in the UO_2 pellets. Scientific studies by Clive Wood and Brian Cox on stress-corrosion cracking, and by Dave Williams and Kit Coleman on stress concentrations, were invaluable in directing thinking towards the graphite layer of CANLUB fuel. Rod MacDonald confirmed the stress-corrosion mechanism for failure by irradiating fresh UO_2 pellets in pre-irradiated sheaths and obtaining no failures under conditions that cause failures in elements with both fuel and sheath pre-irradiated.

Even today, the only other remedy available for this failure mechanism is the 'barrier-clad' fuel developed for Light Water Reactor fuel. This is a much more expensive solution that took many more years to introduce than the very simple CANLUB fuel.

Yet another extensive interdisciplinary program, this one, conducted over many years, established how long the utility could safely leave failed fuel in the reactor without serious deterioration of the fuel bundle and radioactive contamination of the reactor's primary coolant circuit. Rod MacDonald, and many others, have shown that even for a CANDU fuel bundle operating at its maximum power, sufficient time exists to detect the failure and discharge the bundle.

The deposition of thick layers of corrosion products on the sheath was a non-problem for CANDU reactors, thanks to the early control of coolant chemistry provided by Bob Robertson and Merv Allison, and subsequent understanding of the phenomenon by Ken Burrill and Derek Lister. Ironically, the deposit is known as 'crud,' a centuries-old term retroactively attributed to Chalk River Unidentified Deposit.

An account of the engineering development of CANDU fuel would be incomplete without recognition of the contribution by chemical engineers. In the 1950s they developed a flow sheet for the production of UO_2 powder from U_3O_8 ('yellowcake'). Once again, the basic process (via ammonium diuranate) was already known and the achievement of the development by Bill Bourns, Verne Watson, and John Yatabe was in defining conditions to produce powder capable of yielding high-density pellets reproducibly. This characteristic is much more important in CANDU reactors, with their emphasis on neutron economy, than in Light Water Reactors. Their success can be judged from the

fact that their process is still in commercial use thirty years later and that the original development is almost forgotten.

Integrated Economics

To be successful, an engineering product not only has to perform as intended, it also has to be cost-competitive. From the start of CANDU development, W.B. Lewis saw with great clarity that economic nuclear energy depended on low fuelling costs, since relatively high capital costs are inevitable; and that the way to this goal lay in the strict application of neutron economy. He directed the technical development of the CANDU design according to this guiding principle.

In common with those who directed the development programs for other reactor types, he sought to increase the thermodynamic efficiency of CANDU reactors. However, he appreciated that this was only a means to reducing costs, and not an end in itself. Economics is the ultimate design criterion, not the narrower engineering criterion of thermodynamic efficiency.

Neutrons are the currency of nuclear fission. Any neutron wasted has to be replaced by the provision of fresh nuclear fuel – at a cost. This point was made dramatically by W.B. Lewis when he derived a value of $2,860 per gram of neutrons (in 1961, when gold was about $1 per gram). This conversion is an example of how abstruse physical principles must be translated into simple rules if they are to be readily used in design.

During the design of the first CANDU reactor, before the term CANDU had been coined, a simple running indicator of the neutron economics of the design was provided by the estimated fuel burnup achievable with natural uranium (megawatt-days output per tonne of uranium). W.B. Lewis kept his 'fever chart,' similar to the temperature chart of a hospital patient, showing changes in estimated burnup. A measurement showing the neutron-absorption cross-section for a fission-product to be greater than previously accepted made the trace dive to a low that would have been only marginally economic; the achievement of a few per cent increased density for the UO_2 restored the situation. Those who had the responsibility for explaining to W.B. Lewis any of their results that affected his fever chart understand the meaning of 'neutron economy imposed a healthy discipline on the design.' Relieving a reactor designer of this discipline by allowing enriched uranium is like giving a government a printing press: when a problem arises it prints more money instead of solving the problem.

The selection, in 1958, of zirconium over aluminum as sheathing material for CANDU fuel was essentially an economic decision applying the principle of neutron economy. In an alloy-development program, Kim Krenz and his colleagues obtained promising results with respect to the corrosion resistance in hot water of a relatively cheap aluminum-nickel-iron alloy. However, in the thickness needed for strength equivalent to that of the zirconium alloy Zircaloy, then being developed by the Bettis Laboratory, aluminum would have absorbed more neutrons. This operating cost would have more than offset the lower initial cost of aluminum.

The same principle was also responsible for the selection of short fuel bundles, which facilitated on-power refuelling and hence contributed to neutron economy by allowing neutron-absorbing fission products to be removed at the optimum time.

By minimizing the wall thickness, avoiding a gas plenum, and by eliminating the flow-, support-, and control-components found in the fuel assemblies for Light Water Reactors, but not essential to the fuel's function, the UO_2 content of CANDU fuel bundles was brought above 90 per cent by weight. The rest is zirconium alloy, with no stainless steel or other strongly neutron-absorbing materials found in Light Water Reactor fuel.

As a result of the CANDU fuel being designed to maximize neutron economy, Ara Mooradian and I predicted in a 1960 issue of Nucleonics that CANDU reactors would achieve fuelling costs of less than one 'mil' (milli-dollar) per kilowatt-hour. At the time this was thought to be optimistic, even unrealistic, but in the 1970s it was achieved by Ontario Hydro at their Pickering Nuclear Generating Station, and is still being realized in the original 1960 dollars.

Technology Transfer

'Technology transfer' are trendy buzz-words these days, although there is still a very poor understanding of how to achieve it. By way of contrast, means of transferring CANDU technologies developed in the laboratories to commercial exploitation were incorporated in the program from the start, before the phrase was invented. In the early days it was unchallenged government policy that technology developed in government laboratories at public expense should be freely transferred to Canadian industry with the objective of generating new commercial activity. More recently, this policy has been criticized for not providing a source of funds for ongoing research and development. However, there is no question that it was implemented most successfully, particularly in the case of CANDU fuel.

First, the technology for UO_2-powder production was transferred to the appropriate commercial organization, Eldorado Nuclear Limited, which further developed the process. Concurrently, the design of the fuel bundle was being pursued with manufacturing methods very much in mind. In developing the process

for making the fuel pellets, John Runnalls and Geoff Chalder co-opted the expertise for mass production of industrial ceramics that existed in Canadian General Electric's (CGE) Carboloy Division. Later, the processes selected for sealing and assembling the various components into a fuel bundle owed much to developmental programs by CGE and Canadian Westinghouse, which produce CANDU fuel commercially. By now, these two companies have manufactured more than half a million CANDU fuel bundles, worth about a billion dollars, which have generated half a trillion kilowatt-hours, or ten times the total electricity produced in all Canada in 1950.

Several mechanisms were employed to transfer the technology. The conventional one, the transmission of technical reports, predominated in the process for powder preparation. For fuel-bundle fabrication, the awarding of development contracts to the commercial fuel companies proved most effective: the literally day-to-day technical supervision of commercially motivated groups by individuals who were themselves active in relevant research ensured a very detailed two-way exchange of experience. Close cooperation with the utility groups responsible for operational research on the fuel performance not only provided another means to keep the development program relevant to market needs but also meant that resulting improvements were more readily accepted by the utility customer. This benefit was well illustrated by the industry's very rapid introduction of CANLUB fuel.

One mechanism stands out from all others. The best vehicle for technology transfer is people, not paper. Many engineers who participated in the early development of CANDU fuel at Chalk River moved back, or on, to other organizations, carrying with them an intimate understanding of the technology. Others, without working at Chalk River, shared in the development through contracts, visits, and meetings. Personal contacts made while working together towards a common objective proved invaluable later: when problems arose, it was easy and natural to pick up a phone and call a friend who could help. Furthermore, it is valuable not just to the nuclear industry but to the country as a whole to have individuals in education, production, regulation, government, and elsewhere who are thoroughly knowledgeable in the technology.

Public Perception

Even after scientific, engineering, economic, and industrial feasibilities are established, a new technology is not itself established until public acceptance is assured. Back when biotechnology was just baking and brewing, an engineer sponsoring a new technology had a hard enough job selling the idea to decision-makers in industry and government. However, the ground-rules for technical and economic assessments were known, and one could count on the decision-makers being advised by appropriately qualified professionals who would be held responsible for any advice given. Nowadays, the same requirements still hold but, in addition, the public has to be convinced that the technology is desirable. Politicians can afford to support very few unpopular causes, however beneficial these may be for the long-term future of the country.

In this regard, nuclear energy got off to a good start. There was widespread public support for the development of a new energy source to replace the dwindling stocks of conventional oil and to reduce pollution. Internationally, both environmentalist and church organizations supported this development. However, gaining public support is not something that is done once and for all; it must be regained and sustained day after day. Taking this final step in introducing a technology is like having to run up a 'down'-escalator. Ironically, just when the CANDU system delivered on its earlier promise, during the oil crises of the 1970s, public support declined and we found ourselves further down the escalator than we had been.

The CANDU system is recognized as one of the major achievements of Canadian engineering, selected by the Engineering Centennial Board as one of the top ten of the last hundred years. Its outstanding record for reliability, economy, and safety has been established in open competition with the world's best.

Regrettably, the Canadian public is largely unaware of this cause for national pride. Our media are always ready to hold us responsible for failings in foreign nuclear industries, however irrelevant to the CANDU system, but are silent on our successes. An Olympic gold medal, or even an Oscar nomination, will secure major media coverage, but who knows that CANDU reactors have occupied, year after year, about half the top ten places in the international league table for performance of power reactors?

I believe that the present situation is serious. Some countries, less well endowed with alternative energy sources than Canada, have accepted the economic and health penalties inherent in foregoing nuclear energy. Many disparate but well-organized special interest groups believe that they can occupy the moral high ground by opposing nuclear energy. It is their critical views that are predominantly reported by media that are largely ignorant of, if not actively hostile to, technology. The media are the source of virtually all public information, but they are not accountable and are often irresponsible. We have seen how successful the tactics of social activists have been in stopping the seal hunt on purely emotional grounds.

In the near future, I see only increased difficulties. The disposal of nuclear wastes must be resolved in the next few years, for low-level wastes, for mine tailings, and for used fuel. Nobody wants wastes, only the products causing them, so this issue will yield only

negative publicity. Any malfunctions of nuclear stations anywhere, however trivial, will similarly lead to only negative publicity, while even perfect performance will go unrecognized. Oil and natural gas again seem to be plentiful, and electricity shortages are unlikely to occur until the mid-1990s, by which time it will be too late to react. The *absence* of health and environmental harm due to the operation of existing CANDU stations will continue to go unnoticed. What concerns me most is that many of the technical people do not see public acceptance as a problem, and certainly not as *their* problem.

The situation is serious but not hopeless. Engineers do not have to be lectured on social responsibility. Without engineering, Canada would be among the poorest Third World countries, unable to afford all our enviable social programs. The engineering profession in Canada has a code of ethics that could well be adopted or adapted by other professions, including those arch-critics, the media.

But few members of the public are aware of all this. The nuclear industry has been ineffectual in addressing ethical issues associated with nuclear energy, and many individual engineers are uneasy dealing with them. Some of those in the CANDU fuel community are among the exceptions to these generalizations, but all of us must communicate our convictions more effectively if we are to assure a safe, economic, and virtually inexhaustible energy source for our children and for their children.

Future Directions

Those of us who were involved in the development of CANDU fuel can take comfort from the fact that the technical program is now in the hands of those who have already contributed much to the technology, including Ross MacEwan, Mike Notley, Alan Lane, Milan Gacesa, Clive Wood, and Ian Hastings. Theirs is now the responsibility for ensuring that the CANDU system remains available to future generations. That this is possible, through fuel recycling, has already been demonstrated at the scientific stage, and much of the constituent engineering is developed. However, the next step, economic feasibility, is proving much more difficult than we had expected. Although fuel recycling will eventually become economic as depletion of uranium resources causes price increases, the recent glut of uranium has removed any sense of urgency in the commercial introduction of fuel recycling. Those responsible have the difficult task of maintaining and improving the technology in the face of a public perception that their work is not needed.

This difficulty is compounded by two others:

- the necessity to maintain a response capability as insurance against operating troubles arising in the fuel of the CANDU system, which represents a $30 billion national investment; and
- the necessity to remain competitive with international reactor vendors who are constantly improving their products. As in all technologies, those who do not maintain an active, market-oriented development program will soon find themselves overtaken by their competitors.

Fortunately, developing to the point of commercialization the Slightly Enriched Uranium fuel cycle for CANDU reactors provides one means to bridge the gap. If expected reductions in the costs of enrichment services arrive while uranium is still relatively abundant, there will be a significant economic opportunity for this cycle. Much of the required fuel technology is available, but a large-scale demonstration could conceivably reveal both problems and unforeseen opportunities.

I also believe that the currently underchallenged skills of the fuel development team should be exploited to develop an industrial process for the immobilization of nuclear fuel wastes. In at least one interim scenario for fuel recycling, the valuable plutonium would be extracted for recycling with fresh uranium, while the very radioactive fission products would be left with the now depleted uranium as waste for disposal. The process used to produce UO_2 fuel pellets could probably be modified to incorporate the small amounts of fission products, yielding an extremely stable, corrosion-resistant waste form. The fuel team, with experience in processes capable of accepting radioactive feed for a thorium fuel cycle, could develop a remotely operated process with the spinoff benefit of a more economic automated process for fabricating conventional CANDU fuel. Exploiting existing expertise in this way would be preferable to building a new team in the waste management program.

The long-term stability of the design for the CANDU fuel bundle, largely unchanged for 25 years, makes it easy to forget the large number of developmental avenues that were explored but not pursued. These include:

- uranium alloys and aluminum alloys, the original reference materials for fuel and sheath, respectively;
- swaged, and later vibratory-compacted ('Vipac'), UO_2 powder;
- uranium silicide, as an alternative to UO_2 with a greater uranium density (although it is being exploited for research reactors);
- zirconium-chromium-iron alloys as a sheathing material capable of operating in dry-out or even in a steam coolant;
- graphite-coated particles of uranium carbide in a graphite matrix as a fuel element to provide superheated steam;
- annular and tube-in-shell fuel elements to allow more fuel to be packed into a given cross-section;
- fuel elements with conductive graphite discs between hollow UO_2 pellets as a design for high burnup at high power;
- a vertical string of CANDU fuel bundles on a central

support tube for a CANDU Boiling Light Water Reactor; and
- uranium-carbide fuel in zirconium-niobium sheaths for operation in a CANDU reactor with organic liquid as coolant.

All these avenues appeared attractive initially, and for most the exploration successfully surmounted major technical barriers. Ultimately, however, each failed the test of competitiveness with the design of sintered-UO_2 in Zircaloy sheaths assembled in horizontal bundles, on criteria of safety and economy.

Any argument that all this research and development (R&D) was wasted ignores the fact that R&D, like geological exploration, is inherently inefficient, with many avenues having to be explored to determine the best route. According to a dictum of Ara Mooradian, R&D pays for itself if it correctly identifies blind alleys – and wild geese that should not be chased further. Without the very well rounded capability of the CANDU fuel development team, the progress of CANDU fuel could well have been side-tracked into a costly, even calamitous, direction. This capability is going to be just as necessary in the future for the same reason.

Lessons Learned

It is ironic that politicians and academics continue to pontificate on a science-and-technology policy, while ignoring lessons from past Canadian achievements that rank with the world's best. The CANDU nuclear system is just one of these, and its fuel program alone has several valuable lessons for Canadian policy.

Research that is intended to provide economic and industrial benefits should be mission-oriented. Once the mission is assigned, the research program becomes a means to an end, and not an end in itself. Conceptually, it is simple to proceed from the mission to the objectives, and hence to the research program itself. This ensures that the research is relevant, that all necessary research is undertaken, that economics are considered along with technical factors, that the program is performed to an agreed schedule and end-point, and that commercialization is an integral part of the objective.

Since we cannot be world-class in all areas of modern science and technology, the missions must be selected very carefully to match and satisfy national objectives. This is properly a government responsibility, but governments instead concern themselves with the management of research and development, a subject in which they – politicians, bureaucrats, and consultants – are generally neither experienced nor competent.

Canadian governments, both Conservative and Liberal, have a disastrous record for fumbling science and technology. The current Conservative government slashed funds for institutions with proven performance, without having any guiding policy for science and technology. At the 1986 National Liberal Convention, when a resolution on science and technology policy came to the vote, 'messengers and beaters had to be sent into the gossiping corridors to round up a quorum of 50 concerned Liberals before a vote could be legal.' (A. Fotheringham, 1986–12–02).

High-technology industries can be grown out of resource industries. International competition in high technology is intense. Our natural advantages lie in our natural resources, and in our well-educated human resources. Opportunities for Canada are therefore to be found in adapting technologies initiated by other people for other purposes to our own particular circumstances.

Basic, or curiosity-driven, research, while vital for the advancement of mankind in the long-term, rarely benefits directly those who fund it. Basic research in Canada, in universities and elsewhere, had a negligible effect on the development of CANDU fuel. It should be recognized as an altruistic activity. Canada, as an affluent nation, should devote a fair share of its resources to basic research, without expecting economic returns from it.

Even in a mission-oriented program there is a need for research at a fundamental level on topics underlying the mission, to provide a good understanding of the relevant phenomena. This is one essential input for wise guidance of the program, and can be an invaluable resource in tackling unforeseen problems.

Applied research is an essential step from the international pool of basic scientific research to the development of technologies for social benefit. Some countries with outstanding records in basic science, notably the U.K., have been largely unsuccessful in reaping the commercial benefits from the resulting technologies. Germany in the 19th century, the U.S. in the mid-20th century, and latterly Japan, have been very successful in exploiting discoveries made elsewhere. Each in its turn has emphasized applied R&D.

Other essential inputs are economic and market analyses for the product before devoting major resources to the R&D program. Any mission-oriented program should incorporate a plan for commercialization. To this end, Canadian schools of engineering should pay more attention to these aspects in their courses. This could have the spinoff benefit of making the universities' own applied research more relevant to national needs. In a 1986 poll of more than 400 technology-driven companies in Canada, only 6.4 per cent rated university R&D as very important, while 24 per cent said it was not important at all. As necessity is the mother of invention, market forces drive innovation.

The program has to be designed with the transfer of technology to industry in mind from the start. The CANDU fuel program offers useful experience on mechanisms for technology transfer. Those funding

the R&D must decide at a very early stage how they are to obtain a return on their investment, since technology, once transferred, cannot be untransferred.

Even when technology is imported from abroad, a substantial effort in science and technology is required to absorb and support it. To have a proper understanding of the real potential, and problems, of a technology, those responsible should be working on some aspect of it; the published literature probably does not reveal either the most promising prospects or the troublesome problems if these have commercial implications.

Science-and-technology activities are also needed far beyond the innovation stage. Any new product is a delicate transplant from a laboratory, that has to be carefully nurtured if it is to establish itself in its new and competitive environment. Unless technical support is available quickly to solve unforeseen problems that arise, the transplant will be rejected and will probably not have another chance.

Most modern technologies require large inter-disciplinary programs, with expensive, efficient support services combining underlying and applied science, engineering development and design, safety, economic and market analysis, and the operation of test facilities as well as pilot and prototype plants. Universities, with other priorities, do not provide this broad capability, nor do most of the laboratories of Canadian industry, dominated as it is by multinational corporations. In these circumstances, Canada has evolved some excellent government laboratories, but the conventional wisdom is that government R&D is 'bad' while R&D in universities is 'good,' and R&D in industry 'best.' A proven and uniquely Canadian means of managing inovation all the way from fundamental research to commercialization should not be rejected for doctrinaire reasons.

Performing mission-oriented R&D in large, multidisciplinary laboratories is not an automatic recipe for success. Researchers must realize that society does not owe them a living; they must sell the benefits of their work to those who pay for it. The cost of freedom from irrelevance is eternal vigilance. Responding to demands for accountability, the government has introduced extensive and time-consuming review mechanisms that may themselves have already passed the point of optimum efficiency.

However, these reviews have the greatest difficulty assessing the quality and worth of the R&D. I would recommend the formula of backing individuals and institutions with a proven track record for fulfilling their missions. There is an unfortunate tendency to set up new laboratories for new missions, and not to exploit existing resources by modifying a mission in the light of changing circumstances.

These lessons are obvious to many of us who worked in the CANDU fuel program. They have general relevance when tested against other Canadian achievements in engineering. Now they have to be learned by our policy-makers if the Canadian economy is to benefit from the program's excellence in science and engineering.

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Notes

1. A misnomer, it was the demountable nature of the elements in the bundle that was innovative.

Fusion Materials Issues

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Abstract

Progress toward demonstrating the scientific feasibility of fusion energy is strongly governed by materials constraints. The evolution of an economically and environmentally acceptable design for power-producing fusion reactors will be subject to still further materials constraints. Three critical materials areas are addressed: 1) fast neutron damage, 2) tritium breeding, and 3) plasma-materials interactions. Canadian R & D activities in fusion materials are reviewed.

Résumé

Le progrès pour démontrer la faisabilité scientifique de l'énergie de fusion nucléaire est fortement contrôlé par les limites imposées par les matériaux. L'évolution d'une conception économique et acceptable pour l'environnement de réacteur de puissance à fusion sera le sujet d'autres limites encore imposées par les matériaux. Trois aspects critiques des matériaux seront examinés: 1) dégâts causés par les neutrons rapides, 2) génération de tritium, 3) interaction de matériaux / plasma. Les activités canadiennes en R et D sur les matériaux pour la fusion nucléaire sont examinées.

Introduction

Materials is the queen technology of any advanced technical system. The economics eventually depend upon the materials, the reliability depends on the materials and safety depends upon the materials. I assure you that before we are through with fusion, the physicists will give way to the materials engineers as being the leading lights of fusion. ... Yours is the key without which fusion power will remain forever technologically feasible, but practically useless. [1]

E.E. Kintner
Director of U.S. Fusion Program 1975–81

Few people appreciate the pivotal, indeed controlling, role which materials constraints play in the practical application of scientific ideas. Focus is generally directed to the principles of physics or chemistry which underlie the application. Thus, for most people, the essence of steam power is the expansive force exerted by matter changing state from liquid to gas. The immense task of transforming scientific principles into functioning systems is left to the engineers. Engineers, therefore, are very much aware of the fact that for every man-year of effort which went into understanding the principles of two-phase H_2O, a decade of effort was required on ferrous metal development.

So it has been with all of mankind's successful applications of science. If aircraft were still made of wood and fabric, of how much utility would the principles of aerodynamics be? Jet and rocket propulsion has utterly – and not entirely for the best – changed the face of the world we live in; these propulsion methods are conceptually simple, but quite unrealizable without high-temperature alloys, refractory liners, ablative coatings, etc. Microelectronics has possibly had an even greater impact, and while the physics involved is certainly not trivial, the materials purity and modification requirements of microelectronic components are spectacular and controlling.

Fusion energy has been termed the most difficult technical challenge ever undertaken. And this just for the scientific phase! The materials and engineering challenges of fusion have only begun to be addressed. From the analogy with other technical applications, however, we can safely assume that the materials challenges of fusion will be unprecedented. A look at a magnetic fusion reactor design supports this expectation. In close proximity – only a metre or two apart – occur temperatures higher than the centre of the sun: the core of the fusing plasma at a hundred million degrees; and temperatures lower than in deep space: the superconducting magnet dewers. The fusion environment is not only thermomechanically extreme but the materials are also required to withstand intense, high-energy neutron irradiation. The front-line com-

Keywords: fusion, materials, neutron-damage, tritium-breeding, plasma-materials-interaction.

ponents additionally suffer an intense plasma-surface interaction due to the plasma directly contacting them at about one million degrees. Many of the materials must not only withstand this harsh environment for long periods, but must carry out exotic functions as well, such as conducting electricity without resistance, or transmuting lithium into tritium – requirements which severely limit the freedom of materials choice.

Essentially, every component of a fusion reactor requires materials development if fusion is to provide economically and environmentally acceptable power. Many of these developments, fortunately, will be pursued independently of fusion, as they involve problems common to other applications. In this brief survey the focus will be on materials problems peculiar to fusion. Some of these are looming as obvious go, no-go problems; others, more prosaic perhaps, could still quietly kill off fusion as being too expensive or too complicated to attract electrical utilities.

Three Critical Materials Problems for Fusion

Among the materials problem areas which are unique to fusion, and where the achievement of solutions will be critical to the success of the entire enterprise, three stand out: 1) fast neutron damage, 2) the tritium breeding blanket, and 3) plasma-materials interactions.

Fast Neutron Damage

Eighty percent of the DT fusion reaction energy is carried off by 14 MeV neutrons. These neutrons are highly penetrating and leave the plasma zone unobstructed, only stopping deep within the surrounding solid-structure 'blanket.' In the process of slowing to thermal energy, the fusion neutrons cause *bulk* (as distinct from *surface*) damage to the first-wall lining, the vacuum structural materials, the tritium-breeding material, the magnetic coils and their support structures, insulators, windows, coolant channels, etc.

At the atomic level the damage takes two principal forms [2]:

1. During elastic collisions momentum is transferred to the lattice atoms, causing them to be displaced, thus creating voids and interstitials; also, the knock-on atoms themselves have very high velocities, causing displacement cascades.
2. Nuclear transmutations occur, such as (n, α) and (n, p) reactions, which result in the formation of gas within the lattice (helium and hydrogen), while simultaneously the elemental composition of the substrate is changed.

In principle, these damage effects are not unique to fusion, of course, since they also occur in fission systems. The fusion neutron spectrum, however, is much *harder*, i.e., more energetic, than the fission one since the 14 MeV neutrons predominate with only slight moderation. Shown in Figure 1 are neutron spectra for various sources [3]:

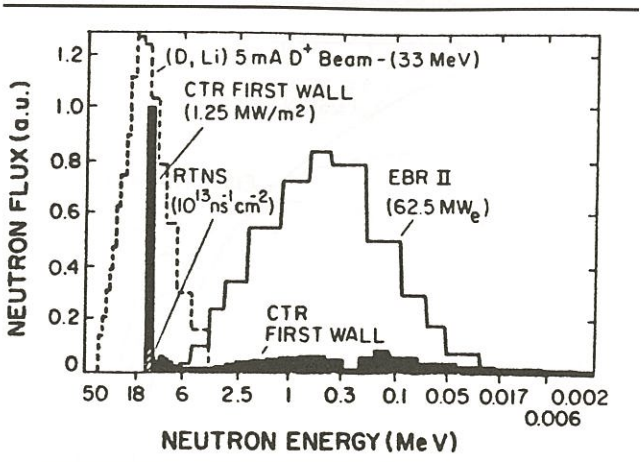

Figure 1: Typical neutron spectra for various nuclear facilities [3].

1. CTR (controlled thermonuclear reactor) first wall – (calculated) spectrum at the fusion reactor first wall for a total power loading of 1.25 MW / m^2 (for economic reasons, a total power loading of ≥1 MW / m^2 is required for a fusion reactor; for DT fusion, 1 MW / m^2 of total power corresponds to 4.4 × 10^{17} (14 MeV) n / m^2 s, and a total neutron flux density about ten times greater).
2. EBR II – a (U.S.) experimental fission reactor used for nuclear materials studies.
3. RTNS – a Rotating Target Neutron Source located at Livermore, California, which employs an accelerated 400 keV D$^+$ beam on a tritiated (titanium) target to produce 14 MeV DT neutrons.
4. (D, Li) – (calculated) neutron spectrum for a proposed D-Li stripping source, using a 33 MeV D$^+$ beam breaking up on a liquid lithium target. The neutron spectrum covers a wide energy band centred at ~16.5 MeV.

As indicated, the fusion neutron spectrum is much harder than the fission one resulting in [2, 3]:

1. higher energy primary knock-on atom, PKA, spectra (Figure 2), causing more lattice displacements (the high-energy knock-ons at ~10^6 eV doing most of the damage);
2. more nuclear transmutations (many transmutations have threshold energies too high to be significant for fission neutrons).

Examples of displacement damage and gas production rates for various materials are given in Table 1 for the fusion spectrum. In brackets are some gas production rates for the EBR-II fission spectrum. One may note the high displacement rate, indicating that each wall atom will be displaced from its lattice position many times each year. The gas production rates in atomic parts per million per year are also very high for fusion.

Synergistic processes may be important, in which case the ratio of gas production to displacement is also relevant. Table 2 indicates that this ratio differs greatly between fission and fusion neutron spectra. The

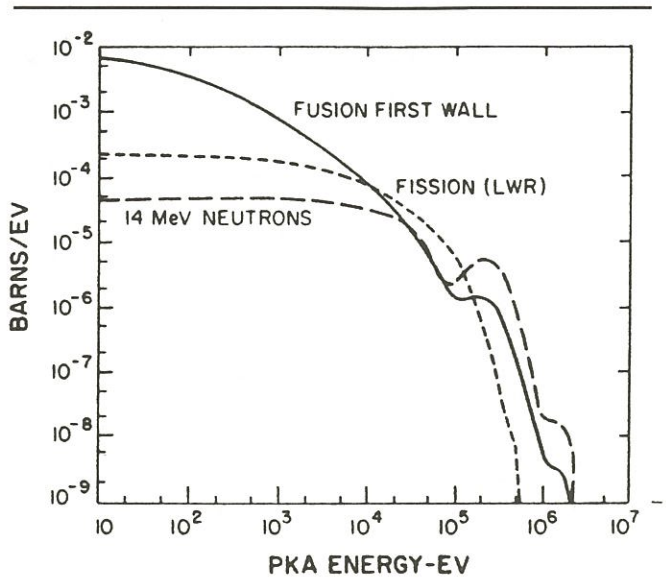

Figure 2: Primary knock-on spectra for copper in various nuclear facilities [3].

elemental composition of the wall changes rapidly (Table 3).

These damage processes occurring at the *microscopic*, or atomic level, then result in a myriad of *macroscopic* materials problems [2, 3]:

1. *Swelling*. Lattice vacancies precipitate into voids causing swelling; the gases – particularly the insoluble helium – also cause swelling due to bubble formation.
2. *Sintering* in some materials reduces available void space, causing contraction.
3. *Growth*. Carbon has enormous advantages as a first wall material (see below), but unfortunately suffers from strong neutron-induced growth, leading to run-away elongation at radiation loadings of 10–20 dpa.
4. *Embrittlement*. Stainless steel suffers virtually total loss of ductility by 100 dpa and 6,000 appm helium. Long before this point, however, the ability of a steel vacuum vessel to maintain ultra-high vacuum integrity over a surface of ~1000 m^2, and subject to thermal cycling, will have been lost.
5. *Creep*. Many of the structural components, such as the vacuum vessel, are subject to high stress and high temperature, resulting in plastic deformation over long periods: creep. The creep rupture life of stainless steel is reduced 50% by neutron irradiation. Even a small plastic deformation will make component disassembly and replacement difficult or impossible.
6. *Fatigue*. Pulsing of the magnets and vacuum system 'works' the metal, inducing fatigue and potential failure. The role of radiation in this process is little understood.
7. *Induced radioactivity*. While DT fusion 'burns clean,' i.e., the fuel ash itself is not radioactive, the neutron bombardment of the reactor walls induces radioactivity via transmutations. In the case of stainless steel, the radioactive (structural) waste to be disposed of at the time of de-commissioning the reactor would not be enormously less than the radioactive (fuel-ash-plus-structural) waste from an advanced fission system (liquid metal fast breeder) (Figure 3a), and a strong incentive therefore exists to develop more exotic metallurgies, such as that of vanadium (Figure 3b), for fusion structural components.

These neutron-related materials problems are daunting, and it remains to be demonstrated that solutions can be found which will permit economical operation of fusion reactors, with reasonable maintenance efforts and acceptable environmental impact.

In light of the seriousness of this problem, considerable thought has gone into potential solutions. The first step has been obvious for many years – although little action has yet occurred – namely, the develop-

Table 1: The Displacement Damage Rate and Gas Production Rates in Typical Fusion Reactor Materials, Based on a Time-Averaged Neutron Wall Loading of 1 MW / m^2

Alloy	*Displacement damage rate dpa / year*	*Helium production rate appm / year*	*Hydrogen production rate appm / year*
SS 316	10–12	140–240 (5)	520–540
PE-16	12–15	160–240	780
Al	15	320–360	300
V-20 Ti	11	59	230
V	12	55–60 (0.3)	105
Nb	7	20–30 (1)	80–105
Mo	7–8	45–50 (2)	95–100
Be	–	3,050	–
C	6	600–3000 (34)	–
B_4C	–	3 600	–
$LiAlO_2$	–	15,500	–
SAP (Al + Al_2O_3)	14	410	780
SiC	–	1,800	580
Al_2O_3	15	435	840

Values in parentheses are helium production rates for a fast fission reactor (EBR-II) [2, 3].

Table 2: Ratio of appm(He) to dpa for a Fast Fission Reactor (FFTF), a Thermal Fission Reactor (HFIR), a Rotating Target Neutron Source of 14 MeV Neutrons (RTNS-II), and a Fusion Reactor Wall [2, 3]

	FFTF	*HFIR*	*RTNS-II*	*Fusion reactor*
Nb	0.033	0.073	5.4	3.3
V	0.004	0.009	9.7	4.9
Mo	0.05	0.012		5.8
Al	0.11	0.31	63	24
316 SS	0.096	95	36	21

Table 3: Solid Transmutation Rates in Fusion Reactor Materials for a Neutron Wall Loading of 1 MW/m^2 [2, 3]

Original metal	*Transmutation product*	*Transmutation rate appm / year*
Al	Mg	400
	Si	40
SS 316	Mn	1200
	V	200
	Ti	50
V	Cr	130
	Ti	80
Nb	Zr	700
Mo	Tc	400
	Ru	30

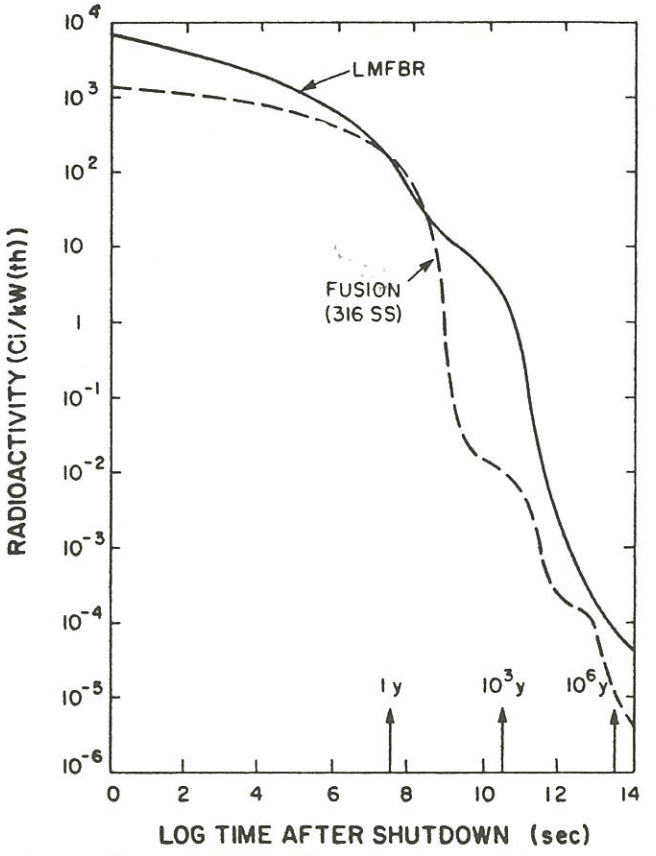

Figure 3a: Comparison of radioactive inventory for fission and fusion reactors with SS 316 structure. From: *Hafele W, Holdren JP, Kessler G, Kulcinski GL* Fusion and fast breeder reactors. Austria: Inter. Inst. Applied Systems Analysis, 1977.

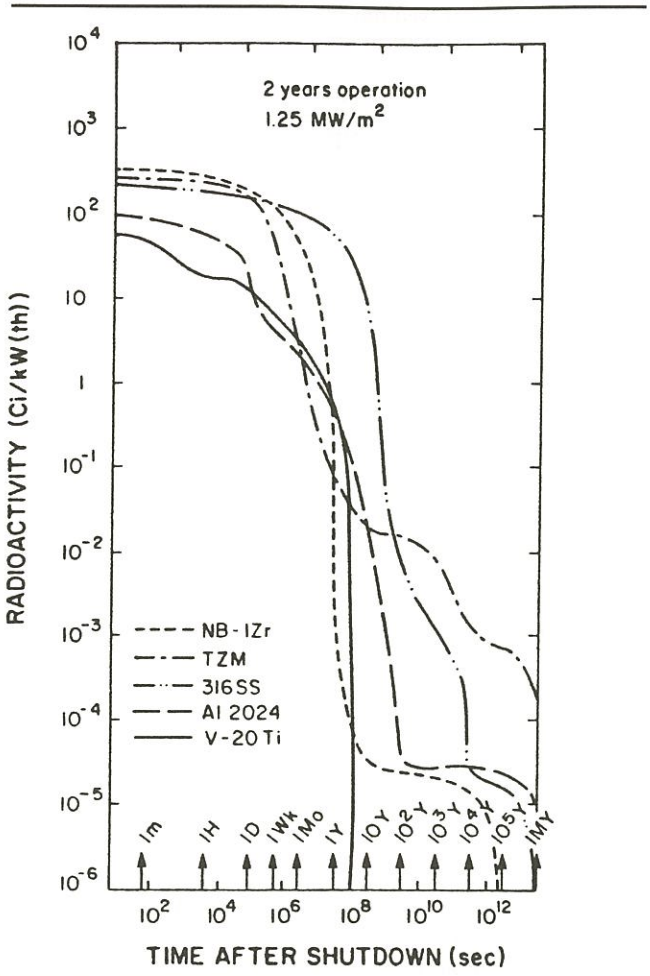

Figure 3b: Radioactivity in fusion reactor first walls after shutdown. As Figure 3a.

ment of intense 14 MeV neutron sources. Without such sources, materials specialists have to rely on test facilities such as fission reactors and ion accelerators, which only partially replicate the fusion neutron environment. In fact it is extremely difficult to achieve a satisfactory fusion neutron source, short of constructing an actual DT, high duty-cycle fusion reactor. The principal requirements of such a materials test source are: 14 MeV neutrons (or at least a very hard spectrum, $\gtrsim$10 MeV) at flux densities $\gtrsim 10^{19}$ n/m^{2}s, with as much test volume as possible $>$ 1 litre. Since a reactor first wall will experience $\sim 10^{18}$ (14 MeV) n/m^{2}s, and since accelerated testing is highly desirable, the test flux density should be at least ten times higher.

In light of these requirements, it is somewhat sobering to consider the currently available and planned neutron sources (Table 4). The RTNS II, currently the most powerful available source, falls short of the above measures by orders of magnitude. The INS (Intense Neutron Source) was initially funded by the U.S. Government ($25 million), but was cancelled in 1977 as part of general economic measures

Table 4: High Energy Neutron Sources for Fusion Materials Studies

Location	*Name*	*Reaction*	*Beam energy (MeV)*	*Target current (mA)*	*Total strength (n/s)*	*Flux intensity ($n/m^2 s$)*	*Test volume (cm^3)*	*Status*
Livermore	RTNSI	d + ^{3}H	0.4	22	6×10^{12}	10^{16}	1	Operated 1970's
Livermore	RTNSII	d + ^{3}H	0.4	150	4×10^{13}	10^{17}	1	Operating
LosAlamos	INS	t + ^{2}H	0.3	1000	10^{15}	10^{18}	1	Cancelled
Hanford	FMIT	d + Li	35	100	10^{16}	10^{19}	10	Cancelled
Needed						10^{19}	$\gtrsim 10^4$	

Table 5: Next Generation of Proposed Fusion Test Facilities [4]

	CIT U.S.	*NET Europe*	*FER Japan*	*INTOR International*
DT fusion power (MW)	300	600	300	600
Pulse length (sec)	4	1000	2000	200
Total pulses	10^4	10^5	10^4	4×10^5
Total burn time[1] (sec)	10^5	10^8	10^7	8×10^7
Total T_2 consumed (kg)	0.05	100	7	94
Fraction of T_2 bred in device	0	0.4	~0	0.6
Breeding material	–	$Li_{17}Pb_{83}$	LiO_2	LiO_2
Time-averaged hard neutron flux[2] ($n/m^2 s$)	2×10^{16}	3×10^{17}	3×10^{16}	3×10^{17}
Total exposure (MW-yr/m^2)	0.02	3	0.3	3

[1]Maximum.
[2]Total exposure averaged over 5 years.

imposed by the Carter Administration. The INS would have employed a d.c. 300 keV 1 Amp T^+ ion beam impacting on a high-speed D_2 flow. A recently considered neutron source is the D-Li stripping FMIT (Fusion Materials Irradiation Test Facility) which the U.S. has proposed to build, at a cost of ~$150 million, provided the cost is shared internationally. The FMIT proposal has been on the table for many years but has not obtained the necessary financial support.

Because of the inherently serious nature of the neutron damage problem, and because the clearly pressing necessity to develop a suitable test facility has not been acted upon, this materials problem may be the most serious one facing fusion reactor development.

The seriousness of this problem is further underlined by considering the next generation of proposed fusion machines [4] (Table 5), the ones which are intended to be the last before an actual Demonstration Reactor, DEMO. These devices will each cost $\gtrsim \$10^9$. Nevertheless, even these impressive facilities will not produce the hard-neutron radiation exposure rates (~10^{19} n/m^2/s), nor even the integrated life-time exposure (~20 MW-yr/m^2) required for fusion reactor development.

The Tritium Breeding Blanket

Tritium is not found to any significant degree in nature and must be manufactured for DT fusion reactors. The most probable method is to utilize the neutrons produced by the DT reaction itself to bombard a lithium-containing blanket surrounding the reactor, breeding tritium via the reaction ^{6}Li (n, α) T (^{6}Li constitutes ~7.5% of natural lithium, ^{7}Li ~92.5%). The blanket must also incorporate a heat-removal system which is capable of handling the fusion power, 80% of which is deposited directly throughout the bulk of the blanket by the neutrons.

'Blanket issues are highly interrelated to material issues and material uncertainties can strongly affect the feasibility of a blanket concept' [5]. The blanket must contain a wide variety of materials, performing different functions (Table 6), with mutual compatibility [5]:

1. *Breeder material* must contain Li in sufficiently high atomic density to achieve a breeding ratio of at least unity, i.e., one tritium atom created for each 14 MeV neutron entering the blanket. Three categories of breeder material are being considered:

Table 6: Primary Blanket Options [5]

	LIQUID BREEDER			SOLID BREEDER	
Breeder	Li	LiPb	Flibe	Li_2O	Ternary ceramics
Coolant	Li, He	LiPb, He, H_2O	Flibe, He	He, H_2O	He, H_2O
Structural materials	Ferritic steels Refractory alloys			Ferritic steels Austenitic steels	
Multiplier	None, beryllium			Beryllium	

(a) liquid lithium;
(b) solid lithium compounds such as LiO_2, $LiAlO_2$, Li_7Pb_2, and $LiSiO_2$; and
(c) molten lithium salts such as LiF-BeF_2 (FLIBE).

2. *Coolants* which remove the heat and possibly also the tritium. Candidate materials: H_2O, He, liquid Li, liquid Na, molten salt, and fluidized LiO_2.
3. *Neutron multipliers* are likely to be required to achieve a breeding ratio ≥ 1. Candidates: Be, Pb.
4. *Neutron moderators* are likely to be needed to enhance the $^6Li(n, \alpha)$ T reaction, which uses thermal neutrons. Candidates: H_2O, C.
5. *Structural material* for the coolant channels, etc., must be compatible with the foregoing exotic materials. Candidates: austenitic, ferritic, and martensitic steels; Fe–Cr–Ni super alloys; and reactive and refractory metals such as V.

Liquid lithium is, in many respects, the ideal breeding material: it has a high Li density and thus a good breeding ratio (Figure 4). It will therefore probably not require ^6Li isotope enrichment, nor a neutron multiplier; liquid Li is an excellent heat-transfer fluid and can be used simultaneously as coolant; the tritium is naturally removed from the blanket for extraction. Unfortunately a number of serious deficiencies may rule out this simple option:

1. Corrosion of the structural materials by the lithium, combined with the large mass transfer, is likely to result in the plugging of valves, etc., with (radioactive) corrosion products.
2. Lithium reacts readily with O_2, N_2, and H_2O, and the risk of a lithium fire releasing the tritium (in the biologically hazardous oxide form) is possibly unacceptable.
3. Lithium is electrically conducting and a liquid lithium coolant would experience pumping losses due to magnetohydrodynamic effects in circulating through the reactors' magnetic fields.

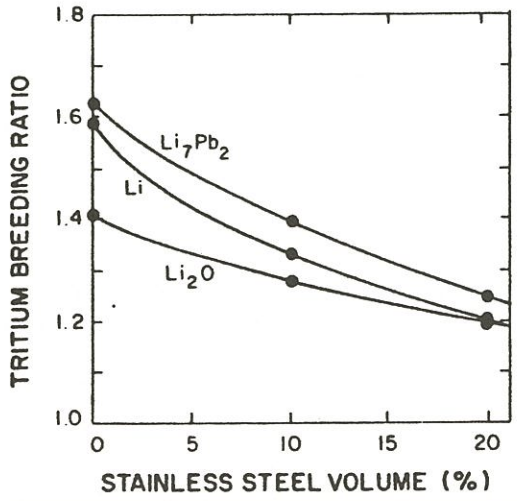

Figure 4: Attainable breeding ratio vs volume per cent stainless steel structure [2].

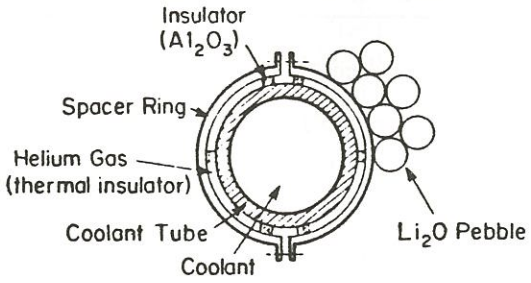

Figure 5: INTOR breeding blanket concept [6].

4. The solubility of tritium in lithium would make it difficult to keep the T-content down to the ppm level required for an acceptable tritium inventory in the blanket: ~1 kg T.

Solid breeders have therefore attracted attention, although this option also encounters materials problems:

1. Tritium self-sufficiency. With lower atomic Li content in solid breeders, the achievement of an adequate breeding ratio is jeopardized. Considering that not all of the periphery is usable for breeding, owing to the presence of structural components (Figure 4), this problem is a serious one. A multiplier is probably necessary, and since the desired neutrons are slow, direct mixing of, e.g., Be and the breeder is desirable. This, however, raises problems of chemical compatibility.
2. Tritium inventory and recovery. The tritium must be rapidly removed from the solid breeder and convected out of the blanket in order to minimize tritium inventory. The transfer of tritium from solid to coolant or purge channel is governed by the largely unknown processes of diffusion, solubility, and surface recombination in materials operating at conditions of elevated temperature, neutron irradiation, and surface chemical contamination.

The INTOR design (International Tokamak Reactor – a conceptional design of the next generation Tokamak experiment by the world's leading fusion countries) [6] calls for a net breeding ratio of >0.6, with a blanket coverage of ~60% and a thickness of ~0.5 m. Since INTOR is an experimental device, not a power reactor, the tritium fuel costs to make up the missing 40% are not prohibitive (although they could still run to ~$1 billion, since an external supply of 4–8 kgm T / year is required, at a cost of $10–100 million / kg). Experimental access limits the blanket coverage, and the bore of the magnets limits the blanket thickness. The latest INTOR blanket option based on LiO_2 is described in Figure 5 and Table 7. The tritium would be carried off in a helium purge circuit, separate from the H_2O coolant circuit.

Plasma Materials Interactions (PMI)

The materials isues associated with the interaction of the plasma with the wall components are the most

Table 7: Parameters for INTOR Reference Li_2O Blanket [6]

Breeder material	Li_2O
Thickness of breeder region (cm)	50
Enrichment of ^6Li (%)	30
Temperature limits for breeder (°C)	410–800
Purge stream for tritium recovery	He (0.1 MPa)
Coolant	H_2O
Coolant temperature (°C)	<280
Structure	316 SS
Maximum structure temperature (°C)	350
Neutron multiplier	Pb
Neutron multiplier thickness (cm)	5
Tritium-breeding ratio	0.65

studied and best understood of the fusion materials problems. This is simply the result of the fact that while the 14 MeV neutron damage and breeding blanket materials questions relate to future machine operation, PMI occurs – indeed dominates – the operating properties of current experimental devices.

Central plasma temperatures, even in current machines, can exceed 10^8 K. Fortunately, the insulating effect of the magnetic field supports a strong temperature gradient across the plasma. Nevertheless, edge plasma temperatures – i.e., of the plasma in actual contact with the walls – are extremely high: one million degrees. Not surprisingly, this results in a strong plasma surface interaction.

PMI leads to erosion of the surface due to a number of processes. Generally, the most serious is *physical sputtering* [7], which is the result of the simple process of momentum transfer from the fast-moving plasma particles to atoms in the solid lattice, knocking them out. The magnitude of this wall erosion process is indicated by the experimentally measured *yield* (Figure 6), which is dependent on the elemental composition

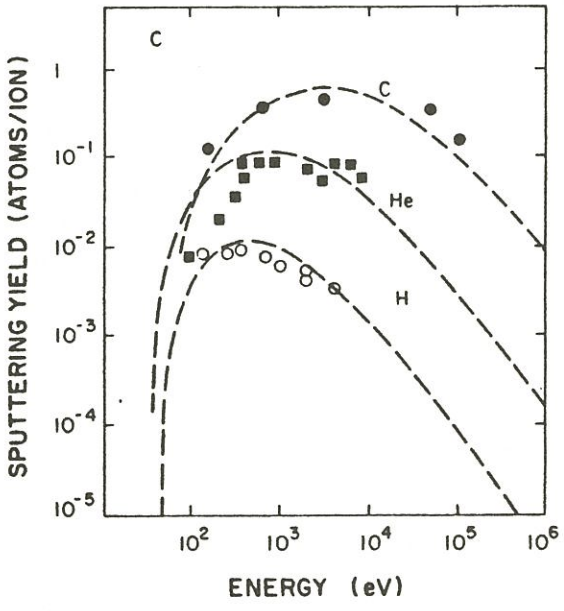

Figure 6a: Sputtering of carbon by H, He, C ions [7].

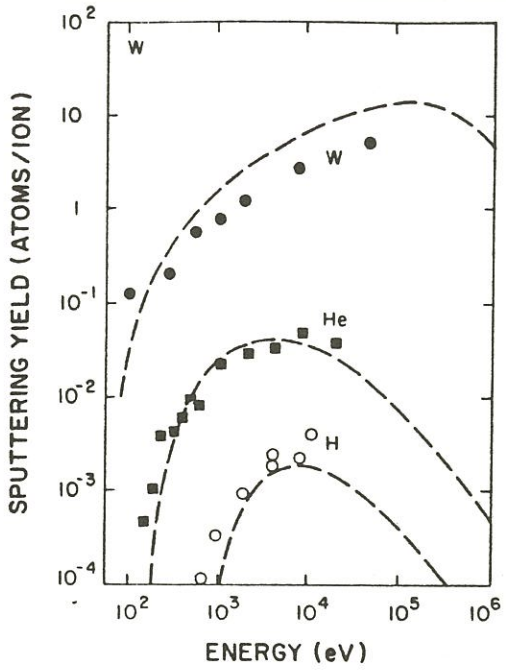

Figure 6b: Sputtering of W by H, He, W ions [7].

of projectile and substrate, and the projectile energy. As indicated, removal rates can exceed ~1 atom removed per ion for impacting energies of a few hundred eV, typical values for an edge plasma of temperature $\sim 10^6$ K. Initially, only the hydrogenic ions cause sputtering. However, the sputtered impurity atoms is quickly ionized upon entering the plasma, and in steady-state returns to the solid surface at the same rate, causing *self-sputtering* by its own impact.

Since impurity ions carry more momentum than hydrogenic ones, their sputtering yield is higher (Figure 6). For steady-state conditions, the impurity removal rate Γ_I [atoms/m^2] is the sum of the hydrogenic and self-sputtering rates: $Y_H\Gamma_H$ and $Y_I\Gamma_I$, respectively; Y_H and Y_I are the hydrogenic and impurity yields; and Γ_H is the hydrogenic flux. Therefore,

$$\Gamma_I = Y_H\Gamma_H + Y_I\Gamma_I \tag{1}$$

or

$$\Gamma_I = \frac{Y_H\Gamma_H}{1 - Y_I}. \tag{2}$$

Thus, self-sputtering is not additive but multiplicative; furthermore, as $Y_I \rightarrow 1$, a catastrophic runaway can occur.

High erosion rates are unacceptable for at least three reasons:

1. The wall material, present in the plasma as an unwanted impurity, thermally radiates away the plasma heat content, preventing net energy production.
2. The wall wears out and its frequent replacement is not compatible with economic plant operation.

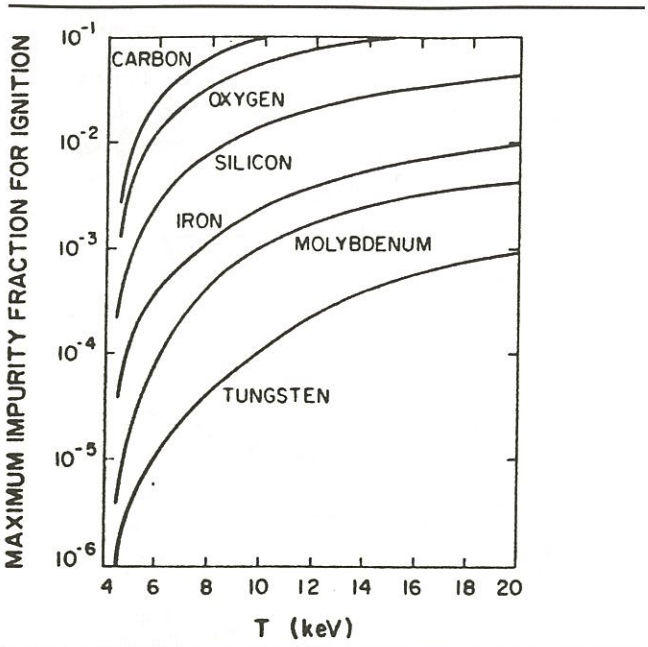

Figure 7: Maximum tolerable impurity fraction for ignition of a DT plasma vs. temperature for various impurity species [8].

3. Gasified impurities such as methane enter the exhaust / clean-up system, which has the task of extracting the highly valuable, unburnt tritium and returning it to the plasma in a completely pure form. Such impurities create one of the biggest problems in the re-cycle loop.

Only the first of these problems is a serious one in currently operating experimental devices. The other two will become more serious as reactor conditions are approached. The plasma contamination problem is perhaps the greatest stumbling block to the demonstration of net fusion power. While the fusion 'flame' burns intensely hot, it is a remarkably vulnerable flame. Were one simply to blow into a magnetic fusion reactor the size of a large room, the fusion flame would be extinguished! (See below.) Eventually, this vulnerability of a fusion plasma will be appreciated as a valuable safety feature: any departure from designed operating conditions of a fusion power reactor will increase the plasma-materials interactions, contaminating and extinguishing the reaction. For the present, however, this vulnerability is an enormous obstacle.

All bodies radiate heat to their surroundings. The hotter the body, the shorter the wavelength of the radiation. Human beings radiate in the infra-red. Fusion plasmas radiate X-rays. This radiation, fortunately, is not black-body, which is so intense that it would result in a hopeless prospect of net energy production at 10^8 K. Rather, the radiation is due to electrons colliding with positively charged nuclei in the plasma. The power of this collisional radiation varies as Z^2 – the charge on the nucleus squared. For this reason, high-Z impurities are particularly damaging (Figure 7). This figure indicates [8] the maximum permitted concentration of various impurities in a DT plasma, which will just permit ignition – i.e., the point at which the self-heating of the plasma by the fusion reaction itself equals the radiative cooling rate. This figure demonstrates how risky it is to employ materials such as tungsten, whose high melting point, good heat-conductivity, and low sputtering rate (Figure 6b) may not be adequate compensations for the low permitted concentration in the plasma (Figure 7). The attraction of low-Z wall materials such as carbon is also evident from Figures 6a and 7. Twenty years ago refractory materials such as tungsten were widely employed in fusion experiments. However, as the seriousness of the impurity problem became clearer, systems were changed to stainless steel and then to carbon. Most operating fusion devices today operate with large quantities of carbon protecting the first walls; indeed, with tons of graphite in the largest machines, such as TFTR at Princeton. Going a step further, beryllium has been experimented with on a small scale in the U.S. machine ISX-B at Oak Ridge [9] and will be tried in the world's largest fusion device, JET, in the next few years. Boron coatings and liquid lithium layers have also been proposed, which, of course, brings one to the end of that line!

With regard to blowing out a fusion flame, let us consider a fusion plasma of typical characteristics: volume of $\sim 100\,m^3$, temperature of $\sim 10^8$ K (corresponding to ~10 keV particle energy), density of $\sim 10^{20}$ D and T per m^3. This yields a total fuel content of $\sim 10^{22}$ D and T ions. With a human breath of about 1 litre of N_2/O_2 ($\sim 2.7 \times 10^{22}$ N_2/O_2 molecules), the plasma contamination would greatly exceed the permitted ~6% level for oxygen (Figure 7).

The direct prevention of ignition by radiative cooling of the *core* plasma, where the fusion reactions occur, is not the most serious problem caused by impurities in current devices. Two other effects are more serious:

1. Fuel dilution. Since the impurities create high-Z ions they fill the plasma with many extraneous electrons for each impurity ion. Each electron adds just as much to the plasma pressure as a D / T fuel ion, and since the confining pressure exerted by the magnetic field, $B^2/2\mu_0$, is limited, the result is fuel dilution. Since the fusions power P_F varies as $n_D n_T$, i.e., n_{fuel}^2, a small impurity fraction reduces P_F enormously, even for low-Z impurities. For example, 5% carbon reduces n_{fuel} by ~30%, hence P_F by ~50%.
2. Density limit. Finite magnetic pressure aside, one would think that n_{fuel} could be raised to any desired level simply by puffing more D_2 or T_2 into the plasma. Unfortunately, an upper density limit occurs for stable operation of the plasma, and at plasma pressures only a small fraction of the available magnetic pressure: ~1%. The cause of this serious limit is not completely understood, but is almost certainly due to impurities, since purer plasmas have higher density limits (Figure 8).

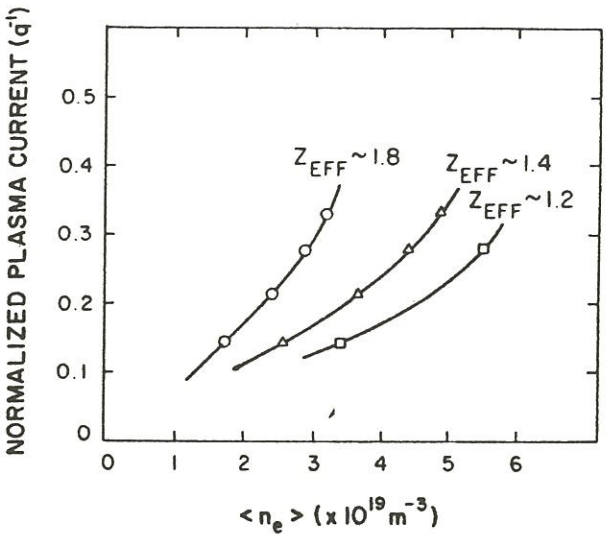

Figure 8: Maximum plasma density achievable at a given plasma current is dependent on impurity level. For pure DT plasma $Z_{eff} = 0$. Calculated result for JT60. *Hirayama T*, et al. J Nucl Mater 1986.

For energy break-even, the fuel density must satisfy the Lawson Criterion:

$$n_{fuel}\,\tau_{confinement} \gtrsim 10^{20}\ [s/m^3], \qquad (3)$$

and thus both fuel dilution and the density limit strike directly at the heart of the entire fusion enterprise.

Because of the critically serious nature of this *near-term* materials problem, fusion research has had to pay greater attention to materials issues than is usually the case for a technology still in its scientific-feasibility phase. The official four objectives of the JET Project [10], for example, given considerable prominence to this area:

scaling of plasma behaviour,
plasma-wall-interactions,
plasma heating,
α-particle behaviour.

The magnitude of the PMI is obviously intimately related to the effectiveness of the magnetic confinement. Perfect confinement would result in no PMI! In reality, of course, the plasma-confining magnetic bottle is quite leaky. Even in the biggest machines, which have the best confinement, the average particle lifetime in the plasma is less than one second. Thus every second, or faster, the entire plasma content of $\sim 10^{22}$ ions strikes the walls. In principle, this load could be distributed evenly over the entire wall surface $\sim 100\ m^2$. For various reasons, a more controlled PMI is desirable, and it can be arranged for the magnetic field to channel most of the plasma outflow to special surfaces, called *'limiters'* or *'divertor plates'* (Figure 9). While the latter are only of area $\lesssim 1\ m^2$, one can afford to make them of special materials and can replace them more readily than the entire wall. The wall still catches

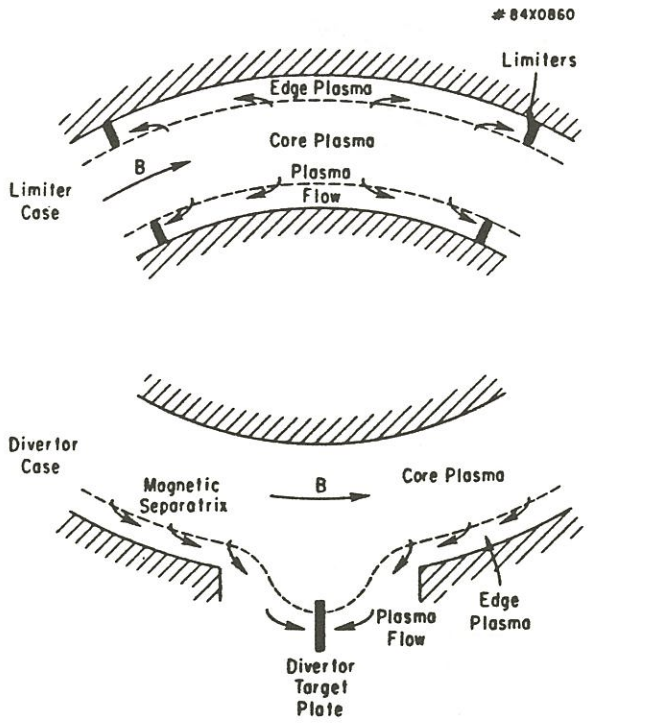

Figure 9: Schematic of limiter and divertor geometry. PMI at divertor target is located further from main plasma than for limiter target.

some action, but orders of magnitude less (per square metre). Thus the PMI occurs at two quite different levels of intensity:

1. Limiters and divertor plates receive 10^{22}–10^{24} ions / m^2s, i.e., ~amps / cm^2.
2. Walls receive 10^{19}–10^{21} ions / m^2s, i.e., ~milliamps / cm^2.

The total incident flux (ions / s) is about the same for the two types of surfaces, and thus the two zones pose about equally important materials problems. The nature of the problems differ, of course. The erosion rate of the limiters, for example, can be quite spectacular. Taking $Y_{D\,on\,C} = 0.1$ for example (Figure 6a), and $\Gamma_D = 10^{23}$ (D^+ / m^2 s), leads to an erosion rate of 10^{22} C / m^2 s (~0.3 mm / hour), just due to the D^+ impact alone. Since much of this removed carbon may be in the form of methane, the impurity load on the re-cycle loop could be unacceptable (operation of cryogenic isotope separators in the loop requires impurity levels $\lesssim 1$ ppm). Fortunately, the impurity atoms and molecules upon entering the plasma are ionized and are then caught up in the hydrogenic flow to the surfaces. While this increases the sputtering, it also leads to *re-deposition*. The latter process is clearly of critical importance if acceptable erosion rates and fuel re-cycle loop conditions are to be achieved. The effectiveness of impurities in contaminating the *centre* of the plasma is also greatly affected by the rate at which this

impurity turn-around occurs. The processes involved in this complex pattern are only starting to be understood. As to the materials side of this, it is evident that whatever surface is initially introduced into the device, it will quickly become a re-deposited surface, with its own unique properties. It is therefore important to carry out materials tests on re-deposited materials, and to create these materials in the same way as is actually encountered in a working device.

The foregoing is focussed on only one aspect of PMI – erosion. A number of other important effects also arise:

1. Hydrogen permeation. Hydrogen (including D and T) can permeate through virtually all solids. This results in embrittlement and swelling, and in the case of (radioactive) tritium, in a containment problem. This problem is enormously exacerbated when the hydrogen arrives at the front surface in an already dissociated form (atoms and ions rather than molecules), causing effective permeation rates to increase by many orders of magnitude. Since the fusion wall receives hydrogen in the pre-dissociated form, the problem of T-contamination of the coolant circuit is a serious one.
2. Hydrogen retention. As above, but the problem is now one of keeping the tritium inventory of the plant to an acceptably low level.

In addition, there are all the thermomechanical materials problems associated with cyclical, high heat-loading in a radiation environment to be dealt with.

From the above, it is obvious that PMI problems are major ones. Fortunately, these problems are being addressed and solid progress is being made. As an example, in initial applications [11] of radio frequency wave heating to fusion experiments, even brief bursts ($\lesssim$0.1 sec) of low power ($\lesssim$0.2 MW) caused such a cloud of impurities to come off the RF antennae, that the plasma was actually cooled! Today, the RF antennae on JET inject ~10 MW of power for seconds-long pulses without appreciably changing the impurity level of the plasma [10]. While many PMI challenges lie ahead, particularly as break-even conditions are approached and α-particles contribute to the PMI, etc., a reasonable basis for optimism exists.

Canadian Fusion Materials Research

Canadian research on fusion materials is concentrated on breeder materials and plasma materials interactions. Two projects in these areas – the Chalk River Fusion Blanket Programme and the Tokamak de Varennes – have been extensively reported on and therefore only a brief review is included here. Other fusion materials R & D work underway at the Ontario Hydro Research Laboratory, McMaster University, and the University of Toronto Institute for Aerospace Studies are described.

Breeder Materials R & D at CRNL

A Fusion Blanket Programme was launched during the first five-year programme of the Canadian Fusion Fuels Technology Project (CFFTP), with joint funding by the CFFTP and AECL. The Blanket Programme utilizes a major fission neutron irradiation source at the Chalk River Nuclear Laboratories of AECL (CRNL) – the NRU research reactor. Focus is on irradiation tests of solid breeder ceramics: $LiAlO_2$ (jointly studied with France), $LiAlO_2$ (with Japan), and Li_2O (with the U.K.). Both vented (CRITIC Project) and unvented (CREATE Project) capsule irradiation tests are performed to establish the release of tritium from the ceramic and to evaluate breeder / cladding interactions, swelling, cracking, etc. The (fission) neutron damage reproduces the displacement rate of a fusion reactor near the blanket rear, but is short by two orders of magnitude for the front (first wall).

With regard to the fabrication of breeder ceramics, the CRNL Programme is focussed on the fabrication of $LiAlO_2$ microspheres by various novel techniques, such as an organic sol-gel process.

Lithium (^{6}Li) isotope enrichment techniques are under study with the purpose of evolving new, environmentally attractive approaches.

Liquid breeder / coolant options are under investigation, including liquid metal breeders, and organic coolants, the latter based on the twenty-year operating experience of AECL's WR-1 organic cooled heavy-water-moderated reactor.

A major alternative to the conventional blanket approach for producing tritium is a new concept based on a water-cooled blanket employing lithium salts dissolved in light or heavy water. This novel approach is the object of a joint study by CFFTP in collaboration with Grumman and Rensselaer Polytechnical Institute.

Materials Studies on the Tokamak de Varennes

The scientific programme of the Tokamak de Varennes is centred around the study of impurity transport and control, plasma-wall interactions, the effect of long pulses, and materials studies. The device will have a divertor system (Figure 9), high-speed pumping of the vacuum vessel and advanced diagnostics – all intended for a strong programme in PMI. The work will include studies of:

1. the equilibrium (long pulses) between the edge plasma and the wall;
2. the thermal fatigue of the materials exposed to large heat deposition (combination of long pulses and a tight, high-energy flux divertor configuration);
3. sputtering, redeposition, and net erosion;
4. coatings, such as titanium-carbide-coated carbon tiles;
5. divertor operation and impurity accumulation;
6. recycling of the hydrogenic species in long pulse discharges, where plasma-solid equilibrium is approached; and

7. the characterization of materials for the limiters and divertor plates.

The Fusion Materials Group at the Institut National de la Récherche Scientifique (INRS) is pursuing two supporting areas of activity:

1. Development of analysis techniques. A 400 keV accelerator is used for the nuclear micro-analysis of H and D in samples. Detection at the ~1% atomic (H) level, with depth resolution to 100Å, has been demonstrated; laser desorption mechanisms are studied in order to make this into a quantitative tool.
2. Materials development and characterization. This work is carried out in collaboration with the Industrial Materials Research Institute at Boucherville. Thick (300 μm) plasma-sprayed coatings have been demonstrated to have valuable properties: thermal shock resistance if deposited under argon atmosphere; absence of hydride formation or blisters under hydrogen implantation; acceptable porosity and gas content.

Fusion Materials Studies at OHRD

Fusion materials work at the Ontario Hydro Research Division (OHRD), part of the CFFTP Project, focuses on the development and study of materials to be used in ancillary systems, such as tritium storage and purification systems, a secondary containment detritiation system, and a reactor exhaust treatment system. The areas of activity are therefore development of tritium gas handling techniques, hydrogen permeation, and hydrogen gas interaction with hydride formers.

Hydriding studies of uranium and titanium sponge have led to the development of 5 KCi and 500 KCi storage beds for tritium.

Studies on the removal of free or chemically bound tritium from process streams, using zirconium alloys, demonstrated that concentrations as high as 7 Ci / m^3 could be removed from inert gases, indicating that routine tritium scavenging from inert gas process streams is viable.

In conjunction with the Jülich Textor Group (Federal Republic of Germany), field work studies on the Textor Tokamak have led to the development of a permeation probe to monitor atomic hydrogen fluxes to the wall. Related to this, the performance of a hydrogen permeation pump based on composite membranes is under investigation.

A tritium test facility at the 1 KCi level is being commissioned at OHRD for fusion and fission studies.

Fusion Materials Studies at McMaster University

Fusion materials research at McMaster University is part of the CFFTP and focusses on tritium in materials. A project is also underway on neutron damage.

Special coatings are under development to act as tritium permeation barriers, specifically SiC and Al_2O_3 deposited on Ni substrates via various techniques, such as Al evaporation followed by anodization. The diffusivity and solubility of tritium in the materials can be monitored continuously by measuring the T content using a high-energy tandem accelerator and nuclear reaction analysis (NRA).

The diffusion of D and T in materials such as Ni is studied subject to temperature gradients and radiation damage. The T is ion-implanted in thin foils, sandwiched between permeation barriers of Al_2O_3. The diffusion is thus constrained to be lateral and is monitored by NRA as a function of time and sample temperature.

A neutron damage project is being carried out in collaboration with the U.S. Sandia Laboratory, who load 316 SS samples with high quantities of T at 20,000 psi, 300°C. The T decays to ^3He, creating levels of up to 500 appm He. The resulting He bubble formation results in the same fatique and creep problems associated with 14 MeV neutron exposure. The samples are fatigue-tested at McMaster under monotonic and cyclic loading.

Plasma Materials Interaction Studies at the University of Toronto

The fusion materials research at the University of Toronto Institute for Aerospace Studies (UTIAS) is focussed on PMI and includes both laboratory testing and field work on fusion devices outside Toronto. The UTIAS work is part of the CFFTP.

World-wide, perhaps half of all PMI research is carried out using relatively small-scale test facilities, such as ion accelerators, to simulate PMI – but in highly controlled conditions. The other part of PMI research is carried out directly on large experimental fusion devices principally tokamaks. PMI research on working devices has the obvious advantages of focus and relevance, but suffers from complexity, experimental inflexibility, and difficulties in interpreting results, since many processes occur simultaneously. The optimal approach, therefore, appears to be to combine these two avenues of research.

Controlled laboratory studies of plasma materials interactions performed at UTIAS include the following:

1. Chemical erosion of graphite and other first wall materials (e.g., amorphous hydrogenated carbon films produced in the TEXTOR tokamak) using a high-current, low-voltage hydrogenic ion accelerator. The released hydrocarbons, principally methane, are detected mass-spectroscopically. Simultaneously, the substrate can be bombarded by known, controlled fluxes of energetic ions, neutral hydrogen atoms, and electrons to investigate *synergistic* erosion – reproducing the combined exposures to which actual working surfaces are subject. The edge structures in fusion devices are bombarded by about equal fluxes of energetic ions H^+ (100's eV energy) H^0 atoms (~1 eV), and electrons (~20–100 eV). By itself, the H^0 is much less

reactive than the H^+; however, a strong synergistic effect has been discovered for combined H^+ and H^0 exposures. The addition of electrons to the bombarding species does not significantly affect the reactivity of graphite. The UTIAS synergistic studies are carried out collaboratively with the Institute for Chemistry Group, Jülich, Germany.

2. Current research on graphite erosion also includes angle-of-incidence dependence of physical sputtering due to low energy (20–100's eV) ions (H^+, D^+, He^+), and radiation-enhanced sublimation at elevated temperatures ($1{,}200 < T < 2{,}200$ K).
3. Hydrogen permeation is studied using a permeation membrane facility with *in situ* surface analysis of contaminants by Auger Spectroscopy. The level of surface impurities strongly influences the permeation and recombinative-release (fuel re-cycle) of hydrogenic species. A small change in carbon impurity levels on the surface of Pd, from ~0.4 to ~0 monolayer, is found to increase the rate of hydrogenic recombinative-release by more than two orders of magnitude. Since all metal surfaces in a working device become contaminated with carbon, it is clearly important to employ materials data for the relevant, i.e., contaminated, state. Permeation studies with bilayer materials for potential use as a unidirectional hydrogen pump are carried out collaboratively with OHRD and the Jülich Textor Group.
4. Hydrogen inventory effects are studied using thermal desorption and laser-induced desorption of wall samples exposed to pre-dissociated hydrogenic species. Such studies on carbon, carried out jointly with the U.S. Sandia Laboratory (Albuquerque), revealed that the highly porous nature of carbon provides additional pathways along porous internal surfaces – beyond normal diffusion – for hydrogen penetration and retention. Both JET and TFTR have recently experienced unusual plasma-pumping effects, which may be related to carbon porosity.
5. A low-level tritium laboratory (~1 Ci) has just been commissioned in which impurity production, carbon erosion, hydrogen permeation, and inventory effects will be studied using the isotope of interest – tritium – rather than simulating it with protium or deuterium, as hitherto.

Field work studies are carried out on U.S. devices – principally TFTR at Princeton – and European devices, JET and the UKAEA's DITE tokamak, both at Culham, Oxfordshire. Studies include the following:

1. Chemical erosion of graphite under actual operating conditions. A special highly instrumented graphite limiter inserted into DITE is examined spectroscopically as its temperature is varied, in order to test for chemical erosion (which is known to have a strong temperature dependence from accelerator studies). The inferred chemical erosion rates for the DITE experiment were found to be low, in agreement with laboratory beam simulation studies, which indicate reduced yields at high incident current densities, low energies, and in the presence of metal surface contaminants.
2. An inverted-geometry limiter installed in DITE is being tested for its capability of reduced impurity contamination of the plasma. The shape of conventional limiters tends to project sputtered atoms toward the plasma; inversion projects atoms toward the wall. It remains to be demonstrated, however, that inversion is consistent with the other PMI constraints (see 4 below).
3. Edge plasma measurements in JET and TFTR using Langmuir probes indicate that PMI conditions differ in these large machines from earlier ones, in that higher plasma temperatures prevail; i.e. $\sim 10^6$ K rather than the $\sim 10^5$ K more typically encountered in smaller tokamaks. Due to the greater distances between limiter surfaces on these big machines, there is more opportunity for various PMI-related effects to arise than in smaller devices; e.g., radiative cooling of the edge plasma, local ionization re-cycle of the hydrogenic fuel, etc. In addition, the re-deposition patterns are clearer since the long pulses lead to greatly increased total exposure times. With individual discharges on TFTR and JET lasting up to 20 seconds, the total edge data accumulation exceeds that from all previous work on other fusion devices.
4. A Monte Carlo computer code has been developed to track the fate of impurity atoms as they are sputtered from the limiter, enter the edge plasma and ionize, wander off into the central plasma radiating energy, and eventually return to the limiter to sputter and re-deposit. Various limiter geometries, such as the inverted one above, were tested out in a search for optimal designs subject to the constraints of a) distributing the heat load, b) minimizing the impurities reaching the main plasma, c) minimizing net erosion via homogeneous distribution of re-deposited ions, and d) minimizing the build-up of trapped tritium by the process of co-deposition (impurity burial of tritium).

   The co-deposition process was only recently identified, on JET, as possibly the major mechanism governing the tritium inventory tied up in the limiters; each carbon atom deposited buries ~0.5 of a hydrogen atom. The co-deposition and re-deposition processes are intimately related, and since all surfaces in working devices, regardless of their installed materials properties, end up in the re-deposited state, it is essential to understand fully and to control these materials-modification processes.

Conclusions

The achievement of an environmentally and economically attractive energy option based on the fusion process will depend on the solution of a formidable array of materials problems. Three of the most critical relate to 14 MeV neutron damage, tritium breeding, and plasma-surface interactions. Canada has now become a significant participant in the world fusion effort and has undertaken substantial initiatives in two of the material areas: tritium breeding and plasma-surface interactions.

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The AC Device for Repositioning of Garter Springs in CANDU Reactors

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Introduction

Proper spacing between the pressure tube and the surrounding calandria tube of the fuel channels in CANDU reactors is maintained by spacers (garter springs). It was established that many garter springs are no longer in their correct positions in Ontario Hydro's operating reactors. Analysis has confirmed that the displaced springs can lead to contact between the pressure tube and the calandria tube, resulting in eventual pressure tube failure, such as the one which occurred 1 August 1983, in channel P2G16 of Pickering 'A' Nuclear Generating Station. To prevent this, a novel AC device was developed at Ontario Hydro Research Division to reposition the garter springs without any harmful effect on the fuel channel integrity.

This paper describes the development of this device. The development started with the recognition of the technical problem, invention [1] of the AC device and proof of principle of operation, theoretical analysis and engineering development, integration into the SLAR system, field trial of the preproduction tooling, and, finally, manufacture of devices for field implementation of the SLAR program.

Recognition of The Repositioning Problem

The problem can be briefly summarized as follows: 1.) access to the garter spring; 2.) moving the garter spring.

Since the only permissible access was stipulated to be from inside the pressure tube, in order to produce a physical force to move the garter springs, there is need to establish some sort of coupling between the garter spring and a device that can exert force on the garter spring. In the case of electromagnetic forces, there are two fundamental difficulties in achieving significant interaction with the garter spring:

1. electromagnetic shielding of the pressure tube (as shown in Figure 2);
2. unfavorable material properties of the garter spring in terms of very low electrical conductivity, non-magnetic characteristics, and similarity to the pressure tube material.

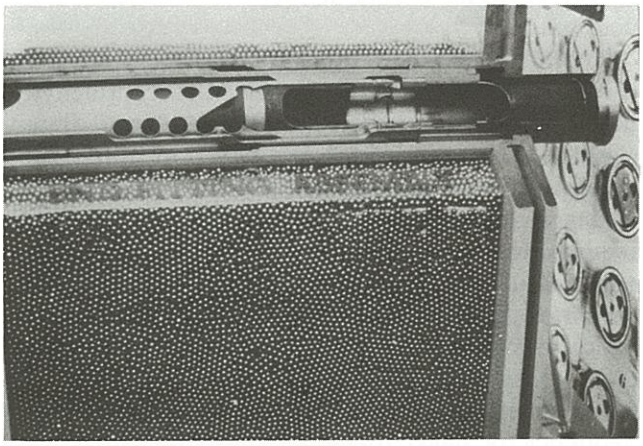

Figure 1: Sectional view of a reactor channel. A garter spring is shown on the extreme right.

An individual solution to each of the above-mentioned difficulties could be achieved by decreasing or increasing the frequency to solve 1. and 2., respectively.

By using too high a frequency, the shielding effect becomes dominant, and in extreme cases no significant field diffuses through the pressure tube wall. A crude

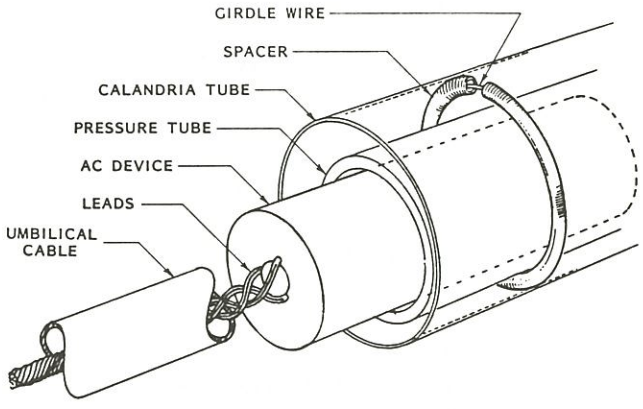

Figure 2: AC device in the fuel channel arrangement.

way of dealing with the extensive shielding effect at high frequency is simply to apply a quantitative rather than qualitative approach, by increasing the operating current. An enormous current is required to obtain some small fraction of the total field in the vicinity of the garter spring. However, there are strict limitations imposed by the fuel channel and the electromechanical system as a whole, which have to withstand the stresses associated with these enormous currents.

Resorting to low-frequency and very high currents will not solve the problem either, for theoretical and practical reasons. At low frequency, the garter spring is essentially resistive, making the electromechanical energy conversion process highly inefficient or impossible.

Another way to view this limitation is from the field theory point of view [2, 3]. Low electrical conductivity, the nonmagnetic property, and the very small dimensions of the garter spring cross-section (girdle wire loop) will result in a very small critical diffusion time constant, and a high critical frequency would be required for any interaction with the garter spring. If one uses a low-frequency source, for which the penetration depth ('skin effect') is much greater than a critical dimension (cross-section of the girdle wire), then, quite simply, the field will not notice the presence of that wire. In the extreme case of zero frequency (DC field), there will be no interaction at all. In conclusion, if one resorts to a brute force increase of the current to enormous values, the practical limitations of the system are obvious. Geometrical confinement and other constraints, and electrodynamic and thermal problems associated with fuel channel and equipment integrity are insurmountable obstacles.

The solution, offered by the AC device, to resolve this conflict between electromagnetic requirements and use of the readily available line frequency, proved to be elusive.

An electromagnetic method and device for repositioning a poorly conductive object (garter spring) behind a conductive wall (pressure tube) was developed on a principle similar to a linear induction motor with an ill-defined and shielded 'rotor'.)

Principles of Operation of the AC Device

The principle feature of the AC device is an efficient conversion of magnetic energy into a continuous mechanical movement of the garter spring, through the conductive wall of the pressure tube, without any adverse effects on the fuel channel integrity. Two fundamental difficulties were overcome, simultaneously, to achieve significant electromagnetic interaction with the garter springs.

The governing principle of the AC device is the creation of a travelling magnetic field in a fashion similar to that in a linear induction motor. The use of low (line) frequency makes the shielding effect of the

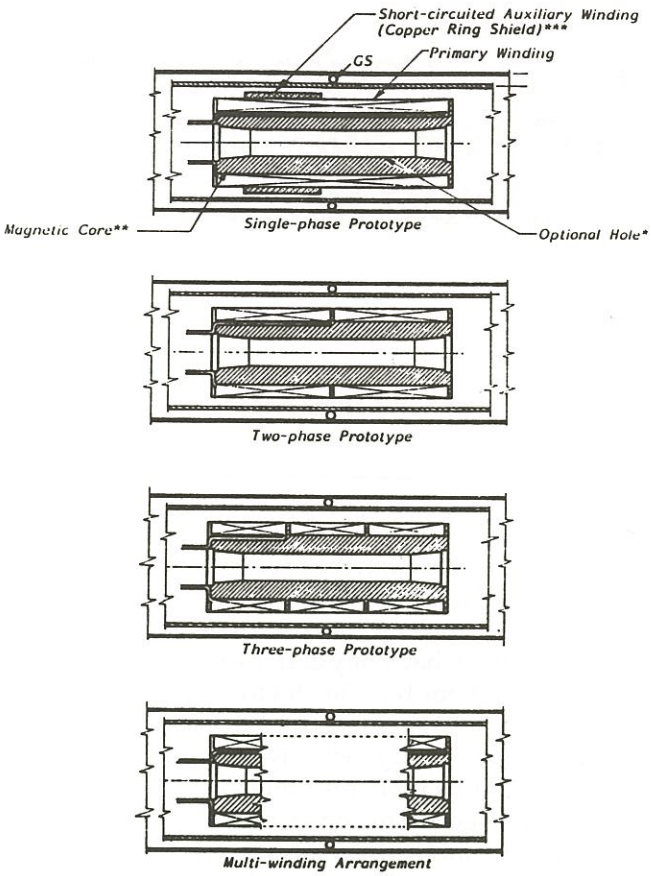

Figure 3: Typical prototypes of the AC device.

pressure tube negligible; yet the proper phase shifts are created between the steady AC currents flowing in the arrangement of windings placed over the special ferromagnetic core, to create the 'stationary' member of a linear motor, while the poorly conductive garter spring represents the 'travelling' part. The travelling magnetic field is created by a system of currents linking a high-permeability magnetic core. These currents are steady-state AC currents, with appropriate spatial and time phase shifts. For convenience, line frequency is used, although this is by no means the optimum value.

The AC device is a compact magnetic device, consisting of an arrangement of windings placed on a ferromagnetic core of high magnetic permeability, and with a high magnetic flux density saturation point. The magnetic core plays a fundamental role in making the CANDU device compact. It can be shown theoretically that the flux density on the same bounding surface in the magnetic field, using magnetic core materials with magnetic flux density B = 1.6 Tesla (which is a typical saturation point for ferromagnetic materials), is about 25,000 times as great as that in the electric field (using the nominal breakdown strength of air of 3×10^6 V/m),

or about 50,000 times as great for a special ferromagnetic material, vanadium permendur (B = 2.4 Teslas), employed in the AC device. Hence, a constant high-energy density is achieved through the use of the magnetic core with the steady-state AC current. In the AC device, the windings can be configured as single-phase or polyphase, and are energized from a continuous AC source. The core can be designed as a straight cylinder in its simplest form, or it can be designed to take more complex forms.

The AC device can take various design forms based on the same principle of operation but different engineering requirements, as shown in Figure 3. For instance, the fundamental single-phase prototype has a metal shield, which provides a low magnetic field region, where the garter spring tends to hide and rest, occupying a minimum energy position. The shield is essentially a copper ring which acts as an auxiliary winding from the circuit theory point of view, or as an electromagnetic shield from the field theory point of view. Induced currents in the copper ring cause the magnetic flux in the shielded portion to be shifted in phase (lag) with respect to the flux in the non-shielded portion (flux window). This results in a magnetic field gradient in space and in time, creating a travelling magnetic field. In case the garter spring needs to be flipped in the inclined position, the coper ring is designed to effect the flipping action (skewed copper ring under the desired angle of inclination). The single-phase AC device prototype is shown in Figure 4.

In the case of a two-phase AC device, the current phase shift in the second phase is predetermined by the source supply, and it is selected by choice to be 120° – but it can be different. In a special case, when the phase shift is 90°, this two-phase prototype becomes identical to a fundamental single-phase prototype.

In the case of the three-phase AC device shown in Figure 5, or, in general, a multi-phase AC device, the principal remains the same, as shown in Figure 3.

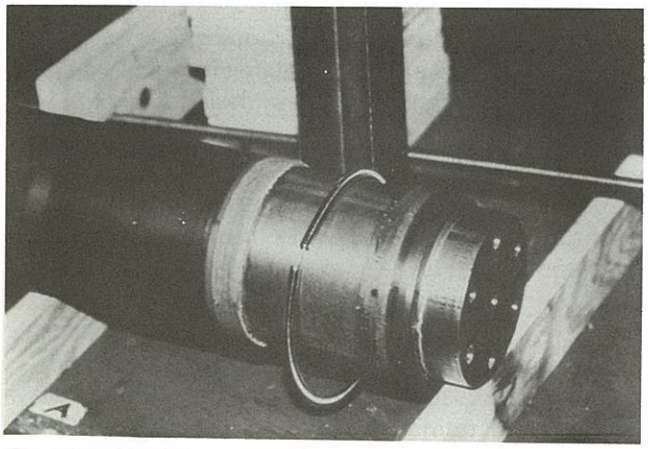

Figure 4: A single-phase AC device.

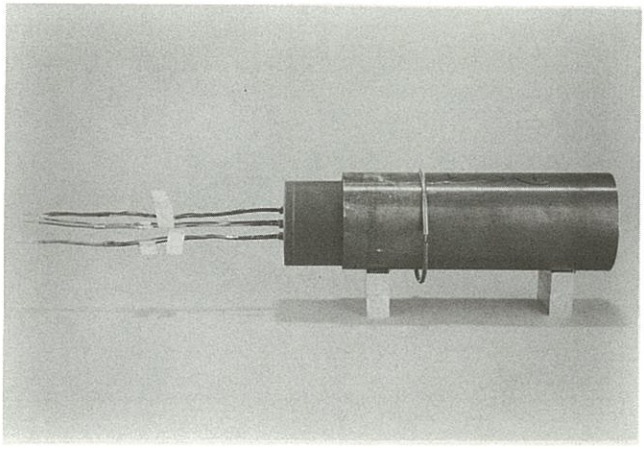

Figure 5: A three-phase AC device (LIM)

Prototypes may incorporate other features, such as more complex winding arrangements, to produce a rotating field superimposed on a linear travelling field, to create a 'screwing' type of motion of the garter spring, and eliminate or reduce the need for unpinching when and if it occurs. Further engineering optimization is possible.

Engineering Development and Integration of the AC Device into the SLAR System

The AC device was conceived prior to the initiation of the SLAR program. Because of its technical merit, cost, and compliance with the SLAR requirements, the AC device was selected for the 'repositioning portion' of the SLAR system.

In addition to the fundamental principles of the operation of the AC device, established initially, detailed circuit modelling was performed to simulate the electromagnetic coupling of the device to a garter spring through the pressure tube. These calculated values were in good agreement with the experimental results.

A number of progressively better prototypes were designed, built, and tested. Simple, two-phase and three-phase prototypes were developed. The geometry of the prototypes was imposed by the pressure tube inner diameter and SLAR tooling configuration, and interfacing requirements.

Initial prototypes were built with a solid magnetic core. Subsequent prototypes were built with a laminated vanadium permendur core to reduce thermal losses, as shown in Figure 6.

The AC device could, in principle, be as long as the fuel channel. However, because of the pressure tube sag, the overall length of the two devices mounted on the SLAR tool was selected to be approximately 0.25 m.

In addition, the entire AC device had to be capable of operating in the heavy water environment of the fuel channel. For this reason it was developed and built

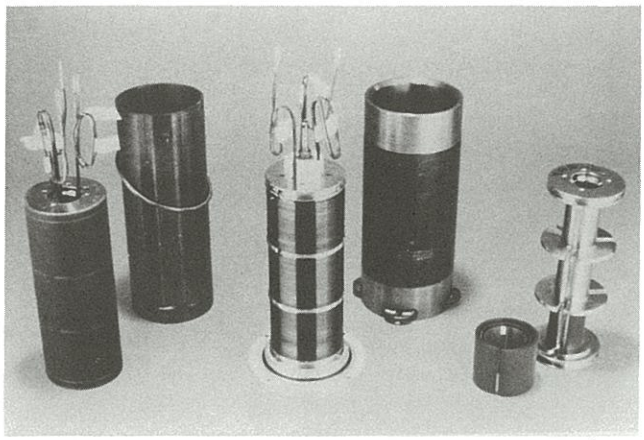

Figure 6: Three-phase AC device with a laminated core.

with a special polyurethane encapsulation, as shown in Figure 6.

Since, the AC device operates at low voltage (typically 400 V, 3 phase), and at low current (typically 100 A per phase), the insulation requirements would not normally be too difficult to meet. However, because of the importance of the SLAR project, and the high reliability requirements, the AC device was designed, built, and tested to be inherently safe. The electromagnetic force acting on the garter spring was developed to be in excess of 30 g, with attained values of up to 50 g, which is about three times the minimum force necessary to move the unpinched garter spring. During the engineering development phase, a number of electromagnetic, thermal, and mechanical parameters had to be established, in order to be able to interface the AC device with the SLAR tool. It was necessary to demonstrate the performance capability of the device under the constraints imposed by the SLAR system, including the environmental requirements (wet channel, radioactive, etc.). The power supply and control system to supply power and control to the AC device (LIM) was also designed, built, and tested, allowing selection of one of the two AC devices on the SLAR tool, and selection of the direction and control of garter spring movement. In addition to force-profile measurements, thermal duty cycle tests were performed to establish the size of the power conductor of the umbilical cable, used by the SLAR tool to feed the hydraulic line, the power line, and the sensor's signals through the reactor channel. Ten different prototypes were produced during this phase, in order to add successive improvements and incorporate design features required by SLAR (i.e., arrangement of leads, size of centre core, etc.). This imposed additional and premature difficulties on design. Near completion of the development phase, the AC device was incorporated into SLAR preproduction tooling, to be used during field trials.

Field Demonstration at Pickering 'A' NGS

The first field trial of the SLAR preproduction tooling took place on 12 April 1986 at Pickering 'A' NGS, Unit 4, Channel K10. The repositioning of the garter spring by using the AC device was successful, and the main objective of the field trial to move a garter spring by more than 50 cm was met. The allocated reactor down-time did not allow for the entire trial program.

Following the field trial, 46 AC Devices for the field implementation of the SLAR program were manufactured.

Conclusion

A key component of the SLAR (Space Location and Reposition) system for commissioned CANDU reactors was developed at Ontario Hydro Research Division. This component is a novel low-power AC device. The AC device was designed to reposition the spacers or garter springs within the gas annulus space of the fuel channels in CANDU reactors. The device operates safely and without any harmful effects on fuel channel integrity. The engineering development and integration into the SLAR system was done according to the SLAR program performance, environmental, and interfacing requirements and constraints.

Two AC devices were incorporated in the preproduction SLAR tooling system and successfully proven in a field trial performed in April 1986 in the wet channel K10 at Pickering 'A' Nuclear Generating Station, Unit 4. The AC device, or LIM, was demonstrated to be safe and easy to operate in the reactor's hostile environment. This will result in very large savings to Ontario Hydro and other CANDU reactor owners in terms of extended reliable reactor life.

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Waterlancing for CANDU Steam Generators

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Introduction

Steam generators, which are critical to the availability of nuclear plants, are very sensitive to corrosion induced by chemical attack, and particularly to attack from deposits left by the concentration of boiler water contaminants. Since steam generators around the world have suffered severely in this regard, high priority has been given to removal of deposits – especially those in the tube sheet area.

A typical CANDU steam generator is shown in Figure 1. This unit, a Darlington SG incorporates 4,550 U-tubes of 16 mm diameter on 24.5 mm triangular pitch. The recirculating boiler water flow enters the hot leg (primary inlet side) just above the tube sheet, whence it penetrates the tube bundle and then flows upward toward the steam drum. CANDU steam generators have special design features to enhance this flow (i.e., high circulation). Nevertheless, the centre of the bundle is inherently an area of weak flow, as well as high heat flux.

These characteristics create an environment that encourages the deposition of any boiler water contaminants onto the tube sheet, and also provides an environment where such a deposit can become highly aggressive in the event of an excursion of feedwater chemistry. Figure 2 shows a typical tube sheet sludge deposit as observed in a Pickering A steam generator.

The Waterlancing Process

Waterlancing is a process for removing sludge deposits by means of high-pressure water jets. The primary application has been in the tube sheet region; however, it can be applied to other areas, such as U-bends and steam separators.

The active component of the system is high-purity water issuing as a coherent, focused jet into the inter-tube gaps of the steam generator tube bundle. The jet impinges on the surface of the deposit, breaking it up and sweeping it out of the bundle. The effectiveness of the jet on deposits is determined by its pressure and the degree of focus that it sustains at the point of impact.

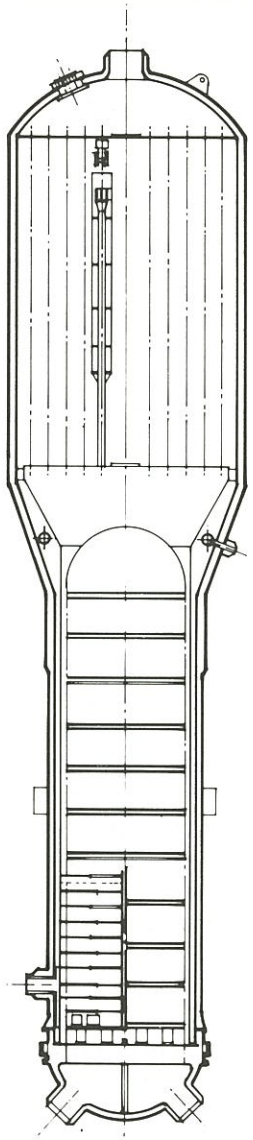

Figure 1: CANDU steam generator (Darlington).

The specific process described in this paper is that used at Pickering 'A', Unit 1, SG No. 3, which is the

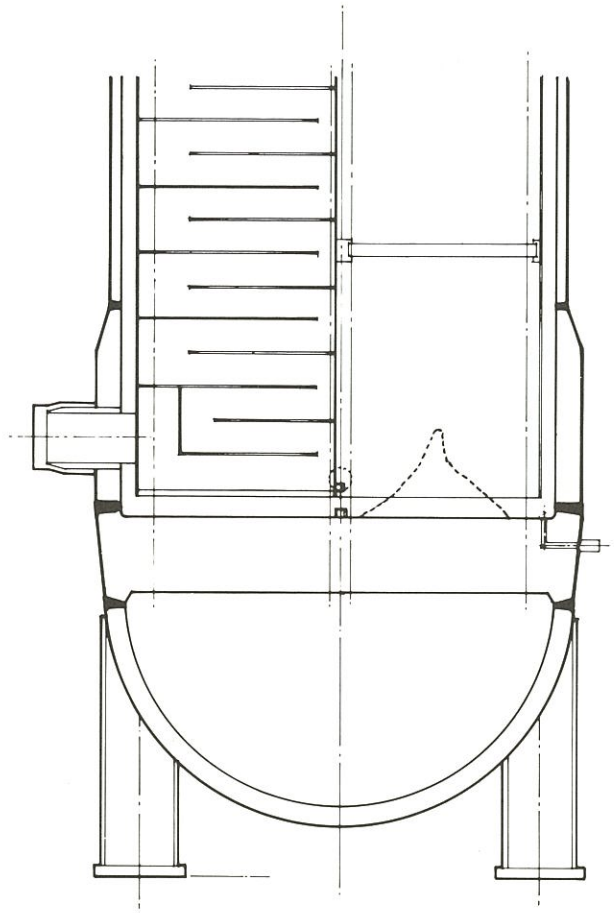

Figure 2: Sludge deposit (Pickering A).

only CANDU application of tubesheet waterlancing at time of writing. This project was carried out using the process of the Booyclean Company of Rotterdam, working together with B&W Canada.

A typical waterlancing setup is shown in Figure 3, and includes a mobile waterlancing unit, control console, lance and drive, and interconnecting hoses. The self-contained mobile unit shown in Figure 4 contains the demineralized water reservoir, the booster and high-pressure (hp) pump, return suction pumps, filters, and system controls. The hp pump is a diesel-driven reciprocating pump that is fed by a centrifugal booster pump and has a pressure control at its outlet, to control pressure at the lance tip. The suction pumps are high-volume, lift-and-force diaphragm pumps designed for air / water service. Their function, along with smaller suction pumps at the SG, is to evacuate the water / sludge mixture from the tube sheet and return it to the mobile unit for filtering. The filtering system incorporates three filter trains in parallel – each with 10.0 micron, 5.0 micron, and 0.5 micron filter elements in series.

The mobile unit also houses the control system, which includes the pump controls, safety shut-downs, and a video monitor for the in-containment cameras.

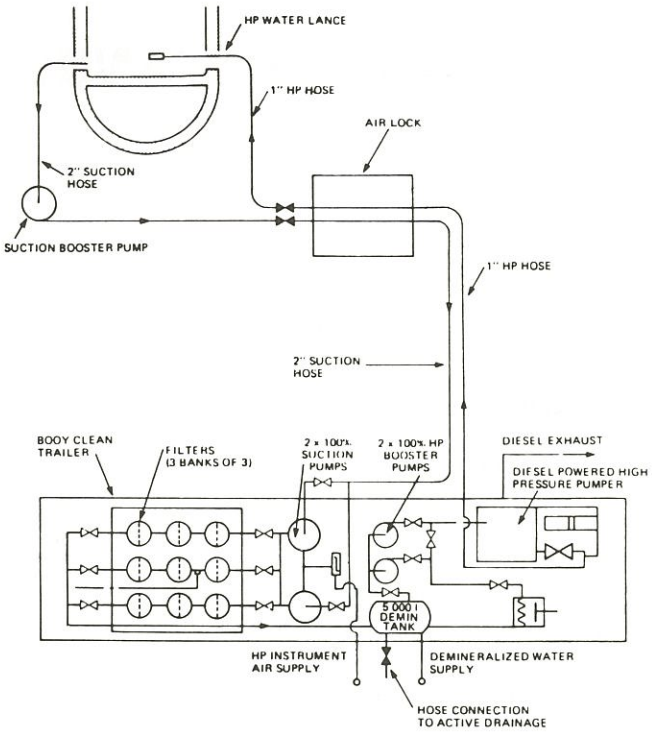

Figure 3: System configuration.

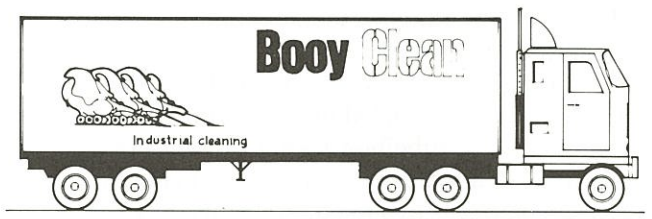

Figure 4: Waterlancing mobile unit.

The lancing equipment at the SG includes the lance and drive, the suction system, and the video cameras to monitor the above. Figure 5 shows a typical lance and drive. The lance is a long tube (usually segmented) with a number of nozzles at an angle of 90° to the lance axis, positioned so that several adjacent lanes may be lanced simultaneously. In operation, the tube rotates about its axis, so that the jets sweep up and down over the surface of the deposit within the inter-tube gap. The lance drive, which indexes and rotates the lance (in / out motion and forward / reverse rotation, respectively), is built onto a flange, which in turn mounts to a handhole at the SG no-tube lane (NTL) at the centre of the bundle.

The above lance and drive is the type used on a SG with inter-tube gaps at an angle of 90° to the NTL. At Pickering 'A', these gaps were at 60° and 120°, necessitating a special lance, as described below.

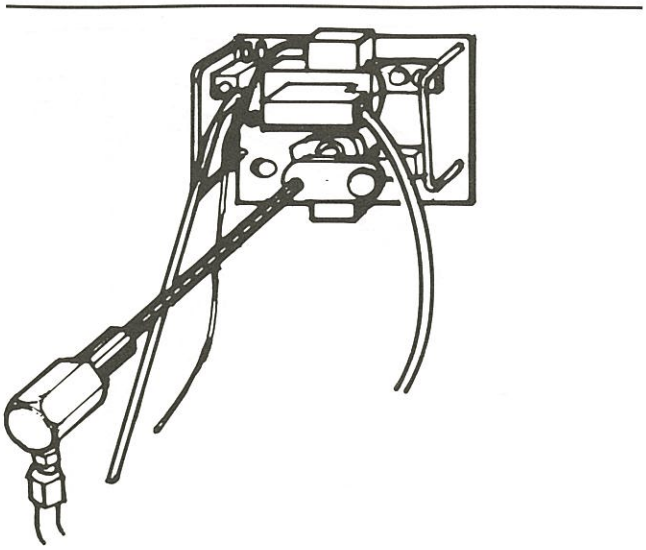

Figure 5: Waterlance with lance drive.

The control console, which is situated in a low-activity area near the SG, contains the indexing controls as well as safety shut-off controls, and a monitor to observe the video from the SG camera.

International Experience

Waterlancing, as a steam generator cleaning technique, began in the 1970's, and has since gained wide acceptance in the U.S. and Europe. In 1986, 45 SG's in Europe and 19 SG's in North America were lanced by the Booyclean process alone.

In many plants, it is now an established practice to waterlance all steam generators at regular intervals and, in some cases, as often as yearly. As noted during the discussions at the EPRI-sponsored conference on SG cleaning [2], SG waterlancing is a matter of routine for many plants.

Technical Issues

The technical issues related to the use of the waterlancing process basically involve: a) The ability to cut hard sludge; b) the ability to totally flush loose material from the SG; and c) the concern of tube erosion.

The ability to cut hard sludge is a function of pressure at the jet tip, the degree of focus of the jet at the point of impingement, and the jet motion as it affects the fracturing of deposits.

It is also a function of the hardness of the sludge, and the amount of fractures and fissures within the deposit. Selection of pressure is a compromise determined by the need to avoid erosion.

A pressure of 200 bar (3,000 psi) has been a widely accepted value and was used at the Pickering 'A' project. The jet focus and sweeping aspects relate to the design of the equipment and the process.

Tube damage is to be avoided at all costs, particularly in CANDU units such as Pickering 'A', in which the 125,000 tubes operating since 1971–3 have had only one in-service failure and, to date, show no sign of deterioration. Erosion testing was a key part of the initial qualifications of the process. Testing was conducted by a number of companies, including Electricité de France, Ontario Hydro, B&W, and others. While some erosion was observed during certain conditions, the tests generally support SG lancing at the normal 200 bar (3,000 psi) operating pressure.

Regarding field experience, there have been no reports of tube damage due to erosion during the years of waterlancing.

The Pickering 'A' Demonstration Project

Ontario Hydro has been pursuing waterlancing as well as chemical cleaning for several years. While it has numerous steam generators of different designs and in-service dates, the units of prime interest were those at Pickering 'A'. The main reason was that inspections had shown that Pickering 'A' SG's had up to 400 mm of sludge on the tube sheets (which have a bundle radius of only about 760 mm). Also, Units 1 and 2 were in an extended shut-down and were available for worthwhile upgrade measures. While these units clearly would benefit from waterlancing, they are also the most difficult to deal with because, as noted below, access is very difficult, and the sludge is extremely hard.

The excellent tube reliability of these SG's provided an unusual constraint. Since there has been only one in-service tube failure and no apparent degradation, it was particularly important that lancing avoid even the slightest effect on the tubes. Additionally, since Pickering 'A' tubes went into service in the as-drawn (not surface ground) condition, it seemed possible that their good fortune night relate to some protective feature built into the outer skin of the tube or the oxide layer. Therefore, these areas also had to remain undamaged.

On 30 September 1986, Ontario Hydro issued instructions to proceed with the design and development of the necessary lancing equipment, and with the demonstration lancing of one steam generator at Pickering 'A' (Unit 1, Boiler 3). The entire program was completed on 7 February 1986, meeting a very tight schedule, considering the uniqueness of the devices required.

The Pickering 'A' Lance Design

The Pickering 'A' steam generator lancing system employed a standard mobile unit, with control console, which was already stationed and in use in North America. However, as already noted, it required a very special lance head and drive because the tube pattern was rotated at 60° and 120° to the NTL, and because the NTL access space allowed a lance head which was only about 31 mm square in cross section. The lance design

had to be small enough for the NTL space, and it had to provide jets which would sweep up and down in the 60° and 120° planes. This required the use of a stationary lance shaft, but with articulated nozzle blocks mounted in the head of the lance. Furthermore, the lance jets had to accurately align with inter-tube gaps of only 3.8 mm (0.15 inch).

The Pickering lance design incorporates a standard lance drive with indexing capability, but without rotation. The lance itself is a tube with five passages. At the inboard end of this tube, there is a non-rotating lance head with two nozzle blocks that sweep up and down and contain the nozzle tips. The sweeping is performed by a special drive at the outboard end of the lance. Alignment of the jet with the inter-tube gaps is confirmed by an infrared eye transmitting the location from the lance head via a fibre optic cable. This device allows initial lance alignment and confirmation of indexing accuracy by the pre-set lance drive indexing system.

Two lances were used at site. The majority of the work was performed by one lance with two jets operating on adjacent lanes at an angle of 120° to the NTL (120° from the NTL handhole onto which the lance drive was mounted). A second lance had two jets – one at a 60° angle and one at a 120° angle. These lances were totally new in concept, as well as being quite intricate. Some mechanical difficulties were encountered during the mock-up testing and during the early part of the site work. Ultimately, however, the equipment performed its function.

Unit 1 Boiler 3 Demonstration

The demonstration waterlancing of this single steam generator at Pickering Unit 1 followed a step-by-step program that was arranged to demonstrate the process with minimum exposure to tube damage. The areas lanced are shown in Figure 6.

Three-Lane Trial

The initial lancing trials were performed on three lanes (designated A, B, C) located at about 120° from the NTL handhole. These lanes were in line with a second handhole, so that extensive inspection was possible. The lancing proceeded in seven steps, with the pressure and time at position increasing up to 3,000 psi, and 50 jet rotations of 30 seconds, respectively. All subsequent jetting was at 3,000 psi – the normal lancing pressure. During this stage, the sludge pile was cut back about 75 mm (measured horizontally from the NTL), with the progress becoming poorer with time. This was attributed to the jet being 'choked' by a buildup of water in the restricted space in the middle rows of the deep pile.

Visual inspection showed no sign of tube erosion or even of loss of the surface layer. It was then decided to try a larger area near the edge of the tube bundle where the pile was less deep and the flow less restricted.

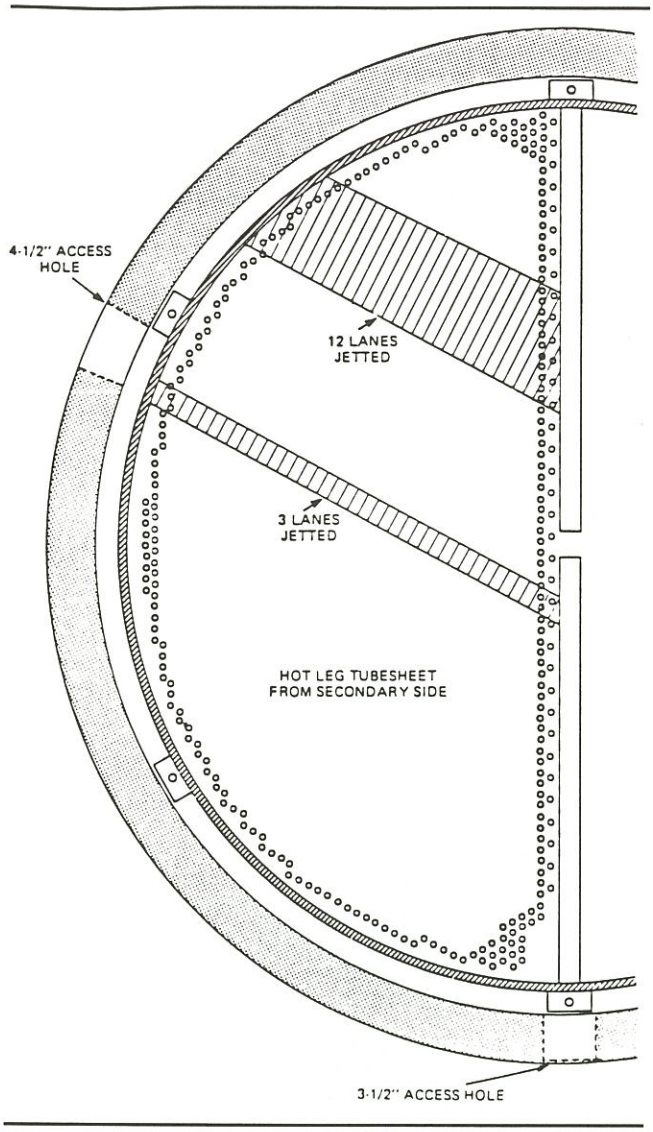

Figure 6: Areas lanced at Pickering Unit 1.

Twelve-Lane Trial

In the twelve-lane area shown in Figure 6, the outer and inner six lanes were jetted for 30 and 60 rotations, respectively. During this stage, the cutting was a bit faster.

With this trial completed, a temporary hold was imposed, while Ontario Hydro removed two tubes for examination (one from the three-lane and one from the twelve-lane area). Visual examination showed no loss of either base metal or of the tubes' protective layer. It was then decided to lance the full hot leg on a 24-hour-per-day basis.

During the earlier part of this program, equipment difficulties were experienced with the prototype lances developed specifically for Pickering. However, there was no problem with the proven equipment. The

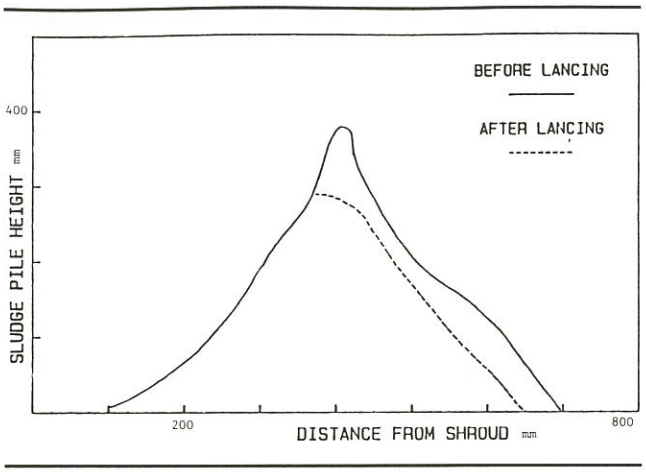

Figure 7: Pickering sludge profile.

difficulties involved design details in the intricate articulated head, and with the initial use of an infrared eye to indicate tube gap location. The need for these particular design features relates to the small NTL space and the rotated pitch orientation of these SG's.

Complete Hot Leg Waterlancing

During this operation, the complete hot leg was lanced in four passes, with each lane undergoing about 120 minutes of jetting at the normal 3,000 psi pressure. This operation altogether took three and a half days, working on a 24-hour-per-day basis.

The result was a steady but slow removal of the very steep face of the bundle on the NTL side. At no time was the lance able to work on the outboard side of the pile, which was initially about 250 mm above the lance axis. By the end of this activity, about 50 mm (measured horizontally) or 100 mm (measured vertically) had been removed from the NTL side of the pile. The 'before' and 'after' profiles are shown in Figure 7 [1].

Overall Results

The overall result of the operation based on the physical measurements was removal of 35–45 kg of deposit, or 12–15% of the total initial deposit. Considering that earlier microanalysis had indicated the sludge to be as hard as 16,000 psi concrete, it is not a bad result; nevertheless, it is clearly necessary to progress faster if the process is to be viable on these particular units.

These results indicate a need for some optimization of the process, to enhance removal rates for this very hard sludge. Some promising possibilities exist in this regard. Optimization would include operation of the system closer to its capacity, i.e., in a normal setup, two to four lances, with up to nine jets each, would operate simultaneously. This compares to the two jets used in this demonstration program.

Examination of the tubes after the twelve-lane trials and after the full hot leg operation showed no tube damage or loss of protective oxide layer.

The recommendation to other plants is to recognize the importance of early lancing in avoiding, as much as possible, hardness. For the same reason, the sludge should not be allowed to air-dry.

Other CANDU SG's

While CANDU SG's in general have had excellent tube reliability experience, they also, in many cases, have deep sludge piles, which can become aggressive at any time. Therefore, the following comments and recommendations are made regarding other CANDU SG's:

The situation at Point Lepreau is different from that at Pickering. The Point Lepreau plant has started up and is operating on phosphate water chemistry – a condition which is known to have caused phosphate wastage attack at a European plant with similar stabilized I800 tubing. Even though no general attack has been observed to date, sludge removal is a high priority – either to reduce the potential for under-deposit attack, or ultimately to allow a change to all-volatile (AVT) water chemistry. At time of writing, instructions have been provided by the utility to proceed with development of special lancing devices, to demonstrate them in a specially constructed mock-up, and to perform full-scale lancing on the four SG's during the May 1987 outage.

The 600 MWe units at Gentilly 2 and Embalse are identical to Point Lepreau in this regard and are waterlanceable with identical equipment. Clearly, such lancing should be planned for and carried out in the near future. The other 600 series SG's are also similar and amenable to similar treatment.

The Pickering 'B' SG's are somewhat easier to lance than the 'A' units, in that they are newer, have small handholes, and have tube gaps at 90° angles to the NTL. These would be amenable to lancing with a variation of the Point Lepreau equipment.

The Bruce 'A' and 'B' SG's with I600 tube material have deposits, and are therefore in need of cleaning. Their physical location makes access exceedingly awkward. Nevertheless, the possibility of some type of lancing should not be dismissed. Even if they are cleaned chemically, some type of lancing is desirable to remove the 15% or so of deposit which is not soluble in the cleaning agents.

The Darlington SG's clearly should be lanced routinely right from the start. While they are carefully designed to minimize deposit, some will undoubtedly occur. Fortunately, these units are well designed for lancing. External access is very good and internally the arrangement for lancing is also very good, and similar to Point Lepreau.

Conclusions

Waterlancing is a very important means of reducing tube sheet deposit accumulation and, in turn, the potential for under-deposit-attack on the critically important steam generator tubing. This conclusion is supported by practice in Europe and in the U.S., where

such lancing is now, in many cases, done at routine intervals.

The Pickering 'A' development and demonstration project has shown that lancing is a viable process, even with the physically difficult and hard deposit conditions of those SG's. It has also been shown that tube erosion is not the concern that many had feared, as evidenced by a variety of tests.

The current Point Lepreau project will provide full-scale lancing of those steam generators. They are newer and have physically easier access, but contain sludge deposits which are very hard.

Sludge lancing should be considered as a routine cleaning measure for all CANDU steam generators, including those which are just in the process of starting up.

Acknowledgements

Recognition is due to the many people who contributed substantially and enthusiastically to the Pickering 'A' first-of-a-kind waterlancing project. Some of these people are Messrs K. Talbot, R. Dyck, M. Upton, D. Bunton, D. Carlson, and D. Evans of Ontario Hydro; Messrs H. Buitendijk and D. Bashford and crew of Booyclean; and Messrs J. Bowman and crew of B&W Canada.

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Locating Gas Pipelines with Radioactive Sources

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Roland Francoeur
Gaz Métropolitain Inc.

Introduction

The increasing cost of excavation work is of concern to every utility that operates buried equipment. It has been estimated that the use of improved pipe locators could result in significant savings.

The location of buried pipes is based on several different principles. A radar-based pipe locator [Hale 1983, Young 1983] sends out a short pulse, which is reflected by the pipe and received by an antenna. The technique of inductive pipe location uses a low-frequency transmitter working in the 10–120 kHz range, and a receiver [Blears and Daniels 1984, Garnett 1983].

These locators can achieve a good success rate in locating the buried pipes, but they exhibit a high signal loss in some parts of the country because of the combination of moisture, clay, and salt in certain soils [Young 1983]. Difficulties arise also when a pipe is buried under an iron-reinforced concrete floor, or when there are many pipes present in close proximity.

With a different physical approach, it may be possible to overcome these problems. The practicability of using radioactivity to locate buried pipes has been evaluated.

Experimental Work

Research work in the laboratory was started by building a test 'gas pipe' consisting of a 2 m long tube covered by concrete blocks, which simulated the soil cover. Weak (20 MBq) radioactive sources containing ^{24}Na and ^{56}Mn were inserted in the middle of the tube. These sources were produced by irradiating $NaNO_3$ and $KMnO_4$ in our nuclear reactor, SLOWPOKE. The layers of concrete blocks covered a 4 m^2 area and varied in thickness from 40 to 80 cm. In the laboratory, the transmitted radiation was measured by a collimated 7.5 cm cylindrical NaI(Tl) crystal, which was shielded on all sides, except its face, by 3 cm of lead, and was coupled to a scaler. In the field, the transmitted radiation was measured by a 5 cm cylindrical (NaI(Tl) detector, which was collimated with a lead ring 2 cm thick, and coupled to a ratemeter.

In the laboratory, readings were taken directly above the source and every 10 cm along a straight line crossing this point.

After obtaining good results in the laboratory, we proceeded with tests on an unused section of real gas main, which was made available to us by Gaz Metropolitain Inc. This pipe is buried 120 cm in the ground, it is 15 cm in diameter and 60 metres long. A valve was installed by Gaz Metropolitain Inc., in order to provide an opening to the pipe. For the tests, 400 MBq of Na-24 and Br-82 were used. The sources were prepared by mixing an epoxy resin with $NaNO_3$, $KMnO_4$, and NaBr, in a ratio of 1:3. The mixture was then irradiated in a SLOWPOKE reactor to obtain the desired activity. The use of radioactive powders, therefore, was avoided. The solid sources were attached to a steel cable and inserted into the gas pipe.

Results and Discussion

The results obtained in the laboratory are listed in Table 1. With these results, the resolution of the location was calculated. This was defined as the distance at which the net count rate decreases to 50 per cent of the maximum count rate obtained when the detector is located directly above the radioactive source. These values indicate the degree of precision that can be achieved in locating a gas pipe by means of this method. An average value of 36 ± 3 cm was obtained in the laboratory. The resolution appears to be independent of the thickness of the material covering the gas pipe. The shape of the curve formed by the readings was also evaluated. In fitting the measured values to the normal distribution function [Spiegel 1975] quite a good fit was obtained, as shown by the values of the chi-square test. The means and the standard deviations of the hypothetical normal distribution functions were calculated by the standard formulas found in all basic books on statistics.

The method was then tested in the field. Using the

Table 1: Variation of Net Count Rates (in Counts / s) with the Position of the Detector (Tests in the Laboratory)

Isotope Thickness		*Na-24*			*Mn-56*	
(cm)	*40*	*60*	*80*	*40*	*60*	*80*
−80	67	23	5.2	6.5	2.4	1.7
−70	88	26	5.6	8.3	4.9	2.8
−60	112	34	7.4	14	7.0	4.4
−50	156	44	7.6	23	9.9	6.0
−40	181	63	9.9	27	15	6.4
−30	318	95	15	66	23	8.0
−20	572	147	21	96	34	8.1
−10	1002	202	30	130	45	9.0
0	1138	225	40	166	47	9.1
+10	922	215	39	129	44	8.0
+20	527	160	35	63	40	7.6
+30	268	101	27	61	26	7.5
+40	165	63	19	21	21	6.9
+50	116	44	7.8	18	15	5.8
+60	100	30	5.2	11	13	5.6
+70	79	23	5.2	9.7	12	4.3
+80	64	21	6.2	7.6	9.5	2.1
Resolution	33	37	40	32	36	38
Chi-square	15.6	6.7	7.7	18.4	7.1	13.7

Table 2: Variation of Net Count Rates (in Counts / s) with the Position of the Detector (Tests in the Field)

	Distance from the opening	*Direction perpendicular to the gas-main (cm)*						
RI	*(m)*	100	40	20	0	−20	−40	−100
	0	15	65	150	320	180	85	13
	2	12	50	140	310	170	65	11
Na-24	4	10	45	120	250	150	50	10
	6	13	55	110	280	170	60	12
	10	14	60	170	310	160	80	14
	20	13	55	160	300	170	85	15
	0	16	130	250	650	280	150	16
Na-24	2	14	120	240	640	250	130	13
&	4	12	100	180	620	220	100	9
Br-82	6	10	90	220	610	230	120	11
	10	11	110	240	650	270	140	12
	20	16	130	230	640	260	130	11
	30	13	120	250	630	270	130	12

steel cable the 400 MBq source of Na-24 and Na-24, combined with Br-82, was inserted into the gas main. The count rates obtained are shown in Table 2.

As clearly indicated, the signal is strongest when the detector is directly above the radioactive source, and it decreases rapidly when the detector is moved away from the source. It was possible to cover the full length of the gas pipe.

The obvious limitation of this pipe-locating method is the need for an opening through which the radioactive source can be introduced into the canalization; consequently, this technique can be used only near the points where gas is consumed. However, the nuclear method described in this paper gives unambiguous results, and it is impossible, for example, to confuse a gas pipe with a water main.

With an activity level of 400 MBq, it is quite easy to locate a gas pipe up to 1.2 m in depth. A weaker radioactive source can be used when the canalization is buried closer to the surface.

The method described should be especially advantageous when a gas pipe under a concrete floor needs to be modified. In this case, an exact location is feasible, and the cost of breaking through the concrete would be minimized.

Another advantage of this method is that it can be used in pipes made of any type of material, some of which (for example plastics) may be difficult to detect

with other techniques. This method should also be effective in locating pipes buried under iron-reinforced concrete, because the radioactivity would be only slightly attenuated by the presence of the iron grid. The whole operation takes about 10 minutes and the basic equipment (counter and ratemeter) can be bought for a few thousand dollars.

Conclusions

Radioactive sources containing Na-24 and / or Br-82 make it possible efficiently and accurately to locate buried gas pipes, provided an opening to the pipe is available. The use of radioactivity gives non-ambiguous results and can be used with gas pipes made of any type of material. The technique is fast and cost-effective.

Acknowledgements

The grant provided by Gaz Metropolitain Inc., which made this research work possible, is gratefully acknowledged.

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Optimization of Darlington Tritium Removal Facility Performance: Effects of Key Process Variables

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Introduction

Ontario Hydro has constructed a Tritium Removal Facility (DTRF) at its Darlington Nuclear Generating Station to extract tritium from the heavy water moderator water of its CANDU heavy water nuclear reactors. Detritiation of moderator water will reduce occupational exposure to tritium and will reduce environmental emissions. The front end of the process system consists of an eight-stage Vapour Phase Catalytic Exchange (VPCE) section which extracts tritium from heavy water into a deuterium gas stream. The tritium is then concentrated to 99.9 percent purity by cryogenic distillation and stored in immobilized form in containers. The DTRF process flowsheet is shown in Figure 1.

For the DTRF, as with any complex process system, the development of an optimum operating strategy requires an understanding of the process and the process parameters which may be set to achieve desired plant performance. For example, one could choose to maximize system throughput at the expense of a lower detritiation factor, or one could choose to minimize operating costs and accept a lower tritium production rate. This paper discusses the major process variables in the DTRF and how they affect system performance. The results presented here are based on computer simulations carried out using Ontario Hydro's FLOSHEET process simulation program [1].

DTRF Performance Objectives

For DTRF operation, the following major performance objectives can be identified:

1. minimize operating costs;
2. maximize the rate at which heavy water is detritiated;
3. maximize the detritiation factor for processed heavy water;
4. maximize the overall tritium production rate; and
5. maximize tritium product purity.

Other objectives may also be defined, but these are probably the most important ones.

Not all of the above objectives are complementary. In general, if plant performance is optimized with respect to only one objective, other objectives are compromised. Therefore, choosing an *overall best* strategy involves tradeoffs.

In this paper, the relative weights which could be assigned to different performance objectives are not discussed, since they depend on factors such as requirements for tritium product purity, feed concentration, and detritiation factor, which will be better defined later.

As with any existing plant, operating strategies must take into account physical and operational constraints to plant operation. For example, it is not helpful to derive an optimum feedrate for the DTRF which is in excess of the processing capabilities of the installed equipment, unless design modifications are an acceptable option. Of course, design modifications involve additional expense and system unavailability, which must be properly accounted for.

Major Process Variables

The major process variables which affect DTRF performance are:

1. feedrate of heavy water;
2. deuterium gas flowrate between the VPCE and the cyogenic distillation cascade;
3. reflux ratio in the Low Tritium Column (LTC); and
4. tritium product drawoff rate from the cryogenic distillation cascade.

The following four sections discuss the effects of adjusting these key process variables.

D_2O Feedrate to the VPCE

It is desirable for the DTRF to be able to process tritiated heavy water at as high a rate as possible. The benefits of a high throughput are high tritium production rate, reduced average tritium levels in reactor moderator systems, and accompanying reduced occupational exposure to tritium. The main drawbacks are a higher overall operating cost (although cost per curie of

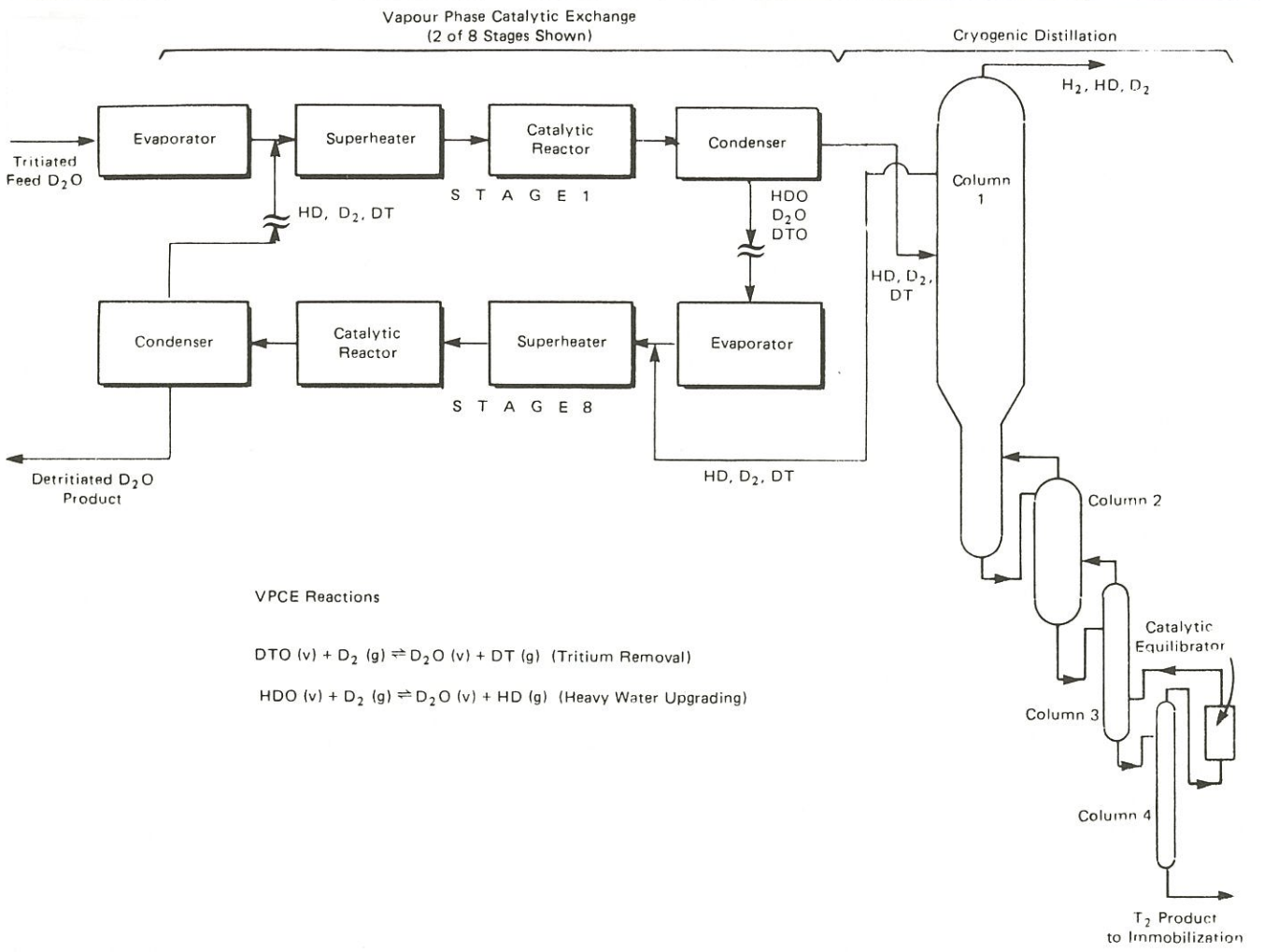

Figure 1: Darlington Tritium Removal Facility process flow diagram.

tritium extracted may be lower) and possibly a lower detritiation factor.

One way to increase the throughput of heavy water through the DTRF is simply to increase the heavy water feedrate to the front end VPCE process while keeping all other variables constant. An increase in the feedrate can be accomplished by taking advantage of system design margins. Calculations show that, provided that the VPCE has sufficient reboiler and superheater capacity, the heavy water throughput can be increased by about 10 percent over the design values without the detritiation factor dropping below 30. The actual system may be even more capable if the designer (Sulzer Canada, Inc.) has assumed conservative values for key design parameters such as height of an equivalent theoretical plate (HETP) in the cryogenic distillation columns, reboiler, and superheater capacity in the VPCE, etc.

For the system as designed, Figure 2 presents simulation results which illustrate the relationship between liquid feedrate, detritiation factor and overall tritium removal rate.

Calculations suggest that the main heavy water

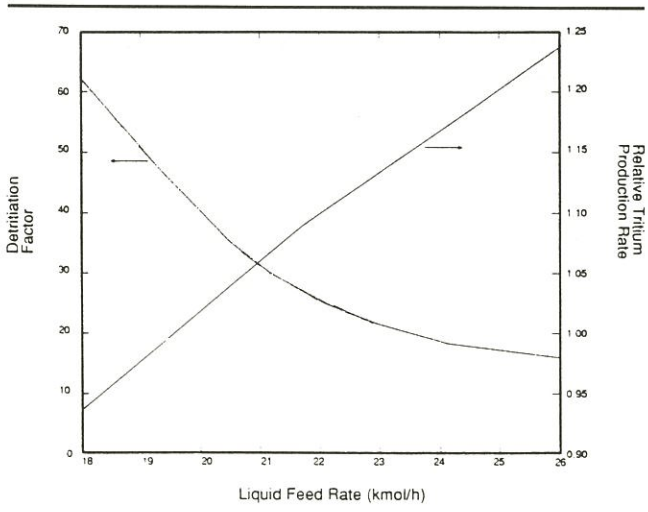

Figure 2: Relationship between liquid feedrate, detritiation factor and overall tritium removal rate.

processing limitation in the DTRF system is the VPCE front end. A possible design change to increase the D_2O throughput would be to add an electrolysis unit

to work in parallel with the VPCE. In this configuration, if the total deuterium gas feedrate to the cryogenic distillation cascade is kept the same as before, the tritium extraction rate will be increased due to a higher concentration of tritium in the gas. This is because the mole fraction of tritium in deuterium gas leaving the VPCE is approximately one half of the tritium mole fraction in the liquid feed. In the case of an electrolysis unit, the mole fraction of tritium in the outlet gas stream is the same as the mole fraction in the input liquid stream. Therefore, if the VPCE were to be completely replaced by an electrolysis front end, the tritium extraction rate would be doubled. If an electrolysis unit is used in parallel with the existing VPCE, the improvement in tritium extraction would be proportional to the fraction of the gas coming from the electrolysis unit.

D_2 Gas Flowrate through the VPCE

The rate of flow of D_2 gas between the VPCE and the cryogenic distillation cascade affects the isotope separation characteristics of both the VPCE and the cryogenic distillation cascade. Figure 3 shows the detritiation factor for heavy water as a function of D_2 gas flow, with all other factors being held constant (34 Ci / kg feed case). If the gas flow is decreased, then the heavy water detritiation factor decreases, although the molar tritium concentration in the deuterium gas feed to the cryogenic distillation cascade increases. Also, if the deuterium gas feed rate to the cryogenic distillation cascade decreases, the reboiler duty in the LTC can be decreased proportionately (the reflux ratio is assumed to be held constant). This results in a saving of energy, since the cryogenic refrigeration load is decreased. In the DTRF design, a decrease in the gas flowrate of 10 percent should be possible while still maintaining a detritiation factor of 30.

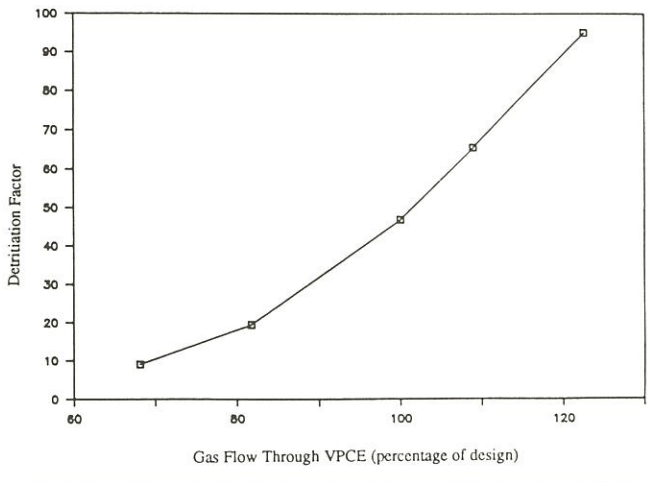

Figure 3: Relationship between detritiation factor and deuterium gas flow through the VPCE. The reflux ratio in the first column of the cryogenic distillation (the Low Tritium Column) is assumed to be held constant.

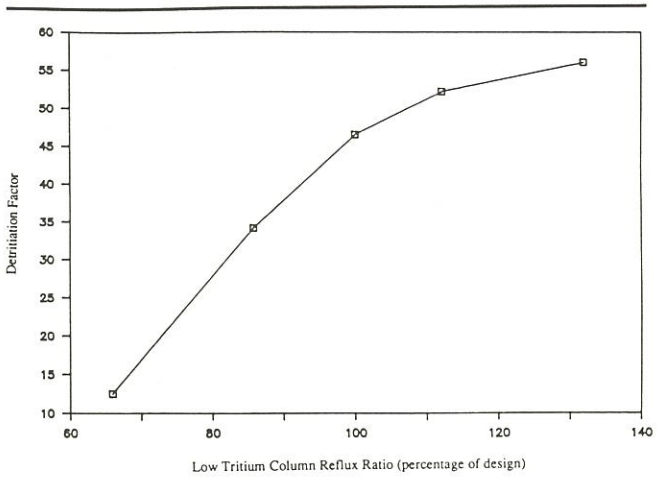

Figure 4: The effect of varying the reflux ratio in the first column (Low Tritium Column) in the cryogenic distillation cascade.

LTC Reflux Ratio

If the reflux ratio in the LTC is varied, while all other factors are kept constant, the heavy water detritiation factor will be affected. Figure 4 illustrates the relationship between detritiation factor and LTC reflux ratio for the 34 Ci / kg feed case. A lower reflux ratio results in energy savings due to decreased cryogenic refrigeration load. However, the tritium concentration in the gas returning to the VPCE from the cryogenic distillation cascade will increase and result in a lower detritiation factor.

Tritium Product Drawoff Rate

In the final column of the cryogenic distillation cascade there is no direct measure of tritium product purity, and tritium drawoffs are based on mass balance calculations of how much tritium has accumulated in the system. Tritium product drawoff rate is related to tritium product purity. If drawoffs are too frequent (by even 1 percent), then a high tritium product purity, such as 99.9 percent purity, will not be met. For each 1 percent excess drawoff rate, the tritium product purity decreases by about 1 percent. Considerable care will be required to operate the system for very high purity product specifications.

Downgraded Heavy Water Feed

Normally the DTRF will process only reactor-grade heavy water (>99.8% D_2O). If downgraded heavy water is introduced into the system, the detritiation efficiency will decrease because the HTO-triated species is not effectively removed by the cryogenic distillation system, and also because the presence of the HDO decreases the tritium extraction efficiency of the VPCE. The effect of small amounts of light water in the heavy water feed to the DTRF is illustrated in Figure 5. Even small amounts of light water (<1%) can significantly decrease the processed heavy water detritiation factor.

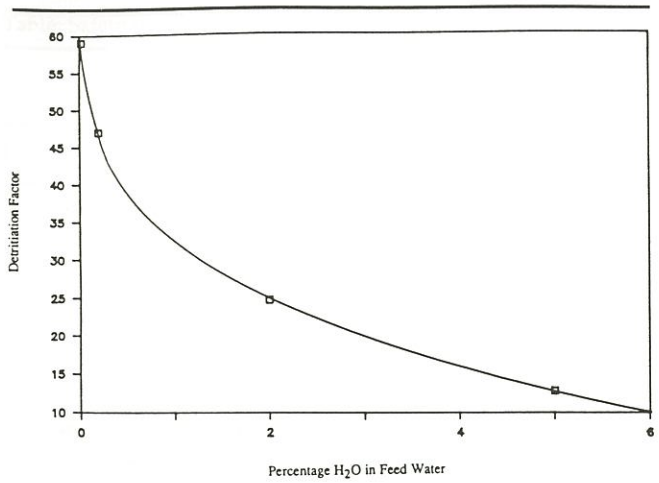

Figure 5: Detritiation factor when processing downgraded heavy water. Even small amounts of light water cause a significant drop in the detritiation factor.

Conclusions

1. The deuterium gas flowrate between the VPCE and the cryogenic distillation cascade affects the detritiation factor for processed heavy water and the refrigeration requirements of the Low Tritium Column (assuming a constant reflux ratio). Decreased gas flow results in energy savings but reduces the heavy water detritiation factor. An optimum value is, therefore, a tradeoff between operating cost and detritiation factor. Figure 3 shows the relationship between detritiation factor and deuterium gas flowrate.
2. The reflux ratio in the Low Tritium Column affects the detritiation factor for processed heavy water and the cryogenic refrigeration load. Reducing the reflux ratio would result in energy savings but would also reduce the detritiation factor, as illustrated in Figure 4. This figure shows that, at the design feedrate (assuming 34 Ci / kg heavy water feed), not much is to be gained by increasing the reflux ratio above the design value, since the detritiation factor increases only marginally.
3. Tritium product drawoff rate from the cryogenic distillation cascade affects product purity. In order to be assured of a tritium product purity of at least 99.9 percent, considerable care will need to be exercised in timing the tritium drawoffs from the final high tritium column.
4. The capability of the DTRF process equipment should be determined as accurately as possible during commissioning, in order that optimum operating strategies may be determined. Some of the calculations reported here will need to be updated to include actual performance data.
5. It may be desirable to operate the DTRF at detritiation factors which are a function of feed concentration. The best operating conditions depend on both the required detritiation factor and a consideration of operating costs.
6. To achieve high detritiation factors for processed heavy water, the heavy water feed to the DTRF should have very low protium levels.

Acknowledgement

The assistance of K.M. Kalyanam in performing some of the calculations reported here is gratefully acknowledged.

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Discussion

A Review of the Health Effects of Energy Development. (Volume 1, No. 1, March 1987, pp. 14–24.)

Two items published recently call for comment. They are the article by Myers and Werner in your first issue and my presentation to the New Democratic Party (NDP) inquiry, condensed in the CNS bulletin of Jan / Feb 1987.

After the Chernobyl experience, are changes needed in the management of safety in general and in nuclear safety in particular? Several of the worst technological disasters which have ever happened were like Chernobyl: the sinking of the Titanic, the Vaiont dam landslide in Italy, the aircraft collision at Tenerife. Each of these was the worst peacetime accident in its category. In all these four cases the principal 'machine' involved started in a substantially normal and viable condition; the disaster would have been easily forestalled and the loss of life averted if the smallest amount of modern safety thinking had been applied to the situation only a few hours or tens of hours beforehand; the trouble was in each case that those responsible for safety had been busily engaged in solving what proved to be only a part of the overall problem. It was not that the 'safety analysis,' such as it was, was inaccurate or lacking in detail; the problem was that it was too narrow.

In papers which I wrote in 1982 and 1985, I made these points about the estimation and management of technological risk:

1. Because we learn, we continually destroy the relevance of failure data. The data apply to the situation before we learnt; we are trying to predict experience in the future after we have learnt.
2. Failure data always suffer from a 'Heisenberg-like' uncertainty: if we want information about closely relevant experience, we find that there is almost none available; if we cast the net wider, the information increases but the relevancy diminishes.
3. Even if unlimited and closely relevant failure data were available, so that risk could be evaluated with certainty, actual future experience will be highly uncertain in cases where the risk is small, as it almost invariably is. For example, if the risk of an accident of the Three Mile Island kind is 0.0008 per reactor year, and 200 reactors are equally exposed to that risk for 6 years, there is about a 70% chance that one reactor will have such an accident and that 199 reactors will not. There may be no such accident at all. There would be no significance in which particular reactor had the accident. A uniform risk would have given rise to a highly stochastic and uncertain actual experience.

These factors are fundamental and cannot be circumvented. No amount of experience or research or calculation will alter the situation, either in general or in particular cases.

Chernobyl invokes all these points. A glaringly wrong misoperation undoubtedly 'went round the end' of a lot of otherwise adequate safety planning by the engineers concerned. Quite obviously, it might never have happened. Now that it has happened, what we need to manage is the risk *after* Chernobyl, that is, after the learning process, and in reactors *other* than those of the Chernobyl type. The data we have amounts to one accident of the kind in 4,000 reactor years globally (not very relevant to Chernobyl-type reactors) or one in 80 reactor-years (not very relevant to non-Chernobyl reactors). It is abundantly clear that we will never be able to achieve certainty in assessing the risk of another Chernobyl accident; estimating the risk points the way to reducing it to insignificance. The Canadian nuclear safety manager must decide how relevant this 'non-western' and 'non-CANDU' event is to his own situation. The USSR safety managers have the somewhat different and more narrow problem of correcting the revealed weaknesses in the Chernobyl reactors and the way in which they were operated, similar to upgrading the hardware and the operation of NRX after its 1952 accident.

I concluded that these problems could only be dealt with by the use of the higher attributes of the human mind, that is, by bringing informed and disciplined judgment to bear. If we recognise this I consider that we can improve the application of judgment. We can operate on our own intellectual processes to some extent. The suggested discipline is to assemble our judgment into a quantitative working hypothesis.

Table: Safety Benefit / Detriment Comparisons.

	More advanced countries	*Less advanced countries*
Lower risk energy source (e.g., nuclear)	GLE 30 days LLE 0.37 days	GLE 200 days LLE 0.37 days
Higher risk energy source (e.g., coal)	GLE 29 days LLE 7.3 days	GLE 220 days LLE 7.3 days
None	GLE 0 LLE 0	GLE 0 LLE 0

GLE – Gain in life expectancy.
LLE – Loss in life expectancy.

One of the tables, reproduced here, from my NDP presentation implements this idea. It is intended to put the global impact of central station energy into the right logical and intellectual framework for decision making. It considers all important direct and indirect impacts and arrives at the total safety impact of each of the three realistic alternatives. The 'zero' option is an 'alternative,' to be considered equally with the other two; it is a clear departure from the general industrial development of the last two centuries. Any more narrow view would be literally reckless. It would raise the possibility, or probability, that in shying away from small risks we may expose ourselves to greater risks, or to loss of safety benefit, which is the same thing.

The numbers shown are a composite of my own work and that of Myers and his colleagues. To the best of my knowledge, they are the only attempts to make unbiased estimates which take both negative and positive safety contributions into account, and which are based as far as possible on real experience. They are disciplined, or try to be. The high safety benefit numbers are simply an attempt to apportion the great and sustained improvement of life expectancy in the modern world, which is hard fact, among the wealth-producing industrial activities which underlie it. The nuclear risk number takes Chernobyl into account.

The table indicates that the safety benefits to people from the production of energy in today's world far exceed the risks in both of the main categories considered. The present hassles between the regulatory bodies and the industry mainly boil down to arguments about increments of risk which are microscopic by comparison. For instance, it is my estimate that a faster second shutdown system for Pickering A would save about 0.000001 lives per reactor-year and would increase the expectation of life in Ontario by about a third of a second. By contrast, even the 0.35 day LLE figure for the total nuclear accident risk is the smallest figure in the table by a factor of 20.

What really matters, therefore, is whether or not the table is roughly right. Are Myers and I and our co-workers roughly right in our high assessment of safety benefits? Is the intermediate mortality figure for coal roughly right? And is the nuclear accident risk figure either roughly right or pessimistic? It is these questions which need to be answered by informed and responsible judgment. All the numbers are uncertain, but there is no 'conservative' escape route; we lose lives by eroding safety benefits just as readily as by underestimating risks.

The table should be dispassionately discussed in the scientific and professional community as part of our normal search for truth. Sadly, at this moment in the history of the human race, this is not likely to happen. If the numbers are accepted as roughly right, they show that human safety will best be furthered if both coal and nuclear energy are developed aggressively. They show that nuclear should displace coal to the greatest extent possible, so that a reduction in the cost of nuclear power could benefit safety even if the direct nuclear risk were increased somewhat as a consequence. In the present social climate, these are *unthinkable* thoughts. Experience shows that most scientists and engineers, like the vicar of Bray, will not take the risk of deviating far from strongly and widely held beliefs of the times. So the issues raised by my table will continue to be carefully ignored.

A final point about disasters of the Titanic / Chernobyl kind; their total effect on expectation of life in the world is negligible. In other words, they have virtually no effect on safety. Another unthinkable thought.

E. Siddall, P. Eng.

The Letter by E. Siddall raises a number of interesting topics. I would agree with many of his suggestions but would like to add some comments. The general area of risk management has attracted considerable scientific interest in recent years and is indeed an area which, in one form or other, accounts for the major proportion of our tax dollars. The question whether social effort would be more wisely spent in one direction or another is a continuing topic of discussion. Some other recent articles on risk management that readers might find of interest include a) B.N. Ames et al. ranking possible carcinogenic hazards. Science 236, April 1987: 271–279; b) M. Konner. Why the reckless survive. The Sciences N.Y. Acad. Sci., May / June 1987: 2–4; c) T. Morsing. Risk philosophy and misunderstanding. Nuclear Europe 6–7, June / July 1987: 25–26; d) R. Wilson and E.A.C. Crouch. Risk assessment and comparisons: An introduction. 236, April 1987: 267–270. As usual, E. Siddall's contribution to this discussion is most interesting and thought-provoking.

D.K. Myers

Book Reviews

Planet Earth in Jeopardy: Environmental Consequences of Nuclear War.
Lydia Dotto. Toronto: John Wiley & Sons, 1986.
ISBN 0 471 90908 4 (paper) $12.95. 134 pp.

In 1983 the Scientific Committee on Problems of the Environment (SCOPE) initiated a study at the request of the International Council of Scientific Unions (ICSU) which resulted in a 2 volume report entitled *The Environmental Consequences of Nuclear War: I. Physical and Atmospheric Effects and II. Ecological and Agricultural Effects.* This report was published by John Wiley and Sons in 1985. Lydia Dotto, a well-known Canadian science writer, was commissioned by an international scientific committee to write an overview of the report in less technical style for the benefit of lay readers. She has succeeded admirably.

The book is factual and the author goes to some pains to point out clearly the many areas of uncertainty, the assumptions used, and the need for more knowledge on such things as smoke production and properties, fire and climate modelling, chemistry in the atmosphere, fuel loading in and adjacent to likely targets of nuclear weapons, and the impact of electromagnetic pulses on communications and electronic systems in the midst of an international crisis.

There are approximately 25,000 nuclear weapons in the arsenals of each of the U.S.A. and the U.S.S.R., with much smaller numbers, totalling in the hundreds, in the hands of China, France, and the U.K. The total world inventory amounts to about 12,000 megatons with an explosive content of approximately one million Hiroshima bombs. Various scenarios are possible for a major nuclear war, but one reasonable possibility is that approximately half the total inventory might be detonated with the other half destroyed by accurate targetting.

Separate chapters are devoted to fire, blast, and other immediate effects; smoke and dust; climatic consequences; changes in the chemistry of the atmosphere; radiation and fallout; and the impacts on biology, agriculture, ecosystems, and people. Depending on the time of year the war occurred, an average temperature drop of 15–35° C in the northern hemisphere, due to smoke and dust obscuring the sun, could lead, in a matter of months, to the death by starvation of billions of people. Since as many as 1,100 million people might have been killed in the initial nuclear attacks, the total surviving population in the world could be barely one per cent of present numbers. The infrastructure of trade and modern agriculture would have been destroyed, setting civilization back thousands of years.

The impact of direct hits by nuclear weapons on nuclear power facilities was also evaluated. The immediate (within 48 hours) additional effect on fallout would be small, perhaps an additional 10 per cent. Because reactors and spent fuel bays contain more long-life isotopes, the long-term effect could be much greater, as much as a threefold increase in total fallout.

As the Chernobyl accident demonstrated, fallout knows no national boundaries. Since a major nuclear war would most likely occur in the northern hemisphere, all the major industrial countries would be affected. Declaring oneself a nuclear-free zone would be quite ineffective protection. Whatever one's views on nuclear deterrence, a reading of this book will make it abundantly clear that a major reduction by the superpowers of their enormous stockpiles of nuclear weapons is an essential first step in lifting the spectre of world destruction. Lydia Dotto has performed a valuable service in putting across this message in clear and unemotional form. The book deserves a wide reading and given, the source of its sponsorship, it is to be hoped that this will come about within the scientific and engineering community.

Alan Wyatt
Chairman,
Social Issues Committee
Canadian Nuclear Association

Understanding Chernobyl
The Uranium Institute, 1986. ISBN 0 946777 09 8

This booklet is made up of seven lectures presented to members of the Uranium Institute in September 1986,

with the objective of assisting Institute members who were not themselves reactor specialists to understand the Chernobyl accident and its implications for the future of the nuclear industry. An introduction to reactor physics is provided by Terence Price (Secretary General, Uranium Institute); the sequence of events at Chernobyl is described by John Gittus (UKAEA's Director of Safety and Reliability); Pierre Tanguy (Inspector General for Nuclear Safety, EdF) discusses the issues raised by Chernobyl and outlines an approach to reactor safety philosophy; John Dunster (Director, National Radiological Protection Board) describes radiation protection policy; and John Wright (Health and Safety Director, CEGB) covers the safety of gas-cooled reactors.

An inevitable problem with this kind of publication, with its diversity of contributors, is the uneveness in the technical level of the various lectures. For example, Price's 'Introduction to reactor physics' is comprehensive and detailed and would be an excellent introduction to the topic for a technically inclined senior high school student – or even a first year undergraduate. In contrast, Gittus, in 'What happened at Chernobyl' adopts an extremely simple approach, perhaps suitable for a non-technical audience of business executives. However, it could be argued that Gittus simplifies to the extent that his readers may form a misleading impression. His opening statement 'The Chernobyl reactor went wrong because it had three design deficiencies: instability, slow shutdown and inadequate automation of safety systems' surely should be preceded by the qualification that what actually brought about the accident was the deliberate violation by the operating crew of a number of operating policies and principles, rather than followed with the comment that, 'Coupled with a maloperation ... these deficiencies' led to the accident.

Pierre Tanguy's two lectures, 'The issues raised by Chernobyl' and 'Reactor safety philosophy' are first rate and would be of interest to any Canadian reader, regardless of technical background. Tanguy places an appropriate degree of emphasis on human and 'institutional' factors, drawing attention to the clear violations of operating regulations (rather than 'errors') and noting that 'At Chernobyl, starting with the superintendant of the plant down to the operator in the control room – and maybe going even higher ... there was not a proper awareness of nuclear risk.' This important point is too often and too easily obscured by detailed technical comparisons of RBMK with other reactor systems. In discussing the safety lessons of the Chernobyl accident, Tanguy issues a salutary reminder to the effect that while 'there is nothing totally new' in reactivity-induced, core disruptive accidents, 'we should not forget one issue. When we look at safety we no longer need to try to imagine new phenomena; but we must verify that already well identified risks are properly taken into account.' One area to which Tanguy could have directed somewhat more attention is that of the definition of 'nuclear safety culture,' a term which made its appearance in the INSAG report issued last year. As yet, the term has not been explicitly defined, yet it is cropping up with increasing frequency in the literature. If it is to be taken into the nuclear safety lexicon, then more rigorous attempts to arrive at an agreed definition should be made.

Tanguy's 'Reactor safety philosophy,' focussing as it does on LWR and GCR, may be of less interest to the Canadian reader – with the possible exception of his discussions of 'human redundancy' in the control room under severe accident conditions, and the use of a form of filtered air discharge system in French 900 MW PWR's to control containment pressure for the long term.

'The safety of gas cooled reactors,' by John Wright, draws attention to some of the attractive features of this kind of reactor, and perhaps lays greater emphasis than is entirely justified on the 'fundamental weakness' of the positive void coefficient in the RBMK design.

'The biological effects of radiation' (Peter Saunders) and 'Radiation protection policy' (John Dunster) are lucid and informative essays which, aimed as they are at the 'intelligent layperson,' will not be unfamiliar in content or argument to a North American reader. Dunster does comment on the actual radiological impact of the Chernobyl accident in Europe and England and has some remarks to make about the quite inappropriate interventions with food supplies made in the three months following the accident – interventions which were costly and 'made on an inadequate technical basis.'

The problem of a less-than-consistent level of technical content, combined with those problems inherent to transferring oral presentations to a written form without extensive emmendation mean that *Understanding Chernobyl* is difficult to recommend in its entirety to any specific audience. It is essentially a 'document of record.' As mentioned above, some sections (particularly those by Tanguy) are interesting reading for specialist and non-specialist alike. But the publication cannot stand alone as a guide for the non-technical person to the Chernobyl accident. For the motivated, intelligent non-specialist, the best introduction to the accident is probably still the INSAG report.

David Mosey

CNS Conferences and Seminars

Canadian Nuclear Society Annual Conference

The 8th Annual Conference of the Canadian Nuclear Society and the 27th annual international conference of the Canadian Nuclear Association were held in Saint John, New Brunswick, 14–17 June 1987. Attendance was 447, including a number of overseas representatives and a strong delegation of speakers from the International Atomic Energy Agency, Vienna (the paper by L.L. Bennett on 'The Outlook for Nuclear Power After Chernobyl' is contained in this issue of the Nuclear Journal of Canada).

Nearly 70 papers were presented at the technical sessions of the CNS Conference. Many subjects were covered, including innovative features of the new CANDU 300 for life extension, and improvements to the CANDU 600 design. The Darlington Probabilistic Safety Evaluation study was unveiled at the conference, the first application of fully integrated event tree / fault tree risk assessment methods to a CANDU reactor (pre-operational) design, which has resulted in nearly 100 design changes.

The following papers were presented:

Plant Life Extension

'Virginia Power's Unit Life Extension Program.' P.R. Beament, Virginia Electric and Power Company.

'EPRI Overview and Generic Aspects of PWR & BWR Studies.' C. Bergeron, Stone and Webster; Melvin E. Lapides, EPRI.

'CANDU 300 Features for Plant Life Extension.' R.S. Hart, AECL.

'Assessment of Plant Life Extension (PLEX) and Long-Term Reliability Assurance (LTRA) Issues for Ontario Hydro's Nuclear Generating Stations' J.A. Chadha, R.J. Strickert, Ontario Hydro.

Safety and the Environment

'Modifications to Augment Reactor Trip Coverage for a Single Pump Trip in a CANDU 600 Reactor.' P.D. Thompson, NB Power; A. Baudouin, Hydro Quebec; L.J. Watt, G. Delorme, GAN.

'Consequences of Pressure / Calandria Tube Failure in a CANDU Reactor Core During Full-Power Operation.' A.P. Muzumdar, G.M. Frescura, Ontario Hydro.

'Gentilly-2 Safety Analysis for Higher Containment Leakage.' R.H. Hu, Hydro Quebec; M.S. Quraishi, AECL SH PK.

'UO_2 Oxidation in Air or Steam – Release or Retention of the Fission Products Ru, Ba, Ce, Eu, Sb, and Nb.' C.E.L. Hunt, F.C. Iglesias, D.S. Cox, N.A. Keller, R.D. Barrand, J.R. Mitchell, R.F. O'Connor, AECL CRNL.

'Fission Product Release from UO_2 in Air During Temperature Ramps.' D.S. Cox, F.C. Iglesias, C.E.L. Hunt, N.A. Keller, R.D. Barrand, R.F. O'Connor, J.R. Mitchell, AECL CRNL.

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